5.4.2 Steam Generators

The AP600 design has two vertical-shell, U-tube SGs. The basic function of these SGs is to transfer heat from the primary reactor coolant through the U-shaped heat exchanger tubes to the secondary side for steam generation. The design of the AP600 SGs, except for the configuration of the channel head, is the same as an upgraded Model F SG with a triangular pitch tube bundle, called the Model Delta-75 SG. In the channel head under the SG tube sheet, a divider plate is used to separate the inlet and outlet chambers. Two canned-motor RCPs are directly attached to the cold leg nozzles on the outlet channel head to provide the driving force for the reactor coolant flow. A passive residual heat removal (PRHR) nozzle is attached to the bottom of the channel head of the Loop 1 SG on the cold leg portion of the head. This nozzle provides recirculated flow from the PRHR heat exchanger (PRHRHX), which cools the primary side under emergency conditions.

The SG channel head, tubesheet, and tubes are a portion of the RCPB, and are designed to satisfy the criteria specified for Class 1 components. The tubes transfer heat to the secondary (steam) system while retaining radioactive contaminants in the primary system.

The SGs remove heat from the RCS during power operation, anticipated transients, and under natural circulation conditions. The SGs' heat transfer function and associated secondary water and steam systems are not required to provide a safety-grade safe shutdown of the AP600. Safe shutdown is achieved and maintained by the safety-related passive core cooling systems.

For the SG operation, the reactor coolant flow from the RCS hot leg enters the primary side of inverted U-tubes, transferring heat to the secondary side during its traverse. The flow then returns to the cold leg side of the primary chamber, exits the SG via two cold leg nozzles and the canned RCPs, to the RV, thus completing a cycle.

If the PRHR system is activated, flow passes from the outlet of the PRHRHX, through the SG's PRHR nozzle connection into the SG channel head. Coolant then flows through the RCPs, into the cold legs and then into the RV.

On the secondary side, feedwater enters the SG at an elevation above the top of the U-tubes through a feedwater nozzle. The feedwater enters a feedring via a welded thermal sleeve connection, and leaves it through nozzles attached to the top of the feedring. This nozzle design minimizes the potential for trapping pockets of steam that can lead to water hammer in the feedwater piping, by discharging feedwater into the SG at an elevation above the top of the tube bundle and below the normal water level, thus reducing the potential for vapor formation in the feedring. After exiting the nozzles, the feedwater mixes with saturated water that has been mechanically separated from the steam flow exiting the SG by internal moisture separators. The combined feedwater/recirculation flow then enters the downcomer annulus between the tube wrapper and the shell. At the bottom of the tube wrapper, the water is directed toward the center of the tube bundle by a flow distribution baffle. The baffle arrangement is designed to minimize low-velocity zones, which present the potential for sludge deposition. As the water passes the tube bundle, it is converted to a steam-water mixture, which, subsequently, rises into the steam drum section, where 18 centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary moisture separators, or dryers, for further moisture removal, increasing its quality to a designed minimum of 99.75 percent (0.25 percent by weight maximum moisture). Water separated from the steam combines with entering feedwater and recirculates through the SG. A sludge collector located amidst the six inner primary moisture separator risers provides a benign region for sludge settling away from the tubesheet and tube support plates. Dry steam exits the SG through the SG outlet nozzle, which has an installed steam-flow restrictor.

The startup feedwater system (SUFS) supplies water to the SGs during startup, shutdown and other times when the normal feedwater system is not needed or not operable. The SUFS is a non-safety grade system that will be used as a defense-in-depth system following a reactor trip or loss of main feedwater event. The SUFS thus provides investment protection for the plant.

During startup and shutdown operations, the SG has enough surface area and a small enough primary-side hydraulic resistance to remove decay heat from the RCS by natural circulation (without operation of the RCPs).

The SG design requirements and design parameters are shown in Tables 5.4-4 and 5.4-5 of the SSAR, respectively. The evaluation of SG thermal performance, including required heat transfer area and steam flow, uses conservative assumptions for parameters such as primary flow rates and heat transfer coefficients. The effective heat transfer coefficient is determined by the physical characteristics of the AP600 SG and the fluid conditions in the primary and secondary systems for the nominal 100 percent design case. It includes a conservative allowance for fouling and uncertainty.

As stated above, the SG heat transfer function is not required for safe shutdown. Because the secondary systems, such as the normal feedwater system and the SUFS are not safety-related systems, they cannot be credited in the SG heat transfer function for mitigation of transients and accidents in the design-basis analyses. The staff reviewed and confirmed that no credit of these non-safety-related systems is taken in the analyses of the design-basis transients and accidents of in Chapter 15. However, in the evaluation of non-design-basis multiple SG tube rupture (MSGTR) events using realistic calculations, the heat transfer function as well as other accident-mitigating characteristics of the SG may be considered. The MSGTR/containment bypass issue is discussed in Section 5.4.2.2 of this report.

5.4.2.1 Steam Generator Materials

GDC 1 of Appendix A of 10 CFR Part 50 requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. The NRC staff reviewed the AP600 SG materials to ensure that the relevant requirements of GDC 1 have been met as they relate to the selection of materials for the SG to determine their adequacy to assure a quality product commensurate with the importance of the safety function to be performed.

GDC 14 requires that the 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. The staff reviewed the SG materials to ensure that they meet the relevant requirements of GDC 14 as they relate to the design, fabrication, and testing of RCPB to achieve an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOO). The staff reviewed the SG materials to ensure that they meet the relevant requirements of GDC 15 as they relate to the provision of margins sufficient to assure that design conditions are not exceeded during normal operation and AOO.

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 the probability of rapidly propagating fracture minimized. The staff reviewed the SG materials to ensure that the relevant requirements of GDC 31 have been met as they relate to an extremely low probability of rapidly propagating fracture or gross rupture of the RCPB.

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing 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 SG 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.

The AP600 SG, discussed in Section 5.4.2 of the SSAR, is a vertical-shell U-tube evaporator with integral moisture separating equipment. The construction of SG components will be in accordance with Section III of the ASME Code, as required by 10 CFR 50.55a(3)(ii)(b)(1). The staff evaluated the SG materials in accordance with Section 5.4.2.1 of the SRP. The areas of review included selection and fabrication of materials, SG design, compatibility of the SG components with the primary and secondary coolant, and cleanup of secondary coolant.

The materials selected for the principal pressure-retaining components of the SG are listed in Table 5.2-1 of the SSAR. These include carbon and low-alloy steels, austenitic stainless steels, and nickel-chromium-iron alloys. Section 5.2.3 of the SSAR provides a general discussion of RCPB materials specifications, including those for the SG materials. The primary side of the SG is designed and fabricated to comply with the ASME Code, Section III, Class 1 criteria while the secondary-side pressure boundary parts are designated as Class 2. However, for the AP600, all pressure-retaining parts of the SG, and thus both the primary and secondary pressure boundaries, are designed to satisfy the criteria specified in Section III of the ASME Code for Class 1 components. The staff's evaluation of the AP600 RCPB materials in general is located in Section 5.2.3 of this report. That section includes discussions of aspects of fabrication, cleaning process specifications, and the fracture toughness of the materials that are applicable to the AP600 SG materials.

Extensive crevices and conditions that promote dryout should be avoided in the tube-to-tube sheet area and tube-to-tube support plate area, to prevent the buildup of potentially corrosive material. In the AP600 SG, the portion of the SG tube that is situated within the tubesheet is expanded hydraulically to close fully the crevice between the tube and tubesheet. The length of the tube expansion is carefully controlled to minimize the potential for an over-expanded condition above the tube sheet or an unexpanded tube within the tubesheet. Because the tubes are expanded to the full depth of insertion in the tubesheet, the crevice between the tubesheet and the inserted tube is minimal. The positive contact pressure between the tube and the tubesheet precludes the formation of impurity buildup in the tube-to-tube sheet crevice region and reduces the probability of crevice boiling.

The AP600 tube support plates have a three-lobed, or trifoil, tube hole design and provide in-plane and out-of-plane strength. The trifoil design eliminates the narrow annular gap at the tube supports, because the support contacts the tube at only three lines on the tube circumference, providing almost complete washing of the tube surface with SG water. The design of the support plate also contributes to a high circulation ratio. This design provides high sweeping velocity at the tube-to-tube support intersections and consequently reduces sludge accumulation in the tube-to-tube support crevices.

The AP600 design utilizes all-volatile treatment (AVT) for the SG water. The AVT control program minimizes the possibility of tube wall thinning from wastage. Successful AVT operation requires maintenance of low concentrations of impurities in the SG water. This reduces the potential for formation of highly concentrated solutions in low-flow zones, which is the precursor of corrosion. The AP600 SG tubes are fabricated of thermally treated Alloy 690 and have a wall thickness of 0.1 cm (0.040 in). Laboratory testing has shown that Alloy 690 is compatible with the AVT environment. On the basis of the guidance in EPRI NP-6239, "PWR Secondary Water Chemistry Guidelines," Revision 2, dated December 1988, this approach is acceptable. However, the staff requested in the DSER that the AP600 design meet other guidance of EPRI NP-6239, Revision 2, and those of EPRI NP-5960 "PWR Primary Water Chemistry Guidelines," Revision 1, dated August 1988. This was Open Item 5.4.2.1-1. The reactor coolant water chemistry specifications for the AP600 were subsequently provided in Table 5.2.2 of Revision 3 of the SSAR, and the secondary-side water chemistry guidelines in Chapter 10 of the SSAR. The information included in the SSAR indicates that the AP600 water chemistry specifications conform with the respective guidelines provided by the two EPRI reports. In a few instances, for example, the aluminum, calcium and magnesium control parameters for the primary-side water, the AP600 specifications are stricter than those contained in the EPRI reports. Thus, the staff concludes that the controls imposed on the primary and secondary water chemistries of the AP600 are appropriate and consistent with current power plant activities in these areas. Therefore, Open Item 5.4.2.1-1 is closed.

The materials of construction for the primary-side components include nickel-based alloys, stainless steels, and low-alloy steels clad with corrosion-resistant material (i.e., nickel-based alloy or stainless steel). Numerous studies have documented the resistance of all of these materials to general corrosion in high-temperature aqueous environments. On the basis of a review of operational experience and test data, a conservative corrosion allowance for Alloy 690 is 0.00025 cm/year (0.01 mils/year) of operation. This indicates that, over 60 years, metal loss as a result of corrosion of both the inside and outside surfaces of Alloy 690 tubing would be about 1.2 mils (0.003 cm) total. The design of the AP600 SG includes a lifetime corrosion allowance of 3 mils (0.008 cm) for the Alloy 690 tubing. Such an allowance is appropriate and acceptable to staff. On the secondary side, the unclad carbon and low-alloy steels used for the SG shells are exposed to the secondary environment, as are the outside surfaces of the tubes, the tube supports and the flow distribution plate (the latter two components made of Type 405 stainless steel). Revision 0 of the SSAR did not include allowances for materials other than Alloy 690. The staff requested that Westinghouse revise the SSAR to include the assumed corrosion allowances for the SG's shell and support plate materials. This was Open

Item 5.4.2.1-2. In the revised SSAR (Revision 5), Westinghouse indicated that the allowance for the shell is 50 mils (0.13 cm) while the stainless steel support plates do not have an allowance. A review of data derived from operational experience and laboratory testing indicates that, for carbon and low-alloy steels in secondary-side applications, a corrosion allowance of .0025 cm/year (1.0 mils/year) of operation would be conservative. The overall allowance of 50 mils (0.13 cm) for the shell is thus reasonable and acceptable. With respect to the support plates, test data have indicated that the corrosion rates of the martensitic stainless steels will be less than 0.1 mil/year (0.00025 cm/yr), leading to a metal loss of 6 mils (0.015 cm) or less over a 60-year period. Such a loss would be insignificant compared with the thickness of the support plates, and the decision not to include a corrosion allowance is, therefore, acceptable. On these bases, the staff concludes that the actions taken in the design of the AP600 SG to account for corrosion-induced metal loss are acceptable. Therefore, Open Item 5.4.2.1-2 is closed.

The components of the AP600 SG will be designed to the rules of the ASME Code and should address the potential influence of environmental effects on the fatigue life of materials over the 60 year design life. A special Steering Committee for Cyclic Life and Environmental Effects in Nuclear Applications of the Pressure Vessel Research Council (PVRC) reviewed the issue of environmental effects on fatigue. These activities were initiated by requests from the ASME Code Committee and the Board on Nuclear Codes and Standards (BNCS). The charter of the PVRC Steering Committee is to provide guidance and direction related to determining the effects of light water-reactor (LWR) service environments on the cyclic life properties of applicable materials. The Steering Committee also evaluated application methodologies that include these effects in the fatigue-analysis process. Preliminary recommendations were provided to the BNCS in September 1992. The initial findings reported to BNCS were that the current serial number (S/N) curves should be appropriate for PWR environments. However, there is not complete agreement on this position among the Code members. Westinghouse will continue to monitor the industry activities on the fatigue curve and fatigue-analysis methodology. The AP600 components are designed to the ASME Code requirements. The staff's evaluation of this issue is discussed in Sections 3.9.3.1 and 3.12.5.7 of this report.

The AP600 SG is designed to permit inspection of pressure boundary parts, including individual tubes. Preservice inspection of the AP600 SGs is performed in accordance with the ASME Code. To meet the ISI requirements of 10 CFR 50.55a of 10 CFR Part 50, the design includes a number of openings to provide access to both primary and secondary sides of the SG. These openings include the following:

- four 53.34 cm (21 in) diameter manways, one for access to each chamber of the reactor coolant channel head, and two in the steam drum for inspection and maintenance of the upper shell internals

- six 15.24 cm (6 in) diameter handholes in the shell, four located just above the tubesheet secondary surface, and two located just above the flow distribution baffle

- two 10.16 cm (4 in) diameter inspection openings at each end of the row 1 tubes

Additional access to the tube bundle U-bend is provided through the internal deck plate at the bottom of the primary separators.

The AP600 SG design permits access to tubes for inspection, repair, or plugging, if necessary, per the guidance described in RG 1.83, "ISI of Pressurized Water Reactor Steam Generator Tubes." The AP600 SG also includes features to enhance robotics inspection of the SG tubes without manned entry of the channel head. These include a cylindrical section of the channel head, primary manways, and provisions to facilitate the remote installation of nozzle dams. The tube location for a large fraction of the tubes is scribed on the tubesheet, to facilitate tube identification for manual activities. The SG tube integrity is verified in accordance with the SG surveillance program, which is the responsibility of the COL applicant. This is COL Action Item 5.4.2.1-1 and is addressed in SSAR Section 5.4.15 and Technical Specification administrative control 5.5.5.

Operating experience indicates that SGs might have to be replaced during the plant lifetime. The SSAR states that the AP600 design facilitates SG replacement. The staff requested that Westinghouse describe its procedures for removing and replacing AP600 SGs. This was Open Item 5.4.2.1-3. Westinghouse responded that full procedures had not yet been developed for the replacement of SGs in the AP600 and they did not consider that inclusion of such detail in the SSAR was a requirement for design certification. They emphasized that consideration of SG removal and replacement was an integral part of the design, and such operations were feasible. The equipment hatch is sized to permit one-piece removal of the SG, and the crane rail girder and the polar crane bridge are designed to permit SG replacement. The design of the containment internal structures and the path through the annex building include consideration of the loads associated with the transport of a SG during its removal and replacement. The staff concludes that consideration of the potential needs associated with SG removal and replacement operations have been adequately addressed in the design of the AP600. The staff further notes that inclusion of detailed procedures for these operations in the SSAR is not essential for design certification. Therefore, Open Item 5.4.2.1-3 is closed.

The staff concludes that the SG materials specified in the SSAR are acceptable and meet the requirements of GDCs 1, 14, 15, and 31, and Appendix B to 10 CFR Part 50. The staff's conclusion is on the basis of the following:

- The applicant has met the requirements of GDC 1 with respect to codes and standards by ensuring that the materials selected for use in Class 1 and Class 2 components will be fabricated and inspected in conformance with Section III of the ASME Code 1. Welding qualification, fabrication, and inspection during manufacture and assembly of the SG will be done in conformance with the requirements of Section IX of the ASME Code.

- The requirements of GDCs 14 and 15 have been met to ensure that the reactor coolant boundary and associated auxiliary systems will be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture during normal operation and anticipated operational occurrences.

- The primary side of the SG is designed and fabricated to comply with the Class 1 criteria of Section III of the ASME Code, as required by the staff. The secondary side is specified as Class 2 but, for the AP600, the secondary pressure boundary parts of the SG are designed to satisfy the Class 1 criteria of Section III of the ASME Code.

- The requirements of GDC 31 have been met with respect to the fracture toughness of the ferritic materials because the pressure boundary materials of the ASME Class 1 components of the SG comply with the fracture toughness requirements and tests of Article NB-2300 of Section III.

- The requirements of Appendix B of 10 CFR Part 50 have been met because the SG construction is subject to a quality assurance program. The pressure boundary components meet requirements established by the ASME Code and ANSI/ASME NQA-1 and NQA-2. The controls placed on the secondary coolant chemistry are in agreement with staff technical positions.

- Reasonable assurance of the satisfactory performance of SG tubing and other generator materials is provided by the following:

  - the design provisions and the manufacturing requirements of the ASME Code

  - rigorous secondary water monitoring and control

The controls described above, combined with conformance with applicable codes, standards, staff positions, and RGs, constitute an acceptable basis for meeting, in part, the requirements of GDCs 1, 14, 15, and 31, and Appendix B to 10 CFR Part 50.

5.4.2.2 Containment Bypass Resulting From Steam Generator Tube Rupture

In SECY-93-087, the staff identifies a containment performance issue where rupture of one or more SG tubes could lead to actuation of the SG safety relief valves, thereby creating the potential for a stuck open safety relief valve, and unisolable LOCA, with discharge of primary system radioactive inventory outside the containment. SECY-93-087 specifies that applicants for design certification for passive or evolutionary PWRs assess design features to mitigate containment bypass leakage during steam generator tube rupture (SGTR) events. The staff also recommends certain design features for consideration that could mitigate the release associated with an SGTR:

(1) a highly reliable (closed loop) SG shell-side heat removal system that relies on natural circulation and stored water sources

(2) a system that returns some of the discharge from the SG relief valve back to the primary containment

(3) increased pressure capacity on the SG shell side with a corresponding increase in the safety valve setpoints

Appendix 1B of the SSAR provides a risk-reduction evaluation of severe accident mitigation design alternatives (SAMDA) for the AP600 design. A total of fifteen design alternatives were selected for evaluation, including the three design features mentioned above. Each design alternative is evaluated to determine whether its safety benefit from risk reduction outweighs the costs of incorporating it in the plant. The results showed that total risk averted by each of these design features ranges from 4.2E-04 to 5.3E-04 man-rem per year, and therefore these SAMDAs are not cost effective.

In response to a staff RAI 440-110, Westinghouse stated that the AP600 has features that reduce the chance of containment bypass following a SGTR, and that those features include the multiple levels of defense described in WCAP-13793, "AP600 Systems/Event Operation Matrix." WCAP-13793 provides a qualitative description of the multiple levels of defense available in the AP600 design. These multiple levels of defense include actuation of many safety- and non-safety-related systems. Equipment combinations noted by the applicant include the following:

- automatic actuation of the non-safety-related RCS makeup chemical and CVS and SUFS in conjunction with manual isolation of the faulted SG to reduce the RCS pressure

- automatic actuation of the core makeup tanks (CMTs), PRHRHX, and passive containment cooling system (PCS)

- automatic actuation of the CMTs, ADS, in conjunction with manual actuation of the RNS

- automatic actuation of the CMTs, ADS, IRWST, accumulator, and PCS

In the DSER, the staff identified Open Item 5.4.2.2-1, stating that the staff's review of the containment bypass resulting from SGTR was not yet complete. This open item now closed because the staff has completed its evaluation, as discussed below.

By letter dated November 11, 1997 (NSD-NRC-97-5431), Westinghouse submitted topical report WCAP-14991, "AP600 WCAP Multiple Steam Generator Tube Rupture Analysis Report," which provides an evaluation of the AP600 plant response to the rupture of multiple SG tubes performed with best-estimate MAAP4 analyses. The staff's evaluation regarding the containment bypass/SGTR issue is discussed below.

5.4.2.2.1 AP600 SGTR Mitigation Design Features

The AP600 design incorporates several automatic protective actions and passive core cooling systems (PXS) for mitigation of the consequences of an SGTR. The automatic protective actions include reactor trip, actuation of the PXS, trip of the RCPs, termination of pressurizer heater operation, and isolation of the CVS flow and the SUFS flow. These protective actions result in automatic cooldown and depressurization of the RCS, termination of the break flow, stabilization of the RCS, prevention of SG overfill, and termination of release of steam to the atmosphere to minimize offsite radiation. The operator may also take actions in accordance with the emergency response guidelines for mitigation and recovery of an SGTR.

The PXS for cooling and makeup of the RCS include the CMTs to provide high pressure RCS recirculation and safety injection, the conventional accumulators, the PRHRHX to transfer decay heat in the RCS to the IRWST, and the ADS to depressurize the primary system to allow for low-pressure gravity injection from the IRWST and the containment recirculation. The CMTs automatically actuate on a safeguards signal or low pressurizer level. The PRHRHX automatically actuates on the CMT actuation signal, high pressurizer pressure, or low SG level. The ADS has four stages that actuate in sequence, with the first stage actuating on the predetermined CMT Low-1 level, the second and third stages actuating sequentially after preset time delays, and the fourth stage subsequently actuating on the CMT Low-2 level. The IRWST actuates upon the actuation of the fourth stage ADS.

On the secondary side, a power-operated relief valve (PORV) is installed on the outlet piping from each SG to provide a means for plant cooldown by discharging steam to the atmosphere when the turbine bypass system is not available. The PORV automatically opens to release steam when the steamline pressure exceeds its predetermined set pressure, which is below the main steam safety valve (MSSV) set pressures; and will close and reseat at a pressure at least 10 psi below the opening setpoint as the steam pressure decreases. A block valve, upstream of the PORV, with a safety-related operator closes automatically on low steamline pressure to terminate steam release in the event of a PORV that is stuck open. In addition, the SG overfill protective actions automatically trip the CVS and the SUFS flow on high SG water level to prevent SG overfill.

These passive systems and automatic protective actions provide a unique plant response to SGTR events. In addition, non-safety-related pumped injection sources are also available to the operator for mitigation of a SGTR accident as defense-in-depth.

In a scenario of a SGTR without operator actions, continued loss of RCS inventory to the SG secondary side through the ruptured tubes leads to a reactor trip on a low pressurizer pressure or overtemperature delta-T signal, and also causes the turbine trip. The CVS injection also actuates on low pressurizer pressure or level to provide non-safety-related coolant makeup to the RCS. A safeguards signal initiated on a low pressurizer pressure or level signal actuates the CMTs as well as the PRHRHX. The CMTs inject water in recirculation mode to exchange cold borated water for hot RCS water, thus providing heat removal and coolant inventory makeup for shrinkage in the RCS. The PRHRHX removes decay heat and reduces the RCS pressure below the pressure of the secondary system, and thus shuts off the break flow to the faulted SG. The heat is removed from the primary system using the PRHRHX instead of the intact SG. Because the CMTs do not drain during the recirculation injection mode, the CMT level remains above the ADS first stage actuation setpoint, and the ADS is not actuated.

On the secondary side, the turbine trip causes the opening of the turbine bypass valve (TBV). The steam pressure does not reach the main steam isolation setpoints of low steamline pressure or high pressure decrease rate, and the main steam isolation valve (MSIV) remains open with the steam release through the TBV. Eventually, the MSIV is closed on the low cold leg temperature. As the SG water level increases, the protective system will trip the CVS and the SUFS flow to prevent overfilling of the SG. When the steam pressure reaches the setpoint of the PORV, the PORV opens to release steam, and closes as the pressure decreases. As the primary system cools down and the pressure decreases to match the secondary pressure, the break flow decreases and eventually terminates.

In the event that the PORV fails to open, the steam pressure will increase further and eventually open the MSSVs to release steam into the atmosphere, and then reseat as the pressure decreases. The scenario is similar to the opening of the PORV except for the MSSV opens at a higher pressure setpoint.

In the unlikely event of a stuck open MSSV, continued RCS leakage through the ruptured tubes will result in the draining of the CMT, the ADS will automatically open on the CMT low level actuation setpoint to rapidly depressurize the RCS so that the IRWST can provide a gravity injection to flood the core.

5.4.2.2.2 MSGTR Analysis

In Section 15.6.3 of the SSAR, Westinghouse provides the design-basis analysis for a single SGTR. The design-basis analysis assumed no operator actions, and assumed a PORV fails to reseat after it opens with continued release through the PORV until the block valve closure at low steamline pressure. The results showed no fuel failure, no SG overfilling, and the resulting offsite radiological doses are within the dose acceptance limits.

In WCAP-14991, Westinghouse provided an evaluation for the beyond design-basis events of a multiple-tube rupture of up to five tubes. The evaluation's intent was to demonstrate the capability of the safety systems and automatic actions for mitigation of the multiple-tube rupture events. No operator actions were modeled in the analysis. The analysis was performed for various cases ranging from a one- to five-tube rupture using the MAAP4 accident analysis code to model the thermal-hydraulic response of the AP600 design. Base cases with the best-estimate PRHR heat exchanger heat removal and the secondary PORV operation were analyzed for ruptures of one through five tubes to bound the range of break flow. In addition, sensitivity analyses were performed on the five-tube rupture case to examine the effects of increased and decreased PRHR heat exchanger capacity, operation of CVS, break elevation uncertainty, and the failure of the PORV to open. A case with no passive systems was performed to show the adequacy of the active systems and the time available before the need for operator action to cool down the RCS. An analysis assuming the opening and stuck-open of the MSSV was performed to demonstrate that the plant response will not uncover the core.

<u>Analysis MAAP4 Model and Assumptions</u>

The analyses assume that the tube rupture occurs at 100- percent power, and all the passive safety systems are available. The MAAP4 code only models the secondary system to the MSIV. The code does not couple the secondary system in the two SGs with the steam header. The MAAP4 modeling assumes the MSIV closes at the time of turbine trip in the analysis. Therefore, the steam flow through the turbine bypass valve to the condenser is not modeled. This is a conservative assumption as it results in earlier pressurization of secondary side. Other assumptions in the analysis include (1) the CMT injection and RCP trip occurs because of low-2 pressurizer level and (2) the reactor trip and turbine trip occur as a result of the CMT injection signal.

<u>MAAP4 Benchmark</u>

MAAP4 is a fast-running thermal-hydraulic computer code designed for severe accident analyses, and chosen by the applicant for the AP600 probabilistic risk assessment (PRA) evaluation, as well as the evaluation of MSGTR. WCAP-14869, "MAAP4/NOTRUMP Benchmarking to Support the Use of MAAP4 for the AP600 PRA Success Criteria Analyses," documents the MAAP4/NOTRUMP benchmark exercises comparing the MAAP4 results with the NOTRUMP analysis results as a part of AP600 PRA evaluation. NOTRUMP is a detailed thermal-hydraulic code, which has been validated and approved for the AP600 design-basis analysis of small break LOCAs. The MAAP4 benchmarking against NOTRUMP provides a basis for the MAAP4 PRA success analyses. To assure appropriateness of using MAAP4 for the MSGTR evaluation, the staff also evaluated whether MAAP4 has proper models and benchmarking for modeling the important systems and phenomena in the MSGTR progression.

In response to RAI 440.676 (NSD-NRC-97-5393, dated October 22, 1997) regarding the appropriateness of MAAP4 evaluation of MSGTR events, the applicant provided a list of important phenomena in a SGTR event progression from the LOFTTR2 phenomena identification and ranking table (PIRT), and a comparison to the PRA PIRT from the MAAP4/NOTRUMP benchmark study. The LOFTTR2 code has been approved for the AP600 design basis analysis of a single steam generator tube rupture event. The phenomena on the LOFTTR2 PIRT, which ranks the SGTR phenomena as high, medium, and low importance, are applicable to the MSGTR sequences that do not involve stuck-open steam generator safety valves, thereby not voiding the RCS or actuating the ADS. The PRA PIRT breaks the small break LOCA sequence into four timeframes (i.e., blowdown, natural circulation, ADS blowdown, and IRWST gravity drain), and ranks the phenomena within each timeframe. Only the phenomena in the blowdown and natural circulation timeframes are of interest to this comparison to LOFTTR2. The phenomena that occur in the RCS for a small break LOCA during the blowdown and the natural circulation phases of the accident are similar to those that occur during the SGTR event. Therefore, the MAAP4/NOTRUMP benchmark is applicable for demonstrating the capability of MAAP4 for modeling the RCS response during SGTR, as well.

For the MSGTR event that results in a stuck-open MSSV, the event progression is essentially a small break LOCA that goes through all four timeframes in the PRA PIRT. With the safety valve stuck open, the faulted SG secondary system is depressurized and filled with water. The RCS voids, the CMTs drain, the ADS actuates, and the IRWST water provides long-term cooling. The behavior on the secondary side of the faulted SG is governed by the stuck open safety valve, and the SG acts as a break flow resistance before ADS actuation. The MAAP4/NOTRUMP benchmark demonstrates that MAAP4 is capable of adequately modeling this sequence.

Among the important phenomena relevant to the MSGTR without a stuck-open MSSV, the break critical flow, core decay heat, the CMT recirculation, and the balance line pressure drop models, and the reactor trip timing have been verified by the MAAP4/NOTRUMP benchmark exercises. The PRHRHX flow and heat transfer was not modeled in the PRA success case. The MAAP4 code models the PRHRHX using the same boiling pool heat transfer model employed for heat transfer from the RCS to the SG during natural circulation, and uses an input correction factor to account for modeling uncertainties. This PRHRHX model is benchmarked against independent design calculations for the maximum, minimum, and best-estimate single-phase water natural circulation heat removal within the range of water temperatures expected in the MSGTR cases. The results of this benchmark, presented in Figures 2-1 through 2-3 in WCAP-14991, show good agreement with the trends and magnitude of heat removal. The sensitivity analyses show that uncertainty in the modeling does not affect the result of the MSGTR analyses.

The LOFTTR2 PIRT ranks the IRWST transient response to be of importance. This is not modeled in the PRA success cases which do not credit the PRHRHX operation. In the MAAP4 MSGTR analysis where the PRHRHX is an important heat removal mechanism, conservative value of the initial temperature of the IRWST water is used.

The steam generator secondary system SGTR modeling is not considered in the MAAP4/NOTRUMP benchmark for the PRA success analysis. The MAAP4 one-node SG modeling includes reasonable models for the secondary system conditions important to the SGTR modeling, especially the secondary system pressure, level and safety/relief valve critical flow calculations. The thermal-hydraulic modeling of the secondary side of the SGs, including the mass and energy balance, thermodynamic conditions and heat transfer from the primary to the secondary system, is described in the MAAP4 user's manual. The secondary system pressure determines the break flow back pressure and the safety/relief valve opening. The SG void fraction is calculated and the water level is tracked via a user input table of level as a function of volume. These parameters determine the timing of the CVS and SUFS isolation and the quality of the relief flow. The safety and relief valve open flow areas are calculated on the basis of the input parameters for the flow rate of the system at the opening pressure setpoint.

Although the MAAP4 code has not been evaluated as rigorously as that for a design-basis analysis code, for the reasons set forth above the staff believes that MAAP4 provides a reasonable analysis tool for the beyond design-basis analysis of MSGTR.

<u>Analysis Results</u>

The results of the analyses can be summarized below for cases ranging from rupture of one to five tubes, sensitivities in the break elevation, PRHRHX performance, failure of the PORV to open, operation of the CVS injection, and the stuck-open MSSV:

(1) As a result of a tube rupture, the secondary pressure increases so that the PORV opens to release the steam into the atmosphere. The MSSVs remain intact as the steam pressure never reaches the MSSV setpoints. The PORV reseats as the steam pressure decreases. The RCS pressure decreases as the decay heat decreases below the heat removal capability of the PRHRHX. The break flow decreases and eventually ceases when the RCS pressure matches the secondary pressures.

(2) In the case where the PORV fails to open, the steam pressure continues to increase until the MSSV with the lowest setpoint opens to release steam into the atmosphere. The MSSV closes as the steam pressure decreases similar to the cases with the PORV. Because of the automatic SG overfill protection, which trips the CVS and SFW flow, the SG is not overfilled and only steam is released through the MSSV.

(3) Throughout the events, the core makeup tanks inject water in the recirculation mode, exchanging cold borated water for the hot reactor coolant. The CMTs do not drain, and therefore, the ADS does not actuate.

(4) In the case that assumes the MSSV fails to reseat after it is actuated, the SGTR scenario turns into a small break LOCA. Continued loss of coolant through the ruptured tubes and the stuck-open MSSV eventually leads to the voiding of the RCS, the draining of the CMT, and the actuation of ADS. The RCS is rapidly depressurized, which results in the actuation of the IRWST and eventual containment recirculation. The core remains covered and cooled. The maximum total release would be limited to the initial activity in the RCS.

5.4.2.2.3 Conclusion

The AP600 design has unique features for mitigation of the SGTR. The analysis indicates that the PORV will automatically open to release steam and reseat within a very short time. Throughout the accident, the core remains covered without voiding, and the SG is not overfilled. The applicant indicates that the probability of the PORV failure to open is in the order of 1E-03. If the PORV fails to open, the MSSV will open and close within a short time. Because of the automatic overfill protection, the SG is not overfilled, and the MSSV will release steam only. The applicant indicates that the failure of the MSSV to reseat when releasing steam is on the order of 1E-03. In the extremely unlikely event of failure of the PORV to open coincident with failure of the MSSV to reseat, an unisolable small-break LOCA scenario occurs with release to the atmosphere. In this event, continued steam release and loss of reactor coolant through the ruptured tubes will result in draining of the CMTs. The ADS will be actuated as the CMT level falls below the ADS actuation setpoint. Rapid depressurization of the RCS eventually results in the gravity injection from the IRWST, as well as the containment recirculation as the IRWST empties. Eventually, the break flow through the ruptured tubes stops. The analysis indicates that, throughout the entire accident, the core remains covered and cooled without core damage.

The staff concludes that there is reasonable assurance that the unique design features of the AP600 are capable of mitigating the consequences of a multiple tube rupture as specified by SECY-93-087. In the extremely unlikely event of PORV failure to open coincident with a stuck-open MSSV, no core damage will occur, and the total release to the atmosphere would be limited to the initial activity of the RCS. The staff concludes that there is reasonable assurance that the containment bypass as a result of multiple tube rupture poses no undue threat to the public health and safety, and the AP600 design satisfies SECY-93-087.

5.4.3 RCS Piping

The RCS piping includes those sections of RCS hot leg and cold leg piping interconnecting the RV, SGs, and RCPs. It also includes piping connected to the reactor coolant loop piping and primary components. The RCS piping accommodates the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. The piping in the AP600 RCS is equipment Class A and fabricated according to ASME Code, Section III, Class 1 requirements, consistent with the requirements of 10 CFR 50.55a(c)(1). Lines with a 0.97 cm (3/8-inch) or less flow-restricting orifice qualify as AP600 equipment Class B and are designed and fabricated with ASME Code, Section III, Class 2 requirements. Because the AP600 CVS provides sufficient makeup of the reactor coolant in the event of a failure of a small line of 0.97 cm (3/8 inch) or less, Class B classification of small piping exempted from ASME Code, Section III, Class 1 requirements in accordance with the exception permitted in accordance with 10 CFR 50.55a(c)(2)(i).

In Section 5.4.3.2.1 of the SSAR, Westinghouse provides a list of the piping connected to the RCS. The detailed RCS P&ID is shown in Figure 5.1-5 of the SSAR. It includes the pressurizer surge, spray, and auxiliary spray lines; pressurizer safety valves; the ADS with the first three stages connected to the pressurizer and the fourth stage connected to the hot legs; the reactor system head vent line; the accumulator lines; the core makeup tank cold leg balance lines and injection lines; the PRHRHX system; the IRWST injection lines; the RNS pump suction line and discharge line; the CVS purification return lines to the SG channel head and the pressurizer spray; the CVS purification intake line from one RCS cold leg; and the drain, sample, and instrumentation lines. The RCS pressure boundary of these connecting lines start from their respective connections to the RCS and end at the second normally-closed isolation valves or check valves in the respective lines, or the code safety valves, as defined in 10 CFR 50.2. All the RCS-connecting piping that constitutes the RCPB is designed to meet the ASME Code Section III requirements (with one exception discussed below), and is acceptable.

One exception to meeting the ASME Code Section III requirements is in the CVS. As discussed in Section 3.9.6 of the SSAR, the safety-related classification of the CVS ends at the third isolation valve in the purification loop intake line. The remainder of the purification subsystem of the CVS downstream of the third isolation valve inside containment consists of non-safety, QG D components. Because the CVS purification intake line contains three isolation valves (CVS-PL-V001, -V002, -V003) that are maintained open during normal operation, the RCPB extends to the containment isolation valves of the CVS. However, because the portion of the CVS downstream of the three isolation valves can be isolated from the RCS, this portion need not be designed to ASME Class 1 in accordance with the exception criterion of 10 CFR 50.55a(c)(2)(ii). Regulatory Position C of RG 1.26 specifies the portion of RCPB that meets the exception criteria of 10 CFR 50.55a(c)(2) consist of safety-related quality Group B or C components. However, Section 5.2.1.3 of the SSAR describes many design enhancements that have been added to the Class D portion of the CVS, such as use of three isolation valves of Class 1 design in the purification loop intake line, and seismic design of piping in the Class D portion. These design enhancements result in an alternate design that provides an acceptable level of quality and safety. As discussed in Section 5.2.1 of this report, the staff evaluation has found this alternative design to be acceptable.

To minimize the potential for thermal stratification that could increase cyclic stresses and fatigue usage, the pressurizer surge line is specifically designed with various degrees of continuous slope up from the hot leg connection to the pressurizer, as shown in Figure 5.4-4 of the SSAR. The surge line is also instrumented with strap-on resistance temperature detectors at three locations, one on the vertical section of pipe directly under the pressurizer and the other two on the top and bottom of the pipe at the same diameter on a more horizontal section of pipe near the pressurizer, to monitor the temperature for indication of thermal stratification.

In Table 5.4.7 of the SSAR, Westinghouse lists the principal design data of the RCS piping, such as pipe sizes, thickness, and design pressure and temperature of the major RCS loop piping, pressurizer surge line, and other reactor coolant branch lines. All of the RCS piping and branch lines have a design pressure of 17.24 MPa (2485 psig). The loading combinations, stress limits, and analytical methods for the structural evaluation of the RCS piping and supports for design conditions, normal conditions, anticipated transients, and postulated accident conditions are discussed in SSAR Section 3.9.3. The RCS piping construction is subject to a quality assurance program with the required testing specified in SSAR Table 5.4-8, and meeting requirements established by the ASME Code. The staff finds the RCS piping design to be acceptable.

The consequences of the RCS piping breaks, including postulated cold leg double-ended guillotine breaks, are analyzed in Section 15.6 of the SSAR to demonstrate their compliance with the respective acceptance criteria. For those low-pressure systems and components outside the containment with connections directly or indirectly to the RCS, the staff requires that those low-pressure portions be designed with the ultimate rupture strength at least equal to the full RCS operating pressure. This is addressed in generic safety issue GSI 105, "Interfacing

System LOCA for LWR," in Chapter 20 of this report. The staff finds the design of the low-pressure piping to be acceptable.

5.4.4 Main Steamline Flow Restriction

Each SG contains a flow restrictor in its steam outlet nozzle. The flow restrictor consists of seven venturi inserts welded to the SG outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle, with the other six equally spaced around it. The steamline flow restrictor limits the steam flow rate from the secondary system to the choked flow of the venturi in the unlikely event of a break in the main steamline. This flow restriction is needed to perform the following functions:

- limit rapid rise in containment pressure
- limit the reactor cooldown rate within acceptable limits
- reduce thrust forces on the main steamline piping
- limit pressure differentials on internal SG components, particularly the SG tube support plates

The steamline flow restrictor is configured to minimize the unrecovered pressure loss across the restrictor during normal operation. The design data of the flow restrictors are specified in SSAR Table 10.3.2-1. The throat size of each venturi is 0.0186 m^2 (0.2 ft^2). With seven venturis in a flow restrictor, the equivalent throat area of the SG outlet is 0.13 m^2 (1.4 ft^2) . The resultant pressure drop through the restrictor at 100-percent steam design flow rate of 1.91 million kg/hr (4.2E+06 lbm/hr) is approximately 30.6 kPa (4.44 psi).

In the DSER, the staff identified Open Item 5.4.4-1, stating that Westinghouse should confirm that the SG steamline flow restrictor will limit the steamline break flow to no more than the design-basis steam flow, and should also confirm that the maximum SG steam flow rate (or a conservatively selected higher rate) is used in the Chapter 15 design-basis analysis for the main steamline break event. Open Item 5.4.4-1 is closed because the staff has reviewed the safety analysis of the design-basis event of steam system piping failure described in Section 15.1.5 of the SSAR. To limit the maximum steam flow for a break at any location, the analysis uses an effective nozzle flow area of 0.13 m^2 (1.4 ft^2) of the main steamline flow restrictors for each SG, which is considerably less than the main steam piping area. Also, Item 8(b)(ii) in ITAAC Table 2.2.4-4 of the AP600 Certified Design Material requires a verification that the installed flow-limiting orifice within the SG main steamline discharge nozzle does not exceed 0.13 m^2 (1.4 ft^2). This is acceptable to the staff.

5.4.5 Pressurizer

The pressurizer is a vertical, cylindrical vessel having hemispherical top and bottom heads, and containing saturated water and vapor. The pressurizer is connected from its bottom to one of the RCS hot legs through a surge line, which allows continuous coolant volume and pressure adjustments between the RCS and the pressurizer. The pressurizer, with the liquid and vapor maintained in equilibrium under saturated conditions, controls the RCS pressure during steady-state operations and transients. Major components of the pressurizer include the pressurizer spray system, electrical heaters, code safety valves, ADS valves, and the surge line. The pressurizer is the principal component of the RCS pressure control equipment. It also accommodates changes in RCS liquid volume, and limits the changes in RCS pressure as a result of reactor coolant temperature changes during all modes of plant operation. The pressurizer also serves as a convenient source of reactor coolant makeup for minor RCS leakage, and is the initial source of water to keep the RCS full in the event of a small-break LOCA in the RCS piping.

During steady-state operation at 100-percent power, approximately 60 percent of the pressurizer volume is water and 40 percent is steam. Electric immersion heaters in the bottom of the vessel keep the pressurizer contents at saturation temperature. A small continuous spray flow is provided through a manual bypass valve around each power-operated spray valve to minimize the boron concentration difference between the liquid in the pressurizer and the reactor coolant. During transient events, pressure increases, caused by insurge of reactor coolant, are mitigated by the pressurizer spray such that the high pressurizer pressure reactor trip setpoint is not reached. Conversely, during pressure decreases, caused by outsurges of reactor coolant, water-to-steam flashing and automatic heater operation keep the RCS pressure above the low pressurizer pressure reactor trip setpoint. The heaters are also energized on the high water level during insurge to heat the subcooled surge water entering the pressurizer from the reactor coolant loop. In "AP600 Design Change Description Report," dated February 1994, Westinghouse described a design change made to the heater control logic so that the power to the pressurizer heaters are automatically blocked upon actuation of passive injection to the RCS. In the DSER, the staff identified Open Item 5.4.5-1, stating that Westinghouse should revise the SSAR to reflect this design change. Open Item 5.4.5-1 is closed as this design feature is described in the SSAR. Section 7.3.1.2.3 of the SSAR describes the signals that actuate the CMT injection. Each of these CMT actuation signals, except for the high pressurizer water level signal, initiates a block of the pressurizer heaters. This function prevents the heaters from attempting to repressurize the RCS during passive safety injection and, therefore, reduces the potential for SG overfill for a SGTR event. This pressurizer heater trip function is credited as a backup protection in the design-basis analyses of a loss of feedwater event, and a SGTR event described in Sections 15.2.7 and 15.6.3, respectively, of the SSAR. In accordance with the technical specification screen criteria specified in 10 CFR 50.36, the pressurizer heater trip function is specified in TS Table 3.3.2-1, Engineered Safeguards Actuation System Instrumentation, and subject to LCO 3.3.2 and associated surveillance requirements.

The pressurizer safety valves provide overpressure protection of the RCS. This is discussed in Section 5.2.2 of this report. In addition, the pressurizer provides for high point venting of noncondensible gases from the RCS by remote manual operation of the first-stage ADS valves to vent the gas accumulated in the pressurizer following an accident. This is discussed in Section 5.4.12 of this report.

The AP600 pressurizer has an internal volume of 45.3 m^3 (1600 ft^3), which is approximately 60 percent more volume than the pressurizers for current PWRs of similar thermal power level. This increased pressurizer volume provides plant operating flexibility, minimizes challenges to the safety/relief valves, and allows for the removal of PORVs from the AP600 RCS design. Section 5.4.5 of the SSAR provides the design bases on the sizing of the AP600 pressurizer to meet the following conditions without the need for a PORV:

- The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

- The water volume is sufficient to prevent (1) a reactor trip during a step-load increase of 10 percent of full power, with automatic reactor control, and (2) uncovering the heaters following reactor trip and turbine trip, with normal operation of control systems and no failures of nuclear steam supply systems.

- The steam volume is large enough to (1) accommodate the surge resulting from a step load reduction from 100-percent power to house loads without reactor trip, assuming normal operation of control systems, and (2) prevent water relief through the safety valves following a complete loss of load with the high-water level initiating a reactor trip, without steam dump.

- A low pressurizer pressure safeguard actuation ("S") signal will not be activated because of a reactor trip and turbine trip, assuming normal operation of control and makeup systems and no failures of the nuclear steam supply systems.

The pressurizer performance during anticipated operational occurrences and postulated accidents is reviewed as part of the design-basis accident analysis review discussed in Chapter 15 of this report. The results of the analyses demonstrate that with the AP600 pressurizer design, the DNBR limit is met for all anticipated operational occurrences, the RCS pressure is within 110 percent of the RCS design pressure for the pressurization events, and the acceptance criteria of 10 CFR 50.46 are met for LOCAs. The staff finds the pressurizer design to be acceptable.

5.4.6 Automatic Depressurization System Valves

The ADS valves are part of the RCS and interface with the PXS. The ADS is divided into two groups and four depressurization stages, with a total of 20 valves. These stages connect to the RCS at different locations. The first, second, and third stage valves are included as part of the pressurizer safety and relief valve (PSARV) module, which is connected to nozzles on top of the pressurizer. The two groups are on different elevations separated by a steel plate. The first stage ADS valves in each group are two motor-operated 10.2-cm (4-in.) valves in series. The second and third stage ADS valves each have two motor-operated 20.3-cm (8-in.) valves in series. The fourth stage ADS valves are 25.4-cm (10-in.) squib valves arranged in series with normally open, dc-powered, motor-operated valves. The outlets of the first three stages in each group are combined into a common discharge line to the IRWST. This discharge line 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 fourth stage ADS valves connect to the RCS hot legs, and are interlocked so that they cannot be opened until RCS pressure has been substantially reduced.

In the DSER, the staff identified Open Item 5.4.6-1, stating that because Section 5.4.6 of the SSAR, Revision 1, contained information that was inconsistent with the February 15, 1994, design change report, it should be revised. Open Item 5.4.6-1 is closed because Westinghouse has revised the ADS design descriptions in Sections 5.4.6, 6.3, and 7.3 of the SSAR to make them consistent with the design features described in the February 1994 design change report. Section 6.3 of the SSAR discusses the operation of the PXS, and Section 7.3 describes the actuation logic and setpoints for opening various stages of the ADS valves. Opening the ADS valves is necessary for the PXS to function as required to provide emergency core cooling following postulated accident conditions. The first stage valves may also be used to remove noncondensable gases from the steam space of the pressurizer, if necessary, following an accident.

In the DSER, the staff identified Open Item 5.4.6-2, stating that the staff review of the adequacy of the ADS system was incomplete at the time. Open Item 5.4.6-2 is closed as the staff has completed its evaluation. The ADS functional performance (as part of the PXS performance) is evaluated in Chapter 6.3 of this report. The safety analyses of various design-basis accidents are evaluated in Chapter 15 of this report. The analysis results of design-basis accidents such as small-break LOCAs described in Section 15.6.5 of the SSAR demonstrate that, with the ADS design and the passive core cooling system, the acceptance criteria specified in 10 CFR 50.46 are met. Therefore, the ADS design is acceptable.

5.4.7 Normal Residual Heat Removal System

The AP600's normal residual heat removal system (RNS) is a non-safety-related system and is not required to operate to mitigate design-basis events. However, the RNS also performs the following safety-related functions:

- containment isolation of RNS lines penetrating containment using containment isolation valves according to the criteria specified in Section 6.2.3 of the SSAR

- preservation of the RCS pressure boundary integrity using pressure isolation valves according to the criteria specified in Section 5.4.8 of the SSAR

5.4.7.1 RNS Design Bases

The RNS performs the following non-safety-related functions. Their design bases are also described below.

- Shutdown Heat Removal

  The RNS is designed to remove both residual and sensible heat from the core and the RCS during shutdown operations, with the capability to (1) reduce the temperature of the RCS from 176.7 °C (350 °F) to 48.9 °C (120 °F) within 96 hours after shutdown during the second phase of plant cooldown (after the initial RCS cooldown is accomplished by the main steam system (MSS)); and (2) maintain the reactor coolant temperature at or below 48.9 °C (120 °F) for the entire plant shutdown, until the plant is started up again

- Shutdown Purification

  The RNS is designed to provide RCS and refueling cavity purification flow to the CVS during refueling operations, with the purification flow rate consistent with that specified in Table 9.3.6-1 of the SSAR.

- In-Containment Refueling Water Storage Tank Cooling

  The RNS is designed to provide cooling for the IRWST during operation of the PRHRHX or during normal plant operations, when required. The RNS is designed to be manually initiated by the operator. During normal operation, the RNS with both subsystems of RNS pumps and heat exchangers available will limit the IRWST water temperature to not greater than 48.9 °C (120 °F). During extended operation of the PRHRHX, the RNS will limit the IRWST water temperature to less than the boiling temperature.

- Low-Pressure RCS Makeup

  The RNS is designed to be manually initiated by the operator following the actuation of the ADS. The RNS provides low-pressure makeup from the IRWST to the RCS (once the pressure in the RCS falls below the shutoff head of the RNS pumps), and thus provides additional margin for core cooling.

- Post-Accident Recovery

  The RNS is designed to remove heat from the core and the RCS following successful mitigation of an accident by the passive core cooling system. The RNS also provides a flow path for long-term postaccident makeup to the containment inventory.

- Low-Temperature Overpressure Protection

  The RNS is designed to provide LTOP for the RCS during refueling, startup, and shutdown operations to limit the RCS pressure within the limits specified in 10 CFR Part 50, Appendix G.

- Spent Fuel Pool Cooling

  The RNS is designed to have the capability to supplement or take over the cooling of the spent fuel pool when it is not needed for normal shutdown cooling.

5.4.7.2 RNS Design and Components

In Section 5.4.7.2 of the SSAR, Westinghouse describes the AP600 RNS design, including specific design features to address the concerns related to mid-loop operation and interfacing system LOCA, respectively. The RNS consists of two mechanical trains of equipment; each consists of one pump and one heat exchanger. The two trains share a common suction line from the RCS and a common discharge header. The RNS is also comprised of piping, valves, and instrumentation necessary for system operation, as shown in Figure 5.4-7 of the SSAR.

Inside containment, the RNS suction header is connected to an RCS hot leg with a single step-nozzle connection. The suction header is comprised of two parallel lines with two sets of two normally closed, motor-operated isolation valves in series, for single failure consideration. These isolation valves comprise the RCS pressure boundary. The two lines are connected to a common suction header. This suction alignment is for reactor cooling during normal shutdown operation. A single line from the IRWST is connected to the suction header to provide a flow path for low-pressure makeup of the RCS.

Once outside containment, the suction header contains a single, normally closed, motor-operated isolation valve. Downstream of the isolation valve, the header branches into two separate lines, one to each pump. In each branch line is a normally open, manual isolation valve, upstream of the RHR pumps, for pump maintenance.

The discharge of each RHR pump is routed directly to its respective RHR heat exchanger. A mini-flow line, which contains an orifice and is sized for a sufficient pump flow rate when the pressure in the RCS is above the RHR pump shutoff head, is routed from downstream of the heat exchanger to upstream of the pump suction. The outlet of each heat exchanger is routed to the common discharge header, which contains a normally closed, motor-operated isolation valve before penetrating the containment.

Once inside containment, the common discharge header contains a check valve that acts as a containment isolation valve. Downstream of the check valve, the discharge header branches into two lines, routed to the direct vessel injection (DVI) lines. These branch lines each contain two check valves, in series, that comprise the RCS pressure boundary. A line is branched from the common header to the CVS demineralizers for shutdown purification of the RCS. Another line is routed from the discharge header to the IRWST for cooling of the tank.

The RNS contains a single safety/relief valve, located off the RNS suction header inside containment that discharges to the IRWST. This relief valve is utilized for low-temperature overpressure protection of the RCS.

In SSAR Table 3.2-3, Westinghouse provides the safety classification and seismic categories of the RNS components. The portions of the RNS piping and components from the RCS up to and including the outer RNS suction isolation valve or outer RNS discharge check valve constitute the RCPB, and are designed with safety Class A requirements. The RNS RCPB valves include V001A, V001B, V002A, V002B, V015A, V015B, V017A, and V017B. These valves are manufactured to the ASME Code Class I requirements in accordance with Section 5.4.8 of the SSAR. The portions form the RCPB to the containment isolation valves outside the containment are designed with safety Class B requirements. The RNS containment isolation valves include V002A, V002B, V011, V012, V013, V021, V022, V023, and V061. These valves (except for RCPB valves V002A and V002B which are ASME Code Class 1) are manufactured to ASME Code Class 2 requirements. The inside containment portions extending to the containment isolation valves outside containment are designed for full RCS pressure. The system piping and components outside containment, including the pumps, valves, and heat exchangers, are safety Class C, and have a design pressure and temperature such that full RCS pressure is below the ultimate rupture strength of the piping. These design classifications comply with GDC 1 which specifies 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. The whole RNS system, except for the heat exchanger shell vents is designed for seismic Category I for pressure retention. This complies with GDC 2 which specifies the SSCs important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. This also complies with RG 1.29 which specifies that the SSCs that constitute the RCPB, are designated seismic category I and should be designed to withstand the effects of the SSE and remain functional. The staff finds that the RNS design for performing its safety-related functions of containment isolation and preservation of the RCPB integrity to be acceptable.

5.4.7.3 Shutdown Operation Design Features

In SECY-93-087, the staff specifies that passive plants must have a reliable means of maintaining decay heat removal capability during all phases of shutdown activities, including refueling and maintenance. The staff review of the AP600 design with respect to shutdown operations is based on the applicant's systematic assessment of shutdown operation concerns identified in NUREG-1449, "Shutdown and Low-Power Operations at Commercial Nuclear Power Plants in the United States," which encompasses mid-loop operation. This assessment is provided in WCAP-14837, Revision 3, "AP600 Shutdown Evaluation Report." The staff evaluation of the shutdown operation issues is based on WCAP-14837 and is addressed in Section 19.3 of this report. This section describes the RNS design features to address NUREG-1449 and Generic Letter 88-17 regarding mid-loop operation.

5.4.7.3.1 Features Related to Mid-Loop Operation

- Loop Piping Offset

  The levels of the RCS hot legs and cold legs are offset vertically with the hot leg nozzles 0.445 m (17.5 in) below the cold leg nozzles so that the RCS can be drained with the hot leg level remaining much higher than traditional designs for venting of the SGs prior to nozzle dam insertion. Furthermore, this loop piping offset allows a RCP to be replaced without removing a full core.

- Step-Nozzle Connection

  The RNS employs a step-nozzle connection to the RCS hot leg to minimize the likelihood of air ingestion into the RHR pumps during RCS mid-loop operations. The step-nozzle connection substantially lowers the RCS hot leg level at which a vortex occurs in the RHR pump suction line as a result of the lower fluid velocity in the hot leg nozzle.

- No RHR Throttling During Mid-Loop

  The RNS is designed with the pumps having sufficient NPSH for operation at full design flow rate without the need for throttling the RHR control valve to minimize susceptibility to cavitation when the level in the RCS is reduced to a mid-loop level.

- Self-Venting Suction Line

  The RNS pump suction line slopes continuously upward from the pump to the RCS hot leg with no local high points (where air could collect and cause a loss of RHR capability). This self-venting suction line will refill after a pump trip. The pumps can be immediately restarted once an adequate level is reestablished in the hot leg.

- Hot Leg Level Instrumentation

  The AP600 RCS contains level instrumentation in each hot leg with a readout in the MCR. Alarms are also provided to alert the operator when the RCS level is approaching a low level. Additionally, the isolation valves in the RCS drain line are interlocked to close on a low RCS level during shutdown operations.

- Reactor Vessel Outlet Temperature

  Each hot leg is provided with a wide-range thermowell-mounted resistance temperature detector for measurement of reactor coolant fluid temperature in the hot leg when in reduced inventory conditions.

- ADS Valves

  The ADS valves of the first three stages are required to be open to provide a vent path to prevent RCS pressurization whenever the CMTs are blocked during shutdown conditions while the RV upper internals are in place.

5.4.7.3.2 Other Features for Shutdown Operations

The RNS contains instrumentation to monitor and control system performance. System parameters necessary for RNS system operation that are monitored in the MCR include the following instrumentation which also allow mid-loop operations to be performed from the MCR:

- RNS pump flow discharge pressure
- RNS heat exchanger inlet and outlet temperatures
- RNS heat exchanger outlet flow and bypass flow
- RCS wide-range pressure

The staff's evaluation of shutdown operations and AP600 design features to support shutdown operations is based on WCAP-14837 and is provided in Section 19.3 of this report. The staff has concluded that the AP600 design features which support shutdown operations, including those of the RNS, is acceptable.

In the DSER, the staff identified Open Items 5.4.7.10-1 and 5.4.7.10-2, stating that the staff was awaiting Westinghouse's response to RAI 440.53 related to the shutdown risk assessment, and that the shutdown risk issue remained open pending resolution of the technical issues identified in RAI 440.54 through RAI 440.72. Open Items 5.4.7.10-1 and 5.4.7.10-2 are now closed as Westinghouse submitted WCAP-14837 which satisfactorily addressed the staff's shutdown operations issues.

5.4.7.4 Interfacing-Systems LOCA Design Features

In SECY-90-016, as well as SECY-93-087, the staff specifies that ALWR designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to full RCS pressure. SECY-90-016 also specifies guidance for those systems that have not been designed to withstand full RCS pressure.

The AP600 RNS design contains the following features that address the interfacing-systems LOCA issue:

- Increased Design Pressure

  The portions of the RNS from the RCS up to and including the containment isolation valves outside containment are designed to the full RCS operating pressure. The portions of the system downstream of the suction line containment isolation valve and upstream of the discharge line containment isolation valve, including the pumps, valves, flanges, fittings, and heat exchangers, have a design pressure of 6.21 MPa (900 psi), approximately 40 percent of the RCS operating pressure, so that its ultimate rupture strength (URS) is not less than the operating pressure of the RCS. An exception to this is the pump seal which does not meet this criterion. This is discussed in the staff evaluation of the ISLOCA in Chapter 20 of this report.

- Additional RCS Isolation Valve

  The AP600 RNS contains an additional isolation valve in the pump suction line from the RCS. This motor-operated containment isolation valve is designed to full RCS pressure, and provides an additional barrier between the RCS and lower pressure portions of the RNS.

- RNS Relief Valve

  The AP600 RNS relief valve is connected to the RHR pump suction line inside containment to provide LTOP of the RCS. It is connected to the high-pressure portion of the pump suction line; as such, it will reduce the risk of overpressurizing the low-pressure portions of the system.

- Features Preventing Inadvertent Opening of Isolation Valves

  The motor-operated isolation valves connected to the RCS hot leg are interlocked to prevent their opening at RCS pressures above 3.21 MPa (450 psig). These valves are also interlocked to prevent their being opened unless the isolation valve from the IRWST to the RHR pump suction header is closed. In addition, the power to these valves is administratively blocked at the valve motor control center to prevent their inadvertent opening.

- RCS Pressure Indication and High Alarm

  The AP600 RNS contains an instrumentation channel that indicates pressure in each RNS pump suction line. A high pressure alarm is provided in the MCR to alert the operator to a condition of rising RCS pressure that could eventually exceed the design pressure of the RNS.

The staff evaluation of the interfacing system LOCA is addressed in GSI 105, "Interfacing System LOCA at LWRs," in Chapter 20 of this report. The staff found that the RNS design features meet the ISLOCA specifications in SECY-90-016 and SECY-93-087.

5.4.7.5 RNS System Operation and Performance

In Section 5.4.7.4 of the SSAR, Westinghouse provides a general description of the RNS operation for the pertinent phases of plant operation (plant startup, plant cooldown, refueling, accident recovery operations, and spent fuel pool cooling). System operations are controlled and monitored from the MCR, even during mid-loop operations.

For accident recovery operations, the RNS can be employed to provide low-pressure RCS makeup upon actuation of ADS. The staff reviewed the AP600 emergency response guidelines to evaluate a possible system interaction, caused by the RNS operation, which may adversely affect the performance of the passive safety systems. For post LOCA recovery, the AP600 emergency response guidelines instruct the operators to actuate the RNS and align the RNS pumps to take suction from the IRWST and inject into the RCS to provide additional core cooling if the CMT level begins to decrease. Operation in this mode provides additional injection flow to the RCS, thereby providing additional core cooling margin. Because the RNS pumps are aligned to inject into the RCS via the DVI lines, which are also the injection paths of the CMTs and IRWST, these shared connections can result in interactions with the PXS.

An evaluation of the potential for adverse system interactions of the RNS and the PXS is provided in WCAP-14477, Revision 1, "The AP600 Adverse System Interactions Evaluation Report." For a small break LOCA, the operation of the RNS pumps in the injection mode increases the backpressure on the CMT and prevents the CMT from draining to the ADS-4 actuation setpoint, thereby preventing the ADS-4 valves from actuating. Operation of the RNS pumps will refill the RCS and recover the water level in the pressurizer without the need to actuate ADS-4 valves. For a large break LOCA, the capacity of the RNS will not be sufficient to prevent the CMT from draining, and subsequent ADS-4 actuation. Therefore, RNS operation has no adverse impact.

However, because the RNS is aligned to the IRWST, continued long-term operation of the RNS pumps could result in the IRWST draining at a faster rate than if the RNS pumps were not operating. This is not a concern as long as the RNS pumps continue to operate, and therefore provide higher injection rate than the gravity injection from the IRWST or the containment recirculation path. However, if the RNS pumps were to fail, the available gravity head from the IRWST or the containment recirculation path for safety injection could be less than would have been available if the pumps had not operated at all. This situation was analyzed in the long-term cooling safety analysis in SSAR Chapter 15, where a case was analyzed with the assumption that both RNS pumps are started by the operator during the IRWST injection of the transient, and failed at the minimum IRWST level. Therefore, this system interaction has been analyzed and found to be acceptable.

5.4.7.6 Design Evaluation

The staff review of the RNS design is for compliance with the following requirements:

- GDC 1, as it relates to the quality standards of the SSCs important to safety
- GDC 2, as it relates to the seismic design of the SSCs important to safety by withstanding an SSE and remaining functional, with acceptability based on meeting RG 1.29

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

- GDC 5, as it relates to SSCs important to safety being prohibited from being shared among nuclear power units

- GDC 19, as it relates to a control room being provided from which actions can be taken to operate the nuclear power unit safely

- GDC 34, as it relates to the ability of the residual heat removal system to transfer fission product decay heat

The RNS is designed for a single nuclear power unit, and is not designed to be shared between units. The RCPB portion of the RNS is designed as safety Class A, and the containment isolation valves of the RNS are designed as Safety Class B, the remaining portions are designed as safety Class C. The pressure boundary is classified as seismic Category I and is designed to withstand a safe shutdown earthquake for pressure retention. The RNS is operated from the MCR. Also, the high energy piping of the RNS (i.e., the RNS suction and discharge portions that constitute the RCPB) are subject to LBB criteria for protection against dynamic effects. This is identified in Table 3B-1, and Figures 3E-2 and 3E-4 of the SSAR. Therefore, the RNS meets GDCs 1, 2, 4, 5, and 19. Because the RNS is not designed to provide safety-related decay heat removal function for mitigation of design-basis events, the safety-related heat removal function of GDC 34 is complied with by the safety-related PRHRHX. The evaluation of the PRHRHX is discussed in Section 6.3 of this report. In the DSER, the staff identified Open Item 5.4.7.7-1, stating that compliance of the RNS with applicable regulations was under staff review. Therefore, Open Item 5.4.7.7-1 is closed and the staff finds the RNS design to be acceptable.

5.4.7.7 Inspection and Testing Requirements

Proper operation of the RNS is verified through preoperational tests, which include valve inspection and testing, flow testing, and verification of heat removal capability. The inspection and test requirements of the RNS valves are consistent with those identified in Sections 5.2.4 and 6.6 of the SSAR, respectively, for the valves that constitute the RCPB and the valves that isolate the line penetrating containment. In addition, these valves are included in SSAR Table 3.9.16 and are subject to inservice testing. The staff finds proper inspection and test requirements are made for the RNS valves performing safety-related functions of containment isolation and preserving RCPB integrity. The set pressure and the relieving capacity of the relief valve, RNS-V021, which is provided for low-temperature overpressure protection, are verified to be consistent with the values specified in SSAR Table 5.4-17. The relief valve relieving capacity will be certified in accordance with ASME Code Section III, NC-7000. The staff finds this acceptable. The minimum flow rates to meet the functional requirements of cooling the RCS during shutdown operations and low pressure makeup to prevent 4th stage ADS actuation for small break LOCA, respectively, are specified in Table 5.4-14. These shutdown cooling and low pressure makeup flow rates are confirmed through the tests with the RNS pump suction aligned to their respective operations, (i.e., with the suction aligned to the RCS hot leg and the IRWST, respectively). The RNS heat exchanger heat removal capability is specified in SSAR Table 5.4-14, and is verified through the manufacturer's test results and data. The staff finds these tests to confirm the RNS flow and heat transfer capabilities to be acceptable.

5.4.7.8 Regulatory Treatment of the RNS

The RNS is a non-safety-related system that is not required to operate to mitigate design-basis events. Therefore, the RNS is not required to meet safety-related system requirements. However, the RNS is a defense-in-depth system that provides the first line of defense during an accident to prevent unnecessary actuation of passive core cooling systems. Regulatory oversight of the active non-safety systems in passive plant designs is subject to a staff evaluation of the regulatory treatment of non-safety systems (RTNSS). In the DSER, the staff identified Open Item 5.4.7.11-1, stating that the staff was still evaluating the AP600 PRA and RTNSS evaluation. Open Item 5.4.7.11-1 is closed as discussed below. A detailed evaluation of the RTNSS issue is described in Chapter 22 of this report.

In SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," the staff describes the RTNSS process. The goal of the RTNSS process is to provide insights on the importance of non-safety related-systems to the overall safety of the AP600 design and assist in determining what, if any, additional regulatory controls should be applied to RTNSS-identified systems. The RTNSS process involves using both probabilistic and deterministic criteria to (1) determine whether regulatory oversight for certain non-safety-related systems is needed, (2) identify the risk significant SSCs for regulatory oversight, and (3) decide on an appropriate level of regulatory oversight for the various identified SSCs commensurate with their risk importance.

As the important non-safety-related SSCs identified through the RTNSS process do not meet the screening criteria specified in 10 CFR 50.36 for inclusion in the technical specification limiting conditions for operation, the applicant proposed a mechanism to provide for short-term availability control of these systems. Section 16.3 of the SSAR provides short-term administrative availability controls for the RTNSS-identified important non-safety-related SSCs. For each RTNSS-identified SSC, the operability requirements for the required functions and system configurations are specified for various modes of operation, and the required actions and completion times are specified for conditions not meeting the operability requirements. Surveillance frequency requirements are also specified to confirm operability of the SSCs. A commitment is included in the SSAR Section 16.3.2 for the COL applicant referencing the AP600 design to develop and implement procedures consistent with the availability controls. These administrative availability controls will also be included in the AP600 design control document. The staff found the approach acceptable as described in Chapter 22 of this report.

In WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Non-Safety-Related Systems Process," the applicant provided the results of its evaluation on the basis of the RTNSS screening process. The RNS was identified as an important system needed for shutdown decay heat removal to support mid-loop operation with reduced reactor coolant inventory and, therefore, subject to additional regulatory controls. In addition, the RNS also provides a non-safety-related means of injecting the IRWST water into the RCS following ADS actuation to provide margin in the PRA sensitivity studies to mitigate at-power and shutdown events. The administrative short-term availability controls of the RNS functions at various modes of operation are specified in the SSAR Table 16.3-2, Sections 2.1 and 2.2. In addition, the availability controls of the RNS supporting systems such as the CCS, the service water system, and the AC power supplies, are specified in SSAR Table 16.3-2. The staff has reviewed Table 16.3-2, and concluded that proper administrative controls are provided to ensure the short-term availability of the RNS to perform its required functions.

5.4.8 Valves

The design bases, design evaluation, qualification testing, ISI and inservice testing of valves associated with the RCS and RCS-connected systems is collectively discussed in Sections 3.9.3, 3.9.6, 3.10, 5.2.3, 5.2.4, and 6.6 of this report.

5.4.9 Reactor Coolant System Pressure Relief Devices

The AP600 does not have PORVs connected to the pressurizer. Instead, the PSVs provide overpressure protection of the RCS during startup, hot shutdown, and power operation. The relief valve on the suction line of the RNS provides low-temperature overpressure protection. The ADS valves provide a means to depressurize the RCS as part of the PXS. The first stage ADS valves can also be used to vent noncondensable gases following an accident.

5.4.9.1 Pressurizer Safety Valves

The PSVs are of the totally enclosed pop type. There is no loop seal in the piping between the pressurizer and the PSVs to collect the steam condensate. The steam condensate will drain back to the pressurizer, and will not be discharged as a water slug during the initial opening of the valve. Each PSV discharge is directed through a rupture disk, located at the end of the discharge piping, to containment atmosphere. The rupture disk is provided to contain leakage past the valve, and is designed with a substantially lower set pressure than the PSV's set pressure to ensure PSV discharge. A small pipe is connected to the discharge piping and directed to the reactor coolant drain tank to drain away condensed steam leaking past the safety valve. Positive position indication is provided for the PSVs, in accordance with the requirements of 10 CFR 50.34(f)(2)(xi), which requires direct indication of relief and safety valve position (open or closed) be provided in the MCR. Temperatures in the discharge lines are measured, and an indication and a high temperature alarm are provided in the control room for indication of any leakage or relief through the associated valve. The PSVs are designed to prevent RCS pressure from exceeding 110 percent of system design pressure. The design parameters of the PSVs are specified in Table 5.4-17 of the SSAR. The sizing of the PSVs with 3-percent accumulation is addressed in Section 5.2.2 of this report.

In 10 CFR 50.34(f)(2)(x), the NRC requires a test program and associated model development, as well as conducting of tests to qualify RCS relief and safety valves for all fluid conditions expected under operating conditions, transients, and accidents. This has been done through the tests of similar safety valves within the EPRI safety and relief valve test program, which found that the safety valves were adequate for steam flow and water flow, even though water flow is not anticipated through the PSVs. Item II.D.1, "Performance Testing of PWR Safety and Relief Valves," in Chapter 20 of this report addresses the resolution of the PSV testing program. The PSVs are also subjected to preservice and inservice hydrostatic tests, seat leakage tests, operational tests, and inspections. This is done through the inservice testing (IST) specified in Table 3.9-16 of the SSAR, as well as the IST for ASME Code Class 2 and 3 components in Section 6.6 of the SSAR.

5.4.9.2 RNS Relief Valve

The RNS relief valve on the RNS pump suction line is designed for water relief, and has an accumulation of 10-percent of the set pressure. The set pressure (setpoint) is the lower of the values determined on the basis of the RNS design pressure or the RV low temperature pressure limit. The design parameters of the RNS relief valve, including the set pressure and relieving capacity, are specified in Table 5.4-17 of SSAR. The determination of the set pressure and relieving capacity, is discussed in Section 5.2.2 of this report. The lowest permissible lift set pressure is determined by the required NPSH for the RCPs. Position indication for the RNS relief valve is provided in accordance with the requirements of 10 CFR 50.34(f)(2)(xi), which requires direct indication of relief and safety valve position (open or closed) be provided in the MCR. Therefore, this is acceptable.

RCS pressure relief devices are required by 10 CFR 50.34(f)(2)(x) to be subjected to tests to qualify for all fluid conditions expected under operating conditions, transients, and accidents. In the DSER, the staff identified Open Item 5.4.9.2-1, stating that Westinghouse should justify its position stated in Section 5.4.9.4 of the SSAR that the RNS relief valve is not required to be tested. Section 5.4.9.4 of the SSAR has been revised to state that the RNS relief valve is designed for water relief and is not a RCS pressure relief device since it has a set pressure less than RCS design pressure. Therefore, the valve selected for the RNS relief valve is independent from the EPRI safety and relief valve test program. Since the RNS relief valve is not a RCPB valve, and is designed for low-temperature overpressure protection, the staff agrees it need not be included in the EPRI test program for the safety and relief valves. As specified in Table 3.2-3 of the SSAR, the RNS relief valve is an AP600 Class 2 component, and will be designed, manufactured, and tested to ASME Section III, Class 2 requirements. In addition, the RNS relief valve is also subject to IST as specified in Table 3.9-16 of the SSAR for its safety related missions and functions. The staff finds these test requirements for the RNS relief valve to be acceptable. Open Item 5.4.9.2-1 is closed.

5.4.10 RCS Component Supports

The design bases and design evaluation of RCS component supports are described in Sections 3.9.3.3 and 3.12.6 of this report. ISI of RCS components is discussed in Sections 5.2.4 and 6.6 of this report.

5.4.11 Pressurizer Relief Discharge

The AP600 design does not have a pressurizer relief discharge system. The AP600 employs neither power-operated pressurizer relief valves nor a pressurizer relief discharge tank. Some of the functions provided by the pressurizer relief discharge system in previous nuclear power plants are provided by portions of other systems in the AP600.

The staff reviewed AP600 pressurizer relief discharge in accordance with SRP Section 5.4.11, "Pressurizer Relief Tank." The SRP acceptance criteria specify that the design meet GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the protection of safety-related systems from the effects of earthquakes, and GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to a failure of the system resulting in missiles or adverse environmental conditions that could result in damage to safety-related systems or components. Conformance with GDC 2 is on the basis of meeting the guidelines of RG 1.29, "Seismic Design Classification," Positions C.2 and C.3. Position C.2 addresses those portions of SSCs which should be designed and constructed such that an SSE could not cause their failure and result in reduced functioning of any seismic Category I equipment or incapacitating injury to occupants in the MCR. Position C.3 addresses the extension of seismic Category I design requirements to the first seismic restraint beyond the defined boundaries. Conformance with GDC 4 is on the basis of meeting the acceptance criteria of SRP 5.4.11(II), as applicable.

The systems and components for AP600 pressurizer relief discharge are discussed in Sections 5.2.2, 5.4.6, 5.4.9, 5.4.11, 5.4.12, and 6.3 of the SSAR. This equipment is located inside containment and is designed to provide overpressure protection for the RCS during power operation. Two pressurizer safety valves are located on top of the RCS pressurizer. Table 3.2-3 and Section 3.2 of the SSAR state that the pressurizer safety valves are classified as AP600 equipment Class A (ANS safety Class 1), seismic Category I, and ASME Code Class 1. These valves are tested in accordance with requirements of the ASME Code, Section XI.

The pressurizer safety valves are spring loaded, self-actuated by direct fluid pressure, and have backpressure compensation features. They are the totally enclosed pop type, and are designed to reclose and prevent further flow of fluid after normal conditions have been restored. Because loop seals are not installed between the pressurizer and safety valves, steam condensation flows back into the pressurizer instead of forming a water slug that would blow out during initial safety valve actuation. Although the valves are designed for the flow of both steam and water, water is not expected to flow through the valves.

The pressurizer safety valves are sized on the basis of the analysis of a complete loss of steam flow to the turbine with the reactor operating at 102 percent of rated power. In the analysis, no credit is taken for the operation of the pressurizer level control system, pressurizer spray system, rod control system, steam dump system, steamline PORVs, or direct reactor trip on turbine trip. The feedwater system is also assumed to be lost. Under these conditions, the total pressurizer safety valve capacity is at least as large as the maximum surge rate into the pressurizer during this postulated event. This results in a safety valve capacity that prevents system pressure from exceeding 110 percent of system design pressure.

Pressurizer safety valve discharge is routed through a rupture disk to the containment atmosphere. The rupture disk is designed to contain any leakage past the safety valves and has a pressure rating much lower than the set pressure of the safety valve. Leakage past the safety valve during normal operation is collected and routed to the RCDT. Each safety valve discharge line includes a temperature indicator and alarm in the MCR.

Pressurizer safety valve discharge is directed away from SSCs inside containment, which could be damaged by the discharge. The containment pressure resulting from a safety valve discharge is significantly less than the containment design pressure (the containment design pressure is determined by LOCA considerations), and the resulting heat load is well within the capacity of the normal fan coolers and the PCS.

Automatic Depressurization System

The ADS is shown in Figure 5.1-5 (sheet 1 of 3 and sheet 2 of 3) of the SSAR. The system is not a pressure relief system. It is designed to depressurize the RCS under emergency plant operations and to vent noncondensable gases from the pressurizer steam space following an accident. Operation of the ADS valves is required for the PXS to function following postulated accident conditions. The first stage valves are used to vent noncondensable gases from the pressurizer steam space. In Table 3.2-3 and Section 3.2 of the SSAR, Westinghouse states that the valves are classified as AP600 equipment Class A (ANS safety Class 1), seismic Category I, and ASME Code Class 1. The valves are tested in accordance with requirements of ASME Code, Section XI.

The ADS consists of twenty valves divided into two divisions, and further divided into four depressurization stages. These valves are connected to the RCS at three locations. The two divisions of the first-, second-, and third- stage valves are connected to the top of the pressurizer while one division of the fourth-stage valves is connected to the hot leg of each RCS loop and vents directly to a SG compartment. The fourth-stage valves are designed such that they cannot open against full system pressure.

The discharge from the first-, second-, and third- stage ADS valves is routed to the IRWST by way of two depressurization spargers (one per division). The spargers are classified as AP600 equipment Class C (ANS safety Class 3) and seismic Category I, and are designed to distribute steam inside the IRWST to ensure effective steam condensation. The IRWST also receives discharges from the relief valve of the RNS, and steam and gas discharges from the PRHR high point vents and the RV high point vents (discussed in Section 5.4.12 of the SSAR).

As described in Sections 5.4.6. and 6.3 of the SSAR, the ADS, consisting of four stages, is part of the RCS and interfaces with the PXS. Two valves are located in each discharge path to prevent inadvertent ADS valve discharges should a valve accidentally open. Diverse and redundant features are provided in the ADS control system to ensure that valves do not inadvertently open. Following ADS actuation, steam can condense in the discharge line creating a vacuum condition that could result in a reverse flow of water from the IRWST. To prevent this, vacuum breakers are provided in the discharge lines to limit the pressure drop that may occur following ADS actuation and thus prevent backflow.

In-Containment Refueling Water Storage Tank

The in-containment refueling water storage tank (IRWST) is a stainless steel-lined compartment inside containment that is integrated into the containment structure underneath the operating deck. The tank is classified as AP600 equipment Class C (ANS safety Class 3) and seismic Category I. The tank is designed to absorb the pressure increase and heat input from the discharge from a first-stage ADS valve (including the water seal, steam, and gases) when venting noncondensable gases from the pressurizer following an accident.

As stated above, the first-, second-, and third-stage ADS valves are divided into two divisions that connect to two separate spargers below the water level of the IRWST. The discharge from the spargers does not result in pressures in excess of the design pressure of the IRWST during a first stage ADS valve discharge of steam, water, and noncondensable gases during an accident. In addition, the IRWST has covered vents that provide tank overpressure protection.

The IRWST does not use a cover gas or a spray system, and does not have a connection to the waste gas processing system. The IRWST is cooled by the RNS and includes level and temperature indicators and alarms.

Conformance with GDC 2 is on the basis of meeting the guidelines of Positions C.2 and C.3 of RG 1.29. Position C.2 states that those portions of the system whose function is not required, but whose failure could reduce the functioning of any seismic Category I system or could result in incapacitating the occupants of the MCR, should be designed and constructed so that an SSE would not cause this failure. As stated above, the pressurizer relief discharge components are seismic Category I, and discharge is directed away from any safety-related SSCs inside containment, that could be damaged by the discharge. Also, the discharges from the ADS valves are routed to the IRWST, which is designed to accommodate these discharges and therefore will not pose a hazard to nearby safety-related SSCs. These processes occur inside containment and therefore do not affect the MCR. In addition, Westinghouse has stated in Appendix 1A of the SSAR that the AP600 design will conform to the guidelines of this position.

Position C.3 states that seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries. Those portions of the system that form interfaces between seismic Category I and non-seismic Category I features should be designed to seismic Category I requirements. Westinghouse has stated in Appendix 1A of the SSAR that the system design will conform to the guidelines of this position.

The pressurizer safety valve discharge is directed away from safety-related SSCs inside containment, that could be damaged by the discharge. In addition, discharges from the ADS valves are routed to the IRWST, which is designed to accommodate these discharges. On the basis of this information, the staff concludes that the pressurizer relief discharge equipment is adequately protected from the dynamic effects associated with failed SSCs inside containment, and also will not pose a hazard to other safety-related SSCs inside containment should any of the pressurizer relief discharge equipment fail.

During the staff's review of systems and components used for AP600 pressurizer relief discharge, several issues were identified for resolution and documented in the DSER. These issues included the following:

- Provide safety valve relief capacity and ADS discharge capacity (Open Item 5.4.11.4-1). Westinghouse addressed the relief capacity of the safety valves in Section 5.2.2 of the SSAR.

- Identify the worst-case load on pressurizer relief discharge equipment, including the IRWST (Open Item 5.4.11.4-2). Westinghouse addressed worse-case loading on applicable pressurizer relief discharge equipment, including the IRWST, in Section 3.8.3 of the SSAR.

- Clarify the scope of the pressurizer relief discharge equipment (Open Item 5.4.11.4-3). Westinghouse provided information clarifying the scope of components and systems included in pressurizer relief discharge functions. The AP600 design does not employ a pressure relief discharge system.

- Provide the inspection and testing requirements for the pressurizer relief discharge equipment (Open Item 5.4.11.4-4). Westinghouse addressed inspection and testing requirements for pressurizer relief discharge equipment in Section 5.4.11 of the SSAR.

- Provide additional information on the instrumentation associated with the pressurizer relief discharge equipment (Open Item 5.4.11.4-5). Westinghouse provided additional information on the instrumentation associated with the pressurizer relief discharge equipment in Section 5.4.11 of the SSAR.

- Address Bulletin 80-05, regarding the susceptibility of the IRWST to vacuum conditions resulting from the cooling of hot water in the tank (Open Item 5.4.11.4-6). With regard to the susceptibility of the IRWST to vacuum conditions resulting from the cooling of hot water in the tank (Bulletin 80-05 issue), Westinghouse described in Section 6.3 of the SSAR an IRWST design feature to prevent tank collapse due to vacuum conditions.

- Identify seismic and safety classes associated with pressurizer relief discharge equipment (Open Item 5.4.11.4-7). Westinghouse identified seismic and safety classes associated with pressurizer relief discharge equipment.

- Provide information regarding divisional separation and isolation of the redundant, safety-related portions of the pressurizer relief discharge equipment (Open Item 5.4.11.4-8). Westinghouse provided information regarding divisional separation and isolation of the redundant, safety-related portions of the pressurizer relief discharge equipment, as applicable. In Section 6.3 of the SSAR, Westinghouse provides information concerning separation of the ADS.

On the basis of the information discussed above, DSER Open Items 5.4.11.4-1 through 5.4.11.4-8 are closed.

Considering the evaluation of information and commitments provided by Westinghouse in the SSAR, the staff concludes that equipment used for AP600 pressurizer relief discharge meets the requirements of GDC 2 on the basis of conformance with Positions C.2 and C.3 of RG 1.29, and also meets the requirements of GDC 4 on the basis of the protection of safety-related SSCs from effects associated with a failure of the equipment. Therefore, the staff concludes that systems and components used for AP600 pressurizer relief discharge conform to the appropriate guidelines of SRP 5.4.11, and are acceptable.

5.4.12 Reactor Coolant System High Point Vents

RCS high-point vents are provided to exhaust noncondensable gases accumulated in the primary system that could inhibit natural circulation core cooling. 10 CFR 50.34(f)(2)(vi) requires that the RCS be provided with high-point vents to maintain adequate core cooling, and that systems to achieve this capability be capable of being operated from the MCR and that their operation not lead to an unacceptable increase in the probability of a LOCA or an unacceptable challenge to containment integrity.

In the AP600 design, noncondensable gases from the RCS are vented using either a reactor head vent or, following an accident, the first-stage valves of the ADS connected to the pressurizer. In addition, the PRHRHX piping and the CMT inlet piping in the PXS also include a high-point vent and are, therefore, in compliance with 50.34(f)(2)(vi).

The review of the AP600 RCS high-point vent design was performed in accordance with Section 5.4.12 of the SRP as discussed below.

5.4.12.1 Reactor Vessel Head Vent System

The reactor vessel head vent system (RVHVS) is designed to remove noncondensable gases or steam from the RCS, with a capacity to vent a volume of hydrogen at system pressure and temperature almost equivalent to one-half of the RCS volume in one hour. The primary function of the RVHVS is for use during plant startup to properly fill the RCS and vessel head. The RVHVS valves also provides an emergency letdown path with a letdown flow rate within the capabilities of the normal makeup system to prevent pressurizer overfill following long-term loss of heat sink events.

The RVHVS consists of two parallel flow paths. Each contains two redundant, 2.54 cm (1 in) open/close, solenoid-operated isolation valves in series, and a flow-limiting orifice downstream. The system discharges to the IRWST.

The solenoid-operated isolation valves are fail-closed, normally closed valves, powered by the safety-related Class 1E dc and uninterruptible power supply (UPS) system. The RVHVS is operated from the MCR, which has individual positive valve position indication and alarm. These valves are included in the AP600 operability program with the IST requirements specified in Table 3.9-16 of SSAR, and are qualified to IEEE-323, IEEE-344, and IEEE-382.

The RVHVS is designed so that a single failure of the remotely operated vent valves, power supply, or control system does not prevent isolation of the vent path. The two redundant isolation valves in series minimize the possibility of RCPB leakage, and ensure that the failure of any one valve does not inadvertently open a vent path.

The flow-limiting orifices limit the flow rate from the head vent path. Acceptance criteria II.5 in Section 5.4.12 of the SRP specifies that the size of the vent line should be kept smaller than the size corresponding to the definition of a LOCA to avoid unnecessary challenges to the emergency core cooling system. Although the size of the vent pipe of 2.54 cm (1 in) is larger than the size corresponding to the definition of a LOCA, the use of the orifices to restrict the flow rate of the head vent to within the capabilities of the normal makeup system allows the AP600 to meet the intent of this criterion.

In the event of a break of the RVHVS line, it would result in a small break LOCA of not greater than 2.54 cm (1 in) diameter. Such a break is similar to the hot leg break LOCA analyzed in Section 15.6.5 of the SSAR. The analysis results indicating no core uncovery apply to a RVHVS line break.

The acceptance criteria of Section 5.4.12 of the SRP specifies that procedures should be developed for use of the vent paths to remove gases that may inhibit core cooling from the U-tubes of the SGs; and that the procedures to operate the vent system should consider when venting is needed, and when it is not needed, with consideration of a variety of initial conditions, operator actions, and necessary instrumentation. The SG tube venting procedures are described in Westinghouse's response to RAI 440.142 and RAI 440.144. In the DSER, the staff identified Open item 5.4.12.4-1, stating that these responses were under staff review. As discussed below, the staff has completed its review. Therefore, Open Item 5.4.12.4-1 is closed.

The primary function of the RVHVS is for use during plant startup to properly fill the RCS and vessel head. During plant startup operations when the RV head is in place and the RCS is filled water solid, the air in the RCS is vented through repeated procedures of (1) starting a RCP in each SG for a short time with the high-point vents closed to allow collection of air in the RCS high points, and (2) opening the vents to allow air trapped in the high points to be vented. During an accident, the AP600 design relies on the passive safety-related systems such as the PRHRHX to provide the safety-related function of core cooling, and therefore does not require the SG U-tubes to be vented to provide coolability of the core. However, the RVHVS is used under loss of heat sink events where the pressurizer level can increase and eventually become water solid following long-term operation of the CMTs. To avoid this occurrence, the functional restoration guidelines for high pressurizer level in the AP600 Emergency Response Guidelines requires that the RV vent flow be established to provide a bleed path in response to high pressurizer level conditions to reduce the RCS inventory and prevent pressurizer overfill. In this case, the operator uses pressurizer level as the primary indication to control operation of the RV head vent.

The RV head vent system consists of safety-grade equipment. The piping and equipment from the vessel head vent up to and including the second solenoid valve constitute the RCPB, and are designed and fabricated to ASME Code Class 1 requirements. The remainder of the piping and equipment are design and fabricated in accordance with ASME Code Class 3 requirements. The piping stresses meet the requirements of ASME Code, Section III, NC-3600, with a design temperature of 343.3 °C (650 °F) and a design pressure of 17.23 MPa (2485 psig). The RVHVS can be operated from the control room or the remote shutdown workstation. Each solenoid-operated isolation vent valve has a position sensor with indication in the control room. Inservice inspection and testing of the RVHVS is in accordance with Section 3.9.6 of the SSAR for valves and Section 5.2.4 of the SSAR for ASME Code, Class 1 components that are part of the RCPB. The RVHVS meets the acceptance criteria specified in Section II or 5.4.12 of the SRP, and is, therefore, acceptable. The resolution of TMI Action Item II.B.1, RCS High-Point Vent, is addressed in Chapter 20 of this report.

5.4.12.2 ADS First-Stage Valves

As discussed in Section 5.4.6 above, the first-stage valves of the ADS provide the capability to remove noncondensable gases from the pressurizer steam space following an accident. Gas accumulations are removed by remote manual operation of the first-stage ADS valves. The discharge of the ADS valves is directed to the IRWST.

The ADS is primarily designed to function as a part of the PXS. The ADS piping up to and including the second isolation valve in series also constitutes the RCPB, and both the piping and valves are designed, constructed, and inspected to ASME Code Class 1 and seismic Category I requirements. The ADS valves are active valves required to provide safe shutdown or to mitigate the consequences of postulated accidents. However, venting of noncondensable gases from the pressurizer steam space is not required to provide safety-related core cooling following a postulated accident. Therefore, the acceptance guidelines of the SRP Section 5.4.12 do not apply to the ADS.

5.4.12.3 Passive RHR Heat Exchanger and Core Makeup Tank High-Point Vents

The PRHRHX inlet piping and the CMT pressure balance line piping in the PXS include high-point vents that provide the capability for removing and preventing the accumulation of noncondensable gases that could interfere with heat exchanger or CMT operation. These gases are normally expected to accumulate when the RCS is refilled and pressurized following refueling. Any noncondensable gases that collect in this high point can be manually vented. The discharge of the PRHRHX high-point vent is directed to the IRWST, and the discharge of the CMT high-point vent is directed to the RCDT.

These high-point vent lines contain two manual isolation valves in series, so that a single failure of either valve to reseat following venting operation does not prevent isolation of the flow path. Each vent line also contains a flow-restricting orifice such that the break flow is within the makeup capability of the CVS and, therefore, would not normally require actuation of the passive safety systems.

5.4.13 Core Makeup Tank

There are two CMTs in the AP600 design as part of the passive core cooling system (PXS). In the CMTs, cold borated water, under system pressure, is stored to provide high-pressure reactor coolant makeup and boration for LOCA and for non-LOCA events, when the normal makeup system is unavailable or insufficient. Section 6.3 of the SSAR describes the operation of the CMTs in the PXS and the connections to the CMTs.

Several changes have been made to the CMT from the original SSAR as described in the "AP600 Design Change Description Reports," dated February 15, 1994, and June 30, 1994. These changes include the (1) installation of a diffuser to the inlet of each of the CMTs, and (2) the elimination of pressurizer-CMT pressure balance lines.

In the DSER, the staff identified Open Item 5.4.13-1, stating that Westinghouse should revise the SSAR to reflect these design changes. Open Item 5.4.13-1 is closed as Sections 5.4.13 and 6.3 of the SSAR have been revised to be consistent with the design changes described above.

5.4.13.1 Design Description

The CMT is a low-alloy steel vessel with a minimum free internal volume of 56.6 m^3 (2000 ft^3), and is supported on columns. The CMT injection line connects from one nozzle on the lower head to the RV DVI piping. The discharge line contains two normally closed, fail-open, parallel isolation valves, and two check valves in series. The CMT pressure balance line connects from the top nozzle in the center of the upper head to one of the RCS cold legs. The pressure balance line with the open flow path to the cold leg maintains system pressure. The top nozzle incorporates a diffuser inside the tank. The bottom of the diffuser, which has the same diameter and thickness as the connecting piping, is plugged and holes are drilled in the side to force the steam flow to turn 90 degrees, which limits the steam penetration into the coolant in the CMT. The diffuser is designed to reduce steam and hot water velocities entering the CMT, thereby minimizing potential water hammer and reducing the amount of mixing that occurs during initial CMT operation. Two sample lines in the upper and lower head, respectively, are provided for sampling the solution in the CMT. A fill connection is provided for makeup water from the CVS.

5.4.13.2 Design Bases

The CMT is a part of the RCPB and AP600 Class A equipment, and is designed and fabricated according to ASME Code, Section III, Class 1 component requirements. Materials of construction are specified to minimize corrosion/erosion and to provide compatibility with the operating environment, including the expected radiation level. The SSAR states, and the staff agrees, that erosion is not an issue because there is normally no flow in the CMT. Those portions of the CMT in contact with reactor coolant are fabricated from or clad with stainless steel. Contamination of stainless steel and nickel-chromium-iron alloys by copper, low-melting-temperature alloys, mercury, and lead is prohibited. The material selection and water chemistry specification, and test and inspections of CMT are discussed in Sections 5.2.3 and 5.2.4, respectively, of this report.

5.4.13.3 Design Evaluation

The loading combinations, stress limits, and analytical methods for the structural evaluation of the CMT for various plant conditions are discussed in SSAR Section 3.9.3. The requirements for dynamic testing and analysis are discussed in Section 3.9.2. The transients used to evaluate the CMT are founded on the system design transients described in SSAR Section 3.9.1.1. In addition to normal RCS transients, the evaluation of component cyclic fatigue of the CMT also assumes 30 occurrences in the plant 60-year lifetime where a small leak draws in hot RCS fluid, and 10 occurrences of increasing containment temperature above normal operating range.

The mechanical component design evaluation with respect to the RCS design transients; requirements for dynamic testing and analysis; and loading combinations; stress limits; and analytical methods for structure evaluation; are discussed in SSAR Sections 3.9.1, 3.9.2, and 3.9.3, respectively. The staff evaluation of these sections are discussed in their respective sections of this report.

The functional performance of the CMTs is evaluated in Chapter 6.3 of this report, as part of the PXS performance, as well as the safety analyses of various design basis transients and accidents in Chapter 15 of this report, to demonstrate its capability to comply with respective acceptance criteria. In addition, various separate effects and integral system tests were performed by Westinghouse to study thermal-hydraulic behavior and the phenomena of the PXS and components, and to validate the codes used for the design basis analysis of transients and accidents. In the DSER, the staff identified its review of the adequacy of the CMT as Open Item 5.4.13-2, pending review of outstanding RAI responses, computer codes, design-basis safety analyses, and information from the ongoing test program. The staff has finished its review. The results of the staff's review are described in Section 6.3 and Chapters 15, and 21 of this report. On the basis of these evaluations the staff concludes that the CMT design meets the guidelines of SRP 6.3 and GDCs 2, 4, 5, 17, 36, and 37, and the PXS as a whole meets GDCs 27, 34, and 35. Therefore the CMT design is acceptable, and Open Item 5.4.13-2 is closed.

5.4.14 Passive Residual Heat Removal Heat Exchanger

The passive residual heat removal heat exchanger (PRHRHX) is part of the passive core cooling system (PXS). Its function is to remove core decay heat for any postulated non-LOCA event where a loss of cooling capability via the SGs occurs. Section 6.3 of this report discusses the operation of the PRHRHX in the PXS.

5.4.14.1 Design Description

The PRHRHX consists of a top and lower tubesheet mounted through the wall of the IRWST. A series of 1.9 cm (0.75 in) outer diameter C-shaped tubes connect to the tubesheets, with the top of the tubes located several feet below the IRWST water surface. An inlet channel head mounted to the top tube sheet is connected through piping to one of the RCS hot legs. An outlet channel head mounted to the bottom tube sheet is connected through piping to the SG cold-side channel head. The primary coolant passes through the tubes, transferring decay heat to the IRWST water. Sufficient thermal driving head is generated in the process to maintain natural circulation flow through the heat exchanger. The design minimizes the diameter of the tubesheets and allows ample flow area between the tubes in the IRWST.

The horizontal lengths of the tubes and lateral support spacing in the vertical section allow for the potential temperature difference between the tubes at both cold and hot conditions. The PRHRHX is welded to the IRWST. The tubes are supported in the IRWST interior with a frame structure. The top of the structure supports a cover that traps and condenses steam during initial activation of the PRHRHX, and helps to minimize the amount of humidity in containment.

Several design changes have been made to the PRHRHX from the original SSAR as described in the "AP600 Design Change Description Report," dated February 15, 1994. These changes include (1) a revision of the PRHRHX actuation logic, and (2) a change in the arrangement of the heat exchanger motor operated valves.

In the DSER, the staff identified Open Item 5.4.14.4-1, stating that Westinghouse should revise the SSAR to include the PRHR design changes discussed above. Westinghouse has revised Sections 5.4.14, 6.3, and 7.3 of the SSAR to be consistent with the design changes described above. Therefore, Open Item 5.4.14.4-1 is closed.

5.4.14.2 Design Bases

The PRHRHX, in conjunction with the PCS, is designed to be able to automatically remove core decay heat for an unlimited period of time. This capability requires a closed-loop mode of operation where the condensate from steam generated in the IRWST is returned to the tank. If no condensate is returned, the PRHRHX provides decay heat removal for at least 72 hours. The PRHRHX and the IRWST are designed to delay significant steam release to the containment for at least one hour. The PRHRHX will keep the reactor coolant subcooled and prevent water relief from the pressurizer. In addition, the PRHRHX will cool the RCS to 204.4 °C (400 °F) in 72 hours, with RCPs operating or, if required, in the natural circulation mode, so that the RCS can be depressurized to reduce stress levels in the system.

The PRHRHX is designed to withstand the design environment of 17.24 MPa (2500 psia) and 343.3 °C (650 °F) for 60 years. The PRHRHX is part of the RCPB, is AP600 class A equipment, and is designed and fabricated according to the ASME Code, Section III, as a Class 1 component. Material specifications, compatibility with the operating environment, including the expected radiation level, as well as the fabrication and processing of the stainless steel for the PRHRHX as the RCPB are discussed in Section 5.2.3 of SSAR, with the staff evaluation provided in the Section 5.2.3 of this report. Section 5.2.4 of the SSAR discusses the ISI and testing of Class 1 components, which are applicable to the PRHRHX.

5.4.14.3 Design Evaluation

The loading combination, stress limits, and analytical methods for the evaluation of structural integrity of the PRHRHX, and the transients used to evaluate the PRHRHX under various plant conditions, are discussed in Sections 3.9.1 through 3.9.3 of SSAR. During normal plant operation, the PRHRHX, without flow through it, is pressurized to the RCS hot leg pressure at the IRWST temperature. Operation of the PRHRHX is evaluated using Service Levels B, C, and D plant conditions, as described in SSAR Section 3.9.1.1. In addition to loads resulting from normal RCS transients and the PRHRHX operation, the evaluation also considers hydraulic loads due to discharge of steam from the ADS valves into the sparger in the IRWST. Seismic, LOCA, sparger activation, and flow-induced vibration loads are derived using dynamic models of the PRHRHX. The dynamic analysis considers the hydraulic interaction between the coolant and system structural elements. The evaluation of component cyclic fatigue also assumes two additional Service Level B transients that affect only the PRHRHX:

- 30 occurrences in the plant 60-year lifetime where a small leak in the manway cover draws in hot RCS fluid

- 10 occurrences of increasing IRWST temperature as a result of an event that activates passive core cooling

The staff evaluation of the mechanical component design with respect to the design transients; requirements for dynamic testing and analysis; and loading combinations, stress limits, and analytical methods for structure evaluation; are discussed in Sections 3.9.1, 3.9.2, and 3.9.3, respectively, of this report.

The PRHRHX functional performance is evaluated in Chapter 6.3 of this report, as part of the PXS performance. The safety analyses of various design-basis transients and accidents is presented in Chapter 15 of this report to demonstrate the PXS capability to comply with applicable acceptance criteria. In addition, various separate effects and integral system tests were performed by Westinghouse to study thermal-hydraulic behavior and phenomena of the PXS and components (including the PRHRHX), and to validate the codes used for the design-basis analysis of transients and accidents. In the DSER, the staff identified Open Item 5.4.14.4-2, stating that the staff's review of the adequacy of the PRHRHX, as a part of the overall operation of the PXS, was an open item pending review of outstanding RAI responses, computer codes, design-basis safety analyses, and information from the ongoing test program. The results of the staff's review of these areas are provided in Section 6.3 and Chapters 15, and 21 of this report. On the basis of these evaluations, the staff concludes that the PRHRHX design meets the guidelines of SRP 6.3 and GDCs 2, 4, 5, 17, 34, 36, and 37. Therefore, the PRHRHX design is acceptable and Open Item 5.4.14.4-2 is closed.

6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

In Section 6.1 of the AP600 Standard Safety Analysis Report (SSAR), Westinghouse provides the AP600 design requirements for engineered safety features (ESFs) materials.

6.1.1 Structural Materials

General Design Criterion (GDC) 1 of Appendix A to 10 CFR Part 50 and 10 CFR 50.55a(a)(1) require that structures, systems, and components (SSCs) important to safety shall 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. The NRC staff reviewed the structural materials of the ESFs to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met as they relate to quality standards for the design, fabrication, erection and testing of ESF components and the identification of applicable codes and standards

GDC 4 requires that structures, systems, and components 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, including loss-of-coolant accidents (LOCAs). The staff reviewed the ESF structural materials to ensure that the relevant requirements of GDC 4 have been met as they relate to the compatibility of structures, systems and components with the various environmental conditions.

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. The staff reviewed the ESF structural materials to ensure that they meet the relevant requirements of GDC 14 as they relate to the design, fabrication, and testing of the RCPB so that an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture is achieved.

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. The staff reviewed the ESF structural materials to ensure that the relevant requirements of GDC 31 have been met as they relate to an extremely low probability of rapidly propagating fracture or gross rupture of the RCPB.

GDC 35 requires, in general, that an emergency core cooling system (ECCS) with abundant capacity be provided. Included within GDC 35 is the requirement that, during activation of the system, clad-metal water reaction is limited to negligible amounts. The staff reviewed the ESF structural materials to ensure that the requirements of GDC 35 have been met as they relate to assurance that the clad-metal water reaction is 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 as they relate to the control of hydrogen concentration in the containment atmosphere following postulated accidents, to ensure that the containment integrity is maintained.

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.

The engineered safety features are defined in Chapter 6 of the AP600 SSAR as consisting of the containment vessel, the passive containment cooling system (PCS), the containment isolation system (CIS), the containment hydrogen control system, the passive core cooling system (PXS), the main control room emergency habitability system, and the fission product system. The containment vessel is a free-standing cylindrical steel vessel with ellipsoidal upper and lower heads. The PCS makes use of the containment vessel and the surrounding concrete shield building, and includes among its component parts a water storage tank, a water distribution system, an air baffle, and an air inlet and air exhaust. The CIS consists of the numerous pipes, valves, and actuators that isolate the containment, including lines that penetrate the containment and are part of the RCPB. The PXS comprises two core makeup tanks, two accumulators, the in-containment refueling water storage tank, the passive residual heat removal heat exchanger, the pH adjustment baskets, and associated piping, valves, instrumentation, and other related equipment. More detailed descriptions of these and the other systems are contained in later subsections of this Chapter.

The components of the ESF used in pressure-retaining situations are fabricated primarily from austenitic stainless steels or other corrosion-resistant material, such as nickel-chromium-iron (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. These are described in Section 6.1.2 of the SSAR. Valve seating surfaces are hardfaced to prevent failure and minimize wear.

Information on the structural materials used in the fabrication of the various ESFs is provided in Section 6.1.1 of the SSAR. The staff reviewed this information in accordance with Section 6.1.1 of the Standard Review Plan (SRP). In the course of its review, the staff transmitted to Westinghouse requests for additional information (RAIs) concerning the ESF structural materials, and received from Westinghouse responses to these RAIs. In addition, several discussions were held between staff and Westinghouse to help clarify and resolve outstanding issues.

A listing of the pressure-retaining materials of the ESF is provided in Table 6.1-1 of the SSAR. The table makes use of cross-referencing other sections in the SSAR for the materials of certain components. The staff pointed out in the DSER that, in some of the cross-referencing instances, the cited sections did not contain the information needed. The staff requested that Westinghouse revise the SSAR to provide this information and to identify the materials used by specification, type, grade and heat treatment. This was Open Item 6.1.1-1. The staff also requested that Westinghouse identify in the SSAR the specific weld metals used in fabricating the components of the ESF by specification, type, grade, and so forth. This was Open Item 6.1.1-2. Westinghouse subsequently revised the SSAR to correct the omissions and to provide specifications for all the ESF structural materials and the weld metals used in fabricating ESF components. The staff review of the materials specifications for the pressure-retaining components finds them in accordance with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. On this basis, Open Items 6.1.1-1 and 6.1.1-2 are closed.

Originally, the SSAR indicated that Ni-Cr-Fe alloy material would be used in fabricating some of the ESF components. The staff requested that Westinghouse identify in the SSAR these nickel-based alloys, together with their specification, type, grade, and heat treatment. This was Open Item 6.1.1-3. The staff also asked Westinghouse to justify each application of a nickel-based alloy and the weld metals for the AP600 (except for the reactor coolant pump flywheel enclosure, discussed in subsection 5.4.1.4) and to address the reason(s) for the choice of one nickel-based alloy over others. This was Open Item 6.1.1-4. Westinghouse subsequently revised the SSAR to state that the use of Ni-Cr-Fe alloy as a structural material in the ESF will be limited to alloy 690. As indicated in the previous paragraph, the revised SSAR provides appropriate specifications for the material. The decision to use alloy 690 was on the basis of its superior 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 other nickel-based alloys, such as Alloy 600. Alloy 600 is the only other nickel-based alloy to be found in the ESF, but its use is limited to cladding or buttering applications. Open Items 6.1.1-3 and 6.1.1-4 are closed.

In its SSAR and in a response to RAI 252.77, Westinghouse indicated that cobalt-based alloys are specified for various ESF applications but that efforts are being made to replace these materials with cobalt-free or low-cobalt content alloys. No definitive statements were made regarding the specific applications where such substitutions would take place or the extent to which cobalt had been eliminated from the ESF design. The staff requested that Westinghouse indicate in the SSAR the base materials and/or the hardfacing materials and processes that are to be used in lieu of the cobalt-based alloys. The staff asked Westinghouse to describe the test programs in place to identify these materials, the results of such programs and the data that ensure a 60-year design life. This was Open Item 6.1.1-5. Westinghouse revised the SSAR to state that hardfacing material in contact with the reactor coolant will be a qualified low- or zero-cobalt alloy, equivalent to Stellite 6. The materials will be qualified through wear and corrosion testing in nuclear industry programs, but results are not yet available. No specific alternative materials have been identified so the staff cannot assess the performance of such materials at this time. However, the reduction or elimination of cobalt in the ESF is associated with as low as is reasonable achievable (ALARA) considerations and has no direct safety implications for the certified design of the AP600. The staff finds that the selection of cobalt-based alloys for wear-resistant applications in the baseline ESF systems design is acceptable on the basis of the adequate performance of such materials in similar applications in current nuclear power plants. With respect to the substitution of low-cobalt or cobalt-free materials for these materials, such action would be prudent for improving the ALARA status of the AP600. However, at this time, no qualified materials have been identified in the SSAR and thus it is not possible to approve any for design certification. Open Item 6.1.1-5 is considered closed.

Section 6.1.1.2 of the SSAR refers to Section 5.2.3 for discussion of the fabrication and processing of austenitic stainless steels and compliance to the guidelines of Regulatory Guides (RGs) 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"; 1.34, "Control of Electroslag Weld Properties"; and 1.44, "Control of the Use of Sensitized Stainless Steel." The Electric Power Research Institute's Advanced Light Water Reactor Utilities Requirements Document (EPRI URD) describes (in Section 5.3.1.1) certain guidelines applicable to austenitic stainless steels for the control of cold work, limitations on its use, and the surface grinding of cold-worked material. The staff requested that Westinghouse address those aspects of the fabrication and processing of ESF components made of austenitic stainless steels pertaining to cold work that had not been addressed in the SSAR, and to identify those positions which differed from the EPRI URD. This was Open Item 6.1.1-6. Westinghouse revised Section 6.1.1.2 of the SSAR to describe the controls placed on cold work in austenitic stainless steels by reference to subsection 5.2.3.4. Section 5.2.3.4 of the revised SSAR specifically addresses the area of cold work in austenitic stainless steels, indicating that it is in conformance with all the guidelines contained in the EPRI URD and that there are no positions that differ from those adopted in the EPRI URD. The staff found (in NUREG-1242) that the EPRI URD criteria for fabrication of cold-worked austenitic stainless steel parts are adequate and thus the positions adopted in the AP600 design are acceptable. Therefore, Open Item 6.1.1-6 is closed.

Fabrication and welding of austenitic stainless steels for ESF components will meet the guidelines of RGs 1.31 and 1.44. A combined license (COL) applicant should review the vendor fabrication and welding procedures to ensure that these guidelines are followed. The staff requested that Westinghouse include COL Action Item 6.1.1-1 to address this requirement. This was Open Item 6.1.1-7. Westinghouse subsequently revised the SSAR by adding Section 6.1.3, "Combined License Information Item," which states that COL applicants will address review of vendor fabrication and welding procedures or other quality assurance methods to judge conformance of austenitic stainless steels with RGs 1.31 and 1.44. On this basis, Open Item 6.1.1-7 is closed.

Materials used in the fabrication of ESF components should be selected after consideration of the possibility of degradation during service. The staff requested that Westinghouse provide information to confirm that 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. This was Open Item 6.1.1-8. 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 AP600 ESF, the materials specifications for the pressure-retaining valves in contact with the reactor coolant are the same as those used for the RCPB valves and piping. Subarticles NB-, NC-, and ND-3120 require, in part, consideration of the effects of corrosion, erosion, and abrasive wear (Subarticles NB-, NC-, and ND-3121) and with environmental effects (specifically, irradiation-induced changes) (Subarticles NB-, NC-, and ND-3124). Discussion of the corrosion and compatibility of those ESF materials in contact with the reactor coolant during service is included in Section 5.2.3 of this report. The materials selected for the ESF have demonstrated satisfactory performance in operating nuclear power plants 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. Open Item 6.1.1-8 is closed.

The discussion in the previous paragraph addressed degradation of materials during normal service. 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 AP600, the potential for this is minimized by the release of trisodium phosphate (TSP) from the pH adjustment basket into the containment sump. This action is controlled so that the pH of the sump fluids 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.

An additional postaccident situation involves the exposure of surfaces of components within containment, that may contain aluminum and zinc, to containment sump fluid. Chemical attack of these surfaces results in the production of hydrogen. The hydrogen generation rate depends on the corrosion rates of these materials, which, in turn, are dependent on factors such as the fluid chemistry and pH, the metal and fluid temperatures, and the surface area exposed to attack by the fluid. The AP600 SSAR analyzes this potential situation using estimates of the inventory of aluminum and zinc in containment and corrosion data for these metals that reflect temperature change over time following a LOCA accident. SRP 6.1.1 provides, in part, for the review of these corrosion rates, as a subset of the review of the complete hydrogen control system, which is addressed in Section 6.2.5 of this report. The corrosion rate data used in the AP600 design are given in Table 6.2.4-5 of the SSAR and were compared with the guidance provided in RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and with published corrosion rate data for aluminum and zinc. RG 1.7 states that an acceptable value for aluminum corrosion rate in alkaline solution is 0.508 cm/yr (200 mils/yr). This value should be adjusted upwards for higher temperatures existing early in the accident sequence. The aluminum corrosion rates provided in Table 6.2.4-5 of the SSAR range from 104 cm/yr (41,000 mils/yr), immediately after an accident, to 0.523 cm/yr (206 mils/yr) after elapse of 11 hours. These values are well above the corrosion rates specified in RG 1.7 and the published corrosion rate data. RG 1.7 provides no guidance concerning acceptable values for zinc corrosion. However, the published corrosion rate for zinc in slightly alkaline water is 0.01 cm/yr (4 mils/yr). This value is significantly below the corrosion rates provided in Table 6.2.4-5 which are 0.03 cm/yr (118 mils/yr), immediately after an accident, and 0.02 cm/yr (8 mils/yr) after elapse of 11 hours. It is thus concluded that the data provided in Table 6.2.4-5 of the SSAR are reasonable, contain a degree of conservatism, and are appropriate for use in estimating hydrogen generation due to corrosion of these materials in a postaccident situation. (The AP600 design does not have a safety-related containment spray system, and accordingly, the review guidelines of Section 6.1.1 of the SRP regarding containment spray systems are not applicable.)

GDC 4 requires that all components important to safety be compatible with the environmental conditions. Materials selected for use in the construction of an ESF should be compatible with the ESF fluids. This includes consideration of any nonmetallic materials that might be used, to ensure that their presence will not lead to deterioration of the ESF materials of construction. A major concern is that the presence of certain non-metallics can lead to the enhanced potential for corrosion and stress corrosion cracking. The EPRI URD (in Section 5.2.8) specifies that the impurity levels of nonmetallic materials used within the nuclear steam supply system and associated systems shall be controlled within certain specified limits. The staff requested that Westinghouse discuss, in the SSAR, its chemical content controls for non-metallic materials to protect ESF components and to identify those positions related to chemical content control that differ from the EPRI URD. This was Open Item 6.1.1-9.

The EPRI URD requirement is intended to address a concern related to those non-metallic materials used infrequently or in the course of construction, installation, and testing where subsequent cleaning is not practical or can be omitted to reduce maintenance time. Thus it includes such materials as cutting fluid, lubricants, abrasive adhesives, and tape. Westinghouse revised the SSAR to indicate that appropriate measures will be taken to avoid such contamination in the handling, storing, and cleaning of the austenitic stainless steel ESF components during the fabrication, installation, and testing phases. The position adopted by Westinghouse related to control of nonmetallic materials is in conformance with the recommendations contained in the EPRI URD. The staff found (in NUREG-1242) that the EPRI URD recommendations with respect to prevention of contamination of austenitic stainless steel parts by nonmetallic materials are adequate; and thus, the position adopted by Westinghouse regarding cleanliness controls on ESF components is acceptable. Open Item 6.1.1-9 is closed.

The thermal insulation used in the AP600 containment will be predominantly of the reflective metallic type. Any fibrous insulation used will be enclosed in stainless steel cans. The SSAR further states that any nonmetallic thermal insulation used in the design of the AP600 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.

The staff concludes that Westinghouse's 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. This conclusion is drawn on the basis of the following observations:

- The materials selected for ESFs satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the 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 fracture toughness of the ferritic materials will meet the requirements of the ASME Code. Thus, the design of the AP600 ESF 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.

- The controls imposed on the austenitic stainless steel of the systems conform to the recommendations of, or proposed acceptable alternative approaches to, RG 1.44 as discussed in detail in Section 5.2.3.4 of the SSAR. The detailed evaluation of meeting these RG recommendations is in Section 5.2.3 of this report.

- Conformance with the ASME Code, the RGs, and staff positions mentioned above constitutes an acceptable basis for meeting the requirements of GDCs 1, 4, 14, 35, and 41, as well as 10 CFR 50.55a and Appendix B to 10 CFR Part 50, in which the systems are to be designed, fabricated, and erected to perform their functions as required.

- The controls to be placed on concentrations of leachable impurities in non-metallic thermal insulation used on components of the AP600 ESF follow the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Compliance with the recommendations of RG 1.36 form a basis 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.

- The controls on the pH and chemistry of the emergency core cooling water, following the LOCA or design-basis accident, are adequate to reduce the probability of stress corrosion cracking of austenitic stainless steel components and welds of the ESF systems in containment throughout the duration of the postulated accident to completion of cleanup. Thus, the ESF components in the AP600 design meet the requirements of GDCs 4, 35, and 41, 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. Because the AP600 design does not have a safety-related containment spray system, review of such systems, required by Section 6.1.1 of the SRP, is not applicable.

- The control of pH of the cooling water, in conjunction with controls on selection of containment materials, is in accordance with RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provides assurance that the cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal, or cause serious deterioration of the materials in containment.

- The controls on the pH and chemistry of the ECCS solutions meet the staff's positions on postaccident chemistry requirements for PWR emergency coolant water, as detailed in the Branch Technical Position MTEB BTP 6-1, "pH for Emergency Coolant Water for PWRs." The design also meets the requirements of GDC 14 for ensuring the low probability of abnormal leakage or failure of the reactor coolant pressure boundary and safety-related structures. The staff concludes that the proposed pH for emergency cooling water is acceptable.

- The controls to be placed upon component and system cleanup are in accordance with recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." These controls provide a basis for concluding that the components and systems will be protected against damage or deterioration by contaminants as stated in the cleaning requirements of Appendix B to 10 CFR Part 50.

6.1.2 Coatings

The staff reviewed the use of coatings in accordance with Section 6.1.2 of the SRP. The protective coating systems are acceptable if the protective coatings that will be applied inside and outside the AP600 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 structures, systems, and components.

To meet the requirements of Appendix B to 10 CFR Part 50, an applicant must specify that the protective coatings in containment conform to the testing requirements of ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Reactor Containment Facilities," and the quality assurance guidelines of RG 1.54. This provides assurance that the protective coatings will not fail under design-basis accident (DBA) conditions, and will not generate significant quantities of solid debris that would adversely affect the performance of the ESF.

However, in the AP600 design, Westinghouse regarded the coatings as being non-safety-related. Westinghouse proposed to eliminate the quality assurance (QA) requirements for applied coatings (paints) of ANSI Standard N101.4-1972, as conditionally endorsed by RG 1.54, on the premise that coatings are classified as non-safety-related.

The staff was not convinced that all coatings for the AP600 containment should be classified as non-safety-related. The staff was concerned that improperly applied coatings within containment may detach and thus prevent safety-related components from performing their functions. On the basis of recent experiences with operating reactors, concerns have been raised over the potential for unqualified coatings, or incorrectly applied qualified coatings, in conjunction with other LOCA-generated debris to cause blockage of the sump screens. This was identified as DSER Open Item 6.1.2-1.

In response to this item, Westinghouse revised its AP600 SSAR. Revision 22 of the SSAR included Table 6.1-2, "AP600 Coated Surfaces, Containment Shell and Surfaces Inside Containment." Table 6.1-2 classified as safety-related the coatings that will be applied over the following surfaces:

(1) all outside containment shell surfaces above Elevation 135'-3"
(2) all inside containment shell surfaces 7 feet above the operating deck
(3) areas inside the containment surrounding the containment recirculation screens.

The rest of the areas inside and outside of the AP600 containment are classified as non-safety-related.

For items 1 and 2 above, the staff agrees with Westinghouse that these coatings should be considered safety-related. Specifically, the inorganic zinc coating on the outside of containment promotes wettability of the outside surface of the containment shell so that the PCS water draining from the passive containment cooling water storage tank (PCCWST) forms a thin film that effectively spreads over the containment shell surface. The inorganic zinc coating on the inside of the containment shell has been included in PCS testing and analysis and as a result is considered safety-related. For item 3 above, the staff agrees with Westinghouse that these coatings should be considered safety-related. This determination is consistent with the assumption that the coatings in this area could cause blockage of the recirculation screens. See section 6.2.1.8 of this report for additional discussion concerning coating transport analysis. In addition, Westinghouse states in SSAR Section 6.1.2.1.6 that Appendix B to 10 CFR Part 50 will apply to procurement of non-safety-related coatings used inside containment on internal structures, including walls, floor slabs, structural steel, and the polar crane, except for surfaces located inside the chemical and volume control system room. Non-safety-related coatings used in the chemical and volume control system room are not subject to procurement under 10 CFR 50, Appendix B, because the room is connected to the containment in a limited way through a drain line. The commitment to Appendix B to 10 CFR Part 50 for the non-safety-related coatings mentioned above is consistent with the assumptions in Westinghouse's coating transport analysis discussed in Section 6.2.1.8 of this report. The staff finds the information in SSAR Revision 22 associated with this issue acceptable and, therefore, DSER Open Item 6.1.2-1 is closed.

The AP600 design uses "high-top" coatings, which are significantly thicker than the coatings used in the past. Therefore, high-top coatings will need to be evaluated for their potential effect on other systems if they should fail during accident or LOCA conditions. Westinghouse was required to indicate that coatings inside containment will be qualified and correctly applied to provide adequate corrosion protection of painted structures and components. Westinghouse was also required to supply data and an in-depth analysis to justify use of new coating types (such as "high-top" coatings) inside containment. This was identified as DSER Open Item 6.1.2-2. Westinghouse revised Section 6.1.2.1.6 of the SSAR to address this item. Revision 22 of the SSAR states that safety-related coatings used in the AP600 meet the pertinent provisions of Appendix B to 10 CFR Part 50, and that the quality assurance program for such coatings conforms to the requirements of ASME NQA-1-1983 as endorsed in Regulatory Guide 1.28. The SSAR also states that safety-related coatings used in the AP600 are tested for radiation tolerance for performance under design-basis accident conditions. In addition, the SSAR states that the coating applicator submits and follows acceptable procedures to control surface preparation, application of coatings and inspection of coatings. For safety-related coatings the painters are qualified and certified, and the inspectors are qualified and certified. The SSAR further states that the procurement, application, and monitoring of safety-related coatings are controlled by a program consistent with the positions in RG 1.54 (as identified in SSAR Appendix 1A) and prepared by the Combined License applicant. This is COL Action Item 6.1.2-1. This is acceptable to the staff because only qualified and properly applied safety-related coatings will be used and is consistent with the guidance in RG 1.54. Therefore, DSER Open Item 6.1.2-2 is closed. However, during its review, the staff requested Westinghouse to clarify its description of the conformance of the AP600 with RG 1.54 (discussed in Appendix 1A of the SSAR). In addition, Westinghouse has agreed to make some editorial changes to SSAR Section 6.1.3.2. The incorporation of these changes into the SSAR is FSER Confirmatory Item 6.1.2-1. Subsequent to the issuance of the advance FSER Westinghouse provided the above information is SSAR revision 23. Therefore, FSER Confirmatory Item 6.1.2-1 is closed.

The source term used for estimating the radiolysis of water and its contribution to hydrogen production in Section 6.2.4.3.1.2 of the SSAR differs significantly from the source term recommended in RG 1.7. The stated reason is that the expected radioactive inventory in the coolant and the containment sump will be much lower for a LOCA, because the amount of core damage would be lower than considered in RG 1.7. The staff concluded that the SSAR should indicate whether these changes in the recommended practices result in a reduction of the amount of predicted hydrogen production. This was identified as DSER Open Item 6.1.2-3.

Westinghouse addressed this issue by stating that because of the mechanisms involved in radiological generation of hydrogen, it is expected that the amount of hydrogen generated will be considerably lower than specified in RG 1.7. In addition, in design basis loss of coolant accidents, there is expected to be no damage to the core and thus no release of activity from the core to the sump solution. However, the source term used for determining radiolytic production of hydrogen is conservatively based on guidance of RG 1.7. The staff reviewed this information and found that the prediction of hydrogen generation by radiolysis has a sufficient degree of conservatism. Therefore, DSER Open Item 6.1.2-3 is closed.

6.2 Containment Systems

The containment systems for the AP600 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 passive containment cooling system

(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 411.6 kPa (45 psig). The limiting calculated peak pressure occurs for a main steamline break (a 0.129 m^2 (1.388 ft^2) nominal area break) and full double-ended rupture from 30 percent power with a main steam isolation valve (MSIV) failure, and is 405.4 kPa (44.1 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 AP600 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 4.13 cm (1.625 in.). The wall thickness is increased by 0.317 cm (0.125 in.) in the transition region where the cylindrical shell enters the concrete embedment in order 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 (NSSS) (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; electrical; 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 48,988 m^3 (1,730,000 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 411.6 kPa (45.0 psig) and a design temperature of 138 °C (280 °F).

The following AP600 containment design features are compared with those of a typical Westinghouse two-loop 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

A number of insights and conclusions can be drawn from Table 6.2-1. The AP600 containment design represents a significant change from Westinghouse's previous two-loop design, which consisted of a steel containment vessel with a reinforced concrete, pressure-controlled secondary containment.

The AP600 free volume is considerably larger than that of previous Westinghouse two-loop containments. Although this larger volume is expected when comparing the AP600 design to Westinghouse's lower-power plants, the comparative ratios of containment free volume to power show that the AP600 containment design has a considerably larger free volume-to-power ratio than is found in Westinghouse's other designs. This ratio is 22.6 m^3/MW (800 ft^3/MW) for Westinghouse's earlier two-loop design, and 25.2 m^3/MW (892 ft^3/MW) for the AP600. This ratio indicates that the AP600 containment is a more robust design, because there is more containment volume available per thermal megawatt.

Table 6.2-1 of this report also shows that the external design pressure of the AP600 containment (20.7 kPa (3.0 psig)) is greater than that of a typical two-loop plant (5.5 kPa or (0.8 psid)). It should be noted that the value for the AP600 is founded on ASME Service

Table 6.2-1 Comparison of Westinghouse Containment Design Features

Parameter	AP600	1650 MWt
Power, MWt	1940	1650
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	3.8 cm (1.5 in.) thick steel cylindrical primary containment with top and bottom dome, 0.76 m (2.5 ft) thick reinforced concrete cylindrical secondary containment
Secondary containment	No	Yes
Free volume	(1.73E+06 ft^3) 4.9x10^4 m^3	(1.32E+06 ft^3) 3.7x10^4 m^3
Volume-to power ratio	(892 ft^3/MW) 25.2 m^3/MW	(800 ft^3/MW) 22.6 m^3/MW
Internal design pressure	(45 psig) 411.6 kPa	(46 psig) 418.5 kPa
External design pressure	(3.0 psid) 20.7 kPa	(0.8 psid) 5.5 kPa
Design temperature	(280 °F) 138 °C	(268 °F) 131 °C
Design leak rate, weight %/day	0.12	0.5
Calculated peak internal pressure (design margin)	(44.1 psig) 405.4 kPa (1.5 %)	(42.2 psig) 392.3 kPa (~8 %)
Calculated peak external pressure (design margin)	(2.0 psid) 13.8 kPa (33 %)	(0.5 psid) 3.45 kPa
Heat removal system	PCS and non-safety-grade fan coolers	safety-grade fan coolers and containment sprays
Combustible gas control system	1. DBA - passive autocatalytic recombiners 2. severe accidents - hydrogen igniters	1. DBA - hydrogen thermal recombiners (Class 1E ac power) 2. Severe accidents - NA
Number of penetrations	~40	~100
Motive power for containment isolation valves	1. air-operated valves 2. Class 1E DC motor-operated valves	Class 1E AC motor-operated valves

Level A considerations, while that of the two-loop plant is founded on ASME Section VIII. The external design pressure is further discussed in Section 6.2.1.1 of this report.

At the time the DSER was issued, the source term methodology to be applied to AP600 was not yet firm. This was DSER Open Item 6.2.1-1. The staff evaluated the ability of the AP600 design to comply with the relevant dose limits of 10 CFR 50.34 and GDC 19 in Section 15.3 of this report. That evaluation assumed a 0.10 weight percent per day leak rate from the AP600 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 AP600 and Open Item 6.2.1-1 is closed.

Furthermore, the original AP600 design did not have containment sprays, which makes natural deposition on surfaces in containment far more important than in past designs. The elimination of containment sprays from the design required further staff review. This was Open Item 6.2.1-2. Westinghouse added non-safety-related containment sprays as described in Section 6.5.2 of the SSAR and evaluated by the staff in Section 19.2.3.3.9 of this report. Therefore, Open Item 6.2.1-2 is closed.

The containment design pressure margin is discussed in Section 6.2.1.1 of this report. Westinghouse presented a design capability for external pressure of 20.68 kPa (3.0 psid). However, Westinghouse did not submit any values for the peak calculated external pressure; therefore, this value was not entered in the table. This was Open Item 6.2.1-3. Subsequently, Westinghouse calculated a peak external pressure of 2.0 psid (13.8 kPa). Therefore, Open Item 6.2.1-3 is closed.

The reliance of the AP600 on cooling by naturally occurring physical phenomena represents a significant difference from other Westinghouse designs. The heat removal system for the AP600 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 performance validation of the PCS was still ongoing by Westinghouse, and staff review was continuing. This was Open Item 6.2.1-4. The performance validation of the PCS has been completed by Westinghouse, and the staff review is presented in Section 21.6.5. Therefore, Open Item 6.2.1-4 is closed.

Table 6.2-1 of this report indicates that both the AP600 and the typical Westinghouse two-loop design use thermal recombiners to limit the hydrogen concentration resulting from a DBA. The primary difference between the AP600 and other Westinghouse designs is that the AP600 hydrogen thermal recombiners are powered by non-safety-grade ac power, whereas those of previous plants are powered from safety-grade ac power. While the hydrogen concentrations provided by Westinghouse in Section 6.2.4 of the SSAR appear to be similar to those of previously licensed plants, the staff had not yet performed confirmatory calculations. These calculations were to be performed after the issue of powering the thermal recombiners with non-safety-grade ac power was resolved. This was Open Item 6.2.1-5.

The AP600 has been provided with passive autocatalytic recombiners, as described in Section 6.2.4 of the SSAR, to limit hydrogen concentration resulting from a DBA. The staff's evaluation of the passive autocatalytic recombiners for combustible gas control is contained in Section 6.2.5 of this FSER. This closes Open Item 6.2.1-5.

Table 6.2-1 of this report also shows that the AP600 containment has considerably fewer mechanical penetrations (approximately 40) than a typical two-loop design (approximately 100). Additionally, the containment isolation valves in the AP600 are primarily either air operated or motor operated by safety-grade DC power. In previous designs, containment isolation valves were typically motor-operated valves powered from safety-grade AC. The staff's evaluation of the containment isolation system is contained in Section 6.2.4 of this report.

Compliance with Regulatory Requirements

The Westinghouse AP600 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 taken for the AP600 containment analysis has evolved from the approach used for the WGOTHIC 1.0 and WGOTHIC 1.2 analyses. To address staff concerns with some of the assumptions and modeling features employed, Westinghouse has developed a model and uses assumptions and boundary conditions that are more consistent with current practices for containment analyses for current operating reactors. The approach is consistent with the guidance provided in SRP 6.1.1.2.A, "PWR Dry Containments, Including Subatmospheric Containments." The review of WGOTHIC 4.2 and the AP600 evaluation model can be found in Section 21.6.5

Compliance with 10 CFR 50, Appendix A

The current guidance for demonstrating that a containment design complies with GDCs 16, 38 and 50 is delineated in Chapter 6.2 of the Standard Review Plan (SRP). The SRP addresses acceptance criteria and some specific model assumptions for design basis LOCA and MSLB analyses for all existing containment types. Westinghouse elected to evaluate the PCS performance using these current guidelines. The Westinghouse documentation for the AP600 evaluation model (EM) is consistent with the guidelines in SRP Sections 6.2.1 and 6.2.1.1.A, as well as Regulatory Guide 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 AP600 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 AP600 containment and the PCS, as used in the licensing analyses, are verified prior to operation.

On the basis of the evaluation presented in Section 21.6.5 of this report, the staff has determined that the WGOTHIC computer program, combined with the conservatively biased AP600 evaluation model, is acceptable for the evaluation of the peak containment pressure following a design basic accident. Although the WGOTHIC code itself is essentially a best-estimate tool, Westinghouse has taken a conservative approach in the evaluation methodology (EM) it is using to support design certification. The AP600 WGOTHIC EM 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 AP600 WGOTHIC EM 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 1200 seconds for LOCA, and up to 600 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.

Long Term Pressure Analysis (GDC 38)

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 siting evaluation is performed with a constant, design basis leak rate. Westinghouse had originally proposed that the 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.

Late in the review process, Westinghouse determined that it could not meet the proposed long-term objective with the original analysis approach. Westinghouse therefore revised the analytical procedure to credit the effect of two-dimensional (2-D) heat conduction (between wet and dry regions of the containment shell) when less than full coverage of the containment shell is expected. The revised 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 ACRS meeting in December 1997 (Westinghouse letter NSD-NRC-97-5492, "Presentation Material for December 9, 10, 11 & 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 (after three hours when the PCS flow rate is first cut back to about one-half its initial value) in the selection of the analysis methodology (GOTHIC) and in the development of a model for the PCS (WGOTHIC). With the coverage area less than the initial assumed 90 percent, heat transfer from the hot, dry regions of the shell into the cooler, wet regions of the shell would occur. To account for this 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 (see Section 21.6.5).

During the first three hours of a DBA event, with the PCS flow rate maintained at 1,665.6 liters/min (440 gpm), the pressure performance envelope is similar to existing designs which use active safety systems. When the PCS flow rate is reduced after three hours, there is a tendency to slightly repressurize and maintain a pressure somewhere between 218.5 and 273.7 kPa (17 to 25 psig), well below the 411.6 kPa (45 psig) design value, until 30 hours into the event when a further reduction in the PCS flow rate occurs. When the flow is again reduced after 30 hours, the containment again repressurizes with the resulting pressure being between 225.4 and 322.0 kPa (18 to 32 psig) for the remainder of the three-day design basis performance period of the PCS but continually decreasing as the decay heat decreases. The difference between the low and high pressure estimates are based on the credit given in the analyses to consider the effects of 2-D heat conduction. As discussed in Section 21.6.5, 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 loss-of-coolant accident 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 AP600 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 EM 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 AP600 PCS (as calculated by the WGOTHIC EM) 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 loss-of-coolant accident and maintain them at acceptably low levels has been demonstrated.

Compliance with 10 CFR 52.47(b)(2)

The unique characteristics of passive containment cooling system 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":

- The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;
- Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;
- 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, has 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 has developed test programs to investigate the passive containment safety systems. These programs include both component and phenomenological (separate-effects) tests and integral-systems tests. The cold water distribution test (WDT) 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 AP600 analysis codes, to comply with the last of the three requirements above.

The large-scale test (LST) is the only integral test for the AP600 PCS. While this test 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 AP600.

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 AP600 EM for DBA licensing analyses to support design certification. Specific limitations and restrictions for future analyses are presented in Section 21.6.5.8.3 of this report.
Section 21.6.5.8.3 of the AP600 FSER defines for the staff the calculational method that has been previously 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 AP600 nodal model described in Section 4 of WCAP-14407, "WGOTHIC Application to AP600," Revision 3, April 1998 must be used. Further, the assumptions must be consistent with the limitations and restrictions denoted in Section 21.6.5.8.3.

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

A full description of WGOTHIC 4.2, the current licensing version, and the staff's review of WGOTHIC 4.2 are presented in Section 21.6.5 of this report.

WGOTHIC code and its methodology are founded on a modified version of the GOTHIC code. GOTHIC is a containment analysis package, which has been selected by EPRI and a national users group for development as a reference containment analysis code. On August 10, 1994, Westinghouse provided the staff with EPRI reports on GOTHIC, including the Technical Manual, Users Manual, and Qualification Report. The portions of these manuals that apply to the AP600 analysis were reviewed by the NRC staff, as were the applicable test qualifications. This was Open Item 6.2.1.1-1.

On September 21, 1995, Westinghouse provided revised GOTHIC documentation pertaining to GOTHIC Version 4.0, the version that forms the bases for the current licensing version of WGOTHIC. Westinghouse responded to staff RAIs on the EPRI GOTHIC 4.0 documentation and on the use of the applicable test qualification (RAIs 480.463 to 480.485). Therefore, Open Item 6.2.1.1-1 is closed.

In creating the WGOTHIC 4.2 computer program, used for licensing analyses to support the AP600 design certification, from the GOTHIC computer program, Westinghouse added analytical models to represent the unique features of the AP600 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.

To model the passive cooling features of the AP600, several assumptions were made in creating the plant deck. The external cooling water does not completely wet the containment shell; therefore, both wet and dry sections of the shell are modeled. The original analysis performed with earlier versions of WGOTHIC (Versions 1.0 and 1.2) assumed coverage of 40 percent on the top of the dome and 70 percent on the side walls. Heat conduction from the dry-to-wet sections was not considered in the original analysis, although calculations showed this to be a benefit. Representative external cooling water flow rates, which include a worse-case, single-failure assumption, were used for the wet sections. The original analysis also assumed that the external cooling water was not initiated until 11 minutes into the transient, allowing time to initiate the signal and to fill the headers, the PCS distribution bucket, and weirs on the basis of an initial PCS flow rate of 832.8 liters/min (220 gpm) (i.e., there is no time-dependent film coverage model). Because the air baffle is not leak-tight, air leakage flow paths are included to simulate the effects of air leaking through the baffle, thus bypassing the normal air flow path.

Design changes to the PCS flow rates, to the water coverage model as used in the licensing analyses, as well as changes to WGOTHIC and the modeling of the AP600 have resulted in revised analyses to support the design certification. The initial PCS flow rate, for the first three hours, has been increased to about 1,665.6 liters/min (440 gpm). This increased flow results in a delay time of 337 seconds before taking credit for PCS water. A limiting PCS flow model was developed to account for only the amount of water expected to evaporate and to provide a means to account for the variation in expected wetting, or surface coverage, as the PCS flow rate is decreased over time. This model is described in Section 21.6.5 of this report. In letter NSD-NRC-97-5152, dated May 23, 1997, Westinghouse provided an ancillary analysis to support a further modification to the limited PCS flow model to account for 2-D conduction between the wet and dry sections. This modification is also discussed in Section 21.6.5. Finally, the analysis methodology, in conjunction with the use of the WGOTHIC 4.2 computer program, is now based on a conservative evaluation model.

Design changes to 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 design. New design features include increased inventory in the passive containment cooling water storage tank (PCCWST), the addition of 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 seven days). In addition, the PCCWST now also provides makeup to the spent fuel pool (SFP).

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

It was not clear to the staff whether any of the initial values were technical specification (TS) maximums, or whether they were assumed bounding conditions. Furthermore, the initial assumption of 100 percent relative humidity and the relatively high temperature of 48.9 °C (120 °F) appeared to be nonconservative. The extent to which this was offset by the initial

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	48,988 m^3	1.73E+06 ft^3
External Temperature	46.1 °C dry bulb 26.7 °C wet bulb	115 °F dry bulb 80 °F wet bulb

pressure was unclear. The staff reviewed the input assumptions, along with the calculated results. This was Open Item 6.2.1.1-2.

The current initial internal pressure and temperature, and the external temperature are TS maximums and have been shown, in Section 5 of WCAP-14407, "WGOTHIC Application to AP600," Revision 3, April 1998, to result in a conservative peak pressure calculation. In Section 5 of WCAP-14407, it was demonstrated that 0 percent relative humidity is a conservative assumption. The staff has reviewed these input assumptions and finds them acceptable for the licensing analyses. Therefore, Open Item 6.2.1.1-2 is considered to be closed.

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

PCS Flow from PCCWST	Flow rate during time frame		Area coverage
Time period	liters/min	gpm	---
Start of flow to 3 hours	1,673.2 to 1,608.8	442 to 425	90 %
3 hours to 30 hours	467.5 to 416.4	123.5 to 110	51 %
30 hours to 72 hours	274.4 to 237.4	72.5 to 62.7	30 %
Post 72 hours (from PCCAWST)	237.4	63.7	(25 %)

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, after three hours 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 Sections 6.2.1.3 and 6.2.1.4 of the SSAR.

This is the first licensing application in which WGOTHIC has been used. In addition to reviewing the applicability of the input assumptions, Westinghouse reviewed the data from the LSTs, and the staff reviewed the analytical models and input assumptions. This was Open Item 6.2.1.1-3.

The baseline LST tests were documented in WCAP-13566, "AP600 1/8th Large-Scale Passive Containment Cooling System Heat Transfer Baseline Data Report," dated October 1992. The Phase 2 and Phase 3 tests were documented in WCAP-14135, "Final Data Report for PCS Large-Scale Tests, Phase 2 and 3," July 1994 (and later revised in April 1997 to Revision 1). These reports contained primarily test data, and did not provide evaluation or interpretation of the test results. Westinghouse, in letter NTD-NRC-95-4463, dated May 15, 1995, submitted "AP600 Testing Program Report: Large-Scale Test Data Evaluation (PCS-T2R-050)." This evaluation is discussed in Section 21.3.8. The staff's review of the analytical models and input assumption used for the current SSAR analyses, as performed with WGOTHIC 4.2, is provided in Section 21.6.5. Therefore, Open Item 6.2.1.1-3 is closed.

Containment Pressure Response

The staff has reviewed Westinghouse's analyses of the AP600 containment's pressure response, as discussed below.

Internal Pressure Analysis

The pressure response of the AP600 containment can be divided into two temporal phases - the short-term or blowdown portion of the transient, and the longer term balance of the transient. The AP600 containment response to the high pressure blowdown portion of LOCA and MSLB transients is not significantly different from that for a standard Westinghouse two-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 AP600 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-loop plant. None of the new AP600 passive design features comes into play during this first portion of a postulated transient. In Section 8 of WCAP-14407, Westinghouse performed an analysis for the AP600 during the blowdown portion of the LOCA to compare the current SSAR multinode model to a simple, single-node model (similar to the modeling used for current operating reactors). This analysis showed that the SSAR multinode model for the AP600 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 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 pressure. 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 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	374.4 (39.6)	37.2 (5.4) 9%	199.5 (391.1)	---
LOCA, double-ended, cold-leg guillotine	400.6 (43.4)	11.0 (1.6) 2.7%	138.4 (281.2)	233.7 (19.2)
MSLB, 1.388 ft², full DER, 102% power, MSIV failure	402.6 (43.7)	9.0 (1.3) 2.2%	188.3 (370.9)	---
MSLB, 1.388 ft², full DER, 30% power, MSIV failure	405.4 (44.1)	6.2 (0.9) 1.5%	186.8 (368.2)	---

Notes: 1. Design pressure is 411.6 kPa (45 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 405.4 kPa (44.1 psig) at about 570 seconds after the MSLB begins. Westinghouse determined this peak pressure after analyzing more than 30 different LOCA and MSLB scenarios. This value provides a margin of 1.5 percent to the design pressure of 411.6 kPa (45 psig).

In Item II.2 of Section 6.2.1.1.A of the SRP, the staff cites GDC 50 as requiring a "sufficient margin" for calculated peak containment pressure to assure that the design leakage rate will not be exceeded. To satisfy the requirements of GDC 16 and GDC 50 regarding sufficient margins, for plants at the CP stage of review, the containment design pressure should provide at least 10 percent margin above the accident peak calculated containment pressure following a LOCA or a steam or feedwater line break. For plants at the OL stage of the review, the peak calculated containment pressure following a LOCA, or a steam or feedwater line break, should be less than the containment design pressure. In discussions with Westinghouse, the staff indicated that additional information was needed on uncertainty and margin in the Westinghouse calculations, since a "best-estimate" heat transfer was used in the WGOTHIC 1.0 analyses. In response to these concerns, Westinghouse submitted on June 30, 1994, "AP600 Passive Containment Cooling System Design Basis Analysis Model and Margin Assessment." This document presented calculations using WGOTHIC Version 1.2 (the SSAR employed WGOTHIC Version 1.0).

Westinghouse stated that the WGOTHIC 1.0 SSAR LOCA mass and energy release analysis was performed using assumptions that would maximize the initial system mass and energy available for release to containment.

Westinghouse stated that the WGOTHIC 1.0 SSAR containment model was created using assumptions that would 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 AP600 WGOTHIC 1.0 containment model is as follows:

- The maximum outside air temperature of 46.1 °C (115 °F) was used as a boundary condition to reduce the heat transfer from containment.

- The maximum containment air temperature of 48.9 °C (120 °F), pressure of 108.2 kPa (1 psig), and 100 percent humidity initial conditions were used to increase the initial stored energy inside containment. (The current SSAR analyses are based on 0 percent humidity inside containment.)

- The subcooling of the PCS water temperature was ignored to reduce the heat transfer rate from containment. (Subcooling was added to the WGOTHIC 1.2 and 4.2 models.)

- A single failure of one out of two valves controlling the PCS cooling water flow was assumed. This assumption provided the minimum PCS liquid film flow rate, and reduced the heat transfer rate from containment.

- The PCS liquid film flow was initiated following an 11-minute delay. This corresponds to the time needed to establish a steady liquid film coverage pattern in the liquid film distribution tests. (The revised PCS flow rate results in the current SSAR delay time of 337 seconds.)

- A 40 percent PCS water film coverage was used on the top of the dome, and a 70 percent coverage was used on the side walls. These values were determined by the minimum coverage observed in the liquid film distribution tests. (The current SSAR water coverage is determined by the limiting flow model, as described in Section 21.6.5.)

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

As discussed in Section 6.2.1.1 of the SSAR, the staff reviewed the input assumptions (especially the second bullet above). This was Open Item 6.2.1.1-5.

The input assumptions (especially the second bullet above concerning the initial relative humidity inside containment) have been revised in the current SSAR analysis to account for the design changes in the PCS and to use the conservative 0 percent relative humidity inside containment as an initial condition. Therefore, Open Item 6.2.1.1-5 is closed.

WGOTHIC 1.2 incorporates improved models to represent mixed convection in the containment interior. In contrast, WGOTHIC 1.0 allowed the user to choose free or forced convection, and then verify that the code results justified the choice. WGOTHIC 1.2 also incorporated a liquid film enthalpy transport model for the exterior cooling water on the containment. In addition, a log-mean noncondensible pressure term was added to the boundary layer mass transfer correlation, and several other minor changes were made.

In addition, the June 30, 1994, analysis incorporated several minor changes to the WGOTHIC 1.0 input decks, many of which were necessary to accommodate the model changes in WGOTHIC 1.2. The following is a summary of the changes:

- The annulus flowpath inertia lengths were modified to balance the steady state volume average velocity with the corresponding flowpath velocity.
- Heat and mass transfer multipliers, which are needed to account for entrance effects in the annulus, were calculated and applied to the lower, upward-flowing volumes of the annulus.
- Inner vessel wall heat transfer multipliers were computed and applied to the inside surface of the containment to convert from the Colburn correlation to the flat-plate correlation.
- The annulus hydraulic diameter input was changed to be consistent with the mixed convection heat transfer correlation with entrance effect multipliers applied.
- In the WGOTHIC Version 1.0 analysis, a constant break droplet diameter was used throughout the entire transient, even though liquid was expected to spill from the break after blowdown was complete. A forcing function was applied to eliminate the formation of a drop field after the blowdown phase is complete.
- The number of subregions was reduced in the larger, outer concrete sections of two conductors.
- On the basis of the results of the LOCA analyses, no significant level of zirconium-water reaction occurs; therefore, the energy release from such a reaction was excluded from the mass and energy calculations.

For the WGOTHIC 1.0 and 1.2 analyses, no additional energy source as a result of metal-water reaction was considered in calculating the mass and energy releases to containment. According to Section 6.2.1.3 of the SRP, 10 CFR 50.44, Appendix A to 10 CFR 50, and GDC 50, this energy should have been included in the calculations. Therefore, the treatment of metal-water reaction energy as an energy source should have been considered. This was Open Item 6.2.1.1-4. Acceptability of the other items was determined by test comparisons.

In the current SSAR analyses, performed with WGOTHIC 4.2, the heat addition from a postulated 1 percent reaction of the zirconium cladding in the active fuel zone is included in the energy releases. Therefore, Open Item 6.2.1.1-4 is closed.

To demonstrate the effect of these analysis conservatisms in the WGOTHIC 1.0 studies, Westinghouse reran the most limiting LOCA using more nominal values in both the mass and energy release calculations and the WGOTHIC Version 1.2 input deck.

Specifically, the LOCA mass and energy release calculations for the WGOTHIC 1.2 analyses were revised to be closer to a best-estimate evaluation. The following is a summary of the changes made in the calculation of mass and energy releases for only the double-ended cold leg guillotine (DECLG) LOCA:

- The nominal full power temperature was used without adding 2.8 °C (5 °F) for instrument uncertainty.
- The normal RCS operating pressure was used without adding 206.8 kPa (30 psi) for instrument uncertainty.
- The nominal RCS geometric (cold) volume (without uncertainty) was increased to account for thermal expansion only.
- The core licensed power was used without adding 2 percent for calorimetric error.
- The nominal core stored energy was used without adding 15 percent for tolerance.
- The 1979 decay heat standard (without uncertainty) for an 800-day average burnup was used to estimate core decay heat.
- The nominal full power steam generator secondary mass was used without adding 10 percent.

For the two MSLB accidents, the nominal values for mass and energy release calculations were not used; rather, the same conservative values that were used in the previous WGOTHIC 1.0 analysis were employed for the WGOTHIC 1.2 analyses. However, for both the most limiting LOCA and the MSLB accidents, Westinghouse created WGOTHIC Version 1.2 input decks that contained more nominal values. The following changes were made to create a containment model with nominal operating conditions:

- The initial outside air temperature was set to 21.2 °C (70 °F).
- The initial containment air temperature was set to 37.8 °C (100 °F), and the relative humidity was set to 8 percent.
- The initial containment pressure was set to 101.4 kPa (14.7 psia).
- The corresponding conductor initial temperatures were set to 21.2 °C (70 °F) and 37.8 °C (100 °F).

- The PCS water temperature was set to 26.7 °C (80 °F).

- The material emissivities were modified to remove the 0.9 multiplier.

Westinghouse reran the most limiting SSAR LOCA and both MSLB accidents to assess the effects of the WGOTHIC code modifications and input changes made since the original WGOTHIC 1.0 analyses were completed. The comparison between the original WGOTHIC 1.0 analyses and the WGOTHIC 1.2 and the most recent WGOTHIC analyses is provided in Table 6.2-5.

The approach taken for the AP600 containment analysis has evolved from the approach used for the WGOTHIC 1.0 and WGOTHIC 1.2 analyses. These previous analyses relied, to a greater extent, on best-estimate and nominal conditions and assumptions. To address staff concerns with some of the assumptions and modeling features employed, Westinghouse developed a model and uses assumptions and boundary conditions that are more consistent with previous practices for containment analyses for current operating reactors. The approach is based on the guidance provided in SRP 6.1.1.2.A, "PWR Dry Containments, Including Subatmospheric Containments." The current licensing model is described in Section 4 of WCAP-14407, " WGOTHIC Application to AP600," Revision 3, dated April 1998. The current licensing version is WGOTHIC 4.2.

The WGOTHIC 4.2 SSAR containment model was created using assumptions that would 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 AP600 WGOTHIC 4.2 containment 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 is determined by an assessment of the LST and separate tests as discussed in Section 21.6.5 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 is determined by an assessment of the LST and separate tests as discussed in Section 21.6.5 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 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 technical specification limits. A 0 percent humidity initial condition is used to increase the initial stored energy inside containment.

- A single failure of one out of two 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 in the liquid film on the basis of an initial flow rate of 1,666 liters/min (about 440 gpm).

- The water coverage is obtained from the limiting flow model, as described in Section 21.6.5, 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 after three hours.

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

  The air gap between the steel liner and the concrete used in the licensing analyses is selected by Westinghouse to be conservative, consistent with the guidance in SRP Section 6.2.1.1.A concerning the use of conservative values to maximize the calculated containment pressure. If the gap could be greater than 0.051 cm (20-mil), over the lifetime of the plant, then the calculated peak pressure will increase. In previous submittals, Westinghouse used a 0.013 cm (5-mil) gap. In response to RAI 480.636, Westinghouse provided the results of analyses for postulated air gaps up to 0.318 cm (125-mil). By increasing the gap from 0.013 cm (5-mil) to 0.318 cm (125-mil), the reference LOCA peak pressure would increase 9.7 kPa (1.4 psid) and the reference MSLB peak pressure would increase 2.8 kPa (0.4 psid).

  The steel-jacket to concrete air gap has been further addressed by Westinghouse in response to RAI 640.155 (Letter NSD-NRC-98-5566, dated February 10, 1998). Based on the evaluation presented, the 0.051 cm (20 mil) gap is considered by Westinghouse to be a conservative maximum value. The staff requested additional information on the subject. It was not clear that the construction technique used for the modular AP600 had been fully assessed. If the steel-jacket provides a leak tight boundary around the concrete, then it may be possible for moisture in the concrete to vaporize during a postulated DBA and the resulting pressurization may lead to deformation of the steel-jacket and the resulting average gap may be greater than the value used in the DBA analyses.

  In Revision 1 to RAI 640.155 (Letter NSD-NRC-98-5567, dated April 14, 1998), Westinghouse provided a conservative assessment of the potential pressurization in the air gap from the moisture in the concrete assuming a leak tight boundary. During the time period through the peak pressure for the limiting LOCA, the differential pressure acting on the steel jacket would result in movement toward the concrete (the external containment pressure remains higher than the internal steel-jacket concrete region pressure) and the assessment indicates that the air gap would not increase. For the MSLB, the temperatures in the steel-jacket concrete region remain low enough to exclude this as an issue. The staff therefore accepts the 0.051 cm (20-mil) air gap for licensing analyses as a conservative value.

- The loss coefficient in the external annulus includes a 30 percent increase over the value derived from the test program. In response to staff concerns with the upper annulus drains, a design change was incorporated to move the drains about one foot above the upper annulus drain floor. This effectively shortens the downcomer to riser turning region from about 1.83 m (6 ft) to 1.52 m (5 ft), once sufficient PCS water fills the region to the new drain elevation. In response to RAI 720.440F (Letter dated January 28, 1998), Westinghouse determined that the geometry changes do not impact the WGOTHIC modeling of the AP600 for design-basis accident analyses, and the data from the test (as modified) are still valid for the AP600 evaluation model. The existence of this pool of water will not significantly effects the already low velocity in the air entrance region. The possible effect of the cold water pool on the air density was shown to have a negligible effect on the buoyancy driven air flow.

- 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 calculations provided in support of the design certification are summarized in Table 6.2-5.

The estimated conservatism introduced into the WGOTHIC 4.2 SSAR analyses was evaluated by Westinghouse in Section 10 of WCAP-14407 (the analyses presented in WCAP-14407 are slightly different from the final SSAR analyses). The peak containment pressures were shown to be lower in the nominal case. The results are summarized in Table 6.2-6.

The staff compared the short-term calculated peak containment internal pressures for the AP600 with the SSAR reported pressures from other Westinghouse designs early in the review, based on the WGOTHIC 1.0 analyses. This comparison indicated that the AP600 results were reasonable relative to previous Westinghouse designs and the pressure trends were similar. Westinghouse had not attempted to demonstrate the expected robust nature of this new design by performing a standard COCO calculation to show how the margins in this new design compare with the margins in past designs. Furthermore, the staff had not yet performed independent confirmatory analysis with the CONTAIN computer code. WGOTHIC computer code verification and validation (V&V) was still being conducted by Westinghouse and was concurrently under staff review. This included Westinghouse's application of the experimental database to code V&V, and the staff's review. This was Open Item 6.2.1.1-6.

Subsequently, the staff performed independent confirmatory analysis with the CONTAIN computer code. These analyses indicate similar characteristics for the PCS performance for both the limiting LOCA and limiting MSLB events. The WGOTHIC 4.2 computer program V&V has been completed by Westinghouse and the staff review is presented in Section 21.6.5 of this report. This includes Westinghouse's application of the experimental database to code V&V. Open Item 6.2.1.1-6 is therefore considered to be closed.

Westinghouse informed the staff, during a telephone call on April 17, 1998, that changes were to be made to the AP600 PCS evaluation model to address ITAAC concerns with the verification of

Table 6.2-5 WGOTHIC Comparisons

Case Description	Pressure [kPa (psig)] and margin[1] to design: WGOTHIC 1.0	WGOTHIC 1.2	WGOTHIC 4.2 (SSAR)	Temperature in break node [°C (°F)]: WGOTHIC 1.0	WGOTHIC 1.2	WGOTHIC 4.2 (SSAR)
LOCA double-ended, cold-leg guillotine	373.7 (39.5) 9.2 %	353.0 (36.5) 14.2 %	400.6 (43.4) 2.7 %	139.4 (283.0)	Graph showed little change in peak	138.4 (281.2)
MSLB, 1.388 ft², full DER, 102% power, MSIV failure	385.4 (41.2) 6.4 %	338.5 (34.4) 17.8 %	402.6 (43.7) 2.2 %	160.2 (320.3)	160.7 (321.2)	188.3 (370.9)
MSLB, 1.388 ft², full DER, 30% power, MSIV failure	386.8 (41.4) 6.0 %	345.4 (35.4) 16.1 %	405.4 (44.1) 1.5 %	151.7 (305.1)	160.1 (320.1)	186.8 (368.2)

Note: 1. Margin (in % to design) is computed on the basis of absolute pressure

Table 6.2-6 Peak Containment Pressures

Parameter	**DECLG LOCA**		**MSLB 30 % power**
	Blowdown	Post blowdown	
AP600 design pressure	411.6 kPa [45 psig]	411.6 kPa [45 psig]	411.6 kPa [45 psig]
Reference[1] peak pressure	337.8 kPa [34.3 psig]	404.0 kPa [43.9 psia]	410.2 [44.8 psig]
Margin[2] reference[1] to design	17.9 %	1.8 %	0.3 %
Nominal peak pressure	324.7 kPa [32.4 psig]	312.3 kPa [30.6 psig]	376.4 kPa [39.9 psig]
Margin[2] nominal to design	21.1 %	24.1 %	8.5 %

Notes: 1. The WCAP-14407 analyses differ slightly from the final SSAR analyses.
2. Margin (in % to design) is computed on the basis of absolute pressure

heat sinks used in DBA licensing analyses for design certification. The evaluation model had included numerous small heat sinks such as stairs, gratings and platforms. Westinghouse refers to these as miscellaneous heat sinks. These miscellaneous heat sinks are not identified in the ITAAC.

Revision 3 to WCAP-14407, "WGOTHIC Application to AP600," was provided by Westinghouse letter DCP/NRC1355 dated May 1, 1998, which included a new Appendix 4B that identified those miscellaneous heat sinks that were removed from the model. Appendix 4B also included additional changes to remove existing conservatism in some of the heat sinks remaining in the model and included other changes in volumes and flow path characteristics to be consistent with AP600 document 1100-S0C-001, Revision 7, "Containment Volumes and Heat Sinks." The method used to remove the miscellaneous heat sinks is to turn off heat transfer by user input, not by actual removal of the information from the WGOTHIC input data deck. This is acceptable to the staff. However, as stipulated in Section 21.6.5.8.3, the COL applicant will be required to verify that, for future licensing analyses, the miscellaneous heat sinks have been removed by proper user input.

Table 4B-1 in WCAP-14407 identifies the ITAAC section for the containment shell, for the concrete structures and steel framing, and for the major equipment inside containment that are modeled as metal heat sinks. In a table attached to DCP/NRC1355, "Table of Internal Metal Heat Sinks," additional references to appropriate ITAAC sections are provided for internal metal heat sinks that might not be considered as major equipment. This table is not consistent with the information provided in Section 4 and Appendix 4B of WCAP-14407. The node numbers and heat sink numbers in that table relate to Westinghouse internal calculation files (1100-S0C-001), not necessarily the node and heat sink identifications used in WCAP-14407. Table 4B-1 and the table attached to DCP/NRC1335 do, however, provide sufficient information to determine that the heat sinks included in the AP600 evaluation model will be verified as part of the ITAAC.

The staff has allowed Westinghouse to move the heat sink information from the SSAR (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 4 of WCAP-14407, which is considered to be fully proprietary to Westinghouse Electric Company. The format and content of these data are different from that used in CONTEMPT, which was the previous basis for requiring these data in the SSAR. A reviewer of the WCAP-14407 material must exercise caution when using the information. The material presented in Section 4 must be integrated with the additional material presented in Appendix 4B to develop a clear understanding of the AP600 evaluation model.

Therefore, FSER Open Item 6.2-1, as identified in "Advance Final Safety Evaluation Report Related to the Certification of the AP600 Design," dated May 1998 (Docket No. 50-003), related to the verification of the internal containment heat sinks which were used in the WGOTHIC design-basis accident analyses for peak containment pressure in support of design certification, based on the Westinghouse commitment of April 17, 1998 and the subsequent May 1, 1998 submittal of WCAP-14407, Revision 3, is closed.

Summary of Staff CONTEMPT/CONTAIN Analyses

The staff conducted a limited independent analysis of the AP600 containment design with respect to pressure and temperature response to design basis LOCA and MSLB events (NRC NUREG-1632, "Evaluation of AP600 Containment Thermal-hydraulic Performance," June 1998). Normally, in the case of conventional containment designs, the staff uses the CONTEMPT code to estimate peak pressures and temperatures. CONTEMPT is well suited for this because DBAs in conventional containments are limited to the effects of a relatively brief high-energy blowdown, followed by a quick termination of the transient through the application of active safety systems (e.g., pumps, fans, sprays). The well-mixed containment atmosphere, as well as the engineered safety systems, are modeled adequately by a lumped parameter representation within the code.

Although the AP600 is similar to conventional large dry containments, it has some notable differences. It relies on external cooling of the containment with the PCS, and it does not use any active safety systems. CONTEMPT is not capable of direct modeling of the externally applied PCS. Hence, it was necessary to use the CONTAIN code to supplement CONTEMPT calculations.

The CONTAIN code is a multicell lumped parameter code equipped with a film tracking algorithm that is well suited to modeling the PCS. To establish continuity of methodology, a comparison was made of the two codes. The results for a DECLG indicate that the two codes yield comparable results for a single cell representation of the AP600 containment atmosphere. Furthermore, reasonable agreement was obtained with the CONTAIN code between single and multicell models of the AP600. On this basis, the CONTAIN code was used to do selected confirmatory analysis of the AP600 EM. The results show good agreement with Westinghouse WGOTHIC results. Specifically for a DECLG break, the peak pressure calculated by CONTAIN is 398 kPa (43.2 psig), which is about 7 kPa (0.8 psi) less than the WGOTHIC results shown in Table 6.2-4. Similarly, the corresponding temperature was calculated as 133.2 °C (271.5 °F), which is about 4.7 °C (8.8 °F) less than the WGOTHIC value.

The results obtained with CONTEMPT/CONTAIN codes, just as those obtained by Westinghouse using the WGOTHIC code, are founded on the assumption of a well mixed containment atmosphere. This is a reasonable assumption in the short term, so that the observed agreement indicates that the results of Westinghouse analyses for the EM appear to be reasonable.

It should be noted, however, that postulated AP600 accident transients are comparatively long (measured in hours rather than minutes), and may provide opportunities for establishing flow patterns that are not necessarily representative of a well-mixed containment atmosphere. Hence, long term pressure and temperature estimates using lumped parameter codes will be subject to uncertainties stemming from potentially stratified atmospheric conditions within the containment. However, these uncertainties are not expected to impact the containment design limits as long as the PCS continues to be available and since long-term mass and energy sources are a small fraction of what is present during the short term blowdown.

The staff performed additional analyses in the form of sensitivity calculations using the CONTAIN code to gauge the relative importance of some of the key AP600 containment design parameters. The sensitivity analyses indicate that the containment heat sink areas and PCS flow rate are the most controlling parameters with respect to containment pressure and temperature.

