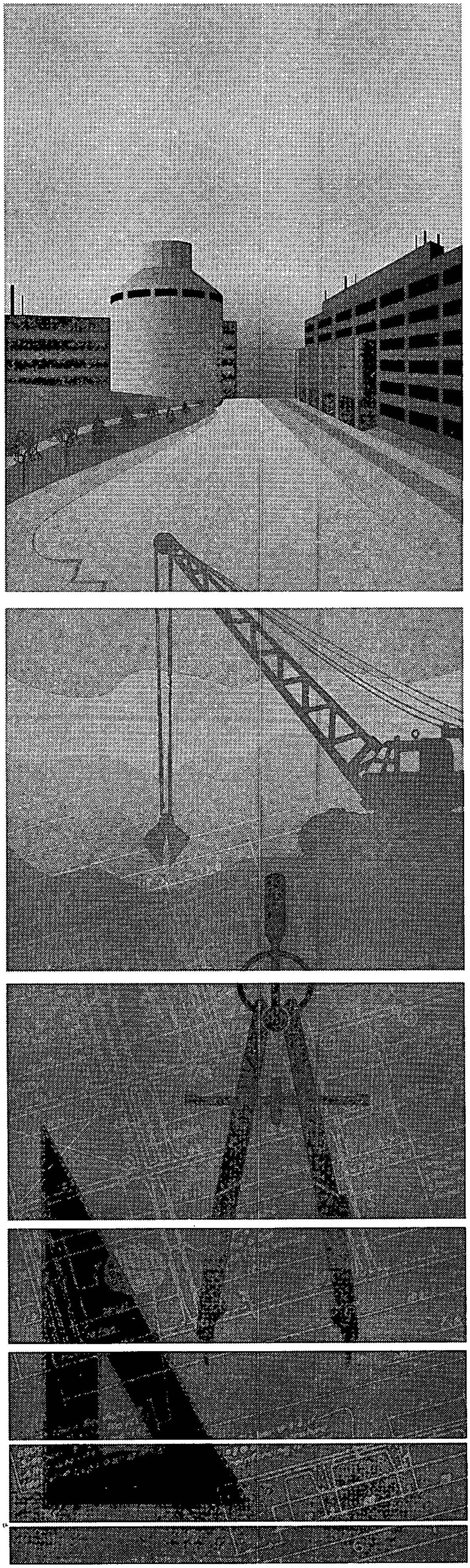

Final Safety Evaluation Report

Related to Certification of the AP600 Standard Design

Volume 1

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

September 1998

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NUREG-1512
Vol. 1

Final Safety Evaluation Report Related to Certification of the AP600 Standard Design Docket No. 52-003

Chapters 1-14

Manuscript Completed: August 1998
Date Published: September 1998

Division of Reactor Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

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. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR 52.53 is provided in Appendix G.

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1.1 Introduction

On June 26, 1992, Westinghouse Electric Corporation (hereinafter referred to as Westinghouse or the applicant) tendered its application for certification of the AP600 standard nuclear reactor design with the U.S. Nuclear Regulatory Commission (the NRC or staff). Westinghouse submitted this application 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 application included the AP600 Standard Safety Analysis Report (SSAR) and the AP600 Probabilistic Risk Assessment (PRA) Report. In addition, on December 15, 1992, to support the design certification application, Westinghouse submitted a comparison of the design to the Electric Power Research Institute's (EPRI's) "Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD);" a discussion of how operating experience was incorporated into the design; a discussion of severe-accident mitigation design alternatives; and the AP600 Tier 1 information, which includes the inspections, tests, analyses, and acceptance criteria (ITAAC). The NRC formally accepted the application as a docketed application for design certification (Docket No. 52-003) on December 31, 1992. Information submitted before that date can be found under Project No. 676.

Westinghouse originally submitted the AP600 SSAR, which describes the design of the facility, on June 26, 1992. Subsequently, the applicant supplemented the information in the SSAR by providing revisions to that document. Westinghouse submitted the most recent SSAR Revision 25 to the Commission on August 19, 1998. Similarly, Westinghouse originally submitted the PRA on June 26, 1992. It has been revised through Revision 13 (submitted by letter dated August 13, 1998). Westinghouse also submitted the AP600 Tier 1 information by letter dated December 15, 1992. Subsequently, Westinghouse submitted revisions (through Revision 7, dated August 13, 1998) incorporating the resolutions of NRC comments. In addition, throughout the course of the review, the NRC staff requested that Westinghouse submit additional information to clarify the description of the AP600 design. Some of Westinghouse's responses to these requests for additional information (RAIs) are discussed throughout this report. Appendix E of this report provides a listing of the issuance and response dates. Note that Westinghouse did not update all of the RAIs because changes to the design or SSAR documentation made it unnecessary or impractical. Unless otherwise noted, this FSER presents the results of the staff's review of information submitted to the NRC through August 24, 1998. The SSAR, PRA, Tier 1 information, and all other pertinent information and materials are available for public inspection at the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20555.

This Final Safety Evaluation Report (FSER) summarizes the staff's safety review of the AP600 design against the requirements of Subpart B of 10 CFR Part 52 and delineates the scope of the technical details considered in evaluating the proposed design. In addition, this FSER is to document the resolution of the open and confirmatory items identified in the draft safety evaluation report (DSER) for the AP600 design, which was issued on November 30, 1994; its supplement, which was issued on May 3, 1996; and the Advance FSER, which was issued on May 6, 1998. A copy of the report by the Advisory Committee on Reactor Safeguards required by 10 CFR 52.53 is provided in Appendix G.

Sections 1.2 and 1.3 of this report summarize the AP600 design. Section 1.4 identifies agents and contractors. Section 1.5 provides a discussion of the principal matters that were the subjects of the staff's review.

1.1.1 Metrication

This report conforms with the Commission's Policy Statement on metrication. Therefore, all measures are expressed as metric units, followed by English units in parentheses. However, in a staff requirements memorandum (SRM) dated January 19, 1994, the Commission exempted the AP600 application from the Commission's policy on metrication. Therefore, the SSAR and the remainder of the application use only inch-pound units.

1.1.2 Proprietary Information

This FSER contains several references to Westinghouse reports. Some of these Westinghouse reports contain information that Westinghouse requested be held exempt from public disclosure, as provided by 10 CFR 2.790. For each such report, Westinghouse provided a nonproprietary version, similar in content except for the omission of the proprietary information. The staff predicated its findings on the proprietary versions of these documents, so the staff refers only to the proprietary versions throughout this report.

In the DSER, the staff stated that much of the material identified as proprietary in the SSAR and PRA did not appear to comply with the criteria of 10 CFR 2.790 for determining the proprietary status of information. In addition, some of this information had previously been released to the public. Consequently, in the DSER, the staff stated that Westinghouse should reevaluate the SSAR and PRA to ensure that the materials requested to be withheld from public disclosure meet the criteria set forth in 10 CFR 2.790. In addition, the staff stated that much of the material that had been submitted in support of the application (besides the SSAR and PRA) did not appear to comply with the criteria of 10 CFR 2.790. The staff concluded that the amount of proprietary information in the design certification should be minimized. This was identified as DSER Open Item 1.1-1 (and later, AFSER Open Item 1.1-1)

In subsequent revisions to the SSAR and PRA, Westinghouse removed all of the proprietary information presented directly in those documents. However, the SSAR does incorporate by reference proprietary versions of Westinghouse reports. In addition, Westinghouse significantly reduced the amount of proprietary information submitted on the AP600 docket. The staff has completed its review of Westinghouse's responses to the staff's determinations on Westinghouse's withholding requests, and concludes that they are acceptable. Therefore, DSER and AFSER Open Items 1.1-1 are closed. However, Westinghouse needs to formally update the non-proprietary versions of some of the proprietary documents to reflect the results of the staff's review. The staff concludes that these revisions are not required to be submitted before the FSER and FDA are issued because the staff relied on the proprietary versions to make their safety determinations. In addition, the staff agrees with the changes to these documents that have been proposed by Westinghouse. Formal submittal of these revisions to non-proprietary versions of these documents is Confirmatory Item 1.1.2-1.

1.1.3 Comparison to the EPRI ALWR Utility Requirements Document

In SRMs dated December 15, 1989 and March 5, 1991, the Commission directed the staff to evaluate any differences between the vendor designs and the EPRI ALWR URD. In its submittal dated December 15, 1992, Westinghouse stated that its goal is "to resolve all differences between the AP600 design and Volume III of the ALWR URD by the time the FDA [final design approval] is issued." Westinghouse identified the nonconformances between the application and Revision 3 of the EPRI URD. Since that time, EPRI has issued several revisions to the URD for passive plants. Because both the AP600 design and the EPRI URD have changed, the staff requested that Westinghouse perform a comprehensive re-evaluation of the current AP600 design to identify and explain the differences from the revised EPRI ALWR URD. This was identified as DSER Open Item 1.1-2 (and later, AFSER Open Item 1.1-2).

Since that time, the staff completed its review of the EPRI URD through Revisions 4 and 5 (depending on the chapter), and issued the results of its evaluation in NUREG-1242, "NRC Review of EPRI's ALWR URD – Passive Plant Designs," dated August 1994. In SSAR Section 1.2.1.1, Westinghouse states that it was a principal participant in the development of the EPRI-sponsored URD, and continues to be involved with EPRI on changes to that document. Therefore, Westinghouse states that the AP600 design remains consistent with the EPRI URD.

In a letter dated January 6, 1998, Westinghouse stated that

> ... an active program is being maintained to assure the AP600 design is and remains consistent with the URD, currently at Revision 7, and the URD is updated to reflect developing technology in advanced plant designs.
>
> During the AP600 detailed design phase, findings have been identified to document features of the plant that are different from the URD requirements. The majority of the occasions where the AP600 did not comply with the URD have been resolved by changing the AP600 design. Design changes have been incorporated using the AP600 procedure for design control and are reflected in the SSAR.
>
> Findings have also been closed by working with EPRI to revise Volume III of the URD. Some modifications included in Revision 7 of the URD may not have been reviewed by the NRC. Consequently, several features of AP600 may respond to different requirements than those reviewed by NRC. Several URD changes were made to accommodate NRC-identified concerns while others were made to facilitate AP600 design solutions for passive safety systems.

In the letter dated January 6, 1998, Westinghouse provided a discussion of those differences that related to design certification issues, and that were not initiated by NRC concerns. In a letter dated August 13, 1998, Westinghouse submitted an update to the January 6, 1998 letter. In that letter, Westinghouse identified changes to the plant design that resulted from resolution of NRC comments and questions that resulted in additional minor variances with the EPRI URD. The staff has completed its review of the AP600 design, and has found these variances to be acceptable. Therefore, DSER and AFSER Open Items 1.1-2 are closed.

1.1.4 Combined License Applicants Referencing the AP600 Design

Applicants who reference the AP600 standard design in the future for specific facilities will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a combined license (COL) that references the AP600 design, the staff will evaluate, for each plant-specific application that references the AP600 design, the technical competence of the COL applicant and its contractors to manage, design, construct, and operate a nuclear power plant. The plant-specific applicants will also be required to satisfy the requirements of Subpart C of 10 CFR Part 52, and any requirements resulting from the staff's review of this standard design. Westinghouse has identified matters to be addressed by plant-specific applicants as "Combined License Information" throughout the SSAR. The staff has also identified such matters, referred to as "COL Action Items," throughout this report. A cross-reference of the COL Action Items and the Combined License Information is provided in Appendix F of this report.

1.1.5 Additional Information

Appendix A to this report provides a chronology of the principal actions, submittals, and amendments related to the processing of the application. Appendix B provides a list of references identified in this report. Appendix C provides a list containing definitions of the acronyms and abbreviations used throughout this report. Appendix D lists the principal technical reviewers who evaluated the AP600 design. Appendix E provides an index of the staff's RAIs and the Westinghouse responses. Appendix F provides a cross-reference of the Combined License Information in the SSAR and COL Action Items discussed in this report. Appendix G contains a copy of the letter received from the ACRS providing the results of its review of the AP600 design.

The NRC's licensing project managers assigned to the AP600 standard design review are Mr. Thomas Kenyon, Mr. William Huffman, Mr. Dino Scaletti, Mr. Joseph Sebrosky, and Mr. Jerry Wilson. They may be reached by calling (301) 415-7000, or by writing to the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

1.2 General Design Description

1.2.1 Scope of the AP600 Design

The requirement that governs the scope of the AP600 design is 10 CFR 52.47(b)(2). Westinghouse chose the option in 10 CFR 52.47(b)(2)(i)(A)(4), which requires that an applicant for certification provide a complete design scope, except for site-specific elements. Therefore, the scope of the AP600 design must include all of the plant structures, systems, and components (SSCs) that can affect the safe operation of the plant, except for its site-specific elements. Westinghouse described the AP600 standard design scope in SSAR Section 1.8, including the site-specific elements that are either partially or wholly outside of the standard design scope. Westinghouse also described interface requirements (see SSAR Table 1.8-1) and representative conceptual designs, as required by 10 CFR 52.47(a)(1)(vii) and 10 CFR 52.47(a)(1)(ix), respectively. The staff has verified the standard design scope and resulting interface requirements in its review of the AP600 ITAAC, and finds it to be acceptable.

1.2.2 Summary of the AP600 Design

The AP600 design has a nuclear steam supply system (NSSS) power rating of 1933 MWt with an electrical output of at least 600 MWe. The plant is designed to accept a step load increase or decrease of 10 percent between 25 and 100 percent power without reactor trip or steam dump system actuation, provided that the rated power level is not exceeded. In Section 1.2 of the SSAR, Westinghouse also indicates that the plant is designed to accept a 100-percent load rejection from full power to house loads without a reactor trip or operation of the pressurizer or steam generator safety valves. The goal for the overall plant availability is projected to be greater than 90 percent, considering all forced and planned outages, with a rate of less than one unplanned reactor trip per year. Westinghouse states that the plant has a design objective of 60 years without a planned replacement of the reactor vessel. However, the design does provide for replaceability of other major components, including the steam generators. The following is a general description of the AP600 design. A detailed description of each system is provided in the section of this report that discusses the given system.

1.2.2.1 Reactor Coolant System Design

The AP600 reactor coolant system (RCS) is designed to effectively remove or enable removal of heat from the reactor during all modes of operation, including shutdown and accident conditions. The system consists of two heat transfer loops, each with the following components:

- a steam generator
- two reactor coolant pumps
- a single hot leg
- two cold legs

In addition, the system includes a pressurizer, interconnecting piping, valves, and the instrumentation necessary for operational control and safeguards actuation. All of the system equipment is located within the reactor containment. Figure 1.2-1 shows a diagram of the AP600 RCS.

The reactor system pressure is controlled by operation of the pressurizer. Overpressure protection for the RCS is provided by the spring-loaded safety valves installed on the pressurizer. These safety valves discharge to the containment atmosphere. The valves for the first three stages of automatic depressurization are also mounted on the pressurizer. These valves discharge steam through spargers to the in-containment refueling water storage tank (IRWST) of the passive core cooling system (PXS). The discharged steam is condensed and cooled by mixing with water in the tank.

The following auxiliary systems interface with the RCS:

- chemical and volume control system (CVS)
- component cooling water system (CWS)
- liquid radwaste system (WLS)
- primary sampling system (PSS)
- passive core cooling system (PXS)
- spent fuel pool cooling system (SFPCS)
- steam generator system (SGS)

1.2.2.2 Reactor Design

An AP600 fuel assembly consists of 264 fuel rods in a 17x17 square array. The fuel grids consist of an egg-crate arrangement of interlocked straps that maintain lateral spacing between the rods. The fuel rods consist of enriched uranium, in the form of cylindrical pellets of uranium dioxide, contained in Zircaloy-4 or ZIRLO tubing. The tubing is plugged and seal welded at the ends to encapsulate the fuel. An axial blanket comprised of fuel pellets with reduced enrichment may be placed at each end of the enriched fuel pellet stack to reduce the neutron leakage and improve fuel utilization. A second type of fuel rod may be used to varying degrees within some fuel assemblies. These rods use an integral fuel burnable absorber (IFBA) containing a thin boride coating on the surface of the fuel pellets. The boride-coated fuel pellets provide burnable absorber integral to the fuel.

Westinghouse states that the reactor core is designed for a 24-month fuel cycle, and that the core is capable of operating on an 18-month fuel cycle. A three-region core design with a radial neutron reflector is maintained for all projected fuel cycles. The reactor core is located low in the vessel to minimize core temperature during a postulated loss-of-coolant accident (LOCA). The core is designed to have a moderator temperature coefficient that is non-positive over the entire fuel cycle and at any power level, with the reactor coolant at the normal operating temperature. The design objective for the reactor vessel is 60 years with a fast neutron fluence that is less than 2E+19 neutrons/centimeter^2 (E > 1.0 MeV).

No vessel penetrations exist below the top of the core because the AP600 does not use bottom-mounted in-core instrumentation. In addition, the design employs an integrated head package that consists of the following components:

- control rod drive mechanisms
- integrated head cooling fans
- instrument columns
- insulation
- seismic support
- package lift rig

A permanent, welded-seal ring is used to provide the seal between the vessel flange and the refueling cavity floor.

1.2.2.3 Steam Generator Design

The AP600 design uses the Model Delta 75 steam generator, which employs thermally-treated nickel-chromium-iron Alloy 690 tubes and a steam separator area sludge trap with clean-out provisions. The channel head is designed to directly attach the two reactor coolant pumps (RCPs), and to allow both manual and robotic access for inspection, plugging, sleeving, and nozzle dam placement operations.

1.2.2.4 Reactor Coolant Pump Design

The four AP600 RCPs are hermetically sealed canned pumps. Two RCPs are attached directly to the steam generator channel head with the motor located below the channel head to simplify the loop piping and eliminate fuel uncovery during postulated small-break LOCA scenarios. Each RCP includes sufficient internal rotating inertia to permit coastdown to avoid departure from nucleate boiling (DNB) following a postulated loss-of-coolant flow accident. Each pump impeller and diffuser vane is ground and polished to minimize radioactive crud deposition and maximize pump efficiency.

1.2.2.5 Pressurizer and Loop Arrangement

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. One spray nozzle and two nozzles for connecting the safety and depressurization valve inlet headers are located in the top head. Electrical heaters are installed through the bottom head. The piping layout for the AP600 is designed to provide adequate thermal expansion flexibility, assuming a fixed vessel and a free-floating steam generator/RCP support system. The pressurizer itself is designed such that, with design spray flow rates, the power-operated relief valve function is neither required nor provided.

1.2.2.6 Steam and Power Conversion System Design

<u>Turbine Generator</u>

The AP600 turbine generator design consists of a double-flow, high-pressure cylinder (high-pressure turbine) and two double-flow, low-pressure cylinders (low-pressure turbines) that exhaust to the condenser. It is a four-flow, tandem-compound, 1800-rpm machine. The turbine system includes the following components:

- stop, control, and intercept valves directly attached to the turbine and in the steam flow path
- crossover and crossunder piping between the turbine cylinders and the moisture separator reheaters

The high-pressure turbine has extraction connections for two stages of feedwater heating, and its exhaust provides steam for one stage of feedwater heating in the deaerator. The low-pressure turbines have extraction connections for four stages of feedwater heating.

The moisture separator reheater is located between the high-pressure turbine exhaust and the low-pressure turbine inlet. The moisture separator reheater is an integral component of the turbine system, which extracts moisture from the steam, and reheats the steam to improve turbine system performance.

The turbine is oriented in a manner that minimizes potential interactions between turbine missiles and safety-related structures and components.

Main Steam System

The main steam system (MSS) is designed to supply steam from the steam generators to the high-pressure turbine over a range of flows and pressures for the entire plant operating range. The MSS is also designed to dissipate the heat generated by the NSSS to the condenser through the steam dump valves, or to the atmosphere through power-operated atmospheric relief valves or spring-loaded main steam safety valves, when either the turbine generator or the condenser is not available.

Main Feedwater and Condensate System

The main feedwater system is designed to supply the steam generators with adequate feedwater during all modes of plant operation, including transient conditions. The condensate system is designed to condense and collect steam from the low-pressure turbines and turbine bypass systems, and then transfer this condensate from the main condenser to the deaerator. Westinghouse states that the main feedwater and condensate systems are designed for increased availability and improved dissolved oxygen control.

1.2.2.7 Engineered Safeguards Systems Design

The engineered safeguards systems consist of the following systems and components. Figure 1.2-2 shows some of the passive safety features, including the containment, the passive containment cooling system, and the passive core cooling system.

- The containment vessel is a free-standing cylindrical steel vessel with ellipsoidal upper and lower heads. Its engineered safety feature (ESF) function is to contain the release of radioactivity following a postulated design-basis accident (DBA). It also functions as the safety-related ultimate heat sink by transferring the heat associated with accident sources to the surrounding environment.

- The passive containment cooling system (PCS) consists of the following components:

    - a water storage tank that is incorporated in the shield building structure above the containment

    - an air baffle that is located between the steel containment vessel and the concrete shield building

    - air inlet and exhaust paths that are incorporated in the shield building structure

    - a water distribution system

- an ancillary water storage tank and two recirculation pumps for onsite storage of additional PCS cooling water

On actuation, the PCS delivers water to the top, external surface of the steel containment shell, which forms a film of water over the dome and side walls of the containment structure. Air is induced to flow over the containment as it is heated, causing a chimney effect. This air flow and cooling water evaporation removes the heat generated within the containment and expels it to the outside air. Westinghouse states that the containment is designed to remain below the design pressure following a DBA without replenishing the PCS water after the initial 7-day inventory is exhausted. Westinghouse states that the containment pressure will not exceed its ultimate pressure during a core melt scenario if only air cooling capability is available. Figure 1.2-3 shows the passive containment cooling system.

- The major function of the containment isolation system is to provide containment isolation to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary. This prevents or limits the escape of fission products that may result from postulated accidents. The containment isolation provisions are designed so that fluid lines that penetrate the primary containment boundary are isolated in the event of a postulated DBA. The system consists of the piping, valves, and actuators that isolate the containment.

- The containment hydrogen control system controls the hydrogen concentration in the containment following a postulated DBA. It consists of the hydrogen monitoring system, passive autocatalytic hydrogen recombiners, and hydrogen ignitors.

- The PXS comprises the following components:

  - two core makeup tanks
  - two accumulators
  - the IRWST
  - a passive residual heat removal (PRHR) heat exchanger
  - pH adjustment baskets
  - associated piping and valves

  The automatic depressurization system (ADS), which is part of the RCS, also provides important passive core cooling functions. This system provides emergency core cooling following a postulated DBA by providing (1) RCS makeup water and boration when the normal makeup supply is lost or insufficient, (2) safety injection to the RCS to provide adequate core cooling during a postulated DBA, and (3) core decay heat removal during transients and accidents. Figure 1.2-4 shows the safety injection systems.

- The main control room (MCR) emergency habitability system comprises a set of emergency air storage tanks connected to a main and an alternate air delivery line. Components common to both lines include a manual isolation valve, a pressure regulating valve, and a flow metering orifice. This system is designed to provide the ventilation and pressurization requirements to maintain a habitable environment in the MCR for 72 hours following any DBA.

In Section 1.2.1.4.1 of the SSAR, Westinghouse states that the engineered safeguards systems are designed to mitigate the consequences of DBAs with a single failure. With the exception of the MCR emergency habitability system, the passive safety systems are designed to cool the RCS from normal operating temperatures to safe-shutdown conditions. In addition, all of these systems are designed to maximize the use of natural driving forces (such as pressurized nitrogen or air, gravity flow, and natural circulation flow). They do not rely on active components (such as pumps, fans, or diesel generators) to function. They do, however, use valves to initially align the safety systems when activated. In addition, the safety systems are designed to function without safety-related support systems (such as alternating current; component cooling water; service water; or heating, ventilation, and air conditioning).

Non-Safety-Related Systems Designs

Westinghouse states that the non-safety-related systems used in the AP600 are not relied on to provide safety functions required to mitigate DBAs. Westinghouse further states that the non-safety-related systems required for normal plant operation provide high plant availability. In addition, the designer states that these systems have appropriate redundancy, are powered by on-site standby power supplies, and have sufficient capacity to prevent automatic passive safety system actuation following anticipated Condition II events.

In Section 1.2.1.4.2 of the SSAR, Westinghouse states that

- The reactor coolant system makeup capability design is sufficient for reactor coolant leaks up to three-eighths of an inch in diameter.

- Steam generator feedwater capability from the startup feedwater system is designed to provide sufficient flow for a loss of main feedwater event.

- The normal containment sump pumps (part of the radioactive waste drain system) are designed to assist in the recovery from leakage to the containment sump.

Although Westinghouse states that the non-safety-related systems used in the AP600 are not relied on to provide safety functions required to mitigate DBAs, the active systems provide defense-in-depth (DID) capabilities for reactor coolant makeup and decay heat removal. These active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. In addition, one of the principal design requirements of EPRI's ALWR URD is that passive systems should be able to perform their safety functions, independent of operator action or offsite support, for 72 hours after an initiating event. After 72 hours, non-safety-related active systems may be required to replenish the passive systems or to perform core and containment heat removal duties directly. As specified in the URD, the following active systems may be needed to provide DID capabilities:

- the chemical and volume control system, which provides reactor coolant makeup

- the reactor shutdown cooling system and backup feedwater system for decay heat removal

- the fuel pool cooling and cleanup system for spent fuel decay heat removal

- the associated systems and structures to support these functions, including the non-safety-related standby diesel generators

The AP600 also includes other active systems, such as the heating, ventilation, and air conditioning (HVAC) system, that are designated as non-safety-related. These systems remove heat from the instrumentation and control (I&C) cabinet rooms and the MCR, and prevent excessive accumulation of radioactive materials in the MCR to limit challenges to the passive safety capabilities for these functions.

In existing plants and the evolutionary LWR designs, many of these active systems are safety-related systems. As stated above, the active systems are not classified as safety-related in the AP600 design. The passive systems involve inherent phenomenological uncertainties. For example, low differential pressures may not create sufficient force to fully open a stuck check valve operating under natural circulation or gravity injection. This differs from the emergency core cooling systems in current operating plants, in which pressure developed by pumps can overcome stuck valves. These uncertainties have been evaluated through Westinghouse's component performance tests, and separate effects and integral system tests over a range of transient and accident conditions in accordance with 10 CFR 52.47(b)(2)(i)(A). Realistic analyses of the performance of AP600 passive systems and components further reduce the uncertainties associated with the passive systems. The staff's evaluation of the testing program is provided in Section 21 of this report.

However, the residual uncertainties associated with passive safety system performance increase the importance of active non-safety-related systems in providing DID functions to the passive systems. The staff does not require that these active systems meet all of the criteria imposed on safety-related systems, but expects a high level of confidence that active systems which have a significant safety role will be available when challenged. As discussed in SECY-94-084, the staff and EPRI developed a process for maintaining appropriate regulatory oversight of these active systems in the passive ALWR designs. In an SRM dated June 30, 1994, the Commission approved the recommendations made in SECY-94-084 concerning the issue of regulatory treatment of non-safety-related systems (RTNSS); however, the Commission directed the staff to accommodate Westinghouse's comments on the subject in its letter dated May 24, 1994.

In the DSER, the staff stated that the regulatory oversight of the active non-safety-related systems was 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," which was under staff review. The general issue regarding RTNSS was identified as DSER Open Item 1.2.2.7-1. The staff's evaluation of the RTNSS issue is discussed in Chapter 22 of this report. In that chapter, the staff finds Westinghouse's treatment of the RTNSS issue to be acceptable and, therefore, DSER Open Item 1.2.2.7-1 is closed.

1.2.2.8 Instrumentation and Control System and Electrical System Designs

Control and Protection Systems Designs

The AP600 control and protection systems are significantly different from instrumentation and control (I&C) systems in operating reactor designs. In particular, the AP600 employs digital microprocessor-based I&C systems with multiplexed data links, instead of the analog electronics, relay logic, and hard-wired systems currently used in operating plants. In Section 1.2.1.5.1 of the SSAR, Westinghouse states that the design of the control and protection system ensures that a single failure in the I&C system will not result in a reactor trip or ESF actuation during normal operation. The design is intended to reduce the potential for a reactor trip and for a safeguards actuation because of failures in the reactor control or protection systems as compared to current operating plants.

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

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

The PMS (1) monitors plant processes using a variety of sensors; (2) performs calculations, comparisons, and logic functions based on those sensor inputs; and (3) actuates a variety of equipment. The PMS is also used to operate safety-related systems and components.

The PLS (1) controls and coordinates the plant during start-up, ascent to power, power operation, and shutdown conditions; (2) integrates the automatic and manual control of the reactor, reactor coolant, and various reactor support processes for required normal and off-normal conditions; (3) controls the non-safety-related decay heat removal systems during shutdown; and (4) permits the operator to control plant components from the MCR or remote shutdown workstation.

The OCS includes the complete operational scope of the MCR, remote shutdown workstation, technical support center, local control stations, and the emergency operations facility.

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 (1) provides a backup to the PMS for some specific diverse automatic actuation, (2) provides diverse indications and controls to assist in operator manual actions, and (3) is a DID system that is also designed to provide essential protection functions in the event of a postulated common-mode failure of the PMS.

Alternating and Direct Current Power Designs

All safety-related electrical power is provided from the Class 1E direct current (dc) power system. The AP600 does not include a separate safety-related ac power system. Safety-related dc power is provided to support reactor trip and engineered safeguards actuation. Batteries are sized to provide the necessary dc power and uninterruptable ac power for items such as the protection and safety monitoring system actuation, control room functions including habitability, dc-powered valves in the passive safety systems, and containment isolation.

Main Control Room (MCR) Design

The MCR controls the plant during normal and anticipated transients as well as DBAs. It includes indications and controls that are capable of monitoring and controlling the plant safety systems and the non-safety-related control systems. The MCR contains the safety-related instrumentation and controls to allow the operator to achieve and maintain safe shutdown following any DBA.

The MCR is serviced by redundant non-safety-related power sources and HVAC systems during normal operation. In the event that either the normal power source or HVAC system becomes unavailable, there are passive systems (batteries and compressed air) available that Westinghouse states will support MCR operation for up to 3 days. The safety-related power sources and passive cooling system are designed to provide a habitable environment for the operating staff assuming that no ac power is available. After 3 days, Westinghouse states that it will be possible to continue operation with the control room cooled and ventilated with the natural circulation of outside air.

The operating staff can transfer control from the MCR to the remote shutdown workstation should they be required to leave the MCR. The remote shutdown workstation contains the safety-related indications and controls that allow an operator to achieve and maintain safe shutdown of the plant following an event when the MCR is unavailable.

1.2.2.9 Plant Arrangement

The AP600 plant is arranged with the following principal building structures:

- the nuclear island
- the turbine building
- the annex building
- the diesel generator building
- the radwaste building

The nuclear island consists of the following:

- a free-standing steel containment building
- a concrete shield building
- an auxiliary building

Figure 1.2-5 shows the AP600 building layout.

The containment building is the containment vessel that primarily contains the RCS, engineered safeguards systems, and portions of the non-safety-related support systems. The shield building comprises the structure and annulus area that surrounds the containment building. The containment and shield buildings are an integral part of the passive containment cooling system.

The auxiliary building protects and separates all of the seismic Category I mechanical and electrical equipment located outside the containment building. It contains the MCR, I&C systems, dc system, fuel handling area, mechanical equipment areas, containment penetration areas, and main steam and feedwater isolation valve compartment.

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

The annex building allows ingress and egress from the nuclear island. The building includes the health physics area, the non-Class 1E ac and dc electric power systems, the ancillary diesel generators and their fuel supply, other electrical equipment, the technical support center, hot machine shop, and personnel facilities (shower and locker rooms). No safety-related equipment is located in the annex buildings.

The diesel generator building houses two diesel generators and their associated HVAC equipment. No safety-related equipment is located in the diesel generator building.

The radwaste building contains facilities for segregated storage of various categories of waste before processing, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building.

1.3 Comparison With Similar Facility Designs

The AP600 standard 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. Some of the design differences between the AP600 and other 600 MWe operating reactors are listed below:

- Reactor and RCS Design
  - low power density
  - moderator displacement rods
  - neutron reflector
  - reduced neutron flux to reactor vessel
  - reduced hot leg temperature
  - larger reactor vessel
  - ring-forged vessel
  - 60 percent larger pressurizer
  - surge line that minimizes thermal stratification
- Passive Safety Features
  - passive safety injection system for residual heat removal (RHR)
  - passive reactor coolant inventory control (using gravity feed or nitrogen injection)
  - natural circulation decay heat removal
  - core makeup tanks
  - IRWST inside containment
  - passive containment cooling (using condensation, gravity, and evaporation)
  - automatic depressurization system
- Other Design Features
  - digital microprocessor-based I&C systems with multiplexed data links
  - revised MCR design
  - modified steam generator channel head
  - hermetically-sealed canned RCP motors mounted to the steam generator
  - increased containment volume
  - battery-operated safety-related valves
  - increased battery capacity
  - modular construction
  - designed for 60-year life
- Design Simplifications
  - no pumps for emergency core cooling
  - fewer pumps
  - no Class 1E diesel generators
  - fewer welds in the reactor coolant piping
  - less piping [greater than 5 cm (2 in)]
  - fewer valves
  - less control cable

Table 1.3-1 of the SSAR provides a detailed comparison of the principal design features of the AP600 standard design, and those of earlier reactor designs.

1.4 Identification of Agents and Contractors

Westinghouse is the principal AP600 designer. The following organizations provided the principal subcontracting services for the design of the AP600:

- Avondale Industries, Incorporated
- Bechtel North American Power Corporation
- Burns & Roe Company
- Chicago Bridge & Iron Services, Incorporated
- MK-Ferguson Company
- Southern Electric International

Westinghouse received additional support from the following organizations:

- Badan Tenaga Atom Nasional (BATAN) of Indonesia
- BPPT of Indonesia
- European Nuclear Energy Association (ENEA) of Italy
- Ente Nazionale per l'Enerfia Elettrica (ENEL) of Italy
- FIAT of Italy
- Empresa Nacional de Ingeniería y TTecnología, S.A. (INITEC) of Spain
- Oregon State University
- PLN of Indonesia
- Società Progettazione Reattori Nucleari, SpA (SOPREN)/ANSALDO of Italy
- UNESA of Spain
- University of Western Ontario of Canada
- UTE of Spain

1.5 Summary of Principal Review Matters

The procedure for certifying a design is conducted in accordance with the requirements of Subpart B of 10 CFR Part 52, and carried out in two stages. The technical review stage is initiated by an application filed in accordance with the requirements of 10 CFR 52.45, continues with reviews by the NRC staff and the ACRS, and concludes with the issuance of an FSER that discusses the staff's conclusions related to the acceptability of the design. The administrative review stage begins with the publication of a Federal Register notice that initiates rulemaking, in accordance with 10 CFR 52.51, and provides a proposed standard design certification rule. The rulemaking will be conducted by the Commission and also provides an opportunity for an informal hearing before an Atomic Safety and Licensing Board. The Board may also request authority from the Commission to use additional procedures, such as direct and cross examination by the parties, or may request that the Commission convene a formal hearing under Subpart G of 10 CFR Part 2 on specific and substantial disputes of fact, necessary for the Commission's decision, that cannot be resolved with sufficient accuracy except in a formal hearing. The rulemaking culminates with the denial or issuance of a design certification rule.

The staff performed its technical review of Westinghouse's application for certification of the AP600 standard design in accordance with Commission guidance and the requirements of

10 CFR Sections 52.47, 52.48, and 52.53. This FSER describes the results of the staff's technical review.

The staff evaluated the technical information required by 10 CFR 52.47(a)(1)(i) in accordance with the standard review plan (NUREG-0800); that evaluation is the subject of this report. Unless otherwise noted, the staff reviewed the application using the newest codes and standards that have been endorsed by the NRC. The staff's evaluation of the technically relevant unresolved safety issues (USIs), generic safety issues (GSIs), and Three Mile Island requirements (Sections 52.47(a)(1)(ii) and (iv)) is discussed in Chapter 20 of this report. The evaluation of the site parameters required by 10 CFR 52.47(a)(1)(iii) is discussed in Chapter 2, the effects of soil amplification at shallow soil sites is discussed in Section 3.7.1, and the nuclear island foundation mat design is discussed in Section 3.8.5 of this report. The staff's evaluation of the design-specific PRA (10 CFR 52.47(a)(1)(v)) is discussed in Section 19.1. The evaluation of the ITAAC required by 10 CFR 52.47(a)(1)(vi) is discussed in Section 14.3 of this report.

10 CFR 52.47(a)(1)(vii) requires that the application for design certification contain interface requirements that must be met by the non-certified portion (the site-specific elements) of a standard plant design, such as the ultimate heat sink. The AP600 design scope is described in SSAR Section 1.8, and SSAR Table 1.8-1 provides an index of the interface requirements for the AP600 design. In the DSER, the staff stated that it was evaluating the acceptability of these proposed interface requirements. This was identified as DSER Open Item 1.9-1. Interface requirements and representative conceptual designs (10 CFR 52.47(a)(1)(vii) through (ix)) are evaluated throughout selected chapters of this report. The staff has completed its evaluation of the interface requirements, and finds them acceptable. The staff also implemented the Commission's Severe Accident Policy Statement, dated August 8, 1985, and the Commission's SRMs on SECYs-90-016, 93-087, 94-084, 95-132, 96-128, and 97-044, in its resolution of severe accident issues. The staff's evaluation of severe accident issues is discussed in Section 19.2 of this report.

The regulations in 10 CFR 52.47(a)(2) describe the level of design information needed to certify a standard design. Determining the acceptable level of design detail necessary for the staff to make its safety findings was one of the most challenging aspects of the staff's review. The December 4, 1990 SRM for SECY-90-377 set forth the Commission's position on the level of design information required for a certification application, and the staff followed that guidance in preparing this document. The staff also followed the guidance of SECY-92-053. To allow for technology improvements and as-procured equipment characteristics, the staff predicated its safety determinations on the use of design acceptance criteria (DAC) for certain technical areas. The DAC are part of the Tier 1 information proposed for the AP600 design. The staff's evaluation of the Tier 1 information, including DAC and ITAAC, is in Section 14.3 of this report.

As part of its technical review, the staff issued numerous RAIs to gain sufficient bases for its safety findings, thereby meeting the requirement in Section 52.47(a)(3) to advise Westinghouse on the staff's requirements for additional technical information. Appendix E provides an index of Westinghouse's responses to these RAIs. Note that Westinghouse did not update all of the RAIs because changes to the design or SSAR documentation made it unnecessary or impractical.

Section 1.2.1 of this report discusses the scope of the design to be certified. Because of the unique nature of the AP600 design, Westinghouse implemented an extensive testing program to provide data on the passive safeguards systems. This data validates the safety analysis methods and computer codes, and provides information to assess the design margins in the passive safety system performance. The staff's evaluation of the testing program required to meet 10 CFR 52.47(b)(2) is discussed in Chapter 21 of this report. Because the AP600 is designed as a single unit (that is, no safety systems will be shared at a multi-unit site), GDC 5 of Appendix A to 10 CFR Part 50 and 10 CFR 52.47(b)(3) do not apply to this design. Any applicant wishing to construct multiple units at a single site will be required to address these regulations in its application.

The staff used the safety standards set forth in 10 CFR 52.48 for its technical review of the AP600 standard design. In addition to these safety standards, the staff followed Commission guidance provided in the SRMs for all applicable Commission papers, including those referenced throughout this report. In particular, SECYs-93-087, 94-084, and 95-132 identified staff positions generic to passive light-water reactor (LWR) design certification policy issues, and SECYs-96-128, 97-044, and 98-161 identified staff positions on issues specific to the AP600 design. In SRMs dated July 21, 1993; June 30, 1994; June 28, 1995; January 15, 1997; and June 30, 1997, the Commission provided its guidance on these matters as they pertain to passive plant designs.

If Westinghouse decides to proceed with certification of the AP600 design, it must prepare a design control document (DCD). The DCD will consist of the Tier 1 and Tier 2 information. Applications that reference the certified AP600 design will be required to conform with the DCD in accordance with the certification rule. Submittal of the DCD is FSER Confirmatory Item 1.5-1. The DCD will be available for public inspection at the NRC's Public Document Room when the proposed rule for design certification is published in the Federal Register.

While the staff was developing the Advance FSER, in certain limited cases, its evaluation was based on expected information being provided in final documentation that was submitted after the report was issued. Submittal of this information was AFSER Confirmatory Item 1.5-2. As discussed throughout this report, Westinghouse has formally submitted this information, and therefore, AFSER Confirmatory Item 1.5-2 is closed.

1.6 Index of Exemptions

In accordance with 10 CFR 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing Westinghouse's application for design certification of the AP600 design. In the DSER, the staff stated that Westinghouse had not yet identified the exemptions that will be required for the AP600 design. This was identified as DSER Open Item 1.8-1. In the AFSER, the staff stated that Westinghouse's request for exemptions required revision. This was AFSER Open Item 1.6-1.

In a letter dated June 19, 1998, Westinghouse submitted an updated list of exemption requests. The staff has determined that the updated list is acceptable, and therefore, AFSER Open Item 1.6-1 (formerly DSER Open Item 1.8-1) is resolved. The exemptions are discussed in the following sections of this report.

Section	Description of Exemption
8.2.4	Exemption from GDC 17 (Appendix A to 10 CFR Part 50)
15.2.9	Exemption from 10 CFR 50.62
15.3	Exemption from 10 CFR 50.34(a)(1)(i)
15.3	Exemption from GDC 19 (Appendix A to 10 CFR Part 50)
18.8.2.3	Exemption from 10 CFR 50.34(f)(2)(iv) (Provide a Safety Parameter Display System console)
20.6	Exemptions from using TID-14844

1.7 Index of Tier 2* Information

The NRC staff has determined that changes to or departures from information in the SSAR (i.e., design commitments) that are proposed by an applicant or licensee who references the certified AP600 design will require NRC approval before implementation of the change in accordance with the design certification rule. This information will be referred to as Tier 2* in the proposed design certification rule.

The staff has identified the Tier 2* information pertaining to the AP600 design, which closes DSER Open Item 1.11-1. Designation of the Tier 2* information is now FSER Confirmatory Item 1.7-1 pending documentation in the DCD. The Tier 2* information is discussed in the sections of this report listed below.

Section	Description
3.7.2.3	Nuclear island structural dimensions
3.8.2.2	ASME Code, Section III
3.8.2.5	ASME Code case N-284
3.8.3.5	ANSI/AISC N690 and ACI 349
3.8.3.7	Design summary of critical sections
3.8.4.6	ANSI/AISC N690, ACI 349, and ACI 318
3.8.4.7	Design summary of critical sections
3.8.5.3.1	ACI 318 and ACI 349
3.8.5.8	Design summary of critical sections
3.9.6.2	Design qualification and testing requirements for MOVs and POVs
3.10	Seismic qualification methods and standards
4.1	Reactor core criteria and design requirements
5.2.1.1	ASME Code, Section III
7.1.4	Instrumentation and control process and standards
9.5.1.1.b	Fire areas
18.14	Human factors engineering program

1.8 COL Action Items

COL applicants and licensees who reference the certified AP600 standard design will be required to satisfy the requirements and commitments in the design control document (DCD), which is the controlling document used in the certification of the AP600 design. Also, certain commitments are identified in the AP600 SSAR as "Combined License Information Items," and in this report as "COL Action Items." These COL action items relate to programs, procedures, and issues that are outside of the scope of the certified design review. These COL action items do not establish requirements; rather, they identify an acceptable set of information for inclusion in a plant-specific SAR. An applicant for a COL must address each of these items in its application. It may deviate from or omit these items, provided that the deviation or omission is identified and justified in the plant-specific SAR.

Westinghouse included a summary of COL action items in SSAR Table 1.8-2, and provided an explanation of the items in the applicable sections of the SSAR. The staff identified a number of COL action items that resulted from its review throughout this report. In the AFSER, the staff stated that it would ensure that the COL action items identified by the staff are consistent with those identified by Westinghouse. This was AFSER Confirmatory Item 1.8-1. In a July 31, 1998 letter, Westinghouse submitted a cross-reference of the COL Action Items. The information from this letter is reflected in the cross-reference that is provided in Appendix F of this report. Therefore, AFSER Confirmatory Item 1.8-1 is closed.

1.9 Summary of Confirmatory Items

In the DSER and AFSER, the staff identified many confirmatory items that required formal documentation of information in the SSAR, PRA, or Tier 1 material. Most of these confirmatory items were resolved as described throughout this report. The following items are confirmatory at the time of issuance of this report, and do not need to be resolved before issuing the FDA on the AP600. These items will be resolved during the staff's review of the AP600 design control document.

Each confirmatory item was assigned a unique identifying number. The number identifies the section in this report where the confirmatory item is described. For example, Confirmatory Item 1.5-1 is discussed in Section 1.5 of this report.

Item	Description
1.1.2-1	Westinghouse will submit updates to the non-proprietary versions of certain documents withheld from public disclosure in accordance with 10 CFR 2.790
1.5-1	Westinghouse will submit the design control document to support its application to certify the AP600 design.
1.7-1	The Tier 2* information must be designated in the DCD.

Figure 1.2-1 AP600 Reactor Coolant System

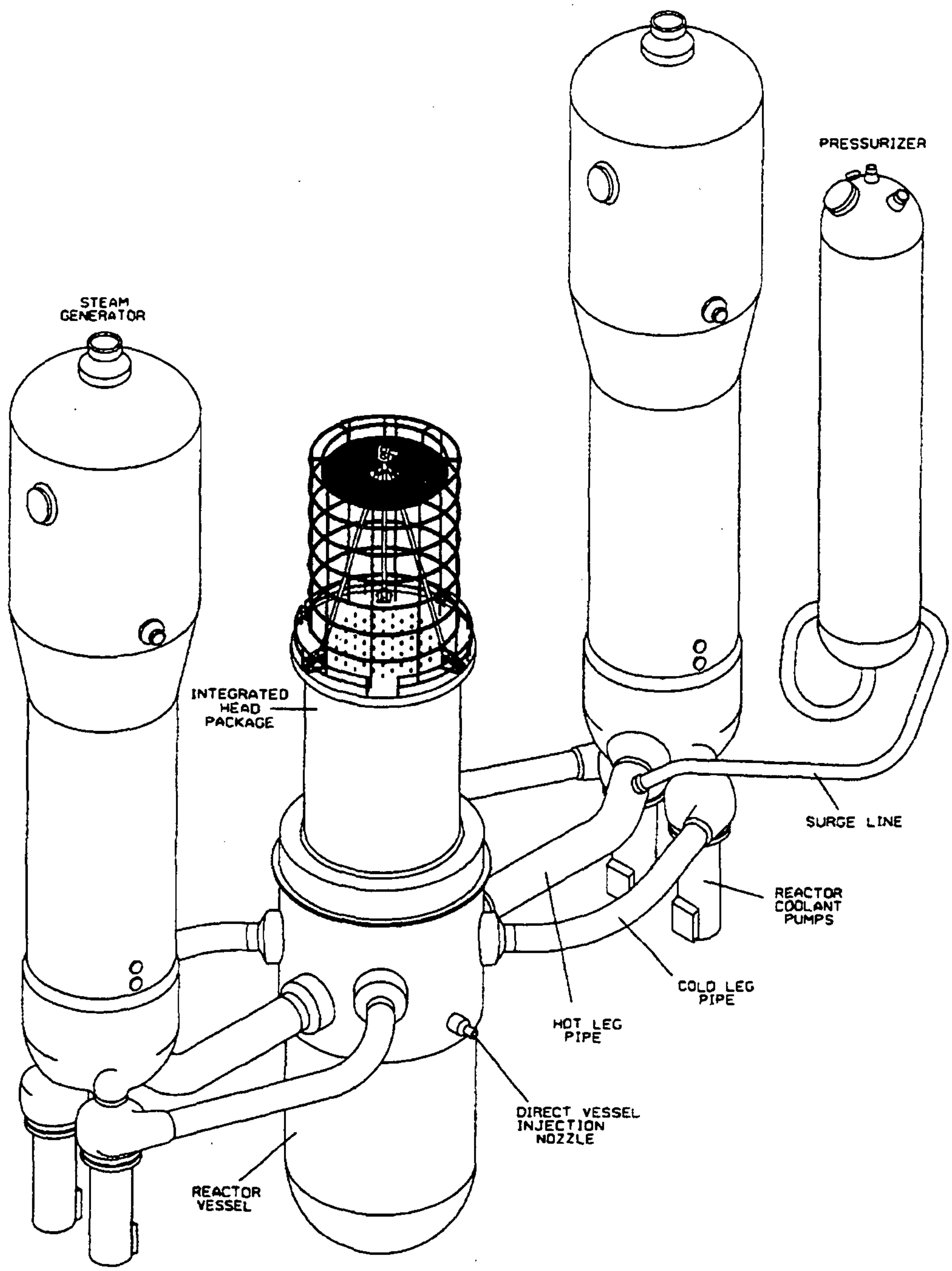


Figure 1.2-2 AP600 Passive Safety Injection System Post-LOCA, Long Term Cooling

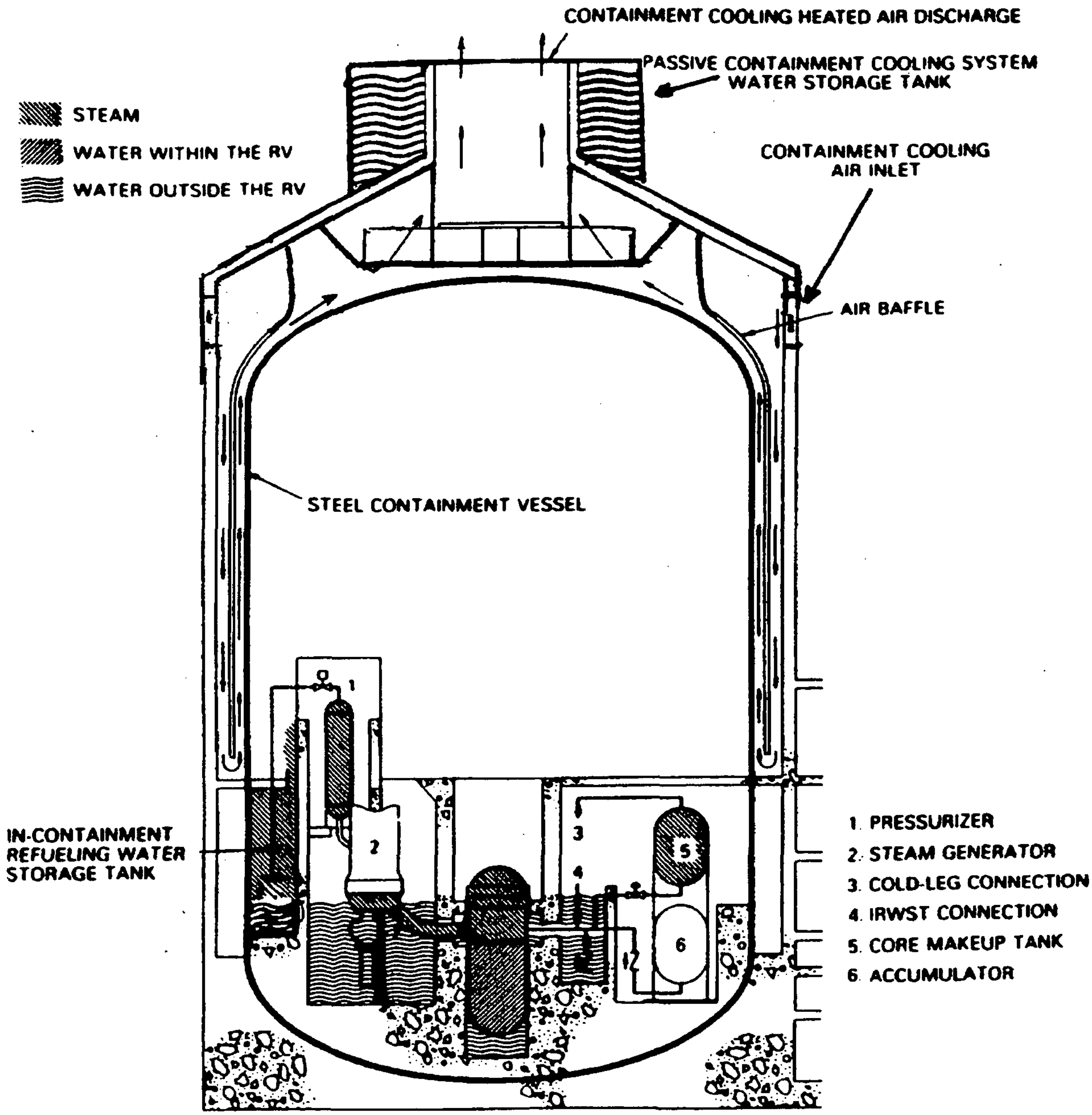


Figure 1.2-3 AP600 Passive Containment Cooling System

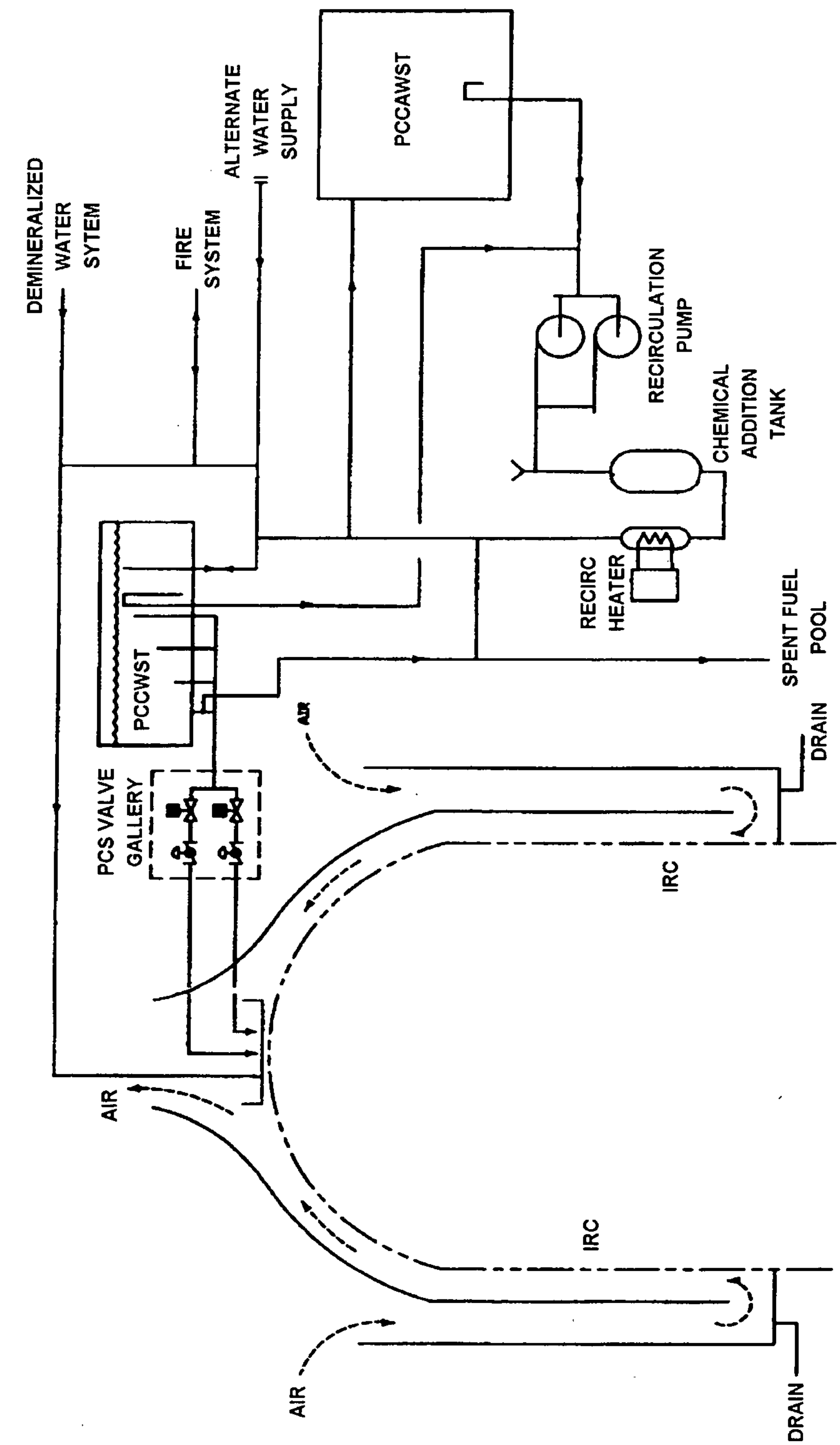


Figure 1.2-4 AP600 Safety Injection Systems

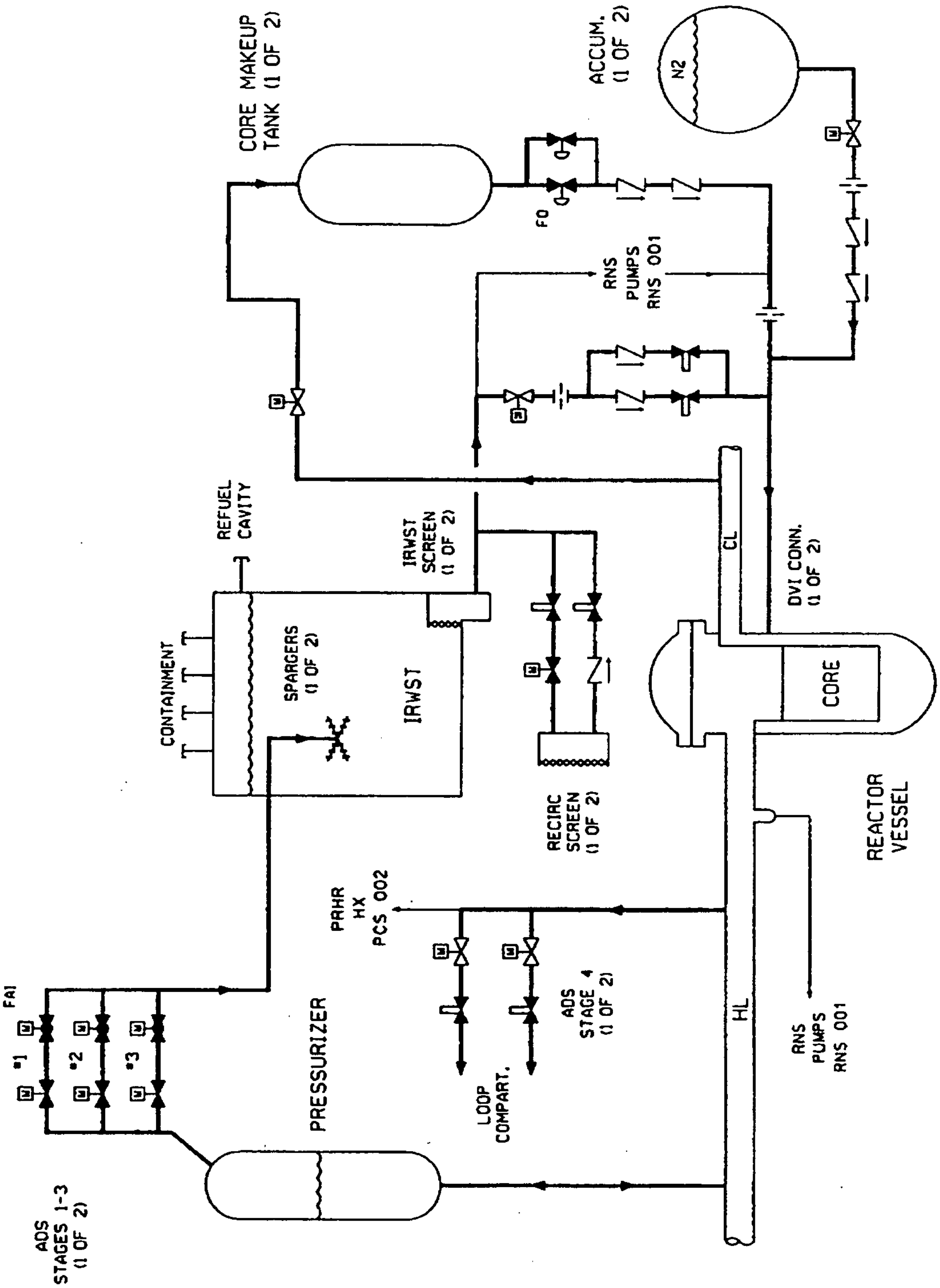


Figure 1.2-5 AP600 Plant Layout
(Sheet 1 of 2)

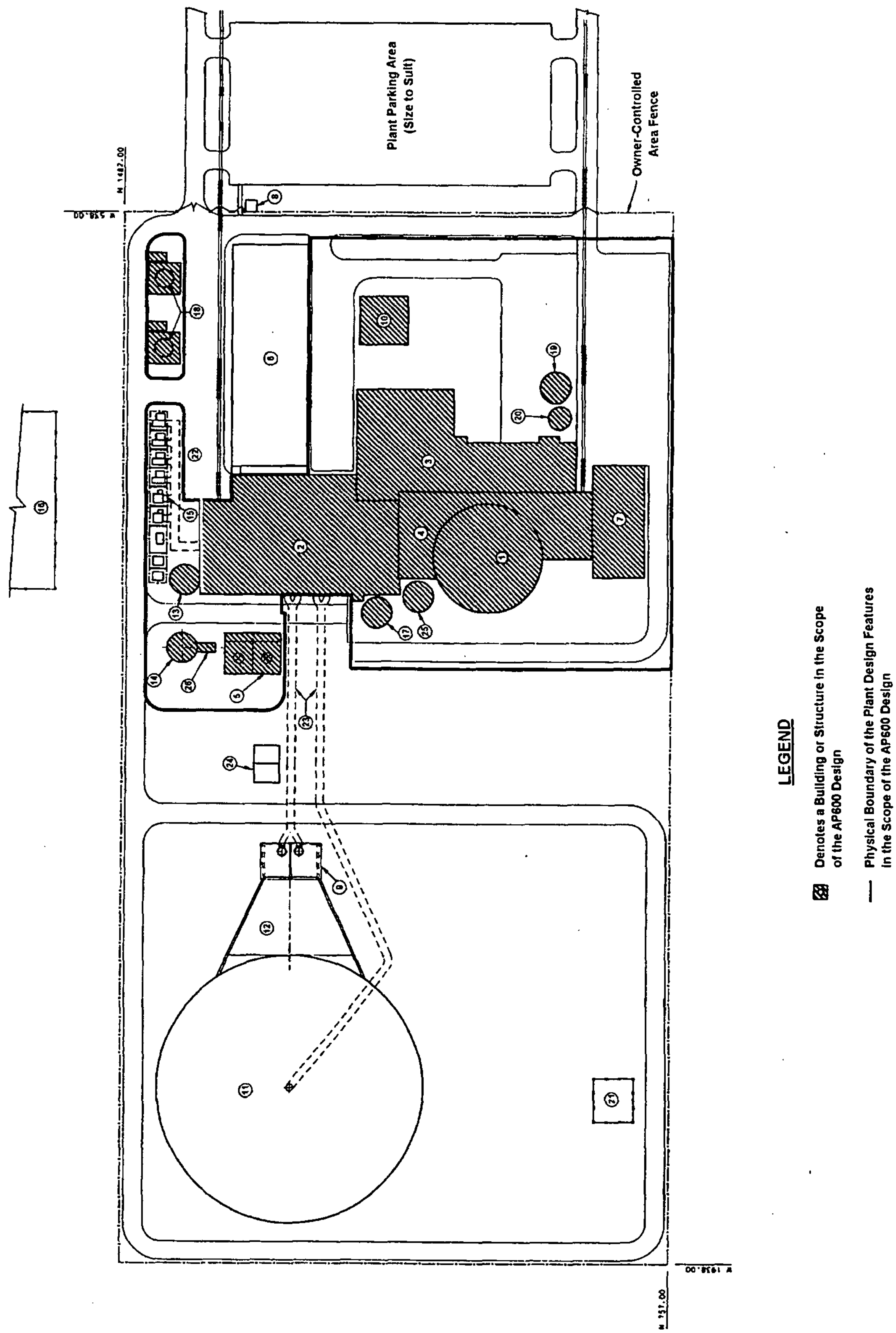


Figure 1.2-5 AP600 Plant Layout
(Sheet 2 of 2)

1. Containment/Shield Building
2. Turbine Building
3. Annex Building
4. Auxiliary Building
5. Service Water System Cooling Towers
6. Administration Building
7. Radwaste Building
8. Plant Entrance
9. Circulating Water Pump Intake Structure
10. Diesel Generator Building
11. Circulating Water System Cooling Tower
12. Circulating Water System Intake Canal
13. Fire Water/Clearwell Storage Tank
14. Fire Water Storage Tank
15. Transformer Area
16. Switchyard
17. Condensate Storage Area
18. Diesel Generator Fuel Oil Storage Tanks
19. Demineralized Water Storage Tank
20. Boric Acid Storage Tank
21. Hydrogen Storage Tank Area
22. Turbine Building Laydown Area
23. Circulating Water Pipe
24. Waste Water Retention Basin
25. Passive Containment Cooling Ancillary Water Storage Tank
26. Diesel-Driven Fire Pump/Enclosure

2 SITE ENVELOPE CHARACTERISTICS

This chapter discusses the geography and demography, nearby facilities, meteorology, hydrologic engineering, and geological, seismological, and geotechnical engineering aspects of the Westinghouse AP600 design. The evaluation is based on the staff's review of the AP600 standard safety analysis report (SSAR) and Westinghouse's responses to the NRC staff's questions.

2.1 Geography and Demography

In the DSER, the staff requested that Westinghouse include in the SSAR that the Combined License (COL) applicants referencing the AP600 certified design will be required to provide site-specific information related to site location and description, exclusion area authority and control, and population distribution. This was identified as DSER Open Item 2.1-1 and COL Action Item 2.1-1. Westinghouse included this requirement in Revision 2 to Section 2.1.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.1-1 is closed.

2.1.1 Site and Location Description

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on the site and its location, including political subdivisions, natural and man-made features, population, highways, railways, waterways, and other significant features of the area. This was identified as DSER Open Item 2.1.1-1 and COL Action Item 2.1.1-1. Westinghouse included this requirement in Revision 2 to Section 2.1.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.1.1-1 is closed.

2.1.2 Exclusion Area Authority and Control

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on the exclusion area authority and control, as well as any activity that may be permitted within the exclusion area. This was identified as DSER Open Item 2.1.2-1 and COL Action Item 2.1.2-1. Westinghouse included this requirement in Revision 2 to Section 2.1.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.1.2-1 is closed.

2.1.3 Population Distribution

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on population distribution. This was identified as DSER Open Item 2.1.3-1 and COL Action Item 2.1.3-1. Westinghouse included this requirement in Revision 2 to Section 2.1.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.1.3-1 is closed.

2.2 Nearby Industrial, Transportation, and Military Facilities

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information related to the identification of potential hazards stemming from nearby industrial, transportation, and military facilities within the site vicinity, including an evaluation of potential accidents. This was identified as DSER Open Item 2.2-1 and COL Action Item 2.2-1. Westinghouse included this requirement in Revision 5 to Section 2.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.2-1 is closed.

2.2.1 Aircraft Hazards

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide a detailed review of aircraft hazards. This was identified as DSER Open Item 2.2.1-1 and COL Action Item 2.2.1-1. Westinghouse included this requirement in Revision 2 to Section 2.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.2.1-1 is closed.

2.2.2 Transportation

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on nearby transportation routes. This was identified as DSER Open Item 2.2.2-1 and COL Action Item 2.2.2-1. Westinghouse included this requirement in Revision 2 to Section 2.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.2.2-1 is closed.

2.2.3 Other Hazards

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on potential industrial and military hazards. This was identified as DSER Open Item 2.2.3-1 and COL Action Item 2.2.3-1. Westinghouse included this requirement in Revision 2 to Section 2.2.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.2.3-1 is closed.

2.3 Meteorology

In Section 2.3 of the SSAR, Westinghouse specifies bounding conditions and assumptions related to regional climatology, local meteorology, onsite meteorological measurements program, short-term (accident) diffusion estimates χ/Q, and long-term diffusion estimates χ/Q. The term χ/Q is the relative atmospheric concentration, χ (Ci/m^3), of radiological releases at the receptor point in terms of the rate of release, Q (Ci/second), from the point of release.

Westinghouse states that site-specific meteorology information would be provided by the COL applicant. Further, Westinghouse notes that, if the site-specific meteorology parameters exceed the bounding χ/Q values in Table 2-1 of the SSAR, the COL applicant will address how the radiological consequences resulting from the design-basis accidents (DBAs) continue to meet the dose reference values given in 10 CFR Part 50.34 and control room operator dose limits given in General Design Criteria (GDC) 19 using site-specific χ/Q values.

2.3.1 Regional Climatology

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to submit site-specific information related to Section 2.3.1, "Regional Climatology" of the SSAR. This was identified as DSER Open Item 2.3.1-1 and COL Action Item 2.3.1-1. Subsequent to issuance of the DSER, Westinghouse specified in Section 2.3.1 of the SSAR that the regional climatology is site specific and will be defined by the COL applicant. The staff finds this acceptable, and therefore, DSER Open Item 2.3.1-1 is closed.

2.3.2 Local Meteorology

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to submit site-specific information related to Section 2.3.2 of the SSAR. This was identified as DSER Open Item 2.3.2-1 and COL Action Item 2.3.2-1. Subsequent to issuance of the DSER, Westinghouse specified in Section 2.3.2 of the SSAR that the local meteorology is site specific and will be defined by the COL applicant. The staff finds this acceptable, and therefore, DSER Open Item 2.3.2-1 is closed.

2.3.2.1 Tornados

Westinghouse specifies in Section 3.3.2.1 and Table 2.0-1 of the SSAR the maximum design-basis tornado (DBT) wind speed of 483 km/h (300 miles per hour) as a site interface parameter for the AP600 design. The current NRC regulatory position with regard to a DBT is in WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," May 1974, and Regulatory Guide (RG) 1.76, "Design Basis Tornado for Nuclear Power Reactors." WASH-1300 states that the probability of occurrence of a tornado that exceeds the DBT should be on the order of 1E-07 per year per nuclear power plant. The RG delineates the maximum wind speeds of 225 to 579 km/h (140 to 360 mph) depending on the region of the contiguous United States.

The staff reevaluated the regulatory positions in RG 1.76 using tornado data that was not available when the RG was developed. The staff's evaluation was published as NUREG/CR-4461, and included the tornado data tape prepared by the National Severe Storm Forecast Center with 30 years of data, 1954 through 1983. This tape contains the data for the approximately 30,000 tornados that occurred during the period.

The reevaluation found that the tornado strike probabilities range from near 1E-07 per year for much of the western United States to about 1E-03 per year in the central United States. Thus, wind speed values associated with a tornado having an annual strike probability of 1E-07 range from less than 246 km/h to 534 km/h (less than 153 mph to 332 mph). These wind speed estimates are 80 to 160 km/h (50 to 100 mph) lower than the wind speed estimates presented in WASH-1300 and RG 1.76 for most of the United States. On the basis of this analysis, NUREG/CR-4461 concluded that it is reasonable to reduce DBT wind speed to 322 km/h

(200 mph) for the United States west of the Rocky Mountains and 531 km/h (330 mph) to east of the Rocky Mountains.

On the basis of updated tornado data and the analysis in NUREG/CR-4461, the staff concluded that it is acceptable to reduce the DBT wind speed to 483 km/hr (300 mph). In SECY-93-087, the staff gives its position on the DBT. The Commission, in its staff requirements memorandum of July 21, 1993, approved the staff recommended position that a maximum tornado wind speed of 483 km/hr (300 mph) be used for the DBT for advanced light-water reactors. Therefore, the staff finds the DBT wind speed specified by Westinghouse to be acceptable.

In the DSER, the staff requested that Westinghouse include in the SSAR the probability of occurrence of a tornado that exceeds the DBT. This was identified as DSER Open Item 2.3.2.1-1. Subsequent to issuance of the DSER, Westinghouse stated in Section 3.3.2.1 of the SSAR that the probability of wind speeds greater than the design basis tornado is between 1E-06 and 1E-07 per year for an AP600 at a "worst location" anywhere within the contiguous United States and specifies in the same section the maximum wind speed for DBT as 300 mph. The staff finds this acceptable, and therefore, DSER Open Item 2.3.2.1-1, to specify the probability of occurrence of a tornado that exceeds the DBT, is closed.

It should be recognized, however, that the DBT requirements have been used in establishing structural requirements (minimum concrete wall thickness) for the protection of nuclear plant safety-related structures, systems, and components (SSCs) against the effects not covered explicitly in review guidance such as RGs or the standard review plan (SRP). Specifically, some aviation (general aviation light aircraft) crashes, nearby explosions, and explosion debris or missiles have been reviewed and evaluated routinely by the staff by considering the existence of the tornado protection requirements.

Hence, the staff's acceptance of a COL application will also necessitate a concurrent review and evaluation of the effect on the protection criteria for some external impact hazards, such as general aviation or nearby explosions.

2.3.3 Onsite Meteorological Measurements Program

Details on the atmospheric diffusion characteristics of a nuclear power plant site are required to determine that postulated accidental, as well as routine operational, releases of radioactive materials are within NRC regulatory guidelines. The meteorological characteristics of a site are determined by staff evaluation of meteorological data collected at the site by the onsite meteorological measurements program in accordance with RG 1.23, "Onsite Meteorological Program."

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design shall submit the onsite meteorological measurements program for review by the staff. This was identified as DSER Open Item 2.3.3-1. Subsequent to issuance of the DSER, Westinghouse specified in Section 2.3.3 of the SSAR that the onsite meteorological measurements program is site specific and will be defined by the COL applicant. The staff finds this acceptable, and therefore, DSER Open Item 2.3.3-1 is closed. This is COL Action Item 2.3.3-1.

2.3.4 Short-Term (Accident) Atmospheric Relative Concentration

In lieu of site-specific meteorological data, Westinghouse provided a bounding set of short-term (accident) atmospheric relative concentration (χ/Q) values for the AP600 design. The meteorological data representative of an 80-90th percentile of United States operating nuclear power plant sites were used to develop these χ/Q values. Westinghouse calculated ground-level 0-2 hour χ/Q values at a 0.8 km (0.5 mile) exclusion area boundary (EAB) using a Gaussian diffusion model modified for source configuration and lateral plume meander under stable atmospheric conditions. In calculating these χ/Q values, Westinghouse used the methodology provided in RG 1.145, "Atmospheric Dispersion Model for Potential Accident Consequence Assessment at Nuclear Power Plants."

In Table 2-1 of the SSAR, Westinghouse provides these short-term (accident) χ/Q values at both the EAB and low population zone (LPZ) receptors as follows, and they are reproduced as follows:

Location	Time Period	Dilution Factor χ/Q (sec/m^3)
EAB	0-2 hours	1.00E-03
LPZ	0-8 hours	1.35E-04
LPZ	8-24 hours	1.00E-04
LPZ	1-4 days	5.40E-05
LPZ	4-30 days	2.20E-05

A site selected for an AP600 facility should have χ/Q values within the bounds specified above. In the event that a site selected for the AP600 design exceeds the bounding χ/Q values, the COL applicant must demonstrate that the radiological consequences associated with the controlling design-basis accident using its site-specific χ/Q values continues to meet the dose reference values given in 10 CFR Part 50.34 and control room operator dose limits given in GDC 19.

In DSER Section 2.3.4, the staff requested that Westinghouse revise the SSAR to state (1) the basis for these χ/Q values; (2) the model and methodology used to calculate ground-level 0-2 hour χ/Q values at 0.5 mile EAB and for time periods greater than 2 hours (i.e., 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days) at a LPZ; and (3) the LPZ distance used to develop the χ/Q values. This was identified as DSER Open Item 2.3.4-1. In Revision 13 to the SSAR, Westinghouse provided this information in Section 2.3.4 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.3.4-1 is closed.

2.3.5 Long-Term (Routine) Diffusion Estimates

In DSER Section 2.3.5, the staff requested that Westinghouse provide in the SSAR the methodology used to determine the annual average long-term relative concentration at the site boundary for evaluation of the AP600 radioactive waste treatment system design. This was identified as DSER Open Item 2.3.5-1. In DSER Section 2.3.5, the staff reflected a Westinghouse statement that the long-term diffusion estimates are site-specific and will be provided by the COL applicant. This was identified as DSER Open Item 2.3.5-2 and COL Action Item 2.3.5-1.

In Revision 13 to the SSAR, Westinghouse stated in Section 2.3.5 that the χ/Q value specified in SSAR Table 2.1 is expected to envelop atmospheric conditions at most U.S. sites and that if a selected AP600 site has a χ/Q value that exceeds this specified value, the release concentrations calculated in Section 11.3 of the SSAR would be adjusted proportionately to the change in χ/Q. The staff finds the Westinghouse specification acceptable, and therefore, DSER Open Item 2.3.5-1 is closed. Subsequent to the issuance of the DSER, Westinghouse specified in Section 2.3.5 of the SSAR that the long-term diffusion estimates are site-specific and will be provided by the COL applicant. The staff finds this acceptable, and therefore, DSER Open Item 2.3.5-2 is closed.

During their review of the AP600, the Advisory Committee on Reactor Safeguards (ACRS) raised a concern about the Site Characteristics. This concern was provided to Westinghouse in a May 27, 1998, letter. Specifically, the concern was that the staff should

> ... ensure that the calculational methodologies used by the Combined License (COL) applicant to derive χ/Q not mask the effects of any unique site meteorological characteristics related to topology, geographical location, directed wind flows during specific times of the day, or any peculiar atmospheric inversion characteristics.

In Revision 24 of the SSAR, Westinghouse augmented the COL Action Items in Section 2.3.6.4 and 2.3.6.5 to state that the COL applicant should consider topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and receptors. The staff finds this acceptable. This is COL Action Item 2.3.5-2.

2.3.6 Onsite Control Room Atmospheric Relative Concentrations

In Table 15A-5 of the SSAR, Westinghouse provided a set of reference control room relative concentration (χ/Q) values for calculating the potential radiation doses to control room personnel following postulated design-basis accidents. In the DSER, the staff requested that Westinghouse provide the methodology used to determine these values, including considerations given to potential radioactive material release points and pathways to the main control room following a design-basis event. This was identified as DSER Open Item 2.3.6-1

In Revision 20 to the SSAR, Westinghouse stated that it used 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 main control room χ/Q values for the AP600 design. Westinghouse also stated that site-specific meteorological information will be provided by the COL applicant. During a COL review, the staff will perform an assessment of all inputs and assumptions used by the COL applicant to calculate χ/Q values to determine the acceptability of the AP600 design for the planned siting. If the assessment results in χ/Q values that exceed the reference χ/Q values provided by Westinghouse, the COL applicant will need to address how the plant design and operation will be capable of meeting the requirements of GDC 19.

The staff finds the main control room χ/Q values specified by Westinghouse acceptable as reference values for the AP600 design, and therefore, DSER Open Item 2.3.6-1 is closed.

2.4 Hydrologic Engineering

COL applicants referencing the AP600 certified design will be required to provide site-specific information related to hydrologic engineering, as discussed in the following sections.

2.4.1 Hydrologic Description

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide a detailed description of all major hydrologic features on or in the vicinity of the site. The COL applicant will also be required to provide specific hydrologic descriptions of the site, including critical elevations of all safety-related structures, exterior accesses, equipment, and systems. This was identified as DSER Open Item 2.4.1-1 and COL Action Item 2.4.1-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.1.-1 is closed.

2.4.2 Floods

Westinghouse stated in Section 2.4 of an earlier version of the SSAR that the plant is designed for a flood level up to grade. This conflicts with Table 1.2-6 in Chapter 1 of Volume II of the Utility Requirements Document (URD), which states that the maximum flood (or tsunami) level site envelope parameter is 0.3 m (1 ft) below grade. The NRC staff agrees with this URD parameter, as documented in NUREG-1242. The staff requested that Westinghouse either justify its selection of plant grade for the maximum flood level, or state that the maximum flood level will be at least 0.3 m (1 ft) below grade. This was identified as DSER Open Item 2.4.2-1. Westinghouse revised the Section 2.4 of the SSAR by stating that, for structural analysis purposes, grade elevation is established at the 30.5-m (100-ft) plant elevation and that actual grade will be a few inches lower to prevent surface water from entering doorways. The staff finds this acceptable, and therefore, DSER Open Item 2.4.2-1 is closed.

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on historical flooding and potential flooding factors, including flood design considerations and the effects of local intense precipitation. This was identified as DSER Open Item 2.4.2-2 and COL Action Item 2.4.2-1. Westinghouse included this requirement in Revision 5 to Section 2.4.1.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.2-2 is closed.

2.4.3 Probable Maximum Flood on Streams and Rivers

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information that will be used to determine the design-basis flooding at the site, including the probable maximum flood on streams and rivers, as well as the extent of flood protection required for safety-related SSCs. This was identified as DSER Open Item 2.4.3-1 and COL Action Item 2.4.3-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.3-1 is closed.

2.4.4 Potential Dam Failures

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on potential dam failures. This was identified as DSER Open Item 2.4.4-1 and COL Action Item 2.4.4-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.2 in the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.4-1 is closed.

2.4.5 Probable Maximum Surge and Seiche Flooding

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will provide site-specific information on probable maximum surge and seiche flooding. This was identified as DSER Open Item 2.4.5-1 and COL Action Item 2.4.5-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.5-1 is closed.

2.4.6 Probable Maximum Tsunami Loading

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will provide site-specific information on probable maximum tsunami loading. This was identified as DSER Open Item 2.4.6-1 and COL Action Item 2.4.6-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.6-1 is closed.

2.4.7 Ice Effects

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on ice effects, and must demonstrate that safety-related facilities and water supply will not be affected by ice flooding or blockage. This was identified as DSER Open Item 2.4.7-1 and COL Action Item 2.4.7-1. Westinghouse stated in Revision 2 to Section 2.4 of the SSAR that adverse effects of flooding due to high water or ice effects do not have to be considered for site-specific non-safety-related structures and water sources outside the scope of the certified design because flooding of water intake structures, cooling canals, reservoirs, or channel diversions would not prevent safe operation of the plant. The staff finds this acceptable, and therefore, DSER Open Item 2.4.7-1 is closed, and COL Action Item 2.4.7-1 is dropped.

2.4.8 Cooling Water Canals and Reservoirs

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on the hydraulic design of canals and reservoirs used to transport and impound cooling water and protect safety-related structures. This was identified as DSER Open Item 2.4.8-1 and COL Action Item 2.4.8-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.3 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.8-1 is closed.

2.4.9 Channel Diversions

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on channel diversions. This was identified as DSER Open Item 2.4.9-1 and COL Action Item 2.4.9-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.3 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.9-1 is closed.

2.4.10 Flood Protection Requirements

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on flood protection requirements. This was identified as DSER Open Item 2.4.10-1 and COL Action Item 2.4.10-1. In Revision 2 to Section 2.4.1.2 of the SSAR, Westinghouse included this requirement. The staff finds this acceptable, and therefore, DSER Open Item 2.4.10-1 is closed.

2.4.11 Cooling Water Supply

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on the cooling water supply, including the natural events that may reduce or limit the available cooling water supply. In addition, COL applicants will be required to ensure that an adequate water supply will exist to operate or shut down the plant as required. This was identified as DSER Open Item 2.4.11-1 and COL Action Item 2.4.11-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.3 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.11-1 is closed.

2.4.12 Groundwater

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on groundwater. This was identified as DSER Open Item 2.4.12-1 and COL Action Item 2.4.12-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.4 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.12-1 is closed.

2.4.13 Accidental Release of Liquid Effluents in Ground and Surface Water

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide site-specific information on the ability of the ground and surface water to disperse, dilute, or concentrate accidental releases of liquid effluents. Effects of these releases on existing and known future use of surface water resources will also be provided. This was identified as DSER Open Item 2.4.13-1 and COL Action Item 2.4.13-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.5 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.13-1 is closed.

2.4.14 Technical Specification and Emergency Operation Requirement

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to establish the technical specifications and emergency procedures required to implement flood protection for safety-related facilities, and to ensure an adequate water supply to shut down and cool the reactor. This was identified as DSER Open Item 2.4.14-1 and COL Action Item 2.4.14-1. Westinghouse included this requirement in Revision 2 to Section 2.4.1.6 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.4.14-1 is closed.

2.5 Geological, Seismological, and Geotechnical Engineering

COL applicants referencing the AP600 design will be required to provide site-specific information related to basic geological, seismological, and geotechnical engineering of the site and the region, as discussed in the following sections.

2.5.1 Basic Geologic and Seismic Information

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 certified design will be required to provide the following site-specific geologic and seismic information:

- regional and site physiography
- geomorphology
- stratigraphy
- lithology
- structural geology
- tectonics
- seismicity

This was identified as DSER Open Item 2.5.1-1 and COL Action Item 2.5.1-1. Westinghouse included this requirement in Revision 2 to Section 2.5.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.1-1 is closed. As requested in Q230.141F in a letter dated December 8, 1997, Westinghouse replaced the word "lithography" by "lithology" in Revision 19 to Section 2.5.1 of the SSAR. Therefore, this issue is resolved.

2.5.2 Vibratory Ground Motion

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 design will be required to provide the following site-specific information related to seismic and tectonic characteristics of the site and region:

- correlation of earthquake activity with geologic structure or tectonic provinces
- maximum earthquake potential
- seismic wave transmission characteristics of the site
- safe-shutdown earthquake (SSE)

This was identified as DSER Open Item 2.5.2-1 and COL Action Item 2.5.2-1. Westinghouse included this requirement in Revision 2 to Section 2.5.2.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.2-1 is closed.

The AP600 is designed for a SSE defined by a peak ground acceleration of 0.3g input at the plant grade level. This ground motion is higher than the SSE of any of the currently licensed nuclear power plants east of the Rocky Mountains. However, there may be areas near the source zones of large earthquakes where this design ground motion would not be adequate. It has become known in recent years that there is the potential, in eastern North America, for the ground motion from smaller nearby earthquakes to exceed the RG 1.60 design response spectrum at frequencies above 10 Hz (approximately). The Amplification Factors for Control Points for the AP600 Design Response Spectra are shown in Table 3.7.1.3 in the SSAR, and the horizontal and vertical ground response spectra (for safe shutdown earthquake) corresponding to 2, 3, 4, 5 and 7 percent of the critical damping are shown in SSAR Figures 3.7.1-1 and 3.7.1-2, respectively. COL applicants will be required to compare site-specific earthquake ground motions to the ground motions used as input for the design certification. COL applicants must demonstrate that the site-specific response spectra at the finished grade level in the free field are enveloped by the ground motions used as input for the design certification (as shown in SSAR Figures 3.7.1-1 and 3.7.1-2). The site-specific response spectra must be developed at the finished grade level taking into account the site specific soil amplification. In addition, COL applicants must assure that the site-specific response spectra at the foundation level (12.2 m (40 ft) below the finished plant grade) in the free field are less than or equal to those given in SSAR Figures 3.7.1-18 and 3.7.1-19.

COL applicants must also check that the foundation material layers are approximately horizontal (dip less than 20 degrees), and that the shear wave velocity of the soil is greater than or equal to 304.8 m/sec (1000 ft/sec). If the site-specific spectra at plant grade and at foundation level exceed the corresponding AP600 certified response spectra given in SSAR Figures 3.7.1-1 and 3.7.1-2, and Figures 3.7.1-18 and 3.7.1-19, respectively, at any frequency, or if soil conditions are outside the four soil profiles included in the AP600 design certification, then COL applicants may perform a site-specific evaluation which, among other things, will consist of dynamic soil-structure interaction (SSI) analysis and generation of in-structure response spectra to be compared with the floor response spectra of the AP600 certified design at 5 percent damping at six locations specified in Revision 17 to the SSAR, as described in detail in Section 3.7.2 of this final safety evaluation report (FSER). In addition, lateral earth pressures computed from the site-specific analysis must not exceed the AP600 certified design values. This was identified as DSER Open Item 2.5.2-2 and COL Action Item 2.5.2-2. Westinghouse included this requirement in Revision 17 to Sections 2.5.2.1 and 2.5.2.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.2-2 is closed, and COL Action Item 2.5.2-2 is dropped.

The AP600 standard seismic design parameter is 0.3g peak ground acceleration with the response spectra shown in the SSAR Figures 3.7.1-1 and 3.7.1-2 for the free field at the ground surface. Westinghouse stated in Revision 17 to Section 2.5.2 of the SSAR that the AP600 has been designed using a set of four design soil profiles (described in Section 3.7.1.4 of the SSAR). These are a hard rock site, a soft rock site, a soft-to-medium soil site, and an upper bound soft-to-medium soil site. The details of these four design profiles is discussed in

Appendices 2A and 2B of the SSAR, and described in Section 2.5.4 of this report. The staff finds this acceptable, and therefore, this issue is resolved.

2.5.3 Surface Faulting

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 design will be required to provide detailed surface and subsurface geological and geophysical information to ensure that the potential does not exist for surface or near-surface faulting affecting the site. This was identified as DSER Open Item 2.5.3-1 and COL Action Item 2.5.3-1. Westinghouse included this requirement in Revision 2 to Section 2.5.3.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.3-1 is closed.

2.5.4 Stability of Subsurface Materials and Foundations

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicant will provide site-specific information related to the geotechnical engineering aspects of the site to demonstrate comparability to the design analyses assumptions given in Table 2-1 of the SSAR. The COL applicant's submittal must meet the guidelines set forth in Section 2.5.4 of RG 1.70. This was identified as DSER Open Item 2.5.4-1 and COL Action Item 2.5.4-1. In response to a staff question as to why Westinghouse has not included this requirement in the SSAR, Westinghouse explained, in a response dated June 5, 1996, that the recommendation to include a COL item to address the stability of subsurface material and foundations is accomplished by a number of specific COL information items already included in the SSAR that address Section 2.5.4 of RG 1.70, and was given in a table in a letter dated June 5, 1996, which provided cross references between the RG 1.70 items and the AP600 SSAR information. The staff finds this approach to be acceptable, and therefore, DSER Open Item 2.5.4-1 is closed, and COL Action Item 2.5.4-1 is dropped.

2.5.4.1 Site and Facilities

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 design will be required to provide site-specific information regarding the underlying site conditions and geologic features. This information will include site topographical features, as well as the locations of various seismic Category I structures and appurtenances (e.g., pipelines and channels) with regard to the source(s) of normal and emergency cooling water. This was identified as DSER Open Item 2.5.4.1-1 and COL Action Item 2.5.4.1-1. Westinghouse included this requirement in Revision 15 to Section 2.5.4.6.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.1-1 is closed.

Based on a detailed survey of the soil conditions at selected, existing nuclear power plant sites and on the results of a series of two-dimensional SSI analyses (as discussed in detail in the SSAR Appendixes 2A and 2B), Westinghouse considered the shear wave velocity profiles and related governing parameters of the following four sites in the seismic SSI analysis of the AP600 Nuclear Island (NI) and its components to develop the floor response spectra for the design of Category I structures (as described in Revision 22 Section 3.7.1.4 of the SSAR):

- for the hard rock site, an upper bound case for firm sites using a fixed base seismic analysis.

- for the soft rock site, a shear wave velocity of 731.5 m/sec (2400 ft/sec) at the ground surface, increasing linearly to 975.3 m/sec (3200 ft/sec) at a depth of 73.2 m (240 ft), and base rock at the depth of 36.6 m (120 ft).

- for the soft-to-medium stiff soil site, a shear wave velocity of 304.8 m/sec (1000 ft/sec) at the ground surface, increasing parabolically to 731.5 m/sec (2400 ft/sec) at a depth of 73.2 m (240 ft), and base rock at the depth of 36.6 m (120 ft).

- for the upper bound to the soft-to-medium soil site, a shear wave velocity of 431 m/sec (1414 ft/sec) at ground surface, increasing parabolically to 1035.5 m/sec (3394 ft/sec) at the depth of 73.2 m (240 ft), and base rock at the depth of 36.6 m (120 ft).

It must be noted that, for the last three sites listed above, the base rock is located at a depth of 36.6 m (120 ft), and is considered rigid. The use of these design soil profiles in seismic analysis and design of structures, systems, and components are discussed in detail in Section 3.7.1 of this report.

In a response dated May 17, 1994, to Q231.27 related to the free field analyses of the generic soil profiles, Westinghouse stated that all free field analyses using the SHAKE program are based on deconvolution analysis with the control motion defined at the finished grade level. Therefore, the free field response at any depth becomes only a function of the soil column properties above that depth, and the strain-compatible shear modulus and damping values at a given depth obtained from a deep (73.2 m (240 ft)) soil column deconvolution analysis are also applicable to soil columns with depths of 36.6 m (120 ft) and 12.2 m (40 ft). In response to a staff question that the results of the free field analyses leading to this finding should be documented in the SSAR, Westinghouse stated, in its June 5, 1996, response that the deconvolution analyses described in the SSAR Appendices 2A and 2B calculate the soil motion at each depth that result in the specified free field motion. Westinghouse further stated that, since the motion at any depth is only a function of the soil column above that depth, the analysis for the 73.2-m (240-ft) depth also gives the motions at the 36.6-m (120-ft) and 12.2-m (40-ft) depths corresponding to the free field motion. In the DSER, the staff requested that Westinghouse include in the SSAR the results of the free field analysis leading to this finding. This was identified as DSER Confirmatory Item 2.5.4.1-1. The staff's position on this issue is described below.

The SSI analysis used to develop the floor response spectra for the design of Category I structures relied on deconvolution of the free field control ground motions defined at the finished grade level, as shown in SSAR Figures 3.7.1-1 and 3.7.1-2, down to the NI foundation level. The ground motion at the plant foundation level (12.2 m (40 ft)) obtained by the deconvolution methodology does not reflect the geology below the foundation level. The NRC staff has performed analyses to evaluate the effect of Westinghouse's deconvolution. In its analyses, the staff considered site geology where the shear wave velocity in the soil is 304.8 m/sec (1000 ft/sec) and the shear wave velocity in the underlying rock is 1828.8 m/sec (6000 ft/sec). The depth of the soil-rock interface was varied from 36.6 m (120 ft) to 18.3 m (60 ft) below ground surface. A rock input spectrum (NUREG/CR-0098 with a 0.2g peak ground acceleration) was propagated up to the plant foundation level (12.2 m (40 ft) below ground surface) and to the ground surface. This representation of the rock input motion was used because it is similar to the site specific motion which could be expected at many sites in the central and eastern United States. The amount of amplification of the motion and its frequency range are a function of the acoustic impedance contrast between the soil and rock, and the thickness of the soil layer.

In Figure 2.5-1 of this report, the staff illustrates the results of the staff's analysis for the horizontal component of ground motion (all the response spectra are at 5 percent of critical damping). FSER Figure 2.5-1A is a plot of the AP600 control motion at the finished grade level (modified RG 1.60 spectrum with a peak ground acceleration of 0.3g, the same as SSAR Figure 3.7.1-1) and the spectrum resulting from the deconvolution down to the 12.2-m (40-ft) depth. The deconvolved spectrum is essentially below the surface spectrum at all frequencies and is significantly lower in the frequency range of 2 to 20 Hz. FSER Figures 2.5-1B through 2.5-1E contain plots of the rock input response spectrum, the AP600 control motion at the finished grade level and the surface spectra and the spectra at 12.2-m (40-ft) depth which resulted from the propagation of the rock input motion up from the assumed rock-soil interface at the depths of 36.6 m (120 ft), 30.5 m (100 ft), 24.4 m (80 ft), and 18.3 m (60 ft), respectively.

A comparison of FSER Figure 2.5-1A with FSER Figures 2.5-1B through 2.5-1E illustrates the problem with deconvolution of surface ground motion and of not considering potential ground motion amplification in developing control ground motions for seismic design. FSER Figure 2.5-1B shows that, for the 36.6-m (120-ft) interface case, the foundation level deconvolved spectrum is lower than the amplified spectrum in the frequency range of 1.2 to 3.5 Hz, and the AP600 surface spectrum is lower than the amplified surface spectrum in the frequency range 1.2 to 3.5 Hz. FSER Figure 2.5-1C shows that for the 30.5-m (100-ft) interface case the foundation level deconvolved spectrum is lower than the amplified spectrum in the frequency range of 1.2 to 3.8 Hz, and the AP600 surface spectrum is lower than the amplified surface spectrum in the frequency range 1.2 to 4.0 Hz. FSER Figure 2.5-1D shows that for the 24.4-m (80-ft) interface case the foundation level deconvolved spectrum is lower than the amplified spectrum in the frequency range of 1.5 to 8 Hz, and the AP600 surface spectrum is lower than the amplified surface spectrum in the frequency range 1.9 to 5.5 Hz. FSER Figure 2.5-1E shows that for the 18.3-m (60-ft) interface case the foundation level deconvolved spectrum is lower than the amplified spectrum in the frequency range of 2.9 to 7 Hz, and the AP600 surface spectrum is lower than the amplified surface spectrum in the frequency range 2.5 to 8 Hz.

In response to this concern, Westinghouse agreed at the review meeting during August 4-7, 1997, to state in the SSAR that COL applicants referencing the AP600 certified design will be required to develop site-specific response spectra at the finished grade level and the foundation level, taking into account the site-specific soil amplification. COL applicants must assure that the site-specific spectra developed for the finished grade are enveloped by the response spectra shown in SSAR Figures 3.7.1-1 and 3.7.1-2. The site-specific spectra must be developed at the finished grade level considering site-specific soil amplification. Furthermore, the site-specific spectra developed for the foundation level must be less than or equal to the spectra given in Figures 3.7.1-18 and 3.7.1-19. Westinghouse included this requirement in Revision 17 to Section 2.5.2.1 of the SSAR. The staff finds this acceptable, and therefore, DSER Confirmatory Item 2.5.4.1-1 is closed. This is COL Action Item 2.5.4.1-2.

In Figure 2.5-1 of the August 8, 1994, response to Q231.1, the value of the shear wave velocity was erroneously printed as 100 ft/sec. This was identified as DSER Confirmatory

Item 2.5.4.1-2. It has been correctly shown as 1000 ft/sec in the SSAR. The staff finds this acceptable, and therefore, DSER Confirmatory Item 2.5.4.1-2 is closed.

2.5.4.2 Properties of Underlying Materials

Field Investigations

In the May 17, 1994, response to Q231.32 concerning the type of geoscience investigations a COL applicant must perform and the type of information that is critical for deciding the acceptability of a site for an AP600 plant, Westinghouse furnished a list of items related to seismology, geology, and geotechnology. The COL applicants will be required to use state-of-the-art methods to determine the static and dynamic engineering properties of all foundation soils and rocks in the site area. In addition, COL applicants will be required to submit a discussion of the type, quantity, extent, and purpose of all field explorations, as well as logs of all borings and test pits. Results of field plate load tests, field permeability tests, and other special field tests (e.g., bore-hole extensometer or pressuremeter tests) will also be required. Results of geophysical surveys will be presented in tables and profiles. COL applicants will also be required to provide all data pertaining to site-specific soil layers (including their thicknesses, densities, moduli, and Poisson's ratios) between the basemat and the underlying rock stratum. Plot plans and profiles of site explorations must be provided in the site-specific safety analysis report (SAR). This was identified as DSER Open Item 2.5.4.2-1 and COL Action Item 2.5.4.2-1. Westinghouse included this requirement in Revision 17 to Section 2.5.4.6.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.2-1 is closed.

Section 2.5.4.5.2 of the SSAR (revised after Revision 11) stated that a series of borings should be drilled on a grid pattern that encompasses the NI footprint and 12.2 m (40 ft) beyond the boundary of the footprint. The staff requested Westinghouse to justify the basis for the proposed 12.2-m (40-ft) limit. At the August 4-7, 1997, meeting with the staff, Westinghouse provided information that justified the 12.2-m (40-ft) limit outside the NI footprint based on a Boussinesq elastic stress solution. The staff finds this acceptable, as it corresponds to standard engineering practice, and therefore, this issue is resolved.

In Section 2.5.4.5.2 of the SSAR (revised after Revision 11), Westinghouse stated that at least one-fourth of the primary borings should penetrate sound rock, or for deep soil sites, to a maximum depth, d_{max}, taken as the depth at which the vertical stress during or after construction for the combined foundation loading is less than 10 percent of in situ effective overburden stress. Other borings may terminate at a depth of 18.8 m (160 ft) below the foundation (equal to the width of the structure). In Revision 17 to Section 2.5.4.5.1 of the SSAR, Westinghouse states that at least one-fourth of the primary borings will extend to a depth of 76.2 m (250 ft) below the foundation mat and the remainder of the primary borings may terminate at a depth of 48.8 m (160 ft) below the foundation mat. These boring depths are acceptable as they exceed the 10 percent stress isobar defined by the Boussinesq elastic stress solution. The staff finds this acceptable, and therefore, this issue is resolved.

As a result of discussions at the review meeting during August 4-7, 1997, Westinghouse included, in Revision 17 to Section 2.5.4.5.3 of the SSAR, certain criteria to be used to define uniform site conditions as discussed below. The subsurface may consist of layers that may dip with respect to the horizontal plane, and whose physical properties may not vary systematically across a horizontal plane. The recommended methodology for checking uniformity of a site is to determine from the boring logs a series of "best estimate" planes that define the top and bottom of each layer beneath the NI footprint. These planes should represent the boundaries between layers having different shear wave velocities (which primarily define the uniformity of a site). For a site to be considered uniform, the variation of shear wave velocity in the material below the foundation to a depth of 36.6 m (120 ft) below finished grade within the NI footprint shall meet the following criteria:

- Case 1: For a layer with a low-strain shear wave velocity greater than or equal to 762 m/sec (2500 ft/sec), the layer should have approximately uniform thickness, should have a dip not greater than 20 degrees, and should have less than 20 percent variation in the shear wave velocity from the average velocity within any layer.

- Case 2: For a layer with a low-strain shear wave velocity less than 762 m/sec (2500 ft/sec), the layer should have approximately uniform thickness, should have a dip not greater than 20 degrees, and should have less than 10 percent variation in the shear wave velocity from the average velocity within any layer.

The above criteria are included in Revision 17 to Section 2.5.4.5.3 of the SSAR. The staff finds the above criteria acceptable because these parameters are within the definition of a uniform site, and therefore, this issue is resolved. In Revision 19 to the SSAR, Westinghouse included the above criteria in SSAR Table 2.1 under the heading "Soil, Uniformity of Site" as requested by Q230.142F in a letter dated December 8, 1997, to Westinghouse. Therefore, this issue is resolved.

By Q230.143F in a letter dated December 8, 1997, the staff requested Westinghouse to revise Section 2.5.4.5.3 of the SSAR to specify that, for a layer with a low strain shear wave velocity greater than or equal to 762 m/sec (2500 ft/sec), the shear wave velocity at any location within any layer should not vary by more than 20 percent from the average velocity within any layer. In Revision 19, Westinghouse adequately revised Section 2.5.4.5.3 of the SSAR. Therefore, this issue is resolved.

Laboratory Investigations

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicant will be required to provide information about the number and type of laboratory tests and the location of samples, and must discuss the results of laboratory tests on disturbed and undisturbed soil and rock samples obtained from field investigations. This was identified as DSER Open Item 2.5.4.2-2 and COL Action Item 2.5.4.2-2. Westinghouse included this requirement in Revision 17 to Section 2.5.4.6.2 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.2-2 is closed.

2.5.4.3 Excavation and Backfill

In the September 1, 1994, response to Q231.23, Westinghouse stated that the excavation will have a vertical face, and that backfill material will not be used against the exterior walls of the NI structures. Westinghouse also described a proposed soil excavation method that uses a soil "nailing" method to retain the earth during excavation for NI structures. In this method, as the

NI excavation progresses downward, metal rods ("nails") are inserted into holes that are drilled horizontally into the adjoining undisturbed soil, and grout is pumped into the holes to anchor the nail rods. Although this method is relatively new in the United States, it has been used successfully to retain soils in excavations of up to 16.8 m (55 ft) in depth in the United States, and up to 27.4 m (90 ft) in depth in Europe (as reported by Westinghouse in the SSAR). The end product of this soil nailing method is a nominal 10.2 cm (4 inch) to 15.2 cm (6 inch) thick soil retaining wall, constructed by blowing a 27.58 MPa (4000 psi) to 34.47 MPa (5000 psi) non-expansive pea gravel shotcrete mix onto a welded wire mesh hung on the vertical soil face and supported by the soil nails. The soil nailing method produces a vertical surface (down to the bottom of the excavation), which is used as the outside form for pouring the concrete for the external basement walls. During a meeting with Westinghouse, the NRC staff expressed concerns about the design and analysis of the soil anchoring system used in the above soil nailing excavation method. This was identified as DSER Open Item 2.5.4.3-1. In response to these concerns, Westinghouse provided in Revision 5 to Section 2.5.4.1 of the SSAR, the following information concerning the soil anchoring system. The "nail" holes (about 8 inches to 10 inches in diameter) will be spaced horizontally and vertically on five- to six-ft centers and will be drilled slightly downward from the horizontal. The nominal length of the nails will be 60 to 70 percent of the wall height depending on the soil conditions. Westinghouse further stated in Revision 5 to Section 2.5.4.3 of the SSAR, that COL applicants will provide the information concerning specific soil nailing systems, including the length and size of soil nails, which is based on actual soil conditions and applied construction surcharge loads.

The excavation support system, defined by the vertical shotcrete, is also required to keep the site dry after the completion of construction to eliminate the need for special corrosion protection of basemat rebar. The shotcrete and mudmat material are required to have a crystalline waterproofing material additive to prevent water from infiltrating through small cracks in these materials. In addition, the shotcrete material must be continuous and contain no windows through which water can penetrate. After discussion at the review meeting on August 4, 1997, Westinghouse included in Revision 17 to Section 2.5.4.6.3 of the SSAR, the requirement that the COL applicant will provide information on the waterproofing system along the vertical wall and the mudmat. The staff finds acceptable the fact that the COL applicant will provide information on the specific soil nailing system and the waterproofing system for NRC review and approval, and therefore, DSER Open Item 2.5.4.3-1 is closed. This is COL Action Item 2.5.4.3-2.

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants will be required to provide data concerning the extent (horizontal and vertical) of all seismic Category I excavations, fills, and slopes. The sources, quantities, and static and dynamic engineering properties of borrow materials must be described in the site-specific SAR. The compaction requirements, results of field compaction tests, and fill material properties (such as moisture content, density, permeability, compressibility, and gradation) should also be provided. This was identified as DSER Open Item 2.5.4.3-3 and COL Action Item 2.5.4.3-1. Westinghouse included these requirements in Revision 17 to Section 2.5.4.6.3 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.3-3 is closed.

In the DSER, the staff stated that the SSAR did not provide details concerning the construction sequence for the large foundation mat of the NI. Westinghouse was requested to consider, in its design of both the foundation mat and the entire NI structural system, the effects of the differential settlement that may be caused by such factors as soil heaving and the construction sequence of the mat and other structural members of the NI. Westinghouse was further requested to include the results in the SSAR. This was identified as DSER Open Item 2.5.4.3-2. In response to DSER Open Item 2.5.4.3-2, Westinghouse provided estimates of total settlements and heave in SSAR Table 2.3 (Revision 5), which was deleted in Revision 12 to the SSAR, dated March 26, 1997. However, in Revision 17 to the SSAR, Westinghouse stated that for the limiting deep soil sites examined, the maximum estimated settlement after placement of the first concrete for the basemat is 11.4 cm (4.5 in) for the alternating sand and clay site and 35.6 cm (14 in) for the all clay site. In a telephone conference with the staff on May 1, 1996, and in its letter to NRC dated June 5, 1996, as well as in Revision 17 to Section 2.5.4.3 of the SSAR, Westinghouse stated that differential settlement between the NI foundation and surrounding buildings does not have an adverse effect on safety-related functions, as the AP600 does not rely on structures, systems, or components located outside the NI to provide safety-related functions. For the main steamlines and the main feedwater lines, anchors located at the exterior walls of the auxiliary building preclude transfer of loads due to differential settlement into the safety related portions of the lines. Westinghouse further stated that the flexibility of the lines in the turbine building minimizes the loads due to differential settlement in the non-safety-related portion of the lines. The staff finds acceptable that differential settlement is not a safety issue, and therefore, DSER Open Item 2.5.4.3-2 is closed.

The question of basemat stresses due to construction sequence was discussed in detail at the August 4-7, 1997, review meeting, and its resolution is documented in Section 3.8.5 of this report, and therefore, this issue is resolved.

2.5.4.4 Groundwater Conditions

Westinghouse considered the effects of groundwater at different depths by performing SSI studies with water table depths assumed at 0 m (0 ft) (ground surface), 12.2 m (40 ft) (bottom of base slab), and very large depths. During discussions with the staff, Westinghouse stated that, in the SASSI calculations, the effect of the ground water was incorporated by using the strain-dependent shear wave velocity determined from the SHAKE calculations while maintaining the compressional wave (P-wave) velocity at 1524 m/sec (5000 ft/sec). Westinghouse further stated in the May 16, 1994, response to Q231.27, that the water table location was considered in the SSI analyses by adjusting the Poisson's ratio of the submerged soil layers, if necessary, so that, in conjunction with the strain-compatible shear wave velocity, the minimum P-wave velocity of 1524 m/sec (5000 ft/sec) is retained. However, the staff noted during an audit that dry soil densities were used in the SSI analyses using the SASSI code. The staff requested Westinghouse to indicate in the SSAR that this procedure was used in the SSI analyses, and the effects of using the dry soil densities for saturated soil conditions. This was identified as DSER Open Item 2.5.4.4-1. At a subsequent meeting with the staff, Westinghouse presented analytical data to show that the effects of using dry soil densities on structural responses were not significant. In its response dated June 5, 1996, Westinghouse proposed a revision to the SSAR stating that it did not adjust the densities for the cases where the water table was shallow, and further stating that the effect of using the total density for saturated soils on the dynamic soil properties is negligible. Accordingly, in Revision 10 to Section 2B.3.6 of the SSAR, Westinghouse included an explanation indicating the insignificant effect of a small variation in soil density between saturated and moist conditions on the SSI analyses results. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.4-1 is closed.

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants will be required to discuss the critical cases of groundwater conditions relative to the foundation stability of the safety-related structures at the site. The COL applicant will also be required to confirm that the soil properties of the various layers, under all possible groundwater conditions during the life of the plant, will fall within the range of values assumed in the SSAR. This was identified as DSER Open Item 2.5.4.4-2 and COL Action Item 2.5.4.4-1. Westinghouse included these requirements in Revision 17 to Section 2.5.4.6.4 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.4-2 is closed.

2.5.4.5 Response of Soil and Rock to Dynamic Loading

In Section 2.5 of the SSAR, Westinghouse presented standard curves showing the variation of shear modulus and material damping with shear strain for soils based on the older (Seed and Idriss, 1970) models and analysis appropriate for sandy soils. The staff requested that Westinghouse's SSI calculations consider more recent soil degradation models that correspond to the lower-bound values presented in the Seed-Idriss model. This was identified as DSER Open Item 2.5.4.5-1. In response to this open item, Westinghouse provided in Revision 5 to Appendix 2B of the SSAR, the results of SSI calculations considering more recent soil degradation models that correspond to the lower-bound values presented in the Seed-Idriss model. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.5-1 is closed.

In the DSER, the staff requested that Westinghouse indicate in the SSAR that the impact of using other soil degradation models appropriate for other soil types (such as silts, clays, gravel, and various combinations) on the SSI response of the NI is small. This was identified as DSER Confirmatory Item 2.5.4.5-1. In response, Westinghouse stated Revision 5 to Appendix 2B, Section 2B.3.2 of the SSAR, that the soil degradation curves for cohesive soils are mainly a function of the plasticity index and differ from those of cohesionless soil. Citing the results of SHAKE analyses and SSI analyses it performed, Westinghouse indicated, in Revision 5 to Appendix 2B, Section 2B.3.2 of the SSAR, that (1) the properties associated with the upper and lower bound sandy soil cases cover the range of properties associated with five clay curves corresponding to plasticity index of 10, 20, 30, 50, and 70; and (2) that the SSI responses obtained by using the clay curves are bounded by the 2-D enveloped SSI responses. This explanation is acceptable to the staff, as far as cohesive soils are concerned. In response to the staff request that Westinghouse should perform similar analyses for gravels and materials with combinations of various soil types and report the results in the SSAR, Westinghouse revised paragraph 1 of Section 2.5.4.5.5 of the SSAR in Revision 9 confirming that the parametric analyses described in Appendices 2A and 2B of the SSAR cover a broad range of dynamic characteristics appropriate for most soil types (sand, silts, clays, gravels, and various combinations). The staff finds this acceptable, and therefore, DSER Confirmatory Item 2.5.4.5-1 is closed.

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants must demonstrate that the assumptions made in the standard design regarding the variation of shear wave velocity and material damping are applicable to the site-specific conditions. This was identified as DSER Open Item 2.5.4.5-2 and COL Action Item 2.5.4.5-1. Westinghouse included this requirement in Revision 17 to Section 2.5.4.6.5 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.5-2 is closed.

In Revision 10 to Section 2.5.4.5.5 of the SSAR, "Response of Soil and Rock to Dynamic Loading," Westinghouse stated that, for sites at which the soil characteristics are outside the range considered in Appendix 2B.2 of the SSAR, site-specific SSI analyses may be performed by the COL applicant to demonstrate acceptability and that the analysis would use the site-specific soil conditions and the site-specific SSE. This issue was discussed in detail at the August 4-7, 1997, meeting and resolved as described in Section 2.5.2 of this report and in the SSAR (Revision 17). The staff finds this acceptable, and therefore, this issue is resolved.

In Section 2.5.4.5.2.2 of the SSAR (revised after Revision 11), Westinghouse indicated that, if a site is classified as non-uniform based on the criteria listed in the SSAR, the investigative effort should be extended in such a way that the site may be demonstrated to be acceptable for AP600 by showing that the in-structure response spectra are enveloped by the design in-structure response spectrum envelopes. However, it should be clearly stated in the SSAR that the demonstration must specifically include a complete reevaluation of the SSI effects for this non-uniform site, because all SSI analyses (2-D or 3-D) performed by Westinghouse were based on uniformly bedded site profiles. The staff, in several review meetings, has raised concerns regarding how the effect of local hills and valleys in the bed rock (or competent material) need to be included in the evaluation. The staff's concern is that these non-uniform conditions would serve to change the input free field ground motions for the site (e.g., local amplification effects). This issue was discussed at the review meeting on August 4-7, 1997, and it was agreed by Westinghouse that non-uniform sites (e.g., sloping bedrock site, undulatory bedrock site, and geologically impacted site), as described in Revision 15 to the SSAR, are not covered by certified design. In Revision 17 to Section 2.5.4.5.3.1 of the SSAR, Westinghouse stated that non-uniform soil conditions may require evaluation of the AP600 seismic response as described in Section 2.5.2.2 of the SSAR. The staff finds this acceptable, and therefore, this issue is resolved.

In Section 2.5.4.5.5 of the SSAR (revised after Revision 11), Westinghouse stated that, for sites with soil characteristics outside the range of the four design profiles (listed in Section 2.5.4 of this report and discussed in SSAR Appendices 2A.2 and 2B.2), the COL applicant may use the site-specific soil conditions and site-specific SSE to perform site-specific SSI analyses and demonstrate acceptability of the site for the AP600 by comparing the floor response spectra at specified locations. This issue was discussed and resolved at the August 4-7, 1997, review meeting as described in Revision 17 to Sections 2.5.2.1 and 2.5.2.2 of the SSAR, and in Section 2.5.2 of this report. The staff finds this acceptable, and therefore, this issue is resolved.

2.5.4.6 Liquefaction Potential

In Table 2-1 of the SSAR, Westinghouse states that sites with liquefaction potential at the site-specific SSE level will be excluded from consideration. This is acceptable. COL applicants must demonstrate that no liquefaction potential exists at the SSE level for the site for soils under and around all seismic Category I structures, including any Category I buried pipelines, tunnels, and electrical ducts. COL applicants must also justify the selection of the soil properties, as well as the magnitude, duration, and number of excitation cycles of the earthquake used in the liquefaction potential evaluation (e.g., laboratory tests, field tests, and published data). The testing methods should be documented and subject to review and approval by the staff. In addition, COL applicants must perform a soil liquefaction evaluation at 1.67 times the site-specific SSE ground motion. This was identified as DSER Open Item 2.5.4.6-1 and COL Action Item 2.5.4.6-1. Westinghouse stated in Revision 17 to

Section 2.5.4.6.6 of the SSAR that the COL applicant will evaluate the liquefaction potential to address seismic margin. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.6-1 is closed.

In the DSER, the staff requested Westinghouse to discuss in the SSAR the need for margin in liquefaction potential beyond the SSE ground motion level, as well as the methods that COL applicants could use to determine the seismic margins that exist against soil liquefaction at prospective sites. Westinghouse was also requested to indicate the criteria COL applicants will be required to use to determine whether the site needs some remediation, such as removal of lenses of liquefiable soils and/or in situ improvement of soils to preclude the potential for liquefaction. This was identified as DSER Open Item 2.5.4.6-2. At the August 4-7, 1997, review meeting, Westinghouse stated that acceptable sites will have no liquefaction potential, and that sites likely to liquefy are outside the certified design. This is documented in Table 2-1 of the SSAR. Therefore, it was the Westinghouse position that the question of improving the sites to preclude the potential of liquefaction did not arise. The staff agrees with this position, and therefore, DSER Open Item 2.5.4.6-2 closed.

2.5.4.7 Bearing Capacity

In Table 2.0-1, "Bearing Strength," of an earlier version of the SSAR, Westinghouse made a general statement that soils must support the AP600 under all specified conditions. In response to a staff question on this subject, Westinghouse stated that the term "bearing strength," as used only in the old Table 2.0-1, refers to the allowable bearing capacity, and that it is defined as the more critical of either (1) the ultimate bearing capacity divided by a safety factor for static loads, or (2) the allowable bearing capacity limited by foundation settlement criteria, depending on the foundation soil. This is acceptable to the staff, because it envelopes the criteria. This was identified as DSER Confirmatory Item 2.5.4.7-1. Westinghouse included a new Table 2.2 in Revision 17 to the SSAR that gives net allowable bearing capacities. Westinghouse further stated that these bearing capacities are preliminary estimates, and that the COL applicant will perform field and laboratory investigations to establish the material type and the associated strength parameters to determine the site-specific bearing capacity value. In view of this assurance, the staff finds this acceptable, and therefore, DSER Confirmatory Item 2.5.4.7-1 is closed.

In the DSER, the staff stated that in response to a staff question on how the high-bearing capacity requirement will be met for a soft soil site with a low shear wave velocity of 304.8 m/sec (1000 ft/sec), Westinghouse stated that the evaluation of the soils is site-specific and within the scope of the COL application. The staff requested Westinghouse to describe in the SSAR the criteria that will be used to determine if the site needed improvement to meet the SSAR requirements. Further, in the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants must demonstrate that the site-specific soil bearing capacity is equal to or greater than the value documented in Table 2-1 of the SSAR. COL applicants will be required to document the method used to establish the site-specific soil bearing capacity and submit it for review and approval by the staff. This was identified as DSER Open Item 2.5.4.7-1 and COL Action Item 2.5.4.7-1. In Revision 15 to Section 2.5.4.2 of the SSAR, Westinghouse stated that for selected soft soil profiles in cohesive soils, soil improvement techniques may be employed to improve the bearing strength. During the discussions with the staff at the August 4-7, 1997, meeting, it was agreed that such soft soils fall outside the range of site certification and accordingly Westinghouse has deleted reference to improving such soft soils in Revision 17 to the SSAR. Westinghouse included these requirements in Revision 5 to Section 2.5.4.5.7 of the SSAR. This is shown as COL Action Item 2.5.4.6.7 in Revision 17 to the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.7-1 is closed.

However, as stated in Q230.144F in a letter dated December 8, 1997, to Westinghouse, the staff did not agree with a statement made in Revision 17 to Section 2.5.4.2 of the SSAR that "generally, once the static bearing capacity at a given site is adequate, dynamic bearing demand will be satisfied," because Westinghouse did not demonstrate the validity of this statement in any of the meetings with the staff. Therefore, COL applicants will be required to demonstrate that the dynamic bearing demand will be satisfied by comparing it with the site-specific seismic bearing capacity of the soils, in addition to satisfying the static bearing demand with respect to the static bearing capacity value given in Table 2-1 of the SSAR. Westinghouse included this requirement in Revision 19 to Section 2.5.4.6.7 of the SSAR, "Bearing Capacity." This is COL Action Item 2.5.4.7-2.

2.5.4.8 Earth Pressures

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicant referencing the AP600 design should provide a site-specific discussion and evaluation of the static and dynamic lateral earth pressures and hydrostatic groundwater pressures acting on plant safety-related facilities. This was identified as DSER Open Item 2.5.4.8-1 and COL Action Item 2.5.4.8-1. In response, Westinghouse stated in Revision 5 to Section 2.5.4.5.8 of the SSAR, that the AP600 is designed for static and dynamic lateral earth pressures and hydrostatic groundwater pressures acting on plant safety-related facilities using soil parameters discussed in previous sections of the SSAR, and that no additional information is required on earth pressures. In clarification of its position, Westinghouse explained during a telephone conference call on October 28, 1996, that there are no safety-related structures other than those constituting the NI, as reported in Revision 10 to Section 1.2 of the SSAR, "General Plant Description," and therefore no additional information is required to be included in the SSAR. The staff agrees with Westinghouse, and therefore, DSER Open Item 2.5.4.8-1 is closed. Additionally, COL Action Item 2.5.4.8-1 is dropped.

2.5.4.9 Soil Properties for Seismic Analysis of Buried Pipes

In the March 24, 1994, response to Q220.43, Westinghouse stated that there are no safety-related underground pipes or tunnels in the AP600 design. This was identified as DSER Confirmatory Item 2.5.4.9-1. Westinghouse included this information in Revision 17 to Section 2.5.4.6.9 of the SSAR. The staff finds this acceptable, and therefore, DSER Confirmatory Item 2.5.4.9-1 is closed.

2.5.4.10 Static and Dynamic Stability of Facilities

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 design will perform stability analyses or evaluations of all safety-related facilities. Such analyses should include foundation rebound, settlement, differential settlement, and bearing capacity. Assumptions made in the stability analyses should be confirmed by as-built data. Settlement monitoring of safety-related structures is an AP600 requirement, and is included in Table 1.8-1 of the SSAR. This was identified as DSER

Open Item 2.5.4.10-1 and COL Action Item 2.5.4.10-1. Westinghouse included this information in Revision 17 to Section 2.5.4.6.10 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.10-1 is closed.

2.5.4.11 Subsurface Instrumentation

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants referencing the AP600 design will be required to describe instrumentation, if any, proposed for monitoring the performance of the foundations for safety-related structures and systems. COL applicants must also specify the type, location, and purpose of each instrument, as well as significant details of installation methods. For example, COL applicants should reference the location and installation procedures for permanent benchmarks and markers required for monitoring the settlement of Category I structures. Similarly, in the case of safety-related water-control structures (such as dams, embankments, slopes, and canals), the installation of instruments such as piezometers, slope indicators, and settlement plates must be described in detail. A schedule for installing and reading all instruments and interpreting the data must also be provided, and limiting values for continued safety should be specified. This was identified as DSER Open Item 2.5.4.11-1 and COL Action Item 2.5.4.11-1. Westinghouse included the instrumentation requirements for monitoring the performance of the foundation of the NI, but has not addressed the instrumentation that may be needed at some sites for monitoring the performance of other safety-related water-control structures such as dams, embankments, and canals. Westinghouse stated during a telephone conference call on October 28, 1996, that there are no safety-related structures other than the NI structures in the AP600 design, as reported in Revision 10 to Section 1.2 of the SSAR, "General Plant Description," and therefore no additional information is required to be included in the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.4.11-1 is closed. However, the staff will review the provisions made by COL applicants for monitoring the performance of safety-related water-control structures (such as dams, embankments, slopes, and canals), should their sites contain such structures, the failures of which may adversely affect the safety of the main power plant.

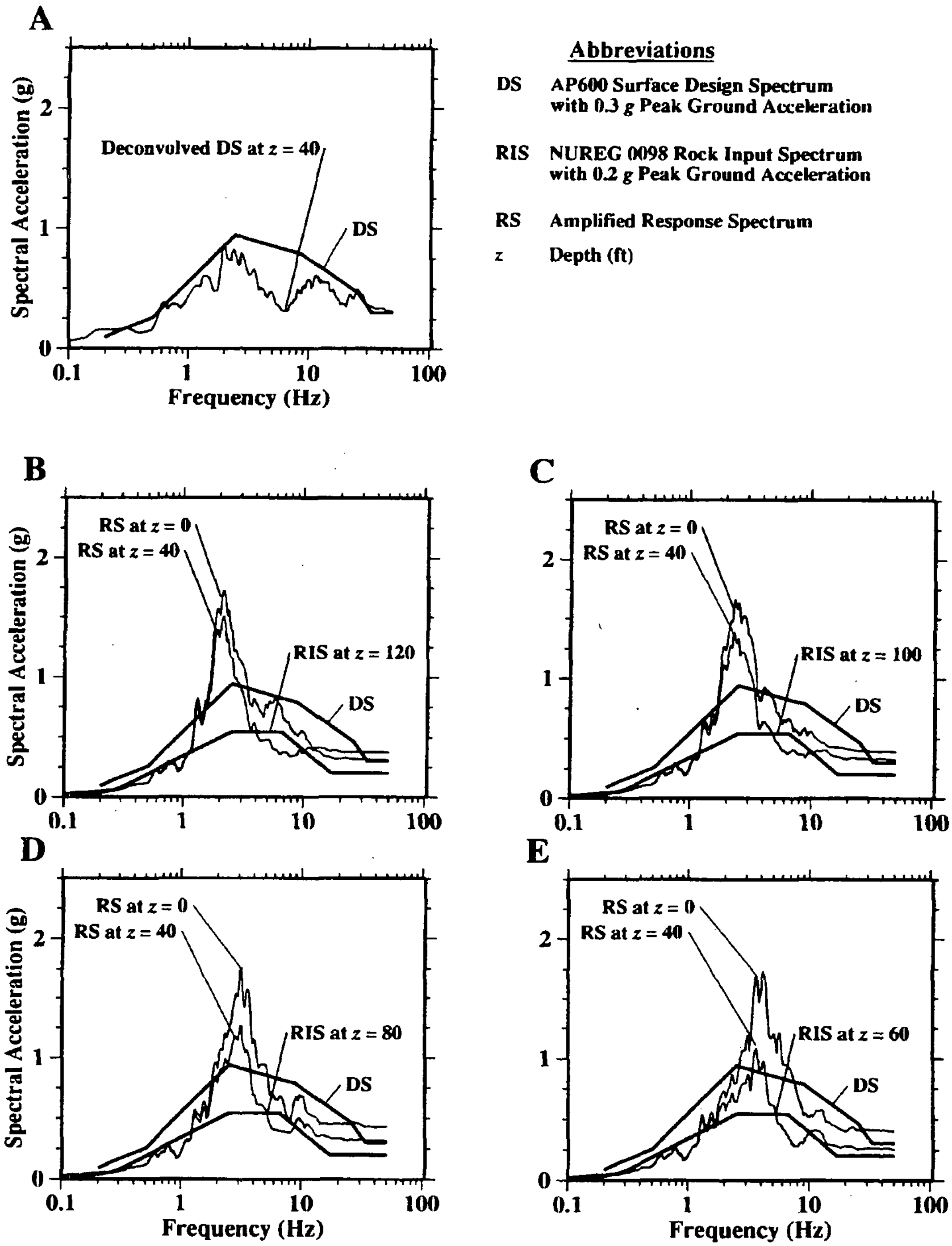
2.5.5 Stability of Slopes

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants will be required to provide site-specific information about the static and dynamic stability of all soil and rock slopes, the failure of which could adversely affect the safety of the plant. This was identified as DSER Open Item 2.5.5-1 and COL Action Item 2.5.5-1. Westinghouse included this requirement in Revision 17 to Section 2.5.5 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.5-1 is closed.

2.5.6 Embankments and Dams

In the DSER, the staff requested that Westinghouse include in the SSAR that the COL applicants will be required to provide site-specific information about the static and dynamic stability of all embankments and dams that will impound water for safe operation and shutdown of the plant. This was identified as DSER Open Item 2.5.6-1 and COL Action Item 2.5.6-1. Westinghouse included this requirement in Revision 17 to Section 2.5.6 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 2.5.6-1 is closed.

Figure 2.5-1 Results of Staff's Analysis for the Horizontal Component of Ground Motion



3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 <u>General</u>

The staff reviewed the information in Section 3.1 of the standard safety analysis report (SSAR) to verify that the AP600 design meets the relevant the general design criteria (GDC) of Appendix A to 10 CFR Part 50.

The staff review of structures, components, equipment, and systems relies, in part, on industry codes and standards that represent accepted industry practices. Unless otherwise noted, the staff found that the codes and standards cited in this report are acceptable.

3.1.1 Elimination of Operating Basis Earthquake from Design Consideration

As a part of its review of Volume III of the Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD), the staff evaluated the use of a single-earthquake design (i.e., elimination of operating basis earthquake (OBE)) for structures, systems, and components (SSCs) and determined that this issue would be reviewed on a plant-specific basis for all ALWRs. In the *Federal Register* Vol. 61, No 239, page 65157, dated December 11, 1996, the NRC amended 10 CFR 50.34 to reference a new Appendix S to Part 50, "Earthquake Engineering Criteria for Nuclear Power Plants." Appendix S, in part, allows use of the single earthquake design by providing the applicant an option to use an OBE value of one-third the maximum vibratory ground acceleration of the safe shutdown earthquake (SSE), and to eliminate the requirement to perform explicit response analyses for the OBE. Section V.B.5 in the Statements of Consideration of this Federal Register Notice contains references to the design criteria that the staff has previously implemented on this issue and documented in the Final Safety Evaluation Reports (FSERs) related to the Design Certification of the General Electric Advanced Boiling Water Reactor (ABWR) and Combustion Engineering System 80+ designs (NUREG-1503 and 1462, respectively). The AP600 design has incorporated the single-earthquake design approach. In a letter to Westinghouse dated April 29, 1994, the staff transmitted an enclosure to request for additional information (RAI) 210.60 that contained the staff's position relative to the types of analyses and information required in the SSAR for the staff to approve the design of SSCs for the AP600 without the OBE. This position contained design criteria identical to that in Issue 1.M of SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and ALWR Designs," subsequently referenced in the December 11, 1996, Federal Register Notice discussed above. The staff concluded that if the AP600 plant design is consistent with the position in the letter dated April 29,1994, the AP600 meets the applicable requirements in 10 CFR Part 50, Appendix S, and the OBE can be eliminated as a design-basis event. The staff's evaluation of this issue in Section 3.12.5.14 of this report concludes that the AP600 SSAR contains criteria that are consistent with the staff's position in the April 29, 1994, letter, and is acceptable. Therefore, the AP600 design for safety-related SSCs meets the applicable requirements in 10 CFR 50, Appendix S, and the OBE can be eliminated as a design-basis event for this plant.