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 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 siting evaluation is performed with a constant, design-basis leak rate. Westinghouse had originally proposed that the pressure reduction be on the basis of 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 of long-term containment pressure resulting from the design-basis LOCA, including the 2-D correction, that appears to demonstrate the desired result, that the long-term (post 24-hour) pressure remains below 50 percent of the design pressure. (See Figure 6.2.1.1-7 of the SSAR.) This analysis is for the cold-leg break LOCA. The same 24-hour analysis should be performed for the entire spectrum of LOCA and MSLB events, or a rationale should be given for why such calculations are not necessary. A table that demonstrates compliance with the GDC 38 requirement should be included in the SSAR. This was Open Item 6.2.1.1-7.

The response for the limiting hot-leg break LOCA is provided in SSAR 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 SSAR Figure 6.2.1.1-1. This limiting case yields a peak containment pressure of 405.4 kPa (44.1 psig) at approximately 570 seconds into the event. 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. Table 6.2.1.1-3 in the SSAR provides the calculated pressure for the most limiting DBA. This table, therefore, 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. Therefore, Open Item 6.2.1.1-7 is closed.

The staff considers the AP600 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. While the PCS flow rate is maintained at 1,666 liters/min (440 gpm) for the first three hours of the event, the performance envelope is similar to existing designs that use active safety systems. When the PCS flow rate is reduced after three hours, there is a tendency to slightly repressurize and maintain a pressure somewhere between 218.5 and 273.7 kPa (17 to 25 psig), well below the 411.6 kPa (45 psig) design value, until 30 hours into the event when a further reduction in the PCS flow rate occurs. After 30 hours, there is again a repressurization tendency with the resulting pressure being maintained at between 225.4 and 322.0 kPa (18 to 32 psig) for the remainder of the three-day design-basis performance period of the PCS. The difference between the low- and high-pressure estimates are determined by the credit given in the analyses to consider the effects of 2-D heat conduction (between wet and dry regions of the containment shell) when less than full coverage of the containment shell is expected. As discussed in Section 21.6.5, 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 AP600 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. Specifically, the pressure evaluation was conducted using the WGOTHIC code, and assumes that all ac power sources are 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, in order to maximize cooling of the containment atmosphere and thus 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 thereby maximize the pressure differential across the containment vessel.

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

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

At one hour after the event, the calculated differential pressure across the containment vessel is approximately 13.8 kPa (2 psid), versus the design external pressure of 20.7 kPa (3.0 psid). To mitigate the event, Westinghouse indicates in SSAR 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 indicates in the SSAR that operators would have sufficient time to restore the pressure before reaching the design external pressure limit.

The staff notes that because the AP600 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. In its RAIs, the staff asked Westinghouse to identify other events that were considered, and to justify why the loss of all ac with maximum ambient cooling is bounding for the maximum external pressure. In its response to RAI 480.1043, Westinghouse indicated that events involving inadvertent PCCS 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 also considered, but were found by Westinghouse not to be bounding.

In particular, the staff asked Westinghouse why the event of an inadvertent actuation of the PCCS with the containment fan coolers in operation would not be considered bounding. This scenario is addressed in Section 2.2.12 of WCAP-14477, "The AP600 Adverse System Interactions Evaluation Report," Revision 1. 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. In response to RAI 480.1043, Westinghouse indicated operation of the containment fan coolers is limited by the minimum temperature (4.4 °C (40 °F)) of the chilled water system. Westinghouse further indicated that the maximum heat transfer from containment for the external pressure transient was chosen without PCCS operation because the heated water within the PCCS 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 3.5 hours when compared to the extreme cold temperature. The staff finds this reasoning acceptable, and 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 containment spray system is described in Section 6.5.2 of the SSAR and the staff's evaluation of the system is found in Section 19.2.3.3.9 of this report. The staff requested that Westinghouse consider the effect of an inadvertent spray actuation on the maximum external pressure analysis.

As noted in SSAR subsection 6.5.2.1.4, the use of the containment spray during power operation requires the opening of two manual valves, including a locked closed valve outside of containment, and a remotely operated valve inside containment, from the MCR or remote access workstation. Because of the isolation valves described above, the staff finds inadvertent actuation of the containment spray system during power operations to be not credible.

In response to RAI 480.1081F, Westinghouse stated that 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. Because of this design configuration, the staff finds inadvertent actuation of the containment spray system during shutdown operations to be not credible.

Westinghouse presented a design capability of 17.24 kPa (2.5 psid) Service Level A, and 20.7 kPa (3.0 psid) Service Level C for the AP600 containment. However, it did not present the supporting analyses to show that these limits are not exceeded, nor did it provide a detailed description of the event leading to the limiting external pressure. The SSAR did not include calculations for the most limiting peak external differential pressure under DBA and severe accident conditions. In the SSAR, Westinghouse stated that the external pressure condition was combined with dead and live loads during normal operation; however, it was not clear whether this also included the loads associated with DBA or severe accident conditions. This was Open Item 6.2.1.1-8.

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 indicated that operators would be able to restore containment pressure before the reverse differential pressure design limit is reached, thereby 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. Open Item 6.2.1.1-8 is closed.

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 Section 6.2.1.2 of the SSAR, Westinghouse has applied the leak-before-break (LBB) concept to RCS high energy piping with a diameter of 15.24 cm (6 in.) 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.

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. At the time the DSER was issued, the application of LBB to AP600 was still under review by the staff. This was DSER Open Item 6.2.1.2-4. Westinghouse has subsequently submitted analyses that demonstrate the validity of LBB to the AP600. The staff's evaluation and acceptance of LBB for the AP600 is discussed in Section 3.6.3 of this report. On the basis of the acceptance of LBB for the AP600, Open Item 6.2.1.2-4 is closed.

Table 6.2-7 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. In the staff's DSER, Open Item 6.2.1.2-1 stated that it was not clear which subcompartment walls were analyzed, and Open Item 6.2.1.2-3 stated that the design pressure of the walls was not made clear. On the basis of the above table, Open Items 6.2.1.2-1 and 6.2.1.2-3 are closed.

In the DSER, the staff questioned why the IRWST and reactor vessel cavity were not analyzed for a 7.62 cm (3 in.) break. As described in Section 3.6.1.2 of the SSAR, 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). Furthermore, the reactor vessel cavity was analyzed for asymmetric pressurization resulting from a five-gpm leak rate crack in the primary piping. Westinghouse did not specifically state the reasons for not applying the 7.62 cm (3 in.) double-ended guillotine (DEG) break as was done with the other subcompartments. This was Open Item 6.2.1.2-2.

In SSAR Section 6.2.1.2.1.1, Westinghouse indicated that the reactor vessel cavity was not analyzed for asymmetric loading caused by vessel pressurization because all of the piping in the reactor vessel cavity is qualified to LBB. To ensure that no breaks were excluded from analysis, the staff considered explicitly whether LBB also applied to the weld joining the RCS piping in the vessel cavity and the "safe-ends," or nozzles, attached to the reactor vessel. Because the staff's acceptance of LBB in Section 3.6.3 of this report encompasses pipe welds, breaks at weld locations do not need to be postulated for LBB piping for the purpose of the subcompartment pressurization analysis.

Table 6.2-7 Postulated Breaks and Subcompartment Design Pressures

Subcompartment	Postulated Break	Subcompartment Design Pressure (psid)
Steam generator compartment and access area	10.16 cm (4 in.) RCS line, 7.62 cm (3 in.) RCS line, 10.16 cm SG blowdown line, and 10.16 cm pressurizer spray line	5
Pressurizer valve room	10.16 cm RCS line	5
CVS room	7.62 cm RCS line	5
CVS pipe tunnel	10.16 cm SG blowdown line	7.5
Maintenance floor and operating compartment walls	0.093 sq. m (1 sq. ft.) main steamline rupture	5

With regard to the IRWST, Westinghouse indicated in its response to RAI 480.1039, that the sparger loads from ADS operation are the bounding loads. On the basis of the aforementioned information concerning the reactor cavity and IRWST, Open Item 6.2.1.2-2 concerning loading analyses for the IRWST and reactor cavity is closed. 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, except for the 5.08 cm (2 in.) pressurizer spray line break in the pressurizer compartment, for which the NOTRUMP code was used because of the relatively small size of that piping, and because it allowed the releases from both sides of the break to be obtained. 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. For all breaks, 10 percent was added to the mass and energy releases to maximize the differential pressure.

In RAI 480.1050, the staff asked Westinghouse why NOTRUMP, instead of SATAN-VI, was used for certain breaks. In its response, Westinghouse stated that 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. Also for the case of NOTRUMP, the piping model does not include friction losses, the exclusion of which results in a higher pressure at the break and thus a greater mass release. Since NOTRUMP more accurately models the AP600 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 to calculate the differential pressure across the subcompartment walls (Reference 2). 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 aforementioned 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.

Westinghouse has noted that several of the subcompartments do not meet the 40 percent pressure margin specified in SRP 6.2.1.2. However, substantial margin still exists. The staff has determined that the few exceptions to the 40 percent margin are acceptable based on Westinghouse's ITAAC commitment to perform the subcompartment analyses using as-built data.

The staff reviewed the short-term mass and energy release data, and the methodology as it applies to the AP600. This was Open Item 6.2.1.3-1.

The staff also 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, "Westinghouse Mass and Energy Release for Containment Design." Furthermore, SATAN-VI has been found acceptable through the staff's review of WCAP-10325, "Westinghouse LOCA Mass and Energy Release Model for Containment" (Reference 1), and NOTRUMP has been found acceptable for use in currently licensed plants for small line breaks as discussed in Section 21.6.2 of this report. This closes Open Item 6.2.1.3-1.

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 AP600, as well as the modeling assumptions made by Westinghouse. This was Open Item 6.2.1.2-5.

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 AP600 is not an ice condenser containment, the staff has previously found TMD acceptable for non-ice condenser operating plants.

The staff finds the aforementioned correlations, computer codes, and methodologies acceptable for the AP600 subcompartment pressurization analysis. Open Item 6.2.1.2-5 regarding the acceptability of SATAN-VI and TMD to the AP600 is now closed.

In conclusion, the staff finds that Westinghouse has satisfied GDC 4 with regard to containment subcompartments by considering the dynamic effects of postulated pipe ruptures within subcompartments. Consistent with GDC 4, Westinghouse has shown, by analysis, that pipe breaks above a certain size can be precluded from that piping for which breaks must be postulated. Furthermore, Westinghouse has satisfied GDC 50 by designing containment subcompartment walls to withstand, with appropriate margin, the calculated differential pressures resulting from pipe breaks postulated in accordance GDC 4. Therefore, the staff finds 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, including subcompartment differential pressure analysis and 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 evaluation in Section 6.2.1.2 of this report provided the basis for closure of Open Item 6.2.1.3-1.

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 DEG severance of the largest RCS pipe. The release rates were calculated for pipe failure at two locations (i.e., 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 PCS 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 AP600 long-term LOCA mass and energy releases were predicted for the blowdown phase for postulated DECLG and DEHLG 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 AP600. This was Open Item 6.2.1.3-2.

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 AP600 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 AP600, for LOCA analyses, the break location switches to the fourth-stage ADS at about 1,000 seconds into the limiting LOCA scenario. Open Item 6.2.1.3-2 is, therefore, closed.

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

As noted in Section 6.2.1.1 of this chapter, the energy release from the zirconium-water reaction has been included as an energy source for the WGOTHIC 4.2 SSAR analyses.

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 (+1.4%)
- allowance in volume for thermal expansion (+1.6%)
- 100% full power operation
- allowance for calorimetric error (+2.0% 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%)
- allowance in initial pressure to account for instrument error and dead band
- margin in steam generator mass inventory (+10.0%)
- 1% 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. This was Open Item 6.2.1.3-3. 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. Open Item 6.2.1.3-3 is, therefore, closed.

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 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 reactor coolant system 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 reactor coolant system mass addition rate from the sum of the accumulator, CMT, and IRWST flow rates.

The staff expressed concern with the use of SATAN-VI for the AP600 (RAI 480.945) because SATAN-VI is not part of the large break LOCA code package submitted by Westinghouse for AP600 large break LOCA analyses, and therefore had not been reviewed by the staff for this application. Therefore, if Westinghouse was to use SATAN-VI to calculate mass and energy releases to the containment during an AP600 large break LOCA, Westinghouse needed to provide adequate justification to demonstrate that SATAN-V I gives conservative results and demonstrate that SATAN-VI is capable of modeling AP600 components (e.g., spherical accumulators, CMTs, 2 x 4 loop layout) and AP600 plant response.

In response to the RAI (Letter NSD-NRC-98-5530, "Revised Response to RAI 480.945," dated January 20, 1998), Westinghouse provided the requested justification.

Westinghouse stated that the variable noding structure of the SATAN model allows the user to simulate current and advanced reactor coolant systems geometries with generalized control volumes. The standard Westinghouse PWR reactor coolant system noding was modified to specifically model the AP600 reactor coolant system geometry. This modeling included two cold legs in the broken loop and the direct vessel injection (DVI) line to the downcomer.

The primary differences between the AP600 design and current operating Westinghouse PWRs are the engineered safety features. The safety features of current operating plants include passive and active systems while the AP600 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 AP600 and is modeled with the NRC-approved LOCA mass and energy release methodology. The AP600 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 core makeup tanks (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 AP600 passive systems.

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 AP600. The behavior of the release of the initial RCS inventory during the initial blowdown for the AP600 is identical to current operating plants. The flexibility of the noding structure in a SATAN-VI model allows for an accurate representation of the AP600 geometry.

Therefore, the SATAN-VI code is acceptable for predicting the mass and energy releases during the blowdown phase for the AP600 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 AP600. This was Open Item 6.2.1.3-4.

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. Therefore, Open Item 6.2.1.3-4 is closed.

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 AP600 mass and energy release calculations conservatively do not assume a single failure. The effects of a single failure in the PCS are taken into account in the containment analysis.

When the DSER was written, it was not clear whether passive system reliability, as identified in Open Item 6.2.1.3-5, was an issue. Passive system reliability and its application to the AP600 was reviewed by the staff, as discussed in Section 22.5.4. Open Item 6.2.1.3-5 is closed.

Metal-Water Reaction

Westinghouse did not consider the metal-water reaction as an energy source in calculating the mass and energy releases to containment. Westinghouse's justification for this was that the calculated fuel temperatures were low enough to preclude zirconium water reaction.

According to Section 6.2.1.3 of the SRP and GDC 50, this energy should have been considered in the calculations. The treatment of metal-water reaction energy as an energy source was discussed with Westinghouse. This was Open Item 6.2.1.3-6.

Westinghouse now considers the metal-water reaction as an energy source in calculating the energy releases to containment. Open Item 6.2.1.3-6 is therefore considered to be closed.

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 steam generator 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 plant, increased heat transfer in the steam generators, 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 steam generators 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, to determine the limiting break case for peak containment pressure, includes a spectrum of four breaks at each of four initial power levels, resulting in a total of sixteen cases. Included are three double-ended ruptures and one split rupture, as follows:

- 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 steam generator with the broken line is controlled by the flow restrictor throat area (0.1 m^2 (1.39 ft^2) nominal). The reverse flow from the intact steam generator 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 steam generator. The reverse flow has been conservatively assumed to be controlled by the flow restrictor in the intact loop steam generator.

- An intermediate size double-ended rupture having an area of 0.037 m^2 (0.4 ft^2).

- A small double-ended rupture having an area of 0.01 m^2 (0.1 ft^2).

- A split rupture representing the largest break which can neither generate a steamline nor a feedwater isolation signal from the primary protection equipment. Steam and feedwater line isolation signals are generated by high containment pressure signals for this type of break.

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.

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 reactor coolant system to the steam generator.

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

- The AP600 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 steam generator secondary system and causes a reduction of mass and energy releases via the break.

Thus, 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 main steam isolation valve.

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, a value of 8 percent revaporization is assumed for all MSLB transients analyzed.

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 405.4 kPa (44.1 psig) at approximately 570 seconds into the event. 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.

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

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

(1) steam generator 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
- steam generator 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 AP600. For example, Westinghouse had not clarified how non-safety system operation could affect the postulated mass and energy releases. This was Open Item 6.2.1.4-1.

Westinghouse has clarified how non-safety system operation could affect the postulated mass and energy releases and determined that continued ac power would be limiting. Open Item 6.2.1.4-1 is, therefore, closed.

Description of Blowdown Model and Mass and Energy Release Data

In this AP600 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 stream line breaks (Carlin, E. L. and U. Bachrach, "LOFTRAN and LOFTTR2 AP600 Code Applicability Document," WCAP-14234, Revision 1 (Proprietary), June 1997).

During the staff's initial reviewed the application of these methodologies to the AP600, it was not clear that the above cited methodologies reflected current technology by including the effect of steam generator superheat. This was Open Item 6.2.1.4-2.

The staff completed its review of the application of these methodologies to the AP600 and found them to be acceptable for licensing analyses. The above cited methodologies reflect current technology by including the effect of steam generator superheat. Therefore, Open Item 6.2.1.4-2 is closed.

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 Emergency Core Cooling Systems

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 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

Section 6.2.1.5 of the SSAR 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 AP600 limiting case, large-break LOCA presented in Section 15.6.5 of the SSAR.

The calculated containment backpressure used by Westinghouse for the AP600 hot-leg and cold-leg guillotine and split breaks for the ECCS analysis is presented graphically in WCAP-14171-P. The "peak" minimized containment pressure is approximately 290 kPa (27.3 psig), as compared to the peak pressure of approximately 412 kPa (45 psig) calculated for containment design and leakage considerations.

As discussed in SSAR Section 6.2.1.5 and WCAP-14171-P, Revision 1, "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident", 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%

- 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. (This was clarified in the Westinghouse response to RAI 480.1136F.)

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 50, and are described in WCAP-14171-P. 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 50.

BTP CSB 6-1 also states that the mixing of subcooled ECCS water from the break with steam atmosphere should be assumed to minimize the pressure. In RAI 480.1046, the staff asked Westinghouse if this assumption had been made. In its response, Westinghouse stated that all mass and energy released from the break during blowdown was assumed to mix with the containment atmosphere, and that spillage of ECCS water into the containment was 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. However, in Westinghouse's minimum backpressure analysis, the containment recirculation cooling system was not assumed to be operating. In RAI 480.1049, the staff asked Westinghouse to provide the rationale for not considering operation of the containment recirculation cooling units, since their operation could be expected to lower the containment pressure. In its response to RAI 480.1049, Westinghouse stated that at 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. Furthermore, Westinghouse indicated that 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.

In response to RAI 480.1045, Westinghouse indicated that PCCS flow was 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. The staff finds Westinghouse's responses to RAIs 480.1045 and 480.1049 acceptable. In addition, prior to actuation of the fourth stage of the ADS there is limited communication between the containment and the RCS. AP600 thermal-hydraulic analyses, performed for the staff using the TRAC 4 computer model, indicate that the actuated 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 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; these releases are acceptable on the basis of the staff's findings in Section 15.6.5 of this report.

The staff reviewed the application of a constant backpressure of 101 kPa (14.7 psia) in performance capability studies of the AP600 ECCS. This was Open Item 6.2.1.5-1.

Westinghouse presented the mass and energy releases to the containment during the blowdown and reflood portions of the limiting DECLG break transient at a Moody discharge coefficient of 0.8 (CD=0.8) in Table 6.2.1.5-1 of the SSAR, as computed by the WCOBRA/TRAC code. The staff reviewed the application of this methodology to the AP600. This was Open Item 6.2.1.5-2.

On the basis of the aforementioned considerations, the staff finds the minimum containment backpressure analysis acceptable. Therefore, Open Item 6.2.1.5-1 regarding the acceptability of the minimum backpressure analysis and Open Item 6.2.1.5-2 concerning the acceptability of the mass and energy releases are closed. However, closure of Open Items 6.2.1.5-1 and 2 applies to the containment backpressure analysis only. The acceptability of the credited backpressure has been evaluated in the overall context of the ECCS performance capability studies. The staff's evaluation concerning these studies are discussed 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 Section 6.2.1.7 of the SSAR. Testing and inservice inspection of the containment vessel is described in Section 3.8.2.6 of the SSAR, while isolation testing is described in Section 6.2.3 of the SSAR, and leak testing is described in Section 6.2.5 of the SSAR. The valves of the passive containment cooling system are periodically stroke tested, and Section 6.2.2 of the SSAR provides a description of the testing and inspection. Testing and inspection will be consistent with regulatory requirements and guidelines. For example, containment isolation valves will be reviewed against the requirements of Appendix J to 10 CFR Part 50, and GL 89-10.

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 Section 3.8.2 of the SSAR for the requirements for inservice inspection of the steel containment vessel. Section 6.2.2 of the SSAR 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 Section 5.2.5 of the SSAR.

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

<u>Preoperational Testing</u>

Preoperational testing of the passive containment cooling system 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 (SSAR Table 6.2.2-1):

1673.2 L/m (442 gpm) at the minimum operating water level
467.5 L/m (123.5 gpm) at a level after the first standpipe is uncovered
274.4 L/m (72.5 gpm) at a level after the second standpipe is uncovered

The containment PCS water coverage fraction (wetted surface area) will also be measured at the base of the upper annulus, in addition to the measurements at the spring line. A full flow test using the PCS water storage tank to deliver the flow will be performed. An additional test will be performed at a lower flow rate using the PCS recirculation pumps to deliver the flow. A throttle valve will be used to obtain the low flow rate (less than the full capacity of the PCS recirculation pumps). This flow rate will be 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 technical specifications. 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 SSAR 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 237.4 L/m (62.7 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 SSAR 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 62.7 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 62.7 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 50 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 400,000 gallons.

Additional details for preoperational testing of the passive containment cooling system are provided in SSAR Chapter 14, and discussed in Section 14.

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 SSAR Section 14.2.9.1.4, "Passive Containment Cooling System."

Operational Testing

Operational testing is performed to:

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

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

- 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 technical specifications (SSAR Section 16.3.6) and inservice testing program (SSAR 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 the following instrumentation to measure, record, and readout in the control room:

- 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 AP600 post-accident monitoring system is described in Chapter 7 of the SSAR, considering the recommendations in RG 1.97. Instrumentation to monitor RCS leakage into containment is described in Section 5.2.5 of the SSAR.

The containment pressure is measured by four independent pressure transmitters, and the signals are fed into the ESF actuation system, as described in Section 7.3.1 of the SSAR. Upon detection of high pressure inside the containment, the appropriate safety actuation signals are generated to actuate the necessary safety-related systems. 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. Section 11.5 of the SSAR provides information on the containment area radiation monitors, while the ESF actuation system operation is described in Section 7.3 of the SSAR.

The containment hydrogen concentration is measured by the hydrogen concentration monitoring subsystem (HCMS). The HCMS is described in Sections 6.2.4 and 7.5 of the SSAR and was evaluated by the staff in Section 6.2.5 of this report. The amount of time for the postaccident hydrogen monitoring system to become operable was Open Item 6.2.1.7-1. The response time of the sensor is at least 90 percent in 10 seconds. 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. As detailed in Section 1.9 of the SSAR, the hydrogen monitoring system is designed in compliance with the recommendations of NUREG-0737. Therefore, Open Item 6.2.1.7-1 is closed.

Table 7.5-1, "Post-Accident Monitoring System," of the SSAR contains the instrumentation provided to meet the guidance of RG 1.97. 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 108 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 AP600, 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 Debris in IRWST and Containment Sumps - Strainer Clogging

On December 3, 1985, the NRC issued GL 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage." This GL informed licensees of operating reactors, applicants for operating licenses, and holders of construction permits of the resolution of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance".

The technical concerns evaluated under USI A-43 are as follows:

- sump hydraulic performance under post-LOCA conditions resulting from potential vortex formation and air ingestion, and subsequent pump failure
- possible transport of large quantities of LOCA-generated insulation debris resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling
- capability of RHR and containment spray system (CSS) pumps to continue pumping when subjected to possible air, debris, or other effects, such as particulate ingestion on pump seal and bearing systems

GL 85-22 did not recommend any actions by the addressees, but did recommend that the findings of USI A-43 be applied to any changes to the thermal insulation used inside containment. The technical data and conclusions related to this topic were discussed in NUREG-0897 (Revision 1) and in Revision 1 to RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems."

Since that time, several significant events have occurred at operating plants, including the plugging of containment spray system suction strainers at the Barsebäck plant in Sweden, and the clogging of ECCS suction strainers at the Perry Nuclear Power Plant in Ohio and the Limerick plant in Pennsylvania. These incidents are discussed in NRC Bulletin 96-03. The Barsebäck event demonstrated the potential for a pipe break to generate insulation debris and transport a sufficient amount of this debris to the suppression pool to clog the ECCS strainers. Two events at the Perry Nuclear Power Plant demonstrated the deleterious effects on strainer pressure drop caused by the filtering of suppression pool particulates (corrosion products) by fibrous glass materials entrained on the ECCS strainer surfaces. The Limerick event demonstrated the need to ensure adequate suppression pool cleanliness. Corrosion products had combined with fibrous material to completely cover the suction strainer screens with a thin layer of "mat" which resulted in a greatly increased pressure drop across the screens. NRC Bulletin 96-03 provides guidance for the final resolution of this issue for BWRs. The Boiling-Water Reactor Owners Group (BWROG) has prepared the Utility Resolution Guidance report (NEDO-32686) and technical support documentation, dated November 20, 1996, to provide guidance on implementing this bulletin. As discussed below, some data from the BWROG Utility Resolution Guidance report was applied to the AP600 design.

The staff had originally proposed that the advanced designs should have the ability to back flush the suction strainers (that is, remove debris from the screens by applying a reverse pressure gradient), which is similar to the resolution taken in Sweden for the Barsebäck plant. However, *in evaluating the events mentioned above, the staff decided that an increase in the strainer size* was adequate. As a result, in the Section 6.2.1.9 of the FSER on the design certification of the Advanced Boiling Water Reactor Design (NUREG-1503), the staff stated that an acceptable resolution for the advanced designs would be to size the ECCS suction strainers in accordance with RG 1.82, Revision 1, but with a factor of three screen area margin. Since that time, understanding of the technical issues has advanced considerably as the result of testing and calculations done by both the NRC and the BWROG. It is the staff's view that the factor of three in margin is not necessary because of this better understanding of the technical issues. Instead, a more mechanistic approach that still provides sufficient conservatism is acceptable. This is the approach taken for the AP600 to provide assurance of the availability of emergency cooling when required.

Section 6.3 of the SSAR provides information on the operation of the passive core cooling system (including coolant recirculation following a LOCA) and a description of the debris screens. SSAR Appendix 1A describes the conformance of the sumps with RG 1.82.

Tests have shown that fibrous insulation deposited on suction screens has the potential, when deposited on a screen mesh, especially when combined with particulates, to significantly increase the pressure drop across the screens. Westinghouse has not provided the staff with the exact amount of fibrous insulation that will be present inside the containment. However, SSAR Section 6.3.2.2.7.1 states the following

> Metal reflective metallic insulation is used on lines subject to loss-of-coolant accidents that are not otherwise shielded from the blowdown jet. As a result, fibrous debris is not generated by loss-of-coolant accidents. 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 and 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.

This is acceptable to the staff. The distances from a reactor coolant system pipe to the nearest allowable locations of fibrous insulation are based on calculations performed for the NRC staff by Science and Engineering Associates, Inc. (SEA), and data taken from tests performed by the BWROG and described in the BWROG Utility Resolution Guidance report NEDO-32686, referenced above. A spherical destruction zone for insulation is permissible in regions in which shock waves from the pipe break will be reflected and attenuated by intervening structures. 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.

Testing performed by the staff and the BWROG as part of the resolution of the issue of ECCS strainer blockage in boiling water reactor (BWR) suppression pools shows that the choice of reflective metallic insulation (RMI) for use inside the AP600 containment will significantly reduce the amount of screen blockage in comparison with fibrous insulation. Screen head loss is caused primarily by the smaller RMI debris sizes resulting from the fragmentation of the inner reflective foils. It would be much less likely that the large internal foils, large pieces of intact foils, the intact RMI assemblies, end disks and cassette sheaths, side panels and other large pieces could be transported in sufficient amount to cause a significant head loss at the recirculation screens and, based on the flow path, could not be transported to the IRWST screens. In addition, the small RMI debris does not interact with particulate debris in the same way that fibrous debris does (the so-called "filtering" effect) to result in a large head loss across the screens.

The low transport velocities of the AP600 and the long (up to five hours) time before the recirculation mode is initiated will further limit the transport of RMI and any other potential debris to the recirculation screens.

There are two sets of screens included in the design of the AP600. These are the IRWST screens and the recirculation screens. The IRWST screens are vertical screens, each 6.5 m^2 (70 ft^2) in area, located inside the IRWST at the bottom of the tank. Two separate screens are provided at opposite ends of the tank. The SSAR states in Section 6.3.2.2.7.2 that the IRWST is closed off from containment and its vents and overflows are normally closed by louvers. Thus, it would be difficult for debris to enter the IRWST during normal operation. In addition, the IRWST is made of stainless steel and will not generate the type of corrosion products that caused problems in operating BWR suppression pools. Section 6.3.8.1 of the AP600 SSAR states that the COL applicant referencing the AP600 will address a program to limit the amount of debris that might be left in containment following refueling and maintenance outages. The staff finds this acceptable. This is COL Action Item 6.2.1.8-1. Technical specification 3.5.6.9 requires visual inspection of the IRWST and recirculation screens every 24 months to ensure that they are not restricted by debris. Technical specification 3.5.4.6 requires a visual inspection of the gutters (which are part of the containment water long-term return and recirculation system) every 24 months. During accident conditions there is a potential for introducing debris to the IRWST. However, for the reasons discussed above, the amount of reflective metallic insulation and debris introduced should be negligible and should not have an adverse effect on the head loss across the IRWST screens.

The containment recirculation screens are also vertically oriented, and each also has a flow area of 6.5 m^2 (70 ft^2). They meet the criteria in RG 1.82. In response to RAI 480.1079, Westinghouse provided the maximum flow and the water velocity at the recirculation screens and 3 m (10 ft) from these screens. These velocities are approximately an order-of-magnitude less than those typical of the ECCS sump screens inside the containments of operating PWRs, even with the RNS pumps operating. These lower flow velocities through the screens reduce the potential of drawing debris into the screens. In addition, when the recirculation lines initially open, the water level in the IRWST is higher than the containment and water flows from the IRWST backwards through the containment recirculation screen. This backflow tends to flush debris located close to the recirculation screens away from the screens.

The water level at the beginning of recirculation is well above the top of the recirculation screens. Thus, any floating debris will remain clear of the screens. Also, there is a two foot clearance between the floor and the bottom of the screen so that any high density debris, swept along the floor, will not block the recirculation screens.

The AP600 design has a non-safety-related containment spray system. Containment spray is capable of washing down insulation debris that might not otherwise be transported to the recirculation system. However, the AP600 containment spray system will be used only in the case of a severe accident. At this point, core heat removal or coolant has been lost and the containment spray's effect in transporting more debris is not significant.

The recirculation piping inlet is slightly above the compartment floor, which is substantially below the expected flood-up water level. This reduces the potential for air ingestion in the piping because recirculation does not initiate until the flood-up water level is well above the piping inlet.

The staff also discussed the issue of possible adverse effects of failed protective coatings inside containment on the recirculation screens with Westinghouse. Westinghouse takes the position that protective coatings below the operating deck do not have to be safety-related because their failure will not clog the recirculation screens and interfere with the recirculation function. In support of this position, Westinghouse performed calculations (reported in its letter dated February 10, 1998) that show that blockage of the recirculation screens by paint particles is unlikely because paint particles cannot be transported to a sufficient fraction of the area of the screen, assuming particle sizes and settling rates similar or more conservative than those previously accepted by the staff in a review of this issue for the Comanche Peak Steam Electric Station Units 1 and 2 (NUREG-0797, Supplement No. 9, dated March 1985). The containment recirculation screens are protected by plates located above them. These plates prevent debris from the failure of protective coatings from entering the water close to the screens, such that the recirculation flow can sweep the debris to the screens before it settles to the floor. To prevent plugging the screens, safety-related coatings are used on the underside of these plates and on the surfaces located below the plates, above the bottom of the screens, 3 m (10 ft) in front and 3 m (10 ft) to the side of the screens. The plates and their dimensions are covered by ITAAC 8.c (v). Another ITAAC, 8.c (vi) covers the use of safety-related coatings on the underside of the plates located above the recirculation screens, down to the bottom of the screens and within 3 m (10 ft) of the trash rack portion of the screen.

The staff completed its review of the AP600 design with respect to possible adverse effects of LOCA-generated debris on the IRWST and recirculation screens and other concerns addressed by USI A-43. The staff finds that these issues have been acceptably addressed.

6.2.2 Containment Heat Removal Systems

In accordance with GDC 38, the system employed by the AP600 to remove heat from the containment atmosphere under postulated DBA conditions is the PCS. As described in Section 6.2.2 of the SSAR, 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 Section 6.2.2 of the SSAR, 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. A passive design like the PCS has not been reviewed before by the staff.

The major components of the PCS are the primary containment vessel, which acts as the safety-grade interface to the ultimate heat sink, shield building, PCS water storage tank (PCCWST), air baffle, air inlets, air diffuser, and a water distribution system comprising a water distribution bucket and distribution weirs. The design of the shield building is fully 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 two 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 2.01E+06 liters (531,000 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 exposed 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 Section 6.2.2 of the SSAR, 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 AP600 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 approximately 20 m (70 ft), a height of 63.1 m (207 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.

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 Section 6.2.3 of the SSAR, Westinghouse provides a description of the containment isolation system. The AP600 has been designed to minimize the number of containment piping penetrations and has less than half the number of penetrations of typical operating plants. 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, 56, and 57. The staff reviewed Westinghouse's proposed use of containment isolation valves, as described in Table 6.2.3-1 of the SSAR, for conformance with these GDC. The staff reviewed the valve arrangement information for each penetration and confirmed that the number, location, and arrangement conform to the acceptance criteria. Table 6.2-2 of the SSAR 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 steam generator (secondary coolant system) piping is typical of PWRs and is acceptable based on credit for use of the secondary coolant system as an extension of the containment.

6.2.4.2 Actuation and Control Features for Isolation Valves

An SRP and TMI (Item II.E.4.2) requirement 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 AP600 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. Westinghouse confirmed that 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.

In the DSER, the staff identified Open Item 6.2.4.2-1 regarding the fact that the normal RHR (NRHR) system isolation instrumentation would not be provided with diverse parameter sensing. Westinghouse subsequently implemented a design change to provide diverse instrumentation for automatic isolation of this system. On the basis of this design change, Open Item 6.2.4.2-1 is closed.

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 AP600 containment isolation system (SSAR 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 Table 6.2-1 of the SSAR. 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 technical specifications.

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 SSAR Table 6.2-2:

- containment isolation signal (SSAR 7.3.1.2.1)

  A containment isolation signal is generated from any of the following monitored variables:

  - initiation of an automatic or manual safeguards actuation signal
  - manual containment isolation actuation
  - manual initiation of the PCCS signal

- safeguards actuation signal (SSAR 7.3.1.1)

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

  - low pressurizer pressure
  - low steamline pressure
  - low T_{COLD}
  - high containment pressure
  - manual initiation

- steamline isolation signal (SSAR 7.3.1.2.10)

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

  - containment high pressure

- low T_{COLD}
- low steamline pressure
- high steamline pressure negative rate
- manual initiation

- main feedwater isolation signal (SSAR 7.3.1.2.6)

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

  - automatic or manual safeguards signal initiation
  - manual initiation
  - high steam generator level
  - low T_{AVG} with P4 permissive
  - low–low T_{AVG} with P4 permissive

- startup feedwater isolation signal (SSAR 7.3.1.2.13)

  This signal occurs as the result of low T_{COLD} in any loop, or high steam generator (SG) narrow range water level in either SG.

- SG blowdown isolation signal (SSAR 7.3.1.2.11)

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

  - PRHR heat exchanger alignment signal
  - low narrow range steam generator water level

- containment High-2 radiation signal (SSAR 7.3.1.2.20)

  Automatic isolation of the NRHR system containment isolation valve is initiated by a containment "High-2" radiation signal. This signal is used in conjunction with a safeguards signal and provides diversity for NRHR system isolation.

- containment air filtration system isolation signal (SSAR 7.3.1.2.19)

  Automatic isolation of the containment air filtration system is initiated on containment "High-1" radiation level.

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

- diverse actuation system (DAS) signal (SSAR 7.7.1 11)

  The DAS is a non-safety-related instrumentation system that provides diverse backup to support risk goals. This system utilizes separate sensors and uninterruptible power supplies to initiate closure of certain containment isolation valves on containment high temperature conditions.

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 an operating license should use pressure history data from similar plants that have operated more than one year. if possible, to arrive at a minimum containment setpoint pressure.

Westinghouse has specified a containment isolation actuation pressure of ≤ 8 psig for the AP600 technical specifications. This setpoint was used in all applicable DBA analyses. The actual setpoint will be determined when the specific instrumentation is procured and installed in the lead AP600 plant. In view of the lack of pressure-history data from similar plants and compliance with all DBA analyses, the ≤ 8 psig specified in the TS is considered acceptable for initial operation of the lead plant.

TMI Item II.E.4.2 states that containment purge and vent isolation valves must close on a high radiation signal. The AP600 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 AP600 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 AP600 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 AP600 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 AP600 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 Table 6.2-1 of the SSAR.

The personnel airlocks 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.

In Open Item 6.2.4.6-1 of the DSER, the staff indicated that the spare penetrations should have redundant barriers or the single barrier should be welded. This open item has been resolved on the basis that the single blind flange in each penetration will be physically located inside the containment. This assures a high degree of reliability and leaktightness, eliminating the need for a redundant closure device. Therefore, Open Item 6.2.4.6-1 is closed.

6.2.4.7 Use of Closed Systems as Isolation Barriers

The steam generator secondary side, as bounded by the main steam, feedwater, and blowdown isolation valves, is a closed system inside containment that serves as an extension of the containment. This feature eliminates the need for inboard containment isolation valves in the steam, feed, and blowdown lines because the steam generator tubes and tubesheet and secondary system piping actually serve as a containment boundary. The steam generator 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. Westinghouse has not identified other instances of the use of closed systems as containment isolation barriers.

6.2.4.8 Protection of Containment Isolation Systems Against Loss of Function As a Results 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 AP600 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 Technical Specifications 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 (TV&D) connections are provided at suitable locations. See Section 6.2.6 of this report for the staff's evaluation of the AP600 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 20-second closure time. This closure time is consistent with assumptions and criteria for radiological dose analyses and ECCS analysis (reflood backpressure) assumptions use in Chapter 15 of the SSAR. 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, ALARA or air quality considerations for personnel entry, or for technical specification surveillances. This was DSER Open Item 6.2.4.13-1. This item is resolved by purging limitations established in the TS (SSAR Chapter 16, SR 3.6.3.1). Therefore, Open Item 6.2.4.13-1 is closed.

6.2.4.16 Conclusion

The Staff 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

Combustible gas within the AP600 containment is controlled by the hydrogen recombination subsystem (HRS) and the hydrogen ignition subsystem (HIS). The HRS is designed to meet the requirements of GDC 41, 42, 43, and 10 CFR 50.44. These requirements define the design-basis case. For this case, there is an initial release of hydrogen caused by the reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel to a depth of 5.8E-03 mm (2.3E-04 in) with water and the hydrogen contained in the reactor coolant system. This initial hydrogen release to containment is not sufficient to approach the flammability limit of 4 volume percent. However, hydrogen generation continues because of radiolysis of water and the corrosion of materials in containment. The HRS is designed to prevent the hydrogen concentration from reaching the flammability limit.

The HIS is designed to meet the requirements of 10 CFR 50.34(f)(2)(ix) and the staff position on hydrogen control as described in SECY-93-087 and approved by the Commission in its, July 21, 1993, SRM. 10 CFR 50.34(f)(2)(ix) requires that a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal water reaction be provided. This requirement was promulgated to address the lessons learned from the accident at Three Mile Island. This type of accident is considered beyond the design basis and will be referred to as the severe accident case in this section. In the severe accident case, the hydrogen generation from the fuel-clad metal water reaction could be sufficiently rapid that it may not be possible to prevent the hydrogen concentration in the containment from exceeding the lower flammability limit. The HIS is designed to promote hydrogen burning soon after the lower flammability limit is reached in the vicinity of the igniter. Initiation of hydrogen burning at the lower level of hydrogen flammability will prevent combustion at higher hydrogen concentrations, and provides confidence that containment integrity can be maintained during hydrogen burns.

Hydrogen Recombination Subsystem

For the design-basis case, the HRS uses four safety-related passive autocatalytic recombiners (PARs) to prevent the hydrogen concentration inside containment from reaching the flammability limit. Two full-size PARs are installed above the operating deck at an approximate elevation of 49 m (162 feet) and 4 m (13 feet) inboard from the containment shell. The PARs are located away from potential high upflow regions, such as the direct plume above the loop compartment. A third partial PAR, which is a quarter of the full-size PAR, is located at one of the vent paths from the IRWST and is utilized to limit hydrogen released from within the IRWST. A fourth partial PAR is located in the CVCS compartment to limit the accumulation of hydrogen within the compartment as a result of radiolysis and corrosion within the compartment from a partially flooded condition following a design-basis LOCA.

In meeting the requirements of 10 CFR 50.44 and 50.46, as well as GDC 41, regarding the functional capability of the combustible gas control systems to ensure that containment integrity is maintained, combustible gas control systems should meet the redundancy and power source requirements for ESFs and should be designed to withstand a single active component failure. Contrary to this requirement, power to the active HRS was supplied by non-safety-related sources. This was Open Item 6.2.5.2-1.

PARs use palladium or platinum as a catalyst to combine hydrogen and oxygen molecules into water vapor. The PARs are passive in nature, with no moving parts, and are independent of the need for electrical power or any other support system. The PARs are safety-related equipment. They are seismic Category 1 and are qualified for the post-LOCA environment. The recombiners are self-actuated in the presence of the reactants (hydrogen and oxygen). Therefore, Open Item 6.2.5.2-1 related to the need for redundant, safety-related power supplies for the PARs, is closed.

The source term used for determining radiolysis production of hydrogen assumed a 100 percent release of the core gap inventory of iodine, cesium, and noble gases. This is equivalent to 3 percent of the core inventory. This source term is a deviation from the guidance of RG 1.7, which states that 100 percent of noble gases, 50 percent of iodines, and 1 percent of other nuclides are assumed to be released from the core. For defense-in-depth, the staff continues to believe in the design basis requirements of RG 1.7. Therefore, the staff found the percentage of core fission product inventory in the sump solution proposed by Westinghouse to be unacceptable. This was Open Item 6.2.5.2-3.

The HRS is designed in accordance with the recommendations of RG 1.7 as discussed in Appendix 1A of the SSAR. Table 1 of RG 1.7 defines conservative values and assumptions that may be used to evaluate the production of combustible gases following a LOCA. The assumptions used in calculating the hydrogen release to containment are listed in Table 6.2.4-4 of the SSAR and are consistent with Table 1 of RG 1.7. Therefore, DSER Open Item 6.2.5.2-3 is closed.

The hydrogen release to containment and the hydrogen production rate are graphically presented in Figures 6.2.4-3 and 6.2.4-4 of the SSAR. Using the cladding oxidation assumptions of RG 1.7 and a cladding outer diameter of 4.7 mm (0.187 in), the staff confirmed Westinghouse's estimate of the hydrogen produced by the reaction of zirconium to be 85 cubic meters (3,000 standard cubic feet). This hydrogen is assumed to be released to the containment atmosphere at the beginning of the accident.

The fission product decay energy used in the calculation of hydrogen and oxygen production from radiolysis of the emergency core cooling water and sump water is acceptable if it is equal to or more conservative than the decay energy model given in BTP 9-2 in SRP Section 9.2.5. Westinghouse used a different decay energy model so the staff requested a comparison of the model used by Westinghouse to the one used in the staff's confirmatory computer program, COGAP. Westinghouse provided the comparison in a June 11, 1997, letter. The comparison showed reasonable agreement between the two models.

The AP600 relies on natural circulation currents enhanced by the PCCS to inhibit stratification of the containment atmosphere. The physical mechanisms of natural circulation mixing that occur in the AP600 are discussed in Appendix 6A of the SSAR. Steam generated by decay heat can vent into the containment atmosphere in the form of a jet plume through the postulated break or the fourth stage of the ADS. The interaction of the plume with the ambient atmosphere can be described in terms of entrainment flow induced by the plume. Entrainment flow results in the mixing of ambient atmosphere with the steam flow in the plume. The plume will rise to the containment dome where the steam will be condensed on the inner surface of the containment shell and the resulting cooler, denser air will fall to the operating deck.

Westinghouse provided an estimate of the degree of mixing by calculating volumetric flow rates of gas entrained by a rising buoyant plume associated with steam generated by decay heat. The calculations were made on the basis of a steam production rate corresponding to decay heat at 1 hour and 24 hours into the accident. Entrainment flow rates were calculated using equations presented in an article by Peterson in Volume 37, Supplement 1, of the International Journal of Heat and Mass Transfer, entitled, "Scaling and Analysis of Mixing in Large Stratified Volumes." In the Westinghouse estimate, no credit was taken for cold plumes falling from the containment dome, which causes further circulation above the operating deck. Westinghouse estimated the circulation time constant at 1 hour to be 490 seconds and at 24 hours to be 670 seconds. Confirmatory calculations by the staff, using the same equations as Westinghouse, but with containment atmospheric conditions calculated by the staff, indicate that the estimates are reasonable.

Westinghouse has arranged containment structures to promote mixing via natural circulation. Two general characteristics have been incorporated into the design of the AP600 to promote mixing and eliminate dead-end compartments. The compartments below deck are large open volumes with relatively large interconnections, which promote mixing throughout the below deck region. All compartments below deck are provided with openings through the top of the compartment to eliminate the potential for a dead pocket of high-hydrogen concentration.

In meeting the requirements of 10 CFR 50.44 to provide the capability for ensuring a mixed atmosphere in the containment, and the requirements of GDC 41 to provide systems as necessary to ensure that containment integrity is maintained, a system should be provided to mix the combustible gases within the containment. An analysis should be presented that shows that excessive stratification of combustible gases will not occur within the containment or within a containment subcompartment. The containment internal structures should have design features that promote the free circulation of the atmosphere. An analysis of the effectiveness of these features for convective mixing should be presented. This analysis is acceptable if it can be shown that combustible gases will not accumulate within a compartment or cubicle to form a combustible mixture. Contrary to this requirement, Westinghouse had not provided this analysis. This was identified as DSER Open Item 6.2.5.2-2.

The accumulator and CVCS compartments and the reactor cavity, including the reactor coolant drain tank room, do not participate in the natural circulation flow as they are dead-ended or filled with water. The IRWST compartment is essentially sealed at the vents by flappers after blowdown. The CVCS and IRWST compartments are included as confined volumes that may have water pools that provide a source of hydrogen. Therefore, each volume has been provided with a quarter-size PAR. The other compartments are either completely water-filled or do not contain a significant pool of water for hydrogen generation. The staff finds this response to be acceptable, and therefore, DSER Open Item 6.2.5.2-2 is closed.

The staff finds the Westinghouse assumption that the fission products and hydrogen released to the containment following a postulated design-basis LOCA are homogeneously distributed in the containment atmosphere within the open compartments that participate in natural circulation to be reasonable. This finding is based on the following: (1) the ability of the PCCS to enhance the condensation of steam and the entrainment of air inside containment, (2) analyses performed by Westinghouse and confirmed by the staff that show the containment atmosphere above the operating deck is recirculated approximately every 10 minutes, 24 hours after a LOCA, (3) containment structures have been arranged to promote mixing via natural circulation, and (4) the CVCS and IRWST compartments have been provided with a quarter-size PAR.

The hydrogen recombination subsystem consists of qualified passive devices that are not susceptible to single failures. However, to provide margin and increased containment coverage, two full-size PARs are provided and credit for only a single unit is assumed in the hydrogen analysis. Westinghouse performed an analysis of the maximum hydrogen concentration present in the containment following a design-basis LOCA. Using the guidance in RG 1.7, Westinghouse calculated the hydrogen production rate as a function of time after a LOCA. Westinghouse analyzed core solution radiolysis, sump solution radiolysis, containment material (i.e., zinc and aluminum) corrosion, core zirconium-water reaction, and reactor coolant dissolved hydrogen. The results of this calculation are graphically presented in Figures 6.2.4-1 and 6.2.4-2 of the SSAR. Figure 6.2.4-1 shows that the flammability limit of 4 volume percent is not reached until after 28 days.

The depletion rate assumed in the analysis is based on testing conducted by Battelle Frankfurt and are described in the EPRI reports, "Qualification of Passive Autocatalytic Recombiners for Combustible Gas Control in ALWR Containments," dated April 8, 1993, and, "NIS Passive Autocatalytic Recombiner Depletion Rate Equation for Evaluation of Hydrogen Recombination During AP600 Design-basis accident," dated November 15, 1995. Subsequent testing conducted by EPRI and Electricité de France supports the conclusion of the Battelle testing as documented in EPRI Report TR-107517, Volumes 1, 2, and 3, "Generic Model Tests of Passive Autocatalytic Recombiners (PARs) for Combustible Gas Control in Nuclear Power Plants," dated June 1997. Although this testing was not conducted by Westinghouse or in accordance with Westinghouse's Quality Assurance Program Plan, WCAP-12600, Revision 2, the staff has concluded that the above cited test programs demonstrate that PARs can be designed and procured with the depletion rates assumed in the analyses to generate SSAR Figure 6.2.4-1. The performance features of PARs will be assured by ITAAC and procurement in accordance with 10 CFR Part 50, Appendix B, quality assurance requirements.

Hydrogen depletion tests of a scaled PAR were performed at Sandia National Laboratories, under the sponsorship and direction of the staff. The experiments were to confirm the hydrogen depletion rate of a PAR in the presence of steam and also to evaluate the effect of scale on the PAR performance for a variety of hydrogen concentrations. Preliminary results show that the depletion rate assumed by Westinghouse in Section 6.2.4.2.2 of the SSAR is acceptable. These results were presented on November 19, 1997, at the American Nuclear Society Winter Meeting, Albuquerque, New Mexico, in a paper by Blanchat and Malliakos titled, "Analysis of Hydrogen Depletion Using a Scaled Passive Autocatalytic Recombiner."

To satisfy the design requirements of GDC 41, Westinghouse describes, in Section 6.2.4.5 of the SSAR, preoperational and surveillance performance testing for the HRS. The PARs are verified to provide a hydrogen depletion rate of greater than or equal to the minimum depletion rate assumed in the design basis analysis. It is also verified that the global PARs are located away from the cold plumes falling from the containment dome. Surveillance bench tests are performed on a sample of the PAR cartridges or plates every 24 months to confirm continued satisfactory performance. The staff finds the proposed preoperational and surveillance testing acceptable.

The environmental qualification of the PARs are performed in accordance with the specifications of Section 3.11 of the SSAR using the methodology defined in Appendix 3D of the SSAR. In a letter dated April 1, 1997, the staff concluded that the chemical environment for environmental qualification should include potential poisons. Specifically, the PARs should be environmentally qualified to include the source term constituents that were conservatively assumed to yield the radioactivity dose rates for environmental qualification. In Section 6.2.4.1.2 of the SSAR, Westinghouse states that the PARs will be qualified pursuant to the guidance of the April 1, 1997, letter.

Based on industry data and catalyst poison literature, Westinghouse considered the effects of possible catalytic poisons and inhibitors on their hydrogen depletion analyses as determined in the EPRI report, "The Effects of Inhibitors and Poisons on the Performance of Passive Autocatalytic Recombiners for Combustible Gas Control in ALWRs," dated May 22, 1997. The report combines qualitative information based on established chemical and physical principles with quantitative information from testing of catalysts systems subjected to a wide range of inhibitors and poisons. The report concludes that, "Even if the accident were to progress to beyond a DBA to substantial in-vessel damage, PAR recombination capacity would be reduced by no more than 25 percent." Figure 6.2.4-2 of the SSAR graphically depicts the assumed hydrogen depletion rate in combination with a 25 percent penalty due to poisons and inhibitors. The curve remains below 1.5 percent hydrogen concentration which is considerably less than the regulatory limit of 4 percent.

In September 1980, operating BWRs in the United States were notified of several chemicals potentially present in the containment atmosphere of a BWR following a postulated LOCA, which may inhibit platinum/palladium alloys used to catalyze hydrogen. Two of the substances identified, phosphates and silicone oils, were not addressed in the EPRI poison report. To address this concern, Section 6.2.4.1.2 of the SSAR states that environmental qualification of the PARs will include exposure to phosphates and silicon oil. The source of phosphate is trisodium phosphate, which is dissolved in the post-accident sump water. The release mechanism to the containment atmosphere post-LOCA is via the ADS stage 4 discharge following the onset of the recirculation phase of PXS operation. The source of silicon oil is from a postulated failed steam generator hydraulic snubber.

The safety related portion of the hydrogen control system is designed to remain functional after a safe-shutdown earthquake (SSE) and to perform its intended function following the postulated hazards of fire, internal missiles, or pipe breaks (GDC 3 and 4). Missile protection is discussed in Section 3.5, pipe break protection in Section 3.6 and fire protection in Section 9.5.1 and Appendix 9A of the SSAR.

The containment recirculation system discussed in SSAR Section 9.4.7 provides the controlled purge capability for the containment as specified in position C.4 of RG 1.7.

The staff evaluated the HRS to determine if the design conformed to the regulations and standards in Section 6.2.5 of the SRP. This was identified as DSER Open Item 6.2.5.2-4. Westinghouse calculates the design-basis LOCA hydrogen production rate in accordance with RG 1.7. The HRS design meets the standards for independence, redundancy, inspection, and quality as given in GDC 41, 42 and 43. Westinghouse's analysis of design-basis LOCA containment hydrogen concentration with conservatively calculated hydrogen production and a single active failure of the HRS shows that the hydrogen concentration will not reach its 4 percent flammability limit. Based on its review, the staff finds that the design conforms to the regulations and standards in SRP Section 6.2.5. Therefore, DSER Open Item 6.2.5.2-4 is closed.

Hydrogen Concentration Monitoring Subsystem

To satisfy the design requirements of GDC 41, combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal or accident conditions. The hydrogen concentration monitoring subsystem (HCMS), as described in SSAR Sections 6.2.4 and 7.5, consists of two groups of eight hydrogen sensors each. Three of these sensors have been designated safety-grade, Class 1E. The three Class 1E sensors are seismic Category 1 and serve to provide a post-accident monitoring function for DBAs. The Class 1E instrument channels are independent of the non-Class 1E instrument channels. The hydrogen monitoring function is designed to accommodate a single failure.

The three Class 1E sensors are designed to withstand the dynamic effects associated with postulated accidents, the environment existing inside the containment following the postulated accident, and a safe-shutdown earthquake. The environmental qualification of the hydrogen monitors are performed in accordance with the specifications of Section 3.11 using the methodology defined in Appendix 3D of the SSAR. The HCMS is also included in the equipment survivability assessment, as described in Appendix D of the AP600 PRA and Section 19.2.3.3.7 of this report.

Hydrogen concentration is continuously indicated in the MCR. Additionally, high hydrogen concentration alarms are provided in the MCR. The sensors are designed to provide a rapid response detection of changes in the containment hydrogen concentration and have a measurement range from 0 to 20 percent hydrogen. The response time of the sensor is at least 90 percent in 10 seconds. As part of the preoperational and in-service 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. The HCMS is designed in compliance with the recommendations of NUREG-0737, as detailed in Section 1.9 of the SSAR. The HCMS meets the guidance of RG 1.97 as described in Section 7.5 of the SSAR.

RG 1.97 endorses ANSI/ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors." Section 6.3.5.3 of the standard states that information display channel accuracy for the hydrogen monitor should be within plus or minus 10 percent of span. The hydrogen analyzer relies on a heated platinum catalyst to function. As mentioned in the preceding section, catalytic efficiency of platinum can be reduced by as much as 25 percent. To address this concern, Subsection 6.2.4.1.2 of the SSAR states that environmental qualification of the hydrogen monitors will include exposure to phosphates and silicon oil.

To satisfy the design requirements of GDC 41, the containment hydrogen monitor should meet the guidelines of Item II.F.1 of NUREG-0737 and NUREG-0718, and the Appendix of RG 1.97. The staff evaluated whether the HCMS satisfied these guidelines. This was identified as DSER Open Item 6.2.5.1-1.

The HCMS has been designed to monitor the concentration of hydrogen in the containment atmosphere as well as performance of the HRS. The HCMS has readout and alarm capability in the control room. The HCMS has been designed to meet the requirements of Item II.F.1 of NUREG-0737 and RG 1.97. The staff concludes that the HCMS design meets the regulations and standards in SRP Section 6.2.5-II.11. Therefore, DSER Open Item 6.2.5.1-1 is closed.

Hydrogen Ignition Subsystem

For severe accident hydrogen control, the AP600 containment has been provided with 64 hydrogen igniters. The igniter assembly is designed to maintain the surface temperature within a range of 870 °C to 927 °C (1600 °F to 1700 °F) in the anticipated containment environment following a LOCA. A spray shield is provided to protect the igniter from falling water drops (resulting from condensation of steam on the containment shell and on nearby equipment and structures).

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

The HIS has been designed to promote hydrogen burning at a low concentration. Igniters have been placed in the major regions of the containment where hydrogen may be released, through which it may flow, or where it may accumulate. The criteria utilized in the evaluation and the application of the criteria to specific compartments is provided in Table 6.2.4-6 of the SSAR. The location of igniters throughout containment is provided in Figures 6.2.4-5 through 6.2.4-11 of the SSAR. The location of igniters is also summarized in Table 6.2.4-7 of the SSAR, identifying subcompartment/regions and which igniters by power group provide protection. The locations identified are considered approximations (± 1 m (2.5 ft)) with the final locations governed by the installation details.

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

Although the PARs have been included solely for the design-basis case, their ability to recombine hydrogen in non-combustible environments (e.g., as a result of steam inertion, limited oxygen, or limited hydrogen) is a complement to the igniters. One disadvantage of the PARs is the possibility of being overwhelmed by severe accident hydrogen production rates. If this is the case, the igniters will burn the mixture.

Flow from the IRWST vents, located at Elevation 135', exhausts into the upper compartment. Igniter coverage for the upper compartment includes 10 igniters at Elevation 162', 4 igniters at Elevation 210', and 4 igniters at Elevation 235'.

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

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

Although igniters have not been located directly over the RCS piping in the north CVCS room, two igniters have been located near the ceiling of the equipment room between the equipment module and the major relief paths from the compartment. In addition, a quarter-size PAR has been provided for the CVCS compartment. The staff finds this exception from the igniter location criteria in Table 6.2.4-6 of the SSAR to be acceptable.

On the basis of the staff's review and Westinghouse's implementation of the igniter location criteria as listed in Table 6.2.4-6 of the SSAR, the staff concludes that adequate igniter coverage has been provided.

An additional consideration is the potential of generating significant concentration gradients within the containment during the course of the event. The staff does not expect significant stratification within the AP600 containment based on the above mixing evaluation for the HRS and the number and location of igniters provided for the AP600 containment.

The hydrogen igniters are turned on upon operator entry into function restoration guideline, AFR-C.1, "Response to Inadequate Core Cooling," of the AP600 ERGs. Status tree, AF-0.2, "Core Cooling," of the AP600 ERGs directs the operator to AFR-C.1 when core exit thermocouples read greater than 650 °C (1200 °F). The HIS has been identified as one of the systems to be included in the equipment survivability program. Equipment survivability is discussed in Appendix D to the AP600 PRA and evaluated in Section 19.2.3.3.7 of this report.

The HIS conforms to the requirements of SECY-93-087 and 10 CFR 50.34(f)(2)(ix) by providing reasonable assurance that uniformly distributed hydrogen concentrations inside containment will not exceed 10 percent. Therefore, Open Item 19.2.3.3-2 is closed.

6.2.6 Containment Leakage Testing

The applicant's top level description of the proposed containment leakage testing program for AP600 facilities is described in SSAR Section 6.2.5 and in the proposed technical specifications of SSAR Section 16. The test program will conform to the performance-based requirements of Option B of 10 CFR 50, Appendix J. The staff reviewed the information in the SSAR for conformance to 10 CFR Part 50, Appendix J, Option B, and to GDC 52, 53, and 54. The staff used the guidance, staff positions and acceptance criteria of SRP Section 6.2.6 in conducting its review. The staff also utilized recent generic guidance promulgated to operating reactor licensees for Option B conversions.

Each COL applicant will develop a "Primary Containment Leakage Rate Testing Program" as specified in SSAR Section 6.2.6 and by TS 5.5.9. This program will identify the specific technical specifications surveillance requirements and test criteria for containment leakage tests. RG 1.163, "Performance-Based Containment Leak-Test Program," guidance will be followed in development of this test program. This is COL Action Item 6.2.6-1.

The staff review of the AP600 containment leakage testing program encompassed the following review areas, as identified in SRP Section 6.2.6:

- Type A 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 Type B leakage rate testing, including identification of containment penetrations, general test methods, test pressures, acceptance criteria, and scheduling of tests

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

- Proposed technical specifications requirements pertaining to containment leakage rate testing

The staff's findings for each of the above areas is 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 containment leakage, 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.

Pretest Requirements for Type A Tests

The SSAR 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, which 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, Section III.

Test Method for Type A Tests

The SSAR indicates that, consistent with ANSI-56.8-1994, "Containment Systems Leakage Testing Requirements," the "absolute" method will be used for Type A tests. The containment will be pressurized with clear dry air to a pressure of Pa. Pa for the AP600 is 412 kPa (45 psig). The accuracy of the test will be verified by a supplemental test using methodology consistent with ANSI-56.8-1994. The ANSI-56.8-1994 test methodology is acceptable for use under Option B.

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

Test duration will be a minimum of eight hours. This is consistent with ANSI-56.8-1994 and is acceptable.

Test Acceptance Criteria

The maximum allowable leakage rate (La) is 0.10 percent of the containment air weight per day at Pa. During the first startup following testing, in accordance with the Primary Containment Leakage Rate Testing Program, the leakage rate acceptance criterion will be 0.75 La. The allowable leak 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 Table 15.6.5-2 of the SSAR, and is the value cited in Section 6.2.6 of the SRP. It is therefore an acceptable leakage rate.

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

ANSI-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 Primary Containment Leakage Rate Test 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 Primary Containment Leakage Rate Testing Program using Option B performance-based criteria.

6.2.6.2 Containment Penetration Leakage Rate Type B Tests

Type B tests are intended to detect or measure leakage across pressure-retaining or leakage-limiting boundaries other than 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: (1) penetrations having resilient seals, gaskets, or sealant compounds; (2) air locks and associated door seals; (3) maintenance and equipment hatches and associated seals; and (4) electrical penetrations. This includes one main equipment hatch, two personnel hatches, one fuel transfer tube, a maintenance hatch, 32 electrical penetration assemblies, and three spare electrical penetration assemblies.

General Test Methods

The SSAR 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 measurement.

Test Pressures

In the SSAR, Westinghouse states that the test pressure will not be less than Pa. Pa for the AP600 is 412 kPa (45 psig).

Acceptance Criteria

In the SSAR, Westinghouse states that the Type B leak 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 La. In addition, air lock chambers and individual doors must meet specific leakage acceptance criteria identified in the technical specifications.

Scheduling of Tests

The schedules for periodic Type B leak rate tests will be in accordance with the Primary Containment Leakage Rate Testing Program to be developed using Appendix J, Option B, and RG 1.163.

6.2.6.3 Containment Isolation Valve Leakage 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 steam generator 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 steam generator secondary side will be opened and vented to the atmosphere). The AP600 is a pressurized-water reactor, therefore, these valves are not encompassed by Appendix J, paragraph II.H, which identifies those isolation valves for which Type C testing requirements are applicable. 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 measurement will be used.

Test Pressures

The test pressure will be Pa for pneumatic tests and 1.1 Pa 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 Primary Containment Leakage Rate Testing Program requirements developed using performance-based Option B criteria.

6.2.6.4 Technical Specifications

The staff reviewed the proposed technical specifications in SSAR Chapter 16. The staff determined that the proposed TS are consistent with staff guidance for format and content of technical specifications for containment leakage testing. As stated above, each COL applicant is required to develop a "Primary Containment Leakage Testing Program." This program is a licensee-controlled document that is invoked by reference in the surveillance requirements.

6.2.6.5 Resolution of DSER Open Items

DSER Open Item 6.2.6.1-1 was the applicant's proposal to establish a maximum Type A test interval and decouple the test schedule from the ASME Inservice Inspection Program 10-year schedule. This open item is resolved by Westinghouse's adoption of Option B, which includes criteria for a performance-based retest schedule. Therefore, DSER Open Item 6.2.6.1-1 is closed.

Appendix J requires that air locks be leakage tested at six-month intervals. DSER Open Item 6.2.6.2-1 was an exemption proposal by the applicant to allow air lock leakage tests to be delayed until after the next air lock opening, if the air lock has not been opened. This proposal was dropped from SSAR Section 6.2.5. Therefore, DSER Open Item 6.2.6.2-1 is closed.

6.2.6.6 Conclusion

On the basis of its review, the staff concludes that the proposed AP600 containment leakage 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.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The staff reviewed the AP600 measures involving fracture prevention of ferritic materials used in the containment pressure boundary in accordance with Section 6.2.7 of the SRP. Containment pressure boundary ferritic materials are acceptable if they meet the requirements of GDC 51 as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions the ferritic materials will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

The AP600 containment, as discussed in Section 3.8 of the SSAR, will be fabricated from ASME SA537, Class 2 material in accordance with the ASME Code, Section III, Subsection NE. The ferritic steel components of the containment meet the requirements of the 1989 Edition, including the 1989 Addenda. Section III, Subsection NE, has fracture toughness requirements that will ensure nonbrittle performance and minimize rapidly propagating fracture. This satisfies GDC 51 as it relates to ferritic materials behaving in a nonbrittle manner and minimizing the probability of rapidly propagating fracture.

6.2.8 In-Containment Refueling Water Storage Tank Hydrodynamic Loads

6.2.8.1 Introduction

The in-containment refueling water storage tank (IRWST) is a stainless steel lined tank located inside containment underneath the operating deck. The IRWST is designed as a Westinghouse Equipment Class C, seismic Category 1 structure, and is an integral part of the containment. It is filled with borated water of approximately 2750 ppm concentration, and has a normal depth of 8.5 m (28 ft), which displaces a volume of 2.12E+06 L (560,000 gallons).

The elevation of the bottom of the IRWST is above the RCS loop elevation so that the borated water can drain under the influence of gravity into the RCS after sufficient depressurization. It is connected to the RCS through two injection lines directly penetrating the reactor vessel. Flow-tuning orifices are included in the injection lines of the IRWST to allow field adjustments of the line resistance. The IRWST can provide sufficient injection until the containment sump floods to a level high enough for recirculation flow. The duration of the injection varies according to the particular event.

The IRWST contains the PRHR heat exchanger and two depressurization spargers. The tank is sized to flood the containment for the purpose of long-term cooling in the event of a LOCA, and to allow for operation of the PRHR heat exchangers. Conservative allowances for spill flow that would occur in the event of a vessel injection line break are incorporated into the design. An overflow to the refueling cavity is provided for the IRWST to accommodate the increases in volume and mass during PRHR heat exchanger or ADS operation, while minimizing the floodup of the containment.

Vents are installed in the roof of the IRWST on the side nearest the containment wall. The vents are normally closed to prevent foreign debris from entering the IRWST and to contain water vapor during normal operation. Vents open upon a slight pressurization of the IRWST to provide a release path into the containment atmosphere for steam released by the spargers or the PRHR heat exchangers. Vents open upon slight pressurization of the containment to equalize the IRWST with the external pressure during a DBA.

Connections to the IRWST are made for transfer to and from the RCS/refueling cavity via the RNS system, purification and sampling via the spent fuel pit cooling system, and for remotely adjusting the boron concentration via the PCCS test panel connection to the CVCS. The IRWST is also capable of being cooled by the RNS heat exchangers and can provide injection from this source.

Instrumentation associated with the IRWST includes level and temperature sensors providing both indication and alarm functions.

The AP600 design utilizes an ADS to depressurize the RCS so that long-term gravity cooling of the of the RCS may be established for 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 will be quenched in the IRWST water. This process of discharging a hot pressurized steam/water mixture into a pool of relatively cool water provides for an efficient method for quenching the hot pressurized mixture but also produces significant oscillatory hydrodynamic load on the IRWST structure. This load must be incorporated into the design and internals of the structure. This section of the report will address the evaluation of the Westinghouse analysis used to quantify the hydrodynamic loads imparted to the IRWST structure from the condensation of the RCS coolant being condensed in the IRWST.

6.2.8.2 Evaluation

Westinghouse's approach to establishing the value of these pool dynamic loads was to conduct a series of tests. The main tests were conducted at the Casaccia Center for Energy Research in Rome, Italy. At this center, the ADS system was modeled as a full-size system. The model became known as the VAPORE test facility. The testing was divided into two parts, phases A and B. Testing included testing the sparger, ADS valves, and piping under AP600 prototypical RCS conditions.

Phase A testing was primarily intended to determine the Westinghouse sparger design performance. Design performance in this sense means to determine the sparger's ability to quench the various expected flow rates of a hot pressurized steam, steam/water mixture, and only hot pressurized water simulating the RCS coolant during an ADS blowdown. The goal of the tests was to show the quencher's ability to condense the RCS coolant and establish that the hydrodynamic forces on the IRWST were limited to well below the design capabilities. In addition the analytical methods used to calculate the hydrodynamic forces acting on the IRWST would be validated by using the measured wall pressures as a means of comparison.

Phase B testing evaluated the ADS stage 1, 2, and 3 system. The specific test objectives were to determine the opening times of the valves and the flow rates through the valves. Secondly, they intended to show the flow behavior through the entire piping system with its associated piping resistances. Lastly, they intended to qualify the computer codes used in the prediction of the ADS blowdown with the associated pool hydrodynamic loads for both single- and two-phase flow conditions.

To achieve these test objectives, a sufficient steam supply system was obtained. The supply was connected to prototypical ADS piping and component systems, including the sparger. Instrumentation to record the test data included pressure transducers, thermocouples, strain gauges, and accelerometers.

There were 19 Phase A tests conducted. The main reason for the number of tests was to vary the RCS pressure and IRWST water level to show that the quencher would properly condense steam under all anticipated flow conditions. It was not the intent of this phase to either determine the associated loads nor the performance of the entire ADS system.

The staff reviewed the test data and made several findings. First of all, the slow opening of the stage one ADS valves has shown that the air clearing load is minimal and can be neglected. The opening times of the valves is the primary difference between the ADS valves and the commonly discussed safety valves associated with BWR systems. Reviewing all the tests results showed the maximum pressure pulse occurred during the steam bubble condensation phase of the blowdown. This was expected because the air clearing loads were very small. Recorded pressure magnitudes peaked at about 7 psi, and were recorded at the first instrument rack. While pressures were measured during the Phase A tests, they were not considered in the establishment of the AP600 design loads because only the sparger was modeled with steam flows well in excess of the design.

The Phase A test results were, however, used to establish the sparger anchor loads. Measurements from strain gauges mounted directly on the sparger were used to develop these loads. The staff reviewed this approach and concluded that these test results are adequate to establish the sparger anchor loads.

The Phase B tests were the basis for the establishment of the condensation loads. Visual examinations showed that strong mixing currents existed throughout the test series. It was this information that led Westinghouse to conclude that there would be no thermal stratification within the tank. The staff concurs with this assessment. The second general observation was that the tank wall pressures decreased as the water was heated from 70° F up through saturation. Finally, through all the tests chugging was not observed. The only loads seen were associated with unstable steam condensation. As a result, the design loads are based on a nominal air clearing load and an unstable steam condensation load.

The unstable steam condensation load was taken directly from the pressure measurements from the wall-mounted transducers. For each frequency, the highest measurement during all tests were taken. With this data, the source term at the sparger was determined using acoustic methodology. The source term was then used in a model of the actual AP600 IRWST tank. The results of this analysis provided tank wall pressures which could then be used to establish design pressure loads. Generally, the design loads were increased from 20 to 100 percent of the calculated loads. The staff reviewed this approach and based on engineering judgment, concluded that it is adequate for establishing wall loads.

The loading condition on the internal columns and piping represented a slightly different issue. Unlike the determination of the wall loads described above, the tests conducted at the VAPORE test facility did not have columns simulated within the tank. Therefore, there had to be a similar approach using the wall measurements to establish the source term as discussed above. Using the acoustic method and the source term, analytical calculations could then be used to determine the local loads on the columns. The staff discussed this approach for resolving the issue with Westinghouse. The results of these discussions was an agreed upon path to resolution. In letter DCP/NRC 1340 dated April 9, 1998, Westinghouse submitted a report, "Hydrodynamic Loads on Vessel Head Support Column and ADS Piping Induced by ADS Blowdown." The report describes the methodology used to calculate the submerged structure drag loads. Enclosure 2 of the same report presents a table of calculated forces and moments using this methodology and comparing the results with the initially assumed design load. The initial design load was developed assuming a constant differential pressure load acting along the entire length of the column of 5 and 7 psi. It is this method that was used to establish the structural requirements of the columns and submerged piping. This new method was established to show that the initial design approach was conservative.

This new method used to calculate the submerged structure drag loads requires the determination of the acoustic source strengths at the ADS sparger location. This was accomplished by conducting a series of tests of the full size sparger within a scaled test tank representing the IRWST tank. The test were known as the "VAPORE" tests and the wall pressure test data was recorded for the test series. Once the wall pressures were obtained during the testing and sound speed determined, the acoustic source strength was determined using incompressible fluid theory and classical acoustic equations, which in this case the licensee had chosen the method of images (MOI) approach. The MOI has been previously accepted by the staff for a similar application on BWR designs.

The bubble condensation theory used was based on work by Dr. F. J. Moody, and referenced in the report. It assumes the formation of the single bubble which grows, detaches, and experiences a sudden collapse. This assumption was one of the main concepts used to model the condensation phenomena steam discharges within a BWR suppression pool. The approach has been substantiated many times by independent researches and accepted by the staff for the forcing function on current boiling water reactor suppression pool submerged structures.

Based upon the methodology presented by Westinghouse in document MT03-S3C-026, the staff has concluded that the analytical principles are very similar to the methodology proposed and accepted for other loading conditions in the submerged portions of the tank. The staff has reviewed those areas that differ and have concluded that these areas are also acceptable. The basis of acceptability has been the use of these same methods relative to BWR plant designs. These methods have been previously found acceptable for BWR applications. Therefore, the acceptability for the AP600 application was to determine applicability between designs. The staff has determined that the conditions are similar for both designs and therefore found acceptable for use for the AP600 design for the determination of both column and piping loads.

Westinghouse, using this new methodology, computed the varying loads along the elevation of the column that is submerged. This is about 17.5 feet in total length. The loads were then compared to those loads computed using a constant pressure differential of both 5 psi and 7 psi. It was found that for the column location nearest to the quencher (central bubble position) the new method computed a 19.4 percent margin on force and 1.8 percent margin on moment when compared to a 5 psi constant pressure differential. When compared to the larger 7 psi pressure differential, the margins obviously increased significantly. The margin on force increased to 39.9 percent and the moment went to 29.9 percent.

For the 2/3 height bubble position the force margin was reduced to 7.0 percent while the moment went to 21.1 percent for the 5 psi case. Similar bubble position for the 7.0 psi case resulted in margins of 30.7 percent on force and 43.6 percent on moment.

Based on these demonstrations of margin as stated above, the staff has concluded that Westinghouse has demonstrated that the design loads for the columns and piping within the IRWST have been verified to be conservative through comparison of test data. Therefore, the staff finds the design loads are acceptable. However, the staff requested Westinghouse to reference the hydrodynamic load forcing function development methodology (used in its April 9, 1998, submittal) in SSAR Section 3.8.3.4.2.1. This was FSER Confirmatory Item 6.2.8-1. In Revision 23 of the SSAR, Westinghouse provided the requested reference and FSER Confirmatory Item 6.2.8-1 is closed.

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 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.

The IRWST is a large tank located above the elevation of the RCS loops that contains more than 1982 m^3 (70,000 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 steam generator 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 direct vessel injection (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 steam generator 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 pH control uses pH adjustment baskets containing granulated TSP, which dissolves when the containment sump water floodup reaches the baskets, to maintain the required recirculation sump pH during severe accident conditions.

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

The AP600 PXS design has undergone many design changes since the original SSAR submittal. In the DSER, the staff identified Open Item 6.3-1 stating that Section 6.3 of the SSAR contained information that was not consistent with the eight major changes described in the AP600 Design Change Description Reports, dated February 15, and June 30, 1994, respectively. Subsequent revisions of the SSAR have incorporated these design changes, as well as additional changes. The staff evaluation and the final design approval is on the basis of this information. Therefore, Open Item 6.3-1 is closed.

The staff review of the PXS uses SRP Section 6.3 as a guidance. Because the AP600 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, 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. Acceptability is on the basis of meeting Position C2 of RG 1.29

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

- GDC 5, 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, 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 RC pressure boundary are not exceeded and that the core is cooled during anticipated operational occurrences and accident conditions

- GDC 27, 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, 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, and 37, 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 Section 6.3.1 of the SSAR, Westinghouse describes the AP600 PXS design bases. The PXS is designed to perform its safety-related functions on the basis of the following considerations:

- It has component redundancy with limiting single failure considerations 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.

- It will be tested and inspected at appropriate intervals as defined by the ASME Code, Section XI, and by technical specifications.

- It is protected from the effects of external events such as earthquakes, tornados, and floods.
- It is 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 steam generators occurs, the PRHR HX is designed to automatically actuate to remove core decay heat to prevent water relief through the pressurizer safety valves (PSVs); 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 steam generator tube rupture event to terminate break flow, without overfilling the steam generator.

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 in a plant 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 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 a 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 Section 7.4 of the SSAR.

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 technical specification 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 115.6 °C (240 °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 PXS is a seismic Category 1, safety-related system located inside the containment. Therefore, the PXS is designed for a single nuclear power plant, 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 UPS system power sources are provided to ensure that the system safety functions can be accomplished under conditions when all ac power is lost, and assuming a single failure has occurred coincident with an event.

The PXS design comprises the six major subsystems or components that function together in various different combinations to perform safety-related functions. A description of the six major subsystems and components follows. The piping and instrumentation drawings of the PXS are shown in Figures 6.3-1 and 6.3-2 of the SSAR. A summary of equipment parameters for the major components is contained in Table 6.3-4 of the SSAR.

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 at an 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 56.6 m^3 (2000 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 at 3300 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 RCDT. The operator can open the isolation valves to remove and prevent the accumulation of noncondensible gases that could interfere with CMT operation. The discharge line has two normally-closed, parallel 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 (AOVs), the outlet lines combine into one line, which contains two tilt-disc check valves in series to prevent backflow from the DVI line. The discharge line from each CMT contains a flow-tuning orifice to provide for field adjustment of the injection line resistance to establish the required flow rates for the associated plant conditions assumed in the CMT design. The flow-tuning orifice will be adjusted as part of the preoperational test program.

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

The actuation signals and logic, as well as the permissives and interlocks, to align the CMT for injection are described in subsection 7.3.1.2.3 and Table 7.3-1 of the SSAR, and the actuation setpoints are specified in Table 3.3.2-1 of the AP600 TS. The discharge valve opening delay times used in the safety analyses are provided in Table 15.0-4b of the SSAR.

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 (2000 cubic feet), are filled with borated water at a concentration of about 2500 ppm and pressurized with nitrogen gas to a pressure between 4488 and 5399 kPa (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 reactor coolant system. 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 AP600 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 1982 m^3 (70,000 ft^3) of borated water with a boron concentration of about 2500 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 AP600 Class C equipment, designed to meet seismic Category 1 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, and a flow-tuning orifice, which provides for field adjustment of the injection line resistance. 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 closed 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 MOV and 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 Section 7.3.1.2.2 and Table 7.3-1 of the SSAR, and the actuation setpoints are specified Table 3.3.2-1 of the AP600 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 Section 6.3.2.2.7 of the SSAR. 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 postaccident pH level in the containment sump within a range of 7.0 to 9.5. The baskets, which contain at least 5239 kg (11,550 lbs) of granulated TSP, have a mesh front and are located below the minimum postaccident 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 "Postaccident 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 671 vertical C-shaped tubes 1.9 cm (0.75 in) in diameter. The tubes have vertical sections 5.49 m (18 ft) long, and are 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 1.1E+11 J/hr (1.06E+8 BTU/hr) and 1.3E+5 kg/hr (2.93E+5 lb/hr), respectively, as specified in Table 6.3-4 of the SSAR. The PRHR HX is connected to the RCS by an inlet line from one hot leg and an outlet line to the associated steam generator cold-leg plenum (RC pump suction).

The PRHR HX performs emergency core decay heat removal by natural circulation 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 RC 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 Table 15.0-4b of the SSAR. 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 technical specification 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 passive containment cooling system (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 drain line 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 Section 7.3.1.2.7 and Table 7.3-1 of the SSAR, and the actuation setpoints are specified Table 3.3.2-1 of the AP600 TS. The discharge valve opening delay times used in the safety analyses are provided in Table 15.0-4b of the SSAR.

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 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 steam generator compartment. Section 5.4.6.2 of SSAR 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 25 cm (10 in) valves.

In the DSER, the staff identified Open Item 6.3.2.6.1, which stated that the staff was reviewing Westinghouse's proposed approach of not specifying the ADS valve types. This would allow for flexibility in the valve type to be used in various stages, with the specific valve types to be selected as a COL application activity, and the AP600 safety analysis to be performed using bounding ADS valve and system parameters. The ADS valves are now specified, as described above, therefore, Open Item 6.3.2.6.1 is closed.

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 Section 7.3.1.2.4 and summarized in Table 7.3-1 of SSAR. 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. Table 15.6.5-11 of the SSAR 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 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 AP600 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. Section 1.9.5.3.2 of the SSAR states that the AP600 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. In the DSER, the staff identified DSER Open Item 6.3.2.2-1, which stated that the staff agreed with Westinghouse's position, and would use this position to evaluate the appropriateness of the check valve arrangements in the PXS. The staff has completed its review of the PXS check valves arrangements, as described below. As a result of its review, the staff concludes that DSER Open Item 6.3.2.2-1 is closed.

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. This 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 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 10,600 kPa (1550 psid) during normal operation. They are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating. During a LOCA, these check valves will be forced open by a large differential pressure created by the accumulator pressure as the RCS depressurizes. Therefore, as stated above, they are not subject to single active failure consideration. The staff finds that the check valve arrangements in the PXS are acceptable.

6.3.2.8 System Reliability

The 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

In the DSER, the staff stated that the specific PXS design features to enhance the system reliability were still under staff review. This was identified DSER Open Item 6.3.2.8.1. The staff's review is complete as discussed below. As a result of its review, the staff concludes that the system reliability of the PXS is acceptable, and therefore, DSER Open Item 6.3.2.8.1 is closed.

The PXS system is designed with sufficient redundancy to withstand credible single active and passive failure. The AP600 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 nuclear power plant, 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 AP600 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 Figures 6.3-1 through 6.3-4 of the SSAR 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.

Table 6.3-5 of the SSAR 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 motor-operated valves, 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 Table 15.0-4b of the SSAR. 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 Section 15.6.5.4C.3 of the SSAR 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 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 and 37. Section 6.3.6 of the SSAR describes the inspection and testing requirements, including the preoperational 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 preoperational testing of the PXS is described in Section 14.2.9.1.3 of the SSAR. 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. Section 3.9.6.2 of the SSAR provides a description of the in-service testing of valves. Table 3.9-16 and 3.9-17 of the SSAR, respectively, specify the valve inservice test requirements and system level operability test requirements.

6.3.2.8.5 Seismic and Equipment Classifications

The PXS is a safety-related system, and all the subsystems are designed to meet seismic Category 1 requirements. Table 3.2-3 of SSAR 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 reactor coolant pressure boundary (RCPB) are designated AP600 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. Table 3.2-3 of the SSAR lists many PXS components as AP600 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 AP600 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 Section 3.2.2.5 of the SSAR under the heading "Safety Classification of Passive Core Cooling System," that states that, for systems that provide ECC 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.

Table 6.3-3 of the SSAR 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 AP600 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 Section 7.6.2, "Availability of Engineered Safety Features," of the SSAR, 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 AP600 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 Chapter 7 of the SSAR, and the staff's review is discussed in Chapter 7 of this report.

Section 6.3.7.4.1 of the SSAR 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 Figure 6.3-1 of the SSAR. 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 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 SSAR describe measures taken to protect the system from damage that might result from various events. Section 3.6 of the SSAR 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 SSAR Section 3.9.3, and the requirements for dynamic testing and analysis are discussed in Section 3.9.2. Seismic design is discussed in Sections 3.7, 3.8 and 3.10 of the SSAR. Environmental qualification of equipment is discussed in Section 3.11 of the SSAR. Protection against missiles and from fire are discussed in Sections 3.5 and 9.5.1 of the SSAR, respectively. The staff's evaluations of these SSAR sections are discussed in the corresponding sections of this report.

6.3.3 Performance Evaluation

The 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 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. Section 6.3.3.4 of the SSAR 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 NRHR cooling with the RCS pressure boundary intact
- loss of NRHR cooling during mid-loop operation
- loss of NRHR cooling during refueling

The issue of shutdown and low-power operation is evaluated separately in Section 19.3 of this report.

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 steam generator 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.

Chapter 15 of the SSAR provides an evaluation of the design-basis events, and Section 6.3.3 of the SSAR 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 steam generator 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 steam generator 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 Sections 15.1, 15.2, and 15.6 of the SSAR, respectively. A postulated double-ended rupture of one PRHR HX tube is not analyzed in 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. Section 15.6.5.5 of the SSAR states that, 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 Section 15.6.5.4C of the SSAR.

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. In addition, the analyses of the postulated accidents assume the most reactive control rod stuck out of the core. 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 assumption, to be acceptable. The Chapter 15 analyses results demonstrate the appropriateness of the PXS performance for mitigation of the design-basis events in compliance with GDC 27, 34, 35, and 10 CFR 50.46.

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. In the DSER, the staff identified Open Item 6.3.3-2, stating that the applications of these codes to AP600 design were still being verified and validated through various test programs and were still under staff review. The review of these codes, as well as the test programs, have been completed as discussed in Chapter 21 of this report. Therefore, Open Item 6.3.3-2 is closed.

6.3.4 Post-72 Hour Actions

The AP600 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. For example, the PCS inventory was originally designed for 72-hour capacity. After 72 hours, operator actions may be needed to replenish the PCS inventory. Though the AP600 design includes active systems, such as the RNS, that provide defense-in-depth functions, these active systems are not safety-related systems, and cannot be relied upon to perform safety-related functions.

Section 1.9.5.4 of the SSAR describes the actions and safety functions required following an extended loss of these non-safety-related systems. Westinghouse originally contended that the AP600 design included safety-related connections for use with transportable equipment and supplies to provide the extended support actions for these safety-related functions. These support actions include, for example, using portable engine-driven pumps and ac generators that connect to safety-related connections for water makeup to PCS inventory and electrical power to supply the post-accident and spent fuel pit monitoring instrumentation and hydrogen recombiners. In addition, these extended support actions are implemented as part of the COL applicant's "Site Emergency Response Plan" to provide support for continued long-term operation of the passive safety systems. These actions are accomplished by the site support personnel, in coordination with the MCR operators, and are performed separate from, but in parallel with, other actions taken by the plant operators to directly mitigate the consequences of an event. Westinghouse did not provide information regarding reliability/availability requirements for the portable equipment, or the site emergency procedures to ensure that the equipment would be available in the event of an extended hurricane. Therefore, the staff identified the post-72 hour support actions as DSER Open Item 6.3.4-1. The staff's review is complete as discussed below. As a result of its review, the staff concludes that the post-72 hour action support is acceptable, and therefore, DSER Open Item 6.3.4-1 is closed.

In Section 22.5.3 of this report, the staff discusses the resolution of the concern of post-72-hour support actions. Westinghouse made design changes so that the site is capable of sustaining all design-basis events with onsite equipment and supplies. The supporting equipment and a 7-day supply of consumables will be stored on site. Temporary connections are provided for temporary equipment for the PCS water storage tank fill and the electrical supply to the post-accident monitoring system. The onsite equipment provided for post-72 hours is classified as non-seismic equipment Class D and is protected from credible natural events. The buildings that contain this equipment are classified as AP600 Class D and seismic Category 2. Control of equipment required for post-72 hours is local at the components. Redundancy is provided for active components. The functions provided by this equipment will be tested during plant startup, and are covered by Tier 1 descriptions and ITAAC. Availability controls and testing commitments for the post-72 hour equipment have been incorporated in the administrative control program, described in Section 16.3 of the SSAR, for risk-significant, non-safety-related SSCs identified in the RTNSS evaluation process.

In addition, the PCS design has been modified to provide cooling for 7 days with onsite water. Design provisions have been added to refill this tank using onsite equipment. Containment makeup is not required for more than 30 days.

Two independent 15 kW ancillary ac diesel generator units and fuel source are located together in the annex building, which is protected from SSE and 233 km/hr (145 mph) winds, and is located above site maximum flood level. The Class D equipment is designed with seismic anchors to withstand an SSE.

For the 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. However, there is a potential need for containment inventory makeup because of the containment leakage. The applicant has calculated that, with the maximum allowable containment leak rate, makeup to the containment is not needed for about one month. After that, a safety-related connection is available in the RNS to align a temporary makeup source to containment. Therefore, the long-term cooling capability of the PXS is assured and DSER Open Item 6.3.4-1 is closed.

6.3.5 Limits on System Parameters

The plant TS establish PXS operability requirements for reactor operation. The limiting conditions for operation and surveillance requirements 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 Section 6.3 of the SSAR and other relevant material regarding the 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 AP600 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 AP600 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 nuclear power plant, 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 AP600 design includes preoperational testing for the PXS as discussed in SSAR Section 14.2.9.1.3, and an ITAAC program. Therefore, the staff finds the AP600 PXS design acceptable.

6.4 Control Room Habitability Systems

The staff reviewed the control room habitability systems in accordance with the 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 AP600 design. The staff used a dose criterion of 0.05 Sv (5 rem) TEDE for evaluating the control room radiological consequences resulting from DBAs, pursuant to GDC 19 of Appendix A to 10 CFR Part 50. The justification for the use of this dose criterion and the associated exemption from the regulation is provided in Section 15.3 of this report

- TMI 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

- 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 the DSER, the staff stated that it was unable to complete its review until Westinghouse provided additional information concerning the (1) incorporation of RAI responses into the SSAR, (2) correction of RAI responses, (3) COL applicant responsibilities, (4) conformance with RGs, (5) discrepancies in the SSAR, (6) TSs and testing surveillances, and (7) design requirements for habitability. This was identified as DSER Open Item 6.4-4. After the DSER was issued, Westinghouse provided additional information about the MCR emergency habitability system (VES) that has allowed the staff to complete its review as discussed below. As a result of its review, the staff concludes that the MCR emergency habitability system is acceptable, and therefore, DSER Open Item 6.4-4 is closed.

In Section 3.1.1 of the SSAR, Westinghouse states that the AP600 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 SSAR 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 Section 15.6.5.3.5 of the SSAR, 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 HVAC system (VBS)
- 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 Figures 6.4-1, 1.2-8, and 12.3-1 of the SSAR. Areas adjacent to the MCRE are shown in Figures 1.2-25 through 1.2-31 of the SSAR. SSAR 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 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 of the SSAR, respectively. Details of the radiation monitors, including testing and inspection, are provided in Sections 7.3 and 11.5 of the SSAR. 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 SSAR 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 23.9 °C (67 to 75 °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 SSAR 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 51.7 °C (125 °F) and the temperature in the dc equipment rooms is limited to 48.9 °C (120 °F).

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). SSAR 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 flowpath 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 flowpath contains a normally closed, manually operated valve to manually activate the alternate delivery flowpath in the event the main delivery flowpath is inoperable. The VES piping and penetrations for the MCR envelope are designated as safety Class C. The piping material is alloy steel (SA335P11, ASME Section III, Class 3, Quality Group C), except the piping from the tanks to the sub-headers is stainless steel, as shown in Figure 6.4-2 of the SSAR, 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 SSAR 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 786 kPa (100 psig) via a self-contained pressure control operator. Each flow-metering orifice provides the required flowrate to the MCRE with an upstream pressure of approximately at 786 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 six 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 SSAR 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, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 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 (SMACNA) standards, and duct joints are sealed with qualified non-silicone sealant. SSAR 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. SSAR 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.

The Westinghouse leakage rate analysis originally assumed a MCRE occupancy of five persons. Westinghouse stated that the five-persons occupancy was based on a preliminary post-accident task analysis evaluation that would be finalized with the validation process task analysis in Chapter 18 of the SSAR. However, the staff's concern was not whether a five-person crew was adequate to perform the post-accident tasks, but rather, if it was reasonable to expect that occupancy in the MCRE can be limited to five persons throughout the 72-hour period following an accident. This was identified as DSER Open Item 6.4-2.

After the DSER was issued, Westinghouse revised SSAR Section 6.4.3.2 to add new 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 relative humidity (RH), the thermal analysis resulted in the following:

- 31 °C (87 °F) and 41% RH at 3 hours, when the non-Class 1E battery heat loads are exhausted

- 29 °C (84 °F) and 45% RH at 24 hours, when the battery heat loads are terminated

- 30 °C (86 °F) and 39% 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."

SSAR 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 SSAR 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 eleven 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 eleven persons throughout the 72-hour period following an accident and is predicated on the revised validation process task analysis in Chapter 18 of the SSAR. The staff finds this to be acceptable, and therefore, DSER Open Item 6.4-2 is closed.

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.

Westinghouse stated that chemicals listed in SSAR Table 6.4-1, "Onsite Chemicals," were evaluated using the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following A Postulated Accidental Release," and concluded that these chemicals do not represent a toxic hazard to control room operators. Also, SSAR Section 6.4.4 states that analysis of onsite chemicals, as described in SSAR Table 6.4-1, and their locations, as shown in SSAR Figure 1.2-2, are in accordance with RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and shows that these sources do not represent a toxic hazard to MCRE personnel. The staff performed an independent evaluation using the NRC approved HABIT computer code and the volatility and toxicity limits for these onsite chemicals. 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.

In accordance with TMI Action Plan Item III.D.3.4, COL applicants referencing the AP600 design must demonstrate that control room operators are adequately protected against the effects of the release of toxic substances either onsite or offsite, and that the plant can be safely operated or shut down under conditions created by any DBA. The COL applicant must also determine the amounts and locations of any possible sources of toxic substances near the plant using the methods in RG 1.78 and RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." The COL applicant must provide toxic gas detectors where necessary to permit automatic isolation of the control room. This is COL Action Item 6.4-1. In the text of SSAR Section 6.4.7, Westinghouse states that COL applicants referencing the AP600 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. Additionally, it further states that RGs 1.78 and 1.95 address control room protection for toxic chemicals, and for evaluating offsite releases (including the potential for toxic releases beyond 72 hours in accordance with the guidelines of RGs 1.78 and 1.95) 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 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. In the DSER, the staff stated that in order to provide any credit for fission product removal by HEPA filters and charcoal adsorbers in the supplemental air filtration units in evaluating the control room radiological habitability, the system would be reviewed to the staff's position described in Section A of SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs." This was identified as DSER Open Item 6.4-1.

The staff reviewed the VBS in accordance with its status as a defense-in-depth (DID) system in accordance with SECY-94-084. Portions of the VBS that provide the DID function of filtration of MCR/TSC air during conditions of abnormal airborne radioactivity are designed, constructed, and tested to conform with GSI B-36, RG 1.140, and ASME N509 and N510 standards. Westinghouse describes the design and operation of the MCR and TSC HVAC subsystem during abnormal plant operation with "high" gaseous radiation level detected in the MCR supply air duct in SSAR Section 9.4.1.2.3.1. It states that the system is designed to maintain control room operator doses within the dose acceptance criteria of GDC 19 as applied to the AP600 design. To verify the Westinghouse assertion, the staff performed independent radiological consequence calculations for personnel in the MCR and TSC following a design-basis LOCA. The staff finds that the system design is capable of controlling radioactivity following a design-basis LOCA to meet the dose criteria specified in GDC 19 as applied to the AP600 design - assuming the control room atmospheric relative concentrations proposed by Westinghouse. However, the system is not designed as a post-accident ESF atmospheric cleanup system. The VBS has no safety-grade source of power; therefore, it was not credited in evaluating conformance with GDC 19. The major assumptions used by the staff and the resulting radiological consequence analyses are provided in Table 15.3-10 of this report. On the basis of the above discussion, DSER Open Item 6.4-1 is closed.

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 in the auxiliary building at Elevation 153 ft-0 in. 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 SSAR. The outside air is continuously monitored by redundant smoke monitors at the outside air intake. As stated in SSAR 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 SSAR 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 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, as applied to the AP600 design, requires that the control room be designed to provide adequate radiation protection permitting personnel to access and occupy the control room under accident conditions. 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).

In SSAR Table 15A-5, Westinghouse states that it uses the ARCON96 computer code described in Revision 1 to NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," and meteorological data from three sites (a coastal site, a valley site, and a site with rolling hills) to establish the MCR χ/Q values for the AP600 design.

Westinghouse submitted a radiological consequence analysis for personnel in the MCR during a design-basis LOCA in SSAR Section 15.6.5. The staff reviewed the Westinghouse analysis and finds that the radiological consequences calculated by Westinghouse meet the dose acceptance criteria specified in GDC 19, except for use of the TEDE criterion. The justification for use of TEDE is provided in Section 15.3 of this report.

To verify the Westinghouse assessments, the staff performed an independent radiological consequence calculation for the VES under "high-high" radiation level as described in the AP600 SSAR Section 6.4. The staff used the following information in its analyses:

- the NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," source term

- aerosol removal rates inside the containment, the control room χ/Q values and in-leakage rates provided by Westinghouse

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

The staff finds that the VES, under "high-high" radiation conditions as described in the AP600 SSAR Section 6.4, is capable of mitigating the dose in MCR following DBAs to meet the dose criteria specified in GDC 19 as applied to the AP600 design. The major assumptions used by the staff and the resulting radiological consequence analyses for the control room operators are provided in Table 15.3-7 of this report.

In the DSER, the staff stated that it will perform an independent radiological consequence analysis to verify the Westinghouse conclusion in meeting GDC 19 after resolution of (1) source term related issues in Chapter 15 of this report, and (2) control room χ/Q values in Chapter 2 of this report. This was identified as DSER Open Issue 6.4-3. All source term-related DSER open issues are closed (see Section 15.3) and the staff accepted the control room χ/Q values proposed by Westinghouse. The staff completed its independent radiological consequence analysis, and therefore, DSER Open Issue 6.4-3 is closed.

Westinghouse's position on GSI 83, "Control Room Habitability," concerning the potential impact of operating and maintenance procedures on the performance of the VES and VBS, is that it is a COL applicant's responsibility. Therefore, the COL applicants referencing the AP600 certified design are responsible for verifying that the as-built design, procedures, and training are consistent with the licensing basis documentation and are also responsible for the intent of GSI 83. This is COL Action Item 6.4-2. Subsequently, in response to the above COL action item, the text of SSAR Section 6.4.7 was revised to state that the COL applicants referencing the AP600 certified design are responsible for verifying that the as-built design, procedures, and training for control room habitability are consistent with the licensing basis documentation and the intent of GSI 83.

The VES is tested and inspected at appropriate intervals in accordance with the surveillance and frequency requirements specified in the TS. The leaktightness testing of the MCRE is conducted in accordance with the frequency specified in the TS. Connections are provided for sampling the air supplied from the CAS and for periodic sampling of the air stored in the emergency air storage tanks. 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.

SSAR Table 15.6.5-2 provides the MCRE volume and maximum unfiltered air in-leakage (infiltration) rates as follows. The MCRE volume is 1010 m^3 (35,700 ft^3). The maximum unfiltered air in-leakage (infiltration) into the MCRE under accident conditions is 0.00117–0.00233 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 0.066 m^3/sec (140 cfm). The AP600 design includes a vestibule style entrance to prevent 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. In SSAR Section 6.4.5.4, Westinghouse states that the COL applicant will test the MCRE in-leakage during VES operation every ten years in accordance with ASTM 741, "Standard Test Method for Determining Air Leakages Rate by Tracer Dilution."

SSAR 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 Division 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 SSAR Section 9.4.1.1.2, the post-72 hour design basis of the VBS is (1) to maintain the MCR below a temperature approximately 2.5 °C (4.5 °F) above the average outdoor air temperature and (2) to maintain Divisions B and C instrumentation and control rooms below the qualification temperature of the instrumentation and control equipment. 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 0.0306 ± 0.0023 m^3/sec (60 ± 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 0.00117-0.00233 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 Leakages Rate by Tracer Dilution." The 72-hour capacity of air storage tanks will be verified to be in excess of 8895.3 m^3/sec (314,132 standard cubic feet) at a minimum pressure of 23,440 kPa (3400 psig). Heat loads will be verified to be below the values in SSAR Table 6.4-3. VBS MCRE isolation valves will be tested to verify the leaktightness 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 SSAR 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 SSAR. 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 Sections 3.5 and 3.6 of the SSAR:

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

- Protection from high-energy lines near the control room is evaluated in Section 3.6.1.2 of the SSAR.

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. The staff finds the VES acceptable.

6.5 Engineered Safety Features

6.5.1 Engineered Safety Feature (ESF) Filter Systems

This section is not applicable to the AP600 design.

6.5.2 Containment Spray as a Fission Product Cleanup System

The AP600 design does not have a safety-related containment spray system. The AP600 relies solely on enhanced natural processes for aerosol fission product removal. Credit is taken for removal of airborne radioactivity by natural processes that do not depend on sprays (e.g., sedimentation, diffusiophoresis, and thermophoresis). The design-basis removal of airborne activity is evaluated in Section 15.3 of this report. The AP600 design does include a non-safety-related containment spray which is used to enhance the natural removal mechanisms in the unlikely event of a severe accident. 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 AP600 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 process (e.g., deposition and sedimentation) without the use of containment spray is discussed in Appendix 15B of the SSAR. No active fission product control systems are required in the AP600 design to meet the regulatory requirements. The fission product control mechanisms and the limited containment leakage result in offsite doses that are less than those specified in 10 CFR Part 50.34. In the DSER, the staff stated that it was still reviewing the fission product control systems and structures, and that the results of its review would be discussed in Sections 6.2.1 and 6.2.4, and Chapter 15 of the FSER. This was identified as DSER Open Item 6.5.3-1.

In the March 24, 1994, response to RAI 450.8, Westinghouse stated that the effectiveness of the AP600 containment is enhanced relative to the design of current operating plants by the following:

- the significant reduction in number and size of the containment penetrations
- the simple, reliable PCCS
- the design features addressing potential containment challenges in severe accident scenarios

Westinghouse also states that the effectiveness of the AP600 containment is demonstrated by the low probability of significant offsite releases as addressed in the AP600 PRA. The radiological consequences analysis in Section 15.6.5.3 of the SSAR provides the licensing design-basis evaluation of the AP600 containment function. The source term used to define the fission product release transient to the containment atmosphere is based on the new source term developed by the staff for NUREG-1465. The AP600 analysis accounts for natural processes for the removal of fission products from the containment atmosphere during the event.

Sections 6.2.3, 6.5.1, 6.5.2, and 6.5.3 of the SRP provide the review guidance for the secondary containment functional design for fission product leakage control, ESF atmosphere cleanup systems, containment spray as a fission product cleanup system, and fission product control systems and structures, respectively. The primary difference between the AP600 and other Westinghouse designs is that the AP600 does not have a system specifically designed for fission product control in the containment, whereas those of previous plants have systems designed for the purpose of conforming with the SRP. Because the primary containment for the AP600 is the most significant system for limiting release of radioactivity to the environment in the event of a core-damage event and no other active systems have been identified in the SSAR to control fission products in the containment, the staff's acceptance of the fission product control systems and structures has been based on the conclusions of the reviews addressed in Sections 6.2.1, 6.2.4, and 15.3 of this report. All of the open items in these sections are closed, and therefore, DSER Open Item 6.5.3-1 is closed.

6.6 Inservice Inspection of Class 2 and 3 Components

10 CFR 50.55a requires, in part, that ASME Boiler and Pressure Vessel Code Class 2 and Class 3 components be designed, and be provided with access, to enable the performance of inservice examination of such components and meet the preservice examination requirements set forth in Section XI of the ASME Code. The NRC staff reviewed the designs of the AP600 Class 2 and Class 3 components to ensure that the relevant requirements of 10 CFR 50.55a have been met as they relate to the preservice and inservice inspectability of these components.

GDC 36 requires that the ECCS be designed to permit periodic inspection of important components to assure the integrity and capability of the system. The staff reviewed the design of the AP600 ECCS to ensure that the requirements of GDC 36 have been met.

GDC 37 requires, in part, that the ECCS be designed to permit periodic pressure and functional testing to assure the structural and leaktight integrity of its components. The staff reviewed the design of the AP600 ECCS to ensure that the requirements of GDC 37 have been met.

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. The staff reviewed the design of the AP600 containment heat removal system to ensure that the requirements of GDC 39 have been met.

GDC 40 requires, in part, that the containment heat removal system be designed to permit periodic pressure and functional testing to assure the structural and leaktight integrity of its components. The staff reviewed the design of the AP600 containment heat removal system to ensure that the requirements of GDC 40 have been met.

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. The staff reviewed the design of the AP600 containment atmosphere cleanup system to ensure that the requirements of GDC 42 have been met.

GDC 43 requires, in part, that the containment atmosphere cleanup systems be designed to permit periodic pressure and functional testing to assure the structural and leaktight integrity of their components. The staff reviewed the design of the AP600 containment atmosphere cleanup systems to ensure that the requirements of GDC 43 have been met.

GDC 45 require that the cooling water system be designed to permit periodic inspection of important components to assure the integrity and capability of the system. The staff reviewed the design of the AP600 cooling water system to ensure that the requirements of GDC 45 have been met.

GDC 46 requires, in part, that the cooling system be designed to permit periodic pressure and functional testing to assure the structural and leaktight integrity of its components. The staff reviewed the design of the AP600 cooling water system to ensure that the requirements of GDC 46 have been met.

In essence, GDCs 36, 39, 42 and 45 require that the subject systems be designed to permit appropriate periodic inspection of important component parts to assure system integrity and capability whereas GDCs 37, 40, 43 and 46 require, in part, that the subject systems be designed to permit appropriate periodic pressure testing to assure the structural and leaktight integrity of their components.

For the AP600 design, the applicability of the above GDCs was reviewed. Because of the passive design concepts of the AP600 design, the above systems or portions of the above systems that had been considered safety-related in existing light-water reactor designs and evolutionary plants are not necessarily safety-related in the AP600 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 inservice inspection and periodic pressure and functional 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 AP600 design.

On the basis of its review, the staff found that the emergency core cooling is performed by the AP600 passive core cooling system as described in the SSAR Section 6.3.1. The staff's evaluation of the use of the PCCS in lieu of an ECCS is in Section 6.3 of this report. This system is safety-related and is classified as ASME Code Classes 1, 2 and 3. As such, this system is subject to periodic inspection, pressure testing, and functional testing required by the ASME Code. This system is designed to permit periodic inspection and testing of components. Thus, the staff finds the passive core cooling system meets GDC 36 and 37.

Containment heat removal is performed by the passive containment cooling system as described in SSAR Section 6.2.2. The passive containment cooling system 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 passive containment cooling system is discussed in Section 6.2.2 of this safety evaluation. This system is safety-related and is classified as ASME Code Class 2 and 3. As such, this system is subject to periodic inspection, pressure testing, and functional 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 passive containment cooling system meets GDC 39 and 40.

The AP600 design does not utilize a containment atmosphere cleanup system as found in existing light-water reactors. The AP600 does not have a containment spray system, and the containment atmosphere cleanup system has no safety-related post-accident cleanup function. Fission product control is provided through natural removal processes within containment and by limiting containment leakage. The staff's evaluation of the containment atmosphere cleanup system is provided in Section 15.3 of this report. Because the containment atmosphere cleanup system has no safety function, most of the system is not covered by the ASME Boiler and Pressure Vessel Code. The only exceptions are piping and valves that perform a containment isolation function. These components are ASME Code Class 2. However, the containment atmosphere cleanup system is designed and located so that it can be inspected periodically only, as appropriate. The appropriate portions are designed to permit periodic pressure and functional testing. In this manner, the staff finds that the containment atmosphere cleanup system meets GDC 42 and 43.

The AP600 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 containment cooling system. The staff's evaluation of the component cooling water system is discussed further in Section 9.2.2 of this report. Because this system is not safety-related, the piping, valves, pumps, and other system components are not classified as ASME Code Class except for those portions that function as containment penetration and are classified as ASME Code Class 2. As such, no periodic inspection, pressure, or functional testing requirements apply, except for those portions classified as ASME Code Class 2. The staff finds that inspection and testing of the cooling water system is not necessary because the functions of the cooling water system are subsumed by the containment heat removal system. Therefore, GDC 45 and 46 are not applicable to the AP600 design.

The procedures for the preservice inspection (PSI) and inservice inspection (ISI) of the AP600 Class 2 and Class 3 components are described in Section 6.6 of the SSAR. The staff reviewed this information in accordance with Section 6.6 of the SRP, with particular emphasis placed on access for inspection and inspectability. In the course of its review, the staff transmitted RAIs to Westinghouse concerning these procedures, and received responses to these RAIs from Westinghouse. In addition, several discussions were held between staff and Westinghouse to help clarify and resolve outstanding issues.

The specific areas reviewed included the following:

- the systems, components and supports subject to examination
- accessibility for inspection and examination
- examination categories and methods
- inspection intervals
- evaluation of examination results
- system pressure tests
- augmented ISI to protect against postulated piping failures
- code exemptions

Examples of AP600 design systems that include ASME Code Class 2 and 3 components are the RNS, steam generator system, and the passive core cooling system. All of these systems transport fluids. Inservice testing of safety-related pumps and valves is evaluated separately in Chapter 3 of this report.

10 CFR 50.55a(g)(3)(ii) requires that Class 2 and Class 3 components 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 Section XI of the editions of the ASME Code and Addenda applied to the construction of the particular component. The version of the ASME Code adopted for design certification of the AP600 is specified to be the 1989 Edition together with the 1989 Addenda. The staff noted, in the DSER, that the references in 10 CFR 50.55a to Section XI of the ASME Code refer to Section XI, Division 1, 1989 Edition but did not include reference to the 1989 Addenda. This was identified as DSER Open Item 6.6-2. Subsequent to the issuance of the DSER, the staff determined that, for design certification, the commitment to use the 1989 Addenda is acceptable. Westinghouse included the 1989 Addenda to the ASME Code in the baseline design of the AP600. This is acceptable to the staff, and therefore, DSER Open Item 6.6-2 is closed.

Compliance with the requirements of the regulations for PSI and ISI are based on the following criteria:

- systems, components, and supports are designed so that meaningful PSIs and ISIs can be accomplished

- the COL applicant accomplishes the PSIs and ISIs required by the applicable ASME Code of record

Thus, the Class 2 and Class 3 systems, components, and supports should be designed to be accessible for inspection, such that meaningful inspections can be performed as required by the regulations. For the PSI and ISI examinations for which alternatives to design certification code requirements are sought, the proposed alternatives to these requirements must be identified and NRC approval obtained under 10 CFR 50.55a(a)(3)(i) or (ii) before design certification. The staff requested that Westinghouse state in the SSAR that all ASME Code-required ISI examinations can be accomplished with meaningful results by an operational plant using the inspection equipment and techniques used in the PSIs for all Class 2 and Class 3 systems, components and supports. Where alternatives to ASME Code-required PSIs or ISIs are proposed, the staff requested Westinghouse to identify the specific instances in the SSAR, and obtain NRC approval of the proposed alternative approaches. This was identified as DSER Open Item 6.6-3.

In response to the staff's request, Westinghouse committed that there will be no proposed alternatives to the inservice inspection requirements of Section XI of the ASME Code. This is acceptable to the staff and no revision of the SSAR is necessary to address this item. DSER Open Item 6.6-3 is closed.

The PSI requirements should be established and known at the time each component is ordered, and 10 CFR 50.55a(g) does not have provisions for relief requests at a later stage because of impractical examination requirements. Notwithstanding, there are provisions in Section XI of the ASME Code for the use of certain shop and field examinations in lieu of the onsite preservice examination. However, Section XI indicates that all PSIs should be conducted with equipment and techniques equivalent to those that are expected to be used for the subsequent ISIs.
The PSIs provide the baseline information for reference in subsequent ISIs. For example, if the ISI of a piping weld is expected to be performed using ultrasonic techniques, the PSI should also be based on ultrasonic techniques. The staff requested that Westinghouse provide the means of ensuring that the PSIs will be conducted using equivalent equipment and techniques that would be used for the ISIs. This was Open Item 6.6-4. Revision 3 of the SSAR indicates (in Section 6.6.9.1) that the COL applicant will prepare both a PSI and an ISI program, and that the PSI program will address the equipment and techniques to be used. This is acceptable to the staff. Open Item 6.6-4 is closed.

Section 6.6.2 of the SSAR notes that the possibility exists that relief from Section XI requirements will be requested because of future, unanticipated, changes to Section XI of the ASME Code. The granting of relief from ISI requirements would apply to the COL applicant only for inspections conducted under later editions of the ASME Code and would not apply to the PSI requirements of the ASME Code used for design certification. The staff requested that the SSAR be revised to reflect this position. This was identified as DSER Open Item 6.6-5. Westinghouse subsequently revised the SSAR to indicate that relief from Section XI requirements will not be needed for Class 2 and Class 3 components for design certification. This is acceptable to the staff, and therefore, DSER Open Item 6.6-5 is closed.

Westinghouse stated that provisions have been made in the design and layout of the ASME Code Class 2 and 3 systems to allow access for the examination requirements contained in Articles IWC-2000 and IWD-2000 of Section XI of the ASME Code. In addition, the components and welds requiring ISI are designed to allow for the application of the required ISI methods. That is, the designs include the following characteristics:

- sufficient clearances for personnel and equipment
- maximized examination surface distances
- two-sided access
- favorable materials
- weld joint simplicity
- elimination of geometrical interferences
- proper weld surface preparation

The staff requested Westinghouse to revise the SSAR to describe the means of accomplishing these objectives. This was identified as DSER Open Item 6.6-6. Revision 3 of the SSAR states that examination requirements and examination techniques will be defined by inservice inspection personnel. In line with this, the PSI and ISI programs prepared by the COL applicant will address the actual equipment and techniques to be used, as indicated in Section 6.6.9.1 in the revised SSAR. This is acceptable to the staff, and therefore, DSER Open Item 6.6-6 is closed.

A COL applicant must incorporate plans for non-destructive examination during the construction of systems, components, and supports to meet all the PSI and ISI requirements of the construction code, or as otherwise provided in the code or regulations. The COL applicant is responsible for completing the design and constructing safety-related and non-safety-related systems, components and their supports. The design and construction should be such that the Class 2 and Class 3 systems, components, and supports will not have their accessibility or inspectability for PSI and ISI degraded by construction or other activities. The staff requested Westinghouse to include this COL action item in the SSAR. This was identified as DSER Open Item 6.6-10 and COL Action Item 6.6-4. Westinghouse subsequently revised the SSAR by the addition of Section 6.6.9.2, which identifies the COL applicant as being responsible for addressing the controls needed to preserve accessibility and inspectability during construction or other post-design certification activities. This is acceptable to the staff, and therefore, DSER Open Item 6.6-10 is closed.

The PSI and ISI plans for ASME Code Class 2 and Class 3 components should include the development of the PSI and ISI programs for Class 2 and Class 3 systems, components, and supports. These are the responsibility of the COL applicant. The staff requested Westinghouse to include this COL action item in the SSAR. This was identified as DSER Open Item 6.6-1 and COL Action Item 6.6-1. In an associated item, it was noted that the COL applicant should submit its PSI and ISI program plans for staff review and approval at the appropriate time. The COL applicant should verify that its PSI and ISI programs will incorporate the requirements of Appendices VII and VIII of ASME Section XI, and of GL 89-08, "Erosion/Corrosion Induced Pipe Wall Thinning." The staff requested Westinghouse to include this COL Action Item in the SSAR. This was identified as DSER Open Item 6.6-9 and COL Action Item 6.6-3. Westinghouse subsequently revised the SSAR to include Section 6.6.9.1, in which the preparation of PSI and ISI programs for Class 2 and Class 3 systems, components, and supports is assigned to the COL applicant. The equipment and techniques to be used will be addressed in the PSI program provided by the COL applicant. As discussed below, the guidelines of GL 89-08 will be addressed separately by the COL applicant and are no longer considered by the staff as part of the COL Action 6.6-3 on the PSI and ISI program (SSAR Section 6.6.9.1). The staff finds this acceptable, and therefore, DSER Open Items 6.6-1 and 6.6-9 are closed.

ASME Class 1, Class 2 and Class 3 piping items made of carbon steel have experienced wall thinning as a result of the single-phase or two-phase (water/steam) erosion/corrosion (E/C) phenomenon as documented in GL 89-08. The most effective way of reducing or eliminating this pipe wall thinning is through proper design. Revision 0 of the SSAR indicated that measures had been taken to minimize the occurrence of this phenomenon in the AP600 but few details were provided of how this had been accomplished. The staff requested Westinghouse to discuss its design approaches to reduce the potential for erosion/corrosion in steel piping, apply measures to ensure inspections will be possible and meaningful, and provide provisions for repair or replacement in the SSAR. This was identified as DSER Open Item 6.6-8. Related to this, the COL applicant should be responsible for inspecting pipe wall thinning as a result of E/C during the service life of the AP600 design. This was identified as DSER Open Item 6.6-7 and COL Action Item 6.6-2.

Pipe wall thinning as a result of E/C is of prime concern in high-energy carbon steel systems, such as the main feedwater and other steam conversion systems. Thus, in response, Westinghouse addresses the general aspects of the AP600 E/C control program in Section 10 of the SSAR. This section includes a statement that the COL applicant is responsible for inspection for pipe wall thinning as a result of E/C during the service life of an AP600 plant. A discussion of the staff's review of this program contained in Chapter 10 of this report. Therefore, COL Action Item 6.6-2 has been subsumed by COL Action Item 10.3-1.

In the context of the ISI of Class 2 and Class 3 components, the principal pressure-retaining materials used in the fabrication of these components are the austenitic stainless steels and SA-335, Grade P11, a low-alloy (chromium-molybdenum) steel. These materials are accepted within the nuclear industry to have much increased resistance to E/C-induced damage compared with the plain carbon steels. Inclusion of the guidelines of GL 89-08 in the ISI program for these components is thus obviated. As a consequence, DSER Open Items 6.6-7 and 6.6-8 are closed in the context of ISI of Class 2 and Class 3 components. The generic E/C concerns expressed in these items are embraced within those contained in DSER Open Item 10.3.1-5, the resolution of which is discussed in Chapter 10.

The staff concludes that the AP600 ISI program for Class 2 and Class 3 components is acceptable and meets the inspection and pressure testing requirements of GDCs 36, 37, 39, 40, 43, 45, and 46, as well as the requirements of 10 CFR 50.55a of 10 CFR Part 50 with regard to preservice and inservice inspectability of these components. This conclusion is made on the basis of the following:

- The AP600 Class 2 and Class 3 pressure-retaining components are designed so that access is provided in the installed condition for the visual, surface, and volumetric examinations specified by Section XI of the ASME Code.

- PSI and ISI of the Class 2 and Class 3 components will be performed in accordance with the requirements of Section XI of the ASME Code, including the Mandatory Appendices of Section XI.

- The responsibility for preparing and implementing inspection and testing programs is assigned to the COL applicant. These programs will provide details of the areas subject to inspection, the method and extent of the inspections, and, in the case of the inservice inspection, the frequency of the inspections.

- The examination categories and requirements are in accordance with the criteria specified in Articles IWC-2000 and IWD-2000 of Section XI of the ASME Code. Evaluation of the examination results are in compliance with Articles IWC-3000 and IWD-3000 of Section XI of the ASME Code.

- System pressure tests will be in compliance with Articles IWC-5000 and IWD-5000 of Section XI of the ASME Code.

- The design of the AP600 includes augmented inservice inspection that provides for 100 percent volumetric examination of welds in the high-energy fluids piping system between containment isolation valves and in other critical areas.

- Appropriate measures have been taken in the design of Class 2 and Class 3 components to minimize the effects of E/C during plant operation.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

In Chapter 7 of the AP600 Standard Safety Analysis Report (SSAR), Westinghouse describes the primary instrumentation and control (I&C) systems of the AP600 advanced reactor design and discusses the commitments that pertain to it. The I&C systems provide protection against unsafe reactor operation during steady-state and transient power operations. They also initiate selected protective functions to mitigate the consequences of design-basis events and accidents, and to safely shut down the plant by either automatic means or manual actions.

In Section 7.1 of the SSAR, Westinghouse describes the AP600 general I&C system architecture, with specific emphasis on the protection and safety monitoring system (PMS) design and design process. Section 7.2 discusses the I&C aspects of the reactor trip (protection) function. Section 7.3 addresses the engineered safety feature actuations. Section 7.4 discusses systems required for safe shutdown. Section 7.5 discusses safety-related display instrumentation. Section 7.6 discusses interlocks important to safety. Section 7.7 discusses control systems and the diverse actuation system.

7.1.1 Acceptance Criteria

The acceptance criteria used as the basis for the staff's review are set forth in the NRC's Standard Review Plan (SRP) (NUREG-0800) and 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants." The primary sections of the SRP used for this review are Chapter 7, "Instrumentation and Controls," and Section 9.5.2, "Communications Systems." SECY-91-292 and SECY-92-053 provided additional review guidance. In the staff requirements memoranda (SRM) dated June 26, 1990, and February 15, 1991, the Commission issued guidance pertaining to SECY-90-016 and SECY-90-377, respectively. The Commission also issued guidance in SRMs dated July 21, 1993, and June 30, 1994, pertaining to SECY-93-087, SECY-94-084, and COMSECY-94-024.

Because Chapter 7 of the SRP does not address design certification or the newer digital technology used in the AP600 I&C systems, the staff developed the necessary supplemental acceptance criteria to cover the certification of those systems. The additional acceptance criteria and conformance to the SRP are discussed in the applicable sections of this report. For example, this report describes certain items that will be incorporated in the Tier 1 Material, such as the design descriptions; inspections, tests, analyses and acceptance criteria (ITAAC); and interface requirements for design certification. Chapter 7 of this report discusses only those Tier 1 areas that relate specifically to the I&C systems' certified design process for I&C systems, in addition to specific I&C design characteristics. References to previously reviewed plant designs and topical reports are provided where applicable. To augment this discussion, Chapter 14 of this report describes the general Tier 1 information development process, its bases, and its acceptability.

7.1.2 Basis and Method of Review

The AP600 advanced reactor's I&C system uses a microprocessor-based distributed digital system to perform plant protection functions and safety monitoring, as well as plant control functions. To ensure that the digital I&C systems are implemented properly, the staff considered existing regulatory requirements, guides, and standards in the SRP, and additional standards applicable to digital systems. The use of digital computer technology in protection and control systems raises a concern that the software and hardware for these computer systems could be vulnerable to design and programming errors that could lead to safety-significant common-mode failures. The primary factors for defense against common-mode failures are quality and diversity in the digital I&C system design. This is discussed further throughout this report.

Unlike the current generation of light water reactors, the AP600 reactor uses passive safety systems that rely on natural forces such as density differences, gravity, and stored energy to provide water for core and containment cooling. The AP600 reactor's active systems are not classified as safety-related, and credit is not taken for these active systems in the licensing design-basis accident analyses described in Chapter 15 of this report, unless their operation makes the consequences of an accident more limiting. The non-safety-related active systems in the AP600 reactor provide defense-in-depth functions and supplement the capability of the safety-related passive systems. SECY-94-084 provides the staff position on the regulatory treatment of non-safety-related systems (RTNSS) for passive advanced light-water reactors (ALWRs), and the SRM dated June 30, 1994, provides the Commission's guidance in this area. In SECY-94-084, the staff discussed the need to establish graded safety classifications and graded requirements for I&C systems on the basis of the safety importance of their functional reliability/availability (R/A) missions. This matter is addressed further in Chapter 22 of this report.

This report concentrates on identifying the following items:

- systems with significant design changes compared to previously reviewed and accepted designs
- the appropriate level of detail necessary for design certification
- design issues for which more information is needed in order for the staff to make a finding of acceptability

In addition to reviewing the SSAR and relevant topical reports, the staff also visited Westinghouse's offices and reviewed documentation associated with the SSAR material. All documentation relied on for the safety evaluation has been referenced in the SSAR.

7.1.3 General Findings

The AP600 advanced reactor's I&C systems are significantly different from I&C systems in operating reactor designs. The primary differences result from using digital, microprocessor-based I&C systems with multiplexed data links in place of the analog electronics, relay logic, and hard-wired systems previously approved by the staff. However, the staff has previously reviewed advanced reactor designs using digital equipment similar to that proposed for the AP600 advanced reactor.

7.1.3.1 Compliance with SRP Criteria

The acceptance criteria listed in Table 7-1 of the SRP identifies the Commission's regulations and industry codes and standards applicable to I&C system design. The SRP provides additional review guidance and acceptance criteria that are not provided in the specified requirements, standards, and other references. Table 7-1 of the SRP provides a cross-reference to the SSAR sections that address the applicable standards and criteria. In general, Westinghouse has committed to meet the SRP guidance with few exceptions. The few exceptions that Westinghouse has requested are noted in Sections 1.9 and 3.1 of the SSAR and the applicable sections of this report. The specific I&C standards to which Westinghouse has committed are a significant consideration for the staff's safety findings. The most important aspects of those criteria that are required to be certified by rulemaking will be included in the Tier 1 Material. This information is discussed in Section 7.1.4 of this report.

Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," contains the general design criteria (GDC) applicable to I&C systems. Section 3.1 of the SSAR generally discusses compliance with the requirements of the GDC, and references other SSAR chapters for specifics.

Appendix 7-B of the SRP, "General Agenda, Station Site Visits," provides a general agenda for the station site visit related to the I&C systems, and includes verification of layouts, separation and isolation, test features, and potential for damage as a result of fire, flooding, or other environmental effects. Because the design certification for the AP600 design under 10 CFR Part 52 will be issued before a construction site is selected, this SRP review item cannot be completed at this stage of the review. The inspection tasks identified (in Appendix 7-B of the SRP) as necessary for design certification will be addressed through the ITAAC process and commitments to pre-operational tests described in Chapter 14 of the SSAR. The review described in Appendix 7-B of the SRP will be accomplished as part of the testing and inspections done by the combined license (COL) applicants referencing the AP600 certified design.

7.1.3.2 Compliance with EPRI ALWR Utility Requirements Document

The Electric Power Research Institute (EPRI) prepared a document of technical requirements, referred to as the ALWR Utility Requirements Document (URD). EPRI intended the utilities' design requirements in this document to be applicable to the design of ALWR power plants, including the AP600 advanced reactor. Volume III of the document, "ALWR Passive Plant," applies specifically to the AP600 design. The EPRI URD requirements pertaining to the I&C systems are primarily contained in Volume III, Chapter 10, "Man-Machine Interface Systems."

Although compliance with the EPRI ALWR URD is not a regulatory requirement, the Commission directed the staff to include a discussion of how the ALWR designs compare with the EPRI ALWR URD in its SRM dated December 15, 1989. Accordingly, in Q100.1, the staff requested that Westinghouse provide such a comparison. In its submittal dated December 15, 1992, Westinghouse provided the initial results of that comparison. However, the staff concluded in the draft safety evaluation report (DSER) that Westinghouse should update the response to Q100.1. This was identified as DSER Open Item 7.1.3.2-1.

In Revision 7 to the SSAR, Westinghouse stated that they are a principle participant in the development of the EPRI-sponsored URD and continue to be involved with EPRI on changes to that document. Therefore, the AP600 design remains consistent with the EPRI URD. In its submittal of January 6, 1998, Westinghouse provided an update to the comparison of the URD to the AP600 design. This issue is addressed in Section 1.1 of this report. Therefore, DSER Open Item 7.1.3.2-1 is closed.

7.1.3.3 Compliance with Industry Standards

In the DSER, the staff stated that the SSAR references Institute of Electrical and Electronics Engineers (IEEE) Standards 279, 384, 603, and 796 for the design of the AP600 reactor's I&C systems; there was no reference to digital microprocessor-related standards. That is, the SSAR did not identify any standards regarding electromagnetic compatibility (EMC), multiplexer architecture, communications protocols, and hardware/software design. The staff regards the application of acceptable standards throughout the I&C system design and production process as an important element of the quality demonstration. The application for the design certification must contain a level of information sufficient to enable the staff to make its safety determination. The staff concluded that an explicit commitment to industry hardware- and software-related standards is important to achieving high quality in the digital I&C system product. Therefore, the staff stated that Westinghouse should commit to and reference digital microprocessor-related industry standards. This was identified as DSER Open Item 7.1.3.3-1.

By letter dated June 17, 1996, Westinghouse submitted Revision 1 to topical report WCAP-13383, "AP600 Instrumentation and Control Hardware and Software Design, Verification, and Validation Process Report," to describe the AP600 I&C system design process. This WCAP is referenced in Section 7.1.2.15 of the SSAR. The report states that the software development process is consistent with the following standards:

- ANSI/IEEE ANS-7-4.3.2-1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations"
- IEC 880-1986, "Software for Computers in Safety Systems of Nuclear Power Stations"
- IEEE 828-1990, "IEEE Standard for Software Configuration Management Plans"
- IEEE 829-1983, "IEEE Standard for Software Test Documentation"
- IEEE 830-1984, "IEEE Guide for Software Requirements Specifications"
- IEEE 1012-1986, "IEEE Standard for Software Verification and Validation Plans"
- IEEE 1028-1988, "IEEE Standard for Software Reviews and Audits"
- IEEE 1042-1987, "IEEE Guide to Software Configuration Management"

Westinghouse also provided a discussion of software development in the topical report. The design verification and validation process stresses a "Design for Verification" approach throughout the hardware design, software design, and system integration aspects of the project.

In 10 CFR 50.55a(h), the NRC requires protection systems to meet the requirements of ANSI/IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations." IEEE 603, "Criteria for Safety Systems for Nuclear Power Generating Stations," has since superseded ANSI/IEEE 279. The guidance in IEEE 603, as endorsed by Regulatory Guide 1.153, "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," incorporates the guidance of ANSI/IEEE 279. Both IEEE 279 and 603 were used by the staff in its evaluation of the design, reliability, qualification and testability of the power, instrumentation, and control portion of AP600 safety systems. IEEE 603 does not directly discuss digital systems. It is supplemented by IEEE 7-4.3.2, which provides criteria for applying IEEE 603 to computer systems.

The staff finds the list of standards to be consistent with the staff criteria for the state of the practice for digital system design and is acceptable. Therefore, DSER Open item 7.1.3.3-1 is closed.

7.1.3.4 Compliance with 10 CFR Part 52

Because the AP600 design has been submitted for design certification, the requirements of 10 CFR Part 52 apply in addition to those of 10 CFR Part 50. 10 CFR Part 52 requires a level of design detail beyond a simple commitment to conformance with the existing requirements. 10 CFR 52.47(a)(2) requires the following:

> "The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, prior to design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination."

In 10 CFR 52.47(b)(2)(i), the NRC also states that certification will be granted only if the scope of the design is complete except for site-specific elements. The following sections of this report describe the information provided by Westinghouse and the staff's conclusions concerning conformance with the SRP criteria, additional criteria necessary to address digital I&C technology, and the above requirements of 10 CFR Part 52.

The AP600 digital I&C system described in the SSAR lacks substantial design detail. Therefore, the staff's safety determination under 10 CFR Part 52 will rely on a satisfactory demonstration by Westinghouse that the digital I&C system design development process, as documented in the SSAR, will ensure a quality product. The staff will then confirm the effectiveness of Westinghouse's or the COL applicant's implementation of this process through audits of the ITAAC implementation at various phases of the design development.

7.1.4 Tier 1 Material

On September 4, 1992, Westinghouse submitted a small group of system design descriptions and ITAAC as a pilot program in an effort to reach agreement on the general scope of the Tier 1 design descriptions and ITAAC for all systems. The staff reviewed Tier 1 and ITAAC for the protection and safety monitoring system as representative of the I&C systems, and provided comments in requests for additional information (Q420.8 and Q420.107).

In general, the Tier 1 descriptions and ITAAC that were proposed by Westinghouse were not sufficiently detailed or complete to permit a staff safety finding. Therefore, the staff concluded that both the SSAR and Tier 1 require revision for the staff to reach a safety conclusion. The staff indicated that resolution of this issue should be addressed by providing a detailed description of the digital I&C system design process that incorporates a phased approach to design implementation, as verified through the ITAAC. The staff considers this design process necessary for developing a digital I&C system of sufficient quality to adequately perform its design function. In addition, certain restrictions on changing key SSAR commitments (Tier 2* information) will be incorporated in any design certification for the AP600 to ensure staff agreement before making changes to important details of the design process. This approach is discussed in COMSECY-94-024. Thus, the staff requested that Westinghouse provide a detailed description of the digital system design process in the SSAR and Tier 1 Material with a corresponding ITAAC. This was identified as DSER Open Item 7.1.4-1.

As stated in draft SRP Section 14.3.5, the Tier 1 material should address the hardware and software development process to be used in the design, testing, and installation of safety-related I&C equipment. The Tier 1 material should include a description of the design process to be followed for hardware and software development; the design commitments; the inspections, tests, and analysis to be performed to verify that the design is consistent with the commitments; and the appropriate acceptance criteria against which the design and final product will be judged. The ITAAC should describe attributes of the process to be used to develop the software, as well as attributes of the final software product. The ITAAC for software and hardware that prescribes the proposed design implementation stages should be described in more detail. In Revision 3 of Tier 1, Section 2.5.1, "Diverse Actuation System," and Section 2.5.2, "Protection and Safety Monitoring System," Westinghouse provided additional detail regarding the design process. The Tier 1 material includes a description of the lifecycle stages for which the process is applicable; a description of the elements of the process as relating to software management, configuration management, and verification and validation; a description of the elements of the process relating to dedication of commercial off-the-shelf (COTS) hardware and software; and a description of the inspections, tests, and acceptance criteria which will be used to show that the process was implemented. SSAR Section 7.1.2.15

provides a reference to Revision 1 of WCAP-13383, which specifies a planned design process for hardware and software development with the following lifecycle stages:

- design requirements phase
- system definition phase
- hardware and software development phase
- system test phase
- installation phase

The Tier 1 material and SSAR contain information that describes the methods to develop plans and procedures that will guide the design process throughout the lifecycle stages. The ITAAC provides the acceptance criteria for verifying that the design is implemented through the stages listed above, while the SSAR includes the set of guidelines and standards that provide more detailed criteria for the development of the design. The Tier 1 material was written to incorporate the most important and top-level requirements from the standards. The standards and criteria in the SSAR encompass the guidance for generating the plans that will be used in the computer software and hardware design process throughout the lifecycle. In SSAR Section 7.1.2.15, "Verification and Validation," Westinghouse provides detailed design information and defines design processes for the AP600 I&C system design that follow the lifecycle design process and are, therefore, acceptable for use in meeting the Tier 1 commitment. The staff considers that the design processes and acceptance criteria described in SSAR Section 7.1.2.15 provides the basis for accepting I&C system design.

On the basis of the above, the staff concludes that the top-level design processes, features, and performance characteristic of the I&C design description in the Tier 1 material are acceptable. Therefore, DSER Open Item 7.1.4-1 is closed. See Section 14.3 of this report for further discussion on the staff's evaluation of Tier 1.

As a result of its review of Chapter 7 and the Tier 1 material, the NRC staff has determined that the following information in the AP600 SSAR must be designated as Tier 2* information in the AP600 design control document. Furthermore, any proposed change to Tier 2* information, by a COL applicant or licensee, will require NRC approval prior to implementation.

SSAR Sections:

7 + Table 1.6-1	WCAP-13383, "AP600 Instrumentation and Control Hardware & Software Design, Verification, & Validation Process Report," Revision 1
7 + Table 1.6-1	WCAP-14605, "Westinghouse Setpoint Methodology for Protection Systems, AP600," Revision 0
7.1.2.15	Verification & Validation
7.1.4.1.8	Conformance with Industry Standards

7.1.5 I&C System Architecture

The I&C systems of the AP600 reactor comprise the following major systems:

- protection and safety monitoring system (PMS)
- plant control system (PLS)
- operation and control centers system (OCS)
- data display and processing system (DDS)
- incore instrumentation system (IIS)
- special monitoring system (SMS)
- diverse actuation system (DAS)

The PMS monitors the plant processes using a variety of sensors; performs calculations, comparisons, and logic functions on the basis of those sensor inputs; and actuates a variety of equipment. Most of the time, the PMS operates automatically without input from plant personnel except for system startup, testing, calibration, and maintenance. The PMS is used to operate safety-related systems and components, and includes the following equipment:

- integrated protection cabinets
- engineered safety features actuation cabinets
- protection logic cabinets
- qualified data processing and I/O cabinets
- qualified displays
- reactor trip switchgear
- sensors
- main control room and remote shutdown workstation multiplexers
- main control room and remote shutdown workstation transfer panels

The PLS controls and coordinates the plant during start-up, ascent to power, power operation, and shutdown conditions; integrates the automatic and manual control of the reactor, reactor coolant, and various reactor support processes for required normal and off-normal conditions; controls the non-safety-related decay heat removal systems during shutdown; and permits the operator to control plant components from the main control room or remote shutdown workstation. The PLS includes the following equipment:

- distributed controllers
- signal selectors
- rod control cabinets
- rod drive motor-generator sets
- pressurizer heater control interface
- rod position indication cabinets
- process bus multiplexers
- controls and indications
- process bus
- sensors

The OCS includes the main control room, technical support center, remote shutdown room, emergency operations facility, local control stations, and associated workstations for each of these centers.

The DDS comprises the equipment used for processing data that results in non-Class 1E alarms and displays for both normal and emergency plant operations.

The IIS provides the flux map of the reactor core and in-core thermocouple signals for post-accident monitoring.

The SMS provides loose parts monitoring of the reactor coolant system.

The DAS provides a backup to the PMS for some specific diverse automatic actuation, and provides diverse indications and diverse controls to assist in operator manual actions. The DAS is a defense-in-depth system that is designed to provide essential protection functions in the event of a postulated common-mode failure of the PMS.

7.1.6 Defense-in-Depth and Diversity Assessment of the AP600 Protection System

The first design that the staff reviewed specifically to address defense against potential common-mode failures in digital systems was the Westinghouse RESAR-414 standardized design. The results of the staff's review of RESAR-414 were published in NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," dated March 1979. NUREG-0493 discusses common-mode failures and different types of diversity, and presents a method for assessing the defense-in-depth of the design.

The staff described concerns with common-mode failures and other digital system design issues in SECY-91-292. SECY-91-292 describes how common-mode failures could defeat the redundancy achieved by the hardware architectural structure, and could also result in the loss of more than one echelon of defense-in-depth provided by the monitoring, control, reactor protection, and engineered safety functions performed by the digital I&C systems. The two principle factors for defense against common-mode/common-cause failures are quality and diversity. Maintaining high quality will increase the reliability of both individual components and complete systems. Diversity in assigned functions (for both equipment and human activities), equipment, hardware, and software can reduce the probability of a common-mode failure.

The modules in the AP600 PMS are to be implemented by microprocessor-based designs with identical or similar hardware and software used in all four divisions. Because of this similarity, the concerns expressed in NUREG-0493 and SECY-91-292 apply directly to the PMS.

The following regulations and industry standards address the need for defense against potential common-mode failures:

- GDC 22, "Protection System Independence," requires that "design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function."
- 10 CFR 50.55a(h) (IEEE 279-1971) and IEEE 603-1980, require that "equipment, not subject to failure caused by the same credible event, shall be provided to detect the event..."

- IEEE 379-1988 states that "certain common-cause failures shall be treated as single failures when conducting the single failure analysis. Such failures can be in dissimilar components and can have dissimilar failure modes."

Also, 10 CFR 50.62 addresses common-mode failure issues concerning mitigation of anticipated transients without scram (ATWS).

Common-cause failures not subject to single-failure analysis include those that can result from external environmental effects, design deficiencies, and manufacturing errors. Design qualification and quality assurance programs are intended to afford protection from external environmental effects, design deficiencies, and manufacturing errors. Personnel training, proper control room design, and operating and maintenance procedures are intended to afford protection from maintenance and operator errors.

NUREG-0493 discusses several different types of diversity, each of which offers certain protection against common-mode failures. These forms of diversity include signal diversity and equipment diversity. Signal diversity includes the use of different signals (sensors) to initiate an action. For example, neutron flux and reactor pressure are diverse signals for initiation of reactor scram. Equipment diversity includes using different kinds of equipment to perform a function. An example of equipment diversity described in NUREG-0493 is the use of relay versus solid-state logic in the I&C system. It is difficult to define how much improvement in safety results from a given kind or degree of diversity. For microprocessor design, this is especially difficult because there is no industry consensus on a method to quantify software reliability and/or availability.

As stated above, the staff considers the two principle factors for defense against common-mode failures to be quality and diversity. The quality in the design process aspects of the AP600 I&C systems is addressed in Section 7.1.4 of this report. Quality is achieved, in part, by the use of quality design standards for the hardware and software, and by the I&C system testing to be performed.

In Enclosure 2 of SECY-91-292, the staff discussed regulatory issues that, when properly addressed, are considered to help ensure defense against common-mode failures, as follows:

- assessment of diversity
- engineering activities
- design implementation
- safety classification of I&C systems

The staff's position on I&C system diversity for ALWRs stated in SECY-93-087, as approved by the Commission in an SRM dated July 21, 1993, is as follows:

(1) The applicant shall assess the defense-in-depth and diversity of the proposed I&C system to demonstrate that vulnerabilities to common-mode failures have been adequately addressed.

(2) In performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate methods.

The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.

(3) If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions.

(4) A set of displays and controls located in the main control room shall be provided for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in Items 1 and 3 above.

In response to the staff's position on I&C system diversity for ALWRs, Westinghouse submitted WCAP-13633, "AP600 Instrumentation and Control Defense-in-Depth and Diversity Report," which describes the diversity and defense-in-depth features of the AP600 instrumentation and control architecture following the guidelines outlined in NUREG-0493. The report demonstrated conformance with the guidance of NUREG-0493, which is consistent with the staff position set forth above.

The analysis to protect against common-mode failure in the AP600 I&C architecture was part of the probabilistic risk assessment (PRA) for the AP600 design. In the AP600 PRA, failures of the I&C system architecture including common-cause failures were analyzed. The PMS reliability analysis is described in PRA Chapter 26, "Protection and Safety Monitoring System," and the reliability analyses of the diverse, non-safety-related DAS is described in PRA Chapter 27, "Diverse Actuation System."

In addition, Westinghouse submitted WCAP-13793, "AP600 System/Event Matrix," which describes how the AP600 systems are used to protect the reactor during different events. For each event, WCAP-13793 lists different safety and non-safety-related systems that provide reactor shutdown, reactor coolant system (RCS) makeup, core decay heat removal, and containment cooling. The Westinghouse topical report also includes the type of actuation and electrical power requirements for each system. The purpose of the document is to demonstrate that there are multiple levels of defense for each type of event. The DAS has been credited with providing reactor protection functions in every event analyzed.

On the basis of its review, the staff found that Westinghouse has assessed the defense-in-depth and diversity of the AP600 I&C system and demonstrated that vulnerabilities to common-mode failures have been adequately addressed. Westinghouse has analyzed each postulated common-mode failure for each event that is evaluated in the accident analysis section of the SSAR, and has addressed the diversity requirements within the design for each of these events. The DAS, as proposed, performs the same functions as the PMS when a postulated common-mode failure disables the PMS protection functions. In addition, the DAS, as proposed, has displays, independent and diverse from the PMS, that can support the manual actions to be performed in the event a postulated common-mode failure disables the PMS. The diverse actuation system is designed to actuate components only in a manner that initiates the safety function. The DAS actuation devices are isolated from the PMS actuation devices so as to avoid adverse interaction between the two systems. Therefore, the staff concludes that the proposed design satisfies the Commission's position on I&C system diversity. The evaluation of the DAS is discussed further in Section 7.7.2 of this report.

7.1.7 Commercial-Grade Item Dedication

In SSAR Section 7.7.1.11, Westinghouse states that the diversity of the DAS is achieved by the use of architecture, hardware implementations, and software different from that in the PMS. Software diversity is achieved by running different operating systems and programming in a different language from that of the PMS. This indicates that the COL applicant may procure COTS software and hardware for implementation in the safety-related and non-safety-related I&C systems.

The first aspect of the commercial dedication issue is the use of well-developed operating systems in the development of a plant-specific digital system, such as the DAS. It is essential that the DAS developer ensure that the operating system was developed under strict guidelines and has the quality necessary for its intended function.

The second aspect of the commercial dedication issue is the use of a complete component, such as a programmable logic controller, for which most of the software was developed before the decision to use it in a nuclear application. As with the operating systems, it is necessary for the developer to verify that the equipment selected is of sufficiently high quality for use in a safety system. It is not necessary for the final developer to repeat the verification and validation activities, but it is necessary for the developer to verify that the original equipment designer followed equivalent criteria. Included in the commercial dedication issue is the qualification of the automated tools and design support software. It is necessary for the I&C system developer to verify that the tools function correctly. The staff expects the developer to verify the quality of the tools used in the design.

In the DSER, the staff stated that the design, verification, and validation process for COTS software and hardware should be clearly documented for design certification. This was identified as DSER Open Item 7.1.7-1.

By letter dated June 17, 1996, Westinghouse submitted Revision 1 to topical report WCAP-13383 to address the AP600 reactor's instrumentation and control hardware and software design development process. This topical report also included a description of the commercial-grade item dedication process. The staff reviewed this topical report and finds that it presents an acceptable process for ensuring the quality in the application of COTS products to safety-related functions. The use of commercial-grade hardware and software items in the DAS will be accomplished through a process that specifies requirements for the following:

- review of supplier design control, configuration management, problem reporting, and change control
- review of product performance
- receipt acceptance of the commercial-grade item

- final acceptance on the basis of equipment quality and software and hardware verification and validation in the integrated system

The staff, however, had the following comments:

- Previous operating experience can only be claimed as a basis for demonstration of acceptable performance for COTS products for safety related equipment applications, if the COTS equipment was then and still is under the control of a configuration management program.

- Configuration management is mentioned in Figure 4 of WCAP-13383 in the column on acceptance criteria, providing examples.

The DAS hardware and software is developed using a planned design process that provides for specific design documentation and reviews during the following lifecycle stages:

- design requirement phase
- system definition phase
- hardware and software development phase
- system test phase
- installation phase

The planned design process also provides for the use of COTS hardware and software. On the basis of the above discussion on the commercial dedication process, the staff concludes that the commercial-grade item dedication program for the AP600 reactor's I&C system is acceptable. Therefore, DSER Open Item 7.1.7-1 is closed.

In SSAR Section 7.1.2.15, "Verification and Validation" (by reference to WCAP-13383, Revision 1), Westinghouse provides detailed information for the use of commercial off-the-shelf hardware and software through a commercial dedication process. Control of the hardware and software during the operational and maintenance phase is the responsibility of the COL applicant. The staff considers that the commercial dedication process described in SSAR Section 7.1.2.15 provides an adequate basis for accepting I&C system design.

7.2 Reactor Trip System

7.2.1 General System Description

The reactor trip system (RTS) is part of the PMS and performs the reactor scram function by interrupting electrical power to the rod control system and allowing the control rods to fall by gravity into the reactor core. The RTS includes power sources, sensors, communication links, software/firmware, initiation circuits, logic matrices, bypasses, interlocks, switchgear, actuation logic, and actuated devices that are required to initiate a reactor trip. The RTS is designed to automatically initiate the rapid insertion of the control rods of the reactivity control system to ensure that the specified acceptable fuel design limits are not exceeded. Manual initiation is also provided as a backup to automatic initiation. The RTS also provides status information to the operator, and status and control signals to other systems and annunciators. The RTS, which is qualified as a Class 1E safety system and is environmentally and seismically qualified, provides the following reactor trip functions:

- nuclear startup trips
  - source range high neutron flux trip
  - intermediate range high neutron flux trip
  - power range high neutron flux trip (low setpoint)
- nuclear overpower trips
  - power range high neutron flux trip (high setpoint)
  - power range high positive flux rate trip
- core heat removal trips
  - overtemperature delta T trip
  - overpower delta T trip
  - low pressurizer pressure trip
  - low reactor coolant flow trip
  - reactor coolant pump (RCP) underspeed trip
  - high RCP bearing water temperature trip
- primary system overpressure trips
  - high pressurizer pressure trip
  - high pressurizer water level trip
- loss of heat sink trip
  - low steam generator water level trip (in any steam generator)
- feedwater isolation trip
  - high-2 steam generator (SG) water level in any SG trip
- automatic depressurization system actuation trip
- core makeup tank injection trip
- safeguards actuation trip
- manual trip

The RTS automatically initiates rapid insertion of the control rods to scram the reactor when warranted by any one of the predetermined conditions listed above. The reactor trip is initiated by means of four redundant divisions of sensor channels, trip logic, and trip actuators (except the manual scram function, which can be accomplished from the main control room by redundant momentary switches). The RTS is a four division system where each parameter is monitored by four sensor channels, one in each division. Each division of sensor channels is powered from the respective Class 1E battery-backed power supply. The coincidence logic (two-out-of-four, two-out-of-three, and one-out-of-two voting) is accomplished in the trip logic subsystem, and the voting is done as local coincidence logic, which requires the two tripped signal inputs to be from the same parameter.

Reactor Trip System Interlocks

The interlocks used in the reactor trip functions are designated as P-xx permissives. These permissives are implemented at the channel level rather than at the logic level to improve the plant availability as follows:

P-6: Derivation -- intermediate range neutron flux above setpoint
Function -- allows manual block of source range reactor trip

P-8: Derivation -- power range nuclear power above setpoint
Function -- permits reactor trip on low flow or RCP high bearing water temperature in a single loop

P-10: Derivation -- power range nuclear power above setpoint
Functions -- (a) allows manual block of power range (low setpoint) reactor trip
(b) allows manual block of intermediate range reactor trip
(c) automatically blocks source range reactor trip (backup to P-6)
(d) allows reactor trip on low coolant flow or RCP high bearing water temperature in multiple loops
(e) allows reactor trip on low RCP speed
(f) allows reactor trip on high pressurizer water level
(g) allows reactor trip on low pressurizer pressure

P-11: Derivation -- pressurizer pressure below setpoint
Function -- allows manual block of high-2 SG water level reactor trip

P-17: Derivation -- power range nuclear power negative rate below setpoint
Function -- blocks automatic rod withdrawal

System-Level Manual Inputs to the Reactor Trip Functions

Manual Control	To Divisions			
Manual Reactor Trip Control #1	A	B	C	D
Manual Reactor Trip Control #2	A	B	C	D
Reactor Trip Reset	A	B	C	D
Source Range Block, Division A	A			
Source Range Block, Division B		B		
Source Range Block, Division C			C	
Source Range Block, Division D				D
Intermediate Range Block, Division A	A			
Intermediate Range Block, Division B		B		
Intermediate Range Block, Division C			C	

Manual Control	To Divisions
Intermediate Range Block, Division D	D
Power Range Block (Low Setpoint), Division A	A
Power Range Block (Low Setpoint), Division B	B
Power Range Block (Low Setpoint), Division C	C
Power Range Block (Low Setpoint), Division D	D
Manual Safeguards Actuation Control #1	A B C D
Manual Safeguards Actuation Control #2	A B C D
Manual Core Makeup Tank Injection Control #1	A B C D
Manual Core Makeup Tank Injection Control #2	A B C D
Manual Depressurization System Actuation Control #1 & #2	A B C D
Manual Depressurization System Actuation Control #3 & #4	A B C D

The RTS will periodically be tested during plant operation as defined by the technical specifications. In addition to the standard operator-initiated surveillance, the safety system logic portion of the RTS will be tested by the automatic tester subsystem.

Complete electrical and physical separation is maintained between the four RTS divisions to meet the criteria of IEEE 279-1971 (50.55a(h)). Implementation of this requirement will be verified during the implementation of the ITAAC. When a trip of any sensor or logic channel is annunciated, it causes that channel to lock in the trip mode until manually reset. The RTS is fail-safe in that a loss of power to a channel will result in that channel going to the tripped condition. Other failures, such as a break in a communications link, are detected by the self-diagnostics of the individual microprocessor that will put the output to the tripped state.

The protection system for the AP600 reactor is an integrated microprocessor-based system. The functional level block diagram of the safety-related signal paths, as shown in Figures 7.1-2 through 7.1-9 in the SSAR, is relatively simple and direct. However, in a microprocessor-based system, the software implied in the blocks of the system diagram can mask much of the safety system's design complexity. Several of the significant issues, involving the use of microprocessor-based systems and ensuring their safety functions, were presented to the Commission in SECY-91-292. These issues are addressed in Section 7.1 of this report. As described above, the staff reviewed the RTS for conformance to the requirements of IEEE-279 (10 CFR 50.55a(h)) for meeting the single-failure criterion, independence, control and protection system interaction (isolation), testing, bypass and bypass indication (including removal of bypass), and manual initiation. The staff concludes that the RTS description and drawings in the SSAR establish a clear commitment to comply with the above requirements of IEEE 279 (10 CFR 50.55a(h)).

7.2.2 Protection and Safety Monitoring System Description

The reactor trip function is performed by the protection and safety monitoring system (PMS), an integrated protection system that provides the following safety-related functions:

- tripping the reactor by opening the reactor trip breakers
- actuating the engineered safety features equipment

- monitoring safety-related plant parameters before, during, and after an accident or plant transient

The functions of the AP600 PMS are implemented by software logic installed in programmable digital devices (data processors). Plant data and other signals are exchanged between data processors by means of isolated data links and data highways. The major subsystems of the PMS are described below.

Integrated Protection Cabinets

The four integrated protection cabinets (IPCs) in the plant (one for each division) collect input data from plant sensors and nuclear instrumentation, perform computation or logic operations on variables on the basis of these inputs, provide trip signals to the reactor trip switchgear, and provide engineered safety features (ESF) actuation data to the ESF actuation cabinets, as required. Section 7.3 of this report discusses the ESF actuation systems. The equipment necessary to permit manual trip or bypass of each individual reactor trip function is contained in the IPCs. The IPCs provide data to other systems in the I&C architecture through isolation devices. Each IPC contains the following subsystems:

- reactor trip group 1
- reactor trip group 2
- global trip
- trip enable
- engineered safety features group 1
- engineered safety features group 2
- communication
- automatic tester
- nuclear instrumentation signal & processing and control 1
- nuclear instrumentation signal & processing and control 2

Reactor Trip Subsystems

The reactor trip functions have been divided into two functionally diverse subsystems. Their primary function is to process input data and provide a partial trip signal to the dynamic trip bus whenever the preset limit of each protection function is exceeded. The trip functions are divided among the two subsystems as follows:

- Reactor Trip Group 1 Subsystem

  - high source range neutron flux
  - low reactor coolant flow
  - low reactor coolant pump speed
  - overtemperature delta T
  - overpower delta T
  - high reactor coolant pump bearing water temperature
  - high compensated pressurizer level

- Reactor Trip Group 2 Subsystem

  - high intermediate range neutron flux
  - high power range flux low setpoint
  - high power range flux high setpoint
  - high positive rate power range neutron flux
  - low pressurizer pressure
  - high pressurizer pressure
  - high-2 steam generator narrow range level
  - low steam generator narrow range level

The Group 1 and Group 2 subsystems function as backups to each other. Independence of the functionally diverse trips is maintained in the reactor trip groups from the input circuit to the dynamic trip bus. A failure in one group will not fail the protective function in that division. There is a Trip/Normal/Bypass switch at the dynamic trip bus in each division associated with the two subsystems. Should a failure be detected in the associated input circuitry or sensor in a subsystem, plant personnel can trip or bypass individual trip functions rather than bypassing an entire division. Indicators are located adjacent to the Trip/Normal/Bypass switch to display the partial-trip function status.

Trip Logic Function

The trip logic function is performed by three primary components:

(1) The global trip subsystem provides partial trip/bypass status to the other divisions and division trip/bypass status to the other divisions. The subsystem computes the division's global trip signal.

(2) The trip enable subsystem provides partial-trip enable signals for each partial-trip function and computes the global bypass permissive signal.

(3) The dynamic trip bus performs the logic to combine the partial-trip and partial-trip enable signals, as well as the global trip and global bypass functions. It also outputs a trip signal to the reactor trip switchgear. The manual reactor trip is implemented in the dynamic trip bus.

Reactor Trip Switchgear

The final stage of the dynamic trip bus provides the signal to energize the undervoltage trip attachment on each of the two division reactor trip switchgear breakers. Loss of the signal deenergizes the undervoltage trip attachments and results in the opening of the reactor trip breaker. An additional external relay is deenergized with the loss of the signal. The normally closed contacts of the relay energize the shunt trip attachments on each breaker at the same time that the undervoltage trip attachment is deenergized. This diverse trip actuation is performed external to the PMS cabinets.

The reactor trip switchgear provides four redundant protection divisions, with each division controlling the trip of two circuit breakers in the reactor trip switchgear (eight breakers total, arranged in a two-out-of-four logic configuration). The reactor trip switchgear connects electrical motive power from the motor-generator sets to the rod control system. When the reactor trip breakers open upon a reactor trip signal, power to the rod control system is removed and the control rods drop by gravity into the core, initiating the shutdown process.

Manual Reactor Trip

A manual reactor trip can be accomplished from the main control room (MCR) or the remote shutdown workstation (RSW) by redundant momentary switches. The switches in the MCR directly interrupt power from the dynamic trip bus, actuating the undervoltage and shunt trip attachments. The switches in the RSW deenergize an interposing relay whose contact interrupts power from the dynamic trip bus.

Automatic Tester Subsystem

The automatic tester subsystem performs the periodic surveillance test function on the IPC while a division is bypassed. The tester is manually initiated, with the test itself proceeding without operator intervention. The test involves injection of simulated inputs and monitoring of the output to demonstrate that expected results are obtained. The automatic tester panel in each IPC provides operator interface capability. A key switch on the panel can manually enable automatic testing. In addition to periodic tests, the automatic tester subsystem also continuously monitors for failures and provides diagnostic information from the subsystems during normal operation. Where practical, the online error-detecting feature is designed to automatically place a channel in which an error was detected into a trip or bypass mode. If the channel is not automatically placed into a trip or bypass mode, the online error-detecting feature will annunciate an alarm to the operator.

Communication Subsystem

Information from the IPC is transferred by multiplexer to both the PLS and the DDS. Fiber optic data links are used to provide electrical isolation between the PMS and the external systems. Analog inputs required for both control and protection functions are processed independently with separate input circuitry.

The review and acceptance of the PMS design as described in Section 7.2.2 is addressed in Sections 7.2.3 through 7.2.9 of this report.

7.2.3 Assessment of IEEE 796 Bus in the AP600 Design

Westinghouse states that the AP600 protection system design conforms to IEEE 796-1983. Because this is the first time that this standard has been referenced in a license application for a safety-related system, the staff performed an assessment of this standard for applicability to the AP600 design, as well as an evaluation of the AP600 design against IEEE 796. The following are the major findings of the assessment:

- The IEEE 796 bus is a computer backplane bus used as an internal data path inside cabinets. The bus can be extended to additional printed circuit board (PCB) cages with bus repeaters and short multiconductor flat cables. It is not used for inter-computer communications over longer distances. For the AP600 design, the functions of the PMS have been decomposed into physically and electrically separate microprocessor-based subsystems. Each subsystem is located on an independent IEEE 796 bus internal to the cabinet.

- The IEEE 796 bus design is sensitive to external noise and should be used in well-shielded enclosures with well-shielded power supplies. The AP600 design uses integrated protection cabinets that will be qualified for electromagnetic interference (EMI) protection by the methodology reported in WCAP-11341, "Noise, Surge, and Radio Frequency Interference Test Report," dated November 1986. The review and acceptance of EMI protection is assessed in Section 7.2.8 of this report.

- The IEEE 796 bus design is based on an 18-year old design standard. Its computation capability is limited, and availability of replacement parts is also a concern. In response to the staff's request for additional information, Westinghouse stated that the IEEE 796 bus design for the AP600 protection system is based on proven technology, and its capability is adequate for the intended purpose. Westinghouse was not aware of parts availability problems. The staff believes that replacement parts availability is a potential problem in the future for all digital instrumentation systems because of the rapidly evolving technology. The digital system design process discussed in the SSAR and the Tier 1 Material should include the complete hardware/software lifecycle that would address this concern by allowing flexibility in the specific design features of the digital system. The corresponding ITAAC in the Tier 1 Material should provide a general description of the activities that are to be performed by the COL applicant to verify that the specific I&C components are consistent with the certified design description without specifying or limiting equipment selection.

- The purpose of the IEEE 796 bus standard is to define technical specifications so that independent suppliers of computers or computer-related PCBs can build compatible products that can be combined in an IEEE 796 PCB cage. In view of the history of incompatible products on multibus computer systems, all safety system applications should be reviewed by a technical organization having expertise in computer backplane bus performance. Westinghouse practice in the past has been to test IEEE 796 bus systems in delivered configurations, and supply diagrams indicating board positions. For the AP600 design, because equipment procurement is the COL applicant's responsibility, subsequent adherence by the COL applicant to Westinghouse equipment specifications is required under the configuration management program.

- The bus design described by IEEE 796-1983 has minimum capability to support multiprocessing applications and, therefore, the software must ensure correct system synchronization. Software errors in this area may result in common-mode failures extending over multiple systems. Westinghouse stated that for the AP600 design, only one processor is to be used in any single IEEE 796 bus configuration. This removes multiprocessor concerns as a source of failure. In the DSER, the staff indicated that this clarification should be included in the SSAR and related documents. This was DSER Confirmatory Item 7.2.3-1. The potential for mis-coordination between readers and writers in shared memory still exists, but this is not specifically limited to IEEE 796 bus systems. This latter concern is dealt with during the software development and verification and validation (V&V) programs that are part of the digital system design process discussed in Section 7.1.4 of this report.

- The IEEE 796 bus is a synchronous bus in which data transfers take place by a two-party interaction called a "handshake." In this interaction, a data receiver does not accept data until it receives confirmation that the data is valid, and a data sender does not remove valid data from the bus until it receives confirmation that data has been received. Data transfers are under the control of a "bus master" that initiates bus transactions. Potential for conflict exists if PCBs are configured for incompatible bus master exchanges. If a bus priority arbitrator is used, it represents a single-point failure vulnerability for the system. Westinghouse stated that for the AP600 design, the box marked "System Bus Parallel Arbitration" was not really used as a bus priority arbitrator. There is no bus priority arbitrator required or used in the design. The system is configured using multiple, separate computer subsystems within a safety division. The divisions are configured with four-way redundancy, thus eliminating single-point failure vulnerability. In the DSER, the staff indicated that this clarification should be included in the SSAR and related documents. This was DSER Confirmatory Item 7.2.3-2.

In Revision 3 of SSAR Section 7.1.2.4.1, Westinghouse stated that bus contention conflict is arbitrated by a combination of hardware and software. A failure of the hardware or software that arbitrates bus contention could result in a complete failure of one of the two processing functions performed within a single protection logic cabinet. However, this failure will have no adverse effect because the other redundant processing function will continue to operate. The I/O Bus Selector function will provide the output cards with commands developed by the two functional processor cards that have not experienced the bus contention failure. Even if a protection logic cabinet should completely fail, this failure affects only the equipment associated with a single mechanical train. Multiple mechanical trains of equipment are provided in the AP600 design so that the failure of equipment within a single equipment train does not effect completion of AP600 safety functions.

In WCAP-14080, "AP600 Instrumentation and Control Software Architecture and Operation Description," Westinghouse stated that the software architecture of the AP600 protection system is based on the use of multiprocessing, with single program, non-interrupt driven execution on each processor. Asynchronous communication methods are used. This will ensure that data transmission error is not common to redundant protection logic functions. The staff finds the above clarifications to be acceptable, and Confirmatory Items 7.2.3-1 and 7.2.3-2 are closed. The review of WCAP-14080 is discussed in Section 7.2.6 of this report.

7.2.4 Review of the AP600 Global Trip Subsystem

In addition to the defense-in-depth and diversity assessment of the AP600 I&C system discussed in Section 7.1.6 of this report, the staff performed an independent, defense-in-depth and diversity assessment of the AP600 protection system that was issued to Westinghouse in questions Q420.94 and Q420.99. In this study, the global trip subsystem (GTS) was identified as a source of vulnerability. Specifically, if a common-mode software failure affects all four GTSs, the reactor protection system would be disabled. The vulnerability identified in the Lawrence Livermore National Laboratories (LLNL) study was based on a postulated GTS failure in which each GTS keeps cycling to maintain its dead man pulse input to the dynamic trip bus logic unit, and maintains communications with the trip enable subsystems, sending valid messages with no trip indications contained therein.

In its response to Q420.99 dated June 2, 1994, Westinghouse clarified the design of the GTS as follows:

The GTS communicates with the trip enable subsystems, the other GTSs, and the dynamic trip bus using two distinct methods. Specifically, signals to the dynamic trip bus are discrete digital outputs, while signals to the other GTSs use isolated data links. In each of the places where GTS signals are received, means are provided to detect failure of the transmitting GTS. Detection of the failure of multiple GTSs will result in a reactor trip before the PMS is degraded to the point where it cannot provide a reactor trip. The postulated failure of the GTS in the LLNL report requires failures to occur concurrently in two or more places within the GTS software in order for the dynamic trip buses and trip enable subsystem to fail to detect the simultaneous failures of the four GTSs. Such multiple, concurrent, and specific common-mode failures are very low probability occurrences. However, even if such a failure did occur, the DAS is designed as a defense against this failure mode by providing appropriate backup functions.

The staff finds that Westinghouse has provided sufficient detail describing the design of the GTS and, therefore, the clarification is acceptable. The use of the DAS as a defense against such common-mode failure events is discussed in Section 7.1.6 of this report.

7.2.5 Review of the Bypass Logic in the Protection System

Westinghouse submitted Addendum 2 to WCAP-8897, "Bypass Logic for the Westinghouse Integrated Protection System," to support the RTS design feature that permits continuous operation of the system with one or two of four channels bypassed for testing or maintenance. The reactor trip actuation logic automatically reverts to two-out-of-three coincidence logic if one channel is bypassed, and to one-out-of-two coincidence logic when two channels are bypassed. The following table summarizes the automatic voting logic associated with the number of inputs bypassed.

No. of Inputs Bypassed	No. of Remaining Inputs that Result in a Trip
0	two out of four (2/4)
1	two out of three (2/3) (alarmed)
2	one out of two (1/2) (alarmed)
3	automatic trip
4	automatic trip

Addendum 2 to WCAP-8897 describes the specific implementation of the reactor trip logic system that incorporates the bypasses of the reactor trip functions. On the basis of its review, the staff concludes that the bypass system as described will perform as intended and as identified above because it satisfies the single-failure criteria and will perform its safety function (reactor trip). However, the staff noted in the DSER that during the time the plant is operating with two channels bypassed, any subsequent single failure could lead to an inadvertent reactor trip and thus, from an operational standpoint, operation with two channels bypassed should be limited. In addition, the staff requested that Westinghouse verify that this bypass logic only applies to the RTS and does not apply to the engineered safety feature actuation system. The staff requested Westinghouse to revise the topical report to provide additional descriptions of the bypass logic for the engineered safety feature actuation systems. This was identified as DSER Open Item 7.2.5-1.

In Revision 3 of SSAR Section 7.1.2.10, Westinghouse stated that the reactor trip and engineered safety features actuation logic reverts to two-out-of-three coincidence logic if one channel is bypassed or in test. Thus, a single failure while in test cannot cause a spurious reactor trip or spurious system level engineered safety features actuation. This same two-out-of-three logic also provides that any failure in a single protection channel or safety division cannot prevent a required reactor trip or system level engineered safety features actuation from occurring. Engineered safety features actuation logic is performed redundantly in each engineered safety features actuation cabinet. Redundant microprocessor-based systems perform this logic so that any component failure related to one subsystem does not affect the other redundant subsystem. The system level actuation outputs are transmitted to the protection logic cabinets over two redundant data highways. A single data highway failure cannot prevent engineered safety features actuation. Extensive error checking is continuously performed on these data highways to prevent failures from causing spurious actuation. Four redundant logic processor boards are provided along with two data highway controller boards. Two logic processor boards are associated with each data highway controller board. Failure of one data highway or one data highway controller board does not prevent component level actuation. The component actuation outputs from the logic processors are combined with the power interface cards in a two-out-of-three voting logic. This prevents the failure of a single logic processor from causing spurious actuation or a required actuation.

On the basis of the above clarification, the staff finds that the bypass system as described will perform its intended function while not compromising required safety system actuations. Therefore, DSER Open Item 7.2.5-1 is closed.

7.2.6 Review of the AP600 Software System Architecture

In its response to Q420.120 dated January 11, 1994, which requested a description of the software architecture planned for the AP600 advanced reactor, Westinghouse supplied the proprietary Class 2 document, "AP600 Protection and Safety Monitoring System and Plant Control System Architecture Description." This document gives a brief overview of the software architecture and a summary description of the execution sequence, software module types, and data structure. However, this description was not sufficient to permit the staff to complete its evaluation of the software architecture design.

In Q420.120, the staff also referenced the Westinghouse response to a question raised by the NRC during a meeting on August 11, 1993, wherein Westinghouse stated that the AP600 architecture would be the same as that used for the Sizewell B facility. This response prompted a request for additional information (RAI) citing several concerns regarding the software system architecture for the Sizewell B facility; these concerns were documented in the "Proceedings of a Forum on Safety Related Systems in Nuclear Applications," dated October 28, 1992. This forum was sponsored by the Royal Academy of Engineering, London, England, to enable Nuclear Electric to describe the Sizewell B protection system computer software, and to enable the attendees to debate the issues related to this application of computer software. Two significant software architectural concerns raised at the debate were: (1) the perceived complexity of the protection system software, both with respect to program size in terms of lines of code, and (2) the use of pointers in dynamic memory management. A third concern was age of the original software. These concerns raised questions about the reliability of the AP600 software.

These concerns have been addressed, both in the Westinghouse response to Q420.120, and during a meeting between the NRC staff and Westinghouse held on April 6 and 7, 1994. Subsequent to that meeting, Westinghouse submitted a proprietary Class 2 document, WCAP-14080, "AP600 Instrumentation and Control Software Architecture and Operation Description," that provides a more detailed description of the AP600 software architecture.

On the basis of its review of WCAP-14080, the staff concludes that Westinghouse has addressed the three items discussed above. The first item is the perceived complexity of the protection system software on the basis of the number of lines of code, when compared with other design approaches. The software architecture described in WCAP-14080 clarifies the difference between a centralized and distributed software architecture, especially where the distributed architecture includes software modules that perform "operating system" software functions, as well as the specific application software functions. While the total number of lines of code for the latter is usually greater, especially because the lines of code associated with the operating system usually are not counted, the number of "different lines of code" can be smaller if the number of different software modules is smaller. The concern is then transferred from the abstract view of code size to the evaluation of design features and design process for the software modules that make up the distributed "operating system" with respect to system reliability.

The second item is the comment made at the forum regarding problems arising from the use of pointers. The description of the operation of the software in WCAP-14080 distinguishes between the reliability concerns associated with the operation of a software system where communication between the various software functions includes the use of pointers, and problems that arise from the use of specific tools in the verification of a design that includes the use of pointers.

The third item is the comment made at the forum with respect to the age of the original software. A comparison of the basic software functions for the architectural structure of the AP600 with the software functions for the new TRW Universal Network Architecture Services (UNAS) shows close similarities in key features that TRW claimed would improve the reliability of the software design when compared to their present design approaches. The TRW UNAS project is discussed in some detail in the LLNL report, "Survey of Industry Methods for Producing Highly Reliable Software." This report was published as NUREG/CR-6278.

In summary, the staff's review of WCAP-14080 indicates that it contains specific design information related to the concerns regarding software safety and reliability raised in Q420.120, as well as those raised in the forum sponsored by the Royal Academy of Engineers. However, this information was not provided in Revision 2 of the Tier 1 Material for the staff to complete its evaluation, which resulted in DSER Open Item 7.2.6-1. In Revision 3 of Section 2.5.1, "Diverse Actuation System," and Section 2.5.2, "Protection and Safety Monitoring System," of the AP600 Tier 1 material, Westinghouse provided additional details of the design process. The Tier 1 Material includes a description of the lifecycle stages for which the process is applicable. The design process includes software management, configuration management, and verification and validation. The staff concludes that the planned AP600 software system architecture described in WCAP-14080 can be verified by the ITAAC process described in Sections 2.5.1 and 2.5.2 of the Tier 1 Material. Therefore, DSER Open Item 7.2.6-1 is closed.

7.2.7 Protection Systems Setpoint Methodology

The RTS setpoints are listed in the RTS technical specification (TS); however, the actual setpoints will not be established at this time. The COL applicant will provide the specific setpoints on the basis of the specific I&C system design and equipment before fuel load. The general requirement for setpoints is that they be established high enough to preclude inadvertent actuation, but low enough to ensure that a proper margin is maintained in the setpoint determination. The technical specifications are addressed in Chapter 16 of this report.

In Appendix 1A of the SSAR, Westinghouse stated that the AP600 design conforms to Regulatory Guide (RG) 1.105, "Instrument Setpoints for Safety-Related Systems." In its response to Q420.95 dated June 17, 1993, Westinghouse stated that the details of the AP600 setpoint study will be provided during the equipment procurement phase. In the DSER, the staff stated that the setpoint methodology must be submitted to support the design certification review. Therefore, the staff requested that Westinghouse provide the setpoint methodology document for review before the final SER is written. This was identified as DSER Open Item 7.2.7-1.

By letter dated May 9, 1996, Westinghouse submitted proprietary topical report WCAP-14605, "Westinghouse Setpoint Methodology for AP600 Protection Systems." WCAP-14606 is the non-proprietary version of this document. The document provides the AP600 protection system setpoint methodology. However, it does not present setpoint calculations or uncertainty allowances typically seen in a Westinghouse setpoint study for an operating plant. The setpoint values that appear in the document are those assumed in the AP600 safety analyses. These values are considered nominal trip setpoints instead of the trip setpoint and allowable values as typically specified for an operating plant. The key assumptions used in the development of the setpoint methodology document include the following:

- The fuel cycle is 24 months.
- Digital electronics will be used for the AP600 protection system, using commercially dedicated systems available at the time of construction to take advantage of the latest technology.
- A 95 percent probability/95 percent confidence level is assumed in order to be consistent with thermal design procedures used to develop core limits and DNB correlation limits. This implies that an equipment testing program will be performed by the instrumentation suppliers to produce sufficient determination of uncertainties at the 95/95 level.
- Trending of instrument calibration and drift uncertainties will be a requirement of the COL applicant. This trending will reduce the impact of the uncertainties analysis on the AP600 safety analyses and technical specification setpoints. COL applicant calibration procedures are to be written consistent with the AP600 setpoint methodology.

- The setpoint methodology is consistent with the AP600 technical specifications Tables 3.3.1-1 and 3.3.2-1 single-column nominal setpoint format.

- Instrumentation operability is directly related to instrumentation equipment design. Because there is no specific instrumentation equipment at this time, instrument operability is not addressed in the setpoint methodology.

- The uncertainty algorithm is directly related to the instrumentation equipment design. Because there is no specific instrumentation equipment at this time, the only specific uncertainty algorithm included in the setpoint document for uncertainty components is "square root of the sum of the squares."

In SSAR Section 7.1.6, "Combined License Information," Westinghouse specifies that the COL applicant referencing the AP600 certified design will provide a calculation of setpoints for protective functions consistent with the methodology presented in WCAP-14605. The COL applicant will need to perform a plant-specific setpoint study when the specific instrumentation equipment is obtained. The staff considers this to be a COL action item. WCAP-14605 provides sufficient information on instrument setpoints for the COL applicant to establish setpoints for plant-specific equipment, and therefore, is acceptable. DSER Open Item 7.2.7-1 is closed. This is COL Action Item 7.2.7-1.

7.2.8 Hardware and Software Qualification

The staff identified three areas of qualification for digital systems that are not fully addressed in the SRP criteria. These areas are software qualification, electromagnetic susceptibility, and mild environmental qualification.

The AP600 PMS is dependent on the proper functioning of the software to perform its safety functions. The software standard that specifically addresses software qualification is ANSI/IEEE ANS-7-4.3.2 (1982), "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," which was endorsed by the staff in RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," in 1985. ANSI/IEEE ANS 7-4.3.2 provides a high level description of the software V&V process. Since it was published in 1982, there have been several other standards issued that provide additional guidance on software V&V and other aspects of software qualification. ANSI/IEEE ANS 7-4.3.2 has undergone a significant revision and was reissued in November 1993.

To address the issue of software qualification and to satisfy the requirements of GDC 1 and Appendix B to 10 CFR Part 50 for quality assurance, the staff requested that Westinghouse commit to software development, documentation, and V&V in accordance with the following standards:

- American Society of Mechanical Engineers (ASME) NQA2a, Part 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications"

- American National Standards Institute (ANSI)/IEEE ANS 7-4.3.2-1993, "IEEE Standard Criteria for Digital Computers in Safety Systems for Nuclear Power Generating Stations"

- IEEE 730-1989, "Software Quality Assurance Plans"
- IEEE 828-1990, "Standard for Software Configuration Management Plans"
- IEEE 829-1983, "Standard for Software Test Documentation"
- IEEE 830-1984, "Guide for Software Requirements Specifications"
- International Electrotechnical Commission (IEC) 880-1986, "Software for Computers in the Safety Systems of Nuclear Power Stations"
- IEEE 1012-1986, "Standard Software Verification and Validation Plans"
- IEEE 1033-1985, "IEEE Recommended Practice of Application of IEEE Standard 828 to Nuclear Power Generation Stations"
- IEEE 1228 (draft), "Standard for Software Safety Plans"
- IEEE 1042-1987, "Guide to Software Configuration Management"
- MIL-STD-2167A-1988, "Defense System Software Development"

Westinghouse submitted Revision 1 to topical report WCAP-13383 to address the digital protection system design process. Commitment to the above standards is addressed in the design process review in Section 7.1.3.3 of this report.

The second area of digital system qualification not fully addressed in the SRP concerns the qualification of the PMS and the other digital I&C equipment for the electromagnetic environment to which it will be exposed. This issue includes protection against electromagnetic interference (EMI), electrostatic discharge (ESD), radio frequency interference (RFI), and the capability to withstand power surges. The staff requested that Westinghouse commit to the following standards for electromagnetic environmental qualification considerations:

- ANSI/IEEE C63.12-1987, "American National Standard for Electromagnetic Compatibility Limits - Recommended Practice"
- ANSI/IEEE C37.90.2-1987, "IEEE Trial-Use Standard, Withstand Capability of Relay Systems to Radiated Electromagnetic Interference from Transceivers"
- ANSI/IEEE C62.41-1980, "Guide for Surge Voltages in Low-Voltage AC Power Circuits"
- ANSI/IEEE C62.45-1987, "Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Circuits"
- MIL-STD-461C-1987, "Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference"
- MIL-STD-462-1987, "Measurement of Electromagnetic Interference Characteristics"

- IEC 801-2, "Electromagnetic Comparability for Industrial-Process Measurement and Control Equipment, Part 2: Electrostatic Discharge Requirements"
- IEEE 518-1982, "Guide for the Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources"
- IEEE 1050-1989, "Guide for Instrumentation and Control Equipment Grounding in Generating Stations"

The standards listed above require the selection of specific test categories. Some of the standards are referenced in SRP Chapter 7, Revision 4, August 1997. Others are the IEC standards on environmental and EMC requirements which are similar to the standards referenced in the EPRI topical report TR-102323 which was approved by the staff. Verification of the appropriate selection should be performed by the COL applicant during the ITAAC implementation. The selection at the planning stages includes consideration of installation techniques (e.g., as indicated in IEEE-1050 for shielding and grounding), and verification at the site that the installed condition is enveloped by the qualification testing.

One specific feature that the staff requested be included in the SSAR is that the digital equipment be tested for the low range of the EMI spectrum, as well as the mid to upper ranges. Westinghouse had not addressed the issue of electromagnetic environmental qualification and had not committed to the appropriate standards. This was identified as DSER Open Item 7.2.8-1.

In Revision 3 of SSAR Subsection 7.1.4.1.8, Westinghouse stated that the instrumentation and control systems are to be designed in accordance with guidance provided in applicable portions of the following standards. The portions of the standards considered to be applicable are the portions that apply to instrumentation and control systems performing protection and control functions in an industrial environment:

- IEC 68-2-1, "Basic Environmental Testing Procedures," 1974
- IEC 68-2-6, "Basic Environmental Testing Procedures, Part 2.1 Tests - Tests Fc: Vibration (Sinusoidal)," 1982
- IEC 1000-4 Series; "EMC Part 4: Testing and Measurement Techniques," 1995
- IEC 1000-4-2, "Section 2: Electrostatic Discharge Immunity Test," June 1994
- IEC 1000-4-3, "Section 3: Radiated, Radio-frequency, Electromagnetic Field Immunity Test," August 1994
- IEC 1000-4-4, "Section 4: Electrical Fast Transient/Burst Immunity Test," July 1994
- IEC 1000-4-5, "Section 5: Surge Immunity Test," September 1994
- IEC 1000-4-6, "Section 6: Immunity to Conducted Disturbances, Induced by Radio-Frequency Fields," May 1994

- IEEE C62.45, "IEEE Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Circuits," 1992
- IEEE C37.90.1, "IEEE Standard Surge Withstand Capability (SWC) Tests for Protective Relays and Relay Systems," 1989
- IEEE 1050, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations," 1989
- MIL-STD-461D; "Electromagnetic Emission and Susceptibility Requirements for Control of Electromagnetic Interference" 1993
- MIL-STD-462D; "Measurement of Electromagnetic Interference Characteristics" 1993

In April 1996, the staff approved an EPRI utility working group topical report TR-102323, "Guidelines for Electromagnetic Interference Testing in Power Plants." EPRI TR-102323 contains an acceptable method of qualifying digital instrumentation and control equipment with respect to EMI environments. The staff encouraged Westinghouse to utilize EPRI TR-102323 or a similar methodology to qualify AP600 I&C equipment. In Revision 3 of the Tier 1 material, Westinghouse stated that the Class 1E equipment (including the PMS cabinets, reactor trip switchgears, MCR/RSW transfer panels, and MCR safety-related controls and displays) has electrical surge withstand capability (SWC), and can withstand the EMI, RFI, and ESD conditions that would exist before, during, and following a design-basis accident without loss of safety function for the time required to perform the safety function. With this commitment, which will accomplish the goal of EPRI TR-102323, the staff considers DSER Open Item 7.2.8-1 closed.

The third area of digital system qualification concerns the qualification of the PMS equipment in mild environmental conditions (temperature profiles) to which the equipment could be subjected during plant operation. In Appendix 3D to Section 3.11 of the SSAR, Westinghouse states that the qualification requirements conform with the environmental parameter limits as follows:

Parameter	Limit
Temperature	≤49 °C (120 °F)
Pressure	Atmospheric
Humidity	30 - 65% (typical), ≤95% (abnormal)
Radiation	≤1E+04 rads gamma, ≤1E+03 rads gamma (microprocessors)

On the basis of operating experience, the staff identified an issue concerning the possibility of local hot spots as a result of higher current densities when using digital chip designs. To maintain qualification, the electronic equipment panel is cooled by natural convection of the room air in the panels. Fans may be used to improve long-term reliability of electronic equipment, but no credit is taken for forced air circulation for thermal qualification purposes. It is desirable to have additional margin built into the design. The components should, therefore, be qualified by testing to higher temperatures than specified in the SSAR for a given room environment. The staff requested that Westinghouse address this concern in the SSAR. This was identified as DSER Open Item 7.2.8-2.

In Revision 3 of SSAR Section 7.1.4.1.6, Westinghouse stated that the digital equipment design has additional design margin to accommodate a loss of the normal heating, ventilation, and air conditioning (HVAC). The cabinets containing the digital equipment are provided with temperature sensors that provide an alarm if internal cabinet temperatures reach an excessive value. In its letter dated September 19, 1997, in response to Q640.62 (c), Westinghouse states that Section 3.11 of the SSAR defines the room environment on the basis of a loss of normal HVAC. The equipment is qualified to this specified environment. Margin is included in the process used to develop the qualification envelope as well as in the procedures used to qualify the equipment. With this clarification, the staff has verified the equipment qualification envelope and concludes that DSER Open Item 7.2.8-2 is closed.

7.2.9 RTS Evaluation Findings and Conclusions

On the basis of its review of the SSAR and other docketed references, the staff reached the following conclusions:

- The staff evaluated the PMS design description in the SSAR to confirm commitments to the SRP and the applicable regulatory guides and industry codes and standards, including the information required by Section 3 of IEEE 279-1971. On the basis of its review of the information provided in the SSAR, the staff concludes that the design meets appropriate SRP criteria. The staff also concludes that the PMS meets the design basis requirements of IEEE 279 - 1979, and the standards of IEEE 603 and IEEE 7-4.3.2.

- The PMS includes systems and components that Westinghouse has committed to design to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles as discussed in SSAR Chapter 3. Therefore, the staff concludes that Westinghouse commitments meet the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Missile Design Bases," for the PMS.

- The staff concludes that the design for the PMS described in Chapters 6 and 15 and Section 2.5.2 of the Tier 1 Material provides instrumentation to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. As described above, it appears that appropriate controls are provided to maintain the variables and systems within prescribed ranges. Therefore, the staff finds that the RTS design satisfies the requirements of GDC 13.

- The staff concludes that the design of the PMS has the capability (1) to initiate automatically the operation of the reactivity control systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. Therefore, the requirements of GDC 20, "Protection System Function," are satisfied.

- The staff concludes that periodic testing of the PMS as described in the SSAR conforms with the criteria of RG 1.22 and RG 1.118. The staff further concludes that Westinghouse commitments to IEEE 279 with regard to system reliability and testability are consistent with the requirements of GDC 21, "Protection System Reliability and Testability," and are acceptable.

- The staff concludes that the design of the PMS, as discussed in Section 7.2.2 of this report, meets the criteria of RG 1.75 for protection system independence. The design techniques, such as functional diversity and diversity in components, are designed to the extent practical to prevent loss of protection function. Therefore, the staff concludes that the PMS meets the requirements of GDC 22, "Protection System Independence."

- On the basis of the staff's review of the failure modes and effects analysis (FMEA) for the PMS, in conjunction with the studies of the PMS design for defense against common-mode failures as discussed Section 7.1.6, the staff concludes that the PMS design meets the requirements of GDC 23, "Protection System Failure Modes."

- The staff finds that the PMS is designed to meet the requirements of IEEE-279 regarding protection and control system interaction. The PMS design meets the requirements of GDC 24, "Separation of Protection and Control Systems."

- The staff concludes that the PMS satisfies the protection system requirements for malfunctions of the reactivity control system such as accidental withdrawal of control rods. Chapter 15 of the SSAR addresses the capability of the system to ensure that fuel design limits are not exceeded for such events. Therefore, the staff finds that the PMS satisfies the requirements of GDC 25.

- Based on its review of all of the above GDCs, the staff concludes that the PMS satisfies the requirements of GDC 29 that the PMS provides protection against anticipated operational occurrences.

- On the basis of Westinghouse's commitment to meet the requirements of IEEE 279-1971 and the staff's conclusions noted above, the staff concludes that the requirements of 10 CFR 50.55a(h) are satisfied.

7.3 Engineered Safety Features Actuation Systems

7.3.1 System Description

This section describes the instrumentation and controls for equipment to initiate the various ESFs. Because the engineered safety features actuation system (ESFAS) is part of the PMS, the discussion of the design and qualification of the PMS, as discussed in Section 7.2 of this report, also applies to the ESFAS. The ESFAS consists of the following equipment:

- sensors and manual inputs
- integrated protection cabinets
- ESF actuation cabinets
- protection logic cabinets
- actuation devices
- actuated equipment

The logic in the ESF actuation cabinets is two-out-of-four coincidence for bi-stable trip inputs from the integrated protection cabinets. ESF actuation logic is performed redundantly in each ESF actuation cabinet. Each voting logic element generates an actuation signal if the required coincidence of partial trips exists at its inputs. Within each ESF actuation cabinet, the signals are combined to generate a system-level signal. System-level manual actuation is also processed by the logic in each ESF actuation cabinet.

The system-level signals are broken down to the individual actuation signals through the logic cabinets to actuate each component associated with a system-level ESF. The interposing logic within each logic cabinet accomplishes this function and also performs necessary interlocking so that components are properly aligned for their safety function. Component-level manual actions are also processed in the interposing logic. Each logic cabinet computer signal is triplicated for reliability and to prevent inadvertent actuation.

The triplicated component-level signals are voted in the power interface. The power interface also transforms the low-level signals to voltages and currents commensurate with the actuation devices they operate. The actuation devices, in turn, control motive power to the final safeguards component.

7.3.1.1 Safeguard Actuation (S) Signal

The safeguard actuation (or "S") signal is used in the initiation logic of many of the engineered safety features. The "S" signal is derived from one or more of the following initiating parameters, where the numbers in parenthesis represent the actuation logic (e.g., 2/4 means two-out-of-four coincidence logic):

- low pressurizer pressure (2/4)
- low lead-lag compensated steamline pressure (2/4 in either steamline)
- low T-cold (2/4 either loop)
- high (high-2) containment pressure (2/4)
- manual initiation (1/2)

To permit startup and cooldown, the safeguards actuation signals generated from low pressurizer pressure, low steamline pressure, or low reactor coolant inlet temperature can be manually blocked when pressurizer pressure is below the P-11 setpoint. The signal is automatically unblocked when the pressurizer pressure is above the P-11 setpoint. Separate momentary controls are provided, each of which will manually reset the safeguards actuation signal in a single division. Manual reset of a safeguards actuation signal in coincidence with reactor trip (P-4) blocks the safeguards actuation signal. The absence of the P-4 signal automatically resets the blocking function. Resetting the signal does not reposition any safeguards actuation equipment, as individual components are required to latch in and seal on the safeguards actuation signal.

7.3.1.2 Engineered Safety Features Description

The ESF systems are described in Chapter 6 of the SSAR, and each is briefly described in the sections that follow.

7.3.1.2.1 Containment Isolation

The containment isolation system provides containment isolation to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary. The containment isolation system consists of the piping, valves, and actuators that isolate the containment. A signal to initiate containment isolation is generated from any of the following conditions:

- automatic or manual safeguards actuation signal
- manual initiation (1/2)
- manual actuation of containment cooling

Manual reset is provided to block the automatic actuation signal for containment isolation. Separate momentary controls are provided for resetting each division. No other interlocks or permissive signals apply directly to the containment isolation function.

7.3.1.2.2 In-containment Refueling Water Storage Tank Injection

The in-containment refueling water storage tank (IRWST) is located in the containment at an elevation slightly above the reactor coolant system loop piping. IRWST injection is possible only after the reactor coolant system has been depressurized by the automatic depressurization system (ADS) or by a loss-of-coolant accident. Squib valves in the IRWST injection lines open on a fourth stage ADS signal. Check valves, arranged in series with the squib valves, open when the reactor pressure decreases to below the IRWST injection head. Signals to align the IRWST for injection are generated from the following conditions:

- actuation of the fourth stage of the ADS

- coincidence loop 1 and loop 2 hot leg levels below low-2 setpoint for a duration exceeding a time delay (2/2)

- manual initiation (2/4, 2 switches must act simultaneously)

7.3.1.2.3 Core Makeup Tank Injection

Core makeup tank (CMT) actuation provides the passive injection of borated water into the RCS. Injection provides RCS makeup water and boration during transients or accidents when the normal makeup supply from the chemical and volume control system (CVS) is lost or insufficient. Two tanks are available to provide passive injection of borated water. CMT injection mitigates the effects of high energy line breaks by adding primary side water to maintain or recover reactor vessel water level following a loss-of-coolant accident (LOCA), and by borating to ensure recovery or maintaining the shutdown margin following a steamline break. Signals to align the CMT for injection are generated from the following conditions:

- automatic or manual safeguards actuation

- automatic or manual actuation of the first stage of the ADS

- low-2 pressurizer level (2/4)
- low wide-range steam generator level coincident with high hot leg temperature (2/4 in both steam generators)
- manual initiation (1/2)
- pressurizer water level increasing above P-12 interlock (mid-loop operation)

7.3.1.2.4 Automatic Depressurization System Actuation

The automatic depressurization system (ADS) provides a sequenced depressurization of the RCS to allow passive injection from the CMTs, accumulators, and the IRWST to mitigate the effects of a LOCA. The depressurization is accomplished in four stages, with the first three stages discharging into the IRWST and the fourth stage discharging into containment. Each of the first three stages consists of two parallel paths with each path containing an isolation valve and a depressurization (control) valve. The first stage isolation valves open on any actuation of the first stage of the ADS. The first stage depressurization valves are opened following a preset time delay after the actuation of the isolation valves. The second stage isolation valves are opened following a preset time delay after actuation of the first stage depressurization valve. The second stage depressurization valves are opened following a preset time delay after the second stage isolation valves are actuated, similar to stage one. The third stage isolation valves and depressurization valves are actuated in a similar fashion as the second stage valves. A signal to actuate the first stage of the ADS is generated from any of the following conditions:

- CMT injection alignment signal coincident with CMT level less than the low-1 setpoint (2/4 in either CMT)
- extended loss of ac power sources (1/2 per charger and 2/4 chargers)
- manual initiation (2/4, 2 switches act simultaneously)

The fourth stage of the ADS consists of four parallel paths. Each of these paths consists of a normally open isolation valve and a depressurization valve. The four paths are divided into two redundant groups with two paths in each group. Within each group, one path is designated to be substage A and the other substage B. The fourth stage ADS is actuated by any one of the following conditions:

- Stage 4 manual initiation (2/4 switches) coincident either with low RCS pressure (2/4) or with actuation of stages 1, 2, and 3.
- CMT level less than low-2 setpoint (2/4 either tank) coincident with low RCS pressure (2/4) and coincident with third stage depressurization
- coincidence loop 1 and loop 2 low-2 hot leg levels (after delay) (2/2)

The CMT injection alignment signal is latched in. A deliberate operator action is required to reset this latch, such that an ADS actuation signal is not cleared by the reset of the safeguards actuation signal.

The loss of all ac power for a period of time that approaches the 24-hour Class 1E dc battery capability will activate the ADS valves. The time holds can be manually reset after the batteries are recharged. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four 24 hour Class 1E battery chargers.

7.3.1.2.5 Reactor Coolant Pump Trip

Reactor coolant pump (RCP) trip allows the passive injection of borated water into the RCS. A signal to trip RCPs is generated from any one of the following conditions:

- automatic or manual safeguards actuation signal
- automatic or manual actuation of the first stage of the ADS
- low-2 pressurizer level (2/4)
- low wide-range SG level (2/4 in both steam generators) coincident with high hot leg temperature (2/4)
- manual initiation of CMT injection
- high RCP bearing water temperature (2/4 in affected pump)

7.3.1.2.6 Main Feedwater Isolation

Feedwater system malfunctions that may increase feedwater flow, decrease feedwater temperature, or excessively increase secondary steam flow can cause an overcooling effect on the RCS. The primary function of the main feedwater isolation is to prevent overcooling events by stopping excessive feedwater flow to the steam generators. Failure to isolate the main feedwater system following a steamline break or feed line break can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and increased containment pressure. Signals to isolate the main feedwater supply to the steam generators are generated from any of the following conditions:

- automatic or manual safeguards actuation
- manual initiation (1/2)
- high-2 steam generator narrow range water level (2/4 in either steam generator)
- low-1 RCS T-avg (2/4) coincident with P-4 permissive (2/4 division)
- low-2 RCS T-avg (2/4) coincident with P-4 permissive (2/4 division)

The RCS T-avg low-1 condition results in the closure of the main feedwater control valves. This condition may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint. The T-avg low-2 condition results in the tripping of the main feedwater pumps and closure of the main feedwater isolation and crossover valves. This condition also blocks the steam dump valves and becomes an interlock to the steam dump interlock selector switch.

7.3.1.2.7 Passive Residual Heat Removal Heat Exchanger Alignment

The passive residual heat removal heat exchanger (PRHRHX) provides emergency core decay heat removal when the startup feedwater system is not available to provide a heat sink. A signal to align the PRHRHX to passively remove core heat is generated from any of the following conditions:

- core makeup tank injection alignment signal
- first stage automatic depressurization system actuation
- low wide-range steam generator level (2/4 in either steam generator)
- low narrow-range steam generator level (2/4 in either steam generator) coincident with low startup feedwater flow (1/2 in either feedwater line)
- high-3 pressurizer water level (2/4)
- manual initiation (1/2)

Each of these conditions opens the passive residual heat removal discharge isolation valves and provides a confirmatory open signal to the inlet isolation valve. The inlet isolation valve is normally open but can be closed by the operator. These conditions override any closure signal to this valve and also close the blowdown isolation valves in both steam generators.

7.3.1.2.8 Turbine Trip

The primary function of the turbine trip is to prevent damage to the turbine resulting from water in the steamlines. Failure to trip the turbine following a steamline break or feedline break can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and increased containment pressure. A signal to initiate turbine trip is generated from any of the following conditions:

- automatic or manual safeguards actuation signal
- reactor trip (2/4)
- high-2 steam generator narrow-range water level (2/4 in either steam generator)
- manual feedwater isolation (1/2)

Each of these conditions initiates a turbine trip to prevent or terminate an excessive cooldown of the reactor, or minimize the potential for equipment damage caused by loss of steam supply to the turbine.

7.3.1.2.9 Passive Core Cooling / IRWST Containment Recirculation

The passive core cooling system (PXS) provides core cooling by gravity injection and recirculation to the IRWST for decay heat removal following an accident. There are four parallel containment recirculation paths provided to permit the recirculation of the water provided by the IRWST. Two of these paths are provided with two isolation valves in series while the remaining two paths are provided with a single isolation valve in series with a check valve. Signals to align the IRWST containment recirculation isolation valves are generated from the following conditions:

- low-3 IRWST water level (2/4) coincident with fourth stage ADS actuation
- manual initiation (2/4)
- extended loss of ac power sources (1/2 per charger, 2/4 chargers)

Manual initiation requires simultaneous actuation of two control switches to prevent inadvertent actuation. When loss of all ac power approaches the 24-hour Class 1E dc battery capability, the PXS initiation logic will activate the IRWST containment recirculation isolation valves. The timed output holds on restoration of ac power and is manually reset after the batteries are recharged. The loss of all ac power is detected by undervoltage sensors that are connected to the input of each of the four Class 1E battery chargers. No interlocks or permissive signals apply directly to the activation of the IRWST containment recirculation isolation valves.

7.3.1.2.10 Steamline Isolation

Isolation of the main steamlines provides protection in the event of a steamline break inside or outside containment. Rapid isolation of the steamlines limits the steamline break accident to the blowdown from one steam generator assuming a break upstream of the isolation valves. For a steamline break downstream of the isolation valves, closure of the isolation valves terminates the blowdown as soon as the steamlines depressurize. A signal to isolate the steamlines is generated from any one of the following conditions:

- manual initiation (1/2)
- high-2 containment pressure (2/4)
- low lead-lag compensated steamline pressure (2/4 in either steamline)
- high steamline pressure negative rate (2/4 in either steamline)
- low reactor coolant inlet temperature (2/4 in either loop)

The steamline isolation signal closes the main steamline isolation valves and the turbine stop and bypass valves. In addition to manual system-level steamline isolation, the steamline isolation valves can be closed individually.

7.3.1.2.11 Steam Generator Blowdown System Isolation

The primary function of the steam generator blowdown isolation is to ensure that sufficient water inventory is present in the steam generators to remove the excess heat being generated until the decay heat has decreased to within the PRHRHX capability. Signals to close the isolation valves of the steam generator blowdown system in both steam generators are generated from the following conditions:

- PRHRHX alignment signal
- low narrow range steam generator level (2/4 in either SG isolates the affected steam generator)

7.3.1.2.12 Containment Cooling Actuation

The passive containment cooling system (PCS) transfers heat from the reactor containment to the environment. This function is necessary to prevent the containment design pressure and temperature from being exceeded following a design-basis accident such as a LOCA or steamline break. A signal to actuate the PCS is generated from either of the following conditions:

- manual initiation (1/2)
- high-2 containment pressure (2/4)

The passive containment cooling system actuation signal opens valves that initiate gravity flow of cooling water from the passive containment cooling system water storage tank to the top of the containment shell. The evaporation of the water on the containment shell provides passive containment cooling.

7.3.1.2.13 Startup Feedwater Isolation

The primary function of the startup feedwater isolation is to stop excessive flow of feedwater into the steam generators. This function is needed in Modes 1, 2, 3, and 4 to mitigate the effects of a large steamline break or a large feed line break. Failure to isolate the startup feedwater system following a steamline or feed line break can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and increased containment pressure. Signals to isolate the startup feedwater supply to the steam generators are generated from either of the following conditions:

- low reactor coolant inlet temperature (2/4 in either loop)
- high-2 steam generator narrow range water level (2/4 in either steam generator)
- manual initiation of main feedwater isolation

The isolation by low T-cold may be manually blocked when pressurizer pressure is below the P-11 setpoint. Either of these conditions isolates the startup feedwater supply by tripping the startup feedwater pumps and closing the startup feedwater isolation and control valves.

7.3.1.2.14 Boron Dilution Block

The block of boron dilution is accomplished by closing the CVS suction valves to the demineralized water storage tanks and aligning the boric acid tank to the CVS makeup pumps.

Signals to block boron dilution are generated from any of the following conditions:

- excessive increasing rate of source range nuclear power (2/4)
- loss of ac power sources (2/2 per charger, 2/4 chargers)
- reactor trip (2/4)

The source range flux doubling signal may be manually blocked to permit plant startup and normal power operation. It is automatically reinstated when reactor power is decreased below the P-6 power level during shutdown. The loss of all ac power is detected by two-out-of-four logic on the basis of undervoltage to the battery chargers for divisions A or C coincident with an undervoltage to the battery chargers for divisions B or D.

7.3.1.2.15 Chemical and Volume Control System Isolation

The CVS makeup line is isolated following certain events to prevent overfilling of the RCS. This line is isolated on high-2 containment radioactivity to provide containment isolation following an accident. This line is not isolated on a containment isolation signal, to allow the CVS makeup pumps to perform their defense-in-depth functions. A signal to close the isolation valves of the CVS is generated from any of the following conditions:

- high-2 pressurizer water level (2/4)

- high-2 steam generator narrow range water level (2/4 in either steam generator)

- automatic or manual safeguards actuation signal coincident with high-1 pressurizer level (2/4)

- high-2 containment radioactivity (2/4)

- manual initiation (1/2)

7.3.1.2.16 Steam Dump Block

Signals to block steam dump (turbine bypass) are generated from either of the following conditions:

- low-2 RCS T-avg coincident with P-4 permissive (2/4)
- manual initiation (1/2)

The automatic block signal is also an input to the steam dump interlock selector switch for unblocking the steam dump valves used for plant cooldown. This function may be manually blocked when the pressurizer pressure is below the P-11 setpoint. The block is automatically removed when the pressurizer pressure is above the P-11 setpoint.

7.3.1.2.17 Control Room Isolation and Air Supply Initiation

Isolation of the main control room and initiation of the air supply provides a protected environment from which the operators can control the plant following an uncontrolled release of radioactivity. This function is required to be operable in Modes 1, 2, 3, and 4, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident or other design-basis accidents. Signals to initiate isolation of the main control room and to initiate the air supply are generated from any of the following conditions:

- high-2 control room air supply radioactivity level (1/2)
- loss of ac power sources (2/2 per charger, 2/4 chargers)
- manual initiation (1/2)

For loss of all ac power sources, a preset time delay is provided to permit the restoration of ac power from the offsite sources or from the onsite diesel generator before control room isolation and air supply initiation. The loss of all ac power is detected by two-out-of-four logic on undervoltage to the battery charger for division A or C coincident with an undervoltage to the battery charger for division B or D.

7.3.1.2.18 Letdown Purification Line Isolation

The CVS maintains the RCS fluid purity and activity level within acceptable limits during normal operation. The CVS purification line receives flow from the discharge of the RCPs. To preserve the reactor coolant pressure in the event of a break in the CVS loop piping, the purification line is isolated on a pressurizer water level low-1 setpoint. This aids in maintaining reactor coolant system inventory. A signal to isolate the letdown purification line is based on low-1 pressurizer water level (2/4). This function can be manually blocked when the pressurizer water level is below the P-12 setpoint. It is automatically unblocked when the pressurizer water level is above the P-12 setpoint. Letdown purification line isolation is also initiated on manual CVS isolation.

7.3.1.2.19 Containment Air Filtration System Isolation

Some design-basis accidents such as a LOCA may release radioactivity into containment where the potential exists for releases to the atmosphere in excess of acceptable site dose limits. Isolation of the containment air filtration system provides protection against radioactivity release to the atmosphere. A signal to isolate the containment air filtration system is generated by:

- containment isolation
- high-1 containment radioactivity (2/4)

7.3.1.2.20 Normal Residual Heat Removal System Isolation

The normal residual heat removal system (RNS) suction line is isolated by closing the containment isolation valves on high-2 containment radioactivity to provide containment isolation following an accident. This line is not isolated on a containment isolation signal to allow the RNS pumps to perform their defense-in-depth functions. A signal for isolating the RNS suction line is generated by the following:

- automatic or manual safeguards actuation signal
- high-2 containment radioactivity (2/4)
- manual initiation (2/4)

The RCS inner/outer isolation valves are normally closed; they are opened only for normal cooldown after the RCS is depressurized to 450 psig. An interlock is provided to protect against inadvertent opening when RCS pressure is above 450 psig. Power to these valves is administratively blocked during normal power operation.

7.3.1.2.21 Spent Fuel Pool Isolation

The spent fuel pool lines from the refueling cavity/IRWST to the spent fuel pool cooling system suction and return header to the refueling cavity/IRWST are isolated on a low level setpoint to maintain the water inventory in the spent fuel pool resulting from line leakage. The isolation signal is generated from low spent fuel pool level (2/3). No permissives or interlocks apply to this function. This function is required to be operable in Modes 1 through 6 to maintain water inventory in the spent fuel pool.

7.3.1.2.22 CVS Letdown Isolation

The CVS letdown isolation helps to maintain reactor system inventory. The isolation signal is generated from low-1 hot leg level (1/2).

As discussed above, because the ESFAS is part of the PMS, Westinghouse has committed to meet the requirements of IEEE-279-1971. This will be verified during the implementation of the ITAAC that is specified in Section 2.5.2 of the Tier 1 material. Therefore, the staff concludes that the requirements of 10 CFR 50.55(h) are met.

7.3.1.2.23 Pressurizer Heater Block

Automatically tripping the pressurizer heaters reduces the pressurizer level swell for certain non-LOCA events, such as loss of normal feedwater or inadvertent CMT operation. For small-break LOCA analysis, tripping the pressurizer heaters supports depressurization of RCS following actuation of the CMTs. Signals for blocking the operation of the pressurizer heaters are generated by the following:

- core makeup tank injection alignment signal
- high-3 pressurizer water level (2/4)

7.3.1.2.24 Steam Generator Relief Isolation

The function of the steam generator power-operated relief valve (PORV) and block valve isolation is to ensure that steam generator PORV flow path can be isolated during a steam generator tube rupture event. Signals to isolate the steam generator PORV flow path are:

- manual initiation (1/2)
- low steamline pressure (2/4 per steamline)

7.3.2 Blocks, Permissives, and Interlocks for Engineered Safety Features Actuation

The interlocks used for engineered safety feature actuation are designated as "P-xx" and are described as follows:

P-4: Derivation -- reactor trip switchgear open

Functions -- (a) permits manual reset of safeguards actuation signal to block automatic safeguards actuation

(b) isolates main feedwater if coincident with low reactor coolant temperature

(c) trip turbine

(d) block boron dilution

P-6: Derivation -- intermediate range neutron flux channels above setpoint

Functions -- allows manual block of flux doubling actuation of the boron dilution block

P-11: Derivation -- pressurizer pressure below setpoint

Functions -- (a) permits manual block of safeguards actuation on low pressurizer pressure, low compensated steamline pressure, or low reactor coolant inlet temperature

(b) permits manual block of steamline isolation on low reactor coolant inlet temperature

(c) permits manual block of steamline isolation and steam generator power operated relief valve (PORV) block valve closure on low compensated steamline pressure

(d) coincident with manual actions of (b) or (c), automatically unblocks steamline isolation on high negative steamline pressure rate

(e) permits manual block of main feedwater isolation on low reactor coolant (RC) temperature

(f) permits manual block of startup feedwater isolation on low RC inlet temperature

(g) permits manual block of steam dump block on low RC temperature

(h) permits manual block of normal residual heat removal system isolation on high containment radioactivity

P-12: Derivation -- pressurizer level below setpoint

Functions -- (a) permits manual block of core makeup tank actuation on low pressurizer level to allow mid-loop operation

(b) permits manual block of RCP trip on low pressurizer level to allow mid-loop operation

(c) permits manual block of auxiliary spray and purification line isolation on low pressurizer level to allow mid-loop operation

(d) coincident with manual action of (a), automatically unblocks IRWST injection and 4th stage ADS initiation on low hot leg level to provide protection during mid- loop operation

P-19: Derivation -- reactor coolant system pressure below setpoint.

Functions -- (a) permits manual block of chemical and volume control system isolation on high pressurizer water level

(b) permits manual block of passive residual heat removal heat exchanger alignment on high pressurizer water level

7.3.3 System Level Manual Input to the Engineered Safety Features Actuation System

Manual Controls	To Divisions			
	A	B	C	D
Manual safeguards actuation #1	A	B	C	D
Manual safeguards actuation #2	A	B	C	D
Manual chemical and volume control system isolation #1	A		C	D
Manual chemical and volume control system isolation #2	A		C	D
Manual passive residual heat removal actuation #1	A	B		
Manual passive residual heat removal actuation #2	A	B		
Manual steamline isolation #1		B		D
Manual steamline isolation #2		B		D
Manual steam generator relief isolation #1		B		D
Manual steam generator relief isolation #2		B		D
Steam/feedwater (FW) isolation and safeguards block control #1		B		
Steam/FW isolation and safeguards block control #2				D
Manual FW isolation #1		B		D
Manual FW isolation #2		B		D
Manual steam dump interlock selector #1		B		
Manual steam dump interlock selector #2				D
Pressurizer pressure safeguards block control #1	A			
Pressurizer pressure safeguards block control #2		B		
Pressurizer pressure safeguards block control #3			C	
Pressurizer pressure safeguards block control #4				D
Manual core makeup tank actuation #1	A	B	C	D
Manual core makeup tank actuation #2	A	B	C	D

Manual Controls	To Divisions
Core makeup tank actuation block control #1	A
Core makeup tank actuation block control #2	B
Core makeup tank actuation block control #3	C
Core makeup tank actuation block control #4	D
Manual containment cooling actuation #1	A B
Manual containment cooling actuation #2	A B
Manual containment isolation actuation #1	A B C D
Manual containment isolation actuation #2	A B C D
Manual depressurization system Stages 1, 2, and 3 actuation #1 & #2	A B C D
Manual depressurization system Stages 1, 2, and 3 actuation #3 & #4	A B C D
Manual depressurization system Stage 4 actuation #1 & #2	A B C D
Manual depressurization system Stage 4 actuation #3 & #4	A B C D
Manual IRWST actuation #1 & #2	A B C D
Manual IRWST actuation #3 & #4	A B C D
Manual containment recirculation actuation #1 & #2	A B C D
Manual containment recirculation actuation #3 & #4	A B C D
Manual control room isolation and air supply initiation #1	A B C D
Manual control room isolation and air supply initiation #2	A B C D
RCS pressure CVS/PRHR block control #1	A
RCS pressure CVS/PRHR block control #2	B
RCS pressure CVS/PRHR block control #3	C
RCS pressure CVS/PRHR block control #4	D
RNS isolation safeguards block control #1	A B
RNS isolation safeguards block control #2	A B
Boron dilution block control #1	A
Boron dilution block control #2	B
Boron dilution block control #3	C
Boron dilution block control #4	D
Manual RNS isolation #1 & 2	A B D
Manual RNS isolation #3 & 4	A B D

7.3.4 *Essential Auxiliary Supporting Systems*

In the AP600 advanced reactor design, many essential auxiliary supporting systems traditionally classified as safety-related are classified as non-safety-related, defense-in-depth (DID) systems that are important to safety. Their implementation requires regulatory oversight in accordance with SECY-94-084, which identifies the staff positions on technical and policy issues pertaining to the regulatory treatment of non-safety systems (RTNSS) for passive ALWRs. Systems and equipment that fall into this category are documented in a topical report WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Non-Safety-Related System Process." In the DSER, the staff stated that its review of the RTNSS issue was not complete. This was identified as DSER Open Item 7.3.3-1. The staff's evaluation of this matter is now addressed in Chapter 22 of this report. DSER Open Item 7.3.3-1 has been subsumed into that review and, therefore, is considered closed.

7.3.5 Soft Control System

The soft controls are located on each operator workstation. The supervisor's workstation and the remote shutdown workstation have the same capability as the operator workstation, but the controls are normally locked out during plant operation. On the basis of the staff's review of the interfaces between the ESF and plant operation control systems, depending on the configuration of the workstations, the soft control may be a source of vulnerability. A postulated common-mode failure of the hardware/software for the soft control system could adversely affect system level actuation of ESF equipment and components. Additional information was required with respect to workstation operation, soft control of the safety- and non-safety-related equipment, and data management between protection and control systems to enable the staff to evaluate the consequence of failures in the control system. This was identified as DSER Open Item 7.3.4-1.

In Revision 3 of SSAR Section 7.1.3.4, Westinghouse stated that these soft controls are linked to the process bus multiplexers by individual data links. The data links use fiber optic cable and are provided with error-detection capability. Incoming data consist of messages received from operator display and field devices. Outgoing messages are sent to the appropriate control device to await operator confirmation. The operator confirmation function is provided by a device that is electrically separate from the soft control device. Each soft control device can control safety-related and non-safety-related equipment; however, it is designed such that it can only communicate with a single division at any one time. When the operator desires to operate a component, the graphical operator display that is indicating the component status is presented on the operator console. This results in a message being sent to the soft control device. The soft control device then displays the appropriate control template. The operator then selects the desired control action on the template. After the operator verifies that the desired control action is properly selected, the operator actuates the confirmation device, causing the selected control action to be transmitted to the control device.

In SSAR Section 18.8, "Human System Interface Design," Westinghouse states that the soft control units are used to provide a compact alternative to the traditional control board switches. The final configuration is dependent on the results of the human system interface design process. Rapid prototyping and man-in-the-loop concept testing are used to establish that the human system interface design of the main control room adequately supports the operator performance in the anticipated range of activities and situations.

On the basis of the information provided in the SSAR and the related topical reports on the human system interface design process, the staff finds the additional information to be sufficient to address its concerns. Therefore, DSER Open Item 7.3.4-1 is closed. The soft control design interface with the operator is part of the human system interface design process that is addressed in Chapter 18 of this report.

7.3.6 ESFAS Evaluation Findings and Conclusions

The staff evaluated the ESFAS design description in the SSAR against the criteria of the SRP and the applicable regulatory guides and industry codes and standards, including the requirements of Section 3 of IEEE 279-1971, as indicated in 10 CFR 50.55a(h). The ESFAS detects a plant condition requiring the operation of ESF systems and initiates operation of those systems.

Because the ESFAS is part of the PMS, the evaluation of the design and qualification of the PMS, as discussed in Section 7.2 of this report, also applies to the ESFAS. The staff concludes that the design of the ESFAS is acceptable and meets the relevant requirements of General Design Criteria (GDC) 1, 2, 4, 13, 19-24, 29, 34, 35, 38, and 41, and 10 CFR 50.34(f), 10 CFR 50.55a(a)(1), and 10 CFR 50.55a(h).

This review was concerned with the trip parameter sensors, PMS, and protection actuation circuits. On the basis of the staff's review of the information provided in the SSAR, the staff concludes that the SSAR provides an acceptable design description and commitments to the appropriate SRP criteria which, if implemented properly, will lead to an acceptable design.

One exception to the above conclusion is the design of the CMT level instrumentation. On October 21, 1997, Westinghouse submitted a design description that shows the arrangement of differential pressure instruments used to measure the CMT level. Ten level channels are installed on each CMT, with eight of these being narrow-range level switches that are qualified for postaccident monitoring. Four of these narrow-range level switches are used to actuate the ADS Stage 1 valves and the other four level switches to actuate ADS Stage 4 valves. The remaining two level channels are wide-range level indication channels used to verify the level during normal operation, but are not qualified for postaccident monitoring.

The staff was concerned that because the CMT is full during normal operation and four level switches share one set of level taps, a postulated common-mode failure in the level sensing line could make all four level switches at each CMT falsely stick at the high position without being detected until the next surveillance period. The instrument channel operation test for CMT level is performed every 92 days and the channel calibration is performed every 24 months. A common undetectable failure will inhibit a protective action.

By letter dated January 9, 1998, Westinghouse provided information on the detailed arrangement of the differential pressure (DP) instruments used to measure the CMT level to address the staff's concern. The Westinghouse submittal provides additional details on one set of the narrow-range DP level instruments, including all valves and their FMEA. Failures like the mispositioning of calibration valves and leaks in the sensing lines are discussed. The FMEA addresses all components affecting the CMT narrow-range level switches. Westinghouse concludes that the FMEA evaluation of the CMT narrow-range level switch arrangement shows that there are no credible single failures that can result in unacceptable multiple failures of these CMT level instruments.

The staff reviewed the CMT level instruments FMEA and agrees with the Westinghouse conclusion. This conclusion is drawn on the basis of certain failure detection mechanisms and administrative controls. To validate this conclusion, the failure detection mechanisms and the administrative controls will be implemented in the CMT level instrumentation design.

The staff requested that the FMEA for the CMT level instrumentation be included in the PMS FMEA documentation, and Westinghouse committed to include the CMT level instruments FMEA as part of WCAPs 13594 (proprietary) and 13662 (nonproprietary), "Advanced Passive Plant Protection System FMEA." A reference to these WCAPs was added to SSAR

Section 6.3.7.4.1. In addition, Westinghouse included a note with the CMT level instrumentation drawing that states that the location of the upper CMT level headers should be 1 inch lower than their connection to the CMT. This is an important design feature that should be included for verification in ITAAC. Therefore, Westinghouse added Item 8.c.ix to ITAAC Table 2.2.3-4 to address this issue. The staff finds the proposed wording acceptable and, therefore, concludes the design is acceptable.

7.4 Systems Required for Safe Shutdown

7.4.1 System Description

The instrumentation and controls necessary to establish and maintain safe-shutdown conditions following an accident are designed to achieve two basic functions:

(1) maintain the core in a subcritical condition
(2) maintain adequate core cooling by removing residual heat

There are no systems specifically and solely dedicated as safe-shutdown systems. To accomplish a safe shutdown, the required functions are reactor trip, coolant circulation, boration, heat removal, and depressurization. The ESF systems are designed to establish and maintain postaccident safe-shutdown conditions for the plant. Non-safety-related systems are not required for safe shutdown of the plant. The following ESF systems and components automatically function to place the plant in a safe-shutdown condition without operator action:

- protection and safety monitoring system, including the following equipment:
  - integrated protection cabinets
  - engineered safety features actuation cabinets
  - qualified data processing system
  - protection logic cabinets
  - reactor trip switchgear
- passive core cooling system, including the following equipment:
  - PRHRHX
  - CMT
  - accumulators
  - IRWST
  - automatic depressurization valves
- passive containment cooling system
- Class 1E dc and uninterruptable power supply system
- containment isolation valves
- reactor system, including the control rods

For establishing safe-shutdown conditions, control is possible from either the main control room or the remote shutdown workstation. The monitoring instrumentation available in the main control room for safe shutdown is safety-related and is part of the qualified data processing system (QDPS). Information on the QDPS is also available at the remote shutdown workstation.

The following is a description of a typical postaccident plant safe shutdown sequence, using only safety-related equipment and with the following assumptions:

- The turbine is tripped.
- The reactor is tripped.
- ESF actuation is provided by safety-related on site dc power.
- The RCS is intact.
- No safeguards actuation signal has been generated.

At the start of the event, loss of ac power trips the reactor coolant pumps, and RCS natural circulation flow initiates and transfers core heat either to the steam generators or to the PRHRHX. The PRHR system is initiated by a low steam generator water level signal, and the PRHRHX removes core decay heat by transferring it to the IRWST. As RCS cooldown continues, the RCS pressure decreases. A safeguards signal is initiated on low RCS pressure, which actuates the CMTs. The CMTs, in turn, provide borated water injection to the RCS. If there is RCS leakage, the CMT level decreases to a setpoint that actuates the ADS. Once the ADS sequence initiates, the plant enters lower pressure and temperature conditions for the long-term safe-shutdown cooling mode.

The safeguards signal initiated on low reactor coolant pressure also actuates isolation of the containment penetrations. This prevents loss of water inventory from the containment and permits indefinite operation of the PRHRHX and the IRWST. Boiling in the IRWST starts about one to two hours after PRHR operation is initiated. Once boiling occurs, steam is generated inside containment and heat is transferred to the air flowing on the outside of the containment shell. As the steam condenses to water inside the containment shell, the water is returned to the IRWST via gutters at openings in the containment floor elevations. The gutter drain valves to the containment sump close on PRHR actuation. Core decay heat removal is provided indefinitely in this configuration. Westinghouse states that the IRWST has sufficient inventory to permit PRHR operation for at least 72 hours, even without the gutter drain water returning to the tank.

As containment pressure slowly increases following an accident, it reaches the setpoint that actuates the PCCS. PCCS actuation initiates water flow on the outside of the containment shell to improve heat removal from the containment through evaporative cooling to the outside air. Once the RCS and the safety systems are in this configuration, the plant is in a stable shutdown condition and RCS temperature and pressure slowly continue to decrease. The PRHRHX cools the RCS to 216 °C (420 °F) in 36 hours. However, if ac power sources have been lost, a timer automatically actuates the ADS before the batteries are discharged. The dc batteries that power the ADS valves provide power for at least 24 hours. Once the automatic depressurization system sequence initiates, the plant automatically transitions to lower pressure and temperature conditions that establish and maintain long-term safe shutdown of the plant.

In Revision 4 of SSAR Section 7.4.1, Westinghouse provided additional discussion on the safe-shutdown capability including the use of non-safety-related systems. The non-safety-related systems are normally used to support plant shutdown operation. Offsite power is also expected to be available to support safe-shutdown operation. Westinghouse states that the safe-shutdown capability using safety-related systems described in Section 7.4.1.1 is only expected to be used in the event that the non-safety-related systems are not available. The non-safety-related systems automatically actuate to establish and maintain the short-term safe-shutdown conditions. The operational philosophy following any event is to maintain appropriate safe-shutdown conditions on the basis of the duration of the shutdown, until the plant is able to restart. Cold-shutdown conditions would only be established if it becomes necessary for equipment repair or because of limitations in the ability of the non-safety-related systems to maintain safe-shutdown conditions.

7.4.2 Safe Shutdown From Outside the Main Control Room

If evacuation of the main control room (MCR) is required because of some abnormal MCR condition, the operator can establish and maintain safe-shutdown conditions at the remote shutdown workstation (RSW). The design basis for safe shutdown at the RSW is an event that requires evacuation of the MCR, coincident with the loss of offsite power and a single active failure without a concurrent design-basis accident.

One RSW is provided for the plant that is similar to the operator workstations in the MCR and is designed to the same standards. The RSW contains controls for the safety-related equipment required to establish and maintain safe shutdown. Additionally, control of non-safety-related components is available, allowing operation and control when ac power is available. The design basis for the RSW does not require the installation of safety-related, dedicated, fixed-position displays, alarms, and controls. The RSW has the same capabilities as the reactor operator's workstation in the MCR.

The RSW is provided for use following an evacuation of the MCR only. No actions are anticipated from the RSW during normal, emergency, routine shutdown, refueling, or maintenance operations. Operation from the RSW is implemented by multiple transfer switches located outside the MCR.

Each individual transfer switch is associated with only a single safety-related or single non-safety-related division. These switches are located behind an unlocked access panel. Entry into the access panel will result in alarms at the MCR workstations and the RSW. The access panel is located within the same fire zone of the RSW, which is separate from the MCR fire zone. Actuation of these transfer switches results in additional alarms at the workstations in the MCR and the RSW. Each division has one transfer switch. The transfer switch will activate the RSW multiplexer and deactivate the MCR multiplexer on a division-by-division basis. The safety-related and non-safety-related operator displays located in the MCR and at the RSW are not affected by this control transfer function.

In addition to the controls and indications provided on the RSW, the following controls are provided outside the control room:

- reactor trip capability at the reactor trip switchgear

- turbine trip capability at the turbine
- start/stop controls for the diesel generators located at each diesel generator local control panel

7.4.3 Evaluation Findings and Conclusions

In Section 7.4 of the SSAR, Westinghouse states that, in the event of a turbine or reactor trip, non-safety-related plant systems automatically function to place the plant in hot standby (i.e., a safe-shutdown condition). Additional non-safety-related systems are available to permit the operator to manually perform normal routine plant depressurization and cooldown. The SSAR also states that the ESF systems are designed to establish and maintain safe-shutdown conditions for the plant following an accident. Section 7.3 of this report discusses the evaluation of the design and qualification of the ESF actuation systems. Non-safety-related systems are not required for postaccident safe shutdown of the plant. When available, the operator will rely on the non-safety-related shutdown systems before actuating ESF systems for safe shutdown.

In the DSER, the staff stated that the SSAR did not provide sufficient information to confirm that the safety-related monitoring instrumentation provided for safe shutdown at the RSW is operational without the transfer switch in the local position. The SSAR did not indicate whether operation of the transfer switch to local disables all indications in the MCR. To maintain continuity of operation between the MCR and the RSW, the indication of the status of the parameters required for safe shutdown should be available to the operators at both locations before, during, and following transfer between the MCR and the RSW, and vice-versa. This was identified as DSER Open Item 7.4.3-1. In Revision 4 of SSAR Section 7.4.3.11, Westinghouse states that the safety-related and non-safety-related operator displays located in the MCR and on the RSW are not affected by this control transfer function. The staff finds this acceptable and, therefore, DSER Open Item 7.4.3-1 is closed.

In the DSER, the staff stated that the SSAR did not provide the design features of the transfer switch located outside the MCR. Details regarding the separation features of the transfer switch (between safety divisions, and between safety and non-safety divisions), its single failure vulnerability, and its access were needed for the staff to complete its safety determination. This was identified as DSER Open Item 7.4.3-2. In Revision 4 of the SSAR Section 7.4.3.1.1, Westinghouse states that each individual transfer switch is associated with only a single safety-related or non-safety-related division. This operator control transfer capability cannot be disabled by any single active failure coincident with the loss of offsite power. With this clarification, the staff concludes that DSER Open Item 7.4.3-2 is closed.

7.5 Safety-Related Display Information

7.5.1 System Description

This section describes the instrumentation used by the operator to monitor and maintain safe operation of the AP600 advanced reactor through operational occurrences and postaccident conditions. Westinghouse classified the variables for this instrumentation in accordance with the guidance of RG 1.97, except for the addition of the Type F classification, which is unique to the AP600 design. The six types of variables that provide information to the control room operator are as follows:

(1) Type A variables are needed to diagnose the plant status per emergency operating instructions. These variables also provide information to assist the operator in taking specified, preplanned, manually controlled actions where automatic actions are not provided to recover from design-basis accidents and achieve a safe-shutdown condition.

There are no specific preplanned, manually controlled actions for postaccident safe shutdown in the AP600 design. Therefore, no Type A variables were identified in the SSAR.

(2) Type B variables are needed to assess the process of accomplishing or maintaining the following:

- reactivity control
- reactor coolant system integrity
- reactor coolant system inventory control
- reactor core cooling
- heat sink maintenance
- containment integrity

(3) Type C variables monitor the potential for causing a gross breach of a fission product barrier.

(4) Type D variables monitor the performance of the plant safety-related systems used to attain a safe-shutdown condition by mitigating the consequences of an accident and subsequent plant recovery.

(5) Type E variables monitor the habitability of the main control room. These variables are also used in determining the magnitude of radioactivity releases, assessing releases of radioactive materials, and monitoring radiation levels and radioactivity in the environment surrounding the plant.

(6) Type F variables provide information to take specified, preplanned, manually controlled actions using non-safety-related systems to prevent the unnecessary actuation of safety-related systems. These variables are also used to monitor the non-safety-related systems used to mitigate the consequences of an accident and subsequent plant recovery, and are normally used for plant cooldown-to-shutdown conditions.

The design and qualification requirements of the instrumentation for the different variable types are divided into three categories, as follows:

(1) Category 1 instrumentation requires seismic and environmental qualification, application of the single-failure criterion, use of Class 1E power, and an immediately accessible display.

(2) Category 2 instrumentation requires environmental and seismic qualification commensurate with the required function. It may require Class 1E power, but not the application of the single-failure criterion or an immediately accessible display. It requires a rigorous performance verification for a single instrument channel.

(3) Category 3 instrumentation does not require qualification, application of the single-failure criterion, use of Class 1E power, or an immediately accessible display. It meets high-quality, commercial-grade qualification.

Westinghouse identified all postaccident monitoring variables in Table 7.5-1 of the SSAR. Table 7.5-1 also provides information associated with each variable, including instrument ranges, type and category, qualification status, number of instruments required, power supply classification, and whether or not information is available as part of qualified postaccident indication on the QDPS. On the basis of its review of the information provided in the SSAR, the staff concludes that the safety-related display information system for the AP600 plant is designed in accordance with the guidelines of RG 1.97.

7.5.2 Alarm System

The alarm system is a monitoring system required to operate for normal and off-normal plant conditions. It is not required for postaccident management or accident mitigation, and it is not a Class 1E system.

The objective of the alarm system is to alert the operating staff when the plant has deviated from its expected operating envelope. The audible alarm directs the operating staff's attention to the visual portions of the alarm system to investigate the abnormality.

Graphic presentations of the alarm messages are displayed at the overview panel in the MCR with spatially dedicated indications of the process abnormality. To reduce the overload messages for the operating staff, the alarm messages are prioritized into three groups:

(1) process abnormalities that can be corrected from the control room

(2) process abnormalities that can be corrected at local control panels outside the control room

(3) process abnormalities related to automatic system status

Group 1 abnormalities (corrected from the control room) have the highest priority. Groups 2 and 3 have a lower priority and are displayed on separate and distinct portions of the overview panel or on the alarm system support displays. To attract the operating staff's attention to the overview panel messages, a number of display dynamics (such as flashing, dimming, underlining, and color coding) are used in the design of the graphical displays.

Alarm response procedures are developed as part of the human-system interfaces (HSI) design process, as discussed in Chapter 18 of this report.

The alarm system is designed to be very reliable because it is the starting point for many of the operator's decision-making activities and the entry point, or index, into specific control operations. The alarm system is designed to be available during the following modes of plant operation:

- normal power operation
- plant startup
- hot standby
- hot shutdown
- cold shutdown
- refueling
- abnormal plant conditions

The alarm system is considered to be unavailable in the following cases:

- when information about those messages that are normally presented on the overview panel displays cannot be provided, no matter what the cause may be

- when time responses and data latency requirements cannot be achieved (including loss of communication with the rest of the instrumentation and control system)

- when there are no means for displaying the alarm system support panel displays (Without the alarm system support panel displays, the main control room area operators do not have access to the queues of alarm messages currently not displayed on the overview panel.)

When there are no means to display the queues of alarm messages, the operator can use the information displays, where many of the alarm setpoints are displayed, and observe the condition of the process variables. If information cannot be displayed on a video display unit, the operator is instructed to go to the center area where the QDPS displays the critical safety function status. The operator can obtain information regarding high-level plant activities from the QDPS display.

The alarm system contains internal redundancy of its constituent hardware. The alarm support panel displays are redundant in the MCR. The alarm system support panel displays can be used as a backup for the overview panel message display. The software reliability of the alarm system is handled through software structure analysis/structure design, and the software V&V program. The alarm system is able, upon the loss of power, to automatically restart once power has been restored. The power supply for the alarm system is provided by redundant non-Class 1E dc power supplies. The alarm system supports a "dark board" philosophy; that is, when there are no abnormalities in the plant's processes, there are no alarm messages to the operator.

7.5.3 Plant Information System

The non-Class 1E plant information system generates dynamic displays of plant parameters and alarm status for the operators. The system also provides plant data analysis, plant data logging, historical storage and retrieval, and operational support for plant personnel. It uses high-resolution, color graphic video display units (VDUs) located on the various operations and control center workstations to display plant process data, and uses the data highway concept to make process data available throughout the plant. The user station consists of an alarm, log, data link, and other software servers to convert the process data for its specific usage. The amount and type of plant data presented via the plant information system is determined from the output of the function-based task analysis discussed in Chapter 18 of this report.

7.5.4 Operation and Control Centers System

The non-Class 1E operation and control center system (OCS) includes the MCR, remote shutdown room, technical support center, emergency operations facility, local control stations, and associated workstations for each of these centers. Each workstation includes the following components:

- VDUs
- control display units
- plant communication equipment
- screen selector controls
- component selector controls
- keyboard

The physical makeup of the operator workstations is identical. However, different graphics and controls are assigned to each workstation during startup and shutdown operations. The supervisor's workstation is nearly identical to the operator's workstations except that its controls are normally locked out. If an operator's workstation encounters operational problems, the failed workstation is locked out, the supervisor's workstation controls are unlocked, and the operator can use the supervisor's workstation as a backup to the failed workstation.

A dedicated display panel provides the qualified plant information system VDUs and the dedicated safety system controls. These VDUs are the only monitoring display devices in the MCR that are seismically qualified and provide qualified postaccident monitoring capabilities.

7.5.5 The Qualified Data Processing System

The qualified data processing system (QDPS) is a Class 1E system that provides postaccident monitoring information at the MCR and remote shutdown workstation. The QDPS provides status information on the postaccident Category 1 variables and selected Category 2 or 3 variables, as determined from the function-based task analysis. It also provides a set of system-level displays to support the emergency procedures and aid the operator in implementing function restoration and plant recovery.

The QDPS consists of the division-separated data processors and is comprised of three qualified major subunits:

(1) remote input/output (I/O) unit
(2) display processing unit
(3) qualified display unit

The AP600 I&C architecture design incorporates two sets of remote I/O units and display processing units in individual single-bay cabinets assigned to separate divisions. Each remote I/O unit transmits its data to its associated display processing unit. Each display processing unit, in turn, drives qualified display units in the MCR, and displays at the remote shutdown room and locally. The two display processing units receive data from the IPCs in all four divisions and share data over isolated data links. The QDPS uses the same microprocessor subsystem architecture and I/O cards as the PMS, and is housed in standard seismically qualified cabinets. Therefore, the evaluation of the design and qualification of the PMS, as discussed in Section 7.2 of this report, also applies to the QDPS.

7.5.6 Bypass and Inoperable Status Information

In RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," the staff states that the operator needs to know the operating status of safety-related systems, and the extent to which safety-related systems have been bypassed. The AP600 design incorporates this information into the alarm system, the operator's workstation, and the wall panel information system in the MCR. High-level plant status during any plant state is continuously available on the wall panel information system. At the operator's workstation, physical and functional displays show how a component's availability or unavailability impacts the alignment and availability of the system. This is indicated on the display that includes the bypassed or deliberately induced inoperability of the protection system and the systems actuated or controlled by that protection system. Alarms on the operator's workstation and the wall panel information system indicate abnormal conditions. Improper safety system alignments, safety-related component unavailability, and bypassed protective functions are considered in the alarm logic. This information is continuously monitored by the alarm system.

Indication is provided for the following conditions in accordance with RG 1.47:

- inoperability of any redundant portion of the reactor protection system, systems actuated or controlled by the reactor protection system, and auxiliary or supporting systems that must be operable for the protection system and the system it actuates to perform their safety-related functions

- inoperability expected to occur more frequently than once a year

- inoperability expected to occur when the affected system is normally required to be operable

- manually initiated inoperability

On the basis of the above discussion, the staff concludes that the AP600 design conforms with the guidelines of RG 1.47 regarding indicating the operating status, including the bypassed status, of safety-related systems.

7.5.7 Incore Instrumentation System

Instead of the traditional movable detector system used in most of the operating PWR plants, the AP600 plant design includes a fixed in-core detector system to measure in-core neutron flux distribution. In the DSER, the staff stated that the SSAR did not describe this system. In its response to Q492.5 dated July 25, 1994, Westinghouse agreed to provide information on the employment of fixed in-core detectors in conjunction with an online power distribution monitoring system. This was identified as DSER Open Item 7.5.8-1.

In Revision 3 of the SSAR Section 4.4.6.1, Westinghouse states that the primary function of the in-core instrumentation system (IIS) is to provide a three-dimensional flux map of the reactor core. This map is used to calibrate neutron detectors used by the PMS as well as to optimize core performance. A secondary function of the IIS is to provide the PMS with the signals necessary for the inadequate core cooling monitor.

The IIS consists of in-core instrument thimble assemblies, which house fixed in-core detectors; core exit thermocouple assemblies contained within an inner and outer sheath assembly; and associated signal processing and data processing equipment. There are 38 in-core instrument thimble assemblies, each of which is composed of multiple fixed in-core detectors and one thermocouple. The thimbles are inserted into the active core through the upper head and internals of the reactor vessel. The signal output from the fixed in-core detectors are digitized inside containment and multiplexed out of containment. The signal processing software integral to the IIS allows the fixed in-core detector signals to be used in an accurate three-dimensional core power distribution that is suitable for developing calibration information for the ex-core nuclear instrumentation input to the overtemperature and overpower Delta T reactor trip setpoints. The system is also capable of accurately determining whether the reactor power distribution is currently within the operating limits defined in the technical specifications while the reactor is operating above approximately 20 percent of rated thermal power.

The IIS data processor receives the transmitted digitized fixed in-core detector signals from the signal processor and combines the measured data with analytically derived constants and certain other plant instrumentation sensor signals to generate a full three-dimensional indication of nuclear power distribution in the reactor core. The minimum set of in-core monitor assemblies necessary to support operation is at least 27, with at least two in each quadrant before nuclear model calibration, and at least 19, with at least two operating assemblies in each quadrant after nuclear model calibration. The nuclear model calibration is performed after each new core load. The hardware that performs the on-line power distribution monitoring is configured such that a single hardware failure will not necessitate a reactor maximum power reduction or restrict normal reactor operations.

During plant operation, the in-core instrument thimble assembly is positioned within the fuel assembly and exits through the top of the reactor vessel to containment. The fixed in-core detector and core exit thermocouple cables are then routed to different data conditioning and processing stations. The data is processed and the results are available for display in the MCR.

Because the signal from the IIS is used to calibrate the ex-core nuclear instrumentation input to the overtemperature and overpower reactor trip setpoints, the quality of the hardware and software of the IIS needs to be equivalent to the PMS. To ensure the quality of the IIS, the ITAAC for the system design process will be applied. In Revision 3 of Section 2.5.5 of the AP600 Tier 1 Material, Westinghouse provided the design description of the in-core instrumentation and design commitment in the ITAAC table. Because this ITAAC applies to the IIS, the staff finds the information acceptable. Therefore, DSER Open item 7.5.8-1 is closed.

7.5.8 Special Monitoring System

The digital metal impact monitoring system is a non-safety-related special monitoring system that monitors the reactor coolant system for metallic loose parts. It consists of several active instrument channels, each comprising a piezoelectric accelerometer (sensor), signal conditioning, and diagnostic equipment. In Section 4.4.6.4 of the SSAR, Westinghouse states that conformance with RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," is described in Section 1.9.1. However, in the DSER, the staff stated that RG 1.133 was not addressed in that section. This was identified as DSER Open Item 7.5.9-1.

In Revision 3 of SSAR Section 4.4.6.4, Westinghouse provided detailed information with respect to conformance of this system with RG 1.133. The staff's evaluation of this issue is addressed in Section 4.4 of this report. In SSAR Section 4.4.7, Westinghouse stated that the COL applicant referencing the AP600 certified design will address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the SSAR. The staff finds this acceptable, therefore, DSER Open Item 7.5.9-1 closed.

7.5.9 Evaluation Findings and Conclusions

The information systems important to safety provide the operator with information on the status of the plant to allow manual safety actions to be performed when necessary. The scope of the staff's review included tables of system variables and component states to be indicated, and descriptive information. The review included the applicable acceptance criteria, guidelines, and design bases, including those indicating bypassed or inoperable status of safety-related systems. The review also included Westinghouse's analyses of the manner in which the design of the information systems conforms to the acceptance criteria and guidelines that apply to these systems as noted in the SRP. The staff concludes that the information systems meet the criteria of RG 1.47 for indication of bypass and inoperable status and, therefore, are acceptable.

In Table 7.5-1 of the SSAR, Westinghouse identifies the variables and their appropriate design bases and qualification criteria for instrumentation employed by the operator for monitoring conditions in the reactor coolant system, the secondary heat removal system, the containment, and the systems used for attaining a post-accident safe-shutdown condition. The instrumentation is used by the operator to monitor and maintain the safety of the plant throughout operating conditions that include anticipated operational occurrences and post-accident conditions. The staff concludes that the AP600 design meets the criteria of RG 1.97 for post-accident monitoring, as described in Section 7.5.1 of this report, and is, therefore, acceptable.

The QDPS is part of the PMS; therefore, the issue of hardware/software qualification of the QDPS was also a part of DSER Open Item 7.1.4-1 identified in Section 7.1.4 of this report. Westinghouse classified the following as non-safety-related systems:

- alarm system
- plant information system

- operation and control centers system
- in-core instrumentation system
- special monitoring system

The amount and type of information provided will be determined from the output of the function-based task analysis. The staff's evaluation of the task analysis is addressed in Chapter 18 of this report.

7.6 Interlock Systems Important to Safety

The areas reviewed in Section 7.6 of the SSAR include those interlock systems that reduce the probability of occurrence of specific events or verify the state of a safety system. These systems include interlocks to prevent overpressurization of low-pressure systems and interlocks to verify availability of safeguard functions. Isolation of non-safety-related I&C systems from safety-related I&C systems is accomplished by engineered safeguard signals, as discussed in Section 7.3 of this report.

The staff reviewed the interlock systems to confirm that such design considerations as redundancy, independence, single failures, qualification, bypasses, status indication, and testing are consistent with the design bases of these systems and commensurate with the importance of the safety functions to be performed.

7.6.1 Normal Residual Heat Removal Isolation Valves

An interlock is provided for the normally closed, motor-operated normal residual heat removal system (RNS) inner and outer suction isolation valves (RNS-V001A and B, V002A and B). The interlock prevents the RNS suction valves from being opened by operator action unless the RCS pressure is less than a preset pressure of 3100 kPa (450 psig) and the IRWST suction and discharge valves are in a closed position.

There are two parallel sets of motor-operated valves in series in the RNS pumps' suction lines from the RCS hot leg. The two valves near the RCS hot leg are designated as the inner isolation valves; the two valves near the RNS pumps are designated as the outer isolation valves. Logic for operation of the outer valves is similar to that provided for the inner valves. The logic for an outer valve has an additional interlock for valve opening upon high containment radiation input. The pressure transmitter used for valve interlocks on the inner valves is diverse from the pressure transmitter used for the outer valve interlocks. These four motor-operated valves are powered from safety-grade 125-V dc busses. The inner valve is powered by a separate power supply from the outer valve of each series combination. The valves may be closed by operator action from the MCR at any time. During extended normal residual heat removal operations following cooldown, the isolation valves' motor breakers are locked open to prevent an inadvertent closure of the valves. Alarms are provided in the MCR and on the remote shutdown workstation to alert the operator if RCS pressure exceeds the RNS design pressure after the valves are opened.

The safety function of the RNS isolation valves is to remain closed (i.e., the interlocks prevent the valves from being opened while the reactor is pressurized). In the unlikely event that two RNS isolation valves are opened at power, the RNS relief valves provide system overpressure protection. The potential for the creation of an intersystem LOCA as a result of RNS overpressurization is addressed in Sections 3.9.3.1, 5.2.5.3, 19.2.2.1.5, and 20.3 (Generic Safety Issue 105) of this report. In the DSER, the staff stated that no information was provided in the SSAR regarding how many pressure transmitters are used as input to the interlock logic, how these valves are controlled, and how the logic meets IEEE 279 criteria. This was identified as DSER Open Item 7.6.1-1.

In Revision 3 of the SSAR, Westinghouse identifies the number of pressure transmitters and the interlock logic for the RNS isolation valves. The isolation valve interlock logic is provided in SSAR Figure 7.2-1, Sheet 18. Because of the possible severity of the consequences of loss of function, the RNS has the following design features:

- The protection system function for RNS isolation is provided by two parallel sets of two valves in series. The interlock components are redundant with the inner valve powered by a separate power supply from the outer valve of each series combination.

- The pressure interlock signals and logic are tested online. This test includes the initiating signals for the interlocks from the protection logic cabinets.

The staff finds that the design for RNS isolation satisfies the single failure criterion and the online testability requirements of IEEE 279-1971, and is acceptable. Therefore, DSER Open Item 7.6.1-1 is closed.

7.6.2 Accumulator Isolation Valves

In the DSER, the staff stated that it had determined that the accumulator isolation valve interlock signals were not shown on the Process Block Diagrams in Section 1.7 or on Figure 7.2-1 of the SSAR. Therefore, there was insufficient design detail to confirm conformance to IEEE 279 criteria. The staff also requested additional detail regarding how many pressure transmitters are used in the interlock logic. This was identified as DSER Open Item 7.6.2-1.

In Revision 13 to the SSAR, Westinghouse states that the accumulator isolation valves are safety-related to retain their pressure boundary and remain in their open position. The accumulator isolation valve operators are non-safety-related as the valves are not required to change position to mitigate an accident. The SSAR Chapter 15 safety analyses assume that these valves are not subject to valve mispositioning (before an accident) or spurious closure (during an accident). Valve mispositioning and spurious closure are prevented by the following:

- The technical specifications (SSAR Chapter 16) require these valves to be open and power locked out whenever these injection paths are required to be available when the reactor coolant system pressure is above 1000 psia.

- The technical specifications (SSAR Chapter 16) require verification that the motor-operated valves are open every 24 hours. They also require verification that power is removed every 31 days.

- With power locked out, redundant (non-safety-related) valve position indication is provided in the MCR and remote shutdown workstation. Valve position indication and alarm are provided to alert the operator if these valves are mispositioned. These indications are powered by different non-safety-related power supplies.

In addition, the valves have a confirmatory open signal during an accident (safeguards actuation signal for accumulator motor-operated valves). The valves also have an automatic open signal when their close permissives (P-11 for accumulator motor-operated valves) clear during plant startup. The confirmatory open and the automatic open control signals are provided to the valve operator by the non-safety-related plant control system. The staff finds this clarification acceptable. Therefore, DSER Open Item 7.6.2-1 is closed.

7.6.3 IRWST Motor-Operated Discharge Valves

In the DSER, the staff stated that it had determined that the IRWST motor-operated discharge valve interlock signals were not shown on the Process Block Diagrams in Section 1.7 or on Figure 7.2-1 of the SSAR. There was insufficient design detail to confirm conformance to IEEE 279 criteria on how the interlock logic and valve controls are configured. This was identified as DSER Open Item 7.6.3-1.

In Revision 13 to the SSAR, Westinghouse states that the in-containment refueling water storage tank injection isolation valves are safety-related to retain their pressure boundary and remain in their open position. The in-containment refueling water storage tank injection valve operators are non-safety-related as the valves are not required to change position to mitigate an accident. The SSAR Chapter 15 safety analyses assume that these valves are not subject to valve mispositioning (before an accident) or spurious closure (during an accident). Valve mispositioning and spurious closure are prevented by the following:

- The technical specifications (SSAR Chapter 16) require these valves to be open and power locked out whenever these injection paths are required to be available. Both in-containment refueling water storage tank injection lines are required to be available in Modes 1, 2, 3, and 4. One in-containment refueling water storage tank injection line is required to be available in Mode 5, and in Mode 6 with the reactor upper internals not removed and the refueling cavity not filled.

- The technical specifications (SSAR Chapter 16) require verification that the motor-operated valves are open every 24 hours. Verification that power is removed every 31 days is also required.

- With power locked out, redundant (non-safety-related) valve position indication is provided in the MCR and remote shutdown workstation. Valve position indication and alarm is provided to alert the operator if these valves are mispositioned. These indications are powered by different non-safety-related power supplies.

In addition, the valves have a confirmatory open signal during an accident (safeguards actuation signal for automatic depressurization system Stage 4 signal for in-containment refueling water storage tank motor-operated valves). The valves also have an automatic open signal when their close permissives (P-12 for in-containment refueling water storage tank motor-operated valves) clear during plant startup. The confirmatory open and the automatic open control signals are provided to the valve operator by the non-safety-related plant control system. The staff finds this clarification acceptable. Therefore, DSER Open Item 7.6.3-1 is closed.

7.6.4 Passive Residual Heat Removal Heat Exchanger Inlet Isolation Valve

In the DSER, the staff stated that it had determined that the passive residual heat removal heat exchanger (PRHRHX) inlet isolation valve interlock signals were not shown on the Process Block Diagrams in Section 1.7 or on Figure 7.2-1 of the SSAR. There was insufficient design detail to confirm conformance to IEEE 279 criteria on how the interlock logic and valve controls are configured. This was identified as DSER Open Item 7.6.4-1.

In Revision 5 of the SSAR, Westinghouse provides design information on the PRHRHX inlet isolation valve interlock. The isolation valve interlock logic is shown in SSAR Figure 7.2-1, Sheet 17. The PRHRHX inlet line includes a normally open motor-operated isolation valve that can be manually controlled from either the MCR or the remote shutdown workstation. The valve is interlocked so that the following conditions occur:

- If the maintain-closed actuation signal has not been manually initiated, it opens automatically on receipt of a confirmatory open signal with the control circuit in automatic or manual control.

- It cannot be manually closed when a confirmatory open signal is present.

During normal plant operation and shutdown, the PRHRHX inlet isolation valve is open. To prevent an inadvertent closure of the valve, redundant output cards are used in the protection logic cabinet.

This normally open motor-operated valve has alarms indicating valve mispositioning (with regard to the passive core cooling function). The alarm actuates in the MCR and the remote shutdown workstation.

The staff finds that the interlock logic design of the PRHRHX inlet isolation valves is consistent with the requirements of IEEE 279 criteria for safety-related functions and is acceptable. Therefore, DSER Open Item 7.6.4-1 is closed.

7.6.5 Core Makeup Tank Cold-Leg Balance Line Isolation Valves

In Revision 5 of the SSAR, Westinghouse provides the CMT cold-leg balance line isolation logic in SSAR Figure 7.2-1, Sheet 17 and describes additional design information, as follows:

Each CMT has a cold-leg balance line, which is provided with a normally open, motor-operated, isolation valve. The balance line isolation valves for each CMT may be manually controlled from either the MCR or the remote shutdown workstation. A confirmatory open signal to these valves automatically overrides any bypass features that are provided to allow the balance line isolation valve to be closed for short periods of time. The control circuit has a valve maintain-closed actuation function to provide an administratively controlled manual block of the automatic opening of the valve when the pressurizer level is greater than the P-12 interlock.

This function allows the valve to be maintained closed if needed for leakage isolation. The maximum permissible time that a CMT cold-leg balance line isolation valve can be closed is specified in technical specifications. An alarm is actuated when the maintain closed function is reinstated.

Each valve is interlocked so that the following conditions occur:

- If the maintain-closed actuation signal has not been manually initiated, the valve opens automatically on receipt of a confirmatory open signal with the control circuit in automatic or manual control.

- The valve opens automatically whenever the pressurizer water level increases above the P-12 interlock and the control circuit is in automatic control.

- The valve cannot be manually closed when a confirmatory open signal is present.

During power and shutdown operations, the CMT cold leg balance line isolation valve remains open. To prevent an inadvertent closure of the valve, redundant output cards are used in the protection logic cabinet.

These normally open motor-operated valves have alarms indicating valve mispositioning (with regard to their passive core cooling function). The alarms actuate in the MCR and the remote shutdown workstation.

The staff finds that the interlock logic design of the CMT cold-leg balance line isolation valves is consistent with the requirements of IEEE 279 criteria for safety related functions and, therefore, is acceptable.

7.6.6 Evaluation Findings and Conclusions

The review of the interlock systems important to safety included the interlocks for the following valves:

- RNS isolation valves to prevent overpressurization of low pressure systems when connected to the primary coolant system

- Accumulator isolation valves

- IRWST discharge isolation valves

- PRHR heat exchanger inlet isolation valve

- CMT cold leg balance line isolation valves

The staff concludes that the interlock system design meets the criteria of IEEE 279 for safety related functions and, therefore, is acceptable.

7.7 Control and Instrumentation Systems

7.7.1 System Description

The I&C systems reviewed in this section include control systems used for normal operation (that is, systems which are not relied on to perform safety functions following anticipated operational occurrences or accidents, but that control plant processes that may affect plant safety). These control systems perform the following normal operating and normal startup/shutdown functions:

- power control
- rod control
- pressurizer pressure control
- pressurizer water level control
- feedwater control
- steam dump control
- rapid power reduction
- defense-in-depth control

In addition, this section addresses the review of the DAS that provides a diverse backup to the protection system and mitigates the consequences of ATWS events. This section also addresses the review of non-safety-related DID systems that the PRA review has determined to be risk significant. These systems were reviewed on the basis of the guidance provided in SECY-94-084 on the RTNSS process.

7.7.1.1 Power Control System

The power control system performs automatic reactor power control and power distribution control by varying the position of the control rods. Separate control banks are used to regulate reactor power and power distribution. The power control system enables the plant to respond to load changes for plus and minus ten percent step load changes, and ramp load increases and decreases of five percent per minute. The system also enables daily load-follow operation. These capabilities are accomplished without resulting in a reactor trip or steam dump actuation. The power control system uses a control strategy for the control rods that regulate core power separate from that for the rods that regulate axial offset control. The power control system controls the reactor coolant average temperature by regulating the positions of control rod banks. The reactor coolant loop average temperatures are determined from hot and cold leg measurements in each reactor coolant loop. The programmed coolant temperature increases linearly with turbine load. The temperature input signals are fed from protection channels via isolation devices. Deviation of the reactor coolant temperature from the programmed value is the basic control variable for reactor power control.

A separate control strategy is used at low-power levels when the turbine is offline and the steam dump system is used to regulate coolant temperature. In this mode, the operator enters a power level setpoint and a desired rate of change into the setpoint calculator. The nuclear power setpoint calculator then supplies a linear ramp change in core power at the selected rate to the new setpoint.

Axial offset control is performed by the axial offset rods. Measurements of axial offset are put into the control system and then compared to an axial offset control range or window. When the axial offset error is outside the acceptable control window, the axial offset rods actuate at a fixed speed to recover the axial offset.

To minimize the potential for interactions between the reactor power and the axial offset rod control systems, the reactor power control system takes precedence. If a demand signal exists for movement of the power control rods, the axial offset rods are blocked from moving.

7.7.1.2 Rod Control System

The rod control system receives rod speed and direction signals from the power control system and the axial offset control systems. For power control, the rod speed demand signals vary over the range of 5 to 45 inches per minute (8 to 72 steps per minute). For axial offset control, the rod speed demand signals are fixed at a constant speed of 5 inches per minute (8 steps per minute). Manual control is provided to move a bank in or out at a prescribed fixed speed. In the automatic mode, the rods are withdrawn or inserted within the limits imposed by the control interlocks. The power and axial offset control banks are the only rods that can be manipulated under automatic control.

The three shutdown banks are always in the fully withdrawn position during normal operation, and are moved to this position at a constant speed by manual control before criticality. A reactor trip causes them to fall by gravity into the core.

The variable speed rod drive programmer used in the power control system inserts small amounts of reactivity at low speed. This permits fine control of reactor coolant average temperature about a small temperature deadband, as well as furnishing controls at high speed for transients such as load rejections.

The digital rod position indication system measures the position of each rod using a detector consisting of discrete coils mounted concentrically within the rod drive pressure housing. The coils are located axially along the pressure housing and magnetically sense the entry and presence of the rod drive shaft through its center line. The demand position system counts the pulses generated in the rod drive control system and provides a digital readout of the demanded bank position. The demanded and measured rod positions are displayed in the MCR. An audible alarm is generated whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. Alarms are also generated if any shutdown rod is detected to have left its fully withdrawn position, or if any control rod bank control rods are detected at the bottom position, except as part of the normal insertion sequence. The purpose of the control bank rod insertion alarms and interlocks is to provide warning to the operator of excessive rod insertion and to terminate the insertion.

Rod stops are provided to prevent abnormal power conditions that could result from excessive control rod withdrawal initiated by either a control system malfunction or operator violation of administrative procedures. Automatic turbine load runback is initiated by an approach to core limit or overpower thermal conditions. This prevents high-power operation that might lead to an undesirable condition. A reactor trip occurs if the limit is reached.

7.7.1.3 Pressurizer Pressure Control

Pressurizer pressure control is designed to provide stable and accurate control of the primary system pressure to its predetermined setpoint. During steady-state operating conditions, the pressurizer heater output is regulated to compensate for pressurizer heat loss and a small continuous pressurizer spray. During normal transient operation, pressurizer pressure is regulated to provide an adequate margin to limit unnecessary safety systems actuation or reactor trip.

Small or slowly varying changes in pressure are regulated by modulation of the variable heater control. Decreases in pressure larger than that which can be accommodated by the variable heater control results in the actuation of the backup heaters, as does a large increase in the pressurizer water level. Pressure increases that are too fast to be handled by reducing the variable heater output result in pressurizer spray actuation. Spray continues until pressure decreases to the point that the variable heater alone is capable of regulating pressure. For normal transients, including a full-load rejection, the pressurizer pressure control system acts promptly to prevent reaching the high pressurizer pressure reactor trip setpoint.

7.7.1.4 Pressurizer Water Level Control

Pressurizer water level control provides stable and accurate control of pressurizer level within a prescribed deadband around a programmed value. As the reactor coolant system temperature is increased from zero-load to full-load value, the reactor coolant system fluid volume expands. The pressurizer level is programmed to absorb this change. A deadband is provided around the pressurizer level program to intermittently control charging and letdown. When pressurizer level reaches the lower limit of the deadband, the charging system is actuated, which continues to operate until pressurizer level is restored to a limit above the nominal programmed value. When the level reaches the upper limit of the deadband, letdown to the liquid waste processing system is actuated. Automatic pressurizer level control is supplied from the point in the startup cycle where the zero-load level is established up through 100-percent power.

7.7.1.5 Feedwater Control

Two modes of feedwater control are incorporated in the feedwater control system:

(1) In the high-power control mode, feedwater flow is regulated in response to changes in steam flow and proportional plus integral (PI) -compensated steam generator narrow-range level deviation from a setpoint.

(2) In the low-power control mode, feedwater flow is regulated in response to changes in steam generator wide-range water level, and PI-compensated steam generator narrow-range level deviation from the pre-established setpoint.

A separate low-range feedwater flow measurement is used in the low-power feedwater control mode. The transition from the low- to the high-power control mode is initiated when the transition point is low enough to allow effective feedwater control using the wide-range level. In the high-power control mode, feedwater flow indication is provided within the upper limit of the low-range feedwater flow measurement instrument. Tracking of steam generator level deviation is provided to allow a smooth transition between control modes and between manual and automatic control.

7.7.1.6 Startup Feedwater Control

The startup feedwater control system maintains a programmed water level in the shell (secondary) side of the steam generator during low power (below 10 percent), no-load, and plant heatup and cooldown modes. Transition between the main and startup feedwater line is automatically controlled on the basis of flow measurement within the respective lines. The startup feedwater system is automatically actuated on signals that indicate a loss of water inventory or heat sink in the secondary side of the steam generator. It will attempt to recover the inventory loss and return the steam generator water level to the programmed level. If the startup feedwater control system cannot recover the inventory deficit, reactor cooling is initiated by the passive residual heat removal system.

7.7.1.7 Steam Dump Control

The AP600 advanced reactor has a design objective to sustain a 100-percent load rejection, or a turbine trip from 100-percent power, without generating a reactor trip, requiring atmospheric steam relief, or opening a pressurizer or steam generator safety valve. The automatic steam dump control system, in conjunction with the rapid power reduction system, is provided to accommodate this abnormal load rejection and to reduce the effects of the transient imposed on the reactor coolant system. The steam dump system is sized to pass 40 percent of the total nominal steam flow. This capacity is sufficient to handle reactor trips from any power level, turbine trips from 50-percent power or less, or load rejections of 50 percent or less.

The steam dump control system has two modes of operation:

(1) The T_{avg} mode uses the difference between measured auctioneered loop T_{avg} and a reference temperature derived from turbine first-stage impulse pressure to generate a steam dump demand signal. This mode is used for at-power transients requiring steam dump.

(2) The pressure mode uses the difference between measured steam header pressure and a pressure setpoint to generate a steam dump demand signal. This mode is used for low-power conditions and for plant cooldown.

The load rejection steam dump controller prevents a large increase in reactor coolant temperature following a large, sudden load decrease. The error signal is the difference between the lead-lag compensated selected T_{avg} and the selected reference T_{avg} (designated T_{ref}), on the basis of turbine impulse chamber pressure. Following a sudden load decrease, T_{ref} is immediately decreased and T_{avg} tends to increase. This generates an immediate demand signal for steam dump. Following the opening of the steam dump valve, the control rods insert in a normal controlled manner to reduce the reactor power to match the turbine load. For a reactor trip situation, the load rejection steam dump controller is defeated and the plant steam dump controller becomes active. Because control rods are not available in this situation, the demand signal is the error signal between the lead-lag compensated auctioneered T_{avg} and the no-load reference T_{avg}. When the error signal exceeds a predetermined setpoint, the dump valves are opened in a prescribed sequence.

The steam header pressure control mode is manually selected by the operator. The pressure setpoint is manually adjusted on the basis of the desired reactor coolant system temperature. The controller also has a feature that allows for automatically controlled plant cooldowns at a chosen rate (within limits). The operator can enter the desired cooldown rate and the desired targeted reactor coolant system temperature. The control system then dumps the required steam to achieve the setpoint cooldown rate and the cooldown stops at the target reactor coolant system temperature setpoint.

7.7.1.8 Rapid Power Reduction

The rapid power reduction system reduces nuclear power to a level capable of being handled by the steam dump system for a large load rejection. When a large and rapid turbine load rejection (via a lead/lag circuit) is detected, the circuit provides a signal demanding the release of a preselected number of control rods. Dropping these preselected rods causes the reactor power to rapidly reduce to approximately 50-percent power. The large load rejection also actuates the steam dump system and the power control system via a primary-to-secondary power mismatch signal. Following the initial opening, the steam dump valves modulate closed based upon the (T_{avg} - T_{ref}) signal.

Controlled rod insertion and steam dump modulation continues until power is reduced to approximately 15-percent power. The plant stabilizes with the steam dump maintained to match the steam flow to the thermal load. The operators can then switch to the pressure mode of control on the steam dump system, recover the released rods, and establish normal rod control. A normal power escalation can then be performed.

7.7.2 Diverse Actuation System

The SSAR states that the DAS is a non-safety-related system that provides diverse backup to the protection system. Westinghouse states that this backup is included to support the AP600 advanced reactor's risk goals by reducing the probability of a severe accident as a result of the coincidence of postulated transients and a postulated common-mode failure in the protection and control systems. The specific functions performed by the DAS are selected on the basis of the PRA evaluation. The DAS functional requirements are determined on the basis of an assessment of the protection system instrumentation common-mode failure probabilities combined with the initiating event probability. The DAS provides automatic actuation signals, manual actuation signals, and indications for the plant operators. These signals are generated in a functionally diverse manner from the protection system actuation signals. The common-mode failure of sensors of a similar design is also considered in the selection of these functions.

The DAS automatic actuation is accomplished by a microprocessor-based system. Diversity from the PMS is achieved by using a different architecture, different hardware implementation, and different software. Software diversity is achieved by running different operating systems and programming in a different language. The DAS is subject to the following automatic actuations:

- trip control rods via the motor-generator set, trip turbine, and initiate the PRHR system on low wide range steam generator water level

- open the passive heat removal discharge isolation valves and close the IRWST gutter isolation valves on high hot leg temperature

- trip control rods via the motor-generator set, trip turbine, actuate the CMTs, and trip the reactor coolant pumps on low pressurizer water level

- isolate selected containment penetrations and start passive containment cooling water flow on high containment temperature

The selection of setpoints and time responses is determined so that the DAS automatic functions do not actuate unless the protection system has failed to actuate to control plant conditions. The DAS automatic logic combines the signals from two redundant subsystems in a two-out-of-two logic. The two-out-of-two logic is implemented by connecting the final actuation devices in series. Actuation signals are output to the loads in the form of normally deenergized, energize-to-actuate contacts, with a nominal voltage of 120 V ac or 125 V dc. The use of the normally deenergized output state, along with the dual, two-out-of-two logic, reduces the probability of inadvertent actuation.

The manual actuation of the DAS is implemented by wiring the control board-mounted switches directly to the final loads in a way that completely bypasses the normal path through the control board multiplexers, the engineered safety features actuation cabinets, the integrated logic cabinets, and the DAS automatic logic. The diverse manual functions are as follows:

- reactor and turbine trip
- passive containment cooling actuation
- CMT actuation and RCP trip
- open Stage 1 ADS valves
- open Stage 2 ADS valves
- open Stage 3 ADS valves
- open Stage 4 ADS valves
- open PRHR discharge isolation valves and close IRWST gutter isolation valves
- selected containment penetration isolation
- containment hydrogen igniter actuation
- IRWST injection initiation
- initiate containment recirculation
- initiate IRWST drain to containment

To support the diverse manual actuations, sensor outputs are displayed in the MCR in a manner that is diverse from the protection system display functions. The following indications are provided from at least two sensors per function:

- wide-range steam generator water level for reactor trip and PRHR actuations, and for overfill prevention by manual actuation of ADS valves

- hot leg temperature for PRHR actuation

- core exit temperature for ADS actuation and subsequent initiation of IRWST injection

- pressurizer level for CMT actuation and reactor coolant pump trip

- containment temperature for containment isolation and PCCS actuation
- containment hydrogen for containment hydrogen igniter actuation

The diverse actuation system uses sensors that are separate from those used by the PMS and the plant control system. This prevents failures from propagating to other plant systems through the use of shared sensors. Signal isolation is provided between the two subsystems within the diverse actuation system, one for each input and output path, to provide isolation against faults. Actuation interfaces are shared between the diverse actuation system and the PMS. The DAS actuation devices are isolated from the PMS actuation devices, so as to avoid adverse interaction between the two systems. The actuation devices of each system are capable of independent operation, which is not affected by the operation of the other system. The DAS and the PMS use independent and separate uninterruptible power supplies.

As stated in SSAR Section 7.7.1.11, the DAS is needed to mitigate ATWS events. For Westinghouse plants, the ATWS rule (10 CFR 50.62) requires diverse actuation of auxiliary feedwater and turbine trip. The DAS provides for the ATWS protection features mandated for Westinghouse plants plus a diverse reactor scram. As discussed in Section 7.1.6 of this report, the DAS is also provided as the system designed to meet the Commission-approved position on I&C system defense-in-depth and diversity, and performs the same functions as the PMS for accident mitigation when a postulated common-mode failure disables the PMS. In the DSER, the staff requested Westinghouse to provide the following additional information in the SSAR on the design of the DAS:

- **Diversity**

  DAS equipment diversity is required from the sensor output to the final actuation device. Sensor and instrument sensing lines should be selected to avoid adverse interactions with existing control systems.

- **Power Supplies**

  DAS power should be from an instrument power supply that is independent from the reactor protection system power supplies.

- **Safety-Related Interface**

  DAS hardware is not required to meet safety-related system requirements (IEEE 279). However, implementation of the DAS must ensure that the existing protection system continues to meet all applicable safety criteria (IEEE-279). All I/O equipment must be qualified to avoid generating faults that would degrade Class 1E signals from the protection system. Non-safety actuation devices (e.g., solenoid valves attached to safety system valves) should not prevent the safety system from performing its function when demanded. Non-safety-related actuation devices should be both seismically and environmentally qualified to the same level as the safety-related actuation device. DAS software should be high quality, including a fully documented software lifecycle development process similar to the development process for the safety-related protection system, but commensurate with the intended DAS function.

- **Completion of Mitigation Action**

  The DAS should be designed so that, once actuated, the mitigation action goes to completion, and the subsequent return to operation requires deliberate operator action.

- **Quality Assurance**

  The quality of the DAS should be in accordance with the guidelines of Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment that is Not Safety-Related."

- **Operability and Availability**

  Requirements for DAS operability and availability (including surveillance, testing, maintenance, and bypass of inoperable processors and sensors) should be addressed.

- **Environmental Qualification**

  DAS equipment should be qualified to the environment for anticipated operational occurrences and accidents that it is designed to mitigate.

The above request for additional information was identified as DSER Open Item 7.7.2-1. In Revision 3 of SSAR Section 7.7.1.11, Westinghouse provides the additional information necessary to address the DSER concerns. In addition, Section 2.5.1, "Diverse Actuation System" of the Tier 1 Material provides a detailed design description and design commitment in the ITAAC table. On the basis of the staff's review of this additional information, the staff concludes that this ITAAC applies to the DAS and finds that the DAS has met the requirements stated above. Therefore, DSER Open Item 7.7.2-1 is closed.

As discussed previously, the DAS design must satisfy the diversity, independent power supply, and actuation devices isolation requirements. The DAS is designed to complete the mitigation action once it actuates. The staff concludes that the mitigation action requirement is satisfied. In SSAR Section 7.7.1.11, Westinghouse has committed to meet the quality guidelines established by GL 85-06, "Quality Assurance Guidelines for ATWS Equipment that is not Safety-Related." The operability and availability issue is addressed in SSAR Section 16.3.1, "Investment Protection Short-Term Availability Controls."

In SSAR Section 7.7.1.11, Westinghouse stated that the DAS is located in a controlled environment and is capable of functioning during and after normal and abnormal events and conditions. In Revision 4 of the Tier 1 Material, submitted by letter dated April 6, 1998, Table 2.5.1-5 identified that the DAS process cabinets are located in the annex building, which is not part of the nuclear island structure. The staff was concerned that the annex building is a seismic Category II, not a seismic Category I, structure. SSAR Section 3.2.1.1.2 states that seismic Category II structures, systems, and components are designed so that a safe shutdown earthquake does not cause unacceptable structural failure or interaction with seismic Category I items. On this basis, the staff finds that the placement of DAS components in the annex building is acceptable.

7.7.3 RTNSS Review of Other Systems

As stated in Section 7.3.4 of this report, some traditional ESF auxiliary supporting systems are classified as DID systems for the AP600 design. The AP600 RTNSS are documented in a topical report WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Non-Safety-Related System Process." The evaluation of AP600 RTNSS is addressed in Chapter 22 of this report.

7.7.4 Signal Selector

The integrated control system for the AP600 advanced reactor derives some of its control inputs from signals that are also used in the integrated protection system. Isolation devices are provided to guard the protection system against possible electrical faults in the control system to meet the criterion of IEEE 279 for control and protection system interaction. Redundant signal selectors provide the plant control system with the ability to obtain inputs from the integrated protection cabinets in the protection and safety monitoring system. The signal selector function maintains the independence of the plant control system and the PMS. The signal selectors select those protection system signals that represent the actual status of the plant and reject erroneous signals. Therefore, the control system does not cause an unsafe control action to occur even if two of four redundant protection channels are degraded by random failure or by being bypassed for test or maintenance.

Each signal selector receives data from each of the redundant divisions of the integrated protection cabinets. The data is received from each integrated protection cabinet on a serial data link via fiber optic cable. The signal selectors provide validated process values to the plant control system via the process bus. They also provide the validation status, the average of the valid process values, the number of valid process values, and an alarm (if two process values have been rejected).

For the logic values received from the PMS, such as permissives, the signal selectors perform voting on the logic values to provide a valid logic value to the plant control system via the process bus. They also provide the validation status, the number of valid logic values, an alarm if one logic value differs from the voted value, and another alarm if two logic values differ from the voted value. Each signal selector provides information to the process bus.

The redundant signal selectors receive identical data from the integrated protection cabinets and perform identical selection algorithms. Each selector provides validated data to one of the redundant highways of the process bus. When there is no failure of either signal selector or either highway, the distributed controllers are free to use data from either of the signal selectors via the appropriate highway. The signal selector subsystem redundancy serves two purposes; it protects against a failure disrupting the control system, and it provides the capability to remove one of the selectors from service for testing while maintaining normal control using data from the other selector.

The staff concludes that implementing redundant signal selector subsystems in the plant control system design will improve the reliability of the control system and minimize challenges to the protection systems. The staff therefore concludes that the signal selector design is acceptable.

7.7.5 Evaluation Findings and Conclusions

The control systems that are used for normal operation are not relied on to perform safety functions, but to control plant processes having a significant impact on plant safety. These control systems include the reactivity control systems, as well as control systems for primary and secondary coolant flow. The staff review of the control systems included features of these systems for both manual and automatic control of non-safety-related process systems. The staff concludes that the control systems permit actions to be taken to operate the plant safely during normal operation, including operational occurrences.

On the basis of its review, the staff further concludes that the isolation between control and protection systems meets the guidelines of IEEE-603 as endorsed in RG 1.153, and therefore meets the requirements of 10 CFR 50.55a(h) and GDC 24 for assurance of safety functions in the event of control system failures.

8 ELECTRIC POWER SYSTEMS

8.1 Introduction

As the bases for evaluating the adequacy of the design of the Westinghouse AP600 simplified passive advanced light-water reactor (ALWR) electric power systems presented in Chapter 8 of the standard safety analysis report (SSAR), the staff used the acceptance criteria and guidelines for electric power systems contained in Chapter 8 of the NRC Standard Review Plan (SRP) and Regulatory Guides (RGs) 1.153 and 1.155.

It should be noted, however, that the AP600 design as presented does not require Class 1E electrical power, except that provided by the Class 1E dc batteries and their inverters, to accomplish the plant's safety-related functions. Therefore, for a large portion of the electrical design, the staff used the above documents as guidance. In addition, on the basis of the input from the applicant, the staff examined the possibility that additional requirements should be placed on the non-Class 1E portion of the electrical design as part of the resolution of the regulatory treatment of non-safety systems (RTNSS) issue. See Section 8.6.2.4 of this report for resolution of the RTNSS.

8.2 Offsite Electric Power System

Westinghouse shares the AP600 design responsibility for this system with the combined license (COL) applicant referencing the design. The requirements imposed on the COL applicant's design by the AP600 design are specified by interface requirements or COL action items. See Section 8.2.3 of this report for COL action items.

8.2.1 Offsite Circuits Outside the AP600 Scope of Design

The utility company grid system and interconnection to the other grid systems and generating stations are site specific. Section 8.2.3 of this report discusses specific COL action items within the subject areas to which they apply.

8.2.2 Offsite Circuits Within the AP600 Scope of Design

The AP600 electrical system design scope extends into the plant from the high side of the main power transformer, and from the high side of the reserve auxiliary transformer (provided for maintenance).

The power to the main ac power system is normally provided from the main generator. When the main generator is not available, the generator output breaker is opened and the plant auxiliary power comes from the switchyard by backfeeding through the main step-up transformers and the unit auxiliary transformers. In addition, two non-Class 1E onsite standby diesel generators supply power to selected loads in the event of loss of both of these sources. There is also a maintenance source provided through a reserve auxiliary transformer to supply power to a limited number of selected loads during conditions such as refueling. Bus transfer to the maintenance source is manual.

In Revision 3 to the SSAR, Westinghouse clarified that the maintenance source and the associated reserve auxiliary transformer (RAT) primary voltage are to be site specific. The RAT is sized to replace any one of the unit auxiliary transformers (UAT), if needed. Connection of the preferred and maintenance power supplies to the utility grid or other power sources is site specific.

The normal feed to the onsite power system is through the two unit auxiliary transformers, which satisfy the functional requirements of the total plant electrical loads.

8.2.3 Offsite Power System Interfaces

The following design criteria and studies involve site-specific aspects and, therefore, will be addressed in the COL application. Further review of these areas will take place when a COL applicant submits the application.

8.2.3.1 Analysis for AP600 Voltage Operating Range

The operating voltage for the high side of the AP600 transformer and transmission switchyard will be determined by the COL applicant. This was identified as COL Action Item 8.2.3.1-1 and DSER Open Item 8.2.3.1-1.

In response to COL Action Item 8.2.3.1-1 and Open Item 8.2.3.1-1, Westinghouse further clarified in Revision 3 to the SSAR that the COL applicants referencing the AP600 certified design will address the analysis for AP600 voltage operating range. Therefore, DSER Open Item 8.2.3.1-1 is closed.

8.2.3.2 Analysis for AP600 Transient Stability

A site-specific transient stability study simulating peak/valley loading conditions shall be made such that the stability performance of the AP600 is analyzed under all reactive loading or power factor conditions. The unit and transmission grid should remain stable for the following simulated contingencies:

- three-phase fault with breaker failure anywhere in the system
- sudden loss of any large generating plant
- sudden loss of all lines on any common right-of-way
- sudden loss of any large aggregation of load or load center anywhere in the system

This was identified as COL Action Item 8.2.3.2-1 and DSER Open Item 8.2.3.2-1.

In response to COL Action Item 8.2.3.2-1 and Open Item 8.2.3.2-1, Westinghouse further clarified in Revision 3 to the SSAR that the COL applicants referencing the AP600 certified design will address the analysis for AP600 transient stability. Therefore, Open Item 8.2.3.2-1 is closed.

8.2.3.3 Analysis for Frequency Decay Rate

Frequency decay rate shall be determined upon site selection and switchyard voltage selection. This is COL Action Item 8.2.3.3-1 and Open Item 8.2.3.3-1.

In response to COL Action Item 8.2.3.3-1 and Open Item 8.2.3.3-1, Westinghouse further clarified in Revision 3 to the SSAR that the COL applicants referencing the AP600 certified design will address the analysis for the frequency decay rate. Therefore, Open item 8.2.3.3-1 is closed.

8.2.3.4 Testing for the Offsite Power System

General design criterion (GDC) 18 requires, in part, that electric power systems important to safety (which includes the offsite power system) be designed to permit periodic testing. With regard to periodic testing of the systems, equipment, and components, Westinghouse indicated the following:

- The COL applicant will periodically test and inspect the capability of the system, and then conduct a comprehensive testing and surveillance program.
- The switchyard breakers, disconnects, and the transmission line protective relaying will be tested and inspected on a routine basis, without removing the unit from service.

In response to RAI 435.6, Westinghouse stated that the design of the offsite power system is a site-specific issue. The inspection and testing plans will be documented by the COL applicant. The COL applicant's design will be required to meet all applicable regulatory requirements and should meet all associated regulatory and industry guidance as part of the COL application.

Pursuant to GDC 18, the COL applicant will be required to include the above specified periodic tests and inspections in appropriate plant procedures. This was identified as COL Action Item 8.2.3.4-1 and DSER Open Item 8.2.3.4-1.

In Revision 3 to the SSAR, responding to COL Action Item 8.2.3.4-1 and Open Item 8.2.3.4-1, Westinghouse stated that it deleted Section 8.2.2.5 where General Design Criterion (GDC) 18 was referenced and replaced it with a cross reference to SSAR Section 3.1 where GDC compliance is provided. Compliance with GDC 18 for offsite power systems will be addressed by the COL applicant 's test and inspection procedures. The staff finds this acceptable and, therefore, DSER Open Item 8.2.3.4-1 is closed.

8.2.3.5 Capacity and Capability of the Offsite Power System Outside the AP600 Scope of Design

In response to RAIs 435.14 and 435.12, Westinghouse again stated that the design of the offsite power system is a site-specific issue. These requests for additional information (RAIs) addressed the grid stability analysis and the maximum loading of each supply circuit during normal and abnormal operating conditions, including accident conditions and plant shutdown conditions. Westinghouse asserted, however, that the requested information cannot be determined until the physical characteristics of the offsite power system have been designed.

This information will therefore be determined by the COL applicant during site selection and design of the transmission system. This was identified as COL Action Item 8.2.3.5-1 and DSER Open Item 8.2.3.5-1.

In response to COL Action Item 8.2.3.5-1 and Open Item 8.2.3.5-1, Westinghouse stated in Revision 3 to the SSAR that the COL applicants referencing the AP600 certified design will address the design of the ac power transmission system as well as its testing and inspection program. The staff finds this acceptable and, therefore, DSER Open Item 8.2.3.5-1 is closed.

Westinghouse specified certain interface requirements, as follows:

- steady-state load
- inrush kVA for motors
- nominal voltage
- allowable voltage
- regulation
- nominal frequency
- allowable frequency fluctuation
- maximum frequency decay rate
- limiting underfrequency value of the reactor coolant pump (RCP)
- *minimum number of engineered safety feature (ESF) trains to be energized* simultaneously

In response to RAI 435.15, Westinghouse stated that several factors will be used to define the operating limits for the real and reactive power, voltage, frequency, and other characteristics. These include, for example, the equipment rating/limitations defined and designed by the manufacturer, system stability limits, system voltage requirements, and station auxiliary operating limits. As a result, the facility operating limits cannot be fully defined until specific equipment for the electrical system is selected.

The real power and reactive maximum power limits will be selected such that the unit will be turbine-limited, not generator-limited, and can provide the necessary ac voltage support to transmit this power to the utility grid. The capability will be depicted by the generator capability curves, known as Vee curves. The minimum limits are established for real and reactive power on the basis of the specific design parameters of the machine and on the stability limits of the unit with relation to other units connected to the grid.

The minimum and maximum voltage limits of the unit will be within $\pm$ 5 percent of the generator nominal voltage as defined by the relevant ANSI standards. The exact limits within this range will be determined after the offsite power grid has been designed, and the station auxiliary system and main step-up transformer impedance is finalized. These limits will be determined so that the unit will remain stable and not be subjected to under-excitation or over-excitation. The frequency limits will be determined by the turbine manufacturer to limit potential damage of the turbine resulting from excessive vibration of the unit associated with resonance. The plant operating procedures will define these limits once they have been determined. Protective relaying devices will also be installed to maintain unit operation within the allowable limits.

Westinghouse classified all offsite power interfaces as nonnuclear safety (NNS), which implies that none are needed to support plant safety assumptions. However, a number of the interfaces involve power system frequency (e.g., maximum frequency decay rate and limiting underfrequency value of RCP). Because of the importance of RCP coast-down and the possible effects of frequency decay, the staff had some concern regarding the potential safety significance of these interfaces.

In addition, Westinghouse had included an interface designated as the "minimum number of ESF trains to be energized simultaneously." This interface appeared to convey some safety significance; the staff required further clarification of this interface. This was Open Item 8.2.3.5-2.

In their response to Open Item 8.2.3.5-2, Westinghouse has stated that "ESF simultaneously energized" in Table 1.8-1 will be deleted because there is no requirement for the ESF trains to be energized simultaneously on the AP600 design. Westinghouse also explained that RCP underfrequency is not required for plant protection in the AP600 design. The signal will be used for RCP equipment protection only. The AP600 includes an RCP underspeed trip and this trip (not underfrequency) is credited in the safety analysis. Westinghouse has committed to revise Table 1.8-1.

In Revision 15 to the SSAR, Westinghouse deleted the item, "ESF simultaneously energized," from Table 1.8-1. In addition, Offsite ac requirements in Table 1.8-1 were revised to include a maximum frequency decay rate, allowable frequency fluctuations, and nominal frequency. Therefore, Open Item 8.2.3.5-2 is closed.

8.2.3.6 Specific Interface Requirements for Supporting Chapter 15 Analyses

As a result of a Westinghouse design requirement that power should be available to the RCPs following a turbine trip for a minimum of 3 seconds, Westinghouse revised the SSAR and incorporated offsite interface requirements in Revision 13, as discussed below.

There is some possibility, although rare, that the loss of generation to the grid following a turbine trip and subsequent generator trip could cause a system-wide grid instability that could result in a loss of offsite power (LOOP). There is generally a delay, unique to the grid system configuration at the time of the event, before offsite power is lost. In this regard Westinghouse has specified an interface requirement in Item 8.3, Table 1.8-1, of the AP600 SSAR that states:

> "Transient stability must be maintained and the RCP bus voltage must remain above the voltage required to maintain the flow assumed in Chapter 15 analyses for a minimum of three (3) seconds following a turbine trip."

The intent of this analysis is for the COL applicant to demonstrate that for his unique grid system configuration, a grid instability condition would take at least 3 seconds before it resulted in a loss of offsite power to the RCPs.

If the main generator were separated from the grid at the same time as the turbine trip it could result in an immediate and sustained voltage drop in the plant switchyard voltage, resulting In a drop in RCP voltage. If the voltage drop were large enough it could result in insufficient voltage to maintain the required flow of the RCPs. Although the voltage is not normally expected to go that low, Westinghouse stated in Section 8.2.2 of the SSAR that the reverse power relay that separates the generator from the grid following the turbine trip will not open the generator for at least 15 seconds following the turbine trip. This provides additional assurance that the main generator will be available to support grid voltage if needed, for the 3 seconds assumed in the Chapter 15 analysis.

In some plant designs, following a turbine trip, power to the RCPs is automatically transferred from transformers connected to the main generator to transformers connected to the offsite power system. If these transfers initiate at the same time as the turbine trip and fail, it would result in immediate loss of RCP power. The AP600 design has no such automatic transfers of the RCPs.

The AP600 design has a generator circuit breaker on the output of the main generator and utilizes backfeed from the grid to maintain power to the RCPs following a turbine/generator trip. Although the generator output breaker design is believed to be a more reliable scheme than the automatic transfer design described above, it potentially has some vulnerabilities of its own. In the AP600 generator circuit breaker design, tripping of the generator switchyard circuit breakers results in the loss of the offsite power source to the RCPs. Therefore, the signals that cause those breakers to open are more critical. In this regard Westinghouse has specified an interface requirement in Item 8.3, Table 1.8-1, of the AP600 SSAR that states:

> "The protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip."

If a turbine trip occurs when the main generator is not connected to the grid (generator supplying plant house loads only), the offsite source is not available to the RCP buses. In this situation, Westinghouse indicates in Section 8.2.2 of the SSAR that the house loads (including reactor coolant pumps) will be supplied by the rotational inertia of the generator until the generator breaker is tripped on generator undervoltage or exciter overcurrent. The SSAR states that the coastdown will last 3 seconds before the generator breaker trips.

Therefore, the electrical features described in the AP600 SSAR, as well as the COL interface requirements that were specified in the SSAR provide good assurance that the RCPs can receive power from either the main generator or the grid for a minimum of 3 seconds following a turbine trip.

8.2.4 Lack of a Second Offsite Power Supply Circuit

The staff raised a concern regarding the lack of a second offsite power supply circuit. A failure of the normal offsite power circuit in the passive plant designs will result in a LOOP and the operation of the standby power sources, or the operation of the passive safety systems if the standby power sources also fail. The staff recognizes that the loss of offsite power as well as standby sources (i.e., station blackout) may not degrade the passive safety systems, but it does challenge those safety systems. With a second offsite circuit also available, these challenges to the safety systems could be reduced. The lack of a second offsite power source could also impact shutdown risk. If the configuration of the AP600 during shutdown is such that the passive makeup and decay heat removal systems are not available, an increased importance is placed on the ac systems. The staff, therefore, concluded that a second offsite power circuit should be provided during power operation, as well as during maintenance. This is consistent with the principle of defense-in-depth (DID) to which the AP600 has been designed, and the philosophy that the safety systems should not be unnecessarily challenged.

In response to RAI 435.3, Westinghouse stated that the AP600 configuration during shutdown provides availability of reactor coolant system (RCS) makeup and decay heat removal. Operation of either of the two onsite standby ac sources provides power to essential DID equipment and avoids challenges to safety systems. Additional levels of defense have a negligible effect on probabilistic risk assessment (PRA) results and are not warranted. This area was to be resolved as part of the RTNSS process. This was Open Item 8.2.4-1.

The RTNSS identification process did not reveal the need for the availability of a second offsite ac power source in accordance with the requirements of GDC 17. On the basis of this result, no requirements will be placed on electrical design for a second offsite power source. Therefore, Open Item 8.2.4-1 is closed. See Section 8.6.2.4 of this report for additional discussion or the RTNSS resolution of this item.

However, because the AP600 design is not in accordance with GDC 17 offsite power requirements, the staff indicated that there was a need for Westinghouse to request an exemption to GDC 17 regarding the lack of two offsite power sources. The AP600 plant design supports an exemption to the requirement of GDC 17 regarding two physically independent offsite circuits by providing safety-related "passive" systems. These passive safety-related systems only require electric power for valves and the related instrumentation. The onsite Class 1E batteries and associated dc and ac distribution systems can provide the power for these valves and instrumentation. In addition, if no offsite power is available, it is expected that the non-safety-related onsite diesel generators would be available for important plant functions; however, this non-safety- related ac power is not relied on to maintain core cooling or containment integrity.

The staff concludes that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that the regulation need not be applied in this particular circumstance to achieve the underlying purpose of having two offsite power sources because the AP600 design includes an acceptable alternative approach to accomplish safety functions that does not rely on power from the offsite system and therefore accomplishes the intent of the regulation. On this basis, the staff concludes that a partial exemption from the requirements of GDC 17 of Appendix A is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

8.2.5 Grounding and Lightning Protection

In Section 8.2.5 of the SSAR, Westinghouse describes it's design for grounding and lightning protection for the AP600.

8.2.5.1 Grounding Protection

This section addresses the staff's evaluation of the Westinghouse responses to RAIs 435.73 and 435.64 respectively. In these responses, Westinghouse stated that the AP600 grounding system will comply with the guidelines provided in IEEE 665-1987, "Guide for Generating Station Grounding." Specifically, the grounding system consists of the following four subsystems:

- station grounding grid
- system grounding
- equipment grounding
- instrument and computer grounding

The station grounding grid subsystem consists of buried, interconnected bare copper conductors and ground rods forming a plant ground grid matrix. The subsystem will maintain a uniform ground potential and will limit the step-and-touch potentials to safe values under all fault conditions.

The system grounding subsystem will provide grounds of the neutral points of the main generator, main step-up transformers, auxiliary transformers, load center transformers, and onsite standby diesel generators. The main and diesel generator neutrals will be grounded through grounding transformers providing high-impedance grounding. The main step-up and load center transformer neutrals will be grounded solidly. The auxiliary (unit and reserve) transformer secondary winding neutrals will be resistance-grounded.

The equipment grounding subsystem will ground the equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, motor control centers (MCCs), and control cabinets with ground connections to the station ground grid.

The instrument and computer grounding subsystem will ground plant instruments and computers through a separate radial grounding system consisting of isolated instrumentation ground buses and insulated cables. The radial grounding systems will be connected to the station grounding grid at only one point, and will be insulated from all other grounding circuits. The design of the grounding system depends on the soil resistivity. Therefore, the design is site specific and is the responsibility of the COL applicant.

8.2.5.2 Lightning Protection

In accordance with Lightning Protection Code, National Fire Protection Association (NFPA) Standard 780-1992, the lightning protection system, consisting of air terminals and ground conductors, will protect the containment/shield building, cooling towers, switchyard, and other exposed structures and buildings housing safety-related and fire protection equipment. In addition, lightning arresters will be provided in each phase of the transmission lines and at the high-voltage terminals of the outdoor transformers. The isophase bus connecting the main generator, main transformer, and medium voltage switchgear will also be provided with lightning arresters. In addition, a surge suppressor will be provided to protect the plant instrumentation and monitoring system from lightning-induced surges in the signal and power cables connected to a device located outside.

In the response to RAI 435.17, Westinghouse stated that the AP600 design includes two distinct mechanisms to protect the main step-up transformers from lightning coming from two sources. This is because lightning can affect transformers both by a direct strike to the transformer, and by propagating to the transformer over the transmission lines connected to it. In the AP600 design, grounded shield wires are located above the equipment in the transformer area, including the main step-up transformers, to intercept lightning strikes in the area and conduct them to ground. Suitably rated surge arresters are located on the high-voltage side of the main step-up transformers to reduce the magnitude of incoming voltage surges to levels that are well within the insulation withstand capability of the transformer.

Direct strike lightning protection for facilities is accomplished by providing a low-impedance path by which the lightning strike discharge can enter the earth directly. The direct strike lightning protection system (consisting of air terminals, interconnecting cables, connection of the conductors to ground, and other components) will be provided external to the facility in accordance with the guidelines included in NFPA 780. The system will be connected directly to the station ground to facilitate dissipation of the large current of a direct lightning strike. The lightning arresters and the surge suppressor connected directly to the ground provide a low-impedance path to ground for the surges caused or induced by lightning. Thus, damage to facilities and equipment resulting from a lightning strike is avoided.

The final design of direct lightning protection and the associated grounding depends on the lightning activity at the plant site and the soil resistivity of the ground. Consequently, the design is site specific and will be described by the COL applicant. This is COL Action Item 8.2.5.2-1 and Open Item 8.2.5.2-1.

In response to COL Action Item 8.2.5.2-1 and Open Item 8.2.5.2-1, Westinghouse stated in Revision 3 to the SSAR that the COL applicants referencing the AP600 certified design will address the design of the ac power transmission system (which will include lighting protection and grounding), as well as its testing and inspection program. The staff finds this acceptable and, therefore, Open Item 8.2.5.2-1 is closed.

8.2.6 Conclusion

Because the AP600 design does not require ac power for design-basis events, Westinghouse has taken exception to some of the requirements of GDC 17 for the offsite system (e.g., two separate offsite power sources). The plant design meets the intent of GDC 17 by providing safety-related passive systems for core cooling and containment integrity, and multiple non-safety related onsite and offsite electric power sources for other functions.

The staff concluded that the AP600 design approach obviates the need for meeting all of the requirements of GDC 17, by accomplishing the plant safety-related functions using safety systems that do not require electrical power (other than that provided by the Class 1E batteries). However, this area could be impacted by decisions made as part of the RTNSS process and the staff concluded that it should therefore remain open until the final resolution of the RTNSS issue. This was Open Item 8.2.6-1.

In response to Open Item 8.2.6-1, the RTNSS identification process has not revealed any offsite power electrical systems for inclusion (see Section 8.6.2.4 of this report). On the basis of this result, Open Item 8.2.6-1 is closed.

In GDC 18, the NRC requires that the offsite electric power systems be designed with the following capabilities:

- permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) to assess the continuity of the systems and the condition of their components
- test the operability and functional performance of the components of the systems
- periodically test the operability of the systems as a whole and (under conditions as close to design as practical) the full operation sequence that brings the systems into operation

In GDC 18, the NRC requires that the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system should meet the capability to be tested. The COL applicant is responsible for the overall testability of the offsite power system design as outlined in COL action items and interface requirements in Table 1.8-1 of the SSAR. On the basis of the above, the staff concludes that the design of the offsite power system for the AP600 is acceptable.

8.3 Onsite Power Systems

In Section 8.3 of the SSAR, Westinghouse describes it's design for the AP600 onsite power system.

8.3.1 Onsite ac Power System

The main onsite ac power system is a non-Class 1E system and does not perform any safety-related function. However, there was some possibility that additional requirements would be placed on the non-Class 1E portion of the electrical system as part of the resolution of the RTNSS issue. This was identified as DSER Open Item 8.3.1-1. On the basis of the results identified in Section 8.6.2.4 in this report regarding resolution of RTNSS, Open Item 8.3.1-1 is closed.

8.3.1.1 Electrical Distribution System

During power generation mode, the turbine generator normally supplies electric power to the plant auxiliary loads through the unit auxiliary transformers (UATs). The plant is designed to sustain a load rejection from 100 percent power, with the turbine generator supplying the plant house loads.

During plant startup, shutdown, and maintenance, the generator breaker is opened. Under this condition, the main ac power is provided by the preferred power supply system from the high-voltage switchyard (switchyard voltage is site-specific) through the main step-up transformers and two UATs. Each UAT supplies power to about 50 percent of the plant loads.

In Revision 3 to the SSAR, Westinghouse stated that the maintenance source and the associated RAT primary voltage are site specific. The RAT is sized to replace any one of the UATs, if needed. The RAT is powered from the switchyard and is available if any of the UATs is out of service to provide operational flexibility. The staff finds this to be acceptable.

In response to RAI 435.16, Westinghouse stated that the start time for the diesel generators is 20 seconds, and actuation times for the non-safety-related DID systems are fast enough to prevent actuation of the safety-related passive systems following transients or RCS leaks. However, Amendment 1 to Section 8.3.1.1.2.3 of the SSAR stated that the diesel generators are available to accept loads in 120 seconds. These statements appeared to conflict. This was Open Item 8.3.1.1-1.

In response to Open Item 8.3.3.1-1, Westinghouse stated that their response to RAI 435.16 regarding the 20 second start time was incorrect. As the final SSAR supersedes the RAIs, Westinghouse stated that there is no need to change the response to RAI 435.16. The staff agrees that the 120 second start time is acceptable for this DID system and therefore, Open Item 8.3.3.1-1 is closed.

The buses tagged with odd numbers (ES1, ES3, etc.) are connected to one UAT, while the buses tagged with even numbers (ES2, ES4, etc.) are connected to the other UAT. These 4.16 kV buses are provided with access to the maintenance source through normally open circuit breakers. The arrangement of the 4.16 kV buses permits feeding functionally redundant pumps or groups of loads from separate buses and enhances the plant's operational flexibility.

A maintenance source is provided to supply power through a reserve auxiliary transformer. Bus transfer to the maintenance source is manual.

8.3.1.2 Standby Diesel Generators

The onsite standby power system, powered by the two onsite standby diesel generators, supplies power to selected loads in the event of a loss of normal and preferred ac power supplies. Those loads that are priority loads for DID functions based on their specific functions (permanent non-safety loads), are assigned to annex building buses ES1 and ES2. These permanent non-safety loads are divided into two functionally redundant load groups. Each bus is backed by a non-Class 1E onsite standby diesel generator. In the event of a loss of voltage on these buses as a result of a turbine generator trip concurrent with a loss of preferred power source, the diesel generators are automatically started and connected to the respective buses. The source incoming breakers on switchgear ES1 and ES2 are interlocked to prevent inadvertent connection of the onsite standby diesel generator and preferred/maintenance ac power sources to the 4.16 kV buses at the same time. The diesel generator, however, is capable of being manually paralleled with the preferred power supply for periodic testing. Design provisions protect the diesel generators from excessive loading beyond the design maximum rating should the preferred power be lost during periodic testing. The control scheme, while protecting the diesel generators from excessive loading, does not compromise the onsite power supply capabilities to support the DID loads.

Generally, periodic diesel generator testing is performed, one diesel generator at a time, during normal plant operation. During this testing, the diesel generator is synchronized with the offsite power source and supplies the plant auxiliaries via the UATs. Should offsite power be lost during this operational mode, the main turbine generator has the capability to withstand 100-percent load rejection and continues to supply the plant auxiliaries. If the onsite standby diesel generator periodic testing is in progress during a LOOP under the stated conditions, additional design controls are not required. The periodic testing may be terminated using the standard test procedure without causing any adverse effects on the onsite standby power source.

For a postulated LOOP concurrent with the main turbine generator trip while a diesel generator is undergoing periodic testing, the system loads required for normal plant operation (which are fed from the permanent non-safety-related bus) would immediately overload the associated diesel generator. This excessive loading would cause a drop in generator frequency. The following design features protect the diesel generator from adverse effects.

Should the diesel generator frequency drop below the set frequency value, a signal would be generated to trip the diesel generator breaker. The under-frequency tripping would take place only when the following additional permissives are satisfied simultaneously:

- diesel generator is in a "test" mode
- diesel generator is operating and is connected to the associated permanent non-safety-related bus as signified by "diesel generator breaker closed position"

When the diesel generator breaker trips, the generator is isolated from the bus load, but is still maintained in a "no load running" condition via the operation of the governor controls. With no power source at the permanent non-safety-related bus, all designated load breakers and the normal power source incoming breaker would be tripped via the bus undervoltage relay operation. The diesel generator test mode would be reset. Because the generator would still produce power at the rated voltage and frequency, the diesel generator breaker would be reconnected to the associated dead bus, and operation of the automatic load sequencer would be initiated to supply the defense-in-depth system loads.

In response to RAI 435.9, Westinghouse provided the proposed maintenance and testing program for the diesel generators, as described in the following paragraphs.

<u>Maintenance</u>

The diesel generators are not safety-related and will be maintained in accordance with the requirements of the overall plant maintenance program. This program will cover the preventive, corrective, and predictive maintenance activities of the plant systems and equipment and will be presented in the COL application. This is COL Action Item 8.3.1.2-1.

<u>Periodic Testing</u>

The periodic testing program will be performed in accordance with the recommendations of the engine manufacturers.

The specific engine loading level will be determined by the engine manufacturer's recommendations, such that the load operation will not cause incomplete fuel combustion that may result in gum and varnish deposits in the engine exhaust system.

Support Systems Operation

The diesel generators are automatically started and connected to the associated medium-voltage buses in the event of a loss of voltage on these buses as a result of a loss of preferred power source concurrent with the turbine generator trip. The following conditions are prerequisites for the diesel generator automatic start:

- starting air pressure within acceptable limits
- dc control power availability for fuel oil valve solenoid operation and the starting air motor solenoid
- fuel supply availability
- diesel generator controls in the automatic mode
- diesel generator breaker lockout trip permissive not activated by any of the trouble conditions
- engine prelubrication provided

Satisfactory status of these "prestart" conditions is continuously monitored, and any failure is annunciated in the main control room (MCR).

The starting air system has an accumulator that stores the pressurized air required for the diesel engine cranking power. The starting air compressor, one per diesel generator, keeps the accumulators pressurized. An alarm in the MCR detects the low starting air pressure. For operation of the starting air valve and fuel oil valve solenoid, dc control power is provided from the non-Class 1E dc buses that are fed from the battery chargers and backed by the non-Class 1E battery bank. Availability of dc control power at the diesel generator control panel is continuously monitored, and loss of control power is annunciated in the MCR. The standby generators have permanent magnet excitation, thus eliminating dc power requirements for the field flashing circuit.

Each diesel generator has a fuel day tank located to provide the necessary suction head for the engine-driven fuel pump. The day tank fuel level is continuously monitored and annunciated for low level indication in the MCR.

The diesel generators are normally maintained in the standby mode ready to accept the signal for auto start. The standby mode status is provided in the MCR, which monitors the diesel generator control switch "auto" position and the MCR/local transfer switch MCR position. If the diesel generator control switch (located in the MCR) is not in the "auto" position, annunciation is provided to the control room operator. The MCR transfer switch is placed in the "local" position to allow local start for testing purposes. After test completion, if the MCR transfer switch is not repositioned to the MCR position, annunciation is provided in the control room. The diesel generator breaker controls are provided with a lockout relay that would trip and lock out the breaker for selected electrical faults and engine trouble conditions that may be detrimental to unit operation. The lockout relay operation is annunciated in the MCR.

Before the automatic start, the engine must be prelubricated. Diesel engine prelubrication is continuously provided while in standby mode by a normal ac motor-driven lubrication pump backed by a dc motor. Should the ac motor pump fail, transfer to the dc-driven motor is automatic. A minimum level of oil in the cylinder block is required to fulfill the starting system interlock. Low oil level is annunciated in the MCR.

The conditions that may render the diesel generator incapable of automatic start are monitored or alarmed in the MCR.

Based on the above, the staff concludes that the standby diesel generators are acceptable for their DID function.

8.3.1.3 Ancillary ac Diesel Generators

Westinghouse included two ancillary diesel generators located in the annex building to provide power to meet the post-72-hour power requirements following an extended LOOP. Each ancillary diesel generator output is connected to a distribution panel. The outgoing feeder circuits from the distribution panel are connected to cables that are routed to the Divisions B and C voltage regulating transformers and to the passive containment cooling system (PCS) pumps. A Class 1E voltage regulating transformer powers the postaccident monitoring loads, the lighting in the MCR, and ventilation in the MCR and Divisions B and C instrumentation and control (I&C) rooms. It also provides power to support operation of the ancillary generator's lighting and fuel tank heating equipment. The ancillary diesel generators are not needed for refilling the PCS water storage tank, post accident monitoring, or lighting for the first 72 hours following a loss of all other ac sources and are not needed for spent fuel makeup for the first seven days following the loss of all other ac sources. The generators are commercial-grade, skid-mounted, packaged units. These generators are located in the portion of the Annex Building that is classified as seismic Category II structure and are, therefore, to some extent seismically protected. Generator control is manual from a control panel mounted with the diesel skid package. The fuel for ancillary generators is stored in a tank located in the same room as the generators. The tank is seismic Category II and holds sufficient fuel for four days of operation.

Each diesel generator will be started and operated for at least one hour, connected to a test load that is representative of its total postaccident load. The test frequency will be at least once every quarter. Each diesel generator will be started and operated for four hours while providing power to the regulating transformer and the PCS water storage tank makeup pump and ancillary fans. The test frequency will be at least once every ten years.

Based on the above, the staff concludes that the Ancillary ac Diesel Generators are acceptable as backup power sources for the longer term (post-72-hour) following a loss of all other ac power sources.

8.3.1.4 Segregation of Non-Class 1E Electrical Equipment

In response to RAI 435.7, Westinghouse provided information regarding the physical and electrical independence requirements. Specifically, Westinghouse stated that the non-Class 1E ac power is for investment protection only and need not meet the electrical independence guidance of RG 1.75. The following statements describe the electrical separation aspects of the non-safety-related ac power supply system design:

- Non-Class 1E circuits are electrically isolated from the Class 1E circuits in compliance with the RG 1.75 stipulations and IEEE 384-1974 standard.
- Non-Class 1E ac power system design includes two divisions of permanent non-Class 1E loads, each supported by its own onsite diesel generator unit.
- Each onsite diesel generator unit is located in a separate enclosure.
- The ac switchgear units pertaining to each of the non-Class 1E ac divisions are located in separate rooms in the annex building.
- The control power for the control of the non-Class 1E ac switchgear breakers is provided from separate non-Class 1E related dc power sources.
- RG 1.75 separation criteria are not going to be implemented for the non-Class 1E related ac raceway design.

RG 1.75 is not applicable to separation among redundant non-Class 1E ac power supply systems; therefore, the non-Class 1E circuits are not required to meet the electrical independence guidance of RG 1.75. They are, however, electrically isolated to the maximum extent practicable.

8.3.1.5 Electric Circuit Protection

Major types of protection systems employed for the AP600 include the following:

<u>Medium-Voltage Switchgear</u>

Each medium-voltage switchgear bus is provided with a bus differential relay to protect against a bus fault. The actuation of this relay initiates tripping of the source incoming circuit breaker and all branch circuit load breakers. The differential protection scheme employs high-speed relays. Motors rated 1500 hp and above are generally provided with differential protection (Device 87M).

Subsequently, Westinghouse replaced the differential relays (87M) with high dropout overcurrent relay (device 50D) for differential protection. The staff finds that Device 50D will provide as good a protection as provided by the Device 87M and the change is, therefore, acceptable.

The source incoming circuit breakers are equipped with an inverse time overcurrent relay on each phase and an inverse time ground fault relay for bus protection.

In Revision 3 of the SSAR, Westinghouse replaced the above relays on source incoming breakers with a microprocessor-based multifunction relay to provide backup protection for the buses. For each medium-voltage motor feeder breaker, the electro-mechanical relays are being replaced with a microprocessor-based motor protection relay which provides protection against various types of faults (phase and ground) and abnormal conditions such as locked rotor and phase unbalance. For each medium-voltage power feeder to a 480 V load center, the electro-mechanical relays providing short circuit and overload protection are being replaced with a microprocessor-based multifunction protection. The staff finds these changes acceptable because using the state-of-the-art microprocessor-based protection provides at least equivalent protection as provided by the electro-mechanical relays.

Medium-voltage buses are provided with a set of three undervoltage relays (27B) that trip feeder circuit breakers connected to the bus upon a complete loss of ac power, using two-out-of-three logic to prevent spurious actuation. In addition, another set of undervoltage relays (27N) is provided on the line side of incoming supply breakers of buses ES1 and ES2. These relays initiate an alarm in the MCR if a sustained low- or high-voltage condition occurs.

After the original SSAR was submitted, Westinghouse in Revision 3 replaced the solid state relays (27B & 27N) with microprocessor-based multifunction protection. The staff finds that this change using the state of the art microprocessor-based protective device provides at least the equivalent protection as provided by the solid state relays and the change is, therefore, acceptable.

480-V Load Centers

Each motor feeder breaker in load centers is equipped with a solid-state, adjustable trip device, which has long-time, short-time, instantaneous, and ground fault tripping features. Each load center bus has an undervoltage relay that initiates an alarm in the MCR upon loss of bus voltage.

In Revision 3 of the SSAR, Westinghouse replaced the solid-state relays with microprocessor-based multifunction protection. The staff finds this change acceptable because using the state-of-the-art microprocessor-based protective device provides at least the equivalent protection as provided by the solid-state relays.

480-V Motor Control Center

MCC feeders for low voltage (460 V) motors have molded-case circuit breakers (magnetic or motor circuit protectors) and motor starters. These motor starters are provided with thermal overload protection.

8.3.1.6 Electrical Equipment Layout

Medium-voltage switchgears ES1, ES2, ES5, and ES6 are located in the electrical switchgear rooms 1 and 2 of the annex building. Switchgears ES3, ES4, ES7, and ES8 are located in the turbine building electrical room. The Class 1E medium-voltage switchgear for four RCPs is located in the auxiliary building. The 480 V load centers are located in the turbine building electrical room and in annex building electrical switchgear rooms 1 and 2, on the basis of the proximity of loads and the associated 4.16 kV switchgear.

8.3.1.7 Heat Tracing System

The non-Class 1E electric heat tracing system provides electrical heating where temperatures above ambient are required for system operation and freeze protection. This system is part of the AP600 permanent non-Class 1E loads and is powered from the diesel-backed 480 Vac MCCs.

8.3.1.8 Raceway and Cable Installation

There are two non-safety-related load groups associated with different transformers, buses, and onsite standby diesel generators. Westinghouse has not provided any physical separation between the two ac load groups because they are non-Class 1E. The power cable ampacities are in accordance with the Insulated Cable Engineers Association publications and the National Electric code. The derating is founded on the type of installation, the conductor and ambient temperature, the number of cables in a raceway, and the groupings of the raceways. A further derating of the cables is applied for cables that pass through a fire barrier. Westinghouse has determined the derating factors from the Insulated Cable Engineers Association publications and other applicable standards.

For circuits that are routed through conduit and partly through trays or underground ducts, the cable size is determined by the ampacity in that portion of the circuit with the lowest indicated current carrying capacity.

Cable tray design is determined by a random cable fill of 30 percent of usable tray depth. Westinghouse has committed to analyze the tray fill if it exceeds the above stated maximum fill. Separate raceways are provided for medium-voltage power, low-voltage power, and control, as well as instrumentation cables. Non-Class 1E raceways and supports installed in seismic Category I structures are designed and/or physically arranged so that an SSE could not cause unacceptable structure interaction or failure of seismic Category I components.

The raceway system for non-Class 1E ac circuits complies with IEEE 422 with respect to installation and support of cable runs between electrical equipment, including physical protection.

8.3.1.9 Conclusion

The staff concludes that the AP600 design approach obviates the need for meeting all the Class 1E requirements for onsite ac power by accomplishing the plant safety-related functions using safety systems that do not require electrical power (other than that provided by the Class 1E batteries See Section 8.3.2). In addition, the staff concludes that Westinghouse has specified appropriate design criteria for the non-Class 1E onsite ac power system and its DID function, and is, therefore, acceptable. However, the final resolution of the RTNSS process resulted in the determination that administrative controls on the ac power from the diesel generators is appropriate for all modes of operation and is further discussed in Section 8.6.2.4 of this report.

8.3.2 Direct Current (dc) Power and Uninterruptible Power Systems

The dc power system consists of Class 1E and non-Class 1E dc power systems. Each system consists of ungrounded batteries, dc distribution equipment, and uninterruptible power supplies (UPSs).

The Class 1E dc and UPS system supplies power for Class 1E equipment required for the plant instrumentation, control, monitoring, and other vital functions needed for plant safety. In addition, the Class 1E dc and UPS system provides power to lighting in the MCR and in the remote shutdown area.

The Class 1E dc and UPS system supplies power for the safe shutdown of the plant without the support of battery chargers during a loss of all ac power sources coincident with a design-basis accident (DBA). The system is designed so that no single failure will result in a condition that will prevent the safe shutdown of the plant.

The non-Class 1E dc and UPS system provides power to the plant's non-Class 1E control and instrumentation equipment and loads that are required for plant operation and investment protection, and to the hydrogen igniters located inside containment. Operation of the non-Class 1E dc and UPS systems is not required for plant safety.

The batteries of the Class 1E and non-Class 1E dc and UPS systems are sized in accordance with IEEE 485. The operating voltage range of dc loads is 105 to 140 V dc. The maximum equalizing charge for the batteries is 140 V dc.

8.3.2.1 Class 1E dc and UPS System

In Section 8.3.2.1 of the SSAR, Westinghouse describes the Class 1E dc and UPS system design for the AP600.

<u>Class 1E DC Distribution</u>

The Class 1E dc power system consists of four independent 125-V Class 1E dc safety system divisions (Divisions A, B, C, and D). Divisions B and C are each comprised of two battery banks, two switchboards, and two battery chargers. A battery bank in each of the four divisions, designated as a 24-hour battery bank, is used to provide power to the loads required for the first 24 hours following a loss of all ac power sources concurrent with a DBA. The second battery bank in Divisions B and C, designated as a 72-hour battery bank, is used for loads requiring power for 72 hours following the same event. Each switchboard connected with a 72-hour battery bank supplies power to an inverter. Because of the size of the Class 1E battery banks, there is no need for a load shedding or load management program to maintain power during the accomplishment of the required safety-related functions.

The "Safety Design Bases" for the Class 1E dc and UPS system for the AP600, includes the following:

> "The Class 1E dc and UPS system is divided into four divisions. Any three out of four divisions can shut down the plant safely and maintain it in a safe-shutdown condition."

Although such a design meets the single-failure criteria, the staff was concerned that the provision for four independent divisions might be construed to imply additional failure tolerance. This can impact such things as establishing allowed outage times for divisional equipment in the technical specifications (TS). Therefore, to clarify this aspect, the staff required that Westinghouse provide a clear indication of the divisions of the Class 1E dc and UPS system which make up a "safety group" (as defined in IEEE 308-1980). This was DSER Open Item 8.3.2.1-1.

In response to Open Item 8.3.2.1-1, Westinghouse stated that "Safety Group" as defined in IEEE 308-1980 may include one or more minimal sets of interconnected components, modules, and equipment that can accomplish a safety function. The staff concluded that this is acceptable and the review of the TS was conducted on this basis. Therefore, Open Item 8.3.2.1-1 is closed.

The operating voltage range for the equipment and the associated loads will be specified in accordance with IEEE 946-1985 to ensure reliable operation of the dc power system for the full range of operating voltages, including charging, equalizing, and end-of-discharge. The batteries have been sized to provide steady-state and switching transient power within the required voltage range of 105 to 140 V dc.

The staff was concerned whether the dc power supply system should be designed with sufficient redundancy to ensure that, in the case of a LOOP concurrent with a turbine trip, the loss of any plant battery or dc bus concurrent with a single independent failure in any other system required for shutdown cooling will not result in a total loss of reactor cooling capability. The staff asked Westinghouse whether the AP600 design conforms to this guidance and, if not, to justify the deviation.

In response to RAI 435.30, Westinghouse referred to work on the EPRI Utility Requirements Document (URD). To address the NRC concern about the loss of a Class 1E dc or vital ac bus, and industry concerns about plant availability, Revision 4 of the URD stated the following:

> "The Class 1E dc and vital ac power supply system shall be designed with sufficient redundancy to ensure that the loss of either a dc or a vital ac bus will not result in a plant transient and simultaneous loss of single-failure protection in the system needed to respond to the event.
>
> <u>Rationale:</u>
>
> This requirement is a particular case of the generic statement of the single failure criterion, endorsed by the URD for the passive plants."

The Class 1E dc and UPS system is designed to accommodate component failures such as the loss of a battery charger, a battery, or an inverter, without the loss of power to either the dc bus or the ac instrumentation and control power bus. In response to RAI 435.31, Westinghouse stated that a single failure of a battery, a battery charger, or an inverter will not deenergize the associated dc or ac buses; therefore, plant operation will be unaffected.

The AP600 design satisfies the requirements of the proposed revision to Section 7.2.1 of the EPRI ALWR URD because the loss of a Class 1E dc bus or ac I&C power bus will not result in a plant transient and the simultaneous loss of single failure protection for safety-related systems needed to respond to the event. In response to RAI 435.22, Westinghouse stated that the required postaccident monitoring loads are included in the 72-hour battery load profiles. The plant control loads required for licensing design-basis events are included in the 24-hour battery load profiles.

In response to RAI 435.21, Westinghouse stated that the AP600 passive systems are actuated with power from the Class 1E dc and UPS system. Once the passive safety system actuation is completed, there is no further requirement from the Class 1E dc and UPS system; the systems will continue to perform their functions. The passive safety systems will be automatically actuated 24 hours after loss of battery charging capability (i.e., loss of ac power) to the Class 1E dc and UPS system. This ensures that the passive safety systems are actuated regardless of which licensing design-basis event is postulated to occur. The time of 24 hours after loss of all ac power sources for automatic passive system actuation was selected on the basis of the following considerations:

- a time long enough to provide a high probability of recovering ac power and avoiding automatic depressurization system actuation
- a time short enough not to unduly increase battery capacity requirements.

The Class 1E dc system is ungrounded; thus, a single ground fault does not cause immediate loss of the faulted system. However, a ground fault followed by a second ground can produce ground currents of sufficient magnitude to initiate operation of deenergized dc loads or inhibit dropout of energized dc loads. Detection with alarms is provided for each power division so that ground faults can be located and removed before a second fault could disable the affected circuit. This was the subject of Information Notice (IN) 88-86 and its Supplement 1. The staff asked Westinghouse to describe the ground detection system for the ungrounded dc auxiliary system. In response to RAI 435.25, Westinghouse committed that during the detail design phase, the concerns expressed in IN 88-86 and its Supplement 1 will be addressed for the Class 1E dc ground detection system. Also, COL plant procedures will be established so that prompt action is taken to clear any ground fault on the Class 1E dc system. This was identified as COL Action Item 8.3.2.1-1 and DSER Open Item 8.3.2.1-2.

In response to COL Action Item 8.3.2.1-1 and Open Item 8.3.2.1-2, in Revision 3 to the SSAR, Westinghouse stated that the COL applicants referencing the AP600 certified design will establish plant procedures for clearing ground faults on the Class 1E dc system. The staff finds this is acceptable and, therefore, Open Item 8.3.2.1-2 is closed

Westinghouse has provided a single spare battery bank with a spare battery charger for the Class 1E dc and UPS system. In the case of a failure or unavailability of any normal battery bank and the battery charger, permanently installed cable connections allow the spare to be connected to the affected bus by a plug-in type twist-lock disconnect. The twist-lock disconnect permits connection of only one battery bank and battery charger at a time so that the independence of each battery division is preserved.

The staff was concerned with how the AP600 design will maintain the electrical and physical separation between the redundant safety systems when a system is powered from the backup dc source.

In response to RAI 435.51, Westinghouse stated that separate cables are permanently installed between each of the four divisions and the spare battery bank with twist-lock, plug-in connectors (male) at each end. Each division and the spare battery include a fused transfer switch between the battery and the switchboard. These transfer switches include a single twist-lock, plug-in connector (female). To connect one division to the spare battery, the permanently installed cable for that division is plugged into its associated connector at one end. The other end is plugged into the connector for the spare battery. Because the spare battery transfer switch has only one plug, it is possible to connect only one division's cable at a time. When the spare battery is connected to a given division, the spare battery bank becomes that division, and only the cable of that division is energized. The cables associated with the other divisions remain disconnected at both ends. Thus, electrical and physical separation between the redundant safety systems is maintained when a system is powered from the spare battery.

The Class 1E dc switchboards employ fusible disconnect switches and have adequate short circuit and continuous current ratings. Fused transfer switch boxes, equipped with double-pole, double-throw transfer switches, are provided to facilitate battery testing and maintenance. The fuses are housed in the fused transfer switch boxes. To provide maximum protection coverage from short circuit, each fused transfer switch box is located as close to the battery terminals as possible. The fuses are sized in accordance with the criteria stated in Section 7.1 of IEEE 946-1985. The continuous current rating of the fuses is sufficiently high to prevent damage to the fuse element at the one-minute current rating of the battery, and sufficiently low to ensure interruption of the short-circuit current available from the battery at end-of-discharge voltage, which is 1.75 V/Cell.

The input ac power for the Class 1E dc battery chargers is supplied from non-Class 1E 480 Vac diesel generator-backed MCCs.

The staff was concerned with whether the standard molded-case ac breakers will be used in dc circuits, because the dc interrupting rating will generally be one-half to one-third of the ac value. Many manufacturers publish no dc application data for these breakers. In response to RAI 435.27, Westinghouse stated that the AP600 design generally uses fusible disconnect switches in the Class 1E dc system. If a molded-case circuit breaker is used in a particular circuit, it will be sized to meet the dc interrupting rating requirement. Proper documentation will be obtained by the COL applicant to ensure that the molded-case breakers have an adequate dc interrupting rating. This was identified as COL Action Item 8.3.2.1-2 and DSER Open Item 8.3.2.1-3.

In response to COL Action Item 8.3.2.1-2 and Open Item 8.3.2.1-3, in Revision 3 to the SSAR Westinghouse stated that the AP600 generally uses fusible disconnect switches in the Class 1E dc system. If molded-case breakers are used for dc applications, they will be sized to meet the dc interrupting rating requirements. The staff find this to be in accordance with good engineering practices prevalent in the industry and is acceptable and, therefore, Open Item 8.3.2.1-3 is closed and COL Action Item 8.3.2.1-2 is not required.

The staff was concerned that, when deenergized, highly inductive loads may generate surges that, if not suppressed, may impress voltage spikes on the dc system, because the battery charger is powered from the ac system. In response to RAI 435.53, Westinghouse stated that the dc system will be protected from surges generated on the ac system by the isolation provided by the Class 1E battery chargers and regulating transformers, which are power-regulating devices. To further ensure protection, this equipment will employ metal-oxide varister surge suppressors at the input terminals to all battery chargers and inverters to minimize the potential for component damage resulting from voltage surges due to electrical transients. This surge suppressor does not emit toxic fumes upon failure and arcing or burning. The surge protection provided in the design will slope off the steep wave front associated with switching surges and the design is therefore, acceptable.

The AP600 design will consider the effective transient armature resistance data (rd) obtained from the motor manufacturer. In cases for which the specific rdis not available, 10 times the motor full-load current will be considered for the motor short-circuit current contribution.

Because of the large capacity of the AP600 Class 1E battery (4800 ampere-hour), the short-circuit current contribution to a fault from the battery is predominant over the contribution from the dc motors. There are only a few small motor-operated valves (MOVs) connected to the Class 1E dc buses. As such, the short-circuit duty of the Class 1E dc distribution system is primarily dictated by the contribution from the battery. The AP600 Class 1E dc system utilizes fusible disconnect switches rated for 100,000A interrupting, and it will envelope the maximum current that a motor will contribute to a fault which is 10 times the motor full load current.

Battery Charger

The AP600 safety-grade battery chargers are sized to meet the largest combined demands of the various steady-state loads plus the charging capacity to restore the battery from the design minimum charged state to the fully charged state in 24 hours (not 12 hours as recommended by IEEE 946). The staff concludes that a longer charging time of up to 24 hours is acceptable, given the large overall battery capacity.

Each battery charger has an input ac and output dc circuit breaker for the purpose of power source isolation. Each battery charger prevents the ac supply from becoming a load on the battery as a result of power feedback result from the loss of ac power to the chargers. Each battery charger has a built-in current limiting circuit, adjustable between 110 to 125 percent of its rating to hold down the output current in the event of a short circuit or overload on the dc side. The output of the charger is ungrounded and filtered. The output float and equalizing voltages are adjustable. The battery chargers have an equalizing timer and a manual bypass switch to permit periodic equalizing charges. Each charger is capable of providing continuous Class 1E loads while providing sufficient power to charge a fully discharged battery within a 24- hour period.

In the response to RAI 435.23, Westinghouse stated that the battery chargers are equipped with a special circuit so that, in the event of an unscheduled disconnection of a battery resulting from a fuse failure or otherwise, the battery chargers will continue to operate in a stable manner. Except during the transition time from normal to spare battery, the AP600 design does not require disconnection of the batteries for any mode of operation including battery equalizing mode. Scheduled maintenance and testing of the batteries will be conducted after the spare battery is connected. The downstream loads will be specified to continuously withstand the maximum equalizing voltage of 140 Vdc in accordance with the recommendation of IEEE 946-1985.

In the response to RAI 435.24, Westinghouse stated that the battery chargers are provided with a high dc voltage charger shutdown feature, which includes an alarm relay and indicating light. This functionally disables and locks out the battery charger whenever the output dc voltage exceeds the preset upper limit of charging voltage. Thus, the dc system equipment is protected from overvoltage damage, because the dc equipment and components are rated for maximum equalization voltage of 140 V.

Sizing Class 1E Batteries

The AP600 Class 1E-125 Vdc batteries are sized to meet the design requirements of their connected load without the charger support for the corresponding times of 24 and 72 hours. The batteries have been sized in accordance with IEEE 485-1983, "Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations." The momentary loads existing for a fraction of a second have been considered to last for a full minute. Also, all momentary loads occurring within the same one-minute period have been added to get the total load for that minute. An aging factor of 25 percent has been provided over and above the minimum operating temperature factor. Therefore, the rating of the Class 1E batteries will be 25 percent greater than that required to supply the load requirements at the minimum expected operating temperature.

The governing factor for the AP600 Class 1E battery size is the steady-state loading condition. The steady-state loads are required to operate for a long period of time (0 to 24 hours and 0 to 72 hours, compared to 0 to 2 hours normally considered for operating plants). Maximum inrush current has been considered for the motor-operated valves and solenoid-operated valves that are required to operate in the first minute. The duration for the inrush current is conservatively assumed as one minute. The rest of the loads are steady-state loads.

There are 14 MOVs with a total locked rotor current of 1864 amps. The AP600 design uses two parallel strings of 2400 amp-hour cells (Type C&D LCY-39) for each battery bank. Each LCY-39 has 19 positive plates for a total of 38 positive plates per battery bank. Each LCY-39 can produce 3040 amps for 1 minute. Battery sizing calculation assumed the worst case for the MOVs and solenoid valves on battery A in accordance with IEEE 485. The minimum battery size to meet all margins, including the recommended 10 percent design margin, is 3252 amp-hours. This leaves a significant additional margin to the 4800 amp-hour size.

Monitoring and Alarms

Each battery bank, including the spare, has a battery monitor system that detects battery open circuit conditions and monitors battery voltage. The battery monitor provides a trouble alarm locally and in the MCR. The battery monitors are not required to support any function.