For the ultimate heat sink features and the radwaste buildings, Regulatory Guide (RG) 1.27, Revision 2, and RG 1.143, Revision 1, respectively, recommend a seismic design based on the OBE. Since the OBE has been eliminated as a design requirement for the AP600, the staff's RAI 230.3 requested that Westinghouse provide the seismic design basis for these facilities. This was draft safety evaluation report (DSER) Open Item 3.1.1.4-1. The response to this open item, relative to the ultimate heat sink, is that for the AP600 the ultimate heat sink is the atmosphere. Heat is transferred to the atmosphere by the passive containment cooling system, including the primary containment and shield building, which are seismic Category I and are designed for the SSE. This exceeds the guidelines in RG 1.27 and is acceptable. With respect to RG 1.143, in Appendix 1A of the SSAR, "Conformance With Regulatory Guides," Westinghouse commits to applicable seismic design criteria which exceed the guidelines in this regulatory guide and are acceptable. Therefore, DSER Open Item 3.1.1.4-1 is closed.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

In GDC 2, the NRC requires, in part, that nuclear power plant SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Some of these features are safety-related and necessary to ensure the following:

- the integrity of the reactor coolant pressure boundary (RCPB)
- the capability to shut down the reactor and maintain it in a safe-shutdown condition
- the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures that are comparable to the guidelines in 10 CFR Part 100

The earthquake for which these safety-related plant features are designed is defined as the SSE in Appendix A to 10 CFR Part 100. The SSE is based on an evaluation of the maximum earthquake potential and it is that earthquake that produces the maximum vibratory ground motion for which SSCs are designed to remain functional. Those plant features that are designed to remain functional, if an SSE occurs, are designated seismic Category I in Revision 3 of RG 1.29. In addition, in Regulatory Position C.1 in RG 1.29, the NRC states that the pertinent quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of seismic Category I SSCs. The staff reviewed the AP600 SSAR in accordance with Section 3.2.1 of the standard review plan (SRP), which references RG 1.29. The details of this review are discussed below.

The safety-related SSCs and equipment of the AP600 standard plant that are required to be designed to withstand the effects of an SSE and remain functional, and therefore, be classified as seismic Category I, are identified by comparing information in Sections 3.2.1.2 and 3.2.4, and Tables 3.2-1, 3.2-2, and 3.2-3 of the SSAR, and applicable piping and instrumentation diagrams (P&IDs) in the SSAR. Table 3.2-3, "AP600 Classification of Systems and Components," includes seismic classifications for fluid systems and some components in these systems. However, this table does not explicitly include piping and piping supports. The P&IDs in the SSAR identify the interconnecting piping and valves and the interface between the safety-related and non-safety-related portions of each system. According to Section 3.2.1.2 of the SSAR, these interfaces are synonymous with the interface between seismic Category I and non-seismic portions of each system. In the response to RAI 210.34 dated June 27, 1994, Westinghouse agreed to revise Section 3.2.4 of the SSAR to state that supports for piping and components have the same seismic and safety classifications as the component or piping supported. This was identified as DSER Confirmatory Item 3.2.1-1. In Revision 7 to Section 3.2.4, Westinghouse included this commitment. Therefore, DSER Confirmatory Item 3.2.1-1 is closed. Based on the review of Sections 3.2.1.2 and 3.2.4, Tables 3.2-1, 3.2-1, and 3.2-3, and the P&IDs of the SSAR as discussed above, the staff concludes that the safety-related SSCs in the AP600 are acceptably classified as seismic Category I in accordance with Position C.1 in RG 1.29.

In Position C.2 in RG 1.29, the NRC states that those portions of non-seismic SSCs whose continued function is not required, but whose failure could reduce the functioning of any seismic Category I SSC to an unacceptable level, or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSE would not cause such failure. In Section 3.2.1.1.2 of the SSAR, Westinghouse classifies such SSCs as seismic Category II. The design criteria for seismic Category II SSCs are discussed in Section 3.7 of the SSAR. In Position C.3 in RG 1.29, the NRC addresses recommended guidelines for designing interfaces between seismic Category I and non-seismic SSCs. The AP600 information relative to Positions C.2 and C.3 are provided in Section 3.7.3.13 of the SSAR, and the staff's evaluations of this information for structures and piping are contained in Sections 3.7.2 and 3.12.3.7, respectively, of this report.

In Positions C.1 and C.4 in RG 1.29, the NRC states that the pertinent quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of (1) all seismic Category I SSCs, and (2) those portions of SSCs covered under Positions C.2 and C.3. Sections 3.2.2.3, 3.2.2.4, and 3.2.2.5, and Table 3.2-1 of the SSAR state that 10 CFR 50, Appendix B applies to all AP600 Equipment Class A, B, and C (ASME Class 1, 2, and 3) SSCs, which are all classified as seismic Category I. The staff concludes that this is an acceptable commitment to Item (1) above. However, Westinghouse does not address Item (2) in the SSAR. To satisfy Position C.4 in RG 1.29, the pertinent QA requirements of Appendix B to 10 CFR Part 50 should be applied to all seismic Category II SSCs. This was identified as DSER Open Item 3.2.1-1. In Revision 17 to Section 3.2.1.1.2 of the SSAR, Westinghouse states that pertinent portions of 10 CFR 50, Appendix B apply to seismic Category II SSCs. This is consistent with Position C.4 in RG 1.29 and is acceptable. Therefore, DSER Open Item 3.2.1-1 is closed.

In Sheet 7 of Table 3.2-3 of the SSAR, Westinghouse properly identifies the new and spent fuel storage racks as seismic Category I. Although these items are also classified as AP600 Class D, the staff's position is that the new and spent fuel storage racks are important to safety and, as a minimum, should meet the applicable quality assurance requirements of Appendix B to 10 CFR 50 in addition to being classified as seismic Category I. SSAR Section 3.2, Revision 0, did not contain this commitment. This was identified as DSER Open Item 3.2.1-2. In Revision 9 to the SSAR, Westinghouse revised Section 3.2.2.6 to state that 10 CFR 50, Appendix B applies to AP600 Class D structures, systems, and components that are seismic Category I. The staff concludes that this commitment is consistent with the guidelines in RG 1.29, and is acceptable. Therefore, DSER Open Item 3.2.1-2 is closed.

3.2.1.1 Conclusion

On the basis of its review of Tables 3.2-1, 3.2-2, and 3.2-3, the applicable P&IDs, and other relevant information in the SSAR, as discussed above, the staff concludes that the AP600 safety-related SSCs, including their supports, are properly classified as seismic Category I in accordance with Position C.1 of RG 1.29, and that acceptable commitments to Positions C.2, C.3, and C.4 of RG 1.29 are included in the SSAR. This constitutes an acceptable basis for satisfying, in part, GDC 2, in which the NRC requires that all SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes.

3.2.2 Quality Group Classification

In GDC 1, the NRC requires that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement is applicable to both pressure-retaining and non-pressure-retaining SSCs that are part of the RCPB and other systems important to safety, when reliance is placed on these items to:

- *prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB*
- permit shutdown of the reactor and maintain it in a safe-shutdown condition
- retain radioactive material

In addition to the seismic classifications discussed in Section 3.2.1 of this report, Tables 3.2-1 and 3.2-3 of the SSAR identify the AP600 safety classification, NRC Quality Group (QG) classification, and QA requirements necessary to satisfy the requirements of GDC 1 for all safety-related SSCs and equipment. Applicable P&IDs of the SSAR identify the classification boundaries of interconnecting piping and valves. The staff reviewed Tables 3.2-1 and 3.2-3 and the P&IDs of the SSAR in accordance with Section 3.2.2 of the SRP, which references Revision 3 of RG 1.26 as the principal document used in the staff review for identifying, on a functional basis, the pressure-retaining components of those systems important to safety as NRC QG A, B, C, or D. Conformance of ASME Code, Section III, Class 1 components, that are part of the RCPB, to 10 CFR 50.55a is discussed in Section 5.2.1.1 of this report. These RCPB components are designated in RG 1.26 as QG A; certain other RCPB components that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified as QG B, with the exception of a portion of the chemical and volume control system inside containment, which is QG D. The basis for this alternate quality group classification is discussed in detail in Section 5.2.1.1 of this report.

In 10 CFR 50.55a, the NRC requires that safety-related equipment be designed and fabricated to the requirements of ASME Code Section III. Westinghouse proposed to use the rules of ASME Code, Section VIII, Appendix 22 for the design and construction of the air gas storage tanks in the main control room emergency habitability system. The NRC staff reviewed the proposal and concluded that the use of Appendix 22 to ASME Code, Section VIII provides an acceptable alternative to the requirements of ASME Code, Section III for the design and construction of the air storage tanks. This conclusion is based on the following reasoning:

a) The air storage tanks are constructed of forged, seamless pipe without welding. The material for the integrally forged tanks are ordered to material specification SA-372 which has been specifically developed for forged tanks fabricated without welding.

b) To construct the tanks, the forged pipe ends are swaged down to reduce the size of the opening. After completion of the tank forming operation the tanks are heat treated. No welding is permitted in the fabrication of the tank and the material is not permitted to be weld repaired.

c) Westinghouse also specified that 10 CFR 50, Appendix B and 10 CFR Part 21 will apply to the manufacture of the air storage tanks.

d) The tank material is specified to be Charpy V-notch tested per supplement S3 of material specification SA-372 and it is required to exhibit an average of 20 to 30 mills of lateral expansion at the lowest anticipated service temperature. This value is consistent with the values specified in Table NC-2332.1-1 of ASME Code, Section III. Thus, the proposed alternative provides an acceptable level of quality and safety and is acceptable.

In Section 3.2.2 of the SSAR, Westinghouse describes the AP600 safety classification system. Safety-related SSCs are classified as AP600 Equipment Class A, B, or C. In Table 3.2-1 of the SSAR, Westinghouse provides a correlation between the following three methods of classification: (1) AP600 Class A, B, C, and D, (2) NRC QG A, B, C, and D in RG 1.26, and (3) ASME Code Section III classes. The relationship between the three methods of classification in the SSAR is shown below.

NRC QG	AP600 CLASS	ASME Section III Class
A	A	1
B	B	2
C	C	3
D	D	-

All pressure-retaining components and component supports classified as AP600 Class A, B, or C are constructed in accordance with ASME Code, Section III, Class 1, 2, or 3 rules, respectively. Construction, as defined in Subsections NB/NC/ND-1110(a) of Section III of the ASME Code, and as used herein, is an all-inclusive term comprising of materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components. Components classified as QG D are designed to the applicable standards identified in Section 3.2.2.6 of the SSAR. The staff concludes that the above table, which reflects information in the SSAR as discussed above, is acceptable for defining the relationship between the three methods of classification.

The staff's review of the information in Section 3.2.2, Table 3.2-1, Table 3.2-3, and applicable P&IDs of the SSAR concludes that QG classifications for the majority of the AP600 SSCs are consistent with the guidelines in RG 1.26, and are, therefore, acceptable. However, the DSER discussed several issues that remained unresolved. Listed below is the status of these issues through Revision 19 to the SSAR.

- Safety Classification of Passive Core Cooling System (PXS)

  Section 3.2.2.5, Table 3.2-3, and P&IDs in Figures 6.3-1 and 6.3-2 of the SSAR collectively identify the following portions of the PXS as AP600 Class C (QG C and ASME Class 3):

  - the accumulators and vessel injection piping system up to the ASME Class 1 check valves

  - the vessel injection piping system from the in-containment refueling water storage tank (IRWST) to the ASME Class 1 check valves

  - the injection piping system from the containment sump to the vessel injection piping coming from the IRWST

  All of the above systems and components perform an emergency core cooling function following postulated design-basis events. In RG 1.26, the NRC recommends that such systems be classified as QG B (ASME Class 2). In the responses to RAI 210.1 and 210.29 dated December 22, 1992, and June 27, 1994, respectively, Westinghouse stated that the basis for classifying these systems and components as QG C is as follows:

  - QG C is essentially equivalent to QG B, except that it has less stringent construction inspection and inservice inspection (ISI) rules.

  - All of these systems and components are located inside containment, therefore, activity releases are contained.

  - Minor leakage does not affect the functional performance of these systems and components.

  - There is continuous water level monitoring of the accumulators and IRWST that detects leaks.

  - The QG C classifications agree with the recently developed standard ANS-58.14, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors."

  After evaluating the first four items above, the staff concluded that the classifications of the PXS and components identified in the first paragraph above can satisfy the guidelines in RG 1.26 if the QG C classification remains and a commitment is made in the SSAR that, during construction, portions of these systems will be inspected to rules that are similar to ASME Class 2 (QG 2) rules. The basis for this staff position is that the enhanced quality of the items inspected to ASME Class 2 rules is sufficient to satisfy the guidelines of RG 1.26, and the features described in the first four items above are sufficient to allow the less stringent in-service inspection rules of ASME Class 3 (QG C). This was identified as DSER Open Item 3.2.2-1. In a meeting on July 25, 1995, Westinghouse proposed to resolve this issue by revising Section 3.2.2.5 of the SSAR to state that for AP600 Class 3 lines that provide an ECCS function, the welds will be required to be spot radiographed. The staff determined that this commitment would result in a piping system whose construction is somewhat enhanced. However, the staff finds that the weld quality for ECCS needs to be consistent with the system's safety functions. Therefore, in a letter to Westinghouse dated December 4, 1997, (RAI 210.235F), the staff requested that the butt welds in ECCS piping should be examined in accordance with the rules of ASME Section III, ND-5222, using only the full radiography option. The commitment to fully radiograph welds is made in Section 3.6.3.2 of the SSAR for the ASME Class 3 accumulator discharge piping as a part of the staff's evaluation of the leak-before-break (LBB) issue. This is discussed in Section 3.6.3.5 of this report. Therefore, the staff's position is that in order to provide reasonable assurance that the affected systems will perform their safety function when required, Section 3.2.2.5 of the SSAR should be revised to provide a similar commitment for the remaining of the ECCS welds in the systems that are listed in Section 3.2.2.5. In Revision 19 to Section 3.2.2.5 of the SSAR, Westinghouse states that for systems that provide emergency core cooling 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. This conforms to the staff's position on this issue and is acceptable. Therefore, DSER Open Item 3.2.2-1 is closed.

With respect to the last item above, the staff has not reviewed or endorsed the ANS-58.14 standard for the design of passive plants. Therefore, in the DSER, the staff requested that the reference to this standard in the response to RAI 210.29 be deleted. This was identified as DSER Open Item 3.2.2-2. In a meeting on July 25 and 26, 1995, Westinghouse explained that the inclusion of ANS-58.14 in the response to RAI 210.29 was only intended to be an aid to explain the AP600 safety classification approach rather than a justification for the classifications in the SSAR. In addition, ANS-58.14 is not referenced in the SSAR. Therefore, since this standard is not referenced in the SSAR, it was not included as input to the staff's evaluation of quality group classification for the AP600. On this basis, DSER Open Item 3.2.2-2 is closed.

- Safety Classifications of Piping and Supports

In RAI 210.34, the staff requested that Table 3.2-3 of the SSAR include safety classifications of piping and supports for components and equipment. In the response dated June 27, 1994, Westinghouse stated that the safety classification for piping is contained in applicable P&IDs. For supports, the response committed to revise Section 3.2.4 of the SSAR to include a statement that supports for piping and components have the same safety and seismic classification as the piping or component being supported. This commitment was added to Section 3.2.4 in Revision 7. The staff concludes that this commitment is consistent with the applicable guidelines in RG 1.26 and is acceptable. The staff further concludes that because this information is in Section 3.2.4, it does not have to be included in Table 3.2-3 of the SSAR. This was part of DSER Confirmatory Item 3.2.1-1 which is closed as discussed in Section 3.2.1 of this report.

- Safety Classification of the Normal Residual Heat Removal System (RNS)

  In RAI 210.37, the staff requested that Section 5.4.7.1.2 of the SSAR be revised to clarify a statement that did not appear to be consistent with the information relative to safety classification in Table 3.2-3 and Figure 5.4-7 of the SSAR. In the response dated June 16, 1994, Westinghouse agreed to revise Sections 5.4.7.1.1 and 5.4.7.1.2 of the SSAR to include more explicitly that the portion of the RNS outside containment (between the outside containment isolation valves) is classified as AP600 Class C. As discussed above, this requires that these components be constructed to ASME Class 3 rules. The staff concluded that this response results in consistent information in the SSAR between Table 3.2-3, Section 5.4.7, and Figure 5.4-7, and is acceptable. This was identified as DSER Confirmatory Item 3.2.2-1. In Revision 5 to the SSAR, the above response was incorporated into Sections 5.4.7.1.1 and 5.4.7.1.2. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.2.2-1 is closed.

- Safety Classification of Supports for the Passive Residual Heat Removal Heat Exchanger

  In the response to RAI 210.38 dated June 27, 1994, Westinghouse agreed to revise Section 5.4.14.1 of the SSAR to reclassify the supports for the passive residual heat removal heat exchanger from AP600 Class C to AP600 Class A. This commitment is consistent with the response to RAI 210.34 discussed above under "Safety Classifications of Piping and Supports." The staff concluded that this response is acceptable because it agrees with the staff's position that component supports should have the same safety classification as the supported component. This was identified as DSER Confirmatory Item 3.2.2-2. In Revision 5 to the SSAR, the above commitment was added to Section 5.4.14.1. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.2.2-2 is closed.

- Safety Classification Interfaces in the Passive Residual Heat Removal Heat Exchanger Design

  In the response to RAI 210.39, Westinghouse agreed to revise Section 6.3.2.2.5 of the SSAR and add Figure 6.3-5, "Passive Heat Removal Heat Exchanger ASME Code Classification and Boundary" to the SSAR to (1) more clearly identify the safety classification interface between the AP600 Class A passive residual heat removal heat exchanger and the AP600 Class C IRWST, and (2) to provide a brief description of the heat exchanger inlet and outlet designs. This was identified as DSER Confirmatory Item 3.2.2-3. In Revision 2 to the SSAR, Westinghouse added this information. The staff concludes that the information included in this revision of the SSAR adequately describes the interfaces between the AP600 Class A heat exchanger inlet and outlet channel heads, tubes, tubesheet, and supports, and the AP600 Class C IRWST. This includes an acceptable description of the jurisdictional boundary between the IRWST building structure and the ASME Class 1, Subsection NF supports. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.2.2-3 is closed.

- In the response to RAI 210.71 dated June 16, 1994, Westinghouse agreed to revise Table 3.2-3 of the SSAR, to add the core barrel to Sheet 53, "Reactor System." This was identified as DSER Confirmatory Item 3.2.2-4. In Revision 11 to the SSAR, the core barrel and core barrel nozzle were added to Table 3.2-3. In this table, the core barrel and nozzle are classified as AP600 Class C, seismic Category I, and are identified as being constructed to the requirements in Subsection NG of Section III of the ASME Code. The staff concludes that this information is consistent with applicable guidelines in Section 3.9.5 of the SRP and is acceptable. Therefore, DSER Confirmatory Item 3.2.2-4 is closed.

- In a letter to Westinghouse dated August 25, 1997, the staff identified a concern relative to the quality group classification of a portion of the chemical volume and control system (CVS) inside containment. Because this issue is related to the exclusion requirements in 10 CFR 50.55a(c)(2), the staff's evaluation and resolution of the response to the above letter is discussed in Section 5.2.1.1 of this report.

3.2.2.1 Conclusion

On the basis of reviews of applicable information in the SSAR, and the above discussions, the staff concludes that the quality group classifications of all pressure-retaining and non-pressure-retaining SSCs important to safety that are identified in Tables 3.2-1 and 3.2-3, and in related P&IDs of the SSAR, are in conformance with RG 1.26, and with applicable staff positions on previously licensed pressurized water reactor (PWR) plants, and are, therefore, acceptable. These tables and P&IDs identify major components in fluid systems (such as pressure vessels, heat exchangers, storage tanks, piping, pumps, valves, and applicable supports) and in mechanical systems (such as cranes, fuel handling machines, and other miscellaneous handling equipment). In addition, P&IDs of the SSAR identify the classification boundaries of interconnecting piping and valves. All of the above SSCs will be constructed in conformance with applicable ASME Code and industry standards. Conformance to RG 1.26, previous staff positions, and applicable ASME Codes and industry standards provide assurance that component quality will be commensurate with the importance of the safety function of these systems. This constitutes the basis for satisfying GDC 1 and is, therefore, acceptable.

3.3 Wind and Tornado Loadings

3.3.1 Wind Design Criteria

The design-basis wind is specified in the Section 3.3.1.1 of the SSAR and is established on the basis of the basic wind speed of 177 km/hr (110 mph) at an elevation of 10.1 m (33 ft) above grade with a recurrence interval of 50 years. This basic wind speed is to be scaled by an importance factor (as defined in ANSI/ASCE 7-88, "Minimum Design Loadings for Buildings and Other Structures") of 1.0 and 1.11 for non-safety-related and safety-related structures, respectively.

The importance factor, I, provides basic wind speeds associated with mean intervals of 100 years (annual probability of being exceeded equal to 0.01). The basic wind speed values of the map in the ANSI/ASCE 7-88 (formerly ANSI A58.1-1982) are for a 50-year mean recurrence interval (annual probability of 0.02). In ANSI A58.1, it is stated that the basic wind speeds associated with a 100-year mean recurrence interval be used for the design of buildings and other structures where a high degree of hazard to life and property exists and when these buildings or other structures are considered to be essential facilities. The guidelines in

Section 3.3.1 of the SSAR refer to the design wind speed as the 100-year return period "fastest mile of wind." Therefore, the use of an importance factor of 1.11 is acceptable because it adjusts the recurrence interval from 50 to 100 years for the design of safety-related structures.

Westinghouse established an exposure category that adequately reflects the characteristics of ground surface irregularities for the site at which the building or structure is to be constructed. In establishing the exposure category, Westinghouse considered large variations in ground surface roughness that arise from natural topography and vegetation as well as constructed features. In Section 3.3.1.1 of the SSAR, Exposure D is used to compute the velocity pressure exposures and the gust response factors for all seismic Category I structures. The use of Exposure D is acceptable because it represents flat, unobstructed areas exposed to wind flowing over large bodies of water and is more conservative than Exposure C specified in the Section 3.3.1 of the SRP for open country.

Therefore, all seismic Category I structures within the AP600 design that will be exposed to wind forces are designed to withstand the effects of the design wind which has a velocity of 196.5 km/hr (122.1 mph) based on a recurrence period (as specified in the Section 3.3.1 of the SRP) of 100 years and Exposure D.

The procedures used in Sections 3.3.1.1 and 3.3.1.2 of the SSAR to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with ANSI/ASCE 7-88, and shape coefficients for the shield building are calculated using American Society of Civil Engineers (ASCE) Paper 3269 (1961). The plant design with respect to the capability of the structures to withstand design wind loadings is acceptable and meets the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading.

The design reflects the following considerations, as described in Section 3.3.1 of the SRP:

- appropriate consideration for the most severe wind not to exceed the velocities presented in Table 2.0-1 of the SSAR for future sites
- appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena
- the importance of the safety function to be performed

These requirements are being met by the use of ANSI/ASCE 7-88 and ASCE Paper 3269 as specified in Sections 3.3.1.1 and 3.3.1.2 of the SSAR to transform the wind velocity into an effective pressure on structures and for selecting pressure coefficients corresponding to the structural geometry and physical configuration.

Westinghouse designed the plant structures with sufficient margin to prevent structural damage during the most severe wind loadings that were determined appropriate for the wind velocities mentioned above. In addition, the design of seismic Category I structures includes, in an acceptable manner, load combinations that occur as a result of the most severe wind load and the loads resulting from normal and accident conditions.

The procedures used in Sections 3.3.1.1 and 3.3.1.2 of the SSAR to determine the loadings on structures induced by the design-basis wind specified for the plant are acceptable because they meet the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading and have been successfully used in the design of conventional structures. The procedures, therefore, provide a reasonable basis that, together with other engineering design considerations, such as the combination of wind load with other loads as indicated in Section 3.8.4 of this report, ensures that the structures will withstand such environmental forces. The use of these procedures provides reasonable assurance that, in the event of design-basis winds, the structural integrity of the plant structures that have to be designed for the design wind will not be impaired and, consequently, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions, if needed.

Because the collapse of non-safety-related structures should not have any adverse impact on nearby safety-related SSCs, the NRC staff requested in RAI 220.24 that Westinghouse discuss the effects of wind-induced failure of non-safety-related structures on safety-related SSCs. If the collapse of non-safety-related SSCs due to wind loading does not adversely impact the function of the safety-related SSCs, the use of 1.0 as the importance factor is suitable. For the design of safety-related SSCs, an importance factor of 1.11 should be used. All SSCs not designed for wind loads should be analyzed using the 1.11 importance factor or checked that their failure mode will not affect the ability of safety-related SSCs to perform their intended safety functions. In the response to RAI 220.24 dated April 14, 1994, Westinghouse stated that wind-induced failure of non-safety-related SSCs due to the design-basis wind of 177 km/hr (110 mph) does not adversely impact the function of the safety-related SSCs. Wind-induced failure of non-safety-related SSCs is evaluated for the 483 km/hr (300 mph) tornado which is more severe than the design-basis wind. However, this evaluation requirement for the non-safety-related SSCs was not provided in the SSAR. In the DSER, the staff requested that Westinghouse provide a commitment in the SSAR that non-safety-related SSCs be designed such that wind and tornado loads should not cause their failure and impact nearby seismic Category I SSCs. This was identified as DSER Open Item 3.3.1-1.

In Revision 2 to Section 3.3.2.3 of the SSAR, Westinghouse states that the structures adjacent to the nuclear island are the annex building, the radwaste building, and the turbine building. The portion of the annex building adjacent to the nuclear island is classified as seismic Category II and is designed to seismic Category I structure tornado loading. The radwaste building is a small steel-frame building. If it were to collapse in the event of a tornado, it would not impair the integrity of the reinforced concrete nuclear island. The turbine building is classified as non-seismic and is designed to the same tornado loading as seismic Category I structures.

If external events produced missiles beyond those postulated for the design (equivalent to Spectrum I missiles specified in SRP Section 3.5.1.4), Section 3.5.4 of the SSAR requires the COL applicant to evaluate these events and provide the design features necessary to protect safety-related SSCs from the identified hazard.

Tornado loads are usually less significant than seismic loads, and similar issues were raised and are discussed in Section 3.7.2 of this report. On this basis, the staff finds that these requirements will provide adequate bases to ensure that failures of non-safety-related SSCs do not adversely impact nearby seismic Category I SSCs and are acceptable. Therefore, DSER Open Item 3.3.1-1 is closed.

In addition, the staff stated in the DSER that certain non-safety-related structures that house systems and components identified as important in the probabilistic risk assessment (PRA) (e.g., the diesel generator building and service water intake structures) should be designed with the importance factor of higher than 1.0 to protect those non-safety-related systems and components identified as important in the regulatory treatment of non-safety systems (RTNSS) process. This was identified as DSER Open Item 3.3.1-2.

In Revision 2 to Section 3.5 of the SSAR, Westinghouse states that there are no systems or components identified as important in the evaluation of RTNSS that require protection from missiles. Therefore, DSER Open Item 3.3.1-2 is not applicable for the AP600 design and is closed.

3.3.2 Tornado Loading

3.3.2.1 Tornado Loads on Exterior Structures

The staff position with regard to design-basis tornados was previously based on two documents published in 1974: WASH-1300 and RG 1.76. According to WASH-1300, the probability of occurrence of a tornado that exceeds the design-basis tornado should be on the order of 1.0E-7 per year for each nuclear power plant. RG 1.76 delineates the maximum wind speeds of 579 km/hr (360 mph) for the contiguous United States.

The staff re-evaluated the regulatory positions in RG 1.76 using the considerable quantity of tornado data which became available since the RG was developed. The re-evaluation is discussed in NUREG/CR-4461. The staff's interim position ("ALWR Design Basis Tornado") on RG 1.76 was issued on March 25, 1988. In this interim position, the staff concluded that the maximum tornado wind speed of 531 km/hr (330 mph) is acceptable. However, in SECY-93-087, the staff recommended that the Commission approve its position that a design-basis tornado with a maximum tornado wind speed of 483 km/hr (300 mph) be employed in the design of evolutionary and passive ALWRs. In its staff requirements memorandum (SRM) dated July 21, 1993, the Commission approved the staff position.

Westinghouse indicates in Section 3.3.2.1 of the SSAR, that all seismic Category I structures exposed to tornado forces and needed for the safe shutdown of the plant are designed to resist tornado loads, and the tornado missile spectrum (Spectrum I) is in accordance with Section 3.5.1.4 of the SRP. Section 3.3.2.1 of the SSAR specifies a maximum tornado wind speed of 483 km/hr (300 mph), a maximum rotational tornado wind speed of 386 km/hr (240 mph), and a maximum translational tornado wind speed of 97 km/hr (60 mph). Also specified are a simultaneous atmospheric pressure drop to 13.8 kPa (2.0 psi) at the rate of 8.3 kPa/sec (1.2 psi/sec) and the radius of 45.7 m (150 ft). Because these parameters meet the design-basis tornado requirements specified in SECY-93-087 as approved in the July 21, 1993, SRM, the staff concludes that the Westinghouse AP600 design-basis tornado is acceptable.

The procedures used to transform the tornado wind velocity into pressure loadings are the same as those used for wind as discussed in Section 3.3.1 of this report. The tornado missile effects are determined using procedures discussed in Section 3.5 of the SSAR and the acceptability of these procedures is given in Section 3.5 of this report. The tornado loadings include tornado wind pressure, internal pressure by tornado-created atmospheric pressure drop, and forces generated by the impact of tornado missiles. These loadings are combined with other loads as described in Section 3.8.4 of the SSAR. The acceptability of these loading combinations is discussed in Section 3.8.4 of this report.

3.3.2.2 Effects of Failure of Non-Safety Structures

Westinghouse states in Section 3.3.2.3 of the SSAR that the failure of structures not designed for tornado loadings does not affect the capability of seismic Category I structures or the performance of safety-related systems. This can be ensured by accomplishing one of the following:

- designing the adjacent non-safety-related structure to the design-basis tornado loading
- investigating the effect of failure of adjacent structures on seismic Category I SSCs to determine that no impairment of safety function results
- designing a structural barrier to protect seismic Category I SSCs from adjacent structural failure

In the DSER, the staff requested that Westinghouse provide a commitment in Section 3.3.2.3 of the SSAR that the COL applicant follow the above criteria to ensure that the collapse of non-seismic Category I structures will not impair the functions of seismic Category I SSCs. This was identified as DSER Open Item 3.3.2.2-1 and COL Action Item 3.3.2.2-1.

In Revision 2 to Section 3.3.3 of the SSAR, Westinghouse requires that the COL applicant address the site interface criteria for wind and tornado. The staff finds this acceptable, and therefore, DSER Open Item 3.3.2.2-1 is closed.

In Section 2.2 of the SSAR, Westinghouse requires that the COL applicant ensure that accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels will be considered for facilities and activities in the vicinity of the plant where such materials are processed, stored, used, or transported in quantity. On the basis of RG 1.91, with respect to the protection against dynamic effects resulting from equipment failure occurring outside of the nuclear power plant, this is acceptable.

3.3.2.3 Tornado Loads on Containment Shell and Air Baffle

For the air baffle, the staff raised a question on the turbulent air flow due to wind load and its impact on the containment structure. Also, in RAI 220.63, the staff requested the following information:

- the magnitude, distribution and number of cycles of the stresses induced by the wind on the fatigue aspects of the containment shell design
- the consideration of the potential for tornado missiles to impact the air baffle

- the protection of the air deflector against tornado missiles

Westinghouse informed the staff that wind tunnel tests were performed on the models with these components and the results of the tests are reported in WCAP-13323-P. The staff reviewed the data from these tests and found them acceptable. A detailed discussion of the tests and the staff's evaluation are in Sections 21.3.7 and 21.5.7 of this report. These tests results were used to determine wind pressure loads on the containment vessel and air baffle for the design-bases wind and tornado.

Westinghouse considered the impact of reduced pressure on the outside of containment in the annulus and above the dome. This reduced pressure is equivalent to an increase in containment internal pressure. The maximum pressure reductions on the portion of the containment vessel supporting the air baffle (cylinder and part of the knuckle region) are calculated by Westinghouse to be 4.48 kPa (0.65 psi) for the design-basis wind and 9.58 kPa (1.39 psi) for the design-basis tornado. These pressures act on the outer surface of the air baffle. These pressures are small in comparison with the design pressure of 411.62 kPa (45 psig). No fatigue consideration of the containment shell design due to wind-induced stresses is deemed necessary because of the low frequency of tornado load and low stresses caused by it. Therefore, the containment design for these reduced pressures is acceptable to the staff.

On the cylindrical portion, the maximum pressure reductions are 3.38 kPa (0.49 psi) and 3.52 kPa (0.51 psi) at the inner annulus and outer annulus, respectively. The pressure inside the inner annulus acts on both the air baffle and the outside of the containment vessel and these loads are resisted by the air baffle supports. The staff finds these reduced pressures to be adequate for the design on the basis of the test results specified in WCAP-13323-P. These reduced pressures are considered in the design of the containment (see Section 3.8.2 of this report) and air baffle (see Section 3.8.4 of this report).

Westinghouse also investigated the impact of resultant wind load on the containment. This load achieves its maximum value at a location opposite the air intakes where positive pressures occur on the windward side and negative pressures occur on the leeward side. Total wind loads across the vessel opposite the air intakes are calculated as 105.4 and 224.5 newtons per meter of height (2,170 and 4,619 pounds per foot of height) on the containment vessel due to the design wind and the tornado, respectively. The staff finds these total wind loads to be adequate for design on the basis of the test results specified in WCAP-13323-P. These loads are considered in the design of the containment (see Section 3.8.2 of this report) and air baffle (see Section 3.8.4 of this report).

Westinghouse stated that the air baffle is designed for the wind and pressure loads from the tornado and hence it will not fail and generate missiles. Further, the air baffle is protected from tornado missiles by the shield building. The upper portion of the air baffle may be subjected to missile impact by missiles that could pass through the air inlets. This portion of the air baffle is constructed from 0.64-cm (0.25-in.) thick plate, which would stop small missiles but would experience local damage from the large tornado missiles. Such damage would not prevent function of the air baffle. The acceptability of the air baffle design is discussed in Section 3.8.4 of this report.

3.3.2.4 Compliance with Regulatory Provisions

The AP600 meets the requirements of GDC 2 and the guidelines of Section 3.3.2 of the SRP with respect to the structure capacity to withstand design tornado wind loading and tornado missiles so that their design reflects the following requirements:

- appropriate consideration of the most severe tornado characterized by the tornado parameters mentioned above
- appropriate combinations of the effects of this severe natural phenomena with those resulting from normal plant operation
- the importance of the safety function to be performed

For the design of safety-related structures, these requirements are met by using criteria specified in SECY-93-087 and the methods of transforming the tornado wind velocity into an effective pressure on structures as described in Section 3.3.1 of the SSAR.

By using design loads and load combinations to meet the guidelines of Section 3.8 of the SRP, the plant structures are designed with sufficient margin to prevent the failure of structures during the most severe tornado loads, so the requirements of the first item above are met. In addition, the design of seismic Category I structures as required by the second item above includes, in an acceptable manner, load combinations of the most severe tornado load and loads resulting from normal plant operation. The procedures utilized to determine the loads on structures induced by the design-basis tornado specified for the plant have been used in the design of conventional structures for most severe winds which, together with other engineering design considerations, will ensure that the structures will withstand such severe environmental forces.

The use of these procedures gives reasonable assurance that, in the event of a design-basis tornado, the structural integrity of applicable plant structures will not be impaired and, consequently, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed; thus, satisfying the third item above.

The staff concludes that the AP600 design is acceptable and meets the requirements of GDC 2 and the guidelines of Section 3.3.2 of the SRP.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

The staff reviewed the AP600 flood design in accordance with Section 3.4.1 of the SRP. Staff acceptance of the flood design is based on the design meeting the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Section IV.C, "Required Investigation for Seismically Induced Floods and Water Waves," as they relate to protecting safety-related SSCs from the effects of floods. Acceptance is based on meeting the guidelines of RG 1.59, "Design Basis Floods for Nuclear Power Plants," with regard to the methods used for establishing the probable maximum flood (PMF) and probable maximum precipitation (PMP) and the guidelines of RG 1.102, "Flood Protection for Nuclear Power Plants," with regard to the means used for protection of safety-related SSCs from the effects of the PMF and PMP. The review addressed the overall flood protection design, including safety-related SSCs whose failure as a result of flooding could prevent safe shutdown or result in an uncontrolled release of radioactivity.

In Section 3.4.1 of the SSAR, Westinghouse discusses the flood protection measures that are applicable to the AP600 design for postulated external flooding and internal flooding from system and component failures. The seismic Category I SSCs identified in Section 3.2 of the SSAR are designed to withstand the effects of flooding due to natural phenomena or postulated component failures. None of the non-safety-related SSCs were found to be important to the RTNSS process, based on flooding considerations. As a result, non-safety-related SSCs, including systems found to be important by the RTNSS process and defense-in-depth (DID) systems, are not important in the mitigation of flood events and are not required to be protected from either internal or external flooding.

Based on this information, the staff concludes that Westinghouse has identified the safety-related SSCs which require protection from internal and external floods.

3.4.1.1 External Flooding

The maximum flood level generally includes PMF generated by PMP or other combinations of less severe environmental and man-made events, along with seismic and wind effects. In Section 2.4 of the SSAR, Westinghouse states that the AP600 is designed for a normal groundwater elevation up to 29.9 m (98 ft) (which is 0.6 m (2 ft) below grade elevation) and for a PMF up to grade elevation. Although the grade elevation is defined as 30.5 m (100 ft), the actual grade will be a few inches lower to prevent surface water from entering doorways. The PMF results from site-specific events such as river flooding, upstream dam failure, or other natural causes. The COL applicant will evaluate events leading to potential flooding and demonstrate that the site meets interface requirements. If these criteria cannot be met because of site-specific flooding hazards, the COL applicant may propose protective measures (as described in Section 2.4 of the SSAR).

External flooding does not occur from PMP. AP600 roofs do not have drains or parapets and are sloped such that rainfall is directed towards gutters along roof edges. Therefore, ponding on the roof tops does not occur. Water from roof drains and/or scuppers flow to catch basins, underground pipes, or open ditches by sloping site yard areas. Table 2-1 in the SSAR, "Site Parameters," defines PMP as 49.3 cm/hr (19.4 in/hr) and the maximum static roof load due to snow and ice buildup as 3.6 kPa (75 lb/ft-sq). The roofs are designed for snow loads in accordance with ASCE 7-88 (formerly ANSI A58.1-82), "Minimum Design Loads for Building and Other Structures."

Westinghouse identified the following components that are postulated to be sources of external flooding:

(1) two fire water tanks (1230.3 and 1514.2 kL (325,000 and 400,000 gallons)), which are located near the turbine building (Section 9.5.1 of the SSAR)

(2) the condensate storage tank (1135.6 kL (300,000 gallons)), which is located near the turbine building (Section 9.2.4 of the SSAR)

(3) the demineralized water storage tank (378.5 kL (100,000 gallons)), which is located near the annex building (Section 9.2.4 of the SSAR)

(4) the boric acid storage tank (234.7 kL (62,000 gallons)), which is located next to the demineralized water storage tank (Section 9.3.6 of the SSAR)

(5) two diesel fuel oil tanks (each with 378.5 kL (100,000 gallons)), which are not located near structures housing safety-related equipment and include dikes to retain leaks and spills. (SSAR Section 9.5.4 of the SSAR)

Failure of the cooling tower, service water piping, or circulating water piping also constitute potential sources of external flooding. However, they are not located near structures housing safety-related equipment and all are bounded by the analysis provided in Section 10.4.5 of the SSAR.

AP600 safety-related systems and components are housed exclusively in seismic Category I structures (i.e., containment and auxiliary buildings). Seismic Category I structures are located such that the land slopes away from the structures. This assures that external flood water will drain away from the structure and prevent water pooling near the structure. In addition, and as stated previously, the actual grade is a few inches lower than building entrances to prevent surface water from entering doorways.

The portions of seismic Category I structures located below the grade elevation are protected from external flooding by waterstops and a waterproofing system. Crystalline waterproofing material is applied to both vertical and horizontal exterior surfaces below grade. Waterstops are installed in exterior construction joints below grade.

The AP600 design minimizes the number of penetrations through exterior walls below grade. Penetrations below the maximum flood level (Elevation 100') will be watertight. Process piping and electrical raceway that penetrate an exterior wall below grade either will be embedded in the wall or will be welded to a steel sleeve embedded in the wall. Exterior walls are designed for maximum hydrostatic loads as are penetrations through the walls. Below grade there are no access openings or tunnels penetrating the exterior walls of the nuclear island, which consists of the containment and auxiliary buildings.

The base mat and exterior walls of seismic Category I structures are designed to withstand the maximum lateral and buoyancy forces associated with the PMF and high groundwater level. Hydrodynamic forces were not considered in the structural design because the PMF and high groundwater level are below the finished grade.

In RG 1.59, the NRC discusses the design-basis floods that nuclear power plants should be designed to withstand without loss of capability to achieve and maintain a cold shutdown condition. In Position 1 of RG 1.59, the NRC states that the conditions resulting from the worst-probable site-related flood at a nuclear power plant, with attendant wind-generated wave activity, constitutes the design-basis flood condition from which safety-related SSCs must be protected. AP600 safety-related SSCs are designed to withstand the effects of external flooding in accordance with the criteria of Position C.1 of RG 1.59.

In Section 2.4 and Table 2-1 of the SSAR, Westinghouse provides design-basis flood information, as discussed above. The COL applicant will verify that the site-specific flood conditions are within the interface parameters assumed in the AP600 design. If the site-specific conditions exceed those assumed for the design, then the COL applicant will provide additional protective features to ensure that safety-related SSCs are protected from the additional flood hazards.

Based on this information, the staff concludes that Westinghouse has identified the design-basis flood assumed for the AP600 design and provided adequate guidance for the COL applicant to ensure that safety-related SSCs will be adequately protected from the worst-probable site-related flood conditions. Therefore, the staff concludes that the AP600 design conforms to the guidelines of Position C.1 of RG 1.59.

In Position C.2 of RG 1.59, the NRC provides alternate guidance for flood protection when the "hardened protection" method is not used. The hardened protection method requires that passive structural provisions be incorporated into the plant design to protect safety-related SSCs from the static and dynamic effects of floods. AP600 reinforced concrete seismic Category I structures, incorporating the waterproofing and sealing features previously described, provide hardened protection for safety-related SSCs as defined in RG 1.59. Therefore, it is not necessary to utilize Position C.2 of RG 1.59 for the flood design.

In RG 1.102, the NRC describes the types of flood protection acceptable to the NRC staff for safety-related SSCs. In Position C.1 of RG 1.102, the NRC provides definitions of the various types of flood protection acceptable to the staff. One of the acceptable methods of flood protection incorporates a special design of walls and penetrations. The walls are reinforced concrete, designed to resist the static and dynamic forces of the design-basis flood and incorporates waterstops at construction joints to prevent in-leakage. Penetrations are sealed and also capable of withstanding the static and dynamic forces of the design-basis flood. The AP600 flood design incorporated these protective features. Therefore, the staff concludes that the flood design conforms with the guidelines of Position C.1 of RG 1.102.

In Position C.2 of RG 1.102, the NRC discusses the technical specification and emergency operating procedures necessary to utilize position C.2 of RG 1.59. However as discussed above, Position C.2 of RG 1.59 does not apply to the AP600 flood design, which incorporates hardened protection. Consequently, Position C.2 of RG 1.102 is not applicable as well.

Based on the evaluation of the information provided in the SSAR, the staff concludes that Westinghouse adequately characterized the PMP and PMF for the AP600 flood design and provided design features to protect safety-related equipment from external flood effects associated with the PMP, PMF, groundwater seepage, and system and component failures. Therefore, as applicable, the flood design meets the guidelines of RG 1.59 with regard to the methods used for establishing the PMF and PMP and meets the guidelines of RG 1.102 with regard to acceptable external flood protection methods.

3.4.1.2 Internal Flooding

Safety-related systems and components are located in the containment and auxiliary buildings. Redundant safety-related systems and components are physically separated from each other as well as from non-safety-related components. Therefore, the failure of a system or component may render one division of a safety-related system inoperable while the redundant division is available to perform its safety function. Other protective features used to minimize the consequences of internal flooding are as follows:

- structural enclosures
- structural barriers
- curbs and elevated thresholds
- systems and components used for leakage detection
- drainage systems

In the SSAR, Westinghouse included the results of internal flooding analysis which described the consequences of compartment flooding for various postulated component failures. The analysis included the following tasks:

- identification of flood sources
- identification of essential equipment in each area
- determination of maximum flood levels
- evaluation of flood effects on essential equipment

The flood sources which were considered in the analysis are as follows:

- high-energy piping (breaks and cracks)
- moderate-energy (breaks and through-wall cracks)
- pump mechanical seal failures
- storage tank ruptures
- actuation of fire suppression systems
- flow from upper elevations and adjacent areas

The criteria of Section 3.6 of the SSAR, "Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping," were used to define break and crack configurations and locations for both high- and moderate-energy fluid piping failures. In addition, storage tanks were assumed to fully discharge their inventory when a postulated tank rupture was assumed. Except for floor drains, no credit was taken for non-safety-related equipment.

Because the PMF for AP600 design is less than grade elevation, the exterior doors are not required to be watertight for protection from external flooding. There are no watertight doors used for internal flood protection because they are not needed to protect safe shutdown components from the effects of internal flooding. Safety-related equipment is located above the maximum anticipated flood levels for the area. Interior walls are designed to withstand the maximum hydrostatic loads associated with the maximum flood level in a given area. The design minimizes the number of penetrations through interior walls below the maximum flood level. Those penetrations below the maximum flood level are watertight and can withstand the maximum hydrostatic loads. Process piping penetrating below the maximum flood level either will be embedded in the wall or will be welded to a steel sleeve embedded in the wall.

Safety-related systems and components needed for safe shutdown are identified in Section 7.4 of the SSAR. The safe shutdown systems and components located in containment are associated with the PXS, the automatic depressurization system (ADS), and containment isolation valves.

In the SSAR, Westinghouse identifies seven compartments in containment which are subject to full or partial flooding. These are the reactor vessel cavity, two steam generator compartments, a vertical access tunnel, the CVS compartment, and two PXS compartments (PXS-A in the southeast quadrant of containment and PXS-B in the northeast quadrant of containment). Of these compartments, only the two PXS compartments contain safe-shutdown equipment. Both compartments are below the maximum flood water level (Elevation 108'-2"). The RCS cavity and the two steam generator compartments are interconnected by the vertical access tunnel. These compartments are combined into one floodable volume called the RCS compartment. The PXS-A, PXS-B, and CVS compartments comprise the remaining separate flood volumes and are isolated from each other as well as from the RCS compartment. Flooding in the PXS-A, PXS-B, or CVS compartments may result in some flooding of the RCS compartment, but will not result in flooding of any other compartment. The maximum flood level in containment assumes that the combined water inventory from all available sources in containment flood the reactor and steam generator compartments to a level above the RCS piping during a loss-of-coolant accident (LOCA). The flood water would cover the break location and allow backflow either through the break or via the PXS recirculation system flow path. The available flooding sources are the RCS, two accumulators, two core makeup tanks (CMTs), and the IRWST. The resulting maximum flood level in containment is at Elevation 108'-2".

The reactor vessel cavity and the adjoining equipment room are located at the lowest level of the containment (Elevation 66'-6"). The equipment room contains the containment sump pumps. Floor drains from the PXS-A, PXS-B, and CVS compartments are routed to the containment sump. Reverse flow to these three compartments is prevented by the use of redundant safety-related backflow preventers. Each compartment drain line is monitored by its own non-safety-related flow sensor. Flow through each drain line, as well as total flow from all drain lines, is monitored in the main control room (MCR). Containment flooding is detected through the use of the containment sump level monitoring system and the containment flood-up level instrumentation. The containment sump level monitoring system uses redundant, seismically qualified level sensors to detect sump level. Level signals are transmitted to the MCR and to the leakage detection monitoring equipment. The leakage detection monitors cause the initiation of appropriate safety actions when there is an indication of leakage (Section 5.2.5 of the SSAR). The containment flood-up level instrumentation consists of redundant, Class 1E sensor racks which monitor the water level from the bottom of the reactor vessel cavity to the top of the vertical access tunnel. Level indications are transmitted to the MCR.

The PXS-A and PXS-B compartments and the CVS compartment in containment are physically separated and isolated from each other by a structural wall so that flooding in one compartment cannot cause flooding in the other compartment. They are located below the maintenance floor level (Elevation 107'-2"). A curb is provided around the openings that penetrate the maintenance floor, thus providing the required protection up to the maximum flood level to Elevation 108'-2". The curb for the CVS compartment is lower so that it floods first. Should flooding continue, the water would overflow the curb and spread over the maintenance floor at Elevation 107'-2". From there, the water would flow into the RCS compartment via the vertical access tunnel.

Inside the PXS compartments, automatically actuated containment isolation valves (CIVs) include one normally closed CIV for the spent fuel pit cooling system in PXS-A and three normally closed CIVs for the normal residual heat removal (RHR) system in PXS-B. These CIVs are not required for safe shutdown operation and will not fail open under flooded conditions. In addition, redundant CIVs are provided on each line outside of containment. Each PXS compartment also contains a set of normally closed air-operated CMT isolation valves. These compartments also contain one normally open accumulator isolation valve and one normally open IRWST isolation valve. Because these valves are normally open, they do not require repositioning during flooded conditions. In addition, each PXS compartment contains two normally closed, motor-operated valves arranged in series as part of the PXS recirculation subsystem. These valves are opened during a flood event to provide a redundant flow path from the RCS compartment to the reactor vessel. Valves below the flood level are the only active, safety-related valves in the compartments.

The internal flood analysis considered single failures such as a break of the 20.3-cm (8-in.) direct vessel injection line, the 30.5-cm (12-in.) normal RHR line, the 20.3-cm (8-in.) accumulator injection line, and the 15.2-cm (6-in.) and 25.4-cm (10-in.) IRWST lines. The worst flood conditions result from a break in the 20.3-cm (8-in.) direct vessel injection line. In this case, flooding would occur as a result of blowdown of the RCS, as well as from the CMT and the accumulator. The resulting flood would affect only one PXS compartment, allowing the redundant PXS division to perform its safety function.

There are several duct penetrations into the CVS and PXS compartments. These penetrations (through the floor at Elevation 107'-2") are designed to prevent the flooding of these rooms from the maintenance floor level.

The fire protection system (FPS) and demineralized water transfer and storage system (DMWS) are open-cycle systems that enter the containment. These systems are isolated during plant operation and are not a potential flooding source.

The auxiliary building upper annulus provides the air flow path for the passive containment cooling system (PCS). The annulus floor has a curb on the outside with a flexible seal connected to the shield building, which blocks communication with the middle annulus below. The outside wall of the upper annulus has redundant, physically separated drains which discharge to the yard drainage system to limit water accumulation. These drains are required for operation of the PCS. The worst-case flooding in the annulus occurs when non-safety floor drains are blocked concurrent with an inadvertent opening of a PCS cooling water isolation valve. During this postulated event, the maximum water height is approximately 61 cm (24 in.). This will not affect any other safety-related equipment. Flooding in the annulus is detected by Class 1E level sensors which provide an alarm in the MCR.

The PCS valve room contains two redundant safety-related valve trains for the PCS. A through-wall crack of the PCS piping is the only flooding source for this room. The valve room door is not watertight. Leakage flows under the door and down the containment wall to the upper annulus floor, where it drains to the yard drainage system. The leakage under the valve room door or through floor drains is sufficient to prevent excessive water accumulation in the valve room. The isolation valves are located above the maximum flood level. Level sensors, in the valve room drain sump, alarm in the MCR. No safety-related equipment is affected by this worst-case flood scenario.

Based on the evaluation of the SSAR information, the staff concludes that Westinghouse properly identified safety-related equipment and flood hazards in containment and provided adequate means of protecting safety-related equipment from the identified flood hazards in containment.

In the SSAR, Westinghouse identifies the safety-related equipment in the auxiliary building which require flood protection on a room-by-room basis, depending on the relative location of the equipment. The auxiliary building is separated into radiologically controlled areas (RCAs) and nonradiologically controlled areas (NRCAs). On each floor, these areas are separated by structural walls and floor slabs 0.61- to 0.91-m (2- to 3-ft) wide. These structures are designed to prevent floods which may occur in one area from propagating to another area. Electrical penetrations between RCAs and NRCAs are located above the maximum flood level. Process piping penetrations between the two areas are embedded in the wall or are welded to a steel sleeve in the wall.

The NRCA is divided into a mechanical equipment area and an electrical equipment area. The electrical equipment area is further divided into an area housing Class 1E electrical equipment and non-Class 1E electrical equipment.

The safe-shutdown equipment located in the NRCA is associated with the protection and safety monitoring system (I&C cabinets on Level 3), the Class 1E dc system (Class 1E batteries on Levels 1 and 2, and dc electrical equipment on Level 2), and containment isolation. The NRCAs are designed to provide maximum separation between the mechanical equipment and electrical equipment areas.

The mechanical equipment areas located in the NRCAs include the valve/piping penetration room (Level 3) and two main steam isolation valve rooms and mechanical equipment rooms (Levels 4 and 5). Flood water in these areas is routed to the turbine building or the annex building via drain lines, controlled access ways, or blowout panels which vent from the main steam isolation valve room to the turbine building.

The NRCAs are also designed to provide maximum separation between Class 1E and non-Class 1E electrical equipment. These areas drain to a sump on Level 1 (Elevation 66'-6").

The AP600 design minimizes water sources in those portions of the NRCAs housing Class 1E electrical equipment. In these areas, the only water sources are associated with firefighting, emergency eyewash/shower, and battery washdown. No water accumulates on the upper floors of the auxiliary building in these areas. Instead, flooding from these sources is directed to Level 1 via floor drains, stairwells, and elevator shafts. The maximum postulated water height on Level 1 is 20.3 cm (8 in.). The terminal height on the first row of batteries on Level 1 is

76.2 cm (30 in.). Therefore, the safety-related electrical equipment on Level 1 is adequately protected from the anticipated worst-case flood conditions. Although the operation of the sump pumps is not required for flood protection, the Level 1 sump pumps are designed to remove approximately 946.4 L/min (250 gpm), which is equivalent to the maximum flow associated with the operation of two fire hose stations.

The MCR and the remote shutdown workstation (RSW) are also located in the NRCA. The MCR and RSW are adequately protected from flooding due to limited sources of flood water, pipe routing, and drainage paths.

In Section 3.11 of the SSAR, Westinghouse states that in the event of potential flooding/wetting, one of the following criteria is applied for protection of equipment for service in such an environment:

(1) Equipment will be qualified for submergence due to flooding/wetting.

(2) Equipment will be protected from wetting due to spray.

(3) Equipment will be evaluated to show that failure of the equipment due to flooding/wetting is acceptable since its safety-related function is not required or has otherwise been accomplished.

In the NRCA, mechanical and electrical equipment are separated by concrete walls and floors that form a watertight barrier. The Class 1E components in the mechanical equipment area are the CIVs, the main steam (MS) and feedwater (FW) isolation valves and the MS and FW line instrumentation. This equipment is either protected from spray wetting or is environmentally qualified for spray conditions. The doors for the battery rooms are normally closed because they also serve as fire barriers (these doors utilize automatic closers). These doors will prevent spray from sources outside the battery room from affecting equipment in the room.

The four Class 1E electrical divisions in the NRCA of the auxiliary building are separated by 3-hour rated fire barriers. Portions of these fire barriers also serve as flood barriers. The HVAC ducts, that penetrate these barriers and are below the maximum flood level, are required to be watertight. Because the maximum flood level in most of the Class 1E electrical areas is 7.6 cm (3 in.), none of the wall penetrations will need to be watertight. Floor penetrations between rooms of the same division are not required to be watertight.

The FPS is the only open-cycle system that enters the mechanical equipment area of the NRCA. Fire water will drain from this area to the turbine building or annex building. FPS and DMWS are open-cycle systems that enter the electrical equipment area of the NRCA. The maximum diameter of the DMWS piping is 2.54 cm (1 in.) and therefore, is not considered a credible flood source. Limited water volume hose stations are used in the Class 1E electrical equipment areas.

Based on the evaluation of the SSAR information, the staff concludes that Westinghouse properly identified safety-related equipment and flood hazards in the NRCA and provided adequate means of protecting safety-related equipment from the identified flood hazards in the NRCA.

The safe-shutdown equipment located in RCAs are primarily CIVs that are located near the containment vessel and above the maximum flood level for the area. These valves are either normally closed or are closed during a safe shutdown operation.

Flood sources in the RCA include component cooling water (CCS), central chilled water, hot water, spent fuel pit cooling, normal RHR, FPS, and CVS and various tanks. Flood water which results from component failures in the RCA is directed to the Level 1 drain collection sump via the vertical pipe chase, floor gratings, floor drains, stairwells, and elevator shafts. There is no safe-shutdown equipment on Level 1. Safe-shutdown equipment in the RCA is located on Level 2 and at the upper levels of the vertical pipe chase. Because flood water is directed to Level 1, there is little accumulation of water in the RCAs at higher levels inside the building. The HVAC duct penetrations in the walls in these areas are above the maximum flood levels. Therefore, safety-related systems and equipment in the RCA in the auxiliary building are protected from the effects of flooding.

The FPS, DMWS, and CVS are open-cycle systems which enter the RCA. The FPS has the largest volume. All water drains to the lowest level where no safe-shutdown equipment is located. Safety-related valves are located above Elevation 82'-6". If the contents of both fire water storage tanks were emptied into the building, the resulting flood height would be less than Elevation 82'-6".

Some doorways between the auxiliary building and the adjacent turbine, annex, and radwaste buildings are double doors located above grade elevation. These doors are not water tight. Water from internal flooding in areas adjacent to the auxiliary building is directed away from or prevented from entering the auxiliary building. The containment and auxiliary buildings (which house all of the safety-related equipment) have a common basemat and there are no tunnels below grade between these buildings and any other buildings.

As stated above, open-cycle systems serve the containment, RCA, and NRCA. The FPS and DMWS are open-cycle systems that enter the containment. These systems are isolated during plant operation and are not a potential flooding source. The FPS is an open-cycle system that enters the mechanical equipment area of the NRCA. Fire water will drain from this area to the turbine building or annex building. The FPS and DMWS are open-cycle systems that enter the electrical equipment area of the NRCA. The maximum diameter of the DMWS piping is 2.54 cm (1 in.) and therefore, is not considered a credible flooding source. Limited water volume hose stations are used in the Class 1E electrical equipment areas. The FPS, DMWS, and CVS are open-cycle systems which enter the RCA. The FPS has the largest volume. All water drains to the lowest level where no safe-shutdown equipment is located. Safety-related valves are located above Elevation 82'-6". If the contents of both fire water storage tanks were emptied into the building, the resulting flood height would be less than Elevation 82'-6".

Based on the evaluation of the SSAR information, the staff concludes that Westinghouse properly identified safety-related equipment and flood hazards in the RCA and provided adequate means of protecting safety-related equipment from the identified flood hazards in the RCA.

Based on the evaluation of the SSAR information, the staff concludes that Westinghouse provided adequate features in the AP600 flood design to ensure that safety-related systems will be adequately protected from flood-related effects associated with both natural phenomena and system and component failures. Therefore, the staff concludes that the flood design meets the requirements of GDC 2 as it relates to protecting safety-related SSCs from the effects of floods.

The COL applicant is responsible for identifying external flood and precipitation hazards beyond those assumed in the AP600 flood analysis and providing protective features to ensure that safety-related equipment is adequately protected from these hazards. In addition, the COL applicant must verify that the as-built design conforms with the certified design.

During the staff's initial review of flood protection for the AP600 standard design, the following issues were identified and documented in the DSER for resolution:

- incorporation of RAI responses into the SSAR
- provide information regarding flood protection for systems classified under RTNSS and DID systems
- correction of RAI responses
- COL applicant responsibilities
- conformance with RGs
- discrepancies in the SSAR
- design requirements for instrumentation
- locations and design requirements for certain isolation valves and structural walls
- design requirements for drains
- backflow protection from buildings not housing safety-related equipment to buildings housing safety-related equipment
- interconnecting tunnels between buildings

Collectively, these issues constituted DSER Open Item 3.4.1-1. During the process of issue resolution, Westinghouse provided applicable information regarding this open item. Based on the information as discussed above, the staff closed DSER Open Item 3.4.1-1.

The staff review of the flood protection design included systems and components whose failure could prevent safe shutdown of the plant and maintenance thereof, or result in significant uncontrolled release of radioactivity. Based on the review of the proposed flood protection criteria for safety-related SSCs necessary for safe shutdown during and following flood conditions resulting from external or internal causes, the staff concludes that the AP600 design for flood protection conforms to the regulations set forth in GDC 2 and 10 CFR Part 100,

Appendix A. This conclusion is based on the capability of the design to protect safety-related SSCs from the effects of floods in accordance with the following criteria:

- meeting Position C.1 of RG 1.59 regarding the criteria used for the design of safety-related SSCs to withstand the worst-probable site-related flood
- meeting Position C.1 of RG 1.102 regarding the type of flood protection provided

As a result, the staff concludes that the AP600 flood protection design meets the requirements of GDC 2 and 10 CFR Part 100, Appendix A, Section IV.C as they relate to protecting safety-related SSCs from the effects of external and internal floods. Therefore, the staff concludes that the AP600 design meets the guidelines of Section 3.4.1 of the SRP and is acceptable.

3.4.2 Analysis Procedures

In Section 2.4 of the SSAR, Westinghouse specifies that the design-basis flood and groundwater levels for the AP600 design are up to the finished grade and 0.6 m (2 ft) below grade, respectively. Criteria for the design-basis flood are in accordance with RGs 1.59 and 1.102. The maximum flood level for a specific site is established at less than the finished grade.

The basemat and exterior walls of the seismic Category I structures are designed to resist the upward and the lateral pressures caused by the probable maximum flood and the high groundwater levels. All seismic Category I structures will be stable when subjected to both overturning moment and uplift forces resulting from the load combinations considered to be appropriate, including the forces from the design-basis flood level.

Westinghouse satisfied the review criteria of Section 3.4.2 of the SRP and met the requirements of GDC 2, with respect to the structural capability to withstand the effects of the maximum flood and highest groundwater levels by reflecting the following design criteria:

- appropriate consideration for the most severe flood not to exceed the flood level identified above for any future site
- appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena
- the importance of the safety function to be performed

The design-basis flood is a site parameter for the standard design and by limiting the design-basis flood elevation up to the finished grade and the design groundwater level to 0.6 m (2 ft) below the grade, Westinghouse provides sufficient margin to prevent structural damage during the most severe flood as described above so the requirement of the first item above is met. In addition, as required by the second item above, the design of seismic Category I structures includes, in an acceptable manner, the combination of the most severe flood or groundwater-related loads described above and loads resulting from normal and accident conditions.

The loadings on seismic Category I structures induced by the design-basis flood or highest groundwater level were used in the design of conventional structures and proven to serve as a conservative baseline that ensures that structures will withstand such environmental forces. Therefore, these procedures are acceptable.

The use of these procedures gives reasonable assurance that, in the event of floods as described above, the structural integrity of the plant seismic Category I structures will not be impaired; therefore, seismic Category I systems and components located within these structures will be adequately protected and are expected to perform the necessary safety functions, thus satisfying the requirements of the third item above.

The staff therefore concludes that the AP600 design is acceptable and meets the guidelines of Section 3.4.2 of the SRP and the requirements of GDC 2.

However, in order to assure that the above design requirements for the design-basis flood and the groundwater are met, the COL applicant must provide a specific description of the site and the elevation for all safety-related structures, exterior accesses, equipment, and systems, from the standpoint of hydrology considerations and flood history (including date, level, peak discharge and related information for major historical flood events in the site region). The COL applicant will also address the following topics:

- probable maximum precipitation (PMP)
- precipitation losses over the applicable drainage area
- runoff and stream course models
- maximum flood flow
- water level determination (flood and groundwater)
- coincident wind wave activity
- flow from postulated break of upstream dams that cannot withstand site-specific seismic motion

This was identified as DSER Open Item 3.4.2-1 and COL Action Item 3.4.2-1.

In Revision 2 to Section 2.4 of the SSAR, Westinghouse states that the COL applicant will evaluate events leading to potential flooding to demonstrate that the site meets the interface requirements. Events to be considered are those identified in Section 2.4.2 of the SRP. Also, in Revision 2 to Section 2.4.1 of the SSAR, Westinghouse requires that the COL applicant describe the site-specific information on major hydrological features on or in the vicinity of the plant including critical elevations of the nuclear island, historical flooding and potential flooding factors including the effects of local intense precipitation, water supply sources to the service water system cooling tower, groundwater, etc. For cases where a site characteristic exceeds the design envelop parameter, it is necessary for the COL applicant to demonstrate that the site characteristic does not exceed the capability of the design. These requirements for the COL applicant meet the guidelines in Sections 2.4.1 through 2.4.14 of the SRP, and, are acceptable. Therefore, DSER Open Item 3.4.2-1 is closed.

The COL applicant must ensure and demonstrate in the site-specific application that all seismic Category I structures are either protected against flood damage or are not subject to such damage. Hydrostatic and hydrodynamic effects of the flood, if applicable, must be considered and described for all postulated design flood levels for the conditions set for the future site, as outlined above. This was identified as DSER Open Item 3.4.2-2 and COL Action Item 3.4.2-2.

In Section 3.4.1.1 of the SSAR, Westinghouse states that the seismic Category I SSCs identified in Section 3.2 of the SSAR are designed to withstand the effects of flooding due to natural phenomena or postulated component failures. In Revision 2 to Section 2.4 of the SSAR, Westinghouse requires, if flood protection is required, that the COL applicant will provide flood protection measures to protect the plant according to the guidelines in Section 2.4.10 of the SRP.

The staff finds that the COL requirement will provide adequate protection against flood damage for all seismic Category I structures and is acceptable. Therefore, DSER Open Item 3.4.2-2 is closed.

3.5 Missile Protection

GDC 4 of Appendix A to 10 CFR Part 50, requires that SSCs important to safety be protected from the effects of missiles. Missiles may be generated by pressurized components, rotating machinery, explosions, tornados, transportation accidents, and dropped loads. In the AP600 design, protection of SSCs from these missiles is achieved by minimizing the sources of the missiles and by arranging structures and equipment to minimize or prevent missile damage.

Westinghouse provided criteria for the identification of missiles and protection requirements for equipment, as well as an evaluation procedure to determine if the identification criteria and protection requirements have been met.

3.5.1 Missile Selection and Description

3.5.1.1 Internally-Generated Missiles (Outside Containment)

The staff reviewed the AP600 design for protecting SSCs important to safety against internally-generated missiles (outside containment) in accordance with Section 3.5.1.1 of the SRP. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the design of the facility for providing protection against internally-generated missiles satisfies the requirements of GDC 4 as it relates to protecting SSCs outside containment against the effects of missiles outside the containment that may result from equipment failures. The missiles outside containment considered in this review include those missiles generated by rotating or pressurized (high-energy fluid system) equipment. The design adequacy of the facility for protection against low-trajectory turbine missiles, including compliance with RG 1.115, "Protection Against Low-Trajectory Turbine Missiles," is discussed separately in Section 3.5.1.3 of this report.

The staff's review of missile protection considered the following:

- plant design features for protecting SSCs outside containment important to safety against internally-generated missiles
- equipment design features that could reduce missile sources

- physical separation or orientation of missile sources away from safety-related SSCs
- adequate protective shielding and barriers that could confine potential internally-generated missiles
- adequate hardening of safety-related equipment and components to withstand missile impact if a missile strike cannot be reasonably avoided

The staff identified the following issues as DSER Open Item 3.5.1.1-1:

- incorporation of RAI responses into the SSAR
- provide information regarding missile protection for systems classified under RTNSS and DID systems
- discrepancies between the SSAR and RAI responses
- nonconservatisms in missile evaluation

Westinghouse provided additional information and revisions to the SSAR to address the issues which the staff reviewed in the following evaluation and found acceptable. Therefore, DSER Open Item 3.5.1.1-1 is closed.

In response to staff's questions concerning the protection of safety-related SSCs, Westinghouse stated that the AP600 design uses only safety-related systems to establish and maintain safe-shutdown conditions. There is no non-safety-related equipment whose failure could adversely affect the ability of safety-related equipment to perform its safety function. The design provides physical separation between the safety-related equipment and non-seismic SSCs to the maximum extent practical. The safety-related systems and components needed to bring a plant to a safe shutdown are inside the containment shield building and auxiliary building that are protected from missiles generated in other portions of the plant or tornado missiles. In addition, there are no systems or components identified as important in the evaluation of RTNSS that require protection from missiles.

In RAI 410.51, the staff requested that Westinghouse identify all safety-related systems and equipment located outside of the containment that require protection from internally-generated missiles. Westinghouse stated that there is no safety-related equipment which requires protection from internally-generated missiles since the AP600 design has no credible missile sources outside the containment. Safety-related systems and components are located within the seismic Category I nuclear island structures and are protected from missiles. The consequences of scabbing are evaluated if the wall thickness is less than the minimum thickness to preclude scabbing and no credible secondary missiles due to fragments from scabbing of concrete are identified. In addition, Westinghouse added in Revision 18 to the SSAR the reference of Section 7.4 of the SSAR for the safety-related systems required for safe shutdown to Section 3.5.2. The staff finds that the added reference answered RAI 410.51.

The staff reviewed the internally-generated missiles from rotating equipment, such as motor-driven pumps and fans. Westinghouse stated in its response to RAIs that the only missile sources in this category are the non-safety-related pumps and fans inside the auxiliary building and they are not considered credible missile sources for the following reasons:

- Rotating equipment with a housing or an enclosure that contains the fragments of a postulated impeller failure is not considered a credible missile source.
- Rotating equipment (pumps, motors for valve operators, and mechanical handling equipment, etc.) in use less than 2 percent of the time is not considered a credible missile source because of the limited risk for missile generation.
- The remote shutdown workstation in the auxiliary building is separated from rotating equipment or pressurized components by compartment walls.

In reviewing the missiles generated by pressurized components of high-energy fluid system, Westinghouse indicated that they are not considered credible missile sources for the following reasons:

- The pressurized components of high-energy systems inside the auxiliary building are constructed to the ASME Code, Section III and are not considered a credible source of missiles.
- The high-pressure gas storage cylinders and attached piping and valves inside the auxiliary building are constructed to the ASME Code, Section VIII (for the gas storage cylinders) and Section III (for the attached piping and valves) and are not considered a credible source of missiles.
- Systems that exceed 93.3 °C (200 °F) or 1999.5 kPa (275 psig) for 2 percent or less of the time during which the system is in operation or that experience high-energy pressure or temperature for less than 1 percent of the plant operation time are considered moderate-energy for the purpose of missile generation and are not considered to be the source of missiles.
- Missiles generated from hydrogen explosions are not considered credible due to the design of the systems which use or generate hydrogen. The hydrogen concentration in the supply line from the hydrogen storage area is within the limits of NUREG/CR-2017. A failure of this line will not lead to an explosion. The battery compartments are well ventilated and the hydrogen bottles have a limited release volume. Furthermore, the storage areas for plant gases are located away from the nuclear island.
- The bonnets of pressure-seal valves are designed in accordance with the requirements of NB/NC/ND-3000 and NB/NC/ND-3500 of Section III of the ASME Code and are not considered to be a potential source of missiles.
- The yoke attached to the valve body is not considered to be a pressure-retaining part. Bolt and nuts do not become missiles unless they break, and the stored energy in nuts, bolts, and combinations is not sufficient to generate a credible missile when struck by a falling object.

In RAI 410.313, the staff identified that Westinghouse did not address the possibility of safety relief valves becoming internally-generated missiles (outside containment). Westinghouse had not assessed the potential missiles that could be generated outside the containment from piping systems and associated monitoring equipment, such as temperature probes, flow measurement devices, pressure gauges, and vessel monitoring devices, if any. In response to the RAI, Westinghouse stated that safety relief valves in high-energy systems use the bolt bonnet design that will preclude missile generation. The piping and tubing that connect instrumentation such as pressure, level, and flow transmitters to the pressure boundary of piping and components in high energy systems are designed with welded joints or compression fittings for the tubing. Threaded connections are not used to connect instrumentation to high-energy systems or components. The quantity of high-energy fluid in these instruments is limited and will not result in missile generation. The staff finds that Westinghouse has adequately assessed the potential missiles outside the containment and, therefore, the response is found acceptable. In Revision 18 to the SSAR, Westinghouse revised Section 3.5.1.1.2.1 to include the above response.

In RAI 410.210, the staff requested Westinghouse to address potential gravitational missiles outside containment. In Revision 18 to the SSAR, Westinghouse states in Section 3.5.1.1.2.4 that the safety-related equipment outside containment is located in the auxiliary building. Falling objects (e.g., gravitational missiles) heavy enough to generate a secondary missile outside containment are postulated as a result of movement of a heavy load. Protecting safety-related SSCs from missiles during movement of heavy loads is addressed in Section 9.1.5 of the SSAR. The staff reviewed this section and finds that this issue is adequately addressed as discussed in Section 9.1.5 of this report.

In response to RAI 410.216, regarding missile protection for the control room and remote shutdown workstation, Westinghouse stated that both the main control room and remote shutdown workstation are located in the auxiliary building. These areas are protected from internally-generated missiles by the structural concrete walls and floors of the auxiliary building. The staff finds the response acceptable.

In response to RAI 410.215, Westinghouse stated that no safety-related systems or component are protected from missiles solely by providing sufficient distance between the system or component and the missile sources. Spatial separation may be used to demonstrate protection from a missile hazard when it is shown that the range and trajectory of the generated missile is less than the distance to or is directed away from the potential target. The staff's review confirms that the safety-related equipment has been adequately protected from internally-generated missiles. The staff finds the response acceptable.

On the basis of its review, the staff concludes that the design of the facility meets the guidelines of Section 3.5.1.1 of the SRP. Therefore, the staff concludes that the AP600 design conforms with GDC 4 as it relates to protection against internally-generated missiles (outside containment).

3.5.1.2 Internally-Generated Missiles (Inside Containment)

The staff reviewed the design of the facility for protecting SSCs important to safety against internally-generated missiles inside containment in accordance with Section 3.5.1.2 of the SRP.

Conformance with the acceptance criteria of the SRP forms the basis for concluding that the SSCs to be protected from internally-generated missiles inside containment meet the requirements of GDC 4, as it relates to protecting SSCs against the effects of missiles that can be internally generated during facility operation. Specifically, the staff's review concentrated on the missiles associated with component overspeed failures, missiles that could originate from high-energy fluid system failures, and missiles due to gravitational effects.

The staff identified the following issues as DSER Open Item 3.5.1.2-1:

- incorporation of RAI responses into the SSAR
- provide information regarding missile protection for systems classified under RTNSS and DID systems
- consideration of all possible missiles

Westinghouse provided additional information and revisions to the SSAR to address the issues which the staff reviewed in the following evaluation and found acceptable. Therefore, DSER Open Item 3.5.1.2-1 is closed.

In reviewing these potential missiles, the staff requested Westinghouse to identify safety-related equipment inside containment that requires missile protection. Westinghouse stated that there is no safety-related equipment which requires protection from internally-generated missiles inside containment since there are no credible missile sources. The only missile sources which may impact on safety-related SSCs are a few non-safety-related fans inside the containment. The safety-related systems and components needed to bring the plant to a safe shutdown are located inside the containment shield building that are protected from missiles.

In Section 3.5.1.2.1.1 of the SSAR, Westinghouse listed the following potential sources of internally-generated missiles:

- any failure of rotating parts of the reactor coolant pump
- catastrophic failure of rotating equipment such as pumps, fans, and compressors leading to the generation of missiles
- failure of the reactor vessel, steam generator, pressurizer, core makeup tanks, accumulators, reactor coolant pump castings, passive residual heat exchangers and piping leading to the generation of missiles
- gross failure of a control rod drive mechanism housing sufficient to create a missile from a piece of the housing or to allow a control rod to be ejected rapidly from the core
- valves, valve stems, nuts and bolts, thermowells, and missiles originating in non-high-energy fluid systems

Westinghouse stated that the above-mentioned potential missile sources are not considered credible because there is insufficient energy to produce a missile or by design the probability of creating a missile is negligible. Westinghouse evaluated the potential failure of the rotating

76.2 cm (30 in.). Therefore, the safety-related electrical equipment on Level 1 is adequately protected from the anticipated worst-case flood conditions. Although the operation of the sump pumps is not required for flood protection, the Level 1 sump pumps are designed to remove approximately 946.4 L/min (250 gpm), which is equivalent to the maximum flow associated with the operation of two fire hose stations.

The MCR and the remote shutdown workstation (RSW) are also located in the NRCA. The MCR and RSW are adequately protected from flooding due to limited sources of flood water, pipe routing, and drainage paths.

In Section 3.11 of the SSAR, Westinghouse states that in the event of potential flooding/wetting, one of the following criteria is applied for protection of equipment for service in such an environment:

(1) Equipment will be qualified for submergence due to flooding/wetting.

(2) Equipment will be protected from wetting due to spray.

(3) Equipment will be evaluated to show that failure of the equipment due to flooding/wetting is acceptable since its safety-related function is not required or has otherwise been accomplished.

In the NRCA, mechanical and electrical equipment are separated by concrete walls and floors that form a watertight barrier. The Class 1E components in the mechanical equipment area are the CIVs, the main steam (MS) and feedwater (FW) isolation valves and the MS and FW line instrumentation. This equipment is either protected from spray wetting or is environmentally qualified for spray conditions. The doors for the battery rooms are normally closed because they also serve as fire barriers (these doors utilize automatic closers). These doors will prevent spray from sources outside the battery room from affecting equipment in the room.

The four Class 1E electrical divisions in the NRCA of the auxiliary building are separated by 3-hour rated fire barriers. Portions of these fire barriers also serve as flood barriers. The HVAC ducts, that penetrate these barriers and are below the maximum flood level, are required to be watertight. Because the maximum flood level in most of the Class 1E electrical areas is 7.6 cm (3 in.), none of the wall penetrations will need to be watertight. Floor penetrations between rooms of the same division are not required to be watertight.

The FPS is the only open-cycle system that enters the mechanical equipment area of the NRCA. Fire water will drain from this area to the turbine building or annex building. FPS and DMWS are open-cycle systems that enter the electrical equipment area of the NRCA. The maximum diameter of the DMWS piping is 2.54 cm (1 in.) and therefore, is not considered a credible flood source. Limited water volume hose stations are used in the Class 1E electrical equipment areas.

Based on the evaluation of the SSAR information, the staff concludes that Westinghouse properly identified safety-related equipment and flood hazards in the NRCA and provided adequate means of protecting safety-related equipment from the identified flood hazards in the NRCA.

Westinghouse stated that the quantity that could be released inside the containment in the event of a hydrogen supply line failure is limited to the contents of a single bottle. Because the volume percent of hydrogen that could be accumulated in the containment is less than the detonation limit, the amount of hydrogen released to the containment would not lead to an explosion. The staff agrees with Westinghouse's response and, therefore, finds it acceptable.

Gas storage cylinders and attached valves and piping systems are considered to have the potential to generate a missile when struck by a dropped object. In response to RAI 410.223, Westinghouse stated that there are no high-pressure gas storage cylinders inside the containment shield building. The staff finds the response acceptable.

On the basis of its review, the staff concludes that the design meets the guidelines of Section 3.5.1.2 of the SRP. Therefore, the staff concludes that the AP600 design for protection from internally-generated missiles inside the containment conforms with GDC 4 as it relates to protection against internally-generated missiles.

3.5.1.3 Turbine Missiles

The staff utilized the guidelines of Section 3.5.1.3 of the SRP to review and evaluate the information submitted by Westinghouse to ensure a low probability of turbine rotor failure. The staff's acceptance criteria are based on the plant design and layout satisfying the requirements of GDC 4, which require that SSCs important to safety be protected against the effects of missiles that might result from equipment failures, in this case the steam turbine.

The probability of unacceptable damage from turbine missiles is expressed as the product of the following items:

- the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing (P_1)
- the probability of ejected missiles perforating intervening barriers and striking safety-related SSCs (P_2)
- the probability of struck SSCs failing to perform their safety functions (P_3)

The evaluation of the materials and the inspection and maintenance of the turbine rotors in Section 10.2.3 of this report form the basis of P_1 used in the calculations. With the use of proper turbine rotor design, proper materials (heat treated correctly and tested to determine that property requirements are met), and meaningful nondestructive examination (NDE) inspection methods and acceptance criteria (preservice and inservice), the probability of turbine missile generation (P_1) will be reasonable.

Consistent with the staff's position taken in the ABWR and System 80+ standard designs, the probability of turbine missile generation should be kept to no greater than 1E-5 per reactor-year for an unfavorably oriented turbine and 1E-4 for a favorably oriented turbine.

The AP600 plant will utilize a favorable turbine generator placement and orientation, and is committed to meet RG 1.115, which should ensure a low probability of unacceptable damage from turbine missiles. With respect to the reactor building, the turbine system will be oriented so that any postulated high-energy, low-trajectory turbine missile will not strike the reactor building.

Upon review of the information provided in Revision 1 to the SSAR, the staff identified DSER open items and COL action items. The staff's review of Westinghouse's responses and the disposition of those items is provided below.

The staff requested that Westinghouse demonstrate that the probability of unacceptable damage to the safety systems protected by the roof of building structures housing safety-related systems required for safe shutdown will provide adequate barriers for high-trajectory, high energy turbine missiles. This was identified as DSER Open Item 3.5.1.3-1. In Revision 5 to Section 3.5.1.3 of the SSAR, "Turbine Missiles," Westinghouse states that the potential for a high-trajectory missile to impact safety-related areas of the AP600 is very low. In the AP600, the safety-related area is contained within the containment shield building and the auxiliary building. Therefore, the risk of a high trajectory turbine missile affecting safety-related functions in the AP600 design is extremely low (1E-7). The staff finds this acceptable, and therefore, DSER Open Item 3.5.1.3-1 is closed.

In the response to RAI 251.31 dated November 30, 1992, Westinghouse referenced its topical report WSTG-4-P, "Analysis of Probability of Generation of Missiles from Fully Integral Nuclear Low Pressure Rotor." In Revision 0 to the SSAR, this report was also referenced; however, in Revision 1, this reference was removed. Subsequently, this report was submitted in the June 27, 1994, response to RAI 410.211. The staff concluded that because the staff had not yet completed its review of this report, the information included in WSTG-4-P could not be incorporated by reference into the SSAR until the staff reviewed and accepted the report for reference. This was identified as DSER Open Item 3.5.1.3-2. Subsequently, the NRC staff reviewed WSTG-4-P and used the information included in the report to address several of the open items identified below. The report is referenced in Section 3.5.5 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 3.5.1.3-2 is closed.

In Section 3.5.1.3 of the SSAR, Westinghouse states, "The turbine and disk design is described in Section 10.2." In Section 10.2.3 of the SSAR, Westinghouse also states, "...The probability of destructive overspeed condition and missile generation, assuming the recommended inspection frequency, is less than 1×10^{-5}...." The staff requested that Westinghouse provide, in the SSAR, a discussion describing how this value was determined. This was identified as DSER Open Item 3.5.1.3-3. The requested information concerning the probability of turbine missile generation was included in Westinghouse Report WSTG-4-P and is referenced in Section 3.5.5 of the SSAR. The assessment of probability of missile generation included the consideration of four potential failure modes: ductile burst, high cycle fatigue cracking, low cycle fatigue cracking and fracture resulting from stress corrosion cracking. In all fracture modes the probability of missile generation was shown to be less than 1E-5 per reactor year. The staff finds this acceptable to the staff based on the staff acceptance criteria discussed above, and therefore, DSER Open Item 3.5.1.3-3 is closed.

Paragraphs 2.3.2.2, 2.3.2.3, and 2.3.2.4 of Section 2.3.2 of Chapter 13 of the EPRI URD specify, and the staff accepts (see NUREG-1242), requirements for: initial design brittle fracture analysis, initial design fatigue analysis, flaw sizing and growth of flaws identified by

NDE at the time of manufacture and not repaired, and brittle fracture analysis of flaws identified by NDE at time of manufacture, respectively.

The staff requested that Westinghouse supply this analysis. This was identified as DSER Open Item 3.5.1.3-4. The requested information is included in Westinghouse Report WSTG-4-P and is referenced in Section 3.5.5 of the SSAR. The assessment of probability of missile generation included the consideration of four potential failure modes, (1) ductile burst, (2) high cycle fatigue cracking, (3) low cycle fatigue cracking, and (4) fracture resulting from stress corrosion cracking. In all fracture modes, the probability of missile generation was shown to be less than 1E-5 per reactor year. Further, a fully integral rotor design reduces residual stresses and stress concentrations and, therefore, decreases the probability of stress-related failure when compared to the shrunk-on disk designs that were used in older plants. The staff's review of the analysis finds it to be acceptable. Therefore, DSER Open Item 3.5.1.3-4 is closed.

In the DSER, the staff stated that the COL applicant should provide brittle fracture analyses of installed turbine rotors to the staff for review. This was identified as DSER Open Item 3.5.1-5 and COL Action Item 3.5.1.3-1. In Revision 5 to the SSAR, Westinghouse included Section 10.2.6, "Combined License Information." In this section, Westinghouse requires that the COL applicant have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the rotor analysis. The staff finds this acceptable, and therefore, DSER Open Item 3.5.1-5 is closed.

In Revision 0 to Section 10.2.3.4 of the SSAR, Westinghouse did not include the results of a fatigue analysis on turbine missile generation. The staff requested that fatigue analysis be provided to address the fatigue effects on probability of turbine missile generation. This was identified as DSER Open Item 3.5.1.3-6. The requested information is included in Westinghouse Report WSTG-4-P which is referenced in Section 3.5.5 of the SSAR. The assessment of probability of missile generation included the consideration of four potential failure modes: ductile burst, high cycle fatigue cracking, low cycle fatigue cracking and fracture resulting from stress corrosion cracking. In all fracture modes, the probability of missile generation was shown to be less than 1E-5 per reactor year. The staff finds this acceptable because the analysis demonstrated that the fatigue effects were adequately treated in the probability of missile generation. Therefore, DSER Open Item 3.5.1.3-6 is closed.

In Section 10.2.2.1 of the SSAR, Westinghouse describes the turbine generator foundation as being a spring-mounted support system. The staff requested that Westinghouse discuss the effects of this system on the fatigue analysis, including a harmonic analysis of the combined spring-mounted support system, controller and turbine generator assembly. This was identified as DSER Open Item 3.5.1.3-7. In Revision 5 to the SSAR, Westinghouse included Section 10.2.3.2.2, "Rotor Fatigue Analysis." In this section, Westinghouse explains that the turbine generator is mounted on a spring-mounted support system in order to isolate the dynamic behavior of the turbine generator equipment from the foundation structure. The support system includes a reinforced concrete deck on which the turbine generator is mounted. The deck is sized to maintain the gravity load and misalignment load bending stresses within allowable limits. The evaluation of loads includes the full consideration of the dynamic behavior of the combined turbine generator and foundation structure. The staff finds this acceptable, and therefore, DSER Open Item 3.5.1.3-7 is closed.

The staff stated in the DSER that the COL applicant should provide fatigue analyses of installed turbine rotors to the staff for review. This was identified as DSER Open Item 3.5.1.3-8 and COL Action Item 3.5.1.3-2. In Revision 5 to the SSAR, Westinghouse included Section 10.2.6, "Combined License Information." Section 10.2.6 includes a requirement that the COL applicant have an available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis. The turbine rotor integrity analysis includes consideration of brittle fracture, fatigue due to combined rotation and thermal stress, and flaw growth analysis. The staff finds this acceptable, and therefore, DSER Open Item 3.5.1.3-8 is closed.

In Section 10.2.3.2 of the SSAR, Westinghouse discusses the method of disposition of the presence of any NDE indications in turbine rotors, and in general terms, describes the factors that determine the acceptability of the indications for 60-year service. Paragraph 13.2.3.2.4 of the URD specifies, and the staff accepts (see NUREG-1242) that for flaws identified during NDE at the time of manufacture and not repaired, justification for use of the component shall be based upon crack growth and brittle fracture analyses. Predicted fatigue crack growth should not result in a flaw size large enough to cause failure in the design life of the component under the worst limiting operating conditions.

The staff requested that Westinghouse provide the assumptions and values that are to be applied in the fatigue crack growth rate calculations which will be used to justify the use of a rotor with flaws detected during manufacture. This was identified as DSER Open Item 3.5.1.3-9. The requested information is included in Westinghouse Report WSTG-4-P which is referenced in Section 3.5.5 of the SSAR. The assessment of probability of missile generation included the consideration of four potential failure modes: (1) ductile burst, (2) high cycle fatigue cracking, (3) low cycle fatigue cracking, and (4) fracture resulting from stress corrosion cracking. In all fracture modes the probability of missile generation was shown to be less than 1E-5 per reactor year. Further, in Revision 5 to the SSAR, Westinghouse added Section 10.2.6, "Combined License Information." In Section 10.2.6, Westinghouse requires that the COL applicant have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis. This COL action is identified in the discussion below as COL Action Item 3.5.1.3-3. The staff finds this acceptable, and therefore, DSER Open Item 3.5.1.3-9 is closed.

The staff concluded in the DSER that the COL applicant should provide the flaw growth analyses of NDE-detected flaws in installed turbine rotors to the staff for review. This was identified as DSER Open Item 3.5.1.3-10 and COL Action Item 3.5.1.3-3. In Revision 5 to the SSAR, Westinghouse included Section 10.2.6, "Combined License Information." In Revision 5 to the SSAR, Westinghouse added Section 10.2.6, "Combined License Information." In Section 10.2.6, Westinghouse requires that the COL applicant have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis. The turbine rotor integrity analysis includes consideration of brittle fracture, fatigue due to combined rotation and thermal stress, and flaw growth analysis. The staff finds this acceptable, and therefore, DSER Open Item 3.5.1.3-10 is closed.

Paragraph 2.3.2.4 of the URD specifies, and the staff accepts (see NUREG-1242), that for flaws identified during NDE at the time of manufacture and not repaired, justification for the use of the component shall be based upon brittle fracture analysis. The brittle fracture analysis for flaws identified during NDE at the time of fabrication shall be based on the following:

- the end-of-life flaw size
- geometry factors based on the configuration and location of the flaw, e.g., elliptical, circular, subsurface, surface connected
- peak stress normal to the detected flaw. For transients, several temperatures shall be considered to determine worst case temperature-stress combinations
- the fracture toughness, based upon the measured value or the lower bound correlation of K_{Ic} with fracture appearance transition temperature (FATT)
- an embrittled FATT shifted from that measured beginning of life FATT to account for the effects of temper embrittlement
- a minimum margin of two between the applied stress intensity factor and K_{Ic}

The staff concluded in the DSER that the COL applicant should provide the brittle fracture analyses of flaws detected by NDE during manufacture which are not removed in installed turbine rotors to the staff for review. This was identified as DSER Open Item 3.5.1.3-11 and COL Action Item 3.5.1.3-4. In Revision 5 to the SSAR, Westinghouse added Section 10.2.6, "Combined License Information." In Section 10.2.6, Westinghouse requires that the COL applicant have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis. The turbine rotor integrity analysis includes consideration of brittle fracture, fatigue due to combined rotation and thermal stress, and flaw growth analysis. The staff finds this acceptable, and therefore, DSER Open Item 3.5.1.3-11 is closed.

The staff requested that Westinghouse provide the assumptions and values that are to be applied in the fatigue crack growth rate calculations which will be used to justify the use of a rotor with flaws detected during manufacture. This was identified as DSER Open Item 3.5.1.3-12. In Revision 5 to the SSAR, Westinghouse added Section 10.2.6, "Combined License Information." In Section 10.2.6, Westinghouse requires that the COL applicant have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis. The turbine rotor integrity analysis includes consideration of brittle fracture, fatigue due to combined rotation and thermal stress, and flaw growth analysis. The staff finds this acceptable, and therefore, DSER Open Item 3.5.1.3-12 is closed.

Consistent with the staff's position taken in the ABWR and System 80+ standard designs, the licensee must submit for NRC approval, within 3 years of obtaining a COL, a turbine system maintenance program including probability calculations of turbine missile generation based on the methodology approved by the NRC, or commit to volumetrically inspect all low-pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff. This was identified as DSER Open Item 3.5.1.3-13 and COL Action Item 3.5.1.3-5. DSER Open Item 3.5.1.3-13 and COL Action Item 3.5.1.3-5 have been consolidated with DSER Open Item 10.2.9-6 and COL Action

Item 10.2.9-1. The staff's review and disposition of these items is documented in Chapter 10, "Steam and Power Conversion Systems" of this report. DSER Open Item 3.5.1.3-13 is closed. COL Action Item 3.5.1.3-5 is dropped.

The staff concludes that the risk for the proposed plant design is acceptable and meets the relevant requirements of GDC 4. This conclusion is on the basis of Westinghouse having sufficiently demonstrated to the staff, in accordance with the guidance of RG 1.115, that the probability of turbine missile damage to structures, systems, and components important to safety is acceptably low.

3.5.1.4 Missiles Generated by Natural Phenomena

GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena, and GDC 4 requires that these same plant features be protected against missiles. The staff reviewed the design of the facility for protecting SSCs important to safety from missiles generated by natural phenomena in accordance with Section 3.5.1.4 of the SRP. The design is considered to be in compliance with GDC 2 and 4 if it meets the guidance of RGs 1.76, "Design Basis Tornado for Nuclear Power Plants," Positions C.1 and C.2, and RG 1.117, "Tornado Design Classification," Positions C.1 through C.3. Conformance with the SRP acceptance criteria forms the basis for concluding that the design of the facility for providing protection against missiles generated by natural phenomena meets the applicable requirements of GDC 2 and 4 with respect to protection against natural phenomena and missiles.

The staff identified the following issues as DSER Open Item 3.5.1.4-1:

- incorporation of RAI responses into the SSAR
- provide information regarding missile protection for systems classified under RTNSS and DID systems

Westinghouse provided additional information and revisions to the SSAR to address the issues which the staff has reviewed in the following evaluation and found acceptable. Therefore, DSER Open Item 3.5.1.4-1 is closed.

The missiles generated by natural phenomena that are of concern are those resulting from tornados. The tornado missile spectrum used by Westinghouse is Spectrum I, as identified in SRP 3.5.1.4. The URD of EPRI for the ALWR Passive Plant requires that the selection of a tornado missile spectrum shall be in accordance with ANSI/ANS 2.3, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites" and meets the intent of current SRP criteria.

In Section 3.3.2.1 of the SSAR, Westinghouse provides the following design parameters for the design-basis tornado (DBT):

- maximum wind speed - 483 km/hr (300 mph)
- maximum rotational speed - 386 km/hr (240 mph)
- maximum translational speed - 97 km/hr (60 mph)

- radius of maximum rotational wind from center of DBT - 46 m (150 ft)
- atmospheric pressure drop - 14 kPa (2 psia)
- rate of pressure change - 8 kPa/sec (1.2 psia/sec)

These design parameters are selected based on the maximum wind speed of the eastern region of the United States in accordance with NUREG/CR-4664, "Tornado Climatology of the Contiguous United States," dated May 1, 1988. Westinghouse stated that the design parameters are consistent with the ALWR URD for passive plant design that bound the tornado hazard anywhere in the contiguous United States. The staff finds that the selected spectrum conforms to a site with a tornado velocity less than 483 km/hr (300 mph) and the parameters for the design basis tornado are acceptable.

An evaluation of the protection afforded safety-related equipment from the identified tornado missiles including compliance with RG 1.117 is discussed separately in Section 3.5.2 of this report. An evaluation of the design of missile barriers and protective structures to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this report.

On the basis of its review, the staff concludes that the AP600 design for protecting against tornado missiles meets the requirements of GDC 2 and 4 with respect to protection against natural phenomena and missiles and meets the guidance of RGs 1.76 and 1.117 with respect to identification of missiles generated by natural phenomena. Therefore, the staff concludes that the tornado-missile spectrum is properly selected for a reference site, if the reference site meets the guidelines Section 3.5.1.4 of the SRP.

3.5.1.5 Missiles Generated by an Event Near the Site

In Section 3.5.1.5 of the SSAR, Westinghouse states that the site interface is established to address site-specific missiles in the COL application. The AP600 missile interface criteria are based on the tornado missiles described in Section 3.5.1.4 of the SSAR. Additional analyses are required to evaluate other site-specific missiles. Each COL applicant referencing the AP600 will provide analyses of accidents external to the nuclear plant. The determination of the probability of occurrence of potential accidents which have severe consequences are based on analyses of available statistical data on the occurrence of the accident on the plant's safety-related structures and components. If an accident is identified for which the probability of severe consequences is unacceptable, specific changes to the AP600 plant will be identified in the COL application. However, in the DSER, the staff requested that Westinghouse provide the site-specific probability of severe accident consequences in the SSAR. In Section 3.5.1.5 of the SRP, the staff recommends the threshold probability of the missiles striking a vulnerable critical area of plant as 1.0E-7 per year. This was identified as DSER Open Item 3.5.1.5-1.

In Revision 2 to Section 2.2 of the SSAR, Westinghouse specifies the threshold of the total annual frequency of occurrence as 1.0E-6 per year for all external event-induced accidents leading to severe consequences. Based on the SRM dated June 26, 1990, in which the Commission approved the overall mean frequency of a large release of radioactive material to the environment from a reactor accident as less than one in one million per year of reactor operation, this is acceptable. Therefore, DSER Open Item 3.5.1.5-1 is closed.

3.5.1.6 Aircraft Hazards

In Section 3.5.1.6 of the SSAR, Westinghouse establishes the site interface to address aircraft hazards in the COL application. The AP600 missile interface criteria are based on the tornado missiles described in Section 3.5.1.4 of the SSAR. Additional analyses are required to evaluate other site-specific missiles. Each COL applicant referencing the AP600 will provide analyses of accidents external to the nuclear plant. The determination of the probability of occurrence of potential accidents which could have severe consequences will be based on analyses of available statistical data on the occurrence of the accident on the plant's safety-related structures and components. If an accident is identified for which the probability of severe consequences is unacceptable, specific changes to the AP600 will be identified in the COL application. However, in the DSER, the staff requested that Westinghouse provide the site specific probability of occurrence of aircraft accidents in the SSAR. In Section 3.5.1.6 of the SRP, the staff recommends the threshold probability of aircraft accidents as 1.0E-7 per year. This was identified as DSER Open Item 3.5.1.6-1.

In Revision 2 to Section 2.2 of the SSAR, Westinghouse specifies the threshold of the total annual frequency of occurrence as 1.0E-6 per year for all external event-induced accidents leading to severe consequences. Based on the SRM dated June 26, 1990, in which the Commission approved the overall mean frequency of a large release of radioactive material to the environment from a reactor accident as less than one in one million per year of reactor operation, this is acceptable. Therefore, DSER Open Item 3.5.1.6-1 is closed.

3.5.2 Protection From Externally-Generated Missiles

The staff reviewed the AP600 design for its ability to protect SSCs important to safety against tornado-generated missiles in accordance with Section 3.5.2 of the SRP. The SRP acceptance criteria specify that the design shall meet GDC 2 and 4 with respect to protection against natural phenomena and missiles and RGs 1.13 and 1.27 concerning tornado missile protection for safety-related SSCs, including stored spent fuel and ultimate heat sink. The selection of tornado regions as identified in RG 1.76 for a specific plant site will be determined by a COL applicant. The tornado missile spectrum for the AP600 design is discussed in Section 3.5.1.4 of this report. The staff's review of externally-generated missiles does not include turbine missiles to meet RG 1.115; turbine missiles are evaluated in Section 3.5.1.3 of this report.

The staff identified the following issues as DSER Open Item 3.5.2-1:

- incorporation of RAI responses into the SSAR
- provide information regarding missile protection for systems classified under RTNSS and DID systems
- SSAR discrepancies

Westinghouse provided additional information and revisions to the SSAR to address the issues which the staff has reviewed in the following evaluation and found acceptable. Therefore, DSER Open Item 3.5.2-1 is closed.

In Section 3.5.2 of the SRP, the staff states that the SSCs required for safe shutdown of the reactor be identified. The identification of SSCs to be protected against externally-generated missiles is acceptable if it is in accordance with the requirements of GDC 2 and 4. These SSCs are identified in Section 7.4 of the SSAR. The structural design requirements for the shield building and auxiliary building are outlined in Section 3.8.4 of the SSAR. Opening through external walls are evaluated on a case-by-case basis to ensure that a missile passing through the opening would not prevent a safe shutdown of the plant and would not result in an offsite release exceeding the limits of 10 CFR Part 100. The COL applicant will evaluate site-specific hazards for external events that may produce missiles more energetic than tornado missiles.

In Section 3.5.2 of the SRP, the staff states that the SSCs to be protected from externally-generated missiles must meet the requirements of GDC 2 and 4 by meeting Regulatory Position C.2 of RG 1.13, "Spent Fuel Facility Design Basis;" Positions C.2 and C.3 of RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants;" Position C.1 of RG 1.115, "Protection Against Low Trajectory Turbine Missiles;" and Positions C.1, C.2, and C.3 of the Appendix to RG 1.117, "Tornado Design Classification."

The spent fuel pool meets Regulatory Position C.2 of RG 1.13, "Spent Fuel Facility Design Basis," because it is protected from externally-generated missiles by the reinforced concrete walls and roof of the auxiliary building.

Positions C.2 and C.3 of RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants," address the use of water sources as heat sinks. The AP600 plant uses the atmosphere as the ultimate heat sink. A baffle located between the containment and the shield building sustains the natural circulation that provides for air flow over the containment shell to carry heat away. In Revision 18 to Appendix A of the SSAR, Westinghouse revised the clarification for criteria C.2 of RG 1.27 to include this response.

On the basis of its review, the staff concludes that the AP600 design for protecting SSCs against externally-generated missiles is in accordance with requirements of GDC 2 and 4 with respect to missile and environmental effects and the guidelines of RG 1.13 (Position C.2), RG 1.27 (Positions C.1, C.2, and C.3), RG 1.115, and RG 1.117 (Position C.1, C.2, and C.3) with respect to protection of safety-related plant features from tornado missiles, including stored spent fuel and the ultimate heat sink. Therefore, the staff concludes that AP600 design for providing protection from externally-generated missiles meets the guidelines of Section 3.5.2 of the SRP.

3.5.3 Barrier Design Procedures

Missile barriers and protective structures are designed to withstand and absorb missile impact loads to avoid damage to safety-related SSCs to satisfy the requirements of GDC 2 and 4 with respect to the capability of withstanding and the protection from the tornado-generated missiles. The staff reviewed the design of seismic Category I SSCs to determine if they are shielded from, or designed to withstand, various postulated missiles using the guidance of Section 3.5.3 of the SRP. Section 3.5.3 of the SSAR contains information on procedures used in the design of the structures, shields, and barriers to resist the effects of missiles. The effects of missile impact on structures are two-fold: local damage and overall damage.

For the prediction of local damage from missiles, Westinghouse provided information on the procedures used in the design of concrete and steel structures. Westinghouse applied the modified National Defense Research Committee (NDRC) formula as shown in Section 3.5.3 of the SSAR, analytically for missile protection in concrete. To prevent missile perforation, Westinghouse used the minimum thickness required for missile shields as the thickness just perforated. The staff finds that the use of the modified NDRC formula for missile penetration and a thickness equal to or greater than the minimum required as specified in Table 1 of Section 3.5.3 of the SRP will result in sufficient concrete barrier thickness to prevent barrier perforation and, when necessary, prevent spalling or scabbing, and is, therefore, acceptable.

Westinghouse used either the Ballistic Research Laboratory (BRL) or Stanford formulae for missile penetration in steel. As discussed in Section 3.5.3 of the SRP, the staff finds the use of either formula to be acceptable. Composite barriers are not used in the AP600 design and are, therefore, not discussed.

For the prediction of overall damage, Westinghouse stated that structural members designed to resist missile impact are designed for flexural, shear, and buckling effects using the equivalent static load obtained from the evaluation of structural response. Stress and strain limits for the equivalent static load comply with applicable codes and RG 1.142. As stated in Section 3.5.3 of the SRP, the staff finds the use of RG 1.142 for concrete to determine the overall damage prediction to be acceptable. In the response to RAI 220.84 dated May 17, 1994, Westinghouse revised penultimate paragraph of Section 3.5.3 of the SSAR and added Section 3.5.3.1 to the SSAR to indicate the limits on ductility of steel structures. These ductility limits are the same as those specified in Appendix A to Section 3.5.3 of the SRP and are therefore, acceptable.

The staff finds that the procedures used for determining the effects and loadings on seismic Category I structures and missile shields and barriers induced by design-basis missiles selected for the plant are acceptable because they provide a conservative basis for engineering design to ensure that the structures or barriers will adequately withstand the effects of such forces.

The use of these procedures provides reasonable assurance that if a design-basis missile should strike seismic Category I structures or other missile shields and barriers, the structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structures will, therefore, be adequately protected against the effects of missiles and will be capable of performing their intended safety functions. Conformance with these procedures is an acceptable basis for satisfying the requirements of GDC 2 and 4, as they relate to the capabilities of the structures, shields, and barriers to provide sufficient protection to equipment that must withstand the effects of natural phenomena (tornado missiles) and environmental effects including the effects of missiles, pipe whipping, and discharging fluids.

As discussed above, Westinghouse used acceptable procedures in its barrier design. Thus, the staff finds that the barrier design procedures are acceptable and meet the guidelines of Section 3.5.3 of the SRP and GDC 2 and 4 with respect to the capabilities of the structures, shields, and barriers to provide sufficient protection to the safety-related SSCs.

In Section 3.5.4 of the SSAR, Westinghouse commits that the COL applicant demonstrate that the site satisfy the interface requirements provided in Section 2.2 of the SSAR. This requires an evaluation for those external events that produce missiles that are more energetic than the tornado missiles postulated for the AP600 design, or additional analyses of the AP600 capability to handle the specific hazard. This is an acceptable interface requirement.

3.6 Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

The staff reviewed the AP600 design as it relates to the protection of safety-related SSCs against postulated piping failures in fluid systems outside containment in accordance with Section 3.6.1 of the SRP. The SRP acceptance criteria specify that the design meet the requirements of GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to accommodating the dynamic effects of postulated pipe rupture, including the effects of pipe whipping and discharging fluids. The design is in compliance with GDC 4 when it conforms with Branch Technical Position (BTP) SPLB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," with regard to high- and moderate-energy fluid systems outside containment.

In BTP SPLB 3-1, the staff specifies that postulated piping failures in fluid systems outside containment should not cause a loss of function of essential safety-related systems. The BTP also specifies that nuclear plants should be able to withstand postulated failures of any fluid system piping outside containment, taking into account the direct results of such failure and the further failure of any single active component with acceptable offsite consequences.

In Section 3.6.1 of the SSAR, Westinghouse provides the design basis and criteria for the analysis required to demonstrate that safety-related systems are protected from pipe ruptures. This SSAR section enumerates the high- and moderate-energy systems which are potential sources of the dynamic effects associated with pipe ruptures and defines separation criteria. In Section 3.6.2 of the SSAR, Westinghouse provides the criteria for postulated pipe rupture location and configuration.

Non-safety-related systems for the AP600 plant, including those that are determined to be important by the RTNSS process and DID systems, are not required to be protected from the dynamic and environmental effects associated with pipe rupture. Protection against pipe rupture is not an RTNSS important mission for non-safety-related systems.

By design, non-seismic piping is not routed near safety-related piping or equipment. If there is non-seismic, moderate-energy piping whose continued function is not required, but whose failure or interaction could degrade the functioning of safety-related equipment to an unacceptable level, then this piping is analyzed and designed for the SSE using the same methods as specified for seismic Category I piping. Safety-related systems required for safe shutdown are not expected to be adversely affected by the dynamic effects of postulated pipe ruptures in non-seismic, moderate-energy piping.

Evaluations of the dynamic effects for postulated pipe breaks which meet the mechanistic pipe break or LBB criteria are eliminated from pipe failure analysis for the AP600 design. Such evaluations include reactor coolant loop branch piping and MS piping out to containment penetrations adjacent to containment isolation valves, and other primary piping inside containment (which is equal to or greater than 15.2-cm (6-in) nominal pipe size). Many of the high- and moderate-energy piping systems meet the LBB criteria, and therefore, are not subject to the dynamic effects associated with a pipe failure.

The AP600 design as it relates to the mechanistic pipe break (or LBB) is addressed in Section 3.6.3 of the SSAR. High-energy fluid system piping that meets the LBB criteria is evaluated for the effects of leakage cracks. High- and moderate-energy fluid system piping which do not meet the LBB criteria are evaluated for the dynamic effects of postulated pipe failures. Safety-related equipment subject to the resulting dynamic effects of pipe failures are protected from these dynamic effects by protective structures, pipe restraints, and separation.

In Section 3.6.1 of the SSAR, Westinghouse identifies those safety-related systems which require protection from the dynamic effects of postulated pipe failures. These systems include the RCS, PXS, PCS, and the steam generator system (SGS). In addition, the protection and safety monitoring system, Class 1E dc system, uninterruptible power supply (UPS) system, and main control room (MCR) and MCR habitability systems are also protected from pipe failures. Finally, containment penetrations and isolation valves, including those for non-safety-related systems, are protected from pipe failures.

In Section 3.6.1 of the SSAR, Westinghouse also provides the design bases related to the evaluation of pipe failure effects. The selection of the pipe failure type is based on whether the system is high- or moderate-energy during normal operating conditions of the system. High-energy systems are defined as those systems or portions of systems containing fluid where the maximum normal operating temperature exceeds 93.3 °C (200 °F) and/or the maximum normal operating pressure exceeds 1999.5 kPa (275 psig). Moderate-energy systems are defined as those systems or portions of systems whose pressures exceed atmospheric pressures during normal operation but are less than 1999.5 kPa (275 psig). In addition, those systems that exceed 93.3 °C (200 °F) and 1999.5 kPa (275 psig) for 2 percent or less of the time during which the system is in operation are defined as moderate-energy. Based on these definitions, in Table 3.6-1 of the SSAR, Westinghouse identifies the high- and moderate-energy fluid systems in the AP600 design.

As determined by the appropriate criteria, pipe failure evaluations are made based on circumferential or longitudinal pipe breaks, through-wall cracks, or leakage cracks. Subcompartment pressurization, jet impingement, jet reaction thrust, internal fluid decompression loads, spray wetting, flooding, and pipe whip are considered for pipe breaks in high-energy fluid piping. Spray wetting and flooding are considered for high- and moderate-energy through-wall and leakage cracks. Pressurization effects on SSCs are considered for both breaks and leakage cracks. Structures inside containment are evaluated for pressurization effects. Through-wall cracks are not postulated in the break exclusion zone. Pressurization, spray wetting, and flooding effects for pipe failures in the break exclusion zone for high-energy piping (including MS and MFW piping) near containment penetrations assume a 0.093 m^2 (1 ft^2) break. Postulated break, through-wall crack, and leakage crack locations are determined in accordance with Sections 3.6.2 and 3.6.3 of the SSAR.

Other design-basis assumptions used in the dynamic effects analysis for pipe failures include:

- Offsite power is not required for actuation of the passive safety systems. Only the Class 1E dc and UPS electrical systems are required to function.

- A single active component failure (SACF) occurs in systems needed to mitigate the consequences of the piping failure or to safely shut down the reactor. The SACF occurs in addition to the pipe failure (including any direct consequences of the pipe failure, such as a unit trip or loss of offsite power (LOOP)).

- Secondary components (e.g., turbine stop, moisture separator reheater stop, and turbine bypass valves) are credited with mitigating the consequences of a postulated steamline break (given a SACF).

- A whipping pipe can break pipes of smaller diameter, regardless of pipe-wall thickness and can cause a through-wall crack in pipe of equal or larger size with equal or thinner wall thickness.

- If the direction of the initial pipe movement caused by the thrust force is such that the pipe impacts a flat surface normal to its direction of travel, it is assumed that the pipe comes to rest against the surface with no pipe whip in other directions. Pipe whip restraints are used wherever pipe breaks could impair the functioning of safety-related systems or components.

- Regarding components impacted by jets from breaks in high-pressure fluid piping, components within 10 diameters of the broken pipe are assumed to fail while components beyond 10 diameters of the broken pipe do not fail.

- When the mechanistic pipe break approach is used, subcompartment pressure loads on safety-related structures and components are determined by the leakage crack used in the mechanistic pipe break approach. In subcompartments containing piping not qualified for LBB, the pressurization effects are determined from the pipe with the greatest effect.

- Where a non-safety-related high-energy system failure could cause a failure of a safety-related system or a non-safety-related system whose failure could affect a safety-related system, pipe whip protection is evaluated.

- Steam, water, gases, heat, and combustible or corrosive fluids which escape from a pipe rupture will not prevent:

  - subsequent access to any areas to recover from the pipe rupture

  - habitability of the MCR

  - capability of safety-related instrumentation, electric power supplies, components, and controls from performing their safety functions

In Section 3.6.1 of the SSAR, Westinghouse states that equipment is adequately separated from the dynamic effects of a postulated pipe failure when the equipment is in a different compartment and the compartment walls are designed to withstand the dynamic effects. For pipe whip, adequate separation is based on the distance between the equipment and the pipe, and the length of the whipping pipe. For jet impingement, equipment located more than 10 pipe diameters from the source of the jet is considered to be adequately protected from the jet.

In subcompartments inside containment (except the IRWST and reactor vessel annulus), which contain piping no greater than 0.93 cm (3 in.) in diameter, the pressurization analysis and the evaluation of venting provisions are based on a 0.93-cm (3-in.) pipe break. The pressurization loads for the IRWST are based on the loads due to the maximum discharge of the first, second, and third stages of the ADS valves. The pressurization loads for the reactor vessel annulus are based on a 18.9 L/min (5 gpm) leakage crack in the primary loop piping.

The MS and MFW lines are the closest piping to the MCR. They are located in the MS isolation valve subcompartment (part of break exclusion area) which is separated from the MCR by two structural walls composed of thick, reinforced concrete. (Between these walls is the portion of the control room used for nonessential office and administrative space for the MCR.) The MS isolation valve subcompartment is evaluated for the effects of flooding, spray wetting, and pressurization resulting from a 0.093 m^2 (1 ft^2) break in the MS or MFW line. The subcompartment wall closest to the MCR is also evaluated for the jet impingement resulting from a 0.093 m^2 (1 ft^2) longitudinal break in the MS or MFW line. The MCR is also evaluated for the dynamic and environmental effects resulting from line breaks in the auxiliary and turbine buildings. (The remote shutdown work station is not subject to adverse effects from high-energy pipe breaks.)

In Section 3.6.1 of the SSAR, Westinghouse provides the protection measures used in the AP600 design to protect safety-related equipment from the dynamic effects of pipe failures. These measures include physical separation of systems and components, barriers, equipment shields, and pipe whip restraints. The specific method used depends on objectives such as adequate allowance for equipment accessibility and maintenance.

Separation between redundant safety systems is the preferred method used to protect against the dynamic effects of pipe failures. Separation is achieved using the following design features:

- locating safety-related systems away from high-energy piping
- locating redundant safety systems in separate compartments
- enclosing specific components to ensure protection and redundancy
- providing drainage systems for flood control

During the staff's initial review of the AP600 design for protection of safety-related SSCs against postulated piping failures, the following issues were identified and documented in the DSER for resolution:

- adequacy of responses to RAI
- incorporation of RAI responses into the SSAR
- information regarding piping failure protection for systems classified under RTNSS and DID systems
- mechanistic pipe break

Collectively, these issues constituted DSER Open Item 3.6.1-1. During the process of issue resolution, Westinghouse provided applicable information regarding this open item. Based on the information as discussed above, DSER Open Item 3.6.1-1 is closed.

The staff concludes that the AP600 design, as it relates to the protection of safety-related SSCs from the effects of piping failures outside containment, meets the requirements of GDC 4 with respect to accommodating the effects of postulated pipe failures and the guidelines of SRP Section 3.6.1, and therefore, is acceptable.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

In GDC 4, the NRC requires, in part, that SSCs important to safety be designed to be compatible with and to accommodate the effects of the environmental conditions resulting from postulated accidents, including LOCAs. The NRC also requires that they be adequately protected against dynamic effects (including the effects of pipe whipping and discharging fluids) that may result from postulated pipe rupture events.

To address this portion of GDC 4, in Section 3.6.2 of the SSAR Westinghouse describes the following items:

- the design bases for locating postulated breaks and cracks in high- and moderate-energy piping systems inside and outside the containment
- the procedures used to define the jet thrust reaction at the break location and the jet impingement loading on adjacent essential SSCs
- design criteria for pipe whip restraints, jet impingement barriers and shields, and guardpipes

The staff reviewed Section 3.6.2 of the SSAR, up to and including Revision 23, and, as discussed in Sections 3.6.2.1, 3.6.2.2, 3.6.2.3, and 3.6.2.4 of this report, found that it conforms with the guidelines of Section 3.6.2 of the SRP, including BTP MEB 3-1, and is acceptable.

In one of the guidelines in BTP MEB 3-1, the staff states that the analyses for the maximum stresses, stress ranges, and usage factors to be used for determining postulated high- and moderate-energy pipe break and crack locations should be based on loads that include the OBE. In SECY-93-087, the staff recommended the elimination of the OBE in the design process on the basis that it would not result in a significant decrease in the overall plant safety margin. In an SRM dated July 21, 1993, the Commission approved the staff's recommendations. The detailed basis for the staff's recommendation is discussed in Section 3.1.1 of this report. Westinghouse incorporated these acceptable criteria in Section 3.6.1 and 3.6.2 of the SSAR. Therefore, the staff's evaluation of these section of the SSAR is based on the Commission-approved staff recommendations that are discussed in Section 3.1.1 of this report.

3.6.2.1 High- and Moderate-Energy Piping Systems

As discussed in Section 3.6.1 of this report, the staff found that the criteria in Item A of Section 3.6.1.1 of the SSAR and Appendix 3E of the SSAR, Revision 0 through 4, for the definition of high- and moderate-energy piping systems are consistent with the criteria in Appendix A of BTP ASB 3-1 in SRP Section 3.6.1, and are, therefore, acceptable. The staff also found that Westinghouse specified that piping systems which operate as high-energy for 2 percent or less of the time during which the system is in operation, or for less than one percent of the plant operation time, are considered moderate-energy systems. In the DSER, the staff questioned the acceptability of the 1 percent or less of plant operating time criterion because it did not appear to be completely consistent with the BTP MEB 3-1 criteria in Section 3.6.2 of the SRP for moderate-energy fluid systems that qualify as high-energy fluid systems for only a short operational period, but qualify as moderate-energy fluid systems for the major operational period. In Footnote 5 to BTP MEB 3-1, the staff specifies that the operational period is considered "short" if the fraction of the time that the system operates within the high-energy pressure-temperature conditions is about two percent of the time that the system operates as a moderate-energy fluid system. This was identified as DSER Open Item 3.6.2.1-1. Subsequent to the issuance of the DSER, the staff evaluated the issue in question and concluded that no matter which definition of short operational period is used (1 vs. 2 percent), the resulting time from either definition is short enough so that the likelihood of a break occurring during either period is small. In addition, it should be noted that the staff approved the 1 percent of plant operating time criterion for several recently licensed PWRs and BWRs. On this basis, the staff concludes that the definitions for a short operational period in Section 3.6.1.1A of the SSAR are consistent with the applicable guidelines in SRP Section 3.6.2, BTP MEB 3-1, and with definitions in recently licensed nuclear plants, and are acceptable. Therefore, DSER Open Item 3.6.2.1-1 is closed.

3.6.2.2 High-Energy Piping in Containment Penetration Areas (Break Exclusion Areas)

In SRP Section 3.6.2, BTP MEB 3-1, the staff states that breaks need not be postulated in portions of high-energy fluid system piping located in the containment penetration area both inside and outside the containment, and that are designed to meet Article NE-1120 of Section III of the ASME Code and additional guidelines specified in BTP MEB 3-1. The staff evaluated the information in Section 3.6.2 of the SSAR to determine if acceptable commitments to these guidelines are provided for the AP600 design. In Section 3.6.2.1.1.4 of the SSAR, Westinghouse identifies those portions of AP600 piping systems that qualify for break exclusion. In Section 3.6.2.1.1.4, Westinghouse also provides the bases for these break exclusion areas which agree with the guidelines in SRP Section 3.6.2, BTP MEB 3-1, and with staff-approved break exclusion areas in several of the most recently licensed PWRs, and are acceptable. One exception to this list of break exclusion areas was due to the staff's position of postulating breaks in the MS isolation valve compartment, which is discussed below as a part of the resolution of DSER Open Item 3.6.2.2-1. The staff's evaluations of several issues related to the break exclusion areas are also discussed below.

One important guideline is that an augmented ISI program be implemented for those portions of piping within the break exclusion zone. In Revision 12 to Section 3.6.2.1.1.4 of the SSAR, Westinghouse provided a commitment to such a program for all piping in the break exclusion zone that is 7.6 cm (3 in.) in diameter or larger. Since this commitment is beyond the requirements of the ASME Section XI Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," and meets the applicable guidelines of SRP 3.6.2, it is acceptable.

In the AP600 design, the east wall of the east MS isolation valve (MSIV) compartment, which houses the MS and FW piping break exclusion zones, is adjacent to the MCR. In addition, safety-related electrical equipment is located in the room below this same compartment. Although these portions of the MS and FW piping are in a break exclusion area, in Section B.1.a(1) of SRP Section 3.6.1, BTP ASB 3-1, the staff states that essential equipment in this area must be protected from a postulated break which has a cross sectional area of at least 0.093 m^2 (1 ft^2). In the response to RAI 210.76 dated July 27, 1994, Westinghouse stated that the east wall is evaluated for jet impingement from a 0.093-m^2 (1-ft^2) break in either the MS or FW line. Structures in the remainder of this compartment are evaluated for subcompartment pressure effects due to the worst case of the 34.5 kPa (5 psi) minimum design pressure and 0.093-m^2 (1-ft^2) ruptures in the MS or FW line. This 0.093-m^2 (1-ft^2) rupture design criterion is consistent with Section B.1.a(1) of SRP Section 3.6.1, BTP ASB 3-1. However, the 0.093-m^2 (1-ft^2) break criterion in Section 3.6.1 of the SRP was based, in part, on a plant design which does not have MS and FW lines routed in the vicinity of the MCR, and which does not have safety-related equipment nearby. Therefore, the staff requested Westinghouse to design: (1) the east wall of the east MSIV compartment between the MCR and the compartment, and (2) the floor slab of the east MSIV compartment between the compartment and the safety-related electrical equipment room to accommodate the worst case MS or FW line break. This was identified as DSER Open Item 3.6.2.2-1. In Revision 4 of the SSAR, Westinghouse revised Sections 3.6.1 and 3.6.1.2.2 of the SSAR to agree with the staff's request. The staff finds this commitment acceptable, and therefore, DSER Open Item 3.6.2.2-1 is closed.

In Part a of RAI 210.40, the staff requested Westinghouse to revise the first paragraph of Section 3.6.2.1.1.4 of the SSAR, Revision 0, to include a commitment that when guard pipes are used in high-energy piping in containment penetration areas, the enclosed portion of the fluid system piping should not only be seamless, but should not contain circumferential welds unless specific access provisions are made in the guard pipe to permit inservice volumetric examination of these welds in accordance with the augmented inservice inspection provisions. If applicable, inspection ports in the guard pipe should not be located in that portion of the guard pipe passing through a shield building annulus.

In the response to RAI 210.40 dated June 27, 1994, Westinghouse stated that Section 3.6.2.1.1.4 of the SSAR, would be revised to state that there are no circumferential or longitudinal welds in the piping enclosed within the guard pipe, thereby obviating the need for augmented inservice inspections in this area. This response is consistent with Section 3.6.2 of the SRP, and is acceptable. The inclusion of this response in the SSAR was identified as

DSER Confirmatory Item 3.6.2.2-1. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.1.1.4 to provide the above commitment. Therefore, DSER Confirmatory Item 3.6.2.2-1 is closed.

In the responses to RAI 210.44 and RAI 210.45 dated July 8, 1994, Westinghouse agreed to revise Sections 3.6.2.4 and 3.6.2.4.2 of the SSAR, to clarify the difference between guard pipes in piping break exclusion zones and auxiliary guard pipes. Guard pipes in the break exclusion zones will be designed to the criteria in Section 3.6.2.1.1.4 of the SSAR, and auxiliary guard pipes will be designed and constructed to the same ASME rules as the enclosed pipe. These criteria are consistent with Section 3.6.2 of the SRP, and are acceptable. This was identified as DSER Confirmatory Item 3.6.2.2-2. In Revision 4 to the SSAR, Westinghouse revised Sections 3.6.2.4 and 3.6.2.4.2 to provide the above commitment. Therefore, DSER Confirmatory Item 3.6.2.2-2 is closed.

The staff concludes that, on the basis of the above discussions, the criteria in Revision 11 to Section 3.6.2.1.1.4 of the SSAR are consistent with SRP Section 3.6.2, BTP MEB 3-1, and with approved break exclusion areas in recently licensed PWRs. Therefore, the AP600 pipe break exclusion areas identified in the SSAR are acceptable.

3.6.2.3 Pipe Rupture Criteria Outside the Containment Penetration Area

In Section 3.6.2.III.1.b of the SRP, the staff states that for final design approval, the SSAR should include the following items:

- sketches of applicable piping systems showing the location, size, and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers

- a summary of the data developed to select postulated break locations, including calculated stress intensities, cumulative usage factors, and stress ranges

These sketches and data summaries were not available. Westinghouse plans to complete the AP600 pipe break analysis in the future as part of design certification, subsequent to which the sketches and data summaries will be available. In the DSER, Westinghouse was requested to inform the staff when these analyses would be available for staff review. This was identified as DSER Open Item 3.6.2.3-1. In Revision 10 of the SSAR, Westinghouse responded to this open item by revising Section 3.6.2.5, "Evaluation of Dynamic Effects of Pipe Ruptures on As-Built Piping Systems" to provide a description of the pipe break hazards analysis activities which will be completed by the COL applicant. The description of the hazards analysis included the information requested in this open item and is acceptable. In Revision 11 to Section 3.6.4.1 of the SSAR, "Combined License Information, Pipe Break Hazards Analysis," Westinghouse included a commitment that the COL applicant will address as-built reconciliation of the pipe break hazards analysis in accordance with the criteria outlined in Sections 3.6.1.3.2 and 3.6.2.5 of the SSAR. In addition, the pipe break hazards analysis is a part of the inspections, tests, analyses, and acceptance criteria in Table 3.3-6 of the AP600 Tier 1 Material. The information and commitments discussed above are consistent with applicable guidelines in Section 3.6.2 of the SRP and are acceptable. Therefore, DSER Open Item 3.6.2.3-1 is closed.

For ASME Class 1 piping, the staff position for postulating pipe breaks is delineated in BTP MEB 3-1 of SRP Section 3.6.2. Before the NRC issued Revision 2 to BTP MEB 3-1 in June 1987, breaks were postulated at intermediate locations between terminal ends of a pipe run if the maximum stress range as calculated by Equation (10) > 2.4 S_m and if the maximum stress range as calculated by either Equations (12) or (13) > 2.4 S_m, where Equations (10), (12), and (13) and S_m are as defined in Subsection NB-3653 of Section III of the ASME Code. This staff position is implemented in many plants operating today. In Revision 2 to BTP MEB 3-1, the same criteria were maintained for break exclusion in the containment penetration areas. However, for other areas, the criteria were revised to require that breaks be postulated at any intermediate locations when only Eq. (10) > 2.4 S_m. The use of Equations (12) and (13) was eliminated. This resulted in an inconsistency in the Revision 2 criteria in that they allow higher limits in the containment penetration areas than in other areas. The break exclusion areas should provide a margin greater than (or at least equal to) the margin for areas outside the break exclusion area. To determine the impact of this inconsistency, the staff obtained several independent analyses for both BWRs and PWRs that compared the number of postulated pipe breaks resulting from the use of Revisions 1 and 2 criteria. These analyses indicated that the Revision 2 criteria will result in a significant increase in the number of postulated breaks, which may be counter productive in terms of improving plant safety. Therefore, the staff recommended that Section 3.6.2 of the SRP be revised to reinstate the Revision 1 criteria related to allowing the use of Equations (12) and (13) for the postulation of intermediate pipe breaks in ASME Class 1 piping systems. During piping design review meetings, Westinghouse committed to revise the pipe break criteria in Sections 3.6.1 and 3.6.2 of the SSAR to be consistent with the Revision 1 criteria as previously discussed. This was identified as DSER Confirmatory Item 3.6.2.3-1. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.1.1.1 to provide the above commitment. On the basis of the above discussion, the staff concludes that this is an acceptable deviation from Revision 2 to BTP MEB 3-1 of SRP Section 3.6.2. Therefore, DSER Confirmatory Item 3.6.2.3-1 is closed.

In the DSER, the staff reported that in Revision 0 to Section 3.6.1 of the SSAR, Westinghouse indicated that structures inside containment containing high-energy piping are evaluated for pressurization loads due to a break area equivalent to a 7.6-cm (3-in.) NPS primary system pipe. During the piping design review meetings, the staff informed Westinghouse that even if LBB is approved in a particular subcompartment, the 7.6-cm (3-in.) break might not be the controlling design criteria. The staff's position is that a minimum subcompartment pressure must be determined for designing the subcompartment walls and floors. This pressure should bound the effects of a high energy intermediate pipe break. This was identified as DSER Open Item 3.6.2-1. In Revision 11 to the SSAR, Westinghouse provided the final AP600 response to this open item. The 7.6-cm (3-in.) break criterion was deleted from Section 3.6.1 of the SSAR, and Sections 3.8.3.5 and 3.8.4.3.1.4 are referenced for pressurization loads on structures. In Sections 3.8.3.5 and 3.8.4.3.1.4, Westinghouse states that subcompartments inside and outside containment containing high-energy piping are designed for a pressurization load of 34.5 kPa (5 psi), with the exception of the pipe tunnel in the CVS room, which is designed to 51.7 kPa (7.5 psi). These SSAR sections further state that both of these subcompartment design pressures bound the pressurization effects due to postulated breaks in high energy pipe inside the subcompartment. These design criteria satisfy the applicable guidelines in Section 3.6.1 of the SRP and are acceptable. Therefore, DSER Open Item 3.6.2-1 is closed.

In RAI 210.77, the staff indicated that the table in Revision 1 to WCAP-13054, "AP600 Compliance with the SRP Acceptance Criteria," that addresses Section 3.6.2 of the SRP lists

Sections B.1.c.(5) and B.3.c.(4) of BTP MEB 3-1 as acceptable for the AP600 design. Both of these guidelines relate to qualifying equipment for environmental (temperature, pressure, and humidity) effects. Several portions of Section 3.6.2 of the SSAR briefly mention requirements for considering environmental effects. For example, Section 3.6.2.1.1.4 of the SSAR provides a commitment to evaluate leakage cracks in MS&FW lines in the containment penetration area. However, Revision 0 to Section 3.6.2 of the SSAR did not appear to contain any detailed discussion relative to the guidelines in the two MEB 3-1 sections. Accordingly, in RAI 210.77, the staff requested that Westinghouse revise Section 3.6.2 to include a commitment to these guidelines and provide a description of how environmental effects will be considered in the AP600 design of high- and moderate-energy piping systems.

Relative to the guidelines of Section B.1.c.(5) of BTP MEB 3-1 that pipe ruptures or leakage cracks be included in the design bases for environmental qualification, in the response to RAI 210.77 dated June 30, 1994, Westinghouse stated that environmental qualification is described in Revision 1 to Section 3.11 of the SSAR, and environmental conditions (temperature, pressure, and humidity) are specified in Appendix 3D. Westinghouse indicated that: (1) inside containment, environmental conditions are based on the LOCA and MS line break inside containment which envelopes other postulated breaks inside containment, and (2) outside containment, environmental conditions are based on a 0.093 m^2 (1 ft^2) break in the MS line. However, staff review of Appendix 3D found that the combined single high-energy line break profile developed for environmental qualification inside containment, as described in Section 3D.5.5.1.5 and shown in Figure 3D.5-9 of the SSAR, is also used for environmental qualification outside containment as described in Section 3D.5.5.2 of the SSAR. Although this appeared to be only a minor discrepancy, it was identified as DSER Open Item 3.6.2.3-2. However, upon further review, the staff determined that Sections 3D.5.5.1.5 and 3D.5.5.2 of the SSAR explain that Figure 3D.5-9 is a combined test envelope for environmental qualification, and is applicable to both inside and outside containment. This is acceptable, and therefore, DSER Open Item 3.6.2.3-2 is closed. On the basis of the response to RAI 210.77 dated June 30, 1994, which is discussed above, the staff concludes that the applicable information in Section 3.11 and Appendix 3D of the SSAR demonstrates that pipe ruptures and leakage cracks are considered in the design bases for environmental qualification in accordance with applicable portions of Section 3.6.2 of the SRP, and is acceptable.

In Section B.3.c.(4) of BTP MEB 3-1 of SRP Section 3.6.2, the staff states that (1) the flow from a leakage crack is assumed to result in an environment that wets all unprotected components within the compartment with consequent flooding in the compartment and communicating compartments, unless an analysis shows otherwise, and (2) flooding effects are determined on the basis of a conservatively estimated time period required to take corrective actions. In the response to RAI 210.77 dated June 30, 1994, Westinghouse committed to revise Section 3.6.2.1.3.2 of the SSAR, to include these guidelines. This was identified as Confirmatory Item 3.6.2.3-2 in the DSER. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.1.3.2 to provide a commitment to the above guidelines. This commitment is consistent with Section 3.6.2 of the SRP, and is acceptable. Therefore, DSER Confirmatory Item 3.6.2.3-2 is closed.

During the piping design review meetings, the staff indicated the following:

- Sections 3.6.2.1.1.1 through 3.6.2.1.1.3 of the SSAR, Revision 0, were not totally in agreement with the guidelines of Sections B.1.c.(1) and B.1.c.(2) of BTP MEB 3-1 in SRP Section 3.6.2 relating to results of piping reanalyses due to differences between the design and as-built configurations. Westinghouse committed to revise these sections to include these BTP MEB 3-1 guidelines. This was identified as DSER Open Item 3.6.2.3-3. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.1.1 (which is applicable to Sections 3.6.2.1.1.1 through 3.6.2.1.1.3) to agree with applicable portions of BTP MEB 3-1. The staff finds this acceptable, and therefore, DSER Open Item 3.6.2.3-3 is closed.

- The criteria in Section 3.6.2.1.1.3 of the SSAR, Revision 0, for break postulation at intermediate fittings in non-analyzed, non-ASME Code high-energy piping systems were not totally in agreement with the guidelines of Section B.1.c.(2).(b).(I) of BTP MEB 3-1 relating to crosses, flanges, and nonstandard fittings. Westinghouse committed to revise Section 3.6.2.1.1.3 of the SSAR to include these BTP MEB 3-1 guidelines. This was identified as DSER Open Item 3.6.2.3-4. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.1.1.3 to add crosses, flanges, and nonstandard fittings to the list of postulated break locations in piping not designed to the ASME Code. This is consistent with SRP 3.6.2 and is acceptable. Therefore, DSER Open Item 3.6.2.3-4 is closed.

In Section B.1.c.(4) BTP MEB 3-1, the staff states that in other than containment penetration areas, if a structure separates a high-energy line from an essential component, the separating structure should be designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure, irrespective of the fact that the pipe rupture criteria in BTP MEB 3-1 might not require such a break location to be postulated. In RAI 210.76, the staff informed Westinghouse of two concerns: (1) Revision 0 to Section 3.6.2 of the SSAR did not appear to address this BTP MEB 3-1 guideline, and (2) Revision 1 to WCAP-13054 takes exception to this criterion and states that separating structures are designed for postulated terminal end breaks and breaks at the high-stress locations. This exception is not completely acceptable. The staff requested Westinghouse to revise Section 3.6.2 of the SSAR to add a commitment to this position and delete the exception to this guideline in WCAP-13054. This was identified as DSER Open Item 3.6.2.3-5. Westinghouse responded to the major portion of this open item in Revision 10 to the SSAR. Section 3.6.2.5 was revised to provide a description of the pipe break hazards analysis. The hazards analysis is discussed above under the resolution of DSER Open Item 3.6.2.3-1. Section 3.6.1.3.2, "Protective Mechanisms," was revised to add a paragraph, in which Westinghouse states that when physical separation is not possible, the pipe rupture hazards analysis includes an evaluation to determine the systems and components that require a structure be added for separation from the effects of a break in a high-energy line. Westinghouse further states that the evaluation of these structures that were added considers that the break may be at the closest point in the line to the separating structure, not only at the break locations identified in Section 3.6.2.1.1 of the SSAR. The staff concludes that this is consistent with Section B.1.c.(4) of SRP Section 3.6.2, BTP MEB 3-1, and is acceptable. In a letter dated February 19, 1997, Westinghouse responded to the remainder of this open item by providing a draft revision of page 3-19 in WCAP-13054, which deletes the exception to SRP Section 3.6.2, BTP MEB 3-1, Section B.1.c.(4). In the latest revision, Westinghouse failed to include this change. Therefore, pending receipt of the revision to WCAP-13054, DSER Open

Item 3.6.2.3-5 is redesignated as FSER Confirmatory Item 3.6.2.3-1. In a letter dated June 2, 1998, Westinghouse provided Revision 4 to WCAP-13054 which deletes the exception to SRP Section 3.6.2, BTP MEB 3-1, Section B.1.c.(4). Therefore, FSER Confirmatory Item 3.6.2.3-1 is closed.

During piping design review meetings, Westinghouse committed to the following revisions:

- To clarify the first paragraph of Section 3.6.2.1.2.1 of the SSAR, Revision 0, to identify "other high-energy piping" as non-ASME Code high-energy piping. This was identified as DSER Open Item 3.6.2.3-6. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.1.2.1 to provide this clarification. This is consistent with SRP Section 3.6.2, and is acceptable. Therefore, Open Item 3.6.2.3-6 is closed.

- To revise Section 3.6.2.1.2.2 of the SSAR, Revision 0, to indicate that the stress limits are applicable to the sum of Equations (9) and (10) of NC/ND-3653 of the ASME Code. This was identified as DSER Open Item 3.6.2.3-7. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.1.2.2 to provide this commitment. This is consistent with SRP Section 3.6.2, and is acceptable. Therefore, DSER Open Item 3.6.2.3-7 is closed.

- To revise Section 3.6.2.1.2.2 of the SSAR, Revision 0, to provide that in the absence of stress analysis, through-wall cracks in high- and moderate-energy piping designed to non-seismic standards are postulated at locations which give the worst effects for flooding and spraying. This was identified as DSER Open Item 3.6.2.3-8. However, subsequent to issuance of the DSER, the staff determined that Section 3.6.2.1.2.2E of the SSAR, Revisions 0 through 4, contain this commitment. This is consistent with SRP Section 3.6.2, and is acceptable. Therefore, DSER Open Item 3.6.2.3-8 is closed.

In RAI 210.41, the staff observed that in Revision 0 to Section 3.6.2.3.1 of the SSAR, Westinghouse stated that if a simplified static analysis is performed instead of a dynamic analysis, the jet impingement force is multiplied by a dynamic load factor (DLF) of 1.2 to 2.0, depending upon the time variance of the jet load. The staff requested Westinghouse to specify a DLF of 2.0 or provide a more detailed basis for a 1.2 factor. In the response to RAI 210.41 dated July 8, 1994, Westinghouse committed to revise Section 3.6.2.3.1 to specify the ANSI/ANS 58.2-1988 criterion that for an equivalent static analysis of the target structure, the jet impingement force is multiplied by a DLF of 1.2 to 2.0 depending on the time variance of the jet load and the elastic plastic behavior of the target. This factor assumes that the target can be represented as essentially a one-degree of freedom system. The staff has endorsed ANSI/ANS 58.2. The revision to the SSAR was identified as DSER Confirmatory Item 3.6.2.3-3. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.3.1 to provide the above commitment. This is consistent with SRP 3.6.2 and is acceptable. Therefore, DSER Confirmatory Item 3.6.2.3-3 is closed.

In RAI 210.43, the staff observed that in Revision 0 to Section 3.6.2.3.4.2 of the SSAR, Westinghouse stated that if energy absorbing material is used in the design of pipe whip restraints, the allowable deflection is 80 percent of the maximum crushable height at uniform crushable strength. In accordance with Section 3.6.2.III.2.a of the SRP, the staff's position is that the allowable capacity of crushable material shall be limited to 80 percent of its rated energy dissipating capacity as determined by dynamic testing at loaded rates within ±50 percent of the specified design loading rate. The rated energy dissipating capacity shall not be greater than the area under the load-deflection curve as illustrated in Figure 3.6.2-1 of Section 3.6.2 of the SRP. The staff requested Westinghouse to revise Section 3.6.2.3.4.2 of the SSAR to be consistent with the staff's position. In the response to RAI 210.43 dated June 27, 1994, Westinghouse committed to revise Section 3.6.2.3.4.2 in accordance with the position in Section 3.6.2 of the SRP. This was identified as DSER Confirmatory Item 3.6.2.3-4. In Revision 4 to the SSAR, Westinghouse revised Section 3.6.2.3.4.2 to commit to the above position in Section 3.6.2 of the SRP. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.6.2.3-4 is closed.

3.6.2.4 Conclusions

On the basis of its review of Revision 22 to Section 3.6.2 of the SSAR, the staff concludes that the criteria for postulating pipe rupture and crack locations and the methodology for evaluating the subsequent dynamic effects resulting from these ruptures comply with Section 3.6.2 of the SRP and meet GDC 4 as it relates to pipe rupture locations, and therefore, are acceptable for ensuring that the AP600 design is adequately protected against the effects of postulated high-energy line breaks. The staff's conclusion is based on the following:

The proposed pipe rupture locations will be adequately determined using the above staff-approved criteria and guidelines. The design methods for high-energy mitigation devices and the measures to deal with the subsequent dynamic effects of pipe whip and jet impingement have been sufficiently and adequately defined by Westinghouse to provide adequate assurance that upon completion of the high-energy line break analyses, the ability of safety-related SSCs to perform their safety functions will not be impaired by the postulated pipe ruptures. The staff will require that the as-built inspections of the high-energy mitigation devices be a part of the ITAAC.

The provisions for protection against the dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside the containment and the resulting discharging fluid provides adequate assurance that design-basis LOCAs will not be aggravated by the sequential failures of safety-related piping and that the performance of the ECCS will not be degraded as a result of these dynamic effects. In addition, these provisions assure that the consequences of pipe ruptures will be adequately mitigated so that the reactor can be safely shut down and be maintained in a safe-shutdown condition in the event of a postulated rupture of a high-or moderate-energy piping system inside and outside containment.

3.6.3 Leak-Before-Break

Under the broad-scope revision to GDC 4 (52 FR 41288; October 27, 1987), the NRC allows the use of fracture mechanics technology to exclude from structural design consideration the dynamic effects of pipe ruptures in nuclear power plants, provided it is demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design bases for the piping. The demonstration of low probability of pipe rupture utilizes a deterministic fracture mechanics analysis that evaluates the stability of postulated, small, through-wall flaws in piping and the ability to detect leakage through the flaws long before the flaw could grow to unstable sizes and break the pipe. The concept underlying such analyses is referred to as leak-before-break (LBB).

3.6.3.1 Leak-Before-Break Acceptance Criteria

In revised GDC 4, the NRC states, in part, that "dynamic effects associated with postulated pipe ruptures in nuclear power units 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."

The analyses referred to in the revised GDC 4 should be based on such specific data as piping geometry, materials, and piping loads. The staff must review the LBB analyses for specific piping design before the applicant can exclude the dynamic effects from the design basis. Applicants seeking design certification for ALWRs under 10 CFR Part 52 may be allowed to incorporate preliminary stress analysis results, provided bounding limits (both upper and lower bound) are determined in order to establish assurance that adequate margins are available for leakage, loads, and flaw sizes. These bounding values and preliminary analyses can be verified when as-built and as-procured information becomes available during the COL phase. Verification of the preliminary LBB analysis should be completed at the COL stage based on actual material properties and final, as-built piping analysis as part of ITAAC associated with 10 CFR Part 52 prior to fuel loading. The above staff position on LBB application is stated in SECY-93-087 and was approved by the Commission in its SRM dated July 21, 1993.

A margin of 10 on leakage is required so that leakage from the postulated flaw size is assured of detection when the pipe is subjected to normal operational loads. A margin of $\sqrt{2}$ (1.0 is acceptable if loads are combined by the absolute sum method) on loads is required to ensure that leakage-size flaws are stable at normal plus accident loads (e.g., SSE and safety relief valve (SRV) discharge loads). A factor of 2 between the leakage-size flaw (postulated under normal loads) and the critical-size flaw (calculated under normal plus SSE loads) is required to ensure an adequate stability margin for the leakage-size flaw. The analysis must be performed for an entire pipe run from anchor to anchor.

In addition, applicants seeking approval of LBB during the design certification phase for an ALWR will be required to perform LBB analyses to establish through-wall flaw sizes and flaw stability. For through-wall flaw sizes, a lower-bound, normal-operational stress limit must be established for dead weight, pressure, and thermal loadings. The mean or best-estimate stress strain curve should be used. For flaw stability, an upper-bound stress limit should be established for normal plus SSE loading. A lower-bound stress-strain curve for base metal should be used regardless of whether the weld or base metal is limiting. In addition, a lower-bound toughness (weld metal or base metal) should be used.

A deterministic fracture mechanics evaluation accounting for material toughness is required. The applicant may propose any fracture mechanics evaluation method for NRC staff review. However, the applicant will have to validate the method by comparing it with other acceptable methods or with experimental data.

3.6.3.2 Leak-Before-Break Limitations

The staff has established certain limitations on the use of the LBB approach for excluding piping that is likely to be susceptible to failure from various degradation mechanisms during service. A significant portion of the LBB review involves the evaluation of the susceptibility of the candidate piping in various degradation mechanisms to demonstrate that the candidate piping is not susceptible to failure from these degradation mechanisms. The NRC staff reviews the operating history and measures to prevent or mitigate these mechanisms.

The LBB approach cannot be applied to piping that can fail in service from effects such as water hammer, creep, erosion, corrosion, erosion-corrosion, fatigue, thermal stratification, and environmental conditions. Such piping is excluded because these degradation mechanisms challenge the assumptions in the LBB acceptance criteria. For example: water hammer may introduce excessive dynamic loads that are not accounted for in the LBB analyses, and corrosion and fatigue may introduce flaws whose geometry may not be bounded by the postulated through-wall flaw in the LBB analyses. Adhering to the DID principle, piping susceptible to failure from these potential degradation mechanisms is excluded from LBB applications.

The limitations and acceptance criteria for LBB used by the NRC staff are discussed in Volume 3 of NUREG-1061, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee."

3.6.3.3 Leak-Before-Break Design Basis

The broad-scope rule introduced an acknowledged inconsistency in the design basis by excluding the dynamic effects of postulated pipe ruptures while retaining non-mechanistic pipe rupture for containments, ECCS, and environmental qualification (EQ) of safety-related electrical and mechanical equipment.

The NRC staff subsequently clarified its intended treatment of the containment, ECCS, and EQ in the context of LBB application in a request for public comment on this issue that was published on April 6, 1988 (53 FR 11311). Effects resulting from postulated pipe breaks can be generally divided into local dynamic effects and global effects. Local dynamic effects of a pipe break are uniquely associated with that of a particular pipe break. These specific effects are not caused by any other source or even by a postulated pipe break at a different location. Examples of local dynamic effects are pipe whip, jet impingement, missiles, local pressurization, pipe break reaction forces, and decompression waves in the intact portions of that piping or communicating piping. Global effects of a pipe break need not be associated with a particular pipe break. Similar effects can be caused by failures from such sources as pump seals, leaking valve packings, flanged connections, bellows, manways, rupture disks, and ruptures of other piping. Examples of global effects are gross pressurizations, temperatures, humidity, flooding, loss of fluid inventory, radiation, and chemical condition.

The application of LBB technology eliminates the local dynamic effects of postulated pipe breaks from the design basis. However, global effects may still be caused by something other than the postulated pipe break. Since the global effects from the postulated pipe break provide a reasonably conservative design envelope, the NRC staff will continue to require the consideration of global effects for various aspects of the plant design, such as EQ, ECCS, and the containment.

The elimination of local dynamic effects of postulated high-energy pipe breaks from the design basis of ALWRs using fracture mechanics analyses (LBB approach) is permitted in the revised GDC 4 of 10 CFR Part 50, Appendix A.

3.6.3.4 Westinghouse Leak-Before-Break Evaluation Approach

The application of the LBB approach to ALWRs seeking design certification under 10 CFR Part 52 is acceptable when appropriate bounding limits are established during the design certification phase using preliminary analyses results and verified during the COL phase by performing the appropriate ITAAC discussed herein. Bounding analyses as described above must be performed by the applicant for ALWR piping requesting the application of LBB. Volume 3 of NUREG-1061 constitutes the current NRC approved LBB methodology and acceptance criteria.

In RAI 252.3, the staff requested Westinghouse to perform bounding LBB analyses for LBB candidate piping systems, including evaluations for susceptibility to potential degradation mechanisms for the projected 60-year AP600 design life. In the response to RAI 252.3 dated December 22, 1992, Westinghouse indicated that bounding LBB analyses were not planned. Instead, sample LBB analyses for the reactor coolant loop (RCL) 78.7-cm (31-in.) inner diameter (I.D.) hot leg and 55.9-cm (22-in.) I.D. cold leg piping were provided in Revision 0 to Appendix 3B of the SSAR. Additional sample analyses for the ADS stages 1, 2, and 3, 10.2-cm (4-in.), 20.3-cm (8-in.), and 35.6-cm (14-in.) piping, MS 81.3-cm (32-in.) piping and the FW 40.6-cm (16-in.) piping were to be provided in later revisions to the SSAR. Westinghouse indicated that the staff should be able to assess the acceptability of the AP600 LBB approach based on these sample analyses and the criteria in Section 3.6.3 of the SSAR.

Additionally, in RAI 252.4, the staff requested Westinghouse to describe the procedure to be used by the COL applicant to verify that the actual material properties and final as-built piping analyses are within the limits in the bounding LBB analyses. In the response to RAI 252.4 dated December 22, 1992, Westinghouse reiterated that bounding LBB analyses were not planned to be performed and stated that a report will be prepared by the COL applicant to reconcile the design analysis. This report will include a review of the as-built piping analyses and the certified material test report for piping in systems qualified by LBB. The staff found that the preparation of the reconciliation report by the COL applicant, as proposed by Westinghouse, is unacceptable. The staff did not understand how as-built analyses for piping not previously included in the sample analyses could be reconciled. Moreover, during a meeting with Westinghouse on May 13, 1993, (meeting summary dated May 21, 1993) in which the Westinghouse sample analysis approach was reviewed, the staff indicated that approval for LBB was treated on a case-specific basis with detailed analyses submitted for each piping subsystem for which LBB approval was requested. The staff also noted that performing preliminary stress analyses and establishing bounding parameters subject to ITAAC verification was an alternative approach for satisfying GDC 4 as proposed by the staff in SECY-93-087, but a sample analysis approach was not in compliance with the regulations nor SECY-93-087.

Further, during the piping design review meeting, the staff requested Westinghouse to address the above staff concern that the sample analysis approach was not in compliance with the regulations nor SECY-93-037. Westinghouse responded that LBB analyses of all LBB candidate lines were to be completed in time to support preparation of the final SSAR for Final Design Approval. However, the staff found that the use of (1) the design configuration and loadings for the Appendix 3B analyses, and (2) actual material test data from a then in-progress but unidentified test program for the MS and FW systems was, as before, neither in conformance with the regulations nor with SECY-93-087. Both require as-built information that would be unavailable for design certification. Accordingly, the staff found that the proposed Westinghouse sample analysis and COL applicant reconciliation report was unacceptable. The staff reiterated its request that Westinghouse perform and submit for staff review bounding LBB analyses for LBB candidate piping systems including evaluations for susceptibility to degradation mechanisms for the projected 60-year AP600 design life. This was identified as DSER Open Item 3.6.3.4-1.

Subsequently, Westinghouse presented a draft revision to Section 3.6.3 of the SSAR in March 1995, to commit to bounding analyses for LBB, and the commitment was later incorporated in Revision 4 to the SSAR. Section 3.6.3 was revised by Revision 4 to state in part that a LBB bounding analysis is to be performed for each piping system. Section 3.6.3.2 was also revised to specify that bounding analysis curves (BACs) are to be developed for each applicable piping system. In response, Westinghouse provided the following information for staff review relating to the bounding analysis:

(1) In a meeting on March 15, 1995, Westinghouse presented a handout which contained methods and procedures to be used for bounding analyses.

(2) In a letter dated June 26, 1995, Westinghouse submitted information related to the BAC for a 10.2-cm (4-in.) diameter stainless steel line attached to the first stage ADS valves.

(3) In the July 25 through 26, 1995, design review meeting, the staff reviewed AP600 piping design including a brief review on status of bounding analyses and some calculations on loads to be used in the analyses.

(4) In a letter dated September 12, 1995, Westinghouse submitted additional information on the bounding analyses program, including selected bounding curves and input parameters and a brief sample calculation.

(5) In a letter dated September 26, 1995, Westinghouse proposed that based on the low probability of a break in the FW line in the turbine building, the depressurization load from such a break be excluded from the load combination for the LBB evaluation of the FW line.

The staff reviewed all of the above stated information provided by Westinghouse and finds that the Westinghouse commitment in Revision 4 to the SSAR to perform bounding LBB analyses for LBB candidate piping system was in partial compliance with the request in DSER Open Item 3.6.3.4-1. However, the staff also identified the following issues relating to the bounding LBB analyses:

(1) The descriptions of the methodology and acceptance criteria of the bounding LBB analyses in Revision 4 to the SSAR should be revised to be consistent with the descriptions provided in the March 15, 1995, handouts and as described in the July 25 through 26, 1995, design review meeting. Westinghouse stated during the meeting that the Revision 0 to Appendix 3B of the SSAR, "Leak-before-break Evaluation of the Reactor Coolant Loop Piping of the AP600," would be extensively revised to address all LBB evaluations. Especially, how a margin of 2 between the leakage crack size and the critical crack size would be verified and calculated under what loads respectively were not clearly stated. The staff expected that Section 3.6.3.2 and Appendix 3B of the

SSAR would be revised to provide consistent descriptions of LBB methods and acceptance criteria in accordance with NUREG-1061, Volume 3 and Section 3.6.3 of the SRP.

(2) The Westinghouse proposal for the load combination for the LBB evaluation for the FW line was unacceptable. Design load combinations for ASME components, including the design-basis pipe break load (DBPB) or MS/FW pipe break load plus SSE load combinations are specified in Section 3.9.3 of the SRP. This SRP section does not permit load combinations to be developed based on probabilistic arguments.

(3) As indicated in the July 25 through 26, 1995 design review meeting, the staff was unable to verify the leak rate in the Westinghouse LBB evaluation of the 10.2-cm (4-in.) ADS line (See DSER Open Item 3.6.3.6-4). The staff utilized the PICEP computer code in its verification analysis. Accordingly, there was a need for further meetings to review the details of Westinghouse bounding LBB analyses to resolve this issue.

On December 20, 1996, Westinghouse presented Revision 10 to Section 3.6.3 and Appendix 3B of the SSAR. Upon review the staff identified the following items:

- LBB Bounding analyses were provided for each applicable piping system listed in Table 3B-1 of the SSAR, with different combinations of piping material type, pipe size, pressure and temperature. Curves satisfying LBB criteria were developed and will be used by the COL applicant to verify that the as-built piping satisfies the LBB requirements. The staff found this approach acceptable, subject to final staff review and evaluation of the bounding curves during a future staff audit.

- Potential failure mechanisms that could affect the integrity of the LBB applicable piping, including erosion-corrosion induced wall thinning, stress corrosion cracking, water hammer, fatigue, thermal aging, thermal stratification, etc. were discussed in Appendix 3B.1 and 3B.2. The discussions were focused on material susceptibility to water chemistry, flow velocity, operating temperature, steam quality etc., and affects of plant operating procedures, operating temperature limits, water chemistry control, experience of past operating events, precaution measures, and design improvements to minimize undesirable occurrences. The staff determined that this is acceptable except regarding water hammer occurrences in the FW lines. The staff found that Westinghouse efforts are effective in minimizing occurrence of water hammer in the FW line but can not totally eliminate its potential occurrence. In addition, the staff had concerns on lack of operating experience of AP600 design and on difficulties to define and incorporate water hammer load in the design of FW lines. In Revision 11 to Section 3.6.3 and Appendix 3B of the SSAR, Westinghouse removed the LBB application to FW line. The staff finds this acceptable, and therefore, this part of the DSER Open Item 3.6.3.4-1 is closed.

- The staff had concerns on the high level of uncertainty in applying LBB methodology to small lines, specifically the lines with a 10.2-cm (4-in.) diameter. These concerns were based on Section 3.5 of the recently published NUREG/CR-6443, regarding uncertainties due to pressure-induced bending effects to calculated leakage flaw size and maximum stresses. To address this concern, Westinghouse was requested to perform sensitivity studies. In Revision 11 to Section 3.6.3 and Appendix 3B to the SSAR, Westinghouse removed the LBB application to 10.2-cm (4-in.) diameter pipe. The staff finds this acceptable.

- Changes incorporated in Sections 3.6.3.3 and 3B.3 of the SSAR, up to Revision 10, presented a detailed description of LBB analysis methodology and acceptance criteria, and procedures for performing LBB bounding analyses and establishing LBB bounding curves. The staff found these descriptions to be comprehensive. However, additional description was needed in Sections 3B.3.1.3 and 3B.3.2.3 on the bounding curve construction procedures, or in Section 3.6.3.3 on the bounding analysis to explain how bounding curves meet LBB acceptance criteria.

On April 16 through 18, 1997, the staff conducted a design review meeting at Westinghouse offices to resolve the remaining LBB open items including those discussed above. Westinghouse provided a detailed explanation of their procedure for developing the LBB BACs given in Appendix 3B of the SSAR. The curves are based on a minimum of two points, one corresponding to a low normal stress and the other to a high normal stress. The loads associated with these high and low normal stresses are used to determine the flaw sizes corresponding to leakage rates equal to 10 times the leak detection capability. The maximum stress associated with each normal stress condition is determined by performing a stability analysis based on a critical flaw size equal to twice the leakage flaw size. The BAC is generated by plotting these two points on a maximum versus normal stress plot and joining the two points by a straight line. A set of bounding curves are generated for each piping system to be qualified for LBB for different pipe sizes and operating conditions. Westinghouse indicated that in these calculations, minimum wall thicknesses and material properties are assumed. To demonstrate the applicability of LBB, the results of the pipe stress analyses are compared to the BACs. The critical location is the location of highest maximum stress based on the absolute combination of pressure, deadweight, thermal, and SSE stresses. The corresponding normal stresses are calculated using the algebraic summation of pressure, deadweight and thermal stresses. In order to verify the computational results for the bounding analysis procedure, Westinghouse was asked to provide information that would allow the staff to independently perform a confirmatory calculation. The piping selected for this calculation was the 81.3-cm (32-in.) MS, ferritic steel pipe. The confirmatory analysis used the methodology described in NUREG/CR-6281. The results of the analysis indicated that the calculated maximum allowable axial plus bending loads are greater than the loads determined by the Westinghouse bounding analysis. Thus, for the selected problem, the confirmatory analysis demonstrated that the Westinghouse bounding analysis methods are conservative and acceptable. Based on the staff's review of the additional information provided by Westinghouse during this meeting and the staff's independent confirmatory calculation, the staff concluded that the BACs are consistent with the guidelines and recommendations given in draft Section 3.6.3 of the SRP and NUREG-1061, Volume 3 and are acceptable.

At the April 1997 meeting, the staff also reviewed a sample of five piping calculation packages to verify proper implementation of the LBB bounding analysis procedures and LBB acceptance criteria. The selected packages covered a wide range of pipe sizes and materials. They included the MS line, the Pressurizer Surge Line, the PRHR Return Line, the CMT-2B Supply Line, and the Direct Vessel Injection Line. The staff verified that the pipe stresses were calculated from the appropriate load cases in accordance with the required load combinations for maximum and normal stresses. Critical stresses were identified and spot checked and found to be reasonable. All five lines met the LBB acceptance criteria. The staff also checked the calculations to verify that the SSE loads considered all of the applicable soil conditions. All of the calculations reviewed used the enveloped response spectrum analysis method. Reference documents were checked to verify that the response spectra represented an envelope of the four soil design cases and included 15 percent broadening. The staff reviewed additional calculations to verify that pipe breaks are postulated at branch pipe connections for which LBB is not applicable. Based on these reviews, the staff concluded that Westinghouse had properly implemented the LBB bounding analysis methods and acceptance criteria.

During the April 1997 meeting, the staff further discussed their concerns regarding the uncertainties in applying LBB methodology to small lines based on the conclusions regarding the potential effects of constraint to pressure-induced bending in piping systems on LBB calculations recently published in NUREG/CR-6443. Although Westinghouse had agreed to delete 10.2-cm (4-in.) diameter lines, the staff was concerned that this may be a problem for lines as small as 15.2 cm (6 in.) in diameter. To assess possible restraint effects, Westinghouse agreed to perform a sensitivity analysis, which included a three-dimensional finite element analysis of a 15.2-cm (6-in.) diameter, austenitic steel pipe for both restrained and unrestrained bending conditions. The restraint effect was simulated by placing a circumferential flaw in the pipe near the fixed end, while the unrestrained condition was simulated by placing the flaw at the pipe mid-span away from any end restraints. The finite element analysis was used to determine the crack opening areas for cracks with restrained and unrestrained bending. The crack opening areas were computed for a flaw length corresponding to an acceptable leakage size flaw in the restrained condition. The results of the study were submitted in letters dated May 1, 1997, and June 10, 1997. The results of the finite element analysis indicated a restraint effect on leak rate when the crack opening area in the restrained region was computed to be smaller than the value computed for the unrestrained region. To assess the restraint effect on load carrying capacity of the pipe, Westinghouse computed the applied J-Integral (J) and tearing modulus (T) for the leakage size flaw in the restrained region at postulated normal plus faulted bounding loads. The results from the applied J/T analysis were compared to representative J/T curves for both the base and weld materials. The representative J/T curves were obtained from EPRI report NP-4768, "Toughness of Austenitic Stainless Steel Pipe Welds", dated October 1986. The results from the J/T analysis indicated that acceptable margins are maintained for the limiting material with a leakage size flaw and loads representative of the bounding loads used in the Westinghouse evaluation of 15.2-cm (6-in.) stainless pipe. The results from the sensitivity analysis indicate that restraint has little effect on the bounding load analysis performed by Westinghouse for AP600 piping 15.2 cm (6 in.) and greater in diameter. In Revision 14 to Section 3B.7 of the SSAR, Westinghouse summarizes this sensitivity study and its results. Based on the review of the results from the restraint sensitivity analysis, the staff concludes that the restraint conditions do not significantly change the results of the bounding analysis and that the bounding analysis satisfies the guidelines of draft Section 3.6.3 of the SRP.

As discussed in Section 3.6.3.6 of this report on the resolution of DSER Open Item 3.6.3.6-4, during the April 1997 meeting, Westinghouse also provided additional detailed information on the procedure that had been used for verification and benchmarking of their leak-rate computer code, and explained some of the differences between their code and the PICEP code and why some correction factors need to be applied to the results for proper comparison. Based on the additional information, the staff concluded that the Westinghouse software had been adequately benchmarked and was acceptable.

On the bases of the staff review of the additional information provided by Westinghouse at the April 16 through 18 design review meeting and the additional information included in Revision 12 to the SSAR, DSER Open Item 3.6.3.4-1 is closed.

In addition, as stated in the DSER, the COL applicant will be required to verify that the actual material properties and final as-built piping analyses meet the acceptance parameters established in the bounding LBB analyses. This was identified as DSER Open Item 3.6.3.4-2 and COL Action Item 3.6.3.4-1. In March 1995, Westinghouse presented a draft SSAR revision to address the staff concern by adding a new Section 3.6.4.2. The draft was partly acceptable. In Revision 10 to the SSAR, Westinghouse revised Section 3.6.4.2 to require that the COL applicants address the following items: (1) verify that the as-built stresses, diameter, wall thickness, material, welding process, pressure, and temperature in the piping applying LBB are bounded by the LBB bounding analysis, (2) review Certified Material Test Reports or Certifications to verify that the ASME Code, Section III strength and Charpy toughness requirements are satisfied, and (3) complete the LBB evaluation by comparing the results of the final piping stress analysis with the BACs documented in Appendix 3B of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 3.6.3.4-2 is closed.

3.6.3.5 Leak-Before-Break Candidate Piping Systems

In Revision 0 to paragraph 7 of Section 3.6.3 of the SSAR, Westinghouse stated that high-energy ASME Code Section III, Class 1, 2, and 3 piping of 10.2-cm (4-in.) nominal diameter or larger was to be evaluated for compliance with LBB criteria. For piping that penetrated the containment vessel, the evaluation was to be continued to the first anchor outside containment and include any branch connections between the penetration and anchor. For systems or portions of systems for which it was not practical or economical to satisfy the LBB criteria, the requirements and criteria in Sections 3.6.1 and 3.6.2 of the SSAR were to apply.

However, in Revision 0 to paragraph 7 of Section 3.6 of the SSAR, Westinghouse indicated that the scope of AP600 piping to be evaluated by LBB was not as extensive as indicated by paragraph 7 of Section 3.6.3, but was to be limited to the RCL, RCL branch lines, MS and FW lines out to anchors adjacent to the isolation valves, and other primary and secondary system piping outside containment equal to or greater than 10.2-cm (4-in.) nominal pipe size (NPS).

Relative to the extent of piping to be evaluated by LBB, first by RAI 252.2, the staff observed that Section 3.6.3 of the SSAR indicated that high-energy ASME Code, Class 1, 2, and 3 piping equal to or greater than 10.2-cm (4-in.) NPS was included in the scope of LBB applications and requested Westinghouse to identify specific piping under consideration for LBB applications. RAI 252.2 also referenced related RAI 210.6, in which Westinghouse was requested to identify high- and moderate-energy piping which failed to satisfy the acceptance criteria in Paragraph II.D of Enclosure 1 of the Draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," dated February 27, 1992.

In the response to RAI 252.2 dated January 14, 1993, Westinghouse indicated that the LBB methodology was to be applied to the candidate high-energy lines in the nuclear island identified in future Appendix 3E of the SSAR to be provided, in part, in response to RAI 210.6. In Revision 1 to the SSAR, Westinghouse provided Appendix 3E which indicated that the LBB methodology was to be applied to the following candidate high-energy piping (NPS) in the nuclear island:

- RCS piping, hot and cold legs (95.25 cm (37.5 in.) and 68.9 cm (27.12 in.), respectively)
- Pressurizer Surge Line (45.7 cm (18 in.))
- Pressurizer Spray Line (15.2 cm and 10.2 cm (6 in. and 4 in.)) and Safety Injection Line (15.2 cm (6 in.))
- Automatic Depressurization System (ADS) Lines (35.6 cm, 30.5 cm, 20.3 cm, and 10.2 cm (14 in., 12 in., 8 in., and 4 in.))
- Normal Residual Heat Removal System (NRHR) Lines (50.8 cm, 30.5 cm and 25.4 cm (20 in., 12 in., and 10 in.))
- Passive Residual Heat Removal System (PRHR) Line (25.4 cm (10 in.))
- Core Makeup Tank Line (20.3 cm (8 in.))
- Direct Vessel Injection Line (20.3 cm (8 in.))
- Main Steam (MS) Line (81.3 cm (32 in.))
- Feedwater (FW) Line (40.6 cm (16 in.))

As identified in Appendix 3E, Westinghouse proposed to expand the AP600 candidate piping systems to be more extensive than systems previously qualified for LBB in existing Westinghouse PWRs. In addition to systems unique to the passive type plant design (e.g., the ADS and PRHR systems), the AP600 LBB candidate piping systems include systems that were not approved for LBB in existing Westinghouse PWR designs such as the MS and FW systems and piping as low as 10.2 cm (4 in.) in diameter. In Revisions 10 and 11 to the SSAR, Westinghouse revised the scope of LBB applicable piping systems, as indicated in Table 3B-1, incorporating changes stated in Westinghouse letter dated February 7, 1997, to delete the FW line and piping with 10.2-cm (4-in.) diameter size from LBB application.

In RAI 252.7, the staff observed that (1) Section 16.1 of the SSAR indicated the technical specifications (TSs) limited the unidentified RCS leakage to less than 1.89 L/min (0.5 gpm), and (2) this leakage limit is used in the LBB analyses of piping inside containment, and requested that Westinghouse describe the administrative controls to ensure that any increase in the unidentified leakage limit in the TSs will initiate a reevaluation of the LBB analyses. In the response to RAI 252.7 dated January 22, 1993, Westinghouse indicated that the TSs included the technical bases for the LBB analyses as described in Section 16.B.3.4.7.b of the SSAR and that any change to the unidentified leakage limit should satisfy this technical basis or require an amendment to the COL applicant.

Additionally, in RAI 252.8, the staff requested Westinghouse to demonstrate the reliability, effectiveness, sensitivity, and timeliness of leakage detection methods and procedures selected for inside containment to detect a 1.89 L/min (0.5 gpm) unidentified leakage. In the response dated January 8, 1993, Westinghouse revised the "Containment Sump Level Monitor" portion of Section 5.2.5.3.1 of the SSAR. Staff evaluation of this response was not complete at that time. However, the staff requested that similar demonstrations for leakage detection methods inside containment should also be provided. This was identified as DSER Open Item 3.6.3.5-1. Subsequently, Westinghouse proposed its resolution to this issue by revising SSAR Section 5.2.5, "Detection of Leakage through RCPB." This information was provided also in response to part of DSER Open Item 5.2.5.3-2. Therefore, DSER Open Item 3.6.3.5-1 was subsumed by DSER Open Item 5.2.5.3-2. DSER Open Item 3.6.3.5-1 is closed. The staff's evaluation of DSER Open Item 5.2.5.3-2 is discussed in Section 5.2.5 of this report.

In RAI 252.5, the staff observed that Revision 0 to Section 3.6.3 of the SSAR indicated that Class 2 and 3 piping of Section III of the ASME Code were included within the LBB scope and requested that Westinghouse discuss the significance of differences between the ASME Code Section III Class 1, 2, and 3 requirements on ensuring piping structural integrity and to describe procedures to address them. In the response to RAI 252.5 dated December 22, 1992, Westinghouse indicated that the LBB methodology was not to be applied to ASME Code Class 3 piping but was to be applied to portions of the Class 2 MS and FW systems inside containment. Westinghouse also committed to (1) verify the fatigue resistance of these Class 2 systems by performing fatigue crack growth calculations for postulated through-wall flaws, and (2) include the critical terminal end weld at the steam generator nozzle for one MS line and one FW line as part of the inservice inspection (ISI) requirements of Section XI of the ASME Code. ISI requirements for Class 2 piping is based on a sampling basis and Westinghouse did not commit to increase the size of the sample and frequency of inspection beyond that required by the ASME Code. In Revision 1 to the SSAR, Westinghouse revised the second bullet in the first paragraph of Section 3.6.3.2 to be in accordance with these commitments. In addition, although no ASME Code Section III Class 3 piping were selected for the LBB candidate piping systems, the differences in analysis, fabrication, and inspection between Class 1 and 2 systems were not adequately addressed by the response to RAI 252.5. Westinghouse was requested to discuss the significance of these differences. This was identified as DSER Open Item 3.6.3.5-2.

In response, Westinghouse submitted the March 1995, draft revision to the SSAR, in which the second item in the first paragraph of Section 3.6.3.2 of the SSAR was revised again to state that for ASME Code Class 2 and 3 piping systems for which LBB is demonstrated, fatigue crack growth analyses were to be performed which, along with the ASME Code Section XI preservice inspection (PSI) and ISI, would provide for the integrity of these systems. The provision for inclusion of welds at connection to the steam generator nozzle for the MS and FW lines in the ISI program, which was added in the Revision to the SSAR was deleted.

During the April 16 through 18, 1997, design review meeting at Westinghouse offices, the staff and Westinghouse discussed the subject of weld inspection requirements for Class 2 and 3 LBB lines. The staff was concerned that one of the LBB lines, the Accumulator Discharge Line which is part of the ECCS system, is classified as an ASME Code Class 3 line. One of the significant differences in fabrication inspection requirements is that Class 2 pipe requires radiographic examination of the welds and Class 3 pipe does not. In Revision 10 to Section 3B.6 of the SSAR, Westinghouse indicated that for Class 3 lines required for emergency core cooling functions, radiography will be conducted on a random sample of the welds. The Class 3 LBB lines are included in the sample to be radiographed. In Section 3B.5 of the SSAR, Westinghouse describes the differences in ISI requirements. Class 2 piping requirements include volumetric inspections while Class 3 piping requires periodic visual inspections in conjunction with pressure testing. As a result of the meeting discussions, Westinghouse agreed to augment the weld inspection requirements for the Class 3 lines. In Revision 12 to the SSAR, Westinghouse included additional requirements in Sections 3.6.3.2 and 3B.6. In these sections, Westinghouse states that the weld and welder qualification, and the weld inspection requirements for Class 3 LBB lines are equivalent to the requirements for Class 2. The ISI requirement for each Class 3 LBB line includes a volumetric inspection equivalent to the requirements for Class 2 for the weld at or closest to the high-stress location. The staff reviewed these additional requirements and concluded that they provide additional and acceptable assurance of LBB integrity for the Class 3 piping, consistent with the requirements for LBB applications for Class 1 and 2 piping. On this basis, DSER Open Item 3.6.3.5-2 is closed.

In RAI 252.6, the staff observed that, in Revision 0 to Section 3.6.3 of the SSAR, Westinghouse indicated that application of LBB was being considered for portions of piping outside containment, and requested that Westinghouse provide information to demonstrate the reliability, effectiveness, sensitivity, and timeliness of leakage detection methods and procedures selected for outside containment. In the response to RAI 252.6, Westinghouse stated in a letter dated December 22, 1992, and in Revision 1 to the SSAR that the LBB methodology was not to be applied outside containment and committed to revise Section 3.6.3 of the SSAR, including Section 3.6.3.1 to reflect this position. Westinghouse committed to revise paragraph 7 of Section 3.6.3 of the SSAR to (1) reference the future Appendix 3E, which would identify piping to be evaluated by the LBB methodology, and (2) indicate that for the MS and FW systems between the steam generators (SG) and auxiliary building anchors, portions of the systems between the SG and the containment penetration flue head inboard welds are evaluated by the LBB methodology and portions between the flued head outboard welds to the auxiliary building anchors satisfy the break exclusion zone requirements. Westinghouse also committed to revise paragraph 9 of Section 3.6.3.1 of the SSAR to delete the requirement that the MS and FW piping inside containment to the first anchor outside containment be designed to the LBB criteria and to revise paragraph 15 to delete descriptions of leak detection procedures outside containment.

The staff found that this Westinghouse response to RAI 252.6 was not totally acceptable. In particular, the staff found that for the MS and FW piping between the steam generator nozzles and the auxiliary building anchors:

- Westinghouse intended to design the portions of the piping inside containment to the LBB criteria and the portions outside containment, including the containment penetration, to the pipe rupture criteria of Section 3.6.2 of the SRP. For the main steamline, the LBB portion is from the steam generator outlet nozzle to the containment penetration flued head inboard weld. However, the LBB methodology is applicable only to entire piping systems or analyzable portions thereof where analyzable portions are typically segments located between anchor points. In Section 3.8 of the SSAR, Westinghouse indicated that the MS and FW containment/shield building penetration designs incorporate guard pipes rather than anchors. However, during the April 12 through 14, 1994, piping design review meeting (Meeting Summary dated May 17, 1994), Westinghouse indicated that the MS and FW piping system anchors were to be relocated to the shield building. Accordingly, Westinghouse was requested to clarify the location of the MS and FW piping anchors and the extent of the MS and FW LBB candidate piping. This was identified as DSER Open Item 3.6.3.5-3. Subsequently, Westinghouse indicated that Section 3.6.3, paragraph 7 and Appendix 3E of the SSAR identifies the scope of the analysis. Furthermore, Westinghouse explained during the design review meeting on July 27, 1995, that because relocation of the MS and FW anchors to the shield building would increase thermal stresses in the MS and FW lines, the anchors are to remain at the exterior wall of the auxiliary building. This clarification is acceptable. However, as discussed previously, the application of LBB technology to the FW line was unacceptable. In Revision 11 to the SSAR, Westinghouse removed FW lines and 10.2-cm (4-in.) diameter lines from the LBB scope piping list. Based on the above discussion, DSER Open Item 3.6.3.5-3 is closed.

- The description of the extent of MS and FW LBB candidate piping in paragraph 7 of Section 3.6.3 of the SSAR was inconsistent with the description in the revised paragraph 7 of Section 3.6.3 and as shown in Appendix 3E of the SSAR. Westinghouse was requested to provide consistent definitions of the MS and FW LBB candidate piping throughout the SSAR. This was identified as DSER Open Item 3.6.3.5-4. During the design review meeting on July 27, 1995, Westinghouse indicated that the inconsistency will be corrected in a SSAR revision. Subsequent, appropriate corrections were made in Revision 4 to the SSAR. This is acceptable. However, as discussed previously, the application of LBB methodology to the FW lines was unacceptable. In Revision 11 to the SSAR, Westinghouse removed FW lines and 10.2-cm (4-in.) diameter lines from the LBB scope piping list. Based on the above discussion, DSER Open Item 3.6.3.5-4 is closed.

In RAI 252.13, the staff observed that application of LBB to MS and FW piping was not approved for power reactors, and requested Westinghouse to provide additional discussions relating to potential susceptibility of these piping systems to degradation mechanisms such as water/steam hammer and erosion/corrosion. In the response to RAI 252.13 dated January 14, 1993, Westinghouse provided a general discussion only. Westinghouse was requested to provide in the SSAR, more detailed discussions with sufficient information to support the conclusion that the MS and FW piping systems do not fall within the limitations delineated in Section 5.1 of Volume 3 of NUREG-1061. This was identified as DSER Open Item 3.6.3.5-5.

During the April 1997, design review meeting, there were discussions on the subject of potential susceptibility for erosion-corrosion (EC) in the MS piping. The staff reviewed the information provided in Revision 11 to Sections 3.6.3.1 and 3B.2.1 of the SSAR and concluded that based on service experience with MS lines in operating PWRs, additional information was needed to justify the low potential for EC degradation required for LBB applications. Westinghouse indicated that in addition to the reasons given in the SSAR, the AP600 MS line has a low flow rate at full power which is less than the currently used industry EC screening criteria of 45.7 m/sec (150 fps). This additional information was subsequently incorporated in Revision 12 to Section 3.6.3.1 of the SSAR. In addition, in Section 3.6.3.2, Westinghouse indicates that the application of mechanistic pipe break in AP600 piping requires that potential degradation by erosion, erosion-corrosion, and erosion cavitation be examined to provide low probability of pipe failure. Based on the additional information, the staff concluded that there is a low potential for EC in the MS line consistent with the guidelines given in draft Section 3.6.3 of the SRP and NUREG-1061 for application of LBB.

During the piping design review meetings, Westinghouse described various design and operating features to address water hammer concerns on the FW system. The staff observed that these features would serve to minimize, but not necessarily eliminate, water hammer occurrences in the AP600 FW system. In addition, the staff also observed that there was no operating experience for the AP600 FW design, and consequently in the DSER, the staff concluded that the application of LBB methodology to the AP600 FW system is unacceptable.

After the DSER was issued, the staff and Westinghouse continued to hold extensive discussions on the FW line issue. In a letter dated May 2, 1995, the staff requested the following additional information which was needed in the SSAR to complete the staff's evaluation of the application of LBB to the FW lines:

- Discuss the steps taken to ensure that water hammer is not a concern in the FW line.
- Explain why thermal stratification is not a concern and what assurance that thermal stratification will not occur in the FW line.
- Demonstrate that the FW nozzle at the steam generator is the controlling location for stress and fatigue effects for the FW line inside containment.
- Commit to perform augmented ISI (100 percent volumetric inspection every inspection interval) at the FW nozzles connecting to the steam generator.
- In addition to performing ASME code Class 1 stress and fatigue evaluation at the nozzle connecting to steam generator, perform a Class 1 equivalent fatigue evaluation for the Class 2 portion of the FW line.
- Discuss any significant differences in the ASME code Class 1 ISI and fabrication requirements from the requirements applicable to the Class 2 portion of the FW line inside containment that affect LBB assumptions.
- Verify that the dynamic load used for design bounds the effects of FW pipe break outside containment (including isolation check valve slamming) and the effects of postulated water hammer event.
- Provide a discussion of the reduced thermal load effects in the FW line resulting from rerouting the auxiliary FW to a separate nozzle on the steam generator.
- Discuss how EC effects have been minimized or eliminated in the FW line inside containment.

- Discuss how fatigue effects due to dynamic operational vibration cycles have been minimized in the FW line.

- Commit to provide instrumentation for monitoring any unanticipated dynamic loads in the FW line inside containment.

In a letter dated September 15, 1995, Westinghouse committed to revise the SSAR Appendix 3B to address the above staff concerns regarding justification for the applicability of LBB to FW lines. Subsequently, the information was presented in Revision 10 to Section 3.6.3 and Appendix 3B of the SSAR dated December 20, 1996. After reviewing the Westinghouse responses and SSAR revisions addressing the above concerns, and participating in extensive discussions with Westinghouse, the staff concluded that their concerns had not been completely eliminated. In a letter to Westinghouse dated January 24, 1997, the staff documented their findings and conclusions regarding application of LBB methodology to the FW line. The staff noted that Westinghouse had made significant improvements in the proposed FW system design to address the concerns in an attempt to qualify the system for LBB consideration. However, two complementary unresolved issues remained, (1) the lack of operating experience with the redesigned FW system and (2) the uncertainty that exists in accurately assessing the frequency and magnitude of water hammer events. The staff had considered all of Westinghouse's proposals and analyses to evaluate the application of LBB methodology on the basis of its technical merits. The approval of the application of LBB to a system is based on demonstrating that the probability of a rupture in a fluid system pipe is extremely low. This, in turn, requires a thorough understanding of the dynamic loadings that may occur in a system, their frequency of occurrence, and the ability to conservatively predict the magnitude of the loadings. It is the staff's position that the lack of operating experience with the FW system as redesigned for AP600 significantly deters its ability to predict the frequency or the magnitude of water hammer events in the AP600. Therefore, it is not possible to predict the likelihood of occurrence of a water hammer event that could invalidate the assumptions of LBB or cause the unanticipated failure of the FW line. As a result, the staff could not conclude that an "extremely low" probability of pipe rupture exists for the AP600 FW system, as is required by GDC 4.

In Revision 11 to the SSAR, Westinghouse removed the LBB application to the main FW line. The staff reviewed this revision and found the changes acceptable and DSER Open Item 3.6.3.5-5 is closed.

In RAI 252.14, the staff observed that the pressurizer surge line is potentially susceptible to thermal stratification and requested Westinghouse to evaluate the impact on Section III of the ASME code fatigue cumulative usage factor (CUF) for the pressurizer surge line for the projected 60-year design life and the consideration of the thermal stratification loads in the LBB analysis. In the response to RAI 252.14 dated January 14, 1993, Westinghouse indicated that the thermal stratification effects will be included in the leak rate and flow stability calculations and combined with the SSE loads. This response is acceptable. However, the consideration of thermal stratification effects needed to be reviewed. More generally, Westinghouse needed to describe what provisions were made to accommodate the effects of unanticipated thermal stratification in presently identified LBB candidate piping systems which may be uncovered during current integral thermal hydraulic testing of the AP600 design. This issue is addressed in Section 3.12.5.10 and DSER Open Item 3.12.5.10-1 of this report.

As discussed in Section 3.6.3.4 of this report, at the April 1997, design review meeting, the staff reviewed a sample of five piping calculation packages to verify proper implementation of the LBB analysis procedures and acceptance criteria. The pressurizer surge line analysis was included in this sample. The staff reviewed the calculation and verified that the thermal stratification loads had been appropriately considered and included in the load combinations for the pressurizer surge line. The broader concern of identifying piping systems which may be subject to thermal stratification loads was investigated and resolved as part of the staff's piping design review and is discussed in Sections 3.12.5.9 and 3.12.5.10 of this report. Under that review, the staff reviewed a calculation which documented the Westinghouse review to identify piping systems susceptible to thermal stratification and thermal cycling. Their review identified three piping systems which may be susceptible. Thermal stratification loads for these lines were subsequently developed and included in the piping analyses. During the April 1997, design review meeting, the staff reviewed the LBB analysis for one of these lines (PRHR Return line) and verified that Westinghouse appropriately accounted for the stratification loads. Therefore, this concern is resolved.

3.6.3.6 Review of Westinghouse Leak-Before-Break Analysis

In Appendix 3B of the SSAR, Westinghouse originally provided an LBB analysis for the RCL. This analysis was to serve as an example for similar analyses for other high-energy piping systems. Various aspects of the LBB analyses included in the Appendix 3B analysis included LBB criteria, potential failure mechanisms, material properties, pipe geometry and loads, selection of critical locations for postulating flaws, and leak rate and stability evaluations of specific postulated flaws. In addition, during the piping design review meeting, Westinghouse made available completed or in-progress LBB analyses for other piping systems, including the pressurizer surge line and the MS and FW systems. As indicated in the preceding sections, Westinghouse planned to revise Appendix 3B to include all AP600 LBB analyses and not just the RCL. Nonetheless, the results of the staff reviews of the original Appendix 3B of the SSAR and these other LBB analyses are provided in the following to identify issues that were considered in the planned revision of Appendix 3B.

In RAI 252.10, the staff observed that Revision 0 to Appendix 3B of the SSAR indicated that two different soil conditions were considered in deriving stresses for the LBB evaluation of the RCL piping and requested Westinghouse to clarify that the piping stresses in the evaluations represented the worst condition of all potential sites within the scope of the AP600 application. In the response to RAI 252.10 dated March 4, 1994, Westinghouse indicated that the two soil conditions for the RCL piping analyses were chosen to provide preliminary stresses for the sample LBB evaluation and analyses for other soil conditions will be performed at a later date. Accordingly, the staff requested that for all LBB candidate piping systems, and not only the RCL system, the worst condition of all potential sites within the scope of the AP600 applications should be used in the bounding LBB analysis. This was identified as DSER Open Item 3.6.3.6-1.

During the design review meeting on July 27, 1995, Westinghouse indicated that Appendix 3B would be extensively revised and that the worst case soil conditions would be used in the bounding analyses. The NRC staff would review calculations of LBB bounding analysis for verifying that the seismic loads used in the analysis have adequately considered all applicable soil conditions, and that the critical flaw stability in all LBB applications are based on N+SSE loading. This was also related to the staff effort stated in DSER Open Item 3.6.3.4-1 for verifying the acceptability of bounding curves, specifically under limiting loading conditions. As discussed in Section 3.6.3.4 above, during the design review meeting at Westinghouse offices on April 16 through 18, 1997, the staff reviewed the LBB bounding analysis calculations and verified that the applicable soil conditions were used in the analysis. On this basis, DSER Open Item 3.6.3.6-1 is closed.

In RAI 252.11, the staff requested Westinghouse to clarify whether the stresses in Tables 3B-3 and 3B-4 of the SSAR used in the LBB evaluation of the RCL, were from the analysis of routed or unrouted RCL piping. In the response to RAI 252.11 dated December 22, 1992, Westinghouse indicated that the sample analysis for the RCL piping was based on routed RCL piping supported by primary equipment supports, but interconnected piping (e.g., the pressurizer surge line) was not included in the model. The staff intended to review these stresses in future piping design review meetings. This was identified as DSER Open Item 3.6.3.6-2. According to Section V in draft Section 3.6.3 of the SRP, pipe breaks at branch connections to the LBB applicable main piping run should still be postulated, unless the branch lines are also qualified for LBB. Westinghouse agreed with the SRP position in a meeting on February 14 through 15, 1995. However, the LBB analyses were not completed at that time. As discussed in Section 3.6.3.4 above, during the design review meeting at Westinghouse offices on April 16 through 18, 1997, the staff reviewed a sample of piping calculation packages to verify proper implementation of the LBB bounding analysis procedures and LBB acceptance criteria. During this design review meeting, the staff reviewed additional calculations which verified that pipe breaks are postulated at branch pipe connections for which LBB is not applicable. On this basis, DSER Open Item 3.6.3.6-2 is closed.

During the April 12 through 14, 1994, piping design review meeting (meeting summary dated May 17, 1994), Westinghouse indicated that the AP600 LBB evaluations were based on margins of 10 on leak rate, 1.5 on flaw size, and $\sqrt{2}$ on loads except that a margin of 1.0 on load is used if the loads are summed absolutely. In addition, as indicated earlier in this report, an unidentified leakage rate of 1.9 L/min (0.5 gpm) was used. The staff indicated that 1.9 L/min (0.5 gpm) unidentified leakage rate and the 1.5 margin on flaw size were unacceptable. In NUREG-1061, Volume 3 the NRC specifies a margin of 2 on leakage flaw size. Accordingly, Westinghouse was requested to use a 1.0 gpm leakage rate and a margin of 2 on leakage flaw size in the bounding LBB analyses to be presented for staff review. This was identified as DSER Open Item 3.6.3.6-3.

In a separate review, the staff found that the leakage detection capability of 1.9 L/min (0.5 gpm) is acceptable. In subsequent revisions to the SSAR, the following were incorporated, (1) capability to monitor 1.9 L/min (0.5 gpm) leak rate is demonstrated, (2) using absolute sum combination of normal and SSE load with load factor of 1.0, and (3) using a factor of two between critical crack length under normal plus SSE and the leakage crack length under normal load. The staff finds this acceptable, and therefore, DSER Open Item 3.6.3.6-3 is closed.

In addition, during the design review meeting, the staff found that the methodology for the Westinghouse evaluation of LBB candidate piping systems appeared to be similar to that employed by Westinghouse in previous LBB evaluations submitted to the staff for approval. However, the staff was concerned with the benchmarking of the Westinghouse method for leak rate evaluations and requested Westinghouse to benchmark their leak rate evaluation methodology against currently staff-accepted methods, (e.g., using the PICEP computer code).

This was identified as DSER Open Item 3.6.3.6-4. Westinghouse responded to this request by stating that their leak rate evaluation methodology had previously been reviewed and accepted by the staff during LBB reviews for the South Texas Project Units 1 and 2. However, as discussed in the July 1995, design review meeting, the staff was concerned that the Westinghouse leakage rate evaluation methodology may not be acceptable for calculating leakage rates in small size piping and especially in the single-phase, low-temperature flow state. As described in DSER Open Item 3.6.3.4-1, the staff was unable to verify the leakage rate in the Westinghouse LBB analyses. This issue was further discussed during the design review meeting on April 16 through 18, 1997. Westinghouse provided additional detailed information on the procedure that had been used for computer code verification. They benchmarked their codes against other leak rate codes from EPRI and NRC, as well as laboratory test data from Battelle leak rate experiments and in-service leak rate data from the Duane Arnold Nuclear Plant. They stated that the codes had been previously reviewed and accepted by NRC for other plant applications. Westinghouse also explained some of the differences between their code and the PICEP code and explained why the staff results using PICEP may have differences from their results. They indicated that a correction factor needs to be applied to compensate for the different formulas used to calculate section modulus, as well as for different assumptions regarding the shape and smoothness of the postulated crack surface. Based on the additional information, the staff concluded that the Westinghouse software had been adequately benchmarked. Westinghouse provided a description of their code benchmarking methods in Revision 12 to Section 3.6.3.3 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 3.6.3.6-4 is closed.

Furthermore, although not observed in the AP600 LBB evaluations reviewed, Westinghouse was requested to clarify the provision in paragraph 4 of Section 3.6.3.3 of the SSAR that where applied normal operating stress is low in comparison with faulted stress at critical locations, stability is established by analyzing part-through-wall flaws. In RAI 252.12, the staff requested that Westinghouse clarify this criterion. Westinghouse indicated that this criterion may be of relevance for small diameter piping systems (10.2- to 15.2-cm (4- to 6-in.) diameter). The staff concluded that additional justification was required. Westinghouse was requested to provide additional justification for this criterion. This was identified as DSER Open Item 3.6.3.6-5. Subsequently, in Revision 4 to the SSAR, Section 3.6.3.3 was modified to delete the part-through flaw criterion. The staff finds this acceptable, and therefore, DSER Open Item 3.6.3.6-5 is closed.

Preliminary results from small-break LOCA tests performed at Oregon State University indicated that rapid condensation events have the potential to cause unanticipated dynamic loads to occur in the AP600 RCS. PRHR subcooling and sweeping direct vessel injection flow toward the pressurizer appear to be responsible for the condensation events. Although the applicability of these water hammer type loadings to a full-size AP600 system was not yet determined, the staff was concerned that a new potential exists for unanticipated dynamic loads to occur in the AP600 reactor coolant pressure boundary. These water hammer type loads had not been considered in the piping design loads to justify a LBB approach for the AP600 main coolant loop and attached piping. Westinghouse was requested to address whether these water hammer type loads from condensation events need to be considered in its LBB analyses or, if not, justify why these loads could be excluded. This was identified as DSER Open Item 3.6.3.6-6. Westinghouse indicated that evaluations of plant loadings from condensation events were being performed, using in part, data from the AP600 test facility at Oregon State

University. Upon completion of the evaluation, loadings from condensation events would be included in the plant analysis as appropriate. During the design review meeting on April 16 through 18, 1997, Westinghouse discussed the results of their evaluation. In order to measure the effects of the condensation events that were occurring in the test facility, more sensitive instrumentation was installed. Based on these measurements, they found that the events were causing noise but no significant stresses. The pressure peaks from these events were very small (only a few psi) which would not produce high stresses in the piping. Westinghouse also discussed the application of the water hammer screening criteria developed by Professor Griffith of the Massachusetts Institute of Technology (MIT) to the AP600 LBB lines. These criteria were developed for the staff and documented in Section 5.29 of NUREG/CR-6519, "Screening Reactor Steam/Water Piping Systems for Water Hammer," November 1996. The criteria include geometric characteristics such as slope and length of horizontal pipe and thermal-hydraulic criteria such as potential for steam voids and the pressure at time of interest. Based on this review, Westinghouse concluded that the AP600 LBB lines do not have significant potential for water hammer. The staff found the Westinghouse evaluations acceptable and concurs with their conclusion that water hammer type loads from condensation events need not be considered in the LBB analyses of these lines. On this basis, DSER Open Item 3.6.3.6-6 is closed.

3.7 Seismic Design

Using the guideline provided in Sections 3.7.1 through 3.7.4 of the SRP and related RGs, the staff reviewed Revisions 0 through 23 of Sections 3.7.1 through 3.7.3 of the Westinghouse AP600 SSAR. In particular, this portion of the review focused on the seismic analysis and design of the nuclear island (NI) structures of the Westinghouse AP600 standard plant.

During this review, the staff evaluated Westinghouse's responses to the open items identified in the staff's DSER, issued in November 1994, related to the AP600 design certification. The staff also conducted a series of design calculation review meetings at the Westinghouse office in Monroeville, Pennsylvania, and the office of Bechtel Power Corporation (BPC) (a consultant to Westinghouse) in San Francisco, California. In addition, the staff used the public-domain System for Analysis of Soil Structure Interaction (SASSI) computer program and the related model developed by Westinghouse to perform a set of confirmatory analyses regarding soil-structure interaction.

In conducting these review activities, the staff focused on the following objectives:

- Review Westinghouse's design calculations to confirm that the technical issues identified by the staff from the SSAR review are adequately resolved and are properly implemented.

- Compare the staff's confirmatory analysis results with Westinghouse's design calculations to identify and resolve discrepancies.

- Discuss the resolution of open issues in accordance with recent revisions to the SSAR and Westinghouse's responses to the open items identified in the DSER.

As stated in Sections 3.2.1 and 3.7 of the SSAR (in Revisions 8 and 12, respectively), the SSCs of the AP600 standard plant are seismically classified into three categories, depending on their functions. Sections 3.2.1.1 and 3.7.2.8 of the SSAR define these three categories, as well as the requirements for seismic analysis and design of items classified into these categories, as follows:

(1) Seismic Category I (SC-I), in general, applies to all safety-related SSCs, as well as other non-safety-related SSCs that are required to support or protect safety-related SSCs. These SSCs must be designed to withstand the seismic loads associated with the SSE as discussed in Section 3.7 of the SSAR, as well as other applicable loads, without any loss of structural integrity or functional capability. In addition, SC-I structures must be sufficiently isolated from non-seismic Category I structures (defined below).

(2) Seismic Category II (SC-II) applies to SSCs that do not perform a safety-related function, given that the structural failure of those SSCs during an SSE or interaction with SC-I items could degrade the functioning of any safety-related SSC to an unacceptable level, or could result in incapacitating injury to occupants of the MCR. These SSCs must be designed so that an SSE will not cause unacceptable structural failure of or interaction with SC-I items.

(3) Non-seismic (NS) SSCs include those SSCs that are not classified as SC-I or SC-II and do not perform any safety-related function. Section 3.7.2. of the SSAR defines the criteria used for the design of these NS SSCs.

Using these three categories, Westinghouse's early SSAR revisions classified all NI structures (including the foundation mat) as SC-I. Similarly, SC-II SSCs included Annex I and II buildings, the high bay area of the radwaste building, and the turbine building. In addition, Westinghouse applied the NS designation to all areas of the radwaste building other than the SC-II high bay areas.

The staff's review of the early SSAR revisions raised a number of concerns in the following areas:

- definition of SC-II and NS structures
- difference between non-Category I and NS structures
- seismic design requirements for non-Category I and SC-II structures
- inclusion of general design requirements for SC-II structures in Section 3.7 of the SSAR
- justification of using the Zone 3 requirements of the Uniform Building Code (UBC) instead of the SC-I seismic design criteria for the analysis and design of the SC-II structures

The staff then reviewed Westinghouse's responses to the issues identified, as well as Revision 2 of SSAR Sections 1.2.5 through 1.2.8 and Section 3.7.2.8. The staff also held discussions with Westinghouse and its consultants during review meetings. This additional review revealed that Westinghouse had reclassified the Annex I and II buildings as SC-II, and the turbine and the radwaste buildings as NS structures.

The staff concluded, however, that it is not acceptable to classify the turbine and the radwaste buildings as NS structures because (as discussed in SSAR Section 3.7.2.8 and shown in SSAR Figure 1.2-2) the radwaste building is very close to the NS structures. Also, one end of the floors between the turbine building main structure and the NI are supported on the NI structures. Consequently, these structures must be designed to criteria equivalent to those used for the SC-II structures, so that any collapse of the radwaste building structure or the floors between the turbine building and the NI will not cause these structures to either strike or impair the integrity of the NI structures in the event of an SSE. The staff identified this issue as Open Item 3.7-1. However, Open Item 3.7-1 is considered closed because its similarity to Open Items 3.7.2.8-3 and 3.7.2.8-5, which addressed related staff concerns regarding the acceptability of classifying the radwaste and turbine buildings as NS structures. Section 3.7.2.8 of this report discusses the staff's evaluation and conclusions regarding these open items. Section 3.7.2.8 of this report also discusses the staff's evaluation of the analysis method and design criteria for SC-II structures and NS structures, which Westinghouse discussed in Section 3.7.2.8 of the SSAR.

3.7.1 Seismic Input

As described in Section 3.7.1.1 of the SSAR, the input seismic design ground motion response spectra for the SSE are defined at plant-finished grade in the free field. The horizontal and vertical design ground motion response spectra for the AP600 standard plant were developed using the RG 1.60 response spectra as the basis and considering the high-frequency amplification effects. The relative values of the spectral amplification factors for the design response spectra are shown in SSAR Table 3.7.1-3. Also, the horizontal and vertical ground motion response spectra corresponding to 2, 3, 4, 5, and 7 percent of the critical damping are shown in Figures 3.7.1-1 and 3.7.1-2, respectively, of the SSAR as well as Figures 3.7-1 and 3.7-2 of this report. In addition, to ensure that the input ground motion satisfies the 60-percent guidance specified in SRP Section 3.7.2.II.4, Westinghouse provided the envelope of the horizontal and vertical response spectra at the foundation level (in the free field) in SSAR Figures 3.7.1-18 and 3.7.1-19, respectively as well as Figures 3.7-3 and 3.7-4 of this report. The staff's review of the adequacy of these response spectra is discussed in Section 2.5.2 of this report.

The peak ground acceleration (PGA) for both horizontal and vertical components of the SSE is 0.3g. For the standard plant design, Westinghouse employed the SSE ground motion to calculate the responses of the SC-I SSCs. The staff's evaluation of the proposed design ground motion for the SSE is in Section 2.5.2 of this report.

In Section 3.7.1 of the SSAR, Westinghouse stated that it used a single set of three components of the synthetic SSE ground motion acceleration time history as input motion for the seismic analysis and design of the AP600 SC-I SSCs. Specifically, Westinghouse generated these three components of the ground motion time history by modifying a set of actual recorded time histories from the Taft earthquake (the Kern County earthquake recorded at Taft, California). These time histories included a total duration equal to 20 seconds with a corresponding, stationary phase characterized by strong motion with a duration greater than 6 seconds. Westinghouse then adjusted the amplitude and frequency of the resulting response spectra to obtain response spectra for 2, 3, 4, 5, and 7 percent of the critical damping. In that way, the adjusted response spectra envelop the SSE design ground response spectra at a sufficient number of frequency points as recommended by Section 3.7.1 of the SRP.

In Revision 9 of SSAR Section 3.7.1.2, Westinghouse also stated that the three components of the ground motion time histories were generated with time step size of 0.01 second for applications in the seismic analyses. For applications in the fixed-base modal superposition time-history analyses, the time step size was reduced by linear interpolation to 0.005 second. The cutoff frequency used in the horizontal and vertical seismic analyses of the NI for the hard rock site is 33 Hertz (Hz). The cutoff frequencies used in the SSI analyses are 33 Hz for the soft rock site, 15 Hz horizontal and 21 Hz vertical for the soft-to-medium soil site, and 20 Hz horizontal and 33 Hz vertical for the upper bound soft-to-medium soil site.

Use of the time step size of 0.01 second for the SSI analyses and 0.005 second for the fixed-base seismic analyses is consistent with both common industry practice and the staff's past review and approval of the conventional nuclear power plants. Therefore, the time step size proposed by Westinghouse is acceptable. As far as the cutoff frequencies used in the various seismic analyses, the results from the confirmatory SSI analyses performed by the staff indicate that the seismic responses of the NI structures above the cutoff frequencies are negligible. On this basis, the staff concludes that the cutoff frequencies used by Westinghouse are acceptable.

The power spectral density function (PSDF) of the horizontal synthetic SSE ground motion time history envelops the target PSDF specified in Appendix A to Section 3.7.1 of the SRP for a frequency range of 0.3 to 24 Hz. Consequently, the PSDF of the horizontal synthetic SSE ground motion time history ensures a conservative design and, therefore, the use of a single set of ground motion time histories as input for the seismic analysis is acceptable.

Westinghouse did not generate a target PSDF for the vertical component of the ground motion time history. Instead, Westinghouse used the horizontal target PSDF as the vertical target PSDF and demonstrated that the PSDF of the vertical synthetic SSE ground motion time history satisfied the PSDF enveloping requirement. Because the vertical response spectra specified by RG 1.60 are completely enveloped by the horizontal response spectra in RG 1.60, the use of the horizontal target PSDF as the vertical target PSDF is acceptable to the staff.

In addition, Westinghouse showed that the three components of synthetic time history are statistically independent from each other. To do so, Westinghouse demonstrated that the cross-correlation of coefficients at zero time lag between these three earthquake components are less than 0.16, as referenced in Section 3.7.1 of the SRP.

In light of the factors discussed above, the staff concludes that the SSE input ground motion (the design ground motion response spectra and ground motion acceleration time histories) documented in Section 3.7.1 of the SSAR meets the guidelines of SRP Section 3.7.1 and RG 1.60 and, therefore, is acceptable. Consequently, it is also acceptable to define the design ground motion in accordance with the SRP guidelines at plant- finished grade in the free field, and to use that definition in calculating the seismic responses (both structural member forces and floor response spectra) for the plant structures founded on a uniform site such as deep soil and rock sites.

For a shallow soil site, the input ground motion should be defined at a hypothetical rock outcrop, as indicated in Section 3.7.1 of the SRP. Consequently, as stated in the DSER, the exclusion of the shallow soil site as one of the standard site conditions was unacceptable and identified as DSER Open Item 3.7.1-1.

In Revision 9 of SSAR Section 3.7.1, Table 2-1 and Appendix 2A, Westinghouse indicated that the soil shear wave velocity of the selected site conditions is greater than 304.8 m/sec (1000 ft/sec). Westinghouse also indicated that the depth of the soil layer overlying the bedrock is equal to or greater than 36.58 m (120 ft). On that basis, Westinghouse concluded that shallow soil sites need not be included in the AP600 standard plant design. Results of the confirmatory analysis performed by the staff verified the adequacy of Westinghouse's conclusion. Therefore, Open Item 3.7.1-1 was considered closed as a result of the staff's review of SSAR Revision 9.

In mid-1996, Westinghouse proposed to allow the AP600 NI to be founded on potential shallow soil sites by considering this issue as a site-specific COL action item. The 0.3g SSE is a fundamental site parameter for the AP600 design, as documented in Section 5.0 of the Tier 1 information. However, because Westinghouse has not demonstrated that the AP600 design is adequate for shallow soil sites using a standard design SSE value of 0.3g, the staff finds no basis to include shallow soil sites in its final design approval. The staff recognizes that a standard design that envelops many site conditions may be suitable for certain shallow soil sites where the seismic hazard is low (for example, a site with a 0.1g site-specific SSE); nonetheless, the acceptability of such a site would have to be determined under the license application. In addition, in 10 CFR Part 52, the NRC requires that the detailed design (including the piping design, seismic qualification of equipment, and seismic assessment of the plant) must be analyzed and evaluated using the seismic site parameter which in this case is an SSE with a PGA of 0.3g. On these bases, the staff cannot accept Westinghouse's proposal to use certain shallow soil sites for the AP600 standard design by considering this a site-specific COL issue.

In letters from T. Quay (NRC) to N. Liparulo (Westinghouse), dated November 4, 1996, and January 31, 1997, the staff stated its position and provided three options for resolution of this issue:

(1) Westinghouse could adopt the design site-specific response spectrum (DSSRS) approach used for the System 80+ reactor design. The suitability of a future AP600 site would then be established through a simple comparison of the DSSRS and the site-specific response spectrum.

(2) The SSAR could state that the AP600 is not designed for shallow soil sites. (That is, the approved design site parameter could exclude shallow soil sites.)

(3) The SSAR could evaluate the design against a separately specified design site parameter for shallow soil sites (e.g., 0.2g with design response spectra specified by RG 1.60).

Westinghouse responded to the staff's position in letters dated January 28 and February 10, 1997. In addition, as discussed in a letter dated March 26, 1997, Westinghouse addressed the staff's concern regarding the shallow soil site by revising SSAR Section 2.5. Section 2.5.4.1 of this report provides additional discussion regarding the staff's review and conclusions concerning the issue of the shallow soil site. On that basis, Open Item 3.7.1-1 is considered closed.

The damping ratios used in the analysis of the AP600 SC-I structures (as documented in SSAR Table 3.7-1) comply with the SSE damping ratios specified in RG 1.61. For soil foundations, damping values (soil material damping limited to 15 percent of critical damping) are determined on the basis of the soil shear strains induced in the free field. The approach used by Westinghouse for considering soil damping (including limiting the soil material damping to 15 percent) meets the guidelines prescribed in Section 3.7.2 of the SRP and, therefore, is acceptable. However, in Table 3.7-1 of early SSAR revisions, Westinghouse proposed to use 20-percent damping for analyzing the cable tray and supports; 7-percent damping for the heating, ventilation, and air conditioning (HVAC) duct systems; and 20-percent damping for fuel assemblies. By contrast, the staff's previous review experience suggested that the damping values of 20 percent for the cable tray and supports, and 7 percent for the HVAC duct systems with welded construction, are too high and are not acceptable for the following reasons:

- Welded HVAC ductwork should be considered as welded steel structures. According to RG 1.61, 4 percent damping should be used for the seismic analysis.

- In Revision 2 of SSAR Table 3.7.1-1 and Figure 3.7.1-13, Westinghouse specified a constant damping value of 20 percent for the seismic analysis of cable tray systems. However, the test reports referenced by Westinghouse (Reference 19) indicated that high damping ratio (e.g., 20 percent) was recorded only for cases including a high percentage of cable fill, no application of fire protection spray, and/or bolted hanger and tray connections.

- A damping ratio of 20 percent is not justifiable for cable tray systems with welded frame type supports, because the damping ratio specified in RG 1.61 for welded steel structures is only 5 percent.

For these reasons, the issue regarding the damping values for cable tray and HVAC systems was identified as Open Item 3.7.1-2.

As a result of the staff's review of SSAR Revision 2 and discussions during review meetings, Westinghouse changed the damping ratio for HVAC ductwork from 5 percent to 4 percent in Revision 2 of the SSAR (Table 3.7.1-1), and modified the damping ratios of the cable tray seismic analysis in Revision 12 to the SSAR (SSAR Section 3.7.1.3, Table 3.7.1-1, and Appendix 3F). In Revision 12 to the SSAR, Westinghouse stated that 10-percent damping is used for both full and empty cable tray and support systems. If the configuration of cable tray systems is demonstrated to be similar to the configurations tested in SSAR Reference 19, the damping ratio shown in SSAR Figure 3.7.1-13 is used with the specified application procedure for the seismic analysis. In reviewing Revisions 2 and 12 of the SSAR, the staff found that Westinghouse's commitment for the use damping ratios of both HVAC and cable tray systems is consistent with those accepted by the staff in the review of other advanced reactors such as the advanced boiling-water reactor (ABWR), and meets the guidelines prescribed in RG 1.61 and recommended in the Brookhaven National Laboratory report, "Recommendations for Revision of Seismic Damping Values in Regulatory Guide 1.61," dated November 1995. On this basis, the staff concludes that the damping ratios proposed by Westinghouse are acceptable, and Open Item 3.7.1-2 is considered closed.

With regard to the damping to be used for fuel assemblies in the AP600 seismic analysis, the staff found that the value of 20 percent used by Westinghouse appears high in comparison with the fuel assembly damping values used in other nuclear power plants. Consequently, the staff requested that Westinghouse justify the use of such a high damping value for AP600 fuel assemblies. The concern regarding the use of 20-percent damping for the fuel assembles was identified as Open Item 3.7.1-3. Westinghouse subsequently provided the requested justification as a part of the responses to open issues related to the analysis and design of major components. The staff's evaluation with regard to the adequacy of using the 20-percent damping ratio for AP600 fuel assemblies is in Section 3.9.5 (Open Item 3.9.5-4) of this report. On the basis of the resolution of Open Item 3.9.5-4, Open Item 3.7.1-3 is considered closed.

In Revision 7 of SSAR Section 3.7.1.3 and Table 3.7.1-1, Westinghouse proposed to use a damping ratio of 5 percent for the AP600 piping analysis. For the seismic analysis of the reactor coolant loop piping, Westinghouse proposed to use the damping ratios specified in its report "Damping Values of Nuclear Power Plant Components" (WCAP-7921-AR), May 1974. Section 3.12.5.4 (Open Item 3.12.5.4-1) of this report discusses the staff's evaluation regarding the adequacy of the specified damping values for the AP600 piping analysis.

As described in Section 3.7.1 of early SSAR revisions, the four AP600 SC-I structures (shield building, containment building, containment internal structures, and auxiliary building) are supported on a common foundation mat and form the NI. This NI foundation is approximately rectangular, with dimensions of 77.4 m (254 ft) long and 35.2 m (115.5 ft) wide. The foundation embedment depth (measured from finished grade to the bottom of the foundation mat) is 12.04 m (39.5 ft). In the auxiliary building area, the foundation mat is 1.83 m (6.0 ft) thick and its thickness in the shield building area varies from 6.5 m (22 ft) at the edge to 1.83 m (6.0 ft) at the center.

During its review, the staff questioned the adequacy of the 1.83 m (6.0 ft) thickness of the foundation mat, given that the AP600 standard plant has to be designed for a full range of site conditions. The staff designated this concern as Open Item 3.7.1-4. Section 3.8.5 (Open Item 3.8.5-9) of this report discusses Westinghouse's resolution of a similar issue raised by the staff, with regard to the design of the foundation mat. On the basis of the resolution for Open Item 3.8.5-9, Open Item 3.7.1-4 is closed.

The staff also expressed a concern that Westinghouse's early SSAR revisions did not provide any key dimensions (such as the size of the foundation mat), the radius of the shield building, the geometry of the shield building roof, or the thickness of the periphery walls, shield building walls and major structural walls. These dimensions are the key parameters for the seismic analyses, and any changes to these dimensions will significantly affect the dynamic responses (structural member forces and floor response spectra) of the NI structures. The staff identified this omission as Open Item 3.7.1-5.

As a result of discussions during the meeting on June 12 through 17, 1995, Westinghouse revised Figures 3.7.1-16 and 3.7.2-12 (in Revision 7 of the SSAR) to provide the overall dimensions for the foundation mat, the distance between column lines, and the distance between the edge of the foundation mat and the center of the containment shell. The staff's subsequent evaluation revealed that Westinghouse has provided sufficient information (dimensions) for the development of the seismic model of the NI structures including the foundation mat. On this basis, Open Item 3.7.1-5 is considered closed.

For the design of the NI structures, Westinghouse initially considered a set of three design site-conditions with various shear wave velocities. Specifically, these three site conditions are hard rock site, soft rock site, and soft-to-medium stiff soil site. For the hard rock site, Westinghouse assumed that a uniform shear wave velocity of 2438.4 m/sec (8000 ft/sec). For the soft rock site, Westinghouse considered a shear wave velocity of 731.5 m/sec (2400 ft/sec) at ground surface, increasing linearly to 975.4 m/sec (3200 ft/sec) at a depth of 93.2 m (240 ft), with base rock (bedrock) at a depth of 36.6 m (120 ft). For the soft-to-medium stiff soil site, Westinghouse used a shear wave velocity of 304.8 m/sec (1000 ft/sec) at ground surface, increasing linearly to 731.5 m/sec (2400 ft/sec) at 73.2 m (240 ft) below ground surface, with base rock (bedrock) at a depth of 36.6 m (120 ft). For each of these three site-conditions, Westinghouse assumed that ground water is at grade level.

As discussed in the DSER, the staff raised the following concerns regarding the adequacy of using only three generic design site-conditions to generate seismic response envelopes (structural member forces and floor response spectra) for use in designing SC-I structures and subsystems (such as piping systems and major components) of the AP600 standard plant:

- For a more realistic representation of soil property (shear wave velocity) distribution through the thickness of the soil layer, the staff requested that for soil sites, Westinghouse vary the shear wave velocity parabolically from ground surface to the bedrock. In Revision 7 of SSAR Section 3.7.1.4, Westinghouse committed to vary the shear wave velocity parabolically for both the soft-to-medium soil site and the upper bound soft-to-medium soil site. The revised SSAR commitment by Westinghouse is acceptable.

- The staff requested that Westinghouse demonstrates that the analysis results generated using the three selected design site-conditions will envelop the seismic responses at sites with different shear wave velocities, such as 457.2 m/sec (1500 ft/sec), 1066.8 m/sec (3500 ft/sec), and so forth. The staff identified this request as Open Item 3.7.1-6.

  As a result of discussions during the meeting on February 28 through March 2, 1995, Westinghouse added an additional site condition for the AP600 standard plant design, as documented in SSAR Revision 7 (Section 3.7.1). The dynamic properties of this site-condition include foundation soil with a shear wave velocity of 430.99 m/sec (1414 ft/sec) at ground surface, increasing parabolically to 731.52 m/sec (2400 ft/sec) at 73.2 m (240 ft) below ground surface, with the bedrock at 36.6 m (120 ft) below ground surface.

  Westinghouse also compared the newly generated floor response spectrum envelopes (four design site-conditions) with those initially generated (three site conditions). Through this comparison, Westinghouse demonstrated that the envelopes cover the uncertainties in the range of soil shear wave velocities between 304.8 m/sec (1,000 ft/sec) and 731.5 m/sec (2,400 ft/sec). On that basis, Open Item 3.7.1-6 is considered closed. (Section 3.7.2.4, Open Item 3.7.2.4-12, of this report also addressed this issue).

- Westinghouse should use a soil shear strain degradation model that is more recent than the model recommended by Seed and Idriss in 1970 for the SSI analysis of AP600 NI structures. The staff's evaluation regarding this issue is discussed in detail in Section 3.7.2 of this report (Open Item 3.7.2.4-5). On the basis of the resolution for Open Item 3.7.2.4-5, the issue is considered resolved.

3.7.1.1 Site Interface Parameters

In Section 2.0 and Table 2.0-1 of the early SSAR revisions, Westinghouse specified that the COL applicant will use the following design site-parameters to confirm the adequacy of the AP600 seismic design for a specific site:

- The site-specific ground motion response spectra are bounded by the modified RG 1.60 design response spectra anchored to 0.3g as shown in SSAR Figures 3.7.1-1 and 3.7.1-2.

- No potential for fault displacement is expected at the site.

- No liquefaction is expected at the site.

- The maximum soil bearing reaction at a corner of the AP600 NI foundation is below 526.68 kPa (11000 lb/ft^2).

- Soil shear wave velocity is equal to or greater than 304.8 m/sec (1000 ft/sec).

The staff concludes that the above design site-parameters are reasonable and acceptable bounding limits for the COL applicant to use in confirming the adequacy of the AP600 seismic design. Section 2.5.4 of this report discusses the staff's evaluation of potential soil liquefaction. However, because the AP600 standard plant has to be designed for a full range of site conditions (including a shallow soil site), Westinghouse is required to commit, in the SSAR, that a potential plant site also needs to meet the following bounding parameters:

- For a shallow soil site, the site-specific ground motion response spectra and associated time histories should be specified as the free field ground motion at a level that complies with the guidelines prescribed in Section 3.7.1.I.1 of the SRP.

- When a seismic SSI analysis is performed for the SSE ground motion, the three components of the site-specific ground motion time history must have a PGA equal to 0.3g. The response spectra generated from these three components of the ground motion time history must satisfy the response spectrum enveloping criterion of Section 3.7.1 of the SRP for all damping values to be assigned for the structural elements and the enveloping criterion for the PSDF.

The staff identified this requirement as Open Item 3.7.1.1-1.

To resolve this issue, Westinghouse agreed (at the meeting on August 4 through 8, 1997) with the staff to revise the SSAR to state that any applicant using the AP600 certified design must compare the free field site specific SSE ground motion response spectra at both the ground surface and the foundation level (12.19 m (40 ft) depth) to the corresponding ground motion response spectra shown in SSAR Figures 3.7.1-1, 3.7.1-2, 3.7.1-18 and 3.7.1-19, respectively. The COL applicant should also demonstrate that the site-specific SSE ground motion response spectra are enveloped by the SSAR ground motion response spectra at both locations. The staff's evaluation of this issue is in Section 2.5 of this report. On the basis discussed above, Open Item 3.7.1.1-1 is closed.

3.7.1.2 Conclusion

On the basis discussed above, the staff concludes that Westinghouse meets the relevant requirements of GDC 2 and Appendix A to 10 CFR Part 100. Specifically, the staff finds that Westinghouse has given appropriate consideration to the most severe earthquake (SSE) to which the AP600 SC-I SSCs are expected to be subjected. In particular, the following factors demonstrate Westinghouse's fulfillment of these requirements:

- SSE design response spectra developed from RG 1.60 with enrichment in the frequency range from 15 to 33 Hz
- synthetic ground motion time histories that comply with the design response spectrum and PSDF enveloping criteria specified by Section 3.7.1 of the SRP
- specific percentage of critical damping values in the seismic analysis of AP600 SC-I SSCs that conform to the guidelines of RG 1.61

These factors ensure that the seismic inputs are adequately defined to form a reasonable basis for the design of AP600 SC-I SSCs to withstand seismic loadings.

3.7.2 Seismic System Analysis

The review scope of the seismic system analysis for the AP600 standard plant considers the seismic analysis methods and acceptance criteria for all SC-I structures, systems and components. It includes the review of basic assumptions, procedures for modeling, seismic SSI analyses, development of in-structure response spectrum envelopes, inclusion of torsional effects, overturning and sliding evaluation of SC-I structures, and determination of composite damping. The review also covered design criteria and procedures for evaluation of the interaction of non-seismic Category I structures with SC-I structures and the effects of parameter variations on floor response spectra.

AP600 structures, systems and components have been classified in accordance with Regulatory Guide (RG) 1.29. However, non-seismic Category I structures, systems and components are further classified into SC-II and non-seismic. The staff's evaluation of the seismic classification of structures, systems and components is discussed in Sections 3.2 and 3.7 of this report. In Section 3.7.2 of the SSAR, Westinghouse states that the AP600 SC-I building structures consist of the steel containment vessel, containment internal structures, and coupled shield and auxiliary buildings. These structures are founded on a common foundation mat and form the NI structures. The NI foundation mat is also classified as a SC-I structure. All other building structures are classified as either SC-II or non-seismic.

As described in Section 3.7.2 of the SSAR, SC-I structures are analyzed and designed for the SSE specified in Section 3.7.1 of the SSAR and the criteria described in Section 3.7.2 of the SSAR. SC-II building structures are designed for the SSE using the same methods as are used for SC-I structures. Non-seismic structures are analyzed and designed for seismic loads according to the UBC requirements for either Zone 2A with an importance factor of 1.25 or Zone 3 with an importance factor of 1.0. The staff's review of the analysis and design results are discussed in the following sections.

3.7.2.1 Seismic Analysis Methods

In Section 3.7.2.1 of the SSAR, Westinghouse states that the analysis of SC-I building structures (designated in the SSAR as the NI structures) consists of (1) determination of seismic loads (forces and moments) for the design of NI building structural components, and (2) the development of in-structure response spectra (or floor response spectra) which are to be used as input motions for the subsystem (piping and equipment) analysis and design.

As described in early SSAR revisions, four major procedural steps were followed for performing seismic analyses and calculating seismic design loads for the design of floors and walls of the NI structures, except the foundation mat, embedded peripheral walls, and shield building roof structures:

(1) For the hard rock site with a foundation shear wave velocity of 2438.4 m/sec (8000 ft/sec) or greater, a response spectrum analysis, using a three-dimensional (3D) fixed-base finite element model (Model A) of the NI structures and the ground motion defined in Section 3.7.1, and Figures 3.7.1-1 and 3.7.1-2 of the SSAR as input at the rock surface (fixed base model), was performed. In this model, the walls and floors were explicitly represented by plate and shell elements. SSI effects attributable to the soil or rock flexibility was not included in this model, and the computer program BSAP was used to perform the analysis. Seismic loads (forces and moments) obtained from this fixed-base model analysis were multiplied by a factor (hereafter called SSI factor) to include the SSI of other site conditions and to determine the final design seismic loads for the floor and walls of the NI structures. This SSI factor was determined as described below.

(2) Using BSAP computer code and a 3D fixed-base lumped-mass stick model (Model B) with the same input ground motion applied at the rock surface used in Step (1) above and corresponding ground motion time histories, response spectrum and time history analyses were performed. When this model was developed, the lateral restraints were used at floor elevations 25.1 m (82.5 ft) and 30.5 m (100 ft) and the stiffness contribution from the peripheral walls below grade were included in the Model B. The purpose of these analyses is to generate floor response spectra for the hard rock site and member forces for developing SSI factors.

(3) Westinghouse used the SASSI Computer Code (frequency domain solution) to perform seismic SSI analyses of the NI structures for two design site conditions (soft-to-medium stiff soil site and soft rock site). In these analyses, Westinghouse chose to use a 3D

lumped-mass stick model to represent the NI structures, and to use the finite element model to represent the foundation mat and peripheral walls below grade. Results from these analyses include seismic forces (axial forces, shear forces and bending moments) and floor response spectra.

(4) Then, Westinghouse compared the seismic loads (forces and moments) at various locations of the NI structures obtained from SASSI analyses in Step (3) above with those from the Model "B" BSAP analysis of Step (2) above. For elements where seismic loads obtained from Step (3) SASSI analyses are larger, the SSI factor was computed as the ratio between SASSI loads (Step (3)) to Model "B" BSAP loads (Step (2)). If, for a particular element, the seismic loads from the SASSI analyses for the two design soil conditions were larger than those from the Model "B" BSAP analysis, the larger of the two calculated SSI factors was used to compute the design seismic loads (see Step (1)). If SASSI loads for any structural element were less than those from Model "B" BSAP analysis, the SSI factor was taken as unity. Thus, the structural design of individual structural elements (shear walls, slabs, etc.), was carried out using the Model "A" BSAP results modified by the SSI factors, as appropriate.

For the foundation mat and shield building roof structures, design loads were obtained by equivalent static analyses using nodal accelerations from the analysis on the basis of the 3D lumped-mass stick model. However, the SSAR did not indicate how these nodal accelerations were calculated on the basis of Step (2) analysis (Model "B," BSAP analysis) or Step (3) analysis (SASSI analysis). Also, the SSAR did not address how the loads used for the design of embedded peripheral walls were calculated. The concern regarding the calculation of nodal accelerations based on Step (2) or Step (3), and the design loads for the embedded walls is addressed in separated sections below.

The design in-structure response spectrum envelopes were generated by enveloping the in-structure response spectra computed from the SASSI analyses for the two design soil site conditions in Step (3) above and the in-structure response spectra obtained from the Model "B" BSAP fixed-base analysis in Step (2) above.

As a result of its review of the early versions of the SSAR and Westinghouse's response to the questions raised in its October 1, 1992, January 26, 1994, and March 16, 1994, letters and discussions with Westinghouse during design review meetings, the staff found that the four major procedural steps used by Westinghouse for generating seismic responses generally meet the guideline prescribed in Section 3.7.2 of the SRP, except that the following concerns need to be resolved by Westinghouse:

(1) To avoid underestimating the seismic response (acceleration, forces and moments), the guidelines prescribed in Section 3.7.2 of the SRP state that the effects of high frequency modes should be adequately considered. To account for such effects in the calculation of seismic forces and moments, the SRP guidelines suggest two methods. One uses a sensitivity test in which it is demonstrated that the inclusion of additional modes does not result in more than a 10 percent increase in response. The other (alternative) method uses all the modes having frequencies up to 33 Hertz (Hz), and accounts for the remaining higher frequency modes in accordance with the guidelines stated in Appendix A to Section 3.7.2 of the SRP.

Section 3.7.2.1 of early SSAR revisions states that when the response spectrum method was used to determine the seismic forces and moments for certain structures (specifically, containment internal structures), the method in Appendix A to Section 3.7.2 of the SRP was used. However, when the modal time-history analysis method was used (for all structures other than containment internal structures), Westinghouse did not demonstrate how the effects of higher frequency modes were adequately accounted for in determining the in-structure response spectra, and the seismic forces and moments. The staff review of Tables 3.7.2-1 and 3.7.2-4 of the SSAR found that when a 33 Hz cut-off frequency was used (1) only 72 percent of the total structural mass participated in the horizontal response and 45 percent of total mass participated in the vertical response for the coupled shield and auxiliary building, and (2) only 51 percent of the total mass participated in the horizontal response and 47 percent of total mass participated in the vertical response for overall NI structures. The staff concluded that if the modal time history analysis method is used for seismic analyses of the NI structures, Westinghouse should justify the adequacy for not including the high frequency modes in the analyses. More details about this issue are discussed in Section 3.7.2.2 below.

(2) During the meeting on July 11 through 14, 1994, the staff found that, in order to verify the appropriateness of the stiffness value of certain hypothetical "rigid" beams used in the SASSI model, Westinghouse compared the seismic moments and shears calculated by the SASSI computer code to those obtained by an equivalent static method with a simplified stick model at various elevations. The staff's concern is that the use of results obtained from a single stick model and the equivalent static analysis to verify the adequacy of the results from the SASSI soil-structure model might not be appropriate. The staff requested Westinghouse to demonstrate that results from the SASSI analysis (moments and shears), which are used to determine the SSI factors (see Step "1" procedure described above), are not sensitive to the "rigid" beam stiffness values used. The staff identified this concern as Open Item 3.7.2.1-1.

During the June 12 through 16, 1995 meeting, the staff reviewed the design calculations presented by Westinghouse and found that the difference between the moments and shears calculated on the basis of the assumed "rigid" beam stiffness and those on the basis of the assumed stiffness multiplied by 1000 is insignificant. This finding implies that the SSI factors are not sensitive to the "rigid" beam stiffness values used. The staff accepted Westinghouse's demonstration and concludes that Open Item 3.7.2.1-1 is closed.

(3) It is the staff's understanding that like many other computer programs, the SASSI computer code results can be sensitive to modification of the code. As such, any modification to the computer program should be documented and validated. In its letter dated January 26, 1994, the staff requested that Westinghouse inform the staff of any changes made to the version of the SASSI code which was used for the SSI analyses of the AP600 design and validate the revised computer code. During the July 1994 meeting, Westinghouse agreed to provide the validation package of the code for the staff review. The staff identified this concern as Open Item 3.7.2.1-2.

During the meeting on June 12 through 16, 1995, the staff reviewed the validation package of the SASSI code presented by Westinghouse and found that only minor changes were made to this computer code and these changes would not affect the results generated for the AP600 design. On this basis, Open Item 3.7.2.1-2 is considered closed.

(4) In its letter dated March 16, 1994, the staff questioned the adequacy of using the fixed-base model and the BSAP computer code to perform seismic analysis for the hard rock, for which the foundation shear wave velocity is equal to or greater than the 2438.4 m/sec (8000 ft/sec). The staff identified this concern as Open Item 3.7.2.1-3.

At the meeting on February 28 through March 2, 1995, the staff reviewed a comparison of results from the fixed-base model analysis using the BSAP computer code and the SSI analysis of hard rock sites using the SASSI computer code in which the shear wave velocity for the hard rock was assumed to be 2438.4 m/sec (8000 ft/sec). The comparison showed that the difference between the results obtained from these two analyses is insignificant. On the basis of the comparison presented by Westinghouse, the staff concludes that the use of the fixed-base model to calculate seismic responses of the NI structures for hard rock sites is reasonable and acceptable. On this basis, Open Item 3.7.2.1-3 is considered closed.

(5) From its review of early SSAR revisions, the staff found that many different 3D dynamic models (3D stick model, 3D finite element model, 3D stick model coupled with 3D finite element soil foundation model, etc.) and analysis methods (response spectrum analysis method, time history analysis method using BSAP code, time history analysis method using SASSI code, etc.) were used for the seismic analyses of the NI structures. However, it was not clear to the staff which model combined with which analysis method was used for generating what kind of dynamic responses for the design. In its letter dated March 16, 1994, the staff requested Westinghouse to clarify this issue. In Revision 2 of the SSAR, Westinghouse provided Table 3.7.2-14, which summarizes the types of model and analysis methods that were used in the seismic analyses of the NI structures, and the types of results that were obtained and where they were used in the design. The staff reviewed the response provided by Westinghouse and found that the information provided in this table was acceptable in general, except that Westinghouse should complete the following actions:

- Add a description of the axisymmetrical model used for calculating containment shell stresses

- Clarify the procedure for using member forces obtained from the stick model to establish the scaling factor which was applied to the in-plane forces of the finite element model for the design of walls and floors

- Describe in detail and justify the perturbation made to correct the SASSI member forces to account for erroneous rigid beam stiffness.

The staff identified this concern as Open Item 3.7.2.1-4.

The staff reviewed Revision 7 of the SSAR and found that Westinghouse (1) provided a description of the axisymmetric model used for calculating containment shell stress in SSAR Subsection 3.8.2.4.1.1, (2) described the procedure for using member forces obtained from the stick model to establish the scaling factor which was applied to the in-plane forces of the finite element model for the design of walls and floors, and (3) presented its justification for the perturbation made to correct the SASSI member forces to account for erroneous rigid beam stiffness. The staff also found that the description of the process for using the axisymmetric model to compute containment shell stresses, the procedure for developing scaling factors, and the justification for the perturbation made to correct the SASSI member forces either meet the guideline prescribed in SRP Section 3.7.2 or are consistent with the common industry practices. Westinghouse's SSAR statements (Revision 7) resolved the staff's technical concern for this open item. On this basis, Open Item 3.7.2.1-4 is closed.

In Revision 13 of the SSAR, Westinghouse added two new paragraphs to Section 3.7.2.1.1. In these two new paragraphs, Westinghouse stated that response spectrum analyses are also performed using the coupled auxiliary and shield buildings on a flexible base. The same 3D finite element model used for the fixed-base hard rock site response spectrum analysis was used, except that plate elements representing the foundation mat, and horizontal and vertical soil springs were added to represent the flexibility of the subgrade. The purpose of the flexible base model is to provide the relative distribution of loads to the various shear walls when plant is located on a soil site. SSI adjustment factors were applied so that the overall forces in the structure match corresponding results from the SSI analyses performed using the SASSI computer code. The staff's review of this new SSAR change and analysis results during the meeting on August 11 through 15, 1997, found that the use of flexible base model is consistent with the common industry practice and the results are more conservative in comparison with those from the analyses on the basis of a rigid foundation mat. Therefore, this new SSAR change is acceptable.

Also, during the meeting on August 11 through 15, 1997, Westinghouse provided a markup of Revision 17 of the SSAR for review. In the draft SSAR, Westinghouse made two additional changes to Section 3.7.2.1.1. The description of these two changes and the staff's review results are discussed below:

(1) Westinghouse replaced the sentence, "the analyses are performed using the 3D, finite element models of the coupled shield and auxiliary buildings and the containment internal structures ...," in the second paragraph with the sentence, "the analyses are performed using the 3D, finite element models of the coupled shield and auxiliary buildings and the stick models of the steel containment vessel and the containment internal structures...."

(2) Westinghouse added a new paragraph for the analysis of the containment internal structures. This new paragraph states that response spectrum analyses of the containment internal structures on a fixed-base are performed using the 3D, finite element models of the containment internal structures. The forces obtained from the response spectrum analyses of the finite element models for the hard rock site are increased by a scaling factor as described for the coupled auxiliary and shield buildings.

During this meeting, Westinghouse added that the fixed-base of the coupled shield and auxiliary buildings model (Item 1 above) is located at Elevation 20.27 m (66.5 ft) and the fixed-base of the containment internal structures model (Item 2 above) is located at Elevation 25.15 m (82.5 ft). The input ground motion for the analysis of both models is the ground motion specified in SSAR Figures 3.7.1-1 and 3.7.1-2. The staff reviewed these SSAR changes and found that the modeling technique used for these two fixed-base models meets the SRP Section 3.7.2.3 guideline, and therefore, is acceptable. The use of fixed-base model for structures founded on hard rock is acceptable on the basis of the discussion in Open Item 3.7.2.1-3. With regard to the use of same ground motion for the analysis of these two models supported at different elevations, the staff reviewed the response spectra at Elevations 20.27 m (66.5 ft) and 25.15 m (82.5 ft) and found that the differences between the two sets of response spectra are very small. On this basis, the input ground motions are also acceptable.

3.7.2.2 Natural Frequencies and Response Loads

Tables 3.7.2-1 through 3.7.2-4 of early SSAR revisions provided a summary of the modal properties of the stick models representing the coupled shield and auxiliary buildings, the steel containment vessel, the containment internal structures and the overall stick model of the NI structures. As shown in these SSAR tables and discussed in Section 3.7.2.1 of this report, when a 33 Hz cut-off frequency is used, only 72 percent of the total structural mass participates in the horizontal response and 45 percent of the total mass participates in the vertical response for the coupled shield and auxiliary building, and only 51 percent of the total mass participates in the horizontal response and 47 percent of the total mass participates in the vertical response for overall NI structures.

In letters dated January 26, 1994 and March 16, 1994, the staff expressed its concerns regarding the total mass participation and missing mass correction in the seismic analyses performed for the NI structures. Westinghouse provided responses to these concerns in its April 14, 1994 and May 16, 1994, submittals. On the basis of the review of these submittals and discussions with Westinghouse in review meetings, the staff concluded that the analysis results (member forces and in-structure response spectra) are acceptable, if the response spectrum analysis method or frequency domain time history analysis method are used. This is because the missing mass would be automatically included when the response spectrum analysis method and frequency domain time history analysis method are applied. However, if the modal time history analysis method is used for seismic analyses of the NI structures, Westinghouse should justify not including the high frequency modes (or missing mass) in the analyses. The staff identified this concern as Open Item 3.7.2.2-1.

During meetings on February 28 through March 2, 1995 and June 12 through 16, 1995, Westinghouse presented a comparison of seismic responses calculated, using modal time history analysis method, by setting the cut-off frequency at 34 Hz (rigid frequency defined in RG 1.60) and 64 Hz, and showed that the difference between the two sets of results is insignificant. On this basis, Open Item 3.7.2.2-1 is considered closed.

3.7.2.3 Procedure Used for Modeling

Revision 22 of SSAR Figures 3.7.1-16 and 3.7.2-12 (Sheets 1 through 12) show the general arrangement of structural elements and key dimensions (overall dimension of NI foundation mat, elevations, distance between column lines, location of the containment vessel and reactor vessel centers, and thickness of walls and floor slabs). Westinghouse, using the information in these figures, developed the analytical models (both 3D finite element model and 3D lumped mass stick model) for the seismic analysis of the NI structures. The staff's review of these figures found that the general arrangement of structural elements and key dimensions provided in these SSAR figures are sufficient for the development of analytical models and, therefore, are acceptable. Furthermore, any proposed change to SSAR Figures 3.7.1-16 and 3.7.2-12 (Sheets 1 through 12) will require NRC approval prior to implementation of the change.

The procedure used for developing analytical models for the seismic analysis of the NI structures is discussed in Section 3.7.2.3 of the SSAR and is summarized in this section.

On the basis of the general arrangement drawings, three explicit 3D finite element models were developed for the NI structures. These three finite element models were developed to represent the coupled shield/auxiliary building, the containment internal structures, and the steel containment vessel, respectively. These models were used for determining stiffness properties of the equivalent lumped-mass stick seismic model and for calculating detailed moment and force distribution in the individual structural components.

From these three finite element models, an equivalent lumped-mass stick model was developed for each of the three SC-I buildings (i.e., coupled shield and auxiliary buildings, steel containment vessel and containment internal structures) on the basis of the rotational and translational stiffness of the explicit 3D models. The stiffness values of the stick models were determined by applying unit static forces or moments in the segment of the explicit 3D models corresponding to the element of the lumped-mass stick model.

The lumped-mass stick model for each of the SC-I building structures (coupled shield and auxiliary buildings, containment steel vessel and containment internal structures) were combined, using rigid links and beams, with the NI foundation mat and soil foundation to form the soil-structure system model for the seismic analyses of the NI structures founded on soil foundations.

The staff, using the guideline prescribed in SRP Section 3.7.2.3, reviewed the methods and procedures used by Westinghouse for modeling the NI structures. On the basis of its review of early SSAR revisions, Westinghouse's submittal related to structural modeling (letters dated October 1, 1992, January 26, 1994, and March 16, 1994) and Westinghouse's design calculations, the staff raised a number of technical concerns. These concerns and their resolutions are summarized below:

- In the SSAR, Westinghouse did not provide a list of dynamic properties (masses and fundamental frequencies) of major subsystems and major equipment such as steam generators, reactor vessel, etc. from which the conformance to the decoupling criteria guided in Section 3.7.2.II.3.b of the SRP can be evaluated. The staff identified this concern as Open Item 3.7.2.3-1.

  In Revision 7 of Section 3.7.2.2, Table 3.7.2-3 and Figure 3.7.2-7 of the SSAR, Westinghouse provided the dynamic properties and model for the major subsystems and major equipment. The staff review of this SSAR revision found that the development procedures and results of dynamic properties of major items and the application of decoupling criteria meet the guideline prescribed in SRP Section 3.7.2.3. On this basis, Open Item 3.7.2.3-1 is considered closed.

- The staff's review of early SSAR revisions found that the procedure for using 3D finite element structural models to determine stiffness properties of the equivalent lumped-mass stick seismic model of the NI structures meets the guideline of SRP Section 3.7.2.3 and, therefore, is acceptable. However, the staff did not concur with the use of an equivalent static analysis method and a simplified model for calculating detailed moment and force distribution in the individual structural components, because the NI structures are very complicated and it is not possible to be realistically represented by a simple model.

  In Revision 7 of SSAR Section 3.7.2 and Table 3.7.2-16, Westinghouse stated that for calculating detailed structural member forces (shear forces, bending moments, axial forces, etc.), the response spectrum analyses with the computer code BSAP were performed using the 3D fixed-base finite element NI structural model (i.e., Model "A" as discussed in Section 3.7.2.1 above). The use of the response spectrum analysis method for performing seismic analyses of a complicated structure meets the guideline prescribed in SRP Section 3.7.2.2 and, therefore, is acceptable to the staff.

- The seismic models used in the analyses had not been updated and modified to reconcile with the latest version of the general arrangement and design drawings. The concern of inconsistency between the original model and the later version of drawings was identified as Open Item 3.7.2.3-2.

  At the review meeting on February 28 through March 2, 1995, Westinghouse presented its revised 3D fixed-based finite element model and 3D lumped-mass stick model of the NI structures for the staff review. These revised models (1) included changes to the general arrangement and design drawings, (2) included those structural elements that had not been considered in the original model, and (3) incorporated the staff's concerns identified during the meeting on October 31 through November 3, 1994. The staff reviewed the revised modeling procedure together with calculations and found that the modeling procedure meets the guideline prescribed in SRP Section 3.7.2. On this basis, Open Item 3.7.2.3-2 is considered closed.

- In the submittal dated January 14, 1993, Westinghouse indicated that the second vertical mode for the explicit 3D steel containment vessel has a frequency of 23.59 Hz, but that for the equivalent seismic lumped-mass stick model is 30.06 Hz. Westinghouse should demonstrate the adequacy of the equivalent seismic lumped-mass model whose second mode frequency is so much higher than that of the detailed model. The staff identified this concern as Open Item 3.7.2.3-3.

  A similar technical concern regarding the modeling of the containment vessel dome and its resolution are discussed under Open Item 3.7.2.3-7 of this report. On this basis, Open Item 3.7.2.3-3 is considered closed.

- Two types of rigid links or rigid elements have been used in the SSI seismic model to simulate connections between the shield building and auxiliary building at various elevations. One type of rigid link simulates the offset between the center of rigidity and the center of mass, and the other simulates the following elements:

- the in-plane rigidity of floors linking the vertical sticks to the sub-grade peripheral walls

- the in-plane rigidity of the foundation mat

- the stiffening effects provided by the internal walls to the foundation mat

The seismic response of the NI structures can potentially be influenced by the arrangement of these rigid links and elements. The modeling arrangement should be detailed with a sketch (or sketches) in the SSAR. Also, the descriptive explanation of the modeling of the NI structures provided by Westinghouse in the submittal dated April 14, 1994 should be incorporated into the SSAR. In addition, resulting member forces in these rigid links should be compared with the capacity and stiffness of actual members connecting these two buildings. If the actual capacity and stiffness are significantly smaller than the calculated values, the seismic response calculations may be erroneous and the connecting members are likely to be overstressed during an SSE. This concern was identified as Open Item 3.7.2.3-4.

During the design calculation review meetings, the staff conducted a review of modeling procedures for the NI structures including sensitivity study results generated with various rigid element properties and found that, within the range of properties used in the analysis, the results are not sensitive to the rigid element properties. The staff also found that these rigid elements are used to simulate the floor slabs in connecting the embedded exterior walls and lumped masses. The technique of using rigid elements to simulate the floors and slabs is consistent with the common industry practice and has been previously accepted by the staff during its review of early nuclear power plants for which the in-plane behavior of the floor slabs can be assumed as rigid when a lumped-mass stick dynamic model is developed. On this basis, the staff concludes that the use of rigid elements for the dynamic model by Westinghouse is reasonable and acceptable. This closes Open Item 3.7.2.3-4.

- The seismic model for the NI structures was developed on the basis of uncracked concrete section properties. This is acceptable provided that the major concrete structural elements (e.g. shear walls) are not significantly cracked. In the SSAR, Westinghouse did not demonstrate, by comparing uncracked and cracked section properties or by comparing floor spectra developed on the basis of cracked and uncracked section properties, that the effect of concrete cracking to the seismic response is negligible. The staff identified this concern as Open Item 3.7.2.3-5.

During review meetings, Westinghouse presented a comparison of floor response spectra on the basis of uncracked concrete structural elements of the auxiliary and shield buildings with those on the basis of cracked concrete structural elements, and demonstrated that the effect of concrete cracking to the seismic responses of the NI structures is relatively small. The basis of Westinghouse's conclusion, from its comparison of the two sets of floor response spectra, is that there is no significant cracking in reinforced concrete structural elements expected under the combined load conditions. In addition, Westinghouse stated that the uncertainties attributable to possible concrete cracking can be covered by the ±15 percent peak broadening of the floor response spectra. Broadening the peak of floor response spectra by ±15 percent to cover the uncertainty attributable to structural modeling for the case of reinforced concrete structures with minor cracks meets the guideline prescribed in RG 1.122. On this basis, the staff concludes that Open Item 3.7.2.3-5 is closed.

- In its letter dated January 26, 1994, the staff raised a concern regarding the possibility of the out-of-phase interaction between the shield building, the steel containment vessel, and the containment air baffle. As a result of the staff's review of the submittal dated April 14, 1994, and the discussion during the review meetings, Westinghouse agreed to provide, in the SSAR (1) figures showing the rigid link connectivity of the stick model to the foundation mat and the wall elements below grade, and (2) the criteria used to establish relative displacements between the shield building and the steel containment vessel for the design of the air baffle. The staff identified this concern as Open Item 3.7.2.3-6.

  In response to this open item, Westinghouse showed the rigid link connectivity of the stick model to the foundation mat and wall elements below grade in Revision 7 of SSAR Figure 3.7.2-13 and provide a vertical sliding plate, as described in Revision 7 of SSAR Section 3.8.4.1.3, to accommodate the differential movement between the containment vessel and the shield building. However, in SSAR Section 3.8.4.1.3 and Figure 3.8.4-1, Westinghouse did not show the size of the sliding plate to ensure that the displacement because of seismic forces will not affect the integrity of the air baffle. Therefore, Westinghouse's SSAR commitment did not satisfy this staff concern regarding the size of the sliding plate.

  In addressing this issue, Westinghouse provided the maximum displacement (relative to the shield building) of the containment vessel because of the SSE, design pressure and design temperature loads in the letter dated January 7, 1998 (NSD-NRC-98-5512). Also, in Revision 20 of SSAR Section 3.8.4.1.3, Westinghouse specified the limits of vertical and horizontal movements and the size of the sliding plate which can accommodate the maximum displacements of the containment vessel. Westinghouse's response satisfied the staff's concern. On this basis, the staff concludes that Open Item 3.7.2.3-6 is closed.

- The staff raised two concerns in the letters dated October 1, 1992 and January 26, 1994, (1) how the containment shell lumped-mass stick model was constructed from the axisymmetric finite element containment shell model, and (2) how the eccentric masses, such as the polar crane system, equipment hatches and personnel air-locks were included in the 3D lumped-mass model.

  On the basis of the staff's review of submittals dated the November 30, 1992 and March 24, 1994, and early revisions of the SSAR, and the discussions during the review meetings, the staff found that the modeling procedure used for developing the 3D lumped-mass stick model of the containment vessel meet the guideline prescribed in SRP Section 3.7.2.3 and, therefore, are acceptable. However, Westinghouse should document the modeling procedure used for the containment vessel in the SSAR. In addition, the staff identified, from its review of early revisions of the SSAR, that the second vertical modal frequency of the containment vessel calculated from the lumped-mass stick model deviates significantly from that obtained on the basis of the axisymmetric finite element shell model. As a result of meeting discussions, Westinghouse agreed, regarding the development of the stick model for the containment vessel, to document its responses in the SSAR and justify the deviation of the second vertical mode of vibration between the two models. The staff identified this concern as Open Item 3.7.2.3-7.

In response to this open item and the staff's concerns raised during the review meeting on June 17 through 21, 1996, Westinghouse made the following changes in Revision 9 of SSAR Section 3.7.2.3.2:

- Provide procedures for developing the 3D lumped-mass stick model from the axisymmetric finite element containment shell model.

- Explain that the shell of revolution vertical model (n=0 harmonic) has a series of local shell modes for the containment top head above Elevation 73.15 m (240 ft) between the frequency range of 23 to 30 Hz. These modes are predominant in a direction normal to the shell surface and cannot be represented by a stick model. Also, these local modes have a small contribution to the total response to vertical earthquake motion when they are at a high frequency where seismic excitation is small.

- State that the only SC-I components attached to this portion of the top head are the water distribution weirs of the passive containment cooling system, and these weirs are designed so that their fundamental frequencies are outside the 23 to 30 Hz range.

The staff's review of this SSAR revision found that the model procedures used meet the guideline of SRP Section 3.7.2.3. The staff also agreed with the basis provided by Westinghouse for justifying that the difference between the two second vertical modal frequencies of the containment top head calculated from the stick model and the finite element shell model is insignificant to the total response of the containment vessel and will not affect the safety function of SC-I items. On this basis, the staff concludes that Open Item 3.7.2.3-7 is closed.

With regard to the concern of the effects of eccentricities attributable to major components and equipment supported on the containment vessel, the staff requested Westinghouse to verify how significant these eccentricities would be to the seismic responses of the containment shell. The issue of including the eccentricities attributable to major components and equipment in the lumped-mass stick model was identified as Open Item 3.7.2.3-8.

During the meeting on February 28 through March 2, 1995, the staff reviewed Westinghouse's design calculations and found that Westinghouse included the eccentricities attributable to major components and equipment in its revised 3D lumped-mass stick model for the final seismic analyses. The staff also found that the modeling procedure for including the eccentricities attributable to major components and equipment meets the guideline prescribed in SRP Section 3.7.2.3. In addition, the staff's review of Revision 9 of SSAR Section 3.7.2.3.2 found that Westinghouse provided a description to show how the polar crane unit was modeled in the 3D lumped-mass stick model. On the basis of its finding from the design calculation review that the modeling procedure used by Westinghouse for modeling the major components and equipment meets the guideline of SRP Section 3.7.2, the staff concludes that the dynamic model developed for the containment vessel is acceptable. The staff conclusion was also confirmed by a comparison of Westinghouse's analysis results with those obtained from the staff's confirmatory analysis (see Enclosure 1 of letter to Westinghouse dated April 9, 1998). On this basis, Open Item 3.7.2.3-8 is closed.

- In the figures contained in Sections 1.2 and 3.8 of early SSAR revisions, Westinghouse did not provide any key dimensions such as the size of the foundation mat (thickness and overall dimensions), the radius of the shield building, the geometry of the shield building roof, the geometry of the containment internal structures (structural modules), and the thickness of the walls (the periphery walls, shield building wall and major structural walls). Westinghouse contended that to provide exact dimensions in the SSAR would sacrifice the flexibility in the final design of safety-related plant structures, systems and components during the plant construction. The staff was concerned that the use of approximate dimensions for developing structural models and a soil-foundation model of the NI structures and foundation mat was not acceptable for generating seismic responses (structural member forces and in-structure response spectra) of the NI structures. Westinghouse was requested to finalize all the dimensions of structural elements and document the dimensions in the SSAR. The staff identified this concern as Open Item 3.7.2.3-9.

  As discussed in Section 3.7.1 of this report, Open Item 3.7.1-5 raised a similar concern regarding the key dimensions of the NI structures. On the basis of the resolution for Open Item 3.7.1-5, Open Item 3.7.2.3-9 is closed.

In addition to the open items discussed above, the staff raised two issues related to seismic modeling of the NI structures during review meetings. These two issues and their resolution are discussed below:

(1) During the meeting on June 12 through 16, 1995, the staff was concerned with the adequacy of the overall 3D lumped-mass stick model. It is the staff's understanding that Westinghouse used a multi-stick (containment vessel, containment internal structures and shield/auxiliary building) lumped-mass model for the seismic analysis of NI structures. This lumped-mass model was developed on the basis of a 3D finite element model of each building. However, Westinghouse chose to use an equivalent lumped-mass stick model to represent the shield building roof structures. This lumped-mass stick model was developed by Westinghouse's consultant ANSALDO. From its review of the seismic analysis and design calculations of the NI structures, the staff found that the seismic member forces of the shield building roof structures calculated by combining the stick model of the roof structures with the finite element model of the other structures are significantly different from those calculated by a complete stick model. It is the common understanding in the engineering field that a finite element model can simulate the actual behavior of a structure much more closely

than a lumped-mass model. Westinghouse should provide basis for the multi-stick model which was used for generating seismic responses (structural member forces and floor response spectra) for the design of safety-related structures, systems, and components.

In its letter dated September 25, 1996, Westinghouse provided, using the response spectrum analysis method and updated 3D lumped-mass stick model, a comparison of the hard rock site seismic member forces (shear forces and bending moments) of the shield building roof structures calculated on the basis of the 3D lumped-mass stick model with those on the basis of the 3D finite element model. The comparison showed that the maximum differences between the results obtained from the two models are 7 percent in axial forces, 5 percent in shear forces (N-S shear), 14 percent in bending moments (N-S bending), and 37 percent in torsional moments. This comparison also showed that the results from the 3D finite element model are consistently higher than those from the lumped-mass stick model.

On the basis of the engineering judgement, the staff agreed with Westinghouse that the differences in axial forces, shear forces, and bending moments are within engineering approximations. With regard to the significant difference in the torsional moments, the staff, from its detailed review of SSAR tables and confirmatory analysis results (see Enclosure 1 of letter to Westinghouse dated April 9, 1998), found that the contribution of the calculated torsional response to the total seismic response of the NI structures is small. The basis of the staff's conclusion is as follows:

(a) According to Revision 7 of SSAR Table 3.7.2-5 which documented the time history analysis results for the coupled shield and auxiliary buildings (all Elevations below 73.46 m [241 ft]) founded on the hard rock site, the maximum acceleration at the edge of floors is only slightly higher than those at the center of mass. This implies that the torsional response (torsional moment) has insignificant effect to the floor accelerations at the edge of the building below Elevation 73.46 m (241 ft) (air inlet of the shield building structures).

(b) From its confirmatory calculation, the staff found that the torsional induced shear force is less than or about 10 percent of the total shear force of the corresponding structural member. This finding implies that the torsional response has only minor effect on the total seismic response of the coupled shield and auxiliary buildings above Elevation 73.46 m (241 ft).

On the basis of the discussion above, the 3D lumped-mass stick model for the coupled auxiliary and shield buildings developed from the 3D finite element model is acceptable.

(2) In SECY-96-128, the staff recommended that the Commission approves its position that the site be capable of sustaining all design-basis events with onsite equipment and supplies for the long term. After 7 days, replenishment of consumables such as diesel fuel oil from offsite suppliers can be credited. In the SRM dated January 15, 1997, the Commission approved this staff position. As a result, in order to provide additional

onsite cooling water in the PCCWS tank, Westinghouse made structural design changes to the PCCWS tank. These design changes are as follows:

(a) increasing the tank water level from Elevation 90.83 m (298 ft) to 91.90 m (301.5 ft)

(b) raising the top of the tank by 0.3 m (1.0 ft)

(c) reducing the thickness of the inner tank wall from 60.96 cm (24 inches) to 45.72 cm (18 inches)

(d) decreasing the thickness of the tank roof from 60.96 cm (24 inches) to 38.10 cm (15 inches)

(e) placing the PCCWS tank floor liner directly on the structural concrete and deleting the 10 cm (4 inches) of grout.

Because of these design changes, Westinghouse revised the 3D lumped-mass seismic stick model of the shield building roof structures to include these structural design changes of the PCCWS tank. Also, in resolving Open Item 3.8.4.3-1 regarding the inclusion of live load in the seismic model, Westinghouse, as stated in the SSAR, included 25 percent of the total live load and 75 percent of snow load in the revised seismic model.

During the meeting on August 11 through 15, 1997, the staff reviewed Westinghouse's Calculation Nos. 1070-S3R-010, Revision 0 and 1000-S2R-054, Revision 0, and found that Westinghouse has properly followed the guideline prescribed in SRP Section 3.7.2.3 to incorporate the design changes attributable to post-72-hour actions requirements, and live and snow loads in the 3D lumped-mass stick model. Also, the staff's review of Revision 17 of SSAR Section 3.7.2.3.1 found that the approach used by Westinghouse to incorporate design changes to the shield building structures and live and snow loads meets the guideline prescribed in SRP Section 3.7.2.3. On the basis of the discussion above, the staff concludes that the revised seismic model used by Westinghouse for performing the updated SASSI analysis is reasonable and acceptable.

3.7.2.4 Soil-Structure Interaction

The early revisions of Section 3.7.2 of the SSAR and Appendix 2A to the SSAR stated that SSI analyses of the NI structures were performed to determine seismic design loads for SC-I building structures, and to develop in-structure response spectra to be used for the design of SC-I subsystems. Westinghouse used the SASSI computer code to perform SSI analyses. The selection of design site conditions for the 3D SSI analysis cases was on the basis of a series of two-dimensional (2D) SASSI parametric analyses in which the following four parameters were varied (1) shear wave velocity of soil and rock, (2) soil layering, (3) depth to base-rock, and (4) water table. Westinghouse's selection of these parameters for the 2D SASSI analyses was on the basis of a survey of subsurface soil profiles and a range of soil properties in twenty-two commercial nuclear power plants located in the United States. Seismic loads for the structural design and in-structure response spectra for the subsystem design were developed by enveloping the responses from two 3D SASSI SSI analyses and one 2D fixed-base BSAP analysis, with one exception. For the design of peripheral walls below the grade, Westinghouse used soil pressures calculated from the 2D SASSI analyses. The fixed-base analysis of the NI structures was performed to calculate the seismic response for the hard-rock foundation. Two soil site conditions (e.g., soft rock site and soft-to-medium stiff soil site) were selected by Westinghouse to cover the entire range of soil conditions for performing SSI analyses. The siting geometry and dynamic soil properties of these two site conditions are as follows:

- For the soft rock site, the depth of soil layer measured from ground surface to bedrock is 36.6 m (120 ft), and the shear wave velocity varies linearly from 731.5 m/sec (2400 ft/sec) to 853.4 m/sec (2800 ft/sec) with ground water at the grade.

- For the soft-to-medium stiff soil site, the depth to bedrock is 36.6 m (120 ft) and the shear wave velocity varies linearly from 304.8 m/sec (1000 ft/sec) to 518.2 m/sec (1700 ft/sec) with ground water at the grade.

Westinghouse used the SASSI computer code to perform the final 3D SSI analysis for each of these two site conditions.

The staff reviewed the early revisions of the SSAR and Westinghouse's submittals dated January 22, 1993, March 24, 1994, May 11, 1994, May 20, 1994, May 11, 1994, May 17, 1994, May 11, 1994, May 20, 1994, and May 17, 1994. In addition, in order to develop bases for its conclusions, the staff, using the lumped-mass stick models (2D and 3D) provided by Westinghouse, performed a set of SSI confirmatory analyses. The staff's review of early revisions of the SSAR and Westinghouse's submittals, and the comparison of results obtained from the confirmatory analyses with those from Westinghouse's analyses resulted in a number of open items. These open items and their resolution are summarized as follows:

- In its letter dated October 1, 1992, the staff raised a concern that the SRP guideline states that the spectral amplitude of the acceleration response spectra at the foundation level in the free field should not be less than 60 percent of the corresponding design response spectra at the finished grade in the free field. However, the spectral amplitudes of the acceleration response spectra (for the two soil site conditions at 12.2 m [40 ft] below grade in the free field) shown in Figures 2A-21 through 2A-24 of early revisions of the SSAR show that the spectral amplitudes at the foundation depth did not satisfy this guideline. Westinghouse, in its submittal dated May 20, 1994, stated that the AP600 design is on the basis of enveloping seismic responses of all soil and rock cases considered. For the rock profile, the reduction of motion with depth is insignificant. Thus, the enveloping ground motion at the foundation level in the free field for all soil and rock cases meets the guideline of Section 3.7.2 of the SRP with respect to the reduction of motion with depth. On the basis of the justification stated above and the analysis results shown in the SSAR, Westinghouse concluded that the results obtained from the fixed-base (hard rock site condition) envelop most of the results from the analysis of soil site conditions. The response provided by Westinghouse was not acceptable because of the following three reasons:

  (1) The guideline of Section 3.7.2 of the SRP states that when variation in soil properties are considered (best estimate soil shear modulus values, twice the

best estimate value, and half the best-estimate value) for the uncertainty of soil properties, the 60 percent limitation may be satisfied using an envelope of the three response spectra corresponding to the three soil properties. This guideline is applicable only for the consideration of soil property variations of any specific site condition and can not be applied for different site conditions. Therefore, Westinghouse's justification of the enveloping ground motion at foundation level in the free field for all soil and rock did not meet the SRP guideline.

(2) For the AP600 standard design, the safety-related structures, systems and components should be designed for the envelope of seismic responses (forces and in-structure response spectra) obtained from all of the individual soil and rock site conditions. For each of these sites, the ground motion should satisfy the 60 percent limitation by itself. It is not the intent of the SRP guideline that the structures, systems, and components should be designed to the motion for which the envelope of response spectra corresponding to each of site conditions at foundation level satisfies the 60 percent limitation.

3) From the results of its confirmatory SSI analyses, the staff found that the in-structure response spectra obtained on the basis of one of the design soil site conditions, and the ground motion time history for which the 60 percent limitation is satisfied at foundation, exceed Westinghouse's envelope of the in-structure response spectra on the basis of the three design site conditions. Westinghouse should perform analyses and calculate seismic responses, and develop the response envelope for the design, using the ground motion time history which satisfies the 60 percent limitation of the surface ground motion for all site conditions or time histories, each of which satisfies the 60-percent limitation for the design site conditions.

The concern of satisfying the requirements that the spectral amplitude of the acceleration response spectra at the foundation level in the free field not be less then 60 percent of the corresponding design response spectra at the finished grade in the free field was Open Item 3.7.2.4-1.

During the February 28 through March 2, 1995, review meeting, Westinghouse demonstrated that the envelope of calculated acceleration response spectra corresponding to the three design site conditions at the foundation level in the free field envelops 60 percent of the design ground response spectrum over the entire frequency range of interest. On the basis of its demonstration, Westinghouse contended that the spectral amplitude of the acceleration response spectra at the foundation level in the free field is not less then 60 percent of the design ground motion response spectra at the finished grade in the free field. Westinghouse also concluded that ground motion time histories meet the 60-percent guideline of SRP Section 3.7.2.II.4. In addition, in its letter dated September 25, 1996, Westinghouse, for each of the four design soil profiles (in the early SSAR revisions, Westinghouse committed to three site conditions for the AP600 standard design. To resolve the staff's concern, Westinghouse added the fourth site condition for the standard design. Detailed review of this issue is discussed under Open Item 3.7.2.4-12 below), provided the following details:

(1) the acceleration response spectra (2 percent damping) from the free-field east-west analyses at the depth of 12.19 m (40 ft) corresponding to the foundation mat elevation

(2) the acceleration response spectra (2 percent damping) corresponding to the foundation mat of the NI from the east-west SSI analysis of the NI

(3) a table of frequencies showing the soil column (upper 12.19 m [40 ft]), the SSI and the fixed base frequencies of the NI

In this submittal, Westinghouse stated that because the fixed-base structural frequency and soil column frequencies are consistently higher than the SSI frequencies, the staff's concern of satisfying the SRP guideline that the spectral amplitude of the acceleration response spectra at the foundation level in the free field not be less then 60 percent of the corresponding design response spectra at the finished grade in the free field does not apply to the AP600 design. From the meeting discussion and its review of this document, the staff found that Westinghouse's justification meets the guideline of SRP Section 3.7.2. On the basis discussion above, the staff concludes that the demonstration provided by Westinghouse meets the SRP guideline and Open Item 3.7.2.4-1 is closed.