As indicated in RAI 435.26, the specific acceptance criteria for monitoring the dc power systems are derived from the generic standards in Section 7.4 of IEEE 946-1985. Although IEEE 946-1985 is not endorsed by a regulatory guide, the staff used this standard in the evaluation of the AP600 dc auxiliary system. In summary, these general standards state that the dc systems (batteries, distribution system, and chargers) shall be monitored to the extent that they can be shown to be ready to perform their intended functions. On November 14, 1993, Westinghouse responded that the AP600 Class 1E dc power system conforms to the indication and alarm requirements recommended in Section 7.4 of IEEE 946-1985, as follows:

Instrument/Alarm/Control	Main Control Room	Local
battery current (ammeter charge/discharge)		X
battery charger output current (ammeter)		X
dc bus voltage (voltmeter)	X	X
battery charger output voltage (voltage)	X	
ground-detector (voltmeter)	X	
dc bus undervoltage alarm	X	
dc system ground alarm	X	
battery breaker/switch open alarm	X	
battery charger output breaker open alarm	X	
battery charger dc output failure alarm	X	
battery charger ac power failure alarm	X	
charger low dc voltage alarm	X	
charger high dc voltage shutdown relay (opens main ac supply breaker to the charger)	X	
battery test breaker closed alarm	X	X

Westinghouse has provided all of the alarms and indications specified in response to RAI 435.26 except the battery high discharge rate alarm. This alarm is needed in case the battery is short circuited. This was identified as DSER Open Item 8.3.2.1-4.

In response to Open Item 8.3.2.1-4, in Revision 3 to the SSAR Westinghouse stated that the monitoring and alarm of dc current and voltages will be through the plant control system, which will include a battery discharge rate alarm. The staff finds this is acceptable because the battery high discharge rate alarm is already included in the plant control system and, therefore, Open Item 8.3.2.1-4 is closed.

In the response to RAI 435.28, Westinghouse stated the following:

- The AP600 design conforms to the recommendation of RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." The position of the fused disconnect switches associated with the battery charger, battery, and dc bus supply will be monitored to provide information to the alarm system.

- The operating range for the safety-related dc power system is 105 to 140 Vdc. This voltage range envelopes the design-basis accident conditions; the batteries have been sized to provide adequate voltage at the end of the battery duty cycle.

In response to RAI 435.38, Westinghouse stated that the AP600 battery monitoring system detects the battery open-circuit conditions including the open-circuited intercell connectors. It also monitors the battery high- and low-terminal voltage. Continuous monitoring of the battery parameters (such as voltage and current) will provide the pre-programmed alarm for high terminal resistance. Checking sulfated plates or other anomalous conditions requires periodic inspections, which will be covered in the plant procedures to be prepared by the COL applicant. This was identified as COL Action Item 8.3.2.1-3 and DSER Open Item 8.3.2.1-5.

In response to COL Action Item 8.3.2.1-3 and Open Item 8.3.2.1-5, in Revision 3 to the SSAR Westinghouse stated that the COL applicants referencing the AP600 certified design will establish plant procedures for checking sulfated battery plates or other anomalous conditions through periodic inspections. The staff finds this acceptable, therefore, Open Item 8.3.2.1-5 is closed.

Physical Independence of Redundant Circuits

There are four safety-related separation groups for the cable and raceway system (Groups A, B, C, and D). Separation group A contains safety-related circuits from Division A. Similarly, separation Groups B, C, and D contain Divisions B, C, and D, respectively. There is also a Group N, which contains non-safety-related circuits. Cables of each separation group are run in separate raceways and are physically separated from cables of other separation groups. Group N raceways are separated from safety-related Groups A, B, C, and D. Raceways from Group N are routed in the same areas as the safety-related groups according to spatial separation as stipulated in RG 1.75 and IEEE 384-1974.

The separation of safety-related systems will be maintained by the physical layout and the unique identification of equipment, raceways, and cables. Different color codes will be used for different safety-related electrical equipment, power, control and instrumentation cables, and raceways. In addition, each circuit and cable will carry a unique identification that will provide a means of distinguishing between circuits of different separation divisions.

Cables associated with the same division will require no physical separation and may be routed in common raceways. Signal cables associated with the same division will require no physical separation and may be routed in common raceways.

Physical separation between electrical equipment and components associated with redundant divisions will be consistent with the criteria established in IEEE 384-1974, Sections 5 and 6. The Class 1E equipment of each division will be located in safety-related structures as discussed in Chapter 3 of this report.

Within the MCR and remote shutdown area, the minimum vertical separation for an open top cable tray is 0.9 m (3 ft) and the minimum horizontal separation is 0.3 m (1 ft). The minimum separation between enclosed raceways qualified as barriers is 2.5 cm (1 in.). Within general plant areas, the minimum vertical separation is 1.5 m (5 ft), and the minimum horizontal separation is 0.9 m (3 ft) for open cable trays. The minimum separation between enclosed raceways qualified as barriers is 2.5 cm (1 in.).

Within panels and control switchboards, the minimum spatial separation groups (both field-routed and vendor-supplied internal wiring) is 15.2 cm (6 in.). Where it is not possible to maintain this separation, barriers are installed between components and wiring of different separation groups. Where spatial separation requirements between raceways of different separation groups are not met, fire barriers are installed. Where the minimum vertical separation is not maintained, a barrier is installed that extends 0.3 m (1 ft) on each side of the tray system or to the wall, if a wall is within 0.3 m (1 ft) when the trays are arranged in stacks. Where the minimum horizontal separation is not maintained, a barrier is installed that extends from 0.3 m (1 ft) above (or to the ceiling) to at least 0.3 m (1 ft) below the tray system (or to the floor).

In addition, Westinghouse specified that the minimum separation between non-Class 1E conduit and Class 1E open top cable trays is 2.5 cm (1 in.). The staff did not agree that this was in accordance with IEEE 384-1974. The staff required that Westinghouse give justifications and bases supported by tests to justify this approach. This was identified as Open Item 8.3.2.1-6.

In response to Open Item 8.3.2.1-6, Westinghouse stated in Revision 3 to the SSAR that the raceways from group N are routed in the same areas as the Class 1E Groups according to spatial separation stipulated in RG 1.75 and IEEE 384-1974 with the following exceptions:

- Within the MCR and remote shutdown area (non-hazard areas), the minimum vertical separation for open cable tray is 3 inches (7.6 cm) and the minimum horizontal separation is 1 inch (2.54 cm).

- Within general plant areas (limited hazard areas), the minimum vertical separation is 12 inches (0.3 m), and the minimum horizontal separation is 6 inches (0.15 m) for the open cable trays with low voltage power circuits for cable sizes less than 2/0 American wire gauge (AWG). *For configurations that involve exclusively limited energy content* cables (I&C), these minimum distances are reduced to 3 inches (7.65 cm) and 1 inch (2.54 cm) respectively.

- Within panels and control switchboards, the minimum horizontal separation between components or cables of different separation groups (both field-routed and vendor-supplied internal wiring) is 1 inch (2.54 cm), and the vertical separation distance is 6 inches (0.15 m).

Section 5.1.1.2 of IEEE 384-1974 states that the minimum separation distance can be established by analysis of the proposed cable installation. This analysis shall be on the basis of tests performed to determine the flame-retardant characteristics of the proposed cable installation, considering features such as cable insulation and jacket materials, cable tray fill, and cable tray arrangement. Westinghouse has established the minimum separation distances by analysis of the proposed cable installation. This analysis is founded on tests performed and the findings published in the IEEE Transactions on Energy Conversion, Volume 5, No.3, September 1990 titled, "Cable Separation - What Do Industry Testing Program Show?" and published by the IEEE Working Group on Independence Criteria, SC-6.5 of the Nuclear Power Engineering Committee. The staff has reviewed the test results and concluded that these results support the Westinghouse analysis. On this basis, the staff finds the lesser distances used by the applicant in their design to be acceptable. Therefore, Open Item 8.3.2.1-6 is closed.

Non-Class 1E circuits are electrically isolated from Class 1E circuits by isolation devices and are physically separated from Class 1E circuits in accordance with the above separation criteria. Class 1E circuits from different separation groups are electrically isolated by isolation devices, shielding, and physical separation in accordance with RG 1.75 for circuits in raceways. Non-Class 1E raceways and supports installed in seismic Category I structures are designed and/or physically arranged so that an SSE could not cause failure of seismic Category I components.

Power and control cables are installed in conduits or ventilated bottom trays (ladder type). Solid tray covers are used in outdoor locations. Instrumentation cables are routed in conduits or solid bottom cable trays with solid tray covers.

Raceways are kept at a reasonable distance from actual or potential heat sources (such as steam piping, steam generators, boilers, high- and low-pressure heaters, and others). Cases of heat source crossings are evaluated, and supplemental heat shielding is used if necessary.

Separate trays are provided for each voltage level (4.16 kV, 120 Vac, 125 Vdc, I&C). Cable trays are physically arranged from top to bottom, in accordance with the function and voltage class of the cables and with the highest voltage at the top. Vertically stacked trays are arranged from top to bottom with a minimum of 12 inches (0.3 meters) vertical spacing maintained between trays of different service levels within the stack.

Raceways installed in seismic Category I structures have seismically designed supports, or are shown not to affect safety-related equipment should they fail. Conduits are attached to Seismic Category I equipment with flexible type connections.

Where hazards to safety-related raceways are identified, a minimum separation is maintained between the break and/or missile source and any safety-related raceway. Alternatively, a barrier designed to withstand the effects of the hazard is placed to prevent damage to raceways of redundant systems. Spatial separation is provided where redundant circuits, devices, or equipment (different separation groups) are exposed to the same external hazards. Otherwise, qualified barriers are installed.

For the Class 1E dc system, the 24-hour and 72-hour battery banks are located in the auxiliary building in rooms apart from chargers and distribution equipment. The battery rooms are ventilated to limit hydrogen accumulation. Each of the four divisions of dc systems is electrically isolated and physically separated to prevent the failure or unavailability of a single battery, battery charger, or inverter from adversely affecting a redundant division.

Class 1E Uninterruptible Power System

The Class 1E UPS provides power at 208/120 Vac to four independent divisions of Class 1E instrument and control power buses. Divisions A and D each consist of one Class 1E inverter with an I&C distribution panel and a backup regulating transformer. The inverter is powered from the respective 24-hour battery bank. Divisions B and C each consist of two inverters, two I&C distribution panels, and a backup regulating transformer. One inverter is powered by the 24-hour battery bank and the other by the 72-hour battery bank. Under normal operation, the Class 1E inverters receive power from the associated battery bank. If an inverter is inoperable, or the Class 1E 125-Vdc input to the inverter is unavailable, the power is transferred automatically to the backup ac source by a static transfer switch, featuring a make-before-break contact arrangement. The backup power is received from the diesel generator backed non-Class 1E 480-Vac bus through the Class 1E regulating transformer. In addition, a manual mechanical bypass switch is provided to allow connection of backup power source when the inverter is removed from service for maintenance.

The staff was concerned that failures of the UPS system have been shown to constitute one of the main causes of forced plant outages. Therefore, the staff requested that Westinghouse verify that the failure or unavailability of a single battery, battery charger, or inverter will not result in a plant trip or a forced outage. Westinghouse provided the following response:

> "Following the loss of either a Class 1E or a non-Class 1E inverter, the associated dc bus remains energized, and the dc loads are not affected. The 208/120-Vac instrumentation and control power bus associated with the failed inverter remains continuously energized. Each uninterruptible power supply (UPS) includes an inverter and a backup regulating transformer that can supply the associated instrumentation and control bus if the inverter fails. The UPS includes a static transfer switch that automatically transfers the bus to the regulated power source if power is unavailable from the inverter. A manual mechanical bypass switch is also included in the UPS to provide a second connection for the bus to the backup regulated power source when the inverter is removed from service for maintenance.
>
> Therefore, with a failure of a single battery charger or a single battery, power is continuously maintained to the dc buses. With a failure of an inverter, power to the instrumentation and control power bus is automatically transferred to a regulated backup power source. With a single failure or the unavailability of these components, the associated buses remain energized, thereby preventing a plant trip or forced outage."

The design as described above is in accordance with IEEE 379 and is, therefore, acceptable.

The staff was concerned about the possibility of age-related failures of inverters and chargers, especially with an increase in ambient temperatures being considered as the main cause of age-related failures (particularly for capacitors, transformers, and semiconductors). The staff asked Westinghouse to describe the conservatism included in the AP600 design with respect to temperature margins, and whether any forced air cooling for the battery chargers is required.

In the response to RAI 435.39, Westinghouse stated that the Class 1E and non-Class 1E inverters and chargers (UPS equipment) are located in a controlled environment during normal operation. The room ambient temperature is maintained between 21.1 °C (70 °F) and 25 °C (77 °F) for the Class 1E and non-Class 1E UPS equipment, respectively. The UPS equipment is rated for continuous operation at 40 °C (104 °F) ambient. In addition, the temperature-sensitive components (such as capacitors, transformers, and semiconductors) used in the UPS equipment are designed to continuously withstand a higher temperature, about 60 °C to 70 °C (140 °F to 158 °F). Therefore, considering the temperature margins conservatively provided in the AP600 design, an age-related failure of the UPS equipment is not expected. The staff finds the conservatism in the design acceptable.

Air cooling is provided by the nuclear island nonradioactive ventilation system for the Class 1E UPS equipment located in the auxiliary building. The non-Class 1E UPS equipment is located in the annex building. Air cooling is provided in this building by the annex building nonradioactive ventilation system.

The staff was concerned that most of the UPS vendors comply with specification requirements for total harmonic distortion (THD), with the provision that these requirements are met for linear loads. The loads used for digital control power supplies and computers in the AP600 are inherently nonlinear in nature. Also, variable-speed drive systems and fluorescent lighting blasts introduce harmonics into the plant distribution system.

In the response to RAI 435.46, Westinghouse stated that the problem of THD resulting from the nonlinear loads inherently present in the AP600 has been recognized. The UPS inverters have harmonic filters designed specifically to reduce the effects of large third-, fifth-, seventh-, and higher-order harmonics that may result from anticipated 100-percent nonlinear loads. To provide high-quality power from the UPS system, the inverters will be specified to power nonlinear loads with a crest factor of 2 or higher (ratio of peak to rms value). This design is in accordance with IEEE 944-1986 and is, therefore, acceptable. In addition, the variable-speed drives used for the main feedwater pumps will have special filters to eliminate the introduction of harmonics into the distribution system. Also, the battery chargers will be furnished with output filtering to limit ripple currents feeding into the dc power supply for the inverters as specified by IEEE Standard 519-1992.

Analysis of dc Power and UPS System

Westinghouse provided a failure mode and effects analysis for the Class 1E dc and UPS system.

In the event of a LOOP coincident with a generator trip, ac power to the battery charger is provided from two separate non-Class 1E onsite standby diesel generators. Division A and C chargers receive power from one diesel generator, and division B and D chargers from the second diesel generator. Provisions are also made to power divisions B and C chargers from the ancillary ac generators during the post-72-hour period. The Class 1E battery chargers and Class 1E regulating transformers are designed to limit the input ac current to an acceptable value under faulted conditions on the output side. They have circuit breakers at the input and output sides for protection. Westinghouse stated that there are two breakers in series to act as isolation devices between the Class 1E and non-Class 1E portions of the circuits; however, it did not address whether these breakers will be coordinated and periodically tested. This was Open Item 8.3.2.1-7.

In response to Open Item 8.3.2.1-7, Westinghouse stated in Revision 3 to the SSAR that the circuit breakers used for ac circuits will be properly coordinated and periodically tested to verify their current-limiting characteristics have not been lost. They are qualified as isolation devices between Class 1E and non-Class 1E circuits. Westinghouse included the appropriate changes in Section 8.3.2.1.1.1 of SSAR Revision 3, which is acceptable. Therefore, Open Item 8.3.2.1-7 is closed.

The four divisions are completely independent and located in separate rooms, have no shared equipment, and cannot be interconnected. Their circuits are routed in dedicated, physically separated raceways. This electrical and physical separation prevents the failure or unavailability of a single battery, battery charger, or inverter from adversely affecting a redundant division. The Class 1E dc and UPS system is designed in accordance with IEEE 308-1980 and IEEE 946-1985. The battery monitoring system detects battery open circuit conditions and monitors battery voltage. The Class 1E dc system is ungrounded; thus, a single ground fault does not cause immediate loss of a faulted system. A spare battery bank and charger enables testing, maintenance, and equalization of battery banks offline.

Design Features for Simultaneous Failures of Batteries

In the January 22, 1994, response to Q435.42, Westinghouse stated that the AP600 Class 1E dc and UPS system design has the following provisions to reduce the risk of simultaneous failure (common-cause failures) affecting more than one battery division.

The AP600 Class 1E dc and UPS system consists of four completely independent divisions located in separate rooms. Their circuits are routed in dedicated, physically separate raceways. The electrical independence between the Class 1E and non-Class 1E systems and between the Class 1E divisions is maintained in accordance with IEEE 384-1981 and RG 1.75.

The AP600 design uses state-of-the-art battery monitors. These monitors continuously monitor the condition of the battery by measuring intercell resistance to provide advance indication of a maintenance requirement. Also, the batteries are in a controlled environment during normal operation. The controlled environment helps eliminate the possibility of a common-cause failure caused by high ambient room temperature.

Industry experience does not indicate any failure mode trends of lead-calcium batteries that could result in common-cause failures when proper maintenance and quality control are exercised. Proper maintenance and surveillance procedures, as determined by the manufacturer's recommendations, will be developed by the COL applicant. This was COL Action Item 8.3.2.1-4 and Open Item 8.3.2.1-8.

In response to COL Action Item 8.3.2.1-4 and Open Item 8.3.2.1-8, Westinghouse in Revision 3 to the SSAR stated that the COL applicants referencing the AP600 certified design will address the manufacturer's recommendation for proper battery maintenance and surveillance. The staff concludes that this is acceptable, therefore, Open Item 8.3.2.1-8 is closed.

Conclusion

GDC 17 requires that the Class 1E 125-Vdc battery supply have sufficient capacity and capability to permit the safety systems to perform their required safety functions. It must also have sufficient independence, redundancy, and testability to allow for performing the safety functions assuming a single failure. GDC 18 of 10 CFR Part 50, Appendix A, requires that Class 1E 125-Vdc battery supply systems be designed with the following capabilities:

- permit appropriate periodic inspection and testing of important areas and features (such as wiring, insulation, connections, and switchboards) in order to assess the continuity of the systems and the condition of their components

- periodically test the operability and functional performance of the components of the systems

- periodically test the operability of the systems as a whole and (under conditions as close to design as practical) the full operational sequence that brings the systems into operation

Westinghouse states that components of the 125-Vdc system undergo periodic tests to determine the condition of the system. Batteries are checked for electrolyte level, specific gravity, and cell voltage. The surveillance testing of the Class 1E 125-Vdc system is performed as required by the AP600 technical specifications.

On the basis of the above evaluation of the dc systems, the staff concluded that the Class 1E dc and UPS system design meets GDCs 17 and 18, and is acceptable.

8.3.2.2 Non-Class 1E dc and UPS System

The non-Class 1E dc and UPS system consists of the dc electric power supply and distribution equipment, which together provide uninterruptible dc power backup to the plant non-Class 1E dc and ac loads that are needed for plant operation and investment protection. The non-Class 1E dc and UPS system consists of two subsystems representing two separate power trains. Both subsystems are located in separate rooms. Each subsystem consists of separate dc distribution buses, but these can be connected by a normally open circuit breaker. Each dc subsystem includes battery chargers, batteries, dc distribution equipment, and associated monitoring and protective devices.

There are four 125-Vdc buses. Buses 1, 2, and 3 provide 125-Vdc power to the associated inverter units that supply the ac power to the non-Class 1E UPS system. An alternative regulated ac power source for the UPS buses is supplied from the associated regulating transformers. Bus 4 supplies large dc motors and other dc power loads, but not inverter loads.

A 480-Vac distribution system backed by the onsite standby diesel generator provides the normal ac power to the battery chargers. The batteries supply the dc power in case the battery chargers fail to supply the dc distribution bus system loads. The batteries are sized to supply the system loads for a period of at least two hours after loss of all ac power sources. Each non-Class 1E dc distribution subsystem bus has provisions to allow connection of a spare non-Class 1E battery charger in case its non-Class 1E battery charger is unavailable because of maintenance, testing, or failure. There are also provisions for the non-Class 1E dc system to use the Class 1E spare battery bank as a temporary replacement for any non-Class 1E battery bank. In this configuration, the spare Class 1E battery bank does not simultaneously supply Class 1E safety loads. Additionally, the design includes two current interrupting devices to preserve the spare Class 1E battery integrity should the non-Class 1E bus experience an electrical fault. Therefore, the Class 1E spare battery would not be degraded.

Deterministic evaluations by Westinghouse, as part of their RTNSS evaluation identified the need for administrative controls on the non-Class 1E dc system needed to operate the diverse actuation system (DAS) for mitigation of an anticipated transient without scram (ATWS) event. See Section 8.6.2.4 of this report for additional discussion of the RTNSS resolution of this item.

<u>Conclusion</u>

The non-Class 1E 125-Vdc system provides dc and UPS to the plant's non-Class 1E dc and ac loads that are needed for plant operation and investment protection. This helps to ensure the independence of the Class 1E dc system from faults or failures in the non-Class 1E systems, and is therefore acceptable.

8.4 Other Electrical Features and Requirements for Safety

The staff reviewed certain safety-related electrical features of the AP600 design to determine whether they are implemented in accordance with the applicable criteria set forth in Section 8.1 of this report.

8.4.1 Containment Electrical Penetrations

To meet the guidelines set forth in IEEE 317-1983, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," as augmented by the recommendations of RG 1.63, the containment electrical penetration assemblies for the AP600 are designed to withstand, without loss of mechanical integrity, the maximum available fault current for a sufficient period of time to allow backup circuit protection to operate, assuming a failure of the primary protective device. The circuit overload protection systems for electrical penetration assemblies meet the single-failure criterion.

Westinghouse has applied the following design criteria to the electrical penetration circuits.

Westinghouse stated that Class 1E and non-Class 1E electrical penetration assemblies are maintained in separate nozzles. The physical separation of the Class 1E penetration assemblies comply with RG 1.75. The penetrations conform to the same functional service level as the cables (e.g., low level instrumentation is separated from power and control penetrations). Individual electrical penetrations are provided for each electrical service level, and are arranged physically from top to bottom in accordance with the function and voltage class of the cables. Specifically, medium-voltage power (4.16 kV) penetrations are at the top, and instrumentation (analog and digital) penetrations are at the bottom. For modular type penetrations (three penetration modules in one nozzle), Westinghouse has assigned the following modules:

- one module for low voltage ac power
- one module for ac control
- one module for instrumentation signal

Penetrations carrying medium-voltage power cables have thermocouples for monitoring the temperature within the assembly at the spot expected to have the hottest temperature.

Electrical circuits passing through electrical penetrations have primary and backup protective devices. These devices are to be selected to coordinate with the thermal capability (I^2t) of the penetration assemblies. Westinghouse stated that the penetrations can withstand the maximum short-circuit currents available either continuously without exceeding their thermal limit, or at least longer than the field cables of the circuits, so that the faults or overload currents are interrupted by the protective devices before a potential failure of a penetration.

In the response to RAI 435.60, Westinghouse stated that the electrical penetration assemblies will comply with the guidelines stipulated in IEEE 317-1983 as endorsed by RG 1.63, Revision 3. Primary and backup protection for the penetration conductors will be provided in accordance with IEEE 741-1990. The connectors, splices, or terminal blocks required for connection of cables to the containment electrical penetrations will be qualified pursuant to 10 CFR 50.49 to withstand a loss-of-coolant (LOCA) or steamline break. Section 5.4.2.2 of IEEE 741-1990 states that the time-current curves of the dual primary protection or the primary and backup protection shall coordinate with the time-current capability curve of the electrical penetration to be protected. This means that the protection shall cover the full range of currents up to the maximum short circuit currents. The staff has taken a position that the penetrations should be protected for the full range of currents up to the maximum short-circuit currents. The staff required further clarification in this area for the AP600 design. This was identified as DSER Open Item 8.4.1-1.

In response to Open Item 8.4.1-1, Westinghouse in Revision 3 to the SSAR stated that the penetrations are protected and rated to withstand the full range of currents up to the maximum short circuit current available in accordance with IEEE Standard 741-1990. The staff concludes that this is acceptable, therefore, Open Item 8.4.1-2 is closed.

The staff also concluded that the protective devices need periodic testing. Westinghouse had not stated that the COL applicant will address the provisions for periodically testing these penetration protective devices. This was COL Action Item 8.4.1-1 and DSER Open Item 8.4.1-2.

In response to COL Action Item 8.4.1-1 and Open Item 8.4.1-2, Westinghouse in Revision 3 to the SSAR stated that the COL applicants referencing the AP600 certified design will address the provisions for periodically testing penetration protection devices. The staff finds this acceptable, therefore, Open Item 8.4.1.2 is closed.

8.4.2 RCP Breakers

Two separate 4.16 kV switchgear buses, ES5 and ES6, power four RCPs. Each RCP is powered through two Class 1E circuit breakers connected in series. The purpose is to satisfy the safety-related tripping requirements associated with the AP600 design. In the response to RAI 435.11, Westinghouse stated that the RCP circuit breakers receive a signal to trip from the protection and safety monitoring system upon generation of the following signals:

- core makeup tank actuation signal
- first stage automatic depressurization (ADS) initiation
- high pump bearing water temperature

The pumps are tripped on core makeup tank actuation and first stage ADS initiation to support LOCA mitigation by precluding interaction of the RCP pressure head with gravity injection makeup to the RCS from the core makeup tanks. The RCP trip function is part of the engineered safeguards response to a design-basis LOCA, and is therefore implemented with Class 1E circuit breakers. This ensures that the RCPs are tripped before the passive systems start. The pumps are also tripped on a high pump bearing water temperature to protect pump flow coastdown capability following the loss of component cooling water flow to the RCPs.

The staff concluded that these RCP trip functions are part of the engineered safeguards needed to respond to design-basis LOCAs and, as a result, are implemented with Class 1E circuit breakers. The staff finds this acceptable, because it will allow the operation of the passive systems as designed.

8.4.3 Thermal Overload Protection Bypass

Motor-operated valves with thermal overload protection devices for the valve motors are used in safety systems and their auxiliary supporting systems. Operating experience has shown that indiscriminate application of thermal overload protection devices to the motors associated with these valves could result in needless hindrance to the successful completion of safety-related functions. RG 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves" (November 1975), recommends (Position C.1) bypassing thermal overload devices during accident conditions, or (Position C.2) properly selecting the setpoints for the thermal overloads in a manner that precludes spurious trips. For the AP600 design, Westinghouse committed to comply with Position C.1. In the response to RAI 435.49, Westinghouse stated that the thermal overload protection for the Class 1E dc valve motor operators will be bypassed only during DBA conditions, and will be maintained in service during normal and testing operations. This is in accordance with Position C.1 of RG 1.106 and the design is, therefore, acceptable. For non-Class 1E valve motor operators, the thermal overload protection will remain in service at all times. The overload elements will be sized in accordance with industry standards and manufacturer's guidelines.

The staff finds that the above proposed design conforms to RG 1.106, and is therefore acceptable.

8.4.4 Power Lockout to Motor-Operated Valves

Westinghouse has identified the following valves that require removal of power consistent with the guidelines of BTP ICSB-18, dated July 1981:

- accumulator MOVs: Isolation valves in the accumulator discharge lines have power removed in the open position during normal operation. Power is removed from the valves at the MCC.

- in-containment refueling water storage tank (IRWST) gravity injection line MOVs: The isolation valves for gravity injection of the IRWST have power removed in the open position during normal operation. The removal of power ensures IRWST injection following RCS depressurization to provide long-term cooling. Power is removed from the valves at the MCC.

- passive residual heat removal (PRHR) heat exchanger inlet isolation valve: The inlet isolation valve to the PRHR heat exchanger provides emergency core decay heat removal for non-LOCA events. The removal of power from the isolation valve ensures the capability of heat removal under high pressure and temperature conditions in the RCS. Power is removed from the valve at the MCC.

The staff concludes that disconnecting power to these fluid system electrical components is preferred over the possibility of a single failure that might cause an undesirable component actuation and the design is, therefore, acceptable. Although not required for single-failure purposes, power is removed from the normal residual heat removal (RHR) suction isolation valves to minimize the potential for an intersystem LOCA. The inner and outer RCS isolation valves in the normal RHR path have power removed during normal operation. Power is removed from these valves at the MCC.

In BTP ICSB-18, Item B-4, the staff states that these valves which have the electrical power removed to meet the single-failure criterion should have redundant position indication in the MCR, and the position indication system should, itself, meet the single-failure criterion. The staff required that the redundant indication be powered from different sources. This was identified as DSER Open Item 8.4.4-1.

Westinghouse has addressed Open Item 8.4.4-1 in Chapter 7.5.4 of Revision 11 to SSAR. It states that electrically operated valves, which have the electrical power removed to meet the single-failure criterion, are provided with redundant valve position sensors. Each of the two position sensors is powered from a different power source. The staff finds this acceptable and, therefore, Open Item 8.4.4-1 is closed.

8.4.5 Submerged Class 1E Electrical Equipment as a Result of a LOCA

In the response to RAI 435.56, Westinghouse stated that, in the event of potential flooding or wetting, one of the following criteria is applied to protect equipment for service in such an environment:

- Equipment will be qualified pursuant to 10 CFR 50.49 for submergence resulting from flooding or wetting.

- Equipment will be protected from flooding or wetting.

- As an alternative to protecting the equipment, the equipment will be evaluated to show that failure of the equipment because of flooding or wetting is acceptable, because its safety-related function is not required or has otherwise been accomplished once it is submerged.

In Section 3.11 of the SSAR, Westinghouse discusses the environmental qualification of the electrical and mechanical equipment. The environmental zones (equipment room locations) and list of safety-related electrical and mechanical equipment are provided in SSAR Table 3.11-1. On the basis of its review, the staff concluded that the required Class 1E equipment will be adequately protected from submergence. This design approach is, therefore, acceptable.

8.5 SSAR Documentation of Responses to Requests for Additional Information

With respect to the electrical systems, Westinghouse has responded to the staff's RAIs by providing supplemental information in a section dedicated to questions and responses. In most cases, Westinghouse concluded that no SSAR revision is needed. The supplemental information primarily involves additional description of the design, or additional analysis or rationale for the particular design feature or approach. The staff believes that the supplemental information involving additional description of the design for grounding should be integrated into Chapter 8 of the SSAR. This was DSER Open Item 8.5-1.

Westinghouse has incorporated the information on grounding in Revision 13 to Section 8.3.1.1.7 of the SSAR. On this basis, Open Item 8.5-1 is closed.

8.6 Compliance With Regulatory Issues

8.6.1 Generic Issues and Operational Experience

The following generic issues and operational experience (bulletins and generic letters) were evaluated on the basis of the assumption that the electrical power system requirements would not be significantly impacted by the resolution of the RTNSS issue.

- A-25, "Non-Safety Loads on Class 1E Power Sources"
- A-35, "Adequacy of Offsite Power Systems"
- A-44, "Station Blackout"
- B-53, "Load Break Switch"
- B-56, "Diesel Reliability"
- 128, "Electrical Power Reliability"
- NUREG-0737, II.E.3.1, "Emergency Power Supply for Pressurizer Heaters"
- NUREG-0737, II.G.1, "Emergency Power for Pressurizer Equipment"
- Generic Letter 88-15, "Electrical Power Systems - Inadequate Control Over Design Process"

The staff stated that if the electrical design requirements changed as a result of the RTNSS, each of the above issues would have to be reevaluated. This was DSER Open Item 8.6.1-1. These issues are discussed in Chapter 20 of this report.

The RTNSS identification process has been resolved (see Section 8.6.2.4 of this report). All of the above issues were reevaluated with respect to that resolution and the staff concluded that there was no impact on these issues. On the basis of this result, Open Item 8.6.1-1 is closed.

8.6.2 Advanced Light-Water Reactor Certification Issues

The following paragraphs discuss the policy, technical, and licensing issues pertaining to passive plant designs that relate to the electrical portion of the AP600 design.

8.6.2.1 Station Blackout

The NRC issued NUREG-0649, NUREG-1032, and NUREG-1109 to address the unresolved safety issue concerning station blackout (USI-A44). To resolve this issue, the NRC published 10 CFR 50.63, which established new requirements so that an operating plant can withstand a station blackout as defined in 10 CFR 50.2, and recover from such an event. The NRC published RG 1.155 as guidance for satisfying 10 CFR 50.63.

The AP600 design minimizes the potential risk contribution of station blackout (SBO) by not requiring ac power sources for design-basis events. Safety-related systems do not need non-safety-related ac power sources to perform safety-related functions. The AP600 safety-related passive systems automatically establish and maintain safe-shutdown conditions for the plant following design-basis events, including an extended loss of ac power sources. The passive systems can maintain these safe-shutdown conditions after design-basis events, without operator action, following a loss of both onsite and offsite ac power sources.
Section 1.9.5.4 of the SSAR provided additional information on long-term actions following an extended station blackout beyond 72 hours.

The AP600 design also includes redundant, non-safety-related, onsite ac power sources (diesel generators) to provide electrical power for non-safety-related active systems that provide a DID function.

The following AP600 design features mitigate the consequences of an SBO:

- a full load rejection capability to reduce the probability of loss of onsite power
- safety-related PRHR heat exchanger
- safety-related passive containment cooling
- bleed and feed capability, using the safety-related automatic depressurization system in conjunction with the water available from the core makeup tanks, accumulators, and IRWST
- Class 1E batteries sized for 72 hours of operation under station blackout conditions
- RCPs without shaft seals
- passive cooling for the rooms containing equipment assumed to operate during SBO conditions (the protection and safety monitoring system cabinet rooms and the MCR) so that this equipment continues to operate for 72 hours

The staff decided to resolve the station blackout (SBO) issue for the AP600 design as part of the process defined for resolving the RTNSS issue. This was Open Item 8.6.2.1-1. See Section 8.6.2.4 for resolution of the RTNSS. On the basis of this evaluation, Open Item 8.6.2.1-1 is closed.

8.6.2.2 Electrical Distribution

The Commission approved the following recommendations in SECY-91-078 for plant designs:

- An alternative offsite power source will be available for non-safety-related loads, unless the design margins for loss of non-safety-related loads are no more severe than turbine-trip-only events in current plants.
- At least one offsite circuit to each redundant safety division will be supplied directly from offsite power sources with no intervening non-safety-related buses.

The AP600 design does not rely on active systems for safe shutdown. Both of the recommendations on this issue are closely tied to the lack of a second normally available offsite circuit. The staff decided to resolve the electrical distribution issue for the AP600 by evaluating the ac power system features using the process defined herein for resolving the RTNSS. This was Open Item 8.6.2.2-1. See Section 8.6.2.4 of this report for resolution of the RTNSS. On the basis of this evaluation, Open Item 8.6.2.2-1 is closed.

8.6.2.3 Industry Codes and Standards

This issue deals with the use of recently developed or modified industry standards or codes, which the vendors are using in applications, but which have not yet been reviewed by the NRC for acceptability. The NRC has recommended the use of the newest codes and standards endorsed by the NRC. Unapproved revisions are to be reviewed on a case-by-case basis.

With respect to the AP600 electrical design, Westinghouse proposed to use the most recent revision of applicable industry standards as of January 1, 1990.

For standards endorsed by regulatory guides and subsequently superseded by a more recent revision (up to January 1, 1990), Westinghouse submitted an analysis that supports their conclusion that these revisions to the standards do not deviate from the design philosophy of the existing regulatory guides and endorsed standards. The staff reviewed the analysis and agreed with the Westinghouse assessment, except with regard to IEEE 323-1983. This standard addresses environmental qualification, and the 1974 version of this standard is cited in the regulations. Therefore, the reference to the 1983 version has not been accepted to date. The staff will require that this standard be deleted from the analysis. This was Open Item 8.6.2.3-1.

In response to Open Item 8.6.2.3-1, Westinghouse in Revision 9 to the SSAR stated that Westinghouse will conform with IEEE 323-1974. On this basis, Open Item 8.6.2.3-1 is closed.

In addition, for some electrical features, Westinghouse has used industry standards that have not been endorsed by any regulatory guide. In these cases, the staff reviewed the use of each standard in the AP600 design, and concluded that the use of the standards is appropriate and acceptable.

8.6.2.4 Regulatory Treatment of Nonsafety Systems

The NRC concluded that its review of passive designs requires not only a review of the passive safety-related systems, but also a review of the functional capability and availability of the active non-safety-related systems to provide significant DID and accident and core damage prevention capability. In March 1994, the NRC issued Commission policy paper SECY-94-084, which outlines the process for resolving the RTNSS issue. This process includes a combination of probabilistic and deterministic criteria to identify risk-significant, non-safety-related systems. Westinghouse's implementation of this process is documented in WCAP-13856, "AP600 Implementation of Regulatory Treatment of Nonsafety-Related System Process."

The AP600 non-safety-related active systems are designed to provide reliable support for normal plant operations, and to provide DID to minimize unnecessary challenges to the safety-related passive systems. These active systems are designed for more probable component and system failures. They are capable of being powered by the non-safety-related diesel generators, and they have non-safety-related automatic actuation and controls that are separate from those of the safety-related systems. In addition, these systems are designed to provide highly reliable performance. As such, each system is to include reliable, proven equipment and component designs. The continuing equipment performance is to be controlled by requirements concerning the availability, inservice inspection and testing, and overall long-term performance, based on the assumptions in the PRA evaluation. For electrical systems, this was Open Item 8.6.2.4-1.

The resolution of RTNSS has resulted in a number of administrative availability controls for the electrical systems. Specifically, the final resolution of the RTNSS open item resulted in the determination that administrative availability controls of ac power from the diesel generators are needed in all modes. This has provided the staff added confidence that power to the DID systems (such as the normal residual heat removal system (RNS) and its support systems) will have a reasonable chance of being available when called upon, will permit DID mitigation of many accidents and transients without challenging all the passive safety systems, and will provide a non-safety related means of supplying power to the safety-related passive protection and safety monitoring system (PMS) for actuation and postaccident monitoring. The RNS provides a non-safety-related means to inject water into the RCS following ADS actuation in Modes 1, 2, 3, and 4. The availability administrative controls are provided in Section 16.3 of the AP600 SSAR and include controls for the ac power supply function in Modes 1, 2, 3, and 4 when RNS injection and PMS actuation are more risk significant.

The final resolution of the RTNSS open item has also resulted in the determination that ac power is risk significant during shutdown Modes 5 and 6, especially during reduced inventory conditions. The administrative availability controls specify that one offsite and one onsite ac power supply should be available during Modes 5 and 6 with reduced inventory when the loss of RNS cooling is important. The offsite power source is available through the transmission switchyard and either the main step-up transformer/unit auxiliary transformer or the reserve auxiliary transformer. The onsite power source is available from one of the two diesel generators. If both of these ac power sources are not available, the plant should not enter reduced inventory conditions.

Also, the final resolution of the RTNSS open item resulted in the determination that the non-Class 1E dc and UPS Systems are important for DAS on the basis of 10 CFR 50.62 (ATWS) and to support ESF actuation. DAS relies upon the functioning of non-Class 1E dc and the UPS system to actuate the components and power the logic needed to meet the requirements of 10 CFR 50.62. The availability controls specify that the non-Class 1E dc and UPS system be available to the DAS sensors, DAS actuation, and the devices that control the actuated components in Mode 1 for DAS ATWS mitigation function and in Modes 1, 2, 3, 4, 5, and 6 for DAS ESF actuation.

In addition, the final resolution of the RTNSS open item has also resulted in the determination that one Ancillary Diesel Generator and its fuel oil storage tank should be available during all Modes of plant operation. After 72 hours with no alternate sources of power available, ancillary diesel generators will be used to power the MCR and I&C room ancillary fans, the PCS recirculation pumps, and MCR lighting, and provide extended post-accident monitoring instrumentation. The availability controls specify that one ancillary diesel generator and its fuel oil storage tank should be available during all Modes of plant operation.

Therefore, Westinghouse has provided availability controls for these electrical areas that are RTNSS important, and has committed to include these controls in the SSAR and in the design certification rule to make the commitment binding on the COL applicant. This is acceptable and, therefore, Open Item 8.6.2.4-1 is closed with respect to the electrical systems. The overall discussion and evaluation of the RTNSS process is in Chapter 22 of this report.

9 AUXILIARY SYSTEMS

The staff's review of the AP600 auxiliary systems is provided in the following sections: 9.1, Fuel Storage and Handling; 9.2, Water Systems; 9.3, Process Auxiliaries; 9.4, Air-conditioning, Heating, Cooling, and Ventilation Systems; 9.5, Other Auxiliary Systems.

9.1 Fuel Storage and Handling

The staff's review of the AP600 fuel storage and handling systems is provided in the following sections: 9.1.1, New Fuel Storage; 9.1.2, Spent Fuel Storage; 9.1.3, Spent Fuel Pool Cooling and Purification; 9.1.4, Light Load Handling System (Related To Refueling); and 9.1.5, Overhead Heavy Load Handling Systems.

9.1.1 New Fuel Storage

The staff has reviewed the Westinghouse AP600 advanced reactor's new fuel storage capability in accordance with Section 9.1.1 of the NRC Standard Review Plan (SRP). The staff's acceptance of the new fuel storage facility is contingent upon whether the design complies with the following requirements:

- General Design Criterion (GDC) 2, as it relates to the ability of the facility and the structures housing it to withstand the effects of natural phenomena, such as earthquakes.

- GDC 5, as it relates to whether shared structures, systems, and components (SSCs) important to safety are capable of performing required safety functions

- GDC 61, as it relates to the facility design for fuel storage

- GDC 62, as it relates to the prevention of criticality

Compliance with GDC 2 is on the basis of adherence to the guidance of Position C.1.1 of Regulatory Guide (RG) 1.29, as it relates to the seismic classification of facility components. Specific criteria necessary to meet the requirements of GDC 61 and 62 are American Nuclear Society (ANS) 57.1 and ANS 57.3 as they relate to preventing criticality and to aspects of the radiological design. In Section 9.1.1 of the standard safety analysis report (SSAR), Westinghouse provides the design bases, a description, and a safety evaluation of the new fuel storage arrangement for the AP600 design.

In Section 9.1.1 of the SSAR, Westinghouse states that the new fuel will be stored in a high-density rack that includes integral neutron absorbing material to maintain the required degree of subcriticality. The rack is designed to store fuel of the maximum design-basis enrichment. The rack will include storage locations for 56 fuel assemblies. The rack array will have a center-to-center spacing of 27.7 cm (10.9 in). This spacing provides the minimum separation between adjacent fuel assemblies that is sufficient to maintain a subcritical array, even if the building is flooded with unborated water or fire extinguishant aerosols, or during any design-basis event. The new fuel storage facility will be located within the seismic Category I auxiliary building fuel handling area. The dry, unlined, approximately 4.7 m (15.5 ft) deep reinforced concrete pit is designed to support the new fuel storage rack. The rack will be supported by the pit floor and laterally supported at the rack top grid structure by the pit wall structures. The new fuel pit will normally be covered to prevent foreign objects from entering the new fuel storage rack.

In Section 9.1.1.3 of the SSAR, Westinghouse provides a safety evaluation to demonstrate that the new fuel storage rack design complies with the design bases. Section 9.1.1.3 also states that the new fuel racks are purchased equipment, and that the purchase specification will require the vendor to perform a criticality analysis of the new fuel storage racks. Westinghouse considered normal and postulated accident conditions such as flooding with pure water and low density optimum moderator "misting." The following design features are used to minimize the possibility of these accidents:

- travel limits on handling equipment capable of carrying loads heavier than fuel components
- rack designed for safe-shutdown earthquake (SSE) conditions
- rack designed for dropped fuel assembly (and handling tool) conditions
- new fuel storage pit cover to protect new fuel from dropped objects and debris

In addition, the new fuel pit is not accessed by the fuel handling machine or by the cask-handling crane. This precludes moving loads greater than that of the fuel components over new fuel assemblies.

The staff performed its review in accordance with the guidance and acceptance criteria in Section 9.1.1 of the SRP. The staff directed its evaluation to determine whether or not the new fuel storage design complies with the requirements of GDC 2, 5, 61, and 62. On the basis of its review, the staff concludes that:

- The new fuel storage facility will be located within the seismic Category I auxiliary building fuel handling area per Section 9.1.1 of the SSAR. The new fuel storage rack is designed to meet the seismic Category I requirements of RG 1.29. Therefore, the staff finds that the new fuel storage facility meets the requirements of GDC 2.
- The AP600 design can be used at either single-unit or multiple-unit sites. Nonetheless, in Section 3.1.1 of the SSAR, Westinghouse states that the AP600 design is a single-unit plant and that "if more than one unit is built on the same site, none of the safety-related systems will be shared." Should a multiple-unit site be proposed, the combined license (COL) applicant referencing the AP600 design will be required to apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs important to safety to perform their required safety functions.

- In Section 9.1.1.3 of the SSAR, Westinghouse states that the design of the rack is such that K_{eff} remains less than or equal to 0.95 with new fuel of the maximum design-basis enrichment. For a postulated accident condition of flooding the new fuel storage area with unborated water, K_{eff} will not exceed 0.98. Section 4.3.2.6.1 of the SSAR states that the two principal methods of preventing criticality of fuel assemblies outside the reactor are to limit the fuel assembly array size and limit interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies. The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95-percent probability at a 95-percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95, as recommended in American National Standards Institute (ANSI) 57.1 and 57.3. Therefore, the staff finds that the new fuel facility meets the requirements of GDC 61 and 62.

The staff reviewed the new fuel storage capability for compliance with GDC 2, 5, 61, and 62, as referenced in Section 9.1.1 of the SRP. However, the staff was unable to reach a final conclusion regarding the acceptability of the new fuel storage facility because it was still reviewing the seismic classification of the new fuel storage structure, as well as information about protecting the fuel inside the fuel storage pit. These were draft safety evaluation report (DSER) Open Items 9.1.1-1 and 9.1.1-2, respectively.

The staff has now completed its review of the new fuel storage facility, including the seismic classification, and the protection of fuel inside the fuel storage pit. The new fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. Accordingly, the staff finds this acceptable to meet the requirements of GDC 2 as it relates to the ability of the facility and the structures housing it to withstand the effects of natural phenomena, such as earthquakes. In addition, in that the AP600 is a single unit design, and a COL applicant must comply with GDC 5 for a multiple-unit site, the staff finds that the requirements of GDC 5 are satisfied as it relates to whether shared SSCs important to safety are capable of performing required safety functions (see Section 9.2 of this report). Based on the analysis set forth above, the staff also finds that the requirements of GDC 61 are met as it relates to the facility design for fuel storage, and the requirements of GDC 62 are met as it relates to the prevention of criticality. Consequently, Open Items 9.1.1-1 and 9.1.1-2 are closed.

9.1.2 Spent Fuel Storage

The staff reviewed the spent fuel storage capability in accordance with Section 9.1.2 of the SRP. The staff's acceptance of the spent fuel storage facility is on the basis of compliance with the following requirements:

- GDC 2, as it relates to the ability of the facility and the structures housing it to withstand the effects of natural phenomena, such as earthquakes, tornados, and hurricanes.

- GDC 4, as it relates to the ability of the facility and the structures housing it to withstand the effects of external missiles, and internally-generated missiles, pipe whip, jet impingement forces, and adverse environmental conditions associated with pipe breaks, such that safety functions will not be impaired

- GDC 5, as it relates to whether shared SSCs important to safety are capable of performing required safety functions

- GDC 61, as it relates to the facility design for fuel storage and handling of radioactive materials

- GDC 62, as it relates to the prevention of criticality

- GDC 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions

Compliance with the requirements of GDC 2 is on the basis of adherence to the guidance of Position C.3 of RG 1.13; the applicable portions of RG 1.29 and RG 1.117; and paragraphs 5.1.1, 5.1.3, 5.1.12, 5.3.2, and 5.3.4 of ANS 57.2. Compliance with the requirements of GDC 4 is on the basis of adherence to the guidance of Position C.3 of RG 1.13, as well as RGs 1.115 and 1.117, and the appropriate paragraphs of ANS 57.2. Compliance with the requirements of GDC 61 is on the basis of adherence to the guidance of Positions C.1 and C.4 of RG 1.13, the appropriate paragraphs of ANS 57.2, and adherence to the fuel storage capacity guidelines noted in Subsection III.1 of Section 9.1.2 of the SRP. Compliance with the requirements of GDC 62 is on the basis of adherence to the guidance of Positions C.1 and C.4 of RG 1.13, as well as the appropriate paragraphs of ANS 57.2. Finally, compliance with the requirements of GDC 63 is on the basis of adherence to the guidance of paragraph 5.4 of ANS 57.2.

In Section 9.1.2 of the SSAR, Westinghouse presents the design bases, facilities description, and a safety evaluation of the spent fuel storage arrangement. In addition, Westinghouse indicates that the spent fuel will be stored in high-density racks that include integral, neutron-absorbing material to maintain the required degree of subcriticality. The racks are designed to store fuel of the maximum design-basis enrichment. The rack arrays will have a center-to-center spacing of 27.7 cm (10.9 in.), and storage locations for 619 fuel assemblies. In addition, the rack module will contain integral storage locations for five defective fuel storage containers. The spent fuel storage racks will be seismic Category I, and will be located within the spent fuel pool. The racks will consist of an array of cells interconnected to each other at several elevations, and to supporting grid structures at the top and bottom elevations. The rack modules will be free-standing, neither anchored to the pool floor nor braced to the pool wall.

The spent fuel storage facility (spent fuel pool) will be located within the seismic Category I auxiliary building fuel handling area (see Chapter 3 of this report for the staff's evaluation seismic Category I structures). The SSAR states that the facility will be protected from the effects of natural phenomena, such as earthquakes, wind, tornados, floods, and external missiles. However, the SSAR did not state whether the pool itself will be seismic Category I and whether it will be protected from internal missiles. In its response to Q410.232 dated July 8, 1994, Westinghouse stated that the spent fuel pool will be seismic Category I. Additionally, SSAR Section 9.1.1.2.1.E was changed to state that internally-generated missiles are of no concern because the fuel handling area does not contain any credible sources of internally-generated missiles.

In Section 9.1.2.3 of the SSAR, Westinghouse provides a safety evaluation to demonstrate that the spent-fuel storage rack design and location comply with its design bases. The safety evaluation includes postulated accidents and criticality safety assumptions. The following postulated accidents were considered:

- fuel handling accidents (e.g., dropped fuel assembly)
- uplift force on the fuel racks
- a misplaced fuel assembly

The following design features will be used to minimize the possibility of these accidents:

- The cask handling crane (capable of carrying loads heavier than the fuel components) is prevented, by design, from carrying loads over the fuel storage area.

- The racks are designed for SSE conditions.

- The racks are designed for dropped fuel assembly (and handling tool) conditions.

- The fuel-handling machine is designed to seismic Category I requirements.

The staff based its review of Section 9.1.2 of the SSAR on the guidance and acceptance criteria in Section 9.1.2 of the SRP. The staff directed its evaluation at determining whether or not the spent fuel storage facility complies with the requirements of GDC 2, 4, 5, 61, 62, and 63. Acceptability for meeting these criteria is on the basis of conformance to Positions C.1, C.3, and C.4 of RG 1.13; applicable portions of RG 1.29 and RG 1.117; and the appropriate paragraphs of ANS 57.2. On the basis of its review, the staff concludes that:

- Heavy loads are prevented, by design, from being lifted over the spent fuel pool. In addition, the fuel racks are designed to withstand a load drop equivalent to that from a fuel assembly and its associated handling tool when dropped from its operating height.

- The spent fuel storage racks are in the spent fuel storage pool, which is located within the seismic Category I auxiliary building fuel handling area. The auxiliary building is designed to maintain its structural integrity following an SSE and to perform its intended function following a postulated event such as a fire. The spent fuel pool and racks are designed to seismic Category I requirements. See Section 3.8.4.4.1 of this report.

- The spent fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. This portion of the auxiliary building is served by the radiologically controlled area ventilation system (VAS). The VAS consists of a fuel handling area ventilation subsystem and an auxiliary/annex building ventilation subsystem (see Section 9.4.3 of the SSAR). As stated in Table 3.2-3 of the SSAR, the VAS is non-seismic. The VAS serves no safety-related function. The staff's review of the VAS is discussed in Section 9.4 of this report.

The staff reviewed the spent fuel storage facility for compliance with GDC 2, 4, 5, 61 and 63. However, on the basis of its review of the information provided in Section 9.1.2 of the SSAR, the staff concluded that additional information was required before it could reach a final conclusion regarding the acceptability of the spent fuel storage facility. This was Open Item 9.1.2-1 in the DSER.

After the AP600 DSER was issued, Westinghouse provided additional information about the spent fuel storage facility in the form of SSAR revisions, RAI responses, and telephone conferences that allowed the staff to complete its review as discussed below. Therefore, Open Item 9.1.2-1 is closed. As described above, the staff reached the following conclusions. The staff found that SSAR Section 9.1.2 (regarding spent fuel storage) is in compliance with GDC 2 as it relates to the ability of the facility and the structures housing it to withstand the effects of natural phenomena, such as earthquakes, tornados, and hurricanes; and that it is in compliance with GDC 4, as it relates to the ability of the facility and the structures housing it to withstand the effects of external missiles, pipe whip, jet impingement forces, and adverse environmental conditions associated with pipe breaks, such that safety functions will not be impaired. SSAR Section 9.1.1.2.1.E states and the staff agrees that internally-generated missiles are of no concern because the fuel handling area does not contain any credible sources of internally-generated missiles. SSAR Section 9.1.2 is in compliance with the intent of GDC 5 as it relates to whether shared SSCs important to safety are capable of performing required safety functions. The staff found SSAR Section 9.1.2 in compliance with GDC 61 as it relates to the facility design for fuel storage and handling radioactive materials, and GDC 62 as it relates to the prevention of criticality. In addition, SSAR Section 9.1.2 is in compliance with GDC 63 as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions.

9.1.3 Spent Fuel Pool Cooling and Pool Purification

The staff has reviewed the spent fuel pool cooling and purification system (SFPCPS) in accordance with Section 9.1.3 of the SRP. The staff's acceptance of the SFPCPS design is on the basis of design compliance with the following requirements:

- GDC 2, as it relates to the ability of system and the structures housing it to withstand the effects of natural phenomena, such as earthquakes, tornados, and hurricanes
- GDC 4, as it relates to the ability of the system and the structures housing it to withstand the effects of external missiles
- GDC 5, as it relates to whether shared SSCs important to safety are capable of performing required safety functions
- GDC 44, as it relates to the following:
  - the system's ability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions
  - suitable redundancy of components so that safety functions can be performed assuming a single active failure of a component coincident with a loss of offsite power (LOOP) event
  - the system's ability to isolate components, systems, or piping so that the system's safety function will not be compromised

- GDC 45, as it relates to allowing periodic inspection of safety-related components and equipment

- GDC 46, as it relates to allowing operational functional testing of safety-related systems or components to ensure structural integrity and system leak tightness, operability, and adequate performance of active system components, as well as the capability of the integrated system to perform the required functions during normal, shutdown, and accident conditions

- GDC 61, as it relates to the following system design criteria for fuel storage and handling of radioactive materials:

  - capability for periodic testing of components important to safety

  - provisions for containment

  - provisions for decay heat removal

  - capability to prevent reduction in fuel storage coolant inventory under accident conditions in accordance with Position C.6 of RG 1.13

  - capability and capacity to remove fission products, radioactive materials, and impurities from the pool water and reduce occupational exposures

- GDC 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, detect excessive radiation levels, and initiate appropriate safety actions

- 10 CFR 20.1101(b), as it relates to radiation doses being kept as low as reasonably achievable (ALARA)

Compliance with the requirements of GDC 2 is founded on adherence to the guidance of Positions C.1, C.2, C.6, and C.8 of RG 1.13, as well as Position C.1 (safety-related portions of the system) and Position C.2 (non-safety-related portions of the system) of RG 1.29. Compliance with the requirements of GDC 4 is founded on adherence to the guidance of Position C.2 of RG 1.13. Compliance with the requirements of GDC 44 is founded on adherence to the recommendations of Branch Technical Position (BTP) ASB-9-2 for calculating the heat loads, the assumptions set forth in Item 1.h of Subsection III of Section 9.1.3 of the SRP, and the pool temperature limitations identified in Item 1.d of Subsection III of Section 9.1.3 of the SRP. Compliance with the requirements of 10 CFR 20.1101(b) is founded on adherence to the guidance of Positions C.2(f)(2) and C.2.f(3) of RG 8.8.

In Section 9.1.3.2 of the SSAR, Westinghouse states that the spent fuel pool cooling system is a non-safety-related system. The safety-related function of cooling and shielding the fuel in the spent fuel pool is performed by the water in the pool. A simplified sketch of the system is provided as Figure 9.1-5 of the SSAR.

Westinghouse states that the spent fuel pool cooling system consists of two mechanical trains of equipment. Each train consists of one spent fuel pool pump, one spent fuel pool heat exchanger, one spent fuel pool demineralizer, and one spent fuel pool filter. The two trains of equipment share common suction and discharge headers. In addition, the spent fuel pool cooling system is comprised of piping, valves, and instrumentation necessary for system operation. Either train of equipment can be operated to perform any of the functions required of the spent fuel pool cooling system independently of the other train. One train is continuously cooling and purifying the spent fuel pool while the other train is available for water transfers or in-containment refueling water storage tank (IRWST) purification, or is aligned as a backup to the operating train of equipment.

Both trains are designed to process spent fuel pool water. Each pump takes suction from the common suction header and discharges directly to its respective heat exchanger. The outlet piping branches into parallel lines. The purification branch is designed to process one-third of the cooling flow while the bypass branch passes the remaining two-thirds. Each purification branch is routed directly to a spent fuel pool demineralizer. The outlet of the demineralizer is routed to a spent fuel pool filter. The outlet of the filter is then connected to the bypass branch, which forms a common line that connects to the discharge header.

The staff finds the spent fuel pool cooling acceptable to demonstrate compliance with 10 CFR 20.1101(b) as it relates to the design of the fuel pool cooling system purification capability to minimize the occupational radiation exposure, and thereby keep radiation doses as low as reasonably achievable.

The spent fuel pool cooling system suction header is connected to the spent fuel pool at two locations. The main suction line connects to the spent fuel pool at an elevation 0.6 m (2 ft) below the normal water level of the pool. Two skimmer connections take suction from the water surface of the spent fuel pool. This suction arrangement prevents the spent fuel pool from inadvertently being drained below a level that would prevent the water in the spent fuel pool from performing its safety functions. This arrangement also eliminates the need for a separate skimmer circuit arrangement.

The spent fuel pool pump suction header is connected to the IRWST and the refueling cavity. This enables purification of the IRWST or the refueling cavity, and allows for the transfer of water between the IRWST and the refueling cavity. The spent fuel pool pump suction header is also connected to the fuel transfer canal and the cask loading pit. These connections are provided primarily for transferring water from the fuel transfer canal to the cask loading pit. Water that is normally stored in the fuel transfer canal can be sent to the cask loading pit and from the cask loading pit back to the transfer canal.

The spent fuel pool is initially filled for use with water having a boron concentration of approximately 2500 ppm. Demineralized water can be added for makeup purposes, including replacement of evaporative losses, from the demineralized water transfer and storage system. Boron may be added to the spent fuel pool from the chemical and volume control system (CVS).

The spent fuel pool water may be separated from the water in the transfer canal by a gate. The gate enables the transfer canal to be drained to permit maintenance of the fuel transfer equipment. The water in the transfer canal may be transferred to the cask loading pit by the spent fuel pool cooling pumps. The water may then be returned directly to the transfer canal by the spent fuel pool cooling pumps, when required.

On the basis of its review of the above information and other information provided in Section 9.1.3 of the SSAR, the staff concluded in the DSER that additional information was required before it could reach a final conclusion regarding the acceptability of the spent fuel pool cooling and pool purification system. This was DSER Open Item 9.1.3-1. However, Westinghouse provided additional information in the form of revisions to the SSAR, meetings with the NRC staff, and responses to RAIs that allowed the staff to complete its review as discussed below. Therefore, DSER Open Item 9.1.3-1 is closed.

The staff reviewed the spent fuel pool cooling system for compliance with the requirements of GDC 2, 4, 5, 44, 45, 46, 61, 63 and 10 CFR 20.1101(b), as referenced in Section 9.1.3 of the SRP. The staff found that the AP600 spent fuel pool cooling system is not a safety-related system and is not required to operate following events such as earthquakes, fires, passive failures, or multiple active failures. The spent fuel pool cooling system has the safety-related functions of containment isolation and providing safety-related connections for temporary emergency makeup to the spent fuel pool for cooling. Spent fuel pool makeup for a long term station blackout can be provided through seismically qualified safety-related makeup connections from the passive containment cooling system. These connections are located in an area of the auxiliary building that can be accessed without exposing operating personnel to excessive levels of radiation or adverse environmental conditions during boiling of the pool. The spent fuel pool is designed such that, using only safety-related makeup, water is maintained above the spent fuel assemblies for at least seven days following a loss of the spent fuel pool cooling system. In accordance with the design, the minimum water level to achieve sufficient cooling is the sub-cooled, collapsed level (without vapor voids) required to cover the top of the fuel assemblies. Therefore, the applicable portion of the requirements of GDC 2 are that the structure housing the system must have the ability to withstand the effects of natural phenomena, such as earthquakes, tornados, and hurricanes. In this regard, the spent fuel pool is located in a seismic Category I building in the fuel handling area. Therefore, the system is protects from external missiles and thereby complies with the requirements of GDC 2. It is also in compliance with the applicable portions of the following requirements:

- GDC 4, as it relates to the ability of the structure housing the system to withstand the effects of external missiles.

- GDC 5, as it relates to whether shared SSCs important to safety are capable of performing required safety functions. Compliance with GDC 5 is discussed in Section 9.2 of this report.

- GDC 44, as it relates to the system's ability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions. There are no safety-related SSCs involved in the spent fuel pool system under normal operating; however, during accident conditions, the spent fuel pool is designed to cool by boiling and transferring the heat to the atmosphere.

- GDC 45, as it relates to allowing periodic inspection of safety-related components and equipment. The spent pool cooling system is not a safety-related system; however, SSAR Section 9.1.3 states that periodic visual inspections and preventive maintenance will be performed.

- GDC 46, as it relates to the capability of the system to perform required functions during normal, shutdown, and accident situations. As discussed above the required functions are containment isolation and providing safety-related connections for temporary emergency makeup for spent fuel pool cooling. These capabilities have been acceptably demonstrated.

- GDC 61, as it relates to provisions for decay heat removal; the capability to prevent reduction in fuel storage coolant inventory under accident conditions; and the capability and capacity to remove fission products, radioactive materials, and impurities from the pool water and reduce occupational exposures. Under accident conditions the system is designed to provide safety-related makeup for seven days. In addition, the purification system is found acceptable to remove fission products and radioactive materials from the pool water thereby reducing occupational exposures.

- GDC 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, detect excessive radiation levels, and initiate appropriate safety actions. The AP600 design provides acceptable instrumentation to measure temperature, pressure, flow and level in the spent fuel pool. The design also limits exposure rates at the surface of the refueling cavity to less than 2.5 mrem/hr. In addition, the tritium concentration is maintained at an acceptable level.

- 10 CFR 20.110(b), as it relates to the design of the fuel pool cooling system purification capability to minimize the occupational radiation exposure, and thereby keep radiation doses as low as reasonably achievable. The staff finds the spent fuel pool cooling system provides a purification and filtration system design that will minimize the occupational radiation exposure, and thereby keep radiation doses as low as reasonably achievable.

On the basis of the above discussion, the staff concludes that the additional information provided by Westinghouse regarding the design of the spent fuel pool cooling and purification system is acceptable. The staff has noted that Figure 9.1-6, sheets 1 of 2, has identified notes 8, 9, and 14; however, the corresponding notes do not exist. Westinghouse has agreed to correct this problem. This was Confirmatory Item 9.1-1. SSAR Revision 23, dated May 18, 1998, corrected Figure 9.1-6 by adding notes 8, 9, and 14; therefore, Confirmatory Item 9.1-1 is closed.

9.1.4 Light Load Handling System (Related To Refueling)

The staff reviewed the light load handling system (LLHS) in accordance with Section 9.1.4 of the SRP. Staff acceptance of the design of the system is contingent on design compliance with the following requirements:

- GDC 2, as it relates to the ability of SSCs to withstand the effects of earthquakes

- GDC 5, as it relates to whether shared SSCs important to safety are capable of performing required safety functions

- GDC 61, as it relates to a radioactivity release as a result of fuel damage and the avoidance of excessive personnel radiation exposure

- GDC 62, as it relates to criticality accidents

Compliance with the requirements of GDC 2 is on the basis of adherence to the guidance of Positions C.1 and C.6 of RG 1.13, as well as Positions C.1 and C.2 of RG 1.29. Compliance with the requirements of GDC 61 is on the basis of adherence to the guidance of Position C.3 of RG 1.13, as well as ANS 57.1/ANSI-N208. Compliance with the requirements of GDC 62 is on the basis of adherence to the guidance of Position C.3 of RG 1.13, as well as ANS 57.1/ANSI N208.

In Section 9.1.4.2 of the SSAR, Westinghouse states that the LLHS consists of the equipment and structures needed for the refueling operation. This equipment is comprised of fuel assemblies, core component and reactor component hoisting equipment, handling equipment, and a dual-basket fuel transfer system. The following structures are associated with the fuel handling equipment:

- refueling cavity
- transfer canal
- fuel transfer tube
- spent fuel pit
- cask loading area
- new fuel storage area
- new fuel receiving and inspection area

The fuel handling equipment is designed to handle the spent fuel assemblies underwater from the time they leave the reactor vessel until they are placed in a container for shipment from the site. As described below, underwater transfer of spent fuel assemblies provides an effective and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. The boric acid concentration in the water is sufficient to preclude criticality.

The associated fuel handling structures may be generally divided into two areas:

(1) the refueling cavity, which is flooded only during plant shutdown for refueling
(2) the spent fuel pool and transfer canal, which are kept full of water

The refueling cavity and fuel storage area are connected by the fuel transfer tube, which is fitted with a quick-opening hatch on the canal end and a valve on the fuel storage area end. The hatch is in place, except during refueling, to provide containment integrity. Fuel is carried through the tube on an underwater transfer car.

Fuel is moved between the reactor vessel and the fuel transfer system by the non-seismic refueling machine. The fuel transfer system is used to move up to two fuel assemblies at a time between the containment building and the auxiliary building fuel handling area. After a fuel assembly is placed in the fuel container, the lifting arm pivots the fuel assembly to the horizontal position for passage through the seismic Category I fuel transfer tube. After the transfer car transports the fuel assembly through the transfer tube, the lifting arm at that end of the tube pivots the assembly to a vertical position so that the assembly can be lifted out of the fuel container.

In the fuel handling area, fuel assemblies are moved about by the seismic Category I fuel handling machine. Initially, a short tool is used to handle new fuel assemblies, but the non-seismic new fuel elevator must be used to lower the assembly to a depth at which the fuel handling machine can place the new fuel assemblies into or out of the spent fuel storage racks.

New fuel assemblies received for refueling are removed one at a time from the shipping container and moved into the new fuel assembly inspection area using the seismic Category II new fuel jib crane.

The transfer car controls for the fuel transfer system are located in the fuel handling area. Therefore, conditions in the containment are not visible to the operator. The transfer car permissive switch allows the fuel transfer system containment operator to exercise some control over car movement if conditions visible to the operator warrant such control.

An interlock on the fuel transfer system prevents the upender from being moved from the horizontal to the vertical position if the transfer car has not reached the end of its travel. An interlock on the transfer tube valve permits transfer car operation only when the transfer tube valve position switch indicates that the valve is fully open.

The fuel transfer system is also interlocked with the refueling machine. Whenever the transfer car is located in the refueling cavity, the fuel transfer system cannot be operated unless the refueling machine mast is in the fully retracted position, the refueling machine is over the core, or the gripper is released and inside the core.

On the spent fuel pool side, the fuel transfer system is interlocked with the fuel handling machine. The fuel transfer system cannot be operated until the fuel handling machine is moved away from the fuel transfer system area.

Fuel handling tools and equipment handled over an open reactor vessel are designed to prevent inadvertent decoupling from machine hooks. In addition, lifting rigs are pinned to the machine hook, and safety latches are provided on hook supporting tools. Tools required for handling internal reactor components are designed with the following fail-safe features that prevent disengagement of the component in the event of operating mechanism malfunction:

- The air cylinders actuating the gripper mechanism for the non-seismic control rod drive shaft unlatching tool are equipped with backup springs that close the gripper in the event of loss of air to the cylinder. Air-operated valves are equipped with safety locking rings to prevent inadvertent actuation.

- When the fingers for the non-seismic new fuel assembly handling tool are latched, the actuating handle is positively locked, preventing inadvertent actuation. The tool is preoperationally tested at 125 percent of the weight of one fuel assembly.

During spent fuel transfer, the gamma dose rate at the surface of the water is 20 mrem/hour or less. This is accomplished by maintaining a minimum of 3 m (10 ft) of water above the top of the active fuel height during handling operations. The three fuel handling devices used to lift spent fuel assemblies are the refueling machine, the fuel handling machine, and the spent fuel handling tool. Both the refueling machine and fuel handling machine contain positive stops that prevent the fuel assembly from being raised above a safe shielding height.

On the basis of its review of the above information and the information provided in Section 9.1.4 of the SSAR, the staff concluded that additional information was required before it could reach a final conclusion regarding the acceptability of the LLHS. This was Open Item 9.1.4-1 in the DSER. However, Westinghouse provided additional information in the form of meetings, telephone conversations, and revisions to the SSAR, which allowed the staff to complete its review of SSAR Section 9.1.4, as discussed in the above evaluation and found it acceptable. Therefore, DSER Open Item 9.1.4-1 is closed.

The staff found that the LLHS for the AP600 design is in compliance with GDC 2, as it relates to the ability of SSCs to withstand the effects of natural phenomena. It is in compliance with GDC, 5 as it relates to whether shared SSCs important to safety are capable of performing required safety functions, per Section 3.1.1 of the SSAR, which states that "The AP600 is a single-unit plant. If more that one unit were built on the same site, none of the safety-related systems would be shared." The LLHS is also in compliance with the intent of GDCs 61 and 62, as related to a radioactivity release as a result of fuel damage and the avoidance of excessive personnel radiation exposure, and criticality accidents respectively.

9.1.5 Overhead Heavy Load Handling Systems

The staff's acceptance of the design of a heavy load handling system (HLHS) is contingent on compliance with the following requirements:

- GDC 2, as it relates to the ability of SSCs to withstand the effects of natural phenomena such as earthquakes
- GDC 4, as it relates to the protection of safety-related equipment form the effects of internally-generated missiles (i.e., dropped loads)
- GDC 5, as it relates to whether shared SSCs important to safety are capable of performing their required safety functions
- GDC 61, as it relates to the safe handling and storage of fuel

Compliance with the requirements of GDC 2 is on the basis of adherence to the guidance of Positions C.1 and C.6 of RG 1.13, as well as Positions C.1 and C.2 of RG 1.29. Compliance with the requirements of GDC 4 is on the basis of adherence to the guidance of Positions C.3 and C.5 of RG 1.13. Other guidelines used in the evaluation of this system include NUREG-0612, ANS 57.1/ANSI N208, and ANS 57.2/ANSI N210.

On the basis of its review of the information originally provided in Section 9.1.5 of the SSAR, the staff concluded that additional information was required before it could reach a final conclusion regarding the acceptability of the overhead HLHSs, including compliance with GDC 2, 4, 5, and 61. This was identified as DSER Open Item 9.1.5-1. Subsequent to the issuance of the DSER, Westinghouse provided additional information in the form of meeting, telephone conversations, and revisions to the SSAR that addressed DSER Open Item 9.1.5-1, as discussed below.

For the AP600 design, Westinghouse defines a heavy load to be a load that weighs more than the combined weight [about 1270 kg (2800 lbs)] of a fuel assembly with a rod cluster control, and the associated handling device (consisting of the inner mast of the fuel handling machine and the fuel gripper assembly). This equipment is part of the mechanical handling system (MHS) and is located throughout the plant. HLHSs are generally classified as non-safety-related, non-seismic systems. The components of single-failure-proof systems necessary to prevent uncontrolled lowering of a critical load are classified as safety-related.

The containment polar crane and the equipment hatch hoist system are single-failure-proof systems. They are classified as seismic Category I, and are designed to support a critical load during and after an SSE (see Section 3.7.2 of this report). A critical load is a heavy load that, if dropped, could cause unacceptable damage to reactor fuel elements, or a loss of safe shutdown or decay heat removal capability.

For the AP600 design, the plant arrangement and the design of HLHSs are predicated on the following criteria:

- To the extent practicable, heavy loads are not carried over or near safety-related components, including irradiated fuel and safe shutdown components. Safe load paths are designed for heavy load handling in safety-related areas.

- The likelihood of a load drop is extremely small (that is, the handling system is single-failure-proof), or the consequences of a postulated load drop are within acceptable limits.

- Single-failure-proof systems can stop and hold a critical load following the credible failure of a single component.

- Single-failure-proof systems can support a critical load during and after an SSE.

Except for the containment polar crane and the equipment hatch hoist system, the HLHSs are not single-failure-proof. The SSAR states that overhead cranes are designed according to ASME NOG-1. Other cranes and hoists handling heavy loads are designed according to the applicable ANSI standard.

In Section 9.1.5.3 of the SSAR, Westinghouse states that, for the polar crane and the equipment hatch hoist crane systems, redundancy is provided for load bearing components such as hoisting ropes, sheaves, equalizer assembly, hooks, and holding brakes. These systems are designed to support a critical load during and after an SSE.

The spent fuel shipping cask storage pit is separated from the spent fuel pool. The spent fuel shipping cask crane cannot move over the spent fuel pool because the crane rails do not extend over the pool. Mechanical stops prevent the spent fuel shipping cask crane from going beyond the ends of the rails.

In Section 9.1.5.3 of the SSAR, Westinghouse also states that a heavy load analysis is performed to evaluate postulated load drops from HLHSs located in safety-related areas of the plant, specifically the nuclear island. Westinghouse further states that no evaluations are required for critical loads handled by the single-failure-proof containment polar crane or equipment hatch hoist, because a load drop is unlikely. The heavy load analysis is meant to confirm that a postulated load drop does not cause unacceptable damage to reactor fuel elements, or a loss of safe shutdown or decay heat removal capability.

The staff has completed its review of the additional information as discussed in the above evaluation and found it acceptable. Therefore, DSER Open Item 9.1.5-1 is closed.

As described above, the staff concludes that the design of the AP600 HLHSs is in compliance with the requirements of GDC 2, as it relates to the ability of SSCs to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods. It is also in compliance with GDC 4, as it relates to protection of safety-related equipment from the effects of internally-generated missiles, because in Section 9.1.1.2.1.E of the SSAR, Westinghouse states that the fuel handling area does not contain any credible sources of internally-generated missiles. The design of the HLHSs is also in compliance with GDCs 5 and 61, relating to whether shared SSCs important to safety are capable of performing required safety functions, and the safe handling and storage of fuel, respectively.

9.2 Water Systems

The staff's review of the AP600 water systems is provided in the following sections: 9.2.1, Service Water System; 9.2.2, Component Cooling Water System; 9.2.3, Demineralized Water Treatment System; 9.2.4, Demineralized Water Transfer and Storage System; 9.2.5, Potable Water System; 9.2.6, Sanitary Drainage System; 9.2.7, Central Chilled Water System; 9.2.8, Turbine Building Closed Cooling System; 9.2.9, Waste Water System; and 9.2.10, Hot Water Heating System.

The AP600 design can be used at either single-unit or multiple-unit sites. Nonetheless, in Section 3.1.1 of the SSAR, Westinghouse states that the AP600 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. Should a multiple-unit site be proposed, the COL applicant referencing the AP600 design will be required to apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs to perform their required safety functions.

9.2.1 Service Water System

The staff reviewed the design of the service water system (SWS) in accordance with SRP Section 9.2.1, "Station Service Water System." However, the SWS for the AP600 differs from that of the previous pressurized-water reactor (PWR) designs in that the AP600 SWS is completely non-safety-related. In previous SWS designs, portions of the system were required to perform safety-related functions. The reason that the AP600 SWS is non-safety-related is that the SWS removes heat only from the component cooling water system (CCS), which is not a safety-related system. The staff's evaluation of the CCS being non-safety-related is provided in Section 9.2.2 of this report. Therefore, the portions of Section 9.2.1 of the SRP that apply to safety-related systems are not applicable for the AP600 SWS. As for the non-safety-related SWS meeting the requirements of GDC 2, as it relates to structures and systems being capable of withstanding the effects of natural phenomena, acceptance is predicated on meeting the guidance of the portions of RG, 1.29 Position C.2 applicable to non-safety-related systems.

In addition, Westinghouse indicated in the response to RAI Q100.11 (dated May 11, 1994) that the SWS system is a non-safety-related system performing a defense-in-depth (DID) function. The staff reviewed the SWS according to the review guidance described in the October 24, 1994, staff position for evaluating non-safety-related systems that have been identified to be important, by the process concerning the regulatory treatment of non-safety-related systems (RTNSS).

In DSER Open Item 9.2.1-1, the staff identified that Westinghouse must resolve RAIs Q410.108-Q410.115. Westinghouse provided responses and additional information in a meeting with the staff on February 22-23, 1995, that allowed the staff to continue its review as discussed below.

The AP600 SWS is described in SSAR Section 9.2.1 and shown in Figure 9.2.1-1. The SWS supplies cooling water to remove heat from the CCS heat exchangers, which are located in the turbine building. The system consists of two trains of components and piping. Each train provides 100-percent capacity cooling for normal power operation. Each train includes one service water pump, one component cooling heat exchanger, one strainer, and one cooling tower cell. Cross connections between the trains upstream and downstream of the heat exchanger allow either service water pump to supply either heat exchanger, and allow either heat exchanger to discharge to either cooling tower. Flooding of the turbine building resulting from a service water failure is less severe than that from the circulating water system, which is discussed in Section 10.4.5 of this report.

The service water pumps are centrifugal pumps driven by electric motors; each pump has a design flow rate of 23 m^3/min (6200 gpm). These pumps take suction from the service water pump basin through fixed screens to the pump suction piping. The service water pumps discharge through strainers to the component cooling water heat exchangers. The heated water from the heat exchangers is passed through the discharge piping to the mechanical draft cooling tower, where the system heat is rejected. The cool water, collected in the tower basin, provides the source for the suction of service water pumps.

The SWS operates during startup, normal plant operation, normal plant cooldown, and refueling, and is available following a LOOP event. Under normal plant operation, one of two service water system trains removes the heat from one of the two component cooling water heat exchangers and discharges it to the cooling tower. The standby train is automatically started on combined low-flow and low-pressure values when the operating train fails. During accident conditions, the SWS remains in the same operating modes as for normal operations. During plant startup, shutdown, and refueling, two service water trains are used.

In Section 9.2.1.1.1 of the SSAR, Westinghouse states that the SWS serves no safety-related function. Failure of the SWS or its components will not affect the ability of any other safety-related systems to perform their intended safety functions. Postulated breaks in the SWS piping will not impact safety-related components because the SWS is not located in the vicinity of any safety-related equipment, and the water from the break will not reach any safety-related equipment. Therefore, the staff finds that the SWS complies with GDC 2 by meeting the guidance of Position C.2 of RG 1.29 for ensuring that the non-safety-related SWS could withstand the effects of earthquakes without affecting safety-related systems.

In Section 8.5 of the SER concerning the Electric Power Research Institute's (EPRI's) Advanced Light-Water Reactor (ALWR) Utility Requirements Document (URD) for passive plants, the staff identified the SWS to be one of the systems that require further evaluation for the appropriate regulatory treatment of non-safety-related systems. In WCAP-13856, entitled "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," Westinghouse indicates that the SWS provides DID functions during shutdown, when at reduced inventory operations. During shutdown operation, the SWS serves the CCS, and the CCS provides cooling water for the spent fuel pool cooling and normal residual heat removal system (RNS), which is required for decay heat removal. In Table 3.2-3 of the SSAR, the cooling tower, cooling tower fans, pumps, and applicable valves of the SWS are classified as AP600 Class D, Seismic Category NS (non-seismic).

The staff initially reviewed the design of the SWS in accordance with the review guidance described in the October 24, 1994, staff position for evaluating non-safety-related systems that have been identified to be important by the RTNSS process. The specific review items, which were identified by the staff in RAI Q410.109. Westinghouse provided the following information in response to the RAI.

- Redundancy: Appropriate redundancy (pumps, valves, piping, heat exchangers, controls, and instrumentation) is provided such that the SWS can support normal operation and its DID function, assuming a single active component failure.

- Power Supply: The power supplies for the SWS pumps and associated active components are independent, permanent, non-safety-related electrical buses. Each bus is capable of being supplied from one of two onsite standby diesel generators. During a LOOP event, the ac power needed for the operating pump and motor strainer is supplied by one of two standby diesel generators that is started automatically on a signal from the integrated protection and control system. No separation is provided for the 480 volt load centers or the supply cables for the system electrical loads.

- Environmental Qualification: The design of the SWS does not ensure functional operability, maintenance access, or support plant recovery following design-basis events.

- Internal Hazards: Protection from internal hazards such as internal flooding, pipe ruptures, jet impingement, fires, and missiles is not provided.

- Natural Phenomena: The SWS is not protected from natural phenomena such as a seismic event. It is designed to seismic category NS.

- Quality Assurance Program: The SWS is classified as an AP600 Class D system. Standard industrial quality assurance standards are applied to provide appropriate integrity and function, although 10 CFR Part 50, Appendix B, and 10 CFR Part 21 do not apply.

- Reliability Assurance and Maintenance Programs: The SWS is included in the plant reliability assurance and maintenance programs. In response to RAI Q410.108 regarding testing and inspection, Westinghouse indicated that the SWS provides no safety-related function and does not require any testing or inspection plan. The performance of system components is demonstrated by the operation of the system and periodically switching between two trains. The reliability and maintenance plans for the DID systems include provisions to check for operability, including appropriate testing and inspection, and to repair out-of-service components. These provisions are documented and administered in the plant reliability assurance plan and operating and maintenance procedures. The probabilistic risk assessment (PRA) assumption determines the test frequency.

- Technical Specification Control: The SWS does not have technical specification controls on limiting conditions for operation. In a letter dated July 16, 1997, Westinghouse proposed investment protection short-term availability controls for the RNS, CCS, and SWS, which have required actions and associated completion time, and surveillance requirements during Modes 5 and 6. These short-term availability controls are shown in SSAR Table 16.3.2.

- Administrative Controls for Shutdown Configurations: In fulfilling the SWS's DID function, Westinghouse proposes the following recommendations in WCAP-13856 for regulatory oversight:

  - Both subsystems of the SWS should be available to supply cooling flow to the component cooling water heat exchangers during reduced reactor coolant system inventory operations.

  - Maintenance affecting the SWS components required to support shutdown decay heat removal operation should normally be scheduled during Mode 1, 2, or 3 power operation.

If the preceding recommendations are not met, the plant should not initiate reduced reactor coolant inventory operations. If the conditions cannot be maintained during reduced reactor coolant system inventory operations, the plant should take action to restore system conditions or to leave these conditions.

Westinghouse's response to Q410.109 indicates that the SWS performs no safety-related functions and need not meet the criteria listed in RAI Q410.109. In addition, Westinghouse used a focused PRA to determine risk-significant non-safety-related SSCs, their reliability/availability (R/A), and level of regulatory oversight for the identified systems. The staff's evaluation of this approach is presented in Chapter 22 of this report. As a result of this review, Westinghouse provided additional short-term availability controls for the SWS and CCS during Modes 5 and 6 with reactor coolant system reduced inventory as shown in SSAR Table 16.3-2. On the basis of the above review, the staff concludes that the design of the SWS is acceptable.

Measures to preclude long-term corrosion and organic fouling for the SWS are provided by water treatment using the turbine island chemical feed system. The equipment injects the required chemicals into the SWS, maintaining a non-corrosive condition and limiting biological film formation. SSAR Section 9.2.1.2.2 discusses the turbine island chemical feed system and describes the measures taken to control long-term corrosion.

The maximum ambient air wet bulb temperature specified in Chapter 2 of the SSAR for site interface parameters is 27 °C (81 °F). This maximum wet bulb temperature is consistent with the EPRI ALWR URD interface requirements, and should apply to most U.S. plant sites. Actual site-specific data will dictate design parameters of the cooling tower, and lower approach temperatures may be used by a COL applicant. The CCS is cooled by the SWS using plate-type heat exchangers, which afford the plant designer the flexibility to use a lower approach temperature if required. Specific site conditions that exceed the 27 °C (81 °F) wet bulb temperature may be accommodated by specific site analysis to adjust cooling system capability. No degradation of safety systems will result.

Compliance with GDC 5 is discussed in Section 9.2 of this report.

As described above, the staff has reviewed the SWS in accordance with Section 9.2.1 of the SRP. Because the AP600 SWS is not safety-related, and its failure does not lead to the failure of any safety systems, the requirements of GDCs 4, 44, 45, and 46 and the guidance of Section 9.2.1 of the SRP do not apply.

RAI Q410.107 identified discrepancies in the SSAR and PRA documents. In response to Q410.107, Westinghouse committed to revise SSAR Sections 14.2.8.1.4 and 14.2.8.1.5, and the AP600 PRA to reflect that the SWS only provides cooling water to the CCW, and not to the turbine building closed cooling water system. The staff reviewed the revised SSAR and PRA and found that the previous discrepancies were removed.

RAI Q410.110 raised a concern of radioactive leakage into and out of the SWS. In response to Q410.110, Westinghouse stated that the SWS has a radiation monitor to monitor effluent, and has the provision for taking samples both upstream and downstream of the component cooling water heat exchangers. If radioactive fluid is detected in the SWS, tower blowdown flow can be isolated by remote manual control.

RAIs Q410.114 and Q410.115 raised the concerns of net positive suction head (NPSH) and the potential for water hammer, respectively. In response to Q410.114, Westinghouse indicated that the design requirements for the SWS specify that the minimum NPSH available at the SWS pumps exceed the required NPSH by either 25 percent or 10 feet, whichever is less. In response to Q410.115, Westinghouse revised SSAR Section 9.2.1.2.1 to state that temperatures in the system are moderate and that the pressure of the system is kept above saturation at all locations. The system pressure and temperature relation, and other design features of the system arrangement and control of valves, minimize the potential for thermodynamic or transient water hammer. The staff found Westinghouse's responses acceptable.

On the basis of the above review, the staff finds the SWS design is acceptable and meets the applicable provisions described in SRP Section 9.2.1, as well as DID systems short-term availability control. Therefore, DSER Open Item 9.2.1-1 is closed.

9.2.2 Component Cooling Water System

The staff reviewed the design of the component cooling water system (CCS) in accordance with Section 9.2.2, "Reactor Auxiliary Cooling Water Systems," of the SRP and against the review criteria described in the October 1994, staff position for evaluating non-safety-related systems that have been identified to be important by the RTNSS process.

DSER Open Item 9.2.2-1 identified that RAIs Q410.118, Q410.121-Q410.124 needed to be resolved. Westinghouse provided responses and additional information in a meeting with the staff on February 22-23, 1995, which allowed the staff to complete its review as discussed below.

The CCS is described in SSAR Section 9.2.2 and shown in SSAR Figures 9.2.2-1 and 9.2.2-2. The CCS is a closed-loop cooling system that transfers heat from various plant components to the SWS during normal phases of operation, including power operation, normal cooldown, and refueling. The plant components cooled by this system are as follows:

- reactor coolant pumps
- letdown heat exchanger
- reactor coolant drain tank (RCDT) heat exchanger
- RNS heat exchangers
- RNS pumps
- spent fuel pool heat exchangers
- chillers
- sample heat exchangers
- miniflow heat exchangers
- air compressors
- cooler
- condensate pumps

The CCS consists of two trains and one component cooling water surge tank. Each train consists of one component cooling water pump and one component cooling water heat exchanger, as well as associated valves, piping, and instrumentation. The component cooling water surge tank, which accommodates thermal expansion and contraction, is connected to a shared portion of the return header. The two trains of equipment take suction from a single return header. The discharge of each heat exchanger is routed directly to the common supply header. Component cooling water is distributed to the components by this single supply/return header. Loads inside containment are automatically isolated in response to a safety injection signal, which trips the reactor coolant pumps. Individual components, except the reactor coolant pumps, can be isolated locally to permit maintenance, while supplying the remaining components with cooling water.

The two component cooling water pumps are horizontal, centrifugal pumps. Each pump has a design flow rate of 22.7 m^3/min (6000 gpm). The pumps are redundant for normal operation heat loads. Both pumps are required for the design basis cooldown; however, an extended cooldown can be achieved with only one pump in operation. Each pump can be aligned to either heat exchanger. The component cooling water heat exchangers are plate heat exchangers made of austenitic stainless steel. Component cooling water circulates through one side of the heat exchanger, while service water circulates through the other side.

Component cooling water in the heat exchanger is maintained at a higher pressure than the service water to prevent corrosive in-leakage from untreated service water. Most of the valves in the CCS are manual valves used to isolate cooling flow from components for which cooling is not required in a given plant operating mode. Three motor-operated isolation valves and a check valve provide containment isolation for the two CCS lines that penetrate the containment barrier. The motor-operated valves are normally open, and are closed upon receipt of a safety injection signal. The CCS components require both cooling water flow from the SWS and electrical support from the main ac power and non-Class 1E dc systems.

In Section 9.2.2.1.1 of the SSAR, Westinghouse states that the CCS serves no safety-related function, except for containment isolation. In Q210.36, the staff identified three safety components being cooled by the CCS, including reactor coolant pumps, chemical volume and control system letdown heat exchangers, and normal residual heat removal (RHR) heat exchangers and pumps. In its response to Q210.36 dated June 27, 1994, and its response to Q410.117 dated June 30, 1994, Westinghouse stated that the CCS provides cooling water to various non-safety-related components and the following three safety-related components:

- reactor coolant pumps
- chemical volume and control system letdown heat exchangers
- normal RHR heat exchangers and pumps

The CCS provides cooling water to support the normal operation of the safety-related components identified above. However, none of these safety-related components requires cooling water to perform its safety-related functions. The safety-related functions of these components are limited to maintaining primary coolant system integrity and providing reactor coolant pump coast-down capability; these functions are independent of CCS operation. The CCS is required for normal RHR heat removal operation to provide plant cooldown and core decay heat removal functions during shutdown conditions. Safety-related cooldown and decay heat removal functions are provided by the passive core cooling system and containment cooling systems. Therefore, CCS operation is not required to perform any safety-related functions. Segments of the CCS piping that penetrate the containment and the associated containment isolation valves are safety-related and perform a safety-related containment isolation function; therefore, these segments are designed to accommodate environmental and dynamic effects and satisfy GDC 4.

In Chapter 8 of NUREG-1242, Vol. 3, "NRC Review of Electric Research Institute's Advanced Light Water Reactor Utility Requirements Document - Passive Plant Designs," the staff identified the CCS to be one of the systems that required further evaluation for the appropriate regulatory treatment of non-safety systems. In WCAP-13856, Westinghouse indicates that the CCS provides DID functions during reduced inventory operations at shutdown. During shutdown operation, the CCS provides cooling water for the normal RHR system, which is required for decay heat removal. Table 3.2-3 of the SSAR classifies CCS pumps and valves as AP600 Class D, Seismic Category NS, with the exception of containment isolation valves. The containment penetration isolation valves are Safety Class B, as is the pipe between the isolation valves. The remainder of the CCS piping is designed to ANSI B31.1.

On the basis of the analysis discussed above, the staff agrees with Westinghouse that the CCS does not perform any safety-related function except for containment isolation. Therefore, portions of Section 9.2.2 of the SRP that apply to safety-related systems are not applicable for the AP600 CCS.

The staff initially reviewed the design of the CCS according to the review criteria described in the October 24, 1994, staff position for evaluating non-safety-related systems that have been identified to be important by the RTNSS process. The specific review items were identified by the staff in RAI Q410.120. Westinghouse provided the following information in response to the RAI:

- Redundancy: Appropriate redundancy is provided such that the CCS can support normal operation and its DID function assuming, a single active component failure.
- Power Supply: The power supplies for the CCS pumps and associated active components are independent, permanent, non-safety-related electrical buses. Each bus is capable of being supplied from one of two onsite standby diesel generators. The onsite standby diesel generators and permanent onsite 4160 volt buses are physically separated. No separation is provided for the 480 volt load centers or the supply cables for the system electrical loads.
- Environmental Qualification: The containment penetrations and isolation valves are designed to the requirements of environmental qualification. The design of the non-safety-related portions of the CCS does not ensure functional operability, maintenance access, or support plant recovery following design-basis events.
- Internal Hazards: Except for the safety-related containment isolation function, protection from internal hazards is not provided.
- Natural Phenomena: Except for its containment isolation function, the CCS is not protected from natural phenomena.
- Quality Assurance Program: The CCS is classified as an AP600 Class D system. Standard industrial quality assurance standards are applied to provide appropriate integrity and function. The portions of CCS that perform the containment isolation function are classified as AP600 Class B, for which 10 CFR Part 50, Appendix B applies.
- Reliability Assurance and Maintenance Programs: The CCS is included in the plant reliability assurance and maintenance programs.
- Technical Specification Control: Except for the containment isolation valves, the CCS does not have technical specification controls on limiting conditions for operations. In a letter dated July 16, 1997, Westinghouse proposed investment protection short term availability controls for the RNS, CCS, and SWS, which have required actions and associated completion time, and surveillance requirements during Modes 5 and 6. These short term availability controls are shown in Table 16.3-2 of the SSAR.

- Administrative Controls for Shutdown Configurations: In fulfilling the CCS's DID function, Westinghouse proposes the following recommendations for regulatory oversight:

    - Both pumps of the CCS should be available to supply cooling flow to the normal RHR pumps and heat exchangers during reduced reactor coolant system inventory operations.

    - Maintenance affecting the CCS components required to support shutdown decay heat removal operation should normally be scheduled during Mode 1, 2 or 3.

    If the preceding recommendations are not met, the plant should not initiate reduced reactor coolant inventory operations. If the conditions cannot be maintained during reduced reactor coolant system inventory operations, the plant should take action to restore system conditions or to leave these conditions.

Westinghouse's response to Q410.120 indicated that, except for containment isolation, the CCS performs no safety-related functions and need not meet the criteria listed in RAI 410.120. In addition, Westinghouse used a focused PRA to determine risk-significant non-safety-related SSCs, their R/A, and level of regulatory oversight for the identified systems. The staff evaluation of this approach is in Chapter 22 of this report. As a result of this review, Westinghouse provided additional short-term availability controls for SWS and CCS during Modes 5 and 6 with reactor coolant system reduced inventory, as shown in SSAR Table 16.3-2. On the basis of the above review, the staff concludes that the design of the CCS is acceptable.

In Section 9.2.2.1.1 of the SSAR, Westinghouse states that failure of the CCS or its components will not affect the ability of safety-related systems to perform their intended safety functions; this conforms to the guidance of Position C.2 of Regulatory Guide 1.29. GDC 44, 45, and 46 do not apply to the CCS because the CCS heat loads are not safety-related. Further, none of the CCS instrumentation performs a safety-related function, so the CCS instrumentation need not comply with IEEE 279.

The operating temperature of the CCS components will normally be well below 93.3 °C (200 °F), and the pressure will be maintained above atmospheric. Because the CCS will normally operate at temperatures and pressures that prevent formation of steam bubbles, water hammer issues will be avoided. The safety-related portions of the CCS (the containment penetrations and isolation valves) are subject to the analysis of postulated cracks in moderate-energy piping systems, as described in Section 3.6 of the SSAR.

Compliance with GDC 5 is discussed in Section 9.2 of this report.

Section 9.2.2 of the SSAR states that the CCS transfers heat from "various plant components" to the SWS. The components that were referred to were listed in the proprietary version of the SSAR. In order to evaluate and determine if the NRC staff agrees that the system should be non-safety-related, the specific components referenced are needed in the non-proprietary version of the SSAR. The staff addressed the above concern in RAI Q410.118. In response, Westinghouse initially indicated that no SSAR description of those components was needed. In a meeting with Westinghouse on February 22-23, 1995, the staff indicated that Westinghouse's response to Q410.118 was not acceptable because it did not include the requested SSAR change. In a subsequent SSAR revision, Westinghouse incorporated the list of components in Table 9.2.2, and therefore, resolved the above documentation issue.

On the basis of the above review, the staff finds the CCS design acceptable in meeting applicable provisions described in SRP Section 9.2.2 and DID systems requirements. Therefore, DSER Open Item 9.2.2-1 is closed.

9.2.3 Demineralized Water Treatment System

The demineralized water treatment system (DTS) receives water from the raw water system (RWS), processes this water by removing ionic impurities and dissolved gases, and provides demineralized water to the demineralized water transfer and storage system (DWS). The DTS does not perform any safety-related function or accident mitigation, and its failure would not reduce the safety of the plant.

The DTS consists of the following components:

- two 100-percent cartridge filters
- two reverse osmosis feed pumps
- two 100-percent reverse osmosis units with auxiliary cleaning system
- one electrodeionization unit for secondary demineralization and carbon dioxide removal
- two brine pumps

The staff evaluated the design and operational requirements of the DTS and concluded that it includes all components associated with the system from the source of raw water to a discharge to the DWS. On the basis of its review, the staff determined that the proposed design criteria for the DTS provided an adequate supply of reactor coolant water purity during all conditions of plant operation. Therefore, the guidelines for demineralized water given in Table 9.2.3-1 of the SSAR are acceptable. In the DSER the staff requested that Westinghouse specify maximum concentrations of halogens and sulfates. This was DSER Open Item 9.2.3-1. Westinghouse revised its SSAR, Table 9.2.3-1 to specify maximum concentrations of halogens and sulfates and, therefore, Open Item 9.2.3-1 is closed.

The staff concludes that the design of the DTS is acceptable because it conforms to the guidelines of Section 9.2.3 of the SRP which requires the system to have the capability for an adequate supply of reactor coolant purity water during all conditions of plant operation.

9.2.4 Demineralized Water Transfer and Storage System

The staff reviewed the DWS in accordance with Section 9.2.3, "Demineralized Water Makeup System," of the SRP which provides acceptance criteria and guidelines for reviewing the DWS. Specifically, the staff reviewed the system to ensure its capability to provide the required supply of reactor coolant pure makeup water to all systems. Acceptability of the DWS is based upon meeting the guidance of Position C.2 of RG 1.29 for non-safety-related systems, the failure of which could affect the functioning of any safety-related system. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the DWS satisfies the applicable requirements of GDC 2, as it relates to the system being capable of withstanding the effects of earthquakes, and GDC 5, as it relates to the effect of shared SSCs important to safety being capable of performing their required safety functions.

The DWS consists of the following equipment:

- a 379 m^3 (100,000 gallons) capacity demineralized water storage tank
- a 1136 m^3 (300,000 gallons) capacity condensate storage tank
- two motor-driven demineralized water transfer pumps
- two catalytic oxygen reduction units

The DWS is a non-safety-related system that supplies demineralized water, through the demineralized water storage tank, to fill the condensate storage tank and to the plant systems that demand a demineralized water supply. The demineralized water storage tank, which receives water from the DTS, supplies demineralized water to the makeup pumps of the CVS during startup and required boron dilution evolutions. A low level alarm on the tank signals the plant operator to isolate demands on the tank, other than CVS supply. The condensate storage tank serves as a reservoir to supply or receive condensate as required by the condenser hotwell level control system. In the event of loss of main feedwater when the deaerator storage tank is not available, the condensate storage tank will serve as a backup water supply for the startup feedwater pumps. The condensate storage tank will provide sufficient water to the startup feedwater system to permit eight hours of hot standby operation. Each of the water tanks is a vertical cylindrical stainless steel tank and the water oxygen content in the condensate storage tank is maintained below 100 ppb.

Two catalytic oxygen reduction units are used to degasify the stored demineralized water. One unit is provided for the demineralized water distribution system and the other unit is provided at the condensate storage tank. Each unit consists of a mixing chamber, a catalytic resin vessel, and a resin trap. Dissolved oxygen is removed chemically by mixing the effluent from the storage tank with hydrogen gas. A check valve, in conjunction with a block valve, is used to prevent backflow of fluids from systems that interface with the DWS. Westinghouse stated that the condensate storage tank normally contains no significant radioactive contaminants.

The DWS is classified as a non-seismic category (Class E) system that is designed to industrial standards. Because the DWS is not safety-related, the quality assurance requirements of Appendix B of 10 CFR Part 50 do not apply. As specified in Section 3.2.2 of the SSAR, the non-seismic category structures are designed for seismic loads according to the Uniform Building Code and are physically arranged so that an SSE could not cause unacceptable structural interaction with or failure of seismic Category I SSCs. Furthermore, the system has no safety-related function other than containment isolation, and its failure does not affect the ability of safety-related systems to perform their intended safety functions. Therefore, the design conforms to the guideline of Position C.2 of RG 1.29. Compliance with Position C.1 of RG 1.29 does not apply to the DWS because the system performs no safety-related function.

Compliance with GDC 5 is discussed in Section 9.2 of this report. In addition, the effect of flooding on other plant systems resulting from the failure of the demineralized storage tank is evaluated in Section 3.4.1.2 of this report, which has concluded that the DWS is not a potential flooding source.

On the basis of its review, the staff concludes that the DWS has the capability to provide adequate supply of reactor coolant pure makeup water to all plant systems during all modes of plant operation. The design of the system complies with Position C.2 of RG 1.29 concerning its seismic classification and satisfies the applicable requirements of GDC 2 with respect to the need for protection against natural phenomena. Therefore, the staff concludes that the DWS meets the guidance of Section 9.2.3 of the SRP, and is acceptable.

9.2.5 Potable Water System

The staff reviewed the potable water system (PWS) in accordance with Section 9.2.4, "Potable and Sanitary Water Systems," of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the PWS satisfies GDC 60, as it relates to design provisions provided to control the release of water containing radioactive material and prevent contamination of the potable water.

The PWS is a non-safety-related system that is designed to provide clean water from the raw water system for domestic use and human consumption. The system consists of the following equipment:

- a carbon steel tank with a capacity less than 37.85 m^3 (10,000 gallons)
- two motor-driven potable water pumps
- a system jockey pump
- a distribution header around the power block
- hot water storage heaters
- necessary interconnecting piping and valves

Potable water is supplied at a maximum flow rate of 1.7 m^3/min (440 gpm) determined by a quantity of 0.4 m^3 (100 gallons) per person per day for the largest number of persons expected at the plant during a 24-hour period. A minimum pressure of 20 psig is maintained in the distribution system. The potable water storage tank was designed with a capacity of up to 10,000 gallons.

The potable water is treated to prevent harmful physiological effects and its bacteriological and chemical quality conforms to the requirements of the EPA "National Primary Drinking Water Standards" (40 CFR Part 141). Disinfection is provided upstream of the potable water storage tank by the turbine island chemical feed system to disinfect the raw water supply into the tank. The PWS distribution is in compliance with 29 CFR 1910, "Occupational Safety and Health Standards, Part 141."

High- and low-water level signals and alarms are provided for the PWS water storage tank to automatically control the operation of the plant RWS clearwell water pumps and maintain the water level in the tank. Manual override of the automatic level controls is also available. The potable water distribution system is sized to limit flow velocity between 6 to 10 fps and water hammer arresters are installed at appropriate locations of the system piping. A continuously operated jockey pump with smaller capacity is used to maintain the pressure of the system during low-flow requirement periods. The PWS is tested hydrostatically for leak-tightness in accordance with the Uniform Plumbing Code.

In the SSAR, Westinghouse states that no interconnections exist between the PWS and any potentially radioactive system or any system using water for purposes other than domestic water service. To prevent contamination of the PWS from other systems supplied by the RWS, the common supply from the onsite RWS will be designed to use either an air gap or reduced-pressure-zone type backflow prevention device. Branches of the PWS supplying plumbing fixtures, located in areas of potential radiological hazard where access is restricted, are provided with the reduced-pressure-zone type backflow prevention devices.

On the basis of its review, the staff concludes that the design of the PWS, as described above, satisfies GDC 60 with respect to prevention of contamination by radioactive water. Therefore, the staff concludes that the potable water system meets the guidance of Section 9.2.4 of the SRP and is acceptable.

9.2.6 Sanitary Drainage System

The staff reviewed the sanitary drainage system (SDS) in accordance with Section 9.2.4, "Potable and Sanitary Water Systems," of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the SDS satisfies GDC 60, as it relates to design provisions provided to control the release of radioactive materials to the environment.

The SDS is a non-safety-related system that collects sanitary wastes from plant restrooms and locker room facilities in the turbine building, auxiliary building, and annex building for treatment, dilution, and discharge. The system is designed to accommodate 0.1 m^3 (25 gallon) per person per day, for up to 500 persons during a 24-hour period. The system will be tested and inspected in accordance with the Uniform Plumbing Code. The SDS components, such as branch lines, lift stations, and waste treatment plant are site-specific and outside the scope of the ABWR design.

In the SSAR, Westinghouse states that the SDS does not serve the facilities in radiologically controlled areas and has no connection to the systems having the potential for containing radioactive material. Therefore, the design of the SDS satisfies GDC 60, with respect to prevention of contamination by the radioactive waste drain system.

On the basis of its review, the staff concludes that the design of the sanitary drainage system, as described above, satisfies GDC 60 with respect to control of the release of water containing radioactive material. Therefore, the staff concludes that the sanitary drainage system meets the guidance of Section 9.2.4 of the SRP, and is acceptable.

9.2.7 Central Chilled Water System

The staff reviewed the central chilled water system (VWS) in accordance with Section 9.2.2, "Reactor Auxiliary Cooling Water Systems," of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the central chilled water system satisfies GDC 2, 44, 45, and 46.

The VWS is a non-safety-related system that provides chilled water to the cooling coils of the supply air handling units and unit coolers of the following plant heating, ventilation, and air conditioning (HVAC) systems during normal modes of plant operation:

- radiologically controlled area ventilation system
- containment recirculation cooling system
- containment air filtration system
- health physics/control access area HVAC system
- radwaste building ventilation system
- Annex I and auxiliary building nonradioactive ventilation system

The VWS also supplies chilled water to the components of the liquid radwaste system, gaseous radwaste system, containment leakrate-test system components, secondary sampling system, the portable and mobile radwaste system, and the electrical switchgear room and personal work area air handling units of the turbine building ventilation system.

The plant HVAC systems require chilled water as a cooling medium to satisfy the ambient temperature requirements for the plant. The cooling water to the chiller condensers is supplied from the component cooling water system. The VWS is divided into two closed-loop subsystems, i.e., the high-capacity subsystem and the low-capacity subsystem.

The high-capacity subsystem, located in the turbine building, is the primary system to provide chilled water to the above major HVAC systems and other plant equipment requiring chilled water cooling. The high-capacity subsystem consists of the following equipment:

- two 100-percent capacity chilled water pumps
- two 100-percent capacity water-cooled chillers
- a chemical feed tank
- an expansion tank
- associated valves, piping, and instrumentation

The high-capacity subsystem, located in the turbine building, is arranged in two parallel trains with common supply and return headers. Each train includes one pump and one chiller. During normal operation of the subsystem, one pump/chiller train is required to provide chilled water to plant components at a normal temperature of 4.4 °C (40 °F). The standby train would be started manually if the operating train fails. The design cooling capacity of the high-capacity subsystem is founded on the ambient design temperature of 38 °C (100 °F), dry bulb/29 °C (77 °F) coincident wet bulb maximum, and -23 °C (-10 °F) minimum.

In Section 9.2.7.2.1 of the SSAR (Revision 3), Westinghouse eliminated an air separator from the high-capacity subsystem that had been specified in the initial design. In a letter dated May 20, 1996, Westinghouse stated that air separators are not required because the expansion tank is sized and located to service this function. The staff agrees and finds this revision to be acceptable.

The low-capacity subsystem, located in the auxiliary building, is designed to provide chilled water to the HVAC systems in the main control room, the technical support center, and the

Class 1E electrical equipment room. The low-capacity subsystem consists of two 100-percent capacity chilled water loops, each with the following equipment:

- a chilled water pump
- an air-cooled chiller
- an expansion tank
- associated valves, piping, and instrumentation

This subsystem is arranged in two independent trains with separate supply and return headers. During normal operation of the subsystem, one pump/chiller train is required to supply chilled water to the components of the nuclear island nonradioactive ventilation system and the radiologically controlled area ventilation system at a normal temperature of 4.4 °C (40 °F). In the event that one train is inoperable, the standby train can be manually aligned to supply chilled water to these components. The design cooling-capacity for the low capacity subsystem is founded on the ambient design temperatures of 46 °C (115 °F) dry bulb/26.7 °C (80 °F) coincident wet bulb maximum.

The staff reviewed the information provided in Section 9.2.7 and Figures 9.2.7-1 (3 sheets) of the SSAR and confirmed that there is sufficient redundancy and power supply to ensure continued operation of the VWS. Should the normal offsite power supply be unavailable, the system will be powered by the standby onsite ac power source (diesel). Westinghouse also performed reliability analysis for the VWS in SSAR Chapter 20, for the AP600 PRA. The PRA review is addressed in Section 19.1 of this report.

The VWS is not required to achieve safe shutdown or to mitigate any postulated accidents and serves no safety-related function, except for the portion of the system lines routed into the containment that require containment isolation. The high-capacity subsystem supply and return lines that penetrate the containment are provided with two air-operated containment isolation valves. These valves automatically close upon receipt of a containment isolation signal. A bypass mode, with indication in the control room, is also provided to restore containment recirculation system cooling during containment isolation. The containment isolation valves and the piping between them are designed in accordance with the requirements of Section III of the ASME Code for Class 2 components and are classified as seismic Category I, safety class 2, quality group B. This portion of piping and valves is safety-related and is designed as part of the containment isolation system.

The VWS and its associated equipment are located in the auxiliary building, which is seismic Category I, as well as flood- and tornado missile-protected. The VWS is designed as a non-seismic system and is classified as Class D in Table 3.2-3 of the SSAR. Section 3.2.2 of the SSAR states that Class D is non-safety-related, with some additional requirements on procurement and inspection. The inclusion of the non-safety-related system in Class D recognizes that the system provides an important first level of defense that helps to reduce the calculated PRA core melt frequency. The VWS is not required to support the function of safety-related equipment; a failure of the system will not impact the operation of safety-related equipment and will not impact the equipment for safe shutdown.

Because the VWS has no safety-related function and a failure of the system will not impact the operation of safety-related equipment, the requirements of GDC 44, as related to the capability to transfer heat loads from safety-related systems; GDC 45, as related to inservice inspection of safety-related components and equipment; and GDC 46, as related to operational functional testing of safety-related systems or components, are not applicable.

On the basis of its review, the staff concludes that the safety-related portion of the system (the containment penetrations and the isolation valves) complies with Position C.1 of RG 1.29 on the basis that they are designed in accordance with containment isolation provisions. Because the system serves no safety-related function and its failure as a result of an SSE will not reduce the functioning of any safety-related plant features, the non-safety-related portion of the system complies with Position C.2 of RG 1.29. Therefore, the staff concludes that the design of the central chilled water system meets the guidance of Section 9.2.2 of the SRP and is acceptable.

9.2.8 Turbine Building Closed Cooling System

The staff reviewed the design of the turbine building closed cooling system (TCS) in accordance with applicable provisions of Section 9.2.2, "Reactor Auxiliary Cooling Water System," of the SRP. With respect to GDC 2, as related to structures and systems being capable of withstanding the effects of earthquakes, acceptance is based on meeting the guidance of RG 1.29, Position C.2, for non-safety-related portions of the system.

The TCS is a closed-loop cooling water system that provides chemically treated, demineralized water for the removal of heat from non-safety-related heat exchangers in the turbine building, and rejects the heat to the circulating water system (CWS). The TCS has no safety-related function. The system comprises the following equipment:

- two 100-percent pumps
- three 50-percent heat exchangers
- a surge tank
- a chemical addition tank
- associated piping, valves, and instrumentation and controls

The TCS is classified as an AP600 Class E (designed to industrial standards).

In RAIs Q410.128 and Q410.133, the staff identified discrepancies between the SSAR system description and P&IDs regarding the instrumentation and components of the TCS, and requested clarification. Westinghouse's responses to these RAIs were incomplete, resulting in DSER Open Item 9.2.8-1. In Revision 3, Westinghouse revised Section 9.2.8.5 and Table 9.2.8.1 of the SSAR to clarify the description and eliminate the discrepancies. The staff finds the SSAR revisions acceptable. Therefore, DSER Open Item 9.2.8-1 is closed.

In its response to RAI Q410.132, Westinghouse stated that a sufficient net positive suction head is available at the TCS pump suction because the elevation of the TCS surge tank is about 18.6 m (61 ft) higher than the elevation of TCS pumps. In its response to Q410.131 dated June 30, 1994, concerning water hammer, Westinghouse stated that the TCS is designed to minimize the potential for water hammer because it is a closed water system with moderate water temperature; is always pressurized such that the pressure of the TCS fluid is kept above saturation pressure; and does not have any fast-acting, power-operated valves.

In its response to Q410.130 (dated June 27, 1994) concerning RG 1.29, Westinghouse stated that the TCS will comply with GDC 2 by adhering to the guidance of Position C.2 of RG 1.29 for ensuring that the non-safety-related portions of the system could withstand the effects of earthquakes without affecting adjacent safety-related systems. Piping and fluid-containing components of the TCS are located entirely within the turbine building, and no safety-related equipment is located in the turbine building. Therefore, the failure of the TCS (including the effects of jet impingement and flooding) cannot lead to the failure of any safety-related SSCs.

Because the TCS is not safety-related, and its failure does not lead to the failure of any safety systems, the remaining requirements of GDC 4, 44, 45, and 46, as reflected in the guidance of Section 9.2.2, of the SRP do not apply. The TCS meets the requirements of GDC 2, on the basis of meeting RG 1.29, Position C.2, as described above, for the non-safety-related portions. Therefore, the TCS meets the applicable (non-safety-related) provisions of Section 9.2.2 of the SRP.

9.2.9 Waste Water System

The staff reviewed the waste water system (WWS) in accordance with Section 9.3.3, "Equipment and Floor Drainage System," of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the WWS satisfies the requirements of GDC 2, 4, and 60, as they relate to the protection against natural phenomena, flooding, and radiological release to the environment, respectively.

The WWS is a non-safety-related system that collects and processes the waste water from the equipment and floor drains in the nonradioactive building areas during plant operation and outages. Low-volume wastes from the turbine building floor and equipment drains are collected in the two turbine building drain tanks for temporary storage. This waste liquid is then transported by the drain tank pumps through the oil separator for discharge. Other sources of waste water collected in the drain tank are the nonradioactive sumps in the auxiliary building and diesel generator building. The waste water from either of the two drain tanks is pumped to the oil separator for removal of oily waste. The waste water from the oil separator flows by gravity to the waste water retention basin, if required, for settling of suspended solids and treatment before discharge. The effluent in the retention basin is pumped to either the cooling tower basin or to the cooling tower blowdown sump, depending on the quality of the waste water.

In the event that radioactivity is present in the drain tanks, a manual three-way valve allows for the waste water to be diverted from the drain tanks to the liquid radwaste system (WLS) for processing and disposal. A radiation monitor is installed on the common discharge piping of the drain tank pumps to detect and isolate the contaminated waste water. The radiation monitor will alarm upon detecting radioactivity in the waste water and trip the drain tank pumps. Westinghouse states in the SSAR that provisions for sampling the drain tanks for radioactive contamination is included in the design. Therefore, the staff concludes that the design of the WWS satisfies GDC 60 with respect to control of the release of water from the WWS containing radioactive material.

In Section 9.2.9.2 of the SSAR, Westinghouse indicates that level controls will be provided for the building drain tanks and the waste water retention basin to prevent overflow of these waste water collection points. High- and low-water level alarms will be installed to alert the operator to take action. However, the staff is concerned that the piping of the waste water drainage system to the drain tanks may fail because of an event such as an earthquake. In Q410.185, the staff requested Westinghouse to provide information on flood levels caused by WWS failure and methods for draining the waste water after a limiting pipe break, assuming a period of water leakage while the operator isolates the problem area. Also, safety-related equipment in other plant areas shall not be affected if a pipe rupture and flooding occur in any of the drain tanks and basin areas.

In its response, Westinghouse stated that the limiting pipe break in the turbine building is a postulated failure of the WWS piping or expansion joint. No credit is taken for turbine drain lines, drain tank pumps, or level controls for pipe breaks in the turbine building. This limiting break will not result in detrimental effects to safety-related equipment because the turbine building has no safety-related equipment. Westinghouse analyzed the WWS piping and expansion joint failures that may cause internal flood in the turbine area and determined that WWS failure will not be a dominant risk. Westinghouse analyzed the postulated WWS failures and concluded that they will not cause a flood hazard to the safety-related equipment. Therefore, the staff concludes that the design of the WWS complies with GDC 4 with respect to flood protection.

The staff also finds that the WWS design complies with GDC 2, as related to the ability of withstanding the effects of earthquakes. Compliance with GDC 2 is predicated on meeting the guidance of Positions C.1 and C.2 of RG 1.29 concerning its seismic classification. The WWS need not comply with Position C.1 because the system is not safety-related. Instead, the WWS complies with the guidelines of Position C.2 of RG 1.29 because failure of the system during an SSE will not reduce the function of any safety-related plant features.

- On the basis of its review, the staff concludes that the WWS meets the NRC regulations set forth in the following review criteria:

- GDC 2, with respect to protecting the system against natural phenomena

- GDC 4, with respect to preventing flooding that could result in adverse effects on safety-related systems

- GDC 60, with respect to preventing the inadvertent transfer of contaminated fluids to the noncontaminated drainage system for disposal

- RG 1.29, Position C.2, for non-safety-related functions

Therefore, the design of the WWS meets the guidance of Section 9.3.3 of the SRP, and is acceptable.

9.2.10 Hot Water Heating System

The hot water heating system (VYS) is a non-safety-related system that supplies heated water to selected non-safety air handling units and unit heaters in the plant during cold weather operation, and to the containment recirculation fan coil units during plant outages in cold weather. In the event of an accident or LOOP, the system will be powered from the onsite diesel generators. There are no GDC or SRP guidelines that are directly applicable to the review of the hot water heating system; therefore, the staff's review is founded on the relevant regulatory guidance and industry standards that apply to the evaluation of the VYS.

The hot water heating system consists of a heat transfer package (including two 50-percent capacity heat exchangers, two 50-percent capacity system pumps, a surge tank, and a chemical feed tank) and a distribution system to the various HVAC systems and unit heaters. The VYS is manually actuated and may operate when the site ambient temperature is 23 °C (73 °F) or below. The system takes high-pressure extraction steam from the high-pressure turbine crossunder piping to heat water by transferring the heat energy through the heat exchangers. During a plant outage, the auxiliary steam taken from the auxiliary boiler is used to heat water. The water is heated to 160 °C (320 °F) by the heat exchanger and is pumped to the hot water coils of the selected air handling unit. The surge tank will maintain the system pressure within the design limit of 1028 kPa (150 psia). The chemical feed tank has the capability to provide chemical mixing in the system for corrosion control. The water to the closed loop VYS is supplied by the demineralized water transfer and storage system.

In the SSAR, Westinghouse states that the VYS has no safety-related function and interfaces with only non-safety related systems. Therefore, the requirements of GDC 5, 44, 45, and 46 are not applicable. The VYS and its associated equipment are designed to be non-seismic Category (seismic designed according to the Uniform Building Code) and are classified as Class E in Table 3.2-3 of the SSAR. Based on the above, the staff agrees that the VYS is not a safety-related system; therefore, the quality assurance requirements of Appendix B of 10 CFR Part 50 do not apply.

During its initial review, the staff requested Westinghouse (in Q410.261) to provide the operating pressures and temperatures of the VYS piping lines that supply hot water to major areas of the plant and information on certain VYS lines that route to the safety-related areas. Westinghouse provided the following information in its response:

- The operating temperature and pressure of the VYS is about 149 °C (300 °F, supply), 104 °C (220 °F, return) and 819 kPa (120 psig). The piping system design conditions are 160 °C (320 °F) and 1380 kPa (200 psig).

- No VYS lines are routed inside the containment. Hot water routed into the containment is via the central chilled water system.

- No VYS piping is routed in rooms that contain safety-related equipment.

- The VYS lines are not routed over or through the main control room.

The VYS was initially designed as a moderate-energy system (at 35 psia and 200 °F). In Section 9.2.10.3 (Revision 3) of the SSAR, the design of the VYS was changed to high-energy system. Westinghouse stated that the system piping routed in the safety-related areas are one-inch lines and smaller and are not evaluated for high-energy line breaks. However, the staff is concerned about the potential consequences of a VYS pipe break in the safety-related areas that could directly or indirectly result in lose of required redundancy in any portion of the systems or equipment. Westinghouse stated that there are no adverse consequences on safety-related components or equipment due to postulated breaks of the VYS piping because the VYS piping is generally excluded from safety-related plan t areas outside containment.

On the basis of its review, the staff has concluded the following:

- The VYS meets GDC 2 because it is not important to safety and its failure will not affect the functions of the safety-related systems, and Position C.2 of RG 1.29 because it interfaces with only non-safety-related systems. The system is not required to comply with Position C.1 of RG 1.29 because it is not a safety-related system.

- The VYS, as designed to industrial standards as a non-seismic Category and classified as Class E, is acceptable because it is not a safety-related system.

Therefore, the staff concludes that the design of the VYS is acceptable.

9.3 Process Auxiliaries

The staff's review of the AP600 process auxiliaries is provided in the following sections: 9.3.1, Compressed and Instrument Air System; 9.3.2, Plant Gas System; 9.3.3, Primary Sampling System; 9.3.4, Secondary Sampling System; 9.3.5, Equipment and Floor Drainage System; 9.3.6, Chemical and Volume Control System (CVS); and 9.4 Air-Conditioning, Heating, Cooling, and Ventilation System.

9.3.1 Compressed and Instrument Air System

The staff reviewed the compressed and instrument air system (CAS) in accordance with the guidance of Section 9.3.1 of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding whether the instrument air subsystem of the CAS satisfies the following requirements:

- GDC 1, as it relates to systems and components being designed, fabricated, and tested to quality standards in accordance with the importance of the safety functions to be performed

- GDC 2, as it relates to the capability of safety-related CAS components to withstand the effects of earthquakes

- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions

After the AP600 DSER was issued, Westinghouse changed the design of the CAS. Therefore, the information originally contained in Section 9.3.1 of the DSER has been superseded by this report. Also, Westinghouse provided additional information about the CAS in the form of SSAR revisions and telephone conferences that has allowed the staff to continue its review as discussed below. Therefore, DSER Open Item 9.3.1-1, concerning the modification of certain RAI responses, clarification of COL applicant responsibilities, conformance with RG 1.68.3, and the resolution of Generic Issue 43, is closed.

The AP600 design can be used at either single-unit or multiple-unit sites. Nonetheless, in Section 3.1.1 of the SSAR, Westinghouse states that the AP600 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. Should a multiple-unit site be proposed, the COL applicant referencing the AP600 design will be required to apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs important to safety to perform their required safety functions.

As identified in Table 3.2-3 of the SSAR, the CAS components, with the exception of the containment penetration piping and isolation valves, are classified as non-nuclear safety and non-seismic. The quality assurance requirements of Appendix B to 10 CFR Part 50 do not apply. The containment penetration piping and isolation valves are classified as safety Class 2, seismic Category I, quality group B. The system description, components, and flow diagrams are provided in Section 9.3.1, Tables 9.3.1-1 to 9.3.1-4, and Figure 9.3.1-1 of the SSAR, respectively.

The CAS consists of the following subsystems:

- the instrument air system
- the service air system
- the high-pressure air systems

The CAS serves no safety-related function other than containment isolation. The major components of the CAS are located in the turbine building. The design of the three subsystems is summarized below.

The instrument air subsystem consists of two 100 percent redundant air compressor divisions. Normally, one compressor runs continuously and the second compressor starts automatically if the first unit fails or if demand exceeds the capacity of the operating compressor. Each air compressor division consists of the following:

- a multistage, low-pressure, rotary screw air compressor package
- a desiccant dryer with a prefilter and an afterfilter
- an air receiver

Each air compressor package includes an intake filter, silencer, intercooler, aftercooler, moisture separators, bleed-off cooler, oil cooler, oil reservoir, automatic load controls, relief valves, and a discharge air check valve. Downstream of the air receivers, the two air compressor divisions join together to form a single instrument air distribution header. The distribution header supplies clean, dry, oil-free air to pneumatically operated valves and instruments inside containment, the turbine building, the annex building, the radwaste building, and the auxiliary building.

One instrument air compressor division can be connected to each of the non-safety-related onsite standby diesel generators. The air compressors are cooled by water supplied from the CCS.

A normally closed connection flange on the distribution header provides the capability to temporarily connect to the service air subsystem, should the instrument air compressors become inoperable. In this arrangement, the service air subsystem would then supply high-quality backup air to the instrument air subsystem by way of a temporary mechanical jumper.

The instrument air line to the containment is normally open; however, air flow to the containment is monitored and a high flow alarm is provided to indicate a possible instrument air line rupture inside containment.

Generic Issue (GI) 43 was resolved by the issuance of Generic Letter (GL) 88-14 which required licensees and applicants to review the recommendations of NUREG-1275, and perform a design and operations verification of the system. A complete discussion of how the AP600 design addresses GI 43 is provided in Section 20.3 of this report.

In accordance with NUREG-1275, the instrument air subsystem of the CAS is (1) designed to provide compressed air which meets the manufacturer's air quality standards for pneumatic equipment supplied as a part of the plant, and (2) initially tested in accordance with the guidance of RG 1.68.3, and (3) periodically tested in accordance with ANSI MC II.1-1976 (ANSI/ISA-S-7.3) as discussed below.

The intake filters for the instrument air subsystem prevent particulates 10 microns and larger from entering the air supply to the compressors. The instrument air receivers function as storage devices for the compressed air. Each dryer assembly consists of a coalescing prefilter, two desiccant air dryer towers, and a particulate afterfilter. The coalescing prefilter removes oil aerosols and moisture droplets. The desiccant air dryers maintain the instrument air at a dewpoint of -33.3 °C (-28 °F) at operating line pressure. The instrument air subsystem provides high-quality instrument air, as specified in the ANSI/ISA S7.3 standard.

Sample points are provided downstream of the air dryers in the instrument air subsystem to monitor the air quality supplied by each compressor. Periodic checks are made to assure high quality instrument air as specified in the ANSI/ISA S7.3 standard.

Air-operated valves that are essential for safe shutdown and accident mitigation are designed to actuate to the fail-safe position upon loss of air pressure. A list of the safety-related air-operated valves supplied by the instrument air subsystem are identified in Table 9.3.1-1 of the SSAR. There are no safety-related air-operated valves that rely on safety-related air accumulators to actuate to the fail-safe position upon loss of air pressure.

Section 9.3.1.4 of the SSAR states that, during the initial plant testing prior to reactor startup, safety systems utilizing instrument air will be tested as part of the safety system test to verify fail-safe operation of air-operated valves upon sudden loss of instrument air or gradual reduction of air pressure, as described in RG 1.68.3. In addition, Section 14.2.9.4.10 of the SSAR states that testing is performed to verify the fail-safe positioning of safety-related air-operated valves for sudden loss of instrument air or gradual loss of pressure, as described in Section 9.3.1.4 of the SSAR.

Therefore, the AP600 design complies with the guidance of ANSI MC 11.1-1976 (ANSI/ISA-S7.3), as it relates to supplying clean, dry, oil-free air to safety-related components, and the guidance of RG 1.68.3, as it relates to the testing of the CAS. On this basis, the staff concludes that the CAS complies with the requirements of GDC 1, with respect to systems and components important to safety being designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The service air subsystem consists of two 100 percent redundant air compressor divisions. Normally, one compressor runs continuously and the second compressor starts automatically if the first unit fails or if demand exceeds the capacity of the operating compressor. Each compressor division consists of:

- a multistage, low-pressure, rotary screw air compressor package
- a desiccant dryer with a prefilter and an afterfilter

Each air compressor package includes an intake filter, silencer, intercooler, aftercooler, moisture separators, bleed-off cooler, oil cooler, oil reservoir, automatic load controls, relief valves, and a discharge air check valve. Downstream of the afterfilters, the two air compressor divisions join together and connect to a common air receiver. The air receiver discharges to a service air distribution header, which supplies clean, dry, oil-free air to service outlets inside containment, the turbine building, the annex building, the diesel generator building, the radwaste building, and the auxiliary building.

Cooling water to the service air compressors is supplied from the CCS, and the service air line to the containment is normally closed.

The intake filters for the service air subsystem prevent particulates 10 microns and larger from entering the air supply to the compressors. The service air receivers function as storage devices for the compressed air. Each dryer assembly consists of a coalescing prefilter, two desiccant air dryer towers, and a particulate afterfilter. The coalescing prefilter removes oil aerosols and moisture droplets. The desiccant air dryers maintain the instrument air at a dewpoint of -33.3 °C (-28 °F) at operating line pressure.

Plant breathing air requirements are satisfied by using the service air subsystem as a supply source. Portable, individually packaged, air purification equipment can be attached to any service air subsystem outlet to improve the service air quality to a minimum of Quality Verification Level D as defined in ANSI/CGA G-7.1. The breathing air purification package consists of replaceable cartridge-type filters, a pressure regulator, carbon monoxide monitoring equipment, air supply hoses, and air supply devices. Carbon monoxide is controlled by a catalytic conversion to carbon dioxide within the package. The service air subsystem is not connected to the instrument air subsystem.

The high-pressure air subsystem consists of one air compressor division that consists of the following components:

- an intake filter
- a four-stage, high pressure, oil-lubricated, reciprocating air compressor
- air-cooled intercoolers

- interstage oil/water separators
- an air-cooled aftercooler
- a final oil/water separator
- relief valves
- an integral air purification system
- discharge check valves
- a high-pressure air receiver

The high-pressure air subsystem supplies clean, dry, oil-free air to the main control room emergency habitability system (VES), the generator breaker, and the fire-fighting apparatus recharge station. The isolation valves to these locations are normally closed and are manually opened on an as needed basis to refill the specified equipment air storage reservoirs from the high-pressure receiver. The compressor is then started to replenish the air stored in the high-pressure receiver.

The air compressor has an integral air purification system to produce air for high-pressure applications, This integral high-pressure air purification system utilizes a series of replaceable cartridge-type filters to produce breathing quality air. The high-pressure air subsystem supplies Quality Verification Level E air, as defined in ANSI/CGA G-7.1, and periodic checks on the high-pressure air compressor are made on a regular basis to verify that the breathing air meets these standards. Carbon monoxide is controlled by a catalytic conversion to carbon dioxide within the package. Breathing air connections to the high-pressure air subsystem are incompatible with the breathing air connections of the service air subsystem to prevent attaching the portable air purification equipment to the high pressure air subsystem.

The onsite standby diesel generators provide an alternate source of electrical power for the high-pressure air compressor.

The high-pressure air subsystem is classified as a high-energy system. The high-pressure compressor and receiver are located in the turbine building, which contains no safety-related equipment or structures. Air piping in safety-related areas is one inch or less in diameter and the dynamic consequences of a rupture are not required to be analyzed, This subsystem is not required to operate following a design-basis accident, nor is it used for safe shutdown of the plant.

Compliance with Position C.1 of RG 1.29 is not required because the CAS is non-safety-related. Instead, the CAS complies with RG 1.29, Position C.2, because the CAS is not required to remain functional, and its failure as a result of an SSE will not reduce the functioning of any plant feature included in items 1.A through 1.Q of RG 1.29, Position C.1, to an unacceptable safety level. The SSCs are non-nuclear safety class, but the structure housing the CAS (turbine building) is designated as seismic Category II and is designed and constructed so that the SSE will not cause any failure in a manner that would adversely affect other safety systems, as stated in Section 3.2.1 and Table 3.2-1 of the SSAR. Therefore, the system complies with GDC 2, as it relates to the ability of the system to withstand the effects of earthquakes.

On the basis of the above review, the staff concludes that the CAS complies with GDC 1, 2, and 5, as referenced in Section 9.3.1 of the SRP and is, therefore, acceptable.

9.3.2 Plant Gas System

The plant gas system (PGS) provides hydrogen, carbon dioxide, and nitrogen gases to plant systems as required. Other gases, such as oxygen, methane, acetylene, and argon are supplied in smaller individual containers and are not supplied by the PGS. The hydrogen portion of the PGS supplies hydrogen to the main plant electrical generator for cooling as well as to other plant auxiliary systems. The carbon dioxide portion stores and supplies carbon dioxide to the generator to purge hydrogen and air during layup or plant outages. The nitrogen portion of the PGS supplies nitrogen for pressurizing, blanketing, and purging various plant components.

The PGS is required for normal plant operation and startup of the plant. The PGS serves no safety-related function. Failure of the system does not compromise any safety-related system, nor does it prevent safe reactor shutdown.

All three portions of the PGS are packaged systems consisting of liquid gas storage tanks and vaporizers. All liquid gas storage tanks are built and hydrostatically tested in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, 1989. In addition, each vessel is examined using the magnetic particle method. Low-level indication alarms are provided in the main control room for the liquid nitrogen and hydrogen storage tank levels. Temperature and pressure indications are provided at various points within the plant gas system.

The main steam isolation valves (MSIVs) and the main feedwater isolation valves (MFIVs) are safety-related valves that use compressed nitrogen stored within the valve operators as the motive force to close the valves. Note 21 on Figure 10.3.2-1 of the SSAR specifies that the MSIVs and MFIVs are pneumatic-hydraulically actuated with a sealed nitrogen accumulator that provides the stored energy to close the valve. Nitrogen makeup for these valves (if needed) is provided from portable high-pressure nitrogen bottles using temporary connections on the valves.

In Section 6.4 of the SSAR, Westinghouse addresses the effect of the PGS on main control room habitability, including explosive gases and burn conditions for those gases. The PGS is designed in conformance with RG 1.91 for explosions.

The nitrogen and carbon dioxide portions of the PGS are located inside the turbine building, and the hydrogen system storage is located outdoors at the hydrogen storage tank area. The storage tanks are analyzed as a potential missile source in Section 3.5 of the SSAR. This is also discussed in Section 3.5.1.1 of this report.

The staff reviewed the PGS on the basis of the above and concludes that it is acceptable.

9.3.3 Primary Sampling System

The primary sampling system is designed to collect and deliver representative samples of liquids and gases from various process fluids, including reactor coolant system and containment air, in a safe, convenient manner during normal and post-accident conditions. The system includes provisions to route sample flow to a laboratory for continuous or intermittent sample analysis. The proposed design uses common sampling lines and points for both normal operation and post-accident sampling. The primary sampling system includes piping, valves, heat exchangers, and other components associated with the system from the point of sample withdrawal from a fluid system up to the analyzing station, sampling station, or local sampling point.

The staff reviewed the primary sampling system in accordance with the guidelines of Section 9.3.2 of the SRP. The review included the following areas:

- design objectives
- design criteria
- technical specifications
- process streams
- piping and instruments drawings
- locations of sampling points and sample stations
- purging of sample lines
- seismic design and quality group classifications
- isolation
- promptness of obtaining samples under accident conditions

Although the primary sampling system has no safety-related function, some of its sampling lines may be connected to safety-related systems. Therefore, in order to meet the requirements of GDC 1 and 2, the seismic and quality group classification of these lines, associated components, and instruments should conform to the classification of the system to which they are connected. The staff concluded that the SSAR should be modified accordingly. This was identified as DSER Open Item 9.3.3-1.

Westinghouse subsequently clarified that the component classification for the primary sampling system was addressed in Table 3.2-3 of the SSAR. The primary sampling system components are classified as ASME Class 2 and 3, seismic Category I. This will ensure that the primary sampling system will perform in service as designed. The staff concluded that no further revision to the SSAR was necessary. Therefore, Open Item 9.3.3-1 is closed.

Process Sampling

In process sampling, liquid grab samples are obtained from the following areas:

- reactor vessel hot leg
- pressurizer vapor space
- pressurizer liquid space
- demineralizer
- accumulator
- core makeup tank (CMT)
- containment sump

Gaseous samples are obtained from the containment building atmosphere. These samples are analyzed for the following:

- boron
- chloride
- fluoride
- lithium
- sodium
- dissolved gases concentrations
- fission and corrosion product activity levels
- pH
- conductivity levels
- fission gas content
- gas composition in various vessels
- hydrogen content in the containment

The staff's review of the equipment and procedures specified by Westinghouse for the primary sampling system finds that, under normal operating conditions, the system has provisions for performing satisfactory sampling and analysis for all these variables.

Post-Accident Sampling

The operators use the same primary sampling system that is used for process sampling to perform post-accident sampling system (PASS) activities. The system is operated manually and grab samples are taken from the RCS hot leg, the containment sump, and the containment atmosphere for subsequent chemical and radiochemical analyses similar to that during normal plant operations. All excess sample and purge volumes from inside containment sample points can be returned to the containment. In the DSER, the staff stated that in order to meet the requirements of GDC 19 and 10 CFR 50.34 (f)(2)(viii), samples that may contain TID-14844 source term radioactive materials should be taken without radiation exposures to any individual that exceed 5 rem to the whole body and 50 rem to the extremities. The SSAR had to be modified accordingly. This was DSER Open Item 9.3.3-2. Westinghouse addressed this item by clarifying that the AP600 is in compliance with the 5 rem limit and provided explanatory information in its response to RAI 471.22. The staff concluded that no further revision to the SSAR was necessary. Therefore, DSER Open Item 9.3.3-2 is closed.

In Section 9.3.3.1.2.2 of the SSAR, Westinghouse stated that the primary sampling system was designed to operate in a post-accident environment by complying with the guidance of NUREG-4330, as supplemented by additional NRC regulations, guidance, and recommendations, including the following:

- 10 CFR 50.34(f)(2)(viii)
- 10 CFR 50.34(f)(2)(xxvi)
- 10 CFR 50.34(f)(2)(xxvii)
- RG 1.97

- NRC staff positions presented in "Major Technical and Policy Issues Concerning the Evolutionary and Passive Plant Design"

Since the SSAR was written, the position paper "Major Technical and Policy Issues Concerning the Evolutionary and Passive Plant Design" was superseded by SECY-93-087. In the DSER, the staff stated that Westinghouse should ensure that the PASS conforms with SECY-93-087, and modify the SSAR accordingly. This was DSER Open Item 9.3.3-3. Westinghouse provided a discussion concerning compliance with SECY-93-087 in its response to RAI 281.20 and included a reference to SECY-93-087. The staff found the response to RAI 281.20 acceptable. Therefore, DSER Open Item 9.3.3-3 is closed.

In SECY-93-087, the staff recommended that the Commission approve its position that evolutionary and passive ALWRs (PWR-type reactor) be required to have the capability to analyze for dissolved gases in the reactor coolant in accordance with Item III.B.3 of NUREG-0737 and the requirements of 10 CFR 50.34(f)(2)(viii). However, the time for taking these samples can be extended to 24 hours following the accident. The staff acknowledged that determination of chloride concentrations, although helpful in ensuring that plant personnel take appropriate actions to minimize the likelihood of accelerated primary system corrosion following the accident, is a secondary consideration. Therefore, the ability to determine chloride concentrations is not a mandatory requirement of the PASS. The staff also recommended that the Commission approve the deviation from Item II.B.3 of NUREG-0737 with regard to sampling reactor coolant for boron concentration and activity measurements using the PASS in evolutionary and passive ALWRs. The rationale for this deviation is that both these measurements are used only to confirm the accident mitigation measures and conditions of the core obtained by other methods, and do not need to be performed in an early phase of an accident. Neutron flux monitoring instrumentation that complies with Category I criteria of RG 1.97 will have fully qualified, redundant channels that monitor neutron flux over the required power range. Therefore, sampling for boron concentration will not be needed for the first eight hours after an accident. Samples for activity measurements provide the information used to evaluate the condition of the core. However, this information will be made available during the accident management phase by monitoring other pertinent variables. Accordingly, sampling for activity and boron measurement could be postponed until 24 hours and 8 hours, respectively, following an accident. The need for obtaining radiological information within 3 hours following an accident for emergency response purposes is addressed in Section 13.3 of this report.

In a staff requirements memorandum (SRM) dated July 21, 1993, the Commission approved the staff positions to remove from the PASS in evolutionary and passive ALWRs the requirement for determining the concentration of hydrogen in the containment atmosphere in accordance with Item III.B.3 of NUREG-0737 and the requirements of 10 CFR 50.34(f)(2)(viii). It also approved extending the time limit for analysis of the coolant for boron and activity to 8 hours and 24 hours, respectively. The Commission directed the staff to modify its recommendations to ensure that the PASS has the capability to determine the gross amount of dissolved gases in a PWR-type reactor (not necessarily pressurized) as a means to meet the intent of 10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737.

The AP600 PASS design provides the capability to obtain liquid reactor water samples for boron analysis eight hours after an accident, and to analyze the sample for gross activity, dissolved gases and chloride 24 hours after an accident. The PASS design also provides the capability to obtain samples of the containment atmosphere for radionuclide analysis within 24 hours after an accident. There is no need to have a unique system to measure hydrogen concentration in the containment atmosphere because hydrogen concentration can be determined by the hydrogen analyzer in the containment hydrogen control system. These post-accident capabilities of the primary sampling system are in conformance with the staff recommendations in SECY-93-087, as modified by the Commission in the SRM dated April 2, 1993, and based on the commitments in Section 1.9.3 and 9.3.3 of the SSAR, the AP600 PASS meets the requirements of 10 CFR 50.34(f)(2)(viii), except for use of the TID 14844 source term. The justification for exempting the AP600 design from the requirements to use TID 14844 source term is provided in Section 20.6 of this report.

Isolation of the primary sampling system from the containment during post-accident conditions is specified in Section 6.2.3 of the SSAR. In the DSER, the staff requested that Westinghouse provide assurance that the isolation signal can be overridden to allow opening of these valves for taking post-accident samples. This was DSER Open Item 9.3.3-4. SSAR Section 6.2.3 states that containment isolation valves close on a containment isolation signal and that manual controls of these valves are provided at the local sample panel in the sample room. This closes Open Item 9.3.3-4.

In view of the above the NRC staff concludes that the primary sampling system is acceptable because it meets the following requirements:

- GDC 1 and 2, by designing the sampling lines and components of the sampling system to conform to the classification of the system to which each sampling line is connected

- GDC 13, as relates to monitoring variables that can affect the fission process, the integrity of the reactor core, and the reactor coolant system pressure boundary, by sampling the reactor coolant and the liquids contained in the following components:

  - accumulators
  - CMTs
  - containment sump
  - boric acid tank
  - boric acid batching tank
  - IRWST tank for boron

- GDC 14, by sampling the primary coolant for chemical impurities that can affect the reactor coolant pressure boundary

- GDC 60, as it relates to the capability of the primary sampling system to control the release of radioactive materials to the environment (by purging draining sample to streams back to the system of origin, or the appropriate radwaste treatment system)

- GDC 64, by sampling the containment building atmosphere and the containment building sump for radioactivity

9.3.4 Secondary Sampling System

The function of the secondary sampling system is to collect and deliver representative samples of fluids from various plant fluid systems to a laboratory for analysis. It does not serve any safety-related function, and has no nuclear safety design basis. The secondary sampling system relies on continuous inline analyses for monitoring the secondary chemistry that is required for assessing and understanding integrated secondary plant operations. It samples water from the turbine cycle, demineralized water treatment system, and circulated water system. The secondary sampling system can provide analyses for several parameters, including the following:

- chloride
- sulfate
- silica
- iron
- copper content
- dissolved oxygen
- pH
- conductivity levels

Grab sample capability is provided as a backup method to obtaining samples, and is also used for calibrating the inline instrumentation. The steam generator blowdown lines are continuously monitored for radioactivity caused by primary to secondary tube leaks. In case of high radioactivity, this flow path is automatically isolated. This prevents introduction of radioactive fluids into the secondary sampling system.

The staff reviewed the secondary sampling system in accordance with the guidelines of Section 9.3.2 of the SRP. The staff's review included the following areas:

- design objectives
- design criteria
- process streams
- piping and instruments drawings
- locations of sampling points and sample stations

The review indicated that the design and operational procedures for the secondary sampling system meet the SRP guidelines of Section 9.3.2 with regard to GDC 13. The secondary sampling system is acceptable because it meets the following requirements:

- GDC 13, regarding the capability to monitor variables and systems over their anticipated ranges for normal operation and anticipated operational occurrences to ensure adequate safety (by providing early diagnosis of system chemistry excursions in the plant). This is achieved by sampling water in turbine cycle, demineralized water, and water treatment systems.

- GDC 64, regarding the capability to monitor radioactivity that may be released from normal operation, anticipated operational occurrences, and postulated accidents by continuously monitoring steam generator blowdown lines by a process radiation monitor on the common blowdown header.

9.3.5 Equipment and Floor Drainage System

The staff reviewed the equipment and floor drainage system (EFDS) in accordance with the guidance of Section 9.3.3 of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding whether the EFDS satisfies the following requirements:

- GDC 2, as it relates to the capability of safety-related portions of the system to withstand the effects of earthquakes

- GDC 4, as it relates to the capability of the system to withstand the effects of flooding and the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents

- GDC 60, as it relates to providing a means to suitably control the release of radioactive materials in liquid effluent, including anticipated operational occurrences

After the AP600 DSER was issued, Westinghouse provided additional information about the EFDS in the form of SSAR revisions and telephone conferences that has allowed the staff to continue its review as discussed below. As a result of the additional information, DSER Open Item 9.3.5-1, concerning the modification of certain RAI responses, clarification of COL applicant responsibilities, and clarification of flood protection aspects, is closed.

The EFDS consists of the radioactive waste drain system (WRS) and the nonradioactive waste water system (WWS). These systems collect liquid wastes from equipment and floor drains during normal operation, startup, shutdown, and refueling. The liquid wastes are separated according to the type of waste, and are then transferred to appropriate processing and disposal systems. The WWS is discussed in Section 9.2.9 of this report.

The WRS consists of the following equipment:

- equipment drains
- floor drains
- collection piping
- vents
- traps
- cleanouts
- sampling connections
- valves
- collection sumps
- drain tanks
- sump pumps
- drain tank pumps
- discharge piping

The WRS collects radioactive, borated, chemical, and detergent liquid wastes at atmospheric pressure from equipment and floor drainage of the radioactive portions of the auxiliary building, the annex building, the radwaste building, and the containment building. These radioactive liquid wastes are routed to either the auxiliary building sump, the containment sump, or the reactor coolant drain tank. The contents of the sumps and the drain tank are pumped to the WLS for processing. The WRS system description, components, and flow diagrams are provided in Sections 9.3.5 and 11.2, Tables 9.3.5-1, 11.2-2 and 11.2-4, and Figures 9.3.5-1, 11.2-1 and 11.2-2 of the SSAR, respectively.

The containment building contains the passive core cooling system (PXS) PXS-A compartment, the PXS-B compartment, and the CVS compartment. These compartments are physically separated and isolated from each other by structural walls so that flooding in one compartment will not cause flooding in the other compartments. Only the PXS compartments contain safe-shutdown equipment below the maximum flood level. The safe-shutdown-related components of the PXS, located in these two compartments, are redundant and essentially identical. Thus, flooding in one of the compartments will not prevent the PXS from performing its safe shutdown function. Each compartment has a separate and independent floor drain line routed to the containment sump. Any leakage that occurs within containment drains by gravity to the containment sump. In the event a drain is blocked, the individual compartment will flood up to the stairwell curb at Elevation 108'-2", overflow to the maintenance floor, and then drain to the containment sump. Each compartment drain line has two safety-related backflow prevention check valves in series; these prevent reverse flow from the containment sump to the compartments. Having these valves in series ensures that a single failure will not compromise the backflow prevention safety function. Each of these drain lines is monitored by a flow sensor that provides plant operators with an indication of the source of water flow. Flooding in any compartment of the containment is detected by the containment sump-level monitoring system and the containment flood-up level instrumentation. Both of these level detection systems are safety-related. The containment sump-level monitoring system is discussed in Section 5.2.5 of this report, and the flood-up level instrumentation is discussed in Section 3.4.1 of this report.

The auxiliary building consists of a radiologically controlled area (RCA) and a non-radiologically controlled area (non-RCA) that are physically separated by 0.6 m (2 ft) structural walls and 0.9 m (3 ft) floor slabs, so that flooding in the RCA will not cause flooding in the non-RCA. The drain system in the RCA is completely separated from non-RCA drains to prevent cross-contamination of nonradioactive areas. There are no permanent connections between the WRS and nonradioactive piping. However, provisions are included for temporary diversion of contaminated water from normally nonradioactive drains to the WLS. The detection and diversion of radioactive fluids in the nonradioactive WWS is discussed in Section 9.2.9 of this report. Based on the foregoing, the WRS is designed to prevent the inadvertent transfer of contaminated fluids to a noncontaminated drainage system for disposal. On the basis of its review, the staff concludes that the WRS complies with the requirements of GDC 60, with respect to preventing the inadvertent transfer of contaminated fluids to a non-contaminated drainage system for disposal.

As identified in Table 3.2-3 of the SSAR, the WRS components are classified as non-safety-related, non-seismic, Quality Group D, with the following exceptions, which are classified as Safety Class 2 or 3, seismic Category I, Quality Group B or C:

- containment isolation valves in the discharge line from the containment sump and the reactor coolant drain tank

- backflow preventers in the drain lines from containment cavities to the containment sump
- drain line piping from the backflow preventers to the containment cavities

Section 9.3.5.1.1 of the SSAR states that the EFDS is designed to prevent damage to safety-related systems, structures, and equipment. Safety-related components are not damaged as a result of EFDS component failure from a seismic event. Single failures of the EFDS and its equipment will not prevent the proper function of any safety-related equipment. Therefore, the staff concludes that the design presented in the SSAR complies with Regulatory Positions C.1 and C.2 of RG 1.29, and that the WRS complies with the requirements of GDC 2, with respect to the capability to withstand the effects of earthquakes.

Operation of the sump pumps and drain tank pumps are not required to mitigate the consequences of design-basis accidents or flooding events. Section 3.4.1 of this report describes the flood protection aspects of the AP600. Sump pumps inside the containment are interlocked with the associated containment isolation valves. The pumps trip and the isolation valves close on receipt of containment isolation signals to prevent the uncontrollable release of primary coolant outside the containment. Equipment drains are adequately sized to meet the flow requirements. Sump pumps and drain tank pumps discharge at a flow rate adequate to prevent sump overflow for drain rates anticipated during normal plant operation, maintenance, decontamination, fire suppression system testing, and fire fighting activities. Sump and drain tank capacities provide a storage capacity consistent with an operating period of approximately 10 minutes with one pump operating. Plugging of the drain headers is minimized by designing them at least 10.2 cm (4 inches) in diameter, which is large enough to accommodate more than the design flow, and by making the flow path as straight as possible. On the basis of its review, the staff concludes that the WRS complies with the requirements of GDC 4, with respect to the capability to withstand the effects of flooding and the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

Pump operation is automatic with manual override. The pumps are automatically started and stopped by preset high, high-high, and low level instrumentation. Level indication is provided in the main control room for the sump in containment to provide indication of the presence of reactor coolant from unidentified leaks. Operating status of the remaining pumps is provided to the plant control system.

The radioactive sump vents outside containment are directed to the ventilation system exhaust ducts serving the areas where the sumps are located. (See Section 9.4.7 of this report for a discussion of how the ventilation system exhaust is monitored and filtered). The containment sump vents directly to the containment. The reactor coolant drain tank is vented to the gaseous radwaste system. Inlet radioactive drain lines to the sumps or drain tank are kept submerged a minimum of 15.2 cm (6 inches) below pump shutoff level to prevent backgassing. The containment sump inlet is not submerged.

On the basis of the above review, the staff concludes that the WRS complies with GDC 2, 4, and 60 as referenced in Section 9.3.3 of the SRP, and is, therefore, acceptable.

9.3.6 Chemical and Volume Control System

The CVS in the AP600 design is a non-safety-related system, and its operation is not required to mitigate design-basis events. Section 9.3.6.1.2 of the SSAR describes the following non-safety-related functions performed by the CVS:

- Purification: The CVS removes radioactive corrosion products, ionic fission products, and fission gases from the RCS to maintain RCS fluid purity and an activity level within the acceptable technical specification limits.
- RCS Inventory Control and Makeup: The CVS provides a means to add and remove mass from the RCS, as required, to maintain the programmed inventory during normal plant operations.
- Chemical Shim and Chemical Control: The CVS provides the means to vary the boron concentration in the RCS.
- pH Control: The CVS maintains the proper pH in the reactor coolant.
- Oxygen Control: The CVS maintains the proper level of dissolved hydrogen in the reactor coolant during power operation and achieves the proper oxygen level before startup after each shutdown.
- Filling and Pressure Testing the RCS: The CVS provides a means for filling and pressure testing the RCS. The CVS also provides connections for a temporary hydrotest pump.
- Borated Makeup: The CVS provides makeup to the PXS accumulators, CMTs, IRWST, and spent fuel pit at various boron concentrations.
- Pressurizer Auxiliary Spray: The CVS provides pressurizer auxiliary spray water for depressurization.

The CVS also performs the following safety-related functions:

- containment isolation of the CVS lines penetrating containment
- termination of inadvertent RCS boron dilution
- isolation of makeup on a steam generator or pressurizer high level signal
- preservation of the RCS pressure boundary, including isolation of normal CVS letdown from the RCS

During an accident, the CVS is not required to provide emergency core cooling or boration. However, the makeup pumps are available to furnish pressurizer auxiliary spray as an additional means to improve the reliability of the plant. The reliability of the CVS itself is enhanced by the use of redundant equipment.

In Section 9.3.6.2 of the SSAR, Westinghouse describes the CVS design, and Figure 9.3.6-1 of the SSAR shows the detailed P&ID. The CVS consists of regenerative and letdown heat exchangers; demineralizers and filters; makeup pumps; tanks; and associated valves, piping, and instrumentation. The CVS purification loop is located entirely inside the containment, and operates at RCS pressure in a closed loop without the CVS makeup pumps, using the developed head of the reactor coolant pumps as the motive force for the purification flow. The primary coolant is drawn at the discharge of the reactor coolant pump, passes through the regenerative and letdown heat exchangers, where it is cooled to the temperature compatible with the resin in the demineralizer. The coolant then passes through one of the two demineralizers containing mixed bed resin. The other demineralizer serves as a backup, in case the active unit should become exhausted during operation. The demineralizer removes ionic corrosion and certain fission products, and also acts as a filter. Coolant can also be directed to the demineralizer containing cation resin, which removes mostly lithium and cesium isotopes.

After passing through the demineralizers, the coolant is directed through the secondary side of the regenerative heat exchanger back to the primary loop. During plant shutdown, when the pumps are not operating, the normal RHR system provides the motive force for the purification loop. The coolant is routed through the WLS degasifier when degasification of the coolant is required because of high fuel defects or during shutdown operation.

The CVS has enough capacity to accommodate minor leakage from the reactor coolant system, and provides inventory control during plant heatups and cooldowns. Inventory control is achieved through the letdown and makeup connections to the CVS purification loop. Two centrifugal makeup pumps transfer water from the boric acid tank, which blends it with the water from the demineralized water tank. The concentration of boron in the makeup water is controlled by the position of a three-way control valve. To reduce coolant inventory, water is let out from the coolant system through the purification loop. The water is depressurized by flowing through the letdown orifice, and is passed to the liquid waste processing system. In addition to controlling coolant inventory in the primary coolant system, the CVS is used to provide borated water for passive core cooling system accumulators, CMTs, and the spent fuel pool. It is also used for filling and pressure testing of the reactor coolant system after maintenance and refueling.

The makeup system, including the makeup pumps and suction line piping, are all outside the containment. Because the letdown and makeup system interface with the RCS and have components outside the containment, all low-pressure portions outside the containment should be designed with an ultimate rupture strength at least equal to the RCS operating pressure. The CVS is evaluated under Generic Issue 105, "Interfacing System Loss-of-Coolant-Accident at LWRs," which is discussed in Chapter 20 of this report.

Control of pH is achieved by injecting lithium hydroxide from the chemical mixing tank into the makeup water. Concentration of boron is controlled by adding 2.5 percent boric acid solution from the boric acid tank to the makeup water. During startup from cold shutdown conditions, oxygen is controlled by introducing hydrazine through the chemical mixing tank and, during power operation, by injecting high pressure hydrogen through a separate containment penetration into the fluid returning from the CVS purification loop. The bottles containing compressed hydrogen will be stored at least 15 m (50 ft) from any building containing safety-related equipment. In the DSER, the staff requested that Westinghouse ensure that the total amount of stored hydrogen does not exceed the quantity which, when accidentally released, could cause an explosion large enough to cause damage to the safety-related equipment. This was DSER Open Item 9.3.6-1. Westinghouse revised its SSAR Section 3.5.1.1.2.2 to state that one tank at a time will be connected to the system at any time. This will ensure that, if a small quantity of hydrogen (i.e., the amount in one bottle)is accidentally released, it will not cause an explosion large enough to damage safety-related equipment. This is acceptable and closes Open Item 9.3.6-1

The CVS is a non-safety-related system, and its operation is not required to mitigate the design-basis events. Therefore, the CVS is not required to meet the safety-related system requirements. However, the CVS is a DID system that provides the first line of defense during an accident to prevent unnecessary actuation of passive core cooling systems. Regulatory oversight of the active non-safety-related system is subject to an evaluation using the RTNSS process described in SECY-94-084. In September 1993, Westinghouse provided a summary report entitled, "AP600 Implementation of the Regulatory Treatment of Non-Safety-Related System Process." Regulatory treatment of the CVS and other non-safety-related systems was Open Item 9.3.6-2. This is a generic issue and the closure of this issue is addressed in Section 20 of this report. Open Item 9.3.6-2 is closed.

The staff reviewed the CVS in accordance with Section 9.3.4 of the SRP. The review indicated that the description in the SSAR did not provide enough information for the staff to complete its evaluation of the design and operation of the system. In the DSER, the staff requested that Westinghouse should address the following areas concerning the CVS (paragraph numbers in parentheses are from Section 9.3.4 of the SRP):

- effect of adverse environmental occurrences, and abnormal and accident conditions (I.A.1)
- malfunction of components needed for performing safety-related functions (I.A.2)
- seismic design requirements (I.A.3)
- instrumentation and control for verifying operability of the system (I.A.5)
- radiation protection, instrumentation, design features to withstand natural phenomena, materials and design codes for the components, and inservice testing (I.B)
- design provisions of the CVS to withstand earthquakes (II.B)
- provisions of the CVS components for venting and draining through closed systems (II.G)
- monitoring temperature and differential pressure in demineralizers (II.1a and II.1b)
- leakage from makeup and letdown lines (II.3)
- limits of performance for degraded components (III.A)

- adequacy of the CVS for control of water chemistry (III.A.5)

- performance of the system under adverse conditions (III.B.1 and III.B.2)

Westinghouse responded to this request by revising its SSAR, Section 9.3.6.5, Design Evaluation. Revision 7 of the SSAR indicated that the design of the CVS is based on specific General Design Criteria (GDC) and regulatory guides specified in subsection 9.3.4 "Chemical and Volume Control System (PWR) (Including Boron Recovery System," Revision 2, of the Standard Review Plan (SRP). The specific GDCs identified in the SRP are GDCs 1, 2, 3, 4, 14, 29, 30, 31, 32, 33, 53, 56, and 61. Although the AP600 CVS is not a safety related system and its design need not strictly adhere to the criteria listed in the SRP, Westinghouse compared the AP600 CVS design to the requirements of the GDCs and concluded that it meets these requirements which were discussed for the AP600 in Section 3.1 of the SSAR. On that basis Open Item 9.3.6-3 is closed.

9.4 Air-Conditioning, Heating, Cooling, and Ventilation System

In Section 3.1.1 of the SSAR, Westinghouse states that the AP600 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 HVAC design described in the SSAR does not share SSCs with other nuclear power units. Therefore, the air conditioning, heating, cooling and ventilation systems meet the requirements of GDC 5. Table 9.4-1 lists the standards to which the various components of the HVAC system are designed.

The staff's review of the AP600 air-conditioning, heating, cooling, and ventilation systems is provided in the following Sections: 9.4.1, Nuclear Island Nonradioactive Ventilation System; 9.4.2, Annex/Auxiliary Buildings Non-Radioactive HVAC System; 9.4.3, Radiologically Controlled Area Ventilation System; 9.4.4, Balance of Plant Interfaces; 9.4.5, Engineered Safety Features - Ventilation System; 9.4.6, Containment Recirculation Cooling System; 9.4.7, Containment Air Filtration System; 9.4.8, Radwaste Building HVAC System; 9.4.9, Turbine Building Ventilation System; 9.4.10, Diesel Generator Building Heating and Ventilation System; and 9.4.11, Health Physics and Hot Machine Shop HVAC System.

The staff's review of SSAR Section 9.4, "Air-Conditioning, Heating, Cooling, and Ventilation System" was based on in part Westinghouse proprietary figures that are not part of the SSAR. The staff requested a complete set of the AP600 proprietary figures for SSAR Section 9.4 in order to review any changes from the set the staff used to review the design. This was Confirmatory Item 9.4-1. Westinghouse provided the requested figures by letter dated May 6, 1998. Therefore, Confirmatory Item 9.4-1 is closed.

9.4.1 Nuclear Island Nonradioactive Ventilation System

The staff reviewed the nuclear island nonradioactive ventilation system (VBS) in accordance with SRP Section 9.4.1, "Control Room Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VBS satisfies the following requirements:

- GDC 2, regarding the capability to withstand earthquakes
- GDC 4, regarding maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions
- GDC 5, regarding sharing 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 AP600 design. The staff used a dose criterion of 0.05 Sv (5 rem) TEDE for evaluating the control room radiological consequences resulting from DBAs, pursuant to GDC 19 of Appendix A to 10 CFR Part 50. The justification for the use of this dose criterion and the associated exemption from the regulation is provided in Section 15.3 of this report
- GDC 60, regarding the capability to suitably control release of gaseous radioactive effluent to the environment

Westinghouse provided additional information about the VBS that allowed the staff to continue its review as discussed below. As a result of the additional information concerning the modifications of some responses to RAIs, clarifications on COL applicant responsibilities, conformance with RGs, discrepancies in the SSAR, technical specifications and testing surveillances, and design requirements for habitability, DSER Open Item 9.4.1-1 is closed.

The VBS provides safety-related design basis functions to (1) monitor air supply for radioactive particulate and iodine concentrations inside the main control room envelope (MCRE), and (2) isolate the safety-related, seismic Category I HVAC piping penetrating the MCRE based upon "high-high" particulate or iodine radioactivity detection in the supplied air. The system is designed to maintain proper environmental conditions and control contaminant levels. The VBS maintains the main control room (MCR) and technical support center (TSC) carbon dioxide levels below 0.5 percent concentration and the air quality within the guidelines of Table 1 and Appendix C, Table C-1 of American Society of Heating, Refrigerating, and Air-conditioning Engineers (ASHRAE) Standard 62. Westinghouse states that the VBS is non-safety-related; however, if the system is operational and ac power is available, the system provides for habitability inside the main control room envelope within the guideline of SRP 6.4, and the technical support center within the guideline of NUREG-0696, "Functional Criteria for Emergency Response Facilities," because the filtration subsystem is designed, constructed, and tested to conform with Generic Safety Issue B-36, RG 1.140 as discussed in SSAR Chapter 1.0, Appendix 1A, and ASME N510-1989. Generic Safety Issue B-36, "Develop, Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems," is discussed in Chapter 20 of this report. In addition, Westinghouse stated in Section 9.4.1.13 of the SSAR that COL applicants referencing the AP600 design will implement a program to maintain compliance with ASME N509-1989, ASME N510-1989, and RG 1.140 for portions of the VBS and VFS identified in SSAR Sections 9.4.1 and 9.4.7. The staff finds this acceptable.

For the post 72-hour design basis, the specific function the nuclear island nonradioactive ventilation system is to maintain the MCR below a temperature approximately 2.5 °C (4.5 °F) above the average outdoor temperature. For Divisions B and C instrumentation and control rooms post 72-hour design basis, the specific function of the VBS is to maintain the I&C rooms below the qualification temperature of the instrumentation and control equipment (49 °C(120 °F)). The ancillary fans are intended to meet the above post 72-hours ventilation criteria for the MCR and Class 1E I&C rooms. The staff's evaluation of the post 72-hours power supply is discussed in Section 8.3 of this report.

The VBS consists of the following subsystems:

- The main control room/technical support center HVAC subsystem serves the main control room and technical support center.

- The Class 1E electrical room HVAC subsystem serves the Class 1E dc equipment rooms, electrical penetration rooms, battery rooms, and I&C rooms, remote shutdown area, reactor cooling pump trip switchgear rooms, and adjacent corridors.

- The passive containment cooling system valve room heating and ventilating subsystem serves the passive containment cooling system valve room.

Descriptions, design parameters, instrumentation (including indications and alarms), and figures for the VBS and the interfacing MCR emergency habitability system (VES) are provided in Sections 6.4, 9.4.1, and 15.6.5.3; Tables 3.2-1, 6.4-1 through 6.4-3, 9.4.1-1, and 15.6.5-2; and Figures 1.2-8, 6.4-1, 6.4-2, and 9.4.1-1 of the SSAR, respectively. Instrumentation for the VES and VBS are discussed in Section 7.3 of the SSAR. Details of radiation monitors, including testing and inspection, are provided in Section 11.5 of the SSAR. The staff's evaluation of the chilled water system is discussed in Section 9.2.7 of this report. The staff's evaluation of fire protection is discussed in Section 9.5.1 of this report. Table 9.4-1 of this report describes the industry standards applicable to the HVAC system, including components of the VBS.

The MCRE penetrations include isolation valves, interconnecting piping, and vent and test connections with manual valves that are classified a safety Class C and seismic Category I. The MCRE isolation valves have electro-hydraulic operators and are designed to fail closed during a LOOP event. The safety-related isolation valves are included in the technical specifications for periodic testing and are also included in the inservice testing (IST) program.

The MCR/TSC HVAC subsystem filtration unit configurations, including housing, internal components, ductwork, dampers, fans and controls, and the location of the fans on the filtered side of units are designed, constructed, and tested in accordance with ASME N509-1989, ASME N510-1989, and RG 1.140-1979, Revision 1, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." The ductwork for the supplemental air filtration subsystem and portions of the MCR/TSC HVAC subsystem that maintains the integrity of the MCR/TSC pressure boundary during conditions of abnormal airborne radioactivity is tested for leak tightness in accordance with ASME N510-1989.

The remaining supply and return/exhaust ductwork is tested in place for leakage in accordance with Sheet Metal and Air-conditioning National Association (SMACNA), 1983, "HVAC Duct Leakage Test Manual." The high-efficiency particulate air (HEPA) filters are shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol and constructed, qualified, and tested in accordance with ASME N509-1989 and UL-586, 1985 "High-Efficiency, Particular, Air-Filter Units." Post filters downstream of the charcoal adsorbers have a minimum dioctyl-phthalate polydispersed (DOP) test efficiency of 95 percent. Each charcoal adsorber is a single assembly with welded construction and 100 mm (4-inch) deep type III rechargeable adsorber cell, conforming with IE Bulletin 80-03, "Loss of Charcoal from Adsorber Cells," and qualified, constructed, and tested in accordance with ASME N509-1989, ASME N510-1989, and RG 1.140-1979, Revision 1. A representative charcoal sample, used or new, is laboratory tested to verify a minimum charcoal efficiency of 90 percent in accordance with RG 1.140 and conforming to ASME N510-1989 for test procedures and test frequency.

The system ductwork flow is tested, balanced and adjusted in accordance with SMACNA, 1985, "Testing, Adjusting, and Balancing." Fire dampers or combination fire/smoke dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The MCR/TSC HVAC and Class 1E electrical room HVAC subsystems are designed so that smoke, hot gases, and fire suppressant does not migrate from one fire area to another to the extent that they could adversely affect the safe shutdown capabilities, including operator actions. Fire or combination fire and smoke dampers are provided for MCRE areas, Class 1E equipment rooms, and the remote shutdown workstation room to isolate each fire area from adjacent fire areas during and following a fire in accordance with the NFPA 90A, 1993, "Installation of Air Conditioning and Ventilation Systems."

If the VBS is not available during the 72-hour period following the onset of a postulated design-basis accident, the VES provides passive heat sinks to limit the temperature rise in the MCRE, I&C rooms, and dc equipment rooms. The heat sinks consist primarily of the thermal mass of the concrete that makes up the ceilings and walls of these rooms. As described in SAR Section 6.4.2.2, to enhance the heat absorbing capacity of the ceilings, a metal form is attached to the surface of the concrete at selected locations. 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 VBS cooling and heating capacity is dependent on the site interface parameters for maximum and minimum normal temperature conditions, as defined in SSAR Chapter 2, Table 2-1 and summarized as follows:

- The MCR/TSC HVAC subsystem maintains the MCR and TSC between 19.4 to 23.9 °C (67 to 75 °F) and 25 percent to 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 to 23.9 °C (67 to 75 °F); 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 to 22.8 °C (67 to 73 °F); and HVAC equipment rooms between 10 to 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).

- The VBS maintains the Class 1 battery rooms to limit the hydrogen gas concentration to less than 2 percent by volume.

- The passive containment cooling system (PCS) valve room heating and ventilation subsystem maintains the PCS valve room at 10 °C to 48.9 °C (50 to 120 °F).

The single outside air intake serving the VBS conforms with the guidance of Section 6.4 of the SRP and RG 1.95. The outside air intake that is protected by an intake enclosure located on the roof of the auxiliary building at Elevation 153'-0". As stated in SSAR Section 6.4.4, the fresh air intake of the MCR is located in excess of 45.7 m (150 ft) from the flue gas exhaust stacks of the onsite standby power diesel generators, and 91.4 m (300 ft) from the onsite standby power system fuel oil storage tanks. This location precludes the combustion fumes or smoke from an oil fire from being drawn into the MCR. It is located more than 15.2 m (50 ft) below and more than 30.5 m (100 ft) laterally away from the plant vent discharge. The split-wing type tornado protection dampers close automatically and can withstand the effects of 134 m/s (300 mph) wind.

As shown in SSAR Figure 9.4.1-1, the fresh air supply from an air intake is automatically isolatable by a fail-closed, electro-hydraulically operated isolation damper at the inlet of each air filtration train. Normally, one VBS air filtration unit train isolation damper is opened, and the other air filtration unit train isolation damper is closed. There are two fail-closed isolation dampers in series in the common outside air supply to each of the normal air handling units. The radiation monitors and outside air isolation dampers are shown on Figure 9.4.1-1 of the SSAR. The outside air is continuously monitored by redundant smoke monitors at the outside air intake. Redundant safety-related radiation monitors are located in the MCRE upstream of the supply air isolation valves. As described in SSAR Section 9.4.1.2.3.1, these monitors initiate operation of the non-safety-related supplemental air filtration units on “high” gaseous radioactivity concentrations and isolate the MCR from the VBS on “high-high” particulate or iodine radioactivity concentrations.

In Section 9.4.12 of the SSAR, Westinghouse states that the COL applicant will provide a description of the MCR/TSC HVAC subsystem's recirculation mode during emergencies involving toxic substances, and how the subsystem equipment isolates and operates, as applicable, consistent with the issues regarding toxic substances to be addressed by the COL applicant, as discussed in SSAR Section 6.4.7.

Portions of the VBS that provide the DID function of filtration of MCR/TSC air during conditions of abnormal airborne radioactivity are designed, constructed, and tested to conform with GI B-36, RG 1.140, and ASME N509 and N510 standards. The staff reviewed the VBS in accordance with its status as a non-safety-related system identified as important by the RTNSS process, or as a DID system in accordance with SECY-94-084. System redundancy for the MCR/TSC HVAC subsystem is provided and the system is automatically transferred to the onsite non-safety-related diesel generators on LOOP. The equipment is located in separate fire areas and high in the building to protect it from flooding. The system quality assurance, availability, and administrative controls are addressed in SSAR Chapters 17, 16, and 13. The

VBS is located in the auxiliary building and equipment will be procured to manufacturer's standards, and as with the other equipment of the nuclear island, it is protected from defined natural phenomena.

The VBS is controlled by the plant control system, except for the MCRE isolation valves, which are controlled by the protection and safety monitoring system. The plant control and plant safety and monitoring systems are discussed in Section 7.1.1 of this report. The instrumentation to satisfy Table 4-2 of ASME N509-1989 is discussed in SSAR Section 9.4.1.5 for the DID VBS supplemental air filtration units. Radioactivity indication and alarms are provided to inform the MCR operators of gaseous, particulate, and iodine radioactivity concentrations in the MCR supply air duct. In Section 11.5 of the SSAR, Westinghouse provides the description of the MCR supply air duct radiation monitors and their actuation functions. Smoke monitors are provided to detect smoke in the outside air intake duct to the MCR and the MCR and Class 1E electrical room return air ducts. Temperature indications and alarms are provided in the return air ducts to control the room air temperatures within the predetermined range. Temperature indications and alarms for the MCR return air, Class 1E electrical return room air, air handling unit (AHU) supply air, supplemental filtration unit prefilter inlet air, and charcoal adsorbers are provided to inform plant operators of abnormal temperature conditions. Pressure differential indications and alarms are provided to control the MCR and monitor the TSC ambient pressure differentials with respect to the surrounding areas. Airflow indication and alarms are provided to monitor operation of the supply and exhaust fans.

9.4.1.1 Main Control Room/Technical Support Center HVAC Subsystem

The MCR/TSC subsystem serves: (1) the MCRE as shown in SSAR Figure 6.4-1, which consists of the main control area, shift supervisor office, tagging room, toilet (area), clerk room, kitchen/ operator area, hallway and double door vestibule; and (2) the TSC areas consisting of the main TSC operating area, conference rooms, NRC room, computer rooms, shift turnover room, kitchen/rest area, and rest rooms. The MCR and TSC toilets each have a separate exhaust fan.

The MCR/TSC subsystem consists of redundant 100-percent capacity supply AHUs, supplemental air filtration units, return/exhaust air fans, associated dampers, instrumentation and controls, and common ducts for the MCR and TSC. Each supply AHU consists of a mixing box section, supply air fan, a low-efficiency filter bank, a high-efficiency filter bank, an electric heating coil, chilled water cooling coil bank, and a humidifier. The supply of the chilled water is normally provided from air-cooled chillers in the VWS.

The supply AHUs and return/exhaust air fans are connected to a common duct that distributes air to the MCR/TSC HVAC subsystem. The only HVAC penetrations in the MCRE are MCR supply, return, and toilet exhaust ducts and include redundant safety-related seismic Category I isolation valves that are physically located in the MCR envelope. The isolation valves isolate the non-safety-related portions of the subsystem from the MCRE when the VES is operating.

The normal outside makeup air is provided to the subsystem through an outside air intake duct that is protected by a non-seismic Category I intake enclosure. Westinghouse states that the non-seismic Category I enclosure is acceptable because failure of the VBS air intake enclosure will not affect the safety-related operation of the VES, including the initial pressure assumptions required by the VES to maintain the control room habitability during a design-basis LOCA. The outside supply air intake enclosure for the MCR/TSC HVAC subsystem is shared by the A and C Class 1E electrical room HVAC subsystem. The staff agrees with Westinghouse's justification for a non seismic Category I intake enclosure.

The tempered air through each AHU is controlled by temperature sensors located in the MCR return air duct to maintain the ambient air design temperature within its normal design temperature range by modulating electric heating or chilled water cooling coil.

Each supplemental air filtration unit includes a high-efficiency filter bank, an electric heating coil, charcoal adsorber with upstream HEPA filter bank and a downstream bank, and a fan. Both redundant trains of supplemental filtration units and one train of supply AHU are located in the MCR mechanical equipment room at Elevation 135'-3" of the auxiliary building. The other supply AHU is located in the MCR mechanical equipment room at Elevation 135'-3" of the annex building. The MCR toilet exhaust fan is located at Elevation 135'-3" of the auxiliary building. The filtration unit's housings are located outside the MCRE, designed to meet the performance requirements of ASME N509 and N510 standards, and operated at a negative pressure.

In SSAR Section 9.4.1, Figure 9.4.1-1, and Tables 9.4.-1 and 9.4.1-1, Westinghouse indicates that the non-safety-related MCR/TSC HVAC subsystem, with single intake and common supply and return ducting for the MCRE and TSC areas, provides a capability similar to that of the engineered safety features (ESF) systems in operating plants with respect to air filtration and adsorption capability. In Table 9.4.1-1 of the SSAR, Westinghouse shows that the depth of the activated charcoal adsorber is 102 mm (4 inches) with an adsorber efficiency of 90 percent, HEPA filter efficiency of 99 percent, and maximum in- leakage of 0.0658 m^3/sec (140 scfm) based on SSAR Table 9.4.1-1 [in-leakages of 0.0047 m^3/sec (10 scfm) through MCR access doors, 0.0047 m^3/sec (10 scfm) through TSC access doors, and 0564 m^3/sec (120 scfm) through MCR/TSC HVAC equipment and ductwork (operating)]. The MCR/TSC HVAC equipment ductwork that forms an extension of the MCR/TSC pressure boundary limits the overall infiltration (negative operating pressure) and exfiltration (positive operating pressure) rates to those values shown in SSAR Table 9.4.1-1 to maintain operator doses within the allowable GDC 19 limits as applied to the AP600 design.

During normal operation, one of the two 100-percent capacity supply AHUs and supply/exhaust air fans operate continuously. Outside makeup air to supply AHUs is provided through an air intake duct. The outside air flow rate is automatically controlled to maintain the MCR/TSC areas at a slightly positive pressure with respect to the surrounding areas and outside environment. The standby AHU and its corresponding return/exhaust fans start automatically if (1) the operating fan air flow drops below predetermined setpoints, (2) return air temperature is above or drops below predetermined setpoints, (3) differential pressure between the MCR and the surrounding areas and outside environment is above or below predetermined setpoints, or (4) electrical and/or control power to the operating unit is lost.

Westinghouse described the design and operation of the MCR/TSC HVAC subsystem in SSAR Section 9.4.1.2.3.1. During abnormal plant operation with high gaseous radioactivity detected in the MCR supply air duct, the system is designed to maintain control room operator doses within the dose acceptance criteria of GDC 19 as applied to the AP600 design. When high gaseous radioactivity is detected in the MCR supply air duct and the MCR/TSC HVAC subsystem is operable, both supplemental air filtration units automatically start to pressurize the MCR/TSC areas to at least 0.03 kPa (1/8" water gauge) using filtered makeup. 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 close. The smoke/purge isolation dampers close, if open. The subsystem AHU continues to provide cooling in the recirculation mode by maintaining the MCRE passive heat sink below its initial ambient air design temperature and maintaining the MCR/TSC areas within their design temperature. The supplemental filtration pressurizes the combined volume of the MCR and TSC concurrently with filtered air. A portion of the recirculated air (approximately, 4,000 cfm) from the MCR and TSC is also filtered for cleanup of airborne radioactivity.

To verify Westinghouse's assertion, the staff performed independent radiological consequence calculations for personnel in the MCR and TSC following a design-basis LOCA. The staff finds that the system design, as bounded by the control room atmospheric relative concentrations proposed by Westinghouse, is capable of controlling radioactivity following a design basis LOCA to meet the dose criteria specified in GDC 19 as applied to the AP600 design. However, the system is not designed as a post-accident engineered safety-feature atmospheric cleanup system. The VBS has no safety grade source of power; therefore, it was not credited in evaluating conformance with GDC 19 as applied to the AP600 design. The major assumptions used by the staff and the resulting radiological consequence analyses are provided in Table 15.3-9 of this report.

During abnormal operation, if ac power is unavailable for more than ten minutes or "high-high" particulate or iodine radioactivity is detected in the MCR supply air duct, which would lead to exceeding operator dose limits of GDC 19 as applied to the AP600 design, 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 MCR occupants from a potential radiation release, because the radiation monitors are effective only when there is air flow through the VBS ductwork. Section 6.4 of this report discusses the emergency mode of operation.

The MCR/TSC subsystem complies with GDC 60, as it relates to protecting those who access the control room during accidental radioactive releases by initiating the supplemental filtration subsystem, which conforms with RG 1.140, regarding detection of high radioactivity or isolating the MCRE and initiating the VES on detection of high-high airborne radioactivity through the redundant nuclear safety-related radiation monitors. The MCR and TSC areas ventilation supply and return/exhaust air ducts can be manually isolated from the MCR.

If a high concentration of smoke is detected in the outside air intake, an alarm is initiated in the MCR. The subsystem is then manually realigned to the recirculation mode by closing the outside air and toilet exhaust duct isolation dampers. The MCR and TSC toilet/kitchen exhaust fans are tripped upon closure of the isolation valves. During the recirculation mode, the MCR/TSC areas are not pressurized. The subsystem continues to provide cooling and ventilation to maintain the emergency passive heat sink below its initial ambient air design temperature and maintains the MCR/TSC areas within their design temperature.

In the event of a fire in the MCR/TSC, the fire/smoke dampers close automatically to isolate the fire area, while the unaffected areas are maintained at a slight positive pressure by the subsystem. The subsystem continues to provide ventilation and cooling to the unaffected areas and maintain them at a slight positive pressure. The subsystem can be realigned manually to the once-through ventilation mode to supply 100 percent outside air to the unaffected areas.

Power is supplied to the subsystem by the plant ac electrical system. In the event of a LOOP when the plant ac electrical system is unavailable, the subsystem is automatically transferred to the onsite standby diesel generators.

In the event that complete ac power is lost and the outside air is acceptable on the basis of compliance with the requirements of GDC 19 as applied to the AP600 design concerning the MCR operator doses and toxic releases, MCRE habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCRE. The outside air supply pathways to the ancillary fans and warm air vent pathways are described in SSAR Section 9.4.1.2.3.1. Power to the ancillary fans is from the respective division B or C regulating transformers, which receive power from the ancillary diesel generators. The ancillary fans' flow paths are located within the auxiliary building which is a seismic Category I structure. Once the normal ventilation is restored, the ancillary fan circuits are disabled manually. Westinghouse states that the ancillary fans are centrifugal type with non-overloading horse power characteristics; conform to ANSI/AMCA 210, 211, and 300 standards; and each fan can provide a minimum of 1530 cfm. The capacity and air flow rate maintain the MCRE environment near the daily average outdoor air temperature. As discussed in Section 22.5.4.3 of this report, short-term administrative controls are provided for the MCR ancillary fans as part of the RTNSS process.

9.4.1.2 Class 1E Electrical Room HVAC Subsystem

The Class 1E electrical room (ER) HVAC subsystem has two ventilation trains; one serves the A and C electrical divisions, spare battery rooms (non-Class 1E), Class 1E spare battery rooms, and reactor pump trip switchgear rooms, and the other serves the B and D electrical divisions and the remote shutdown work station area.

Each subsystem consists of two 100-percent capacity AHUs, return/smoke exhaust air fans, associated dampers, instrumentation and controls, and common ductwork. The AHUs and return/exhaust fans are connected to a common duct that distributes supply air to the Class 1E electrical rooms. Each supply AHU consists of a mixing box section, supply air fan, a low-efficiency filter bank, a high-efficiency filter bank, an electric heating coil, and chilled water cooling coil bank. The supply of the chilled water is normally provided from air-cooled chillers in the VWS. Additionally, the Class 1E battery rooms are provided with duct-mounted electric heating coils and two 100-percent capacity exhaust fans. The HVAC equipment serving the A and C electrical divisions is located in the MCR/A and C equipment room at Elevation 135'-3" of the auxiliary building. The HVAC equipment serving the B and D electrical divisions is located in the upper and lower B and D equipment rooms at Elevation 117'-0" and Elevation 135'-3" of the auxiliary building.

During normal operation, one of the redundant supply AHUs, return fans, and battery room exhaust fans operate continuously to maintain acceptable environmental conditions, maintain the Class 1E electrical room emergency passive heat sink below its initial ambient air temperature, and prevent hydrogen gas buildup in the Class 1E battery rooms. The battery exhaust is vented directly to the turbine building vent to limit the hydrogen gas concentration to less than 2 percent by volume in accordance with RG 1.128, Revision 1, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants."

The normal outside makeup air is provided to the subsystem through an outside air intake duct that is protected by an intake enclosure. The outside supply air intake enclosure for the A and C Class 1E ER HVAC subsystem is common for the MCR/TSC HVAC subsystem air intake located on the roof of the auxiliary building at Elevation 153'-0". The outside supply air intake for the B and D Class 1E ER HVAC subsystem is located and separated from the MCR/TSC HVAC subsystem air intake enclosure on the roof of the auxiliary building at Elevation 153'-0".

The tempered air through each AHU is controlled by temperature sensors located in the return air duct to maintain the room air temperature within the normal design range by modulating electric heating or the chilled water cooling coil. The standby supply AHU and corresponding return/smoke exhaust fans are started automatically if the operating fan air flow drops below a predetermined set point, the return air temperature is above or drops below predetermined setpoints, or the electrical and/or control power is lost to the operating unit.

The operation of the Class 1E ER HVAC subsystem is not affected by abnormal events resulting in detection of airborne radioactivity in the MCR supply air duct of the MCR/TSC HVAC subsystem. During a design-basis accident (DBA), if both onsite and offsite power are lost, the Class 1E electrical room emergency passive heat sink will provide area temperature control as discussed in Section 6.4 of this report.

If a high concentration of smoke is detected in the air intake, an alarm is initiated in the MCR. The Class 1E ER HVAC subsystem is then manually realigned to the recirculation mode of operation by closing the outside air intake damper to the AHU mixing plenum, allowing 100 percent room air to return to the supply air subsystem AHU. During the recirculation mode of operation, the subsystem continues to maintain the served areas within their air design temperatures and pressures.

In the event of a fire in a Class 1E electrical room, fire/smoke dampers close automatically to isolate the fire area and the unaffected areas are maintained at a slight positive pressure by the subsystem. One or both trains of the subsystem can be manually realigned to the once-through ventilation mode to provide 100 percent outside air to the unaffected areas.

Realignment to the once-through ventilation mode minimizes the potential for migration of smoke and hot gases from a non-Class 1E electrical room or a non-Class 1E electrical room of one division into the Class 1E electrical room of another division. Smoke and hot gases can be removed from the affected areas by reopening the closed combination fire/smoke dampers from outside of the affected fire area during the once-through ventilation mode. In the once-through ventilation mode, the outside air intake damper to the AHU mixing plenum opens and the return air damper to the supply AHU closes to allow 100 percent outside air to the supply AHU. The subsystem exhaust air isolation damper also opens to exhaust room air directly to the turbine building vent.

Power is supplied to the subsystem by the plant ac electrical system. In the event of a LOOP and the plant's ac electrical system is unavailable, the subsystem is automatically transferred to the onsite standby diesel generators.

When complete ac power is lost, division B and C I&C room temperature is maintained by operating their respective MCR ancillary fans to supply outside air to the I&C rooms. The outside air supply pathways to the ancillary fans and warm air vent pathways are described in SSAR Section 9.4.1.2.3.2. Power supply to the ancillary fans is from the respective division B or C regulating transformers, which receive power from the ancillary diesel generators. The ancillary fans flow path are located within the auxiliary building which is a seismic Category I structure. Once the normal ventilation is restored, the ancillary fan circuits are disabled manually. As discussed in Section 22.5.4.3 of this report, short-term administrative controls are provided for the MCR ancillary fans as part of the RTNSS process.

9.4.1.3 Passive Containment Cooling System Valve Room Heating and Ventilating Subsystem

The passive containment cooling system valve room heating and ventilation subsystem consists of one 100-percent capacity exhaust fan, two 100-percent capacity electric unit heaters, and associated dampers, instrumentation, and controls. The subsystem equipment is located in the PCS valve room in the containment dome area at Elevation 266'-0".

During normal operation, the exhaust fan draws outside air through an intake louver damper and directly exhausts it to the environment to maintain room temperature within its normal design temperature range. The lead electric unit heater starts or stops when the room air temperature is above or drops below predetermined setpoints. The standby electric unit heater starts automatically if the airflow temperature of the operating electric unit heater drops below a predetermined setpoint.

The exhaust fan and electric heaters are powered by the plant ac electrical system and, in the event of a LOOP, the power source is automatically transferred to the onsite standby diesel generators for the electric unit heaters. Following a fire in the PCS valve room, smoke and hot gases can be removed from the area using portable exhaust fans and flexible ductwork.

9.4.1.4 Conclusion

The VBS is located inside seismic Category I (auxiliary building) and seismic Category II (annex building) buildings, which provide flood and tornado-missile protection. The safety-related MCR isolation dampers are seismic Category I, as shown in Table 3.2-1 of the SSAR. Therefore, the system's safety-related portions comply with the guidelines of Position C.1 of RG 1.29. The system's non-safety-related portions comply with Position C.2 of RG 1.29, because the tornado damper installed at the outside air intake and the MCR fire dampers are designed to meet seismic Category II requirements, so that the failure of system components during an SSE will not reduce the functioning of any safety-related plant features. System equipment and ductwork located in the nuclear island the failure of which could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portions of the system are non-seismic. Therefore, the system complies with GDC 2 requirements, as it relates to protection of the system against natural phenomena.

Redundant safety-related components of the MCR/TSC HVAC subsystem are physically separated, and are protected from internally-generated missiles, pipe breaks, and water spray. All safety-related components are seismic Category I, and designed to function following an SSE. The system design maintains its function with the loss of any single active component. 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 design to protect against floods, internally and externally-generated missiles, and high- and moderate-energy pipe breaks. On the basis of the design for maintaining environmental conditions and considering dynamic effects, the staff concludes that the control room habitability systems satisfy GDC 4 as it relates to protecting the system against floods, internally-generated missiles, and piping failures.

As stated previously in Section 9.4 of this report, the HVAC design described in the SSAR does not share SSCs with other nuclear power units. Therefore, the VBS meets the requirements of GDC 5.

COL applicants referencing the AP600 design will identify the toxic gases to be monitored. Section 6.4 of this report discusses the specifics relating to compliance with GDC 19 as applied to the AP600 design, as it pertains to protection of the control room against intrusion of toxic gases. As stated in that section, the compliance with GDC 19 in this regard is demonstrated by complying with the guidance of RGs 1.78 and 1.95, which is within the scope of the COL applicant because the guidance is site dependent. Section 6.4 of this report also discusses compliance with GDC 19, as it relates to radiation dose limits for the control room operator.

As a result of the RTNSS process, short-term administrative controls are provided for the MCR ancillary fans that provide long-term cooling to the MCR and I&C rooms in the event of a total loss of ac power.

On the basis of the above review, the staff concludes that the VBS complies with GDC 2, 4, 5, 19, and 60, as referenced in Section 9.4.1 of the SRP.

9.4.2 Annex/Auxiliary Buildings Nonradioactive HVAC System

The staff reviewed the annex/auxiliary buildings nonradioactive HVAC system (VXS) in accordance with the SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VXS satisfies the following requirements:

- GDC 2, regarding the capability of the system to withstand earthquakes
- GDC 5, regarding sharing systems and components important to safety
- GDC 60, regarding the capability of the system to suitably control release of gaseous radioactive effluent to the environment

Westinghouse provided additional information about the VXS that allowed the staff to continue its review, as discussed below. As a result of the additional information, concerning the protection limit of a hydrogen concentration buildup in non-Class 1E battery rooms and discrepancies in the SSAR, DSER Open Item 9.4.2-1 is closed.

The VXS is a nonradioactive HVAC system that serves the nonradioactive personnel and equipment areas; the electrical equipment rooms, clean corridors, ancillary diesel generator room, and demineralized water deoxygenating room in the annex building; and the MSIV compartments, reactor trip switchgear rooms, and piping and electrical penetration areas in the auxiliary building.

The VXS is not required to support any functions or operation of any equipment or systems listed in Position C.1 of RG 1.29. Therefore, Position C.1 of RG 1.29 is not applicable to the VXS. Additionally, the portions of the VXS located in areas containing safety-related components will be seismically supported in accordance with Position C.2 of RG 1.29. Therefore, the VXS conforms with GDC 2, "Design-Basis Protection Against Natural Phenomena."

The system description, design parameters, and P&IDs are provided in SSAR Section 9.4.2, Table 9.4.2-1 through 9.4.2-7, and Figure 9.4.2-1, respectively. In Table 3.2-3 of the SSAR, Westinghouse provides the classification of the VXS system and components. Table 9.4-1 of this report describes the industry standards applicable to the components of the VXS. The VXS supply airflow is balanced in accordance with the guidelines of SMACNA-1983, "HVAC Systems - Testing, Adjusting, and Balancing,"

The VXS is designed to maintain proper operating temperatures in the following areas on the basis of the site interface parameters for maximum and minimum normal temperature conditions defined in SSAR Table 2-1 and summarized as follows:

- For the annex building:
    - offices, corridors, locker rooms, toilet rooms, central alarm stations, and security areas between 22.8 °C to 25.6 °C (73 °F to 78 °F)
    - non-class 1E battery rooms between 15.6 °C to 32.2 °C (60 °F to 90 °F)
    - switchgear and battery charger rooms, HVAC and mechanical equipment room, and ancillary diesel generator room between 10 °C to 40.6 °C (50 °F to 105 °F)
    - switchgear rooms, battery charger rooms, and ancillary diesel generator room during upset conditions (LOOP), with diesel generators operating, maximum temperature of 50 °C (122 °F).
- for the auxiliary building:
    - MSIV compartments, non-safety electrical penetration rooms, reactor trip switchgear rooms, and valve/piping penetration room between 10 °C to 40.6 °C (50 °F to 105 °F)
    - Demineralized water deoxygenating room, elevator machine room, and boric acid batching room between 10 °C to 40.6 °C (50 °F to 105 °F)

The VXS protects against the buildup of hydrogen concentrations to less than 2 percent in the non-Class 1E battery rooms in the annex building.

Section 7.3 of the SSAR describes the VXS instrumentation. The VXS is controlled by the plant control system (PLS). The temperature controllers maintain the proper air temperatures and provide indication and alarms that are accessible locally via the PLS. Temperature is indicated for each AHU supply air discharge duct, except for local recirculation units such as those in the MSIV compartments and valve/piping penetration room.

The VXS has the following six independent subsystems, as shown on Figure 9.4.2-1 of the SSAR:

- general area HVAC subsystem
- switchgear HVAC subsystem
- equipment room HVAC subsystem
- MSIV compartment heating and cooling subsystem
- mechanical equipment areas HVAC subsystem
- valve and piping penetration room HVAC subsystems

These subsystems are discussed below.

9.4.2.1 General Area HVAC Subsystem

The general area HVAC subsystem serves personnel areas in the annex building outside the security area, which include the men's and women's change rooms, shower/toilet areas, the ALARA briefing room and operational support center, offices, and corridors. The subsystem is not credited for plant abnormal conditions.

This subsystem consists of two 50-percent capacity supply AHUs, (5,100 scfm each), a humidifier, a ducted supply and return air system, diffusers and registers, exhaust fans, as well as associated dampers, instrumentation, and controls. The subsystem AHUs are located on the low roof of the annex building at Elevation 117'-6". During normal operation, both AHUs and the toilet/shower exhaust fan operate continuously to maintain the served areas within the design temperature range.

Each AHU of the general area HVAC subsystem consists of a centrifugal supply air fan, low-efficiency filter bank, high-efficiency filter bank, a hot water heating coil with integral face/bypass damper, and a chilled water cooling coil. The units discharge into a ducted supply distribution system, which is routed through the building to provide air into the various rooms and areas served via registers. The AHUs are controlled by temperature controllers, with their sensors located in the annex building main entrance. The temperature controllers modulate the chilled water control valves and the face and bypass dampers of the hot water heating coil in the AHUs. The switchover between cooling and heating modes is automatically controlled by the temperature controllers. Chilled and hot water is provided from the VWS and the plant hot water system (VYS), respectively. Outdoor makeup air is added at the AHU to replace air exhausted from the toilet and shower facilities in the annex building. A common steam humidifier is located in the ductwork downstream of the AHUs to provide a minimum space relative humidity of 35 percent with a humidistat control that is located in the main entrance of the annex building. The men's and women's locker, and the toilet and shower facilities in the annex building have an exhaust fan that exhausts directly to the outside environment.

An electric heating coil provides tempering of the supply air inside the men's and women's facilities, and heating coil elements are controlled by a temperature controller with a sensor located in the women's facility.

To replace the AHU filters, the affected supply fan is stopped and isolated from the duct system by subsystem isolation dampers, and the toilet/shower exhaust fan is also stopped. During filter replacement mode, the subsystem runs at 50-percent capacity and maintains the served areas in the annex building at a slight positive pressure.

9.4.2.2 Switchgear HVAC Subsystem

This subsystem serves the electrical switchgear rooms in the annex building. The subsystem consists of two 100-percent capacity AHUs, a ducted supply and return air system, and automatic controls and accessories. During normal plant operation, one AHU operates continuously to maintain the served areas within the design temperature. The AHU is controlled by a temperature controller to maintain the tempered air supply at 16.7 °C (62 °F) dependent on the outdoor ambient temperature conditions. Each subsystem AHU consists of a centrifugal return/exhaust fan, return/exhaust air plenum, low-efficiency filter bank, high-efficiency filter bank, a hot water heating coil with integral face and bypass dampers, a chilled water cooling coil, and centrifugal supply fan. The AHUs discharge into a common duct supply distribution system, which is routed through the building to the various areas served, and air is returned to the AHU. Chilled and hot water is provided from the VWS and VYS, respectively. The AHUs are located in the north air handling equipment room in the annex building at Elevation 135'-3". The AHUs are connected to a common plenum, which also supplies the outdoor air to the equipment room HVAC subsystem, located along the east wall adjacent to the air handling equipment room.

When the outdoor air temperature is above 16.7 °C (62 °F), the outdoor air, return air, and exhaust air dampers automatically reposition to provide minimum outdoor air, and the temperature controller modulates the chilled water control valves to maintain the supply air at 16.7 °C (62 °F). When the outside air temperature is below 16.7 °C (62 °F), each temperature controller modulates the outdoor air, return air, and exhaust air dampers to control a mixture of the return and minimum outdoor air in the proper proportion and modulates the face and bypass dampers of the hot water heating coils to maintain a mixed air temperature of 16.7 °C (62 °F).

To replace the AHU filters, the affected supply fan is stopped and the affected AHU is isolated from the duct system by subsystem isolation dampers. During filter replacement mode, the second AHU of the subsystem runs at full system capacity.

This subsystem is designed to remove smoke after a fire by placing the subsystem in a once-through smoke exhaust ventilation mode. Additionally, the alternate subsystem, which consists of 100-percent capacity supply and exhaust propeller fans mounted in the annex building wall and controls, provides cooling to electrical switchgear rooms 1 and 2 in the event that the primary subsystem AHUs are unavailable due to a fire. The switchgear rooms will be maintained at or below 50 °C (122 °F). The fans will be controlled to maintain the area above freezing.

In the event of a LOOP, the supply and return/exhaust fans are connected to the standby power system to provide the DID cooling function to the diesel bus switchgear. In this mode, outdoor air and return air volume dampers are positioned to the once-through flow mode to maintain the switchgear rooms at or below 50 °C (122 °F), where equipment is designed for continuous operation under this environment. To maintain the areas above freezing, the mixing dampers will modulate to maintain a supply air temperature of 16.7 °C (62 °F) for outdoor temperatures below 16.7 °C (62 °F). For outdoor temperatures above 16.7 °C (62 °F), the outdoor air, return air, and exhaust air dampers are positioned for a once-through flow.

9.4.2.3 Equipment Room HVAC Subsystem

This subsystem serves the electrical and mechanical equipment rooms in the annex and auxiliary buildings. These rooms include non-Class 1E battery charger rooms 1 and 2, non-Class 1E battery rooms 1 and 2, non-class 1E penetration room on Elevation 100'-0" and non-class 1E penetration room on Elevation 117'-6", and reactor trip switch gear rooms I and II. It also serves the security area offices, and central alarm stations in the annex building including rest rooms, access areas, and corridors.

This subsystem consists of two 100-percent capacity AHUs, two battery room exhaust fans, a toilet exhaust fan, a ducted supply and return air system, and automatic controls and accessories. During normal plant operation, one AHU operates continuously to maintain the served areas within the design temperature. The AHU is controlled by a temperature controller to maintain the tempered air supply at 16.7 °C (62 °F). Each subsystem AHU consists of a centrifugal return/exhaust fan, return/exhaust air plenum, low-efficiency filter bank, high-efficiency filter bank, a hot water heating coil with integral face and bypass, a chilled water cooling coil, and centrifugal supply fan. The AHUs discharge into a common duct supply distribution system, which is routed through the building to the various areas served and air is returned to the AHU, except for the battery rooms and rest rooms. Chilled and hot water is provided from the VWS and VYS, respectively. The AHUs are located in the north air handling equipment room in the annex building at Elevation 135'-3". The AHUs are connected to a common plenum, which also supplies the outdoor air to the switchgear room HVAC subsystem, located along the east wall adjacent to the air handling equipment room.

When the outdoor air temperature is above 16.7 °C (62 °F), the outdoor air, return air, and exhaust air dampers automatically reposition to provide minimum outdoor air, and the temperature controller modulates the chilled water control valves to maintain the supply air at 16.7 °C (62 °F). When the outside air temperature is below 16.7 °C (62 °F), each temperature controller modulates the outdoor air, return air, and exhaust air dampers to control a mixture of the return and minimum outdoor air in the proper proportion, and modulates the face and bypass dampers of the hot water heating coils to maintain a mixed air temperature of 16.7 °C (62 °F).

The electrical reheat coils are installed in the ductwork of non-Class 1E battery rooms, security area offices, and central alarm stations. The hot water unit heaters are provided in the north air handling equipment room and operate intermittently to maintain the area above 10 °C (50 °F). A steam humidifier is installed in the security areas ductwork to provide a minimum relative humidity of 35 percent.

A temperature controller opens the outdoor intake for the elevator machine room and starts and stops the elevator machine room exhaust fan, as required, to maintain the elevator machine room at design temperature conditions. A local thermostat controls the electric unit heater in the elevator machine room.

Each non-Class 1E battery room exhaust system consists of an exhaust fan, gravity back-draft damper, and associated ductwork located in the fan discharge, and exhausts to the atmosphere to prevent a hydrogen gas buildup below 2 percent. Air supplied to the battery rooms by the AHUs is exhausted to the atmosphere. Air from the rest room is exhausted to the atmosphere by a separate exhaust fan.

To replace the AHU filters, the affected supply fan is stopped and the affected AHU is isolated from the duct system by subsystem isolation dampers. During filter replacement mode, the second AHU of the subsystem runs at full system capacity.

The portion of the subsystem servicing the auxiliary building is designed so that smoke, hot gases, and fire suppressant will not migrate from one fire area to another to the extent that they could adversely affect safe shutdown capabilities, including operator actions. Fire or combination fire and smoke dampers, which close in response to smoke detector signals or in response to the heat from a fire, are provided to isolate each fire area from adjacent fire areas during and following a fire in accordance with NFPA 9A requirements.

In the event of a LOOP, the supply and return/exhaust fans are connected to the standby power system to provide the DID cooling function to the dc switchgear and inverters. In this mode, outdoor air and return air volume dampers are positioned to the once-through flow mode to maintain the dc switchgear and inverter areas at or below 50 °C (122 °F), where equipment is designed for continuous operation under this environment. To maintain the areas above freezing, the mixing dampers will modulate to maintain a supply air temperature of 16.7 °C (62 °F) for outdoor temperature below 16.7 °C (62 °F). For outdoor temperatures above 16.7 °C (62 °F), the outdoor air, return air, and exhaust air dampers are positioned for a once-through flow.

9.4.2.4 Main Steam Isolation Valve HVAC Subsystem

The MSIV HVAC subsystem serves the two MSIV compartments in the auxiliary building that contain the main steam and feedwater piping. The main steam and feed water lines between the turbine building and containment are routed through two separate compartments in the auxiliary building. This subsystem is not credited for plant abnormal conditions.

The MSIV HVAC subsystem consists of two 100-percent capacity AHUs in each compartment (3300 scfm each), supply and air distribution ducting, and automatic controls and accessories. During normal plant operation, one of the AHUs in each compartment operates continuously in a recirculation mode to maintain the design temperature range in the area served by the system.

Each AHU consists of a low-efficiency filter bank, a hot water heating coil, a chilled water cooling coil, a centrifugal supply air fan, and associated instrumentation and controls. Chilled and hot water is provided from the VWS and VYS, respectively. Two inside air temperature indicators are provided for each compartment. The switchover between cooling and heating modes is automatically controlled by the area temperature controller.

The temperature of the MSIV compartment is maintained at or less than 40.6 °C (105 °F) and above a minimum of 10 °C (50 °F) by a temperature controller that modulates the chilled water and hot water control valves serving each unit. For investment protection, the subsystem can be powered by the standby power system in the event of a LOOP.

The AHU may be shut down for replacement while another AHU in the same MSIV compartment operates to maintain the served area within the design temperature range. The AHUs can be connected to the standby power system during a LOOP event for investment protection.

9.4.2.5 Mechanical Equipment Areas HVAC Subsystem

The mechanical equipment areas HVAC subsystem serves the ancillary diesel generator (DG) room, the demineralized deoxygenating room, boric acid batching room, and upper and lower south air handling equipment rooms in the auxiliary building. This subsystem maintains served areas at a slightly positive pressure with respect to the adjacent buildings by a constant volume of outside air. This subsystem is not credited for plant abnormal conditions.

The mechanical equipment areas HVAC subsystem consists of two 50-percent capacity AHUs with supply fans and return/exhaust fans (2,200 scfm each), a ducted supply and return air system, and automatic controls and accessories. During normal plant operation, the AHUs operate continuously to maintain the served areas within design temperature range. To replace the AHU filters, the affected AHU is stopped and isolated from the duct system by subsystem isolation dampers. During filter replacement mode, the subsystem operates at approximately 50-percent capacity.

Each subsystem AHU consists of a return/exhaust fan, return/exhaust air plenum, low-efficiency filter bank, high-efficiency filter bank, a hot water heating coil with integral face/bypass damper, chilled water cooling coil, centrifugal fan, and associated instrumentation and controls. Chilled and hot water is provided from the VWS and VYS, respectively. The AHUs are located in the lower south air handling equipment room on Elevation 135'-13" of the annex building. The outdoor air is supplied from the (non-tornado missile protected) air intake plenum #2, which also serves the VAS, VHS, and VFS, located at the extreme south end of the annex building between Elevations 135'-3" and 158'-0".

Two inside air temperature indicators are provided in the lower south air handling equipment room. The temperature of the area is maintained by temperature controllers with sensors located in the south air handling equipment room. The temperature controllers modulate the chilled water control valves and the face and bypass dampers of the hot water heating coil in the AHUs to maintain the areas served by the subsystem within the design temperature range. The switchover between cooling and heating modes is automatically controlled by the area temperature controller.

The ancillary DG room is served by the subsystem and maintained within the design temperature range when the ancillary DGs are not in operation. The air supplied to the DG room is exhausted to the outdoors by means of a separate exhaust fan, which runs continuously. The ancillary DG room is maintained within the design temperature during ancillary diesel generators operation by opening a manual damper and a room door to allow DG radiator discharge air to be exhausted to the outdoors.

9.4.2.6 Valve and Piping Penetration Room HVAC Subsystem

The valve and piping penetration room HVAC subsystems consist of local recirculation HVAC units to cool the valve/piping penetration room located at Elevation 100'-0" of the auxiliary building. The subsystem is not credited for plant abnormal conditions.

The valve/piping penetration room HVAC subsystem consists of two 100-percent capacity AHUs, and is provided with an automatic ducted supply and return air system, and automatic controls and accessories. The AHUs are located directly within the space served on Elevation 100'-0". During normal operation, one AHU operates continuously in a recirculation mode to maintain the served area within the design temperature range.

Each subsystem AHU consists of a low-efficiency filter, a hot water heating coil, a chilled water cooling coil, centrifugal supply air fan, and associated instrumentation and controls. Chilled and hot water is provided from the VWS and VYS, respectively. Two inside air temperature indicators are provided for each valve and piping penetration room HVAC subsystem. The switchover between cooling and heating modes is automatically controlled by the area temperature controller.

The temperature controllers, with sensors located in the room, modulate the chilled water control valves and the hot water control valves in the operating AHU. The temperature of valve/piping penetration room is maintained at or less than 40.6 °C (105 °F) and above a minimum of 10 °C (50 °F) by a temperature controller that modulates the chilled water and hot water control valves serving each unit.

The hot water and electric unit heaters are controlled by local thermostats. The area temperature indication is accessible from the MCR. The pressure indication and high differential pressure alarm are provided for each of the filters in the AHUs for filter replacement. The operational status of fans is indicated in the MCR and position indicating lights are provided for automatic dampers. The air flow is indicated for the discharge ducts of the AHUs and alarms for low flow rates are provided in the fan discharge ducts. Smoke alarms are provided in the discharge ducts of the AHUs.

9.4.2.7 Conclusion

The VXS is non-safety-related; therefore, Position C.1 of RG 1.29 is not applicable to the VXS. Additionally, the portions of the VXS located in areas containing safety-related components will be seismically supported in accordance with Position C.2 of RG 1.29. System equipment and ductwork located in the nuclear island the failure of which could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portions of the system are non-seismic. The VXS conforms with GDC 2.

As stated previously in Section 9.4 of this report, the HVAC design described in the SSAR does not share SSCs with other nuclear power units, therefore, the VXS meets the requirements of GDC 5.

The VXS is a non-radioactive HVAC system that serves areas where no radioactive sources are anticipated. Therefore, GDC 60, "Control of Releases of Radioactive Materials to the Environment," is not applicable.

The staff evaluated the VXS for conformance with GDC 2, 5, and 60, as referenced in Section 9.4.3 of the SRP. The staff finds the VXS acceptable.

9.4.3 Radiologically Controlled Area Ventilation System

The staff reviewed the radiologically controlled area ventilation system (VAS) in accordance with SRP Sections 9.4.2, "Spent Fuel Pool Area Ventilation System," and 9.4.3, "Auxiliary and Radwaste Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VAS satisfies the following requirements:

- GDC 2, regarding the capability of the system to withstand earthquakes
- GDC 5, regarding sharing systems and components important to safety
- GDC 60, regarding the capability of the system to suitably control release of gaseous radioactive effluent to the environment
- GDC 61, regarding the capability of the system to provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment from the fuel storage facility under normal and postulated accident conditions

Westinghouse provided additional information about the VAS that allowed the staff to continue its review, as discussed below. As a result of this additional information, concerning the safe hydrogen concentration levels for the gaseous radwaste module areas and discrepancies in the SSAR, DSER Open Item 9.4.3-1 is closed.

The VAS neither serves nor supports the plant safety-related functions; therefore, the system has no nuclear safety design basis. The VAS consists of the following two subsystems:

(1) the auxiliary/annex building ventilation subsystem (AABVS)
(2) the fuel-handling area ventilation subsystem (FHAVS)

The VAS serves the fuel-handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings. The VAS maintains environmental conditions appropriate for equipment operation, for performing maintenance and testing, and to allow personnel access. The VAS ventilation airflow rate dilutes potential airborne contamination to within the effluent concentration limits allowed by 10 CFR Part 20 at the site boundary during normal plant operation. The plant's internal airborne concentration levels will conform to the 10 CFR Part 20 occupational derived air concentration (DAC) limits. The VAS maintains normal airflow direction from lower to higher potential airborne concentrations for ALARA considerations. The design of the VAS exhaust subsystems conforms with the requirements of Appendix I to 10 CFR Part 50 for releases during normal operation as follows. Upon detection of high airborne radioactivity in the air exhaust duct or high ambient pressure differential resulting from an imbalance in supply and exhaust airflow rates, unfiltered normal VAS exhaust is isolated, and the containment air filtration system (VFS) filtered exhaust subsystem starts to provide filtration of the exhaust air from the fuel-handling area, as well as the auxiliary and annex buildings to minimize unfiltered offsite releases. These areas are maintained at a slight negative pressure with respect to the adjacent clean areas when high airborne radioactivity is detected. The VFS mitigates exfiltration of unfiltered airborne radioactivity by maintaining the isolated zone at a slightly negative pressure with respect to the outside environment and adjacent unaffected plant areas. The VFS maintains a slightly negative pressure differential with respect to the outside environment until operation of the AABVS and FHAVS are restored.

The AABVS supply and exhaust ducts are configured into two independently isolatable building zones. A radiation monitor is located in the exhaust duct, upstream of an isolation damper in each zone served by AABVS. The FHAVS supply and exhaust ductwork is arranged to exhaust the spent fuel pool plume and to provide directional airflow from the rail car bay/filter storage area into the spent fuel resin equipment rooms. The FHAVS contains a radiation monitor that is mounted upstream of an isolation damper in the exhaust duct. The AABVS' and FHAVS' unfiltered, but monitored, exhaust is routed to the plant vent during normal operation. However, if high radioactivity is detected, the subsystems' exhaust is filtered through the VFS and then routed to the plant vent. The staff's review of radiation monitoring is discussed in Section 11.5 of this report. In conjunction with the VFS as described above, the VAS provides appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment from the fuel storage facility during normal and postulated accident (non-DBA) conditions, in accordance with GDC 61. However, no credit for these features is assumed in the determination of the radiological consequences of a fuel-handling accident. The radiological consequences of a fuel-handling accident are discussed in Section 15 of this report.

Each radwaste effluent holdup tank exhaust is connected to the AABVS exhaust ducting, to prevent the potential buildup of airborne radioactivity or hydrogen gas that may leak within the tanks, which is routed to the plant vent through the VFS. The AABVS provides sufficient ventilation to the gaseous radwaste equipment areas to dilute hydrogen gas that may leak from the radwaste equipment into the equipment rooms. The hydrogen gas concentration is maintained below a safe level of about 1 percent which conforms with the guidelines of SRP Section 11.3. The hydrogen monitoring instrumentation is described in SSAR Table 11.3-2. An evaluation of the gaseous waste management system is provided in Section 11.3 of this report.

The VAS description, component design parameters, and P&IDs are given in Section 9.4.3, Tables 3.2-3 and 9.4.3-1, and Figure 9.4.3-1 of the SSAR, respectively. Table 9.4-1 in this report describes the industry standards applicable to the components of the VAS. The VAS supply airflow is balanced in accordance with the guidelines of SMACNA-1983, "HVAC Systems - Testing, Adjusting, and Balancing."

The VAS is a once-through design that draws outdoor air and exhausts the air to the plant vent. During normal plant operation, VAS supply air handling units and exhaust fans for both the AABVS and FHAVS operate continuously to ventilate the areas on a once-through basis. The VAS air handling units are automatically shut down if airflow or supply air temperature is below a predetermined setpoint. Temperature controllers maintain proper supply air temperature to maintain the ambient room temperature within the normal range. The temperature of the air supplied by each AHU of the AABVS and FHAVS is controlled by temperature sensors in the supply air duct. When the outdoor air temperature is low, the face and bypass dampers across the supply air heating coil are modulated to maintain the ambient room temperature, and when the outdoor air temperature is high, the supply air is tempered by the chilled water coil. The supply of chilled and hot water for the VAS is provided from the VWS and VYS, respectively.

The VAS cooling and heating capacity is dependent on the site interface parameters for maximum and minimum normal temperature conditions, as defined in SSAR Table 2-1. The annex building staging areas and storage areas, as well as other corridors and staging areas are maintained between 10 °C and 40 °C (50 °F and 104 °F). The corridors and access areas served by the FHAVS is maintained between 10 °C and 40 °C (50 °F and 104 °F). The radiation chemistry laboratory and security rooms are served by the AABVS and are maintained between 22.8 °C and 25.6 °C (73 °F and 78 °F). The primary sample room is served by the AABVS and is maintained between 10 °C and 54.4 °C (50 °F and 130 °F).

Section 7.3 of the SSAR describes the VAS instrumentation. The VAS is controlled by the PLS. The temperature controllers maintain the proper air temperatures and provide indications and alarms. The pressure differential indications and alarms monitor the outside air and inside ambient pressures of the fuel-handling area and auxiliary/annex buildings, and control the supply air flow to maintain a slightly negative pressure differential with respect to adjacent clean areas and outdoors. The area temperature indication is accessible from the MCR for the RNS and makeup pump rooms without requiring personnel access to these rooms. Auxiliary building and fuel-handling area radiation monitoring instrumentation (radiation detectors) is provided in the system exhaust ducts upstream of the isolation dampers. As described above, upon detection of high airborne radioactivity in the air exhaust duct or high ambient pressure differential, the radiation detectors automatically isolate the unfiltered exhaust air ducts and start the VFS filtered exhaust. The MCR is provided with indication and alarms in the exhaust ducts from the fuel-handling area and radiologically controlled areas of the auxiliary and annex buildings. Operational status of fans and dampers is indicated in the MCR. All fans and AHUs can be operated or shutdown from the MCR. Pressure indications and high differential pressure alarms for the system filters and unit coolers are provided.

The AABVS and FHAVS ventilation air is continuously monitored by smoke monitors located downstream of the AHUs and upstream of the exhaust fans. If smoke is detected in the supply or exhaust ducts, an alarm is initiated in the MCR, fire dampers automatically isolate the HVAC ductwork penetrating the affected fire area upon exceeding the predetermined setpoints of the local temperature, and HVAC subsystems remain in operation but may be shutdown manually by MCR operators, as needed.

9.4.3.1 Auxiliary/Annex Building Ventilation Subsystem (AABVS)

The AABVS supply and exhaust ducts are configured into two zones. The annex building staging and storage area, containment air filtration rooms, containment access corridor, and adjacent auxiliary building staging area, equipment areas, middle annulus, middle annulus access room, and security rooms are served by one zone. The other zone includes the remaining rooms and corridors shown in Figure 9.4.3-1, Sheet 2 of 3, including but not limited to the radiation chemistry laboratory, primary sample room, spent fuel pool cooling water pump and heat exchanger rooms, RNS pump and heat exchanger rooms, CVS makeup pump room, lower annulus, and various radwaste equipment rooms, pipe chases, and accesses corridors.

The AABVS provides conditioned air to maintain the proper operating temperatures in the following areas:

- the RNS and CVS pump rooms (pumps not operating), containment purge exhaust filter rooms (fans not operating), liquid radwaste pump rooms, HVAC equipment room, gaseous equipment rooms, and spent fuel pool pump and heat exchanger rooms between 10 °C and 40 °C (50 °F and 104 °F)

- the degasifier column, RNS and CVS pump rooms (pumps operating), containment purge exhaust filter rooms (fans operating), and liquid radwaste tank rooms between 10 °C and 54.4 °C (50 °F and 130 °F)

The supply air flow rate is modulated to maintain a slightly negative pressure differential with respect to the outside environment. Provision for supplementary heating is made for the exterior annex/auxiliary building areas through the annex building heating coil. The AABVS consists of two 50-percent capacity supply AHUs (18,000 scfm each), two 50-percent capacity exhaust fans sized to allow the AABVS to maintain a negative pressure in served areas with respect to the adjacent areas, associated ductwork, dampers, diffusers and registers, and instrumentation and controls. Each AHU consists of a centrifugal supply air fan, a low-efficiency filter bank, a high-efficiency filter bank, a water heating coil with integral face and bypass dampers, a chilled water cooling coil, and associated instrumentation and controls. The supply AHUs are located in the south air handling equipment room of the annex building at Elevation 158'-0" and connected to a common air intake plenum #3. The common, non-tornado missile protected, air intake plenum #3 is located at the extreme south end of the annex building between Elevation 158'-0" and about 180'-0". Each subsystem AHU discharges into a ducted supply distribution system, which is routed through the radiologically controlled areas of the auxiliary and annex buildings.

The AABVS exhaust fans are located in the auxiliary building at Elevation 145'-9". The supply and exhaust ducts have isolation dampers. During normal operation, the subsystem's exhaust is unfiltered and directed to the plant vent for discharge. During high radiation isolation mode, the normal unfiltered ventilation subsystem is isolated from the affected zone when high airborne radioactivity is detected, and the isolated area is exhausted through the VFS to the monitored plant vent. The VFS exhaust fans prevent unfiltered airborne releases by maintaining these areas at a slight negative pressure with respect to the outside environment and adjacent clean plant areas.

Each CVS makeup and normal RNS pump room is provided with a dedicated, 100-percent capacity unit cooler, for a total of two per CVS and RNS pump room, to provide supplemental cooling during pump operation. The chilled water to coolers is supplied from redundant trains of the central chilled water system (VWS). Each unit cooler consists of a low-efficiency filter bank, a cooling coil bank and fan, except that each RNS pump room cooler has redundant cooling coil banks such that either redundant train of the VWS can support the operation of both RNS pumps simultaneously. The CVS pump room coolers are connected to redundant trains of the VWS; however, either train unit cooler can maintain the common makeup pump room temperature conditions and support either makeup pump operation. The pump room coolers automatically start whenever the associated pump receives a start signal or a high room temperature signal. In a LOOP event, the unit coolers can be powered by the onsite diesel generators.

The upper annulus is separated from the middle annulus area of the auxiliary building by a concrete floor section and flexible seals that connect the containment steel shell to the shield building. The annulus seal provides a passive ventilation barrier during normal operation or during isolation of the auxiliary building to prevent the exfiltration of unmonitored releases from the middle annulus to the environment.

The radiation chemistry laboratory and security room supply air ducts are supplemented with locally installed electric coils and humidifiers to maintain environmental conditions within the areas suitable for personnel comfort. The electric unit heaters provide supplemental heating in the middle annulus as shown in SSAR Figure 9.4.3-1.

9.4.3.2 Fuel-Handling Area Ventilation Subsystem (FHAVS)

The FHAVS serves the-fuel handling area, rail car bay/filter storage area, resin transfer pump/valve room, spent resin tank room, waste disposal container area, WSS (spent resin) valve/piping area, and elevator machine room.

The FHAVS provides air to maintain the following conditions:

- *the rail car bay/filter storage area between 10 °C and 40 °C (50 °F and 104 °F)*
- the spent resin equipment rooms between 10 °C and 54.4 °C (50 °F and 130 °F)
- the fuel handling area between 10 °C and 35.6 °C (50 °F and 96 °F)
- a maximum wet bulb temperature of 26.7 °C (80 °F), within the guidelines of EPRI NP-4453, in the areas occupied by plant personnel during refueling activities

The supply air flow is modulated to maintain the served areas at a slight negative pressure differential with respect to the outside environment. Provision for supplementary heating is made for the rail car bay through the rail car bay heating coil. The FHAVS consists of two 50-percent capacity supply AHUs (9,500 scfm each), two 50-percent capacity exhaust fans sized to allow the AABVS to maintain a negative pressure in served areas with respect to the adjacent areas, associated ductwork, dampers, and instrumentation and controls. Each AHU consists of a centrifugal supply air fan, a low-efficiency filter bank, a high-efficiency filter bank, a water heating coil with integral face and bypass damper, a chilled water cooling coil, and associated instrumentation and controls. These areas form a single isolation zone when high airborne radioactivity is detected in the exhaust air. The supply AHUs are located in the south air handling equipment room of the annex building at Elevation 135'-3" and connected to a common air intake plenum #2 located at the south end of annex building. The common, non-tornado missile protected, air intake plenum #2 is located at the extreme south end of the annex building between Elevation 135'-3" and about 158'-0". Each subsystem AHU discharges into a ducted supply distribution system, which is routed to the fuel-handling area and rail car bay/filter storage areas of the auxiliary building.

The FHAVS supply AHUs are located in the south air handling equipment room of the annex building at Elevation 135'-3", and the exhaust fans are located in the upper radiologically controlled area ventilation system equipment room at Elevation 145'-9" of the auxiliary building. The exhaust has isolation dampers. During normal operation, the subsystem's exhaust is unfiltered and directed to the plant vent for discharge and monitoring of offsite gaseous releases. During high radiation isolation mode, the normal unfiltered ventilation subsystem is isolated from the affected zone when high airborne radioactivity is detected, and the isolated area is exhausted through the VFS to the monitored plant vent.

9.4.3.3 Conclusion

As described in Section 9.4.3 and Table 3.2-3 of the SSAR, the VAS is located completely within a seismic Category I auxiliary building and seismic Category II annex building structures, and all system components are non-safety-related. Position C.1 of RG 1.29 is not applicable because the FHAVS is not credited for the mitigation of the fuel-handling accident (FHA), and therefore, it is not required to remain functional after an SSE. Compliance with Position C.2 of RG 1.29 is demonstrated in the evaluation of the FHAVS for interaction with seismic Category I systems in Section 3.7.3 of this report. The evaluation demonstrates that seismic failure of the FHAVS does not reduce the functioning of the safety-related plant features. System equipment and ductwork located in the nuclear island, the failure of which could affect the operability of safety-related systems or components, are designed to seismic Category II requirements and the remaining portion of the system is non-seismic. The AABVS is not credited for any DBA conditions, and therefore, it is not required to remain functional after an SSE. The makeup pump and RNS pump room coolers and exhaust fans for the AABVS are located in the seismic Category I auxiliary building. Therefore, the VAS complies with the requirements of GDC 2.

As stated previously in Section 9.4 of this report, the HVAC design described in the SSAR does not share SSCs with the other nuclear power units. Therefore, the VAS meets the requirements of GDC 5.

As discussed above, the subsystems' unfiltered, but monitored exhaust is routed to the plant vent during normal operation. If high radioactivity is detected, the subsystems' exhaust is filtered through the VFS, and then routed to the plant vent in compliance with the guidelines of RG 1.140 for controlling the release of radioactivity as discussed in Chapter 1.0, Appendix 1A of the SSAR. As discussed in Section 9.4.7 of this report, the VFS filtration units are designed, constructed, and tested to conform with ASME N509 and N510 standards, and the guidelines of RG 1.140. Therefore, the VAS also complies with the requirements of GDC 60, as it relates to the capability of the system to suitably control the release of gaseous radioactive effluent to the environment. The VFS filtration units, where the VAS radioactive exhaust is filtered, are evaluated against Position C.4 of RG 1.13, as discussed in SSAR Chapter 1, Appendix 1A; consequently, the system complies with the requirements of GDC 61, as it relates to the capability of the system to provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment.

The staff evaluated the VAS for conformance with GDC 2, 4, 5, 17, and 60, as referenced in Section 9.4.5 of the SRP. The staff finds the VAS acceptable.

9.4.4 Balance of Plant Interfaces

The AP600 is a complete design; therefore, balance-of-plant interfaces are not applicable to this design.

9.4.5 Engineered Safety Features Ventilation System

Following the issuance of the DSER, the staff evaluated the non-safety-related HVAC systems against the RTNSS criteria. The staff concluded that none of the HVAC systems are ESF ventilation systems, and that no HVAC system is required to support non-safety-related systems that are determined to be important by the RTNSS process, on the basis of the following:

- All HVAC systems are non-safety-related, non-RTNSS systems. The safety-related main control room emergency habitability system (VES) is credited to meet the requirements of GDC 19 as applied to AP600 during a design-basis LOCA and is evaluated in Section 6.4 of this report.

- The nuclear island nonradioactive ventilation system (VBS) provides safety-related design basis functions to (1) monitor the air supply for radioactive particulate and iodine concentrations inside the MCRE, and (2) isolate the safety-related, seismic Category I HVAC piping penetrating the MCRE upon detecting "high-high" particulate or iodine radioactivity in the supplied air. The VBS is non-safety-related except as described above and is not credited to meet the requirements of GDC 19 as applied to AP600 during a design-basis LOCA. The non-safety-related VBS is evaluated in Section 9.4.1 of this report.

- The containment air filtration system (VFS) is not required to mitigate the consequences of a design-basis fuel-handling accident or a LOCA. The VFS was evaluated in accordance with the SRP Section 9.4.5, "Engineered Safety Features Ventilation System." The system serves no safety-related function other than containment isolation, and its operation is not required following a DBA. The non-safety-related VFS is evaluated in Section 9.4.7 of this report.

- No other non-safety-related HVAC systems are credited in the accident dose analyses or provide any safety-related design-basis functions; other HVAC systems are evaluated in Sections 9.4.2, 9.4.3, 9.4.6, and 9.4.8 through 9.4.11 of this report.

Therefore, DSER Open Item 9.4.5-1, concerning those ventilation systems that may be required to support non-safety-related systems that are determined to be important by the RTNSS process, is closed.

9.4.6 Containment Recirculation Cooling System

The staff reviewed the containment recirculation cooling system (VCS) in accordance with SRP Section 9.4.5, "Engineered Safety Feature Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VCS satisfies the following requirements:

- GDC 2, regarding the capability of the system to withstand earthquakes
- GDC 4, regarding maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions
- GDC 5, regarding sharing systems and components important to safety
- GDC 17, regarding the assurance of proper functioning of essential electric power systems
- GDC 60, regarding the capability to suitably control release of gaseous radioactive effluent to the environment

After the DSER was issued, Westinghouse provided additional information about the VCS that allowed the staff to continue its review as discussed below. As a result of this additional information, concerning the classification of VCS components in SSAR Table 3.2-3; radiation monitoring; conformance with ASME N509-1989, N510-1989 and RG 1.140; and SSAR revisions, DSER Open Item 9.4.6-1 is closed.

The VCS is a non-safety-related ventilation system that is not required to mitigate the consequences of a DBA or LOCA. If the VCS is available following abnormal operational transients, fan coil units can be operated at slow speed for post-event recovery operations to lower the containment temperature and pressure. The VCS is supplemented by a maintenance space ventilation subsystem with a portable exhaust filtration unit, which is used during shutdown and refueling operation to protect maintenance personnel and to control the spread of airborne contamination from the steam generator compartments to the other containment areas. During integrated leak rate testing (ILRT) operation, the VCS fans are operated at slow speed in order not to exceed their rated horsepower, which could affect the ILRT results.

The VCS operates during normal plant operation and shutdown to maintain suitable temperatures in the served areas of the containment building. The two fan coil unit (FCU) assemblies are located on a platform at Elevation 149'-7", approximately 180° apart to provide proper mixing of return and supply air. The top of the ring header is at Elevation 176'-6". System equipment and ductwork located in the nuclear island the failure of which could affect the operability of safety-related systems or components are designed to seismic Category II requirements and the remaining portion of the system is non-seismic.

The system description, component design parameters, and P&ID are provided in Section 9.4.6, Table 9.4.6-1, and Figure 9.4.6-1 of the SSAR, respectively. Table 9.4-1 of this report describes the industry standards applicable to the components of the VCS. The VCS airflow is balanced in accordance with SMACNA-1983, "HVAC Systems - Testing, Adjusting, and Balancing."

The VCS maintains temperatures in the served areas below 48.9 °C (120 °F) during normal operation. The VCS also maintains the reactor cavity area average concrete temperature at

65.6 °C (150 °F), with a local area temperature of 93.3 °C (200 °F). During refueling and plant shutdown, the served areas bulk air temperature is maintained below 21.1 °C (70 °F) and above 10 °C (50 °F) for personnel access and equipment operability.

As stated in SSAR Section 9.4.6.2 and Table 9.4.6-1, and as shown in Figure 9.4.6-1, the VCS has two 100-percent FCU assemblies, each with two separate, but physically connected, 50-percent capacity FCUs. Each FCU assembly draws air from the upper levels of the operating floor and delivers tempered air through the ring header and the secondary duct distribution system to the cubicles, compartments, and access areas above and below the operating floor, including the reactor cavity and reactor support areas. As the tempered air absorbs the heat released from the various components inside containment, return air rises through vertical passages and openings where it is again returned into the FCUs, tempered, dehumidified, and recirculated.

Each FCU assembly consists of two 50-percent capacity FCUs. Each FCU contains a vane axial, upblast, direct-driven fan with a two-speed motor; return air mixing plenum section with a physical barrier in the middle; and three chilled water cooling coils attached to the side of each plenum section. The fans operate in high speed during normal operation and low speed for high ambient air density conditions, such as during ILRT and abnormal post-event recovery operation.

The supply of the chilled and hot water is provided from the central chilled water system (VWS) and hot water heating system (VYS), respectively. The cross-connections for VWS and VYS are located outside containment. The water piping inside containment is common to both the VWS and VYS.

To meet the environmental design criteria during various modes of VCS operation, temperature controllers are provided in the ring headers of the corresponding FCU, which provide an input signal to modulate the VWS supply valves to the cooling coils to maintain the normal air supply at 15.6 °C (60 °F). The standby FCUs start automatically if the discharge flow rate from the operating FCU drops below a predetermined setpoint, if the discharge temperature from the operating FCU is above or drops below a predetermined setpoint, and if electrical and/or control power is lost. The FCU fans are connected to 480 V buses with backup power supply from the onsite standby diesel generators.

A steam generator maintenance space ventilation subsystem is employed through the compartment supply air ducts during reactor shutdown for personnel access and maintenance activities. This vacuum system protects personnel and controls the spread of airborne radioactive contamination from the steam generator compartments to the other containment areas. The subsystem consists of permanently installed exhaust ductwork with flexible hose connections in the vicinity of steam generator channel heads which can be connected to a portable exhaust filtration unit which does not exhaust outside of containment. During subsystem operation, the supply air distribution ductwork is isolated by closing relevant supply dampers.

The VCS instrumentation is described in Section 7.3 of the SSAR. The VCS is monitored by the plant monitoring system and controlled by the plant control system. The indication of the operational status and controls of the equipment inside the containment is provided in the MCR.

Temperature indications and alarms are provided in the equipment compartment or areas of the containment to maintain the supply air temperature within a predetermined range.

The containment and equipment compartment temperatures are monitored from the MCR. The FCU discharge flow is monitored and low flow is alarmed to alert the MCR operator to start the spare FCU manually. The reactor cavity areas are also monitored and alarmed for low flow condition.

Compliance with Position C.1 of RG 1.29 does not apply to the VCS because it is not designed to perform any safety functions. Instead, the VCS complies with Position C.2 of RG 1.29 for the following reasons:

- System equipment and ductwork located in the nuclear island, the failure of which could affect the operability of safety-related systems or components are designed to seismic Category II requirements. The remaining portions of the system are non-seismic.

- The VCS is evaluated for interaction with seismic Category I systems in Section 3.7.3.13 of the SSAR to ensure that the VCS does not reduce the functionability of any safety-related plant features. Seismic interaction is evaluated in Section 3.7.3 of this report.

Therefore, the system complies with the requirements of GDC 2.

As stated previously in Section 9.4 of this report, the HVAC system design described in the SSAR does not share SSCs with other nuclear power units. Therefore, the VCS meets the requirements of GDC 5.

The staff determined that the VCS is not an ESF system, the system is not credited in analyzing the consequences of DBA, and the system does not exhaust to the environment. Therefore, the requirements of GDC 4, 17, and 60 are not applicable.

The staff evaluated the VCS for conformance with GDC 2, 4, 5, 17 and 60 as referenced in Section 9.4.5 of the SRP. The staff finds the VCS acceptable.

9.4.7 Containment Air Filtration System

The staff reviewed the containment recirculation cooling system (VFS) in accordance with SRP Section 9.4.5, "Engineered Safety Feature Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VFS satisfies the following requirements:

- GDC 2, regarding the capability of the system to withstand earthquakes

- GDC 4, regarding maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions

- GDC 5, regarding sharing systems and components important to safety
- GDC 17, regarding the assurance of proper functioning of essential electric power systems
- GDC 60, regarding the capability to suitably control release of gaseous radioactive effluent to the environment
- GDC 61, regarding the capability to provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment

After the DSER was issued, Westinghouse provided additional information about the VFS that allowed the staff to continue its review as discussed below. As a result of this additional information, concerning the outdoor design temperature range, AHU outlet design temperature, component classification, specific applicable codes and standards for the VFS equipment, and air intakes protection and their locations, DSER Open Item 9.4.7-1 is closed.

The VFS is not required to mitigate the consequences of a design-basis fuel-handling accident or a LOCA. The system serves no safety-related function other than containment isolation, and its operation is not required following a DBA. The containment isolation components are safety Class B and seismic Category I, and the quality assurance requirements of 10 CFR Part 50, Appendix B are applicable. The components include air-operated, fail-close (during loss of power or loss of air pressure) containment isolation valves (CIVs), penetrations, interconnecting piping, and vent and test connections with manual valves. The size of supply and exhaust air lines that penetrate the containment pressure boundary is 914.4 mm (36 in.) in diameter. Each penetration includes inboard and outboard branch connections with 406.4 mm (16 in.) diameter CIVs that are opened when the VFS is aligned to containment. The other ends of the containment penetrations are capped with 914.4 mm (36 in.) diameter blind flanges for installation provisions for a high-volume purge system on a site-specific basis. The seismic Category I debris screens are designed for post-LOCA pressures and mounted on safety Class C, seismic Category I piping between the containment atmosphere and the CIVs to prevent entrainment debris through the supply and exhaust opening that may prevent a tight valve shut off against the containment pressure. The CIVs in the supply and exhaust air subsystems automatically close upon receiving a containment isolation signal or a containment area high radiation signal. The CIVs are designed to shut tight when subject to the containment pressure following a DBA. The containment isolation function is evaluated and found acceptable in Section 6.2.3 of this report.

The VFS also provides the following functions:

- flow of outdoor air for containment purging to reduce the airborne radioactivity to an acceptable level for personnel access intermittently during normal plant operation and continuously during hot or cold plant shutdown conditions
- containment pressure control within its normal design pressure range by intermittent venting of air into and out of the containment
- filtration of exhaust air before discharge to the plant vent in accordance with the guidelines of 10 CFR Part 50, Appendix I for offsite releases and 10 CFR Part 20 allowable effluent

concentration limits, when combined with other gaseous effluent releases, for the site boundary release

- monitoring gaseous, particulate, and iodine concentration levels discharged to the environment through the plant vent

- conditioning and filtration of outside air for suitable environmental conditions for personnel comfort inside the containment during access for maintenance and refueling operations

- filtration of exhaust air from the fuel-handling area, and auxiliary or annex buildings, and maintaining these areas at a slight negative pressure with respect to the adjacent clean areas through the VFS exhaust air subsystem's opposed-blade pressure differential control dampers when these areas are isolated due to high airborne radioactivity or ambient pressure differential

The VFS is designed to maintain the supply air temperature range between 10° C and 21.1° C (50° F and 70° F) inside containment, dependent on maximum and minimum normal outside temperature conditions shown in SSAR Chapter 2, Table 2-1. The supply air is distributed and conditioned within the containment by the VCS.

The VFS supply air subsystem airflow is measured and balanced in accordance with SMACNA-1983. The VFS containment isolation valves, which are located in auxiliary and containment building, conform to ASME Section III - Class 3 for Class B valves, and B31.1 for Class D valves.

The non-safety-related portions of the VFS are designed to accomplish their intended functions assuming a single active failure and a LOOP event. The VFS consists of two 100-percent capacity, 1.9 m^3/s (4000 cfm) supply and exhaust air subsystems. Each train consists of a supply AHU, ducted air supply, registers, valves and piping, automatic controls, and accessories. Each of the exhaust air systems consists of filtration units, exhaust fans, valves and piping, automatic controls, and accessories. Exhaust air subsystems also contain common containment isolation valves and piping prior to the inlet of the air filtration units and common exhaust leading to the plant vent. A gaseous radiation monitor, located downstream of the exhaust air filtration units in the common ductwork, activates an alarm in the MCR when excess activity in the effluent discharge is detected. The plant vent exhaust flow is monitored for gaseous, particulate, and iodine releases to the environment. The radiation monitoring is described in Section 11.5 of this report.

The supply AHUs are located in the south air handling equipment room of the annex building at Elevation 158'-0". The exhaust filtration units are located within the radiologically controlled portion of the annex building at Elevation 135'-3" and Elevation 146'-3". The common air intake plenum #3 for the supply and makeup air for the exhaust fan (which is not protected from turbine missiles) is located at the extreme south end of the annex building between Elevation 158'-0" and Elevation 180'-0". The ductwork located inside containment, the potential failure of which could affect safety-related equipment, is designed to seismic Category II. The VFS description, P&IDs, and component design parameters are provided in AP600 SSAR Section 9.4.7, Tables 3.2-3, 9.4-1, and 9.4.7-1 and Figure 9.4.7-1, respectively. Table 9.4-1 of this report describes the industry standards applicable to the components of the VFS.

Each supply AHU consists of a low-efficiency filter, a high-efficiency filter, a hot water heating coil with integral face and bypass dampers, a chilled water cooling coil, a supply air fan, and associated instrumentation and controls. The AHU air flow rate is controlled to a constant value by modulating the supply fan inlet vanes to compensate for filter loading or changes in containment pressure. The discharge air through each AHU is controlled by temperature sensors located in the supply air duct. When the supply air temperature is low, the integral face and bypass dampers across the hot water heating coil bank are modulated to heat the supply air, and when the supply air temperature is high, the chilled water flow is modulated to maintain the desired temperature in the area. The supply air is continuously monitored by a smoke alarm located in the common discharge ductwork downstream of the AHUs.

Each exhaust air filtration unit consists of a 100-percent capacity electric heater to maintain 70 percent or less RH of the effluent air, an upstream high-efficiency filter bank, a charcoal adsorber with pre- and post-HEPA filter bank, an exhaust fan, and instrumentation and controls. The post-filters downstream of the charcoal adsorbers have a DOP efficiency of 95 percent. The isolation dampers in the exhaust air subsystem are bubble-tight, single-blade or parallel-blade type, and conform to AMCA 500 and ASME N509 standards. The representative samples of charcoal adsorbent are tested to verify a minimum charcoal efficiency of 90 percent, in accordance with the guidance of RG 1.140 at frequencies identified in the ASME N509 standard, and each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent. The exhaust air subsystem filtration units are designed, constructed and tested to conform with ASME N509 and N510 standards, and the guidelines of RG 1.140-1979, Revision 1.

Each charcoal adsorber is a single-tray assembly with welded construction and a 101.6 mm (4") thickness Type III rechargeable adsorber cell, which conforms with IE Bulletin 80-03, "Loss of Charcoal from Absorber Cells."

The air flow rate through the exhaust filters is controlled to a constant value when the exhaust filters are connected to the containment by modulating the exhaust fan inlet vanes to compensate for filter loading or changes in system resistance caused by single or parallel fan operation, or a change in containment pressure. The containment exhaust line consists of isolation valves arranged in parallel, to restrict the airflow to maintain the exhaust plenum at a negative air pressure when the containment is positively pressurized and, therefore, the exfiltration of unfiltered air bypassing the filtration unit filters is prevented.

During normal plant operation, one supply AHU provides outdoor air, which is filtered, cooled or heated, to the containment areas above the operating floor. During single subsystem operation, the standby supply and exhaust air units can be started manually by the operator if the operating train fails. The supply of chilled water is provided from the VWS and hot water is provided from the VYS. The filtered exhaust air from the containment is discharged to the atmosphere through the plant vent by one exhaust fan.

Before and during cold plant shutdown, one or both trains of the VFS can be operated to remove airborne radioactivity before personnel access inside the containment. When both trains are in operation concurrently, the VFS provides a maximum air flow rate equivalent to 0.25 air changes per hour.

During an abnormal operation, if high airborne radioactivity or pressure differential is detected in the fuel-handling area, auxiliary or annex buildings (zone area(s)), the VAS is isolated from the served zone area(s) and the VFS exhaust air subsystem operates to maintain the isolated zone(s) at a slightly negative pressure with respect to adjacent clean areas. The exhaust airflow rate is modulated by differential pressure control dampers to provide outside makeup air to the exhaust fan when the VFS exhaust air subsystem is connected to the VAS-served zone area(s). The VFS is automatically isolated from the containment if purging is in progress and the standby exhaust filtration train does not start. One VFS train can be manually aligned to continue containment purging while the other VFS train is aligned to exhaust effluent from the zone areas. If both exhaust filtration trains are connected to containment, one exhaust filtration train is automatically isolated from the containment and is realigned to the zone area(s). The VFS exhaust air subsystem can be manually connected to the diesel generators during a LOOP. The VFS is not credited for a design basis FHA or a LOCA, but it may be used if it is operational and onsite power is available to support post-event recovery operations.

Section 7.3 of the SSAR describes the VFS instrumentation. The VFS is controlled by the plant control system, except CIVs, which are controlled by the protection and safety monitoring system (PMS) and diverse actuation system (DAS). The instrumentation to satisfy Table 4-2 of ASME N509-1989 for the VFS air filtration units is discussed in SSAR Section 9.7.1.5. The status indication and alarms are provided to monitor fans, control dampers and control valves. All fans and AHUs can be started remotely or shutdown from the MCR or locally. The temperature controllers maintain the proper supply air temperature. The temperature indication and alarms for high or low supply air temperature are accessible locally via the plant control system to inform operators of abnormal temperature conditions for supply air and charcoal adsorbers. The flow indication and alarms are provided for equipment malfunctions. The radioactivity indication and alarms are provided in the MCR for the gaseous radioactivity in the filtration subsystem's common exhaust duct and gaseous, and particulate and iodine concentrations in the plant vent. The pressure drops across all AHUs and exhaust air filtration unit filters (except charcoal filters) are monitored, and a high-pressure drop is alarmed in the MCR.

The safety-related portions of the VFS are protected against internally and externally-generated missiles, as well as high- and moderate-energy pipe breaks, as discussed in Sections 3.5 and 3.6 of this report.

The staff concludes that the system's safety-related portions comply with the guidelines of Position C.1 of RG 1.29, and the system's non-safety-related portions comply with Position C.2 of RG 1.29, on the basis of the design of the following VFS design features:

- it is located inside the seismic Category I (containment and auxiliary building), flood-protected, and tornado-missile-protected buildings

- classification of the safety-related containment isolation valves, penetrations, interconnecting piping, debris screens, and vent and test connections as seismic Category I as shown in Table 3.2-3 of the SSAR

System equipment and ductwork located in the nuclear island, the failure of which could affect the operability of safety-related systems or components, are designed to seismic Category II requirements to preclude them from collapsing onto safety-related equipment or structures during an SSE. The remaining portion of the system is non-seismic. In Section 3.7.3.13 of this report, the staff evaluated the VFS for interaction with seismic Category I systems to verify that its failure does not reduce the ability of any safety-related plant features to perform their functions. The staff found that the system complies with the requirements of GDC 2.

The staff determined that the system is not an ESF system, and that it is not credited in analyzing the consequences of a DBA, except for containment isolation. Therefore, the requirements of GDC 4 and 17 are not applicable to this system.

As stated previously in Section 9.4 of this report, the HVAC design described in the SSAR does not share SSCs with other nuclear power units. Therefore, the VFS meets the requirements of GDC 5.

The filtered VFS exhaust is monitored and then routed to the plant vent in compliance with the guidelines of RG 1.140 for controlling the release of radioactivity as described above. Therefore, the system complies with the requirements of GDC 60, as it relates to the system's capability to suitably control the release of gaseous radioactive effluent to the environment. The VFS filtration units, which also filter the VAS radioactive exhaust, meet the guidance of Position C.4 of RG 1.13, that the spent fuel pool building be equipped with an appropriate ventilation and filtering system to limit the potential release of radioactive iodine and other radioactive materials. Therefore, the staff found that the system complies with the requirements of GDC 61, as it relates to the system capability to provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment.

The staff evaluated the VFS for conformance with GDC 2, 4, 5, 17, 60, and 61, as referenced in Section 9.4.5 of the SRP. The staff finds the VFS acceptable.

9.4.8 Radwaste Building HVAC System

The staff reviewed the radwaste building HVAC system (VRS) in accordance with SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VRS satisfies the following requirements:

- GDC 2, regarding the capability of the system to withstand earthquakes
- GDC 5, regarding sharing systems and components important to safety
- GDC 60, regarding the capability to suitably control release of gaseous radioactive effluents to the environment

After the DSER was issued, Westinghouse provided additional information about the VRS that allowed the staff to continue its review, as discussed below. As a result of this additional information, concerning the components classification including mobile filtration units and fire dampers, shielding of components, specific applicable codes and standards for the VRS equipment, system P&ID, and discrepancies in the SSAR, DSER Open Item 9.4.8-1 is closed.

The VRS serves the radwaste building, which includes the clean electrical/mechanical equipment room, potentially contaminated HVAC equipment room, package waste storage room, waste accumulation room, and mobile system facility. The VRS is located within the radwaste building, except for the portion that connects with the plant vent. The VRS is a non-seismic system and serves no safety-related functions. The VRS is a once-through, non-safety-related ventilation system that operates at 100 percent capacity continuously, with both supply and both exhaust fans on during normal plant operation to maintain suitable temperatures in the radwaste building. During filter replacement operations, the VRS operates at 50 percent capacity, and radwaste processing operations are adjusted to obtain an acceptable temperature in the radwaste building. The supply air system AHUs are located in the electrical/mechanical equipment room at Elevation 100'-0" on the southwest side of the radwaste building. The exhaust air system fans are located in the HVAC equipment room on Elevation 100'-0" in the northwest corner of the radwaste building.

The VRS collects the vented discharges from potentially contaminated equipment and provides for radiation monitoring of exhaust air before release to the environment through the plant vent stack. Radiation monitoring is described in Section 11.5 of this report.

The system description and components classification are provided in SSAR Section 9.4.8, Figure 9.4.8-1, and Table 3.2-3. As identified in Table 3.2-3 of the SSAR, the VRS components are non-nuclear safety class, non-seismic category, and the quality assurance requirements of Appendix B to 10 CFR Part 50 do not apply. Table 9.4-1 of this report describes the industry standards applicable to the components of the VRS.

The VRS is designed to maintain proper operating temperatures in the following areas, depending on the maximum and minimum normal outside temperature conditions shown in SSAR Chapter 2, Table 2-1 and summarized as follows:

- processing areas and storage rooms control between 10 °C and 40.5 °C (50 °F and 105 °F)
- mechanical equipment rooms between 10 °C and 40.5 °C (50 °F and 105 °F)

The radwaste building is maintained at a negative pressure (with respect to ambient environment) to prevent (potentially) unmonitored radioactive releases from the radwaste building. The differential pressure controllers, with sensors located in the general building area and mounted outdoors (shielded from wind effects), automatically modulate the inlet vanes of the AHU supply fans to maintain negative pressure inside the radwaste building with respect to the outdoors. The electric interlocks between the large truck doors and the supply fan flow controller permit the supply air to drop to 2.832 m^3/s (6000 cfm) below the exhaust flow when any truck bay door is open to create a flow into the radwaste building through the open door.

The VRS consists of the supply air system and the exhaust air system. The VRS total flow is 8.495 m^3/s (18,000 cfm), consisting of two 4.248 m^3/s (9,000 cfm) trains. The supply air system consists of two 50-percent capacity AHUs, each with a low-efficiency filter bank, a high-efficiency filter bank, hot water heating coil, chilled water coil, and a centrifugal fan with automatic inlet vanes. The supply of chilled and hot water is provided from the VWS and VYS, respectively. Each AHU draws 100 percent outside air through individual louvered outdoor air intakes. The two AHUs discharge into a common air supply duct distribution system that is routed through the radwaste building. The temperature of the air supplied by the AHUs is controlled by separate cooling and heating temperature controllers with sensors in the general building area. The cooling controllers modulate the control valves on the chilled water supply lines to the AHUs to maintain the desired temperature in the area. The heating controllers modulate the face and bypass dampers of the hot water heating coil in the AHUs to maintain the desired temperature in the area. The hot water unit heaters in the mobile facility, which are controlled by local thermostats, are provided to temper air entering the building when a roll-up door is opened. The hot water unit heater in the electrical/mechanical room operate in response to local thermostats to maintain the minimum required temperature.

The exhaust air system consists of two 50-percent capacity centrifugal fans sized to allow the system to maintain a negative pressure with respect to the adjacent areas, an exhaust air duct collection system, and automatic controls and accessories. The exhaust fans discharge to a common duct, which is routed to the plant vent. A radiation monitor records activity in the common exhaust air system discharge duct and activates an alarm in the MCR when excess activity in the effluent discharge is detected. The exhaust air collection duct inside the radwaste building exhausts air from areas and rooms where low levels of airborne contamination may be present. The exhaust connection points are provided to allow the direct exhaust of equipment located on the mobile systems. The back draft dampers are provided at each mobile system vent connection to prevent blowback through the equipment in the event of exhaust system trip. Where potential high levels of airborne radioactive contamination exist, mobile systems will include HEPA filtration. The mobile processing systems are discussed in SSAR Section 11.2 and Section 11.4 of this report.

The VRS is designed to permit periodic inspection of system components during normal plant operation and is controlled by the PLS. (Refer to SSAR Section 7.1.1 for a discussion of the PLS.) The temperature is indicated for each AHU supply air discharge duct. Local differential pressure indications and high-pressure alarms are provided for the AHU and exhaust air system air filters to alert the operator to the need for filter replacement. An alarm is provided for high radiation in the main exhaust duct to the vent stack. Airflow indications are provided for the AHU and exhaust fan discharge ducts, and low flow alarms are provided in the fan discharge ducts. The operational status indications for the fans are provided in the MCR. The fans and AHUs can be initiated or shutdown from the MCR. An alarm is provided for smoke in the common AHUs discharge duct. Position indicating lights are provided for automatic dampers. The VHS instrumentation is described in Section 7 of this report.

Compliance with Position C.1 of RG 1.29 does not apply because the VRS is not designed to perform any safety functions. The VRS complies with Position C.2 of RG 1.29 as follows:

- The VRS serves no safety-related function. Failure of the system does not affect the operation of safety-related structures, systems, or components because none of these are located in the area served by the system.

- The system is non-safety-related and is not credited to operate during any abnormal plant conditions.

Therefore, the system complies with the requirements of GDC 2.

As stated previously in Section 9.4 of this report, the HVAC design described in the SSAR does not share SSCs with other nuclear power units. Therefore, the VRS meets the requirements of GDC 5.

The VRS collects the vented discharges from potentially contaminated areas, and provides for radioactive particulate removal and radiation monitoring of exhaust air before release to the environment through the plant vent stack. Therefore, the system meets the requirements of GDC 60, as it relates to the system's capability to suitably control the release of gaseous radioactive effluent to the environment.

The staff evaluated the VRS for conformance with GDC 2, 5, and 60, as referenced in Section 9.4.3 of the SRP. The staff finds the VRS acceptable.

9.4.9 Turbine Building Ventilation System

The staff reviewed the turbine building ventilation system (VTS) in accordance with industry standards and SRP Section 9.4.4, "Turbine Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VTS satisfies the following requirements:

- GDC 2, regarding the capability of the system to withstand earthquakes
- GDC 5, regarding sharing systems and components important to safety
- GDC 60, regarding the capability to suitably control release of gaseous radioactive effluent to the environment

After the DSER was issued, Westinghouse provided additional information about the VTS that has allowed the staff to continue its review as discussed below. As a result of this additional information, concerning the system's P&ID, design parameters, and classification of the system and its components, DSER Open Item 9.4.9-1 is closed.

The VTS operates during startup, shutdown, and normal plant operations. The VTS consists of (1) the general area ventilation subsystem, (2) the electrical equipment and personnel work area HVAC subsystem, and (3) the local area heating and ventilation subsystem. The general area ventilation subsystem serves the operating deck, intermediate levels, and base slabs. The electrical equipment and personnel work area HVAC subsystem serves switchgear rooms 1 and 2, the electrical equipment room, the variable frequency drive (VFD) power converter room, and personnel work areas (secondary sampling laboratory and office spaces) at Elevation 149'-0" and the engineering work station at Elevation 171'-0". The local area heating and ventilation subsystem serves the lube oil reservoir room, clean and dirty lube oil storage room, toilet areas (facilities), auxiliary boiler room, and motor-driven fire pump room. The VTS maintains the air temperature of all areas inside the turbine building between 10 °C and 40.6 °C (50 °F and 105 °F), except for the personnel work areas, which are maintained between 22.8 °C and 25.6 °C (73 °F and 78 °F), depending on the maximum and minimum normal outside temperature conditions shown in SSAR Chapter 2, Table 2-1.

The system description and components classification are provided in SSAR Section 9.4.9, Figure 9.4.9-1, and Table 3.2-3. As identified in Table 3.2-3 of the SSAR, the VRS components are non-nuclear safety class, non-seismic category, and the quality assurance requirements of Appendix B to 10 CFR Part 50 do not apply. Table 9.4-1 of this report describes the industry standards applicable to the components of the VTS.

The VTS neither serves nor supports the plant's safety-related functions; therefore, the system need not be designed to meet the guidelines of RGs 1.29 and 1.140 or to withstand the effects of an SSE. Some areas of the turbine building have a potential for radioactive contamination, but any contamination is expected to be low. Radiological monitors are provided in the turbine building to detect system leakage in the condenser air removal, steam generator blowdown, component cooling water, and main steam systems. The radiation monitoring system (RMS) is addressed in Sections 11.5 of this report.

The VTS is designed to permit periodic inspection of system components during normal plant operation. The VTS is monitored by the plant monitoring system, and controlled by the plant control system. Temperature indication is provided to allow temperature surveillance of room and space temperatures in the turbine building. Controllers are provided to control the room air temperatures to within a predetermined range. Differential pressure indication and high-pressure alarms are provided for the AHU air filters.

9.4.9.1 General Area Heating and Ventilation Subsystem

The general area ventilation subsystem serves most of the turbine building and is manually controlled. The subsystem consists of roof mounted exhaust ventilators and wall mounted louvers. The ventilators are hooded, direct driven, propeller type with pneumatically actuated back draft dampers. The wall louvers are located at Elevation 100'-0", Elevation 117'-6", and Elevation 135'-3". During heating operation, the general area ventilation subsystem is not operated. Additionally, the operating floor wall louvers are normally closed during power operation and are manually opened during outage operations for ventilation.

The general area heating subsystem is manually or automatically controlled. The system consists of hot water unit heaters and heater fans, and provides local heating throughout the turbine building. The system heater fan motors are controlled by thermostats in the automatic mode. The hot water is supplied from the VYS.

9.4.9.2 Electrical Equipment and Personnel Work Area HVAC Subsystem

This HVAC subsystem consists of an independent electrical equipment area HVAC system and an independent personnel work area HVAC system.

The subsystem chilled water is supplied from the VWS and hot water is supplied from the VYS. The subsystem maintains served areas at a slight positive pressure by mixing outside air with the recirculated air. The subsystem room thermostats control the chilled water control valves for cooling and the integral face/bypass dampers for heating.

Each independent system, serving corresponding areas, consists of two 50-percent AHUs located at Elevation 149'-0". Each AHU consists of a mixing section, high- and low- efficiency filters, integral face and bypass damper, a hot water heating coil, chilled water cooling coil, and centrifugal fan with automatic inlet vanes. The electrical equipment area HVAC system is provided with a return and exhaust fan and an integral face and bypass damper. The electrical equipment HVAC system's AHU has a capacity of about 14,000 scfm, and the personnel work area HVAC system's AHU has a capacity of about 10,700 scfm. Electric reheat coils are provided in the ductwork to each room served by the personnel work area HVAC system, to maintain close temperature control. During normal operation all AHUs operate continuously.

9.4.9.3 Local Area Heating and Ventilation Subsystem

The lube oil reservoir room, clean and dirty lube oil storage room, toilet areas (facilities), and secondary sampling laboratory fume hood have centrifugal exhaust fans to remove flammable vapors, odors, or chemical fumes as required.

A direct-drive, two-speed, wall exhaust ventilator is provided for each of the auxiliary boiler room, and the motor-driven fire pump room. The ventilators are two speed, propeller type with pneumatically actuated backdraft dampers. The air is pulled from the general area of the turbine building through the fire damper openings and exhausted to the atmosphere. Each exhaust ventilator is automatic or manually controlled. In the automatic mode, the exhaust ventilator motor is controlled by a two-stage room thermostat. In the manual mode, the exhaust fan runs continuously at high speed until it is stopped manually.

The fire pump room is heated by the hot water-to-unit heater supplied from the VYS for freeze protection. The hot water unit heater fan motors are controlled by a thermostat during automatic mode, but run continuously in manual mode.

No hot water heating is provided in the auxiliary boiler room. The auxiliary boiler room exhaust fan pulls air from the general area of the turbine building. A heating thermostat is provided in the boiler room to control the operation of the fan when temperature falls below 10 °C (50 °F). The boiler room exhaust fan starts at low speed and continues to run until the space temperature rises above 10 °C (50 °F).

9.4.9.4 Conclusion

Compliance with Position C.1 of RG 1.29 does not apply because the VTS is non-seismic and is not designed to perform any safety functions. The VTS complies with Position C.2 of RG 1.29 as follows:

- The VTS serves no safety-related function. Failure of the system does not affect the operation of safety-related structures, systems, or components because none of these are located in the area served by the system.

- The system is non-safety-related and is not credited to operate during any abnormal plant conditions.

Therefore, the system complies with the requirements of GDC 2.

As stated previously in Section 9.4 of this report, the HVAC design described in the SSAR does not share SSCs with other nuclear units. Therefore, the VTS meets the requirements of GDC 5.

As stated above, radiological monitoring is provided in the turbine building in the condenser air removal, steam generator blowdown system (BDS), component cooling water (CCS), and main steam systems to detect system leakage for any potential radioactive contamination. The RMS is addressed in Section 11.5 of this report. Also, provisions for temporary barriers are provided around the BDS, CCS, and condensate polishing areas for radiological protection. Therefore, the system is in compliance with the requirements of GDC 60.

As described above, the staff evaluated the VTS for conformance with GDC 2, 5, and 60, as referenced in Section 9.4.4 of the SRP. The staff finds the VTS acceptable.

9.4.10 Diesel Generator Building Heating and Ventilation System

The staff reviewed the diesel generator building heating and ventilation system (VZS) in accordance with SRP Section 9.4.5, "Engineered Safety Feature Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VZS satisfies the following requirements:

- GDC 2, regarding the capability of the system to withstand earthquakes
- GDC 4, regarding maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions
- GDC 5, regarding sharing systems and components important to safety
- GDC 17, regarding the assurance of proper functioning of essential electric power systems
- GDC 60, regarding the capability to suitably control release of gaseous radioactive effluent to the environment

The VZS serves the standby diesel generator (DG) rooms, electric equipment service modules, and diesel fuel oil day tank vaults in the DG building, and the two diesel oil transfer modules located in the yard. The VZS consists of the normal heating and ventilation subsystem, the standby exhaust ventilation subsystem, the fuel oil day tank vault exhaust subsystem, and the diesel oil transfer module enclosures ventilation and heating subsystem.

After the DSER was issued, Westinghouse provided additional information about the VZS that allowed the staff to continue its review, as discussed below. As a result of this additional information concerning the AHU outlet temperature and its control, the air filters and louver locations for outside air intake, and the equipment operability for the equipment located inside DG area exposed to 54 °C (130 °F) while the DG is in operation, DSER Open Item 9.4.10-1 is closed.

As identified in Table 3.2-3 of the SSAR, the VZS components are a non-nuclear safety class and non-seismic category. As such, the quality assurance requirements of Appendix B of 10 CFR Part 50 do not apply. The system description and layout drawings are provided in

Section 9.4.10 and Figure 9.4.10-1 of the SSAR, respectively. Table 9.4-1 of this report describes the industry standards applicable to the components of the VZS.

The two redundant DGs and associated equipment that provide standby ac power in the event of a LOOP are located in separate rooms of the non-safety-related DG building. Each DG room is served by an independent train of the VZS that provides normal heating and ventilation to continuously maintain acceptable environmental conditions in the area when the DGs are not operating. The standby exhaust ventilation portion of the VZS operates when the corresponding DG is in operation to maintain acceptable temperatures for equipment operation and reliability, in order for the onsite standby power system to perform its DID function. Within each DG room, an electrical equipment service module houses the DG electrical and control support equipment. Each DG has its own dedicated diesel oil transfer module with an enclosure and is located in the yard. The DGs are not safety-related and are not essential for the safe shutdown of the plant. The VZS, which supports the operation of the DGs, is also not safety-related.

The VZS is designed to maintain the temperature inside the DG area between 10 °C (50 °F) and 40.6 °C (105 °F) when the DG is not operating and a maximum of 54.4 °C (130 °F) when the DG is operating. In Section 8.3.1.1.2.1 of the SSAR, Westinghouse states that the DGs will be procured to be consistent with the VZS (i.e., design requirement of a maximum of 54.4 °C (130 °F) when the DG is operating), as described in SSAR Section 9.4.10.

The VZS also maintains the temperature inside the electrical equipment service modules between 10 °C (50 °F) and 40.6 °C (105 °F) at all times. Each dedicated diesel oil transfer module is maintained between 10 °C (50 °F) and 40.6 °C (105 °F) inside an enclosure. Two electric unit heaters are provided in each DG room, which maintains the space at 10 °C (50 °F) when the DGs are not operating. The VZS is designed for -20.6 °C to 35 °C (-5 °F to 95 °F) ambient conditions, which are the 5 percent exceedance values.

Each train of the normal heating and ventilation subsystem consists of one 100-percent capacity engine room AHU that ventilates the diesel generator room, one 100-percent capacity service module AHU that ventilates the electrical equipment service module, an exhaust system for the diesel generator room, an exhaust system for the fuel oil day tank vault, and two electric unit heaters in the DG area. The engine room AHUs are located above the electrical equipment service module with HVAC ducting in the DG rooms. The service module AHUs are located above the service module with HVAC ducting into the module. Outside air is supplied to each AHU through a wall-mounted fixed louver. Air intake louvers are located as high in the DG building wall as possible which meets the intent of the guidance of NUREG/CR-0660 to control the dust and other particulates for conformance with GDC 17, as it relates to ensuring proper functioning of the standby onsite ac electric power system.

Each AHU of the normal heating and ventilation subsystem consists of a mixing box section, a high-efficiency filter bank, a low-efficiency filter bank, and a centrifugal fan. During normal plant operation, the engine room AHU runs continuously when the DG is off and outdoor air is required for room cooling. The space thermostats control the proportion of outside air that is mixed with return air to maintain adequate temperature in the engine room served areas. The excess outside air supplied to the engine room is discharged to the outdoors via a gravity relief damper.

Each service module AHU has an electrical heating coil that is controlled by a separate space thermostat. The outside air is supplied to each AHU through a wall-mounted fixed louver. The service module excess outside air flows into the diesel engine area via a wall-mounted relief damper. The service module AHU operates continuously regardless of the DG status.

Each train of the standby exhaust ventilation subsystem consists of two 50-percent capacity roof-mounted exhaust fans and two motor-operated air intake dampers mounted in the exterior walls of the room. The exhaust fans turn on when the DGs start and turn off when the DGs stop. The standby exhaust fans are actuated by a DG start signal, as shown in SSAR Figures 9.4.10-1 and 9.4.10-2. The motor-operated air intake dampers open and close in conjunction with the operation of the exhaust fans. One or both standby exhaust fans are required to operate to maintain the engine room temperature in reference to ambient temperature. The subsystem is required to operate to support DG operation during LOOP.

Each fuel oil day tank vault is continuously ventilated by a 100-percent capacity centrifugal exhaust fan. The exhaust fans are roof mounted and ducted to draw air from 0.3 m (1 ft) above the vault floor to remove any oil fumes generated in the space. Air is drawn into each fuel oil tank vault from the DG room through a fire damper. The fans are manually operated.

The diesel oil transfer module enclosures are serviced by a separate exhaust ventilation system. Each diesel oil transfer module enclosure is ventilated by a 100-percent capacity roof-mounted exhaust fan. Outside air is drawn into the enclosure through manually operated louvered air intakes; these louvers are closed during winter operation when heating is required. An electric unit heater is provided in each enclosure to maintain the space at a minimum temperature of 10 °C (50 °F). The subsystem is required to operate to support DG operation during LOOP.

Because the VZS has two 50-percent capacity exhaust fans for each DG room and one 100-percent capacity AHU for each service module, at least one DG train will be fully operational should a single fan failure occur.

The system design allows for periodic inspection of the system's components. Refer to SSAR Section 7.1.1 for the plant control system. The system temperature indication and alarms are accessible locally and via the plant control system. The operational status indications for the fans are provided in the MCR. All fans and AHUs can be operable locally or from the MCR. Differential pressure indication for each filter in the AHUs and a high-pressure drop alarm for each AHU are provided.

Compliance with Position C.1 of RG 1.29 does not apply because the VZS is not designed to perform any safety functions. The VZS complies with Position C.2 of RG 1.29 as follows:

- The VZS serves no safety-related function. Failure of the system does not affect the operation of safety-related structures, systems, or components because none of these are located in the area served by the system.

- The system is non-safety-related and is not credited to operate during any abnormal plant conditions.

Therefore, the system design complies with the requirements of GDC 2 and is acceptable.

The VZS is not required to maintain a controlled environment in areas containing safety-related equipment, and areas served by the VZS do not contain equipment essential for the safe shutdown of the reactor or necessary to prevent or mitigate the consequences of a DBA. Therefore, GDC 4 is not applicable to the VZS.

As stated previously in Section 9.4 of this report, the HVAC design described in the SSAR does not share SSCs with other nuclear power units. Therefore, the VZS meets the requirements of GDC 5.

The onsite standby power system (ZOS) includes two diesel generators housed in the non-safety-related DG building. Westinghouse states that the ZOS and DGs are not safety-related; therefore, they are not essential for the safe shutdown of the reactor, nor are they necessary to prevent or mitigate the consequences of a DBA. Westinghouse further states that the VZS is physically separated from potentially contaminated areas, and does not contain any radioactive materials. Therefore, compliance with the recommendations of RGs 1.52 and 1.140 and the requirements of GDC 60 are not required for the VZS.

The staff evaluated the VZS for conformance with GDC 2, 4, 5, 17, and 60, as referenced in Section 9.4.5 of the SRP. The staff finds the VZS acceptable.

9.4.11 Health Physics and Hot Machine Shop HVAC System

The staff reviewed the health physics and hot machine shop HVAC system (VHS) in accordance with SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System." Conformance with the SRP acceptance criteria forms the basis for concluding whether the VHS satisfies the following requirements:

- GDC 2, regarding the capability of the system to withstand earthquakes
- GDC 5, regarding sharing systems and components important to safety
- GDC 60, regarding the capability of the system to suitably control release of gaseous radioactive effluent to the environment

After the DSER was issued, Westinghouse provided additional information about the VHS that allowed the staff to continue its review, as discussed below. As a result of this additional information concerning the AHU outlet temperature and its control, discrepancies in the SSAR, and radiation monitoring for the hot machine shop, DSER Open Item 9.4.11-1 is closed.

The VHS collects the vented discharges from potentially contaminated sumps and equipment in the health physics area, as well as exhaust from welding booths, grinders, and other equipment located in the hot machine shop. It also monitors exhaust air for radiation before its release to the environment through the plant vent stack. The radiation monitoring is described in Section 11.5 of this report.

The VHS serves no plant safety-related functions and there are no safety-related structures, systems, or components in the area serviced by the system. The VHS is a once-through, non-safety-related ventilation system that operates only during normal modes of plant operation. The VHS is located within the annex building, except for the portion that connects with the plant vent. The VHS serves both the health physics/access control area in the annex building located at Elevation 100'-0" and the hot machine shop located at Elevation 107'-2" in the annex building. These areas are maintained at a slight negative pressure with respect to the outdoors and clean areas to ensure that all potentially radioactive releases are monitored before discharge.

The supply air subsystem AHUs are located in the lower south air handling equipment room on Elevation 135'-3" of the annex building, and the exhaust air subsystem fans are located in the staging and storage areas on Elevation 135'-3" of the annex building.

As identified in AP600 SSAR Table 3.2-3, the VHS components are non-nuclear safety class, non-seismic category, and the quality assurance requirements of 10 CFR Part 50, Appendix B do not apply. The system description and layout drawings are provided in Section 9.4.11 and Figure 9.4.11-1 of the SSAR, respectively. Table 9.4-1 of this report describes the industry standards applicable to the components of the VHS.

The VHS maintains the direction of air flow from areas of low potential radioactivity to areas of higher potential radioactivity. The VHS is designed to maintain the temperature of the health physics area at 22.8 to 25.6 °C (73 to 78 °F), and the hot machine shop at 18.3 to 29.4 °C (65 to 85 °F), depending on the maximum and minimum normal outside temperature conditions shown in SSAR Chapter 2, Table 2-1. The VHS is designed to maintain a minimum RH of 35 percent in normally occupied areas via a steam humidifier located in the main system supply duct. The water for the system humidifier is provided by the demineralized water system.

The differential pressure controllers, with sensors in the general health physics area and sensors mounted outdoors (shielded from wind effects), modulate the automatic inlet vanes of the supply fan to maintain the area at a negative pressure with respect to the surrounding areas which do not have their exhausts monitored for radioactivity. A separate differential pressure controller, with a sensor in the hot machine shop, modulates a damper in the supply air duct to the hot machine shop to maintain a negative pressure with respect to the outdoors.

The VHS consists of the supply air subsystem and the exhaust air subsystem. The supply air subsystem consists of two 100-percent capacity (14,000 scfm each) AHUs, each with a low-efficiency filter bank and a high-efficiency filter bank; hot water heating coil; chilled water cooling coil bank, a centrifugal fan with automatic inlet vanes, associated dampers, instrumentation, and controls; and ductwork. Each AHU draws 100 percent outside air through a common louvered outdoor air intake plenum #2, as described in SSAR Section 9.4.2. The two AHUs discharge into a distribution system to the health physics and hot machine shop areas. The temperature in the health physics and hot machine shop areas are maintained within the design range by a temperature sensor located in the health physics area, which modulates the control valve on the chilled water supply lines to the cooling coil and the face and bypass dampers of the hot water heating coil. The supply of the chilled and hot water is provided from the VWS and VYS, respectively.

The exhaust air subsystem consists of two 100-percent capacity centrifugal exhaust fans, sized to maintain a negative pressure with respect to the adjacent areas, with ductwork and automatic controls. A separate machine shop exhaust fan and high-efficiency filter are provided for exhausting from machine tools and other localized areas in the hot machine shop. The air flow rates are balanced to maintain a constant exhaust air flow across the fans. The exhaust fans discharge to a common duct, which is routed to the plant vent stack. Individual flexible exhaust duct branches are provided to machine tools. The flexible ducts are connected to a hard duct manifold, which is connected to a filter and a fan. The exhaust fan discharges into the main system exhaust ductwork.

One supply AHU and one exhaust fan are capable of maintaining corresponding areas at the designed temperatures, at a slight negative pressure, and with the direction of system air flow from areas of low radioactivity to areas of high radioactivity, should a single fan failure occur.

The health physics area, including the hot machine shop, is monitored by two non-safety-related radiation monitors. One is an area monitor located in the hot machine shop and the other is an air exhaust monitor located in the common VHS exhaust duct. High-radiation alarms are provided both locally and in the MCR. SSAR Sections 9.4.11 and 11.5, and Tables 11.5-1 and 11.5-2 describe these radiation monitors.

Temperature indication is provided for each AHU supply air discharge duct. Local differential pressure indications and MCR high-pressure alarms are provided for the AHU and exhaust air system air filters. The remote manual hand switches and alarms for the system fans are provided in the MCR. The fans and AHUs can be initiated or shutdown from the MCR. An alarm is provided for smoke in the common AHUs discharge duct. Position indicating lights are provided for automatic dampers.

Compliance with Position C.1 of RG 1.29 does not apply because the VHS is not designed to perform any safety functions. The VHS complies with Position C.2 of RG 1.29 as follows:

- The VHS serves no safety-related function. Failure of the system does not affect the operation of safety-related structures, systems, or components because none of these are located in the area served by the system.

- The system is non-safety-related and is not credited to operate during any abnormal plant conditions.

Therefore, the system design complies with the requirements of GDC 2 and is acceptable.

As stated previously in Section 9.4 of this report, the HVAC design described in the SSAR does not share SSCs with other nuclear power units. Therefore, the VHS meets the requirements of GDC 5.

The VHS is not safety-related, performs no safety-related function for safe-shutdown or post-accident operation, and failure of the system does not affect the function of other safety-related equipment. The annex building lower south air handling equipment room, where VHS supply AHUs are located, has no sources of radioactivity during normal plant operation. The hot machine shop mezzanine area, where hot machine shop exhaust fans are located, is not a high-radioactivity area. The shielding of components and personnel is commensurate with radiation sources in the vicinity of the VHS equipment during normal plant operation.

The VHS collects the vented discharges from potentially contaminated areas and provides for radiation monitoring of exhaust air prior to release to the environment through the plant vent stack. Therefore, the requirements of GDC 60 are met.

The staff evaluated the VHS for conformance with GDCs 2, 5, and 60, as referenced in Section 9.4.3 of the SRP. The staff finds the VHS acceptable.

9.5 Other Auxiliary Systems

The staff's review of the other AP600 auxiliary systems is provided in the following sections: 9.5.1, Fire-Protection System; 9.5.2, Communication System; 9.5.3, Plant Lighting System; 9.5.4, Diesel Generator Fuel Oil Storage and Transfer System; 9.5.5, Standby Diesel Engine Cooling System; 9.5.6, Standby Diesel Engine Starting System; 9.5.7, Standby Diesel Lubricating Oil System; and 9.5.8, Standby Diesel Combustion Air Intake and Exhaust System.

9.5.1 Fire Protection Program

The fire protection criteria for the AP600 are specified in SECY-90-016, SECY-93-087, and SECY-94-084. In addition, 10 CFR 52.48 specifies that the design will comply with the requirements specified in 10 CFR 50.48, "Fire protection," and GDC 3, "Fire protection," of Appendix A to 10 CFR Part 50. Conformance with the SRP is addressed in 10 CFR 50.34(g), which specifies that applications include an evaluation of the facility against the SRP. The fire protection guidance for nuclear power plants specified in the SRP is provided in BTP CMEB 9.5-1 "Guidelines for Fire Protection for Nuclear Power Plants." In addition to the guidance specified in the BTP, consistent with the technical positions stated in SECY-90-016, SECY-93-087, and Section 9.3 of NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," Volume 3, the staff specified that the advanced light-water reactors shall provide an enhanced level of fire protection to ensure that safe shutdown can be achieved assuming all equipment in any one fire area is rendered inoperative as a result of fire damage, and that reentry into the fire area by plant personnel for repairs or operator actions is not possible. The control room and the containment are excluded from this criterion, provided an independent alternative shutdown capability is provided for a control room fire, and that fire protection for redundant divisions located inside containment is provided to ensure that one shutdown division will be free of fire damage following a fire inside the containment. The design must also ensure that smoke, hot gases, and fire suppressants do not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. The NRC staff interpretations and positions related to fire protection which are published in generic communications were used as applicable in the review of the AP600. In addition, the applicable National Fire Protection Association (NFPA) codes, standards and recommended practices were applied to the fire protection systems and features provided in the AP600 design.

The following evaluation is based on the staff's review of the markup of Revision 18 of the SSAR dated March 6, 1998. Westinghouse substantially revised the AP600 fire protection design since the staff issued the DSER. Therefore, the evaluation documented herein replaces the DSER in its entirety.

The staff identified a concern regarding the AP600 fire protection design in the Advanced FSER. This concern was identified in a letter to Mr. Nicholas J. Liparulo from Jack W. Roe, dated April 8, 1998. The staff was concerned that the design appears not to meet the intent of

Commission policy identified in SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." SECY 93-087 states that passive plant designs are to be reviewed using the newest industry standards endorsed by the NRC. Concerns regarding this issue centered on the location of the fire pumps in the turbine building. Specifically, the fire pumps should be protected against possible interruption of service caused by explosion and fire as specified by NFPA 20. Westinghouse was requested to reassess the fire pump location and layout to demonstrate how they provide an equivalent level of facility fire safety comparable to that of pumps at operating plants and in the recently certified evolutionary plant designs. Also, Westinghouse was asked to reassess how the AP600 design addresses the latest fire protection standards endorsed by the NRC. This was FSER Open Item 9.5.1-1.

In response to Open Item 9.5.1-1, SSAR, Revision 23 reassessed the AP600 fire pump location. As part of this reassessment, Westinghouse revised the fire pump design such that the diesel-driven fire pump, which had been previously located in the turbine building, is relocated to a separate prefabricated structure located in the yard. The staff concludes that the revised fire pump location meets the applicable regulatory review guidance and provides an equivalent level of fire safety compared to that of pumps at operating plants and the certified evolutionary plant designs. The staff reviewed the AP600 design using guidance and criteria consistent with current NRC fire protection criteria; therefore, the staff concludes that the AP600 fire protection design is acceptable. Open Item 9.5.1-1 is closed.

9.5.1.1 Fire Protection Program Requirements

<u>9.5.1.1.a Fire Protection Program</u>

The establishment of a fire protection program at the facility for the protection of structures, systems, and components important to safety, and the procedures, equipment and personnel required to implement the program is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(a).

<u>9.5.1.1.b Fire Hazard Analysis</u>

Westinghouse has provided the fire hazard analysis for the AP600 design in Section 9.5.1 and Appendix 9A of the SSAR. This analysis demonstrates that the plant will maintain the ability to perform safe shutdown functions, minimize radioactive releases to the environment, identify fire hazards and appropriate protection, and verify that NRC fire protection guidelines have been met. The revision of the fire hazard analysis to reflect the actual plant configuration is the responsibility of the COL applicant. This is COL Action Item 9.5.1-2.

The staff has determined that the design commitments in the following figures of SSAR Section 9.5.1, Appendix 9A, Fire Protection Analysis, if considered for a change by a COL applicant or licensee will require NRC review and approval prior to implementation of the change. The commitment identified below should be listed in the proposed rule certifying the AP600 design as Tier 2* information.

AP600 SSAR	DESCRIPTION
Figure 9A-1	Nuclear Island Fire Area Plan
Figure 9A-2	Turbine Island Fire Area Plan
Figure 9A-3	Annex I & II Building Fire Area Plan
Figure 9A-4	Radwaste Building Fire Area Plan
Figure 9A-5	Diesel Generator Building Fire Area Plan

9.5.1.1.c Fire Suppression System Design Basis

The fire suppression systems located inside the containment and outlying buildings are subject to a single active failure or crack that could impair both the primary and backup fire suppression capabilities, which is not in accordance with the guidance specified in the BTP. The fire suppression systems located inside the containment are qualified to seismic Category I criteria, which reduces the potential for a failure of the system. The buildings outside containment do not contain safety-related equipment, or present an exposure hazard to structures containing safety-related equipment. Manual fire suppression capability using hose lines connected to the outside hydrants of the yard main can be provided in the event of a failure of the interior fire suppression systems. On the basis of the seismic qualification of the fire suppression system located inside containment, the finding that there is no safety-related equipment in the outlying buildings, and that there is manual suppression capability using the outside hydrants, the staff concludes that this alternative means of protection is acceptable. The staff also concludes that Westinghouse identified no other exceptions from the guidance specified in the BTP related to fire suppression system design basis and is, therefore, acceptable. This is Deviation 9.5.1-1.

9.5.1.1.d Alternative/Dedicated Shutdown

Westinghouse concluded that alternative or dedicated shutdown capability is not necessary. In Generic Letter (GL) 86-10, the staff stated its position that, for the purpose of analysis to Section III.G.2 of Appendix R to 10 CFR Part 50 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. Therefore, the staff concludes that the safety-related PXS and PCS used to achieve and maintain safe shutdown following a fire in the AP600 are acceptable as an alternative/dedicated shutdown method for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP.

9.5.1.1.e Implementation of Fire Protection Program

The implementation of the fire protection program prior to receiving fuel onsite for fuel storage areas, and for the entire unit prior to reactor startup is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(b).

9.5.1.2 Administrative Controls

The establishment of administrative controls to maintain the performance of the fire protection systems and personnel is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(c).

9.5.1.3 Fire Brigade

The establishment of a site fire brigade trained and equipped for fire fighting to ensure adequate manual fire fighting capability for all plant areas containing structures, systems, or components important to safety is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(d).

9.5.1.4 Quality Assurance Program

The establishment of a quality assurance program to ensure that the guidelines for the design, procurement, installation, and testing, as well as the administrative controls for fire protection systems are satisfied is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(e).

9.5.1.5 General Plant Guidelines

9.5.1.5.a Building Design

The safety-related structures, the containment and auxiliary building non-RCAs, are separated from non-safety-related structures, the turbine building, annex building, radwaste building, diesel generator building, and auxiliary building (RCAs), by barriers having a minimum fire resistance rating of three hours. With the exception of the control room, the remote shutdown workstation, and the containment, fire barriers with a minimum fire resistance rating of three hours are provided to separate redundant divisions of the passive safety-related systems.

Openings through fire barriers for pipe conduit and cable trays are sealed with noncombustible materials to provide a fire resistance rating equal to that required by the barrier, qualified in accordance with the criteria specified in the BTP. Penetrations for ventilation systems are protected in accordance with the criteria specified in NFPA 90A, "Air Conditioning Systems." Doors installed in fire barriers are qualified in accordance with the criteria specified in NFPA 80, "Fire Doors and Windows." Inspection and maintenance of fire doors, access to keys for the fire brigade, and the marking of exit routes is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(f).

Personnel access and egress routes are provided for each fire area. Stairwells outside containment serving as access or egress routes are enclosed in gypsum towers with a minimum fire resistance rating of two hours equipped with self-closing doors with a fire resistance rating of one and a half hours. The BTP specifies that stair towers should be enclosed in masonry or concrete towers; however, there are no missile hazards in the vicinity of the subject stairwells. Therefore, the alternative protection provides an equivalent level of safety as that prescribed in the BTP and is, therefore, acceptable. This is Deviation 9.5.1-2.

There are no cable spreading rooms in the AP600 design. Therefore, the guidance specified in the BTP addressing separation of cable spreading rooms is not applicable. Gaseous suppression systems are not used in the AP600 design. Therefore, the guidance specified in the BTP addressing gaseous suppression systems is not applicable.

Interior finish, wall, ceiling, structural components, thermal insulation, radiation shielding, and soundproofing materials used in the AP600 are noncombustible. Metal deck roof construction is noncombustible and listed as Class I in the Factory Mutual Approval Guide. With the exception of the combustible cable insulation installed in the underfloor and ceiling spaces in the control room, technical support center and remote shutdown workstation, the concealed spaces are free of combustible materials. The BTP specifies that concealed spaces should be devoid of combustibles. Fire detection is provided in the concealed areas containing cables, and the cables used in the plant are qualified in accordance with the criteria specified in IEEE 1202, "Standard for Flame Testing of Cables for Use in Cable Tray in Industrial and Commercial Occupancies." The alternative protection provides an equivalent level of safety as that specified in the BTP and is, therefore, acceptable. This is Deviation 9.5.1-3.

Transformers installed in safety-related areas are either dry type or contain a noncombustible liquid. Outdoor transformers are located at least 15.2 m (50 ft) from other structures or are separated by blank fire walls with a minimum fire resistance rating of three hours. Outdoor oil-filled transformers are provided with oil containment or drainage away from structures.

Floor drains of adequate capacity are provided in areas containing safety-related equipment to remove fire suppression water discharged from fixed or manual fire suppression systems. The collection and sampling of water drainage from areas that may contain radioactivity is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(g).

Drains installed in areas containing combustible liquids are equipped with backflow prevention to preclude the flow of combustible liquids into areas containing safety-related equipment.

With the exception of the gypsum stair towers and cable insulation installed in the concealed spaces of the main control room, technical support center, and remote shutdown workstation, Westinghouse identified no exceptions from the guidance specified in the BTP; therefore, the staff finds the design acceptable.

9.5.1.5.b Safe Shutdown Capability
9.5.1.5.c Alternative or Dedicated Shutdown Capability

The NRC staff developed the following criteria for the protection of safe and cold shutdown capability following a single fire in any fire area of the AP600:

- Safe shutdown following a fire is defined for the AP600 as the ability to achieve and maintain the reactor coolant system (RCS) temperature below 215.6 °C (420 °F) without venting of the primary coolant from the RCS. This is a departure from the criteria applied to the evolutionary plant designs, and the existing plants where safe shutdown for fires applies to both hot and cold shutdown capability. This is Deviation 9.5.1-4.

- Cold shutdown for the AP600 is defined as the ability to achieve and maintain the RCS below 93.3 °C (200 °F), consistent with the criteria applicable to the evolutionary designs and existing plants.

The use of the non-safety-related normal shutdown systems and/or the safety-related passive systems are acceptable to the staff to achieve and maintain safe shutdown following a fire. The safety-related passive systems are considered an alternate/dedicated shutdown method as described in the BTP for fire areas where the normal shutdown systems have not been protected in accordance with the guidance prescribed in the BTP. Consistent with the fire protection criteria for the advance light-water reactors specified in SECY-90-016 and SECY-93-087, redundant divisions of these systems shall be separated such that a fire in any fire area outside of the containment or the MCR will not impair the plant's capability to achieve and maintain safe shutdown as defined above, assuming a loss of all equipment in the affected fire area.

Consideration in the safe shutdown analysis upon personnel entry into the affected fire area to repair or operate equipment to achieve safe shutdown is prohibited as prescribed in SECY-90-16. Personnel entry into the affected fire area to repair or operate equipment necessary to achieve and maintain cold shutdown of the AP600 is acceptable, as a result of the unique capability of the AP600 to remain in safe shutdown using only passive systems for an extended period of time.

The criteria concerning cold shutdown capability deviates from the criteria applied to the evolutionary reactor designs, but is consistent with the criteria applicable to existing plants. To enhance the survivability of the normal safe shutdown and cold shutdown capability in the event of a fire, and to reduce the reliance on the infrequently utilized safety-related passive systems, automatic suppression shall be provided in those fire areas outside containment where a fire could damage the normal shutdown capability, or result in a spurious operation of equipment that could result in a venting of the RCS. This criterion is unique to the AP600 and does not ensure that the normal shutdown capability will be free of fire damage, or that the equipment necessary to achieve and maintain cold shutdown can be repaired within 72 hours. This is Deviation 9.5.1-5.

As a result of the inability of the fire brigade to rapidly enter the AP600 containment in the event of a fire, and the potential for damage to safety-related and normal shutdown equipment, in addition to potential spurious actuation(s) resulting in a venting of primary coolant from the RCS, the protection of circuits and equipment inside containment should be enhanced beyond the criteria specified in BTP CMEB 9.5.1 for existing plants, consistent with the staff's technical position stated in Section 9.3 of NUREG-1242.

Fire areas outside of the containment or the MCR containing redundant equipment and circuits necessary to achieve and maintain cold shutdown using the normal systems specified in Table 9.5.1-4 of the SSAR have been provided with automatic suppression in accordance with the criteria specified in Section 9.5.1.5.c above, and are therefore, acceptable.

In Section 9.5.1.3 of the SSAR, Westinghouse states that manual partial opening and closing of one of the first stage automatic depressurization system (ADS) valves to reduce RCS pressure to allow initiation of the normal RNS will be used in one fire area in the event that the CVS is not available. The use of the first-stage ADS results in a controlled venting of the RCS to the IRWST and does not result in an increase in containment pressure or temperature. The use of the first stage ADS as described above is acceptable as an alternative shutdown method.

Westinghouse provided adequate fire protection of the equipment and circuits located inside containment that are required for safe shutdown to provide reasonable assurance that one division of safety-related equipment will remain free of fire damage, in accordance with the criteria specified above. Hose stations for manual suppression are provided inside containment; however, because of the potential hazard associated with personnel entry into containment during a plant transient, the response of the plant fire brigade may be significantly delayed. Therefore, no credit for manual suppression of fires inside containment during power operations is considered acceptable by the staff. In fire zone 1100 AF 11300B, Westinghouse provided a manually actuated water spray system over the non-safety-related open cable trays in this zone. Both divisions of the passive residual heat removal (PRHR) control valves and PRHR flow transmitters are located in this zone in close proximity to each other. These valves are separated by a noncombustible steel or steel composite barrier. There are no exposed cables in fire zone 100 AF 11300A, which is adjacent to fire zone 1100 AF 11300B. Westinghouse has provided reasonable assurance that one division of the normal or passive safe shutdown capability located inside containment will be maintained free of fire damage; therefore, this aspect of the design is acceptable.

Westinghouse included the reactor head vents for consideration as a high/low pressure interface in accordance with the guidance provided in GL 81-12. Inside containment the cables for the control of one head vent valve in each flow path are routed in separate conduits to prevent a spurious actuation of both valves in the flow path. In areas outside containment, the control room, and the remote shutdown workstation, the power and control circuits are located in separate fire areas. The soft controls located in the control room and remote work stations are not susceptible to fire induced spurious actuation. The dedicated switches located in the control room are located on separate panels, such that a fire may short the switches on one panel, but the unaffected panel will be deenergized before spurious actuation of two valves in the same flow path. Therefore, the Westinghouse resolution of this issue is acceptable.

The spurious actuation of ADS resulting from of hot shorts of control circuits of motor operated valves from a fire in the MCR, remote shutdown workstation, DC equipment rooms, and Class 1E penetration rooms were addressed by Westinghouse. Separation and prompt operator actions are credited to minimize the potential for spurious actuation of ADS. The spurious actuation of ADS does not result in an unrecoverable plant configuration. Therefore, the Westinghouse resolution of this issue is acceptable.

9.5.1.5.d Control of Combustibles

Safety-related systems are separated from concentrations of combustible materials where practicable. Where separation is not possible, appropriate fire protection based on the fire hazard analysis is provided. The BTP specifies that bulk gas storage tanks should not be located inside structures containing safety-related equipment. Breathing air storage tanks are located in the auxiliary building non-RCA. These tanks are safety-related and are provided with over pressure protection and are, therefore, acceptable. High-pressure gas storage containers are located in accordance with the guidance prescribed in the BTP. This is Deviation 9.5.1-6.

The control of the use of compressed gases inside structures is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(h).

The use of plastic materials in the plant is minimized through design and administrative controls. The storage of flammable liquids complies with the criteria specified in NFPA 30, "Flammable Liquids Code."

Hydrogen lines in safety-related areas are designed to seismic Category 1 requirements. The design of the plant hydrogen system complies with the criteria specified in NFPA 50A, "Gaseous Hydrogen Systems."

With the exception of the breathing air storage tanks for the MCR, Westinghouse identified no exceptions from the guidance specified in the BTP and is, therefore, acceptable.

9.5.1.5.e Cable Construction

Cable trays, conduit, and other electrical raceways are constructed of noncombustible and metallic materials in accordance with the criteria specified in the BTP. Electrical raceways are only used for cables.

Safety-related cable trays located outside of containment are separated from redundant divisions and non-safety-related areas by three-hour fire rated barriers. Cable trays located inside containment containing safety-related cables are enclosed in noncombustible steel or steel composite materials. Safety-related cable trays are provided with line-type heat detection and are designed to allow wetting with fire suppression water without causing electrical faults. With the exception of the containment, safety-related cable trays are accessible for manual firefighting. In fire zone 1100 AF 11300B, Westinghouse provided a manually actuated water spray system over the non-safety-related open cable trays in this zone.

Electrical cable is qualified in accordance with the criteria specified in IEEE 1202, "Standard for Flame Testing of Cables for Use in Cable Tray in Industrial and Commercial Occupancies." Miscellaneous storage and piping for combustible liquids or gases are located so as to not present an exposure hazard to safety-related systems.

Westinghouse provided reasonable assurance that one division of the safety-related cables will remain free of fire damage. The staff finds this aspect of the design acceptable.

9.5.1.5.f Ventilation

The ventilation system is designed such that smoke and other products of combustion following a fire can be discharged to an area that will not affect safety-related equipment. With the exception of the nuclear island nonradioactive ventilation system equipment rooms, main steam isolation compartments, and elevator shaft, all other fire areas located in the non-RCA portion of the auxiliary building can use the auxiliary building ventilation system to exhaust smoke directly to the outside. For these areas, and for the containment and RCA portions of the auxiliary building, portable fans and ductwork will be used for smoke removal. The routing of the portable ductwork will not enter any other fire area that contains safety-related equipment. In the diesel generator building, automatic suppression, and smoke and heat ventilation capability is provided. In lieu of providing separate smoke and heat vents in the ancillary diesel generator room of the annex building, automatic suppression is provided. Roof-mounted smoke and heat vents as well as partial area automatic suppression are provided in the turbine building.

The release of smoke and hot gases to the environment is monitored in accordance with the guidance specified in RG 1.101. Westinghouse evaluated the ventilation systems to ensure that inadvertent operation or single failures will not violate the RCAs of the plant.

The power supply and control for the ventilation systems are routed outside of the fire area served by the system. Air intakes for ventilation systems serving areas containing safety-related equipment are located remote from the exhaust air outlets and smoke vents of other fire areas.

There are no safety-related ventilation systems in the AP600 design; therefore, the guidance related to engineered safety feature filters and gaseous suppression systems are not applicable to the AP600.

Westinghouse evaluated the smoke control capability of the normal ventilation system against the criteria specified in NFPA 92A, "Recommended Practice for Smoke Control Systems," including stair tower pressurization in the auxiliary building. Specifically, Westinghouse provided dedicated fans to maintain the minimum design pressure difference across the doors in the stair towers S01 and S02, in accordance with the guidance specified in NFPA 92A. Therefore, the Westinghouse resolution of this issue is acceptable.

9.5.1.5.g Lighting and Communication

Emergency lighting in the MCR and remote shutdown workstation is powered by the Class 1E dc and uninterruptable power supply that has an expected duration of 72 hours in the event of a loss of normal ac power. A loss of the emergency lighting in either the MCR or the remote shutdown workstation will not result in a loss of the emergency lighting in the other area. The emergency lighting in other plant areas is provided by 8-hour battery-powered, fixed, self-contained units to provide safe ingress and egress of personnel and the operation of equipment following a fire, in the event of a loss of the normal lighting. Portable battery-powered lighting is provided for emergency use by plant personnel. This is Deviation 9.5.1-7.

Fixed emergency communications are provided at selected locations, independent of the normal plant communications system.

Portable radio communication for use by the plant fire brigade is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(i).

Westinghouse demonstrated that the emergency lighting and communications provided in the event of a fire provide a level of protection equivalent to that specified in the BTP and is, therefore, acceptable.

9.5.1.6 Fire Detection and Suppression

The COL applicant is responsible for ensuring that any deviations from the applicable NFPA codes and standards in addition to those specified in the SSAR, are incorporated into the final safety analysis report (FSAR) with appropriate technical justification. This is COL Action Item 9.5.1-3.

9.5.1.6.a Fire Detection

Fire detection systems designed and installed in accordance with the criteria specified in NFPA 72, "Protective Signaling Systems," is provided in all plant areas that contain or present a potential fire exposure to safety-related equipment. Westinghouse identified no exceptions from the guidance specified in the BTP and, therefore, this aspect of the design is acceptable.

9.5.1.6.b Fire Protection Water Supply

An underground yard fire main loop, separate from the sanitary or service water system, and designed and installed in accordance with the criteria specified in NFPA 24, "Private Fire Service Mains," is provided for the AP600. In addition, indicating isolation valves are provided to permit maintenance or repair of the fire main and outside hydrants without interrupting the water supply to both the primary and backup fire suppression capability to areas that contain or present an exposure to safety-related equipment. Westinghouse states that the AP600 design is a single-unit plant; therefore, Cross-connections at multi-unit sites is not part of the AP600 design.

Two redundant 100-percent capacity fire pumps (one diesel and one electric, 7571 L/min [2000 gpm] each) which are designed and installed in accordance with the criteria specified in NFPA 20, "Centrifugal Fire Pumps," have been provided. The electric fire pump is separated from the balance of the turbine building by three-hour fire rated construction. The diesel fire pump is located in a separate structure in the yard. An electric driven jockey pump is provided to maintain system pressure.

The outside manual hose installation is sufficient to provide an effective hose stream to any onsite location that could present a fire exposure hazard to structures containing safety-related equipment. Fire hydrants are installed approximately every 76.2 m (250 ft) on the yard main. Hose houses are provided in accordance with the criteria specified in NFPA 24. Threads compatible with the local fire department are provided on all hydrants, hose couplings and standpipe risers.

Four water storage tanks are used for supplying the fire protection water demand. The primary fire water storage tank has a capacity of 1,230.3 kL (325,000 gallons), and the secondary fire water storage tank has a capacity of 1,514.2 kL (400,000 gallons). The PCS tank can supply 68.2 kL (18,000 gallons) by gravity feed to the seismic standpipe and fixed water spray system located inside containment. The passive containment cooling ancillary water storage tank can supply a minimum of 68.2 kL (18,000 gallons) via the motor driven recirculation pumps (378.5 L/min (100 gpm) capacity each) to the seismic standpipe and fixed water spray system located inside containment, in the event the PCS supply is not available. The primary and secondary fire water storage tanks comply with the criteria specified in NFPA 22, "Water

Tanks." The PCS tank and the ancillary passive containment cooling tank, which supplies the seismically qualified standpipes, are not designed in accordance with the criteria specified in NFPA 22. These tanks are seismically qualified, safety-related water storage tanks and are acceptable for fire protection use. In addition, a manually operated valve connecting the seismic standpipe to the yard loop is provided. The passive containment cooling water recirculation pumps are not designed and installed in accordance with the criteria specified in NFPA 20. These deviations from the NFPA standards will not adversely affect the performance of the seismically qualified portions of the fire protection water supply system and are, therefore, acceptable. These are Deviations 9.5.1-8, -9, and -10.

The fire protection water supply from the primary fire water storage tanks is dedicated for fire protection purposes. The secondary fire protection water storage tank is used to supply the containment spray following a severe accident. This deviation provides adequate defense in depth and will not adversely affect the performance of the fire protection water supply and is, therefore, acceptable. This is Deviation 9.5.1-11.

9.5.1.6.c Sprinkler and Standpipe Systems

Automatic sprinkler systems are designed and installed in accordance with the criteria specified in NFPA 13, "Installation of Sprinkler Systems," with the exception of providing individual fire department connections to each sprinkler system. Because the sprinkler systems are supplied by the plant's fire protection water supply, individual connections are not necessary. This deviation is acceptable. This is Deviation 9.5.1-12.

Standpipes for each building are designed and installed in accordance with the criteria specified in NFPA 14, "Installation of Standpipe and Hose Systems," for Class III service with the exceptions of (1) the water supply to the standpipe inside containment is manually operated, and (2) the containment isolation valves controlling the water supply to standpipes inside containment are not listed by an independent testing laboratories for fire protection service. The staff concludes that these deviations from the code will not adversely affect the performance of the hose station and standpipe system and are, therefore, acceptable. These are Deviations 9.5.1-13, and -14.

9.5.1.6.d Halon Systems

Halon fire suppression systems are not used in the design of the AP600, therefore, the guidance specified in the BTP is not applicable.

9.5.1.6.e Carbon Dioxide Systems

Carbon dioxide fire suppression systems are not used in the design of the AP600. Therefore, the guidance specified in the BTP is not applicable.

9.5.1.6.f Portable Fire Extinguishers

Portable fire extinguishers are provided in accordance with the criteria specified in NFPA 10, "Portable Fire Extinguishers." Westinghouse identified no exceptions from the guidance specified in the BTP. The staff finds this acceptable.

9.5.1.7 Specific Plant Areas

9.5.1.7.a Primary and Secondary Containment

Fire protection for the containment is provided as specified in Westinghouse's fire hazard analysis. A lube oil collection system for the RCPs is not required as the four RCPs do not contain oil. Operation of the fire protection suppression systems located inside containment will not compromise the integrity of the containment or other safety-related systems. Smoke and heat detection is provided in the primary containment; smoke detection is provided in the annulus. Manual hose stations are provided in the primary containment and in the lower annulus. Redundant divisions of safety-related cables located in the middle annulus are separated by three-hour fire barriers. Division B and D cables are located in the upper annulus, and Division A and C cables are located in the lower annulus.

Westinghouse provided adequate fire protection inside primary containment to provide reasonable assurance that one division of safe-shutdown equipment and cables will remain free of fire damage. Hose stations for manual suppression are provided inside containment, however, because of the potential hazard associated with personnel entry into containment during a plant transient, the response of the plant fire brigade may be significantly delayed. Therefore, no credit for manual suppression of fires inside containment during power operations is considered acceptable by the staff. Westinghouse provided a manual (operated from the MCR) water spray system in zone 1100 AF 11300B over the exposed cable trays located in this fire zone. Westinghouse identified no exceptions from the guidance specified in the BTP. The staff finds this acceptable.

Fire protection inside containment during refueling and maintenance is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(j).

9.5.1.7.b Control Room

The MCR complex is separated from other plant areas by three-hour rated fire barriers. The MCR fire zone is separated from the shift supervisors office and kitchen fire zone by one-hour fire barriers. Fire detection is provided in the general area and subfloor areas. Manual hose stations and portable fire extinguishers are provided for fire suppression. Smoke removal is provided by the nonradioactive ventilation system. Breathing apparatus is provided for control room personnel. The above provisions are in accordance with the BTP. Automatic suppression is not provided in the control room or peripheral rooms in this fire area. Fire detection is not provided in the cabinets or consoles. These deviations are acceptable as the control room is continuously occupied, the area fire hazard is low, manual suppression capability is available, and the remote shutdown workstation is located in a separate fire area. These are Deviations 9.5.1-15, and -16.

The staff concludes that the deviations from the guidance specified in the BTP do not adversely affect safety and are, therefore, acceptable.

9.5.1.7.c Cable Spreading Room

There are no cable spreading rooms in the AP600. Therefore, the guidance specified in the BTP is not applicable.

9.5.1.7.d Plant Computer Rooms

There are no computers performing safety-related functions in the MCR complex. Non-safety-related computers outside the MCR are separated from safety-related areas by three-hour fire barriers.

The integrated protection cabinets, engineered safety features actuation cabinets, and protection logic cabinets are located in the instrumentation and control rooms which are separated from other plant areas and from redundant divisions by three-hour fire rated barriers. Automatic fire detection, portable fire extinguishers, and manual hose stations are provided. Floor drains are provided for the removal of firefighting water. Smoke removal using the nuclear island nonradioactive ventilation system or portable fans and ductwork, is provided for these areas. Westinghouse identified no exceptions from the guidance specified in the BTP and is, therefore, acceptable.

9.5.1.7.e Switchgear Rooms

The instrumentation and control, penetration, and reactor trip switchgear rooms are separated from other plant areas and from redundant divisions by three-hour fire rated barriers. Automatic fire detection, portable fire extinguishers, and manual hose stations are provided. Floor drains are provided for the removal of firefighting water. Smoke removal using the nuclear island nonradioactive ventilation system or portable fans and ductwork is provided for these areas. Westinghouse identified no exceptions from the guidance specified in the BTP. The staff finds this acceptable.

9.5.1.7.f Remote Safety-related Panels

Safety-related panels outside of the control room are separated from other plant areas by three-hour fire barriers. Automatic fire detection, portable fire extinguishers, and manual hose stations are provided. Remote shutdown panels located in the remote shutdown workstation can be electrically isolated from the MCR by a transfer switch. The control of combustible materials in these areas is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(k).

Westinghouse identified no deviations from the guidance specified in the BTP. The staff finds this acceptable.

9.5.1.7.g Safety-Related Battery Rooms

Safety-related battery rooms are separated from each other and other plant areas by three-hour fire rated barriers. Switchgear and invertors are not located in the battery rooms as specified in the SRP. Automatic fire detection is provided in the battery rooms. Portable extinguishers and hose stations are readily available outside the battery rooms. Ventilation systems are capable of maintaining the hydrogen concentration in the battery rooms below 2 percent. A loss of the battery room ventilation system alarms in the MCR. Westinghouse identified no exceptions from the guidance specified in the BTP. Therefore, the staff finds it acceptable.

9.5.1.7.h Turbine Building

The turbine building is separated from adjacent structures containing safety-related equipment by fire barriers with a minimum rating of three hours. The fire barriers are designed to maintain structural integrity in the event of a collapse of the turbine building. Openings and penetrations are minimized and are not located in proximity to the turbine lube oil system or generator hydrogen cooling system. Westinghouse identified no exceptions from the guidance specified in the BTP. The staff finds this acceptable.

9.5.1.7.i Diesel Generators
9.5.1.7.j Diesel Fuel Storage

The standby diesel generators are located in a separate structure, remote from safety-related areas, and separated from each other by three-hour fire barriers. The ancillary diesel generators are located in the same fire area, but are separated from other plant areas containing safety-related equipment by three-hour fire barriers. This deviation regarding the lack of three-hour separation between the ancillary diesels, which are not safety-related, does not adversely affect safety and is, therefore, acceptable. This is Deviation 9.5.1-17.

Automatic fire suppression is provided in the diesel generator and fuel storage rooms and is designed to actuate during diesel operation without affecting the diesel. Automatic detection is provided in the diesel generator service modules only; the dry pipe sprinklers provide detection in the diesel generator and fuel storage rooms. This deviation does not adversely affect safety and is, therefore, acceptable. This is Deviation 9.5.1-18.

Portable extinguishers and manual hose stations are readily available outside the fuel storage area. Drainage for firefighting water and a means for manual venting of smoke is provided.

The diesel generator day tanks have a capacity of 5,678 L (1,500 gallons). The tanks for the standby diesel generators are located in separate fire areas enclosed with three-hour fire barriers. The fuel tanks for the ancillary diesels have a capacity of 1,514 L (400 gallons) and are located in the same fire area as the diesels. The fuel storage areas are provided with automatic dry pipe sprinkler protection. The fuel tank enclosure is capable of containing the entire contents of the tank.

In view of the foregoing, the deviations identified by Westinghouse do not adversely affect safety and are, therefore, acceptable.

9.5.1.7.k Safety-related Pumps

There are no safety-related pumps required for safe shutdown following a fire in the design of the AP600. Therefore, the guidance specified in the BTP is not applicable.

9.5.1.7.l New Fuel Storage Area

The new fuel storage pit is provided with automatic fire detection, hose stations, and portable extinguishers; automatic suppression is not provided. Floor drains are provided to prevent the accumulation of water that could result in an inadvertent criticality. The new fuel storage pit is located in the same fire area (1200 AF 02) as the rail car bay/filter storage area. The rail car bay/filter storage area is provided with automatic suppression. Westinghouse identified no exceptions from the guidance specified in the BTP. Therefore, the staff finds the fire protection design for the new fuel storage area acceptable.

9.5.1.7.m Spent Fuel Pool Area

The fuel-handling area is provided with automatic fire detection, hose stations and portable extinguishers; automatic suppression is not provided. The fuel-handling area is located in the same fire area (1200 AF 02) as the rail car bay/filter storage area. The rail car bay/filter storage area is provided with automatic suppression. Westinghouse identified no exceptions from the guidance specified in the BTP. Therefore, the staff finds the fire protection design for the spent fuel pool area acceptable.

9.5.1.7.n Radwaste and Decontamination

The radwaste building is separated from other plant areas containing safety-related equipment by three-hour fire rated barriers. Automatic fire suppression is provided in the mobile systems facility, waste accumulation room, and packaged waste storage room. Fire detection and hose stations are provided throughout the radwaste building.

The cask washdown pit and the waste disposal container area are located in the same fire area (1200 AF 02) as the rail car bay/filter storage area. The rail car bay/filter storage area is provided with automatic suppression. As described above, Westinghouse identified no exceptions from the guidance specified in the BTP. Therefore, the staff finds the fire protection design for the radwaste and decontamination areas acceptable.

9.5.1.7.o Safety-related Water Tanks

The core makeup tanks, IRWST, and PCS tank are not susceptible to damage from an exposure fire. Westinghouse identified no exceptions from the guidance specified in the BTP. Therefore, the staff finds the fire protection design for the safety related water tanks acceptable.

9.5.1.7.p Records Storage Area

Records storage areas are located and protected such that a fire in these areas will not affect safety-related systems or equipment. Westinghouse identified no exceptions to the guidance specified in the BTP. Therefore, the staff finds the fire protection design for the records storage area acceptable.

9.5.1.7. q Cooling Towers

The cooling towers are not used as the ultimate heat sink or for fire protection purposes, therefore, the guidance specified in the BTP is not applicable.

Fire protection for cooling towers is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(l).

9.5.1.7.r Miscellaneous Areas

Miscellaneous areas such as shops, warehouses, auxiliary boiler rooms, fuel oil tanks, and flammable and combustible liquid storage tanks are located and protected such that a fire or the effects of a fire will not affect any safety-related equipment. These areas are outside of the containment, which is separated from other plant areas by a 3-hour fire barrier. Westinghouse identified no deviations from the guidance specified in the BTP. Therefore, the staff finds the fire protection design for these areas acceptable.

9.5.1.8 Special Protection Guidelines

9.5.1.8.a Storage of Oxygen-Acetylene Fuel Gases

The proper storage of welding gas cylinders is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(m).

9.5.1.8.b Storage Areas for Ion Exchange Resins

The proper storage of ion exchange resins is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(n).

9.5.1.8.c Hazardous Chemicals

The proper storage of hazardous chemicals is the responsibility of the COL applicant. This is COL Action Item 9.5.1-1(o).

9.5.1.8.d Materials Containing Radioactivity

Materials that collect and contain radioactivity such as spent resins, charcoal filters, and HEPA filters are stored in closed metal containers. Westinghouse identified no deviations from the guidance specified in the BTP. Therefore, the staff finds the fire protection design associated with the storage of these materials acceptable.

9.5.1.9 Evaluation of DSER Open Items and COL Action Items

DSER Open Item 9.5.1.2-1

Westinghouse did not include the remote shutdown workstation as an area where separate redundant divisions will not be separated by three-hour rated fire barriers. Further information regarding the remote shutdown workstation is required to determine the acceptability of the Westinghouse passive fire protection design. This was identified as DSER Open Item 9.5.1.2-1.

In Section 9.5.1.2.1.1 of the SSAR, "Plant Fire Prevention and Control Features, Plant Arrangement," Westinghouse states that fire barrier separation is not provided within the remote shutdown workstation fire area because the remote shutdown workstation is not required unless a fire requires evacuation of the MCR. A fire in the remote shutdown work station is not postulated to occur simultaneously with a fire in the MCR. Therefore, the staff finds this acceptable; therefore, DSER Open Item 9.5.1.2-1 is closed.

DSER Open Item 9.5.1.2-2

Additional information concerning fire protection features (such as detection, suppression, structural walls, and/or distances between redundant safe-shutdown equipment) is needed to determine the acceptability of the AP600 design concerning fire protection inside primary containment. This was identified as DSER Open Item 9.5.1.2-2.

Westinghouse provided additional information in Section 9.5.1.2.1.1 of the SSAR, "Plant Fire Prevention and Control Features, Plant Arrangement," and in Table 9.5.1-1, "Guidelines 166 - 174," regarding primary and secondary containment. A discussion of primary and secondary containment is in Section 9.5.1.7.a of this report, including the requirement that the COL applicant is responsible for fire protection inside containment during refueling and maintenance. The staff finds this acceptable; therefore, DSER Open Item 9.5.1.2-2 is closed.

DSER Open Item 9.5.1.3-1

Westinghouse should provide more detail regarding detector capability and exceptions to the guidance specified in the BTP. This was identified as DSER Open Item 9.5.1.3-1.

In Table 9.5.1-1, "Guidelines 112 -120," and Section 9.5.1.2.1.2 of the SSAR, Westinghouse described the fire detection and alarm system. The staff finds this acceptable, therefore, DSER Open Item 9.5.1.3-1 is closed.

DSER Open Item 9.5.1.3-2

Guideline 137 (BTP CMEB 9.5-1, Section C.6.b (9)) states that two separate fresh water supplies with a minimum of 1,135.6 kL (300,000 gallons) each should be used for fire service. On the basis of the information provided, it is not clear whether the second fire protection water storage tank also contains a dedicated 1,135.6 kL (300,000 gallons) storage capacity for fire service. Additional information will be required to determine the acceptability of the fire protection water supply. This was identified as DSER Open Item 9.5.1.3-2.

In Table 9.5.1-2 of the SSAR, Westinghouse states that the primary fire water tank has 1,135.6 kL (300,000 gallons) of water dedicated for fire protection and the secondary fire water tank has 1,135.6 kL (300,000 gallons) of water available to fire protection. In Section 9.5.1.2.3, Westinghouse provided additional information on the fire protection water supply (see Section 9.5.1.6.b of this report for more information). The staff finds this acceptable, therefore, DSER Open Item 9.5.1.3-2 is closed.

DSER Open Item 9.5.1.3-3

Additional information regarding the design of the sprinkler system based on results of the fire protection analysis is needed for the staff to make its determination regarding the

Westinghouse design of automatic fire suppression systems. This was identified as DSER Open Item 9.5.1.3-3.

In Section 9.5.1.2.1.4, Westinghouse stated that the automatic sprinkler and water spray systems are provided in accordance with the applicable requirements in NFPA 13 and NFPA 15 and describes the AP600 automatic water suppression systems (see Section 9.5.1.6.c of this report for more information). The staff finds this information acceptable; therefore, DSER Open Item 9.5.1.3-3 is closed.

DSER Open Item 9.5.1.3-4

The staff requires the following additional information:

- availability of the water dedicated to the manual hose stations from the passive containment water storage tank
- the pressure required to produce at least two effective hose streams inside containment utilizing the passive containment water storage tank
- assurance that no possibility exists for channeling water from fire-extinguishing operations in one redundant fire area into another redundant fire area

This was identified as DSER Open Item 9.5.1.3-4.

In Section 9.5.1.2.1.5 of the SSAR, "Manual Fire Suppression Systems," Westinghouse stated that the seismic standpipe system supply is from a portion of the PCS allocated for fire protection. This volume of water is sufficient to supply two hose streams, each with a flow of 285 L/min (75 gpm), for two hours. In addition the PCS ancillary water storage tank has a quantity of water dedicated to fire protection to supply two hose streams, each with a flow of 284 L/min (75 gpm), for two hours. In Section 9.5.1.1.1 of the SSAR, "Safety Design Basis'" Westinghouse stated that floor drains are sized to remove the expected fire fighting water flow without flooding safety related equipment (see Section 9.5.1.6.a of this report for more information). The staff finds this information acceptable; therefore, DSER Open Item 9.5.1.3-4 is closed.

DSER Open Item 9.5.1.4-1

Westinghouse had not provided sufficient information to enable the staff to determine whether the control room emergency lights and the remote shutdown work station emergency lights will be electrically and physically protected from a fire. Additional information was required for the staff to determine the acceptability of AP600 emergency lighting. This was identified as DSER Open Item 9.5.1.4-1.

In Section 9.5.3.2.2 of the SSAR, "Emergency Lighting," Westinghouse stated that the MCR and remote shutdown area are provided with three-hour fire barrier separation between redundant emergency lighting power supplies and cables outside the MCR and remote shutdown area. The emergency lighting system is designed so that, to the extent practical, alternate emergency lighting fixtures are fed from separate divisions of the Class 1E dc and uninterruptable power supply system. The staff finds this acceptable; therefore, DSER Open Item 9.5.1.4-1 is closed.

DSER Open Item 9.5.1.4-2

In Table 9.5.1-1, under BTP CMEB 9.5-1 Guideline 111, Westinghouse did not indicate whether it commits to meeting the emergency communication guidelines of Section C.5.g(4) of BTP CMEB 9.5-1 regarding the use of a portable radio communications system by the fire brigade and other operational personnel required to achieve safe plant shutdown. Additional information was required from Westinghouse for the staff to determine the acceptability of the AP600 portable radio communications systems for the fire brigade and other operations personnel. This was identified as DSER Open Item 9.5.1.4-2.

In Table 9.5.1-1, Westinghouse indicated that the COL applicant will address how it will comply with Guideline 111, regarding a portable radio communication system. Section 9.5.1.5.g of this report provides a discussion of communications including the requirement that the COL applicant is responsible for portable radio communication for the use of the fire brigade. The staff finds this acceptable; therefore, DSER Open Item 9.5.1.4-2 is closed.

DSER Open Item 9.5.1.4-3

In Table 9.5.1-1 of the SSAR, under BTP CMEB Guideline 32, Westinghouse stated that the AP600 is expected to conform to the guidelines or their intent. The COL applicant will need to provide additional information. The guidelines also indicate that the procedures and administrative controls governing the fire protection program during plant operations are developed for specific plants and covered in the COL application. The staff asked Westinghouse to provide additional information concerning reserve air to permit quick and complete replenishment of exhausted air supply bottles as they are returned. They also asked Westinghouse to identify the location of air compressors or other equipment such as cascading air bottles that will be used to replenish the breathing air, as specified in BTP 9.5.1.3.c. This additional information was necessary for the staff to determine the acceptability of the fire brigade emergency breathing air. This was identified as DSER Open Item 9.5.1.4-3.

In Table 9.5.1-1, Westinghouse indicated that satisfying Guideline 32 is the responsibility of the COL applicant. In satisfying Guideline 32, Westinghouse stated that a breathing air compressor and receiver are provided in the compressed and instrument air system to replenish the exhausted air supply bottle used by the fire brigade. Additionally, an equivalent six-hour supply of reserve air (e.g., the 12 additional SCBA bottles) will be maintained in an area located outside of the turbine building. Further information was also provided in Section 9.3.1 of the SSAR. Westinghouse also stated in Note 2 of Table 9.5.1-1 that procedures and administrative controls governing the fire protection programs during plant operation are the responsibility of the COL applicant. Additionally, COL Action Item 9.5.1-1, in part, as described in this report, states that the COL applicant is responsible for the establishment of administrative controls to maintain the performance of the fire protection systems and personnel. The staff finds this acceptable; therefore, DSER Open Item 9.5.1.4-3 is closed.

DSER Open Item 9.5.1.4-4

Westinghouse had not provided sufficient information regarding floor drains sized to remove water flow without flooding safety-related equipment. The staff required additional information regarding the acceptability of the AP600 design for curbs and drains. This was identified as DSER Open Item 9.5.1.4-4.

In Table 9.5.1-1, Westinghouse states that they comply with Guidelines 65-68, and 70, regarding drains. Guideline 69 refers to the gaseous suppression system which does not apply to the AP600. Guideline 71 refers to the drainage of potentially radioactive water and is the responsibility of the COL applicant. Section 9.5.1.5.a of this report provides a discussion of drains including the requirement that the COL applicant is responsible for the collection and sampling of potentially radioactive water (see Section 9.5.1.9 of this report for additional information). The staff finds this acceptable; therefore, DSER Open Item 9.5.1.4-4 is closed.

DSER Open Item 9.5.1.4-5

On the basis of the current AP600 design (i.e., as of the date of issuance of the DSER) concerning smoke control features, it is not clear how smoke will be prevented from migrating to other fire areas. Therefore, the staff requested additional information from Westinghouse to clarify how smoke and hot gases will not adversely affect safe shutdown, including operator action, for all safe shutdown and safety-related areas. Additional information concerning the AP600 smoke-control features was required for the staff to determine the acceptability of the smoke-control features. This was identified as DSER Open Item 9.5.1.4-5.

In Section 9.5.1.2.2 of the SSAR, "System Operation," and Section 9.5.1.3, "Safety Evaluation (Fire Protection Analysis)," Westinghouse stated that the fire protection analysis includes a discussion regarding the control and removal of smoke and hot gases for each fire area. Section 9.5.1.5.f of this report provides a discussion of ventilation system. Based on that discussion, the staff finds this acceptable; therefore, DSER Open Item 9.5.1.4-5 is closed.

DSER Open Item 9.5.1.4-6

Westinghouse had not described the protection provided regarding the inadvertent operation of fire protection systems. Therefore, additional information from Westinghouse was required for the staff to determine the acceptability of the AP600 design concerning interaction with other systems. This was identified as DSER Open Item 9.5.1.4-6.

In Section 9A.2.6 of the SSAR, "Fire Protection System Integrity," Westinghouse stated that for all areas containing safety related equipment, the potential for a credible inadvertent actuation of the automatic suppression systems has been determined and the consequences evaluated. The staff finds that there are no automatic suppression systems in safety related areas; therefore, DSER Open Item 9.5.1.4-6 is closed.

DSER Open Item 9.5.1.4-7

Additional information was required from Westinghouse for the staff to determine the acceptability of the preoperational acceptance test for all active components of the entire fire protection system(s). This was identified as DSER Open Item 9.5.1.4-7.

In Section 9.5.1.4 of the SSAR, Westinghouse stated that preoperational testing will be in accordance with the initial test program, as described in Chapter 14 , which includes the fire protection system. The fire pumps will be initially tested by the manufacturer in accordance with NFPA 20 to verify pressure integrity and performance. The staff finds this acceptable; therefore, DSER Open Item 9.5.1.4-7 is closed.

COL Action Item 9.5.1.2-1

The COL applicant's maintenance program is to be developed to ensure that fire-rated assemblies (such as fire doors, fire dampers, and penetration seals) will be maintained in accordance with their respective NFPA codes or manufacturer's instructions. The staff will review the COL applicant's maintenance program on a plant-specific basis. This was identified as COL Action Item 9.5.1.2-1.

In Section 9.5.1.2.1.1, Westinghouse described the plant fire protection and control features. COL Action Item 9.5.1-1, as identified in this report, requires the establishment of a fire protection program at the facility, which includes the protection of structures, systems, and components important to safety. COL Action Item 9.5.1-2, as identified in this report, requires an update to the fire hazard analysis, which will demonstrate that the plant will maintain its ability to perform safe shutdown and other functions. COL Action Item 9.5.1-1, in part, as identified in this report, addresses maintenance of fire doors, access to keys, and exit routes. These COL action items supersede COL Action Item 9.5.1.2-1; therefore, COL Action Item 9.5.1.2-1 is dropped.

COL Action Item 9.5.1.3-1

In Table 9.5.1-1 of the SSAR, under BTP CMEB 9.5-1 Guidelines 121-144, Westinghouse committed to follow the BTP CMEB 9.5-1 guidelines but noted that, because of conflicting design considerations, there may be a need to take exception to specific guidance. The COL applicant will identify and address these deviations in the fire hazards analysis.

In Table 9.5.1-1 of the SSAR, under BTP CMEB 9.5-1 Guidelines 164-165, Westinghouse indicated that extinguishers will be provided in areas that contain or present a fire exposure hazard to safety-related equipment, in accordance with the guidelines of NFPA 10. The staff expects that these deviations to BTP CMEB 9.5-1 and/or NFPA 10 will be addressed in the Fire hazards analysis submitted by the COL applicant. The staff will review the deviations to BTP CMEB 9.5-1, Section C.6.f, and/or NFPA 10 on a plant-specific basis. This was identified as COL Action Item 9.5.1.3-1.

In Section 9.5.1.8 of the SSAR, Westinghouse stated that the COL applicant will address BTP CMEB 9.5-1 issues identified in Table 9.5.1-1 by the acronym "WA," and that the COL applicant will address updating the list of NFPA exceptions after design certification, if necessary. In Table 9.5.1-1, Westinghouse indicated in Note 3 that, for Guidelines 133, 150, 153, and 164, it fully intended to comply with the NFPA standards referenced in Section 9.5.5, as they apply to the AP600; however, because of conflicting design considerations, there may be a need to take exception to specific guidance. Known exceptions to NFPA requirements are identified in SSAR Table 9.5.1-3. The COL applicant will address updating the list of NFPA exceptions after design certification, if necessary. The staff finds this acceptable. COL Action Item 9.5.1-2, as described in this report, includes the requirement to revise the fire hazard analysis to reflect the actual plant configuration. COL Action Item 9.5.1-3, as described in this report, includes the requirement to incorporate any deviations from NFPA codes and standards, with appropriate justification, into the FSAR. These COL action items supersede COL Action Item 9.5.1.3-1; therefore, COL Action Item 9.5.1.3-1 is dropped.

<u>COL Action Item 9.5.1.5-1</u>

Westinghouse indicated that the procedures and administrative controls governing the fire protection program during plant operation are developed for specific plants and covered in the COL application. The staff will perform a detailed review of the administrative controls during the plant-specific licensing process of a COL application referencing the AP600 system design. Items of interest under the administrative controls review will include the following:

- control of combustible materials such as combustible/flammable liquids and gases, fire-retardant-treated wood, plastic materials, and dry ion-exchange resins
- transient combustible materials and general housekeeping, including health physics materials
- open-flame and hot-work permits, and cutting and welding operations
- quality assurance with respect to fire protection system(s) components, installation, maintenance, and operation
- qualification of fire protection engineering personnel, fire brigade members, and fire protection system(s) maintenance and testing personnel
- instruction, training, and drills provided to fire brigade members

These items were identified as COL Action Item 9.5.1.5-1.