- In its letters dated October 1, 1992 and January 26, 1994, the staff raised two concerns (1) the statement made in Section 3.7.2.4 of early SSAR revisions that the selected soil conditions envelop the potential variation of soil properties and, therefore, the guidelines of Section 3.7.2 of the SRP for the variation of soil properties were not considered in the design, and (2) the statement made in Section 2A.4 of early SSAR revisions that because the generic soil profiles considered include a wide range of shear wave velocities, the customary plus 100 percent and minus 50 percent variation in low strain shear modulus (G_{max}) for each soil profile was not applied in the analysis. Also, in Table 2.0-1 of early SSAR revisions, Westinghouse specified that the minimum shear wave velocity of soil foundation is 304.8 m/sec (1000 ft/sec). It is implied that Westinghouse intended to design the plant for soil sites with a best estimate soil shear wave velocity as low as 304.8 m/sec (1000 ft/sec). According to the guideline stated in Section 3.7.2.II of the SRP, Westinghouse should consider a lower bound soil shear wave velocity of 215.5 m/sec (707 ft/sec) (the shear wave velocity corresponding to half the best estimate shear moduli). During the early review meetings, Westinghouse agreed to include the true lower bound shear wave velocity in its SSI analyses. The staff's above-stated concern was Open Item 3.7.2.4-2.

  In response to the staff's concern regarding how the structural responses such as member forces and floor response spectra were generated for different soil site conditions, Westinghouse, in its submittal dated March 24, 1994 and August 26, 1994, (1) provided explanation of how the structural responses were calculated for each of the soil site conditions, and (2) explained how the SRP Subsection 3.7.2.II.4 guideline are met for various soil properties based on the selected soil site conditions. Westinghouse also revised SSAR Section 3.7.2.4 and Table 3.7.2-16 (Revision 9) to document the response of the August 26, 1994 submittal.

The staff's review of the SSAR, Revision 9 found that the approach for calculating the seismic structural member forces for different soil site conditions meets the guidelines of SRP Section 3.7.2.4. Also, during the June 12 through 17, 1996, meeting, Westinghouse presented the calculation to demonstrate that the lower bound of the soft-to-medium soil with a shear wave velocity of 215.5 m/sec (707 ft/sec) has insignificant effect to the overall structural responses. On the basis discussed above, the staff concludes that Westinghouse's justification is acceptable and Open Item 3.7.2.4-2 is closed.

- The guideline in Section 3.7.1.I.1 of the SRP states that for sites composed of one or more thin soil layers overlying a competent material such as a shallow soil site, the input ground motion should be specified on an outcrop or a hypothetical outcrop at a location on the top of the competent material such as rock. In its letter dated January 26, 1994, the staff raised concerns that Westinghouse did not specify the location of the input ground motion to be applied for each of the design site conditions and requested Westinghouse to demonstrate that the design seismic loads of the NI structures and related subsystems on the basis of the three design site conditions can envelop the design loads calculated from the analysis for a shallow soil site. On the basis of its review of Westinghouse's March 24, 1994, submittal and its confirmatory analysis (see Enclosure 1 of letter to Westinghouse dated April 9, 1998) in which the in-structure response spectra from the shallow soil site exceed the envelope of in-structure response spectra from the three design site conditions, the staff concludes that Westinghouse's seismic design on the basis of the four design site conditions will not be adequate for shallow soil sites. Therefore, the shallow soil sites should be excluded for the AP600 design. This was Open Item 3.7.2.4-3.

The concern of this open item is similar to that of Open Item 3.7.1-1. As described in Section 2.5.2 of this FSER, Westinghouse is including an action item in the SSAR to require that COL applicants demonstrate that the site-specific earthquake ground motion at the finished grade level in the free field are enveloped by the ground motions used as input for the design certification (SSAR Figures 3.7.1 and 3.7.1-2) and COL applicants must assure that the site-specific response spectra at the foundation level (12.19 m [40 ft] below the plant finished grade level) in the field are less than those in SSAR Figures 3.7.1-18 and 3.7.1-19. On the basis of the resolution of Open Item 3.7.1-1 and the commitment in SSAR Section 2.5.2, Open Item 3.7.2.4-3 is closed.

- During the January 20 through 21, 1994, meeting with Westinghouse, the staff found that Westinghouse considered the effect of ground water in the SASSI analysis by maintaining the shear wave velocities determined from SHAKE analysis while changing the P-wave velocity to 1524 m/sec (5000 ft/sec). However, the unit weight of the soil used in the SASSI analysis was on the basis of a dry condition, and was not modified to account for soil saturation effects. Westinghouse should provide a basis for not considering the dry soil condition in the SSI analyses. This was Open Item 3.7.2.4-4.

At the February 28 through March 2, 1995, review meeting, Westinghouse presented its calculations and demonstrated that the effect of unit weight of soil above water table to the soil shear modulus is negligible. On this basis, Open Item 3.7.2.4-4 is closed.

- From reviewing the early revisions of the SSAR, the staff found that the strain-dependent shear modulus and hysteretic damping data used in the SSI analysis were on the basis of 1970 data developed by Seed and Idriss for sandy soils. Since then, newer soil degradation models have been developed and published. A comparison of shear strain degradation curves presented in the SSAR (Seed & Idriss 1970 curves) with the more recent industry results showed that the Seed & Idriss 1970 curves always overestimate the shear strain degradation. According to the independent analyses, using various soil degradation models performed by the staff (NUREG/CR-5956), the staff found that the analysis results on the basis of more recent publications such as the soil strain degradation model developed by Idriss in 1990 are much higher than those on the basis of the 1970 Seed & Idriss curves which were accepted by the staff during the review of conventional nuclear power plants. In the May 11, 1994, submittal and the discussions during the review meetings, Westinghouse agreed to provide additional information for the staff review. This was Open Item 3.7.2.4-5.

  From its review of Revision 7 of SSAR Section 3.7.2, the staff found that Westinghouse has replaced the 1970 Seed-Idriss soil strain degradation model by the soil strain degradation model developed by Idriss in 1990. Also, in the June 17 through 21, 1996, review meeting, Westinghouse presented the updated SSI analyses to show that, for calculating the final seismic responses, the 1970 Seed-Idriss soil strain degradation model was replaced by the Idriss 1990 model. This resolved the staff's concern and Open Item 3.7.2.4-5 is closed.

- AP600 SSI analyses are on the basis of a soil degradation model for sandy soil. The effects of using soil degradation models appropriate for soil types other than sand, such as clay, silt, gravel, and various combinations, on the SSI responses have not been addressed in the SSAR. This was Open Item 3.7.2.4-6.

  At the June 17 through 21, 1996, meeting, Westinghouse presented its calculation results and showed that the use of 1990 Idriss sandy soil strain degradation model will yield more conservative results than other types of soil. On this basis, Open Item 3.7.2.4-6 is closed.

- As stated in early revisions of the SSAR, the peripheral walls of the NI structures below finished grade are designed on the basis of soil pressure obtained from 2D SSI analyses. The model used for the 2D SSI analyses of the NI structures did not include the adjacent SC-II and non-seismic structures. In the January 22, 1993 and May 20, 1994 submittal, Westinghouse stated that these non-seismic Category I structures are relatively lighter than SC-I structures and the effect of structure-to-structure interaction on the NI structures is negligible. The staff's concern is that the localized through-the-soil SSI effect of non-seismic Category I structures on the design of SC-I peripheral walls could be significant and this effect was not included in the design. In addition, the potential for pounding between structures should also be reasonably evaluated. These two issues were Open Item 3.7.2.4-7.

  In response to this open item, Westinghouse, in its submittal dated September 25, 1996, provided the analysis results for review. This analysis is on the basis of a refined 2D SSI model of the NI and annex building, and was performed in the East-West direction for the soft rock site condition for two cases (1) the NI alone and (2) the NI with the annex building. The results in terms of maximum lateral soil pressure on the east wall (annex building side) and the west wall were calculated and compared. The comparison shows that the difference between the soil pressure for the case of the NI alone and the NI with the annex building is insignificant. From its review of Westinghouse's submittal, the staff concurs with Westinghouse's justification that the localized through-the-soil SSI effect of non-seismic Category I structures on the design of SC-I peripheral walls is insignificant. On this basis, the concern related to the through-the-soil SSI effect is resolved.

For the concern of potential pounding between structures, Westinghouse, during the August 11 through 15, 1997, meeting, presented a calculation for review and stated that the impact energy from the potential pounding between the radwaste building and the NI structures during an SSE is less than the elastic strain energy of the NI structures. This calculation was performed on the basis of the energy balance method and the assumption that pounding would occur in the event of an SSE. Westinghouse also documented the analysis method and results in Revision 13 of SSAR Section 3.7.2.8. From this demonstration and its engineering judgement, the staff technically agreed with Westinghouse's conclusion that the impact from the radwaste building will not impair the structural integrity of the NI structures. The concern of potential impact between the NI and the turbine and annex buildings is discussed in Section 3.7.2.8 of this report. On this basis, Open Item 3.7.2.4-7 is closed.

- Westinghouse used a linear variation in the soil profile through the depth of the soil layers. From the review of existing literature, it is evident that the assumption of linear variation of soil profile is unrealistic for typical sandy soils. Westinghouse should use a more realistic variation with depth of soil profile, such as a parabolic distribution, for the AP600 SSI analyses. This was Open Item 3.7.2.4-8.

  The staff review of Revision 9 of SSAR Section 3.7.2.4 and analyses presented in the June 17 through 21, 1996, meeting found that Westinghouse used the parabolic variation of soil profiles for the soft-to-medium and the upper bound soft-to-medium soil site conditions in the SSI analyses. On the basis of this finding, the staff concludes that Open Item 3.7.2.4-8 is closed.

- Westinghouse did not show that the Poisson's ratio values assumed for soils above the water table are consistent with the values for silty sands with densities high enough for a shear wave velocity of 304.8 m/sec (1000 ft/sec). In addition, Westinghouse erroneously indicated in the third paragraph of Section 3.7.2.4 of the SSAR, that the SHAKE computer code was used to compute a strain compatible Poisson's ratio and other parameters. These two issues regarding Poisson's ratio of soil foundation were Open Item 3.7.2.4-9.

  At the February 28 through March 2, 1995, meeting, Westinghouse presented its calculation and demonstrated that the calculated responses (floor response spectra) are insensitive to the Poisson's ratio. Westinghouse also made a correction regarding the use of the SHAKE computer code for computing strain compatible Poisson's ratio in Revision 9 of the SSAR. On this basis, Open Item 3.7.2.4-9 is closed.

- In its letter dated March 16, 1994, the staff raised a concern that the SASSI computer code cannot be relied upon to produce accurate member forces and bending moments because the calculated results may be sensitive to the assumed "rigid beam" stiffness. During the review meeting, Westinghouse agreed to provide justification to show that the results (member forces and moments) from the SASSI analysis are adequate. This was Open Item 3.7.2.4-10.

  At the February 28 through March 2, 1995, meeting, Westinghouse presented a comparison of results from two SASSI analyses. The first analysis assumed that the rigid beam stiffness is 1,000 times the stiffness of the stiffest structural member of the model. In the second analysis, the same rigid beam stiffness is assumed to be 10,000 times the stiffness of the stiffest structural member. The comparison showed that the difference between the results (moments and shear forces) from these two analyses is negligible. On the basis of this finding, the staff agreed with Westinghouse's conclusion that the seismic response of the NI structures is not sensitive to the rigid beam stiffness. Therefore, Open Item 3.7.2.4-10 is closed.

- The staff requested, in its March 16, 1994, letter, that Westinghouse should justify the validity of performing a fixed-base seismic analysis for the site conditions with shear wave velocity equal to or greater than 2438.4 m/sec (8000 ft/sec). In the May 11, 1994 submittal and during review meeting discussions, Westinghouse agreed to provide its justification for the staff review. This was Open Item 3.7.2.4-11.

  At the February 28 through March 2, 1995, meeting, Westinghouse presented a comparison of results from the analysis using a fixed-base NI structural model and from the SSI analysis of hard rock sites, and showed that the difference between the two sets of results is negligible. On this basis, the staff concurred with Westinghouse's conclusion that the use of the fixed-base model to calculate seismic responses of the NI structures for hard rock sites will provide reasonable results. This conclusion is also confirmed by the results of staff's confirmatory analyses. Therefore, Open Item 3.7.2.4-11 is closed.

- In the letter dated March 16, 1994, the staff questioned whether the envelope of seismic responses obtained on the basis of the three selected site conditions can truly envelop the seismic responses calculated for other site conditions, such as a site with shear wave velocity equal to 731.5 m/sec (2400 ft/sec) or 1066.8 m/sec (3500 ft/sec). During the February 28 through March 2, 1995 review meeting, Westinghouse restated its basis documented in Appendix 2A to the SSAR that the selected three site conditions will cover a wide range of potential site conditions in the states. The issue of using additional site condition in the AP600 standard design to cover a full range of site conditions was Open Item 3.7.2.4-12.

  In response to this open item, Westinghouse, as described in Revision 9 of SSAR Section 3.7.1.4, added the fourth design site condition (the upper bound of the soft-to-medium soil site) to the AP600 standard design. The dynamic characteristics of these four site conditions are as follows:

  (1) for hard rock site, an upper bound case for firm sites using fixed base seismic analysis

(2) for the soft rock site, a shear wave velocity of 731.52 meters per second (2400 feet per second) at the ground surface, increasing linearly to 975.36 meters per second (3200 feet per second) at a depth of 73.15 meters (240 feet), base rock at the depth of 36.58 meters (120 feet)

(3) for the soft-to-medium soil site, a shear wave velocity of 304.8 meters per second (1000 feet per second) at ground surface, increasing parabolically to 731.52 meters per second (2400 feet per second) at 73.15 meters (240 feet), base rock at the depth of 36.58 meters (120 feet)

(4) for the upper bound soft-to-medium soil site, a shear wave velocity of 430.99 meters per second (1414 feet per second) at ground surface, increasing parabolically to 1034.49 meters per second (3394 feet per second) at 73.15 meters (240 feet), base rock at the depth of 36.58 meters (120 feet)

During the June 17 through 21, 1996, meeting, Westinghouse presented the floor response spectrum envelopes on the basis of four design site conditions for review, and demonstrated that these floor response spectrum envelopes envelop those corresponding to other site conditions. As a result of its review, the staff concurred with Westinghouse's justification for site with soil shear wave velocity in the range between 304.8 m/sec (1000 ft/sec) and 2438.4 m/sec (8000 ft/sec). In addition, the Westinghouse's conclusions are confirmed by the staff's confirmatory analysis results (see Enclosure 1 of letter to Westinghouse dated April 9, 1998) on the basis of a site condition with a shear wave velocity of the supporting material equal to 1,066.8 m/sec (3,500 ft/sec). On the basis discussed above, the staff concludes that the seismic responses (member forces and floor response spectra) calculated on the basis of the four site conditions will cover the full range of site conditions as indicated in SSAR Table 2-1 (Site Parameters) of the SSAR. Therefore, Open Item 3.7.2.4-12 is closed.

As discussed in Section 3.7.2.3 of this report, Westinghouse revised the 3D lumped-mass stick seismic model of the NI structures on the basis of the design changes attributable to the post-72 hour action requirements. In order to evaluate the impact of these design changes and the incorporation of the live and snow loads, Westinghouse performed a modal frequency analysis of the revised seismic model on the basis of the hard rock site condition (fixed base) using the BSAP computer code, and performed a seismic SSI analysis on the basis of the upper bound of the soft-to-medium soil condition using the SASSI computer program. Westinghouse's basis of using the upper bound of the soft-to-medium soil condition for the reanalysis was reviewed by the staff was found acceptable on the basis that the interaction effect of the geotechnical characteristics corresponding to this site condition with structural properties of the revised model is judged to be higher than those corresponding to other soil or soft rock site conditions.

In the August 11 through 15, 1997, meeting, Westinghouse presented the results of the new BSAP modal frequency analysis and the new SASSI analysis (Calculation Nos. 1000-S2R-054, Revision 0 and 1010- CCC-002, Revision 2) for review. In its presentation, Westinghouse stated that (1) the frequency of significant peaks of the floor response spectra from the updated SASSI analysis differ by less than 5 percent from those of the existing response envelopes, and (2) the magnitude of floor response spectrum peaks and member forces from the updated

SASSI analysis do not exceed those of the existing response envelopes by more than 10 percent. Therefore, Westinghouse concluded that these design changes will not affect the existing design of the NI structures. The staff's eight review findings and conclusions are discussed below:

(1) For the hard rock site, the maximum change in modal frequencies of the revised NI structural modeling comparison with those of the existing structural model is 2.8 percent. On the basis of this small change, Westinghouse concluded and the staff concurred that the change in seismic responses of the NI structures for the hard rock site condition would also be negligible.

(2) The peak floor accelerations from the reanalysis are constantly higher than those from the previous analysis results in the North-South direction. At all locations, the difference is in the range of 1 to 2 percent except at Elevation 62.56 m (205.25 ft) of the containment vessel. The difference is about 7 percent. In the other two directions (East-West and vertical directions), the peak accelerations from the new analysis are either 1 to 2 percent higher or lower than those from the previous analyses, except at Elevation 93.57 m (307 ft) of the shield building, the new result is 5 percent higher than the previous analysis results. The peak acceleration increase in the North-South direction from the new analysis resulted in a 3 percent increase of the base-shear at the foundation mat. The staff reviewed Westinghouse's dynamic stability evaluation results and found that the safety-factor against sliding is 1.101 which barely meets the SRP guideline of 1.1.

(3) The comparison of the newly generated floor response spectra and the existing floor response spectrum envelopes at Elevations 73.46 m (241 ft), 82.91 m (272 ft), 86.56 m (284 ft), 90.53 m (297 ft) and 93.57 m (307 ft) of the shield building, and at Elevation 78.03 m (256 ft) of the containment vessel shows that all newly generated floor response spectra were enveloped by the existing floor response spectrum envelopes, except that the new floor response spectra of the shield building in the vertical direction exceed the existing floor response spectrum envelopes. In resolving these discrepancies, Westinghouse provided the revised Figure 3.7.2-15 (Sheet 9 of 9) in Revision 17 of the SSAR, in which the exceedance of the vertical floor response spectra was incorporated. Westinghouse's resolution for the floor response spectrum envelopes is acceptable.

(4) The comparison of the new member forces against the existing member forces is shown as follows:

    (a) the average exceedance of new axial forces over the existing axial forces is about 7 percent and the maximum exceedance is 25 percent

    (b) the average exceedance of new shear forces in the North-South direction over the existing shear forces is 7 percent and the maximum exceedance is 15 percent

    (c) the average exceedance of new shear forces in the East-West direction over the existing shear forces is 6 percent and the maximum exceedance is 16 percent

(d) the average exceedance of new torsional moments over the existing torsional moments is 1.4 percent and the maximum exceedance is 13 percent

(e) the average exceedance of new bending moments about the North-South axis over the existing bending moments is 8 percent and the maximum exceedance is 15 percent

(f) the average exceedance of new bending moment about the East-West axis over the existing bending moments is 6.5 percent and the maximum exceedance is 11 percent.

During the meeting, Westinghouse provided its design calculations for review and justified them by stating that the design margin (reinforcements provided ratio to reinforcements required) of structural members (shear walls and floor slabs) from the existing design is high enough to cover the seismic response exceedances from the reanalysis. Westinghouse's justification is acceptable to the staff. On this basis, the existing design is adequate.

(5) Westinghouse's comparison of the dynamic soil pressure against the exterior embedded wall (wall at Column Line 11 between Column Lines K and L) with the existing soil pressure shows that the new soil pressure is about 7 percent higher than the existing soil pressure. Westinghouse justified this exceedance by presenting its design calculations that the design margin (reinforcements provided ratio to reinforcements required) of this embedded wall is much higher than 7 percent. On this basis, the existing design of this wall is adequate.

(6) In its submittal dated July 28, 1997, Westinghouse stated that the soil bearing pressures at the corners of the basemat for the new seismic model were calculated (on the basis of a time history analysis) at each time point. These calculated bearing pressures were compared with those calculated on the basis of the original model (model before the structural changes) and response spectrum analysis method. The comparison showed that the soil bearing pressures from the new analysis are smaller. According to the guideline prescribed in SRP Section 3.7.2, the use of time history analysis method is acceptable to the staff. Also, the Westinghouse's foundation mat design was accepted by the staff as described in Section 3.8.5 of this report. On this basis, the staff concludes that the design changes to the shield building roof structures will not affect the design of the foundation mat.

(7) In the seismic design of the PCCWS tank, Westinghouse assumed that the tank roof slab is rigid and the out-of-plane flexibility of the slab was not considered. The staff's review of Westinghouse's Calculation No. 1070-S3R-010 found that the out-of-plane frequency of the modified roof slab is around 14 Hz which is not in the rigid range. According to the vertical floor response spectrum at the tank roof, the out-of-plane design seismic load should increase by 30 to 40 percent. This finding implies that the assumption of a rigid tank roof slab and the existing design of the roof slab are not acceptable. Westinghouse should consider the out-of-plane flexibility of the slab in the design.

The staff's review of Design Calculation No. 1070-S3R-010 during the meeting on August 11 through 15, 1997, and evaluation of Westinghouse's response to this open item (the letter dated December 17, 1997 [NSD-NRC-97-5497]) found that the out-of-plane frequency of the modified shield building roof slab (around 14 Hz) is a very localized mode, and will not have any detrimental effect on the overall seismic response of the shield building roof structures. On this basis, this issue is resolved.

(8) From the results of its updated SASSI analysis, Westinghouse stated in the draft Revision 17 of SSAR Section 3.7.2.6 that the results of this analysis confirm the adequacy of the seismic responses and the floor response spectra with the exception of the vertical response spectra for the shield building roof which is affected by the additional water mass. On the basis of this statement, Westinghouse decided to revise SSAR Figures 3.7.2-4 and 3.7.2-15 (Sheet 9 of 9) but not the design information documented in SSAR Tables 3.7.2-1 through 3.7.2-12. This is not acceptable on the basis of the discussion above. Westinghouse should either replace the existing results by those from the new analysis or provide new tables to document these new results in the SSAR.

In response to this open issue, Westinghouse provided a draft SSAR revision (SSAR Sections 3.7.2.2.1, 3.7.2.3.1 and 3.7.2.5; SSAR Tables 3.7.2-20 through 3.7.2-23) in its submittal dated December 19, 1997, (NSD-NRC-97-5501) for review. In general, Westinghouse's responses are reasonable and acceptable, except for the following concerns:

(a) SSAR Figure 3.7.2-4 should be revised to incorporate the elevations corresponding to the updated seismic model.

(b) The phrase, "... and the design changes of tank structures due to the post 72 hour action requirements," should be added to the end of the last sentence of the first bullet of Section (revised) 3.7.2.2.1.

(c) The SSAR should commit that if any new seismic analysis is performed for site conditions outside the design certification scope, the revised model (Model B) should be used.

(d) As indicated in Westinghouse's submittal (NSD-NRC-97-5251) dated July 28, 1997, the comparison of floor response spectra (FRS) from Models "A" and "B" showed that the vertical FRS at Elevations 272 ft, 284 ft, 297 ft and 307 ft from Model "B" significantly exceed (about 20 to 25 percent) those from Model "A." If the FRS at Elevations 272 ft, 284 ft and 297 ft are used for the design of safety-related subsystems and components (including seismic Category II piping and components), Westinghouse should either commit, in the SSAR, to use the FRS at Elevation 307 ft in the design or include the FRS at Elevations 272 ft, 284 ft and 297 ft in the SSAR.

(e) In Sheet 2 of 2 of SSAR Table 3.7.2-23 (a new table), Westinghouse should include bending moments at Elevation 306 25 ft. These bending moments were shown in its submittal (NSD-NRC-97-5251) dated July 28, 1997.

In response to the concerns discussed above, Westinghouse provided Revision 22 of the SSAR for review. The changes made by Westinghouse are summarized below:

(a) Westinghouse incorporated the elevations at nodal mass points of the revised seismic model in SSAR Figure 3.7.2-4.

(b) In resolving the staff's concerns regarding incorporation of (1) structural design changes of the shield building roof structures (including the PCCWS tank structures water inventory) because of the post-72 hour action requirements and (2) applicable live loads in the seismic model, Westinghouse added a new Section 3.7.2.2.1 in the SSAR. Also, a commitment was made in this new section that the site specific-evaluation, if required in accordance with SSAR Section 2.5.2.2, will use the modified lumped-mass stick model. Westinghouse further made the following modifications to SSAR Section 3.7.2.2.1 and related SSAR tables and figures to satisfy the staff's concerns:

    (I) Revised the last sentence of the first bullet as "the lumped mass stick model for the shield building roof includes the increase in tank volume and the added water inventory in the passive containment cooling tank."

    (ii) Added a new sentence, "the broadened floor response spectra at the base of the passive containment cooling water storage tank (Elevation 272.42 ft) are shown in Figure 3.7.2-20," and a new Figure 3.7.2-20 to provide seismic input (both horizontal and vertical) for the seismic analysis of safety related piping systems supported by the PCCWS tank structures.

    (iii) Added new Tables 3.7.2-20, 3.7.2-21 and 3.7.2.22 to provide a comparison of structural dynamic properties and seismic member forces of the original seismic model and the updated seismic model.

    (iv) Revise Table 3.7.2-23 by adding seismic moments between Elevations 297.08 ft and 306.25 ft.

    (v) Revise Figure 3.7.2-4 (two sheets) to indicate the top two nodal mass elevations of the updated seismic model.

(c) Revise the first paragraph of Section 3.7.2.3.1 to eliminate the sentence, "this mass was demonstrated to be negligible for the analyses for which results are given later in this section," and to add a phrase, "......, is considered as mass in the global seismic models (these masses are only included in the modified model described in Section 3.7.2.2.1)," to the fourth sentence.

The staff reviewed these SSAR changes and found that the revised SSAR satisfied the staff concerns stated above and meets the guideline of SRP Section 3.7.2. On the basis discussed above, the staff concludes that the issue regarding the revised seismic model of the NI structures because of the post-72 hour requirements is closed.

From the discussion above, the staff found that Westinghouse's assumptions, analysis methods, and computer code used for analyzing the NI structures meet the guidelines prescribed in SRP Section 3.7.2.4 and concludes that the analyses performed for the NI structure design is acceptable.

3.7.2.5 Development of Floor Response Spectra

As described in early revisions of the SSAR, floor response spectra at various elevations and locations of the NI structures were first generated for each of the three selected site conditions:

(1) hard rock site with fixed-base time domain modal time-history analysis (BSAP analysis)

(2) soft rock site with frequency-domain time-history analysis (SASSI analysis)

(3) soft-to-medium stiff soil with frequency-domain time history analysis (SASSI analysis)

Then, these floor response spectra were enveloped, peak-broadened by plus and minus fifteen percent (±15 percent), and smoothed to develop a set of design in-structure spectrum envelopes in accordance with RG 1.122. A set of 3D structural stick models (models for the steel containment vessel, the containment internal structures, and the combined shield and auxiliary buildings) combined with the support foundation mat were used for these analyses. The effects of the spatial combination of three components of the earthquake ground motion time history were considered in the analysis. As such, the coupling effects have been accounted for. The staff's evaluation of the adequacy of the approach for combining responses attributable to three components of the input ground motion is discussed in Section 3.7.2.6. On the basis of the staff's review of the SSAR, the review of Westinghouse's November 30, 1992, March 24, 1994, and May 11, 1994, submittal, and the discussions during the review meetings, the staff concludes that the methods used for the development of in-structure response spectra at different locations and the in-structure response spectrum envelopes are in conformance to the guideline of Section 3.7.2.II.5 of the SRP and RG 1.122, except that the issues related to the combined effects of insufficient participating mass, number of design site conditions, low cut-off frequency, non-conformance of 60 percent limitation of surface ground motion at foundation level, concrete cracking, and other SSI issues discussed above need to be resolved. This was Open Item 3.7.2.5-1.

The concerns of this open item are also addressed in Sections 3.7.2.3 and 3.7.2.4 of this report. On the basis of the resolution for Open Item 3.7.2.2-1 (cut-off frequency for seismic analyses), Open Item 3.7.2.3-5 (conformance of RG 1.122), Open Item 3.7.2.4-12 (number of design site conditions) and Open Item 3.7.2.4-1 (60 percent limitation of the ground motion at the foundation in the free field), and the other open items related to SSI concerns discussed in Section 3.7.2.4 above, the Open Item 3.7.2.5-1 is closed.

3.7.2.6 Three Components of Earthquake Motion

Section 3.7.2.6 of early SSAR revisions stated that the seismic analyses of the NI structures were performed considering the simultaneous occurrences of the two horizontal and the vertical components of earthquake ground motion (ground motion time history or ground response spectra). However, in the seismic analyses, the three components of earthquake were applied either simultaneously (time history analysis) or separately (response spectrum analysis and modal time history analysis).

In the time history analyses with the three earthquake components simultaneously applied, the responses of the three earthquake components were combined within the analytical procedure at each time step. When the three earthquake components were separately applied for the case of time history analyses, the corresponding responses from the three individual analyses were combined algebraically, at each time step, to obtain the total acceleration response time history. In some cases, the peak responses from the three individual analyses were combined to obtain the total peak response using either the square root of the sum of squares (SRSS) technique or the 1.0, 0.4, and 0.4 direct combination technique. The SRSS or 1.0, 0.4 and 0.4 direct combination technique were also applied to the response spectrum method of analysis. For axisymmetric structures, such as the steel containment vessel and shield building roof structure, only one horizontal peak response and the vertical peak response were combined using either the SRSS or the 1.0, 0.4, and 0.4 direct combination technique.

The staff found that the algebraical sum for the time history analysis and the SRSS technique for the response spectrum analysis used to combine the responses attributable to the three earthquake components meet the guideline of SRP Section 3.7.2 and will result in a reasonable calculation of peak responses and are, therefore, acceptable. However, Westinghouse should revise the SSAR and provide a list of analysis cases showing how and where each of the three combination techniques was applied. In addition, Westinghouse should justify the adequacy of using the 1.0, 0.4, and 0.4 direct combination method for combining the responses attributable to the three earthquake components. This was Open Item 3.7.2.6-1.

In Revision 7 of SSAR Section 3.7.2.6 and Table 3.7.2-16, Westinghouse provided a description of how and where each of the three techniques for combining seismic responses attributable to the three components of earthquake ground motion is used. Also, at the June 12 through 16, 1995 meeting, Westinghouse provided calculations to demonstrate that the 1.0, 0.4, 0.4 combination method always gives reasonable results by comparing these results with those from the SRSS combination method. From its review of Revision 7 of the SSAR, the staff found that the SSAR description of how and where each of the three combination techniques was applied meets the guideline of RG 1.92. The staff also, from its review of design calculations, found that the difference between results obtained using these two methods is insignificant. Because Westinghouse has (1) demonstrated the use of the 1.0, 0.4, and 0.4 combination method for combining the spatial responses attributable to the three earthquake components always gives reasonable results, and (2) provided, in SSAR Table 3.7.2-16, a list of analysis cases showing how and where each of the three combination techniques was applied, Open Item 3.7.2.6-1 is closed.

3.7.2.7 Combination of Modal Responses

In Section 3.7.2.7 of early SSAR revisions, Westinghouse stated that modal responses on the basis of the response spectrum analysis method were combined using the SRSS technique unless the modes were closely spaced. For closely spaced modes, either the grouping method, the 10 percent method, or the double sum method described in Section C of RG 1.92 was used. On this basis, these modal response combination techniques are acceptable to the staff. However, Westinghouse should revise the SSAR and provide a list of analysis cases showing where each of the three combination techniques for closely spaced modes was applied. This was Open Item 3.7.2.7-1.

The staff's review of Revision 7 of SSAR Section 3.7.2.6 and Table 3.7.2-16 found that Westinghouse provided descriptions of how and where the modal combination technique was applied for calculating seismic responses including the consideration of closely spaced modes. On this basis, Open Item 3.7.2.7-1 is closed.

3.7.2.8 Interaction of SC-II and Non-seismic Structures with SC-I Structures

As described in Section 3.2.1 of the SSAR, non-seismic Category I structures include the SC-II and non-seismic structures. In the June 27, 1994, submittal, Westinghouse classified the structures adjacent to the NI structures as follows:

- Annex Building: non-seismic
- Turbine Building: non-seismic
- High Bay Area of Radwaste Building: Category II
- Single-Story Area of Radwaste Building: non-seismic

In addressing the staff's concern regarding the classification of non-seismic Category I structures, Westinghouse, in Revision 9 of the SSAR, reclassified the annex building as SC-II, and the turbine building and the radwaste building as non-seismic. The staff's review and conclusion for the classification of AP600 plant structures are discussed under Open Items 3.7.2.8-3 and 3.7.2.8-5 below.

In Section 3.7.2.8 of early SSAR revisions and in the June 27, 1994, submittal, Westinghouse also described the interaction requirements for the NI structures with the SC-II structure and non-seismic structures as follows:

- The collapse of a non-seismic structure will not cause the non-seismic structure to strike a SC-I structure or components.

- The collapse of a non-seismic structure will not impair the integrity of SC-I structures or components.

- The structure is classified as SC-II and is analyzed and designed to prevent their collapse under the SSE.

Westinghouse's interaction requirements stated above were found acceptable to the staff, except that Westinghouse was requested to apply these requirements to systems and components as well as to structures. This was Open Item 3.7.2.8-1. According to Westinghouse, the interaction requirements for subsystems are provided in Revision 7 of SSAR Section 3.7.3.13. The staff's review of Westinghouse's response of this open item is discussed in Section 3.12.3.7 (Open Item 3.12.3.7-1) of this report. On this basis, Open Item 3.7.2.8-1 is closed.

As shown in Figure 1.2-2 of the SSAR, the annex building, radwaste building and turbine building are very close to the NI structures. It is obvious that the interaction requirements stated above cannot be met if these building structures are classified as non-seismic and are not analyzed and designed for the SSE. These building structures should be reclassified as SC-II. This was Open Item 3.7.2.8-2. In Revision 12 of SSAR Section 3.7.2.8, Westinghouse stated that the annex building, which is 10 cm (4 in) away from the NI structures, is classified as SC-II and is designed to prevent its collapse under the SSE. This building, according to Revision 7 of SSAR Section 3.7.2, is designed for the SSE using the same methods and criteria as are used for SC-I structures. In addition, the staff, during the review meetings, reviewed the design calculations and found that the 10 cm (4 in) clearance between the annex building and the NI will prevent any interaction of these two buildings. Therefore, to classify the annex building as SC-II is acceptable.

Revision 12 of the SSAR also stated that the radwaste and turbine buildings are classified as non-seismic. The minimum clearance between the structural elements of the radwaste building above grade and the NI is 10 cm (4 in). For the turbine building, the major structure is separated from the NI by approximately 6 m (18 ft). However, there are floors and roof between the turbine building main structure and the NI to provide access to the NI. The staff concern regarding the classification of the radwaste and turbine buildings, and the interaction between these two buildings and the NI are addressed under Open Items 3.7.2.8-3 and 3.7.2.8-5. On the basis discussed above, Open Item 3.7.2.8-2 is closed.

In Section 3.7.2 of early SSAR revisions, Westinghouse stated that the seismic design of SC-II structures is on the basis of the same input ground motion (i.e., SSE) and acceptance criteria used for the SC-I structures. SC-II building structures are to be analyzed for the SSE using the same methods as were used for SC-I structures. For SC-II concrete structures, load combinations and load factors are in accordance with ACI 318, except that the load factor for the SSE is taken as 1.0. Allowable stresses for SC-II steel structures are in accordance with AISC with a 60 percent increase permitted for SSE instead of a 33 percent increase. As for the non-seismic structures, Section 3.7.2 of the SSAR indicated that these structures were analyzed and designed for seismic loads according to UBC requirements for Zone 2A.

From the review of Westinghouse's submittal dated January 22, 1993 and June 27, 1994, the staff identified several issues. These issues and their resolutions are summarized as follows:

- Westinghouse was requested to provide the basis for classifying the single-story portion of the Radwaste Building as non-seismic and the high bay area of the Radwaste Building as SC-II. The staff identified the concern of the seismic design of the radwaste building as Open Item 3.7.2.8-3.

  In Revision 9 of SSAR Section 3.7.2.8, Westinghouse stated that the radwaste building is classified as non-seismic and is designed to the seismic requirements of the UBC Zone 2A with an importance factor of 1.25. However, Westinghouse did not make a commitment that the collapse of this building will not impair the safety function of the NI structures.

  In Revision 12 of SSAR Section 3.7.2.8, Westinghouse, on the basis of the energy balance theory, provided the analysis procedures which are to be used for demonstrating that the collapse of the radwaste building will not cause any damage of the NI structures. Because the application of energy balance for checking potential damages of structures is consistent with the industry practice, it is acceptable to the staff. Also, in the August 11 through 15, 1997, meeting, the staff reviewed the final calculation (Calculation No. 5000-S2C-002) and found that the analysis procedure described in the SSAR was properly applied in the evaluation of the impact between the NI and the radwaste building and that the impact from the radwaste building in the event of an SSE would not impair the integrity of the NI. This is acceptable to the staff regarding the potential interaction between the radwaste building and the NI structures. However, Westinghouse revised its commitment and stated in Revision 17 of SSAR Section 3.7.2.8.2 that the radwaste building is designed to the seismic design requirements of UBC, Zone 2A with an importance factor of 1.0. To design the radwaste building on the basis of the requirements of UBC, Zone 2A with an importance factor of 1.0 is not consistent with the Electric Power Research Institute's (EPRI's) requirements document for passive plant design and is not acceptable to the staff.

In Revision 20 of SSAR Section 3.7.2.8.2, Westinghouse revised the seismic design criteria for the Radwaste building from UBC Zone 2A requirements with the "importance factor" of 1.0 to UBC Zone 2A requirements with the "importance factor" of 1.25. This SSAR revision meets the previous staff's conclusion for DSER Open Item 3.7.2.8-3, and also meets the staff's review conclusion for the EPRI URD. On this basis, the staff concludes that the issue regarding the "importance factor" for the radwaste building design is closed.

- If Category II structures are designed using load factors and allowable stresses as discussed, the stress level can exceed yield stress (for American Institute of Steel Construction (AISC)) or load demand can be equal to the ultimate load capacity (for concrete sections). In such a case, Westinghouse should demonstrate that the SC-II structures, designed using the load factor method and allowable stresses in accordance with AISC with a 60 percent increase permitted for the SSE, will not collapse during an SSE or these structures possess enough margin (ductility reserve) to prevent collapse. However, Westinghouse did not indicate that any such design evaluation has been performed. This was Open Item 3.7.2.8-4.

  In Revision 9 of SSAR Section 3.7.2, Westinghouse stated that SC-II building structures are designed for the SSE using the same methods as are used for SC-I structures. The acceptance criteria are on the basis of ACI 349 Code for concrete structures and on AISC N690 for steel structures including the supplemental requirements described in Sections 3.8.4.4.1 and 3.8.4.5 of the SSAR. However, Westinghouse should commit in the SSAR that the same design allowables specified in ACI 349 Code and AISC N690 Standard used for the seismic Category I structures will be used for the design of seismic Category II structures.

  In Revision 20 of SSAR Section 3.7.2 (the third paragraph), Westinghouse stated that "seismic Category II building structures are designed for the SSE using the same methods and design allowable as are used for seismic Category I structures." This SSAR change technically resolved the staff's concern regarding the design allowable to be used for the design of seismic Category II structures. On this basis, Open Item 3.7.2.8-4 is closed.

- To avoid the collapse of the annex and the turbine buildings towards the NI structures, Westinghouse proposed, during review meetings, a method for the design of bracing systems. In the design, the bracing systems for preventing these structures to be deformed toward the NI structures are twice as strong as those to be used for the opposite direction for which the bracing systems are designed according to UBC Zone 2A requirements. The seismic design of these buildings proposed by Westinghouse is not acceptable as follows:

- There are many conditions (e.g., inherent material variability, differences in tolerances, the effect of construction sequences and temperature conditions) that can cause uneven loading. The collapse strength of two supposedly identical braces can differ by more than 50 percent. Thus, the proposed method does not ensure collapse away from the NI structures.

- For a large structural system when subjected to a seismic level (SSE) higher than the design level (UBC Zone 2A requirements), it is probable that some structural members will be stressed beyond their elastic limit or fail at the initiation of the earthquake motion. The effects of one or several such localized failures on the progression of collapse, especially under continuing vibratory loads (that will cause load reversal), are quite uncertain. It is further complicated by the significant difference in the compression and tension capacity of the braces.

These concerns had not been addressed by Westinghouse. This was Open Item 3.7.2.8-5.

The resolution of the concern related to the classification of the annex and radwaste buildings is discussed under Open Item 3.7.2.8-1 above and, therefore, this issue is resolved.

As for the turbine building, Westinghouse stated in Revision 9 of SSAR Section 3.7.2.8 that this building is classified as non-seismic and the major structure of this building is separated from the NI by approximately 6 m (18 ft). The roof and floors (which are classified as non-seismic) between the turbine building main structure and the NI provide access to the NI. The floor beams are supported on the outside of the NI with a nominal clearance of 30 cm (12 in) between the structural elements of the turbine building and the NI. These beams are of light construction such that they will collapse if the differential deflection of the two buildings exceeds the clearance and will not jeopardize the 0.6 m (2 ft) thick walls of the NI. The roof in this area rests on the roof of the NI and could slide relative to the roof of the NI in a large earthquake. The seismic design of the turbine building including the beams and roof structure is on the basis of the requirements of UBC Zone 3 with an importance factor of 1.0 in order to provide margin against collapse during the SSE. Also, in Revision 12 of this SSAR section 3.7.2.8, Westinghouse stated that for an eccentrically braced structure, the resistance modification factor is 10 using allowable stress design without considering the increase of allowable stresses by one third for seismic loads. In addition, the design of the lateral bracing system complies with the seismic requirements for eccentrically braced steel frames of Section 9.3 of the AISC Seismic Provisions for Structural Steel Buildings. The staff's review of these two SSAR revisions found that Westinghouse's response to the staff's concern regarding the design of the turbine building is acceptable. On the basis discussed above, Open Item 3.7.2.8-5 is closed.

- In the January 22, 1992 and June 27, 1994, submittal, Westinghouse proposed to use the same input ground motion and acceptance criteria as those for the SC-I structures for the design of SC-II structures. On the basis of its review of the design approach and criteria used for the NI structures stated above, the staff concludes that Westinghouse's design approach for SC-II structures is acceptable because it meets the guidelines of Section 3.7.2 of the SRP. However, Westinghouse should demonstrate that these SC-II structures will not be excessively deformed and will not affect the function of any safety related items in the event of an SSE. The issue regarding the design of SC-II structures was Open Item 3.7.2.8-6.

  In response to this open item, Westinghouse stated in Revision 2 of SSAR Section 3.7.2 that SC-II structures are designed for the SSE using the same methods that are used for SC-I structures. Also, Westinghouse stated in this SSAR revision that the annex building is the only building structure adjacent to the NI structures classified as SC-II and this building is designed to prevent its collapse under the SSE. During the December 9 through 13, 1996, meeting, Westinghouse informed the staff that a 10 cm (4 in) gap is provided between these two buildings. The staff reviewed the seismic analysis reports and found that the absolute summation of deflections of these two structures is less than 10 cm (4 in). In Revision 12 of the SSAR, Westinghouse formally documented the 10 cm (4 in) gap in Section 3.7.2.8. On this basis, Open Item 3.7.2.8-6 is considered closed.

- For the evaluation of seismic margin, Westinghouse should demonstrate and document in the SSAR that both SC-II and non-seismic structures can withstand an earthquake with a peak ground acceleration up to 0.5g without collapse. As an alternative, Westinghouse may demonstrate that the collapse of these structures as a result of an earthquake up to 0.5g will not have any impact on the safe function of SC-I structures, systems, and components. This was Open Item 3.7.2.8-7.

  The staff's review of Westinghouse's response on this issue is discussed in the staff's safety evaluation for AP600 Probabilistic Risk Assessment (PRA) (Appendix 19A of this report). On this basis, the staff concludes that Open Item 3.7.2.8-7 is closed.

3.7.2.9 The Effects of Parameter Variations on Floor Response Spectra

In the previous revisions of SSAR Section 3.7.2.9, the effects of parameter uncertainty had not been explicitly considered. To account for such effects, the peaks of the floor spectra were broadened by ±15 percent as recommended in Section 3.7.2 of the SRP and RG 1.122. The staff found this acceptable, except that Open Item 3.7.2.3-5 (see Section 3.7.2.3) concerning the issue of concrete cracking must be resolved. This issue is especially significant when one considers the additional uncertainties associated with the modular construction (e.g., attributable to the presence of cross diaphragms in module walls, module anchorages to the building concrete, module connections, etc.). The staff's evaluation of modular construction used in the AP600 design is discussed in Section 3.8.3 of this report. Another staff concern is that the SSI analyses performed for the two soil sites did not include the usual variation of soil shear moduli as recommended by the SRP. This was Open Item 3.7.2.9-1.

On the basis of the resolution of open items related to the structural modeling (Section 3.7.2.3 of this report), the use of structural modules for the containment internal structures (Sections 3.8.3 and 3.8.4 of this report), and SSI (Section 3.7.2.4 of this report), Open Item 3.7.2.9-1 is closed.

3.7.2.10 The Use of Constant Vertical Static Factors

Vertical seismic response was explicitly considered in the SSI and fixed-base seismic analyses. Therefore, equivalent vertical static factors were not used to compute seismic design loads of major structures. Therefore, this issue is not applicable to the AP600 design.

3.7.2.11 Method Used to Account for Torsional Effects

From its review of the previous revisions of the SSAR Section 3.7.2.3 and the review of design calculations in the review meetings, the staff found that all known eccentricities were explicitly represented in the seismic model, except the eccentricities in the SCV that are associated with (1) equipment hatch, (2) personnel hatch, and (3) polar crane trolley (which is to be parked away from the building center during plant operation). On the basis of the staff's past review experience, the seismic structural responses (in-plane shear in structural elements and in-structure response spectra) would be significantly affected by these eccentricities. Westinghouse was requested to include them in the seismic models. This was Open Item 3.7.2.11-1. At the February 28 through March 2, 1995, meeting, the staff's review of analysis reports found that Westinghouse has included the eccentricity attributable to major components in its modified seismic model and the modeling procedure used meets the guideline of SRP Section 3.7.2. On this basis, Open Item 3.7.2.11-1 is closed.

Section 3.7.2.11 of early SSAR revisions stated that the seismic analysis models of the NI structures incorporate the mass and stiffness eccentricities of these structures and the torsional degrees of freedom and, therefore, accidental torsion is not added to the actual calculated torsional responses. This is not acceptable to the staff. The guideline of Section 3.7.2 of the SRP states that, to account for accidental torsion in the calculation of the seismic shear forces, Westinghouse should include an additional accidental eccentricity of ±5 percent of the maximum building dimension at the level under consideration for each direction in the design. The issue regarding the inclusion of accidental torsion for computing seismic shear forces was Open Item 3.7.2.11-2. In Section 3.7.2.11 of the SSAR, Revision 17, Westinghouse stated that the accidental torsional moments as a result of the eccentricity of each mass are calculated on the basis of the assumption of an accidental eccentricity equal to plus and minus 5 percent of the maximum building dimensions at the elevation of the mass. The staff found that the approach for considering accidental eccentricity used by Westinghouse meets the guideline of SRP Section 3.7.2. Open Item 3.7.2.11-2 is closed.

3.7.2.12 Comparison of Responses

For the fixed-base (representing hard rock site) case, the response spectrum analysis method was used to calculate the moments and forces, and the modal time-history analysis method was used to calculate the in-structure acceleration response spectra. Even though the response spectrum analysis was performed on the basis of a detailed finite element model and the modal time-history analysis was performed on the basis of an equivalent lumped-mass stick model, in-structure response spectra and member forces at various floor elevations obtained by these two methods can be meaningfully compared. Especially, the base-shear comparison will indicate the effect of neglecting the contributions from the higher modes (above 33 Hz) in the modal time-history analysis results. The staff position stated in Section 3.7.2.II.12 of the SRP indicates a need for such a comparison to demonstrate approximate equivalency between the response spectrum and the time history methods of analyses. However, Westinghouse stated, in early SSAR revisions, that in the seismic analyses performed, two different models were used. Therefore, a comparison of responses calculated by alternative methods is not necessary. The staff does not consider this an acceptable justification because the equivalence of the two different methods of analyses has not been established. For structural models with significant contributions from higher modes of vibration, the results of the response spectrum analyses can be defined. This was Open Item 3.7.2.12-1.

In Revision 9 of SSAR Section 3.7.2.12, Westinghouse stated that the 3D lumped mass fixed-base stick model of the NI was analyzed by modal superposition time history analysis and by the response spectrum analysis method for the hard rock site condition. The comparison of results (maximum absolute nodal accelerations, member forces and moments) from these two analyses is shown in Tables 3.7.2-17, 3.7.2-18 and 3.7.2-19, respectively. The staff's review of this SSAR revision found that the maximum absolute nodal accelerations calculated by the response spectrum analysis are consistently higher than those from the modal superposition time history analysis. At some locations of the steel containment vessel, the accelerations from these analyses deviate by 30 percent in the East-West direction and more than 40 percent in the vertical direction. The staff's concern is that if the maximum nodal accelerations calculated by modal time history analyses are always lower than those obtained from response spectrum analyses, it implies that the floor response spectra, generated on the basis of the floor time histories, may not be conservative for the design of subsystems such as piping. Westinghouse should justify the adequacy of the final design floor response spectra documented in the SSAR.

In Revision 13 of SSAR Section 3.7.2.12, Westinghouse stated that the two methods of analysis give similar results with the response spectrum analysis being generally more conservative. Investigations of the two analyses showed that the conservatism in the response spectrum analyses is attributable to cross coupling of the directions in the multi stick model. The double sum modal combination method used in the response spectrum analysis is very conservative when there are closely spaced modes some of which are out-of-phase. Also, in its submittal dated May 2, 1997, Westinghouse provided a comparison of the vertical nodal accelerations from the new modal response spectrum analysis with the those from the existing time history analysis. The staff's review of this submittal and the review of the additional information provided by Westinghouse during the August 11 through 15, 1997, meeting found that the new modal response spectrum analysis reduced the vertical nodal accelerations to about 55 percent of those from the existing response spectrum analysis throughout the height of the containment vessel. As a result, the nodal accelerations from the modal response spectrum analysis became much lower than those from the modal time history analysis at lower elevations of the containment vessel. Westinghouse, during the meeting, could not explain the cause of the significant discrepancies between the vertical seismic response of the containment vessel. On the basis discussed above, the staff concludes that Westinghouse's response to Open Item 3.7.2.12-1 is not acceptable.

Westinghouse responded to this open item through the letter dated August 28, 1997. As described in this submittal, Westinghouse, using a single stick lumped-mass model of the steel containment vessel, performed a new response spectrum analysis and a new modal time history analysis for the steel containment vessel in the vertical direction. This single stick lumped-mass model was developed by setting the mass of the containment internal structures and the coupled shield/auxiliary building in the 3D lumped-mass stick model of the NI structures equal to zero. The purpose of this set of analyses is to eliminate the coupling dynamic effects on the steel containment vessel and other NI structures. The staff's review of this submittal found that for the single stick SCV model, the nodal accelerations in the vertical direction obtained from the new time history analysis are reasonably close in comparison with those from the new response spectrum analysis, when the effects of high frequency modes were considered. On the basis discussed above, the staff concludes that the nodal accelerations of the SCV in the vertical direction (calculated by the response spectrum analysis method) is acceptable and Open Item 3.7.2.12-1 is closed.

3.7.2.13 Methods of Seismic Analysis of Dams

In Section 3.7.2.13 of early SSAR revisions, Westinghouse stated that seismic analysis of dams is a site specific design. The staff agrees with this SSAR statement. However, Westinghouse did not state in the SSAR that a COL applicant referencing the AP600 design will perform seismic analysis for evaluating the safety of existing dams and design the new dams on the basis of the defined SSE. This was COL Action Item 3.7.2.13-1 and Open Item 3.7.2.13-1.

In Revision 2 of SSAR Section 3.7.5.1, Westinghouse stated that COL applicants referencing the AP600 certified design will, using the site-specific SSE, evaluate existing and new dams whose failure could affect the site flood level specified in Section 2.4.1.2. Since dams are site-specific features and on the basis of the rationale used for liquefaction potential evaluation (Section 2.5 of this report), the use of site-specific SSE for evaluating these dams is acceptable. Therefore, Open Item 3.7.2.13-1 is closed.

3.7.2.14 Determination of SC-I Structure Overturning Moments

The staff's evaluation of dynamic stability (e.g., sliding, flotation, and overturning) of the NI structures is discussed in Section 3.8.5 of this report.

3.7.2.15 Analysis Procedure for Damping

The staff's evaluation of the analysis procedure for damping is discussed in Section 3.7.1 of this report.

3.7.2.16 Confirmation of Plant-Specific Seismic Design Adequacy

The seismic design basis earthquake for the AP600 structures, systems, and components are essentially defined at the plant grade level in the free field by an SSE with the peak acceleration of 0.3g and the ground response spectra shown in SSAR Figures 3.7.1-1 and 3.7.1-2. The seismic design of the NI (structures, system, and components) is on the basis of the enveloped results from a limited number of site conditions (soft-to-medium stiff soil site, upper bound of the soft-to-medium stiff soil site, soft rock site, and hard rock site). It is the staff's concern that if these design bases are not satisfied (i.e., the site condition is not within the range of site conditions committed in the SSAR) or if the seismic analysis response envelope used for the design can not envelop the results obtained from some potential plant site conditions not included in the three site conditions stated above, the basis established for the design certification will no longer apply. In its letter dated March 16, 1994, the staff requested Westinghouse to commit in the SSAR that the COL applicant should perform an analysis and evaluation using the design basis earthquake ground motion and plant-specific site conditions to confirm the design adequacy of the AP600 design. This was COL Action Item 3.7.2.16-1 and Open Item 3.7.2.16-1.

During the meeting on August 11 through 15, 1997, and the early review meetings, the staff reviewed the seismic analysis summary report for each of the four site conditions and the seismic analysis summary report of the updated model, and found that the SSAR commitments for the seismic analysis of the NI structures have been properly implemented. On this basis, the staff concludes that for the reconciliation analysis by the COL applicant for sites with site parameters within the bound of those specified in SSAR Table 2-1 is not necessary. However, Westinghouse should commit in the SSAR that these seismic analysis summary reports are available to demonstrate that seismic Category I structures are analyzed according to the procedures described in the SSAR. (For sites with site parameters outside the bounds of those described in SSAR Table 2-1, Westinghouse committed in Revision 15 of SSAR Section 2.5.2.2 that site-specific SSI analyses must be performed by the COL applicant to demonstrate acceptability of sites with seismic and soil characteristics that are outside the site parameters in SSAR Table 2-1. On the basis of the discussion above, the technical concern of this issue is resolved.) Open Item 3.7.2.16-1 will not be closed until Westinghouse commits in the SSAR that the above-mentioned seismic analysis summary reports are available for the future use.

In Revision 20 of the SSAR, Westinghouse added a new paragraph to the end of Section 3.7.2. This new paragraph stated that the seismic analyses of the nuclear island are summarized in a seismic analysis summary report. This report describes the development of the finite element models, soil-structure interaction analyses, and results thereof. A separate report provides the floor response spectra for the nuclear island. Westinghouse's SSAR commitment resolved the staff's concern on this issue. On this basis, Open Item 3.7.2.16-1 is closed, and COL Action Item 3.7.2.16-1 is dropped.

3.7.2.17 Conclusions

On the basis of the above discussion, the staff concludes that the AP600 design is acceptable and meets the requirements of GDC 2 and Appendix A to 10 CFR Part 100. This conclusion is as follows:

Westinghouse has met the requirements of GDC 2 and Appendix A to 10 CFR Part 100 with respect to the capability of the structures to withstand the effects of earthquakes so that their design reflects the following three requirements:

(1) appropriate consideration for the most severe earthquake recorded for most sites east of the Rocky Mountains with an appropriate margin (GDC 2)

(2) appropriate combination of the effects of normal and accident conditions with the effect of the natural phenomena

(3) the importance of the safety functions to be performed (GDC 2). The use of a suitable dynamic analysis to demonstrate that structures, systems and components can withstand the seismic and other concurrent loads

Westinghouse has met the requirements of the first item listed above by using seismic design parameters that meet the guidelines of Section 3.7.1 of the SRP. With respect to the second item above, the combination of earthquake-induced loads with those resulting from normal and accident conditions in the design of Category I structures meet the guidelines of Sections 3.8.2 through 3.8.5 of the SRP and are discussed in corresponding sections in this report.

The structural system and subsystem analyses were performed by Westinghouse on an elastic and linear basis. Time history analysis methods form the bases of the seismic analyses for the NI structures founded on soil sites and for generating in-structure response spectra. The seismic response spectrum analysis method was used for calculating seismic loads to design structural members. When the modal response spectrum method is used, the methods used in combining modal responses are in conformance with RG 1.92 and also meet high-frequency mode contribution guidelines of Appendix A to Section 3.7.2 of the SRP. The square root of the sum of the squares of the maximum codirectional responses is used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. In-structure response spectra used for analysis and design of subsystems are generated from the time history method (both time domain and frequency domain) and are in conformance with RG 1.122. A vertical seismic system dynamic analysis is employed for all structures, systems and components where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning, sliding flotation are considered. A coupled structure and soil model is used to evaluate SSI effects upon seismic responses. Appropriate nonlinear stress-strain and damping relationships for the soil are considered in the analysis.

The staff concludes that the use of the seismic structural analysis procedures and criteria delineated above by Westinghouse provides an adequate basis for the seismic design, which is in conformance with the requirements of the Item (3) above.

3.7.3 Seismic Subsystem Analysis

This section discusses the staff's review of the seismic input motion, seismic analysis methods, and modeling procedure used for the analysis and design of AP600 seismic Category I (SC-I) subsystems. In particular, this review focused on such subsystems as miscellaneous steel platforms; steel frame structures; tanks; cable trays and supports; heating, ventilation, and air conditioning (HVAC) ductwork and supports; and conduit and supports.

Section 3.12 of this report discusses the staff's review of Westinghouse's analysis and design criteria for AP600 piping systems, while Section 3.10 discusses the review of AP600 electrical and mechanical components. Sections 3.8.3 and 3.8.4 of this report discuss the staff's evaluation regarding the design of subsystems other than piping and electrical/mechanical components.

3.7.3.1 Seismic Input Motion

As input motions for the analysis of AP600 SC-I systems, Westinghouse is required to use the envelopes of the in-structure response spectra generated according to the procedures described in Section 3.7.2 of the SSAR. Section 3.7.2 of this report discusses the staff's evaluation of these in-structure response spectrum envelopes.

3.7.3.2 Analysis Methods

In Section 3.7.3.1 of early SSAR revisions, Westinghouse stated that one of four analysis methods is used for the seismic analysis of AP600 subsystems. Specifically these four methods include modal response spectrum analysis, time history analysis, equivalent static analysis, and "design by rule." The following paragraphs summarize the staff's evaluation regarding the adequacy of these analysis methods:

- Use of the modal response spectrum and time history analysis methods for the analysis of AP600 subsystems meets the guidelines prescribed in Section 3.7.2 of the SRP and therefore, is acceptable to the staff. However, in the SSAR, Westinghouse should identify the computer codes used and the validation method for those computer codes. The staff identified this issue as Open Item 3.7.3.2-1.

  According to Westinghouse, Section 3.9.1.2 of the SSAR fulfills this requirement by addressing both the computer programs used and their validation. Consequently, Open Item 3.7.3.2-1 is closed, on the basis of the staff's review, as described in Section 3.9.1 and 3.12 of this report.

- As described in Section 3.7.3.5 of early SSAR revisions, the equivalent static analysis method involves the calculation of equivalent horizontal and vertical static forces applied at the center of gravity of various subsystem masses. If the subsystem can be characterized as a single degree of the freedom system, the equivalent static forces are computed as the spectral acceleration corresponding to the calculated natural frequency of the subsystem, multiplied by the total mass of the subsystem. If the subsystem cannot be characterized or modeled as a single degree of the freedom system (such as multi-degree of the freedom system), the seismic forces are computed as the peak spectral acceleration multiplied by a factor of 1.5 times the total mass of the subsystem. The SSAR also stated that this analysis method may be used for the design of steel platforms, cable tray and supports, conduit and supports, HVAC ducts and supports, and other substructures.

  After reviewing the SSAR description, the staff concluded that use of the equivalent static analysis method to analyze the AP600 subsystems (substructures, systems, and components) does not meet the guideline prescribed in Section 3.7.2 of the SRP. Specifically, SRP Sections 3.7.2 and 3.7.3 recommend that the subsystem must be realistically represented by a simple model, and the design and associated simplified analysis should account for the relative motion between all points of support. However, the staff's review experience suggests that subsystems such as steel platforms and frame structures cannot be modeled as a single degree or simple multi-degree of the freedom system. Westinghouse should therefore justify the use of this method to analyze these subsystems, and the staff identified this issue as Open Item 3.7.3.2-2.

In Revision 7 of SSAR Section 3.7.3.5.1, Westinghouse further addressed the applicability of this method for components that have a distributed mass and for which the dynamic response will be single-mode dominant. For such components, the equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value at the component natural frequency from the applicable floor response spectra, multiplied by a factor of 1.5. (A factor of less than 1.5 may be used if justified. In addition, a factor of 1.0 is used for structures or equipment that can be represented as a uniformly loaded cantilever, simply supported, fixed-simply supported, or fixed-fixed beams.) If the frequency is not determined, the peak spectral acceleration from the applicable floor response spectrum is used.

The staff concluded that the approach proposed by Westinghouse (using the equivalent static loads for the seismic design of subsystems) is not acceptable because this approach does not meet the guideline prescribed in SRP Section 3.7.2.II.1.b. Specifically, the SRP states that, to obtain an equivalent static load of a structure, equipment, or component that can be represented by a simple model, a factor of 1.5 is applied to the peak spectral acceleration of the applicable floor response spectrum. Therefore, if the equivalent static method is used for the analysis of a structure, equipment or component, Westinghouse should commit (in the SSAR) that the equivalent static force is equal to the factor of 1.5 times the peak spectral acceleration of the applicable floor response spectrum. In addition, Westinghouse should commit (in the SSAR) that static factors smaller than 1.5 are not to be used for multi-degree of freedom subsystems such as the piping system and steel frameworks.

In Revision 9 of SSAR Section 3.7.3.5.1, Westinghouse stated that a static factor of less than 1.5 may be used if justified. Static factors smaller than 1.5 will not be used for piping systems. Westinghouse's revised SSAR commitment, therefore, meets the guideline prescribed in SRP Sections 3.7.2 and 3.7.3 and, therefore, is acceptable.

With regard to the analysis of steel frames, the staff is concerned that the amplification of support motion attribute to the flexibility of the steel frame should be considered in the design of safety-related items (such as piping and components) that are to be supported by a steel frame, if the fundamental frequency of the steel frame is not in the rigid range. In Revision 12 of SSAR Section 3.7.3.8.3, Westinghouse committed that when piping systems or components are decoupled from the analysis of the frame using the preceding criteria, the effect of the frame is accounted for in the analysis of the decoupled piping systems or components. Either an amplified response spectra or a coupled model is used. The amplified response spectra are obtained from the time history SSE analysis of the frame. The coupled model consists of a simplified mass and stiffness model of the frame connected to the seismic model of the piping systems or components. Alternative criteria may be applied to simple frames that behave as pipe support miscellaneous steel. The staff reviewed Revision 12 of the SSAR and found that Westinghouse's commitment meets the guideline prescribed in the SRP and, therefore, is acceptable. On that basis, Open Item 3.7.3.2-2 is closed.

- As a result of its review of Westinghouse's submittal dated May 20, 1994, the staff found that Westinghouse will only use the "design by rule" method for the design of small-bore piping systems. As discussed in Section 3.12.3.6 of this report, the staff's evaluation of small-bore piping analysis and design revealed that Westinghouse decided not to use the "design by rule" method for the AP600 subsystems. The staff, therefore, required Westinghouse to revise Section 3.7.3 of the SSAR to remove the reference to this option, and identified this requirement as Confirmatory Item 3.7.3.2-1.

Westinghouse subsequently fulfilled the staff's requirement, indicated in Revision 2 of SSAR Section 3.7.3, by removing the SSAR discussion related to the "design by rule" option for the analysis of AP600 subsystems. Confirmatory Item 3.7.3.2-1 is therefore closed.

3.7.3.3 Procedure Used for Modeling

In Section 3.7.3.3 of early SSAR revisions, Westinghouse did not specifically define the modeling procedures to be used for subsystems other than piping systems. In a letter dated January 26, 1994, the staff requested Westinghouse to take the following actions:

- Provide a detailed modeling procedure and analysis method for steel structural frames and miscellaneous steel platforms.
- Provide modeling procedures for cable trays and supports, conduit and supports, and HVAC systems.
- Consider the potential amplification of motion through the steel frames and platforms in the piping analyses.

In reviewing Westinghouse's submittal dated April 14, 1994, the staff found that the technique used for modeling these subsystems, as described in Section 3.7.3.3 of the SSAR, meets the guideline prescribed in SRP Section 3.7.2 and, therefore, is acceptable. In addition, Westinghouse addressed the third action item listed above in Revision 1 to SSAR Section 3.7.3.8.3. However, Westinghouse should revise Section 3.7.3.3 of the SSAR to incorporate the commitment stated in the submittal of April 14, 1994. The staff identified this requirement as Confirmatory Item 3.7.3.3-1.

In reviewing Revision 2 of the SSAR, the staff found that the subsystems defined by Westinghouse include miscellaneous steel platforms and frames, equipment modules, tanks, components, and distributive systems. (The latter category includes piping and piping supports, electric cable trays and supports, conduit and supports, HVAC ductwork and supports, and instrumentation tubing and supports.) These subsystems are modeled and analyzed using the approach stated in SSAR Sections 3.7.3.3 and 3.7.3.1, respectively.

Westinghouse also addresses the issue related to modifying the piping analyses to consider the potential amplification of motion through the steel frames and platforms (between main structures and piping systems). Specifically, Westinghouse stated in Revision 12 of SSAR Section 3.7.3.8.3 that, when piping or components are decoupled from the analysis of the steel frames or platforms, the analysis accounts for the effect of the steel frame or platforms by developing amplified response spectra from a time history analysis of the steel frames or platforms. These amplified response spectra are used for the seismic analysis of the safety related-items. As an alternative, Westinghouse models the steel frame(s) as part of the supports for the safety-related items. The amplification effect can be excluded if the fundamental frequency of steel frames is in the rigid range. The deformation criteria used for piping supports can also be applied to model these miscellaneous steel structures. The criteria state that if the deflection of the frames attributable to dynamic loading is less than 0.3 cm (0.125 in), the frames are considered to be rigid and the amplification effect through these frames is negligible.

From the discussion above, the staff concludes that Westinghouse's SSAR commitment regarding consideration of the amplification effect in the seismic analysis of AP600 subsystems meets the guideline prescribed by SRP Section 3.7.2. On this basis, Confirmatory Item 3.7.3.3-1 is closed.

3.7.3.4 Analysis Procedure for Damping

Sections 3.7.1 and 3.7.2 of this report discuss the staff's evaluation of damping values assigned to each subsystem, as well as the procedure for calculating composite damping of subsystems.

3.7.3.5 Analysis of Seismic Category I Tanks

In this portion of the review, the staff focused on three SC-I tanks:

(1) The spent fuel pit is a reinforced concrete tank located in the auxiliary building.

(2) The IRWST is an irregularly shaped steel structural module and is located between the steel containment shell and containment internal structures.

(3) The PCCWST is an axisymmetrical reinforced concrete structure located at the top of the shield building.

In the seismic analysis, Westinghouse modeled both the spent fuel pit and the PCCWS tank together with the NI structures, and the IRWST was modeled with the internal structures. Section 3.7.2 of this report discusses the staff's evaluation of the seismic input, modeling procedures and analysis methods that Westinghouse applied for these three tanks, while Sections 3.8.3 and 3.8.4 of this report discuss the tank design.

On the basis of this review, the staff concluded that Westinghouse should indicate in the SSAR that the AP600 design does not include any safety-related flexible wall tanks (field-erected or building-supported) other than the three tanks identified above. The staff therefore designated this observation as Confirmatory Item 3.7.3.5-1.

Westinghouse subsequently fulfilled this requirement in Revision 7 of SSAR Section 3.7.3.16, by stating the AP600 design does not include any SC-I tanks other than the three tanks discussed above. On this basis, Confirmatory Item 3.7.3.5-1 is closed.

3.7.3.6 Conclusion

On the basis of the review and evaluation discussed above, the staff finds that the input motion, treatment of damping, and subsystem analysis methods discussed in Section 3.7.3 of the

SSAR meet the guidelines prescribed in Sections 3.7.1 and 3.7.2 of the SRP and, therefore, are acceptable.

However, because of time constraints, not all changes that the staff requires to be incorporated into SSAR Chapter 3 have been made. Specifically, Westinghouse has agreed to make a change to SSAR Section 3.7.1.4 to clarify that ground water is assumed to be at grade level for the purposes of the analyses done in SSAR Chapter 3 and to make a change to SSAR Section 3.8.4.7 to clarify that there are no other inservice testing or inspection requirements for the seismic Category I shield building and auxiliary building. The incorporation of this information into the SSAR is FSER Confirmatory Item 3.7.3.6-1. Subsequent to the issuance of the advance FSER Westinghouse provided the above information in SSAR Revision 23. Therefore, FSER Confirmatory Item 3.7.3.6-1 is closed.

3.7.4 Seismic Instrumentation

The seismic instrumentation system and the plant response to an earthquake specified for the AP600 plant in SSAR Section 3.7.4 are acceptable and meet the GDC 2 and 10 CFR Part 100 Appendix A guidelines, provided that the changes and additions indicated below are incorporated.

The seismic instrumentation specified for the AP600 plant are as follows:

- one triaxial accelerometer located in the free field, one located on the common basemat of the NI, one located on the shield building structure, and one located on the containment internal structure
- one digital time history recording, analyzer, and playback control system located in a room adjacent to the MCR

Westinghouse has specified that the triaxial acceleration sensors have a frequency range of about 0.5 to 33 Hz. This was not adequate. The triaxial acceleration sensors shall have a dynamic range of 1000:1, and a frequency range (bandwidth) of at least 0.20 Hz to 50 Hz. This was DSER Open Item 3.7.4-1. In Revision 9 of the SSAR, Westinghouse has stated that the triaxial acceleration sensors have a dynamic range of 1000 to 1 (0.0001 to 1.0g) and a frequency range of 0.2 to 50 Hz. This is acceptable, and therefore, DSER Open Item 3.7.4-1 is closed.

Westinghouse has specified that the recording, analyzer, and playback control system shall have a sample rate of at least 200 samples per second and shall be capable of determining both the cumulative absolute velocity (CAV) and the response spectral velocity in the specified frequency range (0.2 - 50 Hz) of the digitized ground motion records. This is acceptable.

Westinghouse has indicated that the system shall have a pre-event memory capable of recording from 1.2 to 15 seconds of the signal before the actuation of the system in case of an earthquake. The SSAR should state that the system will be set to record at least 3 seconds of pre-event signal. This was COL Action Item 3.7.4-1 and DSER Open Item 3.7.4-2. Westinghouse included this requirement in the AP600 SSAR Revision 9, therefore, DSER Open Item 3.7.4-2 is closed.

Westinghouse has specified also that the system shall be capable of recording continuously for at least 25 minutes. Data obtained from such installed seismic instrumentation will be sufficient to ascertain that the seismic assumptions and the analytical models used in the seismic design of the AP-600 system were adequate and that the allowable stresses have not been exceeded under conditions when continuity of operation is intended. This is acceptable.

Westinghouse has specified that the COL applicant is responsible for specifying plant procedures following an earthquake and that the plant procedures following an earthquake are contained in the EPRI reports NP-5930, NP-6695, and TR-100082. These procedures are acceptable provided that the COL applicant includes in its procedures the modifications to the EPRI reports specified by the NRC staff in a letter from J.T. Wiggins of the NRC to J.T. Taylor at EPRI, dated September 13, 1993. With these procedures in place, the system will be capable of accurately determining both the response spectrum and the cumulative absolute velocity of the recorded earthquake ground motion. This data will then yield sufficient information to guide the operator on a timely basis to determine if the level of earthquake ground motion requiring shutdown has been exceeded. This was COL Action Item 3.7.4-2 and DSER Open Item 3.7.4-3. Westinghouse included this requirement in the AP600 SSAR Revision 12. Therefore, DSER Open Item 3.7.4-3 is closed.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

This section is not applicable to the AP600 design.

3.8.2 Steel Containment

Using the guideline prescribed in Section 3.8.2 of the SRP and related RGs, the staff reviewed Revision 0 through 23 of Section 3.8.2 of the Westinghouse AP600 SSAR. In particular, the review under this section focused on the analysis and design of AP600 containment vessel shell structure.

3.8.2.1 Description of Containment

As described in SSAR Section 3.8.2.1 and depicted in SSAR Figures 1.2-12, 1.2-13, and 3.8.2-1, the AP600 containment vessel consists of a thin, cylindrical steel shell with an inner radius of 19.8 m (65 ft) and a wall thickness of 4.13 cm (1.625 inch). The wall thickness increases to 4.44 cm (1.75 in) in the lower course of the cylindrical shell to provide margins for protection in the event of corrosion in the embedment transition region. The top of the cylindrical shell is covered by a smooth ellipsoidal head, and the bottom is enclosed by another ellipsoidal head that is embedded in a concrete foundation below an elevation of 30.5 m (100 ft).

The cylindrical portion of the containment vessel is provided with two T-ring stiffeners, as well as one box-girder stiffener that serves as a crane girder supporting a crane bridge. In addition, the vessel is equipped with two equipment hatches and two personnel airlocks, located as shown in Figure 3.8.2-1 of the SSAR. Other attachments to the vessel include the containment air baffle; walkway; cable trays; HVAC ductwork; concrete on the external stiffeners; other penetrations; and the containment recirculation unit platform.

In reviewing the containment vessel design, the staff found that neither Section 3.8.2 nor Figures 1.2-12, 1.2-13 or 3.8.2-1 of early SSAR revisions provided the radius and thickness of the knuckle region and the dome. In its submittal dated April 28, 1994, Westinghouse addressed this observation stating that the containment vessel head is ellipsoidal with a major diameter of 39.6 m (130 ft), a height of 11.5 m (37.625 ft), and a thickness of 4.13 cm (1.625 in). The staff subsequently requested that Westinghouse include these geometrical properties in the SSAR, because they are important for developing models used in seismic analyses and analyses against combined load conditions. The staff identified this request as Open Item 3.8.2.1-1.

In Revision 3 of SSAR Section 3.8.2.1.1, Westinghouse described both upper and lower containment vessel heads as having the geometrical dimensions stated above. The inclusion of this description of dimensions in the SSAR closes Open Item 3.8.2.1-1.

3.8.2.2 Applicable Codes, Standards, and Specifications

In Sections 3.8.2.2 and 5.2.1.1 of early SSAR revisions, Westinghouse stated that the 1992 Edition of the Boiler and Pressure Vessel Code of the ASME Code, Section III, Subsection NE, "Metal Containment," was used as the basis for the design and construction of the AP600 steel containment vessel. Further, Sections 3.8.2.2 of the SSAR stated that nonpressure parts of steel structures such as ladders, walkways, and handrails were designed to the requirements of the N690 Standard promulgated by the ANSI/AISC N690.

After reviewing this design basis, the staff concluded that use of the 1992 edition of the ASME Code for the design of the AP600 steel containment is not acceptable at this time. Consequently, if the 1992 Edition of the ASME Code is used for the design, Westinghouse should identify the differences between the 1989 and 1992 editions of the ASME Code, and submit an analysis of the differences to the staff for review and acceptance. The staff identified this concern as Open Item 3.8.2.2-1.

Westinghouse subsequently addressed this issue in its submittal dated September 6, 1995, which identified the differences between the 1989 and 1992 editions of the ASME Code. After reviewing Westinghouse's comparison of these two editions, the staff concluded that the changes made in the 1992 Edition are minor and will not affect the design of the AP600 containment vessel. Therefore, Open Item 3.8.2.2-1 is closed. However, any proposed changes to the use of ASME Code (1992 Edition) for the design, fabrication, and construction of the steel containment vessel will require NRC review and approval before implementation of the changes.

Moreover, Westinghouse has committed to use ANSI/AISC N690, modified in accordance with the staff's guideline discussed in Section 3.8.4 of this report for the design of nonpressure parts of steel structures. The staff finds this approach to be acceptable.

3.8.2.3 Loads and Load Combinations

In Table 3.8.2-1 of the SSAR, Westinghouse summarized the loads, load combinations, and ASME service limits applicable to the AP600 containment vessel design. In accordance with the guidelines of Section 3.8.2 of the SRP including the load combinations recommended in Section 3.8.2.II.3.b of the SRP, the staff finds that Table 3.8.2-1 of the early SSAR revisions listed acceptable load combinations for the containment vessel design, with the exception that Westinghouse should resolve the following issues:

- The load combination corresponding to design conditions should include the design external pressure.
- For Level A service limits, the following criteria apply:
  - The design loads should consider the case involving actuation of multiple safety relief valves (SRVs).
  - The external pressure should be included in the case of a LOCA.
  - The design loads should consider the case involving multiple SRV loads with a small or intermediate pipe break accident.
  - For the load combination indicated in the second-to-last column of Table 3.8.2-1 of the SSAR, the external pressure of 17 kPa (2.5 psi) is combined with "T_0" (normal thermal load) and "R_0" (normal reaction load). Westinghouse should clarify whether this external pressure is in combination with the normal plant operating condition or LOCA condition.
- The design should consider appropriate load combinations for Level B service limits.
- For Level C service limits, the following criteria apply:
  - The design should consider the external pressure in the case of a LOCA in combination with the SSE.
  - For the case including a plant operating condition in combination with the SSE, the design should clearly consider the operating pressure associated with T_0 and R_0.
  - The design should consider the load combination related to actuation of multiple SRVs, in combination with a small or intermediate pipe break accident and SSE.
  - For the load combination indicated in the last column of Table 3.8.2-1 of the SSAR, the external pressure of 3.0 psi is combined with "T_0" and "R_0." Westinghouse should clarify whether this external pressure was in combination with load combinations (iii)(c)(1) or (iii)(c)(2) of Section 3.8.2.II of the SRP.

- For Level D service limits, the following criteria apply:
  - The design should consider the external pressure for the case of a LOCA in combination with the SSE and local dynamic loadings.
  - The design should consider the load combination related to actuation of multiple SRV actuation in combination with a small or intermediate pipe break accident, SSE and local dynamic loadings.

To clearly communicate the above concerns regarding the loads and load combinations used for the AP600 containment vessel design, the staff grouped these concerns as Open Item 3.8.2.3-1.

In Revision 7 of SSAR Section 3.8.2, Westinghouse identified the external pressure considered in the design of condition load combinations, including the combination of external pressure with the SSE. Also, in its submittal dated December 18, 1996, Westinghouse justified why the following conditions were considered in the AP600 design:

- In accordance with RG 1.117, the negative external pressure associated with the tornado wind is not postulated concurrent with the LOCA.
- The occurrence of external pressure resulting from a loss of containment heating in extremely cold weather is not included with the LOCA case because such an occurrence is independent from other accidents such as a LOCA.
- The AP600 containment vessel design does not include the load associated with multiple SRVs discharge, because such discharge is not a load case for the PWR containment vessel design.
- The design of the AP600 containment vessel does not consider the loads associated with the ADS, which discharges into the IRWST. This design load is independent from the containment vessel.
- There are no load combinations to be evaluated against Level B service limits for the AP600 nuclear plant.

The staff reviewed the SSAR and Westinghouse's responses, and found that the combined load conditions described in the SSAR (Table 3.8.2-1) meet the guideline prescribed in SRP Section 3.8.2. Therefore, Open Item 3.8.2.3-1 is closed.

3.8.2.4 Design and Analysis Procedures

In Section 3.8.2.4 of the SSAR, Westinghouse stated that the design and analysis procedures used for the steel containment vessel are consistent with the requirements of Section III, Subsection NE, of the ASME Code. Moreover, in performing the analyses, Westinghouse considered various load combinations by performing separate analysis for each individual design load, and combining the stresses obtained according to the required load combinations. An assumption inherent in this process is that individual loads produce linear stresses and the combined response is essentially in a linear state. The staff recognizes that the approach used by Westinghouse is commonly applied in the industry and, thus, is acceptable.

For the evaluation of shell buckling and the determination of buckling margin, Westinghouse used a similar rationale for the linear combination of stresses from individual analyses. The following paragraphs discuss the staff's review of Westinghouse's analysis approach and results for the AP600 containment vessel design.

Analysis of Design Conditions

In Section 3.8.2.4.1 of early SSAR revisions, Westinghouse stated that preliminary analyses were performed using the shell-of-revolution models of the overall containment vessel. These analyses considered the loads that are most significant to the design, such as dead loads, pressure loads, polar crane wheel loads, and seismic loads. In addition, Westinghouse used Fourier harmonics to represent non-axisymmetric loads attributable to earthquake and crane loads.

On the basis of this description, the staff concludes that Westinghouse's consideration of both axisymmetric and non-axisymmetric loads for the design of the AP600 steel containment vessel is consistent with the guideline prescribed in SRP Section 3.8.2 and, therefore, is acceptable. However, the SSAR did not provide the final static and dynamic containment vessel models, the analysis methods used in the analysis and design of the AP600 containment vessel, or the results of the related analyses. The staff designated this omission as Open Item 3.8.2.4-1.

In Revision 3 of the SSAR, Westinghouse addressed this open item by providing containment vessel models (in Figures 3.7.2-5 and 3.8.2-7), as well as the analysis method (in Section 3.8.2.4.1) and the analysis results (in Figure 3.8.2-5 and the containment vessel design report). Inclusion of these information in the SSAR and the design report adequately closes Open Item 3.8.2.4-1.

In evaluating the adequacy of the containment vessel analysis and design, the staff considered the review of the SSAR, the audit of sampled design calculations, and discussions with Westinghouse during the various review meetings as bases for drawing its conclusions. The following paragraphs discuss the staff's review findings regarding the adequacy of the models, analysis methods, and results used in the design of the AP600 containment vessel.

Load Application

During previous design calculation review meetings, the staff found that for the seismic analyses of the containment vessel, Westinghouse used the envelope of peak acceleration profiles obtained from the seismic analyses of NI structures for each design site condition. Specifically, Westinghouse multiplied this envelope by the lumped-masses (lumped at nodal points) to obtain the lateral forces and then applied these forces to the shell as equivalent static forces for the design.

In accordance with the guidelines prescribed in Section 3.7.2 of the SRP, the staff requested that Westinghouse perform a dynamic analysis (response spectrum analysis or time history analysis) of the containment shell to generate seismic stresses for the design. Such an analysis would also show that the stresses on the basis of the equivalent static analysis envelope the stresses obtained from the dynamic analysis. The staff designated this requirement as Open Item 3.8.2.4-2.

During the review meeting held on August 30 and 31, 1995, Westinghouse demonstrated that the results calculated from the equivalent static analysis with input from the acceleration profile (plot of the maximum acceleration at each lumped mass location) are more conservative than those calculated from dynamic analysis. In addition, Westinghouse committed, in SSAR Section 3.8.2, Revision 3, that local analyses are to be performed to assess the responses of local masses using floor response spectra at the appropriate locations in the containment vessel as input motions. The results obtained from the staff's confirmatory analyses substantiate that Westinghouse's demonstration is reasonable and acceptable. Open Item 3.8.2.4-2 is, therefore, closed.

Nevertheless, the staff's review experience with other nuclear power plants suggests that high local stresses may occur in the vicinity of concentrated masses or discontinuities (such as the equipment hatches and personnel air locks). Consequently, the staff requested that Westinghouse demonstrate that stresses in the vicinity of the concentrated masses calculated on the basis of an equivalent static analysis bound the local stresses computed in the dynamic analysis. The staff identified this issue as Open Item 3.8.2.4-3.

During the meeting on August 30 and 31, 1995, Westinghouse stated that detailed analyses and the design of the containment vessel in the vicinity of concentrated masses are beyond the scope of the AP600 standard design. However, Westinghouse agreed to modify the SSAR to provide the analysis procedures. Revision 11 of SSAR Section 3.8.2.4.1.2 fulfill that commitment by providing the following information:

- a detailed description of the methods to be used for the dynamic analysis of local masses
- the approach for analyzing the local buckling potential of the containment shell adjacent to major penetrations
- the criteria for redistributing the stress to be applied to the shell adjacent to local masses
- methods for evaluating the compressive strength of the containment shell in the vicinity of major penetrations

After review this new information, the staff concluded that the SSAR commitment by Westinghouse is acceptable, because the analysis procedures provided are sufficiently detailed for the future design of containment vessel elements adjacent to concentrated masses and are consistent with the industry practice. However, Westinghouse should commit in the SSAR that the design of containment vessel elements adjacent to concentrated masses is an action item for the COL applicant. Therefore, Open Item 3.8.2.4-3 remains unresolved.

In Revision 20 of SSAR Section 3.8.2.4.1.2, Westinghouse added a new paragraph, which stated that the final design of containment vessel elements (reinforcement) adjacent to concentrated masses (penetrations) is completed by the COL applicant and documented in the ASME Code design report. Also, Revision 22 of SSAR Section 3.8.6.1 stated that the final design of containment vessel elements (reinforcement) adjacent to concentrated masses (penetrations) is completed by the COL applicant and documented in the ASME Code design report in accordance with the criteria described in Subsection 3.8.2.4.1.2. Westinghouse's SSAR commitment resolves the staff's concern regarding the design of containment vessel elements adjacent to concentrated masses. On this basis, the staff concludes that Open Item 3.8.2.4-3 is closed. This is COL Action Item 3.8.2-1.

Westinghouse designed the containment vessel with the assumption of uniform internal and external pressures (not including wind). The consideration of these pressures as uniform static loads is acceptable to the staff because the peak internal and external pressures vary slowly with time. The SSAR also specifies the magnitude of design internal pressure loads as 6.89 kPa (1 psi) for the operating condition, 310.26 kPa (45 psig) for the accident condition, and 20.68 kPa (3.0 psid) for the external pressure loads attributable to the loss of containment cooling with extremely low external temperature. Section 6.2 of this report discusses the adequacy of these pressure loads for the AP600 design.

Westinghouse also considered the non-axisymmetric effects of the crane loads (dead loads, lift loads, tangential loads, and radial loads) in the analysis and design of the AP600 containment vessel. To include these loads and to consider their non-axisymmetric effects in the design is common industry practice and, therefore, is acceptable to the staff. However, Westinghouse did not include the eccentric mass effect in the dynamic model when calculating the seismic loads. Section 3.7.2 of this report discusses the resolution of this concern (Open Item 3.7.2.3-8) regarding the omission of crane eccentricity in the seismic analyses. In addition, Westinghouse should consider the effect of crane eccentricity in the seismic stress and buckling analyses. The staff identified this concern as Open Item 3.8.2.4-4.

During the meeting on August 30 and 31, 1995, the staff reviewed the containment vessel design report and found that the revised design calculation included the effect of crane eccentricity and demonstrated that the buckling will not occur under the total stresses attributable to combined load conditions. This conclusion is confirmed by the comparison of stress- resultants from Westinghouse's design report with those from the staff's confirmatory analyses. This comparison shows only insignificant differences between these two sets of results. On these bases, Open Item 3.8.2.4-4 is closed.

During early review meetings, the staff found that Westinghouse assumed a uniform accident temperature of 137.8 °C (280 °F) in the lower portion of the containment. This assumption is consistent with common industry practice and, therefore, is acceptable to the staff. However, Westinghouse's design calculation for the thermal stress and buckling analyses near the base of the containment shell was not available for staff review. The staff identified this concern as Open Item 3.8.2.4-5.

During the meeting on August 30 and 31, 1995, the staff reviewed Westinghouse's calculation of the thermal stress and buckling analyses for the area near the base of the containment shell. As a result of this review, the staff concluded that the method used by Westinghouse meets the guidelines prescribed in Section 3.8.2 of the SRP and yielded reasonable results. This conclusion was confirmed by a comparison of the results obtained by Westinghouse with those obtained by the staff in its confirmatory analysis. (A summary of the confirmatory analysis can be found in Enclosure 2 of a letter to Westinghouse dated April 9, 1998.) On this basis, Open Item 3.8.2.4-5 is closed.

During previous review meetings, the staff found that when non-axisymmetric thermal loads induced by the passive containment cooling process were considered in the design, Westinghouse assumed an alternating strip distribution of temperature corresponding to wet and dry regions around the containment circumference. The wet strips were assumed to be 86.4 cm (34 in) wide with a temperature of 93.3 °C (200 °F), and the dry strips were assumed to be 38.1 cm (15 in) wide with a temperature of 137.8 °C (280 °F). The staff requested that Westinghouse demonstrate the adequacy of this thermal load distribution and designated this request as Open Item 3.8.2.4-6. Section 21.5 (Open Item 21.5.8-2) of this report discusses the staff's review of Westinghouse's subsequent resolution of this concern as well as the staff's conclusion regarding the adequacy of this thermal load distribution. On this basis, Open Item 3.8.2.4-6 is closed.

In early revisions of the SSAR, Westinghouse did not properly define wind loads for the design of the containment vessel. As in previous review meetings, the staff was informed that Westinghouse is in the process of defining the wind load and tornado pressure load for the containment vessel design. The staff identified this outstanding issue as Open Item 3.8.2.4-7.

In Revision 2 of SSAR Section 3.3, Westinghouse responded to this open item by providing the wind and tornado loads for the design of AP600 SC-I structures. Section 3.3 of this report discusses the staff's conclusion regarding the adequacy of the design wind and tornado loads defined in SSAR Section 3.3. On this basis, Open Item 3.8.2.4-7 is closed.

Stress Analysis

Chicago Bridge & Iron (CB&I) (one of Westinghouse's contractors) developed a shell-of-revolution finite difference model for the containment shell and performed stress analyses by using their in-house computer program (E0781B) for various design loads. To model the containment shell as a shell-of-revolution finite difference model is consistent with industry practice and, therefore, is acceptable. However, in order to ensure that the "E0781B" computer code will always generate reasonable results, the staff requested that Westinghouse provide the validation package of this computer code for staff review during early review meetings. The staff identified this request as Open Item 3.8.2.4-8.

At the meeting on August 30 and 31, 1995, the staff reviewed the validation package for the CB&I computer program "E0781B" and found that only relatively simple problems had been tested. The staff questioned whether this computer code was capable of analyzing the complex AP600 containment vessel and stated that a more complicated structural model should be used for the validation.

In order for Westinghouse to complete its validation of the "E0781B" computer code, the staff provided Westinghouse the containment vessel model used in the confirmatory analysis on October 16, 1995, to test the capability of this computer code. In its submittals dated February 12, 1996, and August 23, 1996, Westinghouse presented the containment analysis results for both axisymmetric and asymmetric loading cases, on the basis of the sample problem provided by the staff. The staff reviewed these two submittals and compared the results with those obtained from the staff's confirmatory analysis, and found only insignificant differences between these two sets of results. On this basis, Open Item 3.8.2.4-8 is closed.

Because of the nature of the shell-of-revolution finite difference model and the limitation of the computer code, Westinghouse distributed the mass of the polar crane, major penetrations, and other attached weights around the circumference of the dynamic model used in the seismic stress analysis. In the analyses (as discussed in Section 3.8.2.4.1 of the SSAR), Westinghouse mathematically represented the asymmetric loads such as earthquake loads and crane loads by Fourier harmonics. By contrast, the containment vessel is an axisymmetric structure and is modeled as such. Thus, the staff concluded that the model reasonably represents the containment vessel and applied loads and, therefore, is acceptable. Nonetheless, the envelope of peak acceleration profiles used by Westinghouse, as discussed in Section 3.7.2 of this report, did not include the effect of eccentric masses from major penetrations and polar crane. Also, as discussed in the "Load Application" above, Westinghouse should perform a dynamic analysis instead of an equivalent static analysis to calculate stresses attributable to earthquake loads. The staff's evaluation of the modeling of eccentric masses and the adequacy of using the equivalent static analysis method for the containment vessel is discussed in a later section of this report.

Instead of performing a detailed finite element stress analysis for vessel penetrations, Westinghouse used the area-replacement rule for the penetration reinforcement design, as stated in early SSAR revisions. Such use of the area-replacement rule is consistent with the requirements of the ASME Code (Section III, Division 1, Subsection 3331 through 3335) and results in a conservative design; therefore, the use of this rule is acceptable in tension regions where it is applicable. However, Westinghouse should demonstrate that the area-replacement rule is applicable in the region of concentrated masses, such as the lower equipment hatch and the two personnel airlocks for buckling attributable to compression. In addition, this region is close to the lower spring line and concrete embedment, which may limit the stress redistribution that is implicit in the area-replacement rule. The staff designated these concerns as Open Item 3.8.2.4-9.

In Revision 3 of SSAR Section 3.8.2.4.1.2, Westinghouse responded to this open item by providing the analysis methods used to evaluate the potential buckling of the containment shell in the region of major penetrations (such as equipment hatches and personnel airlocks) and the region close to the lower spring line as a result of compression. Westinghouse also agreed to provide its response to this open item as a part of the resolution for Open Item 3.8.2.4-3. On the basis of the resolution to Open Item 3.8.2.4-3, the staff concludes that Open Item 3.8.2.4-9 is closed.

For small isolated penetrations in compression regions, Westinghouse stated, in its submittal dated December 22, 1992, that penetrations with 80 percent of the required "area-replacement" area have the same buckling strength as the unpenetrated shell. However, Westinghouse did not verify the adequacy of this approach for large or for closely spaced penetrations. (ASME Code Case N-284 limits axisymmetric analyses to shells with penetration diameters less than 10 percent of the containment diameter.) Moreover, Westinghouse should consider penetrations in a compression region as being closely spaced if they are within two buckled wave-lengths. In particular, the region of the lower equipment hatch and the personnel airlocks both have large, closely spaced penetrations. As discussed above, this region is also close to the lower spring line and the concrete embedment. Consequently, the staff requested that Westinghouse perform analyses to show that sufficient compressive strength exists in this region. The staff also identified this request as Open Item 3.8.2.4-10.

During the review meeting held on August 30 and 31, 1995, the staff reviewed samples of Westinghouse's calculations for the containment analysis and shell design. As a result of this review, the staff found that Westinghouse has developed a 3D shell model, which includes the lower portion of the containment shell, the lower equipment hatch, and the lower personnel airlock. On the basis of the assumption that the shell is unpenetrated and the material behavior is linear and elastic, Westinghouse analyzed this model using the ANSYS computer code with both external pressure loading and vertical meridional loading. The results of this analysis showed that the penetrated containment shell has approximately the same compressive strength as the unpenetrated shell.

In addressing the staff's concern regarding the local effect attributable to an SSE, Westinghouse provided the method to be used for evaluating the compressive strength of the containment shell in the region of large or closely spaced penetrations. The staff found that the evaluation method (provided in Revision 3 of SSAR Section 3.8.2.4.1.2) meets the guidelines prescribed in Section 3.8.2 of the SRP. On this basis, Open Item 3.8.2.4-10 is closed.

For the thermal stress analysis, Westinghouse did not provide the analysis results associated with thermal loads for the staff review during the early review meetings. The staff identified this omission as Open Item 3.8.2.4-11. Westinghouse responded to this open item in Revision 3 of the SSAR and the containment vessel design report. In reviewing this new information, the staff found that the shell stress attributable to thermal loads is within the stress limits allowed by the code. The staff's confirmatory analysis results also confirmed this conclusion. Therefore, Open Item 3.8.2.4-11 is closed.

During the early review meetings, Westinghouse neglected to provide the stress analysis reports for all of the combined load conditions for staff review. The staff identified this omission as Open Item 3.8.2.4-12. Westinghouse subsequently responded to this open item during the meeting on August 30 and 31, 1995, by providing the containment vessel design report for all combined load conditions. In reviewing this new information, the staff found that the obtained results met code requirements. Also, the comparison of results obtained by Westinghouse with those obtained from the staff's confirmatory analysis (see Enclosure 3 of a letter to Westinghouse dated April 9, 1998) showed only insignificant differences between these two sets of results. On this basis, Open Item 3.8.2.4-12 is closed.

In its submittal dated April 14, 1994, Westinghouse stated that the flexible seal for the upper air baffle can accommodate the differential deflections of the containment and shield building for all loadings. The staff requested that Westinghouse provide the calculation and magnitude of these relative displacements for staff review. The staff designated this request as Open Item 3.8.2.4-13. To respond to this open item, Westinghouse changed the design concept for this flexible seal, as described in Revision 3 of SSAR Section 3.8.4. Section 3.8.4 of this report discusses the staff's review and conclusions regarding the adequacy of the flexible seal design (Open Item 3.8.4.1-2). On this basis, Open Item 3.8.2.4-13 is closed.

Buckling Evaluation

As described in the submittal dated January 22, 1993, Westinghouse used ASME Code Case N-284 to assess the buckling of the cylindrical portion of the containment remote from the base and penetrations. In addition, Westinghouse used the criteria in Section III, Paragraph NE-3133, of the ASME Code to analyze the vessel head and evaluate buckling under external pressure. (Use of ASME Code Case N-284 and the criteria in Paragraph NE-3133 for the evaluation of containment vessel buckling was previously found acceptable during the staff's review of the System 80+ design.)

In the submittal dated December 22, 1992, Westinghouse claimed that the area-replacement rule, which was used to design the reinforcement for the penetration, can also satisfy stability requirements. However, this is not acceptable, and the staff requested that Westinghouse evaluate the buckling potential in the vicinity of the base and the large penetrations under various load conditions. The staff identified this requirement as Open Item 3.8.2.4-14. Because the concern addressed by this open item is encompassed by the issues raised in Open Items 3.8.2.4-3 and 3.8.2.4-10, Open Item 3.8.2.4-14 is closed.

Westinghouse performed a buckling analysis for the load condition involving the non-axisymmetric temperature distribution. This analysis included a curved panel analysis with peak stress the same as thermal stress, as well as a complete cylinder analysis with an average thermal stress. However, the staff was unable to determine the adequacy of the analysis, which is contingent upon verification of the temperature distribution and definition of suitable boundary conditions. The staff identified this issue as Open Item 3.8.2.4-15. During the review meeting on August 30 and 31, 1995, the staff reviewed the containment vessel design report and found that the definition of boundary conditions is appropriate. Also, the close comparison of results from the confirmatory analysis with those obtained from the thermal stress analysis performed by Westinghouse confirmed that the boundary conditions defined by Westinghouse are acceptable. Therefore, Open Item 3.8.2.4-15 is closed.

During the early review meetings, Westinghouse neglected to provide the buckling analysis calculations for the various combined load conditions, and the staff identified this omission as Open Item 3.8.2.4-16. In the meeting on August 30 and 31, 1995, the staff reviewed the containment vessel design report and compared Westinghouse's results with those obtained from the staff's confirmatory analysis. On this basis, the staff found that the containment vessel possesses adequate margin to prevent buckling as a result of combined load conditions. Open Item 3.8.2.4-16 is closed.

Ultimate Pressure Capacity of the Containment

In the DSER, the staff provided the evaluation of the ultimate pressure capacity of the containment in Section 3.8.2.4 and duplicated the discussion in Section 19.2.6. To prevent a duplicate discussion in the FSER, the evaluation occurs only once and can be found in Section 19.2.6. Because the evaluation of the ultimate pressure capacity in the DSER contained open items and because the evaluation was duplicated, there are two sets of the same open items. Below is a listing of the Section 3.8.2.4 open items and the open item that they correspond to in Section 19.2.6.

DSER open items from Section 3.8.2.4	Corresponding DSER open item in Section 19.2.6	Status of DSER Open Item
3.8.2.4-17	19.2.6.2-1	Closed
3.8.2.4-18	19.2.6.2-2	Closed
3.8.2.4-19	19.2.6.2-3	Closed
3.8.2.4-20	19.2.6.2-4	Closed
3.8.2.4-21	19.2.6.3-1	Closed
3.8.2.4-22	19.2.6.3-2	Closed
3.8.2.4-23	19.2.6.3-3	Closed
3.8.2.4-24	19.2.6.3-4	Closed
3.8.2.4-25	19.2.6.3-5	Closed
3.8.2.4-26	19.2.6.3-6	Closed
3.8.2.4-27	19.2.6.3-7	Closed
3.8.2.4-28	19.2.6.3-9	Closed
3.8.2.4-29	19.2.6.4-1	Closed
3.8.2.4-30	19.2.6.4-3	Closed
3.8.2.4-31	19.2.6.4-4	Closed

In addition, COL Action Item 3.8.2.4-1 corresponds to COL Action Item 19.2.6.4-1. The discussion of this COL action item can be found in Section 19.2.6 of this report. Therefore, COL Action Item 3.8.2.4-1 is dropped.

3.8.2.5 Structural Criteria

As stated in early revisions of SSAR Section 3.8.2.5, the containment vessel is designed, fabricated, installed, and tested according to the requirements of ASME Code, Section III, Subsection NE. The stress intensity limits are according to the ASME Code, Section III, Paragraph NE-3221 and Table NE-3221-1. Critical buckling stresses are checked according to the provisions of ASME Code, Section III, Paragraph NE-3222 or Code Case N-284, Revision 0.

The use of ASME Code, Section III, Subsection NE for evaluating the potential buckling of the AP600 containment vessel meets the guideline prescribed in Section 3.8.2.II.5 of the SRP. In addition, during the staff's review of the System 80+ design, the criteria in ASME Code Case N-284, Revision 0, were previously found acceptable for evaluating containment shell buckling. On this basis, the structural criteria to which Westinghouse committed in early revisions of SSAR Section 3.8.2.5 are acceptable.

However, in Revision 7 of SSAR Section 3.8.2.4.1.1, Westinghouse proposed to use Revision 1 of ASME Code Case N-284 for the evaluation of containment shell buckling. On February 12, 1996, in response to the staff's request, Westinghouse submitted a comparison of Revisions 0 and 1 of Code Case N-284, including its evaluation of the differences between the two revisions. The staff reviewed Revision 1 of ASME Code Case N-284 along with the submittal dated February 12, 1996, and identified a number of errors and typographical errors in Revision 1 of this code case. In its letter dated November 26, 1996, and during the meeting on August 11 through 15, 1997, the staff indicated that the use of Revision 1 of ASME Code Case N-284 is not acceptable for the evaluation of containment shell buckling.

In Revision 17 of SSAR Section 3.8.2 and Appendix 3G to the SSAR, Westinghouse provided criteria for evaluating the local buckling of the AP600 steel containment vessel components such as the ellipsoidal head, cylindrical shell, and equipment hatch covers. These criteria are on the basis of the rules specified in ASME Code Case N-284, Revision 0 with supplemental requirements which are not provided in Revision 0. The supplemental requirements revised some equations (3G.5.2.2, 3G.6.1.1.a, 3G.6.1.1.b, 3G.6.1.3.b, 3G.6.2.1.b, and 3G.6.2.1.c) and added new equations (3G.3.2.1, 3G.3.2.3, 3G.4.1, 3G.4.1.1, 3G.4.1.2, and 3G.4.1.3). However, Westinghouse did not provide any basis to demonstrate that the supplemental requirements are applicable for evaluating potential buckling of containment vessel components. The supplemental requirements rely on an interaction equation that is obtained from the cylindrical shell behavior and not a hemispherical head as applicable to the spring-line region of the AP600 containment vessel. This issue remained unresolved.

In Revision 20 of Appendix 3G to the SSAR, Westinghouse provided bases for the revised equations. The staff reviewed the SSAR, Revision 20 and found that the bases provided by Westinghouse satisfied the theory of shell buckling and therefore are acceptable, except the interaction equation for the biaxial compression condition of spherical shells (Section 3G.6.1.2). Westinghouse contended that this equation was derived from that for the cylindrical shell and was adopted by the code committee in Code Case N-284, Revision 1. The justification provided by Westinghouse is not acceptable to the staff, because there are no theoretical bases to demonstrate the adequacy of this interaction equation.

In Revision 22 of SSAR Section 3G.6.1.2, Westinghouse addressed the staff's concern regarding the buckling criteria for the containment vessel design and stated that the acceptance criteria identified in Items (a) and (b) of SSAR Section 3G.6.1.2 (interaction equations) are used in the design of the containment vessel documented in Subsection 3.8.2. The design resulting from the criteria described in Items (a) and (b) of SSAR Section 3G.6.1.2 has been demonstrated to be adequate for the containment vessel certified design in part on the basis of an independent confirmatory analysis by the NRC staff. Unrestricted use of this equation is not considered acceptable without additional technical justification. Where additional design evaluations are performed by the COL applicant, non-linear buckling analyses shall demonstrate a factor of safety against buckling in accordance with Subsection 3G.2 (safety factor of 2.0 for design conditions, and Level A and B service limits, 1.67 for Level C service limits, and 1.34 for Level D service limits. These factors of safety are consistent with those specified in Code Case N-284). The factor of safety against buckling (ratio of the calculated critical buckling stresses to the membrane compressive stresses because of the specified design loads and load combinations) may be calculated by numerical analyses such as BOSOR-5, ANSYS, or ABAQUS of a portion of the containment vessel with appropriate boundary conditions. The analyses shall consider the effect of geometric imperfections and plasticity. The method and criteria to be used by the COL applicant for the additional design evaluation are documented in Appendix 3G of the SSAR.

The staff's review of Revision 22 to SSAR Section 3G.6.1.2 found that the SSAR commitment by Westinghouse is consistent with the common industry practice and meet the criteria of Code Case N-284. On this basis, the staff concludes that the use of Code Case N-284 (with the supplemental requirements documented in Appendix 3G to the SSAR, Revision 22 for the containment vessel design is acceptable to the staff. However, any proposed changes to the use of ASME Code Case N-284 (including the supplemental requirements documented in Appendix 3G to the SSAR, Revision 22) for the buckling evaluation of the containment vessel will require NRC review and approval prior to implementation.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

According to Section 3.8.2.6 of the SSAR, materials used in the AP600 containment vessel (including the equipment hatches, personnel air locks, penetrations, attachments, and appurtenances) meet the requirements of ASME Code, Section III, Subsection NE-2000. The basic containment material is SA537, Class 2 plate. To provide corrosion protection, the containment vessel is coated with inorganic zinc, except for those portions that are fully embedded in concrete. The inside of the vessel below the operating floor and up to 2.44 m (8 feet) above the operating floor also has a phenolic top coat. Below Elevation of 30.48 m (100 feet), the vessel is fully embedded in concrete, with the exception of the few penetrations at low elevations. Seals are provided inside and outside the vessel so that moisture will not be trapped next to the steel vessel just below the top of the concrete.

Westinghouse also committed (in Section 3.8.2.6 of the SSAR), that the quality control program involving welding procedures, erection tolerances, and nondestructive examination of shop-fabricated and field-fabricated welds conforms with ASME Code, Section III, Subsections NE-4000 and NE-5000.

In addition, as stated in SSAR Sections 3.8.2.5 and 3.8.2.6, Revision 17, the containment vessel is designed, fabricated and tested in accordance with the requirements of ASME Code, Section III, Subsection NE, and will receive a code stamp. The basic material of the containment vessel is SA537, Class 2, plate material, and the material will be impact tested in accordance with the requirements of NE-2000. Meeting the requirements of the ASME Code, Section III, Subsection NE satisfies GDC 51. The staff concludes the design is in compliance with the requirements of GDC 51 because the steel vessel is made of materials that will meet the fracture toughness requirements of ASME Code. This will ensure that the steel containment vessel materials will not undergo brittle fracture and the probability of a rapidly propagating fracture will be minimized.

On the basis of the Westinghouse commitments discussed above, the staff concludes that the materials used in the AP600 containment vessel (including corrosion protection), and the related quality control program meet the guidelines prescribed in SRP Section 3.8.2.6 and, therefore, are acceptable.

With regard to construction techniques, Westinghouse described (in Section 3.8.2.6 of the SSAR) that the containment vessel is designed to allow its construction to use large subassemblies. These subassemblies consist of two heads and three ring sections, which will be assembled in an area near the final location using plates fabricated in a shop facility.

3.8.2.7 Testing and Inservice Inspection Requirements

As stated in Section 3.8.2.7 of the SSAR, Revision 17, testing of the containment vessel and the pipe assemblies which form the pressure boundary within the containment vessel will be according to the provisions of NE-6000 and NC-6000, respectively. The in-service inspection of the AP600 containment vessel will be performed according to the ASME Code, Section XI, Subsection IWE, and is the responsibility of the COL applicant. The SSAR commitments for the structural integrity test and in-service inspection are acceptable. With regard to the leak rate test of the containment system including the containment vessel, the staff's evaluation is discussed in Section 6.2 of this report.

3.8.2.8 Conclusion

On the basis discussed above, the staff concludes that the design of the AP600 steel containment vessel is acceptable and meets the relevant requirements of 10 CFR Part 50 as well as GDC 1, 2, 4, 16, 50, 51 and 53. In particular, this conclusion is on the basis of the following observations:

- By following the guidelines of RG 1.57 and ASME Code, Section III, Subsection NE, Westinghouse has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the steel containment vessel is designed, fabricated, erected, contracted, tested, and inspected to quality standards commensurate with its safety function.

- Westinghouse has met the requirements of GDC 2 by designing the AP600 steel containment vessel to withstand a 0.3g SSE with sufficient margin, and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

- Westinghouse has met the requirements of GDC 4 by ensuring that the design of the AP600 steel containment vessel is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and fluid discharges.

- Westinghouse has met the requirements of GDC 16 by designing the AP600 steel containment vessel so that it essentially provides a leaktight barrier to prevent the uncontrolled release of radioactive effluent to the environment.

- Westinghouse has met the requirements of GDC 50 by designing the AP600 steel containment vessel to accommodate, with sufficient margin, the design leakage rate, calculated pressure, and temperature conditions resulting from postulated accidents, and by ensuring that the design conditions are not exceeded during the full course of the accident. In meeting these design requirements, Westinghouse has followed the recommendations of RG 1.57 and ASME Code, Section III, Subsection NC. Westinghouse has also performed an appropriate analysis, which demonstrates that the ultimate capacity of the containment will not be exceeded and establishes an acceptable margin of safety for the design.

- For the portion of the AP600 containment vessel above Elevation 30.48 m (100 ft), Westinghouse provided access space to perform any necessary periodic inspection of all important areas, as well as surveillance program, and periodic testing. The remainder of the containment vessel is fully embedded in concrete. Therefore, Westinghouse has ensured that this portion of the containment vessel is leaktight and periodic inspection above Elevation 30.48 m (100 ft) would provide the necessary indication of moisture intrusion or evidence of degradation in progress. In addition, as indicated in SSAR Section 3.8.2 (Figure 3.8.2-4), the majority of containment penetrations (both mechanical and electrical) are located above Elevation 30.48 m (100 ft). For those penetrations located below that elevation, Westinghouse provided access (pockets) for testing and inspection from outside the containment vessel. On the basis stated above, Westinghouse has met the requirements of GDC 53.

- The AP600 primary containment is a welded steel vessel fabricated to the requirements of ASME Code, Section III. The ASME Code requires that the vessel materials meet the fracture toughness requirements of Subsection NE-2000. The staff concludes the design is in compliance with the requirements of GDC 51 because the steel vessel is made of materials that will meet the fracture toughness requirements of ASME Code. This will ensure that the steel containment vessel materials will not undergo brittle fracture and the probability of a rapidly propagating fracture will be minimized.

The criteria used in the analysis and design of the AP600 containment vessel, as well as those proposed for its construction, adequately account for anticipated loadings and postulated conditions that may be imposed upon the containment vessel during its service lifetime. These criteria conform with the requirements of ASME Code, Section III, Subsection NE.

In addition, Westinghouse has used these criteria as defined by applicable codes, standards, guides, and specifications regarding the loads and loading combinations; design and analysis procedures; structural acceptance criteria; materials; quality control programs; special construction techniques; and testing and in-service surveillance requirements. Together, these considerations provide reasonable assurance that in the event of winds, tornados, earthquakes and various postulated accidents occurring within and outside the containment, the containment vessel will withstand the specified design conditions without impairment of structural integrity or safety function of limiting the release of radioactive material.

3.8.3 Concrete and Steel Internal Structures of Steel Containment

Using the guideline prescribed in Sections 3.8.3 and 3.8.4 of the SRP and related RGs, the staff reviewed Revisions 0 through 17 of Section 3.8.3 of the Westinghouse AP600 SSAR. In particular, the review under this section focused on the analysis and design of AP600 containment internal structures including modular walls and floors.

3.8.3.1 Description of the Containment Internal Structures

As stated in Section 3.8.3 of the SSAR, the containment internal structures include the concrete and steel structures inside the containment pressure boundary, supports of the RCS, as well as the components and related piping systems and radiation shielding. Specifically, these structures consist of the primary shield wall, reactor cavity, secondary shielding walls, IRWST, refueling cavity walls, operating floor, intermediate floor and steel platforms, and containment vessel support concrete structure.

Most of these concrete and steel containment internal structures are designed using structural modules. At the lower elevations, "L"-type steel modules act as forms for constructing the reinforced concrete base structure. These "L"-type form modules are constructed from steel plates, reinforced by horizontal angles and vertical tee sections. These form modules are left in place following the concrete pour and curing period.

Before Westinghouse submitted Revision 7 of the SSAR, Section 3.8.3.1 and Appendix 3A of earlier SSAR revisions briefly described the form modules. In addition, Figure 3A-1 (Sheets 1 through 10) in Appendix 3A of the SSAR showed the arrangement and layout of the form modules. However, the SSAR did not provide any details concerning the welding, bracing, connections, or other aspects related to construction of form modules. Since the form modules serve only as forms and are not relied upon to take any structural loads during operation of the plant, the staff concurred with Westinghouse that the form modules are not safety-related; thus, the description provided for the form modules is considered sufficient.

In Revision 7 of the SSAR, Westinghouse eliminated Appendix 3A and transferred some of the information into Sections 3.8.3 and 3.8.4. In particular, Westinghouse revised Section 3.8.3.1 to indicate that welded studs or similar embedded steel elements are attached to the steel forms where loads from surface attachments need to be transferred to the concrete fill. During the review meeting held on May 22 and 23, 1996, the staff found that some safety-related components would be supported by these steel form modules. Therefore, the staff concluded that Westinghouse needs to describe these modules in greater detail and should present the design method for review by the staff. The staff documented this conclusion, along with others from the meeting on May 22 and 23, 1996, in a letter to Westinghouse dated July 1, 1996.

In Revision 11 of Sections 3.8.3.1 and 3.8.3.5.6, and SSAR Figure 3.8.3-16 of the SSAR, Westinghouse provided the requested descriptive information including the design criteria, details, and approach for the steel form modules. The staff reviewed this information, and found that the criteria and approach provided in the SSAR are sufficient for the design of these modules and the design approach meets the guideline prescribed in SRP Section 3.8.4. Thus, the information provided in Revision 11 of the SSAR resolved the staff's concern related to the design of the steel form modules.

Above Elevation 29.9 m (98 ft), "M"-type structural steel modules are used for the containment internal wall structures. Revision 7 of the SSAR provided the modified configuration of the structural wall modules. As a result of this modification, the modules consist of two faceplates of 1.27-cm (0.5-in) thick steel plates, positioned, 76.2 cm (30 in.) or 121.9 cm (48 in.) apart, and connected by steel trusses. The primary purpose of the steel trusses is to stiffen and hold the two faceplates during handling, erection, and concrete placement. An array of steel studs are welded to the faceplates in order to connect the plates to the concrete. Following erection, the M modules are filled with concrete. The modules serve as the upper portion of the primary shield wall, refueling cavity walls, secondary shield walls, and a portion of the IRWST tank walls. They also support the operating floor, as well as other steel framing and steel platforms.

As described in SSAR Section 3.8.3.6, all of the structural steel modules are constructed using A36 plates and shapes, except that Nitronic 33 (American Society for Testing and Materials 240, designation S24000, Type XM-29) stainless steel plates are used on the surfaces of modules in contact with water during normal operation or refueling.

In addition, as described in SSAR Sections 3.8.3.1.3 and 3.8.3.6, the M-type structural steel modules are also used as the west wall of the IRWST. The tank wall modules generally consist of stainless steel plates stiffened with structural steel sections in the vertical and horizontal directions. However, on the west side of the IRWST, along the containment wall, the tank wall consists of a stainless steel plate stiffened with structural steel sections in the vertical direction and angles in the horizontal direction.

Structural steel modules are also used for the operating floor at Elevation 41.2 m (135.25 ft). These modules consist of structural tee sections welded to steel plates stiffened by angles. The floor modules are supported by steel girders, with the flange and a portion of the web located above the plate, and steel reinforcing bars are placed above the plate. Following erection of the floor modules, concrete is poured on the modules to create a composite section.

Before Westinghouse submitted Revision 7 of the SSAR, descriptions for all of the structural modules discussed above were provided in Section 3.8.3.1 and Appendix 3A of earlier SSAR revisions. Figures 3A-1 through 3A-6 of the SSAR showed the arrangement, layout, and details of the modules. However, early revisions of the SSAR provided insufficient details regarding the connections among the M-type wall modules, as well as the connections between the M-type modules and other types of modules. In addition, some of the detailed drawings required further evaluation as part of a structural design review to be conducted by the staff for these modules. At that time, the staff designated this issue as Open Item 3.8.3.1-1 because Westinghouse was in the process of developing the connection-related details.

In Revision 7 of the SSAR, Westinghouse eliminated Appendix 3A and transferred some information into SSAR Sections 3.8.3 and 3.8.4. During this process, Westinghouse provided some design details in these SSAR sections. However, the SSAR still lacked the necessary details regarding connections between wall and floor modules and between adjacent wall modules. In addition, the SSAR did not identify the type of welds for the wall connections.

In response to the staff's concerns raised during the January 14 through 16, 1997 meeting, Westinghouse submitted Revision 11 of the SSAR, which provided design-related information including, details for the wall modules and the connections between these modules. Also, SSAR Sections 3.8.3.1.3 and 3.8.3.6.1 were revised to state that the modules are connected by welding adjacent faceplates using full penetration welds so that the weld is at least as strong as the steel faceplates. In addition, Westinghouse added SSAR Figure 3.8.3-17 to show the details of another typical module-to-floor connection. The descriptive information and details included in Revision 11 of the SSAR are acceptable. Nevertheless, the SSAR still lacks the following details:

- the floor plan and connection details at Elevation 32.66 m (107'-2"),

- a statement that the nominal thickness of the steel faceplates in the structural modules is 1.27 cm (0.5 inch),

- correction of the maximum wall thickness of the modules described in SSAR Section 3.8.4.1.2 (this concern is also discussed under Open Items 3.8.3.4-4 and 3.8.4.1-3 of this report).

Previous revisions of SSAR Sections 3.8.3.1.3 and 3.8.4.1.2 stated that the faceplates are 1.27 cm (0.5 inch) thick, but Westinghouse removed this statement from Revision 11. To resolve this staff concern, Westinghouse submitted Revision 12 of SSAR Sections 3.8.3.1.3 and 3.8.4.1.2, which stated that the minimum thickness of the steel faceplates of these modules is 1.27 cm (0.5 inch). Also, the floor plan and connection details at Elevation 32.66 m (107'-2") are provided in Revision 12 of SSAR Figures 3.8.3-7 and 3.8.3-8, respectively. On this basis, Open Item 3.8.3.1-1 is closed.

To support all containment internal structures, the AP600 design includes a containment internal structures basemat, composed of reinforced concrete, which rises from Elevation 21.8 m (71'-6") to Elevation 33.5 m (109'-10"). As shown in Figure 1.2-12 of the SSAR, the AP600 design did not provide any shear studs at the interface between the containment internal structures basemat and the steel containment shell to prevent any potential dynamic instability such as overturning. This is significant because this large, reinforced concrete structure supports the major components from the steel containment shell which is embedded in a concrete foundation mat that underlies the NI structures.

In previous review meetings, the staff raised a concern regarding the potential that the containment internal structures basemat might overturn during a SSE and requested that Westinghouse demonstrate the dynamic stability of this structure. In the submittal dated April 28, 1994, Westinghouse provided its analysis to demonstrate that the containment internal structures basemat will be stable during an SSE and showed that the factor of safety is 2.5 against overturning. This is consistent with the SRP guidelines and is acceptable to the staff. However, Westinghouse did not demonstrate that this structure will not lift up during an SSE. The staff's concern is that any uplifting of the containment internal structures basemat will cause an impact between this structure and the containment shell, and will thereby affect the integrity of safety-related items supported by this structure. The staff therefore identified this concern as Open Item 3.8.3.1-2.

During the review meeting held on April 14 through 18, 1997, Westinghouse provided the analysis report concerning the possibility that the containment internal structures basemat might uplift (Calculation No. 1010-CCC-003). The staff's review of this calculation found that the possibility of the containment internal structures basemat uplifting attributable to the excitation of an SSE is negligible. On this basis, Open Item 3.8.3.1-2 is closed.

3.8.3.2 Applicable Codes, Standards, and Specifications

Section 3.8.3.2 of the SSAR lists the applicable codes, standards, and specifications applicable to the design, materials, fabrication, construction, inspection, and testing of various type of modules. For some modules, the AP600 design conforms to standard N690-1984 promulgated by ANSI and the AISC, which the staff has not officially endorsed. However, the staff has developed an interim technical position which accepts the use of this standard for advanced reactors when supplemented by a number of provisions. The staff provided Westinghouse a copy of that technical position during the meeting on January 21, 1994.

In the submittal dated May 16, 1994, Westinghouse proposed to revise only Section 3.8.4.5 of the SSAR. However, the staff contends that Westinghouse should also revise Section 3.8.3 of the SSAR to reflect the staff's technical position regarding the use of the ANSI/AISC N690-1984 Standard. The staff, therefore, identified this requirement as Open Item 3.8.3.2-1.

Westinghouse subsequently responded to this open item by submitting Revision 3 of the SSAR, which modified Section 3.8.4.5, "Structural Criteria," to include the staff's technical position. For completeness, however, Westinghouse needed to revise Section 3.8.3.5 as well. Consequently, Westinghouse submitted Revision 7 of the SSAR, which modified Sections 3.8.3.2, 3.8.3.5, and 3.8.4.2 to include (by reference to Section 3.8.4.5) the staff's technical position regarding ANSI/AISC N690-1984. Also, Westinghouse modified Sections 3.8.3.2 and 3.8.4.2 of the SSAR to include limitations applicable to this standard. Because the SSAR properly references and commits to be consistent with the staff's technical position on the use of ANSI/AISC N690-1984, Open Item 3.8.3.2-1 is closed.

In Section 3.8.3.2 of early SSAR revisions, Westinghouse stated that the design of the concrete portion of containment internal structures was on the basis of the 1990 Revision of American Concrete Institute (ACI) Code 349-85, "Code Requirements for Nuclear Safety Related Structures." *The use of ACI-349 for the reinforced concrete design is acceptable to the staff*, as discussed in Section 3.8.3 of the SRP. However, the staff has only accepted the 1980 version of ACI-349, with the exception that the staff's position on the design requirements for the steel embedment should be satisfied when referencing Appendix B to this code. The staff believes that if the 1990 revision of ACI-349-85 is used for the design, Westinghouse should first identify the differences between the 1980 version of ACI-349 and the 1990 revision of ACI-349-85, and submit an analysis of these differences to the staff for review and acceptance. The staff, therefore, identified this requirement as Open Item 3.8.3.2-2.

At the meeting with the staff on April 25 through 27, 1995, Westinghouse stated that ACI had previously published journal articles documenting the differences between the 1980 Code and the 1990 Revision to the 1985 Code. ACI 349-85 Code was accepted by the staff during its review of other nuclear reactor licensing applications. Also, it is the staff's understanding that the difference between ACI 349-85 and the 1990 revision to ACI 349-85 is that a new appendix (Appendix B) was added to this revision. The staff's acceptance of Appendix B is discussed in Section 3.8.4.2 (Open Item 3.8.4.2-4) of this report. On the basis discussed above, Open Item 3.8.3.2-2 is closed.

The ACI Code and ANSI/AISC N690 are to be used specifically for the design of reinforced concrete and steel member structures, respectively. However, Westinghouse did not provide the basis and justification to establish the applicability of these industry standards for the design of modular structural elements. The staff, therefore, identified this omission as Open Item 3.8.3.2-3. However, upon further review of the SSAR and design calculations, the staff found that the structural module elements used for the AP600 containment internal structures except the concrete-filled steel M-type modules are comparable to those designed in accordance with the ACI code and ANSI/AISC N690. Therefore, the staff concludes that these standards are applicable for all modules except the M-type modules. Moreover, because the applicability of these standards to M-type modules has been specifically identified as Open Item 3.8.3.2-5, Open Item 3.8.3.2-3 is closed.

In early SSAR revisions, Section 3A.1 of Appendix 3A stated that the structural modules are designed according to the codes and standards previously identified in Section 3.8.2.2.1 of the SSAR. The staff noted, however, that Westinghouse needed to correct the SSAR, because Section 3.8.2.2.1 did not exist. The staff identified this requirement as Open Item 3.8.3.2-4. Westinghouse subsequently responded to this open item by submitting Revision 7 of the SSAR, which eliminated Appendix 3A and incorporated some of the information into Sections 3.8.3 and 3.8.4. As a result of this revision, Westinghouse deleted the incorrect reference to Section 3.8.2.2.1 of the SSAR. On this basis, Open Item 3.8.3.2-4 is closed.

For the concrete-filled steel M-type modules, it is not clear that ANSI/AISC N690 and the ACI-349 Code are directly applicable. For example, AISC/N690 is primarily applicable to steel structures. Although Section Q1.11 of the Standard does cover composite construction, that section states that composite construction shall consist of steel beams or girders supporting a reinforced concrete slab, so interconnected that the beam and slab act together to resist bending. This definition does not cover unreinforced concrete-filled steel shear walls. Similarly, the ACI-349 Code generally covers reinforced concrete structures, not unreinforced concrete-filled steel shear walls. Thus, the staff requested that Westinghouse provide justification to show the applicability of these standards for the design of M-type concrete-filled steel modules. The staff identified this requirement as Open Item 3.8.3.2-5.

Westinghouse subsequently responded to this open item in Revision 7 of the SSAR, by revising the design procedures and acceptance criteria for the concrete-filled steel modules. Specifically, Westinghouse revised Section 3.8.3.5 of the SSAR to specify that concrete-filled wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349 and some supplemental requirements defined in Section 3.8.4.5.1. This approach treats the steel faceplates as reinforcements in conventionally reinforced concrete structures.

The staff reviewed this design method during review meetings held on May 22 and 23, 1996, and January 14 to 16, 1997. As a result of that review, the staff concluded that the revised design approach described in the SSAR, together with Westinghouse design documents (drawings and sample design calculations), demonstrate that ACI-349 is applicable for the design of the concrete-filled steel wall modules. On this basis, Open Item 3.8.3.2-5 is closed. As for the design concepts (such as the assumption of composite action and the connection of the steel faceplates to the concrete using steel studs), the staff's concern will be addressed under Open Item 3.8.3.4-3.

In Revision 3 of Sections 3.8.3.2 and 3.8.4.2 of the SSAR, Westinghouse added a new paragraph related to welding activities. This paragraph stated that the AP600 welding activities for SC-I structural steel, (including building structures; structural modules; cable tray supports and HVAC duct supports) are accomplished in accordance with written procedures and meet the requirements of ANSI/AISC N690. Westinghouse also indicated that the weld acceptance criteria will be as defined in Nuclear Construction Issues Group (NCIG)-01 Standards, Revision 2. In addition, Westinghouse stated that the weld seam of the plates forming part of the IRWST will be examined by liquid penetrant examination and vacuum box examination after fabrication to confirm that the boundary does not leak. Conformance with the above standards, to which Westinghouse committed in the SSAR, will ensure that SC-I steel structures and component supports will perform in service as designed; thus, Westinghouse's welding approach for the AP600 design is acceptable.

A related concern that arose from the staff's review of Revision 7 of the SSAR is whether Westinghouse inspects all SC-I structural steel welds in accordance with ANSI/AISC N690. Westinghouse resolved this concern by submitting Revision 11 of the SSAR Sections 3.8.3.2 and 3.8.4.2, which state that Westinghouse accomplishes all welding and inspection activities for SC-I structural steel in accordance with written procedures meeting the requirements of ANSI/AISC N690. Therefore, the issue related to the welding and inspection of structural steel is considered resolved.

3.8.3.3 Loads and Load Combinations

The loads and load combinations specified for containment internal structures are the same as those for other SC-I structures as described in Section 3.8.4 of the SSAR. These loads and load combinations meet the guidelines of the SRP. Westinghouse did not include wind (W), tornado (W_t), and precipitation (N) loads in the design load combinations, because these loads are not applicable to the structures housed by other structures (such as containment internal structures). Therefore, the loads and load combinations committed by Westinghouse in the SSAR are acceptable. However, Westinghouse neglected the construction-related loads associated with utilization of modular construction methods in the design. Consequently, in a submittal dated May 20, 1994, Westinghouse's proposed revision to Section 3.8.3.6.1 of the SSAR identified Part 2.2 of the ASME Standard NQA-2, 1989 Edition, as the governing standard for the packaging, transportation, receiving, storage, and handling of structures modules. However, the staff raised a concern that the requirements of ASME NQA-2 are more qualitative than quantitative, and the standard has not commonly been applied to massive structural modules, such as the AP600 M-type modules. Consequently, Westinghouse needed to provide a more quantitative definition of construction-related loads for the M-type modules. Moreover, this definition should address the entire construction process, from offsite fabrication to final onsite placement. In general, however, the definition need not address combination with operations-related loads unless a significant residual condition exists which could degrade the in-place structural capacity. The staff identified this issue as Open Item 3.8.3.3-1.

Westinghouse subsequently responded to this open item in revision 7 of the SSAR. Specifically, Section 3.8.3.3.2 discussed the loads on the steel faceplates that result from concrete placement. Also, Section 3.8.3.6 stated that "the structural wall and floor modules are fabricated and erected in accordance with the ANSI/AISC N690 standard. Loads associated with handling and shipping during the fabrication and erection are considered to be normal loads, as described in Section 3.8.4.3.1.1." Section 3.8.3.6.1 discussed the process for transportation, offsite fabrication, onsite assembly, and final placement of the M1 module. In addition, Section 3.8.3.6.3 discussed the process for concrete placement after the module has been lifted into its final position.

In Revision 11 of the SSAR, Westinghouse further modified Section 3.8.3.6.1 to include the loads and load combinations associated with construction related activities. Thus, with the changes in Revisions 7 and 11, the loads and load combinations described in SSAR Section 3.8.3 meet the guidelines of SRP Section 3.8.3. On this basis, Open Item 3.8.3.3-1 is closed.