In Section 9.5.1.2.1.1, "Plant Fire Protection and Control Features, Control of Combustible Materials," Westinghouse described its controls and states that the control of construction and combustible materials are in accordance with BTP CMEB 9.5-1 and NFPA 803. In Table 9.5.1-1, Westinghouse addressed Guidelines 77-83, regarding the control of combustibles.

In Section 9.5.1.7 of the SSAR, "Quality Assurance," Westinghouse commits that the quality assurance controls were applied to the activities involved in the design, and will be applied to the activities involved in the procurement, installation, testing, and maintenance of fire protection systems for safety-related areas, in accordance with the programs outlined in

Chapter 17. Additionally, in Table 9.5.1-1, Westinghouse designates the COL applicant to address Guidelines 30-34, regarding the fire brigade.

In Section 9.5.1.6, Westinghouse stated that the qualification requirement for individuals responsible for training fire fighting personnel is the responsibility of the COL applicant. In Section 9.5.1.8, Westinghouse stated that the COL applicant will address qualification requirements for individuals responsible for the development of the fire protection program, training of firefighting personnel, the administrative procedures and controls governing the fire protection program during plant operation, and fire protection system maintenance.

The staff finds the above statements in the SSAR acceptable. A discussion of the control of combustibles is in Section 9.5.1.5.d and 9.5.1.7.f of this report, including the requirement that the COL applicant is responsible for the control of combustible material in the remote shutdown panels. COL Action Item 9.5.1-1, in part, as described in this report, requires the COL applicant to establish a site fire brigade trained and equipped to fight and to establish a quality assurance program to ensure that the guidelines for the design, procurement, installation and testing, and the administrative control for fire protection systems are satisfied. The staff also finds the COL action items supersede COL Action Item 9.5.1.5-1; therefore, COL Action Item 9.5.1.5-1 is dropped.

Applicable National Fire Protection Association Codes, Standards and Recommended Practices

NFPA 10 - Portable Extinguishers
NFPA 13 - Sprinkler Systems
NFPA 14 - Standpipe and Hose Systems
NFPA 20 - Centrifugal Fire Pumps
NFPA 22 - Water Tanks
NFPA 24 - Private Fire Service Mains
NFPA 30 - Flammable Liquids Code
NFPA 50A Gaseous Hydrogen Systems
NFPA 80 - Fire Doors and Windows
NFPA 90A - Air Conditioning Systems
NFPA 92A - Recommended Practice for Smoke Control Systems

SUMMARY OF APPROVED DEVIATIONS AND COL ACTION ITEMS

I. Approved Deviations

9.5.1 - 1 - Single failure of primary and backup fire suppression inside containment.
9.5.1 - 2 - Gypsum stair towers.
9.5.1 - 3 - Cable insulation in concealed spaces of the control room, technical support center, remote shutdown workstation.
9.5.1 - 4 - Definition of safe shutdown.
9.5.1 - 5 - Achievement of cold shutdown in 72 hours.
9.5.1 - 6 - Breathing air storage tanks located in the auxiliary building.
9.5.1 - 7 - Self contained emergency lighting in the control room and remote shutdown workstation.
9.5.1 - 8 - PCS tanks compliance with NFPA 22.
9.5.1 - 9 - Manual connection between seismic standpipe and yard loop.

9.5.1 - 10 - PCS recirculation pumps compliance with NFPA 20.
9.5.1 - 11 - Fire department connections to the sprinkler systems.
9.5.1 - 12 - Manual operation of standpipe inside containment.
9.5.1 - 13 - Containment isolation valves not listed for fire protection service.
9.5.1 - 14 - Automatic suppression of peripheral rooms in control room complex.
9.5.1 - 15 - Fire detection in MCR cabinets and consoles.
9.5.1 - 16 - Fire separation of ancillary diesels.
9.5.1 - 17 - Fire detection in diesel generator rooms.
9.5.1 - 18 - Automatic detection not provided in the D/G and fuel storage rooms.

II. COL Action Items

9.5.1 - 1(a) - Fire protection program.
9.5.1 - 1(b) - Implementation of fire protection program.
9.5.1 - 1(c) - Administrative controls.
9.5.1 - 1(d) - Fire brigade.
9.5.1 - 1(e) - Quality assurance program.
9.5.1 - 1(f) - Inspection and maintenance of fire doors, keys for the fire brigade, and marking of exit routes.
9.5.1 - 1(g) - Sampling of water drainage for contamination following a fire.
9.5.1 - 1(h) - Control of combustibles.
9.5.1 - 1(i) - Portable radio communications for the fire brigade.
9.5.1 - 1(j) - Fire protection inside containment during refueling and maintenance.
9.5.1 - 1(k) - Control of combustibles in areas containing safety-related equipment.
9.5.1 - 1(l) - Cooling tower fire protection.
9.5.1 - 1(m) - Storage of welding gas cylinders.
9.5.1 - 1(n) - Storage of ion exchange resins.
9.5.1 - 1(o) - Storage of hazardous chemicals.
9.5.1 - 2 - Fire hazard analysis.
9.5.1 - 3 - Deviations from NFPA codes and standards.

9.5.2 Communication System

The staff reviewed the design of the AP600 communication system to determine its conformance to the acceptance criteria in Section 9.5.2 of the SRP, and the guidance provided in the EPRI ALWR URD for passive plants.

The acceptance criteria in the SRP are predicated, in part, on the similarity of the communication system design with that of previously reviewed plants with satisfactory operating experience. According to the SRP criteria for assessing system design capability, the communication system is acceptable if the integrated design of the system will provide effective communication between plant personnel in all vital areas during normal plant operation and the full spectrum of accident or incident conditions (including fire), under the maximum potential noise levels. The scope of the staff's evaluation includes a review of the communication system description and the supporting analyses that demonstrate the effectiveness of the system when maximum noise levels are being generated during accident conditions to verify that the communication system will function effectively.

Section 4.6.1 of Chapter 10 of the EPRI ALWR URD states that the scope of voice communication covers both in-plant operations and maintenance, and communications with outside organizations (including the plant owner load dispatcher, as well as other agencies for which communications are needed to support emergency operations). Section 4.6.2 of Chapter 10 states that the designer of the communication system should include an analysis of the specific communication needs and specific design requirements in the design basis documentation of the man-machine interface system (M-MIS). This section also states that the primary, dedicated means of communication between operators and maintenance technicians during normal or emergency operation will be portable, wireless communication equipment supported by appropriate base stations, antennas, amplifiers, and/or repeaters. A plant-wide paging system and in-plant telephone system should be included. Offsite communications should be accomplished primarily with dedicated phone links.

Westinghouse states that the communications system provides intraplant communication and plant-to-offsite communications during normal, maintenance, transient, fire, and accident conditions, including a LOOP event. The communications system consists of the following subsystems:

- wireless telephone system
- telephone/page system
- private automatic branch exchange (PABX) system
- sound-powered phone system
- emergency response facility communications system
- security communications system

The communications system allows each guard, watchman, or armed response individual on duty to maintain continuous communication with an individual at each manned alarm station (access to vital areas), and with offsite agencies as required by 10 CFR 73.55(e) and 73.55(f) requirements. This is accomplished by both the PABX system and the wireless telephone system. Each system provides these communication functions independently. Communications equipment used with respiratory protection devices will be designed and selected in accordance with EPRI guidance NP-6659.

The wireless telephone system consists of wireless belt-clip portable handsets, hands-free type portable headsets, a comprehensive antenna system, and wireless telephone switches. The wireless telephone system is the primary means of communication for plant operations and maintenance personnel. The telephone/page, PABX, and sound-powered phone systems also provide general plant communications, and serve as an independent backup to the wireless telephone system. The staff notes that extensive use of wireless communication will increase emphasis on consideration of electromagnetic interference in the design of the plant's digital instrumentation and control systems, as discussed in Section 7.2 of this report.

The telephone/page system consists of handsets, amplifiers, loudspeakers, siren tone generators, a centralized test and distribution cabinet, and associated equipment. The system also consists of one paging line and five party lines. A five-tone siren generator annunciates alarms using the telephone page system amplifiers and speakers. Alarm initiation and tone selection capabilities are provided in the MCR.

The PABX system provides communications among the system stations, with the capability for transferring calls and providing conference calls among a maximum of five stations. The PABX system also interfaces with the wireless telephone system, a hotline to the load dispatcher, local area telephone system, page circuit, and PABX locations exterior to the plant. The load dispatcher hot line circuits are dedicated channels that provide direct communication between the MCR and the COL applicant headquarters, or other facilities as required. Power to the PABX is provided from the non-Class 1E dc and uninterruptable power supply system. Backup power is provided to the PABX system for 90 minutes after a loss of ac power.

Two unitized sound-powered phone systems are provided. One is a loop sound-powered phone system used for refueling, and the other is a multi-loop system throughout the plant that is used specifically for startup and maintenance testing.

The MCR and the remote shutdown workstation room are designed and instrumented to bring the plant to a safe-shutdown condition without relying on communications equipment.

Areas of the plant will be subjected to noise surveys during plant startup and initial operation to determine actual noise levels. On the basis of the results of these surveys, noise reduction provisions (such as use of acoustical booths) will be provided to ensure effective communications, as required.

Westinghouse states that the emergency response facility communications system and the security system communication network are site specific and will be described in the COL application. These COL Action Items are identified as 9.5.2-1 and 9.5.2-2, respectively.

On the basis of its review, the staff concludes that the AP600 communication systems will provide effective communication between plant personnel in all vital areas during normal plant operation and the full spectrum of accident or incident conditions, under the maximum potential noise levels. These are consistent with the criteria of Section 9.5.2 of the SRP and are therefore acceptable.

9.5.3 Plant Lighting System

The plant lighting system includes normal, emergency, and security lighting. The normal and emergency lighting in the MCR and remote shutdown area is non-Class 1E. The normal lighting provides normal illumination during plant operating, maintenance, and test conditions. The emergency lighting provides illumination in areas where emergency operations are performed upon loss of normal lighting. The security lighting system is site-specific and will be described by the COL applicant. This was identified as COL Action Item 9.5.3-1.

Westinghouse clarified that the security lighting system identified in COL Action Item 9.5.3-1 is described and evaluated in SSAR Section 13.6. Therefore, COL Action Item 9.5.3-1 is closed.

9.5.3.1 Normal Lighting System

Power to the normal lighting system is supplied from the non-Class 1E power distribution system, and is backed up by the onsite standby diesel generators. The lighting load is distributed between the two diesel generator buses. The motor control centers powering the normal lighting system are energized from the 480V ac load centers. Lighting distribution panel branch circuit breakers are controlled by a lighting control system. Approximately 75 percent of the normal lighting is tripped off automatically upon loss of normal ac power (except in the MCR and in the remote shutdown area) to limit the load on the onsite diesel generators. The lighting control system allows an operator to energize or de-energize lighting in selected areas, depending on the actual need and available power from the onsite standby diesel generator. The circuits to the individual ac fluorescent lighting fixtures are staggered as much as practical. Separate buses feed the staggered circuits to ensure that some lighting is retained in the event of a bus or circuit failure. The lighting fixtures located in the vicinity of safety-related equipment are supported so that they do not adversely impact this equipment when subjected to the seismic loading of an SSE.

The non-Class 1E ac power distribution system supplies power to the normal lighting system at the following voltage levels:

- The 480 V motor control centers feed the 480/277 V, 3-phase, 4-wire, grounded neutral system lighting panels. This source is for the lighting fixtures rated at 480/277 V and for the welding receptacles.

- The 480 V motor control centers feed the 208/120 V, 3-phase, 4-wire, grounded neutral system distribution panels through dry-type 480-208/120 V transformers. This source is for the convenience lighting and utility receptacles.

- The 480 V motor control centers feed the 208/120 V, 3-phase, 4-wire, grounded neutral regulated power through the Class 1E 480-208/120 V voltage regulating transformers (Division B and C). This source is for the normal and emergency lighting in the MCR and remote shutdown area and is isolated through two series of fuses.

9.5.3.2 Emergency Lighting

The Class 1E 125 V dc and uninterruptable power supply (UPS) system supply power to the emergency lighting in the MCR and the remote shutdown area through the Class 1E 208/120 V ac inverters and are isolated through two series fuses. Three-hour barrier separation is provided between redundant emergency power supplies and cables outside the MCR and remote shutdown area. A single fault cannot interrupt all of the lighting on the MCR and remote shutdown working station simultaneously. The Class 1E dc and UPS system are capable of powering the emergency lighting for 72 hours during a station blackout (i.e., with no ac electrical power source). The emergency lighting system is designed to provide illumination levels within 10 to 40 foot-candles in areas of the plant where emergency operations are performed. These areas include the MCR and remote shutdown area. The lighting in these areas consists of 120 V ac fluorescent lighting fixtures, which are continuously energized. The fixtures are powered from the Class 1E, 125 V dc switchboards through the Class 1E, 208/120 V ac inverters. The control room lighting complies with human factors requirements by using semi-indirect, low-glare lighting fixtures and programmable dimming features. The MCR emergency lighting is integrated with the normal lighting, which consists of identical lighting fixtures and dimming features. Both the normal and emergency lighting circuits in the MCR will be configured so that different lighting circuits are powered from different buses. This will ensure that the lighting is retained in the event of a bus or circuit failure. Normal and emergency controllers, dimmers, and associated cables used in the MCR and remote -shutdown area are qualified as Class 1E.

During the post-72-hour period following a loss of all ac power sources, the normal lighting circuits in the MCR and in the remote shutdown area can be powered from two ancillary ac generators.

The AP600 lighting system design for the MCR and remote shutdown areas will be a fabricated total ceiling modular system that meets the seismic Category II requirements. The ceiling grid network, raceways, and fixtures will use seismic Category 1 supports and will meet seismic Category 1 requirements. The lighting fixtures located in the vicinity of safety-related equipment will be supported so that they do not adversely impact this equipment when subjected to the seismic loading of an SSE. The lighting fixtures will be designed and located so that plant personnel can maintain and replace lights effectively and safely.

Westinghouse stated that power to the normal and emergency lighting in the MCR and remote shutdown area is supplied from the Class 1E ac power distribution system. Specifically, 480 V motor control centers will feed 208/120 V, 3-phase, 4-wire, grounded neutral regulated power through the Class 1E, 480 to 208/120 V regulating transformers (Divisions B and C). This appeared to conflict with other information on normal plant lighting. Therefore, the staff needed clarification in this area. This was identified as Open Item 9.5.3.2-1.

In response, Westinghouse revised the SSAR (Revision 11) to state that the normal and emergency lighting in the MCR and in the remote shutdown area is non-Class 1E. A Class 1E UPS feeds power to the emergency lighting in these areas through two series fuses that are coordinated for isolation. A panel lighting system provides lighting for the safety panels in the MCR and is fed from the Divisions B and C Class 1E inverters through Class 1E distribution panels. Panel lighting is designed to provide lighting in the MCR at the safety panels. It consists of lighting fixtures located on or nearby safety panels in the MCR. The panel lights are continuously energized. The Division B and C Class 1E inverters power the fixtures through Class 1E distribution panels. The circuits are treated as Class 1E. The panel lighting circuits up to the lighting fixtures are classified as associated and routed in seismic Category I raceways.

The staff evaluated Westinghouse's response and determined that the panel lighting circuits to the lighting fixtures are powered from the Division B and C Class 1E inverters through Class 1E distribution panels and are routed in seismic Category I raceways. Therefore, the panel lighting design is acceptable and Open Item 9.5.3.2-1 is closed.

9.5.3.3 Non-Class 1E Battery Pack Units

In areas outside the MCR and the remote shutdown area, emergency lighting is provided by 8-hour, self-contained, battery-pack, sealed-beam lighting units. These units are powered from the non-Class 1E diesel-backed buses and provide illumination for safe ingress and egress of personnel following a loss of normal lighting for areas that are involved in power recovery. In addition, these units are provided in areas where normal actions are required for operation of equipment needed during a fire. These units are normally powered from the non-Class 1E 480/277 V ac motor control centers. In the event of loss of ac power, these units automatically switch to their internal dc source. The illumination level of at least 2 foot-candles is maintained in these areas. The sealed-beam lamps will be positioned so that adequate illumination is provided and is not obstructed by plant equipment and components.

The following areas are equipped with this emergency lighting:

- standby diesel generator rooms
- switchgear rooms (annex and auxiliary buildings)
- access route between main control room and remote shutdown area
- connecting corridors and stairwells
- access and egress routes, as determined by the fire analysis
- fire pump rooms

9.5.3.4 Conclusions

On the basis of its review, the staff concludes that the lighting systems for the AP600 are capable of providing adequate lighting during all plant operating conditions and during emergency situations has the capability to provide adequate lighting during conditions, including fire, transients and accident conditions, and the effect of the loss-of-offsite power on the emergency lighting system, and are therefore acceptable. The requirements are in accordance with Section 9.5.3 of the SRP and with lighting levels recommended in Exhibit 6.1-22 of NUREG-0700, "Guidelines for Control Room Design Review," which incorporates the requirements of the Illuminating Engineering Society (IES) Lighting Handbooks.

However, the staff identified that the AP600 design may require additional lighting capabilities as the result of resolution of the RTNSS issue. This was identified as Open Item 9.5.3.4-1.

The RTNSS identification process for the resolution of RTNSS discussed in Section 8.6.2.4 of this report did not reveal any lighting system for inclusion. On the basis of this result, Open Item 9.5.3.4-1 is closed.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

Section 9.5.4.1 of this report addresses compliance of all the diesel generator auxiliary systems with the requirements of GDC 2, 4, and 5. Section 9.5.4.2 of this report addresses issues specific to the standby diesel and auxiliary boiler fuel oil system.

9.5.4.1 Diesel Generator Auxiliary Support Systems (General)

There are two onsite standby DGs in the AP600 design. Each DG engine has the following auxiliary support systems, which are addressed in detail in the SSAR sections indicated below:

- standby diesel and auxiliary boiler fuel oil system (Section 9.5.4.2)
- standby diesel engine cooling system (Section 8.3.1.1.2.1)
- standby diesel engine starting system (Section 8.3.1.1.2.1)
- standby diesel lubricating oil system (Section 8.3.1.1.2.1)
- standby diesel combustion air intake and exhaust system (Section 8.3.1.1.2.1)

The adequacy of these systems is dependent on compliance with the requirements of GDC 2 for protection against natural phenomena, GDC 4 for protection from environmental and dynamic effects of equipment failure, and GDC 5 for sharing SSCs between units, as well as the recommendations of NUREG/CR-0660, "Enhancement of On-Site Emergency Diesel

Generator Reliability." Compliance with the requirements of other GDC is reviewed on a system-specific basis in the following sections of this report.

The diesel engine vendor has not been selected. Therefore, Westinghouse has not fully defined the DG auxiliary support systems. Many components (such as tank size, pump capacity, and system flow rates) are site and DG dependent. However, the requirements and design characteristics specified, and committed to, for the DG auxiliary support systems in the AP600 SSAR provide reasonable assurance that applicable NRC guidance will be satisfied.

In the AP600 DSER, the staff determined that certain RAI responses needed to be modified; the use of equipment procurement specifications needed to be clarified; and the recommendations of NUREG/CR-0660 and certain RAI responses need to be incorporated into the SSAR. This was identified as DSER Open Item 9.5.4.1-1.

After the AP600 DSER was issued, Westinghouse provided additional information about the DG auxiliary support systems in the form of SSAR revisions, RAI responses, and telephone conferences that allowed the staff to continue its review, as discussed below. As a result of the additional information, DSER Open Item 9.5.4.1-1, concerning the need to modify certain RAI responses to clarify the use of equipment procurement specifications, and to incorporate the recommendations of NUREG/CR-0660 and certain RAI responses into the SSAR is closed.

The AP600 design can be used at either single-unit or multiple-unit sites. Nonetheless, in Section 3.1.1 of the SSAR, Westinghouse stated that the AP600 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. Should a multiple-unit site be proposed, the COL applicant must apply for an evaluation of the units' compliance with the requirements of GDC 5, with respect to the capability of shared SSCs important to safety to perform their required safety functions.

The AP600 DG auxiliary support systems have no safety-related functions; therefore, the staff has determined that compliance with the recommendations of RGs 1.115 and 1.117, and the requirements of GDC 2 and 4 is not required for the DG auxiliary support systems.

Sections 9.5.4 through 9.5.8 of the SRP specify that the DG auxiliary support systems should meet the guidance of the Diesel Engine Manufacturers Association (DEMA) standard, "Standard Practices for Low and Medium Speed Stationary Diesel and Gas Engines." In Section 9.5.4.1.3 of the SSAR, Westinghouse states that the portions of the standby diesel and auxiliary boiler fuel oil system (DOS) that support the standby diesel generators follow the guidance for distillate fuel oil supply contained in Chapter 13 of the DEMA standard. The staff has determined that it is acceptable for only the DOS to meet the guidance of the DEMA standard because the DOS is the only DG auxiliary support system that will not be procured with the packaged DG set. The packaged DG set, which includes all the other DG auxiliary support systems, will be procured in accordance with the diesel generators manufacturer's standards.

In response to RAI Q410.173, Westinghouse stated that the following DG trip conditions are in effect during both testing and operation:

- engine overspeed
- high engine crankcase pressure
- low-low lube oil pressure
- low engine jacket water pressure
- high jacket water temperature
- generator differential current
- generator "B" fault lockout
- loss of generator field

Generator reverse power actuation, generator overcurrent, and generator negative sequence overcurrent trip conditions are only in effect during testing, and are bypassed during operation.

The staff reviewed the designs of the DG auxiliary support systems with respect to the recommendations of NUREG/CR-0660, which described specific recommendations for increasing the reliability of DGs at nuclear plants. Table 9.5-1 of this report summarizes compliance of the AP600 DG auxiliary support systems with the recommendations of NUREG/CR-0660.

In Section 8.3.1.1.2.1 of the SSAR, Westinghouse states that the onsite DGs will be procured in accordance with an equipment specification that will include requirements based upon manufacturer's standards and applicable recommendations from documents such as NUREG/CR-0660.

In NUREG/CR-0660, the NRC recommends that provisions should be made to reduce the effect of dust and dirt on DG operation and reliability. Specifically, all contactors and relays should be enclosed in dust-tight steel cabinets with gasketed openings, and should have dust-tight enclosed electrical contacts of the bifurcated type. Also, ventilation air for the DG room should be taken about 20 feet above the adjacent ground surface. In its response to RAI Q410.183 dated August 3, 1994, Westinghouse stated that the AP600 equipment procurement specifications for electrical equipment will require dust-tight enclosures. Section 9.4.10.2.1.1 of the SSAR states that the air intake for the DG building heating and ventilation system will be as high in the DG building wall as possible to limit the intake of dust blown about by wind and/or passing vehicles. The ventilation system for the DG building service module, which includes most of the electrical switchgear, is provided with 80-percent efficient inlet air filters to clean the cooling air. In addition, the supply air handling unit for the DG building heating and ventilation system is also provided with 80-percent efficient inlet air filters to clean the incoming air. Also, Section 8.3.1.1.2 of the SSAR states that the DGs will be procured to be consistent with the DG building heating and ventilation system. The staff has determined that the above mentioned provisions adequately reduce the effect of dust and dirt on DG operation and reliability. Therefore, the design of the DG auxiliary support systems complies with this recommendation of NUREG/CR-0660.

In NUREG/CR-0660, the NRC recommends that operations and maintenance personnel and their immediate supervisors receive training that is provided by the vendor, or that is equivalent to vendor training, because a lack of proper training has contributed to the degradation of DG reliability. In its response to RAI 410.183 dated August 3, 1994, Westinghouse stated that personnel training of the DG operating staff is the responsibility of the COL applicant. Section 8.3.1.1.2.1 of the SSAR states that training is addressed as part of overall plant training described in Section 13.2.1 of the SSAR. Section 13.2.1 states that the COL applicant will develop and implement training programs for plant personnel. The staff concludes that it is acceptable for the COL applicant to develop and implement the training program. This is part of COL Action Item 13.2-1. This was previously identified as DSER COL Action Item 9.5..4.1.1.

In NUREG/CR-0660, the NRC recommends minimizing no-load and light-load operation, following the engine manufacturers' recommendations for testing frequencies, test load size, and testing duration, as well as for performing preventive maintenance. Section 8.3.1.1.2.1 of the SSAR states that testing, test loading, and preventive maintenance are addressed as part of overall plant testing and maintenance in Chapter 13 of the SSAR. Section 8.3.3 of the SSAR states that the COL applicant will establish plant procedures as required for DG operation, inspection, and maintenance in accordance with manufacturers' recommendations. The staff finds this acceptable. This is part of COL Action Item 8.3.1.2-1. This was previously identified as DSER COL Action Item 9.5.4.1.2.

In NUREG/CR-0660, the NRC recommends improving the identification of root causes and the proper choice of corrective actions. Specifically, the obvious root cause should be questioned, closely spaced component failure should be unacceptable, and the use of licensee event report (LER) records as a reliability index should be exercised with caution. In its response to RAI Q410.183 dated August 3, 1994, Westinghouse stated that instrumentation is provided to support diagnostics during operation. Section 8.3.1.1.2.1 of the SSAR states that instrumentation to support diagnostics during operation are shown on Figure 8.3.1-5 of the SSAR. Figure 8.3.1-5 provides indication and alarm points for:

- lube oil pressure, temperature and sump level
- cooling water pressure and temperature
- starting air pressure

The staff has determined that the instrumentation shown on Figures 8.3.1-5, and 9.5.4-1 provides the necessary information to facilitate the adequate identification of root causes. Therefore, the design of the DG auxiliary support systems complies with this recommendation of NUREG/CR-0660.

In NUREG/CR-0660, the NRC recommends the use of high-temperature-rated insulation in the generators for overload protection. In its response to RAI Q410.183 dated August 3, 1994, Westinghouse stated that the equipment procurement specifications for the DG electrical equipment will address the need for high-temperature insulation. Section 8.3.1.1.2 of the SSAR states that the requirements of the high-temperature insulation will be determined by the manufacturer's recommendations. The staff finds this acceptable to meet the intent of this recommendation of NUREG/CR-0660.

In NUREG/CR-0660, the NRC recommends that the floors be painted with concrete or masonry-type paint in all rooms of the DG buildings that house any devices with electrical contacts. In Section 8.3.1.1.2 of the SSAR, Westinghouse states that the DG building floor coatings are described in Sections 6.1.2.1.4 and 6.1.3.2 of the SSAR. In Section 6.1.2.1.4, Westinghouse states that concrete floors subject to high traffic or contaminated liquid spills are coated with self-leveling epoxy. Otherwise, concrete floors are coated with high-build epoxy. A high-build epoxy finish is also applied a minimum of 0.3 m (1 ft) up the wall where liquid spills might splash. Remaining concrete walls and exposed concrete ceilings are coated with a thin film epoxy sealer to reduce dusting. In Section 6.1.3.2, Westinghouse states that COL applicants will address preparation of a program to control testing, application, and monitoring of non-safety-related coatings. The staff finds that applying the above mentioned coatings to all exposed concrete surfaces of the DG building will adequately minimize the production of air borne particulates and, consequently, minimize the degradation of electrical contacts. Therefore, the design of the DG auxiliary support systems complies with this recommendation of NUREG/CR-0660.

In NUREG/CR-0660, the NRC recommends that instruments, and control, monitoring, or indicating elements be supported in or on a free-standing, directly floor-mounted panel, except for the necessary sensors in piping. In its response to RAI Q410.183 dated August 3, 1994, Westinghouse stated that the effects of engine vibration on engine-mounted monitoring and control instrumentation will be addressed in the equipment procurement specifications. In Section 8.3.1.1.2 of the SSAR, Westinghouse states that response to the effects of engine vibration will be dependent upon the manufacturer's recommendations. Therefore, the staff finds this acceptable to meet the intent of this recommendation of NUREG/CR-0660.

The DG auxiliary systems have no safety-related function. While the DG auxiliary systems are subject to availability controls (see Section 8.6.2.4 of this report), Their failure will not prevent safety-related systems from performing their function. Therefore, the staff has determined that compliance with the requirements of GDC 2 and 4 is not required for the DG auxiliary systems. On the basis of the above review, the staff concludes that the DG auxiliary systems comply with GDC 5 and the recommendations of NUREG/CR-0660 and is therefore acceptable.

9.5.4.2 Standby Diesel and Auxiliary Boiler Fuel Oil System

The staff reviewed the DOS in accordance with the guidance of Section 9.5.4 of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding whether the DOS satisfies the following requirements:

- GDC 2, for protection against natural phenomena
- GDC 4, for protection from environmental and dynamic effects of equipment failure
- GDC 5, for sharing SSCs between units
- GDC 17, for availability of electric power systems

Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 are discussed in Section 9.5.4.1 of this report.

In the AP600 DSER, the staff determined that the use of plant installation specification and COL applicant responsibilities needed to be better defined and certain RAI responses needed to be modified and incorporated into the DSER. This was identified as DSER Open Item 9.5.4.2-1.

After the AP600 DSER was issued, Westinghouse provided additional information about the DOS in the form of SSAR revisions and telephone conferences that allowed the staff to continue its review, as discussed below. As a result of the additional information, DSER Open Item 9.5.4.2-1, concerning the need to modify certain RAI responses, incorporate certain RAI responses into the SSAR, further define the COL responsibilities, and clarify the use of plant installation specifications, is closed.

As stated in Table 3.2-3 of the SSAR, the DOS components are non-nuclear safety class and non-seismic category. Therefore, quality assurance requirements of Appendix B of 10 CFR Part 50 do not apply. The system description, components, and flow diagrams are provided in Section 9.5.4, Table 9.5.4-1, and Figure 9.5.4-1 of the SSAR, respectively.

The DOS consists of two fuel oil storage tanks, a DG fuel oil transfer system (DGFOTS), an auxiliary boiler fuel oil supply system (ABFOSS), and an ancillary DG fuel oil supply system (ADGFOSS).

The ADGFOSS is located in the same room in the annex building as the ancillary DGs. It consists of a single ancillary DG fuel oil storage tank, which is sized to provide sufficient capacity for four days of operation for both ancillary DGs. The ancillary DG fuel oil storage tank is seismic Category II and is analyzed to show that it will withstand an SSE (see Section 3.7.3 of this report). The ancillary DG fuel oil storage tank is vented to the atmosphere with a 51 mm (2 in) line that has a ball-float check valve, and flame arrestor at the end. The ancillary DG fuel oil storage tank will be located at an elevation to provide the necessary head for both ancillary DGs. The ancillary DG fuel oil storage tank is equipped with a dip stick for locally monitoring the fuel oil level in the tank.

The ABFOSS consists of a single supply line to the auxiliary boiler, two 100-percent capacity pumps, each supplied by a fuel oil storage tank, and a recirculation fuel oil return line from the boiler to the storage tanks.

The DGFOTS consists of two divisions (one per DG) each of which is independent and physically separated from the other division serving the redundant DG. Thus, a single failure in one of the two divisions will affect only the associated DG. Each division consists of a fuel oil storage tank; a suction strainer; a fuel oil transfer pump; an electric fuel oil heater; a moisture separator; a duplex fuel oil filter; a day tank; an engine-mounted, engine-driven fuel oil supply pump; inline fuel oil filters; a fuel oil recirculation circuit with a fuel oil cooler; and interconnecting piping, valves, and instrumentation.

The DG fuel oil storage tanks are designed and fabricated to American Petroleum Institute standard API-650, "Welded Steel Tanks for Oil Storage." Each DG fuel oil storage tank is sized with sufficient capacity, below the auxiliary boiler suction, to provide for seven days of operation of one DG at continuous rating without fuel replenishment. Also, both storage tanks have a combined capacity, above the auxiliary boiler suction, sufficient to operate the auxiliary boiler for seven days. The location and arrangement of the suction nozzles on the storage tank for the auxiliary boiler limit the ABFOSS fuel supply to the top section of the tank. Therefore, fuel stored below the level of the auxiliary boiler suction is reserved for the DGs, and cannot be used for the auxiliary boiler. In addition to the seven-day supply, the fuel oil storage tanks contain additional fuel oil for testing purposes. Fittings are provided for each DG fuel oil storage tank for external tank fill, water removal, level instrumentation, sampling, and sounding. The DG fuel oil storage tanks are vented to the atmosphere with a 102 mm (4 in) line that has a flame arrestor at the end. Additional tank storage capacity is included in the fuel oil storage tank for this purpose. The DG fuel oil storage tank outside fill line, located at the truck unloading station, is above grade and therefore will not be exposed to floods. The fill line incorporates a normally closed valve and a filler cap at the end to preclude the entrance of water. The fill line has a strainer located downstream of the isolation valve to prevent entrance of deleterious solid material into the tank. In Section 9.5.4.7 of the SSAR, Westinghouse states that the site-specific needs to reduce the effects of sun heat input into the stored fuel oil will be addressed by the COL applicant.

The DG fuel oil transfer pumps are of the motor-driven gear, positive displacement type. Each pump has a capacity of approximately four times the full-load consumption rate of its associated DG and can therefore increase the inventory of the day tank while still supplying its DG. Each pump is powered from the electrical bus on which the DG it serves is connected. Failure of a pump or a DG will not affect the operability of components in the other division.

Each DG fuel oil day tank provides four hours of operation for the associated DG at continuous rating without resupply from a fuel oil storage tank. Fittings are provided for each fuel oil day tank for external tank fill, water removal, recirculation, and instrumentation. The fuel oil day tanks are vented to the atmosphere with a 102 mm (4 in) line, which has a ball-float check valve, and a flame arrestor at the end. The flame arrestor is 3 m (10 ft) above the fuel oil storage tank maximum level. The day tank vent terminates above the DG building roof at a higher elevation than the storage tank fuel oil level to prevent fuel spills and to prevent buildup of combustible fumes within the DG building. The day tank is provided with an overflow line that discharges back to the fuel oil storage tank by way of the recirculation line. To protect the day tank in the event of accidental overfilling, its design pressure is sufficient to withstand the system maximum static head. The day tank will be located at an elevation to provide the necessary suction head for the engine-driven fuel oil pump.

The DOS is designed to allow replenishment of fuel without interrupting operation of the DG or auxiliary boiler. The fill connection on the fuel oil storage tanks includes an internal pipe and diffuser to limit inlet filling velocities to prevent turbulence of sediment on the bottom of the tank. The suction connection at the bottom of the fuel oil storage tank for the DGFOTS will be at least 152 mm (6 in.) above the tank bottom to limit agitation of the accumulated residuals at the bottom of the tank. The suction connection at the bottom of the fuel oil day tank will be above potential residue level. In addition, the moisture separator and the duplex fuel oil filter in the diesel fuel oil piping and the duplex fuel oil filter on the DG prevent detrimental effects on DG performance from sediment that may be disturbed during a refill.

The DOS is designed to maintain the fuel oil temperature above the fuel oil cloud point temperature. All of the components of the DOS - except for the fuel oil storage tanks, the buried piping, and the day tanks - are housed in two separate prefabricated weather enclosures. These enclosures are insulated, heated, and ventilated. The fuel oil storage tanks for the DGFOTS, located at grade level, are erected on a continuous concrete slab totally contained within a concrete dike to contain spills and prevent damage to the environment and seepage into the ground water. The buried sections of piping are enclosed in guard pipes to prevent leakage to the environment.

The DGFOTS maintains the fuel oil temperature above the cloud point automatically on low temperatures with an electric fuel oil heater at the discharge of the fuel oil transfer pump and by burial of the transfer piping below the frost line. In recirculation mode, each fuel oil transfer pump recirculates fuel oil from its associated fuel oil storage tank through its electric fuel oil heater and back to its fuel oil storage tank to maintain the fuel oil pipe line temperature above the cloud point temperature. The electric fuel oil heaters and the fuel oil transfer pumps are supplied with electrical power from their associated onsite standby DG backed 480 V bus. The day tanks are located in the DG buildings, which are maintained at a minimum of 10 °C (50 °F) by the DG building heating and ventilation system. In addition, the day tanks are insulated and provided with a 3 kW exterior electric pad heater on the bottom of each tank to maintain the fuel oil above the cloud point when ambient temperature is -6.7 °C (20 °F) or less. Also, above-grade piping and outdoor inline equipment are insulated.

The ABFOSS has no special provisions for fuel oil heating, moisture removal, or filtration because the ABFOSS only supplies fuel oil to the auxiliary boiler for plant heating.

The ADGFOSS maintains the fuel oil temperature above the cloud point by locating the ancillary DG and the ancillary DG fuel oil storage tank in the same room inside the annex building. When normal ac power source is available, the normal annex building heating and ventilation system maintains the annex building temperature at a minimum of 10 °C (50 °F). The ancillary DG fuel oil storage tank is insulated and provided with two 1.25 kW exterior electric pad heaters to maintain the fuel oil above the cloud point when ambient temperature is -6.7 °C (20 °F) or less. In addition, the fuel oil lines from the ancillary DG fuel oil storage tank to the ancillary DGs are insulated.

Provisions are included in the design of the fuel oil storage tanks and day tanks to check and remove accumulated water (the fuel oil storage tanks and day tanks are equipped with water removal ports). In the event that the diesel fuel oil degrades during storage, biocides and other fuel additives will be introduced to the tanked fuel to prevent deterioration of the oil, accumulation of sludge in the storage tanks, and the growth of algae and fungi. In Section 9.5.4.7 of the SSAR, Westinghouse states that the site-specific needs to (1) protect against fuel oil degradation by a program of fuel oil sampling and testing and (2) ensure that the diesel fuel oil specifications, grade, and properties are consistent with manufacturers' recommendations, will be addressed by the COL applicant.

The exterior and interior surfaces of the fuel oil storage tanks are painted with a primer and finish coat system for corrosion protection of the tank surface. Also, exterior surfaces of the diesel fuel oil piping are painted for corrosion protection. The buried sections of the transfer piping are enclosed in guard pipes made of corrosion resistant plastic and designed and fabricated for the site overburden wheel loads resulting from equipment removal and replacement. In Section 9.5.4.7 of the SSAR, Westinghouse states that the site-specific need for cathodic protection in accordance with the National Association of Corrosion Engineers (NACE) Standard RP-01-69, "Recommended Practice for the Control of External Corrosion on Underground or Submerged Metallic Piping Systems," for external metal surfaces of metal tanks in contact with the ground will be addressed by the COL applicant.

The DG fuel oil transfer pumps can be operated from the MCR. A list of the alarms and indications for the DGFOTS and the ABFOSS are provided in Figure 9.5.4-2 of the SSAR. Alarms and indications of tank levels and transfer pump status are displayed in the MCR. Fuel oil storage tank level, auxiliary boiler supply, and diesel oil day tank level are indicated and alarmed in the MCR. A secondary means of tank level determination is provided by dipsticks or sounding ports. A combined trouble alarm is provided in the MCR for the following:

- diesel oil low fuel oil pressure
- diesel oil moisture separator differential pressure
- diesel oil filter and pump suction strainer differential pressure
- diesel oil fuel oil heater in service
- diesel oil fuel oil heater low temp out
- fuel oil tank fill strainer differential pressure

The DGFOTS fuel oil transfer pumps start and stop on low and high level in the day tank, respectively, and the tank level transmitter activates a day tank high or low level alarm. The DG fuel oil transfer pumps start automatically when the level in the day tank decreases to a set capacity. The day tank low level alarm annunciates when the level decreases to a point where two hours of fuel oil remain. The DG fuel oil transfer pumps are automatically stopped when the day tank level has increased to a higher set level. Low fuel oil level in the DG fuel oil storage tanks section reserved for the DGs is also alarmed.

The ABFOSS alarms on low fuel oil pressure as a result of a loss of supply pump pressure, on low level in the fuel oil storage tanks, and high differential pressure across the suction strainer.

The ADGFOSS is neither monitored nor controlled in the MCR. All controls and instruments are local/manual only.

In NUREG/CR-0660, the NRC recommends that all bulk fuel tanks should have a gravity drain at the bottom of the tank for water removal, that the fuel supply pumps for the DG engine should be engine driven, and that the fuel oil supply to this pump should be either gravity fed or from a dc booster pump powered from the Class 1-E station batteries. As discussed above, all of the fuel oil tanks are equipped with water removal ports, the fuel oil supply pump is engine-driven, and the fuel oil day tank is elevated to provide the necessary suction head for the engine driven fuel oil supply pump. Therefore, the DOS complies with these recommendations of NUREG/CR-0660.

The DOS has no safety-related function. While the DOS system is subject to availability controls (see Section 8.6.2.4 of this report), its failure will not prevent safety-related systems from performing their function. Therefore, the staff has determined that compliance with the recommendations of RGs 1.9 and 1.137, IEEE Standard 387, ANSI Standard N195, and the requirements of GDC 17 is not required for the DOS. Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report. On the basis of the above review, the staff concludes that the design of the DOS is acceptable.

9.5.5 Standby Diesel Engine Cooling System

The staff reviewed the standby diesel engine cooling system (SDECS) in accordance with the guidance of Section 9.5.5 of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding whether the SDECS satisfies the following requirements:

- GDC 2, for protection against natural phenomena
- GDC 4, for protection from environmental and dynamic effects of equipment failure
- GDC 5, for sharing SSCs between units
- GDC 17, for availability of electric power systems
- GDC 44, for provision for cooling systems
- GDC 45, for inspection of cooling systems
- GDC 46, for testing of cooling systems

Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 are discussed in Section 9.5.4.1 of this report.

In the AP600 DSER, the staff determined that information about the SDECS (including specific design criteria) should be incorporated into the SSAR. In addition, COL applicant responsibilities needed to be better defined and certain RAI responses needed to be modified and incorporated into the SSAR. This was identified as DSER Open Item 9.5.5-1.

After the AP600 DSER was issued, Westinghouse provided additional information about the SDECS in the form of SSAR revisions and telephone conferences that allowed the staff to continue its review, as discussed below. As a result of the additional information concerning the need to modify certain RAI responses, incorporate certain RAI responses into the SSAR, and to further define the COL responsibilities, DSER Open Item 9.5.5-1 is closed.

There are two standby DGs onsite in the AP600 design. An independent and physically separate closed-loop SDECS division is provided for each DG engine. Thus, a single failure in one of the two divisions will affect only the associated DG.

The SDECS cools the engine jacket water, lube oil, and combustion air. Each SDECS division consists of the following two cooling loops (1) the jacket water cooling loop and (2) the after-cooler/lube oil cooler loop. The jacket water cooling loop consists of the following components:

- an engine-driven jacket water pump
- an electric motor-driven jacket water heater pump
- an electric jacket water heater
- a three-way thermostatic control valve
- a jacket water radiator
- two radiator fans
- an expansion tank

The after-cooler/lube oil cooler loop consists of the following components:

- an engine-driven after cooler/oil cooler water pump
- a turbo charger after cooler
- a lube oil cooler
- a three-way thermostatic control valve
- an after cooler/oil cooler radiator
- two radiator fans
- an expansion tank

The radiator fans and expansion tank are shared by both loops. The jacket water radiator and the after cooler/oil cooler radiator are located at the highest points of the system (on the DG building roof) and are provided with vents and drains so that the system can be completely filled and vented before startup. The expansion tank serves as a surge tank and provides positive suction pressure for the pumps. The electric jacket water heater and the electric motor-driven jacket water heater pump automatically maintain the engine jacket water of the idle engine at a temperature suitable for fast starts.

In Table 8.3.1-5 of the SSAR, Westinghouse provides indications and alarms that will be available locally and in the MCR for the SDECS. The SDECS indications and alarms provided are for high cooling water temperature and low cooling water pressure.

In NUREG/CR-0660, the NRC recommends minimizing no load and light load operation of the DG to reduce formation of gum and varnish deposits on the DG engine internals. In its response to RAI Q410.181 dated July 15, 1994, Westinghouse stated that the SDECS and DG are designed to operate with the DG engine at partial load for extended periods of time without degrading performance. Therefore, the SDECS and DG are designed to comply with this recommendation of NUREG/CR-0660.

In NUREG/CR-0660, the NRC recommends the use of a three-way thermostat temperature control valve for directing the engine water to the bypass or cooler. It should be of or equivalent to the "Amot" brand with an expanding wax-type, temperature-sensitive element.
Section 8.3.1.1.2.1 of the SSAR states that the cooling water in each loop passes through a three-way self-contained temperature control valve, which modulates the flow of water through or around the radiator, as necessary, to maintain required water temperature. The temperature control valve has an expanding wax-type temperature-sensitive element or equivalent. Therefore, the design of the SDECS complies with this recommendation of NUREG/CR-0660.

The SDECS has no safety-related function. While the SDECS is subject to availability controls (see Section 8.6.2.4 of this report), its failure will not prevent safety-related systems from performing their function. Therefore, the staff determined that compliance with the recommendations of RG 1.9 and IEEE 387, and the requirements of GDC 17, 44, 45, and 46 is not required for the SDECS. Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report. On the basis of the above review, the staff concludes that the design of the SDECS is acceptable.

9.5.6 Standby Diesel Engine Starting System

The staff reviewed the standby diesel engine starting system (SDESS) in accordance with the guidance of Section 9.5.6 of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding whether the SDESS satisfies the following requirements:

- GDC 2, for protection against natural phenomena
- GDC 4, for protection from environmental and dynamic effects of equipment failure
- GDC 5, for sharing SSCs between units
- GDC 17, for availability of electric power systems

Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report.

In the AP600 DSER, the staff determined that information about the SDECS (including specific design criteria) should be incorporated into the SSAR. In addition, COL applicant responsibilities needed to be better defined and certain RAI responses needed to be modified and incorporated into the SSAR. This was identified as DSER Open Item 9.5.6-1.

After the AP600 DSER was issued, Westinghouse provided additional information about the SDESS in the form of SSAR revisions and telephone conferences that allowed the staff to continue its review, as discussed below. As a result of the additional information concerning the need to modify certain RAI responses, incorporate certain RAI responses into the SSAR, and further define the COL responsibilities, DSER Open Item 9.5.6-1 is closed.

There are two standby DGs onsite in the AP600 design. An independent and physically separate closed-loop SDESS division is provided for each DG engine. Thus, a single failure in one of the two divisions will affect only the associated DG.

Each SDESS division consists of the following components:

- an inlet air filter
- an electric motor-driven, air-cooled air compressor
- an air-cooled aftercooler
- inline air filters
- a refrigerant dryer
- an air receiver
- an air admission valve
- a starting air motor

In its response to RAI Q410.180, dated August 3, 1994, Westinghouse stated that the SDESS is designed to provide three consecutive starts of a cold DG engine, without recharging the starting air receivers with the normal coldest ambient temperature in the DG building of 10 °C (50 °F). As a result, Westinghouse revised Section 8.3.1.1.2.1 of the SSAR to state that the air receivers are designed with sufficient capacity for three DG engine starts. In Section 9.5.6 of the SRP, The staff specifies that emergency diesel engine starting systems should be capable of cranking a cold DG a minimum of five times without recharging the air receiver. Because the

SDESS is not safety-related, the staff determined that three consecutive starts provides adequate assurance of being capable of starting the standby DG.

In Table 8.3.1-5 of the SSAR, Westinghouse provides the indication and alarms that will be available locally and in the MCR for the SDESS. The SDESS indication and alarm provided are for low starting air pressure.

In NUREG/CR-0660, the NRC recommends using air dryers (desiccant or refrigerated types) for removing moisture from the starting air system, because water in the starting air is the primary contributor to DG engine starting failures. In its response to RAI Q410.183 dated August 3, 1994, Westinghouse stated that moisture in the SDESS is reduced by using refrigerated-type air dryers that supply air with a maximum dew point temperature of 4.4 °C (40 °F). Westinghouse revised Section 8.3.1.1.2.1 of the SSAR to state that the dew point is maintained at least 5.6 °C (10 °F) less than the lowest expected ambient temperature in the DG building of 10 °C (50 °F). The starting air system supply air will be at the normal DG building temperature, between 10 °C (50 °F) and 40.6 °C (105 °F). The air receivers are equipped with drain lines for periodic manual blowdown of accumulated moisture and foreign material. In addition, Section 8.3.1.1.2.1 of the SSAR states that requirements for the control of moisture in the SDESS will be determined by the manufacturer's recommendations. Therefore, the SDESS complies with the recommendations of the NUREG/CR-0660.

In Section 8.3.1.1.2.1 of the SSAR, Westinghouse states that the SDESS will be consistent with manufacturer's recommendations regarding the devices to crank the engine, the duration of the cranking cycle, the number of engine revolutions per start attempt, the volume and design pressure of the air receivers, and compressor size.

The SDESS has no safety-related function. While SDESS is subject to availability controls (see Section 8.6.2.4 of this report), its failure will not prevent safety-related systems from performing their function. Therefore, the staff has determined that compliance with the recommendations of RG 1.9 and IEEE 387, and the requirements of GDC 17 is not required for the SDESS. Compliance with the requirements of GDC 2, 4, and 5, and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report. On the basis of the above review, the staff concludes that the design of the SDESS is acceptable.

9.5.7 Standby Diesel Lubricating Oil System

The staff reviewed the standby diesel lubricating oil system (SDLOS) in accordance with the guidance of Section 9.5.7 of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding whether the SDLOS satisfies the following requirements:

- GDC 2, for protection against natural phenomena
- GDC 4, for protection from environmental and dynamic effects of equipment failure
- GDC 5, for sharing SSCs between units
- GDC 17, for availability of electric power systems

Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report.

In the AP600 DSER, the staff determined that information about the SDLOS (including specific design criteria) should be incorporated into the SSAR. In addition, the use of equipment procurement specification and COL applicant responsibilities needed to be better defined and certain RAI responses needed to be incorporated into the SSAR. This was identified as DSER Open Item 9.5.7-1.

After the AP600 DSER was issued, Westinghouse provided additional information about the SDLOS in the form of SSAR revisions and telephone conferences that has allowed the staff to continue its review, as discussed below. As a result of the additional information concerning the need to modify certain RAI responses, incorporate certain RAI responses into the SSAR, further define the COL responsibilities, and clarify the use of equipment procurement specifications, DSER Open Item 9.5.7-1 is closed.

There are two standby DGs onsite in the AP600 design. An independent and physically separate closed-loop SDLOS division is provided for each DG engine. Thus, a single failure in one of the two divisions will affect only the associated DG.

The SDLOS is contained on the engine skid and consists of an engine lube oil sump, an engine-driven main lube oil pump, and a continuous engine prelube system that includes an ac motor prelube oil pump, a dc motor prelube pump, and a "keep-warm" lube oil heater. The SDLOS is designed to maintain the engine lubrication system in a keep warm mode. The prelube pump continuously circulates heated lube oil through the engine while it is in the standby mode to provide a fast-start capability.

In Table 8.3.1-5 of the SSAR, Westinghouse provides the indications and alarms that will be available locally and in the MCR for the SDESS. The SDESS indications and alarms provided are for low lube oil pressure, high lube oil temperature, and low lube oil sump level.

In NUREG/CR-0660, the NRC recommends that prelube periods of a maximum of 3 to 5 minutes be required preceding all engine starts (except for an actual or simulated emergency start) because DG engine problems may be caused by excessively long prelube periods with the Fairbanks Morse opposed-piston DG engine. Prelube periods of more than approximately 5 minutes are to be used only by specification or recommendation of the particular engine manufacturer. It is recommended that the engine prelube pump be started by the same signal that initiates the cranking of the DG engine and be stopped when the engine stops cranking to limit metal-to-metal contact in the bearings.

In its response to RAI Q410.183 dated August 3, 1994, Westinghouse stated that the equipment procurement specifications for the DG engine will require that the engine be provided with a continuous keep-warm and prelube system that remains in operation while the engine is in the standby mode. During a meeting (December 13 - 14, 1994), Westinghouse stated that the AP600 design does not utilize the Fairbanks Morse opposed-piston DG engine. Nevertheless, Section 8.3.1.1.2.1 of the SSAR states that the automatic engine prelube equipment will be based upon manufacturer's recommendations. Therefore, the SDLOS complies with the recommendations of the NUREG/CR-0660. Section 8.3.1.1.2.1 of the SSAR also states that the capability to detect system leakage and to prevent crankcase explosions will be based upon manufacturer's recommendations.

The SDLOS has no safety-related function. While the SDLOS is subject to availability controls (see Section 8.6.2.4 of this report), its failure will not prevent safety-related systems from performing their function. Therefore, the staff has determined that compliance with the recommendations of RG 1.9 and IEEE 387, and the requirements of GDC 17 is not required for the SDLOS. Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report. `On the basis of the above review, the staff concludes that the design of the SDLOS is acceptable.

9.5.8 Standby Diesel Combustion Air Intake and Exhaust System

The staff reviewed the standby diesel combustion air intake and exhaust system (SDCAIES) in accordance with the guidance of Section 9.5.8 of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding whether the SDCAIES satisfies the following requirements:

- GDC 2, for protection against natural phenomena
- GDC 4, for protection from environmental and dynamic effects of equipment failure
- GDC 5, for sharing SSCs between units
- GDC 17, for availability of electric power systems

Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report.

In the AP600 DSER, the staff determined that information about the SDCAIES (including specific design criteria) should be incorporated into the SSAR. In addition, the use of equipment procurement specifications and COL applicant responsibilities needed to be modified and incorporated into the SSAR. This was identified as DSER Open Item 9.5.8-1.

After the AP600 DSER was issued, Westinghouse provided additional information about the SDCAIES in the form of SSAR revisions and telephone conferences that allowed the staff to continue its review as discussed below. As a result of the additional information concerning the need to modify certain RAI responses, incorporate certain RAI responses into the SSAR, further define the COL responsibilities, and clarify the use of equipment procurement specifications, DSER Open Item 9.5.8-1 is closed.

There are two standby DGs onsite in the AP600 design. An independent and physically separate closed-loop SDCAIES division is provided for each DG engine. Thus, a single failure in one of the two divisions will affect only the associated DG.

The SDCAIES provides combustion air directly to the DG engine and discharges exhaust gases from the engine to the outside of the DG building. Each SDCAIES division consists of (1) the combustion air circuit and (2) the engine exhaust gas circuit. The combustion air circuit includes two weather-protected, dry-type disposable inlet air filters, two diesel engine-mounted turbochargers, an emergency shut off valve, and an aftercooler. The engine exhaust gas circuit consists of two engine exhaust gas manifolds, two diesel engine-mounted turbochargers, discharge pipes from the turbocharger outlets to a single vertically mounted outdoor silencer, which discharges to the atmosphere. The exhaust gas-driven turbochargers compress the combustion air and deliver it to the DG.

In Section 8.3.1.1.2.1 of the SSAR, Westinghouse states that manufacturer's recommendations are considered in the design of features to protect the silencer module and other system components from possible clogging resulting from adverse atmospheric conditions such as dust storms, rain, ice, and snow.

In NUREG/CR-0660, the NRC recommends that DG engine combustion air should be taken from outside the DG building and at least 6.1 m (20 ft) from the ground level through filters. Also, room ventilation air should be filtered and taken from a level at least 6.1 m (20 ft) above ground level. The piping for the room ventilation air should be separate from that used for the DG engine combustion air. Room ventilation air, hot cooling system air, and DG engine exhaust gas should not be permitted to circulate back into the DG room or into any other part of the power plant building.

In Section 8.3.1.1.2.1 of the SSAR, Westinghouse states that the SDCAIES provides combustion air directly from the outside to the DG engine while protecting it from dust, rain, snow, and other environmental particulates. It then discharges exhaust gases from the DG engine to the outside of the DG building more than 6.1 m (20 ft) higher than the air intake. The SDCAIES is separate from the DG building heating and ventilation system and includes weather-protected, dry-type inlet air filters piped directly to the inlet connections of the DG engine-mounted turbochargers. The combustion air filters are capable of reducing airborne particulate material, assuming the maximum expected airborne particulate concentration at the combustion air intake. Each engine is provided with two filters. A differential pressure gauge is installed across each filter to determine the need for filter replacement. In addition, in Section 9.4.10.2.1.1 of the SSAR, Westinghouse states that the air intake for the DG building heating and ventilation system will be as high in the DG building wall as possible to limit the intake of dust blown about by wind and/or passing vehicles. Therefore, the staff finds this design acceptable to meet the intent of this recommendation of NUREG/CR-0660.

The SDCAIES has no safety-related function. While the SDCAIES is subject to availability controls (see Section 8.6.2.4 of this report), its failure will not prevent safety-related systems from performing their function. Therefore, the staff has determined that compliance with the recommendations of RG 1.9 and IEEE 387, and the requirements of GDC 17 is not required for the SDCAIES. Compliance with the requirements of GDC 2, 4, and 5 and the recommendations of NUREG/CR-0660 is discussed in Section 9.5.4.1 of this report. On the basis of the above review, the staff concludes that the design of the SDCAIES is acceptable.

Table 9.4-1 HVAC System Components

Component	Standard	Title
Supply, return, and exhaust fans	ANSI/AMCA 210-85	Laboratory Methods of Testing Fans for Rating Purposes
	ANSI/AMCA 211-85	Certified Ratings Program Air Performance
	AMCA 300-85	Reverberant Room Method of Testing Fans for Rating Purposes, Standards
Supply air ductwork, supports, and accessories	SMACNA-1975	High Pressure Duct Construction Standards
	SMACNA-1985	HVAC Duct Construction Standards - Metal and Flexible
Low efficiency filters, high-efficiency filters, and post-filters	ASHRAE 52-76	Methods of Testing Air-Cleaning Devices Used in General Ventilation for Removing Particulate Matter, Standards
	UL 900, 1986	Test Performance of Air-Filter Units, Class I Criteria
Cooling coils and hot water heating coils	ANSI/ARI 410-91	Forced Circulation Air Cooling and Air Heating Coils
	ASHRAE 33-78	Methods of Testing for Rating Forced Circulation Air Cooling and Air Heating Coils
Electric unit heaters	UL-1025-1991	Electric Air Heaters
	NFPA 70, 1970	National Electrical Code
Electric heating coils	UL1096-1986	Electric Central Air Heating Equipment
Humidifiers	ARI 620-80	Self-Contained Humidifiers, Standards
Dampers	ANSI/AMCA 500-83	Testing Methods for Louvers, Dampers, and Shutters, Standards
		or
	ASME N509-1989	Nuclear Power Plant Air-Cleaning Units Components
Fire dampers	UL-555, 1990	Fire Dampers
Smoke and fire dampers	UL-555S, 1993	Leakage Rated Dampers for Use in Smoke Control Systems

Table 9.5-1 Conformance To NUREG/CR-0660 Recommendations for Diesel Generator Auxiliary Support Systems

Recommendation		Conformance	FSER Section
(1)	Moisture in air starting system	Yes	9.5.6
(2)	Dust and dirt in diesel generator room	Yes	9.5.4.1
(3)	N/A to support systems	--	--
(4)	Personnel training	Specific site	9.5.4.1
(5)	Automatic prelube	Yes	9.5.7
(6)	Testing, test loading, and preventive maintenance	Specific site	9.5.4.1
(7)	Improve the identification of cause of failures	Specific site	9.5.4.1
(8)	Diesel generator ventilation and combustion air systems	Yes	9.5.8
(9)	Fuel storage and handling	Yes	9.5.4.2
(10)	High-temperature insulation	Yes	9.5.4.1
(11)	Engine cooling water	No	9.5.5
(12)	Concrete dust control	Yes	9.5.4.1
(13)	Vibration of instruments	Yes	9.5.4.1

10 STEAM AND POWER CONVERSION SYSTEM

10.1 Introduction

The steam and power conversion system is designed to convert the heat energy generated by the reactor into electric power. The steam and power conversion system for the AP600 design is described in Chapter 10 of the standard safety analysis report (SSAR). This system generates electricity by using the main steam system to drive a turbine generator unit. Two steam generators produce steam from the heat energy generated by the reactor to supply the turbine for the main steam system.

The turbine exhaust steam is condensed and deaerated in the main condenser. The heat rejected in the main condenser is removed by a closed-loop circulating water system (CWS). The condensate pumps take suction from the condenser and deliver the condensate water through heaters to the suction of the main feedwater booster pump. The water is then discharged to the suction of the main feedwater pumps, which discharge the feedwater through feedwater heaters to the two steam generators.

Steam from each of two steam generators enters the high-pressure turbine through four stop valves and four governing control valves. Crossties are provided upstream of the turbine stop valves to equalize pressure. The turbine bypass system provides the capability to relieve a combined capacity of 40 percent of total full-power steam flow to the condenser during startup, hot shutdown, cooldown, and step-load reductions in generator loads.

The protective features for the steam and power conversion system include the following:

- loss of external electrical load and/or turbine trip protection
- main steamline overpressure protection
- loss of main feedwater flow protection
- turbine overpressure protection
- turbine missile protection
- radioactivity protection
- erosion-corrosion protection

Westinghouse indicated in response to a request for additional information (RAI) 410.138 that among the above protective features only the main steamline overpressure protection is a safety-related function. The other features are for investment protection and personnel safety. Spring-loaded safety valves are provided on both main steamlines for the overpressure protection, in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. The pressure relief capacity of the safety valves is such that the energy generated at the high-flux reactor trip setting can be dissipated through this system. The design capacity of the main steam safety valves equals or exceeds 105 percent of the nuclear steam supply system (NSSS) design steamflow at an accumulation pressure not exceeding 110 percent of the main steam system design pressure.

The steam and power conversion system description, design and performance characteristics are identified in Section 10.1, and Table 10.1-1 of the SSAR. In reviewing the system description, the staff requested that Westinghouse provide a heat balance diagram of the steam and power conversion system in the non-proprietary version of the SSAR. This was identified as draft safety evaluation report (DSER) Confirmatory Item 10.1-1. In Revision 5 to the SSAR, Westinghouse provided Figure 10.1-1, "Heat Balance," to the SSAR, which resolves DSER Confirmatory Item 10.1-1.

10.2 Turbine Generator

The staff reviewed the design of the turbine generator in accordance with Section 10.2 of the standard review plan (SRP). The design of the turbine generator system is acceptable if the integrated design of the system meets the requirement of General Design Criterion (GDC) 4 as related to the protection of the structures, systems, and components (SSCs) that are important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generation of turbine missiles. Specific criteria necessary to meet the requirements of GDC 4 are described in SRP Section 10.2.II.

DSER Open Item 10.2.10-1 identified several questions in the areas of diversity of overspeed trips, closing time for extraction nonreturn lines, and testing intervals for turbine valves as discussed in RAIs 410.139, 410.143, and 410.144. Westinghouse provided responses to the RAIs, discussed the remaining staff concerns resulting from the review of the responses in a meeting of December 12-13, 1994, and provided additional information to address those concerns. The staff reviewed the additional information and has found it acceptable. The staff used this information in developing the following evaluation. Therefore, DSER Open Item 10.2.10-1 is closed. The staff concerns are discussed in the following sections.

The turbine generator converts the thermal energy into electric power. The AP600 turbine generator has a nominal rating of 675,000 kW for the Westinghouse NSSS thermal output of 1,940 MWt. In the DSER, the staff found that the nominal rating was listed inconsistently between Section 10.1.1 and Table 10.2-1 of the SSAR. Westinghouse explained that the differences are there because the KWt in Section 10.1.1 represents the required KWt, and the KWt in Table 10.2-1 represents the KWt that the turbine generator may provide, which has a margin over the requirement. Furthermore, the thermal output of 1940 MWt and 1933 MWt are listed in Section 10.1.1 and Table 4.1-1 of the SSAR respectively. The difference represents the difference between the core power and the steam generator power (i.e., reactor coolant pump heat addition and reactor coolant system heat loss). Therefore, Westinghouse resolved the above discrepancy. Therefore, DSER Open Item 10.2-1 is closed.

The Westinghouse turbine generator is designated as Model TC4F with a 47-inch last-stage blade unit. Section 10.2.2 of the SSAR states that the turbine generator and its components are shown in Figure 10.2-1. However, Figure 10.2-1 was proprietary. In RAI 410.140, the staff requested Westinghouse to present the information as a non-proprietary figure. In Revision 5 to the SSAR, Westinghouse provided Figure 10.2-1, "Turbine Generator Outline," in the SSAR. Therefore, Confirmatory Item 10.2-1 is closed.

The design parameters of the turbine generator are identified in Table 10.2.1 of the SSAR. The piping and instrumentation diagram containing the stop, governing control, intercept, and reheat valves is shown in Figure 10.3.2-2 of the SSAR. The turbine generator consists of a double-flow, high-pressure turbine and two double-flow, low-pressure turbines. Other related system components include: a turbine-generator bearing lubrication oil system, a digital electrohydraulic (DEH) control system, a turbine steam sealing system, overspeed protective devices, turning gear, a generator hydrogen and seal oil system, a generator CO_2 system, an exciter cooler, a rectifier section, and a voltage regulator.

The design of the turbine generator foundation is a spring-mounted support system. The springs dynamically isolate the turbine generator deck from the remainder of the structure in the range of operating frequencies.

Steam from each of two steam generators enters the high-pressure turbine through stop valves and governing control valves. After expanding through the high-pressure turbine, exhaust steam flows through one external moisture separator/reheater. The reheated steam flows through separate reheat stop and intercept valves leading to the inlets of the two low-pressure turbines. Turbine steam is supplied to feedwater heaters.

10.2.1 Overspeed Protection

The overspeed protection control of the DEH control system and the emergency trip system (ETS) are provided to protect the turbine against overspeed.

The overspeed protection control of the DEH control system opens a drain path for the hydraulic fluid in the overspeed protection control header if turbine speed exceeds 103 percent of rated speed. The loss of fluid pressure in the header causes the control and intercept valves to close. Following the above valve closure, if the turbine speed falls below rated speed and the header pressure is reestablished, the control and intercept valves are reopened, and the unit resumes speed control. Additional discussion of the DEH control system is in Section 10.2.2 of this final safety evaluation report (FSER). In addition, an emergency trip system is provided to trip the turbine, in the event that speeds exceed the overspeed protection control trip setpoints (110 percent of rated speed). Additional discussion of the ETS is in Section 10.2.4 of this report.

10.2.2 Digital Electrohydraulic Control System

The turbine generator is equipped with a digital electrohydraulic (DEH) control system. The DEH control system has two modes of operation. The first is the speed control that functions to maintain the desired speed, and the second mode is the overspeed protection control, which operates if the normal speed control should fail or upon a load rejection.

The DEH control system combines the capabilities of solid-state electronics and high-pressure hydraulics to regulate steam flow through the turbine. The control system has a speed control unit, a load control unit, and an automatic turbine control (ATC) unit. The ATC is discussed in Section 10.2.3 of this report. Valve opening actuation in the DEH control system is provided by a hydraulic system while closing actuation is provided by springs and steam forces upon reduction or relief of fluid pressure. A trip signal is sent to a fast acting solenoid valve. Energizing the solenoid valves releases the hydraulic fluid pressure in the valve actuators, allowing springs to close each valve. The system is designed so that loss-of-fluid pressure leads to valve closure and consequent turbine trip. Steam valves are provided in series pairs.

A stop valve is tripped by the overspeed trip system; the control valve is modulated by the governing system and actuated by the trip system.

10.2.3 Automatic Turbine Control

Automatic turbine control (ATC) regulates turbine speed and acceleration through the entire speed range. When the operator selects ATC, the programs both monitor and control the turbine.

The ATC is capable of automatically performing the following activities:

- changing speed
- changing acceleration
- generating speed holds
- changing load rates
- generating load holds

10.2.4 Turbine Protective Trips

The turbine protective trips are independent of the electronic control system, and, when initiated, cause tripping of the turbine stop and control valves. The protective trips are as follows:

- low bearing oil pressure
- low electrohydraulic fluid pressure
- high condenser back pressure
- turbine overspeed
- thrust bearing wear
- remote trip that accepts external trips

The ETS discussed in Section 10.2.1 is designed for the turbine overspeed trip. The ETS can detect undesirable operating conditions of the turbine generator, take appropriate trip actions, and provide information to the operator about the detected conditions and the corrective actions.

The ETS consists of the following: an emergency trip control block with trip solenoid valves and status pressure sensors, three test trip blocks with pressure sensors and test solenoid valves, rotor position pickups, a speed pickup, a cabinet containing electrical and electronic hardware, and a remotely mounted status and test panel.

The ETS utilizes a two-channel configuration which permits online testing with continuous protection afforded during the test sequence. A trip of the ETS opens a drain path for the hydraulic fluid in the auto stop emergency trip header. The loss of fluid pressure in the trip header causes the main stop and reheat stop valves to close. Also, check valves in the connection to the overspeed protection control header open to drop the pressure and cause the control and intercept valves to close. The control and intercept valves are redundant to the main stop and reheat stop valves respectively. Section 10.2.2.8 of the SSAR states that major system components are readily accessible for inspection and are available for testing during normal plant operation, and that turbine trip circuitry is tested prior to unit startup.

The staff reviewed the above information, as described in Sections 10.2.1 through 10.2.3 of the SSAR, to confirm that there is sufficient redundancy on turbine overspeed protection. In the DSER, the staff found that the AP600 turbine generator design is in conformance with Acceptance Criteria II.1 and II.4 of Section 10.2 of the SRP, with one exception. The exception being that the AP600 turbine generator did not have a mechanical overspeed trip device as described in Paragraph III.2.C of Section 10.2 of the SRP. The staff requested Westinghouse to provide the justification for not having a mechanical overspeed trip device. Specifically, the concern of diversity and common-mode failure should be addressed. This was identified as DSER Open Item 10.2.4-1. In Revision 15 of the SSAR, Westinghouse decided to revise its design of the emergency overspeed trip system such that it will consist of a mechanical and an electric trip.

The mechanical overspeed trip device consists of a spring-loaded trip weight mounted in the rotor extension shaft. The mechanical overspeed and manual trip header can be tripped manually via a trip handle mounted on the governor pedestal. The electrical overspeed trip system has separate, redundant speed sensors and provides backup overspeed protection utilizing the trip solenoid valves in the emergency trip control block to drain the emergency trip header. The speed control and overspeed protection function of the DEH control system combined with the ETS electrical and mechanical overspeed trips provide sufficient level of redundancy and diversity. Therefore, DSER Open Item 10.2.4-1 is closed.

10.2.5 Valve Control

Criterion II.2 of Section 10.2 of the SRP states that turbine main steam stop and control valves and reheat steam stop and intercept valves should be provided to protect the turbine from exceeding set speeds and to protect the reactor system from abnormal surges. The reheat stop and intercept valves should be capable of closure concurrent with the main steam stop valves, or of sequential closure within an appropriate time limit, to assure that turbine overspeed is controlled within acceptable limits. The valve arrangements and valve closure times should be such that a failure of any single valve to close will not result in an excessive turbine overspeed in the event of a turbine generator system trip signal.

SSAR Section 10.2.2.4.3 states that the flow of the main steam entering the high-pressure turbine is controlled by four stop valves and four governing control valves. Each stop valve is controlled by an electrohydraulic actuator, so that the stop valve is either fully open or fully closed. The function of the stop valves is to shut off the steam flow to the turbine when required. The stop valves fully close within 0.3 seconds of actuation of the ETS devices, which are independent of the electronic flow control unit.

The turbine control valves are positioned by electrohydraulic servo actuators in response to signals from their respective flow control units. The flow control unit signal positions the control valves for wide-range speed control through the normal turbine operating range, and for load control after the turbine generator unit is synchronized.

The reheat stop and intercept valves, located in the hot reheat lines at the inlet to the low-pressure turbines, control steam flow to the low-pressure turbines. During normal operation of the turbine, the reheat stop and intercept valves are wide open. The intercept valve flow control unit positions the valve during startup and normal operations, and closes the valve rapidly on loss of turbine loads. The reheat stop valves close completely on a turbine overspeed and turbine trips. Quick closure of the steam valves prevents a turbine overspeed. The valve closure time for both the reheat stop valves and intercept valves is 0.3 seconds. Because redundancy is built into the overspeed protection systems, the failure of a single valve will not disable the trip functions.

Westinghouse performed an analysis for the AP600 design in accordance with ASME PTC 20.2 (Overspeed Trip Systems for Steam Turbine-Generator Units). Valves connected to the turbine which could contribute to a turbine overspeed are considered in the analysis. The analysis identified that the emergency stop valve closing time is important, but the closing time of extraction nonreturn valves does not contribute to overspeed. Therefore, a closing time of one second was specified for the extraction nonreturn valves to ensure that they will close during an accident to minimize overall steam flow following a turbine trip.

On the basis of the above discussion, the staff concludes that the AP600 design is in conformance with Criteria II.2 and II.3 of Section 10.2 of the SRP with respect to the availability and adequacy of the control valves.

10.2.6 Turbine Missiles

The turbine generator and associated piping, valves, and controls are located completely within the turbine building. There are no safety-related systems or components located within the turbine building. The orientation of the turbine generator is such that a high-energy missile would be directed at a 90-degree angle away from safety-related structures, systems, or components. Failure of turbine generator equipment does not preclude a safe shutdown of the reactor. The issue of turbine missiles is addressed in Section 3.5.1.3 of this report.

10.2.7 Inservice Inspection

Criteria II.5 of Section 10.2 of the SRP states that an inservice inspection (ISI) program for main steam and reheat valves should be provided and it should include the following provisions:

- At approximately 3-1/3-year intervals, during refueling or maintenance shutdowns coinciding with the ISI schedule required by Section XI of the ASME Code for reactor components, at least one main steam stop valve, one main steam control valve, one reheat stop valve, and one reheat intercept valve should be dismantled and visual and surface examinations conducted of valve seats, disks, and stems. If unacceptable flaws or excessive corrosion is found in a valve, all other valves of that type should be dismantled and inspected. Valve bushings should be inspected and cleaned, and bore diameters should be checked for proper clearance.

- Main steam stop and control valves and reheat stop and intercept valves should be exercised at least once a week by closing each valve and observing by the valve position indicator that it moves smoothly to a fully closed position. At least once a month, this examination should be made by direct observation of the valve motion.

ISIs for the turbine assembly and valves are described in Section 10.2.3.6 of the SSAR.

The maintenance and inspection program for turbine assembly and valves is based on turbine missile probability calculations, operating experiënce of similar equipment, and inspection results. The methodology used for analysis of the missile generation probability calculations used to determine turbine valve test frequency is described in WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency." At least one main steam stop valve, one main steam control valve, one reheat stop valve, and one intercept valve is dismantled approximately every four years during scheduled refueling or maintenance shutdowns. A visual and surface examination of valve internals is conducted. If unacceptable flaws or excessive corrosion is found in a valve, the other valves of its type are inspected. Valve bushings are inspected and cleaned, and bore diameters are checked for proper clearance. Westinghouse provided justification in Revision 8 of the SSAR that the valve inspection frequency of four years noted above is consistent with a 24-month fuel cycle for AP600 and is based on evaluations performed to support this valve inspection interval at operating plants with 24-month fuel cycles. The staff finds the justification acceptable.

Westinghouse refers to EPRI ALWR Utility Requirements Document (URD), Volume III, Chapter 13, Section 2.2.3, where EPRI states that the turbine valves shall have a design goal for the capability to operate for a minimum of six years between inspections. The staff SER for EPRI URD (NUREG-1242) states that this is acceptable based on the added commitment in the maintenance program described in Section 2.3.8.3.2 of the URD Volume III, Chapter 13. It states that the "Plant Designer shall provide recommendations for the plant maintenance program ... These recommendations will be based on both turbine missile generation probability analyses as well as plant availability. The basis shall be provided to the plant owner." Westinghouse states in Revision 5 of the SSAR that a Combined Operating License (COL) holder recommendation for a valve inspection frequency longer than four years may be justified when a longer interval is supported by operating inspection program experience and supported by the missile generation probability calculations. The staff review was based on the approved four year inspection intervals. Inspection intervals longer than four years will have to be approved by the NRC.

In addition, main stop valves, control valves, reheat stop and intercept valves are tested with the turbine online. The DEH control system test panel is used to stroke the valves. SSAR Section 10.2.3.6 states that the turbine valve testing is performed at quarterly intervals, based on nuclear industry experience that turbine-related tests are the most common causes of plant trips at power. Evaluations in WCAP-11525, show that the probability of turbine missile generation with a quarterly valve test is within the evaluation criteria. The staff finds that while the quarterly interval deviates from the specified test intervals of once per week in Criterion II.5.b of Section 10.2 of the SRP, the justification that Westinghouse provided in WCAP-11525 was reviewed and found acceptable by the staff in a safety evaluation, dated November 2, 1989, issued to Westinghouse.

10.2.8 Access to Turbine Areas

Criterion II.6 of Section 10.2 of the SRP states that unlimited access to all levels of the turbine area under all operating conditions should be provided. Radiation shielding should be provided as necessary to permit access.

Under operating conditions, there is access to turbine generator components and instrumentation associated with a turbine generator overspeed protection. Major system components are readily accessible for inspection, and are available for testing during normal plant operation.

Since the steam generated in the steam generators is not normally radioactive, no radiation shielding is provided for the turbine generator and associated components. Radiological considerations do not affect access to system components during normal conditions.

Based on the above discussion, the staff concludes that the turbine generator design is in conformance with Criterion II.6 of Section 10.2 of the SRP. Furthermore, Criterion II.7 of Section 10.2 of the SRP states that connection joints between the low-pressure turbine exhaust and the main condenser should be arranged to prevent adverse effects on any safety-related equipment in the turbine room in the event of rupture (it is preferable not to locate safety-related equipment in the turbine room). Criterion II.7 is satisfied because no safety-related equipment is located in the turbine building.

10.2.9 Turbine Rotor Integrity

GDC 4 requires that SSCs important to safety shall be appropriately protected against environmental and dynamic effects, including the effects of missiles, that may result from equipment failure. Because turbine rotors have large masses and rotate at relatively high speeds during normal reactor operation, failure of a rotor may result in the generation of high-energy missiles and cause excessive vibration of the turbine rotor assembly. The staff reviewed the measures taken by Westinghouse to ensure turbine rotor integrity and reduce the probability of turbine rotor failure.

The staff used the guidelines of Section 10.2.3 of the SRP to review and evaluate the information submitted by Westinghouse to ensure rotor integrity and low probability of turbine rotor failure with the generation of missiles. Section 10.2.3 of the SRP provides that the turbine rotor materials have an acceptable fracture toughness and elevated temperature properties, and the rotor is adequately designed and inspected prior to service and inservice during plant shutdown coinciding with the ISI schedule as required by the ASME Code, Section XI.

Section 10.2.3 of the SSAR provides information concerning the turbine rotor material. The staff reviewed this information and requested additional information concerning rotor design, materials and ISI from Westinghouse in order to evaluate the adequacy of the turbine rotor material. Westinghouse responded to the staff's requests for additional information (RAI), however, the information in some instances was incomplete or was not addressed in the SSAR. Accordingly, several issues were identified in the DSER as open items. The resolution of the DSER open items are discussed below.

As described in responses to staff RAIs, AP600 turbine rotors are made from a vacuum-melted, deoxidized Ni-Cr-Mo-V alloy steel by processes which minimize flaw occurrence and can provide adequate fracture toughness. However, the SSAR did not describe these processes. These processes should be described in Chapter 10 of the SSAR. This was Open Item 10.2.9-1. In response to this open item, Westinghouse provided topical report WSTG-4-P, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors." The report indicated that the turbine rotors are made from forgings made to the requirements of materials specifications ASTM A470 and A471. These materials specifications contain a description of the alloy steel making processes. The report is referenced in section 3.5.6 of the AP600 SSAR. In addition, Westinghouse revised SSAR Section 10.2.3. Revision 5 of the SSAR indicated that turbine rotor forgings will be made to the requirements of ASTM A470 Class 5, 6 and 7 with strict limits being imposed on phosphorous, aluminum, antimony, tin, argon, and copper. Limiting these impurities in the turbine rotor in accordance with ASTM A470 will assure its appropriate fracture toughness. This is acceptable. Open Item 10.2.9-1 is closed.

Residual elements are controlled to the lowest practical concentrations consistent with melting practice. Westinghouse responses to the staff requests for additional information as documented in 250.28 (November 30, 1992) (the parts regarding chemical and material property tests), RAI 251.25 (November 30, 1992), RAI 251.26 (November 30, 1992), RAI 251.27 (November 30, 1992), RAI 251.28 (November 30, 1992), and RAI 251.29 (January 22, 1993) provides, in part, some of the information necessary for an evaluation of the adequacy of the turbine rotor material. In the DSER, the staff noted that the basic chemical and material test requirements (i.e., chemical composition, mechanical properties, number and location of specimens, test temperature, and so forth) were not specified in responses to the RAIs. The staff stated that the SSAR needed to contain all of the necessary chemical and material property requirements to enable the staff to evaluate the adequacy of the turbine rotors for their performance in service. This was Open Item 10.2.9-2. Westinghouse provided topical report WSTG-4-P, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors." The report indicated that the turbine rotors are made from forgings made to the requirements of materials specifications ASTM A470 and A471. Further the report indicated that the forgings have a minimum specified yield strength between 80 ksi and 140 ksi depending upon the particular application's requirements. The report is referenced in section 3.5.6 in the AP600 SSAR. In addition, Westinghouse revised SSAR Section 10.2.3. Revision 5 of the SSAR indicated that turbine rotor forgings will be made to the requirements of ASTM A470 Class 5, 6 and 7 with strict limits being imposed on phosphorous, aluminum, antimony, tin, argon, and copper. This is acceptable. Open Item 10.2.9-2 is closed.

Westinghouse stated that the turbine materials have the lowest fracture appearance transition temperature (FATT) and the highest Charpy V-notch properties obtainable from water-quenched Ni-Cr-Mo-V material of the size and strength level used. The November 30, 1992, responses to RAIs 251.25, 251.26, and 251.27 contain the information needed for an evaluation of fracture requirements. The January 22, 1993, response to RAI 251.29 provides the correlations of Charpy V-notch properties with fracture toughness for the turbine rotor material. In the DSER, the staff stated that the basis for the correlation of Charpy V-notch properties and fracture toughness of the turbine rotor material needs to be included in the SSAR. In addition, the relevance of a well-defined Charpy energy and fracture appearance transition temperature curve and strain-rate sensitivity should also be discussed in the SSAR. This was Open Item 10.2.9-3. Westinghouse provided topical report WSTG-4-P, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Rotors." The report is referenced in section 3.5.6 of the AP600 SSAR. The report indicated that the fracture toughness K_{IC} in ksi$\sqrt{in}$ is calculated from the minimum Charpy V-notch energy and yield strength using Barsom-Rolfe correlation. As stated above Westinghouse is also committed to use rotor forgings that are made to the requirements of ASTM A470 Class 5, 6 and 7. Specification A470 requires that Class 5 ,6 and 7 forgings have at least 90 ksi

(620 Mpa) yield strength. Since the specified materials will have a yield strength greater that 90 ksi (620 Mpa) the use of the Barsom-Rolfe correlation is acceptable. Open Item 10.2.9-3 is closed.

The AP600 turbine rotor design will be a solid-forging, fully-integral rotor rather than disks shrunk on a shaft. The current practice employed by some turbine manufacturers for the large, low-pressure, fully-integral rotors is to bore the center to remove metal impurities and permit internal inspection. The fully-integral, forged rotors will not be as susceptible to stress-corrosion cracking as the shrunk-on disks because of the elimination of interference fits which induce higher stresses and provide susceptible faying surfaces. However, Section 10.2.3.4 of the SSAR indicates that for high-pressure, fully-integral turbine rotors, the centers of the forgings are not bored out. Westinghouse was requested in the DSER to provide a technical justification for not boring the centers of the high pressure turbine rotors. This was Open Item 10.2.9-4. Westinghouse revised Section 10.2.3.4 of the SSAR (Revision 5) and explained that the non-bored design of the high-pressure rotors provides the necessary design margins because of inherently lower centerline stress. The use of solid rotor forgings was qualified by evaluation of the material removed from center bored rotors that were used in fossil power plants. This evaluation demonstrated that the material at the center of the rotors meets the requirements of the material specification. Further, supply of forgings for the high-pressure rotors is limited to suppliers that have been qualified based on bore materials performance. This is acceptable. Open Item 10.3.9-4 is closed.

The SSAR states that the design overspeed of the turbine will be 20 percent above rated speed, and the highest anticipated speed above rated speed with power/load balance and speed control systems operating is 111 percent of rated speed resulting from a loss of load. The SSAR also states that the combined peak stresses of a low-pressure, fully-integral turbine rotor at design over-speed due to centrifugal forces and thermal gradients will not exceed 0.65 of the minimum specified yield strength of the material. The integral rotor profiles are designed to limit surface stress in areas vulnerable to stress corrosion to 50 percent of yield stress. This will reduce the possibility of stress corrosion cracking.

In Section 10.2.3.2 of the SSAR, Westinghouse discusses in general terms maximum initial flaw size, crack growth rates, and so forth The staff evaluation of the application of NDE, initial flaw size, crack growth rates, and fracture toughness, in terms of addressing the probability aspects of turbine missile generation, is discussed in Section 3.5.1.3 of this report.

The preservice inspection (PSI) will include a 100 percent volumetric (ultrasonic) examination of each finished machined rotor and a surface visual examination using established acceptance criteria developed by the turbine manufacturer. Every subsurface sonic indication is either removed or evaluated to ensure that it will not grow in size, and thus compromise the integrity of the turbine during service. All finished machined surfaces are subjected to a magnetic particle examination with no flaw indications permissible in bores or other highly stressed areas. Each turbine rotor assembly is spin tested at 120 percent of rated speed.

Section 3.2.3.5 of the SSAR, "Preservice Tests and Inspections," states that finish-machined surfaces are subjected to a magnetic-particle examination. No magnetic-particle flaw indications are permissible in bores (if present) or other highly stressed regions. In the November 30, 1992, response to 250.28, Westinghouse states, "NDE inspection is performed on the holes after final machining of the rotor and prior to drilling and tapping operations." The

SRP acceptance criteria for PSI are that drilled holes should be subject to a surface examination. Drilling a hole is a machining operation, as is tapping threads in a hole. In the DSER the staff stated that it did not understand how NDE (surface examination) of a hole can be accomplished prior to drilling (and tapping) if finish-machined surfaces are to be subject to magnetic-particle examination. Holes are considered stress risers and they have been a source of fatigue cracking in turbines in the past. Accordingly, if surface examination of drilled and tapped holes is not performed, justification must be provided in the SSAR. This was Open Item 10.2.9-5. Revision 5 of the SSAR explained in Section 10.2.3.5 those rotor areas which require threaded holes are not subjected to magnetic particle examination because the number of threaded holes are minimized and the drilled holes are not located in high-stress areas. In addition, Westinghouse indicated that threaded holes are not unique to the AP600 plant and that Westinghouse-designed turbine rotors in operating nuclear power plants have no history of fatigue crack initiation due to duty cycles. Threaded holes in the turbine are located in areas where the stress is below the endurance limit of the rotor and do not affect the fatigue evaluation for the rotor and, therefore, are acceptable. Open Item 10.3.9-5 is closed.

The ISI for the AP600 turbine assembly includes the disassembly of the turbine and complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, low-pressure turbine blades, and low-pressure and high-pressure rotors. During plant shutdown, turbine inspection will be performed at approximately six year intervals.

The COL applicant referencing the AP600 design must submit to the NRC staff for review and approval, within three years of obtaining a combined license, a turbine maintenance program including probability calculations of turbine missile generation based on the NRC-approved methodology, or volumetrically inspect all low-pressure turbine rotors at the second refueling outage and every other refueling outage thereafter until a maintenance program is approved by the staff. This was COL Action Item 10.2.9-1 and Open Item 10.2.9-6. In its SSAR, revision 11, dated February 28, 1997, Westinghouse revised SSAR Section 10.2.6, Combined License Information. The revised section included a commitment to require that the COL applicant submit a turbine maintenance program to the NRC for review and approval within three years of obtaining a combined license. Therefore, COL Action Item 10.2.9-1 is acceptable and Open Item 10.2.9-6 is closed.

The actual inspection schedule will be based on probability calculations of turbine missile generation. The calculated probability of turbine missile generation is expected to be less than 1.0E-4 per year.

The COL applicant referencing the AP600 design must submit test data for the turbine rotors and the calculated toughness curves to the NRC staff for review and approval. This was COL Action Item 10.2.9-2 and Open Item 10.2.9-7. Revision 5 of the SSAR included a Section 10.2.6, Combined License Information. Section 10.2.6 includes a requirement that the COL will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis. Therefore, COL Action Item 10.2.9-2 is acceptable and Open Item 10.2.9-7 is closed.

The staff concludes that the integrity of the turbine rotor is acceptable and meets the relevant requirements of GDC 4 of Appendix A of 10 CFR Part 50. This conclusion is based upon the following:

Westinghouse has met the requirements of GDC 4 of Appendix A of 10 CFR Part 50 with respect to the use of materials with acceptable fracture toughness and elevated temperature properties, adequate design, and the requirements for preservice and ISIs. Westinghouse has also described its program for assuring the integrity of low-pressure turbine rotors, which include the use of suitable materials of adequate fracture toughness, conservative design practices, and preservice and ISIs. This provides reasonable assurance that the probability of failure with missile generation is low during normal operation, including transients up to design overspeed.

10.2.10 Conclusion

Based on the above evaluation, the staff concludes that the design is acceptable and meets the requirements of GDC 4 with respect to the protection of structures, systems, and components important to safety from the effects of turbine missiles. Westinghouse has met these requirements by providing a turbine overspeed protection system to control the turbine action under all operation conditions and which assures that a full-load turbine trip will not cause the turbine to overspeed beyond acceptable limits resulting in turbine missiles.

10.3 Main Steam Supply System

The staff reviewed the design of the main steam supply system (MSSS) in accordance with Section 10.3 of the SRP. Acceptability of the design of the MSSS is based on meeting the

- GDC 4, with respect to safety-related portions of the system being capable of withstanding the effects of external missiles and internally-generated missiles, pipe whip, and jet impingement forces associated with pipe breaks

- GDC 5, as related to the capability of shared systems and components important to safety to perform required safety functions

- GDC 34, as related to the system function of transferring residual and sensible heat from the reactor system in indirect cycle plants, and the following

    - The system is designed in accordance with the positions in Branch Technical Position RSB 5-1 as related to the design requirements for residual heat removal.

    - The system is designed to NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum From Director, NRR, to NRR Staff," Issue Number 1, which specifies allowable credit being taken for valves downstream of the main steam isolation valve (MSIV) to limit blowdown of a second steam generator in the event of a steamline break upstream of the MSIV.

DSER Open Item 10.3-1 identified RAIs 410.145, 410.253, 410.255, and 410.257-410.260 to be resolved. Westinghouse provided responses and additional information in a meeting with the staff on February 22-23, 1995, that has allowed the staff to continue its review as discussed below. Therefore, DSER Open Item 10.3-1 is closed.

The MSSS includes components of the AP600 steam generator system (SGS), main steam system (MSS), and main turbine system (MTS). The function of the MSSS is to transport steam from the steam generators to the high-pressure turbine over the entire operating range. The system provides steam to the moisture separator/reheater and the steam seal system for the main turbine. The system removes heat generated by the NSSS by means of a steam dump to the condenser through the turbine bypass system or to the atmosphere through power-operated atmospheric relief valves or spring-loaded main steam safety valves, when either the turbine generator or condenser is unavailable.

SSAR Section 10.3.1.1 and Table 3.2-3 steam generator system (SGS) identify all the safety-related mechanical equipment in the MSSS and lists the associated ASME code class. The following MSSS components are classified as safety-related:

- the main steamline piping from the steam generator up to the pipe restraint located on the wall between the auxiliary building and the turbine building, including main steam isolation valve and main steam isolation bypass valves

- the inlet piping from the main steamline up to the main steam safety valve discharge piping and vent stacks, and to power-operated relief line piping, including block valve and power-operated relief valves

- the instrumentation tubing up to, and including, the main steamline pressure instrument root valves

- the vent line and nitrogen connection on the main steamline up to, and including, the first isolation valve

- the main steam drain condensate pot located upstream of the main steam isolation valves, and the drain piping up to, and including, the first isolation valve

- the condensate drain piping from the outlet of the isolation valve to the restraint on the wall between the auxiliary building and the turbine building

The remainder of the MSSS is non-safety-related.

The safety-related portion of the MSSS complies with the QA requirements of Appendix B to 10 CFR Part 50 and is designed to environmental design, and fire protection. No single failure coincident with loss of offsite power compromises the MSSS's safety functions.

Provision III.5.f of SRP Section 10.3 states that in a postulated safe-shutdown earthquake, the design includes the capability to operate atmospheric dump valves remotely from the control room so that cold shutdown can be achieved using only safety-grade components, assuming a concurrent loss of offsite power. In AP600 design, the passive RHR system (see

Section 5.4.14 of the SER), which can be initiated automatically without requiring the control of steamline pressure, provides safety-grade decay heat removal capability. The power-operated atmospheric relief valves provide a non-safety-related means for plant cooldown to the point that the normal residual heat removal system can be initiated to remove the decay heat. The relief valves are automatically controlled by steamline pressure, with remote manual adjustment of the pressure setpoint from the control room. If the relief valve for an individual main steamline is unavailable because of the loss of its control or power supply, the respective spring-loaded safety valves, which are safety-related, will provide overpressure protection. The safety valves are designed to AP600 Class B, ASME Code, Section III, Class 2, and seismic Category I. Therefore, the staff concludes that the position in Branch Technical Position RSB 5-1 as related to the design requirements for residual heat removal is met.

Following a main steamline break, the main steam isolation is designed to limit blowdown to one steam generator so that the fuel design limits and containment design pressure can be maintained. The MSIV and MSIV bypass valves on each main steamline are designed to isolate the secondary side of the steam generators to prevent the uncontrolled blowdown of more than one steam generator and isolate non-safety-related portions of the system. The MSIV automatically closes upon receipt of either of two main steam isolation signals associated with independent Class 1E electrical divisions. Redundant power supplies and power divisions operate the main steam isolation valves (MSIVs) and MSIV bypass valves. The isolation valve is a part of the containment isolation boundary and therefore is specified as Class 1E, active, ASME Code, Section III, Safety Class 2. The conditions that initiate automatic closure of the MSIVs and MSIV bypass valves are high containment pressure, low steamline pressure, high steamline pressure negative rate, and low reactor coolant inlet temperature. The MSIVs are gate valves controlled by a pneumatic/hydraulic operator. The energy required to close the valves is stored in the form of compressed nitrogen in one end of the actuator cylinder. The valves are maintained open by high-pressure hydraulic fluid. For emergency closure, redundant Class 1E solenoids are energized, causing the high-pressure hydraulic fluid to be dumped to a fluid reservoir and the valves to close. The backup isolation valves (such as the turbine stop valves) receive signals derived from the protection and safety monitoring system (PMS) to actuate the valves.

In Section 3.6.1.1 of the SSAR, Westinghouse states that turbine stop valves, moisture separator/reheater stop valves, and turbine bypass valves (which are not safety-related) are credited in single-failure analyses to mitigate postulated steamline ruptures. These valves are included as non-safety-related equipment, and are evaluated for pipe whip protection as part of the evaluation of the affected system as required by GDC 4. Based on the design alternatives identified in Issue No. 1 of NUREG-0138, relative to utilizing the turbine stop valves to provide redundancy for safety-related equipment, the turbine stop valves and control valves are credited for demonstrating that the design will preclude the blowdown of more than one steam generator, assuming a concurrent single active failure. The staff concluded in NUREG-0138 that in accidents involving spontaneous failures of secondary system piping, reliance on non-safety-grade valves in the postulated accident evaluation is permitted based on the reliability of these valves. In addition, Westinghouse proposed in a letter dated August 29, 1997, to add technical specification control for the turbine stop valves, moisture separator reheated stop valves, and turbine bypass valves as a part of MSIV technical specification. The staff agrees with Westinghouse, and believes that including these valves in the technical specification will provide adequate assurance for the valve reliability. Based on the conclusion in NUREG-0138, the staff finds that the AP600 MSSS meets the requirements of GDC 34 as related to limiting blowdown of a second steam generator in the event of a steamline break upstream of the MSIV.

Based on meeting the relevant acceptance criteria specified in the SRP, the staff concludes that the MSSS meets the requirements of GDC 34 as related to the system function of transferring residual and sensible heat from the reactor system.

Compliance with GDC 2 is based on meeting the relevant acceptance criteria specified in the SRP that the safety-related portions of the system are capable of withstanding the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods, and meet the positions of Regulatory Guide 1.29, as related to the seismic design classification of system components, and Regulatory Guide 1.117, as related to the protection of structures, systems, and components important to safety from the effects of tornado missiles.

AP600 piping and valves from the steam generators up to and including each MSIV, are designed in accordance with ASME Code, Section III, Class 2, and seismic Category I requirements, as are branch lines up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic/remote manual closure. Piping and valves downstream of the MSIVs and the valves identified above are designed in accordance with ASME Code, Section III, Class 3, and seismic Category, I up to and including pipe anchors located at the auxiliary building wall. The power supplies and controls necessary for safety-related functions of the MSSS are Class 1E.

In Sections 10.3.1.1 and 10.3.3 of the SSAR, Westinghouse stated that the safety-related portion of the system is designed to withstand the effects of a safe shutdown earthquake, is protected from the effects of natural phenomena, and is capable of performing its intended function following postulated events. The safety-related portion of the MSSS is located in the containment and auxiliary buildings, which are designed to withstand the effects of earthquakes, tornados, hurricanes, floods, external missiles, and other appropriate natural phenomena. The components of the safety-related MSSS are qualified to function in normal, test, and accident environmental conditions. The staff evaluation of flood protection is described in Section 3.4.1 of the SER. The safety-related mechanical equipment in the main steam supply system is identified in Table 3.2-3 and described in Section 10.3.1.1 of the SSAR. Based on the review, the staff concludes that the safety-related portion of the system meets the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A to 10 CFR Part 50 with respect to the ability of structures housing the safety-related portion of the system and the safety-related portions of the system being capable of withstanding the effects of natural phenomena.

Compliance with GDC 4 is based on meeting the relevant requirements specified in the SRP that the safety-related portions of the system are capable of withstanding the effects of external missiles, internally-generated missiles, pipe whip, and jet impingement forces associated with pipe breaks and Position C.1 of Regulatory Guide (RG) 1.115, "Protection Against Low-Trajectory Turbine Missiles," dated July 1977 (as related to the protection of SSCs important to safety from the effects of turbine missiles). In addition, the system design should adequately consider steam hammer and relief valve discharge loads to assure that system safety functions can be achieved and should assure that operating and maintenance procedures include adequate precautions to avoid steam hammer and relief valve discharge loads. The system design should also include protection against water entrainment.

In a response to RAI 410.147, Westinghouse stated that steam hammer prevention is addressed by appropriate precautions in the operating and maintenance procedures, which include system operating procedures that caution against using the MSIVs except when necessary, and operating and maintenance procedures that emphasize proper draining. Westinghouse also stated that the stress analyses for the safety-related portion of the MSSS piping and components include the dynamic loads from rapid valve actuation of the MSIVs and the safety valves. Design features that prevent water formations in the MSSS include the use of drain pots and the proper sloping of lines. Westinghouse included the above information in Section 3B.2.3 of Appendix 3B to the SSAR.

A discussion of high-energy pipe break locations and evaluation of effects is provided in Sections 3.6.1 and 3.6.2 of the SSAR, including pipe whip and jet impingement forces associated with pipe breaks. Section 10.3.2.2.1 of the SSAR states that the main steamlines between the steam generator and the containment penetration are designed to leak-before-break (LBB). A discussion of the LBB application and criteria is presented in Section 3.6.3 of the SSAR. The staff evaluation is in Sections 3.6.1 through 3.6.3 of the SER. In RAI 410.145, the staff stated that to apply LBB to the main steamlines the main steamline leak detection system should be included in the plant technical specifications. In response to RAI 410.145, Westinghouse committed to add secondary side leakage detection in the technical specifications. The staff would determine its acceptability when the technical specifications were available. This was identified as DSER open item. In a letter, dated July 26, 1996, Westinghouse rescinded its commitment of providing steamline leakage detection technical specification. Instead, Westinghouse proposed to use administrative procedures to control main steamline leakage. The staff disagreed with Westinghouse on its approach for dealing with the main steamline leakage for LBB. After discussion with the staff, Westinghouse incorporated into Revision 16 of SSAR Section 16.1, Technical Specification, TS 3.7.8, "Main Steamline Leakage." The staff has reviewed this TS and finds it acceptable.

An evaluation of the protection against externally and internally-generated missiles is provided in Section 3.5 of the SER. Specifically, the staff evaluation on the compliance with Position C.1 of RG 1.115 is in Sections 3.5.1 and 3.5.2 of the SER. The staff's finding on the conformance of GDC 4 is discussed in Sections 3.5 and 3.6 of the SER.

Although the AP600 design can be used at either single-unit or multiple-unit sites, Westinghouse stated in Section 3.1.1 of the SSAR that the AP600 design is a single-unit plant and if more than one unit were built on the same site, none of the safety-related systems would be shared. Should a multiple-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the capability of shared SSCs important to safety to perform their required safety functions.

As described above, the staff has reviewed the MSSS in accordance with Section 10.3 of the SRP and finds that the system design conforms with the Commission regulations given in GDC 2, 4, 5, and 34. Therefore, the design of MSSS is acceptable.

10.3.1 Steam and Feedwater System Materials

GDC 1 requires that systems important to safety shall be designed to quality standards commensurate with the importance to safety of the functions to be performed. GDC 35 requires suitable interconnection, leak detection, isolation, and contaminant capabilities be provided to assure that the safety system function (i.e., emergency core cooling) can be accomplished, assuming a single failure. 10 CFR Part 50.55a requires that SSCs are designed, fabricated, erected, constructed, tested and inspected in accordance with the requirements of the applicable codes and standards commensurate with the importance of the safety function to be performed. 10 CFR Part 50, Appendix B establishes QA requirements for the design, construction, and operation of SSCs that are important to safety.

The staff reviewed the steam and feedwater system materials in accordance with Section 10.3.6 of the SRP. Section 10.3.6 of the SRP provides that steam and feedwater system materials are acceptable if they satisfy the requirements of 10 CFR Part 50.55a, Appendix B and GDC 1 and 35.

Section 10.3.6 of the AP600 SSAR provides information concerning steam and feedwater materials. The staff reviewed this information and requested additional information from Westinghouse in order to evaluate the adequacy of the steam and feedwater materials. In the DSER, the staff noted that Westinghouse responded to the staff's requests for additional information, however, several of the responses were judged to be inadequate.

Specifically, the December 22, 1992, response to RAI 252.137 was inadequate because the response did not confirm that steam and feedwater materials will meet the requirements of Section III of the ASME Code. Westinghouse was requested to provide information to confirm that all of the materials selected for the steam and feedwater systems (piping and fittings) conform to Section III of the ASME Code. It was also requested that the SSAR be revised to identify the materials used, by specification, type, grade, and heat treatment, and the applicable ASME Code Cases used for steam and feedwater system materials. This was Open Item 10.3.1-1. Revision 7 of the SSAR included Table 10.3.2-3. Table 10.3.2-3 lists the materials for feedwater and steam service. The materials for the ASME Code designed portions of the main steamline and main feedwater lines were identified as being ASME SA-106 Grade B and ASME SA-335 Grade P-11. For the non-ASME Code designed portion of the steam and feedwater lines the materials were identified as being ASTM A-106 Grade B and ASTM A-335 Grade P-11. This is acceptable. Open Item 10.3.1-1 is closed.

In the December 22, 1992, response to RAI 252.137, Westinghouse identified the steam and feedwater system pipe materials in the SSAR. There was a typographical error, in that ASTM A-355, "Standard Specification for Alloy Steel Bars for Nitriding," was specified instead of SA-335, which covers alloy steel pipe. This was Confirmatory Item 10.3.1-1. Westinghouse revised its SSAR Table 6.1-1 which lists SA-335 as being the correct piping material. This closes Confirmatory Item 10.3.1-1.

Some of the materials for the main steam and feedwater systems that were identified by specification in the SSAR, and were classified as ASME Code Class 2 materials do not satisfy the requirements of Section III of the ASME Code and parts A, B and C of Section II of the Code, because some ASTM specifications were also listed. The ASME Code requires that only

ASME code approved materials be used for Class 2 components. This was Open Item 10.3.1-2. Westinghouse revised SSAR Section 6.1.1-3 to clarify that materials for nonpressure-retaining portions of fluid systems may be procured under ASTM specifications. This clarification is consistent with Code requirements and is acceptable to the staff. Therefore, Open Item 10.3.1-2 is closed.

In the December 22, 1992, response to RAI 252.137, Westinghouse stated that, "Material selection could be affected by ongoing piping analyses and leak before break qualification for these lines." In the DSER the staff found this response unsatisfactory, because the staff could not reach a safety finding on the acceptability of the materials until the SSAR has a specific commitment as to the materials selected for the steam and feedwater systems. This was Open Item 10.3.1-3. As previously stated, Westinghouse identified the material for the main steamline and the feedwater lines will be SA-106 Grade B and alloy steel SA-335 Grade P11. In addition Westinghouse revised SSAR Section 10.3.2.2.1 to state that the main steamlines between the steam generator and the containment penetration are designed to meet the leak before break criteria described in subsection 3.6.3. This closes Open Item 10.3.1-3.

In the January 22, 1993, and November 30, 1992, responses to RAIs 251.139 and 251.140 respectively, Westinghouse addressed some of the important engineering aspects of nuclear power plants. The discussion on corrosion and erosion-corrosion (E/C) of feedwater materials is especially of concern to present day operating plants. The staff requested that the information be incorporated into the SSAR. In addition, measures used to control and minimize the effects of E/C over the 60-year design life should be addressed in the SSAR. This was Open Item 10.3.1-4. Westinghouse revised SSAR Section 10.3.2.2.1 which commits that main steam piping is designed to consider the effects of E/C. Piping containing dry, single phase steam is constructed of carbon steel, and piping exposed to wet two-phase steam is constructed of E/C resistant low alloy steel or carbon steel with a stainless steel inner liner. Piping low points with low point drains are provided for collecting and draining moisture and to help reduce the potential for water carryover to the high and low pressure turbines. Pipe wall thickness inspections will be performed by the COL applicant to monitor wall E/C rates. Therefore, the staff finds this item, COL action item 10.3-1, acceptable and Open Item 10.3.1-4 closed.

Carbon steel piping in steam and feedwater systems has experienced wall thinning due to the single-phase or two-phase (water) erosion-corrosion (E/C) phenomenon as documented in Generic Letter (GL) 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning." As required by GL 89-08, the COL applicant is responsible for inspecting for pipe wall thinning due to E/C. However, the most effective way of reducing or eliminating this pipe wall thinning is through design. In the DSER, the staff requested that Westinghouse discuss its design approaches to reduce the potential for E/C of steel piping (single-phase and two-phase) and identify measures to ensure that inspections will be possible and meaningful. This was Open Item 10.3.1-5. Westinghouse revised (Revision 5) its SSAR, Section 10.1.2 to include detailed discussion concerning E/C protection. Specifically, the following controls will be incorporated in the AP600 design. Erosion- corrosion resistant materials, such as ASME SA-335 grade P11 (1.5 Cr - 0.5 Mo) will be used in the steam and power conversion system for components exposed to single-phase or two-phase flow where significant erosion can occur. Factors considered in the evaluation of E/C include system piping and component configuration and geometry, water chemistry, piping and component material, fluid temperature, and fluid velocity. Carbon steel with only carbon and manganese alloying agents will not be used for applications subject to

E/C. In addition to material selection, pipe size and layout will be also used to minimize the potential for E/C in systems containing water or two-phase flow. Steam and power conversion systems are designed to facilitate inspection and E/C monitoring programs. An industry-sponsored computer program developed for nuclear and fossil power plant applications is used to evaluate the rate of wall thinning for components and piping potentially susceptible to E/C. In addition, the COL applicant is required to prepare an E/C monitoring program. This program will address industry guidelines and the requirements included in GL 89-08. The staff finds these controls acceptable. Open Item 10.3.1-5 is closed. This is part of COL action item 10.3-1.

Copper is known to have deleterious effect on materials in steam and feedwater systems. The staff requested Westinghouse to identify in the SSAR whether any copper or copper alloys are used in the steam and feedwater systems, including the condenser, or alternatively commit to meet the guidelines of EPRI report NP-2294, "Guide to the Design of Secondary Systems and their Components to Minimize Oxygen Induced Corrosion." This was Open Item 10.3.1-6. Westinghouse revised SSAR Section 10.3.6.2, which requires that no copper or copper-bearing materials are to be used in the steam and feedwater system. This is acceptable. Therefore, Open Item 10.3.1-6 is closed.

The staff concludes that the main steam and feedwater system materials are acceptable and meet the relevant requirements of 10 CFR 50.55a, GDC 1 and 35, and Appendix B to 10 CFR Part 50. The following provides the basis for this conclusion:

- In the SSAR, Westinghouse commits that the fracture toughness properties of the materials will meet the requirements of NC-2300, "Fracture Toughness Requirements for Materials," of Section III of the ASME Code. The fracture toughness and mechanical property tests required by the Code provide reasonable assurance that ferritic materials will have adequate margins against the possibility of nonductile behavior or rapidly propagating fracture. This satisfies in part 10 CFR Part 50.55a and GDC 1 and 35.

- Welders welding on pressure retaining materials will be qualified in accordance with the requirements of the ASME Code, Section IX. Welders welding on component supports and other non-pressure-retaining materials will be qualified in accordance with the requirements of the governing construction code. To address the area of welding in areas of limited accessibility, Westinghouse proposed an acceptable alternative to the recommendations of RG 1.71, which provides guidance concerning qualification of welders for welding involving limited accessibility. The proposed alternative is, for shop welds, the welder's position will be controlled and closely supervised. This is acceptable because joints of limited accessibility are repetitive due to multiple production of similar components. For field welds, the qualifications of the welder will be considered on a case-by-case basis because of the great variety of circumstances encountered. These joints (field and shop) are also subject to the inspections required by the applicable construction code. These precautions should provide adequate assurance of the acceptability of joints welded under conditions of limited accessibility. On this basis, Westinghouse provided an acceptable alternative to the recommendations in RG 1.71.

- The AP600 design will also comply with RG 1.85, which describes acceptable Code Cases that may be used in conjunction with the ASME Code, Section III. Westinghouse takes exception to RG 1.37. The exception relates to the fact that the ANSI N45.2 series of standards have been replaced by ASME NQA-1 and NQA-2. ANSI Standard N45.2.1, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," which is referenced in RG 1.37, has been incorporated into NQA-2, Part 2.1. The technical requirements specified in ANSI 45.2.1 and NQA-2, Part 2.1 are compatible. Westinghouse considers compliance with NQA-2, Part 2.1 to be equivalent to satisfying RG 1.37. This is acceptable to the staff because the staff also considers compliance with NQA-1 and NQA-2 to be an acceptable alternative to the regulatory guidance of RG 1.37.

10.4 Other Features

10.4.1 Main Condenser

The staff reviewed the design of the main condenser in accordance with Section 10.4.1 of the SRP. The acceptability of the system design is contingent on meeting the requirements of GDC 60 as it relates to system design such that failures do not result in excessive releases of radioactivity to the environment, do not cause unacceptable condensate quality, and do not flood areas housing safety-related equipment.

DSER Open Item 10.4.1-1 identified RAI 410.255 concerning condenser tube leakage to be resolved. Westinghouse provided responses and additional information in a meeting with the staff on February 22-23, 1995, that allowed the staff to continue its review as discussed below.

The AP600 design main condenser system is described in SSAR Section 10.4.1 and shown in Figure 10.4.7-1 of the SSAR. Design parameters of the condenser (such as heat transfer capability, surface area, design operating pressure, shell side pressure, circulating water flow, tube side inlet temperature, tube-side temperature rise, condenser outlet temperature, condenser tube material, and so forth) were listed in Table 10.4.1-1, "Main Condenser Design Data," of the SSAR. The table was referenced in DSER Section 10.4.1, and was found acceptable by the staff. However, while preparing the draft FSER, the staff found that SSAR Table 10.4.1-1 was removed by Westinghouse. The staff indicated to Westinghouse that the design parameter information in SSAR Table 10.4.1-1 is necessary for the staff to complete its review of the condenser system. As a result, Westinghouse restored Table 10.4.1-1 in SSAR Revision 19. The staff finds it acceptable.

The main condenser system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine system. When the system functions as the steam cycle heat sink, it receives and condenses exhaust steam from the main turbine and the turbine bypass system. The main condenser is designed to receive and condense the full-load main steam flow exhausted from the main turbine and serves as a collection point for vents and drains from various components of the steam cycle system. Upon actuation of the turbine bypass system, the main condenser is designed to receive and condense steam bypass flows of up to 40 percent of plant full load steam flow without either reaching the condenser over-pressure turbine trip setpoint or exceeding the allowable exhaust temperature. In the event of high condenser pressure or a trip of both circulating water pumps, the turbine bypass valves are prohibited from opening. If the main condenser is unavailable to receive this flow, the steam is discharged to the atmosphere through the main steam power-operated relief valves or spring-loaded safety valves.

The main condenser is a non-safety-related and non-seismic component located in the turbine building. The failure of the main condenser and the resultant flooding will not preclude operation of any essential system, because no safety-related equipment is located in the turbine building, and the water cannot reach safety-related equipment located in Category 1 plant structures.

The main condenser has no significant inventory of radioactive contaminants during normal operation and plant shutdown. Radioactive contaminants can be obtained through primary-to-secondary system leakage resulting from steam generator tube leaks. In the letter of July 15, 1994, response to RAI 410.254 regarding the concern of meeting GDC 60, "Control of Releases of Radioactive Materials to the Environment," Westinghouse stated that early detection of concentrated levels of radioactivity is provided by the MSSS and steam generator blowdown system (BDS) radiation devices. In addition to this monitoring, radioactive effluent monitoring equipment is provided in the turbine island vents, drains, and relief system (TDS) at the combined exhaust of the condenser air removal system (CMS) and the turbine gland seal system (GSS). The plant operator may secure the discharge of the radioactive effluent upon detection of a high radioactivity level. Although the design has radioactivity monitors in the system to detect leakage into and out of the main condenser during normal operation, startup, and shutdown, the main condenser has no radioactive contaminants inventory. Because the above systems continuously monitor and detect the radioactivity leakage into and out of the condenser, GDC 60 is met with respect to failures in the system design that could result in excessive releases of radioactivity to the environment. The radiological monitoring capabilities are discussed in Section 11.5 of the SER.

The main condenser is not subject to ISI testing. The condenser water boxes are hydrostatically tested after erection. Condenser shells are tested by the fluorescent tracer method in accordance with ASME Power Test Code 19.11. Tube joints are leak tested during construction and prior to startup.

The system is provided with the following instrumentation and control features that determine and verify the proper operation of the main condenser:

- the main condenser hotwell level control devices
- control room indicators and alarms of water levels in the condenser hotwell
- control room indicators and alarms of condenser pressure
- a turbine trip on high turbine exhaust pressure
- temperature indicators for monitoring condenser performance

In response to the staff's RAI 410.255 (dated August 8, 1995) concerning leakage and condensate quality, Westinghouse stated that leakage at the connections of the tubes to the tube sheets can be detected at either end of each tube bundle by the collection troughs and conductivity cells. These conductivity measurements are indicated and alarmed. This information helps to identify the leaking tube bundle. The steps that may be taken are, first isolate the circulating water system from the affected water box while at reduced plant power, then drain the water box and finally repair or plug the affected tubes. Leakage occurring in tube locations other than at the tube ends is detected and alarmed by monitoring the condensate leaving the hotwell.

The condensate polishing system removes corrosion products and ionic impurities from the condensate system. This allows for continued operation with a "continuous" condenser tube leakage of 0.001 gpm or a "faulted" leak of 0.1 gpm until repairs can be made or until an orderly shutdown is achieved. The length of time of operation for maintaining the required condensate quality is determined by the value of the control parameters in Table 10.3.5-1 of the SSAR. The table provides the recommended corrective measures in three action levels according to the value of control parameters during an in-leakage condition. Therefore, DSER Open Item 10.4.1-1 is closed.

As discussed above, the staff reviewed the design of the main condenser in accordance with Section 10.4.1 of the SRP. On the basis of this review, the staff concludes that the main condenser system is acceptable and meets the requirements of GDC 60 with respect to the system design such that failures do not result in excessive releases of radioactivity to the environment. Westinghouse meets this requirement by providing radioactive monitors in the system to detect leakage into and out of the main condenser.

10.4.2 Main Condenser Evacuation System

Main condenser evacuation is performed by the condenser air removal system (CMS). The staff reviewed the design of the CMS in accordance with Section 10.4.2 of the SRP. Acceptability of the design of the CMS is based on meeting the following general design criteria as described in the SRP:

- GDC 60, as it relates to the CMS design for the control of releases of radioactive materials to the environment

- GDC 64, as it relates to the CMS design for the monitoring of releases of radioactive materials to the environment

The requirements of the two Commission regulations are met by using the regulatory positions contained in the following RGs and industrial standards:

- RG 1.26 as it relates to the GMS quality group classification that may contain radioactive materials but is not part of the reactor coolant pressure boundary and is not important to safety

- RGs 1.33 and 1.123, as they relate to the QA programs for the GSS components that may contain radioactive materials

- "Standards for Steam Surface Condensers," 6th Ed., 1979, Heat Exchanger Institute, as it relates to the CMS components that may contain radioactive materials

The CMS is a non-safety-related component located in the turbine building. All piping is designed to ANSI B31.1. Using liquid ring vacuum pumps, the system establishes and maintains a vacuum in the condenser during startup and normal operation. It also removes non-condensable gases and air from the main condenser during plant startup, cooldown, and normal operation from the two condenser shells and exhausts them into the atmosphere.

Westinghouse indicated in WCAP-13054, "AP600 Compliance with SRP Acceptance Criteria," that the CMS will be in conformance with 8th edition of "Standards for Steam Surface Condensers, Heat Exchanger Institute." In Section 10.4.2.4 of the SSAR, Westinghouse stated that a performance test will be conducted on each pump in accordance with the "Heat Exchanger Institute Performance Standard for Liquid Ring Vacuum Pumps." The staff finds Westinghouse commitment acceptable.

WCAP-13054 stated that RG 1.33 is not applicable, and that RG 1.123 has been withdrawn. RG 1.33, "Quality Assurance Program Requirements (Operation)," applies only to the operational phase of nuclear power plants. Therefore, the staff will review COL applications to ensure their conformance with RG 1.33. A COL applicant referencing the AP600 certified design should demonstrate compliance with RG 1.33. Westinghouse includes this COL action as a part of the overall plant QA program for operation, which is discussed in Section 17.4 of the SSAR. This approach in dealing with QA for operation is similar to the approach taken for QA in the radwaste systems (see Section 11.2 and 11.3 of the SER) because radioactive contaminants can be introduced in the GSS through primary-to-secondary system leakage resulting from steam generator tube leakage. The staff agrees with Westinghouse that RG 1.123 has been withdrawn and is not applicable to the AP600 CMS.

Provisions 3 and 5 of the specific acceptance criteria in the SRP recommends a discussion on the potential for explosive mixtures and provide specific guidance for the system if the potential exists. In the July 8, 1994, response to RAI 410.256, Westinghouse stated that the potential for explosive mixtures within the CMS does not exist. This is stated in Section 10.4.2.2.1 of the SSAR.

In the July 15, 1994, response to RAI 410.254 regarding compliance with GDC 60, Westinghouse stated that early detection of concentrated levels of radioactivity is provided by the MSSS and steam generator BDS radiation devices. In addition to this monitoring, radioactive effluent monitoring equipment is provided in the TDS at the combined exhaust of the CMS and the GSS. The plant operator may secure the discharge of the radioactive effluent upon detection of high radioactivity level. Although the design has radioactivity monitors in the system to detect leakage into and out of the main condenser during normal operation, startup, and shutdown, the main condenser has no radioactive contaminants inventory. Radioactive contaminants can only be obtained through primary-to-secondary system leakage resulting from steam generator tube leaks. Because the above systems continuously monitor and detect the radioactivity leakage into and out of the condenser, GDC 60 and GDC 64 are met with respect to the control and monitoring radioactivity releases to the environment. The radiological monitoring capabilities are discussed in Section 11.5 of the SER.

In RAI 410.257, the staff raised a question on the compliance of RG 1.26 as related to CMS quality group classification. In response, Westinghouse referred to SSAR Sections 3.2.2.6 and 3.2.2.7, which indicated that the CMS is Class E according to AP600 classification, which treats systems and components that have potential to be contaminated with radioactive fluids but do not normally contain radioactive fluids as Class E. The staff reviewed SSAR Section 3.2.2.6 and 3.2.2.7 and did not find sufficient justification for Westinghouse's position of using Class E for CMS. The referenced SSAR sections stated Westinghouse's classification position, but did not identify it as a deviation from RG 1.26 nor provide any justification for its deviation. This was identified as DSER Open Item 10.4.2-1. In a meeting on February 22-23, 1995, after discussions with the staff, Westinghouse agreed this was a deviation from RG 1.26. However, Westinghouse decided to maintain its position of using Class E for CMS. Subsequently, the staff found Westinghouse's position inconsistent with RG 1.26 and Section 10.4.2 of the SRP. Specifically, RG 1.26 Position C.3 states that Quality Group D should be applied to components and systems that contain or may contain radioactive material but are not part of the reactor coolant system (RCS) or included in quality Groups B or C. Radioactive contaminants in CMS can be introduced through primary-to-secondary system leakage resulting from steam generator tube leakage, although the probability and consequences are low. RG 1.26 does not have a provision to allow low probability, low consequence radioactive contaminated systems to be exempted from Quality Group D. Westinghouse could not provide AP600 design-specific differences that would justify its position. In Revision 13 of the SSAR, Table 3.2-3, Westinghouse revised the CMS to Class D. Therefore, DSER Open Item 10.4.2-1 is closed.

As discussed above, the staff reviewed the design of the CMS in accordance with Section 10.4.2 of the SRP, and finds the system conforms with the Commission regulation given in GDC 60 and GDC 64 and is therefore acceptable.

10.4.3 Gland Seal System

The staff reviewed the design of the gland seal system (GSS) in accordance with Section 10.4.3 of the SRP, "Turbine Gland Sealing System." Acceptability of the design of the GSS is based on meeting the following general design criteria as described in the SRP:

- GDC 60 as it relates to the GSS design for the control of releases of radioactive materials to the environment

- GDC 64 as it relates to the GSS design for the monitoring of releases of radioactive materials to the environment

The requirements of the two Commission regulations are met by using the regulatory positions contained in the following RGs:

- RG 1.26 as it relates to the GSS quality group classification that may contain radioactive materials, but is not part of the reactor coolant pressure boundary and is not important to safety

- RGs 1.33 and 1.123 as they relate to the QA programs for the GSS components that may contain radioactive materials

The GSS is a non-safety-related system designed to prevent air leakage into and steam leakage out of the casings of the turbine generator. The system returns condensed steam to the condenser and exhausts non-condensable gases into the atmosphere. The system is designed to detect the presence of radioactive contamination in the gas exhaust. The system consists of a steam supply header, steam drains/non-condensable gas exhaust header, two motor-driven gland steam condenser blowers, gland seal condenser, vent and drain lines, and associated piping, valves, and controls. The GSS serves no safety-related function.

During the initial startup phase of turbine generator operation, steam is supplied to the GSS from the auxiliary steam header supplied from the auxiliary boiler. At times other than initial startup, GSS steam is supplied from either the auxiliary steam system or from main steam. The system is tested in accordance with written procedures during the initial testing and operation program. In the July 22, 1994, response to RAI 410.149, Westinghouse stated that the testing procedures for the system are provided by the turbine vendor in equipment instruction manuals. During normal operation, the satisfactory operation of the system components will be demonstrated by monitoring essential parameters. Pressure and temperature indication with alarms are provided for monitoring the operation of the system. A pressure controller is provided to maintain steam-seal header pressure by providing signals to the steam-seal feed valve. The gland seal condenser is monitored for shell side pressure and internal liquid level. A radiation detector with an alarm is provided in the TDS.

WCAP-13054 stated that RG 1.33 is not applicable, and that RG 1.123 has been withdrawn. RG 1.33 applies only to the operational phase of nuclear power plants. Therefore, the staff will review COL applications to ensure their conformance with RG 1.33. A COL applicant referencing the AP600 certified design should demonstrate compliance with RG 1.33. Westinghouse includes this COL Action as a part of the overall plant QA program for operation, which is discussed in Section 17.4 of the SSAR. This approach in dealing with QA for operation is similar to the approach taken for QA in the radwaste systems (see Section 11.2 and 11.3 of the SER) because radioactive contaminants can be introduced in the GSS through primary to secondary system leakage resulting from steam generator tube leakage. The staff agrees with Westinghouse that RG 1.123 has been withdrawn and is not applicable to the AP600 GSS.

The mixture of non-condensable gases discharged from the gland steam condenser blower is not normally radioactive; however, in the event of significant primary-to-secondary system leakage as a result of a steam generator tube leak, it is possible for the mixture discharged to be radioactively contaminated. The discharge line vents to the TDS, which contains a radiation monitor for the detection of radioactivity. Upon detection of unacceptable levels of radiation, operating procedures are implemented. The radiological monitoring capabilities are discussed in Section 11.5 of the SER. Because the above systems continuously monitor and detect the radioactivity, GDC 60 and GDC 64 are met with respect to the control and monitoring radioactivity releases to the environment.

In RAI 410.258, the staff raised a question on the compliance of RG 1.26 as related to GSS quality group classification. In response, Westinghouse, referred to SSAR Sections 3.2.2.6 and 3.2.2.7, which indicated that the GSS is Class E according to AP600 classification, which treats systems and components that have potential to be contaminated with radioactive fluids but do not normally contain radioactive fluids are considered as Class E. The staff reviewed SSAR Section 3.2.2.6 and 3.2.2.7 and did not find sufficient justification for Westinghouse's position of using Class E for GSS. This was identified as part of DSER Open Item 10.4.3-1. Subsequently, this issue was discussed and resolved with Westinghouse in the same manner as DSER Open Item 10.4.2-1 and evaluated in Section 10.4.2 of the SER. In Revision 13 of the SSAR, Table 3.2-3, Westinghouse revised the GSS to Class D. Therefore, the portion of DSER Open Item 10.4.3-1 relating to quality classification of the GSS is closed.

In RAI 410.259, the staff requested Westinghouse to provide a system flow diagram piping and instrument diagram (P&ID) for the GSS. This was identified as the second part of DSER Open Item 10.4.3-1. The staff explained to Westinghouse that simply because the system is non-safety-related, it is not an adequate justification for not having a P&ID in the SSAR. The importance of the GSS to AP600 is not much different from the importance of the GSS to other pressurized water reactors (PWR) plants. The staff performed its review according to SRP Section 10.4.3, which identifies the P&ID of the GSS in the review area, recognizing GSS being a non-safety-related system. In SSAR Revision 4, Westinghouse provided the requested diagram as Figure 10.4.12-1. The staff found it acceptable and resolved the open item regarding the P&ID for GSS. However, while preparing the draft FSER, the staff discovered that Figure 10.4.12-1 was removed by Westinghouse. The staff indicated to Westinghouse that the SSAR should have a P&ID for the GSS. As a result of the staff finding, Westinghouse restored the P&ID of GSS in SSAR Revision 18. The staff finds it acceptable. Therefore, DSER Open Item 10.4.3-1 relating to the P&ID is closed.

As discussed above, the staff reviewed the design of the GSS in accordance with Section 10.4.3 of the SRP. The system conforms with the Commission regulation given in GDC 60 and GDC 64, and is therefore acceptable.

10.4.4 Turbine Bypass System

The staff reviewed the design of the turbine bypass system in accordance with Section 10.4.4 of the SRP. The acceptability of the system design is based on meeting the requirements of GDC 4 and GDC 34 as described in the SRP:

- GDC 4, as it relates to the system being designed such that a failure of the system (due to a pipe break or system malfunction) does not adversely affect safety-related systems or components

- GDC 34, as it relates to the ability to use the turbine bypass system for shutting down the plant during normal operations by removing residual heat without using the turbine generator

DSER Open Item 10.4.4-1 identified RAI 410.264 to be resolved. RAI 410.264 indicates that the turbine bypass system should be identified in the P&ID. In response, Westinghouse revised Figure 10.3.2-2 in Revision 3 of the SSAR identifying "Turbine Bypass Banks A and B" as part of the main steam system in the figure to address the staff question in RAI 410.264. Therefore, DSER Open Item 10.4.4-1 is closed.

The turbine bypass system, which is also called the steam dump system, provides the capability to direct main steam from the steam generators bypassing the turbine to the main condenser in a controlled manner to dissipate heat and to minimize transient effects on the RCS during startup, hot shutdown, cooldown, and step-load reductions in generator loads.

The turbine bypass system consists of a manifold connected to the main steamlines upstream of the turbine stop valves and lines from the manifold with regulating valves to each condenser shell. The turbine bypass valves are globe valves and are electro-pneumatically operated. The bypass valves will fail to a closed position upon loss of air or electrical signal. A modulating position responds to the electrical signal from the control system and provides the appropriate air pressure to the valve actuator for modulating the valves open.

Solenoid valves located in the air line to each bypass valve actuator open and close the bypass valve and serve as protective interlocks for bypass valve actuation for tripping the valve open or closed. Two of the blocking solenoid valves for each turbine bypass valve are redundant and prevent bypass valve actuation upon low RCS average temperature (T_{avg}). This minimizes the possibility of excessive RCS cooldown. However, the low T_{avg} block can be manually bypassed for two of the bypass valves to allow operation during plant cooldown. Another blocking solenoid valve prevents actuation of the bypass valve when the condenser is not available.

The turbine bypass system has two modes of operation (1)T_{avg} control and (2)pressure control modes. A discussion of the system operation is in SSAR Section 10.4.4.3. The design basis of the turbine bypass system is to eliminate challenges to the main steam power-operated relief valves, main steam safety valves, and pressurizer safety valves during a reactor trip from 100 percent power or a 100 percent load rejection, or turbine trip from 100 percent power without a reactor trip. The turbine bypass system meets its power generation design basis with its ability to bypass 40 percent of the full load main steam flow to the main condenser. The system's total flow capacity, in combination with bypass valve response time, RCS design, and reactor control system response is sufficient to meet its design basis.

For load rejections greater than 10 percent but less than 50 percent, or a turbine trip from 50 percent power or less, the turbine bypass system operates with the NSSS control systems to meet the design-basis requirements for heat removal. For power changes less than or equal to a 10 percent change in electrical load, the turbine bypass system is not actuated. The total power change is handled by the power control, pressurizer level and pressure control, and the steam generator level control systems. Therefore, the staff concludes that the system is designed to enable sufficient steam to be bypassed to the main condenser so that the plant can be shutdown during normal operation without using the turbine generator. The system therefore meets GDC 34, "Residual Heat Removal," of Appendix A to 10 CFR Part 50 with respect to the ability to use the system for shutting down the plant during normal operations.

In Section 10.4.4.5 of the SSAR, Westinghouse stated that the turbine bypass valves will be tested for operability and the system will by hydrostatically tested to confirm leak tightness before the turbine bypass system is placed in service. The bypass valves may be tested while the unit is in operation. System piping and valves are accessible for inspection. The turbine bypass system does not require ISI and testing.

The failure of a turbine bypass high-energy line will not disable the turbine speed control system. The turbine speed control system is designed such that its failure will cause a turbine trip. If the bypass valves fail open, an additional heat load is placed on the condenser. If this load is great enough, the turbine is tripped on high condenser pressure. Ultimate over-pressure protection for the condenser is provided by turbine rupture discs. If the bypass valves fail closed, the power-operated relief valves permit a controlled cooldown of the reactor. Chapter 15 of the SSAR addresses credible single failures of the turbine bypass system on the NSSS.

The high-energy lines of the turbine bypass system are located in the turbine building, which is a non-seismic category building. No safety-related equipment is located within the turbine building or near the turbine bypass system. Therefore, the staff concludes that the system complies with the requirements of GDC 4 regarding adverse effects of a pipe break or malfunction on those components of the system necessary for shutdown or accident prevention or mitigation, as there are no such systems in the turbine building.

The turbine bypass system includes all components and piping from the branch connection at the main steam system to the main condensers. The scope of review of the turbine bypass system for the AP600 design included layout drawings, P&IDs, and descriptive information for the turbine bypass system and auxiliary supporting systems that are essential to its operation.

The basis for acceptance of the turbine bypass system is conformance to the design, design criteria, and design bases to the Commission's regulations as set forth in GDC 4 and 34 of Appendix A to 10 CFR Part 50 as follows:

- Westinghouse meets the requirements of GDC 4 with respect to the system being designed, such that a safe shutdown will not be prevented as a result of the turbine bypass system failure.

- Westinghouse meets the requirements of GDC 34 with respect to the ability to use the turbine bypass system for shutting down the plant during normal operations. The turbine bypass system is designed such that sufficient steam can be bypassed to the main condenser so that the plant can be shutdown during normal operations without using the turbine generator.

Based on the above, the staff concludes that the design of the turbine bypass system conforms to Section 10.4.4 of the SRP and meets the requirements of GDC 4 and GDC 34.

10.4.5 Circulating Water System

The staff reviewed the circulating water system (CWS) in accordance with Section 10.4.5, of the SRP. Acceptability of the system as described in the SSAR is based on meeting the requirements of GDC 4, as it relates to design provisions provided to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS. Compliance with GDC 4 is based on meeting the relevant acceptance criteria specified in the SRP, such as the following:

- means to prevent, detect, and control flooding of safety-related areas due to leakage from the CWS

- adverse effects of malfunction or failure of CWS piping on functional capabilities of the safety-related systems or components

- control of water chemistry, corrosion, and organic fouling in the CWS

- the effects of high- and moderate-energy line breaks on the CWS design under Section 3.6.1 of the SRP

The CWS is a non-safety-related system that is designed to provide a continuous cooling water supply to the main condenser, the turbine building closed cooling water system (TCS) heat exchangers, and the condenser vacuum pump seal water heat exchangers under all modes of power operation and design weather conditions. The system consists of two 50-percent-capacity circulating water (CW) pumps (mounted in an intake structure), one hyperbolic natural-draft cooling tower, and associated valves, piping, and instrumentation.

Although the SSAR stated that the CWS and the cooling tower are applicable to a broad range of sites, the SSAR did not provide sufficient information or alternative design requirements on the CWS design. Specifically, the design basis should address the need for the safety-related equipment to be protected in the event of the CWS failure, and for the cooling tower to be located far enough from safety-related structures to prevent damage in the event of a cooling tower failure. Because the heat sinks for the CWS are site dependent, conceptual design and interface requirements should be provided for the normal heat sink and, in some cases, for portions of the CWS that are outside of the design certification scope. In RAI 410.186, the staff requested that Westinghouse provide design descriptions and interface requirements for the CWS as required by 10 CFR Part 52.

In its response, Westinghouse stated that the reference design has been evaluated to verify that postulated CWS failures have no adverse impact on any safety-related SSCs. A postulated CWS line break in the yard area or a failure of the cooling tower basin has no detrimental effect on safety-related SSCs. The cooling tower will be located sufficiently distant from the nuclear island structures so that its postulated collapse does not affect equipment, components, or systems required for safe shutdown of the plant. The site is graded to drain water away from the seismic Category I structures. The seismic Category I structures below grade are protected from flooding by waterproofing membranes and water stops. The COL applicant is responsible for determining the system configuration and may modify the design to meet site-specific requirements. Westinghouse revised SSAR Sections 10.4.5.2.3 and 10.4.12.1, Revision 9, and SSAR Table 1.8-2, Revision 14, to address the staff's concern. This is COL Action Item 10.4-1.

The cooling tower, which serves as a heat sink for the CWS, is site specific with its description in the SSAR as a reference design using a hyperbolic natural draft structure. The cooling tower cools circulating water by discharging the water over a network of baffles in the tower and falls through fill material to the basin beneath the tower, so that heat is rejected to the atmosphere. The cooling tower basin serves as a storage facility for the circulating water inventory and allows bypassing of the cooling tower during cold weather operations. The bypass is used only during plant startup in cold weather, or to maintain the CWS temperature above 4.4 °C (40 °F) while operating at partial load during periods of cold weather. Makeup water is supplied to the cooling tower basin by the raw water system for the water losses in the CWS. Blowdown from the CWS is discharged to the waste water system. Makeup to and blowdown from the CWS is controlled by the makeup and blowdown control valves.

In SSAR Table 10.4.5-1, Westinghouse specifies that the circulating water temperature from the cooling tower to the condenser is 32.2 °C (90 °F) when the wet bulb temperature is at 26.7 °C (80 °F) during limiting site conditions. Because the water temperature in the cooling tower varies with weather conditions, the circulating water temperature to the condenser will change accordingly. Higher circulating water temperature results in increased pressure in the condenser due to a decreased rate of steam condensation. In RAIs 410.13 and 410.14, the staff requested Westinghouse to evaluate possible degradation of the CWS function and remedial measures if the maximum temperature at a site is higher than 32.2 °C (90 °F). This was identified as COL Action Item 10.4.5-1 and Open Item 10.4.5-1 in the DSER. Westinghouse stated that the design temperature is limited to typical sites in the United States and is consistent with the interface requirements of ALWR URD of EPRI. Specific site conditions that exceed the wet bulb temperature of 26.8 °C (80 °F) will be accommodated by specific site analysis to adjust cooling system capability. However, in Revision 14 of the SSAR, Westinghouse deleted Table 10.4.5-1. The staff could not complete its review without the reference design information in this table.

In response to the staff's request, Westinghouse reincorporated Table 10.4.5-1 into the SSAR (Revision 19) and revised its title to "Design Parameters for Major Components of the CWS." The change is consistent with the AP600 standard plant scope as defined in SSAR Section 1.8, "Interfaces for Standard Design." Westinghouse also revised SSAR Section 10.4.5.2.1 (Revision 19) for the reinstated table with the following statement:

> The circulating water system and cooling tower are subject to site-specific modification or optimization. The combined license applicant will determine the final system configuration. Table 10.4.5-1 provides CWS design data based on a conceptual design.

The staff finds that the revised SSAR Table 10.4.5-1 and the clarification in SSAR Section 10.4.5.2.1 are acceptable. Therefore, COL Action Item 10.4.5-1 is dropped and Open Item 10.4.5-1 is closed.

Two CW pumps take suction from the CW intake structure and circulate the water through the TCS, the condenser vacuum seal water heat exchangers, and the tube side of the main condenser and discharge to the cooling tower. The underground portion of the CWS piping is concrete pressure pipe; the rest is carbon steel pipe that is coated with a corrosion preventive compound inside the pipe. Section 10.4.5.2.2 of the SSAR states that the CWS piping, expansion joints, butterfly valves, condenser water boxes, and tube bundles are designed for a maximum pump discharge pressure of 414 ka (60 psig). The staff was concerned that any failure of the pressure piping may cause flooding in the turbine building as a result of the large quantity of water under pressure that is contained in the CWS piping. To satisfy GDC 4, flood protection in the turbine area as a result of CW piping failure should be considered in the CWS design. Leak detection in the condenser pit and automatic isolation of the CWS on indication of turbine building flooding should be provided in the control room. In RAI 410.187, the staff requested that Westinghouse analyze the effects of a postulated failure of the CWS piping or expansion joints and verify that any safety-related SSCs in the turbine building will be protected from the resulting flood water level.

In its response, Westinghouse stated that a small CWS leak in the turbine building will drain into the waste water system. A large CWS leak due to pipe failure will be indicated in the control room by a gradual loss of vacuum in the condenser shell. The effects of flooding due to a CWS failure, such as a rupture of an expansion joint, will not result in detrimental effects on safety-related equipment because there is no safety-related equipment in the turbine building. The base slab of the turbine building is located at grade elevation and water from a system rupture will run out of the building through a relief panel in the turbine building reference plant west wall before the water level could rise high enough to cause damage. Furthermore, the site will be graded to carry water away from the safety-related buildings. The staff finds this response acceptable. However, in Revision 14 of the SSAR, Westinghouse deleted the CW pump discharge pressure and revised Section 10.4.5.2.2 to state that the CWS is designed to withstand the maximum operating discharge pressure of the CW pumps. Westinghouse states in the SSAR that the piping design pressure is site specific and will be provided by the COL applicant. The staff finds the change acceptable. This is part of COL Action Item 10.4-1.

Circulating water chemistry is maintained by the turbine island chemical feed system and controlled by cooling tower blowdown and chemical addition. The chemicals can be divided into six categories based on function, i.e., biocide, algicide, pH adjustor, corrosion inhibitor, scale inhibitor, and a silt dispersant. The use of these specific chemicals is determined by the site water conditions. The COL applicant will determine the use of the specific chemicals in the CWS chemistry control. This is part of COL Action Item 10.4-1.

In Section 10.4.5.2.3 of the SSAR, Westinghouse states that when the condenser is not available due a to malfunction of the CW pumps, cooling tower, or the CW piping, cooldown of the reactor may be accomplished by using the power-operated atmospheric steam relief valves or safety valves rather than the turbine bypass system. The staff concurs with this alternate cooldown method because the turbine bypass system will not function during accident conditions and the CWS is not required for safe shutdown following an accident.

On the basis of its review, the staff concludes that the design of the circulation water system meets the requirements of GDC 4, with respect to the effects of discharging water that may result from a failure of a component or piping in the CWS. Acceptance is based on the following design provisions:

- The CWS is designed to prevent flooding of safety-related areas so that the intended safety function of a system or component will not be precluded due to leakage from the CWS.
- The CWS is designed to detect and control flooding of safety-related areas so that the intended safety function of a system or component will not be precluded due to leakage from the CWS
- Malfunction of a component or piping of the CWS, including an expansion joint, will not have unacceptable adverse effects on the functional performance capabilities of safety-related system or components.

Therefore, the staff concludes that the design of the CWS meets the guidelines of SRP 10.4.5.

10.4.6 Condensate Polishing System

The staff utilized the guidelines of Section 10.4.6 of the SRP to review and evaluate the information submitted by Westinghouse concerning the condensate cleanup system (CCS). The acceptance criteria for the CCS is based on meeting the relevant requirements of GDC 14, "Reactor Coolant Pressure Boundary," as it relates to water chemistry control being capable of preventing adverse chemistry conditions that could degrade the primary coolant boundary integrity. The design of the CCS is acceptable if the CCS design is capable of maintaining adequate water purity specified for the AP600 design.

The purpose of the CCS is to remove dissolved and suspended solids from the condensate to maintain high-quality feedwater to the reactor under all normal plant conditions (startup, shutdown, hot standby, and power operation). Such impurities could cause corrosion damage to the secondary system and increase radiation levels. The CCS removes radioisotopes coming from primary coolant in leakage through damaged steam generator tubes and impurities that could enter the system through a condenser circulating water tube leak. The CCS is not required to perform any safety-related function, but is important in maintaining secondary coolant quality.

Secondary coolant quality is maintained by directing one-third of condensate to one of two polishing vessels, which are piped in parallel. The second polisher is on standby or in the process of being cleaned, emptied, or refilled. The two polishing vessels contain mixed-bed, ion-exchange resin with a strainer installed downstream of each vessel. The strainers are used to prevent the release of resin beads into the feed system.

In Section 10.4.6.2.2 of the SSAR, Westinghouse describes the condensate polishing system resin bed vessels as being constructed of carbon steel with a rubber lining or protective coating. Table 10.4.6-1 of the SSAR states that the polishers will be carbon steel with an epoxy coating. In the DSER, the staff requested that Westinghouse identify in the SSAR the type of coating (or lining) to be used. If rubber is to be used, then limits on sulfur content should be specified. This was Open Item 10.4.6-1. Westinghouse revised SSAR Section 10.4.6.2, which specifies that the polishers are constructed of carbon steel with protective rubber lining on the inside of the vessel. The leachable sulfur content in rubber is specified to be less than 20 ppm. This value corresponds to industry standards endorsed by the NRC. Therefore, Open Item 10.4.6-1 is closed.

In Section 10.4.6.3 of the SSAR, Westinghouse states that if condensate polisher resins become contaminated, they can be transferred to a 1500 ft^3 resin storage tank. In the DSER, the staff requested Westinghouse to clarify whether this tank would be used only in the event of resin contamination. This was Open Item 10.4.6-2. Westinghouse revised SSAR Section 10.4.6.2 and clarified that the tank is used for storage of exhausted or spent resin prior to shipping off site for regeneration or disposal. This closes Open Item 10.4.6-2

In Table 10.4.6-1 of the SSAR, the number of polishers is listed as one, whereas in Figure 10.4.6-1 of the SSAR, two polishers were shown. The staff requested in the DSER that the table be corrected. This was Confirmatory Item 10.4.6-1. Westinghouse revised Figure 10.4.6-1 to show one polisher vessel. This closes Confirmatory Item 10.4.6-1

The acceptance criteria in Section 10.4.6 of the SRP call for conformance of the secondary water chemistry program with Branch Technical Position BTP MTEB 5-3, "Monitoring of Secondary Side Water in PWR Steam Generators." The secondary water chemistry program is described in Section 10.3.5 of the SSAR, and the secondary sampling system is described in Section 9.3.4 of the SSAR. The following issues identified in the DSER concerned conformance with BTP MTEB 5-3:

- Section 10.3.5.4 of the SSAR discusses chemical additions to maintain the all-volatile treatment method. The staff requested Westinghouse to specify in the SSAR if an oxygen scavenger would be used. This was Open Item 10.4.6-3. Westinghouse revised SSAR Section 10.3.5.4 to state that an oxygen scavenger would be used to maintain the dissolved oxygen content in the feedwater within specified limits for each mode of operation. This closes Open Item 10.4.6-3.

- Reference 1 of Section 10.3.5.5 of the SSAR was a 1982 EPRI report (EPRI NP-2704-SR) on secondary water chemistry guidelines. These guidelines have been revised since 1982. The most recent revision available (EPRI TR-102134, Revision 3), was published in 1993. Westinghouse was requested to clarify whether its secondary water chemistry is consistent with the latest EPRI guidelines. If not, inconsistencies with the EPRI guidelines should be discussed in the SSAR. This was Open Item 10.4.6-4. Westinghouse provided the action levels for the secondary side water chemistry during power operation in Table 10.3.5-1, during cold shutdown/wet lay up in Table 10.3.5.2 and during heat up in Table 10.3.5.3. The limits specified in the Table meet the EPRI guidelines of EPRI TP-102134, Revision 3. These guidelines are referenced in the ALWR URD, Chapter 1, Section 5.5.2. Open Item 10.4.6-4 is closed.

- Position 3.a of BTP MTEB 5-3 calls for the secondary water chemistry program to cover five operational modes:

  (1) normal power operation
  (2) startup
  (3) hot standby
  (4) hot shutdown
  (5) cold shutdown/cold wet layup

  Westinghouse describes only three operational modes in Tables 10.3.5-1 through 10.3.5-3 in the SSAR:

  (1) normal power operations
  (2) cold shutdown/wet layup
  (3) heat up

- The staff requested Westinghouse to discuss in the SSAR the condensate clean up system functions during hot standby and hot shutdown modes, as well as the secondary water chemistry parameters for these modes. This was Open Item 10.4.6-5. Westinghouse provided information on the secondary water chemistry program, including condensate cleanup system, in Tables 10.3.5-1 through 10.3.5-3 of the AP600 SSAR. The secondary water chemistry program meets the guidelines of EPRI TR-102134, Revision 3, which the staff finds acceptable. This closes Open Item 10.4.6-5.

- The staff requested Westinghouse to discuss the sampling frequency schedule for the secondary system grab sample points as described in Section 9.3.4 of the SSAR. This was Open Item 10.4.6-6. The information provided in the SSAR indicated that, because grab samples are usually taken whenever there is a need to calibrate in-line instrumentation or verify its reading, no specific frequency can be assigned for taking these samples. This closes Open Item 10.4.6-6.

- The staff requested Westinghouse to describe the program for recording, managing, and interpreting secondary water chemistry data. This was Open Item 10.4.6-7. Westinghouse described in Section 10.3.5.5 of the SSAR action levels for abnormal chemistry, and specified in Tables 10.3.5-1 through 10.3.5-3 the limits for chemistry parameters for each action level, including the licensee-described means for managing and interpreting secondary chemistry data. This closes Open Item 10.4.6-7.

Controlled chemistry has to be maintained in order to reduce corrosion of steam generator tubes and materials, thereby reducing the likelihood and magnitude of reactor piping failures and of primary-to-secondary coolant leakage. The AP600 design meets the recommendations of BTP MTEB 5-3, "Monitoring of Secondary Side Water in PWR Steam Generators." Westinghouse's proposed design criteria and design bases for the CCS and the requirements for operation of the system are acceptable. This conclusion is based on Westinghouse having met the requirements of GDC 14 as it relates to maintaining acceptable chemistry control for secondary coolant during normal operation and anticipated operational occurrences.

10.4.7 Condensate and Feedwater System

The staff reviewed the condensate and feedwater system (CFS) in accordance with Section 10.4.7, "Condensate and Feedwater System," of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the CFS satisfies the following:

- GDC 2, with respect to withstanding the effects of natural phenomena (such as earthquakes, tornados, and floods)
- GDC 4, with respect to withstanding the effects of possible fluid flow instabilities (such as water hammers)
- GDC 5, with respect to the capability of the shared systems and components that are important to safety to perform their required safety functions
- GDC 44, with respect to the capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions
- GDC 45, with respect to periodic ISI of systems, components, and equipment
- GDC 46, with respect to design provisions to permit functional testing of the system and components for structural integrity and leak-tightness

The CFS that provides a continuous feedwater supply to the steam generators is composed of piping and components from the condensate system, main feedwater system, and portions of the steam generator system. The condensate system collects condensed steam from the condenser and pumps the condensate to a deaerator. The deaerator removes dissolved gases from the condensate to provide a source of high-quality heated feedwater supply. A 40.8 cm (16 inch) main feedwater line takes suction from the deaerator and supplies heated feedwater to each of the two steam generators during all modes of plant operation.

The CFS contains three 50-percent-capacity motor-driven condensate pumps and two 50-percent-capacity motor-driven feedwater pumps. Two condensate pumps are required during power operation. The spare condensate pump will start automatically on loss of one of the normally running condensate pumps and/or low condensate header discharge pressure. The main feedwater pumps and booster pumps are paired by train. The two main feedwater pumps take suction from the associated feedwater booster pumps which draw water from the deaerator storage tank. Westinghouse states in the SSAR that the feedwater pump and condensate pump and the pump control systems are designed so that loss of one booster/main feedwater assembly or one condensate pump does not result in a trip of the turbine generator or reactor.

The safety-related isolation function of the CFS is accomplished by redundant means. A single active component failure of the safety-related portion of the system does not compromise the safety function of the system. SSAR Table 10.4.7-1 provides the failure analysis results for those occurrences that result in reduced heat transfer in the steam generators. Loss of all feedwater is also evaluated in Section 15.3 of the SSAR.

Each main feedwater line to the steam generator contains a feedwater flow element, a main feedwater isolation valve (MFIV), a main feedwater control valve (MFCV), and a check valve. The MFIVs, installed in each of the two feedwater lines outside the containment, are used to prevent uncontrolled blowdown from the steam generators in the event of a feedwater line break. The MFCVs (located in the auxiliary building) are used to control feedwater flow rate to the steam generator during normal operation and provide a backup isolation to limit high-energy fluid addition through the broken loop in the event of a main steamline break. The feedwater check valves (located outside the containment) provide backup isolation to prevent reverse flow from the steam generators whenever the feedwater pumps are tripped. The check valves prevent blowdown from more than one steam generator in the event of feedwater line break while the engineered safety feature (ESF) signal is generated to isolate the MFIV and MFCV.

On the basis of the above discussion, the staff finds that the CFS is capable of supplying sufficient feedwater water to the steam generators as required during normal operation and has incorporated appropriate redundancy for containment and feedwater isolation.

The feedwater system does have a has no connection with the startup feedwater system but does not have the safety function to transfer heat under accident conditions and, therefore, GDC 44 is not applicable.

During normal plant operation, as well as during plant upset or accident conditions, possible fluid flow instabilities in the feedwater piping when flow is entering the steam generator may cause water hammer in the system piping. Generic Safety Issue (GSI) a-1 was raised after the occurrence of various incidents of water hammer in operating plants that involved steam generator feedrings and feedwater piping. The staff reviewed the dynamic effects associated with possible water hammers in the feedwater piping for compliance with the requirements of GDC 4. Acceptance is on the basis of meeting the guidance contained in the BTP ASB 10-2, "Design Guidelines for Avoiding Water Hammer in Steam Generators," with respect to feedwater control induced water hammer. Specifically, BTP ASB 10-2 recommends the CFS to be designed with the following provisions:

- prevent or delay water draining from the feedring following a drop in steam generator water level

- minimize the volume of feedwater piping external to the steam generator which could pocket steam using the shortest horizontal run of inlet piping to the feedring

- perform tests, acceptable to the NRC, to verify that unacceptable feedwater hammer will not occur and provide test procedures for staff approval
- implement pipe refill flow limits where practical.

In RAI 410.263, the staff requested Westinghouse to review the AP600 feedwater system design against these guidelines. Westinghouse states in the SSAR that the potential for water hammer in the feedwater line would be minimized by the improved design and operation of the feedwater delivery system with the following features:

- The main feedwater pipe connection on each of the steam generators is the highest point of each feedwater line downstream of the MFIV, and the feedwater lines contain no high point pockets that could trap steam.
- The feedwater enters the steam generator at an elevation above the top of the tube bundle through a feedwater nozzle and below the normal water level by a top discharge feedring.
- The feedwater enters a feedring via a welded thermal sleeve connection and leaves it through nozzles attached to the top of the feedring.
- The feedwater line connected to the steam generator is a short, horizontal or downward sloping feedwater pipe at the steam generator inlet which will help keep the feedring full of water.
- Operational limitations on flow to recover steam generator levels and on early feedwater flow into the steam generator to maintain the feedring full of water will minimize the potential for water hammer occurrence.

SSAR Section 5.4.2.2 (Revision 11) states that these features will prevent the formation of steam pockets during steam generator low level conditions and minimize the potential for trapping pockets of steam that could lead to water hammer events. The top discharge of the feedring, through the nozzles, will help to reduce the potential for vapor formation in the feedring and the heated feedwater will reduce the potential for water hammer in the feedwater piping or steam generator feedrings.

The staff reviewed the SSAR using the guidance of BTP ASB 10-2 and finds that the cited design features would minimize, but not necessarily eliminate, water hammer occurrence in the AP600 feedwater system design. Consequently, flow testing to detect possible feedwater hammer in the feedwater piping should be performed. In its response to RAI 410.263, Westinghouse stated that the test requirement of BTP ASB 10-2 is met by the tests performed on feedring-type steam generators of operating plants in the United States and by monitoring the feedwater system for water hammer during the AP600 initial test program. Therefore, they concluded that further design testing is not required. The staff observed that there was no operating experience for the AP600 feedwater system and the test data from operating plants may not be usable. Therefore, testing for detecting feedwater hammer occurrence to meet BTP ASB 10-2 is required for the AP600 design. This was identified as Open Item 10.4.7-1 in the DSER.

The staff previously reviewed the initial test program in Sections 14.2.9.1.7 and 14.2.10.4.18 of the AP600 SSAR concerning preoperational and startup tests, and found that these sections did not incorporate the tests for feedwater hammer prevention. In a letter dated June 3, 1997, the staff requested Westinghouse to perform feedwater flow tests or perform an engineering analysis based on acceptable test data from an identical Westinghouse plant for preventing feedwater hammer. However, Westinghouse declined to perform such tests for the AP600 standard design. After further discussion and meeting with Westinghouse on this issue, Westinghouse committed to perform the test through the initial test program to satisfy BTP ASB 10-2 (see Section 14.2 of this report). The staff finds it acceptable. Therefore, DSER Open Item 10.4.7-1 is closed.

The staff concludes that the CFS design meets the requirements of GDC 4 with respect to testing for water hammer occurrence. The evaluation of the CFS to conform to GDC 4 with respect to protection from the effects of missile and high-energy line breaks is provided in Sections 3.5 and 3.6 of this report.

The staff reviewed the CFS to the requirements of GDC 2. Compliance with the requirements of GDC 2 is based on adherence to the guidance of RG 1.29, "Seismic Design Classification," Position C.1 for the safety-related portion, and Position C.2 for non-safety-related portion. The SSAR indicates that the CFS is non-safety-related and serves no safety function except the portion of the feedwater piping routed into containment that requires containment and feedwater isolation. The portion of the feedwater system from steam generator inlets outward through the containment up to and including the MFIVs is safety-related and has the following safety-related functions:

- automatically isolate the main feedwater flow to the steam generators when it is required to mitigate the consequences of a steamline or feedwater line break

- provide a barrier against the release of containment atmosphere during a loss-of-coolant-accident (LOCA)

- serve as a boundary for ensuring that steam generator levels can be maintained when the main feedwater pumps are not available

The safety-related portion of the CFS is required to remain functional after a design-basis accident to provide containment and feedwater isolation. This portion will be designed and tested in accordance with the requirements of Section III of the ASME Code for Class 2 components which requires the CFS to be seismic Category I and protected from wind, tornado, missile, and dynamic effects. The non-safety-related portion of the CFS, from the MFIV inlets to the piping restraints at the interface between the auxiliary building and the turbine building, is designed in accordance with the requirements of Section III of the ASME Code for Class 3 components and is seismic Category I. Based on this review, the staff concludes that the CFS design satisfies the requirements of GDC 2, as it relates to protection of the system against natural phenomena.

The AP600 design can be used at either single-unit or multiple-unit sites. Criterion 5 of SSAR Section 3.1.1 states that the AP600 design is a single-unit plant and if more than one unit were built on the same site, none of the safety-related systems would be shared. Should a multiple-unit site be proposed, the COL applicant must apply for the evaluation of the units' compliance with the requirements of GDC 5 with respect to the capability of shared systems and components important to safety to perform their required safety functions. A COL applicant must comply with GDC 5 for a multiple-unit site; therefore, the staff finds that the requirements of GDC 5 are satisfied as it relates to whether shared SSCs important to safety are capable of performing required safety functions.

The SSAR states that both the safety-related and non-safety-related portions of the feedwater system are designed and configured to accommodate ISI in accordance with ASME Code, Section XI. Therefore, GDC 45 is satisfied with respect to periodic ISI of system components and equipment. The SSAR also states that the feedwater system is designed so that the active components are capable of limited testing during plant operation. Therefore, GDC 46 is satisfied with respect to design provisions to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness. The evaluation of the CFS with respect to periodic ISI of the system's components and equipment is addressed in Section 6.6 of this report.

On the basis of its review, the staff concludes that the design of the CFS meets the NRC regulations set forth in GDCs 2, 4, 44, 45, and 46 and is acceptable. The following provide the basis for this conclusion:

- Westinghouse meets the requirements of GDC 2 with respect to the system being capable of withstanding the effects of earthquakes by meeting RG 1.29, Position C.1, for the safety-related portion, and Position C.2 for the non-safety-related portion.

- Westinghouse meets the requirements of GDC 4 with respect to the dynamic effects associated with possible fluid flow instabilities by having the feedwater system designed and tested in accordance with the guidance contained in BTP ASB 10-2, and thereby eliminating or reducing the possibility of water hammers in the feedwater system.

- Westinghouse does not have to meet the requirements of GDC 44 because the AP600 design does not have a safety-related auxiliary feedwater system to provide flow to the steam generator via the feedwater system during accident conditions for decay heat removal.

- Westinghouse meets the requirements of GDC 45 and GDC 46 because the safety-related portions of the system are located in accessible areas for inspection and the active components are capable of limited testing during power operation in accordance with the Technical Specifications.

10.4.8 Steam Generator Blowdown System

The staff reviewed the design on the steam generator blowdown system (SGBS) in accordance with Section 10.4.8 of the SRP. The SGBS is acceptable if it satisfies requirements of GDC 1 as it relates to system components being designed, fabricated, erected, and tested to the highest quality standards; GDC 2, as it relates to system components designed to seismic Category I requirements; and GDC 14 as it relates to secondary water chemistry control. The scope of review included P&IDs, seismic and quality group classifications, design process parameters, and Westinghouse's analyses of the proposed system operation.

The SGBS controls the quality of water on the shell side of the steam generators by removing chemical impurities and radioactive materials that accumulate as a result of primary-to-secondary and condenser tube leaks, and corrosion of the steam generator materials. A continuous high-flow blowdown controls the concentration of these impurities.

Each steam generator has its own blowdown line with the capability of blowing down from a location just above the tubesheet where impurities are expected to accumulate. Each system accommodates a continuous blowdown with a minimum blowdown rate of 0.1 percent (17 gal/min total, or 8.5 gal/min per steam generator), up to 1 percent of the maximum steam flow rate and up to 3 percent for a short period of time. In each line, the blowdown fluid is cooled in a regenerative heat exchanger directed to a filter to remove suspended solids and particulate matter from the influent and then directed to the mixed-bed demineralizers where ionic species are removed. The processed fluid then returns to the condenser. The system processes the blowdown fluid at a rate of from 0.1 percent of each steam generator's maximum steam flow, for full-power operation and normal steam generator chemistry, up to 1 percent of maximum steam flow is processed to control chemistry within the normal limits. There is a provision to isolate the portion of the blowdown system exiting the containment by the redundant blowdown line isolation valves that would close upon a containment isolation signal, or upon actuation of the passive residual heat removal system or abnormal conditions in the blowdown system. The valves provide containment integrity in conjunction with the steam generator and main steamline within containment.

Section 10.4.8.2.2.6 of the SSAR states that blowdown is discharged to the waste water system (WWS) for disposal, or recovered if non-radioactive. In the DSER Westinghouse was requested to clarify if the WWS was the same system as the liquid radwaste system (WLS) mentioned in Section 10.4.8.2.2.2 of the SSAR. This was Open Item 10.4.8-1. Westinghouse revised SSAR Sections 10.4.8.2.2.2 and 10.4.8.2.2.6 to clarify how blowdown flow is controlled and recovered. Normal operation is to recover the blowdown flow through the condensate system. However, the plant operator has the option of discharging blowdown and a high level of impurities to the WWS. The blowdown flow is continuously monitored for radioactivity from steam generator primary-to-secondary tube leakage. If such radioactivity is detected, the WLS may be manually aligned to process the blowdown effluent. If radioactivity reaches a preset high level, the blowdown flow control valves and isolation valves automatically close. This explanation clarified that the WWS and the WLS are not the same systems. This closes Open Item 10.4.8-1.

The SGBS is classified as seismic Category I and Quality Group B from its connection to the steam generator inside the primary containment, up to and including the first isolation valve outside the containment, in accordance with RGs 1.26 and 1.29, because this portion of the SGBS is considered an extension of the primary containment. The SGBS downstream of the outer containment isolation valves, up to and including the piping anchors located at the auxiliary building wall, are designed in accordance with the requirements of Class 3 of Section III of the ASME Code and seismic Category I requirements. Piping downstream of the auxiliary wall anchors is not safety-related and not seismic Category I, and meets the quality standards of Position C.1.1 of RG 1.143.

The staff concludes that SGBS meets the quality standard requirements of GDC 1 and the seismic requirements of GDC 2. The SGBS design also meets the primary boundary material integrity requirements of GDC 14, as it relates to maintaining acceptable water chemistry control during normal and anticipated operational occurrences by reducing corrosion of steam generator tubes and materials, thereby reducing the likelihood and magnitude of primary-to-secondary coolant leakage.

10.4.9 Startup Feedwater System

The AP600 plant does not have a safety-related auxiliary feedwater system. Instead, a non-safety-related startup feedwater system (SFS) is used to supply feedwater to the steam generators during plant conditions of startup, hot standby, cooldown, and unavailability of main feedwater pumps. In response to the staff's RAI regarding the use of a non-safety-related SFS, Westinghouse stated that the SFS is not required to supply feedwater under accident conditions, but the system is expected to be available as a non-safety-related first line of defense to provide a source of feedwater for loss of feedwater events. The safety-related passive core cooling system (PXS) will provide safety-grade protection for such events. Therefore, the operation of the SFS will not be credited to mitigate a design-basis accident described in Chapter 15 of the SSAR.

Because the passive design philosophy departs from current licensing practice, the staff may not require the non-safety-related active SFS to meet all the safety-related criteria specified in Section 10.4.9, "Auxiliary Feedwater System," of the SRP. However, the availability of the system must be ensured when needed in its defense-in-depth roles. Consequently, regulatory oversight measures are considered for those significant non-safety active systems. Specifically, the staff's review considered whether the design of the startup feedwater system:

- has sufficient redundancy to ensure defense-in-depth functions
- has electric supplies from both normal station ac and onsite non-safety-related ac power supplies that are separated to the extent practicable
- is designed and arranged for conditions or an environment anticipated during and after events to ensure operability, maintenance accessibility, and plant recovery
- is protected against internal flooding and other in-plant hazards, including the effects of pipe ruptures, jet impingement, fires, and missiles
- can withstand the effects of natural phenomena (such as earthquakes, tornados, and floods) without the loss of capability to perform required function
- has a QA program applied to it
- is included in the design reliability assurance program (DRAP) and under the scope of the Maintenance Rule (10 CFR 50.65) to ensure proper and effective maintenance, surveillance, and ISI and testing.

- has graded safety classifications and graded requirements for instrument and control (I&C) systems based on the importance to safety of their function to meet the reliability/ availability (R/a) missions

- has proper administrative controls for shutdown configurations

- meets RG 1.29, BTP ASB 10-1, and BTP (SRXB) 5-1 concerning seismic classification, power diversity, and design of residual heat removal systems

- meets NUREG-0737 and NUREG-0611 concerning generic improvements to the startup feedwater system design, technical specifications, and SFS reliability

In RAI 410.188, the staff requested that Westinghouse address the above criteria for the SFS. Westinghouse stated that the SFS performs no safety-related functions other than isolation of the SFS to mitigate a main steamline break and does not need to meet the criteria applicable to safety-related systems. Consequently, Technical Specifications (TS) were not provided for the startup feedwater pumps and their associated flow paths, as is required for the auxiliary feedwater system. Instead, Westinghouse provided TS 3.7.7 in SSAR Section 16.1 for the startup feedwater isolation valves and control valves as they are safety-related. The staff concurs with Westinghouse's position regarding administrative controls because the SFS pumps and flow paths are not safety related. The staff's review of the SFS regarding the rest of the review criteria for non-safety systems serving defense-in-depth functions is provided in the following paragraphs.

The SFS has two trains, which share common suction and discharge piping. Two parallel startup feedwater pumps are provided with a single pump capable of satisfying the SFS flow demand for decay heat removal. Each of the two trains was initially designed with a 100-percent capacity motor-driven startup feedwater pump. In Section 10.4.9.2.1 (Revision 4) of the SSAR, Westinghouse changed the pump capacity from 100-percent to 50-percent because of the low flow demand. The staff was concerned that the change of pump capacity may not withstand a single active component failure if one of the two pumps is unavailable. In RAI 410.292, the staff requested Westinghouse to demonstrate that the reduced pump capacity can meet the flow demand and can perform defense-in-depth functions with sufficient redundancy as specified in the regulatory treatment of non-safety systems (RTNSS) process. Westinghouse revised SSAR Section 10.4.9.2.2 (Revision 9) to state that each startup feedwater pump can supply 100-percent of the required flow to the two steam generators to meet the decay heat removal requirements. The staff finds the revision acceptable.

During normal startup and shutdown operations, the two startup feedwater pumps take suction from the condensate storage tank to supply feedwater to the two steam generators. In the event of loss of offsite power that results in a loss of main feedwater supply, the SFS automatically supplies feedwater to the steam generators to cool down the reactor under emergency shutdown conditions. The startup feedwater pumps automatically start following the loss of main feedwater flow in conjunction with an intermediate low steam generator level setpoint. The startup feedwater flow transmitters also provide redundant indication of startup feedwater and automatic safeguards actuation input on low flow coincident with a low, narrow-range steam generator level.

Each of the two startup feedwater pumps and its associated instruments and electric valves are powered by the standby source motor control center circuit. The pump discharge isolation valves are motor operated and are normally closed and interlocked with the startup feedwater pumps. In the event of loss of offsite power, the startup feedwater pumps will be powered by the onsite standby power supply (diesels). If both the normal ac power and the onsite standby ac power are unavailable, these valves will fail as is. The pump suction header isolation valves are pneumatically actuated. The SFS also has temperature instrumentation in the pump discharge that would permit monitoring of the SFS temperature.

On the basis of the above discussion, the staff finds that the startup feedwater pumps possess diversity in motive power source with an electric supply from both normal station ac and onsite non-safety-related ac power supplies that are separated. Therefore, the staff concludes that the design of the startup feedwater pumps meets the redundancy and power source review criteria.

Section 14.2.8.1.91 of the SSAR presents the initial test program for the startup feedwater control system. Each startup feedwater pump is equipped with a recirculation line to the demineralized water storage tank for periodic functional testing. When one pump is being tested, the other pump will remain available for automatic operation. Currently, periodic surveillance tests of the auxiliary feedwater pumps and their associated flow trains for the operation plants are required by the standard technical specifications. Because the SFS serves as the first line of defense for loss of feedwater events, maintenance, surveillance, and ISI and testing of the SFS should be incorporated into its maintenance and reliability assurance programs. This was identified as Open Item 10.4.9-1 in the DSER.

In its response, Westinghouse stated that the maintenance, surveillance, and ISI and testing of the SFS, as it relates to defense-in-depth systems, is incorporated into applicable sections of the SSAR. In SSAR Section 14.2.9.1.2, verification for proper operation of the main and feedwater valves is specified, including automatic open/close valve operation and timing. SSAR Section 14.2.9.2.2 requires verification of the startup feedwater pumps to actuate or start properly. Maintenance requirements of the SFS are covered by the Maintenance Rule. Surveillance of the SFS components important to safety is covered by the TSs in SSAR Section 16.1. ISI of the active valves (i.e., startup feedwater control valve) is covered by SSAR Section 3.9.3.2.2 and Table 3.9-16. Inservice testing of these valves is included in SSAR Table 3.9-12. The Reliability Assurance Program for the startup feedwater pumps is addressed in SSAR Section 17.4. The staff's review finds that the SFS has been incorporated into the maintenance and reliability assurance program. Therefore, DSER Open Item 10.4.9-1 is closed.

Item II.E.1.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (dated November 1980), requires all operating PWR plants to perform auxiliary feedwater system reliability analysis. Generic Safety Issue 124 addresses the use of probabilistic risk assessment (PRA) to evaluate the reliability of the auxiliary feedwater system. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs" (dated April 2, 1993), provides the interim position on the reliability assurance program applicable to AP600 design certification. Accordingly, Westinghouse performed reliability analysis for the main and startup feedwater systems that was addressed in Appendix C8 of the AP600 PRA.

Westinghouse evaluated the SFS reliability for the following two postulated transient and accident scenarios:

(1) loss of main feedwater with offsite power available
(2) loss of main feedwater due to loss of offsite power

The dominant quantitative system unavailability for each of the above two transient/accident conditions is calculated using the guidance and database in Appendix III of NUREG-0611, which is referenced in SRP 10.4.9. Westinghouse modeled the two non-safety-related startup feedwater pumps in the PRA. These pumps are started automatically by the control system and are automatically loaded on the non-safety-related diesels to take suction from the deaerating water storage tank or the condensate storage tank. In addition, the passive residual heat removal heat exchanger and passive feed and bleed will provide safety-related means of core decay heat removal. The two passive features are redundant and diverse from each other. The staff's review confirms that Westinghouse properly analyzed the safety function of the SFS to meet the reliability/availability missions as specified in the RTNSS. The acceptance of the SFS reliability analysis is addressed in Chapter 19 of this report.

Westinghouse also performed a startup feedwater system component failure analysis, with the results identified in Table 10.4.9-1 of the SSAR, which listed several cases where startup feedwater flow is not available to the steam generator. The analysis indicates that failure of startup feedwater supply has no effect on RCS function.

The SFS has no safety-related function other than containment and startup feedwater isolation. The portion of the SFS piping that penetrates the containment from the startup feedwater isolation valve (SFIV) to the connection at the steam generator is safety-related, and is required to function following a design-basis accident to perform safety functions such as containment isolation, steam generator isolation, and feedwater isolation. This portion is designed in accordance with the requirements of Section III of the ASME Code for Class 2 components and is seismic Category I. The portion of the SFS piping from the SFIV inlets to the pipe restraints at the interface between the auxiliary building and turbine building is non-safety-related and is designed in accordance with Section III of the ASME Code for Class 3 components and is seismic Category I. As specified in Table 3.2-3 of the SSAR, other valves and remaining piping of the SFS meet ANSI B31.1 requirements and are classified as Class D. Westinghouse stated that inclusion of the non-safety-related components in Class D recognizes their important first-level-of-defense functions that help to reduce the calculated PRA core melt frequency. The staff's review confirms that the SFS design complies with RG 1.29 concerning its seismic classification.

In SSAR Section 5.4.2.2 (Revision 4), Westinghouse added a paragraph to address a design change to use a separate startup feedwater delivery system to the steam generator. The 10.2 cm (4 inch) startup feedwater line connects directly to the steam generator nozzle rather than via the main feedwater piping. The design change makes the main feedwater system and the startup feedwater system parallel systems. The main feedwater system draws water from the deaerator tank and delivers it to the main feedrings within the steam generator, but the startup feedwater system draws water from the condensate storage tank and delivers it to the startup feedwater nozzle on the steam generator. Westinghouse stated that the piping route change would minimize the potential for thermal stratification in the feedwater piping. The use of a separate startup feedwater line and nozzle in the steam generator would reduce the thermal transient when introducing cold water through the feedwater line. The design allows main feedwater pumps to deliver water to the startup feed headers but does not allow the startup feed pumps to deliver water to the main feed headers. The staff finds that the design change will improve system safety.

In situations where SFS is actuated, the flow control valves automatically control flow to each steam generator. The staff was concerned that the startup feedwater delivery system to the steam generators was not evaluated for water hammer. In RAI 410.294, the staff requested that Westinghouse provide information on the dynamic effects associated with possible water hammers in the startup feedwater delivery system as a result of the SFS design change, and verify that the initial flow does not result in plant damage due to water hammer events.

Westinghouse responded that the startup feedwater piping layout includes the same features as the main feedwater piping layout, such as a downward elbow in close proximity to the startup feedwater nozzle on the steam generator, exclusion of high points for limiting void collection, redundant positive isolation to prevent back leakage, and delivery of startup feedwater to steam generator independent of feedrings. The startup feedwater system is sized, operated, and has water sources consistent with minimizing the potential for water hammer. The staff finds that Westinghouse considered water hammer prevention in the SFS design change.

Double-valve startup feedwater isolation is provided by the SFIV and the startup feedwater control valve (SFCV) located outside the containment. The SFIV and SFCV are powered from separate Class 1E power sources to provide redundant and independent actuation. SSAR Section 10.4.9.1.1 (Revision 4) states that the SFCVs and SFIVs are designed to close on an appropriate engineered safety signal (startup feedwater isolation signal). Before the design change, SSAR Section 10.4.7.1.1 stated that the MFIV and SFIV served as the containment isolation valves and closed on a containment isolation signal or backflow in the line. In RAI 410.293, the staff requested that Westinghouse justify the change in MFIV and SFIV actuation signals.

Westinghouse stated that SSAR Sections 10.4.7.1.1 and 10.4.9.1.1 address the safety-related functions of the MFIVs and SFIVs respectively. The design change of the startup feedwater system has not modified the safety-related logic for automatic isolation of the SFIV. The fact that the SFIV is serving as a containment isolation valve and closing on an ESF signal indicates the need to isolate the startup feedwater line while retaining the defense-in-depth function of the system. The valves also can be opened after isolation or closed by a remote manual signal. The ESF signal is designed to close the remotely operated containment isolation valves. Westinghouse revised SSAR Section 10.4.9.1.1 (Revision 19) to state that both SFIV and SFCV are designed to close on a startup feedwater isolation signal, an appropriate ESF signal, as indicated on SSAR Figure 7.2-1. The startup feedwater control valve also serves as a containment isolation valve. Because of their specific safety function that requires both the valves to close automatically on receipt of a safeguard actuation signal, the staff finds the design of the actuation signals acceptable.

On the basis of its review, the staff concludes that the SFS design meets the review criteria for non-safety systems serving defense-in-depth functions.

10.4.10 Auxiliary Steam System

The current SRP does not include a section specifically addressing the auxiliary steam system. The staff determined that the acceptability of the system will be on the basis of meeting the requirements of GDC 4, in that failure of the auxiliary steam system as a result of a pipe break or malfunction of the system should not adversely affect safety-related systems or components.

DSER Open Item 10.4.10-1 identified RAI 410.260 regarding a clarification of system description. The revised Section 10.4.10.5 in the SSAR Revision 4 addressed the staff question. Therefore, DSER Open Item 9.2.1-1 is closed.

The auxiliary steam system supplies steam required by the unit for a cold start of the main steam system and turbine generator. It also provides steam during plant operation for hot water heating. Main steam supplies the auxiliary steam header during normal operation. The auxiliary boiler provides steam to the header during a plant shutdown. The auxiliary steam boiler has a rated capacity of 110,000 pounds per hour of saturated steam at 195 psig. The system is protected from overpressure by safety valves on the boiler, boiler deaerator, and auxiliary steam header.

The auxiliary steam system is a non-safety-related system classified as AP600 Class E. The system consists of an auxiliary steam system and boiler, pumps, auxiliary boiler deaerator, chemical treatment components, and auxiliary boiler fuel oil components.

Operational safety features are provided within the system for the protection of plant personnel and equipment. The auxiliary steam system does not interface directly with nuclear process systems. The auxiliary boiler is located in the turbine building, and none of the lines pass through areas where safety-related equipment is located. Therefore, the auxiliary steam system meets the requirements of GDC 4 in that failure of the system as a result of a pipe break or malfunction of the system should not adversely affect safety-related systems or components.

Testing of the auxiliary steam system is performed before initial plant operation. Components of the system are monitored during operation to verify satisfactory performance. In the July 29, 1994, response to RAI 410.151, Westinghouse stated that testing procedures for the auxiliary steam system are located in the system specification and vendors' equipment instruction manuals, which are not part of the AP600 design certification review.

On the basis of the above review, the staff finds that the auxiliary steam system meets the requirements of GDC 4 in that failure of the auxiliary steam system as a result of a pipe break or malfunction of the system does not adversely affect safety-related systems or components, and therefore, the staff finds the auxiliary steam system acceptable.

11 RADIOACTIVE WASTE MANAGEMENT

The radioactive waste (radwaste) management systems designed for the AP600 control the handling and treatment of liquid, gaseous, and solid radwaste. The liquid waste management system includes the liquid radwaste system (WLS), which is designed to control, collect, process, store, and dispose of liquid radioactive wastes. The WLS is discussed in Section 11.2 of the standard safety analysis report (SSAR) and Section 11.2 of this report. The WLS contains hold up tanks, process pumps, other processing equipment including monitor tanks, and appropriate instrumentation and controls. The principal waste treatment process in WLS is ion exchange.

The gaseous waste management system (WGS) collects, processes, and monitors gaseous releases; the WGS is discussed in Section 11.3 of the SSAR and Section 11.3 of this report. Gaseous wastes that are potentially radioactive or hydrogen bearing, namely, those from degassing the reactor coolant and the reactor coolant drain tank contents, are collected and decayed in charcoal delay beds and subsequently released to the environment via the plant vent.

The solid waste management system (WSS) controls the processing of solid wastes generated during reactor operation, as well as the packaging and storage of such processed wastes before shipment to a licensed disposal facility. The WSS is discussed in Section 11.4 of the SSAR and Section 11.4 of this report.

The process and effluent radiological monitoring instrumentation and sampling systems detect and measure the radioactive materials in plant liquid and gaseous process and effluent streams; this system is discussed in Section 11.5 of the SSAR.

11.1 <u>Summary Description/Source Terms</u>

Chapter 11 of the SSAR and Westinghouse's responses to NRC staff requests for additional information (RAIs) provide the basis for the staff's review. In addition, the staff's review includes Section 1.8 of the SSAR, as it relates to the identification of combined operating license (COL) and site-dependent interfaces for radwaste management systems. Sections 11.1 through 11.5 of the NRC's Standard review Plan (SRP) provides the acceptance criteria for the staff's evaluation. The subject SRP sections include compliance with 10 CFR Part 50, Appendix I, and applicable General Design Criteria (GDC) as acceptance criteria (GDCs 3, 60, 61, 63, and 64). Additionally, the staff used 10 CFR Part 20 Section 20.1302, which defines the criteria for radionuclide concentration limits in liquid and gaseous effluents in unrestricted areas. For evaluating the source terms, the staff reviewed the adequacy of information in Chapter 11 and Section 1.8 of the SSAR, and Westinghouse's responses to RAIs on radwaste management systems, against the requirements set forth in 10 CFR 50.34a.

In the following sections (11.2 and 11.3), the staff evaluates the capability of the WLS and WGS to keep radioactive effluents in unrestricted areas as low as reasonably achievable (ALARA) in accordance with the requirements of 10 CFR Part 50, Appendix I. The staff's evaluation takes into account the radwaste expected to be produced over the life of the plant. Additionally, the staff evaluates the capability of the systems to maintain the radionuclide releases to unrestricted areas (via the liquid and gaseous effluents) below their respective liquid and gaseous effluent concentration limits specified in 10 CFR Part 20, Section 1302, during periods of fission product leakage.

The staff found that the assumption of a 0.25 percent fuel defect described in SSAR Section 11.1.1.1 for the AP600 design deviates from the fuel defect assumption of the 1.0 percent described in SRP Sections 11.2 and 11.3 for the WLS and the WGS. This was *identified as draft safety evaluation report (DSER) Open Items 11.2-5 and 11.3-7 relating to* compliance with 10 CFR Part 20, Section 1302. To justify its position, Westinghouse stated in Revision 12 of the SSAR, Section 11.1.3, and the note for Table 11.2-3 that the WLS and WGS have the capability to process wastes on the basis of one percent fuel defects. The demonstration of this capability is shown in the results of Westinghouse's analyses, which assume a one percent fuel failure as described in SSAR Sections 11.2 and 11.3, and evaluated is in Sections 11.2 and 11.3 of this report.

In addition, Westinghouse provided fuel leak data for operating plants in a letter dated June 17, 1997, to demonstrate that the 0.25 percent fuel defect is an appropriate assumption. The staff reviewed the recent fuel data for Westinghouse fuel and compared the data with independent information available to the staff. Based on the results of the comparison, the staff agrees with Westinghouse that for Westinghouse 17 x 17 Vantage 5 Hybrid (V5H) fuel the 0.25 percent fuel failure assumption is reasonable. Furthermore, the AP600 Technical Specifications, limiting condition for operation (LCO) 3.4.11, "RCS Specific Activity," specifies dose limits for iodines and noble gases corresponding to a fuel defect level of 0.25 percent to ensure that plant operation remains within the limits consistent with the design assumptions. On the basis of the above, the staff finds Westinghouse's justifications for the deviation from the SRP on the fuel defect assumption to be acceptable. The adequacy of process and storage of the solid waste is discussed in SSAR Section 11.4 and evaluated in Section 11.4 of this report.

On the basis of information supplied by Westinghouse, the staff evaluated the following aspects of the AP600 design:

- compliance of the quality group and seismic design classification applied to equipment and structures housing these systems in accordance with RG 1.143
- compliance of design features with GDC 3 for protecting gaseous waste handling and treatment systems from the effects of a flammable/explosive mixture of hydrogen and oxygen
- compliance with GDC 60 for controlling radioactive releases to the environment
- compliance with GDC 61 for ensuring adequate safety under normal and accident conditions
- compliance with GDC 64 for monitoring radioactivity releases

- the waste gas processing system's conformance with Branch Technical Position of Effluent Treatment Systems Branch (BTP ETSB) 11-5, "Postulated Radioactive Releases Due to Waste Gas System Leak or Failure," (Rev. 2), July 1981

In its evaluation of the WSS, the staff considered design objectives, including volumes and activities of wastes processed for offsite shipment, conformance to federal packaging regulations, provisions for controlling potentially radioactive airborne dusts generated during compacting operations, and provisions for onsite storage before shipping.

Specific compliance with 10 CFR Part 50, Appendix I and the guidelines in American National Standards Institute (ANSI) N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities"; Regulatory Guide (RG) 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants"; and RG 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) — Effluent Streams and the Environment," is not totally within the scope of the AP600 standard design. Therefore, the staff will review COL applications to ensure their conformance with 10 CFR Part 50, Appendix I, ANSI N13.1-1969, and RGs 1.21 and 4.15. A COL applicant referencing the AP600 certified design must demonstrate compliance with 10 CFR Part 50, Appendix I guidelines for maximally exposed offsite individual doses and population doses via liquid and gaseous effluents to the environment as well as the additional guidelines addressed in ANSI N13.1-1969, and RGs 1.21 and 4.15. This was identified as DSER COL Action Item 11.1-1 and Open Item 11.1-1. Westinghouse revised SSAR Section 11.5.7 and Table 1.8-2 to require COL applicants' compliance with the regulatory guidance addressed above. Therefore, DSER Open Item 11.1-1 is closed.

To evaluate the calculation of the expected releases of radioactive materials via liquid and gaseous effluents for the AP600 standard design, the staff used the pressurized-water reactor (PWR) GALE code methodology described in NUREG-0017, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors," Revision 1, dated April 1985. The calculations in the GALE code for estimating the liquid and gaseous effluents during normal plant operation, including anticipated operational occurrences (AOOs), are based on the following factors:

- data from operating reactors
- field and laboratory tests
- standardized primary coolant activities and adjustment factors derived from the American Nuclear Society 18.1 Working Group recommendations
- standardized secondary coolant activities for a PWR with U-tube steam generators (SGs) derived from the above group's recommendations
- release and transport mechanisms that result in the appearance of radioactive material in liquid streams

- the plant's radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environment

The principal parameters used in the calculations are developed from the data given in Sections 11.1, 11.2, and 11.3 of the SSAR.

11.2 Liquid Waste Management System

11.2.1 System Description and Review Discussion

The liquid waste management system consists of the process equipment and instrumentation necessary to collect, process, monitor, and recirculate or discharge the processed liquid radwaste. The staff used the acceptance criteria in Section 11.2 of the SRP to evaluate system compliance with 10 CFR 50.34a and 20.1302; 10 CFR Part 50, Appendix A, GDCs 60, 61, and 64; and 10 CFR Part 50, Appendix I.

10 CFR 50.34a requires that sufficient design information be provided to demonstrate that the design objectives for equipment necessary to control releases of radioactive effluents to the environment have been met. 10 CFR 20.1302 defines the criteria for radionuclide concentration limits in liquid and gaseous effluents in unrestricted areas. GDC 60 relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment. GDC 61 relates to radioactive waste systems being designed to assure adequate safety under normal and postulated accident conditions. The relevant requirements of the regulations are met by using the regulatory positions contained in the following regulatory guides.

- RG 1.110, as it relates to performing a cost-benefit analysis for reducing cumulative dose to the population by using available technology

- RG 1.143, as it relates to the seismic design and quality group classification of components used in the liquid waste treatment system and structures housing systems and the provisions used to control leakages

GDC 64 relates to the radioactive waste management systems being designed for monitoring radioactive levels. 10 CFR Part 50, Appendix I, Sections II.A and II.D relate to the numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion.

The AP600 liquid waste management system includes the systems that may be used to process disposal liquids containing radioactive material. These include the following:

- steam generator blowdown processing system (SGBS)
- radioactive waste drain system (WRS)
- liquid radwaste system (WLS)

The SGBS and WRS are discussed in this report in Sections 10.4.8 and 9.3.5 respectively. These sections discuss the systems with respect to their expected releases from the liquid waste management systems. Section 11.2 of this report primarily discusses the WLS.

The WLS, shown in SSAR Figure 11.2-1, consists of tanks (effluent holdup tanks, waste holdup tanks, chemical waste tank, and monitor tanks), pumps, ion exchangers, and filters. The design data for those components are listed in SSAR Table 11.2-2. The WLS is designed to control, collect, process, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences. The WLS is designed to process radioactively contaminated wastes in four major categories:

(1) borated, waste water from the reactor coolant system (RCS) effluents through the chemical and volume control system (CVS), and also from the reactor coolant drain tank

(2) floor drains from various building sumps and equipment drains

(3) Detergent waste from hot sinks and showers, and some cleanup and decontamination processes

(4) Chemical waste from the laboratory and other relatively small volume sources

Nonradioactive secondary-system waste normally is not processed by the WLS. Secondary-system effluent is normally handled by the SGBS, as described in SSAR Section 10.4.8, and by the turbine building drain system. Radioactivity can enter the secondary systems from steam generator tube leakage. If significant radioactivity is detected in secondary-side systems, blowdown is redirected to the WLS for processing and disposal in a monitored fashion.

The subsystem that processes the borated and hydrogen-bearing liquid from the RCS through CVS and reactor coolant drain tank is called the effluent subsystem. Effluent from the CVS is produced mainly as a result of RCS heatup and boron concentration changes. The reactor coolant drain tank collects leakage and drainage from various primary systems and components inside containment. Input collected by the effluent subsystem normally contains hydrogen and dissolved radiogases. It is routed through the WLS vacuum degasifier, where dissolved hydrogen and fission gases are removed. After the degasifier, it is stored in the effluent holdup tanks. The effluent holdup tanks vent to the aerated vent header of the gaseous radwaste system. Hydrogen monitors are included in the tanks to alert the operator to elevated hydrogen levels. There are two effluent holdup tanks, each with a capacity of 28,000 gallons. The tank contents may be recirculated and sampled, recycled through the degasifier for further gas striping, discharged to the RCS via the CVS makeup pumps, discharged to a mobile concentration and solidification system, or processed.

Normally, these wastes are processed through the pre-filter, ion exchangers, and after-filter. The processed waste is then collected in one of the effluent waste monitor tanks (with a capacity of 57 m^3 (15,000 gallons)), sampled, and discharged (if acceptable) or recirculated for further processing by the subsystem. Westinghouse estimated the normal generation rate of these wastes and listed them in SSAR Table 11.2-1.

Potentially contaminated waste from floor drains, containment sumps, auxiliary building sumps, excess water from the solid radwaste system, and other sources that tend to be high in particulate loading but low in dissolved solids loading is collected in waste holdup tanks.

Additives may be introduced to the tank to improve filtration and ion exchange processes. Tank contents may be recirculated for mixing and sampling.

The principal process equipment for treating liquid radwaste from the effluent holdup tanks and the waste holdup tanks, is a set of four non-regenerative ion exchangers (demineralizers) connected in series. The four ion exchangers have a waste pre-filter (upstream of the ion exchangers), and a waste after-filter (downstream of the ion exchangers) and consist of the following:

- one specific ion demineralizer (containing activated charcoal on zeolite resin) that acts as a deep-bed filter and removes oil from floor drain wastes
- one cation bed demineralizer
- two mixed bed demineralizers

Design flexibility exists to manually bypass any of these ion exchangers and to interchange the order of the last two mixed beds to provide complete usage of the resin. Westinghouse indicated that the media for the cation bed and the mixed bed demineralizers will be selected by the COL applicant to optimize system performance. A COL applicant referencing the AP600 certified design should identify the media it plans to utilize for the cation bed and the mixed bed demineralizers in the WLS. This is DSER COL Action Item 11.2-1 and Open Item 11.2-1. In Revision 6 to the SSAR, Westinghouse identified in Table 1.8-2 and Section 11.2.5.3 that the COL applicant will identify the types of liquid waste ion exchange and adsorbent media to be used in the WLS, dependent upon developments in ion exchange technology and specific characteristics of the liquid radwaste to be processed. Therefore, COL Action Item 11.2.1 is acceptable and Open Item 11.2.1 is closed.

On the basis of SSAR Table 11.2-1, the combined normal generation rate of the wastes serviced by both effluent holdup tanks and the waste holdup tank is 6.2 m^3/day (1647 gal/day) and the capacity of the shared limiting processing equipment (ion-exchanger) is 409 m^3/day (108,000 gal/day). This provides an adequate margin for processing surges in the generation rates of all the wastes serviced by the two subsystems. Therefore, the holdup tanks in these subsystems have an adequate margin for collecting large increases in the generation of wastes.

The WLS piping permits connection of mobile processing equipment. When liquid wastes are processed by mobile equipment, the treated liquid waste is returned to the WLS for eventual discharge to the environs, or to an ultimate disposal point for liquids that are to be removed from the plant site.

The detergent waste subsystem collects wastes that are generally high in dissolved solids but low in radioactivity from plant hot sinks and showers, and some cleanup and decontamination processes. The detergent wastes are generally not compatible with the ion exchange resins and are collected in the chemical waste tank 34 m^3(8,900 gal). The size of the chemical waste tank is adequate. Normally, these wastes are sampled. If the detergent waste activity is low enough, the waste can be discharged without processing. When sufficient detergent wastes are produced and processing is necessary, the waste water may be transferred to a waste holdup tank and processed in the same manner as other radioactively contaminated waste water.

If onsite processing capabilities are not suitable for the composition of the detergent waste, processing can be performed using mobile equipment brought into the radwaste building or the waste water can be shipped offsite for processing. After processing by mobile equipment, the water may be transferred to a waste holdup tank for further processing or transferred to a monitor tank for sampling and discharge. Westinghouse estimates in SSAR Table 11.2-1 that the normal generation rate of these wastes will be 0.6 m^3/day (157 gal/day) and assumes that the waste will be fully discharged to the environs. The capacity of the limiting processing equipment (ion-exchanger) is 409 m^3/day (108,000 gal/day) in this subsystem. This capacity provides an adequate margin for processing a surge in the generation rate of this waste.

Radioactively contaminated chemical wastes are normally generated at a low rate and collected in the chemical waste tank shared with detergent wastes. Chemicals are added to the tank, as needed, for pH or other chemical adjustment. The design includes alternatives for processing or discharge. They may be processed onsite, without being combined with other waste, using mobile equipment. When combined with detergent wastes, they may be treated like detergent wastes described above. If onsite processing capabilities are not suitable, processing can be performed using mobile equipment, or the waste water can be shipped offsite for processing.

Steam generator blowdown is normally processed by the steam generator blowdown treatment system demineralizers (see Section 10.4.8 of the SSAR). In the AP600 design, the steam generator blowdown does not normally contribute to any liquid radwaste discharge to the environs. Under normal conditions, the processed blowdown is totally recycled in the plant (i.e., discharged to the condenser hot well). However, if the blowdown flow is detected to be excessively radioactive, it will be manually aligned to the inlet of the waste holdup tank for processing before its eventual discharge to the environment.

Process discharge is normally aligned to one of the three monitor tanks. The release of processed liquid waste from any monitor tank to the environs is permitted only when sampling of the subject tank's contents shows that such a release is permissible. The effluent discharge line includes a radiation monitor. The discharge flow rate for the borated wastes should be pre-set by the COL applicant to limit the boric acid concentration in the circulating water blowdown stream to an acceptable level in compliance with local requirements. A COL applicant referencing the AP600 certified design should identify its planned discharge flow rate for borated wastes. This was identified as DSER COL Action Item 11.2-2 and Open Item 11.2-2. Westinghouse revised SSAR Sections 11.2.5.4 and Table 1.8-2 to require COL applicant to determine the rate of discharge and the required dilution to maintain acceptable concentration in compliance with local requirements. Therefore, DSER COL Action Item 11.2-2 is acceptable and Open Item 11.2-2 is closed.

The discharge flow rate for any waste stream should be restricted, as necessary, to maintain an acceptable level for concentrations of radionuclides in liquid effluents in any unrestricted area, when the waste discharge flow is diluted by the circulating water blowdown flow of 13.25 m^3/minute (3500 gal/min). The above criterion for liquid waste discharge flow complies with 10 CFR Part 20, Appendix B, Table 2, Column 2, limits for concentrations of radionuclides in liquid effluents in any unrestricted area. All WLS discharges are made through a single liquid waste discharge line to the circulating water blowdown stream. The dilution factor provided for the activity released is site dependent and is provided by the COL applicant.

All WLS releases are monitored, prior to dilution and discharge, by a radiation monitor. The monitor is located on the common discharge line downstream of the WLS monitor tanks to comply with 10 CFR Part 20, Appendix B, Table 2, Column 2, limits for liquid effluent concentrations of radionuclides in unrestricted areas. These radiation monitors will terminate liquid radwaste releases to unrestricted areas before the discharge concentration in the line exceeds a predetermined setpoint. As discussed above, the radiation monitors are provided for controlling and monitoring release of radioactive materials from liquid effluents to unrestricted areas as required by GDC 60 and 64. The operational setpoints of these monitors should ensure that the sum of the ratios of instantaneous discharge concentrations of radionuclides in an unrestricted area to the liquid effluent concentration limits given for the corresponding radionuclides in the table column referenced above does not exceed a factor of 10. Therefore, the staff will review the operational setpoints of the subject radiation monitors on a plant-specific basis for each COL application. A COL applicant referencing the AP600 certified design should identify the planned operational setpoints for its WLS radiation monitors in its plant-specific offsite dose calculation manual (ODCM). This was identified as DSER COL Action Item 11.2-3 and Open Item 11.2-3. Westinghouse revised SSAR Section 11.5.7 and Table 1.8-2 to require COL applicant to develop an ODCM to address operational setpoints for the radiation monitors, and to address programs for monitoring and controlling the release of radioactive material into the environment, which eliminates the potential for unmonitored and uncontrolled release. Therefore, DSER COL Action Item 11.2-3 is acceptable and Open Item 11.2-3 is closed. COL Action Item 11.2-3 is identified in SSAR Table 1.8-2 as part of Item Number 11.5-1.

Westinghouse calculated the annual liquid effluent releases (shown in Table 11.2-7 of the SSAR) using the PWR GALE code methodology. The standard design parameters for running this computer program to calculate expected primary and secondary coolant radionuclide concentrations and liquid effluents are provided in Table 11.2-6 of the SSAR. Table 11.2-2 of the SSAR lists the component data for the WLS. Specifically, the table lists the number of WLS holdup tanks, monitor tanks, pumps, filters and ion exchangers (and their types), and their design capacities or flow rates, whichever is applicable. Table 11.2-1 of the SSAR lists the collection rates and primary coolant activity (PCA) fractions of the individual liquid waste streams. Table 11.2-5 of the SSAR gives the decontamination factors (DFs) for different categories of radionuclides provided by the different types of ion exchangers. Figure 11.2-2 of the SSAR gives the WLS piping and instrumentation drawings (P&IDs). In the DSER, the staff stated that Westinghouse had not responded to staff RAI 460.25, which dealt with the rerun of the GALE code associated with proper input data. This was identified as DSER Open Item 11.2-4. Subsequently, by a letter dated October 23, 1996, Westinghouse revised its response to RAI 460.25(R1), and in Revision 8 to the SSAR, Westinghouse provided the results of the GALE code rerun in SSAR Tables 11.2-7, 11.2-8, and 11.2-9. The GALE code methodology is recommended in SRP 11.2, and the input data are reviewed and found acceptable. Therefore, DSER Open Item 11.2-4 is closed.

Because demonstration of specific compliance with 10 CFR Part 50, Appendix I dose guidelines for liquid effluents is not within the scope of the standard design, the staff will review each compliance demonstration for each COL application. Section 11.2.5.2 of the SSAR states that the calculation of offsite individual doses resulting from liquid effluents is the responsibility of the COL applicant. The staff agrees with Westinghouse on its approach in this regard. However, the staff found in the DSER that in the response to RAI 460.18 (Revision 0), dealing with COL action items, Westinghouse stated that the cost-benefit analysis for population doses resulting from liquid effluents was not needed and that the cost-benefit analysis provided in Appendix 1B of the SSAR should be sufficient to preclude the need for a site-specific cost-benefit analysis in most cases. The staff reviewed Appendix 1B and found that it did not support Westinghouse's conclusion. A COL applicant referencing the AP600 certified design should demonstrate compliance with 10 CFR Part 50, Appendix I requirements for offsite individual doses and population doses resulting from liquid effluents. RG 1.110 provides guidance for performing a cost-benefit analysis in order to reduce cumulative dose to the population by using available technology. This COL action was previously identified, in Section 11.1 of the DSER, as part of DSER COL Action Item 11.1-1. Subsequently, Westinghouse revised the response to RAI 410.18 (Revision 1), SSAR Section 11.2.5.2 and Table 1.8-2, stating that the COL applicant will provide a site specific cost-benefit analysis to address the requirements of 10 CFR 50, Appendix I, regarding population doses resulting from liquid effluents. Therefore, this part of DSER COL Action Item 11.1-1 is acceptable. The compliance with RG 1.110 as it relates to performing a site-specific cost-benefit analysis for reducing dose will be demonstrated by a COL applicant. COL Action Item 11.1-1 is identified in SSAR Table 1.8-2 as Item Number 11.2-2.

Table 11.2-8 of the SSAR shows that the sum of the ratios of the liquid effluent concentrations of radionuclides in any unrestricted area to the liquid effluent concentration limits for the respective radionuclides given in 10 CFR Part 20, Appendix B, Table 2, Column 2, are well below 1.0. In the DSER, the staff indicated that Westinghouse had not responded to RAI 460.21, which dealt with AP600 WLS compliance with 10 CFR Part 20, Section 1302. Section 1302 requires that the annual average concentrations of radioactive materials in liquid effluents in an unrestricted area do not exceed the limits specified in the subject table column. In RAI 460.21, the staff specifically stated that SSAR Table 11.2-8 should be revised based on (1) primary coolant concentration of fission products corresponding to 1.0 percent fuel failure instead of 0.25 percent, and (2) annual average concentrations of radionuclides in liquid effluents in unrestricted areas. This was identified as DSER Open Item 11.2-5. In a letter dated October 23, 1996, Westinghouse revised its response to RAI 460.21 (Revision 1) and SSAR Table 11.2-9 to show the results assuming a maximum defined fuel defect level, which corresponds to 1.0 percent fuel defects for all fission product nuclides except iodine and noble gas. For iodine and noble gas, the Technical Specification limits, corresponding to 0.25 percent fuel defect, were assumed. The results in SSAR Table 11.2-9 demonstrate that the sum of the ratios of the liquid effluent concentrations of radionuclides in any unrestricted area to the liquid effluent concentration limit for the respective radionuclides given in 10 CFR Part 20, Appendix B, Table 2, Column 2, is 0.36, still well below 1.0. Therefore, compliance with 10 CFR Part 20, Section 1302 is demonstrated by AP600 and DSER Open Item 11.2-5 is closed.

Subsequently, the staff found that SSAR Section 11.1 had not been revised to reflect the 1.0 percent fuel failure assumption even though the analysis in SSAR Section 11.2 used a 1.0 percent fuel failure assumption. In response, Westinghouse clarified its position that the 0.25 percent fuel defect assumption is still used as the AP600 design basis for the waste management systems. However, the above SSAR analysis demonstrates that the AP600 liquid waste system has a sufficient margin such that the system capability is able to process liquid waste at 1.0 percent fuel defect in meeting the requirements of 10 CFR Part 20, Section 1302. Therefore, the staff finds that Westinghouse's position for the liquid waste system is acceptable because SRP Section 11.2 regarding fuel failure assumption of 1.0 percent for the liquid waste system is satisfied.

The WLS is a non-safety-related system and serves no safety function except system isolation from the containment, when required. The system interface valves with the containment, which serves the above safety function, are safety-related and seismic Category I, as shown in SSAR Table 3.2-3. In SSAR Table 3.2-3, Westinghouse indicates that the WLS is located in the seismic Category I auxiliary building. All waste collection and waste monitor tanks, the chemical waste tank and the condensate storage tank, are equipped with level indication and provisions for high-level alarms in the control room. Local indication and control are available on portable displays, which may be connected to the data display and processing system.

The staff reviewed SSAR Section 11.2 for the compliance with RG 1.143 and identified several questions in RAI 460.20. In its response (Revision 0) to RAI 460.20, which deals with the system's compliance with RG 1.143, "Design Guidance for Radwaste Management Systems, Structures, and Components Installed in Light Water-Cooled Nuclear Power Plants," Westinghouse states that the components in the WLS are designed and tested to the guidelines set forth in the codes and standards listed in Table 1 of RG 1.143, as supplemented by guidelines 1.1.2 and 1.1.4 of RG 1.143. The staff finds the above WLS design information useful for its evaluation of the system. However, the staff noted that Westinghouse had not fully responded to the staff's question pertaining to the WLS's compliance with the guidelines of RG 1.143. Radwaste systems should be designed in accordance with the guidelines of RG 1.143 to comply with GDC 60 and 61 and 10 CFR 50.34a. Westinghouse's response to RAI 460.20 (Revision 0) did not clarify whether indoor WLS tanks will have curbs or thresholds with floor drains routed to the WLS for treatment. This was identified as DSER Open Item 11.2-6.

In Response (Revision 1) to RAI 460.20, dated October 23, 1996, Westinghouse stated that the revised Appendix 1A, "Conformance with Regulatory Guides," of the SSAR (Revision 7) provided a discussion of the compliance with RG 1.143. In the appendix, Westinghouse committed to comply with RG 1.143 Regulatory Positions C.1.1.1 through C.1.1.4, and C.1.2.1 through C.1.2.5. The tanks of the WLS (effluent holdup tanks, waste holdup tanks, monitor tanks, and chemical waste tank) are located in the auxiliary building, which is designed to seismic Category I criteria. The other components, such as ion-exchangers, filters, degasifier, pumps, applicable valves, and heat exchangers, are in the auxiliary building. All WLS tank overflows are routed to a water-tight room within the auxiliary building and drained to the auxiliary building sump, which is pumped to a waste holdup tank. Therefore, the staff finds the WLS acceptable with respect to meeting the seismic design guidance specified in RG 1.143.

Components (such as heat exchangers, pumps, tanks, degasifier, ion exchangers, filters, and valves) in the WLSs are non-seismic, and are classified as the AP600 Class D (i.e., Quality Group D in RG 1.26) as shown in SSAR Table 3.2-3. The quality assurance (QA) program applied to the radioactive waste systems is in accordance with the overall QA program described in SSAR Chapter 17. SSAR Section 17.4 states that the COL applicant will address its Quality Assurance program. Therefore, the staff finds the WLS acceptable with respect to meeting the QA guidance specified in RG 1.143. On the basis of the above, the staff concludes that DSER Open Item 11.2-6 as it relates to the WLS's compliance with RG 1.143 is closed.

The liquid waste system is designed to handle most liquid effluents and other anticipated events using installed equipment. However, for events occurring at a very low frequency or producing effluents not compatible with the installed equipment, temporary equipment may be brought into the radwaste building mobile treatment facility truck bays. Connections are provided to and from various locations in the liquid waste system to these mobile equipment connections. This allows the mobile equipment to be used in series with installed equipment, as an alternate to it with the treated liquids returned to the liquid waste system, or as an ultimate disposal point. The staff will review any mobile processing equipment that may be used for processing liquid radwaste on a plant-specific basis for applicable COL applications, against the guidelines of RG 1.143. The COL applicant should discuss how any mobile processing equipment, intended for use in the processing of liquid radwaste, meets the guidelines of RG 1.143. This was identified as DSER COL Action Item 11.2-4 and Open Item 11.2-7. SSAR Section 11.2.4.1 (Revision 6) and Table 1.8-2 address this COL action item. Therefore, DSER COL Action Item 11.2-4 is acceptable and Open Item 11.2-7 is closed. By meeting the guidance in RG 1.143 and RG 1.110, the WLS meets the requirements of GDC 61 as specified in SRP Section 10.2.

Westinghouse has reviewed the applicability of NRC Office of Inspection and Enforcement (IE) Bulletin (IEB) 80-05, "Vacuum Conditions Resulting in Damage to CVS Holdup Tanks (sometimes called Clean Waste Receiver Tanks)," 1980, to the AP600 standard design. IEB 80-05 addresses the issues concerning the release of radioactive material or other adverse effects as a result of low vacuum conditions causing tank buckling. The low-vacuum condition is created by the cooling of hot water in a low-pressure tank. On the basis of its review, Westinghouse concluded, in its response to RAI 460.17 (dated June 30, 1994), that except for the reactor coolant drain tank (RCDT) located in the containment building, no other tank in the WLS is exposed to hot water. Westinghouse added that the RCDT has several design features, including an external design pressure of 204.8 kPa (15 psig), which eliminate the possibility of structural collapse of the RCDT resulting from steam condensation. Westinghouse also stated that, because of these design features, the RCDT will not collapse even if it is exposed to a full vacuum. The staff does not consider the RCDT to be part of the WLS. However, the staff notes that all of the WLS tanks have vents that are adequately sized to prevent tank collapse during drain down. The staff finds that the design of the WLS adequately addresses the concern identified in IEB 80-05 and is, therefore, acceptable. This design complies with GDC 61, because it ensures adequate safety for the WLS tanks under normal and postulated accident conditions.

11.2.2 Conclusion

The staff concludes that the design of the WLS is acceptable and meets the requirements of 10 CFR 20.1302, GDC 60, 61, and 64, 10 CFR Part 50, Appendix I, and 10 CFR 50.34a. This conclusion is based on the following:

- A COL applicant referencing the AP600 certified design will demonstrate compliance with 10 CFR Part 50 Appendix I requirements for offsite individual doses and population doses resulting from liquid effluents by a site-specific cost-benefit analysis in accordance with RG 1.110. This is discussed in the resolution of DSER COL Action Item 11.1-1 in Section 11.2.1.

- AP600 has met the requirements of 10 CFR 20.1302 that the annual average concentration of radioactive materials in liquid effluents in an unrestricted area do not exceed the limits specified in 10 CFR Part 20, Appendix B, Table 2, Column 2. This is discussed in the resolution of DSER Open Item 11.2-5 in Section 11.2.1.

- AP600 has met the requirements of GDC 60 and 64 with respect to controlling and monitoring releases of radioactive material to the environment by radiation monitoring the WLS releases. All WLS releases are monitored by a radiation monitor, which will terminate liquid radwaste releases before the discharge concentration exceeds a predetermined set point. A COL applicant will identify the operational setpoint for its WLS radiation monitors in its plant-specific ODCM as discussed in the resolution of DSER COL Action Item 11.2-3 in Section 11.2.1.

- The compliance with the requirements of GDC 61 is demonstrated by meeting the guidelines of RG 1.143 and RG 1.110 as discussed in the resolution of DSER Open Items 11.2-6, 11.2-7, and DSER COL Action Item 11.1-1 in Section 11.2.1.

- The compliance with 10 CFR 50.34a as it relates to sufficient design information being provided is demonstrated by the above discussion.

The staff has identified two figures in SSAR Section 11.2 that need to be changed. Figure 11.2-1 is missing a flow path and Figure 11.2-2, sheet 1 has illegible text, Westinghouse has agreed to correct these figures in the next SSAR amendment. This is Confirmatory Item 11.2-1. Figures 11.2-1 and 11.2-2, Sheet 1, were corrected in SSAR Revision 23 and 24, respectively. Therefore, Confirmatory Item 11.2-1 is closed. Based on the above review and the closure of Confirmatory Item 11.2-1, the staff has determined that the AP600 WLS design meets the guidelines of SRP Section 11.2. Therefore, it is acceptable.

11.3 <u>Gaseous Waste Management System</u>

11.3.1 System Description and Review Discussion

The WGS controls, collects, processes, stores, and disposes of gaseous radioactive wastes generated during normal operation, including AOOs. The staff used the acceptance criteria provided in Section 11.3 of the SRP and the guidelines of BTP ETSB 11-5 to evaluate system compliance with 10 CFR 50.34a; 10 CFR 20.1302; 10 CFR Part 50, Appendix A, GDC 3, 60, and 61; and 10 CFR Part 50, Appendix I.

10 CFR 50.34a requires that sufficient design information be provided to demonstrate that the design objectives for equipment necessary to control releases of radioactive effluents to the environment have been met. The staff used 10 CFR 20.1302, which defines the criteria for radionuclide concentration limits in liquid and gaseous effluents in unrestricted areas, as one of the current evaluation requirements.

BTP ETSB 11-5 provides guidelines to analyze postulated radioactive releases due to a waste gas system leak of failure. GDC 3 relates to providing protection to gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen. GDC 60 relates to the radioactive waste management systems designed to control releases of radioactive materials to the environment. GDC 61 relates to radioactivity control in gaseous waste management systems and ventilation systems associated with fuel storage and handling areas. 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D relate to the numerical guides for dose design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion.

The relevant requirements of the regulations identified above are met by using the regulatory positions contained in the following regulatory guides:

- Regulatory Guide 1.140, as it relates to the design, testing, and maintenance of normal ventilation exhaust systems at nuclear power plants.

- Regulatory Guide 1.143, as it relates to the seismic design and quality group classification of components used in the gaseous waste treatment system and structures housing systems and the provisions used to control leakages.

The WGS involves the gaseous radwaste system, which deals with the management of potentially hydrogen-bearing and radioactive gases generated during plant operation. Additionally, it involves the management of building ventilation, containment purge, and condenser air removal system exhausts, as they relate to gaseous effluents to the environs.

The AP600 WGS is a once-through, ambient-temperature, activated carbon delay system. The system includes a gas cooler, a moisture separator, an activated carbon-filled guard bed, and two activated carbon-filled delay beds. Also, included in the system are an oxygen analyzer subsystem and a gas sampling subsystem. The major inputs to the WGS are RCS gases stripped from the CVS letdown flow by the WLS vacuum degasifier during RCS dilution and boration and during degassing prior to a reactor shutdown. Other inputs to the WGS are the gases from the RCDT vent, and the gases stripped from RCDT liquid by the WLS degasifier.

The flow through the WGS consists of hydrogen and nitrogen (as carrier gases), fission gases, and water vapor. Influent to the WGS passes through the following stages:

(1) a gas cooler, which cools the influent waste gas to 7°C (45°F) by a chilled water system

(2) a moisture separator, which removes the moisture formed when gas stream is cooled

(3) a guard bed, which protects the delay beds from abnormal moisture carryover, or chemical contaminants, by removing them from the waste stream

(4) two 100-percent capacity delay beds

The fission gases in the waste gas stream undergo dynamic adsorption by the activated carbon in the delay beds and, therefore, experience significant delay during their transit through the beds. Radioactive decay of the fission gases during the delay periods significantly reduces the radioactivity of the gas flow leaving the system. The effluent from the delay bed passes through a radiation monitor and is discharged to the environs, via the system ventilation exhaust duct and the plant vent.

The design data for the WGS are provided in Table 11.3-2 of the SSAR. Westinghouse did not provide a complete response to RAI 460.10(c) regarding information on monitoring instrumentation for the WGS. This was identified as DSER Open Item 11.3-1. By letter dated October 23, 1996, Westinghouse indicated in Revision 2 to the response to RAI 460.10 that a list of gaseous radwaste system instrumentation and control items was provided in Revision 8 to SSAR Table 11.3-2. The system contains provisions for continuously monitoring the moisture level at the inlet of the guard bed. Monitoring performance of individual components in the system is done by collecting and analyzing grab samples. Connections between the two delay beds allow for the collection of samples at the inlet and outlet of the guard bed, and at the outlet of the second delay bed. Therefore, the staff finds DSER Open Item 11.3-1 regarding monitoring instrumentation is closed.

The WGS has a radiation monitor, which continuously monitors the discharge from the delay beds. The monitor will automatically terminate the discharge when the radiation level in the discharge stream reaches a predetermined setpoint, which will be determined by the COL applicant. A COL applicant referencing the AP600 certified design should identify its planned operational setpoint for the WGS radiation monitor in its plant-specific ODCM. This was identified as DSER COL Action Item 11.3-1 and Open Item 11.3-2. In Revision 6 to SSAR Section 11.5.7, Westinghouse stated that the COL applicant will identify operational setpoints for the radiation monitors and identify programs for monitoring and controlling the release of radioactive material to the environment, which will eliminate the potential for unmonitored and uncontrolled release. Therefore, COL Action Item 11.3-1 is acceptable and Open Item 11.3-2 is closed. COL Action Item 11.3-1 is identified in SSAR Table 1.8-2 as part of Item Number 11.5-1.

In the DSER the staff identified in Open Item 11.3-3 that RAI 410.10 pertaining to the calculation of delay times for Xenon and Krypton was not answered satisfactorily. By letter dated October 23, 1996, Westinghouse revised its response to RAI 410.10 in Revision 2 and clarified that the revised delay times, which were based on a test report (ORNL CF59-6-47) and shown in SSAR (Revision 8) Table 11.2-6, were used in running the GALE Code. Based on the test data, the staff finds the revised response dealing with the delay times acceptable. Therefore, DSER Open Item 11.3-3 is closed. By providing carbon delay beds, the AP600 WGS design is consistent with GDC 60 with regard to the control of radioactive releases to unrestricted areas.

The WGS is a non-safety-related system and has no accident mitigation functions. According to SSAR Table 3.2.3, the WGS is located in the auxiliary building, which is a seismic Category I structure and, therefore, designed to withstand a safe-shutdown earthquake (SSE). The WGS and the structure housing the system are designed in accordance with the applicable Positions C.2, C.4, C.5, and C.6 of RG 1.143 with respect to the following specific guidelines for gaseous radwaste systems:

- general guidelines for design, construction, and testing criteria for radwaste systems
- specific seismic design criteria for WGS
- general seismic design criteria for structures housing radwaste systems
- general guidelines for providing QA for radwaste management systems

Appendix 1A of the SSAR, and Westinghouse's response to RAI 460.20 (Revision 0), pertaining to the AP600 radwaste management systems' conformance with RG 1.143, provides a detailed discussion of how the design of the WGS and its housing structure meet the applicable guidelines of RG 1.143. Specifically, the response states that the WGS components were designed and tested to the requirements set forth in the codes and standards listed in Table 1 of RG 1.143, as supplemented by Guideline 2.1.2 of RG 1.143. By telephone conversation on August 9, 1994, Westinghouse stated that the guard bed and delay beds, and the supports for the beds, were designed to seismic Category I standards. The staff indicated in the DSER that Westinghouse should document this seismic design certification. This was identified as DSER Confirmatory Item 11.3-1. Subsequently, in Revision 9 to the SSAR, Appendix 1A, C.2.1.3 of RG 1.143, Westinghouse further clarified that the guard bed and the delay beds, including supports, in the gaseous radwaste system are designed for seismic loads according to RG 1.143, not for seismic Category I. Seismic loads for this equipment will be established using one-half of the SSE floor response spectra. The loads resulting from this seismic response spectra are equivalent or greater than those resulting from an operating-basis earthquake (OBE). The WGS is housed in a seismic Category I structure (the auxiliary building) and the guard bed, absorber beds, and their supports are also seismically qualified in accordance with Position C.5 of RG 1.143, with respect to seismic design criteria for the WGS. Therefore, DSER Confirmatory Item 11.3-1 is closed.

The response of RAI 410.20 (Revision 0) also stated that the QA program for the radwaste systems meets the requirements of American Society of Mechanical Engineers (ASME) NQA-1-1989 Edition through NQA-1b-1991 Addenda and, therefore, Position C.6 of RG 1.143. However, because the installation, procurement, and fabrication of the system components are within the scope of the COL applicant, the staff will review the specifics of the QA program for each COL application. A COL applicant referencing the AP600 certified design should provide details of its proposed QA program as it pertains to the installation, procurement, and fabrication of the WGS components. This was identified as DSER COL Action Item 11.3-2 and Open Item 11.3-4. Revision 8 to the SSAR states that the QA program for design, fabrication, procurement, and installation of the gaseous radwaste system is in accordance with the overall QA program described in Chapter 17. SSAR Chapter 17.4 states that the COL applicant will address its QA program for design, procurement, fabrication, installation and construction, and operation. Therefore, COL Action Item 11.3-2 is acceptable and

Open Item 11.3-4 is closed. COL Action Item 11.3-2 is identified in SSAR Table 1.8-2 as part of Item Number 17.5-2. In SSAR Table 3.2-3, Westinghouse classifies WGS equipment and components such as gas cooler, sample pumps, guard and delay beds, moisture separator, and applicable valves as the AP600 Class D (i.e., RG 1.26 Quality Group D).

On the basis of the above discussion, the staff finds that the AP600 WGS meets the applicable guidelines of RG 1.143, and thus complies with GDC 60 and 61, as well as 10 CFR 50.34a, as they relate to controlling radioactive materials released to the environment via gaseous effluents, ensuring adequate safety under normal and postulated accident conditions for equipment that may contain gaseous radwaste, and providing adequate design information for the WGS, respectively.

Because the potential exists for a buildup of explosive mixtures of hydrogen and oxygen in the WGS, as per SRP Section 11.3 guidelines, the system should be designed to either withstand the effects of a hydrogen explosion, or have design features to preclude the formation or buildup of explosive mixtures. Revision 2 to the response to RAI 460.4(c) and Section 11.3.1.2.3.1 of the SSAR, describe the design features for preventing the formation or buildup of explosive mixtures in the WGS. The WGS operates at a slightly positive pressure to prevent air in-leakage. A continuous purge flow of nitrogen is provided at the outlet of the WGS to prevent back-leakage of air through the discharge check valves.

Dual oxygen analyzers are provided for continuous sampling in a side stream taken off the process flow paths. These analyzers sound an alarm both locally, and in the main control room (MCR), upon high oxygen level. The alarm setpoint is at an oxygen concentration level that would allow adequate time for operator action. A hydrogen analyzer is also provided for direct measurement of hydrogen concentration in the sampling side stream. The operator can use the above analyzer reading in conjunction with a flammability chart to assess the flammability potential during an upset situation that permits oxygen into the system. A hydrogen monitor also samples the ambient environment of the charcoal delay bed vault. This monitor sounds an alarm both locally, and in the MCR, upon high hydrogen level (usually set at 1 percent). The entire system is electrically at the same potential, thereby eliminating the buildup of static electricity and sparking. The staff found the above information useful for the DSER in evaluating the design features for preventing the formation or buildup of explosive mixtures in the WGS. However, it was not clear whether the dual oxygen analyzers were independent and, therefore, can provide independent measurements of oxygen concentrations in the WGS process stream upstream of the charcoal beds. Further, the staff considered that the WGS should be designed such that the oxygen source gets automatically isolated and nitrogen gets automatically injected upon a high-high oxygen alarm setting (4-percent oxygen concentration). This was identified in the DSER as Open Item 11.3-5 as related to Acceptance Criterion II.B.6 of SRP Section 11.3, and GDC 3 for the protection from the potential effects of an explosive mixture. Subsequently, in SSAR Section 11.3.1.2.3.1, Westinghouse stated that the dual oxygen analyzers are independent and that at an operator selectable oxygen concentration of 4 percent or less, the systems automatically isolate oxygen inputs and initiate a nitrogen purge. DSER Open Item 11.3-5 is closed.

Besides the WGS exhaust, the other exhaust released to the environs via the radiation monitored plant vents include:

- the containment purge exhaust
- the auxiliary building exhaust
- the annex building release
- the radwaste building exhaust

Except for the containment purge and radwaste building exhausts, these exhausts are not filtered prior to their release to the environs. The radwaste building exhaust goes through a mobile high-efficiency particulate air (HEPA) filter to the plant vent in accordance with RG 1.143. However, as stated in the response to RAI 460.4(d) (Revision 2), Westinghouse has not taken credit for filtration of this airborne release, because this exhaust contributes a small fraction of the normal airborne releases. The containment purge exhaust is filtered by the containment air filtration system (VFS) HEPA filters and a 10.2-cm (4-in) thick charcoal absorber (see Section 9.4.7 and Table 9.4-1 of the SSAR for information on the VFS).

The turbine steam sealing (gland seal) system exhaust, and the condenser air removal system exhaust, which includes the gland seal exhaust during the plant startup, are routed to a common header that discharges the exhausts to the environs via a radiation-monitored turbine building vent. The gland seal system and condenser air removal system exhausts are not filtered prior to their release to the environs, as they are not normally radioactive. However, upon detection of unacceptable levels of radiation in the exhausts, which may occur as a result of a steam generator tube leak, appropriate corrective actions are manually performed. (See Sections 10.4.2.2.1 and 10.4.3.2.2 of the SSAR.) The turbine building exhaust is released to the environs via unmonitored turbine building vents, because it is not expected to have detectable radioactivity. In Revision 1 to its response to RAI 460.4(a), Westinghouse provided release point characteristics for the plant vent and the turbine building vent, through which the combined discharge of the condenser air removal system and the gland seal system occurs. The staff reviewed the above information, and finds that it complies with 10 CFR 34a, as it relates to the adequacy of design information provided by the applicant, and is therefore acceptable.

In Revision 2 to its response to RAI 460.4(d), which deals with the air filtration systems, and Appendix 1A of the SSAR, Westinghouse provided a detailed discussion of how the VFS meets the guidelines of RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants." The staff reviewed the response and Appendix 1A and finds that the VFS has a 10.2 cm (4 in) charcoal absorber, and HEPA filters upstream and downstream of the absorber and that the VFS meets the RG 1.140 guidelines. The staff finds that Westinghouse has credited filter efficiencies of 90 and 99 percent for removal of iodine and other radionuclides in particulate form, respectively, in its calculation of gaseous effluents from containment purge exhaust. These efficiencies are in accordance with the specified efficiencies for 10.2 cm (4 in) charcoal absorber and HEPA filters in RG 1.140. Therefore, the staff agrees with the filter efficiencies credited by Westinghouse in the calculation of containment purge exhaust effluents.

Although the auxiliary building and annex building exhausts are not normally filtered prior to their release, as stated above, the ventilation systems serving the above areas incorporate design features that provide automatic filtration of the exhausts, prior to their release, under certain circumstances. Specifically, a high radiation signal, from any of the monitors in the exhaust ducts of the annex building, the fuel handling area of the auxiliary building and the radiologically controlled portion of the auxiliary building, will isolate the normal supply and (unfiltered) exhaust ducts to the affected area, and connect the VFS exhaust filter and fans to the isolated area. (See Section 9.4.3.2.4 of the SSAR, and Westinghouse's response to RAI 460.13). On the basis of the above discussion, the staff finds that the WGS complies with GDCs 60 and 61, as they relate to control of release of radioactive material from plant areas, including the fuel storage and handling area, to the environs.

Westinghouse calculated the annual gaseous effluent releases (Table 11.3-3 of the SSAR) using the PWR-GALE code. The standard design parameters for running the computer program, which calculates expected primary and secondary coolant radionuclide concentrations and gaseous effluents, are provided in Table 11.2-6 of the SSAR. Tables 11.3-1 and 11.3-2 of the SSAR list the design data for the WGS. Figures 11.3-1 and 11.3-2 of the SSAR depict the schematics and the P&ID of the WGS. When the DSER was prepared, Westinghouse did not respond to RAI 460.25, pertaining to the rerun of the PWR-GALE code. This was identified as DSER Open Item 11.2-4. In a revised response to RAI 460.25 (Revision 1) and in SSAR Revision 8, Westinghouse provided the results of the PWR-GALE code rerun in Table 11.3-3. Therefore, DSER Open Item 11.2-4 is closed.

Demonstration of specific compliance of 10 CFR Part 50, Appendix I requirements for maximally exposed offsite individual and population doses resulting from gaseous effluents is not within the scope of the AP600 standard design. The staff will review demonstration of such compliance for each COL application. This COL action was previously identified in Section 11.1 of this report as a part of COL Action Item 11.1-1. Section 11.5.7 of the SSAR states that the calculation of offsite individual doses is the responsibility of the COL applicant. The staff agrees with Westinghouse's approach in this regard. However, in response to RAI 460.18 (Revision 0), which deals with COL action items, Westinghouse indicated that the cost-benefit analysis for population doses resulting from gaseous effluents during normal plant operation, including AOOs, need not be performed. Westinghouse stated that the cost-benefit analysis provided in Appendix 1B of the SSAR should be sufficient to preclude the need for a site-specific cost-benefit analysis in most cases. In the DSER, the staff reviewed Appendix 1B and found that it did not support Westinghouse's conclusion. Therefore, Westinghouse should identify COL Action Item 11.1-1, related to demonstrating compliance with 10 CFR Part 50, Appendix I guidelines for *both* individual and population doses resulting from gaseous effluents, in the SSAR. This was previously identified in Section 11.1 of this report as part of Open Item 11.1-1. Subsequently, Westinghouse revised SSAR Section 11.3.5.1 and Table 1.8-2, stating that the COL applicant will provide a site specific cost-benefit analysis to address the requirements of 10 CFR 50, Appendix I, regarding population doses resulting from gaseous effluents. The staff finds this COL action acceptable for the compliance with 10 CFR Part 50, Appendix I. Therefore, this part of DSER Open Item 11.1-1 is closed.

Table 11.3-4 of the SSAR gives the ratios of the gaseous effluent concentrations of noble gas and iodine radionuclides at the site boundary to the gaseous effluent concentration limits for the respective radionuclides given in 10 CFR Part 20, Appendix B, Table 2, Column 1. For calculating these ratios, Westinghouse used a site χ/Q of 2×10^{-5} sec/m^3. Table 11.3-4 of the SSAR did not include carbon-14, tritium, argon-41, and other radionuclides in particulate form. Further, the table did not sum the ratios to demonstrate that the sum is below 1.0. This was identified as DSER Open Item 11.3-6. In Revision 12 to the SSAR, Westinghouse revised Table 11.3-4 to include the missing radionuclides and the sum of 0.17. Therefore, DSER Open Item 11.3-6 is closed. On the basis of the results in Table 11.3-4, the staff finds that the WGS design complies with 10 CFR Part 20, Section 1302.

In the DSER, the staff found that Westinghouse did not respond to RAI 460.21 pertaining to site boundary effluent concentrations using proper fuel failure assumption to comply with 10 CFR Part 20 limits. This was identified as DSER Open Item 11.3-7. On October 23, 1996, Westinghouse provided the revised response to RAI 460.21 to address the staff concern of the assumption of 1 percent fuel failure and provided the results in the revised SSAR Table 11.3-4. The staff reviewed the above results based on one percent fuel failure and finds them acceptable. Therefore, DSER Open Item 11.3-7 is closed.

In Revision 1 of its response to RAI 460.4(b), Westinghouse provided a waste gas system leak or failure analysis and the justification for the assumptions used in that analysis. The analysis was performed to demonstrate that the WGS design meets the applicable guidelines of BTP ETSB 11-5. The BTP stipulates that the total body dose at the exclusion area boundary (EAB), as a result of the release of radioactivity for two hours from a postulated failure of the WGS, calculated in accordance with the BTP assumptions, should not exceed 5 mSv (500 mrem). Westinghouse analyzed the accident using a short-term (0-2 hours) χ/Q of 1×10^{-3} sec/m^3 at the EAB, a release duration of one hour instead of two hours, as suggested by the BTP, and other assumptions which agreed with those in the BTP. Westinghouse states in the above response that a release duration of two hours, as suggested by the BTP, is not appropriate for the AP600 design. Westinghouse calculated a 0- to 2-hour total body dose within 5 mSv (500 mrem). The staff independently analyzed the above failure, using all the BTP assumptions, including the release duration and the same EAB χ/Q as used by Westinghouse, and has determined that the 0- to 2-hour total body dose at the EAB will be within 5 mSv (500 mrem). Therefore, the staff finds the analysis acceptable.

11.3.2 Conclusion

In its evaluation of the WGS design, the staff considered the following acceptance criteria:

- the capability of the system to maintain gaseous effluents in unrestricted areas below the limits stated in 10 CFR 20.1302, during periods of fission product leakage at design levels for the fuel (1.0 percent). (see the resolution of DSER Open Items 11.3-6 and 11.3-7)

- the design features incorporated to control the release of radioactive materials to the environs via gaseous effluents in accordance with GDC 60. (see the resolution of DSER Open Items 11.3-2, 11.3-3, 11.3-4, and compliance with RG 1.140, and RG 1.143)

- the design features incorporated in compliance with GDC 61, as it relates to radioactivity control in WGS and ventilation systems associated with fuel storage and handling areas (see the compliance with RG 1.140 and RG 1.143

- the design features incorporated to comply with GDC 3, as it relates to protecting the WGS from the effects of an explosive mixture of hydrogen and oxygen (see the resolution of DSER Open Item 11.3-5)

- the capability of the design features to ensure WGS compliance with BTP ETSB 11-5 guidelines in the analysis of a postulated waste gas system leak or failure (see the resolution of RAI 460.4(b))

The staff reviewed all applicable information submitted in Section 11.3 of the SSAR and in Westinghouse's responses to staff RAIs related to radwaste management systems for the AP600. On the basis of its review, as discussed in Section 11.3.1 of this report regarding meeting RG 1.140 and RG 1.143 guidance, the staff concludes that the AP600 design complies with GDC 60 and 61. The staff further concludes that the WGS complies with BTP ETSB 11-5 guidelines. As discussed in Section 11.3.1 of this report, Westinghouse responded to the staff's questions relating to delay times for xenon and krypton radionuclides in charcoal delay beds, instrumentation for the WGS, gaseous effluent releases, and compliance with 10 CFR Section 20.1302. Also, the staff finds that Westinghouse's responses to staff questions relating to WGS compliance with GDC 3, and the identification of a COL action item for demonstrating WGS compliance with 10 CFR Part 50, Appendix I guidelines for population doses are acceptable. On the basis of the above, the staff finds the AP600 WGS to be in compliance with 10 CFR 20.1302 and 10 CFR 50.34a. Therefore, on the basis of its evaluation of the WGS in accordance with the applicable acceptance criteria of Section 11.3 of the SRP, the staff finds the WGS to be acceptable.

11.4 Solid Waste Management System

11.4.1 System Description and Review Discussion

The WSS consists of equipment and instrumentation to collect, segregate, store, process, sample, and monitor solid wastes. For its evaluation of the WSS, the staff used the acceptance criteria in Section 11.4 of the SRP to assess system compliance with the following regulations:

- 10 CFR 50.34a, as it relates to providing adequate system design information

- 10 CFR 20.1302, as it relates to ensuring that concentrations of radionuclides in gaseous and liquid effluents to unrestricted areas arising from WSS operation are within the limits specified in 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2, respectively

- 10 CFR Part 50, Appendix A, GDC 60 and 64, as they relate to controlling and monitoring the release of radioactive materials to the environment

- GDC 63, as it relates to monitoring radiation levels and leakage

- 10 CFR Part 71, as it relates to packaging of radioactive materials

- 10 CFR Part 61, as it relates to classifying, processing, and disposing of solid wastes

The relevant requirements identified above are reviewed using the regulatory positions identified in RG 1.143, as it relates to the seismic design and quality group classification of components used in the WSS and structures housing the systems and the provisions to control leakages.

The WSS is designed to collect and accumulate wet solid wastes (e.g., spent ion exchange resins, deep bed filtration media, filter cartridges), dry active wastes (e.g., rugs, paper, clothing, HVAC filters), and mixed wastes for shipment to a licensed waste disposal facility. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes are by mobile systems in the auxiliary and radwaste buildings. The packaged waste is stored in the auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

The flows of wastes through the WSS are shown in SSAR Figure 11.4-1. SSAR Table 11.4-10 lists WSS equipment design parameters. SSAR Table 3.2-3 identifies WSS pumps, tanks, filters, and certain valves to be the AP600 Class D. The spent resin system, which is part of the WSS, contains the following major components:

- two spent resin tanks, each with a volume of 8.5 m^3 (300 ft^3)
- a resin mixing pump
- a resin transfer pump
- a resin fines filter
- a resin sampling device

The spent resin tanks provide holdup capacity for spent resin and filter bed media decay before processing. The resin mixing pump fluidizes and mixes the resins in the spent resin tanks; transfers water between spent resin tanks, discharges excess water from the tanks to the WLS for processing and disposal, and flushes the resin transfer lines. The resin transfer pump recirculates spent resins via either one of the spent resin tanks for mixing and sampling, for transferring spent resins between tanks, and for blending high- and low-activity resins to meet the specific activity limit for disposal. The resin transfer pump is also used to transfer spent resins to a waste container in the fill stations or in its shipping cask located in the auxiliary building rail car bay. The resin sampling device collects a representative sample of the spent resin either during spent resin recirculation or during spent resin waste container filling operations. The filter transfer cask permits remote changing of filter cartridges, dripless transport to the storage area, transfer of the filter cartridges into and out of the filter storage, and loading of the filter cartridges into disposal containers. The resin dewatering pump of the potable dewatering system removes water from the spent resin disposal container and discharges it to the spent resin tanks. Waste disposal containers are to be selected from available designs that meet the requirements of the U.S. Department of Transportation Department of Transportation (DOT) and the NRC.

A filter transfer cask is used to change the high-activity filters of the CVS and spent fuel cooling system. The filter vessel is drained. If recent applicable sample analysis for the filter media is available, the filter cartridge can be loaded directly into a disposal container. However, if analysis is required, the filter cartridge is placed in a high-activity filter storage tube until sample analysis results are available. Upon completion of the analysis and determination of packaging requirements, the transfer cask is used to retrieve the cartridge from the storage tube and deposit it in the waste container.

At the radwaste building, low- and moderate-activity filter cartridges are deposited into disposal or storage drums. The drums are stored within portable shield casks in the shielded accumulation room, which is serviced by the mobile systems facility crane. Depending on dose rates and analysis results, stabilization may or may not be required. Cartridges not requiring stabilization are loaded into standard, 0.21 m^3 (55 gallon) shipping drums with absorbent and may be compacted using a mobile system. When stabilization is required, the cartridges may be loaded into either high-integrity containers or standard drums. If standard drums are used, mobile equipment is used to encapsulate the contents of the drums. Section 11.4.2.3.2 of the SSAR provides the details of spent filter processing operations.

Chemical wastes are accumulated in the chemical waste tank; they are normally processed by mobile equipment to reduce the volume and to package into drums, which are stored in the packaged waste storage room of the radwaste building. The mixed wastes are collected in suitable containers and brought to the radwaste building, and are normally sent to an offsite facility having mixed-waste processing and disposal capabilities.

Normally, the spent resin from the condensate polishing system demineralizers is nonradioactive and is transferred directly to a truck or to the spent resin tank until it can be removed offsite. If the condensate resins are radioactive, they are transferred from the condensate polishing vessels or a spent resin tank to a temporary processing unit. In this processing unit, the resins are dewatered and processed, as required, for offsite disposal. Westinghouse estimates the condensate polishing spent resins to have negligible radioactivity (see Table 11.4-6 of the SSAR). Also, Westinghouse estimates the maximum generation volume for radioactive condensate polishing resins to be 5.8 m^3 (206 ft^3) per year. In the DSER, the staff stated that Westinghouse did not provide details on packaging of the secondary system wet wastes. This was identified as DSER Open Item 11.4-1. In Sections 10.4.6.3 and 11.4.2.1 of Revision 8 to the SSAR, Westinghouse stated that nonradioactive spent resins do not require any special packaging and that radioactive condensate polishing resin will be disposed of in containers as permitted by DOT regulations. After packaging, the resins may be stored in the radwaste building. The staff finds the above information acceptable; therefore, DSER Open Item 11.4-1 is closed.

In the DSER, the staff stated that Westinghouse did not indicate what design feature had been provided to contain the contents of the secondary spent resin tank, in the event of its failure. This was DSER Open Item 11.4-2. Section 10.4.6.3 of the SSAR Revision 9 states that a spill containment barrier is provided to contain spent resin tank or condensate polish vessel contents in the event of a tank failure. The spill containment barrier is a curb surrounding the area containing the spent resin tank and condensate polisher vessel with sufficient height to contain the contents of a full tank or vessel. On the basis of the above information, DSER Open Item 11.4-2 is closed.

Dry wastes are segregated by portable shielding on the basis of their contact-dose rates into low-activity, moderate-activity, and high-activity wastes. The bags or containers containing these dry wastes are transported to the radwaste building and placed in low-, moderate-, or high-activity storage areas, depending upon their activity levels. High-activity wastes are normally compacted in drums using a mobile compactor system. Moderate-activity wastes are sorted and compacted by mobile equipment. The packaged wastes may be loaded directly into a truck for shipment or may be stored in the packaged waste storage room until a truck load is accumulated. Low-activity waste generally contains a large amount of nonradioactive material. These wastes normally will be processed through a mobile radiation monitoring and sorting system to remove nonradioactive items for reuse or local disposal. The radioactive wastes are compacted or packaged for disposal. SSAR Section 11.4.2.3.3 and Figure 11.4-1 of the SSAR provide the processing details for dry solid wastes.

With the exception of storage capacity, which is discussed later in this section of the SER, the solid waste management system conforms with the guidelines of Branch Technical Position Effluent Treatment Systems Branch (BTP ETSB) 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water Cooled Nuclear Power Reactor Plants" (Rev. 2), July 1981. The wet solid wastes will be disposed of in accordance with 10 CFR Part 61 requirements by COL applicants.

Section 11.4.1.3 of the SSAR states that the waste disposal containers are to be selected from available designs that meet the disposal requirements of 10 CFR Part 61, specific requirements of the disposal facility chosen, and the radioactive waste transportation requirements of 10 CFR Part 71 and DOT regulations. The verification of waste characteristics, waste packaging, and waste disposal are within the purview of the COL applicant. Similarly, the staff considers that development of a process control program (PCP), which identifies the operating procedures (i.e., boundary conditions for a set of process parameters such as settling time, drain time, drying time, and so forth) for processing wet solid wastes, in compliance with 10 CFR Part 61 requirements, is within the scope of the COL applicant. Therefore, for each COL application, the staff will review the PCP, including dewatering or solidification (if performed), and demonstration of WSS compliance with 10 CFR 61.55 and 61.56, 10 CFR 71, and DOT regulations. In the DSER, the staff found that Westinghouse's response to RAI 460.18, which dealt with COL action items, was inadequate. A COL applicant referencing the AP600 certified design should submit a PCP that identifies the operating procedures for processing wet solid wastes. Additionally, the COL applicant should demonstrate WSS compliance with 10 CFR 61.55 and 61.56, 10 CFR 71, and DOT regulations. This was identified in the DSER as COL Action Item 11.4-1. Westinghouse should include COL Action

Item 11.4-1, related to the submittal of a PCP for processing wet solid wastes, and demonstration of WSS compliance with 10 CFR 61.55 and 61.56, 10 CFR Part 71, and DOT regulations, in the SSAR. This was identified as DSER Open Item 11.4-3. In Revision 8 to the SSAR Section 11.4.6, Westinghouse identified a COL Action Item as described above. Therefore, DSER COL Action Item 11.4-1 is acceptable and Open Item 11.4-3 is closed.

The liquid and gaseous effluents resulting from the WSS operation are released during normal operation, including AOOs, to unrestricted areas through the WLS and the monitored plant vent, respectively. The liquids resulting from wet waste processing are routed to the WLS to be processed before release to the environment. Specifically, the excess water from the spent resin tanks is pumped to the WLS through a resin-fines filter by the resin mixing pump. The radwaste and auxiliary buildings contain and drain spillage to the WLS through the radioactive waste drain system. Sloped floors and floor drains are provided to collect and control the release of radioactive material that could be removed from stored solid waste by water contact.

The primary spent resin tanks are located in the seismic Category I auxiliary building that will retain the maximum liquid and spent resin inventory of the spent resin tanks. The spent resin tank vent and overflow connections have screens to prevent the discharge of spent resins. The WSS has a resin-fines filter to minimize the spread of high-activity resin fines.

The liquids and gases that result from the WSS operation are monitored by the WLS and WGS radiation monitors before their release to the environs. Sections 11.2 and 11.3 of this report evaluate the compliance of liquid and gaseous effluents with 10 CFR 20.1302. In the response to RAI 460.11(b) (Revision 2), Westinghouse identified the design features in the system design to comply with GDCs 60, 63, and 64. GDC 60, Control of Releases of Radioactive Materials to the Environment, requires that means be provided to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Means are provided in the solid waste management system to handle the applicable categories of solid radwaste as indicated in SSAR Section 11.4. To control the release of radioactive materials to the environment, the areas and components in the radwaste building~~s~~ that process or house radioactive solid wastes are located in areas with exhaust ventilation that discharges to the radiologically monitored plant vent, as indicated in SSAR Subsection 9.4.8.2. Sloped floors and floor drains are provided to collect and thereby ~~To~~ control the release of radioactive material that could be removed from stored solid waste by water contact.

GDC 63, Monitoring Fuel and Waste Storage, requires that means be provided to detect conditions that may result in excessive radiation levels and to initiate appropriate actions. For the solid waste management system, the wastes with the most potential for high radiation levels are the spent ion exchange resins and filter cartridges, especially those from the chemical and volume control system ion exchangers and filters. The radiation levels of the spent resin tanks can be monitored without entering the rooms. A floor penetration above the spent resin tanks used for personnel access via a ladder allows the radiation levels in the tank rooms to be monitored by lowering detectors down the outside of the tanks.

As described in SSAR Subsection 11.4.2.3.2, the dose rates of high-activity filter cartridges are measured during the change out process when the filter is raised into the high-activity filter transfer cask (but before the bottom cover of the shield cask is secured) using a long-handled radiation probe. The measured dose rate determines the precautions taken during subsequent handling operations. The high-activity filter cartridges can be transferred into and out of the high-activity filter storage tubes using the high-activity filter transfer cask without direct exposure to personnel. The filters in storage can be monitored with minimal exposure at any time through sampling ports (normally closed by shielded plugs).

Dry, solid wastes are normally monitored when received at the radwaste building and are then transferred to the appropriate temporary storage location depending on the measured dose rate as described in SSAR Subsection 11.4.2.3.3. Local shielding can be used within the temporary and packaged waste storage areas to segregate the higher dose rate items and thereby minimize the dose rate in the rest of the storage areas.

GDC 64, Monitoring Radioactivity Releases, requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. Airborne effluents from the auxiliary and radwaste buildings are monitored by exhaust radiation monitors, as described in SSAR Subsection 11.5.2.3.2. Drums and boxes containing filters and lower-activity dry wastes are also surveyed and decontaminated. Mobile or portable equipment may be used for clean waste monitoring to verify that wastes segregated and sorted for nonradioactive disposal are nonradioactive. Hand-held survey meters are used to prevent removal of radioactivity from the radwaste building by personnel. The arrangement of the radwaste building allows the corridors and vehicle access areas to be very low radioactivity areas, thereby minimizing the need for any decontamination operations. Liquid wastes generated from solid radwaste system operations are discharged directly to the liquid radwaste system (WLS) or are collected by the radioactive waste drain system (WRS) and directed to the WLS for subsequent processing and monitored discharge.

On the basis of the above discussion, the staff finds that the WSS complies with GDC 63, 60, and 64 with respect to monitoring solid waste storage, controlling, and monitoring releases of radioactive materials to the environment, respectively. Additionally, the staff concludes that WSS complies with 10 CFR 20.1302, as discussed in Sections 11.2 and 11.3 of this report. As such, DSER Open Items 11.2-5 and 11.3-7 are closed.

The WSS is a non-safety-related system and has no accident mitigation functions. The bulk of the system is located in the radwaste building, which is not a seismic Category I structure. According to Appendix 1A of the SSAR, which pertains to the radwaste management systems compliance with RG 1.143, the primary spent resin tanks are located in the auxiliary building, which is a seismic Category I structure. The seismic Category I structure will retain the maximum liquid and spent resin inventory of the spent resin tanks. Thus, the WSS complies with Positions C.3.1.3 and C.5 of RG 1.143, regarding seismic design criteria for structures housing solid radwaste management systems. The portion of the WSS that is within the scope of the AP600 standard design (i.e., the design of components and subsystems of mobile systems that are used by contractors to process wet solid wastes and chemical wastes are not within the scope of the AP600 standard design) is designed in accordance with the applicable Positions C.3, C.4, and C.6 of RG 1.143 with respect to specific guidelines for solid radwaste systems; general guidelines for design, construction, and testing criteria for radwaste systems; and general guidelines for providing QA for radwaste management systems. Appendix 1A of the SSAR provides a detailed discussion of how the design of the WSS, and its housing structure, meets the applicable guidelines of RG 1.143. Specifically, the subject appendix states that the WSS components are designed and tested to the guidelines set forth in the codes and standards listed in Table 1 of RG 1.143 as supplemented by Guideline 3.1.2 of RG 1.143.

Table 3.2-3 of the SSAR states that components such as pumps, tanks, filters, and applicable valves of the WSS are designed to the AP600 Class D, which is equivalent to Quality Group D of RG 1.26. Appendix 1A of the SSAR states that the QA program for radwaste systems is in accordance with SSAR Chapter 17 and meets RG 1.143. The staff stated in the DSER that because the installation, procurement, and fabrication of the system components are within the scope of the COL applicant, the staff will review the specifics of the QA program for each COL application. A COL applicant referencing the AP600 certified design should provide details of its proposed QA program as it relates to the installation, procurement, and fabrication of the WSS components. This was identified as DSER COL Action Item 11.4-2 and Open Item 11.4-4. In Revision 8 to the SSAR Section 11.4.5, Westinghouse stated that the QA program for design, installation, procurement, and fabrication issues of the WSS is in accordance with the overall QA program described in Chapter 17. Section 17.4 of the SSAR states that the COL applicant will address its design phase QA program, as well as its QA program for procurement, fabrication, installation, construction, testing and operations of structures, systems, and components in the facility. Therefore, DSER COL Action Item 11.4-2 is acceptable and Open Item 11.4-4 is closed. COL Action Item 11.4-2 is identified in SSAR Table 1.8-2 as part of Item Number 17.5-2.

The staff will review the mobile systems facility operated by the COL applicant (or its contractors) on a plant-specific basis, against the guidelines of RG 1.143. A COL applicant referencing the AP600 certified design should discuss how any mobile processing equipment intended for use in the processing of solid radwaste meets the guidelines of RG 1.143. This was identified as DSER COL Action Item 11.4-3 and Open Item 11.4-5. In Revision 6 to the SSAR Section 11.4.6, Westinghouse identified the above COL actions. Therefore, DSER COL Action Item 11.4-3 is acceptable and Open Item 11.4-5 is closed.

On the basis of the above discussion, the staff finds that the AP600 WSS meets the applicable guidelines of RG 1.143, and thus complies with GDC 60, and 10 CFR 50.34a, as they relate to the control of radioactive materials released to the environment during system operation and adequacy of design information for the WSS, respectively.

The staff's evaluation guidance for the solid waste storage capacity is stated in SRP Section 11.4, Paragraphs II.6 and III.4, and BTP ETSB II-3, Positions B.III. In the DSER, the staff reviewed and found the AP600 solid waste storage capacity acceptable.

However, Westinghouse subsequently changed its design and revised the SSAR information such that the bases for the staff finding in the DSER are not valid. In light of the new design, the staff requested additional information in RAI 460.27 and RAI 460.28F, asking Westinghouse to clarify the use of Westinghouse's terms "expected generation" or "maximum generation" for the demonstration of adequacy in storage capacity and to demonstrate the compliance of the above applicable guidance. The expected generation rates are significantly lower than the maximum generation rates.

In a letter dated June 18, 1997, in response to RAI 460.27, Westinghouse clarified that the maximum waste generation rate, shown in SSAR Table 11.4-1, "Estimated Solid Radwaste Volumes," was used for demonstrating the adequacy in storage capacity. The maximum generation rates were based on the assumption of the design-based fuel failure rate of 0.25 percent. The staff evaluated this assumption and found it acceptable for the AP600 waste management system design, as discussed in Section 11.1 of this report.

In a letter dated December 19, 1997, Westinghouse explained each item listed in SSAR Table 11.4-1 to respond to RAI 460.28F. The maximum generation rates of 30 m^3/yr(1060 ft^3/yr) of primary resins including wet activated carbon are stored in two spent resin tanks (7.8 m^3/tr (275 ft^3) useful capacity per tank) and one 4.5 m^3/yr (158 ft^3)high-integrity container; both tanks and container are located in the auxiliary building. Primary filters at the maximum generation rates of 0.2 m^3/yr (6.5 ft^3/yr) are stored in shielded spent filter storage tubes 0.13 m^3/yr (4.5 ft^3 for total of nine tubes) in the auxiliary building for decay and packaging into drums. The maximum volume of these packaged drums of 1.4 m^3/yr (48 ft^3/yr)are stored in the rail car bay (18 ft x 60 ft x 30 ft) of the auxiliary building. Other wastes such as chemical waste, mixed liquid waste, compactible dry waste, non-compatible dry waste, mixed solid waste, and low activity filters are processed, compacted, packaged, and then stored in the packaged waste storage room of radwaste building. The maximum and expected amounts of these packaged wastes are 65.4 m^3/yr (2309 ft^3/yr) and 36.6 M^3/yr (1292.5 ft^3/yr) respectively, calculated from SSAR Table 11.4-1,"Maximum Shipped Solid" column and "Expected Shipped Solid" column. The size of the packaged waste storage room in the radwaste building is 110.5 m^3/yr (3900 ft^3). In addition, there are 26.5 m^3/yr (939 ft^3/yr) of condensate polishing resin and steam generator blowdown material, which normally are not radioactive and are shipped directly for disposal; however, in case of contamination they may be stored in the radwaste building packaged waste room after processing. Therefore, the waste storage room in the radwaste building has more than a one-year capacity for the maximum amount of the packaged waste, and more than three years of capacity for the expected amount of packaged waste. On the basis of the above information, the staff concludes that sufficient storage capacity for the maximum waste generation rates is available in accordance with BTP ETSB Position B.III regarding solid waste storage requirements.

In Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," 1981, the staff provided guidance to licensees on the addition of onsite storage facilities for low-level radioactive wastes generated onsite. The staff recognizes that the need for additional onsite storage capacity, for low-level radioactive wastes beyond what has been provided in the AP600 standard design, is a site-specific issue because it will depend upon offsite low-level waste storage space availability for the site's wastes. Therefore, when such a need is identified by a COL applicant, the COL applicant should submit the details of any proposed on-site low-level radioactive waste storage facility to the NRC. The staff will review and evaluate such a proposed additional site-specific facility against the guidelines in GL 81-38, which is similar to the guidance in Appendix 11.4-A to SRP Section 11.4.

11.4.2 Conclusion

In its evaluation of the WSS, the staff considered the following acceptance criteria:

- the capability of the system in conjunction with WLS and WGS to maintain any liquid or gaseous effluents in unrestricted areas, arising from the system operation, below the limits in 10 CFR 20.1302 (see the resolution of DSER Open Item 11.2-5 and 11.3-7 in Section 11.4.1)

- design features to comply with GDC 60, 63, and 64 (see the discussion on the response to RAI 460.11(b) in Section 11.4.1)

- provisions for onsite storage of processed solid wastes in accordance with BTP ETSB 11-3 Position B.III (see the discussion on the responses to RAI 460.27 and RAI 460.28F in Section 11.4.1)

- quality group and seismic classification applied to the structures housing the system in compliance with RG 1.143 (see the discussion on the evaluation of Appendix 1A of the SSAR, the resolution of DSER Open Items 11.4-4 and 11.4-5 in Section 11.4.1)

- the requirements of 10 CFR Part 61 and Part 71 (see the resolution of DSER Open Item 11.4-3 in Section 11.4.1)

Based on the above information, the staff concludes that the WSS is in compliance with the requirements of 10 CFR 50.34a on the adequacy of design information. Based on the above evaluation of the WSS in accordance with the applicable acceptance criteria of Section 11.4 of the SRP, the staff finds the WSS to be acceptable.

11.5 <u>Process and Effluent Radiological Monitoring and Sampling System</u>

11.5.1 System Description and Review Discussion

The process and effluent radiological monitoring and sampling system is used to measure, record, and control releases of radioactive materials in plant process streams and effluent streams. The system consists of permanently installed sampling and monitoring equipment designed to indicate routine operational radiation releases, equipment or component failure, system malfunction or misoperation, and potential radiological hazards to plant personnel or to the general public.

The staff reviewed Westinghouse's responses to the staff's RAIs on radwaste management systems for the AP600 and Section 11.5 of the SSAR. The staff used the acceptance criteria provided in Section 11.5 of the SRP to determine system compliance with the following regulations:

- 10 CFR 20.1302, as it relates to ensuring concentrations of radionuclides in gaseous and liquid effluents to unrestricted areas within the limits specified in 10 CFR Part 20, Appendix B, Table 2, Columns 1 and 2

- 10 CFR Part 50, Appendix A, GDC 60 and 64, as they relate to controlling and monitoring the release of radioactive materials to the environment via gaseous and liquid effluents

- GDC 63, as it relates to monitoring fuel and waste storage areas, and monitoring leakages

- 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii), as they relate to monitoring plant gaseous effluents for containment high-range radiation and noble gases, and sampling and analyzing gaseous effluents for radionuclides and particulates, during and following an accident

Specific criteria acceptable to meet GDC 64 are as follows:

- the gaseous and liquid process streams or effluent release points should be monitored and sampled according to Tables 1 and 2 of SRP Section 11.5

- the design of systems should meet the provisions of the applicable positions in RG 1.21, RG 1.97, and RG 4.15.

In the AP600 design, radiation monitors are provided for the following processes and effluents:

- the auxiliary building fuel handling area exhaust

- exhaust from the rest of the auxiliary building areas, excluding two electrical penetration rooms and two reactor trip switchgear rooms

- containment air filtration exhaust

- health physics area (in annex building) and hot machine shop area (annex building) exhaust

- exhaust from the rest of annex building

- radwaste building exhaust

- gaseous radwaste system exhaust

- plant vent exhaust

- containment atmosphere
- turbine island vent discharge
- main control room supply air
- primary sampling gaseous sample
- primary sampling liquid sample
- main steamline
- SG blowdown
- component cooling water system
- service water blowdown
- liquid radwaste discharge
- waste water discharge

The monitored stream, detector type, normal range for the detector, automatic function associated with the monitor, principal radionuclides that are monitored, and the locations of all the liquid and gaseous processes and effluent radiation monitors are provided in Sections 11.5.2.3.1, 11.5.2.3.2, and 11.5.2.3.3, and Table 11.5-1 of the SSAR. Table 11.5-1 indicates that the radiation monitors at the MCR supply air duct are safety-related and the monitors for containment atmosphere are seismic Category I.

Westinghouse's response to RAI 460.12, which deals with unmonitored exhausts, states that the exhausts from the diesel generator rooms (located in the stand alone diesel generator building), personnel areas, the electrical and mechanical equipment rooms of the annex building, and electrical penetration and reactor trip switchgear rooms of the auxiliary building are not monitored, because the subject areas do not contain any radioactive materials. The response further states that the subject areas interface only with clean areas, if at all, thus precluding transfer of radioactive materials from adjoining areas. The response also states that the annex building general area HVAC system normally maintains the personnel areas of the annex building at a slightly positive pressure with respect to adjoining areas.

On the basis of its review of the SSAR sections, the table, and Westinghouse's response mentioned above, the staff finds that the lack of monitoring for the area exhausts mentioned above is acceptable. Additionally, the staff finds that the subject SSAR sections and table cover all of the applicable gaseous and liquid processes and effluent streams identified in Section 11.5, Tables 1 and 2 of the SRP, except for the service water effluent stream. In the DSER, the staff stated that Westinghouse had not provided a complete response to RAI 460.7, which dealt with monitoring of the service water effluent. This was identified as DSER Open Item 11.5-1. In the response to RAI

460.7, Revision 2, Westinghouse stated that the service water system has been changed to include an effluent radiation monitor. This is confirmed in Section 11.5.2.3.1 of the SSAR, Revision 9. Therefore, DSER Open Item 11.5-1 is closed.

In the DSER, the staff found that the AP600 SSAR provided sufficient information, except for the following:

- Table 3.2-3 of the SSAR did not include the RMS. This was identified as DSER Open Item 11.5-2.
- Sections 11.5.2.3.1 through 11.5.2.3.3 of the SSAR made incorrect references to SSAR figures. This was identified as DSER Open Item 11.5-3.
- Section 11.5 of the SSAR did not explain why the containment atmosphere particulate detector (part of the reactor coolant pressure boundary leak detection system) is non-seismic Category I and receives power from a non-Class 1E power supply. This was identified as DSER Open Item 11.5-4.

In a letter dated March 5, 1997, Westinghouse stated that the RMS is not a fluid system and that Table 3.2-3 of the SSAR includes only fluid systems. Therefore, the RMS is not included in Table 3.2-3. All the safety-related monitors are identified in SSAR Tables 11.5-1 and 11.5-2, and SSAR Section 7.1.4 provides information on the requirements for safety-related monitors. Other radiation monitors are provided in SSAR Tables 11.5-1 and 11.5-2 and discussed in SSAR Section 11.5. The staff finds the response acceptable. Therefore, DSER Open Item 11.5-2 is closed.

The incorrect references in SSAR Section 11.5.2.3.1 through 11.5.2.3.3 to figures were corrected in Revision 6 to the SSAR. Therefore, DSER Open Item 11.5-3 is closed.

Westinghouse responded to DSER Open Item 11.5-4 (OITS Item No. 1189) in a letter dated October 17, 1996, stating that the containment atmosphere particulate detector is not required for reactor coolant pressure boundary leak detection in the AP600. Instead, a containment atmosphere N^{13}/F^{18} monitor, which is designed to seismic Category I, will be used for the reactor coolant pressure boundary leak detection. This is discussed in SSAR Section 5.2.5, evaluated and found acceptable in Section 5.2.5 of this report. Therefore, DSER Open Item 11.5-4 is closed.

On the basis of the above discussion, the staff finds that the effluent monitors comply with GDC 64, with regard to monitoring of radioactive liquid and gaseous effluents from the plant.

The area radiation monitors (ARMs) monitor the radiation levels in selected areas throughout the plant. These are provided to supplement the personnel and area radiation survey provisions of the AP600 health physics program. Table 11.5-2 of the

SSAR lists the following areas with ARMS, along with the nominal range and type of radiation measured:

- containment high range
- primary sample room
- containment area personnel hatch
- main control room
- chemistry laboratory area
- fuel handling area
- rail car bay area
- liquid and gaseous radwaste area
- technical support center
- radwaste building mobile systems facility
- hot machine shop
- annex staging and storage area

All of these detectors are gamma-sensitive Geiger-Muller tubes. A local readout and alarm module is located in each area to visually guide personnel before entering the monitored areas. The containment area radiation monitors and fuel handling area radiation monitors provide an alarm locally and in the main control room. On the basis of the review of the above information, the staff finds that the process and effluent monitoring and sampling system for the AP600 standard design provides the needed monitoring for fuel and radioactive waste storage and thus complies with GDC 63.

Besides the plant vent accident range monitor and the condenser air removal exhaust monitor, which monitor radioactive gaseous effluents during accidents, the following special-purpose radiation monitors are provided either for monitoring during an accident or to trigger an automatic control action:

- two main steamline radiation monitors for monitoring radionuclide concentrations in the two main steamlines and using the concentration data for calculating radioactive releases to the environment, when the SG safety relief or power operated relief valves are used to release steam to the atmosphere

- four containment high-range radiation monitors for monitoring gamma radiation intensities inside the containment following an accident and using the data for estimating radioactive material inventory in the containment volume

- radiation-level monitors (two particulate detectors, two iodine detectors, and two noble gas detectors) to monitor the radiation level in the air supply to the MCR and to activate the MCR emergency habitability system, if the concentrations of radioactive materials exceed predetermined setpoints for the monitors

Sections 11.5.2.3.1 and 11.5.6.2 of the SSAR state that the MCR radiation monitors and containment high-range monitors are environmentally and seismically qualified in accordance with the guidelines of RGs 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," and 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants." The monitors for these containment and MCR monitors receive Class 1E power.

Table 11.5-1 of the SSAR indicates that the steamline radiation monitors are non-safety-related and receive non-Class 1E power. Section 11.5.2.3.1 and Table 11.5-1 of the SSAR provide the nominal range, and type of radiation measured, the detector type, the location, and automatic control feature (if provided) for these special purpose radiation monitors. The staff finds the range provided in Table 11.5-1 of the SSAR for gamma radiation measurement inside the containment by the high-range containment radiation-monitors acceptable because it meets the range criterion for such monitors specified in NUREG-0737, TMI Item II.F.1, Attachment 3, containment high-range radiation monitoring, and therefore, it complies with the applicable portions of 10 CFR 50.34(f)(2)(xxvii). Also, the staff finds the ranges specified in Table 11.5-1 of the SSAR for the steamline and MCR radiation monitors acceptable, because they are consistent with industry practice. On the basis of the above discussion, the staff finds that these special purpose monitors comply with GDC 60 and 64, as related to controlling and monitoring the release of radioactive materials to the environment.

The RMS initiates such control actions as reducing or terminating radioactive releases to the environment on detection of high radiation by the monitors in accordance with GDC 60. Specifically, the WGS exhaust monitor and the liquid waste discharge monitor initiate control actions to terminate the applicable discharge on detection of high radiation by the respective monitor. The MCR supply air duct radiation monitor isolates the MCR air intake and exhaust ducts and activates the MCR emergency habitability system (for information on the MCR habitability system, see Section 6.4 of this report) upon detection of high radiation level in the air intake by the MCR radiation monitor. The fuel handling area exhaust radiation monitor, annex building exhaust radiation monitor, or the auxiliary building exhaust radiation monitor, (each located upstream of its respective exhaust air isolation dampers) initiates control action to automatically divert the exhaust from the applicable area to the VFS, on detection of high radiation in the exhaust of the applicable area by its associated radiation monitor. This is accomplished by closure of the affected area supply and exhaust air isolation dampers, opening of the applicable exhaust air isolation dampers of the VFS, and starting of the containment air filtration exhaust unit. The turbine island vent discharge radiation monitor facilitates performance of corrective manual actions in a timely manner, upon detection of high radiation in the subject exhaust. The SG blowdown radiation monitor and the waste water discharge monitor facilitate manual diversion of the applicable stream to the WLS for processing by that system on the detection of high radiation in the applicable stream by the associated radiation monitor. The component cooling water system radiation monitor facilitates manual isolation of the system and timely performance of leak repairs on the detection of radiological leakages into the system by the system radiation monitor. On the basis of the above discussion, the staff finds that the AP600 design permits control of radioactivity releases to the environs in accordance with GDC 60.

To comply with the numerical objectives in 10 CFR Part 50, Appendix I, for the offsite doses resulting from gaseous and liquid effluents during normal plant operation including AOOs, plants will be required to limit annual and three-month offsite doses. Additionally, plants will be required to use treatment systems (e.g., waste gas treatment by delay beds, demineralizers to treat liquid radwaste, and containment air filtration system) if the monthly dose is likely to exceed about 25 percent of the 10 CFR Part 50, Appendix I annual dose guidelines prorated for one month. These requirements will be specified in a plant-controlled document and will be implemented by the COL applicant (holder). Additionally, plants will be required to limit instantaneous discharge concentrations from the WGS and WLS to comply with 10 CFR Part 20.1302, which defines the criteria for radionuclides concentration limits in liquid and gaseous effluents. Plants will be required to provide the associated setpoints for the applicable radiation monitors in the plant-specific ODCM. These requirements, in conjunction with the automatic control (i.e., termination of the discharge, or diversions through the VFS) features of applicable effluent monitors, will ensure that the AP600 effluent monitors are in compliance with 10 CFR 20.1302. The staff will review the plant-specific radiological effluent technical specifications (RETS) that will be provided in the plant-controlled document, as well as setpoints in the plant-specific ODCM for each COL application. (Conformance with 10 CFR Part 50, Appendix I and 10 CFR 20.1302 have been previously identified as COL Action Items 11.1-1, 11.2-3, and 11.3-1.)

The staff reviewed SSAR Section 11.5 against the guidelines of Section 11.5 Table 1 and 2 of the SRP and raised questions in RAI 460.15. The process and effluent radiological monitoring and sampling system should have the capability to sample process and effluent streams during normal plant operation, including AOOs. Westinghouse's revised response (Revision 3) to RAI 460.15 discusses in detail how the system meets the guidelines of Section 11.5, Tables 1 and 2 of the SRP. RAI 460.15 pertains to the sampling systems and the identified SRP tables list the needed sampling provisions for liquid and gaseous process and effluent streams. Westinghouse's response includes Tables 460.15-1 and 460.15-2, which list the sampling provisions for gaseous and liquid streams, respectively. Additionally, the response includes revised SSAR Tables 9.3.3-1 through 9.3.3-3 and 9.3.4-1 and 9.3.4-2. These Chapter 9 tables deal with primary and secondary sampling systems. Further, Section 11.5.2.3.3 of the SSAR describes the sampling provisions provided for the effluent stream through the plant vent. The sampling provisions identified in the response and SSAR are summarized below. The system provides for continuous and representative sampling of airborne particulate and iodine radioactivities for the plant vent discharge, which is a major gaseous effluent stream. The system also provides grab sampling provisions for noble gases, iodines, and particulates, and tritium radioactivities for the gaseous radwaste system discharge.

In the DSER, the staff found that, for liquid streams, the system provided grab sampling and analysis capability for gross radioactivity determination, identification, and concentration determination of principal radionuclides and alpha emitters for:

- the chemical waste tank
- the primary spent resin tank
- the WLS monitor tanks
- the waste holdup tanks

On the basis of its review of the response to RAI 460.15 (Revision 1), Sections 11.5.3 and 11.5.2.3.3 of the SSAR, and the above discussion, the staff identified the following information that Westinghouse needed to provide:

- Westinghouse did not include a grab sampling provision for tritium activity in the effluent via the plant vent. This was identified as DSER Open Item 11.5-5.

- Westinghouse did not include grab sampling and continuous sampling provisions for the condenser air removal system effluent stream. It should be noted that provision of a monitor, which continuously monitors an effluent stream, is not equivalent to provision of continuous sampling capability for that stream. This was identified as DSER Open Item 11.5-6.

- Westinghouse did not include a grab sampling provision for the turbine gland seal system exhaust. This was identified as DSER Open Item 11.5-7.

- Westinghouse did not include grab sampling provisions for noble gas activities in the building ventilation and containment purge exhausts. This was identified as DSER Open Item 11.5-8.

- Westinghouse did not include grab sampling and continuous sampling provisions for iodine activity in the containment purge exhaust. This was identified as DSER Open Item 11.5-9.

- Westinghouse did not include continuous sampling and analysis provisions for service water system effluent. Provision of a continuous radiation monitor in an effluent line is not the same as provision of continuous sampling capability for the line. This was identified as DSER Open Item 11.5-10.

- Westinghouse included a reference to a nonexistent radiation monitor in the service water system in Table 9.3.4-1 of the SSAR. This was identified as DSER Open Item 11.5-11.

- Westinghouse should explain the purposes of the grab sampling and analysis provisions for the component cooling water system, service water system effluent stream, SG blowdown stream, turbine building drains, and waste water drains. Westinghouse should indicate if grab sampling and analysis provisions for tritium activity are included for the above system, streams and drains. This was identified as DSER Open Item 11.5-12.

- Westinghouse did not include grab sampling and analysis provisions for the spent fuel pit treated water. This was identified as DSER Open Item 11.5-13.

- Westinghouse did not include grab sampling and analysis provisions for secondary resin slurry stream. This was identified as DSER Open Item 11.5-14.

- Westinghouse did not include grab sampling and analysis provisions for tritium activity in the WLS tanks, chemical waste tank, and primary spent resin tanks. This was identified as DSER Open Item 11.5-15.

Subsequently, Westinghouse revised SSAR Table 9.3.3-2, Table 11.5-1, Table 11.5-2, and Sections 11.5.2.3.1 through 11.5.2.3.3 in Revisions 8 through 11 to address DSER Open Items 11.5-5 through 11.5-15 above. Specifically, Westinghouse addressed the open items by providing the information requested by the staff and listed in the SSAR as follows:

- DSER Open Item 11.5-5 is addressed in SSAR Table 9.3.3-2, Item 33, Note c.

- DSER Open Item 11.5-6 is addressed in SSAR Table 9.3.3-2, Item 31, Note b.

- DSER Open Item 11.5-7 is addressed in SSAR Table 9.3.3-2, Item 32, Note b and Figure 10.4.3.1.

- DSER Open Item 11.5-8 is addressed in SSAR Table 9.3.3-2, Item 33 and Section 11.5.2.3.3 regarding sampling provisions for plant vent, which is at the flow downstream of building ventilation and containment purge exhausts (shown in SSAR Figure 9.4.7-1).

- DSER Open Item 11.5-9 is addressed in SSAR Table 9.3.3-2, Item 33. DSER Open Item 11.5-10 is addressed in SSAR Table 9.3.3-2, Item 23.

- DSER Open Item 11.5-11 is addressed in SSAR Table 11.5-1, service water radiation monitor.

- DSER Open Item 11.5-12 is addressed in SSAR Sections 9.3.3 and 9.3.4 which explain the purposes of grab sampling and analysis. In addition, grab sampling and analysis provisions for tritium activity are included in Table 9.3.3-2 Items 7, 13, 23, and 24 for SG blowdown, component cooling water system, service water system, and turbine building drains and waste water drains respectively to address DSER Open Item 11.5-12.

- DSER Open Item 11.5-13 is addressed in SSAR Table 9.3.3-2, Item No. 8 and Figure 9.1-6.

- DSER Open Item 11.5-14 is addressed in SSAR Table 9.3.3-2, Items 22 and 25.

- DSER Open Item 11.5-15 is addressed in SSAR Table 9.3.3-2, Items 18, 21, and 22.

On the basis of the above discussion of the resolution of DSER Open Items 11.5-5 through 11.5-15, the staff finds that the applicable liquid and gaseous process and effluent streams will be sampled as specified in Section 11.5, Tables 1 and 2 of the SRP, which provide a list of monitoring and sampling guidelines. Therefore DSER Open Items 11.5-5 through 11.5-15 are closed.

Specific compliance with the guidelines provided in ANSI N13.1-1969 and RGs 1.21 and 4.15 (as stated in Section 11.1 of this report, these documents deal with sampling programs, reporting radioactivity measurements, and QA for radiological monitoring programs) is not within the scope of the AP600 standard design. Consequently, the staff will review the compliance of radiological monitoring and sampling programs with the specific guidelines of the above documents on a plant-specific basis for each COL application. The staff's review will include (but is not limited to) the following:

- written procedures for sample collection, preparation, and analysis in accordance with RG 4.15

- written procedures for reduction, evaluation, and reporting of data

- identification of radioactivity reference standards

- information on detector calibration, calibration frequency, sampling frequency, analysis frequency, type of activity analysis, and lower limit of detection

In response to RAI 460.18, which deals with the COL action items for radwaste management systems, Westinghouse states that the radiation monitoring system for the AP600 will comply with the guidelines of ANSI N13.1-1969, as well as RGs 1.21 and 4.15. Westinghouse further states that the radiation monitoring system will be supplied and maintained in accordance with a QA program per ASME NQA-1-1989 Edition through NQA-1b-1991 Addenda. SSAR Section 11.5 states that the radiation monitoring system is designed in accordance with ANSI N13.1-1969. Demonstrating specific compliance of the radiological monitoring and sampling programs for the individual the AP600 reactor with the guidelines of ANSI N13.1, RG 1.21 and RG 4.15, is the responsibility of the COL applicant. This COL action has been previously identified in Section 11.1 of this report as COL Action Item 11.1-1, which is acceptable.

The following AP600 post-accident radiation monitors are consistent with the ranges specified in RG 1.97 for such monitors:

- the main steamline radiation monitors
- high range radiation monitors
- plant vent accident range radiation monitor
- turbine island vent discharge radiation monitor
- primary sample room monitor

Table 11.5-1 and Section 11.5.2.3.3 of the SSAR, and Westinghouse's response to RAI 460.16 (this question deals with plant gaseous effluent monitoring and sampling under accident conditions), give a detailed discussion of post-accident monitoring and sampling instrumentation, particularly as it relates to the compliance of such instrumentation for plant gaseous effluents with the guidelines of NUREG-0737, TMI Item II.F.1, Attachments 1 and 2. On the basis of its review of the above SSAR sections and Westinghouse's response, the staff finds that only the plant vent discharges gaseous effluents directly to the environs. The plant vent has an accident-range effluent monitor to continuously monitor the noble gas concentrations in gaseous effluents during and following an accident. Also, the vent has the capability to grab sample and analyze plant gaseous effluents for iodine and particulates during and following an accident. The staff was concerned that the AP600 design did not provide continuous sampling of the plant vent gaseous effluent for iodines and particulates during and following an accident. The staff considered that a reasonable estimate of the iodine and particulate radioactive release via the plant vent to the environs resulting from an accident would not be possible without continuous sampling provision. Westinghouse should have included in the AP600 design a continuous sampling capability and onsite analysis capability for the plant vent gaseous effluent for iodines and particulates, during and following an accident. This was identified as DSER Open Item 11.5-16. In Revision 6 of the SSAR Section 11.5.2.3.3, Westinghouse included the capability to sample continuously plant vent effluent during and following an accident for iodine and particulates. In Revision 8 of the SSAR, Westinghouse clarified its onsite analysis capability in an on-site laboratory. On the basis of above, DSER Open Item 11.5-16 is closed.

The turbine island vent discharge has a noble gas effluent monitor which can continuously monitor noble gas concentrations in gaseous effluents through the vent during normal operation, and during and following an accident. The upper limits for concentration measurements for the accident-range effluent monitor for the plant vent, the accident- and low-range effluent monitor for the condenser air removal system exhaust vent, and the high-range monitor for the containment area agree with the limits specified for such monitors in NUREG-0737, TMI Item II.F.1, Attachments 1 and 2. The staff finds the specified ranges for these monitors acceptable.

Furthermore, Westinghouse's response to RAI 460.16 identifies the following information regarding accident monitoring instrumentation:

- procedures and/or methods for converting radiation measurements into release rates of gaseous discharges through the vents
- sampling techniques used to monitor and sample effluent gases to assure that a representative sample is taken
- the sampling system's capability to maintain isokinetic conditions during and following an accident
- collection techniques to extract a representative sample of radioactive iodine and particulates during and following an accident
- calibration frequency and techniques for radiation monitors
- shielding requirements for the sampling systems and the shielding design of the systems to comply with the guidelines of TMI Item II.F.1, Attachment 2
- radiation reading capability (readings are continuous)
- location of instrument readouts

The staff reviewed the identified material outlined above, and found that the accident monitoring instrumentation provided in the AP600 design for monitoring noble gases in gaseous effluents, and sampling and analyzing the plant effluent for post-accident releases of radioiodine and particulates, meets the guidelines of NUREG-0737, TMI Item II.F.1, Attachments 1 and 2 regarding accident monitoring instrumentation and, therefore, complies with the applicable portions of 10 CFR 50.34(f)(2)(xxvii) (incorporating the subject TMI requirements).

The staff reviewed the AP600 design regarding the issues identified in NRC Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to the Environment," 1980. Specifically, the staff reviewed the following the AP600 design features as identified in Sections 9.2.9, 9.3.5, 9.4.2, and 9.4.10 of the SSAR:

- the ability to detect the contamination of nonradioactive systems
- the ability to prevent the potential for unmonitored and uncontrolled release of radioactive material to the environment

On the basis of this review, as discussed in this section and Section 11.2 of this report, the staff finds that in the AP600 design the radioactive systems are segregated from nonradioactive systems, and all radioactive or potentially radioactive effluent pathways are monitored before the effluents are released to the environment. For example, drain systems that carry radioactive wastes generally do not contain piping connections that could allow inadvertent transfer of radioactive fluid into nonradioactive piping systems. Where such connections exist, back flow prevention is provided in the nonradioactive piping. The annex/auxiliary building nonradioactive HVAC system and the diesel generator building HVAC system are segregated from radioactive HVAC systems. The normally nonradioactive secondary coolant system sampling drains and other waste water are diverted to the WLS for processing and monitored disposal if detected to be radioactive (a radiation monitor is provided for monitoring the normally nonradioactive waste water discharge). The staff further notes that Westinghouse considers the issues identified in NRC bulletin 80-10 regarding contamination of nonradioactive systems as part of the COL surveillance issue (Westinghouse's submittal dated December 15, 1992, WCAP-13559). On the basis of the above discussion, the staff finds that Sections 9.2.9, 9.3.5, 9.4.2, and 9.4.10 of the SSAR satisfactorily address the concerns raised in NRC Bulletin 80-10. The design features in the AP600 design are adequate to detect the contamination of nonradioactive systems, and prevent the potential for unmonitored and uncontrolled release of radioactive material to the environment. However, the staff expects that the COL applicant will periodically verify that these design features function as intended. A COL applicant referencing the AP600 certified design must provide details of its proposed surveillance program, which eliminates the potential for unmonitored and uncontrolled release of radioactive material to the environment. This is COL Action Item 11.5-1.

Westinghouse should include in the SSAR COL Action Item 11.5-1, related to the proposed surveillance program that eliminates the potential for unmonitored and uncontrolled release of radioactive material to the environment. This was identified as DSER Open Item 11.5-17. In SSAR Section 11.5.7, Westinghouse clarified that the above requested program will be included in the site-specific ODCM, which is required by the COL action Item as specified in SSAR Section 11.5.7. Therefore, DSER Open Item 11.5-17 is closed.

11.5.2 Conclusion

The staff verifies that sufficient information has been provided in the SSAR Section 11.5 and the RAI responses discussed above. The staff concludes that process and effluent radiological monitoring instrumentation and sampling systems are acceptable and meet the relevant requirements of 10 CFR 20.1302, 10 CFR 50.34(f)(2)(xvii) and 50.34(f)(2)(xxvii), and GDC 60, 63, and 64. This conclusion is based on the following:

The staff review includes the provisions proposed in the SSAR to sample and monitor all plant effluents in accordance with GDC 64, monitoring radioactivity releases. The process and effluent radiological monitoring and sampling systems include the instrumentation for monitoring and sampling radioactivity, contaminated liquid, gaseous, and solid waste process and effluent streams. This is demonstrated by that all the processes and effluent streams identified in Section 11.5, Table 1 and 2 of the SRP are included in the SSAR Section 11.5.2.3 and Table 11.5-1 of the SSAR. The staff reviewed and found the information acceptable as discussed in the closure of DSER Open Items 11.5-1, and 11.5-11.15. In addition, the staff review includes the provisions for conducting sampling and analytical programs in accordance with the guidelines in RG 1.21 and 4.15, and the provisions for sampling and monitoring process and effluent streams during postulated accidents in accordance with the guidelines in RG 1.97.

The staff review includes the provisions proposed in the SSAR to provide automatic termination of effluent releases and ensure control over discharge in accordance with GDC 60, controlling of releases of radioactive materials to the environment. The controlling and terminating radioactive releases of the following are discussed in Section 11.5.1 of the report: WGS exhaust, WLS discharge, MCR air supply, fuel handling area, annex building exhaust, auxiliary building exhaust, turbine island vent discharge, SG blowdown, waste water discharge, and component cooling water. In Section 11.5.1 of this report, the staff reviewed and found the design of controlling of radiation releases acceptable.

As discussed in Section 11.5.1 of this report, the staff review includes the provisions proposed in the SSAR for sampling and monitoring fuel handling and waste storage areas in accordance with GDC 63.

Based on the above review, the staff has determined that the AP600 RMS design meets the guidelines of SRP Section 11.5. Therefore, it is acceptable.

12 RADIATION PROTECTION

12.1 Introduction

Chapter 12 of the Westinghouse AP600 Standard Safety Analysis Report (SSAR) describes the radiation protection measures of the AP600 reactor design and operating policies. The staff evaluated this information against the criteria in Chapter 12 of the NRC's Standard Review Plan (SRP) (NUREG-0800).

The radiation protection measures incorporated into the AP600 reactor design are intended to ensure that internal and external radiation exposures to station personnel, contractors, and the general population, resulting from plant conditions, including anticipated operational occurrences (AOOs), will be within acceptable regulatory criteria and will be as low as is reasonably achievable (ALARA). Westinghouse's radiation protection design and program features should also be consistent with the guidelines of Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Revision 3, dated June 1978, or an acceptable alternative.

The basis for the staff's acceptance of this material is that doses to personnel will be maintained within the limits of 10 CFR Part 20, "Standards for Protection Against Radiation." On May 21, 1991, the Commission issued a revision of 10 CFR Part 20 that changed the system of radiation dose limitation. The previous occupational dose limit for whole-body radiation exposure was 12.5 mSv (1.25 rem) per quarter year with a provision to extend the limit to 120 mSv (12 rem) per year. The new dose limit is 50 mSv (5 rem) per year with a provision (i.e., by planned special exposure) to extend it to 100 mSv (10 rem) per year with a lifetime dose limit of 250 mSv (25 rem) due to planned special exposures. Similarly, the previous 10 CFR Part 20 limits for doses from licensed radioactive material inside the body (deposited through injection, absorption, ingestion, or inhalation) were separate from the dose limits for exposure to licensed sources outside the body. The new 10 CFR Part 20 limits the sum of the external whole-body dose (deep dose equivalent) and the committed effective equivalent doses resulting from radioactive material deposited inside the body. In addition, the new 10 CFR Part 20 requires that this sum (the total effective dose equivalent) be maintained ALARA for each individual. These changes to the regulation do not affect the acceptance criteria used by the staff to review the AP600 reactor design.

The SRP acceptance criteria provide assurance that the radiation doses resulting from exposure to licensed radioactive sources both outside and inside the body can each be maintained ALARA and well within the limits of 10 CFR Part 20. The balancing of internal and external exposure necessary to ensure that their sum is ALARA is an operational concern that will be reviewed in conjunction with a combined operating license (COL) application. 10 CFR Part 20, as amended, contains a number of new programmatic requirements that do not affect plant design. Programmatic and operational radiation protection concerns will be addressed by an applicant seeking a COL.

The staff has received sufficient information from Westinghouse to conclude that the radiation protection measures incorporated in the AP600 reactor design offer reasonable assurance that occupational doses will be maintained ALARA and within the limits of 10 CFR Part 20 during all plant operations. The following sections present the bases for the staff's conclusions.

12.2 Ensuring that Occupational Radiation Doses Are As Low As Is Reasonably Achievable

The staff reviewed the information in Chapter 12 of the SSAR to assess adherence to the guidelines in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," as well as the criteria in Section 12.1 of the SRP regarding the radiation protection aspects of the AP600 reactor design. Specifically, the staff reviewed Section 12.1 of the SSAR to ensure that Westinghouse had either committed to adhere to the criteria of the regulatory guides and staff positions referenced in Section 12.1 of the SRP, or had provided acceptable alternatives.

12.2.1 Policy Considerations

In Section 12.1.1 of the SSAR, Westinghouse described the design, construction, and operational policies that have been implemented to ensure that ALARA considerations are factored into each stage of the AP600 design process. Westinghouse has committed to ensure that the AP600 plant will be designed and constructed in a manner consistent with the guidelines of RG 8.8 (Rev.3). In particular, Section 12.1.1.1 of the SSAR states that Westinghouse has met this commitment by reviewing the plant design during the design phase for ALARA considerations. This ALARA policy is consistent with the guidelines of RG 8.8 and is acceptable.

10 CFR Part 20, as amended, requires that all licensees develop, document, and implement a radiation protection program. Specifically, this program should encompass the ALARA concept and include provisions for maintaining radiation doses and intakes of radioactive materials ALARA. The detailed policy considerations regarding overall plant operations and implementation of such a radiation protection program are outside the scope of this design certification review. The operational ALARA policy forms the basis for the operating station's ALARA manual. In order to maintain doses to plant personnel ALARA, the COL applicant should review all plant procedures and modification plans that involve personnel radiation exposure to ensure that the ALARA policy is applied. In addition, a COL applicant referencing the AP600 certified design should submit an operational ALARA policy which conforms to the requirements of 10 CFR Part 20 and the recommendations of RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," Rev.2; RG 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable;" and RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable." This issue was identified as DSER Open Item 12.2.1-1 and COL Action Item 12.2.1-1.

To address this issue, Westinghouse added information to Section 12.1.3 of the SSAR, "Combined License Information." Specifically, Section 12.1.3 now states that the COL applicant will address operational ALARA considerations and it lists the specific RGs that the COL applicant will need to address in its application. Therefore, DSER Open Item 12.2.1-1 is closed.

12.2.2 Design Considerations

Section 12.1.2 of the SSAR describes the objectives for the general design and shielding to minimize the time employees spend in radiation areas and to minimize radiation levels in routinely occupied areas housing equipment requiring attention by plant personnel. Section 12.1.2 of the SSAR also states that these design considerations are consistent with the guidelines in RGs 8.8 and 8.10. Specifically, Section 12.1.2 of the SSAR states that the basic management philosophy guiding the AP600 design is to ensure that exposures are ALARA by designing structures, systems, and components to achieve the following objectives:

- Attain optimal reliability and maintainability, thereby reducing maintenance requirements for radioactive components.

- Reduce radiation fields, thereby allowing operations, maintenance, and inspection activities to be performed in reduced radiation fields.

- Reduce access, repair, and equipment removal times, thereby reducing the time spent in radiation fields.

- Accommodate remote and semi-remote operation, maintenance, and inspection, thereby reducing the time spent in radiation fields.

In addition, Section 12.1.2 of the SSAR describes several design features which satisfy the objectives of the plant radiation protection program:

- The use of highly reliable equipment reduces the frequency of maintenance and associated personnel exposure.

- Except in limited applications where it is necessary for reliability considerations, materials in contact with the reactor coolant system (RCS) have low concentrations of cobalt and nickel. This reduces the amounts of cobalt-60 and cobalt-58 introduced in the RCS. (Cobalt-60 and cobalt-58 are the major sources of radiation exposure during shutdown, maintenance, and inspection activities at light-water reactors, or LWRs.)

- Adequate spacing and laydown areas facilitate access for maintenance and inspection.

- The amount of time spent in radiation areas will be minimized with enhanced servicing convenience for anticipated maintenance or potential repairs, including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair.

- Radioactive systems are separated from non-radioactive systems, and high-radiation sources are located in separate shielded cubicles.

- Equipment requiring periodic servicing or maintenance (e.g., pumps, valves, and control panels) are separated from more radioactive sources (such as tanks and piping).

- Valves located in high-radiation areas are equipped with reach rods or motor operators to minimize operator exposure.
- Equipment and piping are designed to minimize the accumulation of radioactive materials.
- Drains are located at low points.
- Piping is seamless, and the number of fittings is minimized, thereby reducing the radiation accumulation at seams and welds.
- Use of flushing connections minimizes the buildup of crud in system components.
- Systems that produce radioactive waste are located close to radwaste processing systems to minimize the length of piping runs carrying highly radioactive material.
- Pipes that carry resin slurries are run vertically as much as possible and large-radius bends are used instead of elbows, thereby minimizing the potential for pipe plugging.

The design features described in Section 12.1.2 of the SSAR meet the objectives set forth above, are consistent with the guidelines in RG 8.8, and are, therefore, acceptable.

In addition to the features described above, the AP600 reactor design incorporates several features that represent improvements over many currently operating plants:

- The AP600 design accommodates the use of robotic technology to perform maintenance and surveillance in high-radiation areas.
- The design reduces the number of components containing radioactive fluids and clearly and deliberately separates clean areas from potentially contaminated areas.
- The design eliminates the need for waste and recycle evaporators and the boron recycle system, which have historically required frequent operational and maintenance attention, exposing plant personnel to substantial levels of radiation.

These design features, intended to minimize personnel exposures, comply with the guidelines of RG 8.8 and are, therefore, acceptable.

12.2.3 Operational Considerations

Operational considerations regarding the implementation of a radiation protection program are outside the scope of this design certification review. A COL applicant referencing the AP600 certified design should address operational considerations to the level of detail provided in RG 1.70 (Rev.3). Section 12 of the SRP lists the following RGs that the COL applicant should address:

- RG 8.2, "Guide for Administrative Practices in Radiation Monitoring," February 1973
- RG 8.3, "Film Badge Performance Criteria," February 1973

- RG 8.15, "Acceptable Programs for Respiratory Protection," October 1976
- RG 8.20, "Applications of Bioassay for I-125 and I-131," September 1979
- RG 8.26, "Applications of Bioassay for Fission and Activation Products," September 1980
- RG 8.27, "Radiation Protection Training for Personnel at Light-Water Cooled Nuclear Power Plants," March 1981
- RG 8.28, "Audible-Alarm Dosimeters," August 1981
- RG 8.29, "Instructions Concerning Risks from Occupational Radiation Exposure," July 1981

The staff has revised some existing RGs and has developed additional RGs to address some of the new issues contained in the revised 10 CFR Part 20. Some of the new or revised RGs that pertain to Chapter 12 of the SSAR are listed below:

- RG 8.7, "Instructions for Record Keeping and Recording Occupational Radiation Exposure Data," June 1992
- RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," July 1993
- RG 8.25, "Air Sampling in the Work Place," June 1992
- RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," July 1992
- RG 8.35, "Planned Special Exposures," June 1992
- RG 8.36, "Radiation Dose to the Embryo/Fetus," (Draft, Revised) July 1992
- RG 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," June 1993

Addressing the above RGs is outside the scope of this design certification review. A COL applicant should state its intention to follow the guidance contained in the above listed RGs, or state what alternatives will be used. This issue was identified as DSER Open Item 12.2.3-1 and COL Action Item 12.2.3-1.

To address this issue, Westinghouse added information to Section 12.1.3 of the SSAR, “Combined License Information,” to list the specific RGs that the COL applicant will need to address in its application. Therefore, DSER Open Item 12.2.3-1 is closed.

On the basis of its review of the information supplied by Westinghouse, as described above, the staff concludes that Westinghouse's policy and design considerations meet the criteria of Section 12.1 of the SRP and are, therefore, acceptable.

12.3 Radiation Sources

The staff reviewed the descriptions of the radiation sources given in Chapter 11 and Section 12.2 of the SSAR to assess completeness against the guidelines in RG 1.70 and the criteria in Section 12.2 of the SRP. The contained source terms described in the SSAR are used as the basis for the radiation design calculations (shielding and equipment qualification) and personnel dose assessment. The airborne radioactive source terms in the SSAR are used in the design of ventilation systems and for assessing personnel dose. The staff reviewed the source terms in the SSAR to ensure that Westinghouse had either committed to follow the guidelines of the RGs and staff positions set forth in Section 12.2 of the SRP, or provided acceptable alternatives.

12.3.1 Contained Sources

In Section 12.2.1 of the SSAR, Westinghouse described plant components that can become significant sources of radiation during plant operations including shutdown. To calculate the source terms used for shielding design, Westinghouse assumed 0.25-percent fuel cladding defects at full-power operation. Other than the reactor core, the RCS is the principal contributor to radiation levels in the containment. Sources of radiation in the RCS include the following:

- fission products (which are released from defective fuel cladding)
- activation products
- corrosion products.

Of these radiation sources, the activation product nitrogen-16 (N-16) is the predominant radionuclide in the RCS piping, reactor coolant pumps (RCPs), and steam generators (SGs) during plant operations. The staff reviewed Westinghouse's estimates of N-16 activity levels in various parts of the RCS, and found them comparable to the levels measured at operating plants. Further, N-16 activities outside the containment during normal power operations are insignificant, because no reactor coolant is directed outside the containment during normal operations except RCS sampling.

Westinghouse based its estimates of corrosion product activity on the assumptions contained in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," dated April 1995. These corrosion product activity estimates were adjusted from reference plant values provided in ANSI/ANS-18.1, "Radiation Source Term For Normal Operation of Light Water Reactors," dated 1984. Following plant shutdown, the predominant long-term sources of radiation in the containment are the spent fuel assemblies.

Section 12.2.1 of the SSAR describes contained and airborne radioactive sources, which are used as inputs for dose assessment calculations, as well as shielding and ventilation design considerations. Specifically, the SSAR contains the following information:

- reactor core and spent fuel average gamma ray source strengths
- N-16 activities
- primary system component source strengths and specific activities

Chapter 12 of the SSAR lists all large contained sources of radiation in the reactor containment, fuel handling area, auxiliary building, and radwaste buildings. For each of these contained sources, the associated maximum activity levels are listed by isotope. The SSAR also includes the assumptions that Westinghouse used in arriving at quantitative values for these contained and airborne source terms, on the basis of General Design Criterion (GDC) 61 and 10 CFR Part 20.

Section 12.2.1.1.10 of the SSAR states that the COL applicant will identify additional contained radiation sources used for instrument calibration or radiography. Since the staff needs to review any additional contained radiation sources used for instrument calibration or radiography that are not listed in the SSAR, this issue was identified as DSER Open Item 12.3.1-1 and COL Action Item 12.3.1-1.

To address this issue, Westinghouse added information to Section 12.2.3 of the SSAR, "Combined License Information," stating that the COL applicant will address any additional contained radiation sources not identified in Subsection 12.2.1 of the SSAR, including radiation sources used for instrument calibration. Therefore, DSER Open Item 12.3.1-1 is closed.

12.3.2 Airborne Radioactive Material Sources

In Section 12.2.2 of the SSAR and in responses to the staff's requests for additional information (RAIs), Westinghouse described the sources of airborne radioactivity for the AP600 reactor design. According to these descriptions, the main source of airborne radioactivity within the plant is primary equipment leakage.

Airborne radioactive source terms are used in the design of ventilation systems and for personnel dose assessment. RG 1.70 states that this section should include a tabulation of the calculated concentrations of airborne radioactive material, by nuclide, expected during normal operation and AOOs for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. Section 12.2 of the SSAR describes the assumptions and parameters used to determine the expected airborne radioactivity concentration levels during normal operations in the containment, fuel handling area, radwaste building, and auxiliary building. In response to staff requests, Westinghouse provided tables estimating doses to plant operating personnel from airborne radioactivity in various plant rooms and buildings. For each of these rooms and buildings, these tables detail airborne activity, total occupational time spent, and inhalation whole body dose.

The SSAR provides system layouts within rooms or cubicles, and information about the type and size of components in these systems. In its response to Q471.1, dated January 22, 1993, Westinghouse provided information on dimensions, volumes, material, and equipment self-shielding for dominant radiation sources within the plant.

12.3.3 Sources Used in Post-Accident Shielding Review

In Section 12.2.1.3 of the SSAR, Westinghouse described the use of a degraded core source term model (from "Passive ALWR Source Term", February 1991) to develop source strengths for contained sources within the plant. Westinghouse then used these source strengths to calculate worker doses incurred during vital area access/activities. This degraded core source term model differed from the TID 14844 source term required by 10 CFR 50.34(f)(2)(vii) (Item II.B.2 of NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," dated August 1980), and NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980. Because the Westinghouse source term model differed from the NRC accepted TID 14844 source term model, the staff identified this issue as DSER Open Item 12.3.2-1.

Westinghouse then agreed to base the AP600 core activity release model on the model used for the NRC's new source term defined in NUREG-1465, "Accident Source Term For Light-Water Nuclear Power Plants." In the event of core degradation, no high activity sump solution would be recirculated outside the AP600 containment. In order to insure that the containment shielding would still provide sufficient shielding capability using the revised source term model, Westinghouse analyzed the containment shielding using the NUREG-1465 source term and determined that the AP600 containment addresses this post-LOCA source term. This is acceptable to the staff and, therefore, DSER Open Item 12.3.2-1 is closed.

Even though the staff has accepted Westinghouse's revised source term model, this model uses a source term which differs from the TID-14844 source term requirement in 10 CFR 50.34(f)(2)(vii). An exemption from this requirement is justified in Section 20.6 of this report.

12.4 Radiation Protection Design

The staff reviewed the facility design features, shielding, ventilation, and area and airborne radiation monitoring instrumentation contained in the SSAR for adherence to the guidelines in RG 1.70 and the criteria in Section 12.3 of the SRP. The purpose of this review was to ensure that Westinghouse had either committed to follow the guidelines of the RGs and staff positions, or offered acceptable alternatives. The following sections present the staff's findings.

12.4.1 Facility Design Features

Several facility design features have been incorporated into the AP600 reactor design to help maintain occupational radiation exposures ALARA in accordance with the guidance in RG 8.8. These design features are founded on the ALARA design considerations described in Section 12.1 of the SSAR and discussed in Section 12.2.2 of this report.

The AP600 reactor vessel design includes an integrated head package that incorporates the following design features which will help minimize the time, manpower, and radiation exposure associated with head removal and replacement during refueling operations:

- the head lifting rig
- control rod drive mechanisms (CRDMs)
- lift columns
- missile shield
- CRDM cooling system, and
- power and instrumentation cabling

Permanently installed shielding is integral in the head package for reducing work area dose rates from the CRDM drive shafts and the incore instrumentation (when the incore flux detector thimbles are withdrawn into their conduits).

The AP600 design replaces conventional top-mounted thermocouple and movable incore flux detectors with a combination thermocouple/incore detector system. This system eliminates the need to disassemble and reassemble the instrument port conoseals at each refueling. This task has historically resulted in relatively high radiation exposures.

Insulation in the area of the reactor vessel nozzle welds is fabricated in sections with quick-disconnect clasps to facilitate insulation removal for weld inspection. Permanent identification markings of the insulation sections will accommodate rapid reinstallation.

The AP600 RCPs are hermetically sealed, canned motor pumps. Because the shaft for the impeller and rotor is contained within the reactor coolant pressure boundary (RCPB), seals are not required to restrict RCS leakage out of the pump. The RCPs are designed to require infrequent maintenance and inspection. When maintenance or replacement is required, the pump can be unbolted from a flange connection for quick removal to a low-radiation (background) work area using a specially designed pump removal cart, thereby reducing personnel exposure.

Steam generators (SGs) in the AP600 reactor will be designed to use robotic equipment for inspection and maintenance activities. The lower portion of the primary channel head is hemispherical and merges into a cylindrical portion, which mates with the tube sheet. This arrangement provides enhanced robotic access to all tubes, including those at the periphery of the tube bundle, without the need for a manned entry into the channel head. The area surrounding the SGs has adequate pull and laydown areas and permanent platforms. In addition, the SGs are provided with hand-holes, manways, and removable insulation. These features all enhance accessibility and reduce overall exposure during SG inspection and maintenance activities. To minimize the deposit of radioactive corrosion products on the channel head surfaces, and enhance the decontamination of these surfaces, the SG channel head cladding is machined or electropolished to a smooth surface. The AP600 steam generators also have the provision for draining the channel head area of reactor coolant. These features enhance the ability to inspect, replace, and repair the AP600 SGs.

The SSAR states that the AP600 design will minimize the use of snubbers, thereby reducing maintenance and inspection needs. Resin slurry piping systems will be run vertically as much as practicable, and will have remote backflushing capabilities, to reduce personnel exposure during servicing. Pumps and connected piping will be flanged, where feasible, to facilitate pump removal. Pump casings will be provided with drain connections to facilitate maintenance. Floor drain systems in the radiologically controlled area (RCA) will be completely separated from non-RCA drains to prevent contamination of nonradioactive areas. Radioactive drain lines will terminate below the minimum water level in the sumps to prevent the migration of airborne radioactivity from the sump air space back into other rooms. The heating, ventilation, and air conditioning (HVAC) systems will maintain the airflow direction from areas of lower potential airborne contamination, to areas of higher potential airborne contamination.

In addition to designing equipment to comply with ALARA guidelines, the AP600 plant layout is designed to reduce personnel exposures. Adequate work and laydown space will be provided at each inspection and maintenance station. Rigging and lifting equipment will be provided to facilitate the removal, transport, or replacement of equipment or portable shielding during maintenance activities. Adequate illumination and support services (e.g., power, compressed air, water, ventilation, and communications) will be available at work stations. Tube pull areas for components that handle radioactive fluids will be designed with curbs, drains, and coated floors to prevent the spread of contamination in the event of spills. Valves associated with highly radioactive components will be separated from other components, and will be located in shielded valve galleries. Radioactive piping will be routed through pipe chases to minimize personnel exposures. The equipment and layout design features described above conform with the guidelines of RG 8.8 for maintaining occupational radiation exposures ALARA and are, therefore, acceptable.

The AP600 design also incorporates several features to minimize the buildup, transport, and deposition of activated corrosion products in the RCS and auxiliary systems. The SSAR states that, except in cases where no proven alternative material exists, the AP600 design will reduce or eliminate the use of materials containing cobalt and nickel which are in contact with reactor coolant. The SSAR further states that the majority of materials exposed to high-temperature reactor coolant will have cobalt impurities of no more than 0.05-weight percent cobalt. The major use of nickel-based alloys in the RCS is in the inconel SG tubes. Inconel SG tubing will be limited to 0.015-weight percent cobalt, while the surfaces on the inside of the SGs other than the tubing will have a cobalt limit of 0.10-weight percent cobalt. Materials used for rod cluster control assemblies, gray rod cluster control assemblies, and secondary source rod cladding will be Type 304 stainless steel, with an assumed maximum cobalt limit of 0.12-weight percent cobalt. Bolting materials in reactor internals and other small components in regions of high neutron flux will be limited to 0.20-weight percent cobalt. Auxiliary components such as valves, piping, instrumentation, and welding materials will not be limited in cobalt content, but will have average concentrations of approximately 0.20-weight percent.

The presence of antimony in RCP journal bearings in some current generation plants has increased the number of hot particles at these plants. The AP600 design will restrict the presence of antimony to less than one percent in all materials that contact the RCS and will prohibit antimony completely from the RCP and its bearings. Crud traps created in weld areas will be minimized by using butt welds. Pump casings will be provided with drain connections, and valves in radioactive fluid systems under 5.08 cm (2 in) in diameter will be designed for zero stem leakage. Tanks containing radioactive liquid will have drain pipes connected at the lowest part of the tank, and will have convex or sloped-bottom designs to minimize radioactivity deposition. Piping systems used to transport process resins will be designed to minimize pipe plugging. Smooth curves will replace elbows in piping runs, where practicable, to reduce potential crud traps. Equipment and piping containing radioactive materials will have provisions for draining and flushing. These design features, which are intended to minimize the buildup, transport, and deposition of activated corrosion products in the RCS and auxiliary systems, are based on the guidelines in RG 8.8 and are, therefore, acceptable.

Westinghouse provided the staff with detailed drawings of the AP600 plant layout which indicate the nine radiation zones used in the plant design. These radiation zones serve as a basis for classifying occupancy and access restrictions for various areas within the plant during normal operations and accident conditions. On this basis, the maximum design dose rates are established for each zone, and are used as input for shielding of the respective zones. The staff has performed confirmatory shielding design calculations in several areas of the AP600 design for normal and accident conditions. On the basis of the staff's review of the detailed zoning drawings and consistent satisfactory confirmatory calculations, the staff concludes that Westinghouse's method of plant zoning, for normal operations, is consistent with the guidance in RG 1.70 and the SRP and is acceptable.

As required by 10 CFR 50.34 (f)(2)(vii), an applicant must fulfill the following requirements:

- Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain radioactive materials.

- Design, as necessary, adequate access to important areas and protection of safety equipment from the radiation environment.

In NUREG-0660 and NUREG-0737, Item II.B.2 describes source-term information that should be used to calculate post-accident radiation levels. This item states that the post-accident plant dose rates should be such that the dose to plant personnel should not exceed 5E-2 sieverts (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident (per GDC 19). The dose rate in areas requiring continuous occupancy should be less than 15E-5 sieverts/hr (15 mrem/hr) averaged over 30 days.

Item II.B.2 of NUREG-0737 describes a "vital area" as any area that will, or may, require occupancy to permit an operator to aid in the mitigation of, or recovery from, an accident. Item II.B.2 also recommends that the SSAR include a listing of all vital areas in the plant, and provide a summary of the integrated doses to personnel in each of the plant areas requiring either continuous occupancy or infrequent access for the duration of the accident. (These doses should include exposure received while in transit between vital areas.) In the DSER, the staff stated that the zoning of several vital areas of the AP600 that require access during accident conditions appeared to be inappropriate. Furthermore, the remote shutdown area had not been listed as a vital area for the AP600. These issues were identified as DSER Open Items 12.4.1-1 and 12.4.1-2.

To address these issues, Westinghouse, in response to Q471.22, added Section 12.4.1.8, "Post-Accident Actions," to the SSAR. This section lists all of the vital plant areas requiring post-accident accessibility and states that all vital areas can be accessed following an accident for less than 5E-2 sieverts (5 rem) to the whole body or 7.5E-1 sieverts (75 rem) to the extremities. Figure 12.3-2 of the SSAR contains plant radiation zone maps, which reflect maximum radiation fields over the course of an accident. Therefore, DSER Open Item 12.4.1-1 is closed.

In its letter to the NRC dated May 7, 1997, Westinghouse stated that the remote shutdown area does not have to be classified as a vital area during a design-basis accident because it is designed to allow control of a shutdown following an evacuation of the control room, coincident with the loss of offsite power and a single active failure. Because no other design-basis event is postulated beyond the scenario resulting in evacuation of the main control room, no emergency habitability provisions (including post-accident radiation protection and shielding measures) are provided for the workstation area. The staff concludes that this justification is acceptable and, therefore, DSER Open Item 12.4.1-2 is closed.

12.4.2 Shielding

The objective of the plant's radiation shielding is to protect plant personnel and the public against exposure from the various sources of ionizing radiation in the plant during normal operation (including AOOs and maintenance) and during accident conditions. The AP600 design also includes shielding, where required, to mitigate the possibility of radiation damage to materials. The SSAR states that radioactive components and piping will be separated from non-radioactive components and piping to minimize personnel exposure during maintenance and inspection activities. When radioactive piping must be routed through corridors or other low-radiation zones, shielded pipe chases are provided. Where applicable, pumps and other support equipment for components that contain radioactive material are located outside the component cubicle in separate shielded cubicles. Shielded compartments have labyrinth entrances to minimize radiation streaming directly through access openings. Penetrations are located so there is no direct line from the radioactive source to adjacent occupied areas. Space is allocated, where needed, for the erection of temporary shielding. These shielding techniques comply with the guidelines contained in RG 8.8 for protecting plant personnel and the public against exposure from various sources of ionizing radiation in the plant and are, therefore, acceptable. The staff concludes that the information contained in Section 12.3.2 of the SSAR is acceptable to adequately address the relevant requirements of 10 CFR Part 20, 10 CFR 50.34(f)(2)(vii), Item II.B.2 of NUREG-0660 and NUREG-0737, and GDC 61.

There have been several recent instances of overexposures, or near overexposures, at current generation pressurized-water reactors (PWRs). One area where potentially lethal exposures have occurred is the reactor cavity. Personnel have been overexposed upon entering PWR reactor cavities when the movable incore flux detectors have been withdrawn. The AP600 design does not have movable incore detectors; however, the SSAR did not identify the use of administrative controls (lock and access control procedures) in order to limit access to the reactor cavity as a COL Action Item. This issue was identified as DSER Open Item 12.4.2-1 and COL Action Item 12.4.2-1.

To address this issue, Westinghouse added information to Section 12.3.5 of the SSAR, "Combined License Information," stating that the COL applicant will address the administrative controls to control access to the reactor cavity. Therefore, DSER Open Item 12.4.2-1 is closed.

Another area where personnel can be exposed to potentially lethal doses of radiation is in the vicinity of the fuel transfer tube, when a spent fuel assembly passes through this tube. In the AP600 design, there is a minimum of 1.22 m (4 ft) of concrete shielding surrounding the fuel transfer tube. This shielding is sufficient to reduce radiation levels surrounding the tube to within local radiation zone limits; however, the design has a 5.08 cm (2 in) seismic air gap between the end of the transfer tube shielding and the steel containment wall. This seismic gap provides a potential radiation streaming path for personnel in the vicinity of the fuel transfer tube during fuel transfer. This issue was identified as Open Item 12.4.2-2.

To address this issue, Westinghouse modified Section 12.3.2.2.9 of the SSAR to describe the use of a water-filled bladder that will be used in this seismic gap to provide radiation shielding during fuel transfer. This "expansion gap" radiation shield will also accommodate the relative movement between the containment and the concrete transfer tube shielding with no loss in shield integrity. The staff concludes that the addition of the water shield is acceptable and, therefore, DSER Open Item 12.4.2-2 is closed.

The fuel transfer tube has a removable concrete or steel hatch that allows access for periodic inspection of the fuel transfer tube welds. In the DSER, the staff stated that the COL applicant should administratively control the opening of this hatch (treating it as a very high-radiation area under 10 CFR Part 20), and keep this hatch locked during all spent fuel transfer operations. The COL applicant should provide adequate administrative controls (lock and access control procedures) in order to limit access to the fuel transfer tube during refueling operations. This issue was identified as DSER Open Item 12.4.2-3 and COL Action Item 12.4.2-2.

To address this issue, Westinghouse added information to Section 12.3.2.9 of the SSAR, "Combined License Information," stating that the opening of this hatch will be administratively controlled and the hatch will be treated as an entrance to a very high-radiation area under 10 CFR Part 20. Westinghouse has also amended Section 12.3.5 of the SSAR to state that the COL applicant will address the administrative controls to control access to the fuel transfer tube during refueling operations. Therefore, DSER Open Item 12.4.2-3 is closed.

Section 12.3.2 of the SRP states that an applicant must describe how its shielding parameters were determined, including pertinent codes, assumptions, and techniques used in the shielding calculations. The AP600 SSAR describes the shielding codes used to determine the adequacy of the station shielding design. Specifically, Westinghouse states that it used the shielding codes ANISN, DOT, QAD, and ORIGEN to verify the effectiveness of the primary shield (around the reactor core). The SSAR also describes shielding codes used to verify gamma-source shielding elsewhere in the plant. The ANISN, DOT, QAD, and ORIGEN codes are listed as acceptable shielding codes in the SRP and, therefore, the staff finds the use of these codes to evaluate the adequacy of the AP600 station shielding design to be acceptable.

12.4.3 Ventilation

The AP600 ventilation systems are designed to protect personnel and equipment from extreme environmental conditions, and to ensure that personnel exposure to airborne radioactivity levels is minimized and maintained ALARA and within the limits of 10 CFR Part 20. Further, the design ensures that the dose to control room personnel during accident conditions will not exceed the limits specified in GDC 19. The AP600 design incorporates the following features to maintain personnel exposures ALARA:

- Ventilation air is supplied directly to the clean areas of the plant and exhausted from the potentially contaminated areas, thereby creating a positive flow of air from clean areas to potentially contaminated areas.

- Negative or positive pressure is used appropriately in plant areas to prevent exfiltration or infiltration of possible airborne radioactive contamination, respectively.

- Equipment vents and drains are piped directly to a collection system, preventing contaminated fluid from flowing across the floor to a drain.

- Valves in radioactive systems that are larger than 6.35 cm (2.5 in) are provided with double sets of packing with lantern rings and leakoff connections that are piped to drain headers.

- Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure.

These design criteria adhere to the guidelines of RG 8.8 for maintaining doses ALARA and are acceptable. Westinghouse's response to Q471.3, dated September 1, 1994, describes airborne radioactivity in rooms (or areas) as a function of the airborne concentration, and the removal of activity by the ventilation system. The source of airborne radioactivity for a room or area is primarily from equipment leakage within the specified area. Leakage values are provided from historical data from operating plants. Westinghouse's response includes tables of airborne dose estimates for all buildings and rooms with equipment containing reactor coolant. These dose estimates were calculated using the estimated room/area occupancy times and equilibrium concentrations of all radioisotopes from the AP600 source term.

On the basis of Westinghouse's calculations of airborne concentrations and the estimates of occupancy times to perform typical equipment surveillance, inservice inspection, maintenance, and repair operations, the staff concludes that Westinghouse has demonstrated that worker intakes will be maintained ALARA. These calculational methods and assumptions are consistent with provisions of the SRP, and are acceptable.

12.4.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The purpose of the plant area radiation monitoring equipment is to alert operators and other station personnel to changing or abnormally high radiation conditions in the plant in order to prevent possible personnel overexposures and aid health physics personnel in keeping worker doses ALARA. Control room displays provide information on monitor readings, alarm set points, and operating status. The area radiation monitors consist of microprocessors and Geiger-Mueller tubes for detecting gamma-radiation. The area radiation monitors are located according to the potential for significant radiation levels in an area and the expected occupancy of the area. The SSAR states that area radiation monitors will not be located in high-radiation areas that are normally not accessible. While reviewing Chapter 11 of the SSAR, the staff raised some concerns regarding the types of alarms associated with the plant area radiation monitors. In Q471.26, the staff requested that Westinghouse clarify this information. In

Q471.25, the staff also requested that Westinghouse clarify which radiation monitoring channels are safety-related. Section 11.5.6 of the SSAR stated that criticality monitors, as required in 10 CFR 70.24, are not provided because the design of the fuel pool rack precludes criticality under postulated normal and accident conditions. In Q 471.24, the staff requested that Westinghouse further justify why it believes that criticality monitors are not required for the AP600. These three RAIs (Qs 471.24-26) together were identified as DSER Open Item 12.4.4-1.

In response to the RAI to provide more information on radiation monitor alarms (Q 471.26), Westinghouse modified portions of Chapter 11 in Revision 3 of the SSAR to state that area radiation monitors will have a local readout and audible alarm located in each monitored area. In addition, visible alarms will be provided outside each monitored area so that they can be seen by operating personnel before entering the monitored area. This response is acceptable to the staff. In addition, the location of these monitors, as described in the SSAR, meets the criteria of ANSI/ANS Standard HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light-Water Nuclear Reactors," dated May 1991, and is acceptable to the staff.

In its response to the RAI on which radiation monitors are safety-related (Q 471.25) dated November 4, 1994, Westinghouse provided information stating that the main control room air duct and containment high-range radiation monitors are safety-related. This information is acceptable to the staff.

In Revision 15 to the SSAR, Westinghouse responded to the RAI on criticality monitoring (Q 471.24), by modifying Section 11.5.6.4 to provide two fixed radiation monitors (which meet the radiation sensitivity requirements of 10 CFR 70.24 for criticality monitors) to provide coverage on the operating deck level of the Annex Building where new and spent fuel will be handled. In addition, a portable radiation monitor will be used on the crane handling fuel to detect potential criticalities during fuel handling operations. The staff concludes that Westinghouse's response to the RAI on criticality monitoring is acceptable.

On the basis of Westinghouse's responses to the RAIs on criticality monitors, safety-related radiation monitoring channels, and radiation monitor local readouts (Qs 471.24-26), DSER Open Item 12.4.4-1 is considered closed.

10 CFR 50.34(f)(2)(xvii) (Section II.F.1(3) of NUREG-0737) requires that the reactor containment be equipped with two physically separate radiation monitoring systems that are capable of measuring up to 10^5 Gr/hr (10^7 R/hr) in the containment following an accident. Verification of compliance with this requirement was identified as DSER Confirmatory Item 12.4.4-1. In Section 11.5.6.2 of the SSAR, Westinghouse stated that the AP600 design will incorporate four electrically independent ion chambers located inside the containment to measure high-range gamma radiation. These detectors will be mounted on the inner containment wall in widely separated locations, and will have an unobstructed "view" of a representative volume of the containment atmosphere. The design and qualification of these monitors complies with the guidelines of RG 1.97, "Instrumentation For Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev. 3, dated May 1983; and NUREG-0737, with respect to detector range, response, redundancy, separation, onsite calibration, and environmental design qualification.

The staff, therefore, finds these monitors to be acceptable, and DSER Confirmatory Item 12.4.4-1 is closed.

Chapter 12 of the SSAR describes the airborne radiation monitoring system. The airborne radiation monitoring equipment will be placed in selected areas and ventilation systems to give plant operating personnel continuous information about the airborne radioactivity levels throughout the plant. The airborne radioactivity monitors are located upstream of the filter trains, so that they monitor representative radioactivity concentrations from the areas being sampled. The airborne radiation monitoring system, as described in the SSAR, meets the scope of the post-accident monitoring requirements set forth in GDC 64 and the guidance of RG 1.97, and is acceptable. Section 12.3 of the SRP states that airborne radioactivity monitors shall be able to detect the time integrated change of the most limiting particulate and iodine species equivalent to those concentrations specified in Appendix B of 10 CFR Part 20 (one derived air concentration (DAC)) in each monitored plant area within 10 hours (i.e., monitors should be sensitive enough to measure 10 DAC-HRs). Since Westinghouse had not originally addressed the sensitivity of the airborne radiation monitors with respect to being able to detect one DAC within 10 hours, this issue was identified as DSER Open Item 12.4.4-2.

To address this issue, Westinghouse submitted a response to RAI Q471.23 dated May 7, 1997, which provides information concerning the sensitivity of the airborne radiation monitors and the detector response times. This response indicates that airborne process radiation monitors will be employed to monitor the seven areas of the plant which can be accessed by plant personnel. These monitors will be used to measure airborne radiation levels during normal operations. In Revision 12 of Section 11.5.2.3 of the SSAR, Westinghouse states that these airborne radiation monitors are sensitive enough to detect concentrations of 10 DAC-HRs. The specific radiation monitors that are included in this category are identified in Table 11.5-1 of the SSAR. On the basis of the above, the staff concludes that this design is acceptable. Therefore, DSER Open Item 12.4.4-2 is closed.

Section 12.3 of the SRP states that the SSAR must provide the criteria and methods for obtaining representative in-plant airborne radioactivity concentrations in all work areas. Further, Item III.D.3.3 of NUREG-0660 (10 CFR 50.34(f)(2)(xxvii)), states that each applicant should provide equipment and associated training and procedures for accurately determining the airborne iodine concentrations in areas within the facility where personnel may be present during an accident. Section 1.9 of the SSAR states that portable monitoring, including that required to meet Item III.D.3.3 of NUREG-0737, is the responsibility of the COL applicant. This issue was identified as DSER Open Item 12.4.4-3 and COL Action Item 12.4.4-1.

To address this issue, Westinghouse added information to Section 12.3.5 of the SSAR, "Combined License Information," stating that the COL applicant will address the criteria and methods for obtaining representative airborne radioactivity concentrations in work areas. Therefore, DSER Open Item 12.4.4-3 is closed.

12.5 Dose Assessment

The staff reviewed Westinghouse's dose assessment contained in Section 12.4 of the SSAR for completeness against the guidelines in RG 1.70 (Rev. 3) and the criteria set forth in Sections 12.3 and 12.4 of the SRP. This review consisted of ensuring that Westinghouse had either committed to following the criteria of the RGs and staff positions in Sections 12.3 and

12.4 of NUREG-0800, or provided acceptable alternatives. In addition, the staff selectively compared Westinghouse's dose assessment, for specific functions and activities, against the experience of operating PWRs.

In Section 12.4 of the SSAR, Westinghouse provides an assessment of the annual occupational radiation dose that would be received by the operating staff of an AP600 facility. Tables 12.4-1 through 12 of the SSAR provide estimated doses for various jobs and inspections performed in the plant during maintenance and refueling periods, and for power operations. These activities result in an estimated total annual dose of 0.67 person-sievert (67 person-rem). Section 12.4 contains no separate determination of doses attributable to airborne activity; however, experience at operating LWRs demonstrates that the dose from airborne radioactivity is not a significant contribution to the total dose.

In performing its dose assessment, Westinghouse reviewed exposure data from operating plants to obtain a breakdown of the doses incurred within each dose assessment category referenced in RG 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants - Design Stage Man-Rem Estimates," Rev. 1, dated June 1979. Westinghouse then adjusted these values to account for AP600 design features to obtain the estimated annual dose of 0.67 person-sievert (67 person-rem).

The cumulative annual dose of 0.67 person-sievert (67 person-rem) for operating an AP600 plant is consistent with the Electric Power Research Institute (EPRI) design guideline of 1.0 person-sievert (100 person-rem) per year and compares favorably with the average current PWR experience. Although this assessment is not in the format specified in RG 8.19, it is a detailed dose assessment that meets the intent of RG 8.19, and is acceptable.

As discussed above, the AP600 design incorporates several improvements over current operating PWR designs. These improvements significantly reduce the personnel exposure associated with operational and maintenance activities. The occupational radiation exposure resulting from unscheduled repairs on valves, pumps, and other components will be lower for the AP600 than for current plant designs because of the reduced radiation fields, increased equipment reliability, and reduced number of components relative to currently operating plants. Historically, special maintenance performed on SGs has resulted in significant personnel doses. Westinghouse estimates that the annual dose incurred for special maintenance of the AP600 SGs will be slightly more than 0.01 person-sievert (1 person-rem). These low estimated SG doses are the result of improved SG design and improved primary and secondary water chemistry controls. Westinghouse does not predict any special maintenance activities for the canned motor RCPs used in the AP600 design. However, additional inspection of the RCPs is planned for the first AP600 plant.

The AP600 radwaste system design incorporates an uncomplicated approach to waste processing. The AP600 does not use waste or boron recycle evaporators, and it does not have a catalytic hydrogen recombiner in the gaseous radwaste system. Elimination of these high-maintenance components contributes significantly to lower anticipated doses associated with waste processing activities.

Since the refueling process is labor intensive, detailed planning and coordination are essential in order to maintain personnel doses ALARA. The AP600 design has incorporated advanced technology (e.g., integrated reactor vessel head package, combination thermocouple and flux detectors, permanent reactor cavity seal ring, and "pass and one-half" stud tensioning procedures) into the refueling process, thereby reducing doses during refueling operations.

The direct radiation at the site boundary from the containment and other plant buildings is negligible. The containment shield building walls are a minimum of 0.91 m (3 ft) thick, reducing radiation levels outside the containment to less than 2.5 μSv/hr (0.25 mrem/hr) from sources inside containment. The AP600 design also provides storage for refueling water inside the containment, instead of in an outside storage tank, thereby eliminating the refueling water storage tank as an offsite radiation source.

The Westinghouse AP600 design assumptions, and the models on which dose estimates are based for occupational exposures, meet the criteria of SRP 12.4, and are acceptable.

12.6 Health Physics Facilities Design

The staff reviewed the description of the health physics facilities design given in Section 12.5 of the SSAR for completeness against the guidelines in RG 1.70, and against the criteria given in Section 12.5 of the SRP. Sections 12.5 of RG 1.70 and the SRP contain three areas of review:

- organization
- equipment, instrumentation, and facilities
- procedures

Section 12.5 of the SSAR briefly describes the equipment and facilities contained in the AP600 design, including a discussion of whole body and portable survey instrumentation. The SSAR also discusses the facilities that are displayed on the plant layout drawings, and describes traffic flow patterns that personnel would take through the health physics access control area. However, the SSAR makes no reference to the organization or procedures that will be used to ensure that personnel radiation exposures will be maintained ALARA. The COL applicant should fully address these topics. This issue was identified as DSER Open Item 12.6-1 and COL Action Item 12.6-1.

To address this issue, Westinghouse added information to Section 12.5.5 of the SSAR, "Combined License Information," stating that the COL applicant will address the organization and procedures used for adequate radiological protection and will provide methods so that personnel radiation exposures will be maintained ALARA. Therefore, DSER Open Item 12.6-1 is closed.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of the Applicant

In the DSER, the staff stated that Westinghouse should add a COL action item to address the organizational structure of the applicant. This was identified as DSER Open Item 13.1-1 and COL Action Item 13.1-1. In Section 13.1 of the SSAR, Westinghouse states that the organizational structure of the applicant is the responsibility of the COL applicant. However, Westinghouse further states that the organizational structure should be consistent with the - human system interface (HSI). This staff's evaluation of this matter is discussed in Sections 18.2, 18.6, and 18.10 of this report. In Section 13.1.1, Westinghouse added a statement that a COL applicant referencing the AP600 certified design will address the adequacy of the organizational structure. Therefore, DSER Open Item 13.1-1 is closed.

13.2 Training

In the DSER, the staff stated that Westinghouse should add a COL action item to address personnel training by the COL applicant. This was identified as DSER Open Item 13.2-1 and COL Action Item 13.2-1. In Section 13.2 of the SSAR, Westinghouse states that training programs are the responsibility of the COL applicant. However, Westinghouse further references WCAP-14655, which describes the input from the designer on the training of operations personnel who participate as subjects in the human factors engineering verification and validation. The staff's evaluation of this matter is discussed in Section 18.10 of this report. In Section 13.2.1, Westinghouse added a statement that a COL applicant referencing the AP600 certified design will develop and implement training programs for plant personnel. Therefore, DSER Open Item 13.2-1 is closed.

13.3 Emergency Planning

Section 13.3 of the SSAR indicates that emergency planning is the responsibility of the COL applicant and is not within the scope of the AP600 design certification application. Additionally, the SSAR states that communication interfaces between the plant control room and the emergency response centers discussed in NUREG-0696 are outside the scope of the AP600 design certification application. In addition, although Westinghouse states that the AP600 PRA supports simplification of offsite emergency planning (as described in SECY-93-087), the designer states in Section 1.9.5.3.7 of the SSAR that this issue is the responsibility of the COL applicant. The staff agrees that emergency planning will be addressed by the COL applicant referencing the AP600 design and will significantly depend on plant- and site-specific characteristics. In the DSER, the staff stated that Westinghouse should designate emergency planning as a COL action item. This was identified as DSER Open Item 13.3-1 and COL Action Item 13.3-1. In Revision 15 of the SSAR, Westinghouse added SSAR Section 13.3.1, "Combined License Information Item," which indicates that the COL applicant referencing the AP600 certified design will address emergency planning, including post-72 hour actions and its communication interface. The staff finds this to be acceptable, and therefore, DSER Open Item 13.3-1 is closed.