Another load unique to concrete-filled M-type modules is the hydrostatic pressure against the steel walls, during the onsite concrete pour. This construction-induced stress will remain following the curing of the concrete, and it will act concurrently with all other design loads. Nonetheless, in early revisions of the SSAR, Westinghouse neglected to describe the methods used to consider this hydrostatic pressure. The staff identified this concern as Open Item 3.8.3.3-2.

Westinghouse subsequently responded to this open item in Revision 11 of the SSAR. Specifically, Westinghouse revised Section 3.8.3.3.2 to describe how the structural wall modules would be designed for the hydrostatic pressure loads during concrete placement. That design method utilizes loads determined in accordance with industry common practice, as described in ACI-347. Stresses are kept well below the limit allowed by code, since the faceplate is designed to limit the out-of-plane deflection. In Section 3.8.3.3.2 of the SSAR, Westinghouse also described the basis for considering the stresses in the steel face-plates attributable to the concrete placement as secondary stresses and stated that these stresses need not be combined with other loads after the concrete has hardened.

After reviewing the new information, the staff found that the description provided in SSAR Revision 11 properly evaluated the stresses associated with concrete placement. In addition, the staff found that the design is in accordance with industry engineering practice (ACI-347), and the resulting stresses are below allowable. On this basis, the staff concluded that the approach for considering the hydrostatic pressure loads attributable to concrete placement is acceptable, and Open Item 3.8.3.3-2 is closed.

In the meeting held on July 11 through 14, 1994, however, the staff raised concerns that the design of the IRWST should consider the combination of the load associated with actuation of the ADS and the SSE load. In addition, the staff asserted that Westinghouse should consider the thermal loading in the internal structural steel frame design. These concerns are Open Item 3.8.3.3-3.

Westinghouse subsequently responded to this open item in Revision 7 of the SSAR by revising Section 3.8.3.3.1 to indicate that the ADS loads are combined with those associated with the SSE. Specifically, the dynamic ADS_1 load (load attributable to blowdown of the primary system through spargers) is combined with the SSE loads by the square root of the sum of the squares (SRSS) method. By contrast, the static ADS_2 load (load attributable to heatup of the water in the IRWST) is combined with the SSE by the absolute sum method. This SSAR commitment to combine the ADS loads with those associated with the SSE meets the SRP guidelines and, therefore, is acceptable.

However, during the meeting on April 14 through 18, 1997, the staff reviewed samples of Westinghouse AP600 design calculations for the structural modules. From that review, the staff found that the ADS loads were not combined with the SSE loads as specified in the SSAR. Instead, ADS loads were combined algebraically with other normal loads and then combined with plus/minus SSE loads. (The concern related to combining the ADS loads with other loads including SSE is also discussed under Open Item 3.8.3.4-13.)

The staff also reviewed Westinghouse's consideration of thermal loadings in the internal structural steel frame design. During the review meeting on April 14 through 18, 1997, the staff reviewed samples of design calculations for the structural frames, and found that the stresses under the combined load condition including the thermal load are within the limits allowed by code and, therefore, are acceptable. On this basis, Open Item 3.8.3.3-3 is closed.

In addition, during the meeting on April 25 through 27, 1995, the staff conducted a detailed review related to the hydrodynamic analysis of the IRWST for the ADS actuation, as described in Revision 1 of Appendix 3F to the SSAR. As a result of this review, the staff raised several technical concerns regarding Westinghouse's consideration of the ADS loads in the IRWST design. Westinghouse subsequently indicated that new test results would be incorporated into Appendix 3F and would be included in the analysis.

Despite the assertion, Revision 7 of the SSAR eliminated Appendix 3F and transferred some of the information into Section 3.8.3 of the SSAR. On the basis of the staff's review of the design calculations presented by Westinghouse during the meeting on January 14 through 17, 1997, several of the staff's earlier concerns were resolved. However, concerns remain with regard to the combination method for loads associated with actuation of multiple ADS spargers, as well as the method for the damping treatment, wall pressure distribution, selection of time intervals for the analysis, and the extent of information included in the SSAR. Since these concerns including the combination of ADS loads with other loads are also discussed under Open Item 3.8.3.4-13 below, the staff considers these issues closed.

3.8.3.4 Design and Analysis

In early revisions of the SSAR, Section 3.8.3.4 and Appendix 3A described the design and analysis procedures for structural modules. In Revision 7 of the SSAR, however, Westinghouse eliminated Appendix 3A and transferred some information into SSAR Sections 3.8.3 and 3.8.4. For the development of SSE loads at various locations in the containment internal structures, Section 3.8.3.4 of the SSAR referred to the methods described in Section 3.7.2 of the SSAR. In addition, SSAR Section 3.8.3.4 stated that the SSE loads are derived from the response spectrum analysis of a 3D finite element model representing the containment internal structures in their entirety. This statement was consistent with Section 3.7.2.1.1 of the SSAR.

In the submittals dated April 14, 1994 and May 17, 1994, Westinghouse stated that a 3D lumped mass stick model and the response spectrum analysis method were used to calculate structural member forces. The staff therefore identified Open Item 3.8.3.4-1 to document the inconsistency between the SSAR and Westinghouse's submittals.

In SSAR Revision 2, Westinghouse subsequently added Table 3.7.2-14, which delineates the various structural models and analysis methods employed in the seismic analysis of the NI structures, including the containment internal structures. In this table, Westinghouse stated that, for the generation of design member forces (axial forces, shear forces, and bending moments), the containment internal structures were seismically analyzed using a 3D fixed-base finite element model and response spectrum analysis method. The 3D lumped-mass stick model combined with the response spectrum analysis method is used only for determining the scaling factor associated with SSI. Thus, Table 3.7.2-14 clarified the inconsistencies noted above. (The staff's evaluation of the generation of seismic forces for the design of containment internal structures is further discussed in Section 3.7.2 of this report.)

Moreover, in Revision 11 of the SSAR, Westinghouse added Table 3.8.3-2, which summarizes and explains the analytical method, model, and purpose, as well as the concrete stiffness used for each analysis of the containment internal structures. In this revision, Westinghouse also added Figure 3.8.3-10 (Sheet 2), and revised SSAR Sections 3.8.3.4.1.3, 3.8.3.4.2.2, and 3.8.3.4.3 to clarify the concerns raised by the staff during the review meetings. After reviewing this information, the staff found that Revision 11 of the SSAR provided an adequate description of the analytical model, method, and purpose, as well as the concrete stiffness used for each analysis of the containment internal structures. On this basis, Open Item 3.8.3.4-1 is closed.

As a result of the review of early SSAR revisions and meeting discussions with Westinghouse, the staff raised a number of concerns related to the development of the seismic model of the containment internal structures. According to Westinghouse's submittal dated May 19, 1994, Appendix 3A of the SSAR was to be revised to clarify the seismic modeling of the containment internal structures. Specifically, the revision of the SSAR proposed in the submittal stated that a 3D lumped-mass stick model of the containment internal structures is developed on the basis of the structural properties obtained from the finite element model using 3D shell elements. The equivalent thickness of shell elements and the equivalent modulus of elasticity were derived from the composite axial and bending stiffness computed from the listed equations. However, the staff's review of these equations and description provided with the proposed SSAR revision raised a number of concerns regarding the approach used to develop the model. These issues and their resolution are summarized as follow:

- One equation, labeled "Axial and Shear Stiffness of Module," has only the terms "E" (for Young's modulus) and "A" (for cross-sectional area). This equation would therefore only address the membrane stiffness, and would not adequately consider the shear stiffness. The ratio of the moduli for steel and concrete "n" derived using Young's modulus "E" would not be the same as the modular ratio derived using shear modulus "G" for shear stiffness, because the Poisson's ratios of steel and concrete are different. This discrepancy could be important because the results of the seismic analyses are more sensitive to shear stiffness than axial and bending stiffness. Thus, the staff identified this discrepancy as Open Item 3.8.3.4-2.

  On February 16, 1995, Westinghouse submitted a quantitative evaluation concerning the effect of this approximation on the shear stiffness used for seismic modeling. On the basis of a subsequent discussion with Westinghouse on February 21, 1995, the staff found that the discrepancy associated with using Young's modulus "E" instead of shear modulus "G" is in the range of 1 to 2 percent. The staff therefore concluded that the seismic modeling of internal structures using the Young's modulus "E" is acceptable. On this basis, Open Item 3.8.3.4-2 is closed.

- The equation for bending stiffness is valid only if the steel and concrete truly behave as a composite section. Thus, because the original module design did not include any shear studs to bind together the steel plates and concrete, the staff asserted that Westinghouse needed to demonstrate the adequacy of the design on the basis of the assumption of a composite section. The staff identified this requirement as Open Item 3.8.3.4-3.

In Revision 7 of the SSAR, Westinghouse modified the configuration and design approach for the concrete-filled wall modules. Specifically, the diaphragm plates, which previously connected the two faceplates, were replaced with steel trusses. In addition, the horizontal angles welded to the steel faceplates were replaced by a pattern of steel shear studs. (These shear studs were intended to connect the steel faceplates to the concrete fill, thereby creating composite action of the steel plates and concrete.) This modified design approach treated the faceplates as reinforcing steel, and utilized ACI 349 as the basis for the wall design. This approach is backed-up by a series of tests performed in Japan and has been common practice in the industry, and therefore is acceptable provided that the pattern/location and design of the studs is properly performed. However, in reviewing the sampled design calculations for the studs during the meeting on January 14 to 16, 1997, the staff found that Westinghouse failed to demonstrate the adequacy of the stud design criteria, which are needed to ensure composite action.

Westinghouse addressed the staff's concerns related to the design criteria for the shear studs in its letter dated April 10, 1997. Specifically, that letter described the method used to design the shear studs, which is on the basis of the requirements of ANSI/AISC N690 for composite construction with concrete slabs on steel beams. The criterion used in ANSI/AISC N690 for full composite behavior is that the strength of the shear connectors over the length of the beam from the point of maximum moment to the point of zero moment is greater than the yield strength of the steel beam. This letter also described the approach for considering in-plane loadings which need to be transferred between the steel faceplates and the concrete core.

During the meeting at Westinghouse on April 14 through 18, 1997, The staff raised a concern that the design method described in the submittal dated April 10, 1997, did not consider the simultaneous application of in-plane and out-of-plane loads acting on the shear studs. To address this concern, Westinghouse presented preliminary design calculations for review, during the review meeting on April 14 through 18, 1997. In these calculations, Westinghouse expanded its design method for the shear studs by considering in-plane and out-of-plane loads being applied simultaneously. Westinghouse also demonstrated that, for the design of carbon steel faceplates, the existing design of shear studs meets the limits allowed by code. However, for the stainless steel faceplates used for the modules in contact with water, the staff noted that Westinghouse needed to consider the combined action of shear studs and the attached steel angles to meet the code allowances.

Nonetheless, because the shear stud design approach described above follows ANSI/AISC N690 criteria and considers the simultaneous application of the other in-plane and out-of-plane loads, the staff found the design method acceptable, with the exception that Westinghouse needed to finalize the design calculation for the staff review. Also, Westinghouse needed to modify the design calculation for the studs attached to the stainless steel plates so that it would consider a concrete strength of 27.58 MPa (4,000 psi) instead of 34.47 MPa (5,000 psi) (to determine the angle capacity). In addition, Westinghouse needed to modify the SSAR description of the trusses to reflect that the trusses are also used to develop shear load transfer between the steel faceplates and the concrete core. Therefore, Open Item 3.8.3.4-3 remained unresolved.

During a meeting with Westinghouse on January 20 to 21, 1998, the staff reviewed the final design calculation (Calculation No. 1100-SUC-003, Revision 1, entitled "Structural Modules, Containment Shear Studs, General Design") for the shear studs. As stated above, this calculation followed the approach specified in ANSI/AISC N690 and considered the application of the in-plane and out-of-plane loads, and thus is considered acceptable.

However, the staff raised a concern regarding the design of concrete anchorages that the calculation indicated that any margin in the stud capacity could be used to take additional loads from attached equipment. The calculation used methods described in Revision 20 of SSAR Section 3.8.4.5.1 which stated that the design of fasteners to concrete is in accordance with ACI 349-90, Appendix B with supplementary criteria on the basis of three other references (References 46, 47 and 48 of SSAR Section 3.8.4). This SSAR section also stated that anchors are designed wherever possible with sufficient depth of embedment and side cover such that the steel anchor yields before failure of the concrete. The staff's concern is that the above criteria permit Westinghouse to design fasteners to concrete, such that the concrete fails before the steel yielding (i.e. non-ductile behavior), when the embedment depth and side cover are not sufficient to ensure that the steel anchor will not yield before the concrete. The method described in ACI 349 for design of embedded anchors for non-ductile behavior is not acceptable to the staff. In addition, no specific criteria are presented in the SSAR for the design of this type of behavior.

In response to this staff concern, Westinghouse submitted Revision 22 of SSAR Section 3.8.4.5.1 for review. The following changes were made to Revision 22 of the SSAR:

(a) Eliminate the use of three references mentioned above by deleting the second and third paragraphs from this section.

(b) Rewrite the first sentence of the second paragraph as, "design of fastening to concrete is in accordance with ACI 349-90, Appendix B with supplementary criteria described below."

(c) Retain six bullets of criteria in Revision 7 of SSAR Section 3.8.4.5.1.

(d) Add the sentence, "for those cases where steel fastener yield cannot be demonstrated to occur prior to concrete failure, the design strength is established using a minimum factor of safety of 4.0 between the fastener design load and the fastener ultimate capacity determined from static load tests which simulate the actual conditions of installation."

The staff's review found that revised criteria for steel fasteners meet the staff technical position for Appendix B to ACI 349 and the committed factor of safety of 4.0 for steel fasteners against non-ductile failure meets the NRC IE Bulletin 79-02 criteria for concrete expansion anchors. On this basis, the staff concludes that the concern regarding the criteria used for the design of steel fasteners is technically resolved and, therefore, Open Item 3.8.3.4-3 is closed.

- The equation for bending stiffness included an approximation used in calculating the moment of inertia. (It assumed the thickness of the steel faceplates relative to the concrete is very small.) However, when the bending moment of inertia was calculated, Westinghouse did not properly consider the thickness of the steel plates. This approximation may be applicable for the containment internal structural wall modules such as the M-type modules. However, for other wall modules (such as the modules in the auxiliary building), this assumption may lead to inaccurate results. The staff identified this assumption as Open Item 3.8.3.4-4.

  At the meeting on December 6 through 8, 1994, Westinghouse indicated that the minimum module wall thickness to be used in the auxiliary building is 76.2 cm (30 inches). For this wall thickness, the approximation for seismic modeling introduces an error on the order of 1 percent which the staff considers acceptable.

  Specifically, in Revision 13 of SSAR Section 3.8.4.1.2, Westinghouse indicated that the thickness of the structural wall modules in the auxiliary building ranges from 76.2 cm (30 inches) to 1.52 m (60 inches). Because a minimum wall thickness of 76.2 cm (30 inches) is used for the design of wall modules, the approximation for not considering the faceplate thickness when calculating the bending stiffness of wall modules as described in SSAR Section 3.8.3.4.1.1 is acceptable. On this basis, Open Item 3.8.3.4-4 is closed.

- The behavior of the concrete is 3D in view of the wall thicknesses of the 76.2 cm (30 inches) and 121.9 cm (48 inches) specified in SSAR Section 3.8.3. With this 3D behavior, interaction effects at contact faces between concrete and steel may generate non-negligible through-thickness normal stresses. Deformation compatibility is enforced only at discrete locations (such as the horizontal angle stiffeners). For these reasons and the design details shown the SSAR, it was not clear to the staff whether the equations presented are adequate to develop appropriate equivalent properties for the isotropic shell model. In addition, the staff noted that Westinghouse should demonstrate that the assumptions made are realistic to represent the 3D behavior of the basic concrete-filled steel module. Further, the staff asserted that a local 3D solid model of the module geometry and materials should be used as the basis for developing equivalent isotropic shell properties or for justifying the equations currently used. The staff identified these requirements as Open Item 3.8.3.4-5.

  In Revision 7 of the SSAR, Westinghouse revised the configuration and design approach for the concrete-filled steel wall modules. According to that approach, welded steel studs are used to connect the steel faceplates to the concrete fill to ensure composite action of the wall sections. With this arrangement, the design method then treats the wall section as a reinforced concrete wall. In addition, Section 3.8.3.4 described the analysis procedure for determining the wall stiffness, and compared the calculated results to the available test data. For static loads, the finite element analyses were completed with the assumption of monolithic (uncracked) stiffness of each concrete element. For thermal and dynamic loads, the analyses considered the extent of concrete cracking. The wall stiffness was therefore established on the basis of analyses of the wall behavior and review of the test data related to concrete-filled steel structural modules. Some of the test data included concrete-filled steel wall sections with studs, which are similar to the Westinghouse AP600 configuration. Because the method used to calculate properties of wall modules has been verified and confirmed with actual test data, Open Item 3.8.3.4-5 is closed.

- During discussions at various review meetings, the staff raised concerns regarding the stiffness degradation attributable to cracks in the concrete fill, ductility, and margins of the modules. On the basis of early SSAR revisions and meetings with Westinghouse, the staff found that the justification provided for the resolution of these concerns relied primarily on tests performed in Japan.

Specifically, justification for the seismic modeling of internal structural modules (including the effect of concrete cracks) was provided in the submittal dated May 19, 1994. For that justification, Westinghouse primarily relied upon a few tests conducted in Japan on concrete-filled steel wall structures. In addition, Westinghouse provided comparisons to demonstrate similarities between AP600 modules and test samples. The referenced tests performed in Japan appeared to demonstrate that the use of concrete-filled steel modules results in a better design, compared to conventional reinforced concrete structures. However, the staff noted that a number of differences between the tested configurations and the AP600 modules (e.g., studs versus horizontal angles, tie rods between the two face plates, application of only shear or axial loads versus multiple loads, etc.).

One of the Japanese tests actually demonstrated that, in compression, the initial stiffness is approximately 80 percent of the calculated stiffness. Also, the staff noted that the referenced tests were only performed for one load at a time, either compression or shear. The M-type modules, however, would be subjected to biaxial bending, shear, and compression or tension. In addition, the limited information included in the published technical papers for these tests is insufficient to support generic conclusions. Further review by the staff and discussions with Westinghouse were needed to determine the possible resolution of this issue. Items relating to the seismic modeling of the containment internal structures were collectively identified as Open Item 3.8.3.4-6.

In Revision 7 of the SSAR, Westinghouse revised the configuration of the concrete-filled wall steel modules documented in early revisions of the SSAR. As a result, the calculated results more closely matched the available test data. The test data showed that concrete-filled steel specimens experience less overall stiffness degradation than reinforced concrete sections. The test data also demonstrated that the concrete-filled steel test specimens possess substantial ductility and ultimate capacity. Thus, the use of concrete-filled steel wall modules in place of reinforced concrete walls is acceptable.

Westinghouse also addressed how cracks in concrete-fill affect the seismic model of the Containment internal structures. Specifically, in Revision 7 of the SSAR (Section 3.8.3.4.1.2 and Table 3.8.3-1), Westinghouse stated that the in-plane shear stiffness was calculated on the basis of a 45-degree diagonal concrete compression strut with tensile loads carried by the steel plates. Thus, the calculated stiffness was considerably lower than the test data described in SSAR References 27 and 28, where the overall stiffness was reduced to 60 to 70 percent of the monolithic stiffness. If the calculated stiffness were used for the boundaries of the IRWST, the equivalent shear area of the containment internal structures would be reduced by about 30 percent, with a corresponding reduction in frequency of about 16 percent. The staff reviewed this SSAR revision, and found that the floor response spectra generated in the containment internal structures on the basis of calculated stiffness are not acceptable for the following two reasons:

(1) As shown in Revision 7 of SSAR Figure 3.7.1-7 and Table 3.7.2-3, the first dominant frequency of the internal structures in the north-south direction is 13.6 Hz, and the corresponding ground spectral acceleration is ±0.63g. If the first dominant frequency were reduced from ±13.6 Hz to 11.42 Hz (16 percent), the corresponding ground spectral acceleration would be increased to ±0.72g. Westinghouse did not consider this increased ground spectral acceleration attributable to concrete cracks when calculating the floor response spectra for the containment internal structures.

(2) In following the guideline of RG 1.122, Westinghouse developed the final floor response spectra by applying the ±15 percent peak broadening rule to the enveloped floor response spectra. This would cover the uncertainties associated with material properties of structures and soil, SSI techniques, and approximations in the modeling techniques. However, the ±15 percent peak broadening cannot cover the uncertainties associated with the cracked concrete in the structural modules.

In conclusion, Westinghouse should either regenerate the floor response spectra for the containment internal structures on the basis of the seismic model for which concrete cracks are considered, or justify the adequacy of the floor response spectra documented in the SSAR. On the basis discussed above, the commitment stated in Revision 7 of the SSAR is not acceptable, and Open Item 3.8.3.4-6 remained unresolved.

In a letter dated April 8, 1997 and during the meeting on April 14 through 18, 1997, Westinghouse provided additional information on the effect of concrete cracks on the calculation of in-structure response spectra. Westinghouse contended during the meeting that Revision 7 of SSAR Section 3.8.3.4 used the calculated stiffness of "Case 3" as a conservative estimate of the lower bound in-plane shear stiffness of the structural modules. This case assumed that the concrete in tension has no stiffness. For in-plane stiffness, a 45-degree diagonal concrete compression strut is assumed with tensile loads carried only by the steel plate. Westinghouse also stated that the in-plane stiffness calculated by these assumptions are much lower than the stiffness measured in the tests of similar construction with in-plane loads. Revision 11 of SSAR Section 3.8.3.4 discussed the results of a test program which demonstrated that, for similar concrete-filled steel wall sections, the stiffness degradation is about one-half the stiffness degradation of reinforced concrete walls.

To develop its justification, Westinghouse performed new analyses to include the consideration of the aggregate interlock effect for the cracked concrete. A new stiffness value was then calculated in which Westinghouse considered the effects of aggregate interlock across the preexisting cracks (associated with the PRHR thermal event). In the previous calculation, the stiffening effect associated with the aggregate interlock was conservatively neglected. The reduction in stiffness on the basis of this new analysis is only 30 percent.

The handouts provided to the staff also described tests performed on reinforced concrete panels which were precracked under biaxial tension and was then cycled with pure shear loading. The stiffness reductions from the tests were about 12 percent.

Westinghouse also indicated in the meeting that the reductions in stiffness described above occur only in the boundary walls of the IRWST. The net stiffness reduction on all module walls inside containment are even lower, because cracks will not occur for those walls that are not in touch with the IRWST water. According to Westinghouse, using the 30 percent reduction on the basis of the analysis results, the effect of the stiffness reduction on the frequency in the more critical North-South direction is 6.2 percent. Therefore, Westinghouse concluded that ±15 percent peak broadening of the floor response spectra is sufficient to cover the uncertainties because of the frequency reduction caused by concrete cracking.

On the basis of the new analyses which indicates a 6.2 percent frequency reduction, the test data for reinforced concrete, and the test data on concrete filled steel modules which show improvement over reinforced concrete walls, the staff concurred with Westinghouse's justification and concludes that Open Item 3.8.3.4-6 is closed.

The damping ratio is an important parameter in the seismic analysis of the concrete-filled steel structural modules. The draft revision of Section 3A.8.4 of the SSAR contained in the submittal dated May 19, 1994, stated that a damping ratio of 7 percent is used. Although the response refers to one of the tests in Japan to justify the use of 7-percent damping, the results of that test actually indicate a damping of 5 percent. The staff identified this discrepancy as Open Item 3.8.3.4-7. Westinghouse subsequently responded to this open item in Revision 2 of the SSAR, by modifying SSAR Table 3.7.1-1, “Safe Shutdown Earthquake Damping Values.” Specifically, for concrete-filled steel plate structures, the modified table specified a damping value of 5 percent. The use of 5-percent damping for steel modules is consistent with test results and meets the guideline of RG 1.61. Consequently, Open Item 3.8.3.4-7 is closed.

In early revisions of the SSAR, Appendix 3A presented details of the methods and procedures used in designing the AP600 structural modules inside containment. This appendix stated that the modules with concrete fill were designed with minimal reliance on the concrete fill for strength. Instead, such modules were generally designed as steel structures, in accordance with the requirements of ANSI/AISC N690. In a few cases where credit is taken for the concrete, Appendix 3A stated that ACI 349 Code was used.

Section 3A.3.1 of the early SSAR revisions described the design procedures used for the wall modules. For in-plane loads under axial compression, the design assumed that the compressive loads are distributed to the concrete and steel plates in proportion to the stiffness of the concrete and steel. However, the design of the wall modules allowed buckling of the steel plates between the horizontal stiffeners over a portion of the plates between the vertical diaphragm webs. This approach led to a number of questions that have not yet been addressed. Above all, after the steel plates buckle, the load will completely shift to the concrete. Thus, the staff requested that Westinghouse demonstrate the integrity of the concrete in the wall systems. The staff identified this request as Open Item 3.8.3.4-8.

In Revision 7 of the SSAR, Westinghouse revised the configuration and design approach for the concrete-filled wall modules. In the modified approach, welded steel studs were used to connect the steel faceplates to the concrete to ensure composite action of the wall section. The staff noted, however, that because Westinghouse changed the module configuration and design approach, and eliminated Section 3A from the early SSAR revisions, this concern no longer applies. On this basis, Open Item 3.8.3.4-8 is closed.

Another concern that Westinghouse needed to address is the effect of interaction between the vertical compressive stresses and the other perpendicular in-plane horizontal stresses and shear stresses. The post buckling theory used to calculate an effective width of the steel plates did not consider these other stress components. This was identified as Open Item 3.8.3.4-9.

In Revision 7 of the SSAR, Westinghouse revised the configuration and design approach for the concrete-filled steel wall modules. In the modified approach, welded steel studs were used to connect the steel faceplates to the concrete to ensure composite action of the wall section. The staff noted, however, that because Westinghouse changed the module configuration, and eliminated Section 3A from the SSAR (which previously permitted buckling of the steel faceplates between the horizontal embedded angles), this concern no longer applies. On this basis, Open Item 3.8.3.4-9 is closed.

As described in Section 3A.3.1.2.2 of early SSAR revisions, the diaphragm web plates with the two face plates form a vertical box section. Because they provide the major structural steel strength in this direction, the walls are designed to span in the vertical direction. Thus, out-of-plane loads causing out-of-plane moments are only resisted by one-way action of the wall. The out-of-plane moments about the vertical axis are stated to be secondary. Consequently, the staff noted that Westinghouse needed to justify the adequacy of this assumption because the moment of inertia about the vertical axis does not appear to be much smaller than the moment of inertia about the horizontal axis. Westinghouse also needed to verify the presumed one-way action of walls because the horizontal span of the walls is comparable to the height of the walls. If biaxial bending is required, Westinghouse would also need to revise the combined stress equations in Section 3A.3.1.3 of the SSAR to reflect realistic action of the walls. The staff identified these requirements as Open Item 3.8.3.4-10.

In Revision 7 of the SSAR, Westinghouse revised the configuration and design approach for the concrete-filled steel wall modules. Specifically, these modules were designed as reinforced concrete structures, in accordance with the requirements of ACI-349 and some supplemental requirements defined in SSAR Section 3.8.4.5.1. The new design of the wall modules considered two-way action to resist out-of-plane loadings. Furthermore, Revision 11 of SSAR Sections 3.8.3.5.3.2 and 3.8.3.5.3.3 stated that the wall sections to resist in-plane and out-of-plane loads were designed in accordance with ACI-349. Because the design of the concrete-filled steel wall modules considered two-way action and complied with ACI-349, Open Item 3.8.3.4-10 is closed.

One of the critical areas in designing structures from individual modules is the connection between modules. The submittal dated May 17, 1994 referred to the detailed drawings presented in the SSAR for the various joints and connections. However, these drawings did not provide any details for the welds between adjacent wall modules and at the intersection of modular walls. These connection-related design details should be completed and reviewed by the staff. This omission was identified as Open Item 3.8.3.4-11.

At the meeting on January 14 through 16, 1997, the staff reviewed samples of calculations related to the design of the module connections. Also, during the meeting on April 14 through 18, 1997, the staff reviewed portions of the design calculation involving the connection between the floor module and the IRWST wall, as well as the connection between the wall module and the concrete base. As a result of these reviews, the staff identified an issue regarding the calculation related to the floor-to-wall connection design. Because the design assumed a hinged boundary for this connection, it failed to consider the bending moments from the IRWST walls. This assumption is not consistent with the boundary condition used in the analyses of the IRWST walls.

In resolving this concern, Westinghouse committed to ensure the connection calculation to show that the bending moments from the wall analyses are to be satisfied. Alternatively, Westinghouse will evaluate the effect on the analysis results assuming a pinned connection. Resolution of the design issues involving module connections will be covered under Open Item 3.8.3.4-13. Therefore, Open Item 3.8.3.4-11 is closed.

As stated in the SRP, the staff should review a design report to obtain design and construction information that is more specific than that contained in the SSAR. The design report can also assist the staff in planning and conducting a structural review. Nonetheless, Westinghouse was unable to provide a design report for the containment internal structures during previous review meetings. Thus, in the submittal dated June 30, 1994, Westinghouse committed to compile design summary reports using the format and attributes described in Appendix C to Section 3.8.4 of the SRP. In addition, these design summary reports would incorporate the criteria acceptable to the staff and would be made available for staff review. This commitment was identified as Open Item 3.8.3.4-12.

During the meeting on January 14 through 16, 1997, the staff reviewed samples of the draft design summary reports for the structural modules, including "Design Summary Report - Containment Internal Structures," No. 1100-S3R-001 (Draft), dated January 1997; and Design Summary Report - Auxiliary Building Structures," No. 1200-S3R-001, Revision 0 (Draft), dated January 1997. These draft summary reports described the components of the structural modules, structural loads, structural analysis and design, and results. Because the information included in the design summary reports is sufficiently detailed and the scope of these reports is in accordance with that described in Appendix C to Section 3.8.4 of the SRP, this issue is considered technically resolved.

In completing its response to this concern, Westinghouse presented two design reports for the staff's review during the meeting on April 14 through 18, 1997. However, neither of these reports was finalized. Westinghouse indicated that further information will be added to the design reports, such as the drawing details for the critical sections of structural wall modules, stress/load/required steel area summary tables for the critical wall sections, and the comparison tables for the ADS pressure loading analyses. Therefore, Open Item 3.8.3.4-12 remained unresolved.

Westinghouse informed the staff that the above-mentioned Design Reports are internal Westinghouse documents and would not be finalized because it is a "living document." Additional information will be added in the future to incorporate changes or additions as required. Therefore, Westinghouse decided to include some of the information contained in the design reports in the SSAR. In particular, design summaries of critical sections for structural modules were provided in a draft SSAR submittal.

During the meeting with Westinghouse on January 20 to 21, 1998, the staff reviewed the proposed revisions to Section 3.8.3.5.7 and a new Section 3.8.3.5.8. Section 3.8.3.5.8 summarized the design of three critical structural wall modules inside containment and the in-containment refueling water storage tank steel wall. The three structural wall modules consist of the southwest wall of the refueling cavity, south wall of the west steam generator cavity, and northeast wall of the in-containment refueling water storage tank. The information in the draft SSAR adequately described the configuration, analysis methods, design procedures, results (loads, stresses, and required steel area), and comparisons to allowables. Westinghouse also demonstrated that the resulted stresses are within code allowances. However, the staff identified that some of the design loads presented in SSAR Table 3.8.3-6 were different from those values contained in the corresponding Calculation No. 1100-SUC-101, Revision 6. Westinghouse needs to evaluate the discrepancy and correct the table.

Similar information was also provided for structural modules in the auxiliary building. The design summary information was included in a new Appendix 3H of the SSAR. During the meeting on January 20 to 21, 1998, the new Appendix 3H, which was in draft form, was reviewed. It presented the design summary information for two critical modules in the auxiliary building, the west wall of the spent fuel pool and the finned floor modules in the MCR and the instrumentation and control room. The design information in Appendix 3H adequately described the structural configurations, analysis methods, design procedures, results, and comparisons to allowables, and is in conformance to the SRP guideline and design code requirements. Therefore, they are acceptable.

In Revision 20 of SSAR Table 3.8.3-6, Westinghouse corrected the design loads that were different from those in Calculation No. 1100-SUC-101, Revision 6. On the basis of discussion above, the staff concludes that Open Item 3.8.3.4-12 is closed.

A structural design review is also required for the containment internal structures, particularly because of their unique design details and modular construction techniques. The objectives of the review are threefold. First, the staff will investigate the way the structural design criteria were implemented. Second, the staff will attempt to verify that the key structural design calculations have been performed in an acceptable way. Finally, the staff will identify and assess the safety significance of particular areas where the containment internal structures were designed and analyzed using methods not covered by the SRP guidelines. Thus, the need for a structural design review was identified as Open Item 3.8.3.4-13.

During the meeting on January 14 through 16, 1997, the staff reviewed samples of design calculations for the structural modules. In general, these calculations and analyses did adhere to the commitments stated in the SSAR. However, the implementation details were sometimes difficult to follow, several inconsistencies were noted, and several reports did not appear to be final. In addition, Westinghouse needed to justify or revise the calculation regarding the shear studs (see the discussion concerning Open Item 3.8.3.4-3), and needed to finalize design calculations for the evaluation of ADS loads.

In addressing the concerns described above, Westinghouse presented its final design calculations for structural modules during the meeting on April 14 through 18, 1997. The staff then reviewed the following samples of design calculations:

- Calculation No. 1100-SUC-101, "Structural Wall Modules - Containment Internal Structures," Revision 2
- Calculation No, 1200-SUC-101, "Structural Module in Areas 5 and 6 - Auxiliary Building," Revision 2
- Calculation No. 1150-SMC-101, "Framing Design of Operating Deck at El. 135.25' - Containment Internal Structures," Revision 1
- Calculation No. MT03-S3C-012 and 018, "Hydrodynamic and Pressure Analysis of IRWST"
- Calculation No. 1100-SUC-003, "Structural Modules - Design of Shear Studs" Revision 0

As a result of its review, the staff found that Westinghouse's design procedures specified in the SSAR were not properly implemented in the design calculations. Thus, the staff requested that Westinghouse conduct its own review of design calculations and finalize these design calculations for the staff review. Therefore, Open Item 3.8.3.4-13 remained unresolved.

During the meeting with Westinghouse on January 20 to 21, 1998, the staff reviewed four samples of design calculations. These sampled design calculations are as follows:

(1) Calculation 1100-SUC-101, Revision 6

(2) Calculation 1100-SUC-003, Revision 1

(3) Calculation 1200-SUC-101, Revision 4

(4) Calculation GW-SUP-003, Revision 2

The staff's review found that these calculations have properly implemented the design procedures specified in the SSAR. On this basis, Open Item 3.8.3.4-13 is closed.

3.8.3.5 Acceptance Criteria

Section 3.8.3.5 of the SSAR addresses the general acceptance criteria for the containment internal structures. In addition, the SSAR references ACI-349 for reinforced concrete components and concrete-filled steel wall modules, as well as ANSI/AISC N690 for steel components. Allowable stresses for each load combination are presented in Tables 3.8.4-1 and 3.8.4-2 for steel and concrete structures, respectively. The submittal dated May 17, 1994 provides supplemental acceptance criteria for inclusion in Section 3.8.4 of the SSAR consistent with the staff's position. (See Section 3.8.3.2 above.) The staff noted, however, that these supplemental acceptance criteria should also be included or referenced in Section 3.8.3.5 of the

SSAR, and identified this omission as Open Item 3.8.3.5-1. (This issue is also addressed in Open Item 3.8.3.2-1, where the emphasis is on the status of ANSI/AISC N690 as an applicable standard.)

In SSAR Revision 3, Westinghouse updated Section 3.8.4.5 to include specific supplemental criteria, in accordance with the staff's technical position. However, Westinghouse also needed to update Section 3.8.3.5 to cover the same issue for containment internal structures. Sections 3.8.3.2 and 3.8.4.2 should also include references to acceptance criteria consistent with the staff's technical position.

In Revision 7 of the SSAR, Westinghouse revised Sections 3.8.3.2, 3.8.3.5, and 3.8.4.2 to include (by reference to Section 3.8.4.5) the staff's technical position on the application of ANSI/AISC N690. Because the revised SSAR properly references and commits to follow the staff's technical position on the use of the ANSI/AISC N690 Standard, Open Item 3.8.3.5-1 is closed.

In early revisions, Appendix 3A provided additional guidance on the allowable stresses for wall and floor modules. However, in Revision 7 of the SSAR, Westinghouse transferred this information to Sections 3.8.3 and 3.8.4. Regardless of the change in location, these acceptance criteria remain appropriate and acceptable, provided that Westinghouse resolves the related open items described above.

In addition, the staff noted that Westinghouse should develop acceptance criteria for loads and deformations related to the fabrication, shipping, and construction/erection of the modules. This includes static loads attributable to lifting, handling, tie-down, fit-up, and other operations, as well as dynamic loads such as vibration and impact loads attributable to railway shipment. (Vibration loads should be specified to ensure that they do not contribute to fatigue usage; otherwise, these additional cyclic loads need to be included in the design fatigue analysis.) Excess deformation may also arise beyond the dimensional tolerances accounted for in the design analysis. These distortions might be developed during the fabrication, handling, shipping, storage, and/or fit-up at the time of assembling the modules. The staff, therefore, asserted that the SSAR should describe the additional acceptance criteria needed to address these loads and deformations during the fabrication, shipping, and construction/erection of the modules. This was identified as Open Item 3.8.3.5-2.

At the meeting on December 6 through 8, 1994, Westinghouse committed to revise the SSAR to provide additional information pertaining to acceptance criteria for construction-related loads. In Revision 7 of SSAR Section 3.8.3.6, Westinghouse stated "The structural wall and floor modules are fabricated and erected in accordance with ANSI/AISC N690. Loads during fabrication and erection due to handling and shipping are considered as normal loads as described in SSAR Section 3.8.4.3.1.1." On the basis of this SSAR commitment, Open Item 3.8.3.5-2 is closed.

On the basis discussed above, the staff concludes that the use of ANSI/AISC N690-1984 (with the supplemental requirements documented in SSAR Section 3.8.4.5.2, Revision 22) and ACI 349-90 (with the supplemental requirements documented in SSAR Section 3.8.4.5.1, Revision 22) for the design, fabrication, and construction of the seismic Category I structural modules (both inside and outside the containment vessel) is acceptable to the staff. However, any proposed change to the use of ANSI/AISC N690-1984 and ACI 349-90 including the supplemental requirements documented in SSAR Sections 3.8.4.5.1 and 3.8.4.5.2 (Revision 22) would require NRC review and approval before implementation.

3.8.3.6 Materials, Quality Control, and Construction Techniques

Section 3.8.3.6 of the SSAR referred to Section 3.8.4.6 of the SSAR for the materials and quality control program used in the construction of the containment internal structures. Section 3.8.4.6 of the SSAR described the concrete ingredients and the reinforcing steel (presumably used to anchor the modules). Additional information related to the materials not covered in Section 3.8.4.6 of the SSAR was presented in Sections 3A.5 and 3A.7 of the SSAR. Specifically, Section 3A.5 of early SSAR revisions described the sleeve used to attach the reinforcing steel to the modules. Section 3A.7 stated that the structural steel modules were designed using A36 plates and shapes, and Nitronic 33 (ASTM 240, designation S24000, Type XM-29) stainless steel plates were used on the surfaces of the modules that come in contact with water during normal operation or refueling. In Revision 7 of the SSAR, Westinghouse eliminated Appendix 3A and transferred the information to Sections 3.8.3.6 and 3.8.4.6. These two SSAR sections, along with SSAR Tables 3.8.4-3 to 3.8.4-6, provide the material descriptions for the concrete, structural steel plates and shapes, and reinforcing steel.

As indicated above, the SSAR references Section 3.8.4.6 for the description of the quality control program. The staff's evaluation of this program is discussed in Section 3.8.4 of this report.

Section 3.8.3.6.1 of the SSAR covers special construction techniques. The submittal dated May 20, 1994, proposed to revise this section of the SSAR. The draft revision stated that the use of concrete-filled steel structures is a proven construction method that has been used in the nuclear industry for years. However, most of the cited examples are not comparable to the concrete-filled steel modules for AP600 containment internal structures. In addition, because the SSAR did not fully describe the AP600 construction techniques, the staff was unable to draw a direct comparison. This was identified as Open Item 3.8.3.6-1.

At the meeting on December 6 through 8, 1994, Westinghouse committed to revise the SSAR to provide additional information pertaining to modular construction techniques. In Revision 7 of SSAR Sections 3.8.3.6.1, 3.8.3.6.2, and 3.8.3.6.3, Westinghouse fulfilled this commitment by providing more detailed information pertaining to the special construction techniques to be used for modules in general and also for the concrete-filled steel "M1" module. The information provided is consistent with the techniques commonly used in the industry and is considered sufficient to close Open Item 3.8.3.6-1.

In addition, the submittal dated May 17, 1994, provided information regarding the placement and curing of the concrete inside the M-type modules. Because the steel plates will remain (unlike wood forms), and in view of the height of the walls, the in-place concrete pour is a special process. Thus, the SSAR should specify the process used and steps taken to ensure that voids (especially adjacent to the bottom face of horizontal stiffeners) will not occur. The fitup and joining procedures for onsite assembly of modular units are also special processes that should be described in the SSAR.

The submittal dated May 17, 1994, also referred to a construction plan, which the staff still needs to review. The issue regarding the construction techniques was identified as Open Item 3.8.3.6-2. At the meeting on December 6 through 8, 1994, Westinghouse agreed to provide additional information (in Revision 7 of the SSAR) pertaining to special construction processes such as onsite fitup/joining and in-place concrete pour.

In Revision 7 of SSAR Section 3.8.3.6, Westinghouse stated that steel structural walls and floor modules are fabricated and erected in accordance with ANSI/AISC N690. Also, in SSAR Section 3.8.3.6.3, Westinghouse discussed the concrete placement process for the "M1" module. The concern regarding the determination of concrete placement loads consistent with the pour rate is covered under Open Item 3.8.3.3-2. Therefore, Open Item 3.8.3.6-2 is closed.

3.8.3.7 Conclusion

On the basis discussed above, the staff concludes that the design of the containment internal structures is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDCs 1, 2, 4, and 50. In particular, the staff reached this conclusion on the basis of the following factors:

- By following the guidelines of the RGs and industry standards, Westinghouse has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the containment internal structures are designed, fabricated, erected, constructed, tested, inspected to quality standards commensurate with their safety function.

- Westinghouse has met the requirements of GDC 2 by designing the AP600 containment internal structures to withstand the 0.3g SSE with sufficient margin and the combinations of the effects of normal and accident conditions with effects of environmental loadings such as earthquakes and other natural phenomena.

- Westinghouse has met the requirements of GDC 4 by ensuring that the design of the containment internal structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and fluid discharges.

- Westinghouse has met the requirements of GDC 50 by designing the containment internal structures to accommodate, with sufficient margin, the calculated design leakage rate, pressure and temperature conditions resulting from postulated accidents, and by ensuring that the design conditions are not exceeded during the full course of the accident. In meeting these design requirements, Westinghouse has followed the recommendations of the RGs and industry standards. Westinghouse has also performed an appropriate analysis, which demonstrates that the ultimate capacity of the structures will not be exceeded and establishes an acceptable margin of safety for the design.

The criteria used in the analysis and design of the AP600 containment internal structures, as well as those proposed for their construction adequately account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime.

These criteria conform with established codes, standards, and specifications acceptable to the staff, including RGs 1.15, 1.55, 1.57, 1.94 and 1.142, and the following industry standards:

- ACI-349, "Code Requirements for Nuclear Safety Related Structures"
- ASME Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containments
- ASME Boiler Pressure Vessel Code, Section III, Subsection NE
- ANSI/AISC N690, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities"
- ANSI N45.2.5

In addition, Westinghouse has used these criteria, as defined by applicable codes, standards, and specifications regarding the loads and load combinations; design and analysis procedures; structural acceptance criteria; materials; quality control programs; and special construction techniques; and the testing and in-service surveillance requirements. Together, these considerations provide reasonable assurance that the containment internal structures will withstand the specified design conditions without losing their structural integrity or the performance of required safety functions in the event of earthquakes and various postulated accidents.

Furthermore, the staff's conclusion regarding the design of the containment internal structures is based on its review of the samples of design calculations for the critical sections of the internal structures described in Revision 22 of SSAR Section 3.8.3.5.8, "Design Summary of Critical Sections." Therefore, any proposed change to the text of SSAR Section 3.8.3.5.8, Revision 22 will require NRC review and approval before implementation of the change.

3.8.4 Other Category I Structures

In addition to the containment vessel and containment internal structures, the NI structures include the shield building and auxiliary building. In early revisions to the AP600 SSAR, Figures 1.2-4 through 1.2-13 showed detailed floor plans and cross-sections of these buildings. However, the Westinghouse floor plans provided only the dimensions between column lines, and did not include any key dimensions. The staff therefore asserted that the SSAR should provide such building dimensions as the size (thickness and overall dimensions) of the foundation mat, overall dimensions and wall thickness of the auxiliary building, radius and wall thickness of the shield building, geometry (radii and wall thickness) of the shield building roof structures including the PCCWS tank, geometry of containment internal structures (structural modules), and thickness of major structural walls. The staff identified this omission as Open Item 3.8.4-1.

Section 3.7.1 of this report discusses the staff's review of Westinghouse's resolution of this issue (Open Item 3.7.1-5). On this basis, Open Item 3.8.4-1 is closed.

3.8.4.1 Description of the Structures

In addition to the containment vessel and containment internal structures, the AP600 SC-I structures include the shield building (including roof structures and PCCWS tank structures), auxiliary building (including structural modules and support of SC-I raceway systems), and containment air baffle.

Shield Building and Passive Containment Cooling Water Storage Tank

As described in Section 3.8.4 of the SSAR, the shield building is a SC-I cylindrical reinforced concrete structure. The layout of this structure and its interface with other SC-I building structures is shown in Section 1.2 of the SSAR. Major features of the shield building including building's cylindrical shell structure; roof structure; lower, middle and upper annulus areas; air inlet; PCCWS tank; air diffuser; air baffle; and air inlet plenum. The cylindrical shell of the shield building supports the roof structure. In addition, the floor slabs and structural walls of the auxiliary building are structurally connected to the cylindrical shell at various elevations of the shield building. By contrast, the roof structure of the shield building consists of a conical roof, air inlet columns, and tension and compression ring beams. In the configuration shown in SSAR Figure 3.8.4-7, this reinforced concrete roof shell structure supports the PCCWS tank and air diffuser.

Section 3.8.4.1.1 of early SSAR revisions stated that a permanent flexible water- and air-tight seal is provided between the shield building and the concrete floor slab at Elevation 40.3 m (132.25 ft). This seal provides an environmental barrier between the upper and middle annulus sections of the shield building, and is therefore classified as a safety-related item. However, in its submittal dated June 27, 1994, Westinghouse stated that the seal also maintains an intact holdup volume within the middle annulus for containment leakage of contaminants following a severe accident scenario. The seal is therefore designated as a non-safety-related and non-seismic feature, because the AP600 design does not rely on this seal to mitigate any design-basis events. However, the submittal also stated that the seal is designed to accommodate events as a result of normal operation, as well as design-basis and severe accident scenarios. Open Item 3.8.4.1-1 addressed the staff's concern regarding the inconsistency between the classification of this seal in early SSAR revisions and in Westinghouse's submittal dated June 27, 1994. In Revision 3 of SSAR Section 3.8.4.1.1, Westinghouse clarified that this seal has been reclassified as a non-safety-related, non-seismic feature, because the AP600 design does not rely on this seal to mitigate design-basis events. Therefore, Open Item 3.8.4.1-1 is closed.

In response to the staff's request during the previous review meetings, Westinghouse provided a description and detailed geometry of the original design of the PCCWS tank in Revision 14 of SSAR Section 3.8.4.1.1 and Figure 3.8.4-7. Recently, because of the requirements for the post-72-hour actions (as described in Section 3.7.2.3 of this report), Westinghouse increased the water volume of the PCCWS tank from 1703.44 m^3 (450,000 gallons) to 2089.55 m^3 (552,000 gallons). As a result, Westinghouse modified the design of the PCCWS tank in five significant ways:

(1) Increase the tank water level from Elevation 90.83 m (298 ft) to 91.90 m (301.5 ft).

(2) Raise the top of the tank by 0.3 m (1.0 ft).

(3) Reduce the thickness of the inner tank wall from 60.96 cm (24 inches) to 45.72 cm (18 inches).

(4) Decrease the thickness of the tank roof from 60.96 cm (24 inches) to 38.10 cm (15 inches).

(5) Place the PCCWS tank floor liner directly on the structural concrete and eliminate the 10.16 cm (4 inches) grout.

Figure 3.8.4-4 in Revision 15 of the SSAR illustrates the revised geometry and configuration of the PCCWS tank.

In Revision 14 of SSAR Section 3.8.4.1.1, Westinghouse stated that the PCCWS tank has a stainless steel liner that provides a leaktight barrier on the inside surface of the tank. The wall liner consists of a plate with stiffeners on the concrete side of the plate, and the floor liner is welded to steel plates embedded in the top surface of the concrete. To ensure leaktightness, the liner is welded and inspected during construction of the tank. Any leakage that might occur would be collected at the base of the cylinder walls. This arrangement permits the monitoring of the tank for leakage and also prevents degradation of the reinforced concrete wall as a result of the freezing and thawing of leakage.

Section 3.8.4.4 of this report discusses the adequacy of the PCCWS tank liner design and the related design changes associated with the post-72-hour action requirements.

Auxiliary Building

The SC-I auxiliary building structure composed of reinforced concrete is supported on the common NI foundation mat. The building has a total of five stories, including three stories above ground and two stories located below grade. The floor slabs and the structural walls of this building are structurally connected to the cylindrical portion of the shield building.

SSAR Figure 3.7.2-12 illustrates the details of the auxiliary building. The major structures of the auxiliary building include the MCR, spent fuel pool, fuel transfer canal, new fuel storage area, cask loading and wash down pit, and 1334.47-KNewton (150-ton) cask handling crane. The walls and floors of the spent fuel pool, fuel transfer canal and cask loading and wash down pit are lined (on the inside) with stainless steel plate to prevent corrosion and leakage. The new fuel storage area is a separate reinforced concrete pit. Structural modules are used in the design of AP600 spent fuel pool, fuel transfer canal, and cask loading and washdown pits. The structural details of steel modules are discussed below.

Containment Air Baffle

The containment air baffle, part of the Passive Containment Cooling System (PCS), is located inside the shield building and is primarily supported by the containment vessel. A series of thin metal panels are used to construct a shell, which surrounds the containment vessel. The air baffle separates the downward air flow entering the air inlets from the upward air flow that cools the containment vessel and flows out of the discharge stack located at the top of the shield building.

The air baffle is a SC-I structure designed to withstand the wind and tornado loads defined in Section 3.3 of the SSAR as well as the seismic loads. The air baffle panels are also designed to accommodate displacements between individual panels, which might occur as a result of containment pressure and thermal growth.

The detailed description of the air baffle, including its function, is provided in Section 3.8.4.1.3 of the SSAR, and the detailed configuration is illustrated in Figure 3.8.4-1 of the SSAR. At the meeting on April 25 through 27, 1995, Westinghouse presented a new design for the air baffle and the attachments to the containment vessel. This updated design was described in SSAR Revision 3.

Section 3.8.4.1.3 of early SSAR revisions indicated that a flexible connection exists between the shield building wall and the air baffle panels fixed to the containment vessel. This connection was designed to accommodate the differential movements between the containment vessel and the shield building. In response to the staff's request, Westinghouse stated that this flexible connection would permit significant differential movement between structures. However, Westinghouse did not specify the magnitude of the calculated relative displacements (in radial, tangential, and meridian directions as a result of seismic, thermal, and pressure loadings). Thus, the staff requested that Westinghouse compare the calculated displacement with the manufacturer-specified safe capacity of this flexible connection against, crimping, fatigue life, and stretching. This request was identified as Open Item 3.8.4.1-2.

Revision 3 of SSAR Section 3.8.4.1 describes a major modification of the sealing mechanism. Specifically, Westinghouse replaced the flexible connection with a steel sliding plate, as shown in Sheet 4 of Figure 3.8.4-1 in Revision 7 of the SSAR. The staff subsequently evaluated the new sealing mechanism during the meetings on August 30 through 31, 1995, and May 22 through 23, 1996. In Revision 20 of SSAR Section 3.8.4.1.3, Westinghouse stated that the sliding plate is set at ambient conditions to permit relative movements from minus 2 inches to plus 3 inches radially and minus 1 inch to plus 4 inches vertically. This accommodates the differential between the containment vessel and the shield building, based on the absolute sum of the containment pressure and temperature deflections and of the seismic deflections. Also, SSAR Figure 3.8.4-1 (Sheet 4) shows that the sliding plate can move freely in the tangential direction. From the meeting discussions and the review of the SSAR, the staff found that the sliding plate mechanism can accommodate the relative displacements in radial, tangential and vertical directions as a result of seismic, thermal, and pressure loadings. On the basis of this finding, the staff concluded that the issue regarding the potential failure of the flexible seal connection has been adequately resolved by the design change. Consequently, Open Item 3.8.4.1-2 is closed.

Supports of SC-I Raceway Systems

The SC-I raceway systems include the SC-I cable tray system and ductwork associated with the HVAC system. As indicated in Sections 3.8.4.1.4 and 3.8.4.1.5 of the SSAR, the cable tray systems are supported by channel-type struts fabricated from cold-rolled channel-type sections. The supports for HVAC ductwork systems consist of either structural steel members or cold-rolled channel-type sections. These supports are attached to the walls, floors, and ceiling of structures, as required by the arrangement of the raceway systems. The specific spacing of the supports is determined by the allowable loads and stresses of the raceways and supports, and the design includes longitudinal and transverse bracing where required.

Structural Modules

In early SSAR revisions, Section 3.8.4.1.2 described the structural modules used in the auxiliary building. Specifically, the structural wall and floor modules of the auxiliary building are located at the south side of the building, extending from Elevation 20.3 m (66.5 ft) to Elevation 41.2 m (135.25 ft). These modules include the spent fuel pool, fuel transfer canal, and cask loading/wash down pit. (Figure 3.8.4-5 of the SSAR illustrated the locations of the modules.) The structural modules are built up with steel structural shapes and plates. In addition, concrete fill is used where required for shielding, but reinforcing steel is not normally used.

After reviewing early revisions of the SSAR, the staff was not certain whether the details of the auxiliary building modules are the same as the details of M-type wall modules that are used inside containment. Consequently, the staff noted that Westinghouse should revise the SSAR to provide more details for these modules, and should indicate any difference between these modules and those located inside the containment vessel. The staff identified this request as Open Item 3.8.4.1-3.

In Revision 12 of SSAR Section 3.8.4.1.2, Westinghouse stated that the configuration and typical details of structural modules in the auxiliary building are the same as for the M-type structural modules used for the containment internal structures described in SSAR Section 3.8.3.1. The overall thickness of the walls range from a minimum of 0.76 m (2.5 ft) to 1.52 m (5.0 ft), and the minimum thickness of the faceplates is 1.27 cm (0.5 inch). In reviewing this SSAR revision, the staff found that the description of structural modules located in the auxiliary building is sufficient for the staff to make a safety determination and, therefore, is acceptable. On this basis, Open Item 3.8.4.1-3 is closed.

Other AP600 structural modules include finned floor modules used for the ceiling of the MCR (floor at Elevation 41.22 m [135.25 ft]) and the ceiling of the instrumentation and control room (floor at Elevation 35.81 m [117.5 ft]). As shown in Figure 3.8.4-6 in Revision 3 of the SSAR, the finned floor modules consist of a 61-cm (24-inch) thick concrete slab poured over a stiffened steel plate ceiling. The fins are rectangular steel plates welded perpendicularly to the bottom of the ceiling plate. In addition, shear studs are welded to the top of the ceiling plate to ensure that the concrete slab and steel ceiling plate behave as a composite section. Several shop-fabricated steel panels, placed side-by-side, are used to construct the stiffened plate ceiling in a modular fashion. The stiffened plate is designed to withstand construction loads before concrete hardening. As a result of its review, the staff found that the SSAR provided sufficient information regarding the finned floor modules in the SSAR.

3.8.4.2 Applicable Codes, Standards, and Specifications

Early revisions (before Revision 2) of SSAR Section 3.8.4.2 provided a partial list of codes and standards used as the basis for the design of the AP600 NI structures. This SSAR section also stated that Westinghouse used nationally recognized industry standards as the basis for specifying material properties, testing procedures, fabrication, and construction methods. In particular, Westinghouse relied upon standards promulgated by the ASTM, ACI, and AISC for the design of SC-I structures. In Sections 3.8.4.2 and 3.8.4.4 of the SSAR, Westinghouse stated that the design procedures for the SC-I structures other than the containment vessel and the containment internal structures are in accordance with the 1990 version of the ACI-349

Code for reinforced concrete structures, and the 1984 version of the N690 standard for steel structures promulgated by the ANSI and AISC. In addition, the allowable stresses for cable trays and strut supports are derived from the provisions of the AISI, and the ductwork and its supports are designed according to the AISI provisions and the ANSI/AISC N690, respectively.

In reviewing Section 3.8.4.2 of early SSAR revisions, the staff identified several issues as follows:

- The SSAR should provide a complete list of the codes and standards that were used in the AP600 design, together with where these codes and standards were used. This requirement was identified as Open Item 3.8.4.2-1.

  In response to this open item, Westinghouse submitted Revision 3 of SSAR Sections 3.8.3 and 3.8.4, which identified all codes and standards used in the design of the AP600 NI structures. This revision also clarified where and why each code and standard applied to various parts of the NI structures. The staff's subsequent review of this information revealed that the codes and standards specified for the AP600 design are consistent with those endorsed in Section 3.8.4 of the SRP and RGs. On this basis, Open Item 3.8.4.2-1 is closed.

- In the submittal dated May 17, 1994, Westinghouse proposed to revise the SSAR to conform with the staff's position on the application of the ANSI/AISC N690 (Appendix G to NUREG-1503, Volume 1, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design - Main Report," dated July 1994). The staff's review of Westinghouse's response found that the proposed SSAR revision did address the staff's position and, thus, the use of ANSI/AISC N690 is acceptable. The staff committed to review Revision 3 of the SSAR to ensure that the SSAR incorporates the proposed changes. The staff identified this validation effort as Confirmatory Item 3.8.4.2-1.

  In Revision 3 of SSAR Section 3.8.4.5, Westinghouse incorporated the staff's position on the application of the ANSI/AISC N690. Therefore, Confirmatory Item 3.8.4.2-1 is closed.

- Early revisions of the SSAR indicated that Westinghouse used the ACI-349-90 Code in the design of reinforced concrete structures. However, this use of the ACI-349-90 Code is not entirely acceptable at this time, because the staff has only approved the use of the 1980 version of the ACI-349 Code (Section 3.8.4 of NUREG-1503).

  However, as indicated in the meeting on June 12 through 16, 1995, the staff evaluated the use of the ACI-349-85 Code during its review of the System 80+ standard plant design developed by ABB/CE. On the basis of that evaluation, the staff concluded that use of the ACI-349-85 Code is acceptable for the design of AP600 SC-I reinforced concrete structures, with the exception that the staff's position on the design requirements for steel embedments should be satisfied. However, the staff has never endorsed the use of the 1990 version of the ACI 349-85 Code for the design of the SC-I structures. Therefore, if the 1990 version of the ACI-349-85 Code is used in the AP600 design, Westinghouse should identify the differences between the 1980 and 1990 versions of the ACI-349 Code, and submit an analysis of the differences to the staff for review and acceptance. The staff identified the requirement of using the 1990 version of the ACI-349-85 Code as Open Item 3.8.4.2-2.

From meeting discussions with Westinghouse on April 25 through 27, 1995 and June 12 through 16, 1995, the staff found that the differences between 1980 and 1985 editions of the ACI 349 code are insignificant, with the exception that Appendix B to the code was added to the code. The staff also found that the changes made in the ACI 349-90 code were primarily to make Appendix B requirements consistent with the test results. Therefore, the differences between 1980 and 1990 editions of the ACI 349 code are also insignificant, except Appendix B to the code. (With regard to the concern of using Appendix B to ACI 349 for the design of steel embedment, the staff's evaluation is discussed under Open Item 3.8.4.2-4.) In addition, the staff accepted the use of ACI 349-85 during the review of the ABB/CE System 80+ standard plant design. On the basis discussed above, Open Item 3.8.4.2-2 is closed.

- The AP600 SSAR did not describe how Westinghouse considered the ductility criteria of ACI-318 in the design of reinforced concrete structures. The staff identified this omission as Open Item 3.8.4.2-3.

  In Revision 11 of SSAR Section 3.8.4.4.1, Westinghouse committed to use the ductility criteria of ACI-318 (Chapter 21) in detailing, placing, anchoring, and splicing the reinforcing steel. The revised SSAR sufficiently described how Westinghouse will use the ductility criteria from the ACI Code in design of AP600 SC-I reinforced concrete structures. (The staff's evaluation of the application of ACI 318 ductility criteria in the design of the NI structures is discussed under Open Item 3.8.4.4-3 below.) Thus, the revised SSAR satisfies the staff's concern, and Open Item 3.8.4.2-3 is closed.

- For the design of steel embedments, the staff has not yet accepted the criteria of Appendix B to the ACI-349 Code. In its submittal dated May 17, 1994, Westinghouse indicated conformance to the staff's position on ACI-349 provisions for the design of steel embedments (Appendix B to the ACI-349 Code) would be addressed by April 1995 (when the ACI-349 Code Committee's decision on the staff position became available). This was not acceptable to the staff, because the use of Appendix B to ACI-349 for the design of steel embedments may lead to a non-conservative result. The staff identified this issue as Open Item 3.8.4.2-4.

  In Revision 7 of the SSAR, Westinghouse responded to this open item by adding Section 3.8.4.5.1, "Supplemental Requirements for Concrete Structures." That addition included guidelines for the use of Appendix B to ACI-349 for the design of embedded steel elements. In reviewing this revision of the SSAR, the staff found that these supplemental criteria meet the staff's technical position as documented in Appendix F to NUREG-1503 (Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design). On this basis, Open Item 3.8.4.2-4 is closed.

3.8.4.3 Loads and Load Combinations

Section 3.8.4.3.1 of the SSAR provided the definition for each individual load corresponding to cases of normal loads, severe environmental loads, extreme environmental loads, and abnormal loads. Section 3.8.4.3.2 of the SSAR defines specific combined load conditions for the AP600 design.

In reviewing early SSAR revisions, the staff found that definitions of the design loads and load combinations met the guidelines prescribed in Section 3.8.4 of the SRP and, therefore, are acceptable. However, the staff raised several issues related to design loads and load combinations as follows:

- The civil/structure design criteria submitted by Westinghouse on May 2, 1994, provide the definition of the maximum live load and operating live load. In addition, the submittal stated that the operating live load is the only live load to be considered in the seismic analysis. The criteria also stated that, for the AP600 NI structures, 25 percent of the maximum live load shall be used to represent the operating live load portion to be included in the seismic load for local member design. This later criterion is not acceptable. Instead, the staff asserted that Westinghouse should include the 25 percent of maximum live load in the dynamic model and perform seismic analysis to calculate seismic responses (in-structure response spectra, structural member forces, and dynamic lateral soil pressure attributable to earthquake). In addition, the SSAR did not explain how Westinghouse considered the live load in the dynamic model for calculating seismic responses. The issue regarding consideration of live load in the seismic model was therefore identified as Open Item 3.8.4.3-1.

  As a result of the discussion during the meeting on June 12 through 16, 1995, the staff informed Westinghouse of its position on the use of live load for computing the overall building seismic response (25 percent of the live load). In addition, the staff informed Westinghouse of its position on computing the design-basis forces and moments resulting from local vertical seismic response of floor or roof panels (100 percent of the live load).

  Westinghouse subsequently incorporated the staff's position in Revision 12 of SSAR Section 3.7.2.3.1. Specifically, Westinghouse committed that 25 percent of the floor live load or 75 percent of the roof snow load (whichever is applicable) is considered as mass in the global seismic model. Also, in Revision 12 of SSAR Section 3.8.4.3.2.3, Westinghouse stated that for the design of structural members, such as floors and beams, live loads in combination with the SSE are taken as 100 percent of the specified live loads, or 75 percent of the roof snow load (whichever is applicable), except in the case of the containment operating deck. For the seismic load combination, the containment operating deck is designed for a live load of 9.58 kPa (200 lb/ft^2), which is appropriate for the plant operating condition.

  After reviewing this SSAR revision, the staff found that Westinghouse's commitment for the consideration of live load in the design of the NI structures met the staff's technical position described above. The staff also found that the inclusion of 75 percent of snow load as mass in the global seismic model is consistent with the industry practice. On this basis, Open Item 3.8.4.3-1 is closed.

- For some design loads, Westinghouse provided cross-references in early revisions of the SSAR. These cross-references provided either the method for computing these particular loads, or the numerical value of these loads. (For example, Table 2-1 of the

SSAR listed precipitation loads.) However, Section 3.8.4.3 of the SSAR did not provide cross-references for loads such as earthquake and pressure loads, etc. The staff identified this omission as Open Item 3.8.4.3-2.

In Revision 3 of SSAR Section 3.8.4, Westinghouse responded to this open item by adding cross-references to specify either the method for computing design loads, or the numerical values for particular loads. On this basis, Open Item 3.8.4.3-2 is closed.

- In early SSAR revisions, Footnote 3 in Tables 3.8.4-1 and 3.8.4-2 stated, "seismic loads will only be combined with ruptures of pipes that are not seismically supported." However, the staff found that this statement was misleading and should be deleted from these tables. According to the guideline prescribed in Section 3.8.4 of the SRP, the applicable pipe rupture loads (Y_r, Y_j, and Y_m) should be combined with SSE loads regardless of whether the pipe is seismically supported. Consequently, the staff identified this issue as Open Item 3.8.4.3-3.

  Westinghouse subsequently responded to this open item in Revision 3 of SSAR Section 3.8.4. Specifically, Westinghouse deleted Note 3 from Tables 3.8.4-1 and 3.8.4-2. Open Item 3.8.4.3-3 is closed.

- In early SSAR revisions, Westinghouse did not commit that all subcompartments located in the auxiliary building would be designed to withstand global pressure and temperature effects resulting from pipe rupture. In addition, the early SSAR revisions did not indicate that Westinghouse would use the actual pressure and temperature loads for the design. The staff identified this omission as Open Item 3.8.4.3-4.

  Westinghouse responded to this open item in Revision 3 of SSAR Section 3.8.4. In that SSAR revision, Westinghouse committed to design the auxiliary building subcompartments to withstand a global pressure of 34.47 kPa (5.0 psi) and temperature effects associated with pipe rupture. Therefore, Open Item 3.8.4.3-4 is closed. Nonetheless, this commitment raised a concern regarding the adequacy of the design pressure of 34.47 kPa (5.0 psi) and the design temperatures described in the SSAR. Section 6.2.1.2 (Open Item 6.2.1.2-1) of this report discusses the staff's review of this issue.

Also, at the meeting on April 25 through 27, 1995, the staff conducted a detailed review of the containment air baffle. As noted in Section 3.8.4.1 of this report, Westinghouse presented a new design concept at the meeting, and subsequently documented that design concept in Revision 3 of the SSAR and presented relevant design data in SSAR Revision. Independent of the final design details, the staff raised several issues concerning the loads on the air baffle. First, Westinghouse should address the significance of fluctuations in the air flow with respect to flow-induced vibrations and cyclic fatigue. Second, the magnitude of the differential air pressure across the baffle panels cannot be considered finalized until the staff accepts Westinghouse's scale model wind tunnel test results as being applicable to the full-scale structure, as discussed in Section 21.5.7.4 (Open Item 21.5.7.4-1) of this report. Issues related to loads on the air baffle are also discussed in Section 3.8.4.4 (Open Item 3.8.4.4-7) of this report.

In addition, in Revision 20 of SSAR Tables 3.8.4-1 (Load Combinations and Load Factors for Seismic Category I Steel Structures) and 3.8.4-2 (Load Combinations and Load Factors for Seismic Category I Concrete Structures), Westinghouse deleted Load Combination No. 6, because this load combination was not included in the design criteria developed by Westinghouse to carry out the structural design, and also was not considered in the design of seismic Category I structures. The deletion of Load Combination No. 6 without any technical basis is not acceptable to the staff. To resolve this staff concern, Westinghouse included Load Combination No. 6 in Revision 22 of SSAR Tables 3.8.4-1 and 3.8.4-2, and stated that Load Combination No. 6 is not limiting for the analysis and design. On this basis, the staff concludes that the staff's concern regarding the elimination of Load Combination No. 6 is resolved.

3.8.4.4 Analysis and Design Procedures

This section contains the staff's review of the analysis and design procedures used for AP600 SC-I structures (including modular steel structures) other than the containment vessel and internal structures.

3.8.4.4.1 Reinforced Concrete Structures

As described in Section 3.8.4.1 of the SSAR, the SC-I reinforced concrete structures include the shield building and the auxiliary building, and are supported on a common foundation mat. (The analysis and design of the foundation mat are discussed in Section 3.8.5 of this report.) The floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building, and form the coupled shield/auxiliary building. Specifically, the coupled shield/auxiliary building structures include reinforced shear wall structures consisting of the vertical cylindrical shell, conical shell roof structures, PCCWS tank structures, shear/bearing walls, and floor slabs supported by structural steel framing. The structural steel framing was used to support the concrete slabs and roofs, and was designed for vertical loads.

In Revision 3 to Section 3.8.4.3 of the SSAR, Westinghouse documented the loads considered in the AP600 analysis and design. In addition, as described in Revision 3 to Section 3.8.4.4 of the SSAR, Westinghouse obtained in-plane seismic forces from the response spectrum analysis of the 3D finite element fixed-base models. Westinghouse then modified these results to account for soil-structure interaction and accidental torsion effects. For the shear wall and floor slab design, Westinghouse relied on hand calculations as the source of the out-of-plane bending and shear loads, lateral earth pressure, hydrostatic and hydrodynamic pressure loads, and wind loads. In addition, the exterior auxiliary building walls below grade were designed to resist the worst case of lateral earth pressure loads (both static and dynamic), soil surcharge loads, and loads attributable to external flooding.

For the analysis and design of the shield building roof structures and the PCCWS tank, Westinghouse used a 3D finite element model, as stated in early revisions of SSAR Section 3.7.2. Seismic loads (calculated using the 3D lumped-mass stick model) were considered to be equivalent static loads, which are equal to the product of calculated accelerations and lumped masses. In addition, Westinghouse analyzed the seismic response of the water in the tank by conducting a separate response spectrum analysis to a finite element model with input defined by the floor response spectra.

In reviewing Section 3.8.4.4 of early SSAR revisions and Westinghouse's submittal dated May 17, 1994, the staff found that the SSAR described an approach (modeling techniques and analysis methods) for computing seismic member forces of structures (including raceway systems and HVAC ductwork) considering design loads, and designing the shear wall and floor slab. The staff also found that Westinghouse's approach generally met the guidelines prescribed in Sections 3.7.2 and 3.8.4 of the SRP and, thus, are reasonable and acceptable. However, the staff identified a number of technical concerns as follows:

- In early SSAR revisions, Westinghouse failed to describe the type of model developed for the shield building (including the PCCWS) tank and auxiliary building under design loads other than the SSE. Westinghouse also neglected to identify which computer code was used to perform the analysis. In addition, the SSAR did not described which specific combined design load conditions Westinghouse considered in the design calculation. The staff therefore identified these omissions as Open Item 3.8.4.4-1.

  Westinghouse responded to this open issue in Revision 7 of SSAR Section 3.8.4.4.1 and Table 3.7.2-14. Specifically, Westinghouse stated that for the NI structures (reinforced concrete and structural module shear wall structures), in-plane seismic forces were obtained from the response spectrum analysis using a 3D finite element fixed-base model. Also, for the shear wall and floor slab design, Westinghouse evaluated and considered out-of-plane bending and shear loads, such as live load; dead load; seismic load; lateral earth pressure; and hydrostatic, hydrodynamic and wind pressure. These out-of-plane bending and shear forces were obtained through hand calculations. To analyze the shield building roof and PCCWS tank, Westinghouse used 3D finite element models with the GTSTRUDL computer code. The specific loads and load combinations used in that analysis were consistent with SSAR Section 3.8.4.3 and included construction, dead, live, thermal, wind, and seismic loads. To calculate the response to these loads, Westinghouse used equivalent static analyses for all but the seismic load of the tank structures which was obtained using response spectrum analysis.

  In reviewing Revision 7 of SSAR Section 3.8.4.4, the staff found that Westinghouse's analysis approach, modeling techniques, and load combinations used for the analysis and design of these structures met the guidelines prescribed in SRP Sections 3.7.2 and 3.8.4 and, therefore, are acceptable. Also, in the meeting on December 9 through 13, 1996, the staff reviewed the AP600 design calculations and found that Westinghouse had properly described the model developed for the auxiliary building under design loads other than the SSE. In addition, Westinghouse had properly identified the computer code used to perform the analysis. On this basis, the staff concluded that Westinghouse's response to this issue is acceptable and Open Item 3.8.4.4-1 is closed.

- In reviewing early design calculations, the staff found that the final design calculation for the shield building and the PCCWS tank was not available for review. The staff identified this omission as Open Item 3.8.4.4-2.

During meetings conducted in late 1995 and early 1996, the staff reviewed analysis methods and models used for the design of these two structures, and raised three concerns that Westinghouse needed to address:

(1) The vertical component of the earthquake ground motion tends to increase (add to) the water pressure against the PCCWS tank walls. Westinghouse should consider this pressure in designing the outer tank wall and the connection between the tank wall and the conical roof. However, during discussions with Westinghouse, the staff was informed that the design forces for the outer tank wall are very low. Westinghouse should demonstrate and justify the adequacy of these design loads.

(2) Because the conical shell has a relatively shallow slope (35 degrees), a high horizontal component of the in-plane seismic force in the conical shell (caused by vertical excitation of the tank under an SSE) should be expected to apply at the top of the tension ring beam which supports the conical shell. This horizontal force will (a) induce high hoop stress and significant cracking of tension ring beam, and (b) produce a torsional moment on the tension ring beam and bending moment at the top of columns supporting the tension ring beam.

    Westinghouse should considered these two effects in the tension ring beam design.

(3) During construction, the precast panels of the shield building roof are temporarily supported on the containment vessel. Consequently, Westinghouse calculated the maximum reaction loads applied on the containment vessel dome, but indicated that these maximum reaction loads would be reduced during construction, because of the combination of the following factors:

    (a) an increasing number of conical roof panels are installed

    (b) the stiffness of the overall structure increases as each panel is erected

    Nonetheless, the staff concluded that Westinghouse should evaluate the significance of these construction loads (potential of buckling) with regard to the containment vessel dome.

During the meeting with Westinghouse on December 9 through 13, 1996, the staff reviewed the final design calculations for the shield building roof structures (including the PCCWS tank). As a result of that review, the staff found that the assumptions, modeling techniques, and analysis methods were reasonable and met the guidelines prescribed in the SRP. Moreover, the analysis yielded results comparable to those obtained from the confirmatory analysis by the staff (see Enclosure 4 of a letter to Westinghouse dated April 9, 1998). In addition, Westinghouse demonstrated that the construction loads applied on the containment vessel are insignificant and will not cause any buckling of the vessel. However, the torsional moment calculated by Westinghouse for the tension ring beam was much lower than that obtained from the confirmatory analysis. The staff therefore concluded that in order to ensure the design adequacy of this ring beam, Westinghouse should regenerate the torsional moment on the tension ring beam and upgrade the design as needed.

Early in 1997, Westinghouse revised the PCCWS tank structures as a result of the post-72-hour action requirements. Westinghouse also performed an updated seismic analysis (consistent with the modified 3D tank model) to evaluate the impact of the revised shield building roof structures on the overall AP600 design. In the meeting on August 11 through 15, 1997, the staff reviewed Westinghouse's final analysis and design calculations. As a result of this review, the staff found that Westinghouse had properly incorporated the design changes in the revised model, and the analysis and design met the guideline prescribed in the SRP. Nonetheless, the staff identified the following four technical concerns:

(1) The design drawings do not show the vertical reinforcement in both sides of the air inlet columns. The design drawings should also specify the length of these rebars.

(2) In Section 3.8.4.4.1 of the SSAR Westinghouse committed to meet the requirements defined in Chapter 21 of ACI 318, which state that in order to resist shear forces, stirrups should be provided with 135° hooks at both ends of the rebars. However, Westinghouse used double-U bars for the hoop reinforcement (or stirrups) at the air inlet columns, tension ring beam, and compression ring beam. Also, in the air inlet columns, Westinghouse did not extend the shear hoop reinforcement (stirrups) and cross-ties above and below air inlet openings. The use of double-U bars for the shear reinforcement does not meet the SSAR commitment.

(3) The staff identified certain inconsistencies between the design drawings and the summary table in Appendix 25 to Calculation No. 1277-S3C-006, and design deficiencies as follows:

   (a) The design drawing of the conical shell roof at the tension ring beam shows that a bottom radial reinforcement of 45.29 cm^2/m (2.14 in^2/ft) is needed at the columns, and a bottom radial reinforcement of 37.66 cm^2/m (1.78 in^2/ft) over the air inlet. However, Table 11.6 of Appendix 11 in Calculation No. 1277-S3C-006 shows that only one number 9 rebar is provided above the air inlet, and none at the columns. Also, the drawing shows the bottom reinforcement discontinued at the end of conical roof.

   (b) The conical roof at the compression ring beam shows that an amount of 13.16 cm^2 (2.04 in^2) bottom radial reinforcement is needed. Table 11.6 of Appendix 11 in Calculation No. 1277-S3C-006 shows no reinforcement provided. Also, the drawing also shows the bottom reinforcement discontinued at the compression ring.

   (c) The conical shell roof at the internal PCCWS tank wall shows that nine number 9 rebars were provided for the hoop reinforcement at the top and bottom faces. However, the design drawing shows that 18 #9 rebars were provided, but they are not properly distributed at the top and bottom faces.

(4) As discussed in Section 3.7.2 above, Westinghouse failed to consider the out-of-plane vibration in the roof slab design.

On the basis discussed above, Westinghouse's design of the shield building roof structures is not acceptable and Open Item 3.8.4.4-2 remains unresolved.

In its letter dated January 16, 1998 (NSD-NRC-98-5525), Westinghouse responded to the staff concerns discussed above. Also, in Revision 20 of the SSAR, a new Appendix 3H to the SSAR was provided. In this new appendix, Westinghouse included a summary of the shield building roof (including the PCCWS tank structures) structural design. In addition, Westinghouse revised SSAR Section 3.8.4.4.1 and stated that the provisions are applied to elements that experience reinforcement tensile stresses above yield or concrete compressive stresses above the concrete strength when the SSE loads are increased by a factor of 1.67 (ratio of the seismic margin review level earthquake to the SSE). In this SSAR section, Westinghouse also limited the application of the ductility criteria for the reinforced concrete design (requirements provided in Chapter 21 of ACI 318-95) to the following:

- in-plane behavior of interior and exterior walls of the NI
- in-plane behavior of cylindrical wall of shield building including columns between the air-inlets
- out-of-plane behavior of the NI basemat

The staff's review of the above mentioned letter and Revision 20 of the SSAR drew the following three conclusions:

(1) The responses to the staff's concerns were generally acceptable, because they met the design requirements of ACI 349.

(2) The commitment documented in Revision 20 of SSAR Section 3.8.4.4.1 for considering the ductility criteria (Chapter 21 of ACI 318-95) is not acceptable, because these criteria should be applied to all safety-related reinforced concrete structures. Also, there is no basis for the statement that the provisions are applied to elements that experience reinforcement tensile stresses above yield or concrete compressive stresses above the concrete strength when the SSE loads are increased by a factor of 1.67 (ratio of the seismic margin review level earthquake to the SSE). In addition, this SSAR revision is also not consistent to Westinghouse's previous commitment documented in Revision 17 of the SSAR, which stated that the ductility criteria of ACI-318, Chapters 12 and 21, are considered in detailing, placing, anchoring, and splicing of the reinforcing steel of the seismic Category I structures. The application of ductility criteria committed in Revision 17 of the SSAR were accepted by the staff.

(3) The design information provided in Appendix 3H to the SSAR, Revision 20 is not sufficient for the description of NI structural critical section design. In addition, Westinghouse needs to ensure that the placement of reinforcing steel follows the criteria of Chapter 21 of ACI 318-95.

In response to the staff's concern regarding the use of the ductility criteria of ACI 318-95 Code, Westinghouse submitted Revision 22 of SSAR Section 3.8.4.4.1 and Appendix 3H to the SSAR for review. In the revised SSAR Section 3.8.4.4.1, Westinghouse stated that the criteria of ACI-318-95, Chapter 12, are applied in the development and splicing of the reinforcing steel. The ductility criteria of ACI-318, Chapter 21, are applied in detailing, and anchoring of the reinforcing steel. Westinghouse also stated that the application of Chapter 21 detailing is demonstrated in the reinforcement details of critical section in subsections 3.8.5 and Appendix 3H. In addition, Westinghouse provided description of examples documented in Appendix 3H.

The staff's review found that the revised SSAR Section 3.8.4.4.1 together with the details provided in Appendix 3H to the SSAR are sufficient and meet the requirements of ACI 349 and ACI 318. On the basis discussed above, the staff concludes that the issues regarding the design of the shield building roof structures including the PCCWS tank, and the application of ductility criteria (Chapter 21 of ACI 318-95) to the design of NI structures are resolved.

- Because a massive amount of water is to be contained in the PCCWS tank, the staff raised a concern that the COL applicant should monitor the vertical and radial deformation of the tank during initial filling, and compare the measured values with the tank deformation predicted by calculation. The staff identified this issue as Open Item 3.8.4.4-3 and COL Action Item 3.8.4.4-1.

At the meeting on June 12 through 16, 1995, Westinghouse stated that the water weight is small, in comparison with the total weight of the shield building roof structure (estimated to be about 10 percent). Westinghouse also showed that the deflection of the roof structure resulting from the first fill of water should be negligible. On that basis, Westinghouse contended that there is no need to monitor the tank deflections and compare the deflections against predictions.

During the meeting on December 9 through 13, 1996, Westinghouse repeated its justification concerning this issue. However, the staff did not agree with Westinghouse's basis for not monitoring the deformation of the tank during initial tank filling. Moreover, the staff asserted that post-construction testing is necessary to confirm the adequacy of the PCCWS tank. This is because the staff's review experience suggest that the excessive deformation resulting from the massive amount of water may cause cracking of the tank wall and base slab, as well as water leakage from reinforced concrete tanks with steel liners.

In Revision 17 of SSAR Section 3.8.4.1.1, Westinghouse added the statement that leak chase channels are provided over the liner welds to permit monitoring for leakage and to prevent degradation of the reinforced concrete wall which might result from the freezing and thawing of leakage. Also, Westinghouse indicated that the exterior face of the reinforced concrete boundary of the PCCWS tank is designed to control cracking, in accordance with Paragraph 10.6.4 of ACI-349, with reinforcement steel stress on the basis of sustained loads (including thermal effects). However, Westinghouse still did not commit to monitor the tank deformation during initial filling and compare the measured tank deformation with that predicted by analysis. On the basis of the above discussion, the staff concluded that Westinghouse's response to the staff's concern (as stated in Revision 17 of SSAR Section 3.8.4.1.1) is not acceptable. Therefore, Open Item 3.8.4.4-3 and COL Action Item 3.8.4.4-1 remained unresolved.

To resolve this open issue, Westinghouse responded as follows:

- committed, in Section 3.8.4.7, that structures supporting the PCCWS tank on the shield building roof will be examined before and after first filling of the tank as follows:

  (1) The boundaries of the PCCWS tank and the tension ring beam of the shield building roof will be inspected visually for excessive concrete cracking before and after the first filling of the tank. Any significant concrete cracking will be documented and evaluated in accordance with ACI 349.3R-96.

  (2) The vertical elevation of the PCCWS tank relative to the top of the shield building cylindrical wall at the tension ring beam will be measured before and after the first filling. The change in relative elevation will be compared against the predicted deflection.

  (3) A report will be prepared summarizing the test and evaluating the results.

  Also, during the operation of the plant, the condition of these structures should be monitored by the COL applicant to provide reasonable confidence that the structures are capable of fulfilling their intended functions.

- added a new section (SSAR Section 3.8.6.2) to state that the COL applicant should examine the structures supporting the PCCWS tank on the shield building roof during the initial tank filling as described in SSAR Section 3.8.4.7.

Westinghouse's commitment in Revision 20 of the SSAR is consistent with industry practice and satisfies the staff's concern. Therefore, the issue regarding the PCCWS tank deformation before and after the initial tank filling is resolved. However, Westinghouse made a statement in SSAR Section 3.8.4.7 (Revision 20) that there are no other in-service testing and inspection requirements for the seismic Category I structures. This statement is not acceptable to the staff, because in-service testing and inspection are required by Appendix J to 10 CFR Part 50 for in-service testing and 10 CFR 50.55a(b)(2)(x) for in-service inspection for the containment vessel which is classified as seismic Category I.

To resolve the staff's concern regarding the in-service testing and inspection for the containment vessel, Westinghouse stated, in Revision 23 of SSAR Section 3.8.4.7, that there are no other in-service testing requirements for the seismic Category I shield building and auxiliary building. The revised SSAR statement regarding the exclusion of the containment vessel from the statement that there are no other in-service testing requirements for the seismic Category I shield building and auxiliary building satisfied the staff's concern and, therefore, this issue is resolved.

- Early revisions of the SSAR stated that the exterior walls of Category I structures below grade are designed to resist the worst-case lateral earth pressure loads. Specifically, in its submittal dated May 20, 1994, Westinghouse stated that the embedded portion of the exterior walls of the AP600 NI are designed for dead loads, live loads, SSE loads, hydrostatic loads attributable to groundwater and probable maximum flood, static soil pressure loads, surcharge loads, and soil pressure induced by the SSE. Two dimensional SSI analysis results are used to establish the soil pressure induced by the SSE and to verify the structural integrity of the walls. Westinghouse's submittal dated May 20, 1994, is acceptable because it conforms with the guidelines prescribed in Section 2.5.4 of the SRP, with the following two exceptions:

(1) During the review of early design calculations, Westinghouse agreed that the pressure to be used for the wall design will not be less than the pressure used in the sliding and overturning evaluation of the AP600 NI. However, the staff's review revealed that the soil pressure used for the wall design was much lower the passive soil pressure used for the NI sliding analysis.

(2) The dynamic soil pressure attributable to the structure-to-structure interaction effects from the adjacent structures (turbine building, annex buildings, and radwaste building) was not considered in the wall design.

The staff identified these discrepancies as Open Item 3.8.4.4-4.

In the meeting dated June 12 through 16, 1995, Westinghouse responded to this open item by agreeing to take the following two actions:

(1) Justify why the soil pressure used for the design of exterior embedded walls is much lower than the soil pressure used for the NI sliding analysis.

(2) Consider the dynamic soil pressure attributable to the structure-to-structure interaction effects from the adjacent structures in the design of the exterior embedded NI walls.

Open Item 3.8.4.4-4 therefore remained unsolved pending Westinghouse's completion of these actions.

In Revision 5 of the SSAR, Westinghouse described how the design pressure for the exterior embedded walls was calculated considering the torsional motion (i.e., the "box") effects. Westinghouse also described the effects of structure-to-structure interaction through the soil. The staff reviewed the analysis method documented in the SSAR, and found that it met the guideline prescribed in the SRP. Also, during the meeting on August 4 through 8, 1997, the staff reviewed Revision 14 of SSAR Section 3.8.4.4.1 and the related design calculations. That review revealed that the exterior embedded walls were designed to resist the full passive earth pressures that can develop in the side soils to ensure that a suitable factor of safety exists in the sliding evaluation. On this basis, the staff concludes that Westinghouse's response is acceptable, and Open Item 3.8.4.4-4 is closed.

- Because of concerns regarding degradation and aging, Westinghouse should commit (in the SSAR) to periodically replace the flexible and nonmetallic containment air baffle seal material throughout the life of the plant. The staff identified this issue as Open Item 3.8.4.4-5.

  Westinghouse responded to this open item in Revision 3 of SSAR Section 3.8.4.1.3 and Figure 3.8.4-1. Specifically, Westinghouse stated that a vertical sliding plate, which replaces the flexible seal, is provided between the top row of the air baffle panels supported off the containment dome and the air baffle attached to the shield building roof. This sliding plate can accommodate the differential movement between the containment dome and shield building roof, and will not degrade during the life of the plant. Westinghouse's SSAR commitment resolved the staff's concern regarding the degradation and aging effect of the flexible seal. Therefore, Open Item 3.8.4.4-5 is closed.

- Westinghouse did not provide analysis procedures or design details regarding the spent fuel pool, including fuel racks, fuel transfer canal, and new fuel storage area. The staff identified this omission as Open Item 3.8.4.4-6.

  In Revision 7 of the SSAR Section 3.8.4.1.2, Westinghouse described the structural modules in the auxiliary building, which include the spent fuel pool, fuel transfer canal, and cask loading and washdown pits. Also, in Revision 11 of SSAR Section 3.8.4.1.2, Westinghouse stated that the structural modules are the same as those described in SSAR Section 3.8.3.1 for the containment internal structures. In addition, in Revision 11 of SSAR Section 3.8.4.4.1, Westinghouse stated that the steel structural modules in the auxiliary building are designed using the same procedures as those inside the containment vessel (as described in SSAR Section 3.8.3). On the basis that the procedures for the design of steel modules inside the containment vessel are acceptable as discussed in Section 3.8.3 of this report, the analysis procedures and design approach for the spent fuel pool, fuel transfer canal, and cask loading and washdown pits are acceptable.

  With regard to the design of the spent fuel pool racks, however, Westinghouse asserted (in Revision 7 of SSAR Section 9.1.2.2.1) that the spent fuel pool racks are purchased equipment and are not treated as part of standard design. This SSAR section did provide the relevant analysis procedures and design approach; however, Westinghouse did not define the relevant design loads. To ensure the design adequacy of the spent fuel pool system, Westinghouse should modify the SSAR to provide cross references to the definition of design loads (including seismic loads), and specify the criteria used for the design of the spent fuel pool floor and fuel racks.

  To resolve the staff's concern regarding the design of the spent fuel racks, Westinghouse stated in Revision 12 of the SSAR that the design of racks for the new fuel and spent fuel (SSAR Section 3.8.4) is described in SSAR Section 9.1. Also, Westinghouse stated that the spent fuel racks (in Section 9.1.2.2) are protected from the effects of natural phenomena such as earthquakes (SSAR Section 3.7.2), wind and tornados (SSAR Section 3.3), floods (SSAR Section 3.4) and external missiles (SSAR Section 3.5). Therefore, the staff concluded that the SSAR provided sufficient information regarding the design of the spent fuel rack design. On this basis, Open Item 3.8.4.4-6 is closed.

- As part of design and analysis procedures, Westinghouse should prepare and document design reports for all SC-I structures in accordance with the guideline prescribed in Appendix C to Section 3.8.4 of the SRP. In its submittal dated June 30, 1994, Westinghouse agreed to prepare a design report for each of the following structures and buildings:

  - NI foundation mat
  - auxiliary building
  - containment internal structures
  - shield building

  Westinghouse also stated that these design reports will not be included in the SSAR, but will be available for NRC audit, and will be updated during construction to incorporate as-procured and as-constructed information. The staff finds that Westinghouse's commitment to prepare the design reports for each of the safety-related structures meets the guidelines of Appendix C to Section 3.8.4 of the SRP and, thus, is acceptable. However, the list of components provided in Westinghouse's submittal dated June 30, 1994, should also include the IRWST (as part of the containment internal structures), and the air baffle (as part of shield building). The staff identified this concern as Open Item 3.8.4.4-7.

  In the meeting on April 25 through 27, 1995, the staff raised several technical concerns pertaining to the air baffle design. Westinghouse responded by presenting a new preliminary design for the containment air baffle. During the meeting on January 14 through 17, 1997, Westinghouse presented the design calculations for the containment air baffle. After reviewing the new information, the staff found that Westinghouse's design meets the requirements of ANSI/AISC N690-84 and, therefore, is acceptable. However, Westinghouse's design did not consider the air flow fluctuations and the potential for flow-induced vibration/fatigue failure. In addressing these two technical concerns during the meeting on April 14 through 16, 1997, Westinghouse provided preliminary design information for review. The staff's review of the draft information found that Westinghouse had, in fact, considered the effect of air flow fluctuations and the potential for flow-induced vibration/fatigue failure. On this basis, the staff concluded that the concern regarding the design of the containment air baffle is resolved. However, Open Item 3.8.4.4-7 will not be closed until Westinghouse submits the final design calculations.

  Westinghouse addressed the staff's concern regarding the effect of air flow fluctuations and the potential for flow-induced vibration/fatigue of the containment air baffle in the letter dated June 11, 1997 (NSD-NRC-97-5175). During the meeting with Westinghouse on January 20 and 21, 1998, the staff discussed the results of its review of the above letter. Westinghouse's submittal evaluated the potential effects of dynamic excitation, fatigue of the air baffle, and localized loads because of air flow imposed on the turning vane of the air baffle. This submittal demonstrated that the dynamic excitation effects to the air baffle are very small by showing that the structural (containment vessel and air baffle) frequencies do not coincide with those of the design wind and tornado frequencies. The effect of fatigue was also evaluated by meeting AISC N690 requirements for fatigue/cyclic stresses. The localized loads on the air baffle were addressed by the loads that were used in the design and the physical configuration of the turning vane which was designed to minimize the occurrence of localized eddies or flow separation.

Because the Westinghouse submittal demonstrated that the effects of air flow fluctuations are very small or meet design requirements of ANSI/AISC for fatigue/cyclic stresses, Open Item 3.8.4.4-7 is closed.

- In order to prevent corrosion, Westinghouse should modify the SSAR to commit to the use of coated reinforcing bars (rebars) in the design of embedded exterior reinforced concrete walls. The staff identified this concern as Open Item 3.8.4.4-8.

  In Revision 3 of SSAR Section 3.8.4.6.1.2, Westinghouse stated that SC-I structures located below grade elevation are protected against flooding by water-proofing membranes and water stops. In addition, Westinghouse stated (in Revision 17 of the same SSAR section) that a cementitious crystalline additive is to be added to the shotcrete vertical walls and mudmat placed under the foundation mat together with the nominal reinforcing steel to minimize the effects of cracking and prevent inflow of water into the site. These waterproofing membranes and waterstops, in conjunction with 5 cm (2 in) of concrete cover provide sufficient protection for the reinforcing steel. Therefore, the use of coated reinforcing steel is not necessary. On the basis of common engineering practice, the staff agrees with Westinghouse's justification and concludes that Open Item 3.8.4.4-8 is closed.

- The staff reviewed Revision 3 to Appendix 3G and 3H of the SSAR, which respectively describe the codes and standards, loads, load combinations, analysis, and design methods for the cable trays and supports and the HVAC ducts and supports. As a result of its review, the staff identified the following three items that Westinghouse needs to clarify:

  (1) Appendices 3G and 3H state that the live load of 1.11 kNewton (250 pounds) applied at the center of the cable tray or HVAC duct span is not combined with seismic loads. Westinghouse should clarify if plant procedures will include measures to limit the live loads that are applied during plant operations. Westinghouse should also determine the appropriate seismic load resulting from the specified live load during plant shutdown and combine this seismic load with the live load.

  (2) Appendices 3G and 3H specify an allowable stress of 1.6 times the basic allowable for the load combination that includes dead and seismic loads. Westinghouse should provide the basis for using the stress limit coefficient of 1.6 for the service load condition including SSE. (In particular, Westinghouse needs

to justify this factor for compressive stresses.) In addition, Westinghouse should clarify Appendix 3H to provide the equations and method for calculating duct stresses associated with pressure loads.

(3) The seismic load on HVAC ducts includes both global and local effects. Consequently, Westinghouse should clarify Appendix 3H by describing the global effects to be determined by beam type analyses, as well as the local effects that may be assessed by analyses of panels bounded by stiffeners and subjected to pressures associated with inertial loads. In addition, Appendix 3H states that ductwork within partially or fully vented buildings is subject to wind effects. In addition to pressure resulting from these effects, ductwork exposed to wind/tornado should also be designed for missiles caused by tornados. Finally, Westinghouse should describe the procedure used for the analysis, design, and qualification of cable tray and duct support anchorages into concrete.

In Revision 7 of the SSAR, Westinghouse renamed Appendices 3G and 3H as Appendix 3A for the design of the HVAC ductwork and supports and Appendix 3F for the design of the cable trays and supports. However, this SSAR revision did not provide any response to address the concerns identified above.

In Revision 11 of SSAR Appendix 3A, Westinghouse provided its response to the staff's concern related to the design of HVAC systems. Also, in Revision 12 of Appendix 3F to the SSAR, Westinghouse addressed the issues related to the design of cable tray systems. The staff's review of the new information led to the following four findings:

(1) In Revision 11 of SSAR Appendix 3A, Westinghouse redefined the dead load for HVAC duct systems as including the weight of the duct sheet, stiffeners, and in-line components (such as duct heaters and dampers). The dead load for HVAC systems also includes permanently attached items, such as insulation and fireproofing, as well as the weight of the duct supports. By contrast, Westinghouse clarified that temporary items used during construction or maintenance are removed before operation. Consequently, the load attributable to the these temporary items, estimated at 113.4 kg (250 pounds), was renamed as construction live load. This live load is to be applied only during construction or maintenance on an area of 64.52 cm^2 (10 in^2) of HVAC duct systems at a critical location to maximize the flexural and shear stresses. Westinghouse also stated that this construction live load is not combined with seismic loads. The staff found that the treatment of live loads for the HVAC design appears reasonable.

(2) In Revision 12 of Appendix 3F to the SSAR, Westinghouse redefined the dead loads for cable tray systems as including the weight of the cable trays, their supports, the cables inside the trays, and any permanently attached items. Westinghouse also clarified that temporary items used during construction or maintenance are removed before operation. Consequently, the load attributable to these temporary items, estimated at 113.4 kg (250 pounds), is considered to be construction live load and is not combined with the seismic loads in the

design. The definition of the construction live load and the application of this live load in the cable tray design appear reasonable to the staff.

(3) In Revision 11 of SSAR Appendix 3A, as well as Revision 12 of SSAR Appendix 3F, Westinghouse redefined the allowable stresses for the load combination that includes dead and seismic loads (formally 1.6 times the basic allowable stress). Specifically, the allowable stresses for the raceway (HVAC duct and cable tray) supports utilizing rolled structural shapes are now defined as the basic allowable for service levels A and B, 1.6 times the basic allowable for tension, and 1.4 times the basic allowable for compression. The allowables proposed by Westinghouse meet the guidelines of the staff position for the application of ANSI/AISC N690 which is documented in Appendix G to NUREG-1503 (Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design).

(4) In Revision 11 of SSAR Appendix 3A (Section 3A.3), Westinghouse stated that the design for the pressure loads is founded on the AG-1 standard promulgated by the ASME and ANSI for SC-I ducts, and the Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) for SC-II ducts. The global behavior of the ducts is determined from the overall bending of the ducts between the supports (which is similar to the beam type bending). For determining the section modulus, the corners of the duct are considered effective. The analysis approach for the duct design follows common engineering practice and is acceptable. In addition, in Revision 12 of Appendix 3A to the SSAR, Westinghouse committed that the SC-I HVAC ductwork is protected from impact by tornado missiles.

On the basis discussed above, the staff concluded that Westinghouse has resolved the open issues related to the design of raceway systems (HVAC ducts and cable tray systems).

In addition to the open items discussed above, the staff raised one issue during its review of the AP600 design calculations. Specifically, the SSAR stated that the embedded exterior (peripheral) walls of SC-I structures in the AP600 NI are designed to resist the worst-case lateral earth pressure loads. However, during the design review meetings, the staff found that the soil pressure used for the wall design was much lower than the passive soil pressure used for the NI sliding analysis. The staff also found that the wall design did not account for the dynamic soil pressure attributable to the structure-to-structure interaction effects from the adjacent structures (turbine building, annex buildings, and radwaste building). In addition, to enhance the resistance to the high shear stress attributable to the external earth pressure (both static and dynamic), Westinghouse applied heavy shear reinforcement at various locations, such as the junction between walls and the foundation mat. However, with relatively thin walls (the wall thickness at the junction with the foundation mat is 91.44 cm [3 ft]), the congestion of reinforcement at these locations may cause reduction of shear resistance of the walls. Westinghouse should consider these concerns in the final design of these walls.

In Revision 17 of SSAR Section 3.8.4.4.1, Westinghouse stated that the exterior walls of the SC-I structures below the grade are designed to resist the worst case lateral earth pressure loads (static and dynamic), soil surcharge loads, and loads because of external flooding. These walls are also designed for full passive earth pressure which develops as a result of a lateral sliding motion as described in SSAR Section 3.8.5.5.3. Also, during the review meeting on August 4 through 15, 1997, the staff reviewed the design calculation for the exterior embedded walls and found that these walls were designed for all loads (as listed in the SSAR) including full passive earth pressure and dynamic soil pressure attributable to structure-to-structure interaction effects from the adjacent structures. On this basis, the staff concludes that this issue is resolved.

3.8.4.4.2 Design of Critical Sections

Because of the complication of the coupled shield/auxiliary building structures, Westinghouse informed the staff that the completed structural design of this building will not be performed. (The five-story auxiliary building is structurally connected with the cylindrical shell shield building at six different elevations and forms a coupled structure. The coupled shield/auxiliary building is founded, together with the containment vessel and the containment internal structures, on a irregular shaped foundation mat.) Instead, the detailed design would be completed only for the critical sections of structures. As described in Revision 12 of SSAR Section 3.8.4.5.3, Westinghouse identified nine critical sections for which Westinghouse completed its structural design. The staff reviewed samples of these critical section designs and raised the following concerns:

- In reviewing the design calculations for the auxiliary building roof slab at Elevation 180 ft (Calculation Nos. 1260-SSC-003, Revision 2, and 1260-CCC-003, Revision 3), the staff identified two issues:

  (1) The design did not account for the effect of global out-of-plane seismic moments along the edge of the roof slab.

  (2) Reinforcements for the concrete slab in the north-south direction (parallel to floor steel girders) along the roof edge should be designed assuming no composite action of the concrete slab with the steel girder.

- The design of the shield building roof structures is not adequate as discussed under Open Item 3.8.4.4-2 above.

- Westinghouse should include the detailed design drawing for each of these critical sections in the SSAR.

Westinghouse needs to revise the design calculation to address the staff's concern discussed above and provide figures describing reinforcement details of critical sections in the SSAR.

To resolve this staff concern, Westinghouse provided Revision 22 of Appendix 3H to the SSAR for review. In this revised appendix, Westinghouse summarized the design of all critical sections located in the auxiliary/shield building including the shield building roof structures. Also, figures were provided to indicate the details of the structural design including the placement and anchorage of reinforcements.

The staff's review of Westinghouse's submittal found that Revision 22 of Appendix 3H has provided sufficient details for the design of critical sections in the auxiliary/shield building. Also, the placement of reinforcement meets the ductility requirements of Chapter 21 of ACI 318-95. On this basis, the staff concludes that the issue regarding the documentation of critical design is resolved.

3.8.4.4.3 Passive Containment Cooling Ancillary Water Storage Tank System

As a result of the post-72 hour action requirements, Westinghouse made significant changes to the passive containment cooling system and associated structures. In addition to the design changes for the PCCWS tank and the shield building roof structures, Westinghouse added a passive containment cooling ancillary water storage (PCCAWS) tank and associated systems and components (pumps, valves, and piping systems). These systems and components are listed in SSAR Table 3.2-3 (Sheets 8 through 11).

As described in Revision 13 of SSAR Section 6.2.2.2 and Table 3.2-3, the PCCAWS tank is a cylindrical steel tank located at ground level near the auxiliary building. This tank is filled with demineralized water and has a useable volume of greater than 400,000 gallons for makeup to the PCCWS tank. The tank is classified as a non-seismic item, and is designed to American Petroleum Institute (API) Standard 650 with the capability to withstand seismic loads (SSE) and a 233 km/hr (145 mph) wind.

As a result of the meeting held on February 4, 1997, and the review of SSAR Section 6.2.2.2 and Table 3.2-3, Revision 13, the staff issued its review results in a letter dated July 7, 1997. In this letter, the staff stated its technical position on the civil/structural area as follows:

- The PCCAWS tank and associated systems and components should at least be classified as SC-II items, and should be analyzed, designed and constructed using the method and criteria for SC-II building structures, as defined in SSAR Sections 3.2.1 and 3.7.2. In these two sections of the SSAR, Westinghouse committed that SC-II structures are designed for the SSE using the same method and criteria used for SC-I structures, and would be constructed to the same requirements as non-seismic structures.

- To ensure that the PCCAWS tank and associated systems and components can withstand the effects associated with the high winds of severe hurricanes, this tank and its components should be analyzed and designed for Category 5 hurricanes, including the effects of sustained winds, maximum gusts, and associated wind-borne missiles.

On the basis of the above discussion, Westinghouse should reclassify the PCCAWS tank and associated systems and components, and should incorporate the staff's technical position regarding the design requirements in the SSAR.

In the letter dated October 10, 1997, Westinghouse provided a markup of the SSAR revision for the staff review. In this markup, Westinghouse reclassified the PCCAWS tank and the anchorage of the associated systems and components as SC-II. In addition, Westinghouse made the following commitments for the design of this system:

- The PCCAWS tank, and associated systems and components will be analyzed, designed and constructed using the method and criteria for SC-II building structures, as defined in SSAR Sections 3.2.1 and 3.7.2.

- The PCCAWS tank and associated systems and components will be analyzed and designed for Category 5 hurricanes, including the effects of sustained winds, maximum gusts, and associated wind-borne missiles.

On the basis of the above discussion, the staff concludes that the concern regarding the classification and design criteria for the passive containment cooling system is technically resolved.

3.8.4.5 Structural Modules

Westinghouse's submittal dated May 17, 1994, stated that the steel structural modules in the auxiliary building and the ceilings of the MCR and the instrumentation and control room are similar in design to those used in the containment internal structures described in Appendix 3A (to the early revisions) of the SSAR. The staff was concerned, however, that if there are differences in the details of these modules, as discussed in Section 3.8.3 of this report, Appendix 3A (to early SSAR revision) should include a description of the criteria used for these different configurations and applications. The staff identified this concern as Open Item 3.8.4.5-1.

For the design of finned-floor modules, Revision 3 of SSAR Section 3.8.4.1.2 stated that the finned-floor modules were designed as reinforced concrete slabs in accordance with ACI-349, while the steel panels were designed and constructed in accordance with ANSI/AISC N690. For positive bending, the bottom steel plate with fin stiffeners is in tension and acts as the bottom reinforcement. For negative bending, compression is resisted by the stiffened plate, and the tension side is resisted by the top steel reinforcement. Westinghouse also provided the design details of the finned-floor modules in Figure 3.8.4-6 of the SSAR. Consistent with the resolution of Open Items 3.8.3.2-1 and 3.8.3.2-2, the staff finds that this design met the requirements of ACI-349 for reinforced concrete slabs and ANSI/AISC N690 for steel panels. The staff finds this design acceptable.

For the design of concrete-filled modules, Revision 11 of Section 3.8.4.4.1 of the SSAR stated that the structural modules in the auxiliary building were designed using the same procedures as the structural modules of the containment internal structures described in SSAR Section 3.8.3. Because Westinghouse used the same analysis and design procedures as in SSAR Section 3.8.3 (reviewed and evaluated in Section 3.8.3 of this report), the staff concludes that Westinghouse has resolved the concern regarding the analysis and design procedures used for structural modules outside the containment vessel. Consequently, Open Item 3.8.4.5-1 is closed.

Section 3.8.4.6.2 of early revisions of the SSAR briefly covered quality control for other Category I structures and was also referenced by Section 3.8.3.6 of the SSAR for containment internal structures. Specifically, Section 3.8.4.6.2 of the SSAR stated only that the QA program was described in Chapter 17 of the SSAR and conformed to RG 1.94 as described in Section 1.9 of the SSAR. However, Section 1.9 of the SSAR stated that RG 1.94 is not applicable to AP600 design certification because this conformance is the responsibility of the COL applicant. Chapter 17 of the SSAR discussed QA during design, procurement, fabrication, inspection, and/or testing of nuclear power plant items and services. This section of the SSAR also referenced two Westinghouse topical reports dealing with QA. However, the staff's review of these documents revealed that they do not adequately address certain aspects of QC which are applicable to modular construction. Specifically, the document should address QC requirements related to the entire process from fabrication to erection. The requirements should also address such items as fabrication and assembly tolerances, handling requirements, load testing before lifting/handling operations, verification of proper fitup, erection, and other tolerances. In addition, the document should describe the extent of adherence to industry codes and standards regarding QC requirements (for example, ACI-349, AWS Code, and AISC Specifications). The staff identified these concerns as Open Item 3.8.4.5-2.

In Revision 11 of SSAR Section 3.8.3.6, Westinghouse presented QC requirements for AP600 structural modules. This SSAR section specified that packaging, shipping, receiving, storage, and handling of structural modules are in accordance with NQA-2, Part 2.2 (formerly ANSI/ASME N45.2.2, as specified in ANSI/AISC N690). In addition, SSAR Section 3.8.3.6.1 specified that tolerances for fabrication, assembly, and erection of the structural modules conform to the requirements of ACI-117, AWS D1.1 and ANSI/AISC N690. On the basis that Westinghouse's SSAR commitments for QC meet the code standards, the staff concludes that Open Item 3.8.4.5-2 is closed.

The staff also concludes that the design of AP600 structural modules outside containment meets the design code requirements discussed above and is acceptable.

3.8.4.6 Structural Criteria

In Sections 3.8.4.2 and 3.8.4.5 of the SSAR, Westinghouse stated that the analysis and design of reinforced concrete structures would conform to ACI-349-90, while the analysis and design of steel structures would conform to ANSI/AISC N690. Also, the SSAR stated that the criteria of ACI-318, Chapters 12 and 21, would be considered in detailing, placing, anchoring, and splicing the reinforcing steel. In addition, in Revision 22 of SSAR Section 3.8.4.5.1, Westinghouse provided supplemental criteria for the application of Appendix B to ACI-349 to the structural design of the AP600 NI. As discussed in Section 3.8.4.4 above, these supplemental criteria are acceptable to the staff.

For the use of ANSI/AISC N690 in the design of steel structures (including concrete filled modules), SSAR Section 3.8.4.5.2 incorporated the staff's position regarding the application of ANSI/AISC N690 to the nuclear plant design. On the basis that the ACI-349 and ANSI/AISC N690 were reviewed and found acceptable during the staff's evaluation of the ABWR and System 80+ designs, and because the SSAR incorporated the staff's positions, the staff concludes that the structural criteria adopted by Westinghouse for the AP600 design are acceptable. However, any proposed change to the use of ANSI/AISC N690 (with the exceptions addressed in SSAR Section 3.8.4.5.2), ACI 349-90 (with the supplemental requirements documented in SSAR Section 3.8.4.5.1, Revision 22), and ACI 318-95 (ductility criteria documented in SSAR Section 3.8.4.4.1, Revision 22) will require NRC approval prior to implementation of the change.

3.8.4.7 Conclusion

On the basis of the above discussion, the staff concludes that the design of safety-related structures other than containment vessel and containment internal structures is acceptable and meets the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, and 4. In particular, the staff reached this conclusion on the basis of the following observations:

- By following the guidelines of the relevant RGs and industry standards (indicated below), Westinghouse has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the safety-related structures other than containment vessel and containment internal structures are designed, fabricated, erected, and constructed to quality standards commensurate with their safety function.

- Westinghouse has met the requirements of GDC 2 by designing the safety-related structures other than containment vessel and containment internal structures to withstand the 0.3g SSE with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

- Westinghouse has met the requirements of GDC 4 by ensuring that the design of the safety-related structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

The criteria used in the analysis, design, and construction of the plant's Category I structures (other than the containment vessel and containment internal structures), adequately account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime. These criteria conform with established codes, standards, and specifications acceptable to the staff, including RGs 1.10, 1.15, 1.55, 1.69, 1.91, 1.115, 1.142, and 1.143; ACI-349; ACI-318; and ANSI/AISC N690, "Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities."

In addition, Westinghouse has used these criteria, as defined by applicable codes, standards, and specifications regarding the loads and loading combinations; design and analysis procedures; structural acceptance criteria; materials; QC programs; and special construction techniques; and testing and in-service surveillance requirements. Together, these considerations provide reasonable assurance that the structures will withstand the specified design conditions without losing their structural integrity or the performance of required safety functions in the event of winds, tornados, earthquakes, and various postulated accidents.

Furthermore, the staff's conclusion regarding the design of the auxiliary and shield buildings (including the PCCWS tank structures, the shield building roof structures and the structural modules outside the containment) is based on its review of the samples of design calculations for the critical sections of these structures described in Revision 22 of SSAR Section 3.8.4.5.4, "Design Summary of Critical Sections" and Appendix 3H, "Auxiliary and Shield building Critical Sections," to the SSAR. Therefore, any proposed change to the text of Revision 22 of SSAR

Section 3.8.4.5.4 and Appendix 3H to the SSAR will require NRC review and approval before implementation of the change.

3.8.5 Foundations

Using the guideline provided in Sections 3.8.5 of the SRP and related RGs, the staff reviewed Revisions 0 through 23 of Sections 3.8.4 and 3.8.5 of the SSAR for the design of the AP600 NI foundation mat. In addition to the evaluation described in various sections below, the staff's review history and the resolution status of open items identified in the DSER are documented in Section 3.8.5.6 and Section 3.8.5.7 of this report, respectively.

3.8.5.1 Description of Foundation Mat

The AP600 NI structures consisting of reactor containment vessel, containment internal structures, the shield building, and the auxiliary building are supported on a common foundation mat. This is the only foundation mat for a SC-I structure within the standard design scope. The foundation mat, while not precisely rectangular, is approximately 77.4 m (254 ft) long and 35.2 m (115.5 ft) wide. The thickness of the mat is 1.8 m (6.0 ft) in the auxiliary building area, and is 6.7 m (22.0 ft) at the periphery, and 1.8 m (6.0 ft) at the center in the shield building and containment vessel area. The top of the foundation mat is located at Elevation 20.27 m (66.5 ft) and the nominal elevation of the free grade surface is 30.48 m (100 ft).

Adjoining buildings (e.g., the radwaste, turbine, and annex buildings), are structurally separated from the NI structures by a 5 cm (2 in) gap below the grade. A 10 cm (4 in) minimum gap is provided above grade. This provides spaces to prevent interaction between the NI structures and adjacent structures during a seismic event. The foot print of the foundation mat is shown in SSAR Figure 3.7.1-16. SSAR Figure 3.8.5-1 shows the foundations for the NI structures and the adjoining structures.

3.8.5.2 Standard Design Certification Rule

The AP600 standard design, which includes the foundation mat, must be acceptable for a range of site conditions in accordance with 10 CFR 52.47 and in compliance with the standards set out in 10 CFR Part 50 and its appendices, and 10 CFR Part 100 as they apply to applications for construction permits and operating licenses for nuclear power plants, and as those standards that are technically relevant to the design. On this basis, it is necessary for the NRC staff to determine that (1) the design of the foundation mat meets design guidance stated in the SRP, including the design codes and standards, and (2) the design is suitable for a range of site conditions.

3.8.5.3 Summary of the Foundation Mat Design

3.8.5.3.1 Design Codes and Standards

In Revision 13 of SSAR Section 3.8.5.2, Westinghouse stated that the applicable codes, standards and specifications used for the design of the NI foundation mat are described in SSAR Section 3.8.4.2. As stated in Revision 14 of SSAR Sections 3.8.4.2 and 3.8.4.4, the design and analysis procedures for the SC-I reinforced concrete structures (other than the containment internal structures), including assumptions on boundary conditions and expected behavior under loads is in accordance with ACI-349-90 Code. SSAR Section 3.8.4.4 also stated that the ductility criteria of ACI-318-95 Code, Chapters 12 and 21, are considered in the detailing, placing, anchoring, and splicing of the reinforcing steel.

As discussed in Section 3.8.4 of this report, the staff accepted the use of ACI 349-90 for the design of AP600 reinforced concrete structures. The staff also accepted the use of ACI 318-95 ductility criteria for the detailing, placing, anchoring, and splicing of the reinforcing steel. However, the use of ACI 349-90 for the design of the NI foundation mat is not acceptable to the staff. This is because of an error that exists in the pre-1990 editions of the ACI code regarding shear design of thick sections.

To resolve this issue, Westinghouse submitted Revision 22 of SSAR Section 3.8.4.4.1 for review. As a result of its review of this SSAR revision, the staff concludes that the use of the ACI 349-90 and ACI 318-95 (ductility criteria) for the design of the AP600 foundation mat is acceptable. Details of the staff's review and the basis for the conclusion are discussed in Sections 3.8.5.5 and 3.8.5.7, below. However, any proposed change to the use of the ACI 349-90 and ACI 318-95 (ductility criteria) for the design of the AP600 foundation mat, as documented in SSAR Sections 3.8.4.4.1 and 3.8.5.5, will require NRC approval prior to implementation of the change.

3.8.5.3.2 Design Concept

The AP600 NI houses all Category I structures and is supported on a single foundation mat with exterior and interior walls rigidly connecting the shield building (including the reactor containment vessel and containment internal structures) and auxiliary buildings to the foundation mat. This system of inter-connected vertical shear walls and horizontal slabs results in a monolithic reinforced concrete structure design such that all loads applied to the structure engage all parts of the structure. For example, large lateral seismic loads applied to the shield building will be transferred in part to the auxiliary building walls. The seismic and dead loads generated from the shield building, together with similar loads from the auxiliary building are then transferred to the foundation mat. The foundation mat model developed by Westinghouse considers the effect of interaction between the foundation mat and the supporting soil by representing soil flexibility by means of a system of horizontal and vertical spring elements uniformly distributed along the base of the foundation mat and attached to the foundation mat nodes. Horizontal bearing reactions on the side walls below grade were neglected in this analysis. The foundation mat is then designed as a flexural member spanning between the various walls of the auxiliary building.

3.8.5.3.3 Finite Element Analysis Model of the NI

A large finite element model, as depicted in SSAR Figure 3.8.5-2 (Revision 11), was prepared by INITEC (consultant to Westinghouse) for use as input to the ANSYS Code (Version 4.4.A135). Equivalent pseudostatic nodal loads were generated for this model from the results of seismic analyses performed for the NI structures under an SSE. The finite element model extends to Elevation 71.93 m (236.0 ft) of the shield building and to Elevation 30.48 m (100.0 ft) of the auxiliary building. Interior auxiliary building walls on the north side of the foundation mat at Column Lines K, L, M and P extend to Elevation 47.55 m (156.0 ft).

For the various load combinations evaluated, iterative analyses were performed to incorporate the potential effects of lift-off of the foundation mat. Westinghouse performed the foundation mat analysis using the iterative process for the 12 most critical load combination cases. Because of the coarseness of the finite elements used to model the foundation mat, the results of this analysis were restricted to generating the peak soil bearing pressures and in-plane membrane and shear forces. These analyses assumed a uniform soil stiffness (modulus of subgrade reaction) of 81,709 kNewton per square meter, per meter (520 kips per square foot, per foot). This soil stiffness was represented in the model by means of equivalent Winkler springs connected to the nodes of the foundation mat.

3.8.5.3.4 Local Simplified Analyses and Design

Using the soil bearing pressures generated from the analyses of the large model, further simplified analyses were performed considering sections of the foundation mat spanning as flexural members between the walls of the auxiliary building. On the basis of the bending moments and shears calculated, combined with any principal tensile forces determined from the large ANSYS model, flexural and shear steel requirements were determined.

3.8.5.3.5 Parametric Studies

Westinghouse performed further simplified analyses in which the soil spring stiffness was assumed to vary linearly from the edges of the foundation mat to the center. These results indicated that variations in soil stiffness between the center and edges of the foundation mat would not result in higher shear or bending moments than those for the uniform soil stiffness case.

An additional study was performed for the north area of the foundation mat in which the foundation mat was considered as a continuous span flexural member supported by walls spanning in the east-west direction. The stiffness of soil springs were varied by ±20 percent in alternate spans of the foundation mat. The results of this analysis indicated that an increase in flexural reinforcing steel would be sufficient to cover such increases in bending moments.

3.8.5.3.6 Constructibility and Construction Sequence

Westinghouse indicated in Revision 15 of SSAR Sections 2.5.4 and 3.8.5.4 that the excavation for the NI will be constructed with vertical faces, and that no backfill material will be placed against the embedded exterior walls of the NI structures. In SSAR Section 2.5.4, Westinghouse also described a proposed excavation method that consists of a shotcrete grout wall which is anchored to the face of the excavation by means of soil "nails" (metal rods) to retain the soil during the excavation. As the NI excavation progresses downward, metal rods are inserted into holes that are drilled near-horizontally into the adjoining undisturbed soil. Grout is injected under pressure into holes to bind and anchor the "nails" to the soil. A welded steel wire mesh in then hung onto the nail heads, and a wall of shotcrete, approximately 15 cm (6 in) thick, is sprayed onto the mesh. The mix of the shotcrete typically consists of a non-expansive grout and pea-gravel combination, which is blown against the soil face and onto the mesh fabric under a pressure of about 34 MPa (5,000 psi). The hardened shotcrete then serves to retain the entire soil mass surrounding the NI.

The process of soil nailing produces a vertical surface wall down to the bottom of the excavation, which can be used as the outside form work for pouring the reinforced concrete walls of the NI structures. Because the detailed design and construction of the soil-nailed walls will depend to a great extent on local soil conditions, Westinghouse stated in the SSAR that COL applicants will provide the information concerning the design of specific soil nailing systems. One result of this proposed construction technique is that the soil immediately surrounding the NI consists of natural in-situ materials only, which have continuous properties in the horizontal and vertical directions. Because this configuration complies with the assumptions made in the seismic analyses performed to assess the seismic responses of the NI structures, the proposed excavation method is considered acceptable. However, under conditions where relatively softer, cohesive soils exist at the site, the soil nailed wall may not be an appropriate excavation/construction method.

During the meeting held August 4 through 8, 1997, Westinghouse and its consultant, Paul C. Rizzo Associates (PCRA), indicated that other excavation and construction methods may be used by the COL applicants if the site conditions are different from those proposed in the SSAR for which the soil nailing technique is not appropriate. The important characteristic of any other construction method that may be used by the COL applicant is that a vertical wall of in-situ soils remains in a relatively undisturbed state immediately adjacent to the NI, with no backfill soils placed against the walls. For this case, the staff concludes that these other construction methods satisfy the configuration assumptions made in the seismic response analyses such as the SSI analysis and is considered acceptable. However, if any other construction technique which requires excavation and backfill of large areas surrounding the NI is used, the procedures and criteria for installing the backfill must be submitted by the COL applicants for review and approval. In addition, an evaluation of the effect of any alternative construction procedures on the seismic responses from the SSI analyses of the NI structures must be performed to assess the impact of the backfilled soil.

For use of the shotcrete wall and mudmat construction technique, Westinghouse indicated that after the completion of soil excavation and the placement of vertical shotcrete walls, a concrete mudmat, approximately 25 cm (10 in) in thickness, with a steel mesh reinforcement installed in place will be poured along the bottom of the excavation. Then the shotcrete wall and foundation mudmat will serve to protect the site by providing a working surface during the construction of the NI foundation mat. The entire process of excavation, defined by the vertical shotcrete soil nailed walls and the contiguous foundation mudmat, is also required to maintain the site in a dry condition after the completion of construction to minimize the requirement of special corrosion protection of the foundation mat reinforcements. The shotcrete and mudmat material as specified in Revision 15 of SSAR Section 2.5.4 also incorporates a crystalline waterproofing material additive in the mixes along with water stops placed at the joint between the vertical shotcrete walls and the mudmat. These additions will help to prevent water from infiltrating through small cracks which may develop in these materials during constructions. The steel wire mesh placed in both the soil-nailed wall and the mudmat can limit potential cracks that may develop because of construction and long-term settlements of the NI structures. In addition, the shotcrete material must be continuous with no windows through which water can easily penetrate. As a result of the meeting held on August 4 through 8, 1997, Westinghouse committed in Revision 17 of SSAR Section 2.5.4.6 that this requirement is a COL action item. On the basis of the commitments by Westinghouse, the staff concludes that the procedure to be used for the excavation and construction of the NI foundation is consistent with the common industry practice and is acceptable. Also, because this construction procedure will prevent water from infiltrating through concrete cracks, the staff's concern regarding the use of coated rebars in the foundation mat (DSER Open Item 3.8.5-14) is considered resolved.

During this meeting, the staff's review identified an issue regarding the potential detrimental effects of construction-induced settlements. The shear and bending moments developed by the construction-induced settlements should also be considered in the design of the foundation mat. From the review of calculations presented during the meeting by Westinghouse, the staff concluded that Westinghouse should follow the procedure described below to incorporate the effects of soil settlement and construction schedules on induced stress resultants as additional loads in the design of the foundation mat:

(1) Obtain stress resultants at critical locations on the NI from the following load cases:

    (a) Case 1 of PCRA analysis for the deep clay site

    (b) Case 2 of PCRA analysis for the sand/clay site

    (c) the analyses on the basis of the simplified INITEC's Winkler spring models

(2) Obtain the maximum values of stress resultants at each of the critical locations from the cases in Item (1) above. This list of maximum stress resultant values is then to be considered as the resultant dead load stress resultant solution.

(3) The stress resultants because of dead load from Item (2) above are then to be combined with all other stress resultants to obtain stress resultants in satisfying the various load combination requirements. For each load case, the list of maximum stress resultant values represents the elastic demand on the NI.

(4) The elastic demand is then to be compared with the section capacities of the concrete structural elements of the NI to judge the adequacy of the design.

(5) Because soil bearing pressures calculated from the PCRA and INITEC analyses are not sensitive to the size of finite elements used in the model, they can be used in the localized analysis with more finely discretized finite elements to calculate final design moments and shears.

3.8.5.3.7 Dynamic Stability of the NI

In addition to the above-mentioned finite element analyses of the foundation mat, Westinghouse performed evaluations of the dynamic stability of the NI against overturning and sliding using the moment balance method. In this approach, the factor of safety against overturning is defined as the ratio of the restoring moment to the overturning moment caused either by wind or an SSE. The effect of potential buoyant forces from a high-water table because of either flooding or natural conditions was included in these evaluations.

The use of the moment balance method for the foundation mat dynamic stability evaluation meets the guideline of SRP Section 3.8.5 and is acceptable to the staff.

3.8.5.4 Major Review Areas

The staff review of the foundation mat design focused on the following areas:

- detailed review and evaluation of the ANSYS foundation mat model and analysis results
- review of the post-processor program ARMA2 (an in-house computer code developed by INITEC) used to convert the ANSYS output to design section configurations
- review of the SSAR and its revisions, and design criteria
- review of Westinghouse's responses to the technical concerns raised by the staff as a result of the SSAR review and meeting discussions
- evaluation of the foundation mat during construction and limitations on construction sequences
- review of sampled design calculations

The staff's review of early revisions of the SSAR identified a number of open items (in addition to those discussed in Section 3.8.5.5 below). These open item were documented in the AP600 DSER for the AP600 NI foundation design. Through a series of review meetings, most of these open items were closed. The details and the basis for resolving these open items are summarized in Section 3.8.5.7 of this report.

3.8.5.5 Major Issues Identified

As a result of its review of samples of the final design calculations, the staff raised a number of concerns regarding the design adequacy of the NI foundation mat. These three issues and their resolution are summarized as follows:

(1) The potential effects of construction-induced settlements on developed moments and shears in the NI were reviewed on the basis of the calculations performed by PCRA to incorporate soil settlement and construction schedule effects on induced stress resultants. The results of PCRA's calculation indicate that the construction induced settlements can be suitably accounted for by using the procedure described in Section 3.8.5.3.6 above. Westinghouse should document this procedure in the SSAR and commit to follow this procedure to design the various elements of the NI for incorporating the effects of additional loads induced by construction settlements.

To resolve the issue regarding the inclusion of construction induced loads in the AP600 foundation mat design, Westinghouse submitted Revision 2 of its response to Open Item 3.8.5-10 (the letter dated February 9, 1998, NSD-NRC-5562) and Revision 22 of SSAR Section 3.8.5.4.3 for staff review. In this SSAR revision, Westinghouse incorporated the procedure described in Section 3.8.5.3.6 above. In addition, Westinghouse selected the five most critical locations (from the stress analysis on the basis of the combination of applicable loads except loads induced by settlements because of construction) and performed an evaluation on the adequacy of the existing foundation mat design by indicating that the member forces because of combined load conditions which govern the design of the foundation mat are within design allowables. The changes and response to Open Item 3.8.5-10 by Westinghouse resolved the staff's technical concern regarding the foundation mat design under the combined load conditions including stresses induced by construction settlements. The resolution of this issue is discussed in detail in Section 3.8.5.7 (Open Item 3.8.5-10) of this report.

(2) The effects of local variations in soil stiffness could increase design values for bending moments and shear.

In response to this issue, in Revision 17 of SSAR Section 3.8.5.4.4, Westinghouse stated that the design moments and shears are 20 percent higher than needed for uniform sites to accommodate the nonuniform sites defined in SSAR Section 2.5.4.5. According to the common engineering practice and the staff's review experience, an increase of the design moments and shears by 20 percent will accommodate the effects because of nonuniform sites. This issue is considered resolved.

(3) During the meeting conducted on August 4 through 15, 1997, the staff reviewed the final design of the 6-foot thick foundation mat. As a result, three design concerns were identified by the staff:

(a) According to Chapter 21 (Section 21.3.3.4) of ACI 318-95 Code, stirrups used as shear reinforcement have to be provided with a 135° hook at both the top and bottom faces of the foundation mat. However, only stirrups (with 90° hook at the bottom face and a 135° hook at the top face of the mat) were provided by Westinghouse for resisting shear force. According to Westinghouse, because the flexural steel is spaced 15 cm (6 in) on center (top and bottom), the provision of 135° hooks is not practical. As a result, the 1.8 m (6 ft) thick foundation mat does not appear to be constructible with such heavy reinforcements.

(b) According to the ratio of span-to-depth, the NI foundation mat should be classified as and designed to the requirements of deep flexural members (Section 11.8.1 of ACI 349-90). However, on the basis of some test results, the ACI 318 Code Committee determined that errors were identified in the 1983 code (the same errors were in the ACI 349 code, because the 349 code is on the basis of the 318 code) and these errors could result in a nonconservative design for deep flexural members. As a result, ACI 318-95 was revised to correct these errors. For the AP600 foundation mat with exterior and interior stiffening walls, the foundation mat should be classified as continuous deep flexural members (Sections 11.8.1 and 11.8.3 of ACI 318-95). ACI-318-95 Code (Section 11.8.5) requires that the critical section for shear is to be located at 0.15 times the span length from the support edge with reinforcing steel over the full span and the design should be on the basis of Section 11.8.3. However, Westinghouse did not treat the foundation mat as a deep flexural member. The shear reinforcement used in the design was on the basis of a much reduced shear force at a section which is further away at a distance of the effective depth of the mat. The correct amount of shear reinforcement would require the use of larger reinforcing bars which would be spaced at a distance not more than "d/2" throughout the length of the member.

(c) The design calculation of the foundation mat was performed using soil stiffness variation in alternate spans. While this design approach will maximize bending moments in the mid span, it will not indicate increases in shear force because of soil variation. If the soil variation is such that the soil stiffness is constant over two adjacent spans, and spans on either side are with lower or higher stiffness, the maximum shear force will occur at the wall between the two spans with the greatest stiffness. This geometry was not considered in the Westinghouse design.

In response to these issues, Westinghouse submitted Revision 20 of SSAR Sections 3.8.5.4.4 and 3.8.5.5 for staff review. As discussed in Section 3.8.5.7 (Open Item 3.8.5-9) of this report, the issue regarding the structural criteria for the design of reinforced concrete NI foundation mat is closed.

3.8.5.6 Review History

Since the issuance of the DSER for the AP600 standard plant design, the staff reviewed Revisions 2 through 17 of the SSAR. In addition, the staff also conducted a total of seven review meetings with Westinghouse. The purpose of these review meetings was to discuss the following issues:

- the issues identified as a result of the SSAR review
- Westinghouse's resolution of open items documented in the DSER
- Westinghouse's resolution for the issues identified in the previous meetings
- the staff's confirmatory analysis results

The issues identified and the actions and agreements reached in each meeting are summarized in the following subsections. The final status of the DSER open items and issues identified during the following seven meetings are discussed in Sections 3.8.5.6 and 3.8.5.7 below.

(1) Review Held on July 11 Through 14, 1994

In this meeting, the staff reviewed Westinghouse's Design Calculation 1010-CCC-001 for the AP600 NI foundation mat. Westinghouse, with the 3-D finite element model, combined specified load conditions and analyzed the foundation mat using the ANSYS computer code. The results obtained from the ANSYS analyses were then input into INITEC's in-house post processor program (ARMA2) for the reinforced concrete design. From the staff's review of this report, the following concerns were identified:

(a) Among the 12 most critical load combination cases, only two combined load conditions, namely normal and extreme combined load conditions, were considered in Westinghouse's analyses and design of the NI foundation mat. In neither of the load combinations were the design-basis accident load and associated thermal effect considered. Without inclusion of the accident load and thermal effect in the combined load conditions, the Westinghouse's analyses are inconsistent with the commitment in Table 3.8.4-2 of the SSAR.

(b) While the plots of the foundation mat elements and nodes were provided in the design calculations, the elements of the containment internal structures and walls were not shown.

(c) Even though the number of finite elements in the model is large, the size of the individual finite elements representing the foundation mat are quite large compared to the spans between the auxiliary building walls. Therefore, errors in computed shears and moments could occur.

(d) The basis for determining the element size was not discussed in the design calculations.

(e) When the foundation mat under the shield building was modeled, the horizontal element planes were vertically offset from radially arranged adjacent elements. This offset could cause spurious bending moments and shears if the in-plane forces are present in these elements.

(f) A total of 149 horizontal (north-south and east-west) soil springs were used to connect the soil foundation and the foundation mat nodal points. However, the number of soil springs are much less than the total number of nodal points. Westinghouse assumed that the horizontal spring locations were uniformly distributed among the foundation mat elements. When certain portions of the foundation mat uplifted because of SSE loads, the horizontal soil springs within this portion of foundation mat still provided restraints to the foundation mat. This is unrealistic when compared with the true behavior of the foundation mat.

(g) The development of the axial forces (in-plane membrane forces) in the foundation mat should depend on the locations of, and the magnitudes of, the horizontal restraints. However, the horizontal restraints provided by the exterior walls and the edge of the foundation mat were neglected in the calculations. Neglecting these horizontal restraints may cause under-design of the flexural reinforcements.

(h) The package of the ANSYS computer code output was not available for review.

(i) The review of the verification package for the INITEC post-processor program "ARMA2" indicated that some significant errors were made in the determination of the concrete shear capacity, calculation of applied shear forces, and calculation of flexural reinforcements for bending and axial forces. A summary of these concerns is as follows:

- Concrete shear capacity was not checked for beam action as required by ACI 349-85, Section 11.11.1.

- The shear capacity was determined using the overall depth "h," rather than effective depth "d."

- Applied shear forces were not conservatively estimated using the maximum length of the finite element rather than the shortest element length.

- The calculated axial compressive forces were consider for reducing both flexural and shear reinforcing steel. This approach may lead to an unconservative design.

- Value of shear forces was inconsistently identified in terms of kips or pounds.

Westinghouse agreed to correct the errors identified by the staff and to provide the English language version of this code for review. The staff's review of the code validation package is discussed under Open Item 3.8.5-8 below.

(2) Review Meeting held on March 2, 1995

The following agreements were reached between the staff and Westinghouse:

(a) Foundation mat under the shield building needs additional study.

(b) Additional load combinations including the effects of accidental pressure should be considered.

(c) The Steinbrenner method is acceptable for calculating coefficients of subgrade moduli.

(d) Local analyses of the foundation mat would be performed in order to determine design values for bending moments, shears, and in-plane forces.

(e) A uniform distribution of horizontal springs representing the distribution of lateral soil stiffness is conservative.

(f) The design of sections in the foundation mat will include the calculated shears, bending moments, and axial tensile forces. Compressive axial forces, which could theoretically serve to reduce reinforcing steel requirements for both shear and bending moments, will be assumed to be zero.

(g) A range of soil stiffness will be used to determine the effects of anticipated variation of soil properties on the peak design values.

(h) The INITEC post-processor programs will not be used.

(i) Top and bottom flexural reinforcing size and spacing will be identical in each direction.

(3) Review Meeting held on June 12 through 16, 1996

The following issues were discussed at this meeting:

(a) Settlement effects attributable to construction procedures need to be considered in computing moments and shears developed in the foundation mat.

(b) The potential effects of variations in soil properties likely to be encountered at various sites need to be evaluated. These include the potential effects of local hard and soft spots in various locations of the foundation mat as well as average stiffness properties ranging from soft soils to hard rock.

(c) The analysis and design procedure was changed so that soil bearing pressures, determined on the basis of the soft to medium soil case, would be used as loading on the foundation mat.

(d) In-plane shears should be included in the design in that they would increase principal tension forces.

(e) The analysis for bending moments and shear would be determined on the basis of MATHCAD formatted calculations. The MATHCAD calculations should, in turn, be determined on the basis of ACI 349-85, Sections 11.3 or 11.11, or whichever is greater.

(4) Review Meeting held on July 11, 1996

A set of evaluation results presented by the staff indicated that locally stiffer and softer soil could cause increases in shears and moments in sections of the foundation mat spanning as flexural members between walls. The actions and agreements resulting from the meeting were as follows:

(a) Lateral support from side walls would be conservatively neglected when calculating bearing soil reactions.

(b) Peak soil bearing from the 12 seismic load combination cases would be used in hand computations used to design sections of the foundation mat.

(c) Actual variation in reactions will be considered in the local finite element analysis of critical bays.

(d) East-west reinforcement is designed assuming one-way action calculated using a three span moment equation.

(e) The design of north-south reinforcements at the exterior wall would be presented to the staff in the next review meeting.

(f) North-south reinforcements at the shield building are calculated on the basis of the local finite element analysis.

(g) Local soil variability effects will be addressed by considering a potential rock pinnacle condition.

Westinghouse described the structural system as an "egg crate," such that the auxiliary building walls transfer soil bearing loads to the shield building, or as shield building seismic loads to the auxiliary building mat foundation.

(5) Review Meeting held on December 9 through 13, 1996

The major issues discussed at the meeting concerned methods for incorporating the potential effects of local soil variability in the design as well as issues associated with including the effects of construction settlements. The NRC staff provided the following comments:

(a) The staff noted that when the shear walls act as stiffeners to reduce the out-of-plane bending of the foundation mat, they behave like deep beams.

(b) Validation of INITEC's post processor programs could not be reviewed because of unavailability of English documentation.

(c) Shear reinforcing steel provided by vertically oriented stirrups should have 135° hooks at both the top and bottom.

(d) The torsional provisions in ACI 318-95 should be used in evaluating the design.

(e) The effects of large in-plane foundation mat shears should be considered in the design.

(f) Special considerations should be used in the design of both flexural and shear reinforcing.

(g) For the 46 cm (18 in) thick section of the foundation mat under the elevator pit, the critical section for shear was incorrectly established at a distance on the basis of the 1.83 m (6.0 ft) thickness instead of the 46 cm (18 inch) thickness.

(6) Review Meeting held on August 4 through 8, 1997

The following technical issues and concerns were identified at the meeting:

(a) The Design of the Foundation Mat

- The design of the foundation mat under the auxiliary building utilized finite element analyses and hand computations of areas of the foundation mat.

- The design for shear reinforcing did not consider requirements for deep flexural members. The code provisions for deep flexural members in ACI 349-85 are on the basis of the provisions of ACI 318-83. Research indicated that the provisions in ACI 318-83 are valid for simple span deep members, but they are not adequate for continuous members. The ACI 318-89 provisions for deep flexural members, which distinguished between simple and continuous spans, were continued in ACI 318-95. In fact, the soon-to-be issued revision of ACI-349 code will contain these provisions.

- The ratio of clear span between walls to depth of the foundation mat indicated that the shear design of the foundation mat should be treated as a deep flexural member.

(b) Details of Shear Reinforcement

- SSAR Section 3.8.4.4 commits to the use of the ductility criteria of ACI 318-95 Code, Chapter 21 in detailing, placing, anchoring and splicing of the reinforcing steel. According to Chapter 21 of ACI 318-95 Code, stirrups used as shear reinforcement have to be provided with a 135° hook at both the top and bottom faces of the foundation mat. However, only stirrups with a 90° hook at the bottom face and a 135° hook at the top face were provided by Westinghouse for resisting shear. Because the flexural reinforcing steel is spaced at 15 cm (6 in) on center in both top and bottom faces, the 1.83 m (6 ft) thick foundation mat with 135° hooks stirrups does not appear to be constructible with such heavy reinforcements.

(c) The Variability of Soil Foundation

- The flexural reinforcing steel would be increased by 20 percent. A variation in soil stiffness of ±20 percent in alternate spans indicated that the increased flexural reinforcing steel would be sufficient.

- The staff indicated that if the variation in soil stiffness were constant over two adjacent spans, then there would be an increase in shear.

(d) The Stiffness of Shear Walls

A calculation (1200-CCC-107) performed by Bechtel Corporation (a consultant to Westinghouse) using the BSAP computer code was reviewed by the staff. This calculation considered the effect on shear wall stiffness of openings in walls in Column Lines J and K, which were not considered in the previous finite element models. However, the conclusion that only the reactions at the base of the walls varied from the previous calculations was not consistent with the results reported. Furthermore, the actual stiffness of walls in Column Lines J and K were underestimated and overestimated, respectively.

(e) The Elevator Pit

The thickness of the foundation mat under the elevator pit is only 46 cm (18 in) thick. The critical section for the design of shear reinforcing steel in this area was reviewed and found acceptable.

(f) The Construction Settlement Effects

On the basis of the staff's past licensing review experience, the potential effects of nonuniformly distributed construction loads on the foundation mat, especially for a foundation mat with such large dimensions and irregular shape, can be very significant and may cause severe foundation cracks. Westinghouse was requested to provide the basis for demonstrating the design adequacy in coping with the unevenly distributed construction loads.

(7) Review Meeting held on November 4 and 5, 1997

As a result of the meeting on August 4 through 8, 1997, the staff issued its position regarding the design of AP600 NI foundation mat on August 29, 1997. Westinghouse provided its response to the staff's position on October 17, 1997. The purpose of this meeting was to discuss the staff's review results of Westinghouse's submittal dated October 17, 1997. The two major issues identified and their resolution are summarized below:

(a) For the issue identified during the meeting on August 4 through 8, 1997 regarding the use of stirrups with a 135° hook (seismic hook) at both the top and bottom face of the foundation mat for the shear reinforcement, Westinghouse proposed to use the headed anchor to replace the seismic hooks. Westinghouse also provided, in the meeting, test results published by the manufacturer for the staff review. The staff found that the stirrups with headed anchors are equivalent to the use of 135° bends at both ends of the shear reinforcement and concluded that this issue was resolved.

(b) According to the ratio of span to depth, the AP600 foundation mat should be classified as a continuous deep flexural member. Westinghouse should design the foundation mat in accordance with the ACI 318-95 requirements (Section 11.8.3).

3.8.5.7 Resolution of DSER Open Items and Issues Identified in the Review Meetings

The AP600 DSER issued in November 1994, identified a number of open items related to the design of the NI foundation mat. Since then, the staff conducted a series of review meetings. The purpose of these meetings was to review Westinghouse's responses to these open items, and to discuss with Westinghouse the new issues identified as a result of the staff's review of later SSAR revisions and response to the open items. The final status of these open items is summarized as follows:

<u>Open Item 3.8.5-1</u>

Westinghouse should provide exact dimensions of the foundation mat in the SSAR.

In Revision 7 of SSAR Figures 3.7.1-16 and 3.7.2-12, Westinghouse provided overall dimensions for the foundation mat, distance between column lines and distance between the edge of foundation mat and center of containment shell. These dimensions allow design engineers to develop the dynamic model and perform seismic analyses for the NI structures. Therefore, Open Item 3.8.5-1 is closed.

Open Item 3.8.5-2

For the use of ACI-349-90 Code in the design, Westinghouse should identify the differences between the 1980 version of ACI-349 Code and Revision 1990 of ACI-349 Code and submit an evaluation of the differences to the staff for review and acceptance.

On the basis that (1) the review of Westinghouse's evaluation on the differences between the 1980 code and the 1990 revision to the 1985 code at the April 25 through 27, 1995 and June 12 through 16, 1995, meetings found that these differences are insignificant, (2) the 1985 code has been accepted by the staff for the CE (3) the 1990 revision of the 1985 code pertains to Appendix B of the code, which is addressed under Open Item 3.8.4.2-4 of this report and (4) the evaluation discussed under Open Item 3.8.4.2-2, the staff concludes that Open Item 3.8.5-2 is closed.

Open Item 3.8.5-3

Westinghouse should consider the effect of accident pressure combined with other design loads in the foundation mat design.

Westinghouse, in Revision 7 of SSAR Section 3.8.5.3, stated that the design containment pressure is included in the design of NI foundation mat as an accident pressure in Load Combinations 5, 6, and 7 of SSAR Table 3.8.4-2. Westinghouse's SSAR commitment meets the SRP Section 3.8.4. guideline. Therefore, Open Item 3.8.5-3 is closed.

Open Item 3.8.5-4

In the dynamic stability (sliding and overturning) evaluation of the NI, Westinghouse should (1) include the buoyancy effect, and (2) use the moment balance method (factor of safety against overturning is defined as the ratio of the restoring moment to the overturning moment because of an SSE) rather than the energy balance method in the analysis.

In Revision 3 of SSAR Section 3.8.5.5.4, Westinghouse replaced the energy balance method by the moment balance method for checking the overturning of the NI. Also, In Revision 13 of SSAR Section 3.8.5.5.3 and Table 3.8.5-1, Westinghouse included the buoyancy effect in the load combinations for evaluating the potential of overturning and sliding. Westinghouse's SSAR commitment meets the SRP Section 3.8.5 guideline and, therefore Open Item 3.8.5-4 is closed.

Open Item 3.8.5-5

In the analysis and design of the foundation mat, Westinghouse included the elastic foundation stiffness of the soil included in the basement elements by a system of horizontal spring elements uniformly distributed on the foundation mat nodes to represent the flexibility of the soil foundation. As described in early revisions of the SSAR, Westinghouse considered only the horizontal soil springs to represent the flexibility of the soil foundation without including the vertical soil springs. This is not acceptable to the staff.

In Revision 7 of the SSAR, Westinghouse included the missing soil springs for calculating soil bearing stresses. The staff's review found that the approach used by Westinghouse is consistent with the common industry practice. On this basis, Open Item 3.8.5-5 is closed.

<u>Open Item 3.8.5-6</u>

Because foundation mat lift-off occurs under most of the combined load conditions (40 out of 48 load combination cases), an iterative process was applied in the analyses. Westinghouse committed in the SSAR that the foundation mat analysis using the iterative process is performed for the 12 most critical load combination cases. These 12 critical load combination cases were determined on the basis of the results from the first linear analysis (the analysis without iteration). However, the staff, during review meetings, found that only two of the 12 critical load combination cases were considered in the design. This is not consistent with the Section 3.8.5.4 commitment.

In the December 9 through 13, 1996, meeting, the staff reviewed foundation mat design calculations and found that the 12 worst load combination cases had been considered in the design. These 12 cases cover the combinations of dead, live, SSE and accident pressure loads. As for the design adequacy of the foundation mat under these combined load conditions, the staff's review is discussed under other open items. On this basis, Open Item 3.8.5-6 is closed.

<u>Open Item 3.8.5-7</u>

The factor of safety against sliding and overturning of the NI structures because of tornado and wind was not provided in the SSAR. In addition, Westinghouse should provide rationale for the buoyancy force criterion for the submerged structures.

During the review of design calculations conducted on August 4 through 15, 1997, Westinghouse presented its evaluation results of dynamic stability (sliding and overturning) of the NI structures. The overturning moment and base shear used by Westinghouse included the effects because of the design changes resulted from the post-72 hour action requirements. As shown in the calculation, the factors of safety against both sliding and overturning are 1.1 and 1.101, respectively. The calculated factors of safety is higher than the allowable factor of safety specified in SRP Section 3.8.5 which is 1.1. Therefore, the staff concludes that Open Item 3.8.5-7 is closed.

<u>Open Item 3.8.5-8</u>

Westinghouse should provide the validation package of INITEC's in-house computer programs for review. In addition, Westinghouse should verify the adequacy of the post-processed results which were used to produce the complete reinforcing steel requirements from the results of the ANSYS analysis.

In the August 11 through 15, 1997, meeting, the staff reviewed the validation package (English language version) for INITEC's in-house computer program ARMA2 and found that INITEC had properly tested this computer code by different test problems and the results showed a good comparison with those from other public domain computer codes. On the basis that the computer program ARMA2 will provide reasonable results on the basis of the test problems reviewed, Open Item 3.8.5-8 is closed.

Open Item 3.8.5-9

Westinghouse should demonstrate the adequacy of using a 6-foot thick foundation mat, especially the foundation mat underneath the containment vessel. In its position letter dated November 4, 1994, the staff offered two options for Westinghouse to consider in resolving this issue (1) demonstrating that the final foundation mat design can accommodate the effects of soil stiffness variations of hard and soft spots underneath the foundation mat, and (2) using different foundation mat thicknesses for a foundation mat with uniform soil foundation stiffness (such as rock sites) and for a foundation mat with non-uniform soil foundation stiffness (such as soil sites with hard and soft spots) and submitting the completed design of each foundation mat thickness for the staff review and approval.

During the meeting conducted on August 4 through 15, 1997, the staff reviewed the final design of the 1.83 m (6 ft) thick foundation mat. As a result, on the basis of the commitment made by Westinghouse in SSAR Section 3.8.4 (Revision 15) that the design and analysis procedures for SC-I structures are in accordance with ACI-349 Code for reinforced concrete structures; and the ductility criteria of ACI 318 Code, Chapters 12 and 21, are considered in the detailing, placing, anchoring and splicing of the reinforcing steel. Three design concerns were identified by the staff:

(1) According to Chapter 21 (Section 21.3.3.4) of ACI 318-95 Code, stirrups used as shear reinforcement have to be provided with a 135° hook at both the top and bottom faces of the foundation mat. However, only stirrups, with a 90° hook at the bottom face and a 135° hook at the top face of the mat, were provided by Westinghouse for resisting shear force. The flexural steel is spaced at 15 cm (6 in), on center, top and bottom. Therefore, the provision of a 135° hook is not practical. The 1.83 m (6 ft) thick foundation mat does not appear to be constructible with such heavy reinforcements.

(2) According to the ratio of span to depth, the NI foundation mat should be classified as and designed for the requirements for deep flexural members (Section 11.8.1 of ACI 349-90). However, on the basis of some test results, the ACI 318 Code Committee determined that errors were identified in the 1983 code (the same errors were in the ACI 349 code because the 349 code is on the basis of the 318 code) and these errors could result in an unconservative design for deep flexural members. As a result, ACI 318-95 was revised to correct these errors. For the case of AP600 foundation mat with exterior and interior stiffening walls, the foundation mat should be classified as a continuous deep flexural members (Sections 11.8.1 and 11.8.3 of ACI 318-95). ACI-318-95 Code (Section 11.8.5) requires that the critical section for shear is to be located at 0.15 times the span length from the support edge with reinforcing steel over the full span and the design should be on the basis of Section 11.8.3. However, Westinghouse did not treat the foundation mat as a deep flexural member. The shear reinforcement used in the design was on the basis of a much reduced shear force at a section which is further away from the edge at a distance of the effective depth of the mat. The correct amount of shear reinforcement would require the use of larger reinforcing bars which would be spaced at a distance not more than "d/2" throughout the length of the member.

(3) The foundation mat calculation was performed using soil stiffness variation in alternate spans. While this design approach will maximize bending moments in the mid span, it will not indicate increases in shear force because of soil variation. If the soil variation is such that the soil stiffness is constant over two adjacent spans, and spans on either side are with lower or higher stiffness, the maximum shear force will occur at the wall between the two spans with the greatest stiffness. This geometry was not considered in the Westinghouse design.

On the basis of the discussion above, the staff concluded that Westinghouse failed to demonstrate that the proposed foundation mat design is adequate with respect to the previously issued staff position. The final design of the foundation mat did not meet certain code requirements committed to in the SSAR.

For the concern of seismic hooks used for the shear reinforcement (Item 1 above), Westinghouse proposed the use of headed anchors (instead of 135° bends) at both ends of the shear reinforcement (stirrups) during the meeting on August 4 through 8, 1997. Westinghouse also provided, in the meeting, test results published by the manufacturer for the staff review. As a result, the staff found that the shear reinforcement with headed anchors is equivalent to the use of 135° bends at both ends of the shear reinforcement and concluded that this issue is technically resolved. However, Westinghouse should document this commitment in a future revision of the SSAR.

With regard to the concerns of the use of design code and soil stiffness variation (Items 2 and 3 above), Westinghouse's response and the staff's evaluation are summarized below:

- In the submittal dated November 24, 1997, Westinghouse provided a draft of SSAR Section 3.8.5.5 to commit that the foundation mat below the auxiliary building is designed for shear in accordance with the requirements for continuous deep flexural members in ACI 318-95 (Section 11.8.3). Specifically, Westinghouse committed to the following measures:

  - The design for shear is on the basis of Sections 11.1 through 11.5 of ACI 349-90 except that the critical section measured from the face of the support is taken at a distance of $0.15l_n$.

  - Shear strength, V_n, is not taken greater than $8(f_c')^{1/2}b_wd$ when l_n/d is less than 2. When l_n/d is between 2 and 5,

    $V_n=2/3(10+l_n/d)(f_c')^{1/2}b_wd$

  - Area of vertical shear reinforcement, A_w, is not less than $0.0015b_ws$ and the spacing of shear reinforcement, s, does not exceed d/2 nor 0.6 m (2 ft).

  - Shear reinforcement required at the critical section is used throughout the span.

  Westinghouse's commitments in the draft SSAR meet the requirement of ACI 318-95 and, therefore, are acceptable. However, Westinghouse needs to include the above commitments in the SSAR. This issue remained open.

- In Revision 17 of SSAR Section 3.8.5.4.4, Westinghouse stated that the design moments and shears are increased by 20 percent above the required for uniform sites to accommodate the nonuniform sites defined in SSAR Section 2.5.4.5. According to the common engineering practice and the staff's review experience, to increase the design moments and shears by 20 percent will accommodate the effects because of nonuniform sites. This issue is considered resolved.

In response to the first two concerns, Westinghouse committed in Revision 20 of SSAR Section 3.8.4.6.1.2 that headed shear reinforcement meeting the requirements of ASTM A970 is used where mechanical anchorage is required, such as for shear reinforcement in the NI foundation mat and in the exterior walls below grade. Also, in Revision 20 of SSAR Section 3.8.5.5, Westinghouse documented its commitments. The staff's review of Revision 20 of the SSAR found that the structural criteria committed by Westinghouse for the design of reinforced concrete foundation mat meet the ACI code requirements. On the basis discussed above, the staff concluded that the design of the shear reinforcement for the foundation mat is technically resolved, and Open Item 3.8.5-9 is closed.

<u>Open Item 3.8.5-10</u>

On the basis of the staff's past licensing review experience, the unevenly distributed construction loads on the foundation mat, especially for the foundation mat with large dimensions and irregular shape, can be very significant and may cause severe foundation cracks. Westinghouse should provide the basis for demonstrating the design adequacy in coping with the unevenly distributed construction loads.

In the meeting on August 4 through 8, 1997, Westinghouse presented its design approach and results for considering the effects of construction settlements in the design of the foundation mat. During this meeting, the effects of construction settlements on developed moments and shears in the NI structures were reviewed and discussed. As a result of the discussion regarding the staff's review findings, the following five design procedures for considering loads because of construction sequence and settlements in the foundation mat design are acceptable to the staff:

(1) Obtain stress resultants at critical locations on the NI from the following load cases:

   (a) Case 1 of the PCRA analysis for the deep clay site, which maximizes settlement effects during the later stages of construction and during plant operation,

   (b) Case 2 of the PCRA analysis for the sand/clay site, which maximizes the potential effects of site dewatering and loads applied during the early stages of construction, and

   (c) the analyses on the basis of the simplified INITEC's Winkler spring models.

(2) Obtain the maximum values of stress resultants at each of the critical location from the cases in Item 1 above. This list of maximum stress resultant values is then to be considered as the resultant dead load stress resultant solution.

(3) The stress resultants as a result of dead load from Item 2 above are then to be combined with all other stress resultants to obtain stress resultants in satisfying the various load combination requirements. For each load case, the list of maximum stress resultant values represents the elastic demand on the NI.

(4) The elastic demand is then to be compared with the section capacities of the concrete structural elements of the NI to judge the adequacy of the design.

(5) Because soil bearing pressures calculated from the PCRA and INITEC analyses are not sensitive to the size of finite elements used in the model, they can be used in the localized analysis with more finely discretized finite elements to calculate final design moments and shears.

Westinghouse should follow this procedure to design the NI structures for resisting the loads induced by construction sequence and settlements. In addition, Westinghouse should document this procedure in the SSAR. On the basis discussed above, Open Item 3.8.5-10 remained unresolved.

To resolve the issue regarding the inclusion of construction induced loads in the AP600 foundation mat design, Westinghouse submitted Revision 2 of its response to Open Item 3.8.5-10 (the letter dated February 9, 1998, NSD-NRC-5562) and Revision 22 of SSAR Section 3.8.5.4.3 for the staff review. In Revision 22 of the SSAR, Westinghouse incorporated the procedure described above. In order to demonstrate the adequacy of the existing design of the NI foundation mat, Westinghouse selected five critical locations and performed a confirmatory evaluation of the foundation mat, by including the construction induced loadings due to potential short term and long term settlement effects as a portion of dead load. (These five locations were selected on the basis of maximum bending and shear effects identified from the stress analyses under the combination of all applicable loads except loads induced by foundation settlement because of construction.) The results indicate that the member forces are within design allowable. These five critical locations are:

- north edge of shield building,
- south edge of shield building,
- north-east edge of shield building,
- south-west edge of shield building, and
- middle of north auxiliary building below Wall "K".

The member forces used for the confirmatory evaluation are developed from dead load conditions, including the effects of stresses developed from potential construction settlement effects, and seismic load conditions developed from the effects of the SSE. Also, in Revision 22 of the SSAR, Westinghouse added a paragraph to Section 3.8.5.4.3 and stated that if it is necessary to perform reanalysis or redesign of the basemat, for the evaluation of a nonuniform site in accordance with Subsection 2.5.4.5.3.1, the member forces at the end of construction will be calculated considering the effects of settlement during construction. These member forces will be included as dead loads in each of the post-construction load combination.

The staff review found that the member forces as computed are within the allowable capacity of the foundation mat. On the basis discussed above, the staff concluded that Westinghouse has properly resolved the staff's concern regarding loads induced by foundation settlements because of construction, including the long term settlement effects. The staff also concluded that, from the confirmatory evaluation performed by Westinghouse, the existing foundation mat design has an acceptable design margin to accommodate loads because of construction induced settlements for the site conditions within the scope of the Standard Design. In addition, Westinghouse's SSAR commitment for the reevaluation or redesign for the site condition outside the design certification scope is consistent with the common industry practice and, therefore, is acceptable. On the basis discussed above, the staff concludes that the issue regarding the design of the NI basemat for loads because of construction settlements is technically resolved and Open Item 3.8.5-10 is closed.

Open Item 3.8.5-11

Westinghouse should justify the basis for using a uniform Winkler spring in the foundation analyses instead of the expected variable stiffness from the edge to center of the foundation mat. In addition, the staff raised a concern regarding the basis of using only one soil condition (soft rock case) for the design of the foundation mat. In addressing this open item, Westinghouse performed additional analyses for evaluating the effects of (1) local soft spots of soil foundation, (2) soil springs to the foundation mat design with non-uniform stiffness, and (3) the soil stiffness corresponding to other soil conditions used in the design.

In the August 4 through 8, 1997, meeting and in Revision 15 of SSAR Section 2.5.4.5, Westinghouse provided new criteria for the spacing and depth of borings, and other geophysical site testing to ensure that the variability of in-site shear wave velocities over the footprint of the NI is within 10 percent of the best-estimate values. The staff's review of the new criteria for the site investigation is discussed in Section 2.5 of this report. On this basis, Open Item 3.8.5-11 is closed.

Open Item 3.8.5-12

In the design of NI foundation mat, Westinghouse should consider the seismic shear and moments because of the out-of-phase vibration between the shield building, containment shell, and containment internal structures.

During the August 4 through 8, 1997 meeting, the staff reviewed the calculation by Westinghouse and found that the seismic shear and moments calculated on the basis of the assumption of in-phase vibration between the shield building, containment shell, and containment internal structures are more conservative in comparison with those on the basis of the assumption of out-of-phase vibration of these buildings. On this basis, Open Item 3.8.5-12 is closed.

Open Item 3.8.5-13

In the evaluation of the dynamic stability of the NI structures against overturning and sliding, Westinghouse should provide the formulas for calculating the energy component because of buoyancy "W_b" in Section 3.8.5.5.4 of the SSAR.

Because Westinghouse, in SSAR Section 3.8.5.5.4, committed to use the moment balance method instead of the energy balance method for evaluating the dynamic stability, there is no need to calculate the energy component because of buoyancy "W_b". The staff's evaluation of the dynamic stability of the NI structures is discussed in the evaluation of Open Item 3.8.5-4 above. On this basis, Open Item 3.8.5-13 is closed.

Open Item 3.8.5-14

In order to prevent the potential for rebar corrosion, Westinghouse should commit in the SSAR to using coated reinforcing bars for the design of the NI foundation.

For the protection from external flooding of the NI and to prevent rebar corrosion, in SSAR Sections 2.5.4 (Revision 15) and 3.4.1.1 (Revision 6), Westinghouse committed to include waterstops and a crystalline additive to the waterproofing system in the vertical soil retention systems and base mudmat as well as steel meshes in the mudmat to limit potential cracking during construction and long-term settlements. Also, as shown in Revision 17 of SSAR Section 2.5.4.6.3, Westinghouse added a COL commitment to provide information on the waterproofing system along the vertical face and the mudmat for the staff review. On this basis, Open Item 3.8.5-14 is closed.

Open Item 3.8.5-15

The following two inconsistencies in SSAR figures were identified:

(1) Figure 1.2-23 (Section M-M) of the SSAR shows the distance between Column Lines H.1 and I is 0.8 m (2.5 ft), and Figures 1.2-13 (Section B-B) and 3.8.5-1 (Section B) of the SSAR show the distance between these two column lines is 0.6 m (2.0 ft).

(2) Figure 1.2-34 (Section B-B) of the SSAR shows the distance between Column Lines "11" and "11.1" is 1.2 m (4.0 ft), and Figure 3.8.5-1 (Section C) of the SSAR shows the distance between these two column lines is 0.6 m (2.0 ft).

Westinghouse should correct these inconsistences in the figures in the SSAR.

In Revision 3 of SSAR Figures 1.2-13, 1.2-23, 1.2-34 and 3.8.5-1, Westinghouse made correction on those inconsistencies identified in the DSER. Therefore, Open Item 3.8.5-15 is closed.

Open Item 3.8.5-16

SSAR Figure 1.2-5 (Section A-A) shows that Column Line "11" is located inside of the NI peripheral wall along this column line. However, SSAR Figure 1.2-34 (Section B-B) shows that the foundation overhang of the turbine building extends beyond this column line and sticks into the NI wall. Westinghouse should correct the errors in these SSAR figures.

In Revision 3 of SSAR Figures 1.2-5 and 1.2-34, Westinghouse made corrections on those errors. Open Item 3.8.5-16 is closed.

<u>Open Item 3.8.5-17</u>

Westinghouse did not include the construction loads and the sequence of these loads in the design of the NI foundation mat.

Same issue is addressed under Open Item 3.8.5-10. On the basis of the resolution of Open Item 3.8.5-10, Open Item 3.8.5-17 is closed.

<u>Open Item 3.8.5-18</u>

As indicated in Figures 1.2-12 through 1.2-17 of the SSAR, Westinghouse did not provide overhangs at the end of the NI foundation mat for having enough rebar development length or use special end plates for rebar anchorage to resist the bending moments because of the soil pressure (static and dynamic) against peripheral walls.

During the December 9 through 13, 1996, meeting, the staff reviewed the design calculations for the foundation mat and found that adequate rebar anchorage with longitudinal mat reinforcements was provided in accordance with ACI-349 Code. On this basis, the staff concludes that overhangs at the end of the NI foundation mat are not needed and Open Item 3.8.5-18 is closed.

<u>Open Item 3.8.5-19</u>

For evaluating foundation uplift potential, the hard rock site condition should be considered for determining foundation mat design forces. The effect of impact between the foundation mat and the rock, and the load concentration at edges and corners, should also be considered in the design.

During the December 9 through 13, 1996, meeting, the staff reviewed design calculations for the foundation mat and found that Westinghouse did considered pressures for various soil and rock, including hard rock, cases in the design loads. On this basis, Open Item 3.8.5-19 is closed.

<u>Open Item 3.8.5-20</u>

The staff review of design calculations found that the shear modulus of the subgrade soil used for the foundation design was on the basis of a foreign test, and soil stress attenuation with depth seems counter-intuitive. The references used for the foundation design should be validated by an independent U.S. reference.

During the December 9 through 13, 1996 meeting, the staff's review of design calculations found that Westinghouse did use independent U.S. references to validate the subgrade modulus used by INITEC in the design of the foundation mat. On this basis, Open Item 3.8.5-20 is closed.

Open Item 3.8.5-21

During the week of July 11 through 15, 1994, the staff conducted a design calculation review at the Bechtel Power Corporation (Westinghouse's consultant) office in San Francisco, California. A number of issues were raised as described in Section 5.1 of this FSER. On the basis of the discussions held with Westinghouse at that meeting, the staff concluded that the foundation mat design performed by INITEC was not acceptable. Westinghouse was requested to verify the adequacy of the original foundation design and make corrections, if necessary. In its letter dated August 2, 1994, Westinghouse committed to the following actions:

(1) Perform an independent review of the existing design calculations.

(2) Verify the adequacy of the INITEC's in-house post-process computer programs used for the foundation mat design.

(3) Perform simplified analyses to confirm the adequacy of the existing design results.

(4) Provide the independent review results for the staff review.

In response to this open item, Westinghouse provided a design summary report and calculations for review. The staff review found that Westinghouse (1) did perform an independent review of the existing design calculations, (2) did not use INITEC's in-house postprocess computer programs for the foundation mat design, and (3) performed simplified analyses to confirm the adequacy of the existing design results. As for the adequacy of the original foundation mat design, the same issue is addressed under Open Item 3.8.5-9. On the basis discussed above, Open Item 3.8.5-21 is closed.

In addition to the open issues stated above, issues were raised by the staff during its review of the foundation mat design calculations in the review meetings. The staff's concern with these issues and their resolution are discussed below.

Meeting Open Item 1

In developing bounding pressure distributions for use in the foundation mat design, the soil stiffness parameters used in the analysis should be varied over a range from soft soil to hard rock in determining pressure distribution underneath the foundation mat. In addition, the variation of soil stiffness along the foundation mat length should also be considered in the development of bounding soil pressures.

In the December 9 through 13, 1996, meeting, the staff reviewed design calculations and found that Westinghouse considered variation of subgrade modulus in calculating the bounding soil pressures. This resolves the staff's concern.

Meeting Open Item 2

Since the foundation mat is only 1.83 (6 ft) thick in the auxiliary building area, the effect of large cut-outs of pits to the overall design of foundation mat could be significant.

Westinghouse, in the letter dated April 15, 1997, and in the August 11 through 15, 1997, meeting provided its response and design calculations for review. As a result, the staff found that the design of the 46 cm (18 in) thick part of the foundation mat in the 1.83 m (6 ft) by 2.7 m (9 ft) elevator pit meets the ACI-349 Code requirements and, therefore, is acceptable.

Meeting Open Item 3

Settlements induced by the construction procedure and construction loads may lead to significant locked-in stresses. These settlement-induced stresses (both immediate and long-term) and construction loads should be included in the design of the mat foundation.

This same issue is addressed under Open Item 3.8.5-10 above. On this basis, the concern is resolved.

Meeting Open Item 4

Since normal site investigations may overlook the local soft and/or hard spots existing in the supporting soil foundation, the effect of the possible soft/hard spots on the local soil pressure computation should be evaluated and included in the design.

In the August 4 through 8, 1997, meeting, the staff reviewed the improved site investigation procedures which are to be used to ensure that local soft/hard spots in the foundation soils are properly evaluated and considered and found they are consistent with the industry practice. On the basis discussed above, this issue is resolved.

Meeting Open Item 5

In order to resist high-shear stresses, Westinghouse applied heavy shear reinforcement in the area of the auxiliary building (especially the mat foundation at the junction of the shield and auxiliary buildings). With relative small thickness of foundation mat (the mat thickness at junction between the shielding and auxiliary buildings is 1.83 m (6 ft)), the congestion of reinforcement at these locations may cause reduction of the shear resistance of the foundation mat.

This concern is similar to the issue addressed under Open Item 3.8.5-9. On this basis, this issue is resolved.

3.8.5.8 Conclusions

On the basis discussed above, the staff concludes that the design of the SC-I foundations is acceptable and meets the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, and 4. This conclusion is on the basis of the following factors:

- Westinghouse has met the requirements of Section 50.55a and GDC 1 with respect to assuring that the NI foundation mat is designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of the RGs and industry standards indicated below.

- Westinghouse has met the requirements of GDC 2 by designing the NI foundation mat to withstand the 0.3g SSE with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

- Westinghouse has met the requirements of GDC 4 by assuring that the design of the NI foundation mat is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

The criteria used in the analysis and design, and those proposed for construction of the NI foundation mat to account for anticipated loadings and postulated conditions that may be imposed upon the foundation mat during its service lifetime, are in conformance with the established criteria, codes, standards, and specifications acceptable to the staff. These include meeting the guidelines of RG 1.142 and industry standards ACI 318-95 and ACI-349-90. The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service inspection requirements, provide reasonable assurance that, in the event of winds, tornados, earthquakes, and various postulated events, the NI foundation mat will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions.

In addition, the staff's conclusion regarding the design of the nuclear island foundation mat is based on its review of the samples of design calculations for the critical sections of the foundation mat described in Revision 22 of SSAR Section 3.8.5.4.5, "Design Summary of Critical Sections." Therefore, any proposed change to the text of Revision 22 of SSAR Section 3.8.5.4.5 will require NRC review and approval before implementation of the change.

3.9 Mechanical Systems and Components

Sections 3.9.1 through 3.9.6 of the SRP address the review of the structural integrity and functional capability of various safety-related mechanical components. The review is not limited to ASME Code components and supports, but is extended to other components such as those portions of the control rod drive mechanisms which are considered part of the reactor coolant pressure boundary, certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code. The staff reviewed such issues as load combinations, allowable stresses, methods of analysis, summary of results, pre-operational testing, and inservice testing. The staff's evaluation focused on determining whether there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

In accordance with the guidelines in Section 3.9.1 of the SRP, the staff reviewed the information in Section 3.9.1 of the SSAR related to the design transients used in the design and fatigue evaluations for ASME Class 1 and core support (CS) components, and methods of analysis used for all seismic Category I components, component supports, CS structures, and reactor internals designated as Class 1, 2, 3 and CS under Section III of the ASME Code, and those not covered by the Code. The staff also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports, as well as experimental and inelastic analytical techniques.

3.9.1.1 Design Transients

In Table 3.9-1 of the SSAR, Westinghouse lists the fluid system design transients for five operating conditions and the number of cycles for each transient that are considered in the design and fatigue analyses of RCS ASME Class 1 components, other Class 1 components, RCS supports, and reactor internals. The operating conditions are as follows:

- ASME Service Level A - normal conditions
- ASME Service Level B - upset conditions, incidents of moderate frequency
- ASME Service Level C - emergency conditions, infrequent incidents
- ASME Service Level D - faulted conditions, low-probability postulated events
- testing conditions

The basis for the number of cycles for the transients in Table 3.9.1 of the SSAR is discussed in Section 3.9.1.1 of the SSAR. The number of cycles are a conservative estimate of the magnitude and frequency of the temperature and pressure transients that may occur during plant operation based, in part, on operating experience of current PWRs, and adjusted for a 60-year AP600 plant life. The effects of seismic events are not included in this table because the table only addresses fluid system transients. However, in Section 3.9.1.1 of the SSAR, Westinghouse states that in addition to the cycles because of fluid system transients, the effect of earthquake cycles are considered in the fatigue analyses mentioned above. The discussion of seismic loading conditions that are included in these analyses is in Section 3.9.3 of the SSAR, and the staff's evaluation of these conditions is in Sections 3.9.3 and 3.12 of this report. On the basis of the above discussion, and the evaluations in Sections 3.9.3 and 3.12 of this report, the staff concluded that the use of PWR operating experience, adjusted for a 60-year plant life, plus additional cycles to account for seismic events, provides an acceptable basis for estimating the total number of cycles for each transient. Therefore, the information relative to the AP600 design transients in Section 3.9.1.1 of the SSAR is consistent with the applicable guidelines in Section 3.9.1 of the SRP and is acceptable.

3.9.1.2 Computer Programs

Westinghouse used computer codes to analyze mechanical components. Design control measures to verify the adequacy of the design of safety-related components are required by Appendix B to 10 CFR Part 50. In Section 3.9.1 of the SRP, the staff provides guidelines sufficient to meet Appendix B. In the response to RAI 210.33 dated June 30, 1994, Westinghouse agreed to revise the SSAR to provide a new Table 3.9-15, which listed and stated the application of 22 computer programs used in the hydraulic transient load analyses, and in dynamic and static analyses of mechanical loads, stresses, and deformations of Seismic Category I components and supports. In addition, Westinghouse agreed to revise Section 3.9.1.2 of the SSAR to include a description of the method used to verify these programs. This was identified as DSER Confirmatory Item 3.9.1.2-1. In Revision 4 to the SSAR, Westinghouse revised SSAR Table 3.9-15 and Section 3.9.1.2 to identify the 22 computer programs and the various methods of program verification. This information conforms to the guidelines in Section 3.9.1.II.2 of the SRP and is acceptable. Therefore, DSER Confirmatory Item 3.9.1.2-1 is closed.

The staff's review of this issue was, in part, based on an evaluation of the adequacy of the Westinghouse computer program used in the representative AP600 piping analyses that are currently being audited by the staff. This was accomplished by the staff performing an independent piping analysis to confirm the adequacy of these Westinghouse analyses. The resolution of this issue is discussed in Section 3.12.4.1 of this report.

3.9.1.3 Experimental Stress Analysis

In Section 3.9.1.3 of the SSAR, Westinghouse states that the only experimental stress analysis used for the AP600 is performed in conjunction with the pre-operational flow-induced vibration testing of reactor internals. The staff's evaluation of this issue is discussed in Section 3.9.2.4 of this report.

3.9.1.4 Inelastic Analyses

The staff's evaluation of the inelastic analysis methodology is discussed in Section 3.12.3.5 of this report.

3.9.1.5 Conclusions

On the basis of the evaluations in Sections 3.9.1.1 through 3.9.1.4, and 3.12.3.5 and 3.12.4.1 of this report, the staff concludes that the design transients, computer program validation, and experimental stress analysis and inelastic analysis methodology for Seismic Category I components and supports meet the applicable portions of GDC 1 and 2, Appendix B to 10 CFR Part 50, Appendix A to 10 CFR Part 100, and the guidelines in Section 3.9.1 of the SRP, and are acceptable.

Westinghouse met GDC 2 and Appendix A to 10 CFR Part 100 by including seismic events in design transients that serve as the design basis for withstanding the effects of natural phenomena.

Westinghouse met Appendix B to 10 CFR Part 50, and GDC 1 by submitting information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I structures designated as ASME Code, Class 1, 2, 3, and CS and those not covered by the Code within the present state-of-the-art limits and by having design control measures that are consistent with the applicable guidelines of Section 3.9.1 of the SRP. This is acceptable for ensuring the quality of the computer programs. If the COL applicant opts to use computer programs different from those used by Westinghouse for the design of any safety-related item with the exception of piping systems, the guidelines of Section 3.9.1 of the SRP must be met for such programs. The staff's review of the piping systems is included in Section 3.12 of this report.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The staff reviewed the methodology, testing procedures, and dynamic analyses that Westinghouse used to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loadings. The staff's review acceptance criteria included the following requirements:

- GDC 14 and 15 by conducting the piping vibration, thermal expansion, and dynamic effects testing

- GDC 2 by reviewing the seismic subsystem analysis methods

- GDC 1 and 4 by committing to testing the dynamic responses of structural components in the reactor caused by steady-state and operational flow transient conditions

- GDC 1 and 4 by committing to the flow-induced vibration testing of reactor internals to be conducted during the pre-operational and startup test program

- GDC 2 and 4 by committing to the dynamic analysis methods to confirm the structural design adequacy and functional capability of the reactor internals and piping attached to the reactor vessel when subjected to loads from a LOCA in combination with an SSE

3.9.2.1 Piping Pre-operational Vibration and Dynamic Effects Testing

Piping vibration, thermal expansion, and dynamic effects testing should be conducted on all AP600 plants during the pre-operational testing program. The purpose of these tests is to confirm that the applicable piping systems, restraints, components, and supports have been adequately designed, fabricated, and installed to withstand flow-induced dynamic loadings under the steady-state and operational transient conditions, and to confirm that the piping system can expand thermally in a manner consistent with the design intent. In Revision 0 to the SSAR, Westinghouse stated that these tests will be conducted only on the first AP600 plant. In the response to RAI 210.53 dated June 27, 1994, Westinghouse agreed to revise Sections 3.9.2.1, 14.2.8.1.78, 14.2.8.1.82, and 14.2.8.2.20 of the SSAR to delete the statements relative to testing on only the first plant and provided a commitment that these tests will be conducted on all AP600 plants. In the DSER, the staff concluded that this response is consistent with applicable guidelines in Section 3.9.2 of the SRP, and was, therefore, acceptable pending receipt of the SSAR revision. This was identified as DSER Confirmatory Item 3.9.2.1-1. In Revision 4 to the SSAR, Westinghouse revised Section 3.9.2.1 to delete the statement relative to the first AP600 plant. In Revision 9, Westinghouse replaced the original Sections 14.2.8.1.78, 14.2.8.1.82, and 14.2.8.2.20 with Sections 14.2.9.1.7 and 14.2.10.4.18, and revised these new sections to delete the same statement. Therefore, DSER Confirmatory Item 3.9.2.1-1 is closed.

In Section 3.9.2 of the SRP, the staff states that the following systems should be monitored during these tests:

- ASME Code, Class 1, 2, and 3 piping systems
- 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

In Revision 0 to Section 3.9.2.1 of the SSAR, Westinghouse only stated that these tests will be conducted on ASME Class 1, 2, and 3 and other high energy piping systems, and in the original Sections 14.2.8.1.77, 14.2.8.1.78, 14.2.8.1.82, 14.2.8.2.18, and 14.2.8.2.20 of the SSAR, Westinghouse did not identify the systems to be tested. The staff's position was that all six of the above sections of the SSAR should be revised, if applicable, to state that all of the piping systems listed above will be included in the AP600 pre-operational piping vibration, thermal expansion, and dynamic test programs described in the new SSAR Section 14.2.9.1.7. This was identified as DSER Open Item 3.9.2.1-1. In Revision 4 to the SSAR, Westinghouse revised Section 3.9.2.1 to add a commitment to include all of the piping systems listed above in the AP600 pre-operational vibration and dynamics effects testing programs. In a letter dated October 23, 1996, Westinghouse submitted a further response to this open item which states that the only systems that meet the criteria in the revised SSAR Section 3.9.2.1, that are not already included in Chapter 14 of the SSAR, are the control room habitability system (VES) and the hot water heating system (VYS). The VES is not subjected to vibration because of the low flow rates in this system, and it is classified as a high-energy system on the basis of pressure, not temperature. Therefore, the expansion, vibration, and dynamic effects testing described in SSAR Section 14.2.9.1.7 are not applicable to the VES. The VYS is not a safety-related system, and the high-energy portion of the VYS is not located in the vicinity of safety-related systems and components. Therefore, the types of testing described in SSAR Section 14.2.9.1.7 are also not applicable to the VYS. In Revision 10 to the SSAR, Westinghouse added a paragraph to Section 14.2.9.1.7 to provide information relative to the VES. The staff concludes that, based on this response, these systems need not be included in this portion of the initial test program. The staff further concludes that the systems identified in Section 14.2.9.1.7 as being applicable to these testing programs are consistent with the guidelines in Section 3.9.2 of the SRP, and are acceptable. Therefore, DSER Open Item 3.9.2.1-1 is closed.

In addition, the response to RAI 210.56 dated June 27, 1994, Westinghouse agreed to revise Sections 3.9.2.1 and 3.9.2.1.1 of the SSAR to include a commitment that these test programs will include safety-related instrument sensing lines up to the first support in each of three orthogonal directions from the process pipe or equipment connection point. This was identified as DSER Confirmatory Item 3.9.2.1-2. In Revision 4 to the SSAR, Westinghouse revised Sections 3.9.2.1 and 3.9.2.1.1 to provide this commitment. This is responsive to the staff's request, and is acceptable. Therefore, DSER Confirmatory Item 3.9.2.1-2 is closed.

As mentioned above, during the plant's pre-operational and startup testing program, all AP600 plants should test various piping systems for abnormal, steady-state, or transient vibration and for restraint of thermal growth. Steady-state vibration, whether flow induced or caused by nearby vibrating machinery, could cause up to 1E10 cycles of stress in the pipe during the 60-year design life of the plant. For this reason, the staff requires that the stresses associated with steady-state vibration be minimized and limited to acceptable levels. The test program should consist of a mixture of instrumented measurements and visual observations by qualified personnel. In the June 27, 1994 response to RAI 210.54, Westinghouse agreed to revise Section 3.9.2.1.1 of the SSAR to state that piping vibration testing and assessment will be performed in accordance with ANSI/ASME OM-1987, "Operation and Maintenance of Nuclear Power Plants," Part 3. The staff's position is that this should be ANSI/ASME OM-1990, Part 3. The 1990 revision of this standard provides more recent requirements for the assessment of vibration in all safety-related piping systems during pre-operational and start-up testing. It includes steady state and transient vibration testing, acceptance criteria, and recommendations for corrective action when required. In addition, it provides guidance acceptable to the staff for the assessment of vibration levels of applicable piping systems during plant operation. The staff reviewed the 1990 version of this standard and finds it acceptable for use in the design certification for all advanced light water reactor plant designs. Therefore, the staff requested that the SSAR be revised to change the date of ANSI/ASME OM from 1987 to 1990. This was identified as DSER Open Item 3.9.2.1-2. In Revision 4 to the SSAR, Westinghouse revised Reference 2 in Section 3.9.8 to change the date of ANSI/ASME OM to 1990. The staff finds this acceptable, and therefore, DSER Open Item 3.9.2.1-2 is closed.

In the response to RAI 210.55 dated June 27, 1994, Westinghouse agreed to revise Sections 3.9.2.1.2, 14.2.8.1.67, and 14.2.8.2.18 of the SSAR to state that detailed test specifications for thermal expansion testing of piping systems during pre-operational and start-up testing are in accordance with the ANSI/ASME OM-1990 Standard, Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems." This standard contains procedures to be used for the assessment of thermal expansion response and design verification of piping systems. Implementation of this standard ensures that the piping system can expand and contract as required during all plant conditions by verifying the following requirements:

- Expected expansion can be accommodated by the piping system restraints.
- Movement is not obstructed by any unintentional restraints.
- Responses are within design tolerances.

It also provides guidance for the development of acceptance criteria, instrumentation, and measurement techniques, as well as corrective actions and methodologies for reconciling movements that differ from those specified by the acceptance criteria. The staff has endorsed this standard. In the DSER, the staff reported that pending receipt of the SSAR revision, the response to RAI 210.55 is acceptable. This was DSER Confirmatory Item 3.9.2.1-3. In Revision 4 to the SSAR, Westinghouse revised Section 3.9.2.1.2 to agree with the response to

RAI 210.55. Revision 9 to Section 14.2.9.1.7 of the SSAR also agrees with this response. Therefore, DSER Confirmatory Item 3.9.2.1-3 is closed.

In the response to RAI 210.57 dated June 27, 1994, Westinghouse agreed to revise Section 14.2.8.1.78 of the SSAR 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 response was acceptable, pending receipt of the SSAR revision. This was identified as DSER Confirmatory Item 3.9.2.1-4. In Revision 9 to the SSAR, Westinghouse replaced Section 14.2.8.1.78 with Section 14.2.9.1.7. Also, Section 14.2.9.1.7 contains the reference to Section 3.9.2.1.1. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.9.2.1-4 is closed.

3.9.2.1.1 Conclusion

On the basis of the above evaluation, the staff concludes that the AP600 piping pre-operational vibration, thermal expansion, and dynamic effects test program described in the SSAR meets the relevant requirements of GDC 14 and 15 with regard to the design and testing of the RCPB. This provides reasonable assurance that there is a low probability of rapidly propagating failure and of gross rupture to ensure that design conditions will not be exceeded during normal operation, including anticipated operational occurrences, by having an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during startup and initial operation of specified high- and moderate-energy piping, including all associated restraints and supports. The tests provide adequate assurance that the piping and piping supports are designed to withstand vibrational dynamic effects as a result of valve closures, pump trips, and other operating modes associated with the design-basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. For the planned tests, loads similar to those experienced during transient and normal reactor operations will be developed. The staff finds that these criteria will provide an acceptable level of safety for a piping system to withstand the effects of vibration and thermal expansion during the plant's 60-year design life. The above test program conforms to Section 3.9.2 of the SRP, and is acceptable.

3.9.2.2 Seismic Subsystem Analysis

In Section 3.7.3 of the SSAR, Westinghouse identifies those items that are categorized as seismic subsystems. Of those items listed in Section 3.7.3 of the SSAR, Sections 3.9 and 3.12 of this report are only responsible for the staff's evaluation of the criteria and methodology used for seismic analyses of piping systems and supports and instrumentation lines and supports. A detailed discussion of this evaluation is contained in Section 3.12 of this report. The staff's evaluations of the remainder of the seismic subsystems are contained in Sections 3.7.3 and 3.10 of this report.

On the basis of the applicable evaluations in Section 3.12 of this report, the staff concludes that the AP600 design meets the relevant guidelines of GDC 2 with respect to demonstrating design adequacy of all seismic Category I systems, components, equipment, and their supports to withstand the SSE by meeting the staff positions in RGs 1.61 and 1.92, and the applicable guidelines in Section 3.9.2 of the SRP.

3.9.2.3 Pre-Operational Flow-Induced Vibration Analysis and Testing of Reactor Internals

Reactor internals are subjected to both steady state and transient flow-induced vibratory loads for the service life of the reactor. Dynamic responses of reactor internals to these loads are related to structural type and location of reactor internal components and reactor operational flow conditions. According to RG 1.20, a vibration assessment program should be implemented to ensure structural integrity and safety functions of the internals. With respect to this assessment program, in Revision 0 of Section 3.9.2.4 of the SSAR, Westinghouse stated that the first AP600 plant is classified as a Non-Prototype Category II. According to Regulatory Position C.1.5 of RG 1.20, this means that the reactor internals configuration and operating conditions should be substantially the same as a specified "valid prototype." Regulatory Position C.3.2 of RG 1.20 states that the vibration assessment program for a Non-Prototype Category II should consist of the following:

- Specify structural differences of reactor internals from the referenced valid prototype, and perform vibration prediction analysis to account for the effects of the structural differences to flow-induced vibrations.

- Monitor vibrations of reactor internal components during pre-operational flow testing with sufficient instrumentation to confirm consistent responses with acceptable safety margins.

- *Perform post-test visual inspection of internals to ensure no indications of structural* degradation.

In Revision 0 of Section 3.9.2.3 of the SSAR, Westinghouse indicated that the AP600 reactor vessel internals are similar in size and overall configurations to the reactor internals of H. B. Robinson, the valid prototype plant for the Westinghouse three-loop design. Successive design changes that have been incorporated into the AP600 design have also been verified separately in pre-operational vibration testing with measurements in several individual plants, including inverted hat upper internals and 17x17 guide tubes at Doel 3 and Sequoyah 1; XL lower core support structure at Doel 4; and elimination of reactor vessel shielding outside the core barrel at Paluel 1. In addition, Westinghouse reported vibration testing for 17x17 fuel internals and inverted hat upper internals in WCAP-8766 and WCAP-8516-P. With regard to the vibration assessment program, in Revision 0 of Section 3.9.2.3 of the SSAR, Westinghouse indicated that by studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing functions are deduced. The effects of these forcing functions have been studied in tests performed on models and reference plants. These effects are factored into the analysis models used to evaluate flow-induced vibrations in the AP600 reactor internals.

For the Non-Prototype Category II testing on the first AP600 plant, in Section 3.9.2.4 of the SSAR, Westinghouse stated that during hot functional testing, the internals are subjected to a total operating time at greater than normal full-flow conditions of at least 240 hours with one, two, or three pumps operating. This provides a cyclic loading of greater than 1E6 cycles on the main structural elements of the internals. Instrumentation is designed and installed to measure the vibration of the internals during hot functional testing, including devices attached to the reactor vessel and internals to measure component movement and deflections. The instrumentation will be concentrated in the lower internals where design changes are located, especially at the radial reflector. Internals will also be inspected before and after the hot functional test to confirm functioning and without indications of structural degradation.

The staff reviewed Revision 0 to Sections 3.9.2.3 and 3.9.2.4 of the SSAR, and Westinghouse responses to RAI 210.16 (May 20, 1994), RAI 210.18 (May 20, 1994), RAI 210.23 (January 14, 1993), RAI 210.58 (June 27, 1994), RAI 210.100 (June 30, 1994), RAI 210.102 (June 30, 1994), and RAI 210.104 (June 30, 1994). The staff also audited Westinghouse on July 27 through 28, 1994, and on May 10, 1995, to discuss issues pertaining to reactor internals for the AP600 design. The findings of staff review and audits consist of the following:

- During the July 1994 audit, the staff found that the features of AP600 reactor internals are different from the internals of H. B. Robinson in the lower core support, vortex suppression structure, downcomer configuration, location of incore instrumentation system, fuel design, core upper support skirt length, and especially, replacing the baffles with a new design of radial reflector. In addition, the AP600 reactor has four inlets, two outlets and four canned motor coolant pumps, and the H. B. Robinson reactor has three inlets, three outlets and three shaft sealed pumps. The information is inadequate to verify that the internals of H. B. Robinson can be considered as a valid prototype of the internals of the AP600 plant. Although additional design changes were verified individually by testing and analysis conducted in separate reference reactors, because the complex nature of flow-induced vibrations, more detailed information is needed to verify the effects of simultaneous interaction of all these design changes. The radial reflector is also a new design, and its effects to flow-induced vibration have never been verified in any previous reactors. Information regarding the referenced foreign reactors has not been accessible to the staff, and therefore, has not been reviewed. Westinghouse should either re-classify the first AP600 as the prototype plant, or provide additional information relative to the above concerns for the staff's review. This was identified as DSER Open Item 3.9.2.3-1.

  During the May 1995 design review, Westinghouse indicated that their major concern was whether a scaled-model test of reactor internals is required if the internals are designated as a prototype. The staff indicated that the vibration assessment program presented in the SSAR for the first AP600 reactor internals, which includes vibration prediction analysis, monitored pre-operational testing, and inspection before and after the testing, fulfills all essential criteria for qualifying the first AP600 reactor internals as a prototype as defined in RG 1.20. Because information related to flow-induced vibration of referenced reactors and of component tests is used for vibration prediction purpose, there is no need for an additional scaled-model test. Westinghouse agreed that the reactor internals of the first AP600 plant will be re-classified as a prototype for meeting guidelines of RG 1.20. The change was subsequently reflected in Revision 4 to Section 3.9.2.4 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 3.9.2.3-1 is closed.

- During the July 1994 audit, the staff found that the documentation regarding vibration prediction analysis for the AP600 internals was still under preparation. This document should provide the Westinghouse assessment that the internals are adequately designed to withstand flow-induced vibrations. In the DSER, the staff reported that it will review the document when it becomes available. Anticipated responses of reactor vessel and internals for intended measurements at each transducer location in the pre-operational vibration testing should be analytically estimated. The acceptance criteria should explicitly define an allowable response level at each transducer location with permissible deviation. The basis for the acceptance criteria should be provided. In addition, the document should also be referenced in the SSAR. This was identified as DSER Open Item 3.9.2.3-2.

During the audit in May 10, 1995, Westinghouse provided a draft report entitled, "AP600 Reactor Internals Flow-induced Vibration Assessment Program." In the report, Westinghouse discussed specific design features and their effects on flow-induced vibrations, provided a vibration analysis and evaluation of vibration levels, and described the vibration measurement program and inspections to be conducted before and after the hot functional test. Westinghouse indicated that the draft report was essentially complete except for the portion related to the evaluation of guide tubes. For verifying effects of rotational speed of the reactor coolant pumps in the AP600 to magnitude and frequencies of flow-induced vibrations, Westinghouse was conducting a pump test to confirm adequate separation of the pump blade passing frequencies from the natural frequencies of the guide tubes, and to confirm pump pulsation levels. However, Westinghouse indicated that analyses of guide tubes for other plants with similar frequency separation have shown resulting stresses well within allowable values, and the effective pump-induced vibration loads will be similar to those in the Sizewell B plant, which has no indication that the structural integrity of the guide tubes has been impaired.

Westinghouse indicated that exact transducer locations in as-built reactor internals are likely to be varied, and vibration prediction at transducer locations (tentatively assumed today) may be revised in the future. Westinghouse further proposed that a prediction analysis to verify the overall level of internals vibrations within acceptable limits might be more appropriate. The staff agreed with the approach, but a COL action is needed to ensure that the analytically predicted vibration level at exact transducer locations are submitted to the staff 60 days before the hot functional test as stated in RG 1.20. In addition to the description of the prediction analysis included in the report, a summary table of analysis results should also be provided. In Revision 4 to Section 3.9.8.1 of the SSAR, Westinghouse indicated that the COL applicant will submit a predicted vibration response and an allowable response at sensor locations before the pre-operational vibration testing of the first AP600 plant within a time frame consistent with the guidance of RG 1.20. This is acceptable.

The staff received and reviewed Revision 1 (May 1, 1995) to the draft report MI01-GER-001, "AP-600 Reactor Internals Flow-induced-vibration Assessment Program" during and after the May 10, 1995, audit. The staff's evaluation of the revised draft report concludes that Westinghouse should finalize the report by incorporating the following three items:

(1) add a summary table of vibration prediction analysis results as included in Westinghouse letter dated June 1, 1995

(2) revise the "Introduction" section and other parts of the report for consistency with the SSAR revision, such as including statements of designating the reactor internals of the first AP600 plant as a prototype

(3) show additional sensors at the guide tubes in Table 8.1 for monitoring their vibrations, which will be consistent with the revised Table 3.9-4 of the SSAR

The final report should be submitted to the NRC and should be included in the list of references in a future revision to Section 3.9.9 of the SSAR. Westinghouse agreed to implement the above staff requests. Subsequently, the report was finalized and re-designated as WCAP-14761, "AP600 Reactor Internals Flow-Induced Vibration Assessment Program." The staff review of the final report found that it is in conformance with RG-1.20 and acceptable. On December 20, 1996, Revision 10 to the SSAR was submitted. WCAP-14761 was included in the list of references in Section 3.9.9 of the SSAR. This is acceptable. However, the staff found that the wordings in Section 3.9.2.3 regarding the prototype reactor internals of the first AP600 plant, as defined in Section 3.9.2 of the SRP and RG 1.20 for vibration assessment of AP600 reactor internals, was confused with the vibration assessment from reference plants, which include information from H. B. Robinson, Doel 3 and 4, etc. used in vibration prediction analysis for the prototype. In Revision 11 to the SSAR, Westinghouse revised Section 3.9.2.3 to avoid the confusion between the "prototype" and the "reference plants". The staff finds this acceptable, and therefore, DSER Open Item 3.9.2.3-2 is closed.

- In the response to RAI 210.102 dated June 30, 1994, Westinghouse proposed a revision of Section 3.9.2.4 of the SSAR to clarify the vibration measurement program for internals of the first AP600 plant, including a table to list types and locations of all transducers. A total of 36 accelerometers and strain gages will be installed at locations including the radial reflector; core barrel flange; core barrel mid-elevation; upper support skirt; lower support plate weld; vortex suppression plate support columns; reactor vessel head studs; and upper support column extension. The staff found that although the measurement program is comprehensive, an additional measurement of guide tube response at the location experiencing the most severe cross flow excitation (i.e., near the reactor outlet nozzle) should be included. This was identified as DSER Open Item 3.9.2.3-3.

  During the audit in May 1995, Westinghouse presented a revised response to RAI 210.102, including proposed changes to Section 3.9.2.4 and Table 3.9-4 of the SSAR. The changes consisted of adding four more axially sensitive strain gages to be mounted on two guide tubes located near the exits of the hot legs for detecting lateral deflections. In subsequent Revision 4 to the SSAR, Westinghouse revised Table 3.9-4 to include the four additional sensors on the guide tubes. The staff finds this acceptable, and therefore, DSER Open Item 3.9.2.3-3 is closed.

- In Revision 0 to Section 14.2.8.1.77 of the SSAR, it appeared to the staff that reactor internals flow-induced vibration testing is required only on the first AP600 plant, with no requirement for subsequent plants. In the response to RAI 210.58 dated June 27, 1994, Westinghouse proposed a revision to Sections 3.9.2.4 and 14.2.8.1.77 of the SSAR to clarify that all AP600 plants subsequent to the first plant will be subjected to the pre-and post-hot functional test inspection program. This is in conformance with RG 1.20 and is acceptable, pending completion of the SSAR revision. This was identified as DSER Confirmatory Item 3.9.2.3-1. In Revision 4 of Section 3.9.2.4 of the SSAR, Westinghouse states that reactor internals of AP600 plants subsequent to the first plant will perform hot functional flow test and post test inspection to ensure structural integrity and operability. This is acceptable. In Revision 9 to the SSAR, Section 14.2.8.1.77 was replaced by Section 14.2.9.1.9. This same commitment was also provided in Revision 10 to Section 13.2.9.1.9. Therefore, DSER Confirmatory Item 3.9.2.3-1 is closed.

Because of the closure of DSER Open Items 3.9.2.3-1, 3.9.2.3-2, 3.9.2.3-3, and DSER Confirmatory Item 3.9.2.3-1 discussed above, the staff's conclusion on the pre-operational vibration analysis and testing program for reactor internals is as follows.

The analysis and test program, discussed in Sections 3.9.2.3 and 3.9.2.4 of the SSAR, conforms to applicable portions of Section 3.9.2 of the SRP and is acceptable.

The staff concludes that Westinghouse meets GDC 1 and 4 with regard to the reactor internals being designed and tested to quality standards commensurate with the importance of the safety functions being performed and appropriately protected against dynamic effects (1) by meeting RG 1.20 for the conduct of pre-operational vibration tests, and (2) by having a pre-operational vibration program planned for the reactor internals that provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of predictive analysis, pre-test inspections, tests, and post-test inspections provides adequate assurance that the reactor internals will, during their service life, withstand the flow-induced vibrations of the reactor without loss of structural integrity. The integrity of the reactor internals in service is essential for ensuring the proper positioning of reactor fuel assemblies and the incore instrumentation system to ensure safe operation and shut down of the reactor.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

In Revision 1 of Section 3.9.2.5 of the SSAR, Westinghouse states that reactor internals analysis for ASME Level D service condition events considers a simultaneous seismic event with the intensity of the SSE and pipe rupture transient. The combined effect is determined by considering the maximum stresses and displacements for each condition and combining them with the square-root-of-the-sum-of-squares rule. Forcing functions, analysis methodology, and modeling techniques are described. For ensuring adequate core cooling and safe core shutdown capability, deformation of the internals should be small. Consequently, the design limitations also include deflections and stability of internal components in addition to stress criteria. Maximum deflections allowed are listed in Table 3.9-14 of the SSAR. The pipe rupture conditions are on the basis of the application of mechanistic pipe break criteria, thus pipe break dynamic effects of high-energy piping with a pipe size of 10.2 cm (4 in.) or larger are excluded because of the application of LBB. In Section 3.9.5.2.1 of the SSAR, Westinghouse states that the core barrel, core support plates, support columns, and radial key supports are considered core support structures, and are certified to the requirements of Subsection NG of Section III of the ASME Code. Other internal structures are designed and fabricated using the ASME Code as a guide. For ensuring control rod insertion, in Section 3.9.2.5.3 of the SSAR, Westinghouse indicates that a pipe break size consistent with LBB application is considered, except that the guide tubes are evaluated for a break size of 929 cm^2 (144 square in.) and smaller.

The staff reviewed Sections 3.9.2.5, 3.9.3 and 3.9.5.3 of the SSAR, and the Westinghouse responses to RAI 210.19 (January 8, 1993), RAI 210.20 (January 8, 1993), RAI 210.21 (January 8, 1993), RAI 210,22 (January 8, 1993), RAI 210.70 (June 30, 1994), RAI 210.94 (June 27, 1994), RAI 210.95 (June 30, 1994), RAI 210.96 (June 27, 1994), RAI 210.97 (June 30, 1994), and RAI 210.103 (June 30, 1994). The staff also audited Westinghouse on July 27 through 28, 1994, and on May 10, 1995, to discuss and review information pertaining to the design of reactor internals for the AP600 plants. The findings of staff review and audits are as follows:

- During its review of the initial SSAR, the staff found that excluding break sizes 10.2 cm (4 in.) and larger for LOCA analysis is not yet acceptable and pending resolution of Open Item 3.6.3.5-1. In responses to RAI 210.20, RAI 210.21, RAI 210.22 and RAI 210.95, Westinghouse indicated that although the AP600 design loads for LOCA conditions are on the basis of the use of mechanistic pipe break criteria, to be consistent with past practice, enveloping LOCA loads on the basis of the dynamic effects of one-square-foot, hot-leg and cold-leg breaks are actually used in the analysis of the reactor internals. In the response to RAI 210.95, Westinghouse proposed a revision of the final paragraph of Section 3.9.2.5 of the SSAR to reflect their position. During the July 27 through 28, 1994, audit at Westinghouse, the staff confirmed the existence of such analysis documentation and concluded that, pending receipt of the SSAR revision, this criterion is adequate to account for all possible break sizes resulting from the LBB review mentioned above and is therefore, acceptable. This was identified as DSER Confirmatory Item 3.9.2.4-1. Subsequently, Westinghouse revised Section 3.9.2.5 in Revision 4 to the SSAR. The revision is as Westinghouse proposed and, as stated above, is acceptable. Therefore, DSER Confirmatory Item 3.9.2.4-1 is closed.

- During its initial review of the SSAR, the staff found that stress limits for core supports were not adequately addressed in Revision 0 to Section 3.9.3 of the SSAR. In the response to RAI 210.19, and subsequently in Revision 1 to the SSAR, Westinghouse revised Section 3.9.3.1.3 and Table 3.9-9 of the SSAR to include ASME Code Class CS stress criteria for the core support structures. These criteria are consistent with applicable portions of Sections 3.9.3 and 3.9.5 of the SRP, and are acceptable.

- In its review of the initial SSAR, the staff requested clarification of stability analyses, the locations using Appendix F of Section III of the ASME Code, and the bases for deflection criteria. In the response to RAI 210.21, Westinghouse indicated that upper support columns are evaluated for buckling and the core barrel is analyzed for shell buckling. The response to RAI 210.96 indicates that, to assure their safety function, components of reactor internals are analyzed to meet the allowable stresses in Appendix F of Section III of the ASME Code, while also meeting the deflection limits in Table 3.9-14 of the SSAR. In the response to RAI 210.97, Westinghouse proposed to revise Section 3.9.5.3 of the SSAR to explain the bases for deflection limits in Table 3.9-14 of the SSAR. The response stated that these limits provide assurance that the control rod insertion function will not be impaired and that adequate flow passage for core cooling will be maintained during and after the event of combined occurrences of LOCA and SSE. These criteria are consistent with past practices for reactor internals designed by Westinghouse, and are acceptable pending the receipt of the SSAR revision. This was identified as DSER Confirmatory Item 3.9.2.4-2. Subsequently, in Revision 4 to the SSAR, bases of deflection limits are explained in Section 3.9.5.3.2 as proposed. Thus, DSER Confirmatory Item 3.9.2.4-2 is closed.

- In its review of Revision 1 to the SSAR, the staff found that design requirements under the combined LOCA and SSE event for reactor internal structures, other than those categorized as core support structures, were unclear. In the response to RAI 210.70 dated June 30, 1994, Westinghouse proposed a revision to Section 3.9.5.2.4 and Table 3.2-3 of the SSAR to clarify the design requirements. For internal structures other than core supports, Westinghouse is committed to design requirements of Subsection NG and Appendix F of Section III of the ASME Code. These design requirements are consistent with the guidelines in Section 3.9.5 of the SRP, and are acceptable because they assure adequate design margins for these internal structures. This was identified as DSER Confirmatory Item 3.9.2.4-3. Currently, Section 3.9.5.2.4 was revised as proposed per Revision 4 to the SSAR. The Reactor System section in Table 3.2-3 of the SSAR, through Revision 11, now contains commitments to design all reactor internal structures, other than core supports, to ASME III CS. This means that these structures will be designed to the requirements of ASME III, Subsection NG and Appendix F. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.9.2.4-3 is closed.

- The staff requested more detailed information regarding production tests of the control rod drive mechanism (CRDM) and acceptance standards for ensuring operational adequacy under LOCA and SSE events. In the response to RAI 210.94 dated June 27, 1994, and during a subsequent meeting, Westinghouse indicated that laboratory seismic testing with a combination of a fuel assembly, CRDM, and rod cluster control assembly has been performed in Japan to demonstrate the ability of rod insertion under Japanese standard earthquake levels. A copy of the reference regarding the testing was provided to the staff (Reference 14 in the response to RAI 210.94 and Reference 17 in SSAR Section 3.9.9, Revision 4). The staff's review of this reference determined that Westinghouse should verify whether the Japanese test input meets the seismic qualification level of the AP600 design. This was identified as DSER Open Item 3.9.2.4-1. In addition, the staff was told that other tests of CRDMs to ensure functioning under LOCA loads were performed and documented in WCAP-8446, "17 X 17 Drive Line Components Tests - Phase 1B 11, 111 D-Loop Drop and Deflection." This report has been reviewed and accepted by the staff as a Topical Report, and since 1976, has been referenced in most PWR license applications whose CRDMs were designed by Westinghouse. In the response to RAI 210.94, Westinghouse also proposed a revision of Section 3.9.4.4 of the SSAR to provide more descriptions of the CRDM tests. This issue was resolved pending receipt of the SSAR revision. This was identified as DSER Confirmatory Item 3.9.2.4-4.

  However, during the May 1995 audit, Westinghouse could not establish the basis of using the foreign test results for seismic qualification of the AP600 CRDM. Thus, the referenced foreign test (Reference 17 in SSAR Section 3.9.9) is not suitable to be used for the CRDM seismic qualification for the AP600 plant. Westinghouse indicated that it will delete the reference to the test in SSAR Section 3.9.4.4 in a future revision of the SSAR. Westinghouse also indicated that demonstration of CRDM operability during a seismic event is impractical and insertion of control rods is not required as long as operability of the CRDM is ensured immediately following the earthquake. The staff's subsequent evaluation concurs that demonstration of CRDM operability during a seismic event is not a regulatory requirement as long as its operability can be verified after the seismic event. However, Westinghouse should demonstrate adequacy of seismic qualification to ensure post-SSE operability of the control rod drive system in the AP600 design. In Revision 10 to Section 3.9.4.3 of the SSAR, Westinghouse indicates that functional capability of the CRDM following a seismic event or a pipe break is assured by analysis. The stresses in the CRDM and the rod travel housing are bounded by the ASME Code limits, and their deflections are within the limits specified in the SSAR Section 3.9.7 to ensure that control rods do not bind during insertion. The staff finds this acceptable, and therefore, DSER Open Item 3.9.2.4-1 and DSER Confirmatory Item 3.9.2.4-4 are closed.

With satisfactory closure of the above open and confirmatory items, the staff concludes that the AP600 dynamic system and component analysis meets the applicable portions of GDC 2 and 4 and Section 3.9.2 of the SRP with respect to the design of systems and components important to safety to withstand the effects of earthquakes. The staff further finds that appropriate combinations of the effects of normal and postulated accident conditions with the effects of the SSE by having a dynamic system analysis performed, provides an acceptable basis for confirming the structural design adequacy of the reactor internals to withstand the combined dynamic loads of a postulated LOCA and SSE. The analysis provides adequate assurance that the combined stresses and strains in the components of the CRD system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction and that the resulting deflections or displacements at any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The staff finds the methods used for component analysis to be compatible with those used for the system analysis. The combination of component and system analyses is, therefore, acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

The staff's review under Section 3.9.3 of the SRP concerns the structural integrity and functional capability of pressure-retaining components, their supports, and core support structures that are designed in accordance with Section III of the ASME Code or earlier industrial standards. The staff reviewed loading combinations and their respective stress limits, the design and installation of pressure-relief devices, and the design and structural integrity of ASME Code, Class 1, 2, and 3 components and component supports. The acceptance criteria for the staff's review are on the basis of meeting the following requirements:

- 10 CFR 50.55a and GDC 1 as related to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed

- GDC 2 as related to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions

- GDC 4 as related to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions

- GDC 14 as related to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

- GDC 15 as related to the RCS being designed with sufficient margin to assure that the design conditions are not exceeded

3.9.3.1 Loading Combinations and Stress Limits

Westinghouse evaluated all ASME Code, Class 1, 2, and 3 components, component supports, core support components, control rod drive components, and other reactor internals using the load combinations and stress limits given in Revision 15 to Sections 3.9.3.1 and 3.9.3.2 of the SSAR. As discussed in more detail in Section 3.2.2 of this report, all safety-related items classified as AP600 Equipment Class A, B, or C are constructed to applicable rules of Section III of the ASME Code. The staff's review of Sections 3.9.3.1 and 3.9.3.2 of the SSAR resulted in the following evaluations.

<u>Loads, Loading Combinations, and Stress Limits</u>

In the DSER, the staff reported that in the response to RAI 210.79, Westinghouse proposed to add Table 3.9-16, "Loadings for ASME Class 1, 2, and 3 Piping" to the SSAR. One of the loads in this proposed table was design-basis pipe breaks (DBPB). In DSER Open Item 3.9.3.1-1, the staff requested that if DBPB includes both LOCA and non-LOCA loads, Table 3.9-16 should be revised to add this to the definition of DBPB. In addition, a loading combination which includes SSE plus DBPB should be added to the new Table 3.9-11, "Piping Functional Capability - ASME Class 1, 2, and 3," which was also proposed in the response to RAI 210.79. In Revision 4 to the SSAR, Westinghouse did not include the proposed Table 3.9-16. Instead, loadings for ASME Class 1, 2, and 3 systems and components, including piping and supports are all identified in Table 3.9-3 of the SSAR. In this table, DBPB is defined as including LOCA loads, which is acceptable. Additionally, in Revision 4, Westinghouse also revised Tables 3.9-5 and 3.9-8, which identify loading combinations for ASME Class 1, 2, and 3 components and supports, respectively, to include a Level D combination which contains SSE plus DBPB. Therefore, DSER Open Item 3.9.3.1-1 is closed. However, in the Section 3.12 of the DSER, the staff identified concerns relative to loads and loading combinations in Tables 3.9-3, 3.9-5, 3.9-6, 3.9-7, and 3.9-8 of the SSAR. The staff's evaluations of these tables are discussed in Sections 3.12.5.3, 3.12.5.14, and 3.12.6.3 of this report. Although Section 3.12 addresses piping only, the staff's evaluation of these tables in Section 3.12 is applicable to all ASME Class 1, 2, and 3 components and supports. The evaluation of SSAR Table 3.9-11, "Piping Functional Capability - ASME Class 1, 2, and 3" is discussed in Section 3.12.5.12 of this report.

Active pumps and valves are those whose operability is relied upon to perform a safety-related function during transients or events considered up to and including the Service Level D (faulted) plant condition. There are no active pumps relied upon to perform a safety-related function in the AP600 design. In RAI 210.66, the staff requested that, in addition to testing active valves to demonstrate operability when the valves are subjected to loads up to and including Level D, the calculated maximum stress in the valves under these conditions should be held to a low value (i.e., only slightly above the allowable yield strength (S_y) of the material). This will help to insure that the deformation resulting from these loads will be small enough to allow operability of the valve. In the response to RAI 210.66 dated July 22, 1994, Westinghouse agreed to revise Tables 3.9-9 and 3.9-10 of the SSAR by adding a note to each table which states that for active valves, pressure integrity verification will be on the basis of using the ASME Code allowables one level less than the service loading condition, which means that for Level D loading, Level C allowables will be used. This means that for Class 1 valves, the allowable stress will be approximately 1.2 S_y, and for Class 2 and 3 valves the allowable will be approximately 1.12 S_y. This was identified as DSER Confirmatory Item 3.9.3.1-1. In Revision 4 to the SSAR, Westinghouse revised Tables 3.9-9 and 3.9-10 to provide these notes. The staff concludes that these allowable stresses will not result in excessive deformations, and will help to assure operability of the valves and are, therefore, acceptable. Therefore, DSER Confirmatory Item 3.9.3.1-1 is closed.

On the basis of the above evaluations, and the evaluations in Section 3.12 of this report, Tables 3.9-5, through 3.9-10 in Revision 15 to the SSAR contain criteria for loads, loading combinations, and stress limits used in the design of AP600 ASME Class 1, 2, and 3 systems, components, and supports that are consistent with the guidelines in SRP 3.9.3 and are acceptable.

Environmental Effects on ASME Fatigue Design Curves

Section III of the ASME Code requires that the cumulative damage resulting from fatigue be evaluated for all ASME Code Class 1 SSCs. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients listed in Table 3.9-1 of the SSAR, plus additional cycles induced by seismic events. As Westinghouse stated in Section 1.2.1.1.2 of the SSAR, the AP600 design objective is 60-years. Recent test data to address fatigue concerns indicates that the effects of the reactor environment could significantly reduce the fatigue resistance of certain materials. A comparison of the test data with the Code requirements indicates that the margins in the ASME Code fatigue design curves may be less than originally intended. This could have a significant impact on SSCs designed for a 60-year life. In RAI 210.106, the staff requested that Westinghouse provide a proposed approach to address this concern for the AP600 design. The staff's evaluation of this issue for piping, which is discussed in Section 3.12.5.7 of this report, is also applicable to all ASME Class 1 SSCs and any Class 2 and 3 SSCs that are applicable to the discussion below.

Design of Certain ASME Class 2 and 3 Components for Fatigue

As a part of a design review meeting at Westinghouse, the staff requested that the SSAR identify all ASME Code, Class 2 and 3 SSCs that are subjected to loadings that could result in thermal or dynamic fatigue so severe that the 60-year design life cannot be assured by required Code calculations, and to describe the evaluations proposed for such items. This was identified as DSER Open Item 3.9.3.1-2. During a subsequent design review meeting at Westinghouse on July 26, 1995, the staff determined that the only Class 2 and 3 SSCs subjected to such loadings are the nozzles on the secondary side of the steam generators. In Section 5.4.2.1 of the SSAR, Westinghouse states that although the secondary side of the steam generator is classified as ASME Class 2, all pressure retaining parts of 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 (Subsection NB). Since ASME Subsection NB contains acceptable rules for evaluating fatigue in Class 1 components, DSER Open Item 3.9.3.1-2 is closed.

Thermal Cycling and Thermal Stratification in Piping Systems

The staff's evaluations of these issues are discussed in detail in Section 3.12.5.9 and Section 3.12.5.10 of this report.

Design Specifications

The ASME Code, Section III requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design data inputs. The code also requires a design report for ASME Code, Class 1, 2, and 3 piping and components. In the SSAR, Westinghouse committed to construct all safety-related components, such as vessels, pumps, valves and piping systems, to applicable requirements of Section III of the ASME Code. In RAI 210.73, the staff requested that Westinghouse provide a detailed description of the procedures used for generating design specifications which will meet the requirements of Subsection NCA of Section III of the ASME Code, and will be used for procurement of ASME Class 1, 2, and 3 components. In the response to RAI 210.73 dated June 30, 1994, Westinghouse stated that their internal procedures used for generating such design specifications are available for staff review. During the follow-up piping design review meeting conducted on July 20, 1994, Westinghouse provided some of these procedures for further review. In the DSER, the staff reported that the information provided during this meeting did not contain sufficient detail to determine how the applicable requirements of Subsection NCA of Section III of the ASME Code will be met. This was identified as DSER Open Item 3.9.3.1-3.

During a subsequent design review meeting on July 25 through 26, 1995, Westinghouse responded to this open item by providing the staff with AP600 Document Number MB01-M2-001, "Steam Generator Design Specification," dated June 17, 1994, to demonstrate the procedures that will be used in the AP600 plant design to prepare design specifications for ASME Class 1, 2, and 3 components. Although this document was still under preparation, it contained sufficient information for the staff to review, evaluate, and reach a conclusion relative to the adequacy of these procedures for design certification. For example:

- The design specification contains a commitment that a document will be furnished with the design report that assures the requirements of the design specification are not in conflict with the requirements of the SSAR.

- Tables 15 and 16 in the design specification contain loading combinations and design-basis load nomenclature used in the design of the steam generator that are consistent with the loading combinations design-basis load definitions in Section 3.9.3.3 of the SSAR. Tables 17 and 18 in the design specification contain quantified axial, vertical, and horizontal forces and moments that are imposed on all steam generator nozzles as a result of all design-basis loads except thermal loads. This implies that the design of piping to the steam generator nozzles has been completed to the extent that these loads and moments can be assigned in the design specification.

On the basis of the above evaluations plus a review of the remainder of the document, the staff has concluded that the AP600 Document Number MB01-M2-001 is an acceptable design specification as required by the ASME Code, Section III. The document contains sufficient information for the staff to further conclude that use of these same procedures to prepare design specifications for other AP600 ASME Class 1, 2, and 3 components will result in acceptable design specifications for all such components. Therefore, Open Item 3.9.3.1-3 is closed. However, since most of the Class 1, 2, and 3 component design reports are provided by vendors, and will not be available at the time of design certification, the staff will perform a more detailed audit of design specifications during the COL review. Therefore, in the DSER, the staff reported that Section 3.9.3 of the SSAR should be revised to state that the AP600 design specifications and design reports will be completed by the COL applicant and will be available for audit by the staff at the time of the COL submittal for license application. This was identified as COL Action Item 3.9.3.1-1 and DSER Open Item 3.9.3.1-4. In Revision 4 to the SSAR, Westinghouse revised Section 3.9.3 to state that design specifications and design reports will be completed by the COL applicant. Additionally, in Revision 5, Westinghouse revised SSAR Section 3.9.8.2 to require that COL applicants referencing the AP600 design have these documents available for NRC audit. The staff finds these commitments acceptable, and therefore, DSER Open Item 3.9.3.1-4 is closed.

Intersystem LOCA Design for Piping Systems

In SECY-90-016, the staff recommended that the Commission approve the staff's resolution of the intersystem loss-of-coolant accident (ISLOCA) issue for ALWR plants by requiring that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its SRM dated June 26, 1990, the Commission approved the staff's recommendation, provided that all elements of the low-pressure system are considered.

The standard design must minimize the effects of ISLOCA accidents by designing low-pressure piping systems that interface with the RCPB to withstand full RCS pressure to the extent practical. In Section 20.3 of this report, under New Generic Issue 105, the staff evaluated Westinghouse's approach, in terms of the practicality for systems, components, and equipment, for implementing the ISLOCA resolution for the AP600. In the following, the staff evaluated the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure should the low-pressure system be subjected to full RCS pressure. In establishing the minimum design pressure, the following goals were used as the basis for selection:

- The likelihood of rupture (burst) of the pressure boundary is based on the staff's goal of 10 percent for conditional containment failure probability (or conversely, a goal of 90 percent survival probability) that was established in Section III.D of SECY-90-016.
- The likelihood of intolerable leakage of flange joints or valve bonnets is reasonably low, although some leakage might occur.
- Some piping components might undergo gross yielding and permanent deformation.

Low-Pressure Piping Design

To achieve these objectives, the staff evaluated, first, on a qualitative basis, several possible ratios of the low-pressure system design pressure (P_d) to the RCS normal operating pressure (P_v) to establish the margins on burst and yield of the piping. The results of the staff's evaluation are depicted in Table 3.9-1 of this report for typical carbon steel (SA-106 Grade B) and stainless steel (SA-312 Types 304 and 316) materials and are then discussed for three ratios of the design pressure to the reactor vessel pressure (P_d/P_v). A margin of 1.0 or less represents the condition where burst or yielding is likely to occur. The higher the margin, the less likely burst or yielding is to occur. The low-pressure piping systems are assumed to be designed to the rules of the Subarticle NC/ND-3600 of Section III of the ASME Code for Class 2 and 3 piping systems.

Piping Integrity at P_d/P_v = 1/2 (ASME Code Service Level D)

When P_d/P_v is equal to one-half, the margins on burst and yield are equivalent to approximately those of the Service Level D condition of Section III of the ASME Code. For carbon steel pipe, this ratio will provide a margin of 2.0 on burst and 1.08 on yield for a pipe at 260°C (500°F). For stainless steel piping, a ratio of one-half will provide a sufficient margin on burst (1.7 and 1.65 for SA-312, Type 304 and SA-312, Type 316 materials, respectively). However, a small amount of yielding is likely to occur with a margin of 0.70 for both stainless steels at 260°C (500°F). No leakage of the pressure boundary is likely to occur at P_d/P_v equal to one-half.

As a result, a ratio of one-half will ensure the pressure integrity of the low-pressure piping system with ample margin.

Piping Integrity at P_d/P_v = 1/3

When the ratio P_d/P_v is reduced to one-third, the margins for carbon steel piping are lowered to 1.33 and 0.72 for burst and yield at 260°C (500°F), respectively. For stainless steel piping, the margins are 1.13 and 0.47 for burst and yield at 260°C (500°F), respectively. At these margins, it is expected that burst failure will not occur in either carbon steel or stainless steel piping. However, a significant amount of yielding might occur in stainless steel piping at all temperatures and in carbon steel piping at 260°C (500°F). Where the carbon steel piping is at a lower temperature, some yielding might occur, although to a lesser extent. The consequence of significant pipe yielding (without bursting) is that gross, permanent distortion might occur in the piping components, thereby resulting in some leakage through flanges or valve bonnets. However, it is not expected that such leakage would be uncontrollable or intolerable.

In summary, a ratio of one-third will ensure the pressure boundary of the low-pressure piping although a significant amount of pipe yielding and some leakage through flanges and valve bonnets is likely to occur.

<u>Piping Integrity at $P_d/P_v = 1/4$</u>

At P_d/P_v equal to one-fourth, the pressure integrity of carbon steel piping becomes questionable, and for stainless steel piping, it is likely that burst failure will occur. Prior to bursting, the piping system would undergo gross plastic deformation, experience a significant amount of leakage at flanges, valve bonnets, and pump seals, and possibly lose some pipe supports due to the radial expansion of the pipe.

Therefore, at P_d/P_v equal to one-fourth, the ability of the low-pressure piping system to withstand full RCS pressure is questionable for carbon steel piping and unlikely for stainless steel piping systems.

The staff further evaluated, on a quantitative basis, the survival probabilities of the low-pressure piping at various design pressures using the methodology described in NUREG/CR-5603, "Pressure-Dependent Fragilities for Piping Components." Calculations were performed by Idaho National Engineering Laboratory (INEL) under contract with the NRC's Office of Nuclear Regulatory Research.

The INEL calculations led to results similar to the qualitative conclusions discussed above. A temperature of 177°C (350°F) was used in the calculations of the following survival probabilities. Using a temperature of 260°C (500°F), the survival probabilities decrease about 2 to 5 percent for the different materials and design pressures.

For carbon steel piping (SA-106 Grade B material) with wall thickness equal to the minimum thickness required by the ASME Code for 40 percent of RCS normal operating pressure, that is, a pressure of 6.21 MPa (900 psig) (or approximately $P_d/P_v = 0.4$), the survival probability is 99 percent. For stainless steel piping (SA-312 Types 304 and 316 materials), the survival probability at 6.21 MPa (900 psig) (or approximately $P_d/P_v = 0.4$), was less than 85 percent.

These survival probabilities are based on the minimum wall thickness calculated using Eq. (3) in Subarticle NC/ND-3640 of Section III of the ASME Code. The wall thickness calculated does not account for manufacturing tolerances or the use of the next heavier, commercially available wall thickness, which would increase the piping wall thickness and also increase the survival probability. Increasing the wall thickness to the minimum commercially available thickness required to satisfy the ASME Code minimum required thickness results in minimum survival probabilities of greater than 99, greater than 87, and less than 85 percent for SA-106 Grade B, SA-312 Type 304, and SA-312 Type 316 material, respectively. On this basis, the staff found that, for PWRs, the approach to designing the interfacing systems and subsystems to 40 percent of the RCS normal operating pressure would not attain the 90 percent survival probability goal in the case of stainless steel systems and subsystems.

Subsequently, the staff determined that if the wall thickness of stainless steel piping systems is of standard weight in piping with a diameter of 35.6 cm (14 in.) and less and is a minimum of Schedule 40 in piping with a diameter of 40.6 cm (16 in.) and greater, the 90-percent survival probability goal will be attained. The minimum probabilities for Type 304 and Type 316 material were 92.7 and 87.2 percent, respectively. For carbon steel piping, a commitment to the 40 percent of RCS normal operating pressure alone will achieve the 90-percent goal. However, for stainless steel piping, the wall thicknesses based on this design pressure will be less than those required to attain the 90 percent survival probability goal. Accordingly, the extension of the minimum 40 percent design pressure and the minimum wall thickness of Schedule 40 piping to both carbon and stainless steel low-pressure piping systems will attain the 90 percent goal. As discussed below in "AP600 Design Criteria for ISLOCA," SSAR Section 5.4.7.2.2 states that the low pressure portion of the normal residual heat removal piping (which is constructed of stainless steel) is designed to pipe Schedule 80S, which results in a pipe wall thickness greater than that of Schedule 40, and is acceptable.

Valves in Low-Pressure Systems

For the valves in the low-pressure piping systems (excluding the pressure isolation valves which are already designed for RCS pressure), the selection of the valve class rating is a primary factor for designing against full RCS pressure. For example, ANSI B16.34 valves are shop-tested to 1.5 times their 37.8°C (100°F) rated pressure. This would mean that for a Class 900 A216 WCB (cast carbon steel) valve, the test pressure is 1.5 x 15.3 = 230 MPa (1.5 x 2220 = 3330 psig).

The Class 900 valve that is tested to a pressure of 230 MPa (3330 psig) would be expected to withstand an RCS normal operating pressure of 15.4 MPa reactor (2235 psig). However, it should not be assumed that the valve in the low-pressure system would be able to operate with this full RCS pressure across the disk.

Therefore, the staff finds that a Class 900 valve is adequate for ensuring the pressure of the low-pressure piping system under full RCS pressure (i.e., 15.4 MPa (2235 psig)), but no credit should be taken to consider these valves operable under such conditions without further justification.

Other Components in Low-Pressure Systems

For other components in the low-pressure systems, such as pumps, tanks, heat exchangers, flanges, and instrument lines, the staff finds that establishing an appropriate safety factor involves several complicating factors related to the individual component design. These factors include requirements for shop hydrotests, the method to determine the pressure class rating of the component, the specific material used for bolting, and the bolt tension applied, or whether the component is qualified by test or analysis.

The remaining components in the low-pressure systems should be designed to a design pressure of 0.4 times the normal operating RCS pressure (i.e., 6.21 MPa (900 psig)). The staff finds that the margins to burst for these remaining components are at least equivalent to that of the piping at its minimum wall thickness since these components typically have wall thicknesses greater than that of the pipe minimum wall thickness.

AP600 Design Criteria for ISLOCA

In the response to RAI 210.61 dated June 16, 1994, Westinghouse stated that the July 22, 1994, response to RAI 440.132, identifies the low-pressure portion of the normal residual heat removal system as the only system in the AP600 plant that carries reactor coolant outside containment that could fail because of overpressurization. The staff's evaluation of the response to RAI 440.132 is discussed in Section 20.3 of this report. In the response to RAI 210.61, Westinghouse provided the following design criteria for the low-pressure portion of the normal residual heat removal system:

- The pipe schedule for the normal residual heat removal system AP600 Class C piping outside containment is 80S.

- The American National Standard Class for the valves, flanges, and fittings in the AP600 Class C portions of the normal residual heat removal system outside containment has been specified to be greater than or equal to Class 900.

- The ratio of normal residual heat removal system and component design pressure to RCS normal operating pressure is 16.21 MPa (900 psig) to 15.4 MPa (2235 psig), or 40 percent.

The staff concluded that this AP600 ISLOCA design criteria are consistent with the staff's positions relative to piping, valves, and other components in low-pressure systems discussed above, and is, therefore, acceptable. However, the revisions to Sections 1.9.5.1 and 5.4.7.2.2 of the SSAR, which were proposed in the response to RAI 210.61 dated June 16, 1994, did not include the last item above. This was identified as DSER Open Item 3.9.3.1-5. In Revision 7 to the SSAR, Westinghouse added a new paragraph in Section 1.9.5.1 which references SSAR Section 5.4.7 for design features which are applicable to the ISLOCA for the normal RHR system (RNS) only. The design criteria in Section 5.4.7.2.2 of the SSAR agree with all of the criteria listed above, and are acceptable for the RNS. The implementation of this criteria is discussed in Section 20.3 of this report under Issue 105. Therefore, DSER Open Item 3.9.3.1-5 is closed.

Intersystem LOCA Conclusion

On the basis of the above evaluations, the staff finds for the AP600 low-pressure piping systems that interface with the RCS pressure boundary, that using a design pressure equal to 0.4 times the normal operating RCS pressure of 15.4 MPa (2235 psig) (i.e., 6.21 MPa (900 psig)), and using a minimum wall thickness of the low-pressure piping of schedule 80S, provides an adequate basis for assuring that these systems can withstand full reactor pressure and thus meet the Commission-approved staff recommendations in SECY-90-016 for designing against ISLOCAs. The piping design is in accordance with Subarticle NC/ND-3600 of Section III of the ASME Code. Using these design guidelines, the staff concludes the following:

- The likelihood of the low-pressure piping rupturing under full RCS pressure is low.

- The likelihood of intolerable leakage is low under ISLOCA conditions although some leakage may occur at flanges and valve bonnets.

- Some piping components might undergo gross yielding and permanent deformation under ISLOCA conditions.

On the basis of the above evaluation, the staff concludes that there is reasonable assurance that the low-pressure piping systems interfacing with the RCPB are structurally capable of withstanding the consequences of an ISLOCA.

Design Criteria for Heating, Ventilation, and Air Conditioning Ductwork

In the response to RAI 210.5 dated January 8, 1993, Westinghouse pointed out that Section 3.8.4.4.3 of the SSAR states that heating, ventilation, and air conditioning (HVAC) ductwork is designed in accordance with AISI specification rules. This specification does not contain all of the criteria that the staff needs to evaluate the design of safety-related HVAC ductwork. In the DSER, the staff requested that more detailed design criteria for such ductwork be included in the SSAR. This was identified as DSER Open Item 3.9.3.1-6. In Revision 7 to the SSAR, Westinghouse added Appendix 3A, "HVAC Ducts and Duct Supports," which is referenced in SSAR Section 3.8.4.4.3, and addresses some of the staff's concerns in this open item. The staff's evaluation of this issue is included as a part of Section 3.8.4.4.1 of this report. Therefore, DSER Open Item 3.9.3.1-6 is closed on the basis of the staff's evaluation in Section 3.8.4.4.1 of this report.

Conclusion

On the basis of the evaluations in Section 3.9.3.1 of this report, the staff concludes that Westinghouse meets 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress limits specified for ASME Code, Class 1, 2, and 3 components by ensuring that systems and components are designed to quality standards commensurate with their importance to safety and that these systems can accommodate the effects of such postulated events as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loadings as applied to ASME Code Class 1, 2, and 3 pressure retaining components in systems designed to meet seismic Category I standards provide assurance that, in the event of an earthquake affecting the site or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress limits for the materials of construction. Limiting the stresses under such loading combinations provides an acceptable basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

3.9.3.2 Design and Installation of Pressure-Relief Devices

The staff reviewed Section 3.9.3.3 of the SSAR with regard to the design, installation, and testing criteria applicable to the mounting of pressure-relief devices used for the overpressure protection of ASME Code, Class 1, 2, and 3 components. This review, conducted in accordance with Section 3.9.3 of the SRP, included evaluation of the applicable loading combinations and stress criteria. The review extended to consideration of the means provided to accommodate the rapidly applied reaction force when a safety relief valve (SRV) opens, and the resulting transient fluid-induced loads are applied to the piping downstream of an SRV in a closed discharge piping system. In the response to RAI 210.67 dated July 8, 1994, Westinghouse agreed to revise Sections 3.9.3.3 and 10.3.2.2.2 of the SSAR to state that the design of pressure relieving valves complies with the requirements of Appendix O of Section III of the ASME Code. In addition, the proposed revision describes supplemental design criteria, which are consistent with Section 3.9.3.II.2 of the SRP, and commits to delete a reference to Appendix 2 of ANSI/ASME B31.1. This was identified as DSER Confirmatory Item 3.9.3.2-1. In Revision 3 to the SSAR, Westinghouse revised Sections 3.9.3.3 and 10.3.2.2.2 to provide all of the above commitments. On the basis of the above information, the staff concludes that the criteria in the SSAR for design and installation of pressure-relief devices are consistent with applicable guidelines in Section 3.9.3 of the SRP and are acceptable. Therefore, DSER Confirmatory Item 3.9.3.2-1 is closed.

On the basis of the above evaluation, which states that the criteria in Section 3.9.3.3 of the SSAR as related to the design, installation, and testing of ASME Code, Class 1, 2, and 3 SRV mounting meet the applicable guidelines of Section 3.9.3 of the SRP, the staff concludes that Westinghouse meets 10 CFR 50.55a and GDC 1, 2, and 4 by ensuring that SRVs and their installations are designed to standards that are commensurate with their safety functions and that they will accommodate the effects of discharge caused by normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. Westinghouse also meets GDC 14 and 15 with regard to ensuring that the RCPB design limits for normal operation, including anticipated operational occurrences, will not be exceeded. The criteria used by Westinghouse in the design and installation of ASME Code, Class 1, 2, and 3 SRVs provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure-relief devices provides a conservative basis for the design and installation of the devices for ensuring that the devices will withstand these loads without loss of structural integrity or impairment of the overpressure-protection function.

In accordance with 10 CFR 50.34f(2)(x), PWR and BWR licensees and applicants are required to conduct testing to qualify the RCS SRVs and associated piping and supports under expected operating conditions for design-basis transients and accidents (TMI Action Item II.D.1). In the response to RAI 210.2 dated November 30, 1992 and in Revision 1 to the SSAR, Westinghouse revised Paragraph (2)(x) of Section 1.9.3 to state that the safety valve and discharge piping used in the AP600 design will be either of similar design as those items that were tested by EPRI and documented in EPRI Report EPRI NP-2770-LD, or will be tested in accordance with the guidelines in Item II.D.1 of NUREG-0737. The staff's evaluation of this response is discussed in Section 20.4 of this report.

3.9.3.3 Component Supports

The staff reviewed Section 3.9.3.4 of the SSAR with regard to the methodology used in the design of ASME Code Class 1, 2, and 3 component supports. The review included an assessment of the design and structural integrity of the supports. It addressed three types of supports (1) plate and shell, (2) linear, and (3) component standard types. The staff's review was conducted in accordance with the guidelines in Section 3.9.3.III.3 of the SRP.

In Section 3.9.3.4 of the SSAR, Westinghouse states that all ASME Code Class 1, 2, and 3 component supports for the AP600 design, including piping supports, are constructed in accordance with the 1989 Edition of Subsection NF of Section III of the ASME Code. The jurisdictional boundary between the NF supports and the building structure is based on the rules in Subsection NF. The staff position is that the 1989 Edition of Subsection NF is an acceptable code for the construction of all safety-related component and piping supports. In addition, the jurisdictional boundaries described in the 1989 Edition are sufficiently defined to ensure a clear division between the component or pipe support and the structural steel, and are acceptable. The AP600 design criteria for loadings and loading combinations for supports are discussed in Section 3.9.3.1 of this report. The staff's review of SSAR Section 3.9.3.4 resulted in the following additional comments:

- In the responses to RAI 210.42 and RAI 210.68 dated June 30, 1994 and July 25, 1994 respectively, Westinghouse agreed to revise Sections 3.6.2.3.2, 3.9.3.4, and 3.10.1.3 of the SSAR to delete a reference to a methodology that implied that some piping supports may be allowed to fail if certain conditions are met. This was identified as DSER Confirmatory Item 3.9.3.3-1. In Revision 3 to the SSAR, Westinghouse revised Sections 3.6.2.3.2, 3.9.3.4, and 3.10.1.3 to delete this proposed methodology. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.9.3.3-1 is closed.

- In the response to RAI 210.68 dated July 25, 1994, Westinghouse agreed to revise Section 3.9.3.4 and Tables 3.9-9 and 3.9-10 of the SSAR to state that the maximum allowable stress for supports of active components will be held to ASME Level C. In addition, the response agreed to revise the AP600 position on SRP Section 3.9.3.11.3.a in WCAP-13054 to provide this same commitment. This was identified as DSER Confirmatory Item 3.9.3.3-2. In Revision 3 to the SSAR, Westinghouse revised Section 3.9.3.4 and Tables 3.9-9 and 3.9-10 to provide a commitment that the allowable stresses for supports for active components will be held to ASME Service Level C. In Revision 15 to the SSAR, Westinghouse revised Appendix 1A under RG 1.130 to provide this same commitment. In Revision 3 to WCAP-13054, Westinghouse also revised the AP600 position on Section 3.9.3.11.3.a of the SRP to provide the same commitment. The staff concludes that limiting the allowable stress of supports designed to the rules of Subsection NF of Section III of the ASME Code to the Service Level C limits will not result in support deflections significant enough to prevent operability of supported active components, and is consistent with SRP 3.9.3. Therefore, DSER Confirmatory Item 3.9.3.3-2 is closed.

- In the DSER, the staff requested (1) more information in the SSAR relative to snubber operability assurance, (2) that the SSAR be revised to redefine large-bore snubbers as 50 kips or greater, rather than 1000 kips or greater, and (3) that if a snubber is used as a support for an active component, there should be a commitment in Section 3.9.3.4 and Tables 3.9-9 and 3.9-10 of the SSAR that these snubbers are included in the Service Level C allowable stress limitation discussed above in DSER Confirmatory Item 3.9.3.3-2. This was identified as DSER Open Item 3.9.3.3-1. In Revision 4 to the SSAR, Westinghouse revised Section 3.9.3.4 to provide most of the requested information as discussed below.

  In response to (1) above, Westinghouse revised Section 3.9.3.4.3 in Revision 4 to the SSAR to provide information relative to snubber operability that is consistent with the applicable guidelines in Section 3.9.3 of the SRP. Additionally, in Revision 11, Westinghouse further revised Section 3.9.3.4.3 to add a commitment to include dynamic testing as a part of the operability tests for all snubbers. Since these revisions are consistent with the staff's guidelines as stated above, they are acceptable. Further assurance of snubber operability during plant operation is assured by commitments in SSAR Sections 3.9.6, 5.4.2, and 6.6 to inservice testing and inspection in accordance with ASME Section XI, and the commitment in SSAR Section 1.9.4.2.3 to perform inservice testing of snubbers in accordance with ANSI/ASME OM Code-1990, "Code for Operation and Maintenance of Nuclear Power Plants." Article IWF 5300 of ASME Section XI references ANSI/ASME OM, Part 4 for inservice testing and inspection rules for snubbers. The staff reviewed the 1990 version of this standard, and finds it acceptable for use in the design certification for all ALWR plant designs. These commitments are consistent with the requirements in 10 CFR 50.55a(b)(2)(viii) and 50.55a(f)(4) and are, therefore, acceptable.

In response to (2) above, Westinghouse, in Revision 4 to Section 3.9.3.4.3 of the SSAR, redefined large bore hydraulic snubbers as those with capacities of 50 kips or greater. This is consistent with the staff's position as defined in Generic Issue 113 and is acceptable. With respect to (3) above, Revision 3 to Section 3.9.3.4 and Tables 3.9-9 and 3.9-10 of the SSAR provided a commitment that the allowable stresses for supports for active components will be held to ASME Service Level C. Therefore, if a snubber is used as a support for an active component, the same criterion will apply in the design of the snubber. This is consistent with the staff's guidelines in Section 3.9.3 of the SRP, and is acceptable. Therefore, on the basis of the above discussions DSER Open Item 3.9.3.3-1 is closed.

- In the response to RAI 210.74 dated June 27, 1994, Westinghouse agreed to revise Appendix 1A of the SSAR and WCAP-13054 to eliminate the reference to an obsolete subparagraph of Appendix F of Section III of the ASME Code and just reference the rules of the current Appendix F of Section III of the ASME Code for allowable loads for ASME Class 1 linear-type supports designed by the load rating method. The staff has accepted the rules of Appendix F for load rating in lieu of the RG 1.124 guidelines. Therefore, pending receipt of the SSAR and WCAP-13054 revisions, this response is acceptable. This was identified as DSER Confirmatory Item 3.9.3.3-3. In Revision 3 to the SSAR, Westinghouse revised Appendix 1A as discussed above and is acceptable. In Revision 2 to WCAP-13054, Westinghouse revised the basis for the AP600 exception to Position C.7.b in RG 1.124 to state that Appendix F to ASME III is used in the designs of these supports. This is consistent with the staff's position as discussed above and is also acceptable. Therefore, DSER Confirmatory Item 3.9.3.3-3 is closed.

- In the response to RAI 210.75 dated June 16, 1994, Westinghouse agreed to revise the SSAR, Appendix 1A and WCAP-13054 for RG 1.130, Position C.6.b to delete an unacceptable equation and just reference Appendix F of Section III of the ASME Code for allowable loads for ASME Class 1 plate-and-shell-type supports designed by the load rating method. The staff has accepted the load rating method in Subparagraph F-1332.7 of the ASME Code, Section III, Appendix F in lieu of the RG 1.130 guidelines. Therefore, pending receipt of the SSAR and WCAP-13054 revisions, this response was acceptable. This was identified as DSER Confirmatory Item 3.9.3.3-4. In Revision 15 to the SSAR, Westinghouse revised Appendix 1A to reference Appendix F of ASME Section III for the AP600 position on RG 1.130. As discussed above, this is acceptable. In Revision 3 to WCAP-13054, Westinghouse also revised the AP600 position on

RG 1.130 to agree with the response to RAI 210.75, and the revised Appendix 1A. Therefore, DSER Confirmatory Item 3.9.3.3-4 is closed.

- DSER Open Item 3.9.3.3-2 was related to the factor of safety for undercut type expansion anchor bolts used in the AP600 design for pipe support base plates. This issue was originated because SSAR Section 3.9.3.4, through Revision 9 committed only to the baseplate flexibility guidelines of IE Bulletin 79-02, "Pipe Support Baseplate Designs Using Concrete Expansion Bolts." The factor of safety guidelines in this Bulletin were not addressed. In Revision 11 to Section 3.9.3.4, Westinghouse added a sentence which states that supplemental AP600 design criteria for fastening anchor bolts to concrete are outlined in Section 3.8.4.5.1 of the SSAR. The staff's evaluation of these criteria is contained in Section 3.8.4.2 of this report as a part of DSER Open Item 3.8.4.2-2. Therefore, DSER Open Item 3.9.3.3-2 is considered to be subsumed by DSER Open Item 3.8.4.2-2, and is closed.

On the basis of the evaluation in Section 3.9.3.3 of this report, supplemented by the evaluations in applicable portions of Section 3.12.6 of this report, the staff concludes that Westinghouse meets the requirements of 10 CFR 50.55a and GDC 1, 2, and 4 with regard to the design and service load combinations and associated stress limits specified for ASME Code, Class 1, 2, and 3 component supports by ensuring that component supports are designed to quality standards commensurate with their importance to safety, and that these supports can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, has met the applicable guidelines in SRP 3.9.3, and are acceptable. The specified design and service loading combinations used for the design of ASME Code, Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or other service loadings because of postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative design basis to assure that support components can withstand the most adverse combination of loading events without loss of structural integrity.

The staff's evaluation of Class CS components is given in Section 3.9.5 of this report.

3.9.4 Control Rod Drive Systems

The staff's review under Section 3.9.4 of the SRP included the control rod drive system (CRDS) up to its interface with the control rods. Those components of the CRDS that are part of the primary pressure boundary are classified as SC 1, Q G A, and are designed according to the Class 1 requirements of Section III of the ASME Code and to the QA requirements of Appendix B of 10 CFR Part 50. The CRDS will be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences or under postulated accident conditions. The staff reviewed the information in Section 3.9.4 of the SSAR related to the criteria used to ensure the structural integrity of this system during normal operation and under accident conditions. These criteria conform to Section 3.9.4 of the SRP and are therefore, acceptable. Loading combinations for the CRDS are discussed in Section 3.9.3.1 of this report.

As a part of its review of the dynamic analysis of reactor internals under faulted conditions, the staff is reviewing information relative to production tests of the CRDMs and the acceptance criteria for ensuring operational adequacy under LOCA and SSE events. The staff's evaluation of this issue is discussed in Section 3.9.2.4 of this report. The evaluation of the structural integrity of the seismic restraints for the CRDMs is discussed in Section 3.9.7 of this report. Additional evaluations relative to the functional design and testing of these systems are discussed in Section 4.6 of this report.

The staff's review acceptance criteria for the CRDS are based on the following requirements:

- GDC 1 and 10 CFR 50.55a requiring that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2 requiring that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions

- GDC 14 requiring that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture

The staff concludes that the design of the control rod drive system is acceptable for the AP600 and meets GDC 1, 2, and 14, and 10 CFR 50.55a.

As the staff stated in the first paragraph in this section, by designing the CRDS up to its interface with the control rods to acceptable loading combinations of normal operation and accident conditions using ASME Class 1 and 10 CFR Part 50, Appendix B, requirements, Westinghouse has assured the structural integrity of the CRDS. Therefore, Westinghouse meets GDC 1 and 10 CFR 50.55a with regard to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. In addition, Westinghouse meets GDC 2 and 14 with regard to designing the control rod drive system to withstand the effects of earthquakes and anticipated normal operation occurrences with adequate margins to ensure its structural integrity and functional capability and with an extremely low probability of leakage or gross rupture of the RCPB. The staff's evaluations of the specified design transients, design and service loadings, and combinations of loads, are discussed in Sections 3.9.1 and 3.9.3.1 of this report. By limiting the stresses and deformations of the CRDS under such loading combinations, the design conforms to the appropriate guidelines in Sections 3.9.3 and 3.9.4 of the SRP.

3.9.5 Reactor Pressure Vessel Internals

In accordance with Section 3.9.5 of the SRP, the staff reviewed SSAR Section 3.9.5 relative to the load combinations, allowable stress and deformation limits, and other criteria used in the design of the reactor internals. The staff's review acceptance criteria are based on meeting the following requirements:

- GDC 1 and 10 CFR 50.55a requiring that the reactor internals be designed to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2 requiring that the reactor internals be designed to withstand the effects of earthquakes without loss of capability to perform its safety functions

- GDC 4 requiring that reactor internals be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated LOCA

- GDC 10 requiring that reactor internals be designed with adequate margins to assure that specified acceptable fuel design limits are not exceeded during anticipated normal operational occurrences

The staff's evaluation of SSAR Section 3.9.5 resulted in the following comments:

- In Section 3.9.5.2.4 of the SSAR, Westinghouse states that the core support structures for the AP600 are designed and constructed in accordance with Subsection NG of Section III of the ASME Code. In accordance with Subsection NG-1100, this means that the manufacture and installation of the AP600 core support structures are in accordance with the NG rules required for materials, design, examination, and preparation of reports. For design, this means Service Level A, B, C, D conditions should meet requirements shown in Figures NG-3221-1, NG-3224-1, NG-3232-1 and Appendix F of Subsection NG of Section III of the ASME Code. This conforms to Section 3.9.5 of the SRP and is acceptable. However, in Section 3.9.5.2.4 of the SSAR, Westinghouse does not clearly specify the design requirements for reactor internals other than the core support structures. In the response to RAI 210.70 dated June 30, 1994, Westinghouse proposed a revision to Section 3.9.5.2.4 and Table 3.2-3 of the SSAR to clarify these design requirements. For internal structures other than core supports, Westinghouse is committed to the design requirements of Subsection NG of Section III of the ASME Code. These components are constructed so as not to adversely affect the integrity of the core support structures as required by Subsection NG-1122 of Section III of the ASME Code. These criteria conform to Section 3.9.5 of the SRP and are acceptable, pending receipt of the SSAR revision. (See DSER Confirmatory Item 3.9.2.4-3) Subsequently, Westinghouse revised Section 3.9.5.2.4 in Revision 4 to the SSAR, and revised Table 3.2-3 in Revision 11 to the SSAR. The changes contain commitments to design all reactor internal structures to requirements of ASME III Code Section III, Subsection NG, and Appendix F. This is acceptable; thus, as discussed in Section 3.9.2.4 of this report, DSER Confirmatory Item 3.9.2.4-3 is closed.

- The ASME Code requires that a design specification be prepared for Class CS core support structures and the safety-related reactor internal components. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design data inputs. The Code also requires a stress report for the ASME Code, Class CS components. During the audit on July 27 through 28, 1994, the staff found that detailed drawings, design and analysis specifications and scoping analysis of reactor internals are available, and design assumptions used are generally in conformance with regulatory positions. The staff was told that more detailed analyses to finalize the design are under preparation and near completion. Subsequent to completion of these final analyses, Westinghouse should inform the staff that these documents are available for staff audit. This was identified as DSER Open Item 3.9.5-1.

During the audit on May 10, 1995, the staff found that Westinghouse completed detailed structural and thermal analyses for those reactor internals components either not similar to a component found in earlier Westinghouse plants or similar but traditionally having a low-stress margin. These components include lower core support assembly, vortex ring assembly, lower radial key assembly, lower radial key restraint Clevis insert, upper support assembly, reflector assembly, and core barrel assembly. AP600 specific transient loads and load combinations were used. Analyses of these components are documented in Westinghouse Calculation No. MI01-S3A-001. Westinghouse also performed scoping analyses for other AP600 reactor internal components which have high stress margin in similar existing designs. The scoping analyses are documented in Westinghouse Calculation No. MI01-M2C-001, which are preliminary analyses using best-estimate bounding loads. Westinghouse indicated that all analyses will be updated once the AP600-specific seismic and LOCA loads have been determined. Future updated analyses are unlikely to change the reactor internals design significantly because of conservatively defined loads used in current analyses. The staff review concludes that the scoping analysis provides the basis for the expectation that the reactor internals design will meet functional requirements with possible minor changes. However, specific ITAAC will be needed in this case for verifying Code compliance and existence of stress reports in the COL stage. Further staff review of ITAAC submittals on the subject will be conducted separately. Thus, DSER Open Item 3.9.5-1 is closed.

- To assure the safety function of reactor internals, in RAI 210.96, the staff requested that Westinghouse describe how the use of Appendix F of Section III of the ASME Code will assure the functionality of these internals. In the response to RAI 210.96 dated June 27, 1994, Westinghouse indicated that, to assure their safety function, reactor internals are analyzed and compared to the allowable stresses of Appendix F, while also meeting the deflection limits in Table 3.9-14 and Section 3.9.5.3.2 of the SSAR. As discussed below and in Section 3.9.2.4 of this report, these criteria are acceptable because they provide adequate margins to maintain geometry of internals for control rod insertion and for core flow passage.

- In the response to RAI 210.97 dated June 30, 1994, the bases for deflection limits are explained and Westinghouse also proposed a revision of Section 3.9.5.3.1 of the SSAR to incorporate these bases. The upper barrel radial inward deflection limit is necessary to prevent contact between the barrel and the peripheral upper guide tubes during a LOCA event, such that drop of control rods will not be impaired. The radial outward deflection limit maintains flow in the downcomer annulus between the core barrel and reactor vessel wall. The upper package deflection limit maintains the clearance between the upper core plate and guide tube support pin shoulder and prevents buckling of the guide tube. The rod cluster guide tube lateral deflection limit minimizes interference to rod drop such that acceptable rod drop time can be maintained. The above deflection limits provide assurance that the control rod insertion function will not be impaired and adequate flow passage for core cooling will be maintained during and after the event of combined occurrences of LOCA and SSE, and are, therefore, acceptable pending receipt of the SSAR revision. This was identified as DSER Confirmatory Item 3.9.2.4-2. Subsequently, in Revision 4 to the SSAR, Westinghouse explained the bases of deflection limits in Section 3.9.5.3.2 as proposed. Thus DSER Confirmatory Item 3.9.2.4-2 is closed. (See also DSER Open Item 3.9.7-1).

- As discussed in Section 3.9.2.4 of this report, Westinghouse indicated in responses to RAI 210.20 (January 8, 1993), RAI 210.21 (January 8, 1993), RAI 210.22 (January 8, 1993) and RAI 210.95 (June 30, 1994) that, although the AP600 design loads for LOCA conditions are based on the use of mechanistic pipe break criteria, to be consistent with past practice, enveloping LOCA loads based on the dynamic effects of 0.093 m^2 (1 ft^2) hot leg and cold leg breaks are actually used in the analysis of the reactor internals. In response to RAI 210.95 and in Revision 4 to the SSAR, Westinghouse revised the final paragraph of Section 3.9.2.5 of the SSAR to reflect their position. This is acceptable because dynamic effects of LOCA loads based on 0.093 m^2 (1 ft^2) hot and cold leg breaks for reactor internals design will bound future refined LOCA loads when the leak-before-break concept is applied to major portions of the primary loop piping. (See previous discussion on DSER Open Item 3.6.3-1 and DSER Confirmatory Item 3.9.2.4-1).

- The staff's review of Figures 3.9-5 and 3.9-6 in Revision 1 of the SSAR found that the figures lack detailed descriptions regarding how different parts of reactor internals are connected. Information regarding relative locations of the internals to each other and to the reactor vessel are not given. In addition, key dimensions of the reactor vessel and its internals are not provided. In the response to RAI 210.99 dated June 27, 1994, Westinghouse proposed a revision of Section 3.9.5.1.4 of the SSAR and a new Figure 3.9-8 to the SSAR to provide a more detailed description of reactor internals interface arrangement. Pending SSAR revision, the issue was resolved. This was identified as DSER Confirmatory Item 3.9.5-1. Subsequently, in Revision 4 to the SSAR, Westinghouse revised Section 3.9.5.1.4 as proposed, and in Revision 5 to the SSAR, Westinghouse included new Figure 3.9-8. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.9.5-1 is closed.

- In the response to RAI 210.17 dated January 8, 1993, key dimensions of the lower reactor internals are added to the original Figure 3.9-5 of the SSAR and key dimensions of the upper core support structure are added to the original Figure 3.9-6 of the SSAR. However, no SSAR revision to incorporate changes of Figures 3.9-5 and 3.9-6 of the SSAR was proposed. At the staff's request during the July 27 through 28, 1994, meeting, Westinghouse agreed to revise Figures 3.9-5 and 3.9-6 of the SSAR for adding key dimensions This was identified as DSER Open Item 3.9.5-2. Subsequently, in Revision 5 to the SSAR, Westinghouse added the key dimensions to Figures 3.9-5 and 3.9-6. The staff finds this acceptable, and therefore, DSER Open Item 3.9.5-2 is closed.

- In the response to RAI 210.101 dated June 27, 1994, Westinghouse proposed a revision of Section 3.9.5.2.5 of the SSAR to incorporate internals design conditions and add Figure 5.3.4-1 to the SSAR to show the reactor vessel key dimensions in plan view at the nozzle level cross section. In the July meeting, Westinghouse also agreed to add a side view to this figure to incorporate dimensions of reactor vessel height and thickness of the vessel wall. This was identified as DSER Open Item 3.9.5-3. In Revision 3 to the SSAR, reactor vessel key dimensions at nozzle level plan view are shown in redesignated Figure 5.3-5 and key dimensions at side view are shown in added Figure 5.3-6. The major reactor vessel design parameters are shown in added Table 5.3-5. The staff finds this acceptable, and therefore, DSER Open Item 3.9.5-3 is closed.

- In RAI 210.98, the staff questioned whether thermal stratification effects are considered in the design and analysis of reactor vessel and reactor internals. In the response to RAI 210.98 dated July 22, 1994, Westinghouse indicated that thermal stratification is addressed in the design of the AP600 reactor vessel and reactor internals. The methodology considered includes stratification in the direct vessel injection line and downcomer during PXS operation and stratification in the reactor vessel closure head during natural circulation cooldown. Thermal stresses are calculated and cyclic loading effects on fatigue life are evaluated. Westinghouse also proposed a revision of Section 5.3.4.1 of the SSAR to incorporate these thermal stratification considerations. The staff's evaluation found that this response is adequate to assure that the cyclic effects of thermal stratification are considered in fatigue evaluations for the design and analysis of the AP600 reactor internals, and is therefore acceptable, pending receipt of the SSAR revision. This was identified as DSER Confirmatory Item 3.9.5-2. In Revision 3 to the SSAR, Westinghouse revised Section 5.3.4.1 as proposed. Thus Confirmatory Item 3.9.5-2 is closed.

- The staff found that in Table 3.7-1 of the SSAR, the damping value assigned for the fuel assemblies is 20 percent. Westinghouse was requested to provide basis that justifies the use of this damping value. (See also DSER Open Item 3.7.1-3)

  During the audit on May 10, 1995, Westinghouse presented a response to this issue in a separate proprietary and non-proprietary attachment to a letter, NTD-NRC-95-4460 dated May 10, 1995. Westinghouse's evaluation of fuel assembly damping values by analysis and testing was provided. In the response, Westinghouse states that as a result of combined effects of inter-fuel assembly rubbing and scraping, fuel rod and grid spacer relative motions and frictional forces, and fluid-structure interactions in a closely packed reactor core, damping value increases as amplitude of vibration increases. The fuel assemblies are structurally flexible with low fundamental frequency, and a large amplitude response to postulated seismic loads is expected. Westinghouse's evaluation concludes that a uniform 10 percent damping value is used for all modes higher than the fundamental mode, and use 20 percent damping value for the fundamental mode to account for additional hydrodynamic effects. In a letter dated January 8, 1997, Westinghouse indicated that the damping value is justified by test and is consistent with evaluations for Westinghouse designed fuel in operating nuclear power plants. This is acceptable. In Revision 11 to the SSAR Section 3.9.2.6, Westinghouse included topical report WCAP-8236, "Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant," in the list of references. The report provides test results to support the damping value. Therefore, this issue is resolved.

- The incore neutron monitoring thimble tubes had experienced thinning as a result of flow-induced vibration in operating PWRs of Westinghouse design. In Bulletin 88-09, the NRC requested all licensees of these plants to establish and implement an inspection program to periodically confirm incore thimble tube integrity. Westinghouse was requested to provide information to verify that either such concern does not exist in AP600 because of an improved thimble design, or an inspection program, in conformance with guidelines given in NRC Bulletin 88-09 be established and implemented as a COL Action Item in all AP600 plants. In the latter case, a description of the inspection program would be provided in the SSAR. Subsequently, Westinghouse submitted the letter NSD-NRC-96-4841 dated October 14, 1996, which indicated that the AP600 incore thimble is an improved design with better wear resistant material, larger diameter, stiffer, and smaller gap between thimble and guide tube. All these features result in minimized vibration. In addition, the double-wall design feature will prevent non-isolable leak of reactor coolant, and preclude the need for inservice inspection. Westinghouse also revised the final paragraph of Section 3.9.7.2 in Revision 10 to the SSAR. The staff found that the letter response and the SSAR revision are acceptable, and therefore, this issue is resolved.

On the basis of the above evaluations and resolution of the open and confirmatory items discussed in this section, the staff's conclusions relative to the design of the AP600 reactor internals are as follows:

In accordance with Table 3.2-3 of the SSAR, the core support structures are safety-related reactor internals, are designed as Safety Class 3 components, and are designed to the QA requirements of 10 CFR Part 50, Appendix B. In addition, as discussed in Section 3.9.1, Section 3.9.2.4, and Section 3.9.3.1, of this report, the SSAR contains acceptable criteria for the design of all safety-related reactor internals under normal, upset, emergency, and faulted loading conditions.

On the basis of these evaluations related to designing all safety-related reactor internals:

- as Safety Class 3
- to the QA requirements of 10 CFR Part 50, Appendix B
- to acceptable rules of Section III of the ASME Code

Westinghouse meets GDC 1 and 10 CFR 50.55a with regard to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed.

On the basis of these evaluations related to designing all safety-related reactor internals to acceptable loading combinations and stress limits when the internals are subjected to the loads associated with normal, upset, emergency, and faulted conditions, Westinghouse meets GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquakes and the effects of normal operation, maintenance, testing, and postulated LOCAs with sufficient margin to ensure that their capability to perform their safety functions is maintained and the specified fuel design limits are not exceeded.

The implementation of the criteria discussed above to the design of the reactor internal structures and components provides reasonable assurance that, in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components will not exceed allowable stresses and deformations under such loading combinations. These criteria provide an acceptable design basis for ensuring that these structures and components will withstand the most adverse loading events that were postulated to occur during their service lifetime without loss of structural integrity or impairment of function.

As discussed above, the staff concludes that the design of reactor internals for the AP600 meets GDC 1, 2, 4, and 10 and 10 CFR 50.55a, and is, therefore acceptable.