Nevertheless, certain design features, facilities, functions, and equipment necessary for emergency planning must be considered in the standard design. Specifically, in accordance with 10 CFR 50.34(f)(2)(xxv), the standard design must address the characteristics of the onsite Technical Support Center (TSC) and Operational Support Center (OSC), and demonstrate conformance of these facilities with Supplement 1 to NUREG-0737 and NUREG-0696. The design should also include an onsite decontamination facility in accordance with 10 CFR 50.47(b)(11), Subsection IV.E.3 of Appendix E to 10 CFR Part 50, and NUREG-0654/FEMA-REP-1, to provide the capability for controlling radiological exposures and providing decontamination facilities for onsite individuals.

The mission and major tasks of the TSC and OSC are provided in SSAR Sections 18.8.3.5 and 18.8.3.6, respectively. The OSC location is identified as the ALARA Briefing Room (40318) and is shown as such in Revision 24 of Figure 1.2-18, "Annex Building General Arrangement Plan at Elevation 100'-0" & 107'-2"," of the SSAR. The TSC is located in the Annex building (40403) at elevation 117'-6", as shown in Revision 24 of Figure 1.2-19, "Annex Building General Arrangement Plan at Elevation 117'-6" & 126'-6"," of the SSAR. In its response to Q100.6 dated February 25, 1993, Westinghouse indicates that the onsite decontamination facility will be located in the health physics area in the Annex building. In the DSER, the staff stated that information related to the emergency response facilities associated with the AP600 design should be incorporated into the SSAR. This was identified as DSER Open Item 13.3-2. A clear cross-reference for information related to emergency response facilities is provided in Revision 13 to Section 13.3 of the SSAR. The staff finds this to be acceptable, and therefore, DSER Open Item 13.3-2 is closed.

The design considerations for the TSC are described in Section 18.8.3.5 of the SSAR. In the DSER, the staff stated that Westinghouse should provide information regarding the size of the TSC. This was identified as DSER Open Item 13.3-3. In Revision 19, SSAR Section 18.8.3.5 was revised to provide information on how the size of the TSC conforms to the guidance in NUREG-0696, which directs the TSC working space to be sized for a minimum of 25 persons with a minimum of 7 m^2/person (75 ft^2/person). The staff finds this to be acceptable because the TSC size conforms with NUREG-0696. Therefore, DSER Open Item 13.3-3 is closed. In addition, Section 18.8.3.5 of the SSAR indicates that the TSC has no emergency habitability requirements. Paragraph 8.2.1.f of Supplement 1 to NUREG-0737 requires the TSC to be provided with:

> ... radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

In the DSER, the staff stated that Westinghouse should provide information to demonstrate the ability of the TSC to meet habitability requirements. This was identified as DSER Open Item 13.3-4. In Revision 11, SSAR Section 9.4.1 was revised to indicate that the nuclear island non-radioactive ventilation system (VBS) serves the TSC. SSAR Section 9.4.1.1.2, "Power Generation Design Basis," states that the VBS provides the following functions:

- controls the main control room (MCR) and TSC relative humidity between 25 and 60 percent
- maintains the MCR and TSC at a slightly positive pressure during normal operations
- isolates the MCR and/or TSC from normal outdoor air intake and provides filtered outdoor air to pressurize the MCR and TSC when a high gaseous radioactivity concentration is detected in the MCR supply air duct
- isolates the MCR and/or TSC when a high concentration of smoke is detected in the outside air intake
- provides smoke removal capability for the MCR and TSC

SSAR Section 9.4.1.2.2 indicates that the VBS components include low-efficiency filters, high-efficiency filters, and postfilters; HEPA filters; charcoal adsorbers; and isolation dampers. SSAR Section 9.4.1.2.3.1, "Main Control Room/Technical Support Center HVAC Subsystem," under the section entitled **Abnormal Plant Operations**, states that when a "high" gaseous radioactivity is detected and the HVAC subsystem is operable, both supplemental air filtration units automatically start to pressurize the MCR and TSC to at least 1/8" wg. The normal outside air makeup duct and the MCR and TSC toilet exhaust isolation dampers close. In addition, if ac power is unavailable for more than 10 minutes or if "high-high" particulate or iodine radioactivity is detected, the MCR is isolated from the TSC. In the event of a loss of the normal plant ac electrical system, the MCR/TSC ventilation subsystem is automatically transferred to the onsite standby diesel generators. The staff finds that this design meets the habitability requirements of Supplement 1 to NUREG-0737 and is acceptable. Therefore, DSER Open Item 13.3-4 is closed.

In its February 21, 1997 response to Q100.10, Westinghouse discusses what would happen if a loss of electrical power to the TSC should occur, and if habitability should be challenged within the TSC because of a lack of cooling or radiological concerns. In such an instance, Westinghouse states that the functions normally performed in the TSC would be transferred to an emergency operations facility (EOF), where habitability is not dependent on plant systems and with communication and data transfer links to the MCR to provide essential information. Westinghouse performed a preliminary habitability task analysis for the MCR but did not include the TSC functions. Westinghouse committed to revise the task analysis to include the TSC functions assigned to the control room as a post-design certification activity and verify the results through testing. In the DSER, the staff stated that Westinghouse should perform the analysis and any necessary testing to demonstrate the ability of the MCR to support the additional staff. This was identified as DSER Open Item 13.3-5.

In lieu of a task analysis, Westinghouse revised SSAR Section 18.2.1.2 to provide a description of assumptions and constraints, including utility requirements, that are used as inputs to the human factors engineering program and the human system interface design. As stated, the human system interface design includes the design of the operation and control centers (e.g., MCR, TSC, etc.). The staff concludes that inclusion of these assumptions and restraints adequately addresses the concerns of the staff. Therefore, DSER Open Item 13.3-5 is closed.

Because of the unique design of the AP600, the habitability system for the TSC is not the same as for the MCR. At currently operating reactors, the TSC habitability system is either the same as for the MCR, or the TSC has been provided a separate habitability system. At these sites, should the TSC become uninhabitable, it is usually evacuated to either the MCR or another location onsite where habitability can be established. Not having the TSC in the same habitability envelope as the MCR, as discussed above, increases the likelihood that the TSC will have to be evacuated. In addition, Westinghouse has indicated that, should the TSC become uninhabitable, the functions and staff will be relocated to the EOF, and not the MCR or another facility onsite were habitability can be established. Consequently, the EOF will have to be activated and staffed early in order to ensure that the functions and support provided to the MCR by the TSC are not impeded. In SSAR Section 13.3.1, "Combined License Information Item," Westinghouse states

> Combined License applicants referencing the AP600 certified design will address the activation of the emergency operations facility consistent with current operating practice and NUREG-0654/FEMA-REP-1, except for a loss-of-offsite power <u>and</u> loss of all onsite ac power. For this initiating condition, the Combined License applicant shall immediately activate the emergency operations facility rather than bring it to a standby status.

The staff concludes that this COL action item acceptably addresses this concern. This is COL Action Item 13.3-2.

In RAI Q100.32F, the staff requested Westinghouse to commit to Revision 2 of Regulatory Guide (RG) 1.101. In Revision 19, Westinghouse modified Appendix 1A of the SSAR to indicate that the AP600 design conforms to RG 1.101, Revision 2, dated October 1981. The SSAR indicates that NUREG-0654/FEMA-REP-1 applies to the TSC, OSC, and EOF. Revision 2 to RG 1.101 references NUREG-0654/FEMA-REP-1. Item 11.H, "Emergency Facilities and Equipment," of NUREG-0654/FEMA-REP-1 is applicable to the TSC, OSC, and Decontamination Facility, which are identified in the AP600 design. The staff finds this commitment acceptable.

In accordance with 10 CFR 50.47(b)(9), the COL applicant must employ adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition. To address this regulation, the NRC has concluded that source term information should be obtained and analyzed promptly to continuously assess and refine dose assessments, and confirm or modify initial protective action recommendations. In Revision 23 to Section 12.4.1.8 of the SSAR, Westinghouse states that

> The design provides for prompt access to the primary sampling room to obtain reactor coolant and containment samples early after the accident without radiation exposure to any individual exceeding 5 rem to the whole body or 50 rem to the extremities. In addition to the design provisions, individual exposure for this early sampling operation may be minimized by proper administrative operation controls (for example, splitting tasks among different crews or limiting sample sizes). Special operational controls would only be

considered in the event that radiation fields associated with access to the primary sampling room reached the levels considered in the evaluations. These levels include activity releases as defined in NUREG-1465, maximum design-basis leak rate from the containment into the access areas, and no operable building ventilation systems.

The staff concludes that the AP600 design includes provisions for promptly obtaining reactor coolant and containment samples that can be used by the COL applicant to comply with this aspect of 10 CFR 50.47. In Revision 23 to Section 13.3, Westinghouse provided a Combined License Information Item that states that the COL applicant will address the capability to promptly obtain and analyze the reactor coolant and containment samples, as follows:

> To initially and continuously assess the course of an accident for emergency response purposes, Combined License applicants referencing the AP600 certified design will address the capability for promptly obtaining and analyzing grab samples of reactor coolant and containment atmosphere and sump in accordance with the guidance of Item II.B.3 of NUREG-0737.

Incorporation of this information into the SSAR was identified as AFSER Confirmatory Item 13.3-1. Therefore, AFSER Confirmatory Item 13.3-1 is closed. The staff finds that these commitments acceptability address this matter, and will ensure that they are addressed by a COL applicant referencing the AP600 design. This is COL Action Item 13.3-3.

13.4 Operational Review

In the DSER, the staff stated that Westinghouse should add a COL action item to address the operational review for the plant. This was identified as DSER Open Item 13.4-1 and COL Action Item 13.4-1. In Section 13.4 of the SSAR, Westinghouse states that the operational review is the responsibility of the COL applicant. In Section 13.4.1, Westinghouse added a statement that a COL applicant referencing the AP600 certified design will address each operational review. Therefore, DSER Open Item 13.4-1 is closed.

13.5 Plant Procedures

In the DSER, the staff stated that Westinghouse should add a COL action item to address the administrative procedures for the plant. This was identified as DSER Open Item 13.5.1-1 and COL Action Item 13.5.1-1. In addition, the staff stated that Westinghouse should add a COL action item to address the operating and maintenance procedures for the plant. This was identified as DSER Open Item 13.5.2-1 and COL Action Item 13.5.2-1.

In Section 13.5 of the SSAR, Westinghouse states that the plant procedures are the responsibility of the COL applicant. However, Westinghouse further references WCAP-14690, which provides input to the COL applicant for developing plant procedure, including information on the development and design of the AP600 emergency response guidelines and emergency operating procedures. Westinghouse also references WCAP-14837, which provides input to the COL applicant for developing plant-specific refueling plans.

In Section 13.5.1, Westinghouse added a statement that a COL applicant referencing the AP600 certified design will address plant procedures for the following areas:

- normal operation
- abnormal operation
- emergency operation
- refueling and outage planning
- alarm response
- maintenance, inspection, test, and surveillance
- administrative

Therefore, DSER Open Items 13.5.1-1 and 13.5.2-1 are closed.

13.6 Security

In the DSER, the staff stated that it would evaluate and address the AP600 security design report in the final SER. This was identified as DSER Open Item 13.6-1. To evaluate the security features of the AP600 design, the staff reviewed: (1) Chapter 13.6 of the SSAR through Revision 25; (2) the AP600 Security Design Report (SDR) through Revision 6 dated July 1, 1998, and amended on August 11, 1998; and (3) the AP600 Security Design Vulnerability Analysis Report (SDVAR) through Revision 3 dated July 1, 1998, and amended on August 11 and 17, 1998. Certain revisions to the SSAR and amendments to the SDR and the SDVAR resulted from a concern raised by the ACRS. The concern was about the security design's potential adverse impact on accessibility for operations personnel to certain areas of the plant. This matter is discussed in this report. In addition, as discussed throughout this section, Westinghouse has appropriately addressed the open items identified in the DSER. The staff finds that the measures described in these documents comply with the requirements of 10 CFR 73.55(b) through (h) or provide alternatives, as provided for in 10 CFR 73.55(a), to achieve the same level of assurance, and adhere to pertinent Commission guidance, including related Generic Letters (GLs), Regulatory Guides (RGs), and the SRP. Further, Westinghouse modified the AP600 security design to address the ACRS' concern. The staff considers the modifications acceptable. Therefore, DSER Open Item 13.6-1 is closed.

In the DSER, the staff evaluated Revision 0 of both the SSAR and the SDR dated June 26, 1992. There have been several format changes to the SSAR and SDR as a result of the staff's interactions with the applicant which requires renumbering the COL Action Items. Table 13.6-1 provides a cross-reference between the COL Action Items identified in the DSER and this report.

13.6.1 Preliminary Planning

Westinghouse states that the objectives and functional requirements of the AP600 physical protection system and description of security features are provided in the AP600 SDR, submitted under separate cover in accordance with 10 CFR 2.790(a). The SDR also includes the security boundary drawings and the listing of the vital equipment and components. A vulnerability analysis was also submitted that demonstrates that the AP600 certified security design is adequate to protect the AP600 from radiological sabotage. Westinghouse further states that, as demonstrated by the AP600 SDVAR, reducing the protected area and eliminating the isolation zones result in a reduced requirement for security staffing when compared to current plants.

The staff confirmed that the reference documents were provided to the Commission. While the AP600 approach reduces security staffing when compared to some current plants, it does not achieve the same result when compared to all current plants.

Westinghouse states that personnel screening, selection, performance evaluation, and training aspects of the physical security program will be addressed by the COL applicant . The staff agrees that the COL applicant should address these aspects of the physical security plan. This is COL Action Item 13.6.1-1.

13.6.2 Security Plan

Westinghouse states that the comprehensive physical security program is the responsibility of the COL applicant and will be addressed in the security plan, contingency plan, and guard training plan provided by the COL applicant. The staff agrees with this approach. These measures will satisfy the conditions of NRC Information Notice 89-05, "Use of Deadly Force by Guards Protecting Nuclear Power Reactors Against Radiological Sabotage," and NRC GL 96-02, "Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat." This is COL Action Item 13.6.2-1.

13.6.3 Plant Protection System

13.6.3.1 Introduction

Westinghouse states that a physical protection system and security organization is provided to protect the AP600 from radiological sabotage, as required by 10 CFR 73.55, and that, to achieve this objective, the physical protection system:

(a) includes a security organization

(b) locates vital equipment within vital areas

(c) controls points of personnel, vehicle, and material access into the protected and vital areas

(d) annunciates alarms in a continuously manned central alarm station and at least one other continuously manned alarm station that is physically separated from the central alarm station

(e) provides for continuous communications between the security officers and the continuously manned alarm stations

(f) provides for testing and maintenance of the alarms, communications, and physical barriers

(g) responds to threats of radiological sabotage in accordance with a developed contingency plan

13.6.3.2 Design Basis

Westinghouse states that the physical protection system protects against radiological sabotage events, following the requirements of 10 CFR 73.55(a). The design basis and assumptions for the design are provided by the SDR and SDVAR. The following assumptions were made by Westinghouse in its analysis:

(a) The intrusion detection systems cannot be disabled without detection and timely response by the security force.

(b) Unless precluded by plant design features or prevented by the plant security system, insider sabotage can potentially result in an initiating event requiring the actuation of safe shutdown systems, disabling of safe shutdown systems, disabling of non-safety-related systems (including offsite power), or any combination of these.

(c) In evaluating vulnerability to internal sabotage, onsite security system features, offsite resources or both are effective in preventing undetected penetration into the protected area by outsiders.

(d) While access to containment for maintenance and testing during operation at power is permitted, such access is controlled and typically non-routine.

(e) The continuous presence of several employees precludes acts of sabotage in the control room. However, the control room is a vital area and will be protected in accordance with 10 CFR 73.55.

(f) Equipment and systems designated as vital for full power operation shall be maintained as vital in other modes of plant operation. However, during unit shutdown, a vital area can be declassified to nonvital if approved by the security plan.

(g) Sabotage events do not occur coincident with some other independent single failure or independently initiated event.

(h) The security restrictions for access to equipment and plant regions will be compatible with loss of site power, access requirements, fire protection, health physics, and local operator actions required for event mitigation. During operating modes, security access control restrictions will not excessively impede operator functions.

The staff agrees with these assumptions. In addition, the safeguards plan submitted by the COL applicant will be required to address how the physical protection system will achieve assumptions a, b, c, and d. This is COL Action Item 13.6.3.2-1.

13.6.4 Physical Security Organization

Westinghouse states that the description of the site-specific physical security organization is the responsibility of the COL applicant. This is COL Action Item 13.6.4-1. The size and capabilities of the physical security organization's armed response team are established by the SDVAR, which demonstrates the acceptability of the AP600 certified security design. The required manning for the security force is established in the SDVAR. The established manning level satisfies the requirements of 10 CFR 73.55(h), and therefore, is acceptable.

13.6.5 Physical Barriers

13.6.5.1 Protected Area

Westinghouse states that the AP600 security design features a collapsed protected area boundary that does not require a perimeter fence such as found in conventional security plans. Where there are portals for personnel or material access into the vital areas, protected areas surround the portals. Protected area barriers provide an outer boundary to prevent unauthorized access to the vital areas of the plant without detection. The protected area barriers are constructed, as a minimum, of chain link fence equivalent to the physical barrier defined in 10 CFR 73.2 for fences. The protected area barriers are constructed so that any attempt to penetrate into the protected areas will activate counter measures in response to the threat. The protected area is equipped with closed circuit television (CCTV) and has intrusion detection equipment and alarms that annunciate upon detection of penetration to alert security response forces that the area has been breached. Descriptions of the protected areas and the drawings showing the location of the protected areas are provided in the SDR.

The concept of the AP600 collapsed protected-area boundary is predicated on vital-area exterior walls, portals, and other openings being constructed to achieve stand-alone-barrier specifications with sufficient delay factors to ensure a security response before successful penetration by a potential adversary desiring entry to commit radiological sabotage. This concept mitigates the requirement for a physical barrier, as described in 10 CFR 73.55(c), at most of the protected-area perimeter. The limited amount of protected area is separated by a barrier, alarmed and covered by CCTV.

With the exception of one section of the shield building that is not under direct observation by a response-force member, but is alarmed and monitored by the alarm stations with CCTV, the AP600 concept requires response-force members to be in close enough proximity to the vital areas to provide surveillance generally and to facilitate a rapid response. Except for the south end of the turbine building, the requirement for isolations zones, as described in 10 CFR 73.55(c), has been mitigated.

As demonstrated by Westinghouse, the collapsed protected-area barrier is an alternative measure that has the same high assurance objective, as detailed in 10 CFR 73.55(c), to protect against radiological sabotage. In accordance with 10 CFR 73.55(a), the staff finds that the alternative measure is acceptable.

In the DSER, the staff stated that Westinghouse should indicate whether the AP600 plant layout allows portions of the protected-area perimeter to abut or cross a body of water. The staff requested Westinghouse to address this matter because the EPRI URD has a provision in it that the plant layout should avoid such a setup. This was identified as DSER Open Item 13.6.5.1-1. However, the AP600 is currently based on a collapsed protected-area boundary. This results in vital enclosures serving as stand-alone barriers surrounded by a vehicle barrier system. This configuration, as detailed in the plant drawings included in the SDR, would not be possible if the protected-area boundary abuts or crosses water. Therefore, this item no longer applies, and DSER Open Item 13.6.5.1-1 is closed.

Similarly, in the DSER, the staff identified a COL Action Item 13.5.5.1-1 that the COL applicant will provide the specific details of the protected-area boundary based on specific site topology and features. This is no longer necessary because the unique configuration of the protected-area boundary is not dependent on these aspects of the as-built plant. This COL Action Item is no longer applicable.

13.6.5.2 Vital Areas

Westinghouse states that vital equipment is located within designated vital areas. The AP600 vital areas are encompassed by the boundary formed by the shield building, a reinforced concrete and steel structure surrounding containment, and by portions of the reinforced concrete perimeter and interior wall of the auxiliary and annex buildings. Accessible and unmonitored portions of the boundary walls, floors, ceilings, windows, doors, and penetrations are hardened for security. Accessible is defined as 5.5 m (18 feet) above the base of a wall that can be reached by normal means (e.g., walking, climbing fixed ladders, or using hand-carried step up devices). Unmonitored is defined as not being visible to a continuously-manned location or to intrusion detection alarms. The security hardened barriers are constructed of sufficient structural integrity to significantly delay a perpetrator who is trying to penetrate the boundary in order to gain access to vital equipment. The delay times of these barriers are sufficient to allow timely response by security and plant personnel to neutralize a sabotage attempt. Access points to vital areas are locked and alarmed with active intrusion detection systems. Protected areas that surround the access portions of the vital areas are equipped with CCTV that is monitored at the alarm stations. In the DSER, the staff stated that it had not completed its review of the list of vital equipment identified by Westinghouse. Completion of this review was identified as DSER Open Item 13.6.5.2-1. The vital areas and a listing of the vital equipment are provided in the SDR. The staff finds that these measures are acceptable and, therefore, DSER Open Item 13.6.5.2-1 is closed.

13.6.5.3 Bullet-Resisting Barriers

Westinghouse states that the doors, walls, floor, and ceiling of the control room and the continuously manned alarm stations are designed to meet the bullet-resisting criteria of UP-752, "High Power Rifle Rating," including resistance to a level 4 round. The staff finds that these measures satisfy the requirements of 10 CFR 73.55(c)(6) and, therefore, are acceptable.

13.6.5.4 Vehicle Barrier System

Westinghouse states that the vital areas are surrounded by a vehicle barrier system that provides a barrier such that no location along the perimeter will permit forced entry of a vehicle. The vehicle barrier system is designed to stop the design-basis vehicle before it reaches the safe standoff distance for the vital equipment located inside the vital areas. No point along the perimeter of the vehicle barrier system is located closer than the minimum safe standoff distance for the vital area barrier. Vital equipment/components are not expected to be damaged to the extent that they are no longer able to maintain the plant in a safe condition as a result of detonation of a design-basis bomb at the vehicle barrier system boundary. Active gates are located at the two vehicle portals that provide the only access for vehicles to enter the area enclosed by the vehicle barrier system. Vehicles that are authorized for access and that have a need to enter the area enclosed by the vehicle barrier system are searched at these gates for items that could be used for radiological sabotage. There is no general parking within the area enclosed by the vehicle barrier system. A description of the vehicle barrier system is provided in the SDR.

The staff finds these measures acceptable because they satisfy the guidance of NRC GL 89-07, "Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs," and the requirements of 10 CFR 73.55(c)(8).

13.6.6 Access Requirements

Westinghouse states that positive control features are implemented to provide authorization for personnel and vehicles entering the protected and vital areas. Section 13.6.6 of the SSAR states that the COL applicant is responsible for ensuring that the following access control features are provided:

(a) means for positive identification of authorized personnel entering the protected and vital areas

(b) means for searching individuals, packages, and materials for firearms, explosives, and incendiary devices. This may be accomplished using detection devices such as metal detectors, explosive detectors, and X-ray machines.

In addition, Westinghouse states that the AP600 design certification scope includes:

(a) Identification of access portals entering the protected and vital areas

(b) provision of unmanned portals with alarm annunciation in the continuously manned alarm stations

The protected and vital area ingress and egress are designed to interface with other plant requirements, and will not impair plant operations during emergency conditions.

The staff agrees that the COL applicant should address specific access control measures. These measures should satisfy the guidance of NRC GL 83-21, "Clarification of Access Control Procedures for Law Enforcement Visits," and 10 CFR 73.55(d). This is COL Action Item 13.6.6-1.

13.6.7 Detection Aids

13.6.7.1 Perimeter

Westinghouse states that, except for the special provisions for the shield building, intrusion detection at the perimeter of the plant is not required. A description of the intrusion detection provisions provided for the shield building is provided in the SDR.

With the exception of one section of the shield building that is not under direct observation by a response-force member, but is alarmed and monitored by the alarm stations with CCTV, the AP600 concept requires response-force members to be in close enough proximity to the vital areas to provide surveillance generally and to facilitate a rapid response. This mitigates the requirement for isolations zones, as described in 10 CFR 73.55(c).

As demonstrated by Westinghouse, the collapsed protected-area barrier is an alternative measure that has the same high assurance objective, as detailed in 10 CFR 73.55(c), to protect against radiological sabotage. In accordance with 10 CFR 73.55(a), the staff finds the alternative measure acceptable.

13.6.7.2 Protected Area

Westinghouse states that an intrusion detection system is utilized to notify the security organization of any unauthorized attempt to gain access into the protected areas. CCTV is installed in unmanned protected areas. Doors entering a protected area are equipped with an alarm to detect tampering and unauthorized access into the protected area. A description of the intrusion detection provisions provided for the AP600 is provided in the SDR. The staff finds these measures satisfy the requirements of 10 CFR 73.55(d) and (h) and, therefore, are acceptable.

In the DSER, the staff stated that Westinghouse should provide a discussion on how SECY-93-326, regarding an internal threat, is applied to the AP600 design. This was identified as DSER Open Item 13.6.7.2-1. In both the SDR and SDVAR, Westinghouse discusses the insider threat. Specifically, in the SDVAR section entitled "Insider," Westinghouse states that insider sabotage attempts requiring actuation of safe shutdown systems, disabling of safe shutdown systems, disabling of non-safety-related systems (including offsite power), or any combination of these, are detected by alarms and control circuits so operators can take mitigating actions. The staff finds these measures to be acceptable enhancements to the protection system described in 10 CFR 73.55(a), and therefore, DSER Open Item 13.6.7.2-1 is closed.

13.6.7.3 Vital Area

Westinghouse states that doors entering a vital area are hardened to provide significant delay time and are alarmed to detect tampering and unauthorized access into the vital area. The delay times for doors accessing the vital areas are sufficient to allow timely response by the security organization to attempted penetrations. A description of the intrusion detection provisions provided for the AP600 is provided in the SDR. The staff finds that these measures satisfy the requirements of 10 CFR 73.55(d) and (h), and therefore, are acceptable.

13.6.8 Security Lighting

Westinghouse states that security lighting is provided for the alarm stations and the protected areas. A description of the security lighting provided for the AP600 is provided in the SDR. The staff finds that these measures satisfy the requirements of 10 CFR 73.55(c), and therefore, are acceptable.

13.6.9 Security Power Supply System

Westinghouse states that security equipment that supports critical monitoring functions (e.g., intrusion detection, alarm assessment, and the security communications system) receive power from the security-dedicated uninterruptable power supply (UPS) system. Switchover to the uninterruptable power supply system is automatic and does not cause false alarms on annunciation modules. The UPS is capable of sustaining operation for a minimum of 24 hours. A description of the security power supply system provided for the AP600 is provided in the SDR. The staff finds that these measures satisfy the requirements of 10 CFR 73.55(f), and therefore, are acceptable.

13.6.10 Communications

Westinghouse provides a description of the AP600 security communication system in the SDR. Westinghouse states that the specific details for the security communications system will be addressed by the COL applicant. Two 2-way communications paths are provided between the control room and the alarm stations within the AP600. A single act of sabotage cannot sever both communications paths. Security force members with responsibilities to respond to acts of sabotage have the capability for continuous 2-way communications with the alarm stations and with each other. The centralized communications equipment and radio antennas are located in a controlled area so that they will remain operable during a radiological sabotage event. Non-portable security communications equipment is fed from the security power supply system so that it remains operable in the event of the loss of normal power.

The staff agrees that specific details of the security communications system should be addressed by the COL applicant. This is COL Action Item 13.6.10-1.

13.6.11 Testing and Maintenance

Westinghouse states that the COL applicant will address the testing and maintenance aspects of the plant security system. The staff agrees that testing and maintenance requirements should be addressed by the COL applicant. This is COL Action Item 13.6.11-1.

13.6.12 Response Requirements

Westinghouse states that the COL applicant will address response requirements of the plant security program. The staff agrees that detailed response requirements should be addressed by the COL applicant. This is COL Action Item 13.6.12-1. The plant-specific licensing review of the security and contingency response plan will include an evaluation of whether the security response force's capability to interdict the violent external assault postulated in 10 CFR 73.1(a)(1)(i) properly accounts for the minimum penetration delay provided by the vital area barriers and doors. In the DSER, the staff stated that Westinghouse should evaluate the interdiction capability of the security response force. This was identified as DSER Open Item 13.6.10-1. In Appendix 1 to the SDVAR, Westinghouse provides detailed analyses of security events. The analyses include scenarios, time lines, and interdicting response routes. The staff finds that these measures satisfy the requirements of 10 CFR 73.55(h), and therefore, are acceptable. Therefore, DSER Open Item 13.6.10-1 is closed.

13.6.13 Combined License Information Item

13.6.13.1 Security Plans, Organizations, and Testing

Westinghouse states that COL applicants referencing the AP600 certified design will address site-specific information related to the security, contingency, and guard training plans. Those plans will include descriptions of the tests planned to show operational status, maintenance of the plant security system, the security organization, communications, and response requirements.

The COL applicant will develop a comprehensive physical security program that includes the security plan, contingency plan, and guard training plan. Each COL applicant will describe in its physical security plan how the requirements of 10 CFR Part 26 will be met. At least 60 days before loading fuel, the COL applicant will confirm that the security system and programs described in its physical security plan, safeguards contingency plan, and training and qualification plan have achieved operational status and are available for the (NRC) staff's inspection. Operational status means that the security systems and programs are functioning. The determination that operational status has been achieved will be based on tests conducted under realistic operating conditions of sufficient duration to demonstrate that:

(a) the equipment is properly operating

(b) procedures have been developed, approved, and implemented

(c) personnel responsible for security operations and maintenance have been appropriately trained and have demonstrated their capability to perform their assigned duties and responsibilities

The staff agrees that the COL applicant should address site-specific information relative to its security, contingency, and training and qualification plans and its fitness-for-duty program. These measures will satisfy the conditions of NRC GL 91-16, "Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty," 10 CFR 73.55(b) and (h), and 10 CFR Part 26. This is COL Action Item 13.6.13.1-1.

13.6.13.2 Vital Equipment

Westinghouse states that COL applicants referencing the AP600 certified design will verify that the as-built location of vital equipment is inside the vital areas. The staff expects the COL applicant to evaluate the completeness of the list of vital equipment before issuing a COL. This evaluation should include an assessment of the plant-specific equipment outside the scope of the certified AP600 design. This is COL Action Item 13.6.13.2-1.

13.6.13.3 Plant Security System

Westinghouse states that COL applicants referencing the AP600 certified design will address site-specific information related to the maintenance and testing of the plant security system, including the intrusion detection and assessment system; the access control features specified in subsections 13.6.6, 13.6.7.2, and 13.6.7.3; and the vehicle barrier system. The COL applicant will address in its safeguards plans how the physical protection system will provide the protection stated in Section 13.6.3.2. The staff agrees that the COL applicant should address these matters to satisfy the requirements of 10 CFR 73.55(d), (e) and (g). This is COL Action Item 13.6.13.3-1.

13.6.13.4 Vulnerability Analysis Report

In the DSER, the staff stated that Westinghouse should provide an analysis of the vulnerabilities of the design to sabotage. This was identified as DSER Open Item 13.6.3.2-1. Westinghouse provided its SDVAR through Revision 3 dated July 1, 1998, and amended on August 11 and 17, 1998. The staff reviewed the AP600 SDVAR, and verified that Westinghouse had appropriately addressed all of the outstanding open items. All of the items are closed, and the staff concludes that the report is acceptable. Therefore, DSER Open Item 13.6.3.2-1 is closed.

13.7 References

Section 13.7 of the SSAR lists references for Chapter 13. In the DSER, the staff stated that referencing American Nuclear Society (ANS) 3.3-1988, which has not been endorsed by the staff, was unacceptable. This was identified as DSER Open Item 13.7-1. Westinghouse has removed that reference, and therefore, DSER Open Item 13.7-1 is closed.

Table 13.6-1
COL Action Item Cross-Reference

DSER COL Action Item Designation	SSAR Section (Revision 20)	FSER COL Action Item Designation
13.6.1-1	13.6.1	13.6.1-1
13.6.2-1	13.6.2, 13.6.13.1	13.6.2-1, 13.6.13.1-1
13.6.2-2	13.6.13.1	12.6.13.1-1
13.6.4-1	13.6.4	13.6.4-1
13.6.5.1-1	SDR	Not Applicable
13.6.5.2-1	SDR	13.6.13.2-1
13.6.5.2-2	13.6.13.2	13.6.13.2-1
13.6.6-1	13.6.6, 13.6.13.3	13.6.6-1, 13.6.13.3-1
13.6.7.1-1	13.6.13.3	13.6.13.3-1
13.6.9-1	13.6.11, 13.6.13.3	13.6.11-1, 13.6.13.3-1
13.6.10-1	13.6.12	13.6.12-1
-	13.6.10	13.6.10-1 (new)

14 VERIFICATION PROGRAMS

14.1 Preliminary Safety Analysis Report Information

In Regulatory Guide (RG) 1.70 and NUREG-0800, the NRC specifies that Chapter 14 in the safety analysis report (SAR) must include Section 14.1, "Specific Information To Be Included In Preliminary/Final Safety Analysis Reports." However, the preliminary safety analysis report information specified by RG 1.70 and the standard review plan is not required for a final design approval or design certification application under 10 CFR Part 52. On the basis of this rationale, the staff concurs with the statement in Section 14.1 of the SSAR which indicates that this section is not applicable to the AP600.

14.2 Initial Test Program

In Section 14.2 of the SSAR, Westinghouse describes the test program that is performed during initial startup of an AP600 plant. The AP600 initial test program (ITP) comprises a series of tests categorized as construction and installation, preoperational, and startup tests. The construction and installation tests are performed to determine that plant structures, components, and systems (SSCs) were correctly constructed or installed and are operational. Preoperational tests are performed after construction and installation tests, but before initial fuel loading, to demonstrate the capability of plant systems to meet performance requirements. Startup tests, which begin with initial fuel loading, are performed to demonstrate the capability of the integrated plant to meet performance requirements.

In 1994, the staff performed an initial review of Section 14.2 of the SSAR, through Revision 1, in accordance with Section 14.2 of NUREG-0800 and RG 1.68, Revision 2. On the basis of this review, the staff sent Westinghouse a request for additional information (RAI) consisting of a list of questions and comments in a letter dated May 24, 1994.

In the draft safety evaluation report (DSER), the staff presented its conclusions on the basis of the evaluation of the AP600 ITP, the information included in the RAI and the Westinghouse responses to NRC RAIs as provided in the enclosures to Westinghouse letters dated June 27, July 29, August 3, and August 8, 1994. This chapter identified the questions as either open items or confirmatory items. Subsequent to the issuance of the DSER, the staff continued its interaction with Westinghouse to achieve resolution of these open items. This interaction is documented in NRC and Westinghouse correspondence dating from August 13, 1996, through March 10, 1998.

In draft Revision 10 to the SSAR, Westinghouse modified the information in Chapter 14 substantially and added several sections. The information previously presented in Section 14.2.8, "Individual Test Descriptions," was allocated into two main sections (1) Section 14.2.9, "Preoperational Test Descriptions," and (2) Section 14.2.10, "Startup Test Procedures," each having multiple sections delineating different phases of testing and/or system/component functions.

14.2.1 Summary of Test Program and Objectives

As stated in Section 14.2.1 of the SSAR, the overall objective of the initial plant test program is to demonstrate that the plant is constructed as designed, that the systems perform as required by the plant design, and that activities culminating in operation at full licensed power (including initial fuel load, initial criticality, and power ascension) are performed in a controlled and safe manner.

In Revision 1 to Section 14.2.1 of the SSAR, the following were the specific objectives of the initial plant test program:

- demonstrate that AP600 design features meet the performance criteria identified in Section 14.2 of the SSAR (In addition, for the first AP600 plant, preoperational tests and startup tests will confirm selected design and analysis assumptions and predictions for the AP600.)

- demonstrate that plant construction is complete and acceptable

- demonstrate the capability of SSCs to meet performance requirements

- demonstrate, where necessary, that the plant is capable of withstanding anticipated transients and postulated design-basis events

- validate plant operating and emergency procedures as practical

- achieve initial fuel loading, initial criticality, and power ascension in a controlled and safe manner

- bring the plant to rated capacity for sustained power operation

In this section, Westinghouse also stated that preoperational and/or startup testing will be performed on those systems that meet the following criteria:

- are relied upon for safe shutdown and cooldown of the plant under normal plant conditions, and for maintaining the reactor in a safe condition for an extended shutdown period

- are relied upon for safe shutdown and cooldown of the reactor under transient and postulated accident conditions, and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions

- are relied upon for establishing conformance with safety limits or limiting conditions for operation

- are classified as engineered safety features actuation systems (ESFASs) or are relied upon to support the operation of ESFASs within design limits

- are assumed to function during an accident or for which credit is taken in the accident analysis and in the probabilistic risk assessment (PRA)
- are used to process, store, control, or limit the release of radioactive material

In the DSER, the staff found that in order to be consistent with the guidance of Regulatory Position (RP) C.1 of RG 1.68, Revision 2, dated August 1978, the third, fourth, and fifth items in Section 14.2.1 of the SSAR regarding systems on which preoperational and/or startup testing is to be performed should be revised as follows:

- are relied upon for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications
- are classified as ESFASs or are relied upon to support or ensure operation of ESFASs within design limits
- are assumed to function or for which credit is taken in the accident analysis of the facility, as described in the SSAR and/or in its design-specific PRA

In addition, the staff concluded that Westinghouse should include in this section of the SSAR (or in another section of Chapter 14, as appropriate), a detailed description of those AP600 plant-specific design features, systems, (including those listed in SSAR Table 1.5-1, "AP600 Detailed Design Tests") and/or system configurations or interactions not being tested and/or simulated within the ITP scope of Chapter 14 of the SSAR, which meet the requirements of 10 CFR 52.47(b)(2)(I). For any such systems or design features identified, Westinghouse should provide appropriate justifications for their exclusion from the ITP, or the applicable test abstract(s) should be modified to include them accordingly.

The staff also found that Section 14.2.1 (or alternatively Section 14.2.8) of the SSAR should be revised to identify, if applicable, any startup tests that are performed to demonstrate the operability of SSCs that are not considered essential to meet the criteria of RP C.1 of RG 1.68, Revision 2, dated August 1978. Portions of the issues outlined above were previously identified by the staff as RAI 260.23. This was identified as DSER Open Item 14.2.1-1.

In its response to RAI 260.23 dated August 13, 1996, Westinghouse confirmed that there were no tests in Chapter 14 which demonstrated the operability of SSCs that are not considered essential to meet the criteria of RP C.1 of RG 1.68. Westinghouse also agreed to revise the third, fourth, and fifth items in Section 14.2.1 of the SSAR, as indicated by the staff in the DSER. The staff agreed that Westinghouse was responsive to the issues identified in this open item, except for the following two issues:

(1) In Revision 9 to the SSAR, the AP600 design-specific PRA had not been included in item 14.2.1(e).

(2) Westinghouse had not addressed whether Section 14.2.1, item (g) needed to be revised to reflect Westinghouse's response to this open item, that "applicable systems" identified in RG 1.68 had been included. If the intent of item (g) was to include any remaining

SSCs included in RG 1.68, Appendix A that are not identified in items (a) through (f), then item (g) should have been clarified accordingly.

In its response dated December 6, 1996, Westinghouse stated it did not use the design-specific PRA as a criteria for the selection of systems, structures or components to be included in the ITP. However, applying this criteria did not capture any additional AP600 SSCs not currently captured by the criteria currently provided in Section 14.2.1 of the SSAR. Therefore, Westinghouse did not believe it was necessary to add a reference to the design PRA as a criteria for test selection. Westinghouse added that Section 14.2.1, item (g) would be revised to clarify that other systems identified in Appendix A to RG 1.68, Revision 2 are within the scope of the ITP.

Additionally, Westinghouse clarified that Table 1.5-1 of the SSAR lists specific AP600 design tests that were performed to assess the performance of components and systems in the AP600. However, Table 1.5-1 does not represent a comprehensive list of design features in the AP600 that are significantly different from currently operating light-water reactors or utilize passive systems. However, Westinghouse believes that those design features embodied in the design tests listed in Table 1.5-1 that (1) are significantly different from those found in light-water reactor designs described in 10 CFR 52.47(b)(1), or (2) utilize simplified, inherent, passive, or other innovative means to accomplish their intended safety functions, are conclusively tested as part of the AP600 ITP described in Chapter 14 of the SSAR. On these bases, the staff finds that Westinghouse adequately addressed the issues identified in RAI 260.23 and in DSER Open Item 14.2.1-1. Therefore, DSER Open Item 14.2.1-1 is closed.

14.2.1.1 Construction and Installation Test Objectives

In Section 14.2.1.1 of the SSAR, Westinghouse states that the adequacy of construction, installation, and preliminary operation of components and systems will be verified by a construction and installation test program. In this program, various electrical and mechanical tests will be performed using written procedures and instructions.

These tests will normally include the following items:

- Cleaning and flushing
- Hydrostatic testing
- Checks of electrical wiring
- Valve testing
- Initial energization and operation of equipment
- Initial calibration of instrumentation

On a system basis, completion of this program demonstrates that the system is ready for preoperational testing. Construction and installation test program abstracts are not provided as part of the SSAR and are not required for design certification. Development of the construction and installation tests will be on the basis of the latest approved engineering information for the installed equipment and systems.

The staff concludes that the construction and installation test objectives outlined in Section 14.2.1.1 of the SSAR meet the acceptance criteria in Section 14.2 of the SRP and are acceptable.

14.2.1.2 Preoperational Test Program Objectives

In Section 14.2.1.2 of the SSAR, Westinghouse states that, following construction and installation testing, preoperational tests will be performed to demonstrate that equipment and systems perform in accordance with design criteria so that initial fuel loading, initial criticality, and subsequent power operation can be undertaken safely. Preoperational tests at elevated pressure and temperature are referred to as hot functional tests. The preoperational test program has the following general objectives:

- demonstrate that essential plant components and systems, including alarms and indications, meet appropriate criteria on the basis of the design

- provide documentation of the performance and condition of equipment and systems

- provide baseline test and operating data on equipment and systems for future use and reference

- operate equipment for a sufficient period to demonstrate acceptable performance

- demonstrate that plant systems operate on an integrated basis

Plant operating, emergency, and surveillance procedures will be incorporated into the ITP. These procedures will be verified through use, to the extent practicable, during the preoperational test program and revised if necessary, before fuel loading.

Plant equipment used in the performance of preoperational tests will be operated in accordance with appropriate operating procedures, giving permanent plant staff an opportunity to gain experience in using these procedures and demonstrating their adequacy before the plant's initial criticality.

The staff concludes that the preoperational test program objectives outlined in Section 14.2.1.2 of the SSAR meet the acceptance criteria in Section 14.2 of the SRP and are acceptable.

14.2.1.3 Startup Test Program Objectives

In Section 14.2.1.3 of the SSAR, Westinghouse states that the startup test program will begin with initial fuel loading after the preoperational testing has been successfully completed.

Startup tests are grouped into the following four broad categories:

(1) Tests related to initial fuel loading

(2) Tests performed after initial fuel loading but before initial criticality

(3) Tests related to initial criticality and performed at low power (less than 5 percent)

(4) Tests performed at power levels greater than 5 percent

During performance of the startup test program, the plant operating staff will have the opportunity to obtain practical experience in the use of normal and abnormal operating procedures while the plant progresses through heatup, criticality, and power operations. The startup test program includes the following general objectives:

- install the nuclear fuel in the reactor vessel in a controlled and safe manner
- verify that the reactor core and components, equipment, and systems required for control and shutdown have been assembled according to design and meet specified performance requirements
- achieve initial criticality and operation at power in a controlled and safe manner
- verify that the operating characteristics of the reactor core and associated control and protection equipment are consistent with design requirements and accident analysis assumptions
- obtain the required data and calibrate equipment used to control and protect the plant
- verify that the plant is operating within the limits imposed by the technical specifications

The staff finds that the startup test program objectives outlined in Section 14.2.1.3 of the SSAR meet the acceptance criteria in Section 14.2 of the SRP and are acceptable.

14.2.2 Organization, Staffing, and Responsibilities

In Section 14.2.2 of the SSAR, Westinghouse states that the combined license (COL) applicant is responsible for establishing a management organization with overall responsibility for defining the responsibilities, requirements, and interfaces necessary to safely and efficiently test, operate, and maintain the AP600 plant. As Westinghouse discussed in Section 14.4.1 of the SSAR, the COL applicant is responsible for developing and establishing the specific plant organization and staffing (i.e., staff responsibilities, authorities, and personnel qualifications) appropriate for the AP600 ITP. This testing organization is responsible for planning, executing, and documenting the ITP and related activities occurring between the completion of plant/system/component construction and commencement of plant commercial operation. This is COL Action Item 14.2.2-1.

14.2.3 Test Specifications and Test Procedures

In Section 14.2.3 of the SSAR, Westinghouse states that preoperational and startup tests will be performed using test specifications and test procedures. For each test, the test specification will specify the following items:

- Objectives for performing the test
- Test prerequisites
- Initial test conditions
- Data requirements
- Criteria for test results evaluation and reconciliation methods and analysis as required

For each test, the test procedure will specify the following items:

- Objectives for performing the test
- Prerequisites that must be completed before the test can be performed
- Initial conditions under which the test is started
- Special precautions required for the safety of personnel or equipment
- Instructions delineating how the test is to be performed
- Identification of the required data to be obtained and the methods for documentation
- Data reduction analysis methods as appropriate

Test specifications and procedures will be developed and reviewed by personnel with appropriate technical background and experience. This includes the participation of principal design organizations in establishing test performance requirements and acceptance criteria. Specifically, the principal design organizations will provide the COL applicant with scoping documents (e.g., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to their scope of design responsibility. Available information on operating or testing experiences of operating reactors are factored into the test specifications and procedures as appropriate.

Copies of test specifications and test procedures will be provided to NRC inspection personnel approximately 60 days before the scheduled performance of the preoperational tests, and not less than 60 days before the scheduled fuel loading date for startup tests. Preoperational and startup tests will be performed in accordance with the quality assurance requirements specified in Section 17.5 of the SSAR. The COL applicant is responsible for providing test specifications and test procedures for the preoperational and startup tests for the NRC's review. This is COL Action Item 14.2.3-1 (refer to SSAR Section 14.4.2).

In the DSER, the staff found that the phrase "NRC staff personnel from the Office of Inspection and Enforcement," should be removed from Section 14.2.2, "Test Procedures," of the SSAR and replaced with "NRC inspection personnel." This was identified as DSER Open Item 14.2.2-1. The staff confirmed that the subject text (now in Section 14.2.3, SSAR Revision 19), was revised accordingly. Therefore, DSER Open Item 14.2.2-1 is closed.

On the basis of the above information, the staff finds that the test specifications and test procedures objectives outlined in Section 14.2.3 of the SSAR meet the acceptance criteria in Section 14.2 of the SRP and are acceptable.

14.2.3.1 Conduct of Test Program

In Section 14.2.3.1 of the SSAR, Westinghouse states that administrative procedures and requirements that govern the activities of the conduct of the ITP include the following items:

- Format and content of test procedures
- Process for both initial issue and subsequent revisions of test procedures
- Review process for test results
- Process for resolution of failures to meet performance criteria and other operational problems or design deficiencies
- Various phases of the initial test program and the requirements for progressing from one phase to the next, as well as requirements for moving beyond selected hold points or milestones within a given phase
- Controls to monitor the as-tested status of each system and modifications including retest requirements deemed necessary for systems undergoing or already having completed testing
- Qualifications and responsibilities of the positions within the startup group

The startup administrative procedures will supplement normal plant administrative procedures by addressing those concerns that are unique to the startup program. As discussed in Section 14.4.3, "Conduct of Test Program," the COL applicant is responsible for the generation of a startup administrative manual containing the administrative procedures and requirements governing the activities associated with the ITP. This is COL Action Item 14.2.3.1-1.

The staff finds that the conduct of test program objectives outlined in Section 14.2.3.1 of the SSAR meet the acceptance criteria in Section 14.2 of the SRP and are acceptable.

14.2.3.2 Review of Test Results

In Section 14.2.3.2 of the SSAR, Westinghouse states that final review of the individual tests is the responsibility of plant management, who is also responsible for final review of overall test results and for the review of selected milestones or hold points within the test phases. As discussed in Section 14.4.4 of the SSAR, "Review and Evaluation of Test Results," the COL applicant and holder is responsible for the review and evaluation of individual test results. Test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible design organizations and corrective actions and retests are performed, as required. This is COL Action Item 14.2.3.2-1.

The staff finds that the process for the review, evaluation, and approval of test results, as outlined in Section 14.2.3.2 of the SSAR, meets the acceptance criteria in Section 14.2 of the SRP and is acceptable.

14.2.3.3 Test Records

In Section 14.2.3.3 of the SSAR, Westinghouse states that retention periods for test records are on the basis of considerations of their usefulness in documenting initial plant performance characteristics and are retained in accordance with RG 1.28, "Quality Assurance Program Requirements (Design and Construction)."

The staff finds that the program for retention of test records, as outlined in Section 14.2.3.3 of the SSAR, meets the acceptance criteria in Section 14.2 of the SRP and is acceptable.

14.2.4 Compliance of Test Program With Regulatory Guides

In Section 14.2.4 of the SSAR, Westinghouse states that Section 1.9.1 and Table 1.9-1 of the SSAR discuss compliance with the applicable NRC RGs.

In the DSER, the staff found that the preoperational and startup test phase descriptions in Revision 1 to Section 14.2.8 of the SSAR did not provide assurance that the operability of several of the systems and components listed in Appendix A of RG 1.68 (Revision 2, August 1978) would be demonstrated. The staff concluded that the test abstracts of Section 14.2.8 of the SSAR should be expanded to address the following items identified in Appendix A to RG 1.68, or Appendix 1A of the SSAR should be revised to provide technical justification for any exceptions taken:

<u>Preoperational Testing</u>

1.a.(2)(l) pressurizer safety valves

1.b.(1) control rod withdrawal inhibit and rod runback functions

1.c diverse actuation system, which protects the facility from anticipated transients without a scram (ATWS)

1.e.(4) steam generator pressure safety valves

1.e.(10) feedwater heaters and drains

1.f.(2) cooling towers and associated auxiliaries

1.j.(7)	leak detection systems used to detect failures in the emergency core cooling system (ECCS) and containment recirculation systems located outside containment (for example, potential leakage in the normal residual heat removal system (RNS) or the post-accident sampling systems that could be used to recirculate reactor coolant outside containment after an accident)
1.j.(8)	automatic reactor power control system and primary T-average control system
1.j.(13)	excore neutron instrumentation
1.j.(17)	feedwater heater temperature, level, and bypass controls
1.j.(20)	instrumentation used to detect external and internal flooding conditions
1.j.(22)	instrumentation used to track the course of postulated accidents such as containment wide-range pressure indicators, reactor vessel water level monitors, containment sump level monitors, high radiation detectors, and humidity monitors
1.j.(23)	post-accident hydrogen monitors
1.j.(24)	annunciators for reactor control and engineered safety features
1.k.(2)	personnel monitors and radiation survey instruments (As the calibration program applied to these devices will be site-specific, it would be appropriate to identify this as a COL action item.)
1.k.(3)	laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations
1.l.(5)	isolation features for condenser offgas systems
1.m.(4)	static load testing at 125 percent rated load of cranes, hoists, and associated lifting and rigging equipment
1.n.(5)	secondary sampling systems
1.n.(9)	drain systems and pumping systems serving essential areas
1.n.(12)	boron recovery system
1.n.(13)	communications systems relating to offsite emergency notification
1.n.(14)(c)	class 1E electrical room heating, ventilating, and air conditioning
1.n.(14)(f)	main control room (including proper operation of smoke and toxic chemical detection systems and ventilation shutdown devices, including leak tightness of ducts)

1.n.(15) shield cooling systems

1.o.(1) dynamic and static load tests of reactor components handling system cranes, hoists, and associated lifting and rigging equipment

1.o.(2) protective devices and interlocks of reactor components handling system equipment

1.o.(3) safety devices for reactor components handling systems equipment

Initial Fuel Loading and Precritical Tests

2.f reactor core and other major components differential pressure and vibration testing after fuel loading

Low Power Testing

4.l control rod block and inhibit functions

Power Ascension Tests

5.m reactor core and major reactor coolant system (RCS) components differential pressure

5.r process computer and control room computer

5.t pressurizer safety valves and secondary system safety valves

5.c.c gaseous and liquid radioactive waste processing, storage, and release systems (operating in accordance with design)

5.g.g design features to prevent or mitigate ATWS

5.k.k dynamic response of the plant for loss of feedwater heaters or bypassing feedwater heaters

These issues were previously identified by the staff in RAI 260.30 and were subsequently identified as DSER Open Items 14.2.8.3-1 and 14.2.8-16. In its response dated August 13, 1996, Westinghouse stated that Section 14.2.9 was revised to include test abstracts for appropriate AP600 systems and components as specified in RG 1.68, Revision 2, Appendix A.

Through correspondence from November 8, 1996, through May 9, 1997, (see Section 14.2 of this report), there was extensive interaction between the NRC staff and Westinghouse on this issue. Finally, in its response dated October 23, 1997, Westinghouse stated that the following revisions would be made to address the latest staff's comments:

- Section 14.2.9.1.2, "Steam Generator System Testing," would be revised to clearly specify that valves be tested at nominal operating pressure and temperature.

- Section 14.2.9.1.2 would also be revised to include verification of the capability of the steam generator power-operated relief valves to provide the required heat removal.

- Section 14.2.9.1.1, "Reactor Coolant System Testing," Item d, would be revised to specify that instrumentation related to RCS leak detection be properly calibrated and their operation verified.

- Section 14.2.10.4.27, "Feedwater Heater Out of Service Test," would be revised to include testing simulating the loss of one of two main feedwater heaters with the reactor operating at 50 percent power and at 90 percent power.

Westinghouse incorporated these changes into the ITP in Revision 19 to the SSAR. The staff finds that the revised sections incorporate the design features and testing specified in RG 1.68, Revision 2, Appendix A, and are acceptable. Therefore, DSER Open Items 14.2.8.3-1 and 14.2.8-16 are closed.

Additionally, in the DSER, the staff found that the preoperational and startup test phase descriptions in Section 14.2.8 of the SSAR did not provide assurance that the operability of several of the systems and components listed in the following RGs would be demonstrated. The staff concluded that the test abstracts of Section 14.2.8 of the SSAR should be expanded to address the following items, or Appendix 1A of the SSAR should be revised to provide technical justification for any exceptions taken:

- RG 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants" - Preoperational test abstract 14.2.8.1.94, "Remote Shutdown," does not provide sufficient detail to verify conformance with the following RPs of RG 1.68.2:

  - Hot Standby Demonstration (RP C.3), including the following items:

    (1) With initial conditions of the reactor at a moderate power level (10 to 25 percent), demonstrate that plant systems are in the normal configuration with the turbine generator in operation and with the minimum shift crew.

    (2) Using only credited remote shutdown equipment, demonstrate the capability to achieve hot standby status, and maintain stable hot standby conditions for at least 30 minutes.

  - Cold Shutdown Demonstration (RP C.4), including the following items:

    (1) with the plant at hot standby conditions

(2) with the procedurally designated crew positions

(3) using only credited remote shutdown equipment, demonstrate the capability to perform a partial cooldown by performing the following actions:

(a) lower reactor coolant pressure and temperature sufficiently to permit operation of RHR system

(b) initiate and control operation of the RHR system

(c) establish a heat transfer path to the ultimate heat sink

(d) reduce reactor coolant temperature approximately 50°F using the RHR system

- RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems" - Preoperational test abstract 14.2.8.1.6, "Compressed and Instrument Air Systems," did not provide sufficient detail to verify conformance with the following RPs of RG 1.68.3:

  - after-coolers, oil separators, air receivers, and pressure-reducing stations (RP C.2)
  - flow, temperature, and pressure meet design specifications (RP C.4)
  - total air demand with leakage meets design (RP C.5)
  - single failure criterion (RP C.7)
  - sudden and gradual loss of system pressure and appropriate response of air power equipment (RP C.8)
  - functional test for increase in the air supply system pressure does not cause loss of operability (RP C.11)

- RG 1.140 - Preoperational test abstracts 14.2.8.1.28, "Containment Air Filtration System," 14.2.8.1.29, "Radiologically Controlled Area Ventilation Test," and 14.2.8.1.88, "High-Efficiency Particulate Air Filters and Charcoal Absorbers" did not provide sufficient detail to verify conformance with the following RP of RG 1.140:

  - heaters (RP C.3.a)
  - prefilters (RP C.3.m)
  - high-efficiency particulate air (HEPA) filters dioctyl-phthalate polydispersed (DOP) tests (RPs C.3.b and C.5.c)
  - ductwork (RP C.3.f)

- fans and motors mounting and ductwork (RP C.3.i)

- dampers (RP C.3.l)

- absorber sections/cells and activated charcoal (RPs C.3.h and C.5.d)

These issues were previously identified by the staff in RAI 260.31. This was identified as DSER Open Item 14.2.8.4-1.

During its review of Revision 11 to the SSAR, the staff concluded that Westinghouse had satisfactorily addressed the staff's concerns related to demonstrating conformance with RG 1.140. However, the following issues remained with respect to RG 1.68.2 and RG 1.68.3:

<u>RG 1.68.2</u>

In Appendix 1A of the SSAR, Westinghouse stated that an exception was taken regarding the testing of the AP600 remote shutdown workstation in accordance with RG 1.68.2. The basis for this exception is the similarity of the remote shutdown station (RSS) to the main control room (MCR) workstations, the testing of plant control capability from the MCR, and the testing of the RSS controls and indications during preoperational testing.

The RSS testing in the ITP is described in Sections 14.2.9.1.12 and 14.2.9.2.12. Section 14.2.9.1.12, "Protection and Safety Monitoring System Testing," tests, in part, manual reactor trip capability from the RSS. It also tests the processing of manual actuation commands from the RSS to the protection logic cabinets through simulated command inputs to the logic cabinets and simulated logic cabinet outputs on component status to the RSS. Section 14.2.9.2.12, "Plant Control System Testing," provides testing of RSS control functions based on simulated inputs at the RSS and verification of proper output through contact operation, component actuation, or electrical test.

The staff concluded that while similarity of the RSS workstations to those in the MCR, and successful testing of the MCR workstations and individual RSS process signals can provide a certain level of confidence with regard to proper RSS operation, they do not suffice as a replacement for integrated control system testing of the RSS. In addition, although the MCR and RSS workstations may be similar, the working environment is different to the operator from that of the control room which is the normal workspace. The operators should demonstrate the ability to perform plant control in an abnormal work environment with the minimum set of controls and indications available under postulated control room evacuation scenarios. Therefore, Westinghouse needed to modify the Chapter 14 test abstracts to demonstrate the remote shutdown capability of the plant in accordance with RG 1.68.2.

<u>RG 1.68.3</u>

During a meeting with the NRC on March 21, 1995, Westinghouse committed to resolving RAI 410.161 by including preoperational testing as described in RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems" in Section 14.2.9.4.10 of the SSAR, "Compressed and Instrument Air System Testing." Specifically, the following information needed to be added to the test abstract:

- All safety-related pneumatically operated valves should be verified to fail in the position specified in Table 9.3.1-1 of the SSAR upon a complete and sudden loss of instrument air pressure and a gradual loss of instrument air pressure.

- The instrument air system should be functionally tested to ensure credible failures resulting in an increase in instrument air system pressure will not cause loss of operability.

- The instrument air system air quality should be tested to meet ANSI/ISA S7.3, "Quality Standard for Instrument Air."

- While at normal, steady-state instrument air system conditions, if practical, simultaneously operate those plant components requiring large quantities of instrument air for operation to verify that pressure transients in the distribution system do not exceed acceptable values.

- Verify that the total air demand at normal, steady-state conditions, including leakage from the systems is in accordance with design.

- Additionally, the test abstract should include the following statements:

  - "Demonstrate the operability of the air compressor dryers and filters, intercoolers, aftercoolers, moisture separators, and air receivers."

  - "Verify appropriate differential pressures (e.g., delta P across prefilters and afterfilters)."

  - "Verify relief valve settings."

In its response dated May 9, 1997, Westinghouse stated that (1) to satisfy RG 1.68.2, a test of the remote shutdown workstation was added as Section 14.2.10.4.28;
(2) Section 14.2.9.4.10 of the SSAR, "Compressed and Instrument Air System Testing," was modified to provide sufficient detail to show conformance to the applicable portions of RG 1.68.3.

Upon reviewing Revision 13 to Sections 14.2.10.4.28 and 14.2.9.4.10 of the SSAR, the staff found that the exception to RG 1.68.2 described in Appendix 1A of the SSAR should be deleted on the basis of the new test for the RSS. Additionally, Westinghouse did not satisfactorily address the staff's concerns identified in RAI 410.308. Specifically, in Section 14.2.9.4.10 of the SSAR, Westinghouse referenced Section 9.3.1.4 which specified air-operated valve testing in accordance with RG 1.68.3. As noted in RAI 410.308, in Section 14.2.9.4.10 of the SSAR, Westinghouse stated that air-operated valves in safety systems will be tested as part of the test program for the individual system, and the test abstract for the individual system typically reference Section 3.9.6 for valve testing requirements, which does not address RG 1.68.3 testing. The issue identified in RAI 410.308, therefore, remained unaddressed.

In its letter dated January 23, 1998, Westinghouse proposed to revise Section 14.2.9.4.10, General Test Method and Acceptance Criteria (d) to include testing to verify the fail-safe positioning of safety-related air-operated valves for sudden loss of instrument air or gradual loss of pressure as described in Section 9.3.1.4 of the SSAR and suggested by the staff in a letter dated January 7, 1998. Westinghouse also proposed to remove the exception to RG 1.68.2 in Appendix 1A of the SSAR.

The staff found that in Revision 20 to Sections 14.2.10.4.28 and 14.2.9.4.10 of the SSAR, Westinghouse adequately addressed the issues identified in DSER Open Item 14.2.8.4-1 and RAI 410.308. However, Westinghouse deleted the statement in Appendix 1A indicating conformance of the AP600 design to RG 1.68.2.

In its letter dated March 10, 1998, Westinghouse stated that Appendix 1A to the SSAR would be revised to add a reference for the conformance to RG 1.68.2. In Revision 22 to Appendix 1A of the SSAR, Westinghouse reinstated the AP600 design commitment to the guidance in RG 1.68.2. Therefore, DSER Open Item 14.2.8.4-1 is closed .

Except for the item identified above, the staff finds that the applicable RGs (i.e., those pertaining to the ITP) identified in Section 1.9.1, Table 1.9-1, and Appendix 1A of the SSAR are consistent with the acceptance criteria in Section 14.2 of the SRP.

14.2.5 Utilization of Reactor Operating and Testing Experience

In Section 14.2.5 of the SSAR, Westinghouse states that experience in the design, testing, startup, and operation from previous pressurized water reactor (PWR) plants is utilized in the development of the initial preoperational and startup test program for the AP600 design. Other sources of experience reported and described in various documents, such as NRC reports including Inspection and Enforcement bulletins, and Institute of Nuclear Power Operations (INPO) reports including Significant Operating Event Reports (SOERs) are also utilized in the AP600 initial preoperational and startup test program.

Special tests to further establish a unique phenomenological performance parameter of the AP600 design features beyond testing performed for design certification and that will not change from plant to plant, will be performed for the first plant only. Because of the standardization of the AP600 design, these special tests (designated as first-plant-only tests) will not be required on subsequent plants. These first-plant-only tests are identified in the individual test descriptions in Sections 14.2.9 and 14.2.10 of the SSAR.

The following is a listing of the first-plant-only tests and the corresponding SSAR sections:

First- Plant-Only Tests	SSAR Section
IRWST Heatup Test	14.2.9.1.3 Item (h)
Pressurizer Surge Line Stratification Evaluation	14.2.9.1.7 Item (d)
Reactor Vessel Internals Vibration Testing	14.2.9.1.9 - Prototype Test

Natural Circulation Tests	14.2.9.10.3.6, 14.2.10.3.7
Load Follow Demonstration	14.2.10.4.22

Other special tests which further establish a unique phenomenological performance parameter of the AP600 design features beyond testing performed for design certification and that will not change from plant to plant, will be performed on the first three plants. Because of the standardization of the AP600 design, once these special tests have affirmed consistent passive system function they are not required on subsequent plants. The following is a listing of tests required on the first three plants and the corresponding SSAR sections:

First-Three-Plant Tests	SSAR Section
Core Makeup Up Tank Heated Recirculation Tests	14.2.9.1.3 Items (k) and (w)
ADS Blowdown Test	14.2.9.1.3 Item (s)

In Sections 14.2.5 and 14.4.6, Westinghouse further states that for subsequent plants, if design changes are made that could affect the applicability of the results of first-plant-only tests or first-three-plant tests, or if variances between the as-built plant and the certified design are discovered during the engineering and construction process, the COL applicant shall either perform the subject test, or shall provide justification that the results of the first-plant-only tests or first-three-plant tests are applicable to the subsequent plant. This is COL Action Item 14.2.5-1.

The staff finds that the use of reactor operating and testing experience in the development of the test program meets the acceptance criteria in Section 14.2 of the SRP and is acceptable. Additionally, the staff finds Westinghouse's basis for the performance of special AP600 tests consistent with the guidance in RG 1.68 and past reactor testing experience. In Section 14.2.5 of the SSAR, Westinghouse lists the justifications for first-plant-only tests and first-three-plant tests as provided below. The staff finds that the justifications for performing the designated tests on the first plant only and on the first three plants are appropriate.

IRWST Heatup Test (Section 14.2.9.1.3 item (h))

During preoperational testing of the passive core cooling system, a natural circulation test of the passive residual heat removal (PRHR) heat exchanger is conducted (item f). For the first plant only, thermocouples will be placed in the IRWST to observe the thermal profile developed during the heatup of the IRWST water during PRHR heat exchanger operation. This test will be useful in confirming the results of the AP600 Design Certification Program PRHR tests with regards to IRWST mixing and is useful in quantifying the conservatism in the Chapter 15 transient analyses.

Because of the standardization of the AP600, the heatup and thermal stratification characteristics of the IRWST will not vary from plant to plant. The PRHR heat exchanger design, and the size and configuration of the IRWST are standardized, such that the heatup characteristics will not significantly change from plant to plant.

Therefore, since the phenomenon to be tested (i.e., heatup and mixing characteristics of the IRWST) will not vary significantly from plant to plant because of standardization, a first-plant-only test of the IRWST heatup characteristics is acceptable.

Core Makeup Tank Heated Recirculation Tests (Section 14.2.9.1.3 Items (k) and (w))

During preoperational testing of the passive core cooling system, tests of the core makeup tanks will be performed to verify the core makeup tank (CMT) inlet and inlet piping resistances. In addition, cold draining tests of the CMTs will be conducted that verify proper operation of the CMTs. For the first plant only, two additional CMT tests will be conducted during hot functional testing of the RCS. These tests are a natural circulation heatup of the CMTs followed by a test to verify the ability of the CMTs to transition from a recirculation mode to a draindown mode while at elevated temperature and pressure.

Operation of the CMTs under natural circulation is conducted on the first plant only for the following reasons:

- Natural circulation of the CMTs will not vary from plant to plant, provided that the other verifications discussed above are performed as specified.
- Natural circulation testing of the CMTs was extensively tested in the various Design Certification Tests including the CMT separate affects test and the SPES-2 and OSU integral tests.
- Performance of this test results in significant thermal transients on Class 1 components including the CMTs and the direct vessel injection nozzles.

ADS Blowdown Test (Section 14.2.9.1.3 Item (s))

During preoperational testing of the passive core cooling system, the resistance of the automatic depressurization system Stage 1, 2, and 3 flowpath(s) is verified. For the first plant only, an automatic depressurization blowdown test will be performed. This test will be performed during hot functional testing of the RCS and results in a significant blowdown of the RCS into the IRWST. This tests verifies proper operation of the ADS valves, and demonstrates the proper operation of the ADS spargers to limit the hydrodynamic loads in containment to less than design limits. This test is performed on the first plant only for the following reasons:

- The operation of the ADS and the resultant hydrodynamic loads will not vary from plant to plant.
- Full scale automatic depressurization testing was performed in the AP600 Design Certification Program. Testing was conducted to conservatively bound ADS flow rates and resultant hydrodynamic loads that will be experienced by the plant during ADS operation.
- Performance of this test results in significant thermal transients on Class 1 components including the primary components. It also results in hydrodynamic loads in containment including the IRWST.

Pressurizer Surge Line Stratification Evaluation (Section 14.2.9.1.7 Item (d))

As part of the AP600 conformance to NRC Bulletin 88-11, a monitoring program will be implemented by the COL applicant of the first AP600 to record temperature distributions and thermal displacements of the surge line piping during hot functional testing and during the first fuel cycle, as discussed in Section 3.9.3 of the SSAR.

Reactor Vessel Internals Vibration Testing (Section 14.2.9.1.9)

The preoperational vibration test program for the reactor internals conducted on the first AP600 plant is consistent with the guidelines of RG 1.20 for a comprehensive vibration assessment program. This program is discussed in Section 3.9.2 of the SSAR.

Natural Circulation Tests (Sections 14.2.10.3.6 and 14.2.10.3.7)

Natural circulation tests using the steam generators and the PRHR heat exchangers will be performed at low-core power during the startup test phase of the ITP for the first AP600 plant. This testing of the heat removal systems meets the intent of the requirement to perform natural circulation testing and the results of this testing will be factored into the operator training. Justification for performing these natural circulation tests for the first plant only is provided in Section 1.9.4 of the SSAR.

Load Follow Demonstration (Section 14.2.10.4.22)

A load follow demonstration test is not required by RG 1.68. However, the AP600 performs load follow with grey rods, as opposed to current Westinghouse PWRs which manipulate RCS boron concentration to perform load follow operations. Therefore, Westinghouse has included a proof of principle load follow demonstration for the first AP600 plant to demonstrate the ability of the AP600 plant to follow a design-basis daily load follow cycle.

14.2.6 Use of Plant Operating and Emergency Procedures

In Section 14.2.6 of the SSAR, Westinghouse states that, as appropriate and to the extent practicable, plant normal, abnormal, and emergency operating procedures will be used when performing preoperational and startup tests. The use of these procedures is intended to fulfill the following objectives:

- to demonstrate the adequacy of the specific procedure, or identify changes that may be required
- to increase the level of knowledge of plant personnel on the systems being tested

A test procedure using a normal, abnormal, or emergency operating procedure will reference the procedure directly, or will extract a series of steps from the procedure to accomplish the operator training goals while safely and efficiently performing the specified testing.

The staff finds that the method described herein for incorporating operating, abnormal, or emergency procedures into the test program meets the acceptance criteria in Section 14.2 of the SRP and is acceptable.

14.2.7 Initial Fuel Loading and Initial Criticality

In Section 14.2.7 of the SSAR, Westinghouse states that initial fuel loading and subsequent initial criticality and power ascension to full licensed power will be performed during the startup test program. Before the initiation of these operations, the systems and conditions necessary to bring the plant into compliance with the technical specifications must be operable and satisfied, respectively. These operations will be performed in a controlled and safe manner using test procedures that specify the following items:

- Required prerequisite testing
- Operational status of required systems
- Step-by-step instructions
- Precautions that must be observed
- Actions to be taken in the event of unanticipated or abnormal response

14.2.7.1 Initial Fuel Loading

In Section 14.2.7.1 of the SSAR, Westinghouse states that the minimum conditions for initial core loading are as follows:

- The composition, duties, and emergency procedure responsibilities of the fuel handling crew are established.

- Radiation monitors, nuclear instrumentation, manual initiation controls, and other devices to actuate alarms and ventilation controls are tested and verified to be operable.

- The status of systems required for fuel loading is established and verified.

- The status of protection systems, interlocks, mode switch, alarms, and radiation protection equipment is established and verified for fuel loading.

- Inspections of fuel and control rods have been made.

- Containment integrity has been established to the extent required by the technical specifications.

- The reactor vessel status has been established for fuel loading. Components are verified to be in place or out of the vessel as required for fuel loading.

- Required fuel handling tools are available, operational, and calibrated, including indexing of the manipulator crane with a dummy fuel element. The fuel handling tools are successfully tested.

- Reactor coolant water quality requirements are established and the reactor coolant water quality is verified.

- The reactor vessel is filled with water to a level approximately equal to the center of the vessel outlet nozzles. The reactor coolant water is circulating at a rate which provides uniform mixing.

- The boron concentration in the reactor coolant is verified to be equal to or greater than required by the plant technical specifications for refueling, and is being maintained under a surveillance program.

- Sources of unborated water to the RCS are isolated and under a surveillance program.

- At least two neutron detectors are calibrated, operable, and located such that changes in core reactivity can be detected and recorded. One detector is connected to an audible count rate indicator and a containment alarm.

- A response check of nuclear instruments to a neutron source is required within 8 hours before loading (or resumption of loading, if delayed for 8 hours or more).

Fuel assemblies, together with inserted components (e.g., control rods, burnable poison assemblies, and primary and secondary neutron sources) are placed in the reactor vessel according to an established and approved sequence.

During and following the insertion of each fuel assembly, until the last fuel assembly is loaded, the response of the neutron detectors will be observed and compared with previous fuel loading data or calculations to verify that the observed changes in core reactivity are as expected. Specific instructions will be provided if unexpected changes in reactivity are observed.

Because of the unique conditions existing during initial fuel loading, temporary neutron detectors may be used in the reactor vessel to provide additional reactivity monitoring. Credit for the use of temporary detectors may be taken in meeting technical specification requirements for the number of operable source range channels.

The staff finds that the initial fuel loading described in Sections 14.2.7 and 14.2.7.1 of the SSAR meets the acceptance criteria in Section 14.2 of the SRP and is acceptable.

14.2.7.2 Initial Criticality

In Section 14.2.7.2 of the SSAR, Westinghouse states that following initial fuel loading, the reactor upper internals and pressure vessel head will be installed. Additional mechanical and electrical tests will be performed in preparation for critical and power operations. The following conditions will exist before initial criticality:

- The RCS is filled and vented.

- Tests are completed on the control rod drive system that demonstrate that the control rods are latched, the control and position indication systems are functioning properly, and the rod drop time under hot full flow conditions is less than the technical specification limit.

- Tests are completed that demonstrate that the plant control and protection systems are operable and the reactor trip breakers respond as designed to appropriate trip signals.

- The RCS is at hot no-load temperature and pressure. The reactor coolant boron concentration is such that the shutdown margin requirements of the technical specifications are satisfied for the hot shutdown condition.

Initial criticality will be achieved in an orderly, controlled fashion by a combination of shutdown and control bank withdrawal and RCS boron concentration reduction.

During the approach to initial criticality, the response of the source range nuclear instruments will be used as an indication of the rate of reactivity addition and the proximity to a critical condition so that criticality is achieved in a controlled, predictable fashion.

Rates for rod withdrawal and boron reduction will be specified such that the startup rate will be less than 1 decade per minute. Following criticality and before operation at power levels greater than 5 percent of rated power, physics tests will be performed to verify that the operating characteristics of the reactor core are consistent with design predictions. During these tests, values will be obtained for the reactivity worth of control and shutdown rod banks, isothermal temperature coefficient, and critical boron concentration for selected rod bank configurations.

Other tests at low power include verification of the response of the nuclear instrumentation system and radiation surveys.

The staff finds that the initial criticality programmatic controls described in Sections 14.2.7 and 14.2.7.2 of the SSAR meet the acceptance criteria in Section 14.2 of the SRP and are acceptable.

14.2.7.3 Power Ascension

In Section 14.2.7.3 of the SSAR, Westinghouse states that after the operating characteristics of the reactor have been verified by low-power testing, a power ascension program will bring the unit to its full-rated power level in successive stages. At each successive stage, hold points will be provided to evaluate and approve test results before proceeding to the next stage. The minimum test requirements for each successive stage of power ascension will be specified in the applicable startup test procedures.

During the power ascension program, tests will be performed at various power levels as follows:

- Statepoint data, including secondary system heat balance measurements, will be obtained at various power levels up to full licensed power. This information will be used to project plant performance during power escalation, provide calibration data for the various plant control and protection systems, and provide the bases for plant trip set points.

- At prescribed power levels, the dynamic response characteristics of the primary and secondary systems will be evaluated. System response characteristics will be measured for design step-load changes, rapid load reductions, and plant trips.

- Adequacy of the radiation shielding will be verified by gamma and neutron radiation surveys. Periodic sampling will be performed to verify the chemical and radiochemical analysis of the reactor coolant.

- Using the incore instrumentation, the power distribution of the reactor core will be measured to verify consistency with design predictions and technical specification limits on peaking factors.

The staff finds that the power ascension testing described in Section 14.2.7.3 of the SSAR meets the acceptance criteria in Section 14.2 of the SRP and is acceptable.

14.2.8 Test Program Schedule

In Section 14.2.8 of the SSAR, Westinghouse states that the schedule for the initial fuel load and for each major phase of the ITP includes the timetable for generation, review, and approval of procedures, as well as the actual testing and analysis of results.

Preoperational testing will be performed as system and equipment availability allows. The interdependence of systems will also be considered. Sequencing of the startup tests will depend on specified power and flow conditions and inter-system prerequisites. The startup test schedule will establish that, before core load, the test requirements will be met for those plant SSCs that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. Testing will be sequenced so that the safety of the plant is not dependent on untested systems, components, or features.

In the DSER, the staff found that the SSAR needed to be revised to state that the startup administrative manual (procedures) will be the responsibility of the COL applicant, as will other documents that delineate the test program schedule for the ITP. This was identified as DSER Confirmatory Item 14.2.7-1.

In its response dated August 13, 1996, Westinghouse stated that Section 14.4 of the SSAR was revised to include a COL information item to provide a startup administrative manual that will delineate the test program schedule for staff review.

In its response to Westinghouse dated November 8, 1996, the staff concluded that closure of this issue was contingent upon the satisfactory resolution of DSER Open Item 14.2.9-2, which was also associated with the startup administrative manual. The resolution of DSER Open Item 14.2.9-2 is documented in Section 14.4.3, below. The staff finds this acceptable and therefore DSER Confirmatory Item 14.2.7-1 is closed.

The staff finds that the test program schedule described in Section 14.2.8 of the SSAR meets the acceptance criteria in Section 14.2 of the SRP and is acceptable.

14.2.9 Preoperational Test Descriptions

In Section 14.2.9 of the SSAR, Westinghouse states that test abstracts are provided for the preoperational testing of systems/components that perform safety-related functions; that are non-safety-related but perform functions designated to provide defense-in-depth; systems/components that may contain radioactive material; and other applicable non-safety-related systems in accordance with RG 1.68, Revision 2, Appendix A. A limited number of these test abstracts establish performance parameters of AP600 design features that do not change from plant to plant. Because the AP600 plant is standardized, these tests will only be performed on the first plant. These test abstracts are clearly identified in Section 14.2.5 of the SSAR.

The staff's evaluation in Chapter 14 of the DSER was on the basis of Revision 1 to the SSAR. In draft Revision 10 of the SSAR, however, Westinghouse modified the information in Chapter 14 substantially and added several sections. The information previously presented in Section 14.2.8, "Individual Test Descriptions," was allocated into two main sections, (1) Section 14.2.9, "Preoperational Test Descriptions," and (2) Section 14.2.10, "Startup Test Procedures," each having multiple sections delineating different phases of testing and/or system/component functions. Therefore, the following sections present the staff's conclusions with regard to Section 14.2.8 and individual test description deficiencies documented in the DSER.

14.2.9.1 Evaluation of DSER Section 14.2.8 Open and Confirmatory Items

DSER Confirmatory Item 14.2.8-1

In the DSER, the staff noted that in the response to RAI 260.26, dated August 8, 1994, Westinghouse indicated that the following changes would be made to the following preoperational and startup test abstracts:

- Preoperational test abstracts 14.2.8.1.77, 14.2.8.1.78, and 14.2.8.1.82 would have their headings revised to delete the designation of "(First Plant Only)," as will startup test abstract 14.2.8.2.20. Westinghouse indicated, however, that the first-plant-only designation would be retained for the dynamic response and vibration measurement portions of preoperational test abstract 14.2.8.1.77. (That abstract includes vibration measurement instrumentation installed for monitoring system and component vibration for the first AP600.)

- An inspection program, as defined in Section 3.1.3 of RG 1.20, would also be conducted on the first AP600 plant. This would provide the basis for performing an inspection program (Section 3.4.3 of RG 1.20) for succeeding AP600 plants. The inspections for these plants would be implemented under Section 14.2.8.1.67, Hot Functional Testing (See RAI 210.58 for additional information).

- Preoperational test abstracts 14.2.8.1.78 and 14.2.8.1.82 would each be modified to be performed on each plant, as would startup test abstract 14.2.8.2.20 (See RAI 210.53 for additional information).

The staff found that these responses agreed with the guidance of RG 1.20 and 1.68, met the acceptance criteria of Section 14.2 of the SRP, and were acceptable (subject to satisfactory resolution of RAI 210.53 and RAI 210.58). The revision of the SSAR was identified as DSER Confirmatory Item 14.2.8-1.

Subsequent to the DSER, however, Westinghouse modified Chapter 14 of the SSAR substantially and the treatment of these special initial test program tests (designated as first-plant-only) is discussed in SSAR Section 14.2.5, "Utilization of Reactor Operating and Testing Experience in the Development of Test Program." The staff's evaluation of the special initial test program tests is in Section 14.2.5 of this report. The staff finds this acceptable, and therefore, DSER Confirmatory Item 14.2.8-1 is closed.

DSER Open Item 14.2.8-14

In the DSER, the staff found that Westinghouse should revise Section 14.2.8 of the SSAR to reconcile its contents with that of Section 14.2.2 of the SSAR, as discussed above in relation to RAI 260.24. This was identified as DSER Open Item 14.2.8-14.

In its response dated August 13, 1996, Westinghouse stated that responses to RAIs 260.24 and 260.28 were provided in a letter to the NRC dated July 22, 1994 and Section 14.4 of the SSAR was revised to specify that the COL will provide appropriate ITP documents for review by the staff.

In its response to Westinghouse dated November 8, 1996, the staff concluded that the closure of this issue was contingent upon the satisfactory resolution of DSER Open Item 14.2.2-2. The resolution of DSER Open Item 14.2.2-2 is documented in Section 14.4.4 of this report The staff finds this acceptable, and therefore, DSER Open Item 14.2.8-14 is closed.

DSER Open Item 14.2.8-15

In the DSER, the staff found that Section 14.2.8 of the SSAR, and the individual test methods or performance criteria, should be revised to provide specific references to the basis for determining acceptable system and component performance. This issue was previously identified in RAI 260.29 and subsequently identified as DSER Open Item 14.2.8-15.

In its response dated December 6, 1996, Westinghouse stated that specific references were provided in each preoperational test abstract. These references specify the SSAR section which defines the functions performed by each system that are tested in the ITP. However, on the basis of the staff's review of Revision 11 to Chapter 14 of the SSAR, this issue was superseded by RAI 260.51 as discussed below.

In RAI 260.51, the staff requested that certain test abstracts be revised to provide specific acceptance criteria or design-basis functional requirements traceable to the appropriate SSAR sections. Westinghouse responded in a August 13, 1996, letter that the noted test abstracts were revised. The staff's review of the revised Chapter 14 determined that a number of additional sections required more detailed acceptance criteria. The staff identified 23 sections that needed to be revised to provide or reference specific acceptance criteria or design-basis functional requirements traceable to specific sections or numbered items of the SSAR, the plant technical specifications, or other appropriate references that contain the detailed structure, system, or component design/performance criteria verified by the testing.

In its response dated May 9, 1997, Westinghouse provided the additional performance criteria references that would be added to the 23 sections of the SSAR that were identified by the staff. The staff found the added references acceptable with eight exceptions.

In its response dated October 23, 1997, Westinghouse stated that the performance criteria for these eight test abstracts would be modified to include a reference to "applicable design specifications." Additionally, the following changes would be made to the identified test abstracts to resolve the remaining issues with regards to specific acceptance criteria or design-basis functional requirements traceable to appropriate SSAR sections:

- Section 14.2.10.1.7, "Incore Instrumentation System Precritical Verification," would be revised to add references to Section 7.5 and applicable design specifications.

- Section 14.2.10.1.8, "Resistance Temperature Detectors-Incore Thermocouple Cross Calibration," would be revised to add references to Tables 7.2-1 and 7.3-4 in Sections 7.2 and 7.3, respectively.

- Section 14.2.10.1.16, "Process Instrumentation Alignment," would be revised to delete the reference to Section 7.7.1.1 and add reference to Tables 7.2-1 and 7.3-4 in Sections 7.2 and 7.3 respectively.

- Section 14.2.10.1.20, "Feedwater Valve Stroke Test," would be revised to delete Section 7.7.1.8 and only reference applicable design specifications.

- Section 14.2.10.2.4, "Post-Critical Reactivity Computer Checkout," would be revised to delete the reference to Section 4.3.2.6 and add a reference to Section 7.7.

- Section 14.2.10.3.2, "Determination of Physics Testing Range," would be revised to delete the reference to Section 4.3.2.2.8. No reference performance criteria is provided since the zero-power testing range is the result of the test as determined by the stated test method. This is consistent with previous tests of this nature as performed on current operating plants.

- Section 14.2.10.4.5, "Startup Adjustments of Reactor Control Systems," would be revised to delete the reference to Section 7.7.1.1 and add a reference to Section 5.1.

- Section 14.2.10.4.10, "Process Instrumentation Alignment at Power Conditions," would be revised to delete the reference to Section 4.3.2.4.16 and add a reference to Section 5.1.

Westinghouse incorporated these changes in the ITP in Revision 17 to the SSAR. The staff finds the Westinghouse response acceptable, and therefore, RAI 250.51 and DSER Open Item 14.2.8-15 are closed.

DSER Open Items 14.2.8-1 through 14.2.8-5, -8, -10, and -12

In the DSER, the staff noted that in response to an NRC staff question, Westinghouse indicated that certain preoperational and startup test abstracts would not be modified to address the testing of successive AP600 plants. However, as a result of Revision 9 to the SSAR, these preoperational and startup test abstracts were either subsumed into, or superseded in their entirety, by new test abstracts, as described below:

- Preoperational test abstract 14.2.8.1.80, performed on the first plant system, will prove the depressurization capability of the automatic depressurization system. Each succeeding standard AP600 automatic depressurization system will meet or exceed the plant depressurization criteria as demonstrated by satisfying the appropriate ITAAC. Component functionality will be proven by performing testing described in abstract 14.2.8.1.79. This test will not be amended to include all plants.

  The staff found this response unacceptable. Preoperational test abstract 14.2.8.1.80 should be modified in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants or to provide appropriate justification for this exception to RG 1.68, Appendix A, Items 1.a.(2)(d) and 1.h.(2). This was identified as DSER Open Item 14.2.8-1.

  In Revision 9 to the SSAR, preoperational test abstract 14.2.8.1.80 was subsumed into Section 14.2.9.1.3, "Passive Core Cooling System Testing," which provided appropriate justification for not performing testing on subsequent plants in accordance with the provisions of RG 1.68, Appendix A, Items 1.a.(2)(d) and 1.h.(2), as described in Section 14.2.5 of the SSAR. Therefore, DSER Open Item 14.2.8-1 is closed.

- Preoperational test abstract 14.2.8.1.85, performed on the first plant system, will prove the thermodynamic capabilities of the passive core cooling system (PXS) to meet or exceed performance criteria. Each succeeding standard AP600 PXS will meet or exceed the performance criteria as demonstrated by satisfying the appropriate ITAAC. Component functionality will be proven by performing test abstract 14.2.8.1.84. This test will not be amended to include all plants.

  The staff found this response unacceptable. Preoperational test abstract 14.2.8.1.85 should be modified in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants, or to provide appropriate justification for this exception to RG 1.79 and RG 1.68, Appendix A, Item 1.h.(1). This was identified as DSER Open Item 14.2.8-2.

  In Revision 9 to the SSAR, preoperational test abstract 14.2.8.1.85 was subsumed into Section 14.2.9.1.3, "Passive Core Cooling System Testing," which provided appropriate justification for not performing testing on subsequent plants in accordance with the provisions of RG 1.79 and RG 1.68, Appendix A, Item 1.h.(1), as described in Section 14.2.5 of the SSAR. Therefore, DSER Open Item 14.2.8-2 is closed.

- Preoperational test abstract 14.2.8.1.87, performed on the first plant system will prove the thermodynamic capabilities of the PRHR to meet or exceed performance criteria.

Each succeeding standard AP600 PRHR subsystem will meet or exceed the performance criteria as demonstrated by satisfying the appropriate ITAAC. Component functionality will be proven by performing test abstract 14.2.8.1.86. This test will not be amended to include all plants.

The staff found this response unacceptable. Preoperational test abstract 14.2.8.1.87 should be modified in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants, or to provide appropriate justification for this exception to RG 1.139 and RG 1.68, Appendix A, Items 1.d.(5), 1.d.(8), and 1.h. This was identified as DSER Open Item 14.2.8-3.

In Revision 9 to the SSAR, preoperational test abstract 14.2.8.1.87 was subsumed into Section 14.2.9.1.3, "Passive Core Cooling System Testing," which provided appropriate justification for not performing testing on subsequent plants in accordance with RG 1.139 and RG 1.68, Appendix A, Items 1.d.(5), 1.d.(8), and 1.h, as described in Section 14.2.5 of the SSAR. Therefore, DSER Open Item 14.2.8-3 is closed.

- Preoperational test abstract 14.2.8.1.94, performed on the first plant system, will prove the remote shutdown design capability. The test will also prove remote manual control of the required valves and functions to effect cooldown from hot standby and cooldown to safe-shutdown conditions with safety-related systems. For each succeeding standard AP600 remote shutdown workstation, control functionality will be proven by testing components (switches) in systems required to be checked in accordance with plant technical specifications, as permitted by RG 1.68 RP C.3 and C.4. This test will not be amended to include all plants.

  The staff found this response unacceptable. Preoperational test abstract 14.2.8.1.94 should be modified in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants, or to provide appropriate justification for this exception to RG 1.68.2, RP C.3 and C.4. This was identified as DSER Open Item 14.2.8-4.

  In Revision 9 to the SSAR, preoperational test abstract 14.2.8.1.94 was superseded by Section 14.2.10.4.28, "Remote Shutdown Workstation," which satisfactorily addressed the provisions of RG 1.68.2, RP C.3 and C.4 for all plants. Additionally, this issue is addressed as part of the resolution of DSER Open Item 14.2.8.4-1. Therefore, DSER Open Item 14.2.8-4 is closed.

- Preoperational test abstract 14.2.8.1.97, performed on the first plant, will prove the thermodynamic capabilities of the passive containment cooling system (PCS) to meet or exceed the performance criteria. Each succeeding standard AP600 PCS will meet or exceed the performance criteria, as demonstrated by satisfying their appropriate ITAAC. Component functionality will be proven by performing test abstract 14.2.8.1.96. This test will not be amended to include all plants.

  The staff found this response unacceptable. Preoperational test abstract 14.2.8.1.97 should be modified in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants, or to provide appropriate justification for this exception to RG 1.68, Appendix A, Item 1.h.(3). This was identified as DSER Open Item 14.2.8-5.

In Revision 9 to the SSAR, preoperational test abstract 14.2.8.1.97 was superseded by Section 14.2.9.1.4, "Passive Containment Cooling System Testing," which satisfactorily addresses the provisions of RG 1.68, Appendix A, Item 1.h.(3) for all plants. Therefore, DSER Open Item 14.2.8-5 is closed.

- Startup test abstract 14.2.8.2.38 will confirm that excore detector uncertainties are enveloped by assumptions in the safety analysis. The results of this test will apply to each succeeding standard AP600.

The staff found this response unacceptable. Startup test abstract 14.2.8.2.38 should be modified in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants, or to provide appropriate justification for this exception to RG 1.68, Appendix A, Items 5.b and 5.y. This was identified as DSER Open Item 14.2.8-8.

In Revision 9 to the SSAR, preoperational test abstract 14.2.8.2.38 was subsumed into Section 14.2.10.4.9, "Process Measurement Accuracy Verification," which satisfactorily addresses the provisions of RG 1.68, Appendix A, Items 5.b and 5.y for all plants. Therefore, DSER Open Item 14.2.8-8 is closed.

- Startup test abstract 14.2.8.2.47 will be performed to prove that the core and instrumentation design meets the performance criteria of rod misalignment per the requirements of Items 5.e and 5.f of RG 1.68. Furthermore, the sensitivity of the nuclear instrumentation to rod misalignments will be demonstrated per the requirements of Item 5.l of RG 1.68. The results of this test will apply to each succeeding standard AP600.

The staff found that this response agrees with the guidance of RG 1.68, for Appendix A, Items 5.e and 5.f, and the acceptance criteria of Section 14.2 of the SRP, and is therefore acceptable. However, startup test abstract 14.2.8.2.47 should be modified to include applicability of Appendix A, Item 5.l, testing to subsequent AP600 plants, or to provide appropriate justification for such exception. This was identified as DSER Open Item 14.2.8-10.

In Revision 9 to the SSAR, preoperational test abstract 14.2.8.2.47 was superseded by Section 14.2.10.4.6, "Rod Cluster Control Assembly Out of Bank Measurements," which satisfactorily addresses the provisions of RG 1.68, Appendix A, Items 5.e, 5.f, and 5.l for all plants. Therefore, DSER Open Item 14.2.8-10 is closed.

- Startup test abstract 14.2.8.2.52 will be performed to prove the plant design response during load follow maneuvers. The results of this test will apply to each succeeding standard AP600 plant.

The staff found this response unacceptable. Startup test abstract 14.2.8.2.52 should be modified in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants, or to provide appropriate justification for this exception to RG 1.68, Appendix A, Item 5.h.h. This was identified as DSER Open Item 14.2.8-12.

In Revision 9 to the SSAR, preoperational test abstract 14.2.8.2.52 was subsumed into Sections 14.2.10.4.20, "Load Swing Test," and 14.2.10.4.22, "Load Follow Demonstration (First Plant Only)." Section 14.2.10.4.22 supplements the testing provisions in RG 1.68, Appendix A, Item 5.h.h, which are satisfactorily addressed by Section 14.2.10.4.20 for all plants. Therefore, DSER Open Item 14.2.8-12 is closed.

14.2.9.2 Evaluation of DSER Test Abstract Open Items

Section 14.2.9.1.6, "Main Control Room Emergency Habitability System Testing"

In the DSER (Preoperational test abstract 14.2.8.1.100, "Main Control Room Habitability System"; RG 1.68, Appendix A, Item 1.n.(14)(f)), the staff found that Westinghouse should provide testing of the MCR emergency habitability system on subsequent AP600 plants. This was identified in RAI 260.26 and subsequently as DSER Open Item 14.2.8-6.

Westinghouse responded in a letter dated August 13, 1996, that Section 14.2.9.1.6, "Main Control Room Emergency Habitability System Testing," would be revised to include appropriate testing for each plant, but that a long-term demonstration of this system would be conducted only for the first plant.

Staff review of this section determined that sufficient assurance did not exist to conclude that the heat loads in the MCR area are identical for all AP600 plants. The AP600 does not provide active, safety-related heating, ventilation, and air conditioning (HVAC) for the MCR, I&C equipment rooms, and Class 1E dc equipment rooms. The habitability of the MCR area is provided by operation of the MCR emergency habitability system, and by the passive heat sinks associated with the MCR structure. Likewise, the environmental conditions that the qualified I&C equipment and class 1E equipment will be exposed to on the basis of the passive heat sinks associated with the building and structures that house this equipment. Therefore, the staff concluded that Section 14.2.9.1.6 should be modified to include applicability of this testing to subsequent AP600 plants, or Appendix 1A in the SSAR should provide appropriate justification for this exception to RG 1.68, Appendix A, Item 1.n.(14)(f).

In its response dated May 9, 1997, Westinghouse stated that in the AP600, a design-basis heatup analysis of the MCR, I&C equipment rooms, and Class 1E dc equipment rooms is performed, and the results are discussed in Section 6.4 of the SSAR. This analysis assumes maximum bounding heat loads for the equipment that could be located in the MCR and equipment rooms. In the AP600 Certified Design Material (CDM) for the MCR Emergency Habitability System (Section 2.2.5, Item 8c, and Item 8c in Table 2.2.5-4), Westinghouse specifies that an evaluation will be performed using as-built information and heat loads from installed equipment for the (1) MCR, (2) I&C equipment rooms, and (3) Class 1E dc equipment rooms. In addition, this evaluation considers the as-built passive heat sinks associated with these rooms, as specified in CDM Section 3.3, Nuclear Island Building Structures. The acceptance criteria for this heat sink capacity analysis results are as follows:

(1) The temperature rise for the MCR is less than or equal to 8.33 °C (15 °F) for the 72-hour period.

(2) The maximum temperature for the 72-hour period for the I&C rooms is less than 51.7 °C (125 °F).

(3) The maximum temperature for the 72-hour period for the Class 1E dc equipment rooms is less than or equal to 51.7 °C (125 °F).

The first-plant-only test specified in Section 14.2.9.1.6 is a test of the long-term heatup characteristics of the MCR, I&C, and Class 1E equipment rooms. It is performed to demonstrate the heatup characteristics of these rooms when they are subjected to a known heat load. This test can be used to provide data for comparison to the design-basis analyses. However, testing is not required on subsequent plants, since these plants are required to be built to the requirements specified in the CDM. As a passive heatup of these rooms is not dependent on the proper operation of a system, but is rather a function of the heat loads and passive heat sinks provided in the design, a verification of these parameters (via the ITAAC process) is sufficient to verify the safety of an AP600 built to the specifications contained in the CDM.

In the letter dated May 9, 1997, Westinghouse proposed that the SSAR be revised to state, "An exception to RG 1.68, Appendix A, Item 1.n.(14)(f) was added to SSAR Appendix 1A that states this test needs to be performed for the first plant only provided the design-basis heat loads used as assumptions in the heat sink capacity analysis bound the actual as-built information and heat loads."

While the staff agreed that the ability of the habitability system to maintain the MCR environment as well as temperatures in the protection and safety monitoring system cabinet and emergency switchgear rooms during a long-term loss of the nuclear island nonradioactive ventilation system *may be verified with a limited duration test* [emphasis added], the staff does not agree that such testing (1) is so impractical or burdensome that it should be performed on the first plant only, and/or (2) should not include specific verification and duration acceptance criteria.

In its response dated October 23, 1997, Westinghouse stated that Section 14.2.9.1.6 would be revised to require testing of the MCR habitability (item e) as part of the preoperational testing for all plants. In addition, Section 6.4 of the SSAR would be revised to specify the criteria for air quality and temperature. This test would be specified to be performed for a sufficient duration to verify that criteria are met at 72 hours. Testing to verify that protection and safety monitoring system cabinet and emergency switchgear rooms heatup at a rate consistent with the 72-hour temperature limit criteria (item f), would be revised to be performed as a preoperational test for all plants.

The staff found that the acceptance criteria changes to Section 14.2.9.1.6, items (e) and (f), in Revision 17 to the SSAR, did not establish measurable parameters for verifying that the protection and safety monitoring system cabinet and emergency switchgear rooms heatup at a rate consistent with the 72-hour temperature limit criteria. Westinghouse modeled the MCR, the protection and safety system cabinets, and the emergency switchgear rooms heatup rate. Therefore, Westinghouse should be able to establish a more specific time duration for the test (i.e., the time-frame which is necessary to validate the model). Without an objective validation of the heatup rate model, a complete 72-hour test is required. Additionally, Westinghouse should remove the exception to RG 1.68, Appendix A, Item 1.n.(14)(f) from Appendix 1A of the SSAR.

In its responses to the staff (NSD-NRC-98-5538 and NSD-NRC-98-5539) dated January 23, 1998, Westinghouse proposed to revise Section 14.2.9.1.6, items (c), (e), and (f) to include specific testing methods and time duration for the heatup and air quality of the MCR, and for heatup of the protection and safety monitoring system cabinet and the emergency switchgear rooms. Additionally, the exception to RG 1.68, Appendix A, Item 1.n.(14)(f) would be deleted from Appendix 1A of the SSAR.

The staff reviewed Revision 20 to Section 14.2.9.1.6 of the SSAR and finds that it adequately establishes measurable parameters for verifying that the protection and safety monitoring system cabinet and emergency switchgear rooms heatup at a rate consistent with the 72-hour temperature limit criteria. However, Westinghouse needs to include a reference to SSAR Section 6.4.3.2 under Section 14.2.9.1.6, items (f) and (g).

In its letter dated March 10, 1998, Westinghouse stated that Section 6.4.3.2 would be added as a reference to Section 14.2.9.1.6, items (f) and (g). In Revision 22 to the SSAR, Westinghouse incorporated a reference to the appropriate SSAR section under Section 14.2.9.1.6, items (f) and (g). Therefore, DSER Open Item 14.2.8-6 (RAI 260.26) is closed .

<u>Section 14.2.9.1.7, "Expansion, Vibration and Dynamic Effects Testing"</u>

- In the DSER, the staff found that consistent with the guidance in Section 3.9.2 of the SRP, the systems monitored during preoperational vibration and dynamic effects tests should include the following items:

  - American Society of Mechanical Engineers (ASME) Code, Class 1, 2, and 3 piping

  - high-energy piping systems inside seismic Category I structures

  - high-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level

  - seismic Category I portions of moderate-energy piping systems located outside the containment

  This was identified as DSER Open Item 3.9.2.1-1. In its response dated May 9, 1997, Westinghouse proposed that Section 14.2.9.1.7 of the SSAR be revised to include appropriate criteria for selection of systems to be monitored during preoperational vibration and dynamic effects tests. Westinghouse also stated that the Dynamic Response test (Section 14.2.10.4.18) and Thermal Expansion test (Section 14.2.10.4.25) are startup tests and are a repeat of the tests conducted on the systems that meet the criteria for inclusion in test 14.2.9.1.7 (conducted during preoperational testing) for those portions of these systems whose conditions during power operation are sufficiently different than during the testing conducted under Section 14.2.9.1.7. Also, Sections 14.2.10.4.18 and 14.2.10.4.25 would be revised to explicitly reference Section 14.2.9.1.7.

  The staff confirmed that Westinghouse revised preoperational test Section 14.2.9.1.7 to include the additional systems as requested. Startup test Sections 14.2.10.4.18, "Dynamic Response," and 14.2.10.4.25, "Thermal Expansion," were revised to require testing for those portions of systems that meet the test selection criteria for Section 14.2.9.1.7, but would not be tested during the preoperational test phase because system conditions during hot functional testing are not prototypical. Also, Sections 14.2.10.4.18 and 14.2.10.4.25 were revised to include a reference to Section 3.9.2 of the SSAR for appropriate performance criteria. On this basis, DSER Open Item 3.9.2.1-1 is closed.

- In the DSER, the staff found that to remain consistent with its response to RAI 210.57, Westinghouse should revise the appropriate Chapter 14 test abstract(s) to add a reference to Section 3.9.2.1.1 of the SSAR for the acceptance standard for the alternating stress intensity due to vibration . This was identified as DSER Confirmatory Item 3.9.2.1-4. Westinghouse responded in a letter dated August 13, 1996, that Section 14.2.9.1.7 would be revised to include reference to Section 3.9.2. Staff review determined that a reference to SSAR Section 3.9 is in the first item of Section 14.2.9.1.7, however, as committed to in the response to RAI 210.57, a specific reference to Section 3.9.2.1.1 for the acceptance standard for alternating stress intensity because of vibration should be added to Sections 14.2.9.1.7(b) and 14.2.10.4.18.

  In its response dated May 9, 1997, Westinghouse stated that a specific reference to Section 3.9.2.1.1 for the acceptance standard for alternating stress intensity because of vibration was previously included in Section 14.2.9.1.7(b) and would be added to Section 14.2.10.4.18.

  In Revision 13 to Section 14.2.10.4.18, Westinghouse added a reference to SSAR Section 3.9.2.1.1 as requested. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.9.2.1-4 is closed.

- In the DSER, the staff found that Westinghouse should provide procedures for testing water hammer occurrence in the feedwater piping. This was identified as DSER Open Item 10.4.7-1. Westinghouse responded in a letter dated August 13, 1996, that Section 14.2.9.1.7 would be revised to include testing to start/stop startup feedwater to the steam generators to verify that unacceptable water hammer does not occur. Staff review of this section determined that it does not provide sufficient information for testing water hammer occurrence. Additionally, Section 14.2.9.2.2, "Main and Startup Feedwater System," should be modified to include the following three items:

  (1) Perform feedwater system test and monitor that no effects because of water hammer are detected

  (2) Check for water hammer noise and vibration using suitable instrumentation

  (3) Visual inspection indicates that the integrity of feedwater piping, support, and function have not been violated

  In its response dated May 9, 1997, Westinghouse proposed to include the testing of dynamic events (e.g., water hammer) for all applicable systems in Section 14.2.9.1.7.c (including applicable portions of main and startup feedwater piping) to address the NRC comments.

The staff confirmed that Westinghouse revised Section 14.2.9.1.7 to specifically address the testing, monitoring and visual inspection for the effects of water hammer on the feedwater system as requested. However, these changes would be evaluated in conjunction with Westinghouse's pending response to RAI 410.263.

In its response to RAI 410.263 dated January 29, 1998, Westinghouse clarified how the testing described in Sections 14.2.9.1.7 and 14.2.10.4.18 meet the requirements of Branch Technical Position (BTP) ASB 10-2, "Design guidelines for Avoiding Water Hammers in Steam Generators." Westinghouse also proposed to modify Sections 10.4.7.4.3, 10.4.9.4.3, 14.2.9.1.7, and 14.2.10.4.18 to clarify that tests to monitor for the occurrence of water hammer are conducted having no detection of water hammer effects as the acceptance criteria. Based upon its review of this proposal, however, the staff concluded that Westinghouse was not proposing to subject the AP600 to the effects of a loss of normal feedwater transient as prescribed in BTP ASB 10-2.

In its letter dated March 10, 1998, Westinghouse stated that Section 14.2.9.2.2, "Main and Startup Feedwater System," would be revised to specify the testing that demonstrates the ability to restore normal steam generator water level from the low narrow range water level without causing unacceptable feedwater or steam generator water hammer, in accordance with BTP ASB 10-2.

The staff finds that the incorporation of the proposed testing in Revision 22 to Section 14.2.9.2.2 of the SSAR satisfactorily subjects the AP600 to the effects of a loss of normal feedwater transient as prescribed in BTP ASB 10-2. Therefore, RAI 410.263 and DSER Open Item 10.4.7-1 are closed .

In Revision 24 to Sections 14.2.9.1.7 and 14.2.10.4.18, slight changes were made to the dynamic response testing descriptions to clarify that suitable instrumentation would be used to monitor for the occurrence of water hammer and vibration, and that deflection measurements would be taken to confirm that no effects due to water hammer are detected. The staff finds these changes acceptable because appropriate dynamic and vibration verifications will be performed during the preoperational testing.

Section 14.2.9.1.9, "Reactor Vessel Internals Vibration Testing"

In the DSER, the staff found that Westinghouse should revise Section 14.2.8.1.77, "Reactor Internals and Reactor Coolant System Vibration Test" to reflect the response to RAI 210.58. This was identified as DSER Confirmatory Item 3.9.2.3-1.

Subsequently, Westinghouse replaced Section 14.2.8.1.77 with Section 14.2.9.1.9 in Revision 9 to the SSAR. The staff determined that this new section only addressed the prototype plant (first-plant-only) tests to comply with that portion of RG 1.20. There should be another section in Chapter 14 to provide the same commitment as that found in Section 3.9.2.4 for non-prototype plants.

In its response dated May 9, 1997, Westinghouse stated that Revision 10 to the SSAR satisfactorily addressed this issue. The staff found that Revision 13 to Section 14.2.9.1.9 is consistent with, and references, SSAR Section 3.9.2.4. Additional justification for this prototypical test is discussed in Section 14.2.5 of this report. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.9.2.3-1 is closed.

Section 14.2.9.2.8, "Fire Protection System Testing"

In Section 9.5.1.4.8, "Preoperational Testing," of the DSER, the staff found that additional information was required from Westinghouse to establish the acceptability of the fire protection system(s) preoperational test program in complying with Section C.4.e of BTP CMEB 9.5-1. This issue was identified as DSER Open Item 9.5.1.4-7.

In its response dated August 13, 1996, Westinghouse stated that Section 14.2.9.2.8 would be revised to state that the system operates as specified in Section 9.5.1 and in appropriate design specifications. These documents identify the applicable National Fire Protection Association (NFPA) standards for the testing of individual components in the fire protection system. Sections 14.2.9.2.19 and 14.2.9.4.13 would describe the testing of plant lighting and communication systems, respectively. The breathing apparatus provided at the plant and the use of this equipment would be identified by the COL applicant as part of the fire protection personnel training.

However, upon reviewing Revision 11 to Section 14.2.9.2.8 of the SSAR, the staff concluded that it needed to be modified to encompass testing of the AP600 fire protection system in an integrated manner (i.e., fire doors, fire dampers, smoke control systems, automatic fire detection, underground fire main, fire pumps, automatic suppression systems, electrical isolation devices for non-safety related equipment in opposite divisional fire areas, and trained fire brigade). This conclusion was documented in a March 20, 1997, NRC letter to Westinghouse.

In its response dated May 9, 1997, Westinghouse proposed that Section 14.2.9.2.8 of the SSAR be revised to incorporate the NRC comments with certain exceptions. Upon reviewing Revision 17 to Section 14.2.9.2.8, the staff concluded that it needed further modifications under General Test Method and Acceptance Criteria, items (a) and (c) to include pressure testing for the seismic standpipe water supply, and testing of the HVAC smoke control and exhaust systems section, respectively. This conclusion was documented in a January 7, 1998, NRC letter to Westinghouse.

In its response dated January 23, 1998, Westinghouse provided clarification on test provisions relevant to the hydrostatic pressure testing of the seismic standpipe water supply, and proposed to revise Section 14.2.9.2.8 to include verification of the proper operation of the portions of the HVAC systems used for smoke control and exhaust.

While the staff finds that Revision 20 to Section 14.2.9.2.8 of the SSAR adequately addresses the staff's concerns with the verification of the proper operation of the portions of the HVAC systems used for smoke control and exhaust, Westinghouse still needs to include pressure testing (i.e., a preoperational test that demonstrates system performance in accordance with design), for the seismic standpipe water supply as part of this section.

In its letter dated March 10, 1998, Westinghouse proposed to revise Section 14.2.9.2.8 to include verification that adequate water pressure is available to provide effective fire hose streams. The staff finds the proposed changes incorporated in Revision 22 to the SSAR acceptable, and therefore, DSER Open Item 9.5.1.4-7 is closed .

Section 14.2.10.3.6, "Natural Circulation (First Plant Only);" and Section 14.2.10.3.7, "Passive Residual Heat Removal Heat Exchanger (First Plant Only)"

In the DSER, the staff found that in startup test abstract 14.2.8.2.34, "Natural Circulation," Westinghouse had taken exception to RG 1.68, Appendix A, Item 4.t, for testing natural circulation as had been done for current PWR plants. The Westinghouse justification for this exception was that the performance of a natural circulation test was not necessary to demonstrate flow characteristics of the plant. The physical layout of the plant and key components (e.g., steam generators, pumps, piping, and reactor vessel) remain identical for each unit. Typical manufacturing and construction variations in these parameters would have no significant impact on the natural circulation flow. Since the design and layout is fixed between each AP600 plant, no changes in the natural circulation characteristics would occur. Other system flow and performance measurements taken during the hot functional and power ascension testing would provide assurances that the overall flow characteristics of the plant are equivalent to the reference plant. Therefore, demonstration of the natural circulation characteristics on the first AP600 plant is sufficient to validate the design characteristics. The natural circulation test is prototypical.

The staff found that this response is acceptable for startup test abstract 14.2.8.2.34, provided that the following criteria were met:

(1) Appropriate justification for this exception to RG 1.68, Appendix A, Item 4.t, is included in Appendix 1A of the SSAR, or Section 1.9.3 of the SSAR, as appropriate. (This justification should provide appropriate reference to Westinghouse's response for NUREG-0737, action item I.G.1, as described in the attachments to the letter from Westinghouse (E.P. Rahe) to the NRC (H.R. Denton), dated July 8, 1981).

(2) Westinghouse identifies this issue, in Section 14.2.9 of the SSAR (or its subsequent equivalent), as a COL action item, which will require the COL applicant referencing the AP600 design to perform the following requirements:

   (a) demonstrate that the physical layout and configuration of the proposed plant and key components (e.g., steam generators, pumps, piping, and reactor vessel) remain identical to the reference plant

   (b) validate the acceptance criteria, provided by Westinghouse, for the specific values or ranges of values for other system flow and performance measurements that are to be taken during the hot functional and power ascension testing to confirm that the overall flow characteristics of the proposed plant are equivalent to the reference plant

This was identified as DSER COL Action Item 14.2.8-1 and DSER Open Item 14.2.8-7. In its letter dated August 13, 1996, Westinghouse stated that Section 14.3 provides reference to CDM which commits the COL applicant to conduct the ITP. As part of that ITP, the COL applicant will verify the physical layout and configuration of the components, and component parameters important to the natural circulation of fluid in the RCS. These verifications will establish that AP600 plants subsequent to the first plant will achieve natural circulation flow similar to the flow demonstrated by testing in the first plant.

In its response to Westinghouse dated November 8, 1996, the staff clarified that while the CDM provides that the COL conduct certain testing to satisfy ITAAC requirements, the CDM does not commit the COL to conduct the ITP. In § 50.34, Appendix A to 10 CFR Part 50, and Section XI, "Test Control," of Appendix B to 10 CFR Part 50, the NRC requires that a test program be established to ensure that SSCs will perform satisfactorily in service.

In its letter dated December 6, 1996, Westinghouse stated that Section 3.4 of the AP600 CDM as submitted for staff review on November 8, 1996, contains a high-level commitment to perform an ITP by the COL applicant.

Justification for this exception [to RG 1.68, Appendix A, Item 4.t.] will be provided in Appendix 1A of the SSAR citing the appropriate reference and stating that for the AP600, natural circulation heat removal is not safety-related, as in current plants. This safety-related function is performed by the PRHR. Natural circulation heat removal via the PRHR will be tested for every plant. Therefore, Westinghouse has met the intent of the previous licensing commitments for natural circulation testing. This justification will be provided in Section 1.9.3 of the SSAR. Westinghouse's response to NUREG-0737, action item I.G.1 (issue of natural circulation testing for use as input into operator training) provided a proposal for low-power testing of existing and future Westinghouse PWRs in Attachment 4 to the letter from Westinghouse (E. P. Rahe) to the NRC (H. R. Denton) dated July 8, 1981. For the AP600, Westinghouse proposed the following similar five exceptions; noting that the appropriate tests are contained in the AP600 ITP:

(1) During hot functional testing, before fuel load, with the reactor coolant pumps not running and no onsite power available, the heat removal capability of the PRHR heat exchanger with natural circulation flow is verified (Section 14.2.9.3, item e).

(2) After fuel loading, but before criticality, with the reactor system at no-load operating temperature and pressure and all reactor coolant pumps (RCPs) operating, the depressurization rate is determined by de-energizing the heaters and pressure is further reduced through use of sprays (Section 14.2.10.1.19).

(3) After criticality is achieved and the plant is at ~ 3 percent power, the plant is placed in a natural circulation mode by tripping all reactor coolant pumps and observing the plant response (Section 14.2.10.3.6).

(4) A loss-of-offsite power test is performed with the plant at minimum power level supplying normal house loads. The turbine is tripped and the plant is placed in a stable condition using batteries and the diesel generator (Section 14.2.10.4.26).

(5) Data obtained from the natural circulation tests are provided for operator training on a plant simulator at the earliest opportunity.

The staff found Westinghouse's response on the NUREG-0737, action item I.G.1 issue acceptable. However, Westinghouse did not provide adequate justification or information in the SSAR for the staff to conclude that the ITAAC process would (1) demonstrate that the physical layout and configuration of the proposed plant and key components (e.g., steam generators, pumps, piping, and reactor vessel) remain within the specified tolerances of the reference plant; and (2) validate the acceptance criteria, provided by Westinghouse, for the specific values or ranges of values for other system flow and performance measurements that are to be taken during the hot functional and power ascension testing to confirm that the overall flow characteristics of the proposed plant remain within the accepted bounds of the reference plant.

Additionally, Section 1.9.4.2.1, Item I.G.1, indicates that the PRHR system fulfills the natural circulation heat removal function for the AP600. Therefore, Section 14.2.10.3.6, "Natural Circulation," should be modified to conduct natural circulation testing with the PRHR. Otherwise, testing conducted under Section 14.2.9.1.3, "Passive Core Cooling System Testing," should be conducted in <u>every plant</u> provided that such testing is performed at conditions necessary and sufficient to demonstrate design and operating system parameters commensurate with those that would have been demonstrated by testing at conditions described under Section 14.2.10.3.6. If the latter testing option is selected, Westinghouse would also need to modify its response to NUREG-0737, action item I.G.1, accordingly.

In its response dated May 9, 1997, Westinghouse proposed that a natural circulation heat removal test using the PRHR (Section 14.2.10.3.7) be included in the startup testing portion of the ITP to be performed in conjunction with the existing steam generator natural circulation test (Section 14.2.10.3.6). These tests would fulfill the requirements of NUREG-0737, action item I.G.1. Westinghouse would revise SSAR Section 1.9.4.2.1, item I.G.1 and the exception to RG 1.68, Appendix A.4.t, in SSAR Appendix 1A accordingly.

The staff confirmed that the new Section 14.2.10.3.7, "Passive Residual Heat Removal Heat Exchanger (First Plant Only)," provides for natural circulation testing of the PRHR system during startup testing of the initial plant. In addition, all plants are subjected to natural circulation testing of the PRHR during hot functional testing as described in Section 14.2.9.1.3.(e), (f), and (g). The staff finds that these testing provisions, in conjunction with the issues addressed in SSAR Section 1.9.4.2.1, item I.G.1, and the exception to RG 1.68, Appendix A.4.t in SSAR Appendix 1A, adequately address natural circulation testing for the AP600 design. On these bases , the staff also finds that a COL action item is no longer required on this subject. The staff finds this acceptable, and therefore, DSER COL Action Item 14.2.8-1 is dropped and DSER Open Item 14.2.8-7 is closed .

<u>Section 14.2.10.4.21, "100 Percent Load Rejection," and Section 14.2.10.4.24, "Plant Trip From 100 Percent Power"</u>

In the DSER, the staff found that startup test abstract 14.2.8.2.51, "100 Percent Load Rejection (First Plant Only)," should be modified in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants or to provide appropriate justification for this exception to RG 1.68, Appendix A, Item 5.n.n. This was identified as DSER Open Item 14.2.8-11.

In draft Revision 10 of the SSAR and its response to this open item dated August 13, 1996, Westinghouse stated that Section 14.2.10.4.21 specifies that the 100-percent load rejection test will be performed only on the first AP600 plant. This testing will provide measurements of the plant parameters including reactor power and primary and secondary pressures and temperatures that occur following this transient. Subsequent plants have similar equipment, control systems, and set points. Westinghouse concluded the above first-plant-only test meets the following criteria used to establish which testing will be performed only on the first AP600 plant (1) the performance parameter(s) to be measured is not provided by previous certification, qualification, or prototype testing; and (2) construction and installation inspections and other preoperational tests, performed on every plant, demonstrate that the performance parameter(s) does not change from plant to plant.

In its response dated November 8, 1996, the staff found that Westinghouse's justification for not demonstrating that the dynamic response of the plant is in accordance with design for the condition described in RG 1.68, Appendix A, Item 5.n.n., on all subsequent plants was unacceptable. The staff added that RG 1.68, Appendix A, Item 5.n.n., provides for the demonstration that the dynamic response of the plant is in accordance with design for the case of a full-load rejection transient with the plant's electrical distribution system aligned for normal full power operation, and that the turbine-generator is subjected to the maximum credible overspeed condition. The staff agrees that subsequent AP600 plants will have similar equipment, control systems, and associated set points; however, this test is not conducted solely to demonstrate that the performance parameters do not change from plant to plant. Rather, the purpose of this test is also to demonstrate that the integrated dynamic response of the as-built plant, including all associated systems and/or design features, conforms to the postulated plant response when subjected to this anticipated transient. Therefore, Section 14.2.10.4.21 needed to be modified to include the applicability of this testing to subsequent AP600 plants.

In its response dated December 6, 1996, Westinghouse stated that Section 14.2.10.4.21 would be performed on every plant. However, the staff found that in Revision 10 (and subsequently in Revision 11) to the SSAR, Section 14.2.10.4.21 (to be conducted on each plant) described an external load rejection test which would not subject the turbine to the maximum credible overspeed condition. RG 1.68, Appendix A, Item 5.n.n, specifies that a full load rejection test be conducted on each plant and that the test should subject the turbine to the maximum credible overspeed condition. Therefore, Section 14.2.10.4.21, or other test abstract as appropriate, should be modified to test the full (external and internal) load rejection capability of each plant.

In its response dated May 9, 1997, Westinghouse proposed that the Plant Trip From 100 Percent Power Test (Section 14.2.10.4.24) be modified to initiate the test by opening the main generator output breaker to achieve the maximum credible turbine overspeed condition.

The staff confirmed that Section 14.2.10.4.24 was modified to simulate a full-load rejection event by opening the generator output breakers, thereby, subjecting the turbine-generator to the maximum overspeed condition. This test meets the recommendations of RG 1.68, Appendix A, Section 5.l.l and 5.n.n and is acceptable. Therefore, DSER Open Item 14.2.8-11 is closed.

Section 14.2.10.4.24, "Plant Trip from 100% Power"

In the DSER, the staff found that Westinghouse should conduct a turbine trip test as described in Section 14.2.8.2.55, "Plant Trip from 100-Percent Power," on subsequent AP600 plants. This was identified as DSER Open Item 14.2.8-13.

Westinghouse responded in their August 13, 1996, letter that Section 14.2.10.4.24, "Plant Trip from 100% Power," will be conducted on each plant. Staff review of this section determined that this test was initiated by a reactor trip, not a turbine trip. Therefore, this test, or other test abstracts, should be modified to test the turbine trip response of all AP600 plants, or Appendix 1A in the SSAR should provide appropriate justification for this exception to RG 1.68, Appendix A, Item 5.l.l. Alternatively, consideration should be given to initiating the full-load rejection testing described in Section 14.2.10.4.21 by a main generator breaker trip so that it could be conducted in lieu of the plant trip test described in Section 14.2.10.4.24. This is acceptable in that RG 1.68, Appendix A, Item 5.l.l specifies that a turbine trip test can be combined with testing in accordance with Item 5.n.n if the test is initiated by a main generator breaker trip.

In its response dated May 9, 1997, Westinghouse indicated that its response to DSER Open Item 14.2.8-11 addressed this issue. The staff finds Westinghouse's response to DSER Open Item 14.2.8-11 acceptable as discussed above, and therefore, DSER Open Item 14.2.8-13 is closed.

Section 14.2.10.4.25, "Thermal Expansion"

In the DSER, the staff found that Westinghouse should revise Section 14.2.8.2.18 (startup thermal expansion testing) to reflect the response to RAI 210.55. This was identified as DSER Confirmatory Item 3.9.2.1-3. The staff's review determined that Revision 9 to Chapter 14 replaced Section 14.2.8.2.18 with Section 14.2.10.4.25, "Thermal Expansion," and that the response to RAI 210.55, did not appear in this new section. As committed to in the response to RAI 210.55, the test specifications for thermal expansion testing during preoperational and startup testing are in accordance with ASME OM Standard, Part 7. This commitment is in Section 14.2.9.1.7 for preoperational testing. The staff concluded that this same commitment should be added to Section 14.2.10.4.25 for startup testing.

In its response dated May 9, 1997 and subsequently in Revision 13 to the SSAR, Westinghouse revised Section 14.2.10.4.25 for startup testing to reference ASME OM Standard Part 7, as requested. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.9.2.1-3 is closed.

Section 14.2.10.4.26, "Loss of Offsite Power"

In the DSER, the staff found that Westinghouse should modify startup test abstract 14.2.8.2.41, "Loss of Offsite Power," in Appendix 1A of the SSAR to include applicability of this testing to subsequent AP600 plants, or provide appropriate justification for this exception to RG 1.68, Appendix A, Item 5.j.j. This was identified as DSER Open Item 14.2.8-9.

In its response dated August 13, 1996, Westinghouse stated that Chapter 14 was revised to delete testing which simulates a loss of off-site electrical power with the reactor core at power.

However, each aspect of a loss of off-site power transient is tested separately. These tests include the RCP flow coastdown test (Section 14.2.10.1.18), the diesel generator start, load testing (Section 14.2.9.2.17), the rod control system test (Section 14.2.10.1.11), and the rod drop time measurement test (Section 14.2.10.1.14).

In its response to Westinghouse dated November 8, 1996, the staff found the Westinghouse justification for deleting a test to demonstrate that the dynamic response of the plant is in accordance with the design to be unacceptable for the condition described in RG 1.68, Appendix A, Item 5.j.j. While results obtained when performing discrete systems tests at separate intervals may be indicative of the overall expected plant behavior during postulated operational transients, such testing is not a substitute for demonstrating that the actual dynamic plant response, including anticipated systems interactions, is in accordance with the design during a simulated or actual transient. The staff concluded that Westinghouse should revise Chapter 14 to reinstate testing for the condition described in RG 1.68, Appendix A, Item 5.j.j.

In its response dated December 6, 1996, Westinghouse stated that a test for conditions described in RG 1.68, Appendix A, Item 5.j.j., would be included in Chapter 14 as Section 14.2.10.4.26, "Loss of Offsite Power."

The staff found the reinstatement of testing for the condition described in RG 1.68, Appendix A, Item 5.j.j., under Section 14.2.10.4.26 of the SSAR acceptable. Therefore, DSER Open Item 14.2.8-9 is closed.

14.2.10 ITP Combined License Applicant Responsibilities

In SSAR Section 14.4, Westinghouse describes the COL applicant or licensee responsibilities required to perform the AP600 ITP.

14.2.10.1 Organization and Staffing

In Section 14.4.1 of the SSAR, Westinghouse states that the specific staff, staff responsibilities, authorities, and personnel qualifications for performing the AP600 ITP are the responsibility of the COL applicant. This test organization is responsible for planning, executing, and documenting the plant initial testing and related activities that occur between completion of the plant/system/component construction and commencement of commercial operation. Transfer and retention of experience and knowledge gained during ITP testing for the subsequent commercial operation of the plant is an objective of the test program.

14.2.10.2 Test Specifications and Procedures

In Section 14.4.2 of the SSAR, Westinghouse states that the COL applicant is responsible for providing test specifications and test procedures for the preoperational and startup tests, as identified in Section 14.2.3, for NRC review.

<u>DSER COL Action Item 14.2.2-1</u>

In the DSER, the staff found that the COL applicant should provide for staff review, the scoping document (e.g., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to Westinghouse's scope of design responsibility. This was identified as DSER COL Action Item 14.2.2-1.

In its response dated August 13, 1996, Westinghouse stated Section 14.4 would be revised to include a COL item to provide preoperational and startup test procedures containing test objectives and acceptance criteria for Westinghouse scope systems/components.

Upon reviewing Revision 19 to the SSAR, the staff found that COL issues previously addressed by DSER COL Action Item 14.2.2-1 were satisfactorily identified in Sections 14.2.3 and 14.4.2. These issues are currently addressed by COL Action Item 14.2.3-1. The staff finds this acceptable, and therefore, DSER COL Action Item 14.2.2-1 is dropped.

DSER COL Action Item 14.2.2-2

In the DSER, the staff found that the COL applicant should provide for staff review, the scoping document, and any related documents, which delineate plant operational conditions at which tests are to be conducted, testing methodologies to be utilized, specific data to be collected, and acceptable data reduction techniques to be utilized. This was identified as DSER COL Action Item 14.2.2-2.

In its response dated August 13, 1996, Westinghouse stated that Section 14.4, was revised to include a COL item to provide preoperational and startup test procedures to delineate test conditions, testing method, data to be collected, and data reduction techniques.

In its response to Westinghouse dated November 8, 1996, the staff found that Section 14.4.3, "Conduct of Test Program," states that the COL applicant is responsible for [developing] a startup manual as identified in Section 14.2.3, "Test Procedures." The staff added that Westinghouse addressed the specific issues identified in this open item. Upon reviewing Revision 19 of the SSAR, the staff found that COL issues previously addressed by DSER COL Action Item 14.2.2-2 were satisfactorily identified in Sections 14.2.3, 14.4.2 and 14.4.3. These issues are currently addressed by COL Action Item 14.2.3.1-1. The staff finds this acceptable, and therefore, DSER COL Action Item 14.2.2-2 is dropped.

DSER COL Action Item 14.2.2-4

In the DSER, the staff found that the COL applicant should provide for staff review, the approved preoperational test procedures (to be provided approximately 60 days before their intended use), and startup test procedures (to be provided approximately 60 days before fuel loading). This was identified as DSER COL Action Item 14.2.2-4.

In its response dated August 13, 1996, Westinghouse stated that Section 14.4 would be revised to include a COL item to provide preoperational and startup test procedures for all safety-related systems, and systems that perform defense-in-depth functions approximately 60 days before their intended use; and to provide approved startup test procedures 60 days before fuel loading.

In its letter to Westinghouse dated November 8, 1996, the staff concluded that although Section 14.2.3, "Test Procedures," as referenced in Section 14.4, appeared to address the COL item identified above, Section 14.2.3 appeared to also draw an unacceptable distinction between the availability (for NRC review) of preoperational test procedures for systems/components that perform safety-related functions, or of those that are non-safety-related but perform defense-in-depth functions (in the context of the AP600 design) versus those that do not perform either type of functions but which still satisfy RG 1.68, Regulatory Position (RP) C.1, "Criteria for Selection of Plant Features To Be Tested." RG 1.68 does not provide for this distinction and, therefore, all plant system and/or features identified in accordance with Section 14.2.1, "Summary of Test Program and Objectives," (once found acceptable) are subject to NRC review and approval. This exception to RG 1.68 is unacceptable and needed to be deleted.

Additionally, it is inappropriate for this section to specify that only safety-related ITP testing will be conducted in accordance with the quality assurance requirements of Section 17.5 of the SSAR. While RG 1.68 and Criterion XI, "Test Control," of Appendix B to 10 CFR Part 50 both recognize that not all SSCs have to be tested to the same stringent requirements, they both also hold that the test program must be conducted in a manner that establishes that SSCs will perform satisfactorily in service. Westinghouse's statement in this section implies that all testing of SSCs that do not perform safety-related functions will be performed in accordance with quality assurance requirements not currently described in Section 17.5 of the SSAR. The staff found that Westinghouse should delete this statement or, otherwise, supplement Section 17.5 of the SSAR to include a detailed description of the quality assurance program requirements that will govern testing of SSCs that do not perform safety-related functions.

In its response dated December 6, 1996, Westinghouse stated that RG 1.68 provides for a "graded approach" and that "While it is required that all SSCs important to safety be tested, it is not required that all of them be tested to the same stringent requirements." Westinghouse provided a comprehensive and systematic process to identify SSCs necessary to be included in the ITP to "provide reasonable assurance that the facility (AP600) can be operated without undue risk to the public." On the basis of a review of the AP600 SSCs, the SSCs requiring the highest level test commitments are the safety-related and defense-in-depth systems. The other systems included in the ITP are provided for completeness, and need not and do not require the same level of test commitment with regards to the ITP.

Westinghouse proposed to clarify the distinction between the most important and least important systems. Section 14.2.3 would be revised to state that (1) test specifications and test procedures for SSCs which perform in safety-related or defense-in-depth functions will be available for NRC review prior to performance of the test, and (2) all testing will be performed in accordance with the quality assurance requirements as specified in Section 17.5. Additionally, Section 17.5 of the SSAR would be revised to include testing within the quality assurance program developed by the COL applicant. These changes were incorporated in Revision 19 to Chapters 14 and 17 of the SSAR.

The staff completed its review of Revision 19 to Chapters 14 and 17 of the SSAR with respect to this COL item and finds the contents acceptable. The COL issues previously addressed by DSER COL Action Item 14.2.2-4 are currently addressed by COL Action Item 14.2.3-2. The staff finds this acceptable, and therefore, DSER COL Action Item 14.2.2-4 is dropped.

14.2.10.3 Conduct of Test Program

In Section 14.4.3 of the SSAR, Westinghouse states that the COL applicant is responsible for a startup administration manual (procedure) which contains the administrative procedures and requirements that govern the activities associated with the plant ITP, as identified in Section 14.2.3.

DSER Open Item 14.2.9-2

In the DSER, the staff found that the startup administrative manual, described in Revision 1 to Section 14.2.2.1 of the SSAR, should be identified in Section 14.2.2.1, "Conduct of Test Program," and in others as appropriate, as "COL License Information" (i.e., information to be supplied to the NRC by COL applicants referencing the AP600 design). This issue was identified as DSER Open Item 14.2.9-2. In addition, Westinghouse should include a description of the organizational units and any augmented organizations or other personnel that will manage, supervise, or execute any phase of the ITP consistent with the guidance in Section 14.2.2 of RG 1.70.

In its letter dated August 13, 1996, Westinghouse stated that Section 14.4 was revised to include a COL information item to provide a startup administrative manual that will delineate specific permissions required for the approval of test results and the permission to proceed to the next testing phase.

In its response to Westinghouse dated November 8, 1996, the staff found that Section 14.4.3, "Conduct of Test Program," states that the COL applicant is responsible for [developing] a startup manual as identified in Section 14.2.3, "Test Procedures." Upon review of Revision 19 to Chapter 14 of the SSAR with respect to the startup administrative manual, and the organizational units and any augmented organizations or other personnel that will manage, supervise, or execute any phase of the ITP, the staff found that these issues were satisfactorily addressed in Sections 14.2.3 and 14.4.3. The staff finds this acceptable, and therefore, DSER Open Item 14.2.9-2 is closed.

DSER Open Item 14.2.2.1-1

In the DSER, the staff found that the startup administrative manual, described in Section 14.2.2.1 of the SSAR should be identified in this section, and in others as appropriate, as "COL License Information" (i.e., information to be supplied to the NRC by COL applicants referencing the AP600 design). In addition, Westinghouse should include a description of the organizational units and any augmented organizations or other personnel that will manage, supervise, or execute any phase of the ITP in a manner consistent with the guidance in Section 14.2.2 of RG 1.70. Portions of the issues outlined above were previously identified by the staff in RAI 260.25. This was identified as DSER Open Item 14.2.2.1-1.

In its response dated August 13, 1996, Westinghouse stated that a description of the organizational units and any augmented organizations or other personnel that will manage, supervise, or execute any phase of the ITP is the responsibility of the COL applicant. Accordingly, Section 14.4 of the SSAR would be revised to provide this information.

Upon review, the staff found that Section 14.4.1 of the SSAR adequately addressed the COL applicant's responsibilities in developing and establishing the specific plant organization and staffing appropriate for the AP600 ITP consistent with Section 14.2.2 of RG 1.70. The staff finds this acceptable, and therefore, DSER Open Item 14.2.2.1-1 is closed.

<u>DSER COL Action Item 14.2.2.2-1 and DSER Open Item 14.2.2.2-1</u>

In the DSER, the staff found that the COL applicant should provide the startup administrative manual, which will delineate the review, evaluation, and approval of test results, for staff review. This was identified as DSER COL Action Item 14.2.2.2-1 and DSER Open Item 14.2.2.2-1.

In its response dated August 13, 1996, Westinghouse stated that Section 14.4 was revised to include a COL item to provide the startup administration manual which delineates the review, evaluation, and approval of test results.

Upon review of Revision 19 to the SSAR with respect to this COL issue, the staff found it satisfactorily identified in Sections 14.2.3 and 14.4.3. This issue is currently addressed by COL Action Item 14.2.3.2-1. The staff finds this acceptable, and therefore, DSER Open Item 14.2.2.2-1 is closed and DSER COL Action Item 14.2.2.2-1 is dropped.

14.2.10.4 Review and Evaluation of Test Results

In Section 14.4.4 of the SSAR, Westinghouse states that the COL applicant and holder is responsible for the review and evaluation of individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retests are performed, as required.

<u>DSER Open Item 14.2.2-2 and DSER COL Action Item 14.2.2-3</u>

In the DSER, the staff found that Section 14.2.2 of the SSAR should be revised to clarify that Westinghouse will provide the COL applicant with scoping documents (e.g., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to Westinghouse's scope of design responsibility. Such documents should also include, as appropriate, delineation of the following testing information:

- specific plant operational conditions under which the tests will be conducted
- testing methodologies to be used
- specific data to be collected
- acceptable data reduction techniques
- any reconciliation methods needed to account for test conditions, methods, or results (if testing is performed at conditions other than representative design operating conditions)

The staff also found that Section 14.2.2 (and/or Section 14.2.9, as appropriate) should include the following COL action items to be provided by the prospective COL applicant for staff review:

- The scoping document (e.g., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to Westinghouse's scope of design responsibility. This was identified as DSER COL Action Item 14.2.2-1.

- The scoping document, and any related documents, which delineate plant operational conditions at which tests are to be conducted, testing methodologies to be utilized, specific data to be collected, and acceptable data reduction techniques to be utilized. This was identified as DSER COL Action Item 14.2.2-2.

- The scoping document that delineates any reconciliation methods needed to account for test conditions, methods, or results if testing is performed at conditions other than representative of design operating conditions. This was identified as DSER COL Action Item 14.2.2-3.

- The approved preoperational test procedures (to be provided approximately 60 days before their intended use, and startup test procedures (to be provided approximately 60 days before fuel loading). This was identified as DSER COL Action Item 14.2.2-4.

These issues were previously identified by the staff in RAI 260.24. The addition of these COL action items to the SSAR was identified as DSER Open Item 14.2.2-2.

The resolution of COL Action Items 14.2.2-1, 14.2.2-2, and 14.2.2-4 are discussed in Section 14.2.2 of this report.

In its response to COL Action Item 14.2.2-3 dated August 13, 1996, Westinghouse stated that Section 14.4, was revised to include a COL item to provide preoperational and startup test procedures to delineate any reconciliation methods needed to account for test conditions, methods, or results if testing is performed at conditions not representative of design conditions.

Upon reviewing the Westinghouse response and Revision 19 to Sections 14.2.3 and 14.4.4 of the SSAR, the staff found that the COL issues previously addressed by DSER COL Action Item 14.2.2-3 were adequately identified. These COL issues are currently addressed by COL Action Items 14.2.3-1 and 14.2.3.2-1. The staff finds this acceptable, and therefore, DSER COL Action Item 14.2.2-3 is dropped.

The staff confirmed the satisfactory resolution of DSER COL Action Items 14.2.2-1 through 14.2.2-4 and the revisions to the SSAR. The staff finds this acceptable, and therefore DSER Open Item 14.2.2-2 is closed.

14.2.10.5 Interface Requirements

In Section 14.4.5 of the SSAR, Westinghouse states that the COL applicant is responsible for the testing of interfacing structures and systems that may be required. Test specifications and acceptance criteria are provided by the responsible design organizations as identified in Section 14.2.3.

In the DSER, the staff recommended that the content of Section 14.2.9 of the SSAR be revised to include "site-specific aspects of the plant," such as the following systems that may require testing "to satisfy certain AP600 interface requirements:"

- electrical switchyard equipment
- site security plan equipment
- personnel monitors and radiation survey instruments
- automatic dispatcher control system (if applicable)

This issue was included in RAI 260.32 and was identified as DSER Open Item 14.2.9-1. In its response dated August 13, 1996, Westinghouse stated that Section 14.3 provides reference to COL information items to verify site-specific aspects of the plant (that may require testing) are within the [design] certification envelope.

In its letter dated November 8, 1996, Westinghouse clarified that in its July 22, 1994, letter to the NRC, and in response to RAI 260.32, that it agreed to the staff's proposed revisions and recommendations. However, Revision 9 to the SSAR relocated such information to Section 14.3, "Certified Design Material." In its August 13, 1994, response to this open item, Westinghouse stated that Section 14.3 provides reference to COL information items to verify site specific aspects of the plant (that may require testing) are within the [design] certification envelope.

The staff requested that Westinghouse identify which subsection of Section 14.3, "Certified Design Material," designates "site-specific aspects of the plant" that may require testing by the COL applicant to satisfy certain AP600 interface requirements, such as those identified in RAI 260.32.

In its letter dated December 6, 1996, Westinghouse stated that interface requirements as defined by 10 CFR Part 52.47(a)(1)(vii) are discussed in Section 14.3. Westinghouse added that it was not necessary to provide a list of possible systems that may or may not require testing, as this determination would be made by the NRC at the time of the COL application.

The staff disagreed with Westinghouse's interpretation of §52.47(a)(1)(vii) and added that Westinghouse needed to specifically identify the structures and systems that are wholly or partially outside the design scope and specify the interface requirements for those systems, including testing to be performed by the COL applicant. Additionally, in an April 18, 1997, RAI to Westinghouse with regard to the AP600 ITAAC, the NRC staff stated, in part, that to meet 10 CFR 52.47(a)(vii) and (ix), Westinghouse needed to specifically identify the structures and systems that are wholly or partially outside the design scope and specify the interface requirements for those systems. This was identified as RAI 640.52. The staff concluded that Westinghouse should address DSER Open Item 14.2.9-1 in conjunction with RAI 640.52.

In its response dated May 9, 1997, Westinghouse stated the that Table 1.8-1 provides a summary of the AP600 plant interfaces with the remainder of the plant (outside of design certification). Westinghouse proposed that Section 14.3.4 be revised to state that verified testing of interfacing systems is the responsibility of the COL applicant. In addition, Section 14.4.5, Interface Documents, would be added to include a list of plant interfaces that may require testing to be performed by the COL applicant.

The staff finds that Revision 22 to Sections 14.3.4 and 14.4.5 of the SSAR adequately designates site-specific aspects of the plant which require testing by the COL applicant to satisfy AP600 interface requirements, as identified in RAI 640.52, and this is COL Action Item 14.2.10.5-1. Therefore, DSER Open Item 14.2.9-1 is closed.

14.2.10.6 First-Plant-Only and Three-Plant-Only Tests

In Section 14.4.6 of the SSAR, Westinghouse states that the COL applicant or licensee for the first plant and the first three plants will perform the tests listed in subsection 14.2.5. For subsequent plants, the COL applicant or licensee shall either perform the tests listed in subsection 14.2.5, or shall provide a justification that the results of the first-plant-only tests or first-three-plant tests are applicable to the subsequent plant. This COL issue is currently addressed by COL Action Item 14.2.5-1 and is acceptable.

14.2.11 Conclusions

The staff performed its review of the AP600 ITP in accordance with Section 14.2 of the SRP. Specifically, the staff reviewed the following eight areas relating to initial plant test programs:

(1) Summary of the test program scope and objectives

(2) Test procedures

(3) Conformance of the test program with RGs

(4) Utilization of reactor operating and testing experiences

(5) Use of plant operating and emergency procedures

(6) Initial fuel loading and initial criticality

(7) Test program schedule

(8) Individual test descriptions

The staff concludes that the AP600 ITP described in Chapter 14 of the SSAR encompasses the testing program requirements of 10 CFR 50.34(b)(6)(iii) and Criterion XI of Appendix B to 10 CFR Part 50, as described in the SRP, RG 1.68, and RG 1.70 and is, therefore, acceptable.

14.3 <u>Tier 1 Information</u>

14.3.1 Introduction

This section describes the staff's evaluation of the "AP600 Tier 1 Material." The Tier 1 material was derived from the AP600 SSAR. Specifically, this material (hereinafter Tier 1 information) includes the following:

- definitions and general provisions
- design descriptions

- inspections, tests, analyses, and acceptance criteria (ITAAC)
- significant site parameters
- significant interface requirements

Westinghouse intends to have this Tier 1 information certified by the NRC in a design certification rulemaking pursuant to Subpart B of 10 CFR Part 52. To be certified, the Tier 1 information must verify the complete scope of the design, and the amount of information in the Tier 1 design descriptions is proportional to the safety significance of the structures and systems in the standard design. The Tier 1 design descriptions are binding requirements for the life of a facility that references the certified design.

The staff reviewed the Tier 1 information in accordance with the guidance provided in the draft of Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification," of the NRC's Standard Review Plan (SRP, also known as NUREG-0800) and consistent with the requirements of 10 CFR 52.47, 10 CFR 52.97(b), and Appendices A and B to 10 CFR Part 52. The NRC prepared SRP Section 14.3 on the basis of experience gained in the review of Tier 1 information for the evolutionary designs (ABWR and System 80+) that were certified in 1997.

14.3.2 System-Based Design Descriptions and ITAAC

Westinghouse organized its AP600 Tier 1 information in a manner similar to that used for the evolutionary designs, as described in SRP Section 14.3. Therefore, the design descriptions and ITAAC for all of the systems in the AP600 standard design are set forth in Section 2.0, "System Based Design Descriptions and ITAAC." In Section 2.0 of Tier 1, Westinghouse provided a Tier 1 entry (subsection) for every system in the AP600 design, thereby meeting the requirement to verify the full scope of the standard design. In addition, although Westinghouse provided a Tier 1 entry for every system that is either fully or partially within the scope of the AP600 standard design, the amount of information in a given subsection is proportional to the safety significance of the particular system. The ITAAC portion of the Tier 1 information is used to verify that the as-built facility conforms with the approved design and applicable regulations.

One area where Westinghouse's approach for Section 2.0 differs from that described in SRP Section 14.3 is the ITAAC for the AP600 digital instrumentation and control (I&C) system. The SRP describes the use of a non system-based process with design acceptance criteria (DAC), in lieu of verifying a complete I&C design. By contrast, Westinghouse used system-based ITAAC for I&C, which focus on confirming that the licensee employs a high-quality design development process that incorporates disciplined specifications and implementation of design requirements. The inspections and tests in ITAAC 2.5, "Instrumentation and Control Systems," will be used to verify correct implementation of the I&C system design process (including quality), and to validate that the design ensures the required functionality. In addition to verifying design quality, the staff's review of the I&C system also focused on defense-in-depth and diversity, because of the concern regarding potential common-mode or common-cause failures resulting from software errors in redundant digital I&C safety system channels.

The Tier 1 information for the I&C system hardware and software development process includes a description of the design process to be followed for hardware and software development, the related design commitments, and the ITAAC to be performed to verify that the design is consistent with the commitments. The ITAAC for hardware and software verify the licensee's proposed design at various stages within the overall design process, as described in the SSAR and the referenced topical report. The Tier 1 information also includes criteria that describe the method used to develop plans and procedures that will guide the design process in order to ensure the quality of the digital I&C system throughout the various life cycle stages.

The AP600 Tier 1 information addresses a concern regarding common-mode software failure. Specifically, the diverse actuation system (DAS) performs safety system actuation functions comparable to those handled by the protection and safety monitoring system (PMS), but the two systems use different operating systems and programming languages. The manual initiation functions of the DAS are implemented in a manner that bypasses the control room multiplexers and the signal processing equipment of the DAS in order to ensure manual initiation capability in the event of a loss of the multiplexers.

In addition, the AP600 Tier 1 information addresses the process used to ensure that I&C equipment is able to function properly when subjected to an electromagnetic environment. Both the PMS and the DAS have electrical surge withstand capability (SWC), and are designed to withstand electromagnetic interference (EMI), radio frequency interference (RFI), and electrostatic discharge (ESD) conditions that are postulated to exist in the plant. The EMI immunity compliance plan is included in the design, installation, and testing of the PMS and DAS I&C equipment. Type tests, analyses, or a combination of both will be performed on the PMS and DAS equipment to ensure that it can withstand the SWC, EMI, RFI, and ESD conditions of the plant. In addition, the Tier 1 information addresses both seismic and environmental qualification of the PMS and DAS equipment.

The instrument set points are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation. These set points will be incorporated into the AP600 technical specifications.

14.3.3 Nonsystem-Based Design Descriptions and ITAAC

In Section 3.0 of the Tier 1 information, Westinghouse provided nonsystem-based design descriptions and ITAAC that apply to multiple systems or structures. Some of the non system-based Tier 1 information, such as the Design Reliability Assurance Program, is similar to that provided for the evolutionary designs. However, the non system-based Tier 1 information for piping and radiation protection in the evolutionary designs described a design process with DAC, because the applicants claimed insufficient as-built or as-procured information existed to complete the final design. By contrast, the following subsections describe the approach used for piping and radiation protection in the AP600 design.

Piping ITAAC

For the AP600, Westinghouse proposed to complete the piping design before design certification, thereby eliminating the need for piping DAC. Unlike the ABWR and System 80+ designs, the completion of the piping design for AP600 could be accomplished because the AP600 design uses fewer safety-related piping systems, and the design of the AP600 safety-related piping systems is less reliant on pumps and valves for which as-procured design information would not be available to complete the piping analyses. However, Westinghouse did not fully complete the AP600 piping design, as discussed below.

Westinghouse completed the piping stress analyses for Class 1, 2, and 3 large-bore piping systems (greater than or equal to 3 inch nominal pipe diameter), in accordance with the Boiler and Pressure Vessel Code promulgated by the American Society of Mechanical Engineers (ASME) (10 CFR 50.55a requires that safety-related piping systems meet the requirements for Class 1, 2, and 3 components in Section III of the ASME Code). Westinghouse provided ITAAC for safety-related piping systems to verify that the requirements of ASME Code Section III are followed in designing, constructing, and testing the AP600 piping, and in inspecting the AP600 pressure boundary welds. In addition, Westinghouse provided ITAAC for each safety-related system to verify that the piping design meets (1) functional capability requirements consistent with 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2, and (2) requirements for accommodating dynamic effects of postulated high-energy line breaks and for leak-before-break (LBB) evaluation, as required by GDC 4. The staff finds that the AP600 ITAAC will adequately verify that the constructed piping meets the requirements of ASME Code Section III and will ensure that the piping is built to adequately perform its intended function under appropriate combinations of design and accident conditions. On this basis, the staff finds that the piping-related ITAAC are acceptable.

In reviewing the level of detail and completeness of the AP600 piping design in accordance with 10 CFR 52.47(a)(2), the staff found that Westinghouse did not complete the stress analyses of the small-bore piping systems (less than 3-inch nominal pipe diameter), although most of these small-bore piping systems have been preliminarily routed and space has been allocated in the plant for them. The staff finds this approach consistent with the staff's approval of near-term operating license plants that were licensed under 10 CFR Part 50. Accordingly, Westinghouse proposed in Section 3.9.3 of its SSAR to have the COL applicant complete the small-bore piping analyses as part of the piping as-built reconciliation. The staff determined that small-bore piping designs were typically not included in the FSARs of recently licensed plants at the operating stage and the completion of these design details by the COL applicant does not raise any safety questions. On this basis, the staff finds that this commitment ensures the completion of the AP600 small-bore piping design and, therefore, is acceptable.

The staff found other areas of the AP600 piping design that Westinghouse did not fully complete. Specifically, Westinghouse did not finalize several large-bore piping stress analyses by using the final seismic input loadings. Similarly, Westinghouse was not planning to complete the fatigue analyses for the ASME Code Class 1 piping systems. Westinghouse also did not analyze the large-bore piping systems for LOCA dynamic loads, although it did complete the analyses for LOCA static loads. Lastly, Westinghouse's time-history analysis of the reactor coolant loop (RCL) piping did not consider a ±15% peak broadening to account for structural uncertainties. The following paragraphs discuss the resolution of these issues.

In the first issue, Westinghouse has been updating its seismic design building floor response spectra as the AP600 design progressed. Certain piping systems inside the reactor building initially were not analyzed to the final seismic spectra. Westinghouse found that all piping analyses had been completed using Revision 1 to the seismic design floor response spectra except for four piping systems, the reactor coolant system (RCS), passive core cooling system (PXS), steam generator system (SGS), and chemical and volume control system (CVS). In addition, Westinghouse revised the vertical spectra for the shield building roof to reflect effects of the water mass in the shield building tank and the addition of snow loads. This change (Revision 2) affected the Revision 1 seismic analysis of certain piping systems in the passive containment cooling system (PCS). As a result, there are five piping systems that were not analyzed to the latest seismic floor response spectra. To resolve this issue, Westinghouse committed (in Section 3.9.3 of the SSAR) that an as-built reconciliation of the piping design will be included in the final design report. The staff finds that the as-built reconciliation of the design includes completion of the seismic analyses utilizing the final floor response spectra described in Section 3.7.2 of the SSAR for the above five piping systems. An ITAAC item verifies the existence of a design report for the as-built piping systems and includes the reconciliation of the final piping seismic design. On this basis, the staff finds the ITAAC item provides an adequate verification of the completion of the piping seismic design using the final floor response spectra and, therefore, is acceptable.

Westinghouse did not complete the fatigue analyses of the ASME Code Class 1 piping systems, including the RCS, PXS, CVS, and normal residual heat removal system (RNS). For all of these piping systems, except the PXS, Westinghouse noted that the cyclic loadings considered in the fatigue analysis are similar to the loadings in existing operating plants. Design experience in the fatigue analyses of existing plants provides confidence that the cyclic loadings in the AP600 will not vary significantly to cause major piping design changes when the COL applicant completes the final fatigue analyses for the AP600 Class 1 piping systems. For the PXS, which is new and unique to the AP600 plant, Westinghouse performed the fatigue analyses of the most bounding PXS component affected by the cyclic loadings, and found that the fatigue evaluation met ASME Code requirements. In Section 3.9.3 of the SSAR, Westinghouse committed to include in the final piping design report an as-built reconciliation that will include evaluation of the ASME Code fatigue analysis for Class 1 piping. An ITAAC verifies the existence of a design report for the as-built piping systems and includes ASME Code fatigue analyses for Class 1 piping. On this basis, the staff finds that the ITAAC provides an adequate verification of the ASME Code Class 1 fatigue evaluation and, therefore, is acceptable.

Westinghouse did not analyze the large-bore piping systems for the effects of dynamic loads resulting from a loss-of-coolant accident (LOCA) or other pipe break dynamic loads (e.g., main steam or feedwater line break). For the AP600 design, the bounding pipe break dynamic load is caused by the mass-energy release resulting from a postulated double-ended guillotine break of the 16-inch feedwater piping at the steam generator nozzle. Westinghouse performed a specific limiting case analysis to demonstrate that the dynamic effects of this postulated pipe break would not cause major design changes to the AP600 piping systems. The specific piping chosen was the passive residual heat removal (PRHR) return line that is attached directly to the steam generator. This line is expected to have the most consequential effect as a result of a feedwater line break at the steam generator nozzle. Westinghouse's analysis showed that the maximum loadings in the PRHR line attributable to a main feedwater pipe break are approximately 10% of the corresponding loadings associated with a safe-shutdown earthquake (SSE) and would have no significant effect on piping layout, equipment nozzle loads, or support design. Consequently, Westinghouse committed in Section 3.9.3 of the SSAR to include in the final piping design report a final as-built reconciliation that would provide an evaluation by the COL applicant of the pipe break dynamic loads. An ITAAC verifies the existence of a design report for the as-built piping systems and includes an evaluation of pipe break dynamic loads that concludes that the acceptance criteria are met. On this basis, the staff finds this ITAAC provides adequate verification of the effects of pipe break dynamic loads on affected safety-related piping and, therefore, is acceptable.

Westinghouse did not complete the design of the RCL piping by using time-history seismic analysis that accounts for ±15% peak broadening effects. The impact of ±15% peak broadening is not expected to cause piping over stress or major design changes, but might increase the support and equipment nozzle loads. In Section 3.9.3 of the SSAR, Westinghouse committed to have the COL applicant complete this final confirmatory analysis. An ITAAC verifies the existence of a design report for the as-built piping systems and includes a confirmation of the RCL time history seismic analyses. On this basis, the staff finds this ITAAC provides adequate verification of the RCL piping using a time-history analysis with ±15% peak broadening and, therefore, is acceptable.

In the area of high- and moderate-energy line break analyses, Westinghouse completed its analyses of the 12 auxiliary lines and the reactor coolant system piping for which LBB is assumed. In Section 3.6.4.2 of the SSAR, Westinghouse committed to have the COL applicant verify the (1) piping characteristics, (2) material properties, and (3) bounding assumptions used in the LBB analyses. This verification will be documented in an LBB evaluation report. An ITAAC verifies the existence of an LBB evaluation report that concludes that the LBB acceptance criteria are met by the as-built piping and materials. On this basis, the staff finds that this ITAAC provides adequate verification of the LBB analyses and, therefore, is acceptable.

For those non-LBB high-energy lines, Westinghouse completed the high-energy line break analyses, except for pipe whip restraint design details which will be completed by the COL applicant. In Section 3.6.4.1 of the SSAR, Westinghouse committed to have the COL applicant complete the final pipe whip restraint design and address the as-built reconciliation of the pipe break hazards analyses. The staff finds that pipe whip restraint design details were typically not included in the FSARs of recently licensed plants at the operating stage and the completion of these design details by the COL applicant does not raise any safety questions. The analyses will be documented in an as-built pipe rupture hazards analysis report, as described in Section 3.6.2.5 of the SSAR. An ITAAC verifies the existence of a pipe break evaluation report for the as-built piping systems that concludes that the acceptance criteria are met. On this basis, the staff finds that this ITAAC provides adequate as-built verification of the high-energy line break analyses and, therefore, is acceptable.

Radiation Protection ITAAC

Westinghouse's evaluation of radiation source terms, plant shielding, and ventilation systems in the AP600 SSAR differs significantly from the evaluation conducted by ABB-Combustion Engineering, Inc. (ABB-CE), with regard to these same areas for the System 80+ design. The System 80+ SSAR did not contain a comprehensive tabulation of the contained and airborne material radioactive sources for the System 80+ design. Without a complete knowledge of the location and source strength for each of the contained sources in the System 80+ design, ABB-CE was not able to calculate the shielding thicknesses for the floors and walls of the System 80+ design necessary to establish designated radiation zones throughout the plant. Therefore, the System 80+ ITAAC contained DACs, which specified the design criteria for determining the contained source strengths for each area. Once the contained source strengths are determined, the DAC specified that the floors and walls should be designed to provide sufficient shielding to ensure that the maximum radiation levels did not exceed the radiation levels for the various radiation zones specified in the ITAAC.

The AP600 SSAR contains a comprehensive tabulation of the contained radiation sources for the AP600 design. Using these source terms, Westinghouse determined the floor and wall shielding thicknesses for the Nuclear Island and Annex Buildings necessary to ensure that radiation levels for the various designated plant radiation zones would not be exceeded. These floor and wall shielding thicknesses are included as part of the Tier 1 information in Section 3.3, "Buildings." Since the AP600 SSAR contains a listing of the expected AP600 contained radiation source terms, there is no need for source term DACs for the AP600 design. Section 3.3 also provides ITAAC that will be used to verify these floor and wall shielding thicknesses. In reviewing Section 3.3, the staff used the review guidance and acceptance criteria in SRP Sections 12.2, "Radiation Sources," and 12.3-12.4, "Radiation Protection Design Features."

The AP600 SSAR also contains a listing of expected airborne radioactivity concentrations for various areas within the plant. With this information, Westinghouse is able to determine the necessary ventilation flow rates in various portions of the plant. These ventilation flow rates are included as part of the Tier 1 information in Section 2.7.5, "Radiologically Controlled Area Ventilation System." Because the AP600 SSAR contains information on airborne radioactivity concentrations, there is no need for a DAC on this subject for the AP600 design. In reviewing Section 2.7.5, the staff used the review guidance and acceptance criteria in SRP Sections 12.2 and 12.3-12.4.

The AP600 SSAR contains a tabulation of the airborne radiation monitors that will be part of the AP600 design, along with their planned locations and nominal ranges. The ITAAC in Section 3.5, "Radiation Monitoring," verifies that these monitors will be part of the AP600 design. In reviewing Section 3.5, the staff used the review guidance and acceptance criteria in SRP Section 12.3.

14.3.4 Other Tier 1 Information

Westinghouse provided other Tier 1 information, such as definitions, general provisions, interface requirements, and site parameters. This information is similar to that used for the evolutionary designs, except for the "Basic Configuration" inspection and the interface requirements. No significant interface requirements were identified for the AP600 design because of design features in the standard plant.

Both evolutionary designs used "verifications for basic configuration for systems." This verification process consisted of an inspection of the system functional arrangement in its final as-built condition at the plant site and included four other elements (e.g., dynamic and environmental qualification). Westinghouse adopted a "functional arrangement" inspection but assigned verification of the other four elements to individual ITAAC, as appropriate. For the evolutionary and AP600 designs, this functional arrangement inspection verifies that the as-built facility is in conformance with the approved design and applicable regulations by using as-built drawings, design documentation, and *in situ* plant walkdowns.

14.3.5 Conclusions

The NRC staff has completed its review of the Tier 1 information for the AP600 design in accordance with the guidance provided in draft SRP Section 14.3. On the basis of satisfactory resolution of the issues identified in previous revisions of the Tier 1 information, the staff's review of Revision 7, and its review of the selection criteria and processes for development of the Tier 1 information in SSAR Section 14.3, the staff concludes that the top-level design features and performance characteristics of the structures and systems for the AP600 design are appropriately described in Tier 1 and the Tier 1 information is acceptable. Furthermore, the Tier 1 design descriptions can be adequately verified by ITAAC. Therefore, the staff concludes that the ITAAC are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility referencing the AP600 certified design can be constructed and operated in conformity with the design certification and applicable regulations.

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1512

Volume 1

2. TITLE AND SUBTITLE

Final Safety Evaluation Report Related to Certification of the AP600 Standard Design
Docket No. 52-003

Volume 1
Chapters 1-14

3. DATE REPORT PUBLISHED

MONTH	YEAR
September	1998

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Safety Evaluation Report

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Reactor Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

Project No. 676, Docket No. 52-003

11. ABSTRACT *(200 words or less)*

This final safety evaluation report (FSER) documents the technical review of the AP600 standard nuclear reactor design by the U.S. Nuclear Regulatory Commission (NRC). The application for the AP600 design was submitted on June 26, 1992 by Westinghouse Electric Corporation in accordance with Subpart B, Standard Design Certifications," of Part 52 of Title 10 of the Code of Federal Regulations (10 CFR Part 52), and Appendix O, "Standardization of Design: Staff Review of Standard Designs." The AP600 nuclear reactor design is a pressurized water reactor with a power rating of 1933 MWt with an electrical output of at least 600 MWe. The AP600 design contains many features that are not found in current operating reactor designs. For example, a variety of engineering and operational improvements provide additional safety margins and address the Commission's severe accident, safety goal, and standardization policy statements. The most significant improvement to the design is the use of safety systems that use passive means (such as gravity, natural circulation, condensation and evaporation, and stored energy) for accident prevention and mitigation. These passive safety systems perform safety injection, residual heat removal, and containment cooling functions. Unique features of the AP600 design include an enhanced reactor core design, larger reactor vessel, larger pressurizer, an in-containment refueling water storage tank, automatic depressurization system, revised main control room design with a digital microprocessor-based instrumentation and control system, hermetically-sealed canned reactor coolant pump motors mounted to the steam generator, and increased battery capacity. In addition, the facility is designed for a 60-year life, and employs modular construction techniques in its design.

On the basis of its evaluation and independent analyses, the NRC staff concludes that Westinghouse's application for design certification meets the requirements of Subpart B of 10 CFR Part 52 that are applicable and technically relevant to the AP600 standard design.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Advanced Reactor, Advanced Light Water Reactor (ALWR)
Final Safety Evaluation Report (FSER)
Passive Plant
Standardization
Combined License (COL)
Final Design Approval
Design Certification
Regulatory Treatment of Non-Safety-Related System (RTNSS)

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL STANDARD MAIL
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67