3.9.6 Testing of Pumps and Valves

In Section 3.9.6 of the SSAR, Westinghouse discusses inservice testing (IST) of certain safety-related pumps and valves typically designated as ASME Code Class 1, 2, or 3. The staff's review of SSAR Section 3.9.6 and its acceptance criteria are on the basis of meeting the following requirements:

- GDC 37 as related to periodic functional testing of the ECCS to assure the leak tight integrity and performance of its active components
- GDC 40 as related to periodic functional testing of the containment heat removal system to assure the leak-tight integrity and performance of its active component
- GDC 43 as related to periodic functional testing of the containment atmospheric cleanup systems to assure the leak-tight integrity and the performance of the active components, such as pumps and valves
- GDC 46 as related to periodic functional testing of the cooling water system to assure the leak-tight integrity and performance of the active components
- GDC 54 as related to piping systems penetrating containment being designed with the capability to test periodically the operability of the isolation function and determine valve leakage acceptability
- 10 CFR 50.55a(f) as related to the verification of the operational readiness of pumps and valves by periodic testing and, in particular, the extent to which pumps and valves classified as ASME Code Class 1, 2, and 3 are designed and provided with access to enable the performance of testing of pumps and valves for assessing operational readiness

In Section 3.9.3 of this report, the staff discusses the design of safety-related valves for the AP600 design. There are no safety-related pumps in this design. The load combinations and stress limits used in the design of valves ensure that the integrity of the component pressure boundary will be maintained. In addition, a licensee will periodically test the performance and measure performance parameters of safety-related valves in accordance with Section XI of the ASME Code, as required by 10 CFR 50.55a(f). Periodic measurements of various parameters will be compared to baseline measurements to detect long-term degradation of the valve performance. The tests, measurements, and comparisons will ensure the operational readiness of these valves. However, as discussed in SECY-90-016, "Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements," the staff determined that the requirements of Section XI of the ASME Code alone might not assure the necessary level of component operability that is desired for ALWR designs. Accordingly, in SECY-90-016, as supplemented by the staff's April 27, 1990, response to comments by the ACRS, the staff recommended criteria to the Commission to be used to supplement those of Section XI of the ASME Code. In its SRM dated June 26, 1990, on SECY-90-016, the Commission approved the staff's recommendations. The staff's proposed criteria for pump and valve testing are as follows:

- Piping design must incorporate provisions for full flow testing at maximum design flow of pumps and check valves.

- Check valve testing must incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

- Provisions must be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced non-intrusive techniques.

- Provisions must be incorporated to test motor-operated valves (MOVs) under design-basis differential pressure.

In SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," dated March 28, 1994, and in SECY-95-132 (revision of SECY-94-084) dated May 22, 1995, and in Consolidation of SECY-94-084 and SECY-95-132 dated July 24, 1995, the staff made the following additional recommendations relative to the testing of pumps and valves:

- Important non-safety-related components are not required to meet criteria similar to safety-grade criteria. However, the important non-safety-related pumps and valves as identified by the RTNSS process should be designed to accommodate testing in accordance with the requirements of Section XI of the ASME Code. Specific positions on the IST requirements for those components will be determined as a part of the staff's review of the plant-specific implementation of the RTNSS.

- To the extent practical, the passive ALWR piping systems should be designed to accommodate the applicable Code requirements for the quarterly testing of valves, rather than to allow designs that only accommodate testing during cold shutdowns or refueling outages. However, design configuration changes to accommodate Code-required quarterly testing should be done only if the benefits of the test outweigh the potential risk.

- To the extent practical, the passive system designs should incorporate provisions (1) to permit all critical check valves to be tested for performance in both forward and reverse flow directions, and (2) to verify the movement of each check valve's obturator during IST by observing the direct instrumentation indication of the valve position, such as a position indicator or by performing non-intrusive test methods. The demonstration of the non-safety direction test need not be as rigorous as the corresponding safety direction test.

- The passive system designs should incorporate provisions to test safety-related power-operated valves (POVs) under design-basis differential pressure and flow. Before installation, the design capability of these types of valves should be demonstrated by a qualification test. Before initial startup, the valve capability under design-basis differential pressure and flow should be verified by a pre-operational test. During the operational phase, the valve capability under design-basis differential pressure and flow should be verified through a program similar to that being developed for MOVs. Similarly, to the extent practicable, the design of RTNSS systems should incorporate provisions to periodically test power-operated valves in the system during operations to assure that the valves meet their intended functions under design-basis conditions.

- To the extent practical, provisions should be incorporated in the design to assure that MOVs in safety-related systems are capable of recovering from mispositioning.

The staff's evaluation of Section 3.9.6 of the SSAR and the IST program that was provided by Westinghouse, up to Revision 22 of the SSAR, is based on meeting the requirements of Section XI of the ASME Code as well as the above applicable staff's recommendations. The staff's evaluation is discussed in the sections below. It should be noted that the staff's evaluation of two DSER open items pertaining to valve qualification testing is discussed in Section 3.9.6.5 of this report.

As discussed above, the IST program shall be in accordance with Section XI of the ASME Code, as required by 10 CFR 50.55a(f). The regulations currently require the IST program to comply with the 1989 Edition of ASME Code Section XI, which references ASME/ANSI OMa-1988, Parts 6 and 10. In Revision 11 to Section 3.9.6 of the SSAR, Westinghouse references the 1990 ASME Operation and Maintenance (OM) Code. Westinghouse was requested to revise the Section 3.9.6 to reflect the current regulations, or request a relief. In a letter dated February 14, 1997, Westinghouse requested to use the 1990 Edition of ASME OM Code in lieu of the 1989 Edition of ASME Code Section XI. The use of the 1990 Edition of ASME OM Code is an acceptable alternative, because there are no technical differences between the 1989 Edition of ASME Code Section XI and the 1990 Edition of ASME OM Code with regards to the inservice testing of pumps and valves. Pursuant to 10 CFR 50.55a(a)(3)(i), the staff finds that the use of 1990 Edition of the OM Code provides an acceptable level of quality and safety, and, therefore, the request to use the alternative is authorized.

The 1990 Edition of ASME OM Code requires inservice testing of pumps and valves that are required to bring the plant to cold shutdown conditions, or to maintain it in cold shutdown conditions. However, as discussed in SSAR Section 7.4, the AP600 is designed for safe-shutdown conditions other than cold shutdown conditions. Because of this design, Westinghouse excludes certain valves that are needed to bring the plant to cold shutdown from the scope of its IST program and requires in Section 3.9.6.2 to inservice test those valves that are required to bring the plant to safe-shutdown conditions or to maintain it in safe-shutdown conditions. This is acceptable on the basis of SECY-94-84 that the safe-shutdown conditions, rather than specific cold shutdown conditions, are approved for the AP600 design.

3.9.6.1 Testing of Pumps

In Section 3.9.6.1 of the earlier SSAR, Westinghouse stated that there are no safety-related pumps in the AP600 design and the AP600 IST program does not include testing of non-safety-related pumps. The staff noted in Section 3.9.6 of this report that the important non-safety-related pumps, as identified by the RTNSS process, should be designed to accommodate testing in accordance with the requirements of Section XI of the ASME Code. Westinghouse was requested to address this issue and this was identified as DSER Open Item 3.9.6.1-1. In response to the open item, Westinghouse revised the SSAR in Revision 4 and stated that instrumentation (e.g., flow rate, head and vibration instruments) is installed to allow confirmation of the pump's operability in systems with RTNSS important functions. The design allows in service testing of RTNSS pumps in accordance with the requirements of Section XI of the ASME Code, and is consistent with the staff recommendation for RTNSS components. The staff finds this acceptable, and therefore, DSER Open Item 3.9.6.1-1 is closed.

3.9.6.2 Testing of Safety-Related Valves

As discussed in Section 3.9.6 of this report, the AP600 design should incorporate provisions to test MOVs under design-basis differential pressure and flow. The design-basis capability should be verified before the valves are installed, before startup, and periodically throughout plant life. The concerns and issues identified in GL 89-10, its supplements, and GL 96-05 for MOVs should be addressed before plant startup. Westinghouse should provide specific requirements in the SSAR for design and qualification testing, pre-operational testing, and IST of safety-related MOVs, to demonstrate their design-basis capability before installation, before startup, and throughout the plant's life. The method of assessing the loads, the method of sizing the actuators, and the setting of the torque and limit switches should also be specifically addressed in the SSAR. This was identified as DSER Open Item 3.9.6.2-1. In Revision 13 of the SSAR (Section 5.4.8.1.2), Westinghouse commits to design and qualify the MOVs for a range of conditions up to the design conditions, and incorporates test provisions to qualify the MOVs up to maximum design-basis operating conditions. Westinghouse also provides criteria for sizing the motor operators and performing valve functional qualification tests. In Revision 13 to Section 5.4.8.5 of the SSAR, Westinghouse specifies provisions for a pre-operational testing before startup that should be used by COL applicants to demonstrate that the results of testing under insitu or installed conditions can be used to confirm the capacity of MOVs to operate under design conditions. Additionally, in Revision 20 to Section 3.9.6 and Table 3.9-16 of the SSAR, Westinghouse commits to develop an IST program consistent with staff positions and criteria as identified in GL 89-10, its supplements, and GL 96-05 for MOVs to demonstrate their design-basis capability throughout the plant life. The staff has reviewed this information and finds it acceptable, and therefore, DSER Open Item 3.9.6.2-1 is closed. Furthermore, any proposed change to the requirements for design, qualification, and testing of motor-operated valves in SSAR Sections 5.4.8.1.2 and 5.4.8.5.2 will require NRC approval prior to implementation of the change.

As discussed in Section 3.9.6 of this report, Westinghouse is also requested to provide a commitment that provisions will be incorporated in the design, to the extent practical, to assure that MOVs in safety-related systems are capable of recovering from mispositioning. This was identified as DSER Open Item 3.9.6.2-2. In response to this open item, Westinghouse revised the SSAR and stated in Revision 5 to Section 5.4.8.1.2.1 that MOVs are designed to change their position from an improper position (mispositioned) either before or during accidents. The staff finds this commitment acceptable, and therefore, DSER Open Item 3.9.6.2-2 is closed.

As discussed in Section 3.9.6 of this report, AP600 system designs should incorporate provisions to test safety-related POVs (other than MOVs) under design-basis differential pressure and flow. The design-basis capability of these types of valves should be verified before the valves are installed, before startup, and periodically through a program similar to that recommended for MOVs. Westinghouse should provide requirements in the SSAR for design and qualification testing, pre-operational testing, and IST for these safety-related POVs (other than MOVs), to demonstrate their design-basis capability before installation, prior to startup, and throughout the plant's life. This was identified as DSER Open Item 3.9.6.2-3. In Revision 13 to Section 5.4.8.1.3 of the SSAR, Westinghouse commits to design and qualify the POVs for their respective design basis and required operating conditions, Westinghouse also provides criteria for sizing the operators and performing functional qualification tests. In Revision 13 to Section 5.4.8.5 of the SSAR, Westinghouse specifies provisions for a pre-operational testing before startup that should be used by COL applicants to demonstrate that the results of testing under insitu or installed conditions can be used to confirm the capacity of POVs to operate under design conditions. Additionally, in Revision 20 to Section 3.9.6 and Table 3.9-16 of the SSAR, Westinghouse commits to develop an IST program consistent with staff positions and criteria as identified in GL 89-10, its supplements, and GL 96-05, where applicable, for POVs to demonstrate their design-basis capability throughout the plant's life. The staff has reviewed this information and finds it acceptable, and therefore, DSER Open Item 3.9.6.2-3 is closed. Furthermore, any proposed change to the requirements for design, qualification, and testing of power-operated valves in SSAR Sections 5.4.8.1.3 and 5.4.8.5.3 will require NRC approval prior to implementation of the change.

As discussed in Section 3.9.6 of this report, to the extent practical, AP600 system designs should incorporate provisions to permit all critical check valves to be tested for performance in both forward and reverse flow directions. However, the demonstration of the non-safety direction test need not be as rigorous as the corresponding safety direction test. The AP600 design should also incorporate provisions to allow for movement of the safety-related check valve obturator to be verified. The verification of valve obturator movement may be made by observing a direct indicator such as a position indicator or by other positive means, including non-intrusive test methods.

It is noted that the IST program submitted in response to RAI 210.24R, did not specify check valves to be tested in both directions. Westinghouse was requested to revise the applicable SSAR section to comply with the above staff position. In Revision 4 to the SSAR, Westinghouse deleted the testing direction for check valves in Table 3.9-16, and stated that exercising a check valve confirms the valve's capability to move to the safety position(s). Additionally, Westinghouse states that the exercise test shows that the check valve opens in response to flow and closes when the flow is stopped. Therefore, each check valve will be exercised in both directions and Westinghouse has complied with the staff position.

The staff position on the use of non-intrusive diagnostic techniques as stated in SECY-90-016 is that IST is to incorporate the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves. The system and component design should assure that non-intrusive diagnostic methods can be accommodated.

In Revision 4 to Section 3.9.8 of the SSAR, Westinghouse stated that the IST program will include provisions for non-intrusive check valve testing methods. This is acceptable. However, additional changes to Section 5.4.8.1.1 of the SSAR are necessary. In this section Westinghouse states that design provisions for non-intrusive determination of disk position and potential valve degradation will only be provided for selected valves. As discussed in Section 3.9.6 of this report, to the extent practical, each valve's obturator movement should be capable of observation by direct indication or non-intrusive test methods. This was identified as DSER Open item 3.9.6.2-4. In response to this open item, Westinghouse revised the SSAR in Revision 12 such that all active safety-related check valves include the capability to verify valve obturator movement by a direct indication or by using non-intrusive test methods. The staff finds this commitment acceptable, and therefore, DSER Open Item 3.9.6.2-4 is closed.

With regard to flow testing of check valves, Westinghouse states in Revision 4 to Section 3.9.6.2 of the SSAR that where practical, check valves will be full-flow tested under actual plant conditions. Where full-flow or actual plant conditions are not achievable, alternative test methods will be outlined in the test program. Westinghouse stated in Revision 4 to Section 3.9.6.2.2 that check valve forward flow tests will be performed at sufficient flow to fully open the valve, unless the maximum accident flowrates are not sufficient to fully open the valve, in which case, the maximum accident flowrate will be used. It is acceptable to exercise check valves with sufficient flow to full-open the valve, provided the valve's fully open position can be positively confirmed.

As discussed in Section 3.9.6 of this report, for the AP600 design, safety-related valves are to be periodically disassembled and inspected to determine if there are any indications of unacceptable corrosion or degradation that cannot be detected through the use of advanced non-intrusive techniques. It is the staff's view that information derived from IST alone is not adequate to assess valve condition and to determine required maintenance. The frequency of inspection and the extent of disassembly may vary depending upon the service condition of the valve. The staff requires, as a minimum, a commitment for the COL applicant to develop a program that will establish the frequency and extent of disassembly and inspection of safety-related valves, including the basis for the frequency and the extent of each disassembly. Westinghouse was requested to revise the applicable SSAR section to comply with the staff's position as stated in SECY-90-016. This was identified as DSER Open Item 3.9.6.2-5. In response to this open item, Westinghouse revised Section 3.9.8 in Revision 11 to include a commitment for the COL applicant to develop a program for valve disassembly and inspection outlined in Section 3.9.6.2.3 of the SSAR. The staff finds this acceptable, and therefore, DSER Open Item 3.9.6.2-5 is closed. This is COL Action Item 3.9.6.2-1.

In an early version of the SSAR, Westinghouse stated that fail-safe valves that rely upon non-safety-related systems to provide actuation power are subject to IST. ASME/ANSI Part 10 of Operation and Maintenance (OM Part 10), Section 4.2.1.6, however, requires all fail-safe valves, including those that rely upon safety-related systems to provide actuation power, to be fail-safe tested. Westinghouse was requested to revise the applicable SSAR section to comply with the Code. This was identified as DSER Open item 3.9.6.2-6.

In response to DSER Open Item 3.9.6.2-6, Westinghouse requires, in Revision 4 to Section 3.9.6.2.2 of the SSAR, that all safety-related fail-safe valves are subject to a valve exercise inservice test. The required test verifies that the valve repositions to the safety-related position on loss of actuator power. The staff finds that this valve exercise test performed by removing actuator power to the valve satisfies the requirement of the fail-safe test. The staff finds this acceptable, and therefore, DSER Open Item 3.9.6.2-6 is closed.

Under the discussion of leakage testing in an early version of the SSAR, Westinghouse stated that in some cases, pressure isolation is satisfied by performing a flow test. It was not apparent how a flow test satisfies the requirements of OM Part 10 Section 4.2.2, Valve Seat Leakage Test. This was identified as DSER Open Item 3.9.6.2-7.

In response to DSER Open Item 3.9.6.2-7, Westinghouse deleted the statement of a flow test and stated, in Revision 4 to Section 3.9.6.2.2 of the SSAR, that valves with a safety-related seat leakage limit will be tested to verify their seat leakage. These valves include (1) pressure isolation valves (PIVs) that provide isolation between low and high pressure systems, (2) temperature isolation valves (TIVs) whose leakage may cause unacceptable thermal loading to piping or supports, and (3) containment isolation valves (CIVs) that provide isolation of piping that penetrates containment. This was acceptable and DSER Open Item 3.9.6.2-7 was considered closed. However, in Revisions 8 and 10 of the SSAR, Westinghouse deleted the test requirements for PIVs and TIVs and modified the test requirements for CIVs.

Westinghouse states that the AP600 requires no testing of PIVs and TIVs. This is not acceptable. Westinghouse should confirm either that there are no PIVs and TIVs in the AP600 design or that there are PIVs and TIVs but there is no specified maximum leakage requirement, and therefore, these valves are not required by the Code to be tested.

However, with regards to PIVs, the staff requested in DSER a commitment in the SSAR for the COL applicant to perform periodic leak testing of all safety-related RCS pressure isolation valves in accordance with applicable sections of the AP600 technical specifications (TSs). If the TSs do not specify a list of all PIVs, this list should be provided in the SSAR. This was identified as DSER Open Item 3.9.6.2-8. In Revision 10 of the SSAR, Westinghouse states that the AP600 requires no testing of PIVs that provide isolation between high and low-pressure systems. Additionally, in the AP600 TSs, Westinghouse deleted the section for leak testing of RCS PIVs. The SSAR and TS changes are not in conformance with the staff position and are not acceptable. The TS for PIV leak testing should be reinstated or a list of PIVs should be provided either in the TS or in the SSAR. In response to the staff's comment, Westinghouse revised the SSAR, and states in Revision 18 to Section 3.9.6.2.2 that the AP600 maximum leakage requirement for PIVs that provide isolation between high and low-pressure systems is included in the surveillance requirements for TS 3.4.16. The PIVs that require leakage testing are tabulated in Table 3.9-18 of the SSAR. These TS and SSAR changes for PIV leak testing are acceptable and DSER Open Item 3.9.6.2-8 is closed.

With respect to CIVs, Westinghouse states in Revision 8 of the SSAR that they are tested in accordance with 10 CFR 50, Appendix J and the leak rate test frequency for CIVs is defined in Section 6.2.5 of the SSAR. The stated leak testing of CIVs meets the requirements of the 1989 Edition of Section XI. However, in 10 CFR 50.55a, the NRC takes an exception to the requirements of Section XI with regards to leak testing of CIVs. Specifically, in 10 CFR 50.55a(b)(2)(vii), the NRC requires that CIVs that do not provide a reactor coolant system pressure isolation function must be individually analyzed in accordance with paragraph 4.2.2.3(e) of OM Part 10 and corrective actions for these valves must be made in accordance with paragraph 4.2.2.3(f) of OM Part 10. The SSAR should be revised to reflect exception noted in 10 CFR 50.55a(b)(2)(vii). In Revision 12 to Section 3.9.6.2.2 of the SSAR, Westinghouse states that the provisions in 10 CFR 50.55a(b)(2)(vii) apply to the AP600 CIVs. This is acceptable.

In an early version of the SSAR, Westinghouse stated that the ASME Code does not specify exercise testing for valves that demonstrate operability during the course of plant operation. Therefore, exercise testing for those valves is not identified. However, the Code (OM Part 10, Sections 4.2.1.5 and 4.3.2.3) requires that the results and observations from normal operations required for IST are identified and recorded. Westinghouse was requested to revise the applicable SSAR section to comply with the Code. This was identified as DSER Open Item 3.9.6.2-9.

In response to DSER Open Item 3.9.6.2-9, Westinghouse revised Section 3.9.6.2.2 in Revision 4 to the SSAR and states that valves that operate during the course of normal plant operation at a frequency that satisfies the exercising requirements of the Code need not be additionally exercised, provided that the observations required for testing are made and recorded at intervals no greater than that specified in the SSAR. The SSAR revision is acceptable and DSER Open Item 3.9.6.2.9 is closed.

3.9.6.3 Relief Requests

In Section 3.9.6.3 of the SSAR, Westinghouse states that relief from the testing requirements of Section XI of the ASME Code will be requested when full compliance with requirements of Section XI of the Code is not practical. Westinghouse further states that in such cases, specific information will be provided which identifies the applicable code requirements, justification for the relief request, and the testing method to be used as an alternative.

As discussed in Section 3.9.6 of this report, the staff's position on the ASME Code-required quarterly testing is that to the extent practical, the passive ALWR piping systems should be designed to accommodate the applicable Code-required quarterly testing of valves. The ASME/ANSI OM Part 10 referenced in the 1989 Edition of Section XI of the ASME Code, or the 1990 ASME OM Code, provides for the relaxation in the valve testing frequency from quarterly intervals to cold shutdowns or refueling outages if testing during normal plant operations or cold shutdown conditions are not practical. The vendors for advanced passive reactors, for which the final designs are not complete, have sufficient time to include provisions in their piping system designs to allow for the ASME Code-required quarterly testing. However, design configuration changes to accommodate ASME Code-required quarterly testing should be done only if the benefits of the test outweigh the potential risk (i.e., the effect of a more complex design configuration on system reliability). It is noted that the testing of numerous valves in the AP600 IST program submitted in Revision 4 to the SSAR were deferred to cold shutdowns or refueling outages, without any basis or justification. For those cases that Westinghouse provided the bases for deferring valve testing to cold shutdowns or refueling outages, the bases did not contain sufficient information to support the deferrals. Westinghouse was requested to provide additional information concerning the bases for deferring testing to cold shutdown or refueling outages. This was identified as a DSER Open Item 3.9.6.3-1. In Revision 11 to the SSAR, Westinghouse provided acceptable justification for deferring testing to cold shutdowns or refueling outages for those affected valves. Therefore, DSER Open Item 3.9.6.3-1 is closed.

In a letter dated February 14, 1997, Westinghouse requested an alternative to the Section XI test frequency requirements for the ADS Stage 1, 2, and 3 valves (Table 3.9-16, Note 3). Westinghouse requested the alternative on the basis that it provides an acceptable level of safety, and stated that "Exercise testing of these valves represents a risk of loss of reactor coolant and depressurization of the RCS if the proper sequence is not followed. For this reason, the frequency of this valve exercise testing should be minimized. Conversely, the PRA assumes that valve reliability for these valves is a function of test frequency. The recommended test frequency for the stage 1 through 3 ADS valve is every six months. The PRA results show that the AP600 meets its safety goals."

The staff agrees that exercise testing of these valves during power operation poses a risk of loss of reactor coolant and depressurization of the RCS if the proper sequence is not followed. The staff has accepted this justification for deferring certain valve exercise tests to cold shutdown for other plants. The staff noted that Westinghouse's request to change the test frequency from three months to six months would not eliminate the potential risk of testing at power and yet the request would require specific relief from the ASME Code requirements. The staff cannot justify the relaxation of the quarterly testing requirements without additional information on the reliability of ADS valves or PRA results that show that AP600 would be safer with 6-month test frequency. In response to the staff's comment, Westinghouse withdrew the relief request, and provided cold shutdown justification for these ADS valves in Table 3.9-16 of the SSAR. This is acceptable.

3.9.6.4 Review of Table 3.9-16 of the SSAR, Valve IST Program

In response to RAI 210.24, Westinghouse submitted an IST program for safety-related valves. Westinghouse subsequently revised the IST program in response to a meeting held on March 14 and 15, 1995, and submitted Revision 4 of the SSAR, which contains the IST program in Table 3.9-16. The staff, with the assistance of Brookhaven National Laboratory (BNL), reviewed this IST program. The purpose of this review is to ensure that Westinghouse's commitments regarding the ability to test the safety-related valves can be met. During its review, BNL identified a number of unresolved issues regarding test deferrals to cold shutdown or refueling outage. This was identified as DSER Open Item 3.9.6.4-1. In response to this open item, Westinghouse revised Table 3.9-16, through Revision 22 of SSAR, and provided acceptable justification for test deferrals to cold shutdown or refueling outages. The staff finds this acceptable, and therefore, DSER Open Item 3.9.6.4-1 is closed. Details of the evaluation of the AP600 IST program are provided in Appendix 3A of this report.

In Section 3.9.6 of the SSAR, Westinghouse states that the AP600 IST program does not include the testing of non-safety-related pumps and valves. As discussed in SECY-94-084, the staff might not require important non-safety-related components to meet criteria similar to safety-grade criteria. However, the important non-safety-related pumps and valves as identified by the RTNSS process should be designed to accommodate testing in accordance with the requirements of Section XI of the ASME Code. Specific positions on the IST requirements for those components will be determined as a part of the staff's review of Westinghouse's implementation of the RTNSS process. This was identified as DSER Open Item 3.9.6.4-2. In response to DSER Open Item 3.9.6.4-2, Westinghouse states in Revision 11 to Section 3.9.6 of the SSAR that pumps and valves identified as having RTNSS important missions have provisions to allow testing. The staff finds this acceptable, and therefore, DSER Open Item 3.9.6.4-2 is closed.

In addition, it should be noted that the development of a complete plant-specific IST program is outside the scope of design certification and shall be the responsibility of the COL applicant. The comprehensive plant-specific IST program will include the following items:

- the tests performed on each component and the Code requirement met by each test
- test parameters and frequency of the tests
- the normal, safety, and fail-safe position on each valve
- component type for each component
- P&ID coordinates for each component

In addition, any requests for relief shall be submitted by the COL applicant and will be reviewed by the NRC staff, on the basis of the ASME Code edition referenced in 10 CFR 50.55a(b), the AP600 design, and the IST methods available at the time of the COL application. It should be noted that in 10 CFR 50.55a(f)(4)(i), the NRC requires that IST programs for the initial 120-month interval must comply with the requirements in the latest edition and addenda of the ASME SSAR Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license, or for AP600, the COL. In a revision to Section 3.9.6 of the SSAR dated June 30, 1994, Westinghouse stated that the IST program, which identifies requirements for functional testing, will be submitted to the NRC by the COL applicant. This was identified as DSER Confirmatory Item 3.9.6.4-1.

In Revision 4 to Sections 3.9.6 and 3.9.8 of the SSAR, Westinghouse stated that the COL applicant will develop a detailed plant-specific IST program in accordance with the requirements outlined in SSAR Section 3.9.6, and Table 3.9-16. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.9.6.4-1 is closed.

3.9.6.5 Outstanding AP600 Test Program Issues

<u>Passive Core Cooling System Check Valves Testing Program</u>

The AP600 PXS utilizes check valves which must operate at low differential pressure during gravity driven injection. The staff had a concern over the performance of check valves that have been held shut for an extended period of time with a high differential pressure and high temperature (as could be the case for the PXS check valves). To address the staff's concern, Westinghouse conducted tests at a domestic nuclear power plant. The tests were conducted to investigate the differential pressure required to open a reactor coolant pressure boundary check valve after a full cycle of operation. The valves tested were 15.2-cm (6-in.) swing check valves typical of those which would be utilized in the AP600 PXS. However, because of an erroneous test methodology used during the tests, the test data could not provide a meaningful evaluation of valve performance under the test conditions. Westinghouse committed to supply additional check valve qualification test program that would demonstrate reliable operation of these valves under these conditions. In a revision to Section 3.9.6 of the SSAR dated June 30, 1994, Westinghouse committed to perform periodic testing under low differential pressure for safety-related check valves that have a safety function to open under low differential pressure. This testing is performed in addition to the forward and reverse flow check valve IST. The staff found this additional testing to be acceptable. However, the staff still needs to receive and review a qualification test or analysis program related to these safety-related check valves. This was identified as DSER Open Item 3.9.6.5-1. In response to DSER Open Item 3.9.6.5-1, Westinghouse indicated in a letter dated May 13, 1996 that PXS design was revised. The current design includes check valves in series with the squib valves which eliminate the high closing differential pressure. Since these valves are normal simple check valves and are no longer exposed to high closing differential pressure, there is no additional qualification test requirements. Therefore, DSER Open Item 3.9.6.5-1 is closed.

Automatic Depressurization Valve Testing Program

In a letter dated August 29, 1994, Westinghouse was requested to supply the following information:

- A narrative of the valve testing "roadmap" which was discussed at the April 7, 1994, meeting (RAI 952.96) (refer to Section 6 of Westinghouse's AP600 Design Change Description Report, dated February 15, 1994)

- ITAAC consistent with the "roadmap" that details the testing to be performed by the COL applicant and/or valve vendors in order to qualify the valves that are to be installed in the AP600 (RAI 952.97)

This was identified as DSER Open Item 3.9.6.5-2, because Westinghouse had not provided the requested information at the time the DSER was prepared. In a letter dated May 13, 1996, Westinghouse provided responses to RAI 952.96 and RAI 952.97. In response to RAI 952.96, Westinghouse provides a detailed testing road map which covers valve type selection, qualification testing, production testing, pre-operational testing and inservice testing. In response to RAI 952.97, Westinghouse states that appropriate requirements will be included in the ITAAC for ADS valves. The staff finds the responses and commitments acceptable, and therefore, DSER Open Item 3.9.6.5-2 is closed.

3.9.6.6 Conclusion

The staff concludes that the AP600 design and methods for RTNSS pumps and safety-related valve testing are acceptable and meet the requirements of GDC 37, 40, 46, 54 and 10 CFR 50.55a(f). This conclusion is on the basis of the staffs' finding that the safety-related pumps and valves in the AP600 plant have been adequately designed and provided access to enable performance of testing for assessing operational readiness of pumps and valves through the life of the plant. The staff further concludes that the AP600 plant design meets the Commission-approved staff positions of SECY-90-016, SECY-94-084, and SECY-95-132 (with revisions) for inservice testing of pumps and valves and are, therefore, acceptable.

3.9.7 Integrated Head Package

The integrated head package (IHP) is described in Section 3.9.7 of the SSAR. The IHP combines several components in one assembly to simplify refueling the reactor. This assembly includes a lifting rig, seismic restraints for control rod drive mechanisms, support for reactor head vent piping and valves, messenger tray and cable support structure, in-core instrumentation support structure, and shroud assembly. In Figure 3.9-7 in the SSAR, Westinghouse provides an illustration of the IHP. The following discussion is only concerned with the structural integrity and deflection limits of the seismic restraints and shroud assembly.

The CRDMs seismic restraint structure interfaces with the shroud assembly to transfer seismic loads from the mechanisms to the reactor vessel head. The mechanism seismic restraint structure and the shroud are both classified as seismic Category 1. In the response to RAI 210.72 dated June 27, 1994, Westinghouse agreed to revise Sections 3.9.7.1 and 3.9.7.3, and Table 3.2-3, Sheet 38 of the SSAR to state that both of these items are reclassified as AP600 Class C (ASME Class 3) and constructed in accordance with the rules in Subsection NF of Section III of the ASME Code. This is acceptable pending receipt of the SSAR revision. This was identified as DSER Confirmatory Item 3.9.7-1. In Revision 4 to the SSAR, Westinghouse revised Sections 3.9.7.1, 3.9.7.3, and 3.9.8 as proposed in the response to RAI 210.72. In Revision 11 to the SSAR, Westinghouse revised Table 3.2-3 to indicate that the seismic restraint structure and the shroud are reclassified as Class C and constructed in accordance with ASME Section III, Subsection NF. Therefore, DSER Confirmatory Item 3.9.7-1 is closed.

In Section 3.9.7.3 of the SSAR, Westinghouse states that, under design-basis loads, the deflection at the top of the control rod drive mechanism rod travel housing is limited to ensure that the rod travel housing does not bend to the extent that the drive rod binds during insertion of the control rods. The staff requested that, in the SSAR, Westinghouse provide a description of the analysis and/or test data that was used to establish this deflection limit. This should include a description of the design-basis loads. This was identified as DSER Open Item 3.9.7-1. Subsequently, Westinghouse submitted a letter dated October 14, 1996, which indicated that the deflection limits for the integrated head package are based on limiting deflections of the CRDM housing to ensure control rod insertion following a seismic event or a pipe break. Westinghouse further indicated that the loads from postulated branch line break in reactor coolant loop as result from leak-before-break (LBB) application are inconsequential to functions of the CRDM and the upper head package due to more limiting LOCA loads being postulated in the CRDM design. The staff's evaluations of the operational adequacy of the CRDM under LOCA and SSE loads is discussed in Section 3.9.2.4 of this report. On the basis of these evaluations, DSER Open Item 3.9.7-1 is closed.

On the basis of the above evaluations and resolution of the open and confirmatory items discussed in this section, the staff's conclusions relative to the design of the AP600 upper head package are as follows.

In accordance with Table 3.2-3 of the SSAR, the upper head package structures are safety-related, designed as Safety Class 3 components, and are designed to the QA requirements of 10 CFR Part 50, Appendix B. In addition, as discussed in this report, the SSAR contains acceptable criteria for the design of upper head package under normal, upset, emergency, and faulted loading conditions.

On the basis of these evaluations it has been determined the upper head package structures design conform to the following requirements:

- as Safety Class 3
- Quality assurance requirements of 10 CFR Part 50, Appendix B
- Rules of Section III of the ASME Code

On the basis of above evaluations, the staff concludes that implementation of the criteria discussed above to the design of the structures of the reactor upper head package provides reasonable assurance that, in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures will not exceed allowable stresses and deformations under such loading combinations. The AP600 design meets GDC 1 and 10 CFR 50.55a with regard to designing the upper head package structures to quality standards commensurate with the importance of the safety functions to be performed. These criteria provide an acceptable design basis for ensuring that these UHP structures and components will withstand the most adverse loading events that were postulated to occur during their service lifetime without loss of structural integrity or impairment of function.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

In Sections 3.9.2.2 and 3.10 of the SSAR, Westinghouse provides information on the seismic and dynamic qualification of safety-related mechanical and electrical equipment.
Section 3.9.3.2 of the SSAR also contains the following information related to valve operability assurance (no active pumps in the AP600):

- rationale used to determine if tests, analyses, or combinations of both will be performed
- criteria used to define the seismic and other relevant dynamic load input motions
- the proposed performance criteria demonstrating the adequacy of the qualification program

The acceptance criteria for the staff's review are based on meeting the following requirements:

- GDC 1 and 30 as related to qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed
- GDC 2 and Appendix A to 10 CFR Part 100 as related to qualifying equipment to withstand the effects of natural phenomena such as earthquakes
- GDC 4 as related to qualifying equipment being capable of withstanding the dynamic effects associated with external missiles and internally-generated missiles, pipe whip, and jet impingement forces
- GDC 14 as related to qualifying equipment associated with the reactor coolant boundary so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure and of gross rupture
- Appendix B to 10 CFR Part 50 as related to qualifying equipment using the quality assurance criteria provided

Westinghouse will use the seismic qualification methodology described in Section 3.10 of the SSAR for both mechanical and electrical equipment. This program conforms to IEEE 323 and RG 1.89. The program also meets the criteria in IEEE 344 as modified by RG 1.100, Revision 2 and, furthermore, any proposed change to these qualification standards will require NRC approval prior to implementation of the change. Westinghouse committed to comply with the guidelines of Section 3.10 of the SRP with exceptions identified in WCAP-13054. The staff reviewed those exceptions and generated a number of RAIs concerning certain exceptions, which were forwarded to Westinghouse in the NRC letter dated April 29, 1994. Westinghouse responded to the staff RAIs in letters dated June 27, July 22, July 25, and August 3, 1994. Provided below are the staff's evaluations of Westinghouse's responses to the staff's RAIs pertaining to equipment qualification.

- In the response to RAI 210.7, Westinghouse proposed to revise Sections 3D.4.1.2, E.4.4, E.5.1, and E.5.2.4 in Appendix 3D of the SSAR to agree with the staff positions related to seismic qualification of equipment. These proposed changes provided criteria to be implemented when the OBE was eliminated as a design-basis. These criteria are consistent with those in SECY-93-087 (Ref. Sections 3.1.1, and 3.12.5.14 of this report) and are acceptable. This was identified as DSER Confirmatory Item 3.10-1. In Revision 5 to the SSAR, Westinghouse revised all of the above SSAR sections as proposed in the response to the staff's request. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.10-1 is closed.

- In RAI 210.81, the staff requested that an exception to Section 3.10 of the SRP in WCAP-13054 be deleted. This exception states that safety-related equipment may be qualified, in part, on the basis of properly documented experience data in accordance with Section 9.0 of IEEE 344-1987. As used in IEEE 344, experience data includes both seismic experience and previous qualifications. The staff has not accepted the use of seismic experience data on either evolutionary or passive plants. In accordance with RG 1.100, Revision 2, this method of qualification will be reviewed by the staff on a case-by-case basis.

  The response to RAI 210.81 dated August 3, 1994, Westinghouse proposed to revise Section 3.10.2 of the SSAR to state that where seismic experience data is used, all aspects of the methodology, qualification basis, and supporting data will be properly documented by the COL applicant. Identification of the specific equipment qualified on the basis of experience data, the details of the methodology, and the corresponding experience data for each piece of equipment will be included in the equipment qualification file. The staff did not consider this response to be completely acceptable for design certification of the AP600. The proposed revision to the SSAR, and a revision to WCAP-13054, should state that the COL applicant will submit all of this information to the staff for review and approval before including it in the equipment qualification file. This was identified as DSER Open Item 3.10-1. In Revision 10 to SSAR Section 3.10.6, and Revision 2 to WCAP-13054, Westinghouse committed that the COL applicant, as a part of the COL application, will identify equipment qualified on the basis of experience and include details of the methodology and the corresponding experience data for each piece of equipment. This is consistent with RG 1.100, Revision 2 which, as stated above, conditions its endorsement of IEEE 344-1987 by stating that the use of experience data for qualification of equipment will be evaluated by the NRC staff on a case-by-case basis. The staff finds this acceptable, and therefore, DSER Open Item 3.10-1 is closed. This is COL Action Item 3.10-1. Furthermore, any proposed change to this approach for qualification by experience will require NRC approval prior to implementation of the change.

- The response to RAI 210.82 dated June 27, 1994, Westinghouse agreed to revise WCAP-13054 to delete an exception to Section 1.a.(1) of Section 3.10 of the SRP. The exception stated that for electrical equipment, the only dynamic loads considered in testing are seismic loads, and that these seismic loads are not combined either by test or analysis with other dynamic loads. This was identified as DSER Confirmatory Item 3.10-2. Additionally, Westinghouse pointed out that in Section 3.10.2 of the SSAR the effect of dynamic loads, in addition to seismic loads, are required to be considered in the qualification of electrical equipment, where applicable. The staff concludes that the information in Section 3.10.2 of the SSAR is consistent with Section 3.10 of the SRP, and is therefore, acceptable. In Revision 2 to WCAP-13054, Westinghouse deleted this exception. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.10-2 is closed.

- The response to RAI 210.83 dated June 27, 1994, Westinghouse agreed to revise WCAP-13054 to delete an exception to Section 1.a.(2) of Section 3.10 of the SRP. This exception states that when performing seismic qualification of mechanical and electrical equipment by test, all accident loads are not superimposed on the seismic loads. The response points out that in Section 3.10.2 of the SSAR, Westinghouse requires that the effects of dynamic loads, in addition to seismic loads, be considered in qualification of electrical equipment, where applicable. In addition, in Section 3.10.2.2 of the SSAR, Westinghouse states that active mechanical equipment is qualified by a combination of test and analysis which addresses non-seismic loads, if applicable. The staff concludes that the information in the Sections 3.10.2 and 3.10.2.2 of the SSAR is consistent with Section 3.10 of the SRP, and is therefore, acceptable. In addition, Westinghouse agreed to delete the WCAP-13054 exception discussed above. This was identified as DSER Confirmatory Item 3.10-3. In Revision 2 to WCAP-13054, Westinghouse deleted this exception. The staff finds this acceptable, and therefore, DSER Confirmatory Item 3.10-3 is closed.

- The response to RAI 210.85 dated July 22, 1994, Westinghouse agreed to revise WCAP-13054 to delete an exception to Section II.1.a(14)(b) of Section 3.10 of the SRP, but did not agree to revise the SSAR. This exception states that valve disks are not analyzed for pressure differential or impact energy resulting from a postulated pipe break, except for certain cases where a significant impact from a LOCA is expected. The staff's position is that, in addition to revising WCAP-13054, either Section 3.9.3 or 3.10 of the SSAR should be revised to briefly describe the methodology used in the AP600 design to analyze the feedwater line valve disks when they are subjected to dynamic loads because of a LOCA. This was identified as DSER Open Item 3.10-2. In Revision 12 to SSAR Section 3.10.2.2, Westinghouse responded to this issue by stating that feedwater line valve disks are evaluated, using appropriate ASME Code, Section III limits, for the effect of dynamic loads resulting from accident conditions by considering the effect of an equivalent differential pressure. The equivalent differential pressure is developed from a transient analysis based on wave mechanics that includes consideration of system arrangement and valve closing dynamics. Valve operating conditions are included as part of the valve design specification and are used to evaluate the valve disk. The staff concludes that this description provided reasonable assurance that these disks will be designed to assure their structural and functional integrity, and is acceptable. Therefore, DSER Open Item 3.10-2 is closed.

- In RAI 210.86 and RAI 210.88, the staff indicated that the exceptions to Sections 3 and 5c of Section 3.10 of the SRP concerning equipment qualification files and seismic qualification reports are not acceptable. In response to these RAIs, Westinghouse agreed to revise Section 3.10.4 and Table 1.8-1 of the SSAR to include a commitment that the COL applicant will verify that the equipment qualification file is maintained during the equipment selection and procurement phase. On the basis of the remainder of Section 3.10.4 of the SSAR, it is the staff's understanding that this equipment qualification file contains the results of tests and analyses verifying that the criteria in Section 3.10.1 are satisfied by employing methods described in Sections 3.10.2 and 3.10.3, and Appendix 3D of the SSAR, and that this file will be available to the staff for review and audit.

  The staff concludes that the above commitments satisfy applicable portions of Section 3.10 of the SRP, and are therefore, acceptable. This was identified as DSER Confirmatory Item 3.10-4. Revision 5 to SSAR Sections 3.10.4 and 3.10.6 added commitments that the COL applicant is responsible for maintaining the equipment qualification file during the equipment selection and procurement stage. As stated above, this is consistent with Section 3.10 of the SRP, and is acceptable. Therefore, DSER Confirmatory Item 3.10-4 is closed.

- In RAI 210.87, the staff indicated that the exception to Section 4 of Section 3.10 of the SRP relative to the qualification program to demonstrate that valves that are a part of the reactor coolant pressure boundary will experience minimum leakage, is not acceptable. In the response dated July 22, 1994, Westinghouse agreed to delete this exception and proposed to add a comment in the WCAP to state that the qualification program shall include testing or analysis that demonstrate that these valves will not experience leakage beyond the design criteria when subjected to design-basis loading conditions. However, the response does not propose to revise the SSAR. It is the staff's position that the SSAR should be revised to provide this same commitment in either Section 3.9.3 or 3.10.2.2 of the SSAR. This was identified as DSER Open Item 3.10-3. In Revision 5 to Section 3.10.2.2 of the SSAR, Westinghouse provided this commitment. Westinghouse also, in Revision 2 to WCAP-13054, deleted the original exception to Section 4 of Section 3.10 of the SRP and provided the same commitment as that in the revised SSAR Section 3.10.2.2. These commitments are consistent with Section 3.10 of the SRP, and are acceptable. Therefore, DSER Open Item 3.10-3 is closed.

- In RAI 210.93, the staff indicated that the WCAP-13054 exception to Section 1.c of Section 3.10 of the SRP relative to IEEE-323-1983, should be deleted and replaced by IEEE-323-1974. Westinghouse initially responded that the exception would not be deleted. IEEE-323-1983 has not been endorsed by the staff. The staff's position on this issue is discussed in Section 3.11.3.2 of this report. This was identified as DSER Open Item 3.10-4. In Revision 5, Westinghouse revised Appendix 3D of the SSAR to commit to the staff's position to use IEEE-323-1974 rather than the 1983 edition, and in Revision 2 to WCAP-13054, Westinghouse changed the "exception" to SRP 3.10, Section 1.c to "acceptable." Additionally, in Revision 8, Westinghouse revised Section 3.11 of the SSAR to commit to the 1974 edition. Therefore, Open Item 3.10-4 is closed.

On the basis of its initial review of Sections 3.9.2.2, 3.9.3.2 and 3.10, and Attachment E of Appendix 3D of the SSAR, and WCAP-13054, the DSER identified several guidelines from Section 3.10 of the SRP that should be included in the SSAR as being applicable to the AP600 design. Westinghouse was requested to provide commitments to these guidelines if they were not already included in the SSAR. This was identified as DSER Open Item 3.10-5. In Revision 5 to the SSAR, Westinghouse revised Appendix D, Attachment E, Section E.5 to provide a response to this open item. These revisions complement information in other sections of the SSAR relative to operability and seismic qualification of electrical and mechanical equipment. The staff concludes that the information in SSAR Sections 3.9.2.2, 3.9.3.2, 3D.4.1.2, 3D.6.2, E.3.2, E.4.3, and E.5 collectively provides commitments to the guidelines of Section 3.10 of the SRP and is acceptable. Therefore, DSER Open Item 3.10-5 is closed.

In Section 3.10 of the SSAR, Westinghouse provides qualification methodology only and contains no plant-specific information. Therefore, each COL applicant using this methodology must ensure that specific environmental parameters along with seismic and dynamic input response spectra are properly defined and enveloped in the methodology for its specific plant and implemented in its equipment qualification program. As indicated in the responses to RAI 210.86 and RAI 210.88 (see DSER Confirmatory Item 3.10-4), Westinghouse committed that the COL applicant shall maintain equipment qualification records in a permanent file which shall be readily available for staff audit. The staff may audit these files to review the results of tests and analyses that were performed for the following reason:

- Ensure that the criteria in the SSAR were properly implemented

- Ensure that adequate qualification was demonstrated for all equipment and their supports

- Verify that all applicable loads were properly defined and accounted for in the testing and analyses performed

<u>Conclusion</u>

On the basis of the above evaluations, the staff concludes that Westinghouse has defined appropriate seismic and dynamic qualification programs for mechanical and electrical equipment that meet the guidelines in SRP 3.10. These programs also meet applicable portions of GDC 1, 2, 4, 14, and 30, Appendix B to 10 CFR Part 50, and Appendix A to 10 CFR Part 100, and are therefore, acceptable. This conclusion is on basis the following information.

In Table 3.2-3 of the SSAR, Westinghouse identifies all AP600 safety-related mechanical and electrical equipment as follows:

- safety Class 1, 2, or 3
- seismic Category I
- designed to the quality assurance requirements of 10 CFR Part 50, Appendix B

As discussed in Sections 3.2.1 and 3.2.2 of this report, the staff concludes that Table 3.2-3 of the SSAR is acceptable. On the basis of these evaluations, the staff concludes that the criteria and commitments in the AP600 SSAR meet GDC 1, 30, and 10 CFR Part 50, Appendix B, as they relate to qualifying safety-related mechanical and electrical equipment to appropriate quality standards commensurate with the importance of the safety function to be performed.

The qualification program, which will be implemented for mechanical, instrumentation, and electrical equipment, meets the requirements and recommendations of IEEE 344-1987 and the regulatory positions of RGs 1.61, 1.89, 1.92, 1.100, and Section 3.9.3 of the SRP. This provides adequate assurance that such equipment will function properly under all imposed design and service loads, including the loadings imposed by the SSE, postulated accidents, and LOCAs. On the basis of this program, complemented by the staff evaluations of the following:

- seismic classifications in Section 3.2.1 of this report
- protection from external missiles and internally-generated missiles in Section 3.5 of this report
- analyses to withstand dynamic effects of postulated pipe breaks in Section 3.6.2 of this report
- loading combinations and stress limits in Section 3.9.3.1 of this report

The criteria and commitments in the AP600 SSAR meet GDC 2, Appendix A to 10 CFR Part 100, GDC 4 and GDC 14, as they relate to qualifying equipment to:

- withstand the effects of natural phenomena such as earthquakes
- be capable of withstanding the dynamic effects associated with external missiles, internally-generated missiles, and pipe whip and jet impingement forces
- demonstrate that equipment associated with the reactor coolant pressure boundary has a low probability of abnormal leakage, rapidly propagating failure, or gross failure

3.11 Environmental Qualification of Mechanical and Electrical Equipment

3.11.1 Introduction

Equipment that is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in GDC 1 and GDC 4 and Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50, is applicable to equipment located inside and outside the containment. More detailed requirements and guidance related to the methods and procedures for demonstrating this capability for electrical equipment are in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which supplements IEEE 323, and various RGs and industry standards, and RG 1.89, Revision 1.

3.11.2 Background

The staff issued NUREG-0588 in December 1979, to promote a more orderly and systematic implementation of equipment qualification programs by industry and to guide the staff in its use in ongoing licensing reviews. The positions in NUREG-0588 provide guidance on the following items:

- how to establish Environmental Qualification (EQ) service conditions
- how to select methods that are considered appropriate for qualifying equipment in different areas of the plant
- other areas such as margin, aging, and documentation

A final rule on EQ for electrical equipment important to safety for nuclear power plants became effective on January 21, 1983. This rule, 10 CFR 50.49, specifies the requirements for demonstrating the EQ of electrical equipment important to safety that is located in harsh environments. Each item of electric equipment important to safety must be qualified by one of the following methods:

- testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable
- testing a similar item of equipment with a supporting analyses to show that the equipment to be qualified is acceptable
- experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable
- analysis in combination with partial type test data that supports the analytical assumptions and conclusions

In Revision 1 of RG 1.89, the staff specifies guidelines for complying with the rule. The applicant or licensee shall prepare a list of electrical equipment important to safety covered by the qualification requirements. In addition, the applicant or licensee shall include the following information for electric equipment important to safety in a qualification file:

(1) the performance specifications under conditions existing during and following design-basis accidents

(2) the voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with (1) above can be ensured

(3) the environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with (1) and (2) above.

The COL applicant shall keep the list and information in the file current and retain the file in auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment important to safety meets the requirements. In conformance with 10 CFR 50.49, electrical equipment for PWRs referencing the AP600 design must be qualified according to the criteria in Category I of NUREG-0588 and Revision 1 of RG 1.89.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B to 10 CFR Part 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

To document the degree to which the EQ program for the AP600 design complies with the EQ requirements and criteria, Westinghouse submitted Section 3.11 and Appendix 3D of the SSAR, and responded on November 30, 1992, (ET-NRC- 92-3777) to an NRC staff RAI dated September 23, 1992, and on June 27, 1994, (NTD-NRC-94-4181) and July 15, 1994, (NTD-NRC-94-4202) to an NRC staff RAI dated May 19,1994.

3.11.3 Staff Evaluation

The staff limited its evaluation of the EQ program for the AP600 design to a review of Westinghouse submittals on its approach for selecting and identifying equipment required to be environmentally qualified for the AP600 design, qualification methods proposed, and completeness of information in Appendix 3D of the SSAR. The bases for the staff's evaluation are contained in Revision 2 of Section 3.11 of the SRP; NUREG-0588, Category 1; Revision 1 of RG 1.89; and 10 CFR 50.49. For COL applicants referencing the AP600 certified design, the staff will review specific details of the EQ programs for their plants using the evaluation bases mentioned above.

3.11.3.1 Completeness of Qualification of Electrical Equipment Important to Safety

The following three categories of electrical equipment important to safety must be qualified in accordance with the provisions 10 CFR 50.49(b)(1), (b)(2), and (b)(3):

- (b)(1) – safety-related electrical equipment (relied on to remain functional during and after design-basis events)

- (b)(2) – non-safety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory performance of the safety functions by the safety-related equipment

- (b)(3) – certain postaccident monitoring equipment (Categories 1 and 2 postaccident monitoring equipment as specified in RG 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident")

In Table 3.11-1 of the SSAR, Westinghouse provides a list of safety-related electrical and active mechanical equipment that is essential to emergency reactor shutdown, containment isolation, reactor core cooling, or containment and reactor heat removal or that is otherwise essential in preventing a significant release of radioactive material to the environment. The NRC staff reviewed this list and concluded that additional discussions with Westinghouse were necessary before a final conclusion could be reached. This was identified as DSER Open Item 3.11.3.1-1.

The staff completed its review of additional information provided by Westinghouse in the form of meetings and discussions with the staff and updates to the SSAR. As a result of reviewing the additional information, the staff concludes that the list is acceptable. Therefore, Open Item 3.11.3.1-1 is closed.

The radiation qualifications for individual safety-related components should be developed on the basis of the following two conditions:

- the radiation environment expected at the component location from equipment installation to the end of qualified life, including the time the equipment is required to remain functional after the accident

- the limiting design-basis accident for which the component provides a safety function

These design-basis accident conditions are discussed in Chapter 15 of this report.

For the LOCA source term, the AP600 design adopted the accident source term presented in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants - Final Report." The staff finds this acceptable.

3.11.3.2 Qualification Methods

<u>Electrical Equipment in a Harsh Environment</u>

Detailed procedures for qualifying safety-related electrical equipment located in a harsh environment are defined in NUREG-0588 and RG 1.89. The criteria in these documents are also applicable to other equipment important to safety defined in 10 CFR 50.49.

In the DSER, the staff determined that the methodology used by Westinghouse for the AP600 relied primarily on IEEE Standard 323-1983. To date, the NRC staff has not endorsed IEEE 323-1983; therefore, references to this standard in its entirety or in part are not acceptable. As indicated in the footnote to 10 CFR 50.49, and stated in NUREG-0588 and RG 1.89, the guidance in IEEE Standard 323-1974 is acceptable to the NRC staff for qualifying equipment within the scope of 10 CFR 50.49. On the basis of Westinghouse's response to the staff RAIs on this issue, further discussions between the staff and Westinghouse were necessary for the resolution of this issue. This was identified as DSER Open Item 3.11.3.2-1.

Westinghouse changed its reliance on IEEE 323-1983 and is now referencing IEEE 323-1974 to demonstrate compliance with the requirements of 10 CFR 50.49. The staff finds this acceptable, and therefore, DSER Open Item 3.11.3.2-1 is closed.

In addition, for current-generation operating reactors, the staff's definition of what constitutes a mild radiation environment for electronic components such as semi-conductors, or any electronic component containing organic materials, is different from what it is for other equipment. The staff position is that a mild radiation environment for electronic equipment is a total integrated dose of less than 10 Gy (1E3 Rad). For other equipment it is less than 1E2 Gy (1E4 Rad). With the expected significant increase in the quantity and variety of electronic components in newer generation plants, the staff has increasing concerns about the efforts being made and the ability of these components to be environmentally qualified. This issue was identified as DSER Open Item 3.11.3.2-2.

In Section 3D.4.2 of Appendix 3D of the SSAR, Westinghouse provides a discussion on "Mild versus Harsh Environments." In this discussion, Westinghouse states that "A radiation-harsh environment is defined for equipment designed to operate above certain radiation thresholds where other environmental parameters remain bounded by normal or abnormal conditions. Any equipment that is above 1E4 rads gamma (1E3 for electronics) will be evaluated to determine if a sequential test which includes aging, radiation, and the applicable seismic event is required or if sufficient documentation exists to preclude such a test." The staff determined that this position is consistent with the staff's position, and finds it acceptable. Therefore, DSER Open Item 3.11.3.2-2 is closed.

Safety-Related Mechanical Equipment in a Harsh Environment

Although no detailed requirements exist for mechanical equipment, GDC 1 and 4 and Appendix B to 10 CFR Part 50 (Criteria III, "Design Control," and XVII, "Quality Assurance Records") contain the following requirements related to equipment qualification:

- Components should be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.
- Measures should be established for the selection and review for the suitability of application of materials, parts, and equipment that are essential to safety-related functions.
- Design control measures should be established for verifying the adequacy of design.
- Equipment qualification records should be maintained and should include the results of tests and materials analyses.

For mechanical equipment, the staff concentrates its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). A review and evaluation should be performed to:

- Identify safety-related mechanical equipment located in harsh environment areas, including required operating time.
- Identify non-metallic subcomponents of this equipment.
- Identify the environmental conditions for which this equipment must be qualified. (The environments defined in the electrical equipment program are also applicable to mechanical equipment.)
- Identify non-metallic material capabilities.

- Evaluate environmental effects.

Table 3.11-1 of the SSAR includes both electrical and mechanical equipment without a clear distinction between the two classes of equipment. To eliminate potential confusion in the EQ program, Westinghouse was asked to clearly identify which items of equipment are classified as electrical and separate those items from those that are classified as mechanical equipment. This was identified as DSER Open Item 3.11.3.2-3.

Westinghouse updated the SSAR and made the requested distinction between the classes of equipment. The staff finds this acceptable, and therefore, DSER Open Item 3.11.3.2-3 is closed.

3.11.3.3 Conclusions

On the basis of its review of the SSAR, other Westinghouse submittals, and NRC staff policies and practices, the staff concludes that the program proposed by Westinghouse for environmentally qualifying electrical equipment important to safety and safety-related mechanical equipment for the AP600 design is in compliance with the requirements of 10 CFR 50.49 and other relevant requirements as stated in this section. The staff finds this acceptable.

3.12 Piping Design

3.12.1 Introduction

This section provides the staff's safety evaluation of Westinghouse's design of piping systems for the AP600 design certification, which comprise the seismic Category I, Category II and piping for RTNSS. The staff used the SRP guidelines to evaluate the piping design information in the SSAR and performed a detailed review of the piping design criteria, including sample calculations. The staff evaluated the adequacy of the structural integrity and functional capability of piping systems. The review was not limited only to the ASME Boiler and Pressure Vessel Code Class 1, 2, and 3 piping and supports, but also included buried piping, instrumentation lines, and the interaction of non-seismic Category I piping with seismic Category I piping. The staff's evaluation of the adequacy of the AP600 piping design analysis methods, design procedures, and acceptance criteria that are to be used for the final completion of the AP600 piping design is provided in the following sections of this report. The staff's evaluation includes the following information:

- applicable codes and standards
- analysis methods to be used for completing the piping design
- modeling techniques
- pipe stress analyses criteria
- pipe support design criteria

The staff's report is based, in part, on a review of AP600 piping and pipe support design criteria documents, and of preliminary piping calculations provided by Westinghouse for the main steam piping and the pressurizer surge line.

3.12.2 Codes and Standards

In GDC 1, the NRC requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. In 10 CFR 50.55a, the NRC requires that certain systems and components of boiling and pressurized water-cooled nuclear power reactors must meet certain requirements of the ASME Code. It specifies the use of latest edition and addenda endorsed by the NRC and any limitations. In RGs 1.84 and 1.85, the staff lists ASME Code cases that the NRC staff finds acceptable and any limitations that apply to them.

3.12.2.1 ASME Boiler and Pressure Vessel Code

For the AP600 design certification, Westinghouse established that Section III of the ASME Code will be used for the design of ASME Code Class 1, 2, and 3 pressure retaining components and their supports. The 1989 Edition and Addenda of the ASME Code are specified in Section 5.2.1.1 of the SSAR for the AP600 design. In Section 5.2.1.1 of the DSER, the staff stated that currently 10 CFR 50.55a(b)(1) only endorses ASME Section III Code through the 1989 Edition and has not yet endorsed the use of the 1989 Addenda. This was identified as DSER Open Item 5.2.1.1-1. The resolution of this open item and the process for changing ASME Code editions and addenda is discussed in Section 5.2.1.1 of this report.

3.12.2.2 ASME Code Cases

The only acceptable ASME Code cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the AP600 design are those either conditionally or unconditionally approved in RGs 1.84 and 1.85 in effect at the time of design certification as listed below. However, the COL applicant may submit with its COL application for staff review and approval future code cases that are endorsed in RGs 1.84 and 1.85 at the time of the COL application provided they do not alter the staff's safety findings on the AP600 certified design.

In the response to RAI 210.109 dated June 30, 1994, Westinghouse provided a listing of ASME Code cases to be used in the AP600 design and would be presented in a new Table 5.2-3 in the SSAR. The ASME Code Cases requested by Westinghouse that are applicable to the AP600 piping and support design are listed below.

- ASME Code Case N-71-15, "Additional Materials for Subsection NF, Classes 1, 2, 3, and MC Component Supports Fabricated by Welding, Section III, Division 1." This code case has been endorsed by the staff in RG 1.85.

- ASME Code Case N-122-1, "Stress Indices for Structure Attachments, Class 1, Section III, Division 1." This revision of this code case has not been endorsed by the staff.

- ASME Code Case N-249-11, "Additional Material for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated Without Welding, Section III, Division 1." This code case has not yet been endorsed by the staff in RG 1.85.

- ASME Code Case N-318-4, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1." This code case has been conditionally endorsed by the staff in RG 1.84.

- ASME Code Case N-319-2, "Alternate Procedure for Evaluation of Stress in Butt Weld Elbows in Class 1 Piping, Section III, Division 1." This revision of this code case has not been endorsed by the staff.

- ASME Code Case N-391-1, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1." This code case has been endorsed by the staff in RG 1.84.

- ASME Code Case N-392-2, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1." This revision of this code case has not been endorsed by the staff in RG 1.84.

As noted in the DSER, ASME Code Cases N-122-1, N-249-11, N-319-2 and N-392-2 have not yet been endorsed by the staff. (See the open item in Section 5.2.1 of this report).

In Revision 3 to the SSAR, Westinghouse included a revised Table 5.2-3 which contained other changes to the list. Code Case N-122-2 replaced N-122-1 and Code Case N-319-2 replaced N-319-1. Footnotes to the table indicate that use of Code Case N-249-11 will meet the conditions for Code Case N-249-10 in RG 1.85, and that the use of Code Case N-392-2 will meet the conditions for Code Case N-392-1 in RG 1.84. The staff's evaluation determined that the proposed code cases and their limitations contained in revised Table 5.2-3 are acceptable and DSER Open Item 5.2.1.2-1 is closed (see Section 5.2.1.2 of this report).

Although not listed in the proposed new Table 5-2.3, Code Case N-411, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1," had been referred to in sections of the SSAR addressing the definition of damping. This inconsistency reflected a change in the specification of damping being proposed by Westinghouse for the AP600 design. This issue is addressed further in Section 3.12.5.4 of this report.

In Revision 2 to the SSAR, Westinghouse revised Sections 3.7.1.3, 3.7.3.15 and Table 3.7.1-1 of the SSAR to include 5 percent damping for piping systems analyzed by the response spectrum method and 2 to 3 percent damping by the independent support motion spectral analysis method, in lieu of the damping values in ASME Code Case N-411. Thus, the damping values specified in Code Case N-411 will not be used. This is discussed in Sections 3.12.3.3 and 3.12.5.4 of this report.

3.12.2.3 Design Specifications

Section III of the ASME Code, requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loading combinations, design data, and other design data inputs. The Code also requires a design report for ASME Code, Class 1, 2, and 3 piping and components.

During its review of the SSAR, the staff noted that although it is understood that design reports will not be available at the time of certification, design specifications, or at a minimum, the procedure for preparing them, should be available because design specifications serve as the basis for construction. In the absence of preparing design specifications at the time of design certification, the staff requested Westinghouse to prepare a document that discussed the requirements and methodologies for preparing design specifications that should be followed by the COL applicant or any applicant referencing the AP600 design. During a design review meeting at Westinghouse on July 25 through 26, 1995, Westinghouse provided a sample AP600 design specification document to demonstrate the procedures that will be used to prepare design specifications for ASME Class 1, 2, and 3 components. The staff review of this document determined that, when completed, it will be an acceptable design specification as required by the ASME Code Section III. The staff also concluded that the use of the same procedures to prepare design specifications for other AP600 ASME Class 1, 2, and 3 components will result in design specifications that will be in conformance with the requirements of ASME Code, Section III, and are acceptable. (See Section 3.9.3.1 of this report).

3.12.2.4 Conclusions

The staff finds that in Sections 3.9.3, 5.2.1.1, and 5.2.1.2 of the SSAR, Westinghouse meets the requirements of and the commitments to the applicable codes and standards contained in 10 CFR 50.55a and GDC 1, as they pertain to the codes and standards specified for ASME Code Class 1, 2, and 3 piping, by ensuring that such piping is designed to quality standards commensurate with their importance to safety.

3.12.3 Analysis Methods

The staff reviewed the information in Section 3.9.1 of the SSAR related to the design transients and methods of analysis used for all seismic Category I piping and pipe supports designated as ASME Code Class 1, 2, and 3 under Section III of the ASME Code, as well as those not covered by the Code. The staff reviewed the assumptions and procedures used for the inclusion of transients in the design and fatigue evaluation of ASME Code Class 1 and core support components. The staff also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports, as well as the proposed inelastic analytical techniques.

In the DSER, the staff found that the descriptions provided in the SSAR of the analysis methods that may be used in piping design were not detailed enough to allow assessment by the staff or in some cases, even to characterize the methods. Westinghouse was requested to provide additional descriptions in the SSAR to include the range of applicability, the criteria for the selection of significant parameters, and any limitations in the use of the piping analysis methods. This was identified as DSER Open Item 3.12.3-1. (See Sections 3.12.3.2 through 3.12.3.4 of this report for further discussion.)

On June 2, 1995, Westinghouse submitted SSAR Draft Revision 4 which contained additional information describing the analysis methods that will be used in the design and analysis of piping systems. As discussed in Sections 3.12.3.2, 3.12.3.3, and 3.12.3.4 below, the staff found that the additional information was not adequate to fully resolve the concerns. Westinghouse subsequently provided additional SSAR Revisions 7 and 9 which resolved the staff's concerns described in Sections 3.12.3.2 and 3.12.3.3. The remaining concerns regarding time history analysis methodology discussed in Section 3.12.3.4 were resolved on the basis of information included in a proposed SSAR revision provided in a Westinghouse letter dated March 13, 1997. The proposed changes were subsequently incorporated in Revision 12 to Section 3.7.3.17 of the SSAR. Therefore, DSER Open Item 3.12.3-1 is closed as discussed in the following sections.

3.12.3.1 Experimental Stress Analysis

In Section 3.9.1.3 of the SSAR, Westinghouse does not state that experimental stress analysis methods will be used in piping design. If a COL applicant wishes to use this method in any AP600 piping design, the details of the method, as well as the scope and extent of its application, should be submitted to the staff for approval before its use. The staff's position is that experimental stress analysis methods shall be in compliance with Appendix II of ASME Section III, Division 1.

3.12.3.2 Modal Response Spectrum Method

Modal response spectrum analysis, time history analysis, equivalent static analysis and design by rule are stated in Section 3.7.3.1 of the SSAR to be analysis options for the seismic analysis of subsystems. The particular method used is at the discretion of the analyst and dependent on the specific item. Both the envelope and independent support motion response spectrum methods are specified in Section 3.7.3.9 of the SSAR as modal response spectrum analysis options.

With either response spectrum method, first a mathematical model is constructed to reflect the dynamic characteristics of the system in accordance with the procedures described in Section 3.7.3.3 of the SSAR. Next, the system's natural frequencies, mode shapes, and modal participation factors are calculated. The latter are then amplified by the appropriate spectral accelerations for each excitation and the modal responses associated with the amplified or low frequency modes are determined. These include the modal forces, shears, moments, stresses, and deflections.

In a separate calculation, the modal responses associated with the rigid or high-frequency modes is determined. This calculation is essentially a static analysis, and its methods are described in Section 3.7.3.7.1 of the SSAR. The results of the rigid modes calculation is expressed and treated as the response to a single additional mode.

As Westinghouse described in Section 3.7.3.7.1 of the SSAR, the modal response of the low-frequency modes is combined with the modal response of the single high-frequency mode by the square-root-of-the-sum-of-squares (SRSS) method, to provide the total modal response for one direction of excitation. The modal response calculations are performed for each of the three earthquake directions (two horizontal and the vertical). The total seismic response from the simultaneous application of the three-directional components of earthquake loading are obtained by combining the maximum codirectional responses of each of the three components by the SRSS method, as Westinghouse described in Section 3.7.3.7.1 of the SSAR.

For piping systems that are anchored and restrained to floors and walls of buildings that have differential movements during a seismic event, additional forces and moments are induced in the system. Additional static analyses are performed to determine these loads, as described in Section 3.7.3.9 of the SSAR. The maximum differential displacements are applied to the piping anchors and restraints. Three analyses are performed (two in the horizontal directions and one in the vertical direction). The resulting stresses are placed in the secondary stress category because they are displacement-induced and self-limiting. These secondary loads are combined with the primary (inertia) loads by the absolute sum method as discussed in Section 3.12.5.13 of this report.

In the DSER, the staff reported that in Section 3.7.3.9 of the SSAR, Westinghouse provided only a rudimentary description of the elements of the response spectrum procedure, briefly summarized above. Although the methodology is fundamental to seismic analysis, differences could exist in its specific application, particularly in the criteria for the selection of significant parameters, the definition or selection of spectral accelerations, and the definition of system boundaries. In the DSER, the staff stated that a more detailed description of all analysis methods, including the modal response spectrum method, must be included in the SSAR. (See DSER Open Item 3.12.3-1 in Section 3.12.3 above).

Westinghouse provided additional descriptions of analysis methods, including the modal response spectrum method, in SSAR Draft Revision 4, dated June 2, 1995. In Section 3.7.3.8, Westinghouse discusses the development of the mathematical model to reflect the dynamic characteristics of the system. In Section 3.7.3.9, Westinghouse provides additional information describing the analysis procedure. Enveloped response spectra are developed in three perpendicular directions to include the spectra at all floor elevations of the attachment points and the piping module or equipment if applicable. The response spectrum analysis calculates mode shapes and frequencies up to the cut-off frequency and modal participation factors in each direction. The spectral accelerations for each significant mode are determined from the enveloped spectra in each direction. On the basis of this information, the modal inertia response forces, moments, displacements and accelerations are calculated. For each direction, the modal responses are combined in accordance with one of the procedures described in Section 3.7.3.7.2 of the SSAR. The high-frequency mode responses are determined and combined with the low-frequency mode responses on the basis of one of the methods described in Section 3.7.3.7.1. The total seismic responses are combined by the SRSS method for all three earthquake directions. The response resulting from differential seismic anchor motions is calculated using static analysis as described in Section 3.7.3.9. The results of the seismic inertia analysis are combined with the results of the seismic anchor motion analysis by the absolute sum method. The staff evaluation of specific elements of the analytical procedure is discussed in Sections 3.12.4.2, "Dynamic Piping Model," 3.12.5.5, "Combination of Modal Responses," 3.12.5.6, "High Frequency Modes," and 3.12.5.13, "Combination of Inertial and Seismic Anchor Motion Effects" of this report. The staff review of the combination of responses for the three directional components identified a discrepancy. Although in Section 3.7.3.9, Westinghouse states that the three directional responses are combined by the SRSS method, in Section 3.7.3.6 Westinghouse provided an alternate combination method which allows combination of the responses from one direction with 40 percent of the responses from the other two directions (i.e., 100 percent-40 percent, 40 percent method). The staff accepted this method in structural analysis because of evidence that the method is generally more conservative than using SRSS. However, the staff had not accepted its application in piping analysis because of a lack of evidence that similar conservatism also exists, because piping seismic response generally has narrower frequency bandwidth than response of structures. Thus, to close this part of DSER Open Item 3.12.3-1, further Westinghouse action either to delete the method of directional response combination or to provide justification for its application to piping analysis in a future SSAR revision was needed.

In Revision 7 to the SSAR, Westinghouse revised Section 3.7.3.6 to indicate that the 100-40-40 method is not used for piping systems. On the basis of this revision, this part of DSER Open Item 3.12.3-1 is closed.

3.12.3.3 Independent Support Motion Method

When this method is used, the staff's position is that the responses caused by motions of supports between two or more different support groups may be combined by the SRSS method if a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor, or portions of a floor, of a structure. The response to RAI 210.11 dated December 22, 1992, and the discussion in Revision 1 to Section 3.7.3.9 of the SSAR, contained this acceptable definition of a support group. Furthermore, in this analysis method, neither ASME Code Case N-411-1 damping nor a constant 5 percent damping is used.

The ISM Method is specified by Westinghouse as an analysis option for subsystems and a mention of the method is made in Section 3.7.3.9 of the SSAR. A commitment to comply with the limitation regarding damping is made in Section 3.7.3.9 of the SSAR. In the DSER, the staff reported that a more detailed description of all dynamic analysis methods, including the independent support motion response spectrum method, must be included in the SSAR. (See DSER Open Item 3.12.3-1 in Section 3.12.3 above).

In Revision 7 to the SSAR, Westinghouse provided additional information in Section 3.7.3.9. It stated that each support group is considered to be in a random-phase relationship to the other support groups. The responses caused by each support group are combined by the SRSS method. A support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure. The damping requirements were given in Section 3.7.3.15 and Table 3.7.1-1. For independent support motion analysis, 2 percent damping is used for piping systems less than or equal to 12 inches in diameter, and 3 percent damping is used for piping systems greater than 12 inches in diameter. This is consistent with Regulatory Guides 1.61 and 1.84 and is acceptable. On the basis of this additional information, the staff evaluation concludes that this part of DSER Open Item 3.12.3-1 is closed.

3.12.3.4 Time-History Method

A time history analysis may be performed using either the modal superposition method or the direct integration method. The modal superposition method involves the calculation and utilization of the natural frequencies, mode shapes, and appropriate damping factors of the particular system toward the solution of the equations of dynamic equilibrium. The direct integration method involves the direct step-by-step numerical integration of the equations of motion and does not require the calculation of natural frequencies and mode shapes. In either method, the numerical integration time step, Δt, must be sufficiently small to accurately define the dynamic excitation and to render stability and convergence of the solution up to the highest frequency of significance. In direct integration analysis, the damping is input in the form of α and β damping constants, which give the percentage of critical damping, λ, as a function of the natural frequency, ω.

In Section 3.7.3.1 of the SSAR, the time history analysis methods are stated to be analysis options for the seismic analysis of subsystems. In Section 3.7.3.6 of the SSAR, some guidance was provided regarding the definition of the time history inputs to be used in time history analyses. No further description of these methods for subsystem analysis was provided in the SSAR. In the DSER, the staff reported that a more detailed description of all analysis methods, including time history analysis methods, must be included in the SSAR. (See DSER Open Item 3.12.3-1 in Section 3.12.3 above).

When the time-history method of analysis is used, the time-history data is broadened, plus and minus 15 percent of Δt, in order to account for modeling uncertainties. Westinghouse stated that they do employ time history broadening to account for uncertainties when time history analyses are performed. (See DSER Open Item 3.12.3-1 in Section 3.12.3 above.)

In a draft SSAR revision, Westinghouse added Section 3.7.3.17, "Time History Analysis of Piping Systems." Westinghouse stated that time history dynamic analysis is an alternate seismic analysis method that may be used with time history seismic input. It may also be used for hydraulic transient loadings and for pipe break loadings. It can be used with the GAPPIPE, PS+CAEPIPE, and WECAN computer programs. The modal superposition method is used to solve the equations of motion. The total responses are obtained by the algebraic sum of the modal responses at each time step. The staff reviewed this section but found the information to be incomplete. The use of the modal superposition method is acceptable but Westinghouse was asked to provide more detailed requirements and limitations. This should include the requirements for ensuring that integration time step is sufficiently small to accurately define the dynamic excitation and to render stability and convergence of the solution up to the highest frequency of significance. The statement in Section 3.7.3.17 that the time steps are no larger than the time history input time steps was not sufficient. The SSAR should also include a description of the method to account for modeling uncertainties such as time history broadening. The use of composite modal damping with PS+CAEPIPE or WECAN was specified. The application of composite modal damping should be subject to the limitation described in Section 3.12.5.16 of this report.

In Revision 9 to the SSAR, Westinghouse revised Section 3.7.3.17 to include a statement that the integration time step is no larger than 10 percent of the period of the cutoff frequency. This section also indicated that composite modal damping is used with the damping values for individual components listed in Table 3.7.1-1. For piping, this is 2 percent for diameters less than or equal to 3.5 cm (12 in.), 3 percent for diameters greater than 3.5 cm (12 in.), and 4 percent for the primary coolant loop. The damping values are consistent with the guidelines of RG 1.61 and previously-accepted engineering practice and are acceptable.

In a letter dated November 11, 1996, Westinghouse proposed another revision to SSAR Section 3.7.3.17 to address the issue of modeling uncertainties in time history analysis. This was reviewed and further discussed during the staff design review meeting conducted at NRC offices on December 5 through 6, 1996. Westinghouse described different methods for addressing the uncertainties. Four separate soil cases (See SSAR Section 3.7.1) must be considered. One approach is to perform time history analysis for each soil case. Another approach is to perform time history analysis for the hard rock soil case and a single response spectrum analysis for the remaining three soil cases. For time history analysis of piping systems that include a dynamic model of the supporting concrete building, either the building stiffness is varied by ±30 percent or the time scale is shifted by ±15 percent to account for uncertainties. Alternately, when a time history analysis is performed and the time history is developed from an enveloping response spectrum, modeling uncertainties are accounted for by the broadened response spectra. These changes were subsequently incorporated into Revision 10 of the SSAR.

The staff reviewed the proposed approaches and concluded that either the time scale shifting method or the building stiffness variation method is acceptable to account for uncertainties in the time history analysis of an individual soil case. However, the mixed use of time history analysis and response spectrum analysis was not acceptable. In a letter dated March 13, 1997, Westinghouse agreed to delete the option of mixing the two types of analyses. The letter proposed another SSAR revision to Section 3.7.3.17 which provides the following two options:

(1) Perform time history analysis for each soil case using either the building stiffness or time scale variations to account for uncertainties (as described above).

(2) Perform time history analysis (developed from enveloping response spectrum) on the basis of the four soils in which the uncertainties are accounted for by the spreading that is included in the broadened response spectra.

The staff finds this acceptable because either approach, not a mixed one, is acceptable to account for modeling uncertainties. The proposed changes were subsequently incorporated in Revision 12 to Section 3.7.3.17 of the SSAR. Thus, the part of Open Item 3.12.3-1 for four separate soil cases in a consistent manner is closed.

3.12.3.5 Inelastic Analysis Method

Westinghouse did not provide any information on the use of inelastic analysis methods for the AP600 piping analyses in Revision 0 of the SSAR. The staff position on inelastic analysis methods is described in Section 3.9.1.II.4 of the SRP which indicates that the methods of analysis used to calculate stresses and deformations should conform to the methods outlined in Appendix F of the ASME Code, Section III. In RAI 210.64, Westinghouse was requested to identify each safety-related system, component, or support which will be designed using an elastic-plastic method of analysis and to revise the SSAR to provide a description of the methodology consistent with the SRP guidelines. In the response dated July 25, 1994, Westinghouse stated that for systems where service level D limits are specified for safety-related piping and supports, the method of analysis used to calculate the stresses and deformations shall conform to the methods outlined in Subsection NF and Appendix F of Section III of the ASME Code. The inelastic analysis criteria included in Appendix F will be used as an alternative to the procedures of NB-3652 of Section III of the ASME Code when the inelastic analysis option will provide a more cost-effective design. No particular system, component, or support had been identified for this type of design evaluation.

In a proposed revision to Section 3.9.3.1.5 of the SSAR, included as part of the July 15, 1994, response to RAI 210.64, Westinghouse provided additional conditions that must be satisfied when an elastic analysis is performed for SSE and an inelastic analysis is performed for pipe break loadings. When an inelastic analysis is performed for SSE and pipe break loadings, Westinghouse stated that the method in Table 3.9-6, note 19 (as revised in the July 27, 1994, response to RAI 210.79, which is discussed in Section 3.12.5.3 of this report) would be used.

In reviewing the response, the staff found that the methods of analysis outlined in Appendix F of Section III of the ASME Code were consistent with the SRP guidelines. However, the SRP also requires that the analytical procedures to be used in the analysis be reviewed by the staff to verify their validity. In the proposed SSAR revision, Westinghouse stated that the analytical procedures used in the inelastic analysis are those associated with the WECAN computer code. Westinghouse was asked to provide a detailed description of these procedures to the staff for review. In addition, the strain limits proposed by Westinghouse were not consistent with the requirements of Appendix F of Section III of the ASME Code. Westinghouse proposed two different sets of strain limits. For elastic strains associated with SSE combined with inelastic strains associated with pipe break loads, the proposed limits were 1 percent averaged through the wall, 2 percent for strains at the surface because of equivalent linear distribution through the wall, and 5 percent for local strains at any point. When inelastic analysis was performed for both SSE and pipe break loadings, the limit was 5 percent for the effective ratchet strain averaged through the wall thickness, and the limit on the effective local peak cyclic single-amplitude strain was on the basis of a formula that considers the allowable stress (S_a) value from the ASME Code fatigue curve at 10 cycles. Neither set of strain limits are specified in Appendix F.

In the DSER, the staff concluded that if Westinghouse plans to use inelastic analysis methods in the design of piping systems, it should identify the specific systems, provide a detailed description of the methodology, and either provide additional justification for the proposed acceptance criteria or provide acceptance criteria consistent with the guidelines of Section 3.9.1 of the SRP. This information should be submitted to the staff for review and approval and included in the SSAR. This was identified as DSER Open Item 3.12.3.5-1.

During the staff design review meeting conducted at Westinghouse offices on April 10-11, 1995, Westinghouse informed the staff that it is withdrawing its plans to use inelastic analysis methods. In Revision 4, SSAR Section 3.9.3.1.5, "ASME Classes 1, 2, and 3 Piping," Westinghouse added the statement, "Inelastic analysis methods are not used." On this basis, DSER Open Item 3.12.3.5-1 is closed.

3.12.3.6 Small-Bore Piping Method

Small-bore piping consists of ASME Code Class 1 piping equal to or less than 2.54 cm (1-in.) nominal pipe size and ASME Class 2 and 3 piping with nominal piping sizes less than or equal to 5.1 cm (2 in.). In Section 3.7.3.8.2.2 of the SSAR, Westinghouse specified three analysis options for the analysis of these systems. The options included the response spectrum method, the equivalent static analysis method, and seismic qualification by experience on the basis of the guidelines in EPRI Report No. NP-6628, "Procedure for Seismic Evaluation and Design of Small-bore Piping (NCIG-14)," April 1990.

The response spectrum method is an acceptable analysis methodology for the analysis of both small and large bore piping. Staff comment regarding its use in the AP600 is provided in Section 3.12.3.2 above.

In Section 3.7.3.5 of the SSAR, Westinghouse provides a description of the equivalent static load method. In the method, dynamic response is determined by performing static analyses of the system subjected to static loads which are a conservative equivalent of the dynamic loads. In Section 3.7.3.5 of the SSAR, Westinghouse provides the criteria for calculating the equivalent static loads. In the response to RAI 210.48 dated July 25, 1994, Westinghouse provided a proposed revision to this section of the SSAR, which clarified the specific requirements for applying this method to small-bore piping. It stated that for piping systems, a static load factor of 1.5 is applied to the peak accelerations of the applicable floor response spectra. For piping runs with axial supports, a factor of 1.0 may be used in the axial direction with the spectral acceleration value on the basis of the frequency in the axial direction. The relative motion between support points is considered when significant. The staff found that these static load factors meet the guidelines of Section 3.9.2.II.2.a(2)(c) of the SRP and are acceptable, pending receipt of the revised SSAR. This was identified as DSER Confirmatory Item 3.12.3.6-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided Revision 2 to SSAR Section 3.7.3.5. The staff reviewed this revision and disagreed with the paragraph that stated, "In lieu of using the peak acceleration value, the actual frequency may be calculated and the corresponding acceleration value may be used without amplification." The staff position is that the amplification factor of 1.5 should still be used in such cases. Westinghouse agreed to rewrite this section. On June 2, 1995, a preliminary copy of SSAR Section 3.7 Draft Revision 4 was provided which included a revised Section 3.7.3.5. The staff reviewed this draft revision and disagreed with the sentence in Section 3.7.3.5.1 that stated, "A factor of 1.0 is used for structures or equipment that can be represented as uniformly loaded cantilever, simply supported, fixed-simply supported, or fixed-fixed beams." This is not consistent with the guidelines of SRP 3.9.2 and is not acceptable without further justification. Westinghouse was asked to revise the SSAR to commit to use a 1.5 factor unless adequate justification for a lower factor was provided. Thus, further Westinghouse action was needed for closure of DSER Confirmatory Item 3.12.3.6-1.

During the design review meeting conducted at Westinghouse offices on June 25 through 26, 1996, Westinghouse agreed that a 1.0 factor should not be used for piping analysis. They provided a draft revision to Section 3.7.3.5.1 in which they added the statement that static load factors smaller than 1.5 are not used for piping systems. This was later incorporated into SSAR Revision 9. Revision 14 to Section 3.7.3.5.2 of the SSAR further characterizes one exception case that if the axial direction of the pipe is rigid and has fundamental frequency equal or above 33 Hz, 1.0 times its peak spectral acceleration for piping axial load may be used. The staff finds this acceptable because 33 Hz is well above the dominate frequency range of the earthquake motions, so significant amplification of piping axial response is unlikely. Therefore, DSER Confirmatory Item 3.12.3.6-1 is closed.

The staff has not accepted the use of EPRI-6628 (NCIG-14), which incorporates, in part, the use of a seismic experience-based approach for the design or qualification of safety-related piping in nuclear power plants. In the responses to RAI 210.30 and RAI 210.046 dated May 20, 1994, Westinghouse responded with proposed revisions to the SSAR which delete reference to seismic design by experience and design by rule, and EPRI-6628. The proposed SSAR revisions were acceptable to the staff and must be incorporated into an amendment to the SSAR. This was identified as DSER Confirmatory Item 3.12.3.6-2.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided Revision 2 to SSAR Section 3.7. The staff reviewed this revised section and found that all references to the "design by rule" method and to EPRI Report NP-6628 (NCIG-14) were deleted. On this basis, DSER Confirmatory Item 3.12.3.6-2 is closed.

3.12.3.7 Non-Seismic/Seismic Interaction (II/I)

All non-seismic Category I piping or other non-seismic SSCs should be isolated from seismic Category I piping. This isolation may be achieved by designing a seismic constraint or barrier or by locating the two sufficiently apart to preclude any interaction. If it is impractical to isolate the seismic Category I piping system, the adjacent non-seismic Category I system should be evaluated to the same criteria as the seismic Category I system.

Westinghouse provided the requirements and criteria for protection against non-seismic/seismic interaction in Section 3.7.3.13 of the SSAR. Separation or segregation of seismic Category I piping from non-seismic SSCs are the preferred methods of eliminating the possibility of seismic interaction. As an alternative, an impact analysis may be performed to demonstrate that a potential non-seismic SSC identified as a source would not cause unacceptable damage to the target. If the approaches of separation, segregation or impact analysis cannot prevent unacceptable interaction, the source is classified and supported as seismic Category II to assure that the SSE will not cause unacceptable structural failure of or interaction with seismic Category I piping. The staff finds this approach consistent with Section 3.9.2 of the SRP and is acceptable.

For non-seismic Category I piping systems attached to seismic Category I piping systems, the dynamic effects of the non-seismic Category I systems should be considered in the analysis of the seismic Category I piping. In addition, the non-seismic Category I piping from the attachment point to the first anchor should be evaluated to ensure that, under all loading conditions, it will not cause a failure of the seismic Category I piping system.

In Section 3.7.3.13.3 of the SSAR, Westinghouse stated that interaction of seismic Category I piping and non-seismic piping connected to it is achieved by incorporating into the seismic analysis of the Category I system a length of pipe that represents the actual dynamic behavior of the non-seismic system. The length considered, at a minimum, extends to the first anchor point beyond the point of change from seismic Category I to non-seismic or, alternately, to two seismic restraints in each of the X, Y, and Z directions. Those portions of the non-seismic piping included in the Category I piping analysis are analyzed according to Section III of the ASME Code stress intensity limits for seismic Category I piping.

Westinghouse was requested to provide justification to demonstrate that a length of pipe extending to two seismic restraints beyond the interface (versus an anchor) is sufficient to characterize the dynamic behavior of the system and to explain how the loads on the seismic restraints from the non-seismic piping beyond the analyzed region will be accounted for. As an alternative, Westinghouse should commit to the guidelines of Section 3.9.2 of the SRP, which require an evaluation of the non-seismic piping up to the first anchor beyond the interface. This was identified as DSER Open Item 3.12.3.7-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse presented an alternative methodology to achieve acceptable separation between seismic and attached non-seismic piping. The methodology was also described in a draft revision to SSAR Section 3.7.3.13.4.2 which was submitted on June 2, 1995. Westinghouse proposed to either (1) extend the analysis up to the first anchor beyond the interface and design the anchor for the maximum loads from the non-seismic piping, (2) provide additional supports beyond the first anchor to help accommodate the maximum loads, or (3) provide a close cluster of supports designed to provide a rigid pipe region to dynamically decouple the seismic pipe from the non-seismic pipe and to accommodate the maximum loads. The additional length of piping between the seismic piping attachment point and the interface anchor or interface support would be analyzed as seismic Category II piping. Plastic hinge moments from the non-seismic piping would be applied at the interface anchor or at the last interface support between the non-seismic piping and the seismically analyzed piping. For each case, the supports in the Category II region would be evaluated for SSE loads using the rules of ASME Section III Subsection NF. The seismic Category II piping would be evaluated in accordance with Equation 9 of ASME Section III, Class 3 with a stress limit equal to the smaller of $4.5S_h$ or $3.0S_y$. The staff reviewed the proposed methodology and determined that it ensures the structural integrity of the seismic Category II piping. Therefore, the staff finds the use of 4.5 S_h acceptable for this particular application only. The description of the methodology given in the Westinghouse proposed draft revision to SSAR Section 3.7.3.13.4.2 was found acceptable with one exception. For the third method, Westinghouse must provide a quantitative definition of the "rigid region" (such as a minimum frequency criterion) in order to justify decoupling of the dynamic response of the non-seismic piping from the dynamic response of the seismic piping.

In SSAR Revision 7, Westinghouse incorporated the above description of the methodology and included the definition of the "rigid region". The rigid region was defined as either four bilateral supports around an elbow or six bilateral supports around a tee. The structural behavior of this region should be similar to that of a six-way anchor. The frequency of the piping system in this region must be greater than or equal to 33 Hz. The staff found this acceptable because the 33 Hz assures that no significant amplification of piping response in this region will occur during the seismic event. Therefore, Open Item 3.12.3.7-1 is closed.

3.12.3.8 Buried Piping

In Section 3.7.3.12 of the SSAR, Westinghouse states there are no seismic Category I buried piping systems and tunnels in the AP600 design.

3.12.3.9 ASME Code, Section III, Appendix N

The staff has not endorsed the use of Appendix N of Section III of the ASME Code. During the April 1994, design review meeting, Westinghouse stated that the methodology used in the

AP600 design and described in the SSAR is not on the basis of Appendix N of Section III of the ASME Code. This is acceptable to the staff.

3.12.3.10 Conclusions

Westinghouse has revised the SSAR to incorporates the proposed changes to Section 3.7.3.17 described in Sections 3.12.3 and 3.12.3.4 of this report, the staff concludes that the analysis methods to be used for all seismic Category I piping systems, as well as non-seismic Category I piping systems that are important to safety, are acceptable. The analysis methods utilize piping design practices that are commonly used in the industry and provide an adequate margin of safety to withstand the loadings as a result of normal operating, transient, and accident conditions.

3.12.4 Piping Methodology

The staff has evaluated the piping methodology used in the design of the AP600, as presented in the SSAR, as described below.

3.12.4.1 Computer Codes

This section addresses the computer codes to be used to analyze piping systems in the AP600 design. Table 3.9-15 of the SSAR includes a listing of computer programs used for static and dynamic analyses to determine the structural and functional integrity of seismic Category I and non-seismic Category I items. Design control measures to verify the adequacy of the design of safety-related components are required by Appendix B to 10 CFR Part 50. A proposed revision to Section 3.9.1.2 of the SSAR provided by Westinghouse in the June 30, 1994, response to RAI 210.33, states that computer programs used in analyses for the AP600 comply with an established QA program and are verified by one or more of 10 methods listed in this section. The methods listed are consistent with the methods recommended in Section 3.9.1 of the SRP. In the DSER, the staff reported that the proposed SSAR revision is acceptable and must be incorporated into an amendment to the SSAR. (See DSER Confirmatory Item 3.9.1.2-1 in Section 3.9.1.2 above). The proposed changes were incorporated into Revision 4 of SSAR Section 3.9 dated June 30, 1995, and were reviewed by the staff and found acceptable.

The staff performed independent confirmatory piping stress analyses of representative piping systems in the AP600 design. The purpose of these analyses was to verify the adequacy of the computer program used to generate the sample piping analyses. The confirmatory analyses duplicate the Westinghouse sample analyses but use an independently developed and verified computer code. In the DSER, the staff stated that the adequacy of the Westinghouse computer program would be assessed by direct comparison of Westinghouse and staff results. The results of the confirmatory analysis would then be included in the FSER. This was identified as DSER Open Item 3.12.4.1-1.

The independent confirmatory piping stress analyses were completed in November 1995. The results of the analyses were compared against the results of the same problems analyzed by Westinghouse using their computer program. The comparison of results did not meet the staff acceptance criteria which were the difference in calculated results not exceeding the following: 2 percent in natural frequency, 10 percent in maximum displacements, 5 percent in maximum moments, and 10 percent in supporting reactions, as indicated in NUREG/CR-6049. These acceptance criteria were used previously in other advanced reactor reviews. A detailed comparison of the staff and Westinghouse computer input files identified discrepancies in modeling parameters and input loads. Both sets of models were revised for consistency and the analyses were rerun. The results of the revised analyses were compared during the design review meeting at Westinghouse on June 25 through 26, 1996. The comparison showed that the results of all three analyses met all of the acceptance criteria. On this basis, Open Item 3.12.4.1-1 is closed.

The staff requested Westinghouse to provide sample analyses for the confirmatory evaluation of three methods of analysis, as well as sufficient information to allow staff interpretation of the computer listings of input and output for each analysis. Westinghouse agreed to provide this information for the following three sample calculations:

(1) seismic analysis of the pressurizer surge line using the enveloped response spectrum analysis method

(2) seismic analysis of the pressurizer surge line using the independent support motion (ISM) response spectrum analysis method

(3) fluid transient dynamic analysis of the main steamline using the modal superposition time history analysis method

In the DSER, the staff reported that Westinghouse provided information on the first two piping calculations. During the staff design review meeting conducted at Westinghouse offices on July 19-21, 1994, Westinghouse informed the staff that the third sample calculation was in progress and would be provided at a later date. This was identified as DSER Open Item 3.12.4.1-2.

The third piping sample calculation (a time history analysis of the main steamline) was provided by Westinghouse in February 1995. However, upon detailed review of the three sample calculations, the staff determined that the calculations were extremely large and complex and not suitable for this purpose. Each sample calculation included the subject piping system coupled to the reactor coolant system piping and components. During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse was asked to revise their analyses to uncouple the piping systems from the reactor coolant system. Westinghouse agreed to do this and provided the results of their uncoupled piping analyses in May 1995. The staff analyzed the same three piping calculations. The results of these analyses were used to verify the adequacy of the Westinghouse computer program as discussed above. Therefore, on the basis of receiving this additional information, DSER Open Item 3.12.4.1-2 is closed.

3.12.4.2 Dynamic Piping Model

For the dynamic analysis of seismic Category I piping, each system is idealized as a mathematical model consisting of lumped masses interconnected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as a change in stiffness as a result of member curvature.

A description of a procedure used for modeling systems was provided in Section 3.7.3.3 of the SSAR. The level of detail provided in the description was not sufficient to define the methods used to model piping in the AP600. The staff reviewed an internal Westinghouse document which described the piping design criteria for the AP600 design. This document provided a detailed description of the design criteria and referenced many pertinent AP600 documents currently in development, but did not describe the modeling methods in the level of detail required. In the DSER, the staff stated that a detailed description of piping system analysis, modeling methodology, criteria, and guidelines, must be included in the SSAR. This was identified as DSER Open Item 3.12.4.2-1.

The effect of pipe support stiffness, the flexibility of supplementary steel, and the flexibility of non-rigid model termination points must all be adequately considered in the piping system analysis. In Section 3.9.3.4 of the SSAR, Westinghouse provided some guidance on the definition of support stiffness for piping system analysis, but it was not in sufficient detail for the staff to reach a safety finding. During the July 19 through 21, 1994 design review meeting, the staff discussed with Westinghouse its approach to define displacement limits for pipe support design and generic support stiffness values for piping system analysis. A detailed description of the modeling of supports, supplementary steel, and non-rigid boundaries in piping system analysis must be provided in the SSAR. In the DSER, the staff reported that this information should be included in the description of piping system analysis modeling methodology to be provided to resolve Open Item 3.12.4.2-1 above.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided a draft piping analysis procedure document for review. The staff reviewed the document and determined that although the methodology was generally acceptable, the document contained a significant amount of procedural information which is not appropriate for incorporating into the SSAR. Westinghouse agreed to extract the appropriate technical information from this document and include it in an SSAR revision.

In a draft SSAR revision dated June 2, 1995, Westinghouse included additional information in Section 3.7.3.8 regarding piping system modeling requirements. It describes the requirements for maximum spacing of lumped masses and minimum number of degrees of freedom. It discusses the pipe support stiffness requirements. Either minimum rigid or calculated support stiffness values are used. When minimum rigid support stiffnesses are used, the faulted condition deflection shall not exceed one-eighth inch. Additional information on the stiffness, mass and decoupling requirements for supporting systems including supplementary steel, equipment and other piping systems is included in Sections 3.7.3.8.1 and 3.7.3.8.2 of the draft SSAR revision and in Section 3.9.3.4 of the SSAR, Revision 4. The staff evaluation of these issues is discussed in Sections 3.12.4.4 and 3.12.6.7 of this report. The detailed description of the modeling requirements described above were incorporated in Revision 7 to Section 3.7.3.8 of the SSAR. The staff reviewed these requirements and found them acceptable because they conform with SRP 3.7.3 and good engineering practice. The staff evaluation of the additional items discussed in Section 3.12.4.4 and 3.12.6.7 of this report resulted in the resolution of all issues of concern. On this basis, Open Item 3.12.4.2-1 is closed.

3.12.4.3 Piping Benchmark Program

In the DSER, the staff reported that to verify the adequacy of the computer program used by the COL applicant to complete the AP600 piping system design and analyses, the NRC staff will establish mathematical models of representative piping systems in the AP600 design and use them in a piping analysis benchmark program. The mathematical models will be on the basis of the dynamic piping model and on the piping stress analysis criteria described in Section 3.12.4.2 and Section 3.12.5, respectively, of this report. The benchmark program verifies the adequacy of linear-elastic, dynamic piping analysis using the enveloped response spectrum method, ISM response spectrum method, and time-history method of analyses.

The benchmark program essentially consists of the COL applicant constructing mathematical models of the AP600 pressurizer surge line and the main steamline, using the COL applicant's computer program. The piping configuration for the piping models are described in NUREG/CR-6414, "Piping Benchmark Problems for the Westinghouse AP600," dated August 1996 and includes piping dimensions, pipe sizes, materials, valve weights, support and anchor stiffnesses, and support locations. The piping input parameters for the benchmark analyses are also specified in the piping benchmark program, and include damping values, loading definitions, and load combinations.

When the COL applicant's dynamic piping analyses are completed, the results of the analyses must be compared with the results of the benchmark problems provided in the piping benchmark program. The piping analysis results to be compared and evaluated include the system modal frequencies, the maximum pipe moments, the maximum support load and equipment reactions, and the maximum pipe deflections. The acceptance criteria, or range of acceptable values, are specified in the piping benchmark program and must be satisfied. The COL applicant must document and submit any deviations from these values, as well as the justification for such deviations, to the NRC staff for review and approval before initiating final certified piping analyses. The benchmark program provides assurance that the computer program used to complete the AP600 piping design and analyses produces results that are consistent with results considered acceptable to the NRC staff. In the DSER, the staff reported that a commitment that the COL applicant will comply with the benchmark program must be included in the SSAR. This was identified as DSER Open Item 3.12.4.3-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse agreed to revise the SSAR to include a commitment that the COL applicant will comply with the guidelines in the benchmark program. In SSAR Revision 8, Westinghouse revised Section 3.9.1.2 to include the commitment that the COL applicant will implement the NRC benchmark program using AP600-specific problems if a piping analysis computer program other than those used for design certification is used. The staff finds this acceptable and DSER Open Item 3.12.4.3-1 is closed. This is COL Action Item 3.12.4.3-1.

3.12.4.4 Decoupling Criteria

When analyzing piping systems, the size of the mathematical model might exceed the capacity of the computer program if large-bore and small-bore piping are included. Thus, the small-bore branch lines are generally decoupled from the large-bore main piping. In Section 3.7.3.8.1 of the SSAR it is stated that branch piping can be decoupled from main piping if the ratio of the moment of inertias of the supported pipe to supporting pipe is less than 0.04, or if the ratio of the nominal outside diameter of the supporting pipe to supported pipe is greater than or equal to three. These criteria are consistent with industry practices and are acceptable.

Criteria are also provided in Section 3.7.3.8.1 of the SSAR to address the effects of supports in the branch line close to the intersection point. No criteria were provided in the SSAR to address mass effects of the branch line in the analysis of the main line. In the DSER, the staff reported that detailed criteria and procedures to account for branch line mass and flexibility effects in the main line analysis, when decoupling, must be included in the SSAR. This was identified as DSER Open Item 3.12.4.4-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided proposed criteria for consideration of mass and stiffness effects of branch lines on run lines when the lines are decoupled. It specifies that the stiffness effect is significant when the distance from the run pipe to the first rigid or seismic support on the branch pipe is less than half the deadweight span of the branch pipe (given in ASME Code Section III Subsection NF). The mass effect is significant when the weight of half the span of the branch line (in each direction) is more than 20 percent of the run pipe span in the same direction. If the weight is less than 20 percent but more than 10 percent, the weight can be lumped at the intersection point for the run pipe analysis. If the stiffness and/or mass effects are considered significant, the branch piping is included in the piping analysis model for the run pipe analysis. This proposed criteria were subsequently incorporated in Revision 7 to SSAR Section 3.7.3.8.1. The staff reviewed this criteria and found it to be sound engineering practice and technically reasonable to account for mass and stiffness effects of branch lines on the run lines. On this basis, DSER Open Item 3.12.4.4-1 is closed.

Section 3.7.3.8.2.1 of the SSAR stated that the intersection point between the run pipe and branch pipe is considered to be anchored in the seismic inertial analysis of the branch piping. The response spectra assigned to this analytical anchor are the spectra for the run pipe supports near the intersection point. The application of this procedure, irrespective of the amplification of the spectra at the intersection point, was not acceptable to the staff. In the July 8, 1994, response to RAI 210.49, Westinghouse proposed a screening criteria for the applicability of the method on the basis of the calculated dynamic displacement of the intersection point. In the DSER, the staff indicated that this criterion will be evaluated along with the description of piping analysis methods to be provided to resolve Open Item 3.12.4.2-1. Criterion for the definition of spectra at the intersection point of the run pipe and branch pipe in an analytical model of the branch piping must be provided in the SSAR. This was identified as DSER Open Item 3.12.4.4-2.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, the proposed screening criterion provided in the July 8, 1994 response to RAI 210.49, was further discussed. In order to limit significant amplification by the run pipe at the branch line connection point, Westinghouse had proposed a one-inch deflection limit on inertial displacement. The staff reviewed the Westinghouse justification for this limit and found it technically inadequate. The use of a deflection limit without consideration of branch line or run line frequencies cannot ensure against significant response spectrum amplification at the connection. The possibility of a frequency ratio criterion was discussed and Westinghouse agreed to give it further consideration. However, in Draft Revision 4 to SSAR

Section 3.7.3.8.2.1 submitted on June 2, 1995, Westinghouse included the same one-inch deflection criterion which is unacceptable to the staff.

During the staff design review meeting conducted at Westinghouse offices on June 25 through 26, 1996, Westinghouse provided another draft revision to SSAR Section 3.7.3.8.2.1 which stated that when supported piping is supported by larger piping, then either a coupled dynamic model of the supported piping and the supporting piping is used or the amplified response spectra at the connection point to the supporting piping is used with a decoupled model of the supported piping. The staff reviewed this revised criterion and concludes that it is acceptable to ensure that dynamic coupling of the supporting and supported piping is considered. These design requirements were subsequently incorporated in Revision 9 to the SSAR. On this basis, Open Item 3.12.4.4-2 is closed.

3.12.4.5 Conclusions

The staff concludes that Westinghouse meets Appendix B to 10 CFR Part 50 and GDC 1 by submitting information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I piping designated as ASME Code Class 1, 2, and 3, and those not covered by the code, within the present state-of-the-art limits, and by having design control measures that are acceptable for ensuring the quality of its computer programs. Although COL applicants or licensees referencing the AP600 design are not required to use the Westinghouse computer programs, the computer programs used by the COL applicant or licensee, to complete its analyses of AP600 piping systems, will be validated using the piping benchmark program discussed herein.

3.12.5 Pipe Stress Analysis Criteria

The staff has evaluated the pipe stress analysis criteria described in the SSAR for the AP600, as discussed below.

3.12.5.1 Seismic Input

The AP600 is designed for an SSE ground motion defined by an RG 1.60 response spectrum that is enhanced in the high-frequency range (8-40 Hz) and anchored to a peak ground acceleration of 0.3g. Amplified building response spectra are generated for the AP600 design to account for varying soil properties in the United States.

In Section 3.7.2.5 of the SSAR, Westinghouse describes the development of floor response spectra for the AP600 design. It currently states that, for the design of subsystems and components, they are generated by enveloping the nodal response spectra determined for the different soil profiles. These enveloped spectra are smoothed and spectral peaks broadened by 15 percent.

During the April 12 through 14, 1994 design review meeting at the Westinghouse offices, Westinghouse stated that it would also use the peak shifting method as an analysis option in the response spectrum analysis of piping. In the DSER, the staff indicated that a description of the peak shifting method must be provided in the SSAR. This was identified as DSER Open Item 3.12.5.1-1.

During the staff design review meeting conducted at Westinghouse offices on April 10-11, 1995, Westinghouse provided a written description of the peak shifting method. The staff reviewed the methodology and found it technically adequate because it is consistent with the ASME Code Case N-411 as conditionally accepted by RG 1.84. The staff, however, suggested that the description of the application of peak shifting in three directions be clarified to indicate that the shifting will be done independently in each direction. Westinghouse agreed to make this change. Draft Revision 4 of SSAR Section 3.7.3.9 included an acceptable description of the peak shifting method. This was subsequently incorporated into SSAR Revision 7. Thus, DSER Open Item 3.12.5.1-1 is closed.

3.12.5.2 Design Transients

In Section 3.9.1.1 of the SSAR, Westinghouse discusses the design transients for ASME Code Class 1 components and supports. Table 3.9-1 of the SSAR lists the design transients for five plant operating conditions and the number of either plant operating events, or cycles for each of the design transients, that will be used in the design and fatigue analyses of the ASME Code Class 1 piping systems.

The operating conditions are as follows:

- ASME Service Level A - normal conditions
- ASME Service Level B - upset conditions--incidents of moderate frequency
- ASME Service Level C - emergency conditions--infrequent incidents
- ASME Service Level D - faulted conditions--low-probability postulated events
- testing conditions

Westinghouse states that the number of events or cycles resulting from each of the listed design transients were defined to be consistent with a 60-year design objective. This is acceptable. A more detailed discussion of this issue is contained in Section 3.9.1.1 of this report.

3.12.5.3 Loadings and Load Combinations

The staff reviewed the methodology used for load combinations and the selected values of allowable stress limits. Westinghouse provided the design criteria for all ASME Code Class 1, 2, and 3 piping, using the loads, load combinations and stress limits given in Section 3.9.3.1.5 of the SSAR. Loads were listed in Tables 3.9-3 and 3.9-4 of the SSAR. Loading combinations were provided in Tables 3.9-6 and 3.9-7 of the SSAR. Stress limits were given in Tables 3.9-9, 3.9-10, and 3.9-11 of the SSAR. The staff reviewed the SSAR and raised several questions and issues which were discussed during the staff design review meeting in Westinghouse offices or submitted as RAIs (210.60, 210.62, 210.65, 210.79, and 210.80). Some of these issues are discussed in other sections of this report and involve elimination of OBE (3.12.5.14), functional capability (3.12.5.12), and alternate piping design criteria (3.12.5.19). Westinghouse subsequently proposed several SSAR revisions in these areas as part of their July 27, 1994, response to RAI 210.79. The final version of this response was submitted to the NRC in a letter dated July 25, 1994. The proposed revision to Section 3.9.3.1.5 of the SSAR indicates that piping loads are listed in revised Table 3.9-16 of the SSAR. Revised Tables 3.9-6 and 3.9-7 of the SSAR list the loading combinations and stress limits. Functional capability requirements are given in revised Table 3.9-11 of the SSAR. During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse agreed to make additional extensive revisions to the previously proposed load and load combination tables. These changes were included in Revision 4 of SSAR Section 3.9.3. The staff evaluation of these revised tables is discussed at the end of this section.

The combinations of design loadings are categorized with respect to service levels, identified as Design, and Level A, B, C, and D, and were shown in revised Table 3.9-6 of the SSAR for Class 1 piping and revised Table 3.9-7 of the SSAR for Class 2 and 3 piping. For each load combination, the corresponding ASME Code equation and stress limit was given. The revised tables reflected the use of alternate piping design criteria proposed for the AP600 design. The alternate criteria were on the basis of proposed changes to the ASME Code for piping systems. They involved the use of higher stress limits for service levels C and D, separate treatment of reversing and non-reversing dynamic loads, use of 5 percent damping, and additional stress limits on seismic anchor motions and longitudinal stresses. The current staff position on the alternate criteria is discussed in detail in Section 3.12.5.19 of this report. During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse advised the staff that it no longer plans to use the alternate piping design criteria for AP600. This is further discussed below and in Section 3.12.5.19 of this report.

The staff reviewed the proposed loads and load combinations given in Tables 3.9-6 and 3.9-7 of the SSAR and found them to be generally consistent with the guidelines provided in Section 3.9.3 of the SRP, the staff position on single earthquake design, and the proposed alternate Code changes. The staff, however, noted a number of deficiencies. Under ASME Service Level B load combinations, Westinghouse included dynamic transient events (DU) associated with upset conditions in NB/NC/ND-3650 Equation (9) stress evaluation load combinations. However, in Table 3.9-16 of the SSAR, the relief/safety valve, open system, sustained load (RVOS) was not included in the DU loads. (See Section 3.12.5.19.2 of the SSAR.) The staff evaluation of the revised load and load combination tables in SSAR Section 3.9.3, Revision 4 is discussed at the end of this section.

Under ASME Service Level C and D load combinations, Westinghouse proposed higher dynamic stress limits than those specified in the 1989 ASME Code, Section III. Under Level D, a stress limit of 4.5 S_m (instead of 3.0 S_m) was proposed for combinations that include SSE or reversing dynamic loads. However, in Table 3.9-6 of the SSAR, there were several combinations with the proposed higher stress limit that include non-reversing dynamic loads or combinations of reversing loads which should be treated as non-reversing loads. Westinghouse was requested to revise the stress limits for these combinations or provide additional justification. As noted above, during the April 1995 design review meeting, Westinghouse advised the staff that the proposed higher stress limits associated with the alternate piping design criteria will not be used for AP600. The staff evaluation of the latest proposed stress limits given in Revision 4 to SSAR Section 3.9.3 is given at the end of this section.

In Notes (6) and (16) to Table 3.9-6 and Note (4) to Table 3.9-7 of the SSAR, Westinghouse stated that the timing and causal relationships among the various given dynamic loads are considered to determine appropriate load combinations. The staff position on dynamic load combinations is that dynamic responses of piping loadings may be combined by the SRSS method in accordance with the guidelines of NUREG-0484, Revision 1. In RAI 210.63, the staff requested Westinghouse to incorporate this position in the SSAR and to explain how the notes in the tables relate to the staff position. In the response dated July 25, 1994, Westinghouse stated that the dynamic loads are combined by the SRSS method, which is consistent with industry practices and NUREG-0484. Westinghouse further explained that dynamic loads are postulated as initiating or consequential events. Dynamic loads that are expected to result from the initiating event are considered in combination with the loads resulting from that event, depending on the time phasing of the consequential event and initiating event. Loads resulting from dynamic events will only be combined with loads resulting from an initiating event if the loads can mechanistically and realistically occur simultaneously. Consequential dynamic loads from an SSE will be combined with SSE depending on the time phasing of the consequential event and the SSE. This issue was further discussed during the July 1994 design review meeting at the Westinghouse offices. The staff disagreed with the proposed Westinghouse approach for combining SSE and other dynamic loads. Westinghouse was requested to revise the SSAR to commit to the staff position. During the design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, this issue was again discussed. Westinghouse agreed to revise the tables to clarify how SSE loads will be combined with specific fluid transient loads. The staff evaluation of the SSAR Revision 4 changes that addressed this issue is given at the end of this section.

In Notes (18) and (19) to Table 3.9-6 of the SSAR, Westinghouse provided alternative strain limits for Levels C and D when an inelastic analysis is performed. Limits on both local peak single-amplitude strain and ratchet strain averaged through the wall thickness were provided. The staff did not find these strain limits acceptable without further justification (see Section 3.12.3.5). As discussed in Section 3.12.3.5, Westinghouse withdrew its plans to use inelastic analysis methods for piping. Therefore, the issue of inelastic strain limits is no longer applicable. The staff evaluation of the tables in Revision 4 of SSAR Section 3.9.3 is given below.

In the DSER, the staff stated that Westinghouse should provide another SSAR revision that adequately addresses the issues discussed above. This was identified as DSER Open Item 3.12.5.3-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, there were extensive discussions on these issues. Westinghouse agreed to revise the load and load combination tables given in SSAR Section 3.9.3 to delete the reference to the alternate piping design criteria and to address the other issues. These tables were extensively revised in later SSAR revisions.

In Revision 4 to Table 3.9-3 of the SSAR, Westinghouse provided the loads for AP600 piping design. The staff noted that previous references to reversing and non-reversing loads associated with the alternate piping design criteria were deleted. This is acceptable. However, the relief/safety valve open system sustained load (RVOS) was still not included as a transient dynamic event load (DU) associated with Level B (Upset) service conditions. It was instead included as a design mechanical load (DML). The safety/relief valve open system transient load (RVOT) was not included as either a DU or a DML load. During the April 1995 design review meeting, Westinghouse had committed to clarify these classifications in the SSAR tables. At the June 1996 design review meeting, Westinghouse explained that a relief/safety valve load (RV) generally consists of an initial transient load (RVOT) followed by a sustained steady state load (RVOS). Specific RV loads may be classified under different ASME Code service conditions. Therefore, RVOT should be included under DU while RVOS should not be included. Westinghouse provided a revised draft SSAR Table 3.9-3 which clarified the definitions of DU as well as DN, DE, DF, and DY (for other service conditions) to indicate that they would include such loads as RVOT, RVC, and FV (as applicable). The staff found this acceptable because load definitions are clarified and in conformance with SRP 3.9.3. Westinghouse also explained that according to the ASME Code, design mechanical loads (DML) are those loads for which Service Level A primary stress limits are applicable. In addition, for AP600, Westinghouse agreed to include RVOS loads that are Service Level B. The staff found this acceptable because this was clarified in the DML definition in the draft Table 3.9-3. These proposed revisions were subsequently incorporated into Table 3.9-3 of SSAR Revision 9. On the basis of this revision, this part of Open Item 3.12.5.3-1 is closed.

In Revision 4 to Table 3.9-5 of the SSAR, Westinghouse provided the minimum design loading combinations for ASME components including piping. In its review of this table, the staff found that note (6) stated that timing and causal relationships that exist between SSE and other dynamic events are considered for determination of appropriate load combinations. As indicated above, the staff position on dynamic load combinations is that dynamic responses of piping loadings should be combined by the SRSS method in accordance with NUREG-0484. In SSAR Revision 7, note (6) was changed but notes (13) and (14) included statements that timing and causal relationships are considered for combinations of SSE with other dynamic loads. At the June 1996 design review meeting, Westinghouse agreed to commit to the staff position of combining SSE with other dynamic loads by SRSS and provided a draft revision to Table 3.9-5 in which SSE was deleted from notes (13) and (14). The proposed revisions to Table 3.9-5 were subsequently incorporated into SSAR Revision 9 with revised notes (13) and (14) renumbered as notes (10) and (11). The staff found this acceptable and this part of Open Item 3.12.5.3-1 is closed.

In Revision 4 to Table 3.9-6 of the SSAR, Westinghouse provided additional load combinations and stress limits for Class 1 piping. Previous references to inelastic analysis methods and to reversing and non-reversing loads were deleted. This is acceptable. However, the staff found that the table did not include any ASME Code, Section III, Paragraph NB-3650, Equation (9) load combinations or primary stress limits for Design or Service Level A, B, C, or D conditions. In addition, the table gave two separate load combinations for Equation (13). One combination included RVOS while the other combination included DU which should include RVOS. Westinghouse needed to clarify this. In SSAR Revision 9, Table 3.9-9 which provides the ASME Class 1 stress criteria were revised to include references to the applicable ASME Code, Section III, Equation (9) for the design condition and for each service level condition. In addition, Westinghouse added a note to Table 3.9-9 which states that Table 3.9-6 includes additional stress limits for Class 1 piping. The staff reviewed this revision and concluded that all required Code equations for Class 1 piping are specified in Tables 3.9-6 and 3.9-9. With regard to the question on Table 3.9-6 Equation (13) load combinations, the clarification of the definitions of RVOS and DU in Table 3.9-3 discussed above resolved this issue. On the basis of this revision, this part of Open Item 3.12.5.3-1 is closed.

In Revision 4 to Table 3.9-7 of the SSAR, Westinghouse provided additional load combinations and stress limits for Class 2 and 3 piping. The staff review found that load combinations and stress limits for Design Condition Equation (8) and for Service Level A, B, C and D Equation (9) were not included. In SSAR Revision 9, Westinghouse revised Table 3.9-10 on Class 2 and 3 stress criteria to include references to the applicable ASME Code Equation (8) for the design condition and Equation (9) for each service level condition. Westinghouse also added a note to Table 3.9-10 which states that Table 3.9-7 includes additional stress limits for Class 2 and 3 piping. The staff reviewed this revision and concluded that all required Code equations for Class 2 and 3 piping are specified in Tables 3.9-7 and 3.9-10. On the basis of this revision, this part of Open Item 3.12.5.3-1 is closed.

In SSAR Revision 4, Table 3.9-11 provided the piping functional capability stress limits. The staff found that, similar to the above issue, the table did not include any Equation (9) Level D load combinations or stress limits. These must be included in the table. In addition, the table should include the restrictions from NUREG-1367, "Piping Functional Capability." In SSAR Revision 9, Table 3.9-11 was revised to include the acceptable Equation (9) Level D stress limits. However, the revised table still did not include the following three restrictions from NUREG-1367:

(1) The Equation (9) stress limit is applicable to reversing dynamic loads including fluid hammer pressure wave loads but not to slug-flow loads.

(2) Steady-state stresses shall be limited to $0.25S_y$.

(3) Dynamic moments must be calculated using an elastic response spectrum method with $\pm$15% broadening and with not more than 5-percent damping.

In a letter dated October 23, 1996, Westinghouse stated that their position is that the current Code limit on Service Level D assures the functional capability for all loads and analysis methods and that this is consistent with Westinghouse operating plants.

At the December 1996 design review meeting, the functional capability issue was further discussed. The staff reaffirmed the NRC position on meeting the NUREG-1367 restrictions and stated that for cases that do not meet the restrictions, earlier NRC-accepted functional capability criteria (such as Level C stress limits) may be used. Westinghouse did not want to use the more restrictive Level C limits and cited cases where the staff had accepted alternate limits for operating plants. The staff agreed that alternate stress limits would be acceptable if justified for AP600. In SSAR Revision 10, Westinghouse revised Table 3.9-11 to include alternate stress limits. The staff reviewed the alternate stress limits and found them unacceptable. In a letter dated February 20, 1997, the staff advised Westinghouse that the AP600 piping functional capability stress criteria should implement the guidelines given in Section 9 of NUREG-1367 with an additional clarification on the treatment of slug flow loads. In a letter dated March 13, 1997, Westinghouse agreed to meet the staff's position and provided a draft SSAR markup to show how the revised criteria will be implemented. Additional corrections to the SSAR markup were provided by Westinghouse on March 20, 1997. The staff reviewed the SSAR markup and found it consistent with the staff's position. The proposed changes in Table 3.9-11 were incorporated in Revision 12 to the SSAR. Therefore, DSER Open Item 3.12.5.3-1 is closed.

3.12.5.4 Damping Values

RG 1.61 contains recommended values of damping to be used in the seismic analysis of SSCs. In addition, RG 1.84 conditionally endorses ASME Code Case N-411-1. The damping values specified by Westinghouse for use in the AP600 design were those specified in RG 1.61 or ASME Code Case N-411, except for the primary coolant loop pipeline. For the primary coolant loop piping, a uniform 4 percent of critical damping, or ASME Code Case N-411 damping was specified in Table 3.7.1-1 of the SSAR, with however, only ASME Code Case N-411 damping specified for the seismic analysis of this piping in Appendix 3C of the SSAR. For piping systems analyzed with the ISM response spectrum method, the damping values were limited in Section 3.7.3.9 of the SSAR, to RG 1.61 damping values. Both uniform and modal composite definitions of damping were used.

In the response to RAI 210.79 dated July 27, 1994, Westinghouse provided proposed revisions to Sections 3.7.1.3 and 3.7.3.15, and Table 3.7.1-1 of the SSAR. In these revisions, Westinghouse proposed to use a uniform 5 percent damping for piping systems analyzed by the response spectrum method in lieu of the damping values in RG 1.61 or ASME Code Case N-411. For piping systems analyzed by time history analysis, RG 1.61 damping would be used, except for the primary coolant loop which would use 4-percent damping. As discussed in Section 3.12.5.19 of this report, the use of 5-percent damping is acceptable for piping systems in the AP600 subject to the same limitations as specified in RG 1.84 for Code Case N-411. However, the proposed SSAR revisions did not commit to these limitations. One of the limitations is on the use of certain analysis methods including the independent support motion response spectrum method without justification. In the DSER, the staff reported that Westinghouse should provide an amendment to the SSAR which commits to all of the limitations specified in RG 1.84 for Code Case N-411 for 5 percent damping or provide additional justification. This was identified as DSER Open Item 3.12.5.4-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse agreed to revise the SSAR to commit to all of the limitations specified in RG 1.84 for Code Case N-411 damping with regard to 5-percent damping. In Draft Revision 4 to SSAR Section 3.7, Westinghouse revised Section 3.7.3.15, "Analysis Procedure for Damping" and Table 3.7.1-1, "Safe Shutdown Earthquake Damping Values." The staff reviewed this information to ensure that Westinghouse committed to all of the applicable limitations specified in RG 1.84. The staff found that 5-percent damping would only be used for piping systems analyzed by the uniform envelope response spectrum method and would not be used for piping systems that are susceptible to stress corrosion cracking. Lower damping values were specified for systems analyzed by the time history or independent support motion response spectrum methods. These limitations are consistent with RG 1.84 and are acceptable. However, the staff noted that Westinghouse would apply the 5-percent damping to coupled equipment and valves as well as to the piping. This is inconsistent with RG 1.84 which states that for equipment other than piping, RG 1.61 damping should be used. In addition, an inconsistency was found in Table 3.7.1-1. The table specified 5-percent damping for the primary coolant loop (with no restriction on analysis method) and also an alternative 4-percent damping for the primary loop if time history or independent support motion response spectrum analysis is performed. The staff also reviewed Draft Revision 4 to Appendix 3C, "Reactor Coolant Loop Analysis Methods and Results," Section 3C.4, and found that for the reactor coolant loop analysis, Westinghouse would use either 5-percent damping when the uniform envelope response spectrum method is used or 4-percent when the independent support motion response spectrum method is used. The staff had earlier accepted the use of 4-percent damping for time history analysis of the reactor coolant loop on the basis of a Westinghouse study. However, the application of this damping value to an independent support motion analysis would require additional justification. The use of 5-percent damping for the coupled reactor coolant loop piping and equipment model is inconsistent with the RG 1.84 limitation described above. Thus, further Westinghouse action was needed to address the above concerns.

During the June 1996 design review meeting, Westinghouse provided a copy of topical report WCAP-7921-AR which described their study justifying the 4-percent damping value for the reactor coolant loop. The report recommended a value of 4-percent damping for reactor coolant loop components and large piping. The report included an NRC staff evaluation letter which stated that this damping value is acceptable and imposed no restriction on the analysis method. On this basis, the staff concludes that the use of 4-percent damping in an independent support motion analysis of the coupled reactor coolant loop system is acceptable.

In a letter dated October 23, 1996, Westinghouse agreed to revise the SSAR to reflect the staff position on damping for coupled systems which may include piping, valves, equipment or other structures. Westinghouse provided a proposed revision to SSAR Section 3.7.3.15 which stated that the composite modal damping approach with either the weighted mass or stiffness method will be used to determine the composite modal damping value for coupled models. Damping values for other structures and components were provided in Table 3.7.1-1. Alternately, the minimum damping value may be used for composite systems. In addition, a proposed revision to Table 3.7.1-1 deleted the use of 5-percent damping for the reactor coolant loop analyzed by the uniform envelope response spectrum method. These issues were further discussed during the December 1996 design review meeting at NRC headquarters. Consequently the staff found the proposed SSAR revisions are in conformance with SRP 3.7.3 and acceptable but asked Westinghouse to limit the composite modal damping method to the weighted stiffness method which is more appropriate for these types of composite systems. Westinghouse agreed to do this and all of the above changes were incorporated into SSAR Revision 10. On the basis of this additional information, the staff concludes that DSER Open Item 3.12.5.4-1 is closed.

3.12.5.5 Combination of Modal Responses

The total unidirectional seismic response for a system is obtained by combining the individual modal responses using the SRSS method. If modes are associated with frequencies that are closely spaced, this method is modified to reflect the possible interaction of the modes. Three alternate options were provided in Section 3.7.3.7.2 of the SSAR to combine closely spaced modes. These were the Westinghouse grouping method, the Westinghouse 10-percent grouping method and any method recommended in RG 1.92. The staff reviewed the Westinghouse grouping method and considered it to be comparable to those recommended in RG 1.92. This method is used in the PS+CAEPIPE computer code used to develop the seismic response for the sample analyses. In the DSER, the staff reported that the staff will verify its adequacy in the confirmatory evaluations of the sample piping problems that are discussed in Section 3.12.4.1 of this report.

The Westinghouse 10-percent grouping method is the same as the Westinghouse grouping method with the exception that with the 10-percent method, a given mode can be included in more than one group. Given this, the estimates of response developed with the 10-percent grouping method will be greater, or more conservative, than the estimates of response developed with the grouping method. In the DSER, the staff indicated that because the

Westinghouse 10-percent grouping method is more conservative than the Westinghouse grouping method, its adequacy will be assessed in the same confirmatory evaluations.

In the DSER, the staff stated that the options specified in Section 3.7.3.7.2 of the SSAR, to combine closely spaced modes, are acceptable as discussed above, contingent on a positive finding in the confirmatory evaluations. This was identified as DSER Confirmatory Item 3.12.5.5-1. In SSAR Revision 9, Westinghouse revised Section 3.7.3.7.2 to state that for piping systems, the methods in Regulatory Guide 1.92 are used in modal combinations. The other originally proposed alternate modal combination options were eliminated. On this basis, the staff concluded that confirmatory evaluation of those methods is no longer applicable and DSER Confirmatory Item 3.12.5.5-1 is closed.

3.12.5.6 High-Frequency Modes

For seismic analysis, consideration of high-frequency modes to preclude missing mass effects must be included. The staff's guidelines for this are provided in Appendix A of Section 3.7.2 of the SRP.

Five methods to account for the contribution of the high-frequency modes were presented in Section 3.7.3.7.1 of the SSAR. These are the residual load method (RLM), the full zero period acceleration method (FZPA), the residual load method for multiple response spectrum analysis (RLMM), the analytical method recommended in Appendix A to Section 3.7.2 of the SRP, and the 10-percent rule recommended in Appendix A to Section 3.7.2 of the SRP. Of these, the two recommended in the SRP are acceptable. In the DSER, the staff stated that for the RLM, FZPA, and RLMM methods, either a basis to verify their adequacy must be provided to the staff, or they must be deleted as calculational options, and that a description of all analysis options used to calculate the modal response of the high frequency modes must be provided in the SSAR. This was identified as DSER Open Item 3.12.5.6-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided a copy of Revision 2 to SSAR Section 3.7 dated March 31, 1995. In Section 3.7.3.7.1, the descriptions of the residual load method, the full zero period acceleration method, and the residual load method for multiple response spectrum analysis which had been included in the earlier SSAR revision were deleted. A description of an additional method, the left-out-force method, which is discussed in the next paragraph was also included. On the basis of this revision, DSER Open Item 3.12.5.6-1 is closed.

Another method to calculate the effect of high-frequency modes, the left-out force method, is incorporated in the computer code PS+CAEPIPE. Since this computer code was used to develop the seismic response in the sample piping analyses, the adequacy of the methods incorporated in it could be evaluated in the confirmatory evaluations performed by the staff. In the DSER, the staff reported that the left-out force method is acceptable contingent on a positive finding in the confirmatory evaluation. As a calculational option used for AP600 piping design, a description of this method must be provided in the SSAR. This was identified as DSER Open Item 3.12.5.6-2.

As noted above, a description of the left-out-force method was included in Section 3.7.3.7.1 of SSAR Revision 2. As discussed in Section 3.12.4.1 of this report, the final independent confirmatory piping stress analyses were completed in June 1996, and the comparison of results met the staff acceptance criteria as stated in NUREG/CR-6414. Therefore, on the basis of this positive finding, the staff concludes that the left-out-force method is acceptable and DSER Open Item 3.12.5.6-2 is closed.

3.12.5.7 Fatigue Evaluation for ASME Code Class 1 Piping

Section III of the ASME Code requires that the cumulative damage from fatigue be evaluated for all ASME Code Class 1 piping. The cumulative fatigue usage factor should take into consideration all cyclic effects caused by the plant operating transients for a 60-year design life. However, recent test data indicates that the effects of the reactor environment could reduce the fatigue resistance of certain materials. In the DSER, the staff indicated that until the ASME fatigue design curves are revised for these materials, the SSAR should discuss why the environmental effects in the fatigue analysis for these materials is not a concern for the 60-year life of the plant. In RAI 210.106, the staff requested that Westinghouse address this concern for the AP600 and propose revisions to the SSAR as necessary.

In the response to RAI 210.106 dated June 27, 1994, Westinghouse summarized it's understanding of the concern as being related to the use of carbon steel and low alloy steel in certain BWR environments. Westinghouse further observed that there was no concern regarding stainless steel in PWR or BWR environments, or carbon steel or low alloy steel in PWR environments.

Westinghouse concluded that the AP600 is a PWR and its environment is maintained accordingly. Therefore, until the staff completes its evaluation of this issue on a generic basis, concerns regarding environmental effects on fatigue are not an issue in the AP600. The Westinghouse response is acceptable and no revision of the SSAR is required.

3.12.5.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

The design life for the AP600 is 60 years. The staff raised a concern that the current ASME Code Class 2 and 3 rules for fatigue may be inadequate for some piping system components to assure their design life of 60 years. In the DSER, the staff stated that the SSAR should identify the ASME Code Class 2 and 3 components (e.g., feedwater and main steam nozzles on the steam generator) that have been, or will be, evaluated for cyclic effects, and describe the evaluation that has been, or will be, performed to verify their fatigue adequacy. This was identified as DSER Open Item 3.9.3.1-2 in this chapter. During a subsequent staff design review meeting at Westinghouse on July 26, 1995, the staff determined that the only Class 2 or 3 components subjected to these effects are the nozzles on the secondary side of the steam generators. The SSAR states that these components are designed to satisfy the criteria specified in Section III of the ASME Code for Class 1 components. Because the staff has determined that these criteria are acceptable for evaluating fatigue in the AP600 design, DSER Open Item 3.9.3.1-2 is closed. (See Section 3.9.3.1 above).

3.12.5.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System

In accordance with NRC Bulletin 88-08, the staff requests that licensees and applicants review systems connected to the RCS to determine whether any sections of this piping that cannot be isolated can be subjected to temperature oscillations that could be induced by leaking valves. Identification of systems which may be subjected to thermal cycling because of valve leakage, a description of the design provisions for minimizing these effects, and the stress and evaluation methodology used to assess the integrity of the system affected, should be provided in the SSAR.

The July 27, 1994, revision to the Westinghouse response to RAI 210.59 included proposed revisions to Section 3.9.3.1.2 of the SSAR, which addressed the issue in Bulletin 88-08. Regarding susceptible system identification, it stated that the unisolable portions of systems which are part of the reactor coolant pressure boundary are considered for susceptibility to valve leakage as described in NRC Bulletin 88-08. The methods from EPRI report "Thermal Stratification, Cycling and Striping (TASCS)," would be used to define isolation valve leakage transients, locate sites of thermal penetration cycling, determine numbers of leakage cycles, and calculate the thermal striping fatigue usage factors. Guidelines for the monitoring for thermal stratification, at locations where the calculated fatigue usage factor exceeds 1, were provided.

The TASCS report had not been provided to the staff, and the descriptions of the methodology provided in the SSAR were not sufficient to define them or to allow the staff to complete its evaluation of their adequacy. This observation applied to the identification of susceptible systems, the methods to define the thermal loading and the methods to calculate the effects of the thermal loads on affected systems. The guidelines for temperature monitoring of systems for which a fatigue usage factor greater than 1.0 is calculated, were comprehensive and acceptable.

In the DSER, the staff reported that an identification of systems susceptible to thermal cycling, or a description of the method used to make that identification, and a description of the analysis methods used to assess their integrity, must be provided in the SSAR. This was identified as DSER Open Item 3.12.5.9-1.

Westinghouse provided the EPRI TASCS Program Report TR-103581 to the staff for AP600 design certification review in June 1996. The report provides screening criteria for identifying portions of piping systems that may be susceptible to thermal cycling; analysis methods for defining thermal loads and numbers of cycles in the susceptible systems; and a summary of thermal hydraulic test programs and correlations to develop and verify the methods. The EPRI report had been submitted to NRC in support of Bulletin 88-08 programs for several operating plants. The staff review concludes that the methodology presented in the report is not acceptable. However, for AP600 design certification, the EPRI report was reviewed as background information only and the Westinghouse AP600 methodology described in the SSAR and in supporting calculations was evaluated on its own merits.

In SSAR Revision 9, Westinghouse revised Section 3.9.3.1.2 to provide additional information on the methods and criteria used to identify systems susceptible to adverse stresses because of thermal cycling. It identified a total of 11 lines which contain unisolable sections connected to the reactor coolant system. It stated that all 11 of these lines were reviewed and were found not susceptible to thermal stratification, cycling, and striping. During the design review meeting in December 1996, Westinghouse provided a copy of AP600 calculation GW-PLC-001 for staff review. This document provided a more detailed description of the piping systems review summarized in the SSAR. The calculation covered the Bulletin 88-08 valve leakage issue as well as other potential causes of thermal stratification and cycling in AP600 piping including those described in Bulletin 79-13 and Bulletin 88-11. The calculation identified the pressurizer surge line and the cold leg piping (under certain accident conditions) as lines susceptible to adverse stresses resulting from stratification. For these lines, the stratification loads were defined and considered in the stress evaluation. No lines were found susceptible to thermal cycling associated with valve leakage as described in Bulletin 88-08. The staff reviewed the calculation and associated reference material and concluded that the methodology for identifying susceptible systems was reasonable and acceptable. The methodology was on the basis of operating experience and considered the thermal stratification and thermal cycling mechanisms that have been identified to date. The Westinghouse review appeared to be thorough and complete in identifying systems susceptible to thermal stratification and cycling. However, there were three sections of piping which Westinghouse judged to be acceptable even though there was a high degree of uncertainty regarding their susceptibility to thermal stratification and cycling. They include: (1) ADS Stage 4 lines, (2) normal RHR suction line from the hot leg, and (3) PRHR return line. The calculation recommended additional confirmatory analysis and possibly confirmatory plant monitoring. The staff, however, did not agree that this should be treated as a confirmatory issue and asked that further evaluation of these systems be performed before the design certification to avoid the need for later redesign. Westinghouse agreed to provide a plan of action to resolve this issue.

In a letter dated December 16, 1996, Westinghouse provided their plan to address the uncertainties of the temperature profiles in the three lines. The plan involved the development of detailed axial and diametral temperature distributions using finite element fluid flow and heat transfer analysis methods. The resulting pipe metal temperature distributions would be used in additional pipe stress analyses. Detailed descriptions of the analyses and their results were provided in a Westinghouse letter dated March 13, 1997. The results of the fluid flow and heat transfer analyses showed that thermal stratification or cycling do not occur in the normal RHR suction line but would be expected to occur in the PRHR Return Line and in the ADS Stage 4 Lines. On the basis of these results, conservative stratified temperature profiles were applied in the pipe stress analyses and all lines were shown to satisfy the ASME Code stress limits. The Westinghouse letter also provided a markup of SSAR Section 3.9.3.1.2 which reflected the results of these analyses. The proposed changes were incorporated in Revision 12 to the SSAR. The staff finds this acceptable and DSER Open Item 3.12.5.9-1 is closed.

3.12.5.10 Thermal Stratification

Thermal stratification is a phenomenon which can occur in long runs of horizontal piping when two streams of fluid at different temperatures flow in separate layers without appreciable mixing. Under such stratified flow conditions, the top of the pipe may be at a much higher temperature than the bottom. This thermal gradient produces pipe deflections, support loads, pipe bending stresses, and local stresses.

The effects of thermal stratification have been observed in both BWR and PWR feedwater piping as discussed in NRC Information Notice (IN) 84-87 and NRC IN 91-38. NRC

Bulletin 88-11 was issued in response to the results of an inspection of the pressurizer surge line at the Trojan plant, which showed large, unexpected movements that closed the gaps between the line and pipe whip restraints. The movements were attributed to thermal stratification which occurred under certain operating conditions when large temperature differences existed between the RCS and the pressurizer. The bulletin requested all PWR licensees to establish and implement a program to assure the structural integrity of the surge line when subjected to thermal stratification. The structural reevaluation should consider the cyclic effects of the additional bending stresses in the pipe as well as the local stresses induced by thermal striping (rapid oscillation of the thermal boundary interface along the piping inside surface). The SSAR should identify piping systems which may be subjected to thermal stratification, as described in NRC Bulletin 88-11, and describe the design provisions for minimizing these effects, and the stress and fatigue evaluation methodology used to assess their impact.

The July 27, 1994, revision to the Westinghouse response to RAI 210.59 included proposed revisions to Section 3.9.3.1.2 of the SSAR, which addressed the thermal cycling and thermal stratification issues. The revision included a commitment to perform a plant-specific analysis of the AP600 surge line to demonstrate that all applicable requirements of Section III of the 1989 Edition of the ASME Code are met for the 60-year life. The analysis would include all loads, including thermal stratification and thermal striping. The commitment included the performance of a monitoring program, at the first AP600 plant, to record temperature distributions and thermal displacements of the surge line piping, and other pertinent parameters. The monitored data would be used to evaluate the analytical temperature distributions and displacements. The SSAR revisions did not specifically address further thermal stratifications.

In the DSER, the staff reported that the Westinghouse commitment for the pressurizer surge line is an acceptable method to address potential problems described in NRC Bulletin 88-11. The analysis confirms the design adequacy of the system while the monitoring program incorporates the assumptions used in the analysis. A treatment of thermal stratification in the broader sense, however, was not provided in the proposed revision. Other systems or locations susceptible to thermal stratification were not specified, nor were the methods used to determine the loads induced by stratification, described. For example, following a LOCA event, the slow injection of cold water from the PXS could result in severe stratification. These areas may be addressed in the TASCS report, however, as noted in Section 3.12.5.9 above, this report was not available to the staff. In the DSER, the staff stated that an identification of piping systems susceptible to thermal stratification, and a description of the methods used to assure their structural integrity, must be provided in the SSAR. This was identified as DSER Open Item 3.12.5.10-1.

In Revision 4 to SSAR Section 3.9.3.1.2, Westinghouse committed to performing an analysis of the pressurizer surge line to demonstrate that the applicable requirements of the ASME Section III Code are met. The analysis would include consideration of plant operation, thermal stratification and thermal striping, using temperature distributions and transients developed from experience on existing plant monitoring programs. A monitoring program to record temperature distributions and displacements of the pressurizer surge line will be implemented at the first AP600 plant. The measured data will be evaluated to demonstrate that it is within the bounds of the analytical temperature distributions and displacements. As noted above, the staff finds this commitment acceptable for addressing the concerns described in NRC Bulletin 88-11. In

SSAR Revision 4, Section 3.9.3.1.2, Westinghouse also made a commitment to address the feedwater line cracking issue described in NRC Bulletin 79-13. The feedwater line analyses will consider thermal stratification, thermal cycling and thermal striping using temperature distributions and transients developed from experience gained in resolving the feedwater line cracking issue. The staff finds this commitment acceptable for addressing the concerns described in NRC Bulletin 79-13.

As discussed in Section 3.12.5.9 of this report, during the December 1996 design review meeting at NRC, the staff reviewed AP600 calculation GW-PLC-001 which addressed the broader thermal stratification issue. It included the issues described in Bulletins 88-08, 88-11 and 79-13 as well as other thermal stratification and cycling potential concerns specific to the AP600 plant. The report documented the review of cold leg piping thermal stratification which may occur during certain LOCA scenarios and provides the worst case thermal loads for stress evaluation. In Revision 9 to Section 3.9.3.1.2 of the SSAR, Westinghouse describes this event and states that it is evaluated as a design transient using ASME Code Level B and Level D service condition stress criteria. On the basis of its review of the Westinghouse calculation and the SSAR revision, the staff concluded that Westinghouse adequately addressed the broader issue of thermal stratification for the AP600 plant. On this basis, DSER Open Item 3.12.5.10-1 was considered closed. However, in a letter dated March 13, 1997, Westinghouse provided a markup of SSAR Section 3.9.3.1.2 which deleted the original COL commitment for implementing a monitoring program to verify the temperature distributions and thermal displacements of the pressurizer surge line as discussed above. The staff finds this unacceptable without additional justification. Subsequently, the discrepancy was corrected in a SSAR markup dated March 27, 1997. The proposed changes were subsequently incorporated in Revision 12 to the SSAR. Thus, Open Item 3.12.5.10-1 is closed.

3.12.5.11 Safety-Relief Valve Design, Installation, and Testing

Section 3.9.3.3 of the SSAR contains the design and installation criteria applicable to the mounting of pressure relief devices used for the over-pressure protection of ASME Code Class 1, 2, and 3 components. The staff reviewed this information in accordance with Section 3.9.3 of the SRP, including an evaluation of the applicable loading combinations and stress criteria. The review extended to consideration of the means to accommodate the rapidly applied reaction force when a safety valve or relief valve opens and the transient fluid-induced loads are applied to the piping downstream of a safety valve, or relief valve, in a closed discharge piping system.

The information provided in Section 3.9.3.3 of the SSAR adequately defines the Westinghouse criteria for the design and evaluation of safety-relief valves in the AP600 plant. Not only pressure relief valves, but also the valves in the automatic depressurization system attached to the pressurizer, are designed and installed to this criteria. From the SSAR descriptions, however, it was not apparent that the design and analysis requirements are in compliance with Appendix 0 of Section III of the ASME Code and the additional criteria given in Section 3.9.3 of the SRP. This concern was noted at a staff design review meeting with Westinghouse and raised formally as one aspect of RAI 210.67. Westinghouse provided a proposed revision to Section 3.9.3.3 of the SSAR to address this concern in its July 8, 1994, response to RAI 210.67. The proposed revision includes the clear commitment to comply with Appendix 0 of Section III of the ASME Code. The proposed revision is acceptable. In the DSER, the staff reported that contingent upon Westinghouse providing an amendment to the SSAR reflecting this revision, this item is considered closed. The proposed revision was incorporated in SSAR Revision 3. (See Section 3.9.3.2 of this report.)

3.12.5.12 Functional Capability

All ASME Code Class 1, 2, and 3 piping systems that are essential for safe shutdown must retain their functional capability for all Service Level D loading conditions. Designs meeting the recommendations in NUREG-1367, "Functional Capability of Piping Systems," are accepted by the staff as satisfying the functional capability requirements.

No criteria to establish the functional capability of ASME Class 2 and 3 components were provided in Revision 0 of the SSAR. Westinghouse was requested, at a staff design review meeting at the Westinghouse offices, and formally in RAI 210.65, to include a commitment to the criteria in NUREG-1367 for all seismic Category I piping systems.

At the July 1994, staff design review meeting, Westinghouse provided a preliminary response to RAI 210.79 to address this issue. In their response, Westinghouse proposed several SSAR revisions, including Section 3.9.3.1.3 and Table 3.9-11 of the SSAR. On the basis of preliminary staff review and discussions at the design review meeting, Westinghouse further revised its response and submitted it to the NRC in a letter dated July 25, 1994. In the revised Table 3.9-11, Westinghouse provided functional capability requirements for ASME piping systems that must maintain an adequate fluid flow path to mitigate a Level C or Level D plant event.

In its review of the Westinghouse submittal, the staff found the Table 3.9-11 criteria to be consistent with some, but not all of the NUREG-1367 recommendations. In NUREG-1367, the staff concluded that piping functional capability is ensured by meeting the requirements of the ASME Code (1989 edition) provided that the following conditions are met:

- Dynamic loads are reversing. This includes loads as a result of earthquakes, building filtered loads, and pressure wave loads (not slug-flow fluid hammer).
- Dynamic moments are calculated using an elastic response spectrum analysis with ± 15-percent peak broadening and with not more than 5-percent damping.
- Steady-state stresses do not exceed 0.25 Sy.
- Do/t does not exceed 50.
- External pressure does not exceed internal pressure.

In the proposed Table 3.9-11, Westinghouse applied the Class 1, Equation 9, Level D stress limit ($3S_m$, but not greater than 2 Sy) to Class 1, 2, and 3 piping systems. The staff found that there was no basis for applying Class 1 stress limits to Class 2 and 3 piping systems. The table was requested to be revised to be consistent with the Class 2 and 3, Equation 9, Level D stress limit ($3\ S_h$, but not greater than 2 Sy) for Class 2 and 3 piping systems. (See Section 3.12.5.19 of this report.)

In proposed Table 3.9-11 of the SSAR, Westinghouse also included load combinations and stress limits for Class 2 and 3 Equation 10a (single non-repeated anchor movement). The inclusion of this stress limit in this table and its applicability to functional capability of Class 1, 2, and 3 piping was requested to be clarified. The staff position requires all current (1989) Code requirements to be met to ensure functional capability.

In NUREG-1367, the NRC restrictions on wall thickness, external pressure and steady state stress limits were included in proposed Table 3.9-11 of the SSAR. Westinghouse did not, however, include the restriction on the analysis method listed above. The restriction on reversing dynamic loads was not clearly followed. There were several load combinations that included non-reversing dynamic loads. Some load combinations included non-reversing dynamic loads combined with SSE by the square root of the sum of squares method. In addition, if the non-reversing dynamic loads included sustained loads, the resulting stresses should have been combined with weight stresses to meet the steady state stress limit of 0.25 Sy.

Westinghouse was requested to provide another SSAR revision that addresses the above issues and is consistent with the staff position as described in NUREG-1367. This was identified as DSER Open Item 3.12.5.12-1.

In SSAR Revision 4, Westinghouse revised Table 3.9-11 which provided the piping functional capability stress limits. As discussed in Section 3.12.5.3 above, the staff found several significant omissions in this revision. The table did not include any Equation (9) Level D load combinations or stress limits. These load combinations and stress limits should be provided for both Class 1, 2, and 3 piping systems. The table also did not include the following additional restrictions from NUREG-1367:

- Steady state stresses shall not exceed $0.25S_y$,
- Dynamic loads must be reversing, and
- Dynamic moments must be calculated using an elastic response spectrum method with $\pm$15-percent peak broadening and with not more than 5-percent damping.

Also, as noted above, Westinghouse included load combinations and stress limits for Class 2 and 3 Equation 10a (single non-repeated anchor movement). The inclusion of these stress limits and their applicability to functional capability of piping should be clarified.

During the staff's design review meeting conducted at Westinghouse offices in June 1996, Westinghouse explained that the additional load combinations and stress limits for Class 2 and 3 Equation 10a (single non-repeated anchor movement) were included to provide stress limits for thermal expansion and steel containment vessel anchor motions under service level C and D events for piping systems that must maintain an adequate fluid flow path. These limits were imposed since there is no ASME Code stress limit for thermal expansion for Level C and D conditions. In SSAR Revision 5, Westinghouse also included stress limits for Class 1 piping for these same load combinations in Table 3.9-11. The staff reviewed these additional load combinations and stress limits and found them acceptable because they meet the intent of ASME Code and functional capability criteria in NUREG-1367.

In Revision 9 to the SSAR, Westinghouse revised Table 3.9-11 to incorporate the appropriate Equation (9) Level D stress limits for Class 1, 2, and 3 piping systems. However, the additional restrictions from NUREG-1367 described above were still not included. As discussed in Section 3.12.5.3 of this report, at the December 1996 design review meeting, the functional capability issue was further discussed. Westinghouse stated that they had applied alternate functional capability stress limits for their operating plants and would like to apply those alternate limits for cases where the NUREG-1367 restrictions are not met. The staff agreed that alternate stress limits may be acceptable for AP600 if adequately justified. In SSAR Revision 10, Westinghouse revised Table 3.9-11 to include alternate stress limits. The staff reviewed the alternate stress limits but found them unacceptable. In a letter dated February 20, 1997, the staff advised Westinghouse that the AP600 piping functional capability stress criteria should implement the criteria given in Section 9 of NUREG-1367 with an additional clarification on the treatment of slug flow loads. In a letter dated March 13, 1997, Westinghouse agreed to meet the staff position and provided a draft SSAR markup to show how the revised criteria will be implemented. Additional corrections to the SSAR markup were provided by Westinghouse on March 20, 1997. The staff reviewed the SSAR markup and found it consistent with the staff position. The proposed changes to Table 3.9-11 were subsequently incorporated in Revision 12 to the SSAR. Therefore, DSER Open Item 3.12.5.12-1 is closed.

3.12.5.13 Combination of Inertial and Seismic Anchor Motion Effects

Piping analyses must include the effects of relative building movements at supports and anchors (seismic anchor motion) as well as the seismic inertial loads. This is necessary when piping is supported at multiple locations within a single structure or is attached to separate structures.

As specified in Section 3.9.2 of the SRP, the effects of relative displacements at support points must be considered by imposing the maximum support displacements in the most unfavorable combination. This can be performed using a static analysis procedure. Relative displacements of equipment supports (e.g., pumps or tanks) must be included in the analysis along with the building support movements.

When required for certain evaluations, such as support design, the responses that are the result of the inertia effect and relative displacement effect should be combined by the absolute sum method. In lieu of this method, time histories of support excitations may be used, in which case both inertial and relative displacement effects are already included.

Revision 0 of Table 3.9-8 of the SSAR showed the seismic inertia, seismic anchor movement, and support self weight loads, combined by the SRSS method for Service Level D evaluations of ASME Class 1, 2, and 3 piping and component supports. At the April 12 through 14, 1994, design review meeting at the Westinghouse offices, Westinghouse was requested to provide a basis for this combination, which provides lower estimates of response then the accepted absolute sum method. This issue was again discussed at the July 19 through 21, 1994, design review meeting, at which time Westinghouse agreed to comply with the absolute-sum method. In the July 8, 1994, response to RAI 210.32, Westinghouse agreed to revise Section 3.7.3.9 of the SSAR to state that the results of the modal spectra analysis (multiple input or envelope) are combined with the results from seismic anchor motion by the absolute sum method. In addition, Westinghouse agreed to revise the exception to Section II.2.g of Section 3.9.2 of the

SRP in the next revision to WCAP-13054. In the July 27, 1994, response to RAI 210.79, Westinghouse proposed a revision to Table 3.9-8 of the SSAR, which shows the seismic inertia, seismic anchor movement and support self weight loads, for Service Level D, combined by the absolute sum method. In the DSER, the staff reported that contingent on Westinghouse providing amendments to the SSAR and WCAP-13054 to reflect the proposed revisions, this item was considered closed. This was identified as DSER Confirmatory Item 3.12.5.13-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided Revision 2 to SSAR Section 3.7.3.9. This revision included the statement that the results of the modal spectra analysis (multiple input or envelope) are combined with the results from seismic anchor motion by the absolute sum method. On June 30, 1995, Westinghouse provided Revision 4 to SSAR Section 3.9 which included revised Table 3.9-8 which specifies that for Level D load combinations, loads because of seismic inertia, seismic anchor motions, and seismic support self weight excitation are combined by the absolute sum method. On September 5, 1996, Westinghouse provided Revision 2 to WCAP-13054 which also included the statement that the results of the modal spectra analysis are combined with the results from seismic anchor motion by the absolute sum method. The staff finds this acceptable and DSER Confirmatory Item 3.12.5.13-1 is closed.

3.12.5.14 OBE as a Design Load

As discussed in Section 3.1.1 of this report, a new Appendix S to 10 CFR Part 50, "Earthquake Engineering Criteria for Nuclear Power Plants," was published in the Federal Register on December 11, 1996. Appendix S, in part, allows the elimination of the OBE as a design-basis event. The AP600 design has incorporated the single-earthquake design approach. In a letter to Westinghouse dated April 29, 1994, the staff transmitted an enclosure to RAI 210.60 that contained the staff's position relative to the types of analyses and information required in the SSAR, for the staff to approve design of safety-related SSCs without the OBE. This document included specific supplemental criteria for fatigue, seismic anchor motion, and piping stress limits, that should be applied when the OBE is eliminated. For fatigue evaluation, two SSE events with 10 maximum stress cycles per event (or an equivalent number of fractional cycles) should be considered. The effects of SAM, as a result of the SSE, should be considered in combination with the effects of other normal operational loadings that might concurrently occur. For the Class 1 primary stress evaluation, seismic loads need not be evaluated for consideration of Level B Service Limits for Eq. (9). However, for satisfaction of primary plus secondary stress range limits in Eq. (10), the full SSE stress range or a reduced range corresponding to an equivalent number of fractional cycles, must be included for Level B Service limits. These load sets should also be used for evaluating fatigue effects. In addition, the stress because of the larger of the full range of SSE anchor motion, or the resultant range of thermal expansion plus half the SSE anchor motion range, must not exceed 6.0 S_m. For Class 2 and 3 piping, seismic loads are not required for consideration of occasional loads in satisfying the Level B Service Limits for Eq. (9). Seismic anchor motion stresses are not required for consideration of secondary stresses in Eq. (10). However, stresses as a result of the combination of range of moments caused by thermal expansion and SSE anchor motions must not exceed 3.0 S_h.

It was stated in Section 3.7 of the SSAR that the OBE has been eliminated as a design requirement for the AP600. A description of design methods, the loads considered, and the analyses performed for ASME Class 1, 2, and 3 piping, was provided in Revision 0 of

Section 3.9.3.1.5 of the SSAR. The loads considered in the qualification of the piping were listed in Tables 3.9-3 and 3.9-4 of the SSAR, the load combinations in Tables 3.9-6 and 3.9-7 of the SSAR, and the corresponding stress limits in Tables 3.9-9, 3.9-10 and 3.9-11 of the SSAR. The staff's review of the referenced tables initially found that they did not adequately reflect the requirements of the staff position on single-earthquake design. At the April 12 through 14, 1994, design review meeting at the Westinghouse offices, Westinghouse was requested to revise the referenced tables to meet the staff position on single earthquake design. A formal submittal of the Westinghouse response to RAI 210.79, dated July 25, 1994, included proposed revisions to the SSAR, including referenced tables developed to reflect the staff position on single earthquake design, as well as other issues. In the proposed revision to Section 3.7.3.2 of the SSAR, Westinghouse stated that for ASME Class 1 piping, the fatigue evaluation is performed on the basis of five seismic events with an amplitude equal to one-third of the SSE response. Each event has 63 high stress cycles to provide the equivalent fatigue damage of two SSE events with ten high stress cycles per event on the basis of IEEE-344-1987. This is consistent with the staff position. This proposed revision to Section 3.7.3.2 was subsequently included in Revision 2 to SSAR Section 3.7 dated March 31, 1995. The staff reviewed this revised section and found it acceptable.

Proposed revisions to load combinations and stress limits to ASME Code 1, 2, and 3 piping were included in revisions to Tables 3.9-6 and 3.9-7 of the SSAR. For Class 1 piping, revised Table 3.9-6 did not require consideration of seismic loads for Level B primary stress evaluation in Equation (9). For primary plus secondary stress evaluation in Equation (10) and for fatigue evaluation in Equation (11) or (14), consideration of the inertia and anchor motion portion of one-third SSE was considered. Under Level D, secondary stresses because of the larger of the full range of SSE anchor motion or the resultant range of thermal expansion plus half the SSE anchor motion range must not exceed 6 S_m. These stress requirements are consistent with the staff position. The staff reviewed revised Table 3.9-6 of Revision 4 to SSAR Section 3.9 which was submitted on June 30, 1995, and determined that Westinghouse included the proposed acceptable changes for Class 1 piping load combinations and stress limits with regard to Equations (10), (11), (14), and Level D secondary stresses as described above. However, as discussed in Section 3.12.5.3 above, Westinghouse deleted all Equation (9) load combinations and stress limits from the table. This issue was subsequently resolved through an additional revision to Table 3.9-9 which referenced Equation (9) in SSAR Revision 9 as discussed in Section 3.12.5.3.

For Class 2 and 3 piping, revised Table 3.9-7 did not require consideration of seismic loads as occasional loads in satisfying the Level B Service Limits for Equation (9). Seismic anchor motion stresses were not required in satisfying Equation (11). Under Level D, stresses because of the larger of the full range of SSE anchor motion or the range of thermal expansion plus half the SSE anchor motion range must not exceed 3 S_h. These stress requirements are consistent with the staff position. In reviewing Table 3.9-7 in SSAR Revision 4, the staff noted that Westinghouse included the proposed acceptable changes for Class 2 and 3 piping described above except for the deletion of Equation (9) load combinations and stress limits as discussed in Section 3.12.5.3 above. This issue was subsequently resolved through an additional revision to Table 3.9-10 which referenced Equations (8) and (9) in SSAR Revision 9 as discussed in Section 3.12.5.3.

In addition to the changes and additions to the ASME Code stress criteria required by the staff for elimination of OBE, the staff position requires ASME Code Class 1, 2, and 3 piping to meet all other design requirements of the 1989 Edition of the code. In the revised Tables 3.9-6 and 3.9-7 of the SSAR, Westinghouse included piping design criteria on the basis of proposed changes to Section NB-3600, NC-3600, and ND-3600 of the code. These changes included increased allowable stress limits for dynamic loads. The staff evaluation of the alternate piping design criteria is provided in Section 3.12.5.19 of this report. During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse advised the staff that the proposed higher stress limits associated with the alternate piping criteria will not be used for AP600. Tables 3.9-6 and 3.9-7 in SSAR Revision 4 were extensively revised and references to the alternate piping design criteria were eliminated. However, certain load combinations and stress limits that should be included in these tables were also deleted. This issue was subsequently resolved on the basis of additional changes in SSAR Revision 9. This is discussed in Section 3.12.5.3 above.

With regard to the evaluation of the supplemental criteria for fatigue, seismic anchor motion, and additional piping stress limits that should be applied when OBE is eliminated, the staff found the proposed revisions to the SSAR provided in the response to RAI 210.79 consistent with the staff position on OBE elimination. This issue was considered confirmatory contingent upon Westinghouse providing an amendment to the SSAR reflecting the proposed revision. This was identified as DSER Confirmatory Item 3.12.5.14-1. As discussed above, the staff review of Revision 2 to SSAR Section 3.7 and Revision 4 to SSAR Section 3.9 concluded that Westinghouse incorporated appropriate supplemental criteria consistent with the staff position on OBE elimination. Thus, DSER Confirmatory Item 3.12.5.14-1 is closed.

3.12.5.15 Welded Attachments

Support members, connections, or attachments welded to piping, should be designed such that their failure under unanticipated loads does not cause failure in the pipe pressure boundary. The integrity of welded attachments should be assessed using methods acceptable to the staff.

The design of welded attachments to piping for the AP600 was not described in the SSAR. At the April 12 through 14, 1994 design review meeting at the Westinghouse offices, Westinghouse was requested to include in the SSAR a description of the analysis methods and criteria for the design of welded attachments to piping. Following discussions, it was agreed that a listing of the ASME Code Cases that would be used for this design purpose, provided in the SSAR, would satisfy this request.

A listing of the ASME Code Cases to be used in the AP600 plant design was included in the June 30, 1994, response to RAI 210.109. The listing was in Table 5.2-3 of the SSAR, a new table proposed for inclusion in the SSAR in response to the RAI. The listing included the code cases pertinent to the design of welded attachments. These include ASME Code Cases N-122-1, N-318-4, N-391-1, and N-392-2. Code Cases N-318-4 and N-391-1 were conditionally endorsed by the staff in RG 1.84, dated April 1992. Code Cases N-122-1 and N-392-2 had not yet been endorsed by the staff; an issue that was being addressed under an open item in Section 5.2.1 of this report. In the DSER, the staff reported that pending satisfactory resolution of the open item in Section 5.2.1 of this report, the proposed inclusion of Table 5.2-3 in the SSAR, which includes the Code Cases for welded attachment design, provides an acceptable basis for the design of welded attachments to piping. This item was considered confirmatory contingent on Westinghouse providing an amendment to the SSAR reflecting the proposed revision and satisfactory resolution to DSER Open Item 5.2.1.1-1. This was identified as DSER Confirmatory Item 3.12.5.15-1. As discussed in Sections 3.12.2.2 and 5.2.1 of this report, SSAR Revision 3 included a revised Table 5.2-3 that was acceptable to the staff and closed DSER Open Item 5.2.1.2-1. On this basis, DSER Confirmatory Item 3.12.5.15-1 is closed.

3.12.5.16 Modal Damping for Composite Structures

For the analysis of structures or components composed of elements exhibiting different values of damping, a composite modal definition of damping will be used in the AP600 design. For piping, the method will be used for both systems composed of piping and building elements, and for systems with different types or sizes of pipe.

At the April 12 through 14, 1994, and the July 19 through 21, 1994, design review meetings at the Westinghouse offices, the application of composite modal damping to AP600 piping was reviewed. Westinghouse stated that composite modal damping is calculated using the strain energy method described in Section 3.7.1.3 of the SSAR. In its application to systems comprised only of piping, it would be used to account for the variation of damping with pipe size. To complete its review, the staff requested Westinghouse to provide an example of its application to a piping or building system. Westinghouse responded it would provide a RCL analysis, including the damping, for each mode and mode shape. The staff indicated that it would review the sample analysis when it became available. This was identified as DSER Open Item 3.12.5.16-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided the Reactor Coolant Loop seismic analysis for staff review. The analytical model included the RCL piping and components as well as the reactor building internals. The strain energy method was used to calculate composite modal damping for each mode on the basis of damping values of 4 percent for the piping and components and 5 percent for the building structures. The staff reviewed the calculated composite damping values for selected modes and found them to be reasonable and consistent with the mode shapes. The staff concluded that the use of composite modal damping was acceptable for the RCL analysis. Additional staff evaluation of issues related to composite modal damping is provided in Section 3.12.5.4 of this report. On the basis of the staff review and evaluation of the RCL analysis, DSER Open Item 3.12.5.16-1 is closed.

3.12.5.17 Minimum Temperature for Thermal Analyses

In Section 3.9.3.1 of the SSAR, it is stated that thermal expansion is one of the loads that may be included in the load combinations for the ASME Code evaluations. In Tables 3.9-6, 3.9-7 and 3.9-8 of the SSAR, thermal expansion loadings are included in the load case combinations for the evaluation of Class 1, 2, and 3 piping and supports. This was the only information regarding thermal expansion analysis provided in the SSAR.

At the April 12-14, 1994 design review meeting at the Westinghouse offices, Westinghouse was advised that the information provided in the SSAR, regarding thermal expansion analysis, was not adequate. A comprehensive description of all analysis methods, including the methods to perform thermal expansion analysis, must be included in the SSAR. (See Section 3.12.3.above)

During the April 12 through 14, 1994, design review meeting, Westinghouse was requested to specify the AP600 criteria for the minimum temperature for thermal analyses. Westinghouse responded that the minimum temperature is 65.6 °C (150 °F). This value is consistent with industry practice and is acceptable to the staff. In the DSER, the staff reported that the specification of minimum temperature must be included in the description of thermal analysis methods to be provided in the SSAR. This was identified as DSER Open Item 3.12.5.17-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided a written description of the criteria for thermal analysis. It included the minimum temperature for which a thermal analysis would be required (65.6 °C [150 °F]) as well as other criteria for the thermal design and analysis of piping systems. The same description was included in Revision 4 to SSAR Section 3.9.3.1.5. The staff reviewed this information and found it acceptable. Thus, DSER Open Item 3.12.5.17-1 is closed.

3.12.5.18 Intersystem LOCA

In SECY-90-016, dated January 12, 1990, the NRC staff discussed the resolution of the ISLOCA issue for ALWR plants by requiring that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its SRM dated June 26, 1990, the Commission approved these staff recommendations provided that all elements of the low-pressure systems are considered.

The ISLOCA issue is addressed in Section 1.9.5.1 of the SSAR. In the response to RAI 210.61 dated June 16, 1994, Westinghouse provided proposed revisions to the SSAR which further clarify the design criteria for low-pressure side piping and components used in order to reduce the likelihood of ISLOCA. The RHR system was identified as the only system susceptible to overpressurization. In this system, schedule 80S piping and Class 900 valves, flanges, and fittings are specified to safely survive the possible overpressurization. The staff concluded that the SSAR revisions are acceptable, but must be expanded to assure that the ratio of downstream piping system pressure to RCS pressure is 0.4 or less. In the DSER, the staff reported that the ISLOCA issue will remain open pending the submittal of a SSAR revision. In SSAR Revision 7, Westinghouse added a new paragraph in Section 1.9.5.1 which references SSAR Section 5.4.7 for design features which address intersystem LOCA for the normal RHR system. The design criteria in SSAR Section 5.4.7.2.2 agree with all of the criteria listed above and are acceptable for the normal RHR system as well as for any other systems for which the ISLOCA issue would apply. (See Section 3.9.3.1 of this report for a more detailed discussion).

3.12.5.19 Alternate Piping Design Criteria

As a result of a design review meeting of the AP600 piping design criteria conducted at the Westinghouse offices in Monroeville, Pennsylvania on July 19-21, 1994, the NRC staff developed a better understanding of the proposed AP600 alternate piping design criteria and their implications on the AP600 design. During the design review meeting, the staff discussed Westinghouse's preliminary response to RAI 210.79, and established interim positions that would be appropriate for the AP600. In its preliminary response to RAI 210.79, Westinghouse revised the text and several tables from the SSAR to reflect the use of alternate piping design criteria proposed for the AP600. Subsequently, Westinghouse revised its response to RAI 210.79 and submitted it to the NRC in a letter dated July 25, 1994 (herein referred to as the "docketed response"). The staff completed its review of the AP600 alternate piping design criteria and reached a conclusion that the alternate piping design criteria are best addressed in an agency-wide position on the proposed changes to the piping design criteria in Subsection III of the ASME Code. The staff's evaluation of the AP600 alternate design criteria was transmitted to Westinghouse in a letter from R. Borchardt to N. Liparulo dated August 12, 1994. The following section provides the details of the staff's evaluation of the proposed alternate piping design criteria as they specifically apply to the safety related piping systems in the AP600 plant. During the staff design review meeting conducted at Westinghouse offices on April 10-11, 1995, Westinghouse informed the staff that it has decided to withdraw its plans to use the proposed alternate piping design criteria for AP600. As a result, many portions of the following section are no longer applicable and will be identified as such.

The AP600 alternate piping design criteria were on the basis of proposed changes to Subarticles NB-3600, NC-3600, and ND-3600 of Section III of the ASME Code for Code Class 1, 2, and 3 piping systems, respectively. However, these changes had not yet been incorporated into a formal Edition or Addendum to the ASME Code. In 10 CFR 50.55a, the NRC endorses Section III of the ASME Code, including addenda through 1988 and editions through the 1989 Edition. Accordingly, the staff considered these alternate criteria to be proposed alternatives to the requirements of the ASME Code. In the DSER, the staff indicated that as proposed alternatives, Westinghouse must demonstrate that an acceptable level of quality or safety exists or compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. This was identified as DSER Open Item 3.12.5.19-1. As indicated above, during the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse advised the staff that it no longer plans to use the proposed alternate piping design criteria for AP600. Therefore, DSER Open Item 3.12.5.19-1 is no longer an issue and is closed.

In the staff's review of the technical merits of the alternate piping design criteria, the staff found that there were seven major criteria where Westinghouse proposed alternative criteria are different from those in the 1989 Edition of Section III of the ASME Code. These seven criteria are as follows:

(1) increased piping allowable stress limits
(2) separate treatment of reversing dynamic loads
(3) 5 percent damping for piping systems
(4) use of Class 1 allowable values in Classes 2 and 3 piping design
(5) stress limit for SSE anchor motions
(6) stress limit for longitudinal piping stress
(7) reversing and non-reversing dynamic loads

The staff's evaluation of the above seven criteria was discussed in the DSER. The following is a summary of the staff's evaluation of these criteria.

Increased Piping Allowable Stress Limit

During the July 19 through 22, 1994, design review meeting, the staff discussed the use of higher stress limits proposed by Westinghouse for Service Levels C and D. Westinghouse proposed to use an allowable stress limit of $4.5S_m$ for Level D and $3.15S_m$ for Level C for Code Classes 1, 2, and 3 piping systems. The higher stress limits were provided in Westinghouse's response to RAI 210.79 as indicated in its revised Table 3.9-6 of the SSAR for ASME Code Class 1 piping and in Table 3.9-7 of the SSAR for ASME Code Classes 2 and 3 piping. However, it was also noted that in revised Table 3.9-11 of the SSAR, Westinghouse committed to meet, for essential safety systems*, more restrictive stress limits for ensuring piping functional capability. The staff guidelines for assuring piping functional capability are discussed in NUREG-1367. These stress limits would restrict the essential piping systems to current ASME Code Service Level D stress limits ($3.0S_m$, not to exceed $2S_y$ for Code Class 1 piping; and $3.0S_h$, not to exceed $2S_y$ for Code Classes 2 and 3 piping). In this manner, the proposed higher stress limit ($4.5S_m$) would not apply to these essential systems because the stress limits for assuring functional capability would be more limiting. However, the DSER stated that Westinghouse should revise its functional capability stress limits in Table 3.9-11 of the SSAR to be consistent with the NRC staff's recommendation for Code Classes 2 and 3 piping (i.e., use S_h instead of S_m). This was identified as DSER Open Item 3.12.5.19-2. In Revision 9 to SSAR Section 3.9, Westinghouse provided a revised Table 3.9-11. This revised table included the appropriate Level D stress limits for Class 1 and for Class 2 and 3 piping in accordance with NUREG-1367. On the basis of this SSAR revision, DSER Open Item 3.12.5.19-2 is closed. However, the staff review found that Westinghouse had not committed to all of the restrictions on these stress limits given in NUREG-1367. The staff evaluation of this issue is discussed in Sections 3.12.5.3 and 3.12.5.12 above. The use of the alternate piping design criteria would have had limited applicability to only those ASME Code Class piping systems that are not required to function in order to achieve a safe plant shutdown (e.g., main steam, feedwater, normal residual heat removal, CVSs). For these non-essential piping systems, Westinghouse would have needed to demonstrate that an adequate margin to failure exists in the use of the stress limits higher than those in the current Section III of the ASME Code (e.g., 1989 Edition). This was identified as DSER Open Item 3.12.5.19-3. As indicated above, during the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse informed the staff that it has decided to withdraw its plans to use the proposed alternate piping design criteria for AP600. Therefore, this is no longer an issue and DSER Open Item 3.12.5.19-3 is closed.

One concern identified by the NRC staff was the difficulty in evaluating the adequacy of the higher stress limits on their own merits without observing how the increased limits affect the overall piping design and analysis assumptions. The staff discussed a possible approach to review the results of actual AP600 piping system analyses in order to evaluate the impact of the higher stress limits on the piping design and to assess the degree to which the higher stress limits improve plant safety. Appropriate acceptance criteria for assuring system and component operability (e.g., deflections at motor-operated valves, branch connections, and weight supports) would have to be established as a part of the certified design. Westinghouse

* "Essential" piping systems are used herein to designate those systems in a passive advanced light water reactor design that are required to achieve a safe plant shutdown (e.g., passive safety injection and passive residual heat removal systems).

estimated that it would take approximately 6 to 12 months to complete the initial piping analyses. In the DSER, the staff indicated that using the results of the initial piping analyses, Westinghouse must demonstrate that an acceptable level of quality and safety exists for those non-essential piping systems using the increased stress limits. This was identified as DSER Open Item 3.12.5.19-4. Since Westinghouse has decided to withdraw its plans for using the proposed alternate piping design criteria, this is no longer an issue and DSER Open Item 3.12.5.19-4 is closed.

Separate Treatment of Reversing Dynamic Loads

In its preliminary response to RAI 210.79, Westinghouse revised Tables 3.9-6 and 3.9-7 of the SSAR to include separate treatment of reversing dynamic loads in Service Level B for Code Classes 1, 2, and 3 piping systems. The staff found that this approach did not appear to satisfy GDC 2 which requires, in part, appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena (e.g., earthquake loadings). However, because the OBE had been eliminated from the AP600 design, the staff found that separating OBE loads from other loads was not an issue for the AP600. In addition, system operating transients, including those resulting from sudden opening or closure of valves (e.g., steam- and water-hammer) and the sustained load associated with relief/safety valve discharge in an open system, should continue to be evaluated in combination with sustained loads in Service Level B as specified in Section 3.9.3 of the SRP. With this understanding, there did not appear to be any other significant reversing dynamic loads that may be treated separately in the AP600 design. Thus, the need to separate reversing dynamic loads in Level B appeared to be unnecessary for the AP600 design.

In its docketed response to RAI 210.79, Westinghouse revised Tables 3.9-6 and 3.9-7 of the SSAR to eliminate separate treatment of reversing dynamic loads. This was acceptable. However, in Table 3.9-16, Westinghouse erroneously did not include RVOS sustained load in its dynamic transient events associated with Level B (DU). This was identified as DSER Open Item 3.12.5.19-5. The staff evaluation of this issue is discussed in Section 3.12.5.3 above. In SSAR Revision 4, Westinghouse revised the Section 3.9 load combination tables and eliminated references to separate treatment of reversing dynamic loads. Proposed Table 3.9-16 which defined loadings was eliminated and replaced by Table 3.9-3. However, the RVOS load was still not included as a DU load. During the April 1995 design review meeting, Westinghouse committed to clarify these classifications in the SSAR tables. At the June 1996 design review meeting, Westinghouse explained that an RV load generally consists of an initial transient load (RVOT) followed by a sustained steady state load (RVOS). Specific RV loads may be classified under different ASME Code service conditions. Therefore, RVOT should be included under DU while RVOS should not be included. Westinghouse provided a revised draft SSAR Table 3.9-3 which clarified the definitions of DU as well as DN, DE, DF, and DY (for other service conditions) to indicate that they would include such loads as RVOT, RVC, and FV (as applicable). The staff found this acceptable. These proposed revisions were subsequently incorporated into Table 3.9-3 of SSAR Revision 9. On the basis of this revision, DSER Open Item 3.12.5.19-5 is closed.

Five-Percent Damping for Piping Systems

In revised Sections 3.7.1.3 and 3.7.3.15 and in Table 3.7.1-1 of the SSAR, Westinghouse proposed to use 5-percent damping for piping systems in lieu of the damping values recommended by the staff in RG 1.61, or in ASME Code Case N-411, "Alternate Damping Values for Response Spectra Analysis of Classes 1, 2, and 3 Piping, Section III, Division 1," which has been approved by the staff with certain limitations in RG 1.84. As such, the use of 5-percent damping is not a proposed alternative to the ASME Code requirements, but a deviation from regulatory guidelines.

The use of 5-percent damping would result in the underprediction of support loads and piping deflections. However, it is recognized that the stress criteria adopted by the staff for ALWRs can lead to non-linear piping response, lowering the dominant natural frequency of piping below 10 Hz where 5-percent damping is already acceptable in ASME Code Case N-411. The average underprediction of support loads is approximately 20 percent, and the seismic margins assessment required for ALWRs assures no loss of support function at this variance. Also, because the AP600 seismic criteria are (1) on the basis of ground response spectra as defined in RG 1.60 that are enhanced in the high-frequency range (approximately 8-40 Hz) and (2) anchored at a relatively high peak ground acceleration value of 0.3g, the staff finds that the use of 5-percent damping is applicable to the AP600. On this basis, the use of 5-percent damping is acceptable for piping systems in the AP600 design, subject to the same limitations as specified in RG 1.84 for Code Case N-411.

Use of Class 1 Allowable Values in Classes 2 and 3 Piping Design

In its preliminary response to RAI 210.79, Westinghouse revised Table 3.9-7 of the SSAR to use ASME Code Class 1 design stress intensity values (S_m) instead of the ASME Code Class 2 and 3 allowable stress values (S) in several equations. The staff did not find adequate justification for using ASME Code Class 1 design stress intensity values (S_m) in lieu of Code Class 2 and 3 allowable stress values (S) in the design of Code Classes 2 and 3 piping. Thus, the use of S_m instead of S is not acceptable for Code Classes 2 and 3 piping in the AP600 design. In its docketed response to RAI 210.79, Westinghouse revised Table 3.9-7 of the SSAR to use allowable stress values (S) instead of design stress intensity values (S_m) for Code Class 2 and 3 piping design. This is consistent with current code requirements and is, thus, acceptable. Westinghouse further revised Table 3.9-7 in SSAR Revision 4. The staff review of this table determined that only Class 2 and 3 allowables were used. However, Equations (8) and (9) were deleted. However, this was subsequently revised in SSAR Revision 9 and was found acceptable as discussed in Section 3.12.5.3 of this report.

Stress Limit for SSE Anchor Motions

In its preliminary response to RAI 210.79, Westinghouse revised Tables 3.9-6 and 3.9-7 of the SSAR to use a stress limit of $6.0S_m$ for stresses caused by seismic anchor motions because of the SSE for ASME Code Classes 1, 2, and 3 piping systems. However, this criterion conflicts with the staff position associated with the elimination of the OBE from the design. Therein, the staff recommends evaluating SSE seismic anchor motions in combination with thermal stresses to an allowable limit of $6S_m$ for Code Class 1 and $3.0S_h$ for Code Classes 2 and 3 piping. In its docketed response to RAI 210.79, Westinghouse proposed to adhere to the stricter limitation of the staff's position for eliminating the OBE from design as noted in revised Table 3.9-6 of the

SSAR, including Note 17, and Table 3.9-7 of the SSAR. On this basis, the treatment of SSE anchor motions is acceptable. In Revision 4 to the SSAR, Tables 3.9-6 and 3.9-7 were again revised. In reviewing these tables, the staff identified a number of issues which are summarized in Section 3.12.5.3 above. However, the stress limits for SSE anchor motions were found to be consistent with the staff position and are, therefore, acceptable.

Stress Limit for Longitudinal Piping Stress

In its preliminary response to RAI 210.79, Westinghouse revised Tables 3.9-6 and 3.9-7 of the SSAR to use a stress limit of $1.0S_m$ for the amplitude of the longitudinal force resulting from the anchor motions due to an earthquake and other reversing dynamic loads for ASME Code Classes 1, 2, and 3 piping systems. This is a new stress check not explicitly considered by the Code previously. Although in application its significance is relatively inconsequential, it does provide a limit for a stress not previously evaluated in piping systems. Thus, its use is acceptable provided the allowable stress values (S) are used for Code Classes 2 and 3 piping rather than the Code Class 1 design stress intensity values (S_m). In its docketed response to RAI 210.79, Westinghouse revised Table 3.9-7 of the SSAR to use allowable stress values (S) for Code Classes 2 and 3 piping, and is, thus, acceptable. In Revision 2 to the SSAR, Tables 3.9-6 and 3.9-7 were again revised but included the same allowables for longitudinal stress which are acceptable to the staff.

Reversing and Non-Reversing Dynamic Loads

Lastly, the staff noted that in revised Tables 3.9-6, 3.9-7, and 3.9-16 of the SSAR, Westinghouse established new categories of loads called "reversing" and "non-reversing." During the design review meeting, the question of which loads are "reversing" and which are "non-reversing" resulted in a confusing approach that can be easily misinterpreted when determining which loads are to be included in the appropriate load combinations. In the DSER, the staff reported that for the AP600, Westinghouse should clearly indicate in its load definition table (Table 3.9-16 of the SSAR) those specific loads that are to be categorized as "reversing" and those that are to be categorized as "non-reversing." This was identified as DSER Open Item 3.12.5.19-6. In SSAR Revision 4, Table 3.9-16 was eliminated and the loads for ASME Code Class 1, 2, and 3 piping were incorporated into Table 3.9-3. The staff evaluation of Tables 3.9-3, 3.9-6 and 3.9-7 is discussed in Section 3.12.5.3 above. The staff noted that all references to "reversing" and "non-reversing" loads associated with the alternate piping design criteria were eliminated. This resolves the issue and DSER Open Item 3.12.5.19-6 is closed.

Conclusion on the Use of Alternate Piping Design Criteria

On the basis of the above evaluations, it was evident that a number of the concerns associated with the use of alternate piping design criteria had been resolved. However, a substantial amount of effort would have been needed to resolve the remaining concerns associated with the use of higher stress limits for Levels C and D for the AP600 design. It also appeared likely that even after a prolonged review, the benefits of the higher stress limits would have been minimal, from a safety standpoint.

The staff had already approved other significant relaxations in piping design criteria for ALWRs (e.g., increase in stress limits to assure piping functional capability, elimination of the OBE from design, approval of LBB) and, in doing so, believed it achieved a balanced set of piping design criteria available for the AP600 design that would result in an adequately conservative design for piping systems without decreasing the safety margin from that of currently operating nuclear plants. The adoption of the above relaxations in piping design criteria for ALWRs and application of these criteria to the AP600 piping systems would result in a substantial reduction in the number of seismic restraints and postulated pipe break locations in the AP600 design from that found in many operating nuclear plants, without reducing the overall safety margin.

In conclusion, the staff found that although the major concerns associated with the use of alternate piping design criteria were not yet resolved for the AP600 design, the application of these criteria to the piping systems did not appear to enhance the overall safety of the plant and had minimal, if any, benefit for the AP600 design. In some aspects, such as in the assessment of fatigue damage, the criteria might yield a non-conservative design.

Therefore, the DSER reported that it was the conclusion of the staff that because of the high degree of complexity of the remaining concerns and the expected length of time required to achieve a mutually agreeable resolution, and the fact that there does not appear to be a major benefit in the use of the proposed alternate piping design criteria for the AP600 design. The proposal to use alternate piping design criteria, as described in the preliminary response to RAI 210.79, was not acceptable. Instead, Westinghouse should adopt in its SSAR the latest ASME Code that has been referenced in 10 CFR 50.55a (i.e., the 1989 Edition) for its piping design, combined with the staff positions established for ALWRs. This was identified as DSER Open Item 3.12.5.19-7.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse advised the staff that it had decided to withdraw its plans to use the proposed alternate piping design criteria and will use the latest NRC-accepted Code Edition. The staff reviewed SSAR Section 3.9, Revision 4 and determined that all references to the alternate criteria were eliminated. The staff identified a number of deficiencies and omissions in the revised tables on load definitions, load combinations, and stress limits. However, as discussed in Section 3.12.5.3 above, all issues related to ASME Code criteria were resolved on the basis of the additional information provided in SSAR Revision 9. Therefore, DSER Open Item 3.12.5.19-7 is closed.

3.12.5.20 Conclusions

Westinghouse has provided an acceptable revision to the SSAR that addresses the issues identified above and reflects the staff's position as indicated. Therefore, the staff concludes that Westinghouse has met the following requirements:

- GDC 1 and 10 CFR Part 50, Appendix B with regard to piping systems being designed, fabricated, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed, and with appropriate quality control

- GDC 2 and 10 CFR Part 100, Appendix A with regard to design transients and resulting load combinations for piping and pipe supports to withstand the effects of earthquakes combined with the effects of normal or accident conditions

- GDC 4 with regard to piping systems important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions

- GDC 14 with regard to the reactor coolant pressure boundary of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapid propagating failure, and of gross rupture

- GDC 15 with regard to the reactor coolant piping systems being designed with specific design and service limits to assure sufficient margin that the design conditions are not exceeded

3.12.6 Pipe Support Criteria

The staff reviewed the methodology used in the design of ASME Code Class 1, 2, and 3 pipe supports as described in Section 3.9.3.4 of the SSAR. On the basis of its review, the staff concluded that the SSAR did not include sufficient information to arrive at a final safety determination with regard to the adequacy of the pipe support design analysis methods, design procedures, and acceptance criteria. During the staff design review meeting conducted at Westinghouse offices on April 12 through 14, 1994, the staff requested copies of several AP600 internal design documents. In a letter dated April 14, 1994, Westinghouse provided a copy of their internal AP600 pipe support design criteria document, GW-P1-003, Rev. 0, for review. This document contained a significant amount of additional information required to make a final safety determination. The staff reviewed this document and identified the type of information that should be included in the SSAR. During the staff design review meeting at Westinghouse on July 19 through 21, 1994, Westinghouse agreed to revise the SSAR to include the additional detailed information. This was identified as Open Item 3.12.6-1 in the DSER. The following sections summarize the staff's evaluation of the pipe support design methods, procedures, and criteria on the basis of all of the information made available by Westinghouse to date.

In SSAR Revision 4 dated June 30, 1995, Westinghouse expanded Section 3.9.3.4, "Component and Piping Supports," to include the additional detailed information previously requested by the staff. The staff evaluation found that the information in the SSAR Revision 4 was inadequate to resolve all specific pipe support design issues. Westinghouse subsequently provided additional information in SSAR Revisions 7, 8 and 9 to address the specific pipe support design issues. The staff evaluation of the specific issues is discussed in Sections 3.12.6.1 through 3.12.6.13 below. On the basis of the additional information provided by Westinghouse, all remaining issues were resolved, and DSER Open Item 3.12.6-1 is closed as discussed in the following section.

3.12.6.1 Applicable Codes

In Section 3.9.3.4 of the SSAR, Westinghouse states that all supports for ASME Code, Section III, Class 1, 2, and 3 components, including piping supports for the AP600 design, will satisfy the requirements of Subsection NF of Section III of the ASME Code. While the staff finds Subsection NF generally acceptable for the design of piping supports, it was noted that Subsection NF does not provide adequate weld requirements for ASTM A500 Grade B tube steel members. In the DSER, the staff stated that if these members will be used in AP600 pipe support design, the SSAR should be revised to include the supplemental requirements of AWS D1.1, "Structural Welding Code," for tube steel welded connections. This was identified as DSER Open Item 3.12.6.1-1. During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, this issue was discussed and Westinghouse agreed to revise the SSAR to include these additional requirements for tube steel welded connections. This was incorporated in Revision 4 of SSAR Section 3.9.3.4. The staff finds this acceptable, and DSER Open Item 3.12.6.1-1 is closed.

3.12.6.2 Jurisdictional Boundaries

In Section 3.9.3.4 of the SSAR, Westinghouse defines the jurisdictional boundaries between pipe supports and interface attachment points, such as structural steel, in accordance with Subsection NF of Section III of the ASME Code. The staff's review of the jurisdictional boundaries described in the 1989 edition of this subsection of the Code finds that they are sufficiently defined to ensure a clear division between the pipe support and the structural steel and are, therefore, acceptable.

3.12.6.3 Loads and Load Combinations

Westinghouse provided the loads and loading combinations for the design of piping supports in Section 3.9.3.4 and Tables 3.9-3 and 3.9-8 of the SSAR. Stress limits for the various ASME Code service levels were presented in Tables 3.9-9 and 3.9-10 of the SSAR. The stress limits for pipe supports are in accordance with Subsection NF of Section III of the ASME Code. The criteria of Appendix F of Section III of the ASME Code are used for the evaluation of Level D service conditions. This is acceptable. The SSAR also stated that for certain Service Level D conditions, such as pipe rupture, the system integrity and operability may be demonstrated by allowing the supports to fail. When this is done, the consequences of the support failures would be evaluated. During the April 12 through 14 design review meeting at Westinghouse, the staff asked for a more detailed description and justification for this approach. During the July 19 through 21, 1994, design review meeting, the staff was informed that Westinghouse will not apply this procedure in the AP600 pipe support design. A proposed revision to Section 3.9.3.4 of the SSAR was provided in the June 30, 1994, and July 25, 1994, Westinghouse responses to RAI 210.42 and RAI 210.68 respectively, in which this approach was deleted. This item was considered closed contingent upon Westinghouse providing a revision to the SSAR reflecting this position. This was provided in SSAR Section 3.9.3.4 Revision 4. (See Section 3.9.3.3 of this report.)

In reviewing the load combinations in Table 3.9-8 of the SSAR, the staff asked Westinghouse to justify the proposed SRSS combination of the SSE inertia, anchor motion, and self weight excitation versus the absolute sum combination prescribed in Section 3.9.2 of the SRP. In the response to RAI 210.79, Westinghouse provided a proposed revision to Table 3.9-8 of the SSAR which requires these loads to be combined by the absolute sum method. This item was considered closed contingent upon Westinghouse providing a revision to the SSAR reflecting this position. This was provided in Revision 4 to Table 3.9-8 of the SSAR. (See Section 3.12.5.13 of this report.)

In a footnote in Table 3.9-8 of the SSAR, Westinghouse indicated that in combining earthquake loads and consequential plant transients, the timing of the loads is appropriately considered. The staff position requires earthquake loads to be combined with other dynamic loads by SRSS in accordance with NUREG-0484, Revision 1. In the DSER, the staff reported that Westinghouse should revise the SSAR accordingly or provide further justification for their position. This was identified as DSER Open Item 3.12.6.3-1. This issue was discussed during the staff design review meeting conducted at Westinghouse offices on April 10 through 11,1995. Westinghouse agreed to revise the table to clarify how SSE loads will be combined with specific fluid transient loads. However, in reviewing SSAR Revision 4, the staff found that Table 3.9-8 still contained a footnote stating that timing and causal relationships among SSE and other dynamic loads are considered to determine appropriate load combinations. During the design review meeting at Westinghouse offices in June 1996, Westinghouse agreed to comply with the staff position and to revise Table 3.9-8 to eliminate all footnote references indicating that timing and causal relationships between SSE and other dynamic loads are considered to determine appropriate load combinations. These changes were incorporated in Revision 9 to the SSAR. The staff found this acceptable and DSER Open Item 3.12.6.3-1 is closed.

3.12.6.4 Pipe Support Baseplate and Anchor Bolt Design

In Section 3.9.3.4 of the SSAR, Westinghouse did not provide any information on pipe support baseplate and anchor bolt design. In the DSER, the staff reported that Westinghouse should revise the SSAR to include additional information on pipe support design, procedures, and criteria as discussed in Section 3.12.6 above.

The staff position on pipe support baseplate and anchor bolt design is described in IE Bulletin 79-02, Revision 2, dated November 8, 1979. This document provides the factors of safety for anchor bolts and states that baseplate flexibility should be accounted for in the calculation of concrete anchor bolt loads. The factors of safety apply to all types of expansion anchor bolts (including undercut type anchor bolts), unless justification for alternative safety factors is provided.

In the response to RAI 210.107 dated June 30, 1994, Westinghouse proposed a revision to Section 3.9.3.4 of the SSAR which commits to the baseplate flexibility criteria of IE Bulletin 79-02, Revision 2. This item was considered closed contingent upon Westinghouse providing an amendment to the SSAR reflecting this position. This was identified as DSER Confirmatory Item 3.12.6.4-1. This revision was incorporated into SSAR Revision 4. The staff reviewed revised Section 3.9.3.4 and found it acceptable. Therefore, DSER Confirmatory Item 3.12.6.4-1 is closed.

With regard to undercut type anchor bolts, Westinghouse stated that the factor of safety is determined in accordance with Appendix B of ACI 349. This was not acceptable to the staff. In the DSER, the staff stated that Westinghouse should revise the SSAR to commit to the safety factors contained in Bulletin 79-02, or provide additional justification for the alternative safety factors. (See Section 3.9.3.3 of this report). In Section 3.9.3.4 of SSAR Revision 11, Westinghouse added a sentence which states that supplemental requirements for fastening anchor bolts to concrete are outlined in Section 3.8.4.5.1. The staff's evaluation of these criteria is contained in Section 3.8.4.2 of this report.

3.12.6.5 Use of Energy Absorbers and Limit Stops

In Section 3.7.3.8.4 of the SSAR, Westinghouse discusses the use of limit stops in AP600 piping systems. During the July 1994 design review meeting, Westinghouse stated that limit stops were being considered for use in the pressurizer surge line and in other piping connected to the RCS. They also stated that energy absorbers were not planned for the AP600 at that time, but may be considered in the future. The staff requested Westinghouse to revise the SSAR to adequately describe design and analysis methods and modeling assumptions for special engineered supports to be used with separate sample analysis problems. This was identified as DSER Open Item 3.12.6.5-1.

During the staff design review meeting conducted at Westinghouse offices on April 10 through 11, 1995, Westinghouse provided a detailed written description of the method incorporated in the GAPPIPE program to analyze piping systems with limit stop supports. The staff previously reviewed the GAPPIPE program and found it acceptable. The staff provided an NRC position paper (Enclosure of an NRC letter dated April 11, 1995, from Brian Sheron to R. L. Cloud, containing staff review of the topical report RLCA/P94/04-94/009 issued by R. C. Cloud and Associates on June 1, 1994, regarding methodology, verification and applications of the computer program GAPPIPE) summarizing the staff's conditions of acceptance and Westinghouse agreed to revise the SSAR to include a description of the methodology consistent with the staff position. On June 2, 1995, Westinghouse submitted Draft Revision 4 to SSAR Section 3.7.3.8.4 which included a description of the GAPPIPE methodology that will be used in the design and analysis of gapped supports (limit stops). The staff reviewed this section and found it meets good engineering practice. This description was subsequently incorporated into SSAR Revision 7. Therefore, DSER Open Item 3.12.6.5-1 is closed.

3.12.6.6 Use of Snubbers

In Section 3.9.3.4.3 of the SSAR, Westinghouse provides a summary of requirements for snubbers used as piping supports, including design criteria and analytical modeling requirements, operational and performance testing, and maintenance requirements. It states that the number of snubbers in the AP600 design will be minimized as a result of LBB considerations and the planned use of gapped support devices. From its review of this section of the SSAR, the staff concluded that the information provided is consistent with applicable portions of Section 3.9.3 of the SRP and is acceptable, pending resolution of DSER Open Item 3.9.3.3-1, which is discussed in Section 3.9.3.3 of this report. This open item was subsequently resolved. (See Section 3.9.3.3.)

3.12.6.7 Pipe Support Stiffnesses

In Section 3.9.3.4 of the SSAR, Westinghouse discusses pipe support stiffness values and support deflection limits used in the piping analyses. Rigid stiffness values are used for fabricated supports, and vendor stiffness values are used for standard supports such as snubbers, rigid gapped supports, and energy-absorbing supports. Pipe support miscellaneous steel deflections are limited for dynamic loading to one-eighth inch in each restrained direction. These deflections are defined with respect to the structure to which the miscellaneous steel is attached.

During the April 1994 design review meeting, the staff requested clarification of the types of structures to which the pipe support deflection limit applies and to the specific load combination for which the deflection limit applies. Westinghouse stated that the stiffness and deflection requirements apply to the total displacement of the pipe support structure, module or platform steel, and embedment or baseplate as described in the pipe support design criteria document, GW-P1-003. The staff indicated that this is acceptable, but this additional detailed information should be included in the SSAR as discussed above in DSER Open Item 3.12.6-1. This information was subsequently included in SSAR Revision 4, Section 3.9.3.4

Westinghouse stated that the loading combination used to calculate deflection is the maximum dynamic portion of pipe support load combinations provided in Section 3.9 of the SSAR. During the July 1994 design review meeting, the staff questioned the basis for considering only the dynamic portion of the load combination. The staff believed that it would be more appropriate to calculate the deflection on the basis of the maximum Level D load. Westinghouse believed that this criterion was consistent with the ALWR Utility Requirements Document. However, upon review of this document, it was agreed that the load requirement was not clearly defined. Westinghouse agreed to revise the SSAR to require that the deflection limit be on the basis of the maximum Level D load combination. This requirement was incorporated in Revision 4 to SSAR Section 3.9.3.4 and DSER Open Item 3.12.6.7-1 is closed.

3.12.6.8 Seismic Self-Weight Excitation

In Section 3.9.3.4 of the SSAR, Westinghouse states that the mass of the pipe support miscellaneous steel is evaluated as a self-weight excitation loading on the steel and the structures supporting the steel. This results in a conservative calculation of the pipe support seismic load, and is, therefore, acceptable. In Table 3.9-8 of the SSAR, the SSE self-weight excitation was combined with SSE inertia and anchor motion by SRSS in the Level D load combinations. As noted in Section 3.12.6.3 above, the staff disagreed with the SRSS combination method. Westinghouse provided a proposed revision to Table 3.9-8 of the SSAR in their response to RAI 210.79 in which the combination method is changed to absolute sum. (See Confirmatory Item 3.12.5.13-1 in Section 3.12.5.13 of this report.) As discussed in Sections 3.12.5.13 and 3.12.6.3 above, this revision was incorporated in Table 3.9-8 of SSAR Revision 2 and the issue is closed.

3.12.6.9 Design of Supplementary Steel

In Section 3.9.4 of the SSAR, Westinghouse states that pipe supports are designed in accordance with Subsection NF of Section III of the ASME Code. This shall include supplementary steel within the jurisdictional boundary of Subsection NF. The use of Subsection NF is standard industry practice and has been proven to provide adequate design guidelines for the design of structural steel for use as pipe supports. In addition, as discussed in Section 3.12.6.1 of this report, for ASTM A500 Grade B tube steel members, the NF requirements shall be supplemented by the weld requirements of AWS D1.1, "Structural Welding Code." (See DSER Open Item 3.12.6.1-1.) As reported in Section 3.12.6.1 above, these supplemental requirements were included in Revision 4 of SSAR Section 3.9.3.4 and the issue is closed.

3.12.6.10 Consideration of Friction Forces

The SSAR did not address the consideration of friction forces in AP600 pipe support design. During the April 1994 design review meeting at Westinghouse offices, the staff reviewed the AP600 pipe support design criteria document (GW-P1-003). This document states that friction loads induced by the pipe on the support must be considered in the analysis of sliding type supports, such as guides or box supports, when the resultant unrestrained thermal motion is greater than one-sixteenth inch. The friction force is equal to the coefficient of friction times the pipe load, and acts in the direction of the resultant pipe movement. A coefficient of friction of 0.35 for steel-on-steel sliding surfaces shall be used. If permanently lubricated bearing plate such as lubrite is used, a 0.15 coefficient of friction shall be used. The pipe force from which the friction force is developed includes dead weight and thermal loads. The staff found these coefficients of friction to be reasonable values for the AP600 design. However, as discussed in Section 3.12.6 of this report, Westinghouse was requested to revise the SSAR to include this additional detailed information. This was identified as a part of Open Item 3.12.6-1 in the DSER. In Revision 4 to SSAR Section 3.9.3.4, Westinghouse included the description of the methodology described above but incorrectly reported a friction coefficient of .30 for steel-on-steel sliding surfaces. This was later corrected to 0.35 in SSAR Revision 8 and this part of DSER Open Item 3.12.6-1 is closed.

3.12.6.11 Pipe Support Gaps and Clearances

The SSAR did not provide any information on pipe support gaps and clearances. Small gaps are normally provided for frame type supports built around the pipe. The gaps allow for radial thermal expansion of the pipe as well as allowing for pipe rotation. However, the gaps should be small enough to ensure the validity of a linear analysis which assumes a gap of zero. In the DSER, the staff reported that Westinghouse should revise the SSAR to include this information. This was identified as DSER Open Item 3.12.6.11-1. In Revision 4 to SSAR Section 3.9.3.4, Westinghouse stated that the minimum gap (total of opposing sides) between the pipe and the support is equal to the diametral expansion of the pipe because of temperature and pressure. The maximum gap is equal to the minimum gap plus one-eighth inch. The staff reviewed this information and concluded that it is consistent with industry practice and is acceptable. Therefore, DSER Open Item 3.12.6.11-1 is closed.

3.12.6.12 Instrumentation Line Support Criteria

The SSAR did not provide design requirements for safety-related instrumentation line tubing and supports. The staff requested Westinghouse to revise the SSAR to include these requirements. In the July 25, 1994, response to RAI 210.48, Westinghouse proposed a revision to Section 3.7.3.5 of the SSAR. Westinghouse stated that the equivalent static load method of analysis may be used for design of instrumentation tubing and supports. The staff found the use of the equivalent static load method to be a conservative approach for calculating loads and stresses in instrumentation lines and supports, and was acceptable. However, as discussed in

Section 3.12.6 above, the SSAR should have included more detailed design requirements. This was identified as a part of DSER Open Item 3.12.6-1. For instrumentation line tubing and supports, this should include loads and load combinations and acceptance criteria. In SSAR Revision 4, Westinghouse added Section 3.9.3.5, "Instrumentation Line Supports". Westinghouse committed to applying similar design loads, load combinations, and acceptance criteria for safety-related instrumentation supports as are applied to pipe supports. Design loads will include deadweight, thermal and seismic. The supports would be designed in accordance with ASME Code Section III, Subsection NF. The staff finds this acceptable, and this part of DSER Open Item 3.12.6-1 is closed.

3.12.6.13 Pipe Deflection Limits

The SSAR did not include specific design criteria to ensure that the maximum deflection of the piping at support locations for static and dynamic loadings are within the allowable limits. The information was requested by the staff during the April 1994 design review meeting at Westinghouse offices. In the pipe support design criteria document, GW-P1-003, Westinghouse states that for standard component supports, all manufacturer's functional limitations (travel limits, sway angles, etc.) must be strictly followed. Pipe movements for the normal condition should not result in support sway motion 4° from the support central position. Maximum sway for any loading combination should not exceed 5°. This criterion is applicable to limit stops, snubbers, rods, hangers and sway struts. Snubber settings should be chosen such that pipe movement occurs over the mid-range of snubber travel. Some margin shall be obtained between the expected pipe movement and the maximum or minimum snubber-stroke to accommodate construction tolerance. The staff found these requirements acceptable. However, as discussed in Section 3.12.6 above, this additional information should have been included in the SSAR. This was identified as a part of Open Item 3.12.6-1 in the DSER. The staff reviewed SSAR Revision 4, Section 3.9 but found that these requirements had not been incorporated. However, in SSAR Revision 8, Westinghouse revised Section 3.9.3.4 to include a description and commitment to follow the manufacturers' functional limitations for standard component supports. The staff found this acceptable, and this part of Open Item 3.12.6-1 is closed.

3.12.6.14 Conclusions

The staff concludes that supports of piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff also concludes with the following findings:

- Westinghouse satisfies the requirements of GDC 1 and 10 CFR 50.55a by specifying methods and procedures for the design and construction of safety-related pipe supports in conformance with general engineering practice.

- Westinghouse satisfies the requirements of GDC 2 and GDC 4 by designing and constructing the safety-related pipe supports to withstand the effects of normal operation as well as postulated events such as LOCAs and dynamic effects resulting from the SSE.

3.12.7 Overall Conclusions

Westinghouse has adequately revised the SSAR to provide the requested information that addresses the issues discussed above such that the staff can reach a final safety determination for AP600 piping and pipe support design. On the basis of the evaluation above, the staff concludes the following:

- Westinghouse satisfies 10 CFR Part 50 requirements by identifying applicable codes and standards, design and analysis methods, design transients and load combinations, and design limits and service conditions to ensure adequate design of all safety-related piping and pipe supports in the AP600 for their safety functions.

- Westinghouse satisfies 10 CFR Part 52 requirements by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design. The implementation of these preapproved methods and satisfaction of the acceptance criteria will be verified through the performance of the ITAAC by the COL applicant to ensure that the as-constructed piping systems are in conformance with the certified design for their safety functions.

- Westinghouse satisfies 10 CFR Part 100, Appendix A, requirements by designing the safety-related piping systems, with a reasonable assurance to withstand the dynamic effects of earthquakes with an appropriate combination of other loads of normal operation and postulated events with an adequate margin for ensuring their safety functions.

Table 3.9-1 Margins for Straight Pipe

Material	Temp °C (°F)	P_d/P_v	S MPa (ksi)	S_v MPa (ksi)	S_u MPa (ksi)	S_y MPa (ksi)	Margins on Burst	Yield
SA-106 Grade B	37.8 (100)	1/2	103.4 (15)	206.8 (30)	413.7 (60)	241.3 (35.0)	2.00	1.34
	260 (500)	1/2	103.4 (15)	206.8 (30)	413.7 (60)	195.1 (28.3)	2.00	1.08
	37.8 (100)	1/3	103.4 (15)	310.3 (45)	413.7 (60)	241.3 (35.0)	1.33	0.89
	260 (500)	1/3	103.4 (15)	310.3 (45)	413.7 (60)	195.1 (28.3)	1.33	0.72
	37.8 (100)	1/4	103.4 (15)	413.7 (60)	413.7 (60)	241.3 (35.0)	1.00	0.67
	260 (500)	1/4	103.4 (15)	413.7 (60)	413.7 (60)	195.1 (28.3)	1.00	0.54
SA-312 Type 304	37.8 (100)	1/2	129.6 (18.8)	258.6 (37.5)	517.1 (75.0)	206.8 (30.0)	1.70	0.92
	260 (500)	1/2	109.6 (15.9)	219.3 (31.8)	437.8 (63.5)	133.8 (19.4)	1.70	0.70
	37.8 (100)	1/3	129.6 (18.8)	388.2 (56.3)	517.1 (75.0)	206.8 (30.0)	1.13	0.61
	260 (500)	1/3	109.6 (15.9)	328.9 (47.7)	437.8 (63.5)	133.8 (19.4)	1.13	0.47
	37.8 (100)	1/4	129.6 (18.8)	517.1 (75.0)	517.1 (75.0)	206.8 (30.0)	0.85	0.46
	260 (500)	1/4	109.6 (15.9)	438.5 (63.5)	437.8 (63.5)	133.8 (19.4)	0.85	0.35
SA-312 Type 316	37.8 (100)	1/2	129.6 (18.8)	258.6 (37.5)	482.6 (70.0)	206.8 (30.0)	1.59	0.92
	260 (500)	1/2	109.6 (15.9)	219.3 (31.8)	424.7 (61.6)	137.2 (19.9)	1.65	0.72
	37.8 (100)	1/3	129.6 (18.8)	388.2 (56.3)	482.6 (70.0)	206.8 (30.0)	1.06	0.61
	260 (500)	1/3	109.6 (15.9)	328.9 (47.7)	424.7 (61.6)	137.2 (19.9)	1.10	0.48
	37.8 (100)	1/4	129.6 (18.8)	517.1 (75.0)	482.6 (70.0)	206.8 (30.0)	0.80	0.46
	260 (500)	1/4	109.6 (15.9)	438.5 (63.6)	424.7 (61.6)	137.2 (19.9)	0.85	0.36

S = allowable stress per ASME Code, Section III for Class 2 piping

S_v = hoop stress at $P = P_v$
$= S/(P_d/P_v)$

S_u = ultimate tensile strength; from Section III, Table I-3.1 and I-3.2

S_y = yield strength; from Section III, Table I-2.1 and I-2.2

Margin on burst pressure = $F \times S_u \times (P_d/P_v)/S$
where F = 1.00 for SA-106 Grade B
F = 0.85 for SA-312 Type 304 & Type 316
Margin on yield pressure = $1.15 \times S_y \times (P_d/P_v)/S$

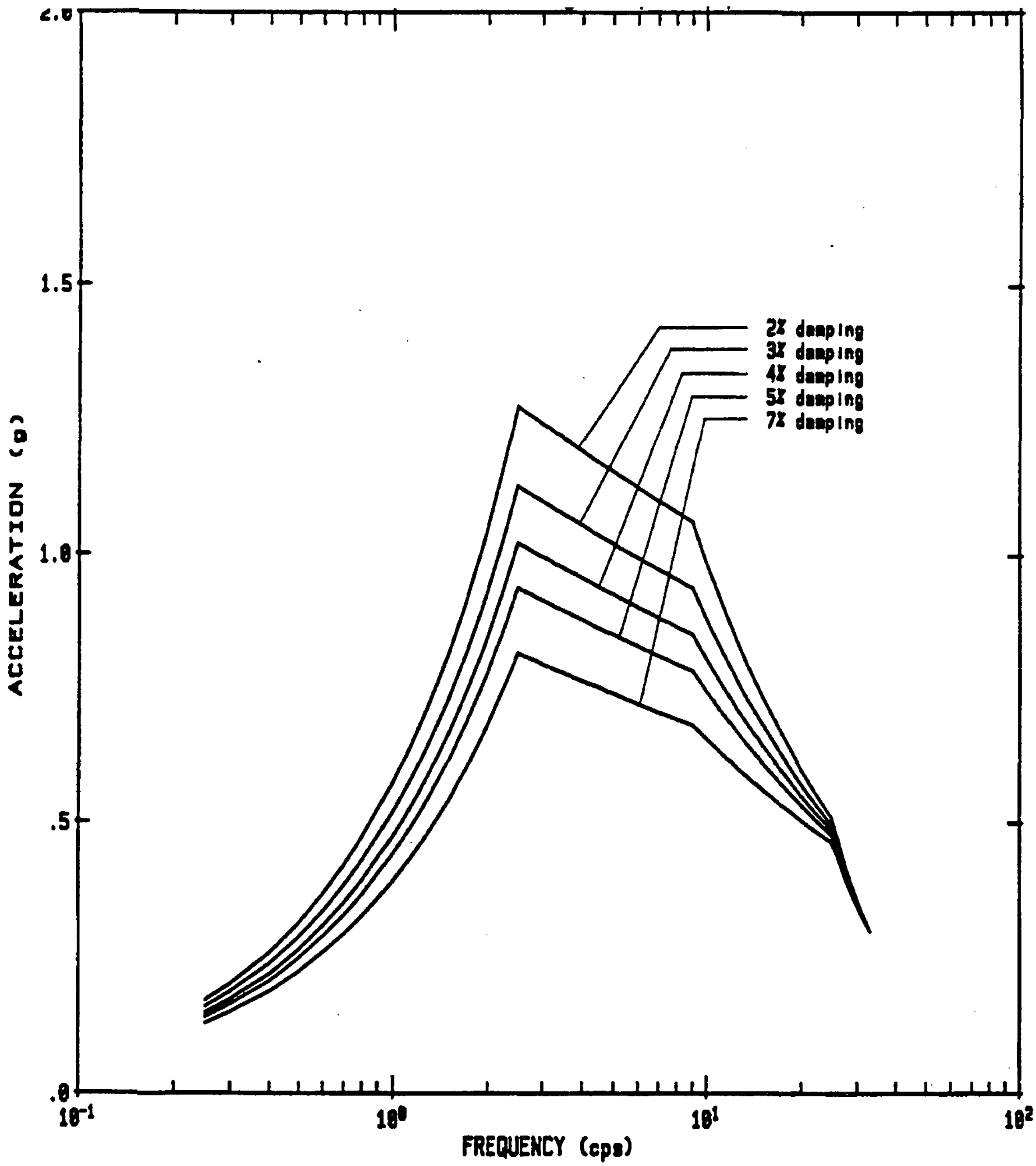


Figure 3.7-1
Horizontal Design Response Spectra
Safe Shutdown Earthquake

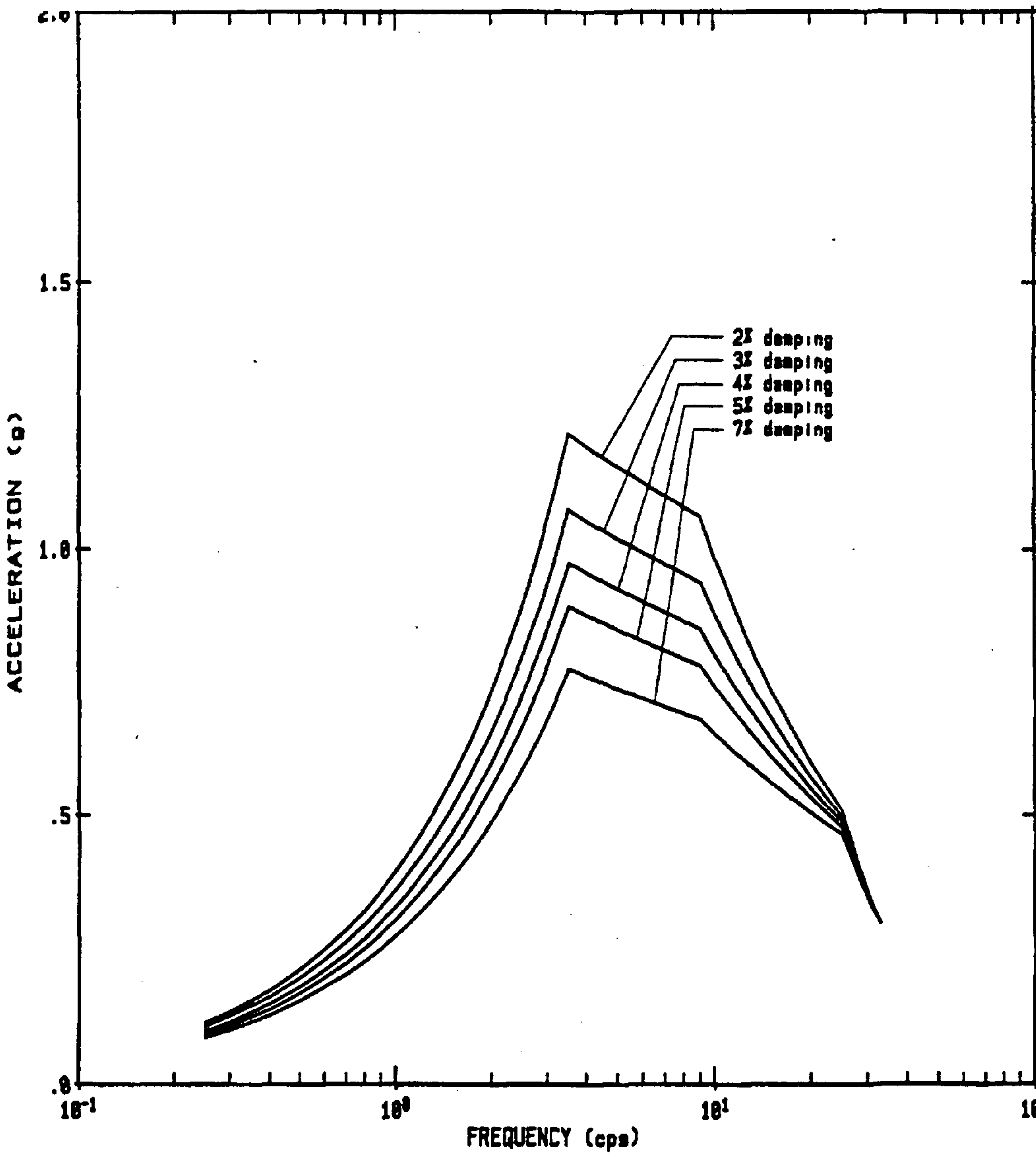


Figure 3.7-2
Vertical Design Response Spectra
Safe Shutdown Earthquake

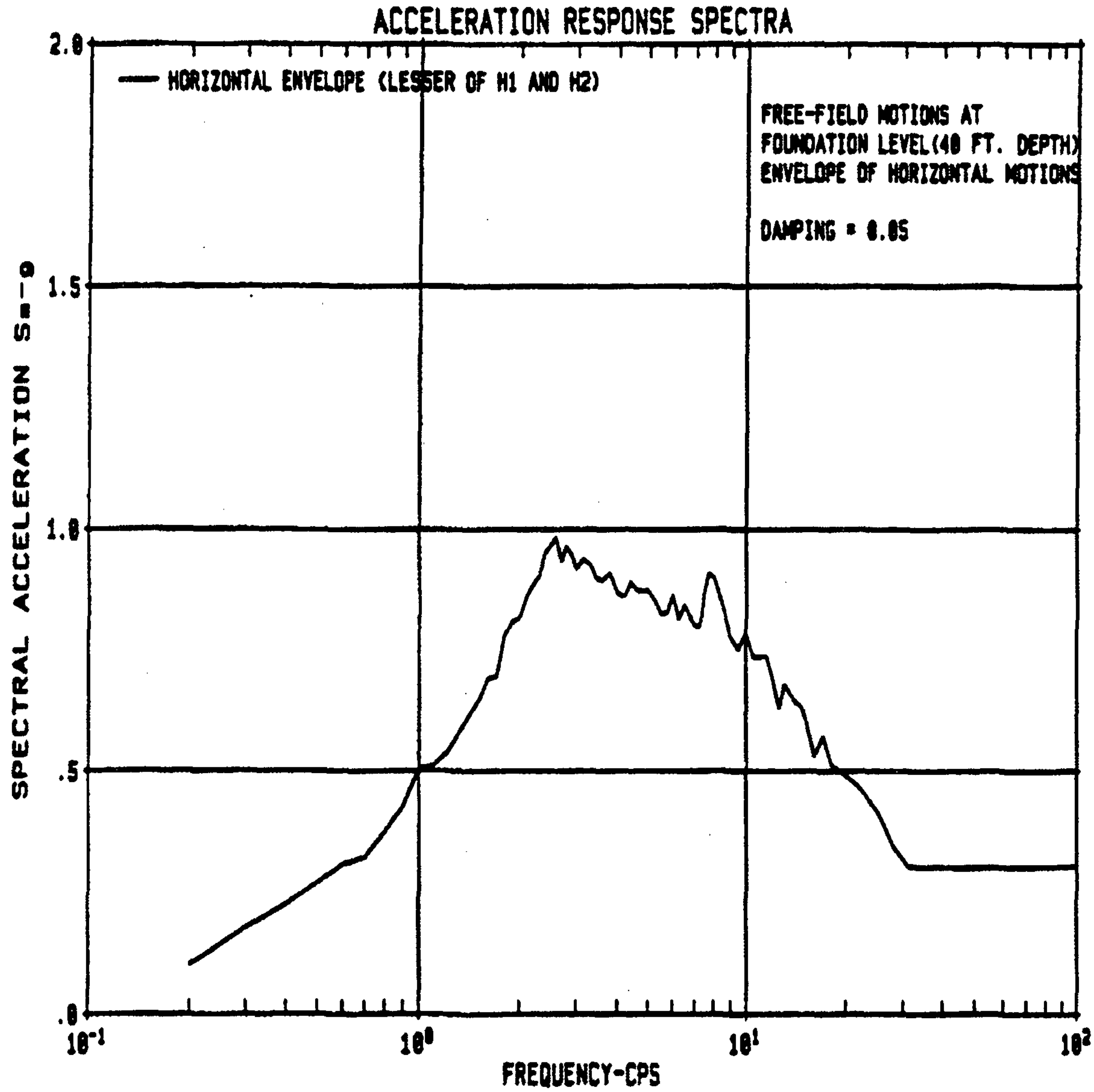


Figure 3.7-3
Free-Field Motions at Foundation Level (40 ft. Depth)
Envelope of Horizontal Motions

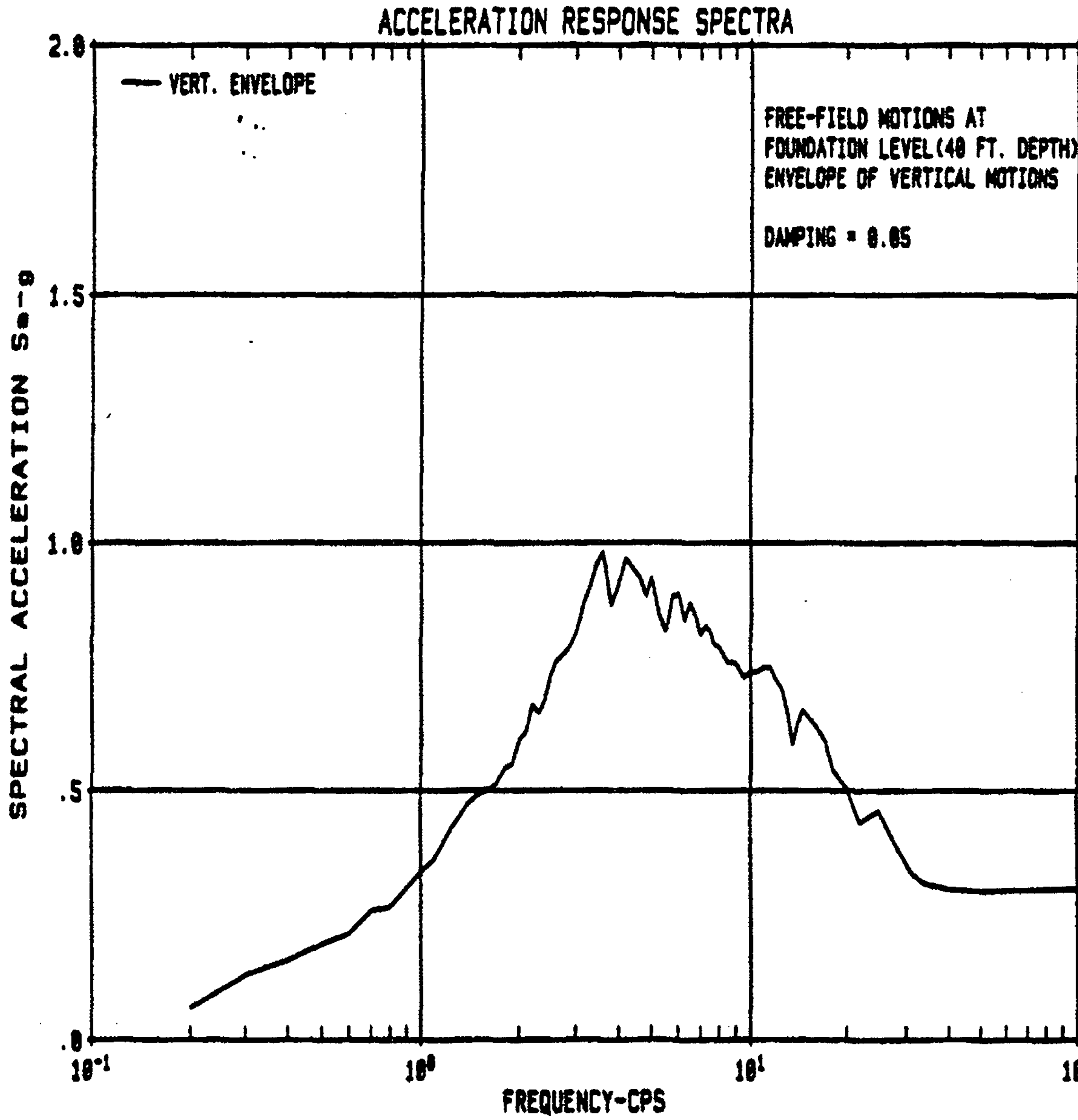


Figure 3.7-4
Free-Field Motions at Foundation Level (40 ft. Depth)
Envelope of Vertical Motions

Appendix 3A: Evaluation of Pumps and Valves Inservice Testing Plan (AP600 SSAR Table 3.9-16)

Introduction

This report presents the results of the review of the Westinghouse AP600 Inservice Testing (IST) Program documented in Section 3.9.6 of the Standard Safety Analysis Report (SSAR) and the IST Plan (Table 3.9-16).

This review was performed to determine if Westinghouse complies with the regulatory and ASME Section XI, 1989 Edition requirements, and included:

- Verification that all pumps and valves classified as safety-related in the SSAR are included in the IST Plan, as required by Section XI. This includes both passive and active components, as required by the Code, and the verification that all containment isolation valves (CIVs) and pressure isolation valves (PIVs) addressed in the SSAR and technical specifications are included in the IST Plan.

- A detailed review of the notes that provide justification for deferral of testing to the cold shutdown or refueling condition

- A review of the testing specified in the IST Plan to ensure that all safety-related functions are tested and that provisions are included to allow testing

- A review of the requests for relief from the code requirements

The review of the IST Program was performed utilizing the Code of Federal Regulations, Section 3.9.6 of the Standard Review Plan; GL 89-04, "Guidance on Developing Acceptable Inservice Testing Programs;" the Minutes of the Public Meeting on GL 89-04, dated October 25, 1989 and September 26, 1991; Supplement 1 to GL 89-04 which references NUREG-1482; "Guidelines for Inservice Testing Programs at Nuclear Power Plants;" and staff positions on passive plants (SECY 90-016 and 94-084).

Evaluation

The Westinghouse AP600 Inservice Testing Program is documented in Section 3.9.6 of the SSAR. Specific safety-related valves subject to IST are identified in Table 3.9-16 of the SSAR. In Revision 20 to the SSAR, Westinghouse clarified in Note 34 that a number of valves in Table 3.9-16 are not safety related. However, they are included in the IST program because they are relied upon in the safety analysis for those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event. The AP600 design does not employ any active safety-related pumps. There are pumps that serve a passive safety-related function (i.e., pressure boundary) and, therefore, they are not required to be tested in accordance with the Code. In Table 3.9-16, Westinghouse lists the individual valves and provides the following information for each component included in the IST Program:

- identification number
- description
- valve type

- safety-related mission(s) and function(s)
- code category
- test parameters
- test frequency

In Table 6.2.3-1 of the SSAR, Westinghouse identifies the containment isolation valves. In Table 3.9-18 of the SSAR, Westinghouse identifies the pressure isolation valves that are subject to Technical Specification LCO 3.4.16. There are no temperature isolation valves with a SSAR or technical specification-specified leakage limit.

The evaluation of the Westinghouse AP600 IST Program consisted of a detailed review of Table 3.9-16 of the SSAR and the referenced piping and instrument diagrams (P&IDs), which are included as figures in the SSAR. This evaluation included verification that all valves classified as safety-related in the SSAR are included in the IST Plan, as required by the Code, and that the testing includes all safety-related functions. Additionally, a review of the P&IDs was performed to verify that the testing committed to in Section 3.9.6 and specified in the table could be accomplished.

In accordance with OMa-1988 Part 10, as referenced by the 1989 Edition of Section XI, and the 1990 Edition of the American Society of Mechanical Engineers (ASME) OM Code, valve testing that is impractical to perform during power operation may be deferred to cold shutdown. If testing is also impractical to perform at cold shutdown, it may then be deferred to refueling. Partial-stroke exercising must be performed, when practical, quarterly or at cold shutdowns. Westinghouse has submitted 22 notes providing the justification for deferring testing to cold shutdown or refueling outages, affecting 87 valves. These justifications document the impracticality of quarterly valve testing during operation as required by Section XI, and were reviewed to verify their technical basis.

As discussed in Generic Letter 91-18, it is not the intent of IST to cause unwarranted plant shutdowns or to unnecessarily challenge other safety systems. In general, those tests involving a plant trip, damage to a system or component, or excessive personnel hazards are not considered practical. Removing one train for testing or entering a limiting condition of operation is not sufficient basis for not performing the required tests, unless the testing renders systems inoperable for extended periods of time. Other factors, such as the effect on plant safety and the difficulty of the test (i.e., the burden) may be considered when determining the impracticality. As discussed in the DSER, the ALWR systems should be designed to accommodate quarterly testing. Design changes to accommodate quarterly testing should only be performed, however, if the benefits of the test outweigh the potential risk.

Each note providing justification for deferring testing to cold shutdowns or refueling outages is discussed below.

<u>Note 3. ADS Stage 1/2/3 valves</u>

Exercising these reactor coolant pressure boundary valves (RCS-V001A & B, 2A &B, 3A & B, 11A & B, 12A & B, and 13A & B) during normal operation is impractical because of the risk of a loss of reactor coolant and depressurization. Westinghouse's proposal to exercise these valves during cold shutdowns is acceptable.

Note 4. RPV head vent valves

The reactor pressure vessel (RPV) head vent valves (RCS-PL-V150A through D) are used to vent non-condensible gases and steam during post-accident conditions.

Westinghouse states that exercise testing these valves at power represents a risk of loss of reactor coolant and depressurization of the RCS if the proper test sequence is not followed. Such testing may also result in the valves developing through-seal leaks. Westinghouse proposes to exercise test these valves at cold shutdowns.

Each branch of the vent line has two 1-inch valves in series. Quarterly testing in current Westinghouse pressurized water reactors (PWRs) is impractical because of a design problem which allows the solenoid valves to open on a pressure surge (i.e., burp open).

Westinghouse stated in Attachment 1 to NSD-NRC-97-4989 that only solenoid valves meet the several disparate design requirements for this application, and that other valve designs, which may allow quarterly testing, are not suitable for this application. Air-operated valves are not well-suited to such an application because they are normally capable of safety-related transfer in only one direction. To achieve safety-related transfer capability in two directions requires the use of a piston operator and a safety-related air supply. Motor-operated valves are not well suited to such an application because they are larger and heavier with an extended operator that makes them difficult to locate and support. They are also less reliable. Both air-operated and motor-operated valves have stem packing which can permit external leakage of reactor coolant.

Therefore, because of the risk of loss of reactor coolant and depressurization of the RCS if the proper test sequence is not followed and the potential for the valves developing through seal leaks, exercise testing of these valves at cold shutdowns is acceptable.

Note 6. CVS RCPB Isolation Valves

Testing the valves (CVS-PL-V001, 2, 3, 80, 81, and 82) would require isolating the RCS purification flow, resulting in undesirable level transients and possible plant trip. Therefore, quarterly testing is impractical and Westinghouse's proposal to test at cold shutdowns is acceptable.

Note 9. Passive Core Cooling Accumulators' Discharge Check Valves

The check valves referenced in Note 9 (PXS-PL-V0028A and B, 29A and B) open to allow the accumulators to inject into the RPV in the event of a large loss-of-coolant accident (LOCA). Quarterly testing is impractical because the accumulator pressure is lower than the RCS pressure. Also, providing flow to RCS during power operation would cause undesirable thermal transients on the RCS. Full stroke exercising during cold shutdowns is impractical because of the potential of adding a significant volume of water to the RCS and lifting the RNS relief valve. There is also a risk of injecting nitrogen into the RCS.

Westinghouse proposes to perform partial stroke testing during cold shutdowns of 48 hours or longer duration in Mode 5. In this test, flow is provided from test connections, through the check valves and into the RCS. It is impractical to perform such testing during cold shutdowns less than 48 hours in duration because the need to install the test setup inside the containment may extend the outage time. Full stroke exercise testing of these valves is conducted during refueling shutdowns (when the RPV head is removed).

In view of the impracticality of testing these valves when the plant is at power and the impracticality of testing these valves during cold shutdowns less than 48 hours in duration because of the need to install a test setup inside the containment, Westinghouse's proposal to partial stroke exercise these valves during cold shutdowns of 48 hours duration or longer and to full stroke exercise these valves during refueling shutdowns is acceptable.

Note 10, Passive Core Cooling CMT Check Valves

These check valves (PXS-PL-V0016A and B, 17A and B) are in series at the discharge of the core makeup tanks (CMT). Although there is normally no flow passing through them (the CMT air-operated discharge valves are normally closed), they are biased open by design. It is impractical to close these valves during operation because the test setup would require personnel entry and manipulation of the manual vent valves (V030A/B and V031A/B). Additionally, the Technical Specifications would require the CMT temperature and boron levels to be restored after the test.

It is also impractical to close these valves during cold shutdown because the test setup and restoration of the CMT boron concentration could delay plant startup. Westinghouse's proposal to exercise these valves at refueling is acceptable.

Note 11, PXS Containment Recirculation Check Valves

Squib valves in line with the normally closed 6-inch containment recirculation check valves (PXS-PL-V119A and B) prevent the use of the in-containment refueling water storage tank (IRWST) water to test the valves. The sumps are normally dry and the squib valves prevent testing these valves with flow (only 20 percent of the squib charges are fired every 2 years).

To exercise these check valves, an operator must enter the containment, remove a cover from the recirculation screens, and insert a test device into the recirculation device to push open the check valve. Westinghouse states that the test device interfaces with the valve without damaging the valve. The device incorporates load measuring sensors to measure the initial opening and full open force.

These check valves are not exercised during power operations or during cold shutdowns because of the need to enter highly radioactive areas and because during this test the recirculation screen is bypassed. Also, during cold shutdowns these actions could require extending the outage time. Westinghouse's proposal to exercise these valves during refueling conditions when the recirculation lines are not required to be available by Technical Specification LCOs 3.5.7 and 3.5.8 and when radiation levels are reduced, using a mechanical exerciser test device as allowed by the Code, is acceptable.

Note 12, PXS IRWST Injection Check Valves

Exercise testing open these normally closed 6-inch check valves (PXS-V122A/B, V124A/B) requires that a test cart be moved into containment and temporary connections made to these valves. Westinghouse confirmed during the October 27, 1995, conference call that the 2-inch test connections are capable, on the basis of calculations, of allowing sufficient flow to fully open these 6-inch check valves. The need to perform significant work inside containment, thereby increases personnel radiation exposure. Also, during cold shutdowns, these actions could require extending the outage time.

As discussed in NUREG-1482, Section 4.1.4, it is acceptable to extend the test interval of check valves that are verified closed by leak testing to refueling outages, on the basis of the need to set up test equipment.

Additionally, the IRWST injection line isolation valves must have their power restored and be closed to permit testing. Exercising these valves during power operations or during cold shutdowns is impractical, because closing the IRWST injection line valve is not permitted by the Technical Specifications.

Therefore, Westinghouse's proposal to exercise test these valves during refueling conditions when the IRWST injection lines are not required to be available by Technical Specifications and the radiation levels are reduced is acceptable.

Note 14, Component Cooling Water System CIVs

Exercising the normally open, motor-operated valves (CCS-PL-V200, V207 and V208) and check valve (CCS-PL-V201) during power operation would isolate cooling water to the reactor coolant pumps and letdown heat exchanger. Because of the potential for equipment damage, this is impractical. Testing the valves during cold shutdowns when component cooling is not required in the containment (i.e., when the reactor coolant pumps are stopped) is acceptable.

Notes 15 and 24, Normal RHR System Reactor Coolant Isolation Valves

Exercising these reactor coolant isolation valves (RNS-PL-V001A and B, 2A and B, 15A and B, 17A and B) during normal operation is impractical because of the potential for equipment damage as a result of overpressurizing the low-pressure normal residual heat removal and passive cooling systems. Westinghouse's proposal to exercise these valves during cold shutdowns is acceptable.

Note 18, Compressed Air CIVs

These instrument air containment isolation valves (CAS-PL-V014 and 15) are normally open and have a safety function only to close. It is impractical to test these valves during operation or cold shutdown since the air-operated valves serviced by the instrument air system (e.g., the containment fan cooler supply and return valves, the passive residual heat exchanger outlet valves, and CMT discharge valves) would close causing system transients, and possible plant transients. Westinghouse's proposal to exercise these valves during refueling is acceptable.

Note 19, Primary Sampling System CIV

The primary sampling system containment isolation check valve (PSS-PL-V024) is located inside containment. The only practical means of verifying the valve's closure capability is by performing a leak test. As discussed in NUREG-1482, Section 4.1.4, it is acceptable to extend the test interval of check valves verified closed by leak testing to refueling outages, on the basis of the need to set up test equipment. Therefore, Westinghouse's proposal to exercise the valve during refueling outages is acceptable.

Note 20, Main Steam Isolation Valves MSIVs and Main Feedwater Isolation Valves MFIVs

The main feedwater and steam isolation valves (SGS-PL-V040A and B, 57A and B) are open during operation. Quarterly full-stroke exercising of the valves would isolate feedwater and steam to the steam generators, causing a steam generator level transient and a plant trip. Westinghouse's proposal to partial-stroke exercise quarterly and full-stroke exercise at cold shutdowns is acceptable.

Note 21, Post 72 Hour Valves

There is one valve in Table 3.9-16 for which Note 20 is indicated (i.e., PCS-PL-V039). This simple check valve opens to allow long-term makeup to the PCS. Westinghouse states that to exercise test this valve open requires transportation and installation of temporary test equipment and pressure/fluid supplies. As discussed in NUREG-1482, Section 4.1.4, the need for test setup and performance limitations may render testing during operation or cold shutdowns impractical. There is no other means for exercising the check valve, therefore, Westinghouse's proposal to test this valve during refueling outages is acceptable.

Note 22, Auxiliary Spray Isolation Valves

The auxiliary spray isolation valves are normally closed check (CVS-PL-V085) and motor-operated (CVS-PL-V084) valves and perform a safety function in the closed direction. It is impractical to open these check valves during operation because of the thermal transients on the pressurizer. Therefore, Westinghouse's proposal to test these valves at cold shutdowns is acceptable.

Note 23, Thermal Relief Valves Inside Containment

The subject valves (RNS-PL-V003A and B, CVS-PL-V100) are thermal relief check valves installed in the normal residual heat removal suction from the RCS hot leg and the CVS makeup line. It is impractical to exercise these valves open or closed quarterly or during cold shutdowns, as they are located inside containment and the only practical means of exercising them is by utilizing test connections. As discussed in NUREG-1482, Section 4.1.4, it is acceptable to extend the test interval of check valves verified closed by leak testing to refueling outages, on the basis of the need to set up test equipment. Westinghouse's proposal is acceptable on the basis of the same justification.

Note 25. Main Feedwater Control Valves, MSR Steam Control Valve, And Turbine Control Valves

The main feedwater control valves (SGS-PL-V250A and B), moisture separator reheater (MSR) steam control valve (MSS-PL-V016), and turbine control valves (MTS-PL-V002A and B, and 4A and B) are normally modulating open and have a function to close. Note 34 indicates that the MSR steam control valve and turbine control valves are not safety related. They are included in the IST program because they are relied upon in the safety analysis for those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event.

Full-stroke exercising these valves during operation would isolate flow to either the steam generators, MSRs, or turbine and cause a plant transient or trip. During normal operation these valves are partial-stroke exercised. Therefore, Westinghouse's proposal to full-stroke exercise them during cold shutdowns is acceptable.

Note 26. Containment Compartment Drain Line Check Valves

The containment compartment drain line check valves (WLS-PL-V071A, B, and C, 72A, B, and C) are normally closed and have a safety function only to close or remain closed. It is impractical to exercise these valves closed quarterly or during cold shutdowns, as they are located inside containment and the only practical means of verifying their closure capability is by leak testing. These valves are not supplied with position indication. As discussed in NUREG-1482, Section 4.1.4, it is acceptable to extend the test interval of check valves verified closed by leak testing to refueling outages as proposed by Westinghouse.

Note 28. Chilled Water System Containment Isolation Valves

The 25.4-cm (10-in.) motor-operated butterfly valves (VWS-PL-V058, 62, 82, and 86) are normally open containment isolation valves. Closure of these valves isolates the chilled water system to the containment recirculation fan coolers. Westinghouse stated that the water flow is necessary to maintain the air temperature within Technical Specification limits. Technical Specification 3.6.5 requires the average air temperature to be less than 120 °F, or restore the temperature within 24 hours. Quarterly testing may be impractical on the basis of the Technical Specification limits. Westinghouse's proposal to test the valves quarterly, unless site climatic conditions would cause the containment temperature to exceed the limit during testing is acceptable.

Note 29. Turbine Bypass Control Valves

The turbine bypass control valves (MSS-PL-V001, 2, 3, and 4) are normally closed. Opening of these valves would result in power transients and temperature transients on the condenser and bypass lines. These valves are only used for rapid load reductions and at low power levels during startup and shutdown. Therefore, Westinghouse's proposal to full-stroke exercise them during cold shutdowns is acceptable.

Note 33, Fuel Transfer Tube Manual Valve

Westinghouse proposed to defer the exercise of manual valve FHS-PL-V001 to refueling outages. This valve is closed and is not required to be operable during normal operation or cold shutdowns. It is only required to be operable during refueling when the blind flange is removed from the tube inside containment. In accordance with Part 10, Section 4.2.1.7, the exercising test schedule (i.e., quarterly) is not required to be followed for this valve. Therefore, Westinghouse's proposal to perform exercise testing of this valve during refueling shutdowns prior to removing the fuel transfer tube flange is acceptable.

Note 35, Turbine Stop Valves

The turbine stop valves (MTS-PL-V001A and B, and 3A and B) are normally open and have a function to close. Note 34 indicates that these valves are not safety related. They are included in the IST program because they are relied upon in the safety analysis for those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event. Exercising them during operation would isolate flow to the turbines and cause a plant transient or trip. Therefore, Westinghouse's proposal to full-stroke exercise them during cold shutdowns is acceptable.

Summary and Conclusion

The AP600 IST Program complies with the 1989 Edition of Section XI. Westinghouse has proposed preparing the IST program in accordance with the 1990 Edition of the ASME OM Code. As discussed above, this Code is identical in technical requirements for pump and valve testing to the 1989 Edition of Section XI. Therefore, the Code edition proposed by Westinghouse provides an acceptable level of quality and safety and the alternative is acceptable pursuant to 10 CFR 50.55a(a)(3)(i).

On the basis of the above evaluations, Westinghouse adequately addressed the impracticality of exercising certain valves during power operation, and the proposed testing frequencies comply with OMa-1988, Part 10, Sections 4.2.1 and 4.3.2.

In conclusion, the Westinghouse AP600 IST Program was reviewed in accordance with the requirements in the 1989 Edition of Section XI and applicable staff guidance. On the basis of the results of this evaluation, it is concluded that the AP600 IST Program is in accordance with the regulatory, ASME Code requirements, and staff positions on inservice testing of pumps and valves used in passive reactor designs.

4 REACTOR

4.1 Introduction

Chapter 4.0 of the Standard Safety Analysis Report (SSAR) describes the mechanical components of the AP600 reactor and reactor core, including the fuel system design (rods and assemblies), the nuclear design, and the thermal-hydraulic design. As a result of its review of Chapter 4.0, 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 the COL applicant or licensee will require NRC approval prior to implementation.

SSAR Sections:

4 + Table 1.6-1	WCAP-12488-A, "Fuel Criteria Evaluation Process"
4.1	Maximum Fuel Rod Average Burnup
4.1.1	Principal Design Requirements
Table 4.3-1	Reactor Core Description (First Cycle)
Table 4.3-2	Nuclear Design Parameters (First Cycle)
Table 4.3-3	Reactivity Requirements for Rod Cluster Control Assemblies

4.2 Fuel System Design

Information contained in the SSAR and referenced topical reports represented the basis for the staff's review of the AP600 fuel design. In addition, the staff conducted its review in accordance with the guidelines provided in Section 4.2 of the Standard Review Plan (SRP), which prescribes acceptance criteria to ensure that General Design Criteria (GDC) 10, 27, and 35 are met. The SSAR describes how the AP600 meets these criteria (and the other guidance in Section 4.2 of the SRP) by reference to fuel designs previously approved by the NRC, or to fuel designs that meet the acceptance criteria approved by the NRC for Westinghouse fuel. Thus, in reviewing the AP600 fuel system design, the staff's primary objective was to ensure that the design fulfills the following criteria:

- The fuel system will not be damaged by normal operations and anticipated operational occurrences (AOOs).
- Fuel system damage will never be so severe as to prevent control rods from being inserted when required.

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

The term "not damaged" means that the fuel rods do not fail, the fuel system's dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis. These objectives address GDC 10, and the design limits that accomplish these objectives are called specified acceptable fuel design limits (SAFDLs). In a "fuel rod failure," the fuel rod leaks, and the first fission product barrier (the fuel cladding) is, therefore, breached. Fuel rod failure must be accounted for in the dose analysis for postulated accidents as required by 10 CFR Part 100. "Coolability," which is sometimes termed "coolable geometry," is the ability of the fuel assembly to retain the geometrical configuration of its rod bundle with adequate coolant channel spacing for removal of residual heat. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (e.g., GDC 27 and 35). In addition, 10 CFR 50.46 establishes specific requirements for emergency core cooling system performance following postulated loss-of-coolant accidents (LOCAs).

4.2.1 Fuel Design Description

The AP600 reactor fuel assemblies consists of 264 fuel rods in a 17x17 square array. The assemblies are very similar to the VANTAGE 5 HYBRID (VANTAGE 5H) fuel assemblies, which evolved from other Westinghouse fuel designs, such as the VANTAGE 5, optimized, and standard fuel assemblies.

Each AP600 fuel assembly consists of a total of nine structural grids, including low pressure drop intermediate grids and either four or five (if an optional fifth grid is used) intermediate flow mixing grids. Each fuel assembly has a reconstitutable top nozzle and a debris filter bottom nozzle to filter debris present in the reactor coolant. In the AP600 reactor, incore reactivity control is provided by control rods, gray rods, burnable absorber rods, and a soluble chemical neutron absorber (boric acid).

4.2.2 Fuel Rod Description

The AP600 fuel rods consist of cylindrical, ceramic pellets of slightly enriched uranium dioxide (UO_2). These pellets are contained in either cold-worked and stress-relieved Zircaloy-4 tubing or an advanced zirconium based alloy (ZIRLO) tubing, which is plugged and seal-welded at the ends to encapsulate the fuel. The UO_2 pellets are slightly dished to better accommodate thermal expansion and fuel swelling, and to increase the void volume for fission product release. The void volume will also accommodate the differential thermal expansion between the clad and the fuel as the pellet density changes in response to irradiation.

The AP600 fuel rod is designed with two plenums (upper and lower) to accommodate fission gas release. A hold-down spring keeps the upper plenum in place, while a standoff assembly holds the lower plenum in position. A stainless steel compression spring located at the top of the fuel pellet column restrains the column in its proper position during shipping and handling. A tapered, solid bottom-end plug grips the end of the fuel rod, making it easier to handle fuel assemblies during fabrication and reconstitution. The end plug is sufficiently long to extend through the bottom grid. This precludes any breach in the fuel rod pressure boundary as a result of clad fretting wear, which is induced by debris trapped at the bottom of the grid location.

The fuel rods are internally pressurized with helium during fabrication. This internal pressurization reduces stress from differential pressure, reduces cladding stress-strain limits, and prevents clad flattening during the lifetime of the fuel in the core.

The AP600 fuel rod design may also include axial blankets consisting of fuel pellets of reduced enrichments. Axial blankets help to reduce axial neutron leakage and enhance fuel utilization. The presence of these axial blankets will not impact the operation of the AP600 excore source range neutron detectors, since the expected reduction in neutron flux is limited to the top and bottom 15.2 cm (0.5 ft) of the core, while the source range detectors are typically located 91.4 cm (3 ft) from the bottom of the core.

The AP600 design may also include a second type of fuel rod, which uses an integral fuel burnable absorber (IFBA) containing a less than 0.03-mm (0.001-inch) boride coating on the surface of the fuel pellets. The utilization of these IFBA rods within individual fuel assemblies will vary, depending on the specific application.

4.2.3 Burnable Absorber Rod Description

Discrete burnable absorber rods are inserted into selected thimbles within the fuel assemblies to reduce the beginning-of-life moderator temperature coefficient (MTC). The burnable absorber rods in each fuel assembly are grouped and attached together at the top end of the rod hold-down assembly. The burnable absorber rods are made of wet, annular absorber boron-carbide material contained within two concentric zirconium alloy tubes. The tubes are plugged, pressurized with helium, and seal-welded to encapsulate the annular stack of absorber material. Released helium gas is trapped in an annular plenum, as the absorber material is depleted during irradiation. These burnable absorber rods have been used previously in Westinghouse-designed reactors. The design of these absorber rods was approved by the staff in its Safety Evaluation Reports (SERs) of WCAP-9179, issued in July 1978, and WCAP-10021-P-A, issued in October 1983.

4.2.4 Rod Cluster Control Assembly Description

Reactivity control in the AP600 reactor is comprised of the typical Westinghouse burnable absorber rods, Integral Fuel Burnable Absorbers (IFBA) and/or Wet Annular Burnable Absorber rods (WABA), and a soluble chemical neutron absorber (boric acid). The design also includes two types of rod control assemblies known as rod cluster control assemblies (RCCAs) and gray rod cluster assemblies (GRCAs). Both consist of neutron absorbing rods fastened at the top end to a common spider assembly. The various components of the spider assembly are made of 304- and 308-type stainless steel. The assembly retainer is made of 17-4 PH material, and the impact springs are made of nickel-chromium-iron Alloy 718.

The AP600 reactor uses 45 RCCAs and 16 gray rod cluster assemblies. The absorber material is a very high thermal neutron absorber (essentially "black") silver-indium-cadmium alloy, with additional resonance absorption to enhance rod worth. Bullet-shaped tips are used as plugs at the bottom of the rods to reduce hydraulic drag during reactor trip and also to help guide the rods smoothly into the dashpot of the fuel assembly.

Typically, the gray rod cluster assemblies are used in load-follow maneuvering. These assemblies provide a mechanical shim reactivity mechanism (versus chemical shim, which is achieved by means of changing the concentration of the soluble boron). Each gray rod assembly has 24 rodlets fastened at the top end to a common hub or spider. Of the 24 rodlets, 20 are made of stainless steel, while the remaining 4 are made of silver-indium-cadmium absorber material. The mechanical design of the gray rod cluster assemblies, gray rod drive mechanisms, and the interface with the fuel assemblies and guide thimbles are identical to those of the RCCAs.

4.2.5 Design Bases

The AP600 fuel rod and fuel assembly design bases were established to satisfy the general performance and safety criteria presented in Section 4.2 of the SRP. Rod burnup (including extended burnup) design criteria, methods, and evaluations are described in WCAP-10125-P-A. The acceptance limits, as used by Westinghouse to analyze the AP600 fuel rods and assemblies, are described in Westinghouse topical report WCAP-12488, "Fuel Criterion Evaluation Process." The staff has approved this process in its SER on WCAP-12488, issued in April 1995.

Fuel integrity is ensured by design limits imposed on various stresses and deformations resulting from non-operational loads (such as shipping), normal loads (as defined for Westinghouse AOO Conditions I and II, which are normal operation and operational transients and events of moderate frequency, respectively), and abnormal loads (as defined for Condition III and IV accidents, which are infrequent incidents and limiting faults, respectively). At each stage of the overall fuel rod and fuel assembly analysis, the performance of the limiting rod, with appropriate consideration for uncertainties, did not exceed the limits specified by the design bases. Moreover, the design bases for the incore components were subject to Conditions I and II, the stress categories, and the theory presented in Section III of the Boiler and Pressure Vessel code promulgated by the American Society of Mechanical Engineers (ASME).

4.2.6 Design Evaluations

Chapter 4 of the SSAR and associated (referenced) topical reports present a variety of methods for use in demonstrating that the AP600 fuel rods, fuel assemblies, and control assemblies meet the established design criteria. These methods include operating experience, prototype testing, and analytical predictions.

4.2.7 Testing and Inspection Plan

The AP600 fuel is subjected to a quality assurance (QA) program similar to those associated with earlier Westinghouse fuel designs. This QA program ensures that the fuel is fabricated in accordance with the design requirements, reaches the plant site undamaged, and is correctly loaded into the core without damage. Online fuel rod failure monitoring and post-irradiation surveillance will be performed to detect anomalies or confirm that the fuel system is performing as expected. The QA program is described in Westinghouse topical reports WCAP-7800, Revision 11A and WCAP-8370, Revision 7A and have been previously approved by the NRC.

4.2.8 Conclusion

In the draft safety evaluation report (DSER), the staff identified a concern related to flow-induced vibration of the VANTAGE 5H fuel in operating plants. This issue was identified as DSER Open Item 4.2.8-1. Westinghouse resolved the VANTAGE 5H flow-induced vibration problem without an imposed thermal margin penalty in their submittal dated June 24, 1996. Specifically, Westinghouse redesigned the flow grid and flow tested it at Columbia University to make sure that the low-flow thermal-hydraulic requirements were not altered by this grid redesign. The redesigned grid precludes any flow-induced vibration problems, and ensures that the pressure drop across the grid is not adversely effected. Westinghouse pointed out that the design basis for the AP600 core outlined in SSAR Section 4.4 will be consistent with the use of this grid. Westinghouse revised Section 4.2.5 of the SSAR to reflect this change. The staff reviewed the June 24, 1996, submittal and concluded that the comparison of the empirical and the calculated data indicates that the redesigned grid is acceptable, therefore, DSER Open Item 4.2.8-1 is closed. Section 4.4 of this report contains more information on this subject.

Consequently, the staff concludes that Westinghouse has designed the AP600 fuel system to meet the following objectives:

- The fuel system will not be damaged by normal operation and AOOs.
- Fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when required.
- Core coolability will always be maintained for design-basis transients and accidents.

The fuel system design addresses certain requirements of 10 CFR 50.46; GDCs 10, 27, and 35 of Appendix A to 10 CFR Part 50; Appendix K to 10 CFR Part 50; and 10 CFR Part 100. The staff also concludes that the AP600 fuel system design, control assembly design, and initial core design satisfy all the requirements of 10 CFR Part 50, and GDCs 10 and 27 of Appendix A to 10 CFR Part 50, and the guidance of Regulatory Guides 1.60 and 1.77 and Sections 4.2 and 4.3 of the SRP. In addition, the fuel design and control assembly design have been specified and the associated analysis results have been presented in the SSAR. Startup tests to confirm specified nuclear and thermal-hydraulic design parameters are described in Chapter 14 of the SSAR.

DSER Open Item 4.2.8-2 and COL Action Item 4.2.8-1 required identification of any changes to the reference design of the fuel, the burnable absorber rods, the rod cluster control assemblies, or the initial core design from that presented in the SSAR. To address this issue, Westinghouse has added this commitment to Section 4.2.5 of the SSAR, "Combined License Information." Therefore, DSER Open Item 4.2.8-2 is closed.

4.3 <u>Nuclear Design</u>

The staff's review of the nuclear design relied on information contained in the SSAR, responses to staff requests for additional information (RAIs), and the referenced topical reports. In addition, the staff conducted its review in accordance with the guidelines provided by Section 4.3 of the SRP.

4.3.1 Design Bases

Chapter 4 of the SSAR presents the design bases for the AP600 nuclear design, which, as described in the chapter, complies with the following General Design Criteria:

- GDC 10, reactor design (core, coolant, control and protection systems) to assure acceptable fuel design limits are not exceeded
- GDC 11, negative prompt feedback coefficient
- GDC 12, power oscillation suppression
- GDC 13, control and monitoring system to monitor normal operation, AOOs, and accident conditions
- GDC 20, protection system function to assure acceptable fuel design limits not exceeded and to initiate operation of systems important to safety
- GDC 25, protection system requirements for reactivity control malfunction
- GDC 26, reactivity control system redundancy and capability
- GDC 27, combined reactivity control system capability to assure the capability to cool the core is maintained
- GDC 28, reactivity limits to assure that reactivity accidents do not result in specified damage to the RCPB or significantly impair the capability to cool the core

The staff concludes that the fuel design bases presented in the SSAR for the AP600 include the requirements in the GDC listed above, are in accordance with Section 4.3 of the SRP and are, therefore, acceptable.

4.3.2 Nuclear Design Description

The SSAR describes the first cycle fuel loading, which consists of a specified number of fuel bundles. Each fuel bundle (assembly) contains a 17x17 rod array comprised nominally of 264 fuel rods, 24 rod cluster control thimbles, and an incore instrumentation thimble. The fuel rods within a given assembly have the same uranium enrichment in both the radial and axial planes. To attain a desired radial power distribution, three batches of fuel assemblies contain rods of different fuel enrichment. The central region of the core consists of the lower enrichment, while the higher enriched assemblies are placed on the periphery. Axial blankets are included in the reload core design basis to reduce neutron leakage and improve fuel utilization. Reload cores, as well as the initial cycle, are anticipated to operate approximately 24 months between refueling, accumulating a cycle burnup of approximately 18,360 MWD/MTU.

Chapter 4 of the SSAR presents the critical soluble boron concentrations and worths, as well as the plutonium buildup. Values presented for the delayed neutron fraction and prompt neutron lifetime at beginning and end of cycle are consistent with those normally used and are, therefore, acceptable.

4.3.2.1 Power Distribution

The total peaking factor, FQ, for the AP600 is 2.60 compared to 2.32 for the standard 17x17 fuel assembly. In addition, the following design bases affect the power distribution of the AP600:

- The peaking factor in the core will not be greater than 2.60 during normal operation at full power, in order to meet the initial conditions assumed in the LOCA analysis.
- Under normal conditions (including maximum overpower) the peak fuel power will not produce fuel centerline melting.
- The core will not operate with a power distribution that will cause the departure from nucleate boiling (DNB) ratio to fall below 1.23 during normal operation or AOOs using Westinghouse's WRB-2 DNB correlation (previously approved by the staff) and corresponding statistical uncertainties.

The AP600 will use the on-line core monitoring system to continuously monitor important reactor core characteristics and establish margins for the operating limits. This system will give the operator detailed power distribution information, in both the radial and axial directions, on demand. The incore instrument system data processor receives the transmitted digitized fixed incore 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 processing algorithms contained within the online monitoring system are functionally identical to those historically used for the evaluation of power distribution measurements in Westinghouse pressurized-water reactors (PWRs). These algorithms are described in topical report WCAP-12472-P-A issued in August, 1994.

Excore detectors register signals that are then processed and calibrated against incore measurements, allowing comparison of the power at the top and bottom halves of the core. These calibrated measurements (referred to as the flux difference, delta I), are displayed on a panel in the control room. Thus, the operator can use this data to determine the shape penalty function for the overtemperature delta T calculation (for DNB protection) and the overpower delta T calculation (for overpower protection).

The online monitoring system also evaluates the consequences of limiting power distributions consistent with the conditions prevalent in the reactor at the current time. In addition, the system provides the operator with the current allowable operating space, current power distribution information, thermal margin assessment, and operational recommendations to manage and maintain required thermal margins. As such, the online monitoring system provides the primary means of managing and maintaining required operating thermal margins during normal operations.

4.3.2.2 Reactivity Coefficients

Reactivity coefficients are expressions of the effect on core reactivity of changes in the following core conditions, among others:

- reactor power
- fuel temperature
- moderator temperature
- moderator density
- boron concentration

These coefficients vary with fuel burnup and power level. In the SSAR, Westinghouse provided calculated values of the coefficients and evaluated the accuracy of these calculations. The staff reviewed Westinghouse's calculated values of the reactivity coefficients and concludes that they adequately represent the full range of expected values. The staff also reviewed the reactivity coefficients used in the transient and accident analyses, and concludes that they conservatively bound the expected values, including uncertainties as discussed in Section 15.2.4 of this report. Further, the startup physics testing will measure moderator and Doppler coefficients, along with boron worth to ensure that actual values are within the bounds of those used in these analyses.

The staff requires that advanced light-water reactors (ALWRs) be designed such that the MTC will be negative under all conditions. Although the SSAR predicts that MTC values will be negative for the full range of expected operating conditions during the initial fuel cycle, the net value of the coefficient could be positive if the concentration of soluble boron is too high. However, the AP600 fuel design can use discrete or integral fuel burnable absorbers in reload cores to ensure that the MTC remains negative over the full range of power operation. The effect of the burnable rods is to make the MTC more negative.

4.3.2.3 Control Requirements

Core reactivity is controlled by means of a chemical poison (boric acid) dissolved in the coolant, RCCAs, gray rod cluster assemblies, and burnable absorbers. To allow for changes in reactivity as a result of reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of reactivity is built into the core. The SSAR provides a sufficient amount of information regarding the core reactivity balance for the first core, and shows that the AP600 design incorporates the means to control excess reactivity at all times.

Both excess reactivity and reactor power level are controlled by moving the RCCAs and/or varying the boron concentration in the reactor coolant. Excess reactivity is controlled by adding soluble boron to the coolant and using burnable absorbers, when necessary. The SSAR describes the boron concentrations for several AP600 core configurations, including the unit boron worth for the initial cycle. The combination of reactivity control systems in the AP600 design satisfies the requirements of GDC 26.

Typically, AP600 plants will operate at steady-state full power. Gray rods and/or RCCAs will permit operators to compensate for rapid changes in reactivity (e.g., power level changes and the effects of minor variations in moderator temperature and boron concentration) without impairing shutdown capability.

Gray rods and/or RCCAs will primarily assist in controlling core reactivity and power distribution, including xenon-induced axial power oscillations during power operations, and axial power shape during load-following transients. The rod control system automatically modulates the insertion of the axial offset (AO) control bank controlling the axial power distribution simultaneous with the MSHIM gray and control rod banks to maintain programmed coolant temperature. Gray rods and/or RCCAs can also control reactivity to compensate for minor variations in moderator temperature and boron concentration during power operations and assist in compensating for reactivity changes resulting from power level and xenon changes during load-following transients. The total reactivity worth of these rods will enable licensees to control load-following transients without changing boron concentrations as much as current generation PWRs, or even at all.

Rod insertion is controlled by the power-dependent rod insertion limits given in the Technical Specifications (TS). The AP600 design meets the following criteria:

- Sufficient negative reactivity is available to quickly shut down the reactor with ample margin.
- If a control rod were ejected (an unlikely event), the worth of a control rod would be no worse than the worth assumed in the accident analysis.

Soluble boron absorber is used to compensate for slow reactivity changes, including changes associated with fuel burnup, changes in xenon and samarium concentrations, buildup of long-lived fission products, burnable absorber rod depletion, and the large moderator temperature change from cold shutdown to hot standby.

The staff reviewed the AP600 calculated rod worths and the related uncertainties. These calculations represent many reactor-years of startup test data for PWR designs and critical experiments (See Section 15.2 of this report for additional information). On the basis of its review of Westinghouse's calculations, the staff concludes that Westinghouse's assessment of reactivity control presented in Chapter 4 of the SSAR is suitably conservative, and that the control system has adequate negative reactivity worth to ensure shutdown capability if the most reactive RCCA is assumed stuck in the fully withdrawn position. Therefore, the rod cluster control assemblies and soluble boron worths are acceptable for use in the accident analysis.

On the basis of its review of the information provided by Westinghouse, the staff also concludes that the functional design of the AP600 reactivity control system meets the requirements of GDC 21, 23, 25, 26, and 27, with respect to its reliability and testability, fail-safe design, malfunction protection design, redundancy and capability, and combined systems capability. Typical codes used are NRC-approved codes such as THINC IV (WCAP-7667-P-A), WESTAR (WCAP-10951-P-A), and LOFTRAN (See Section 21.6.1 of this report). Therefore, the staff concludes that the functional design of the AP600 reactivity control system is acceptable.

4.3.2.4 Stability

Chapter 4 of the SSAR, discusses the stability of the reactor in response to xenon-induced power distribution oscillations and the control of such transients. Because of the negative power coefficient, the reactor is inherently stable to oscillations in total reactor power.

Calculational analysis of a PWR core containing 17x17 fuel assemblies, and approved codes (the PANDA code, as described in WCAP-7084-P-A, and the TURTLE code, described in WCAP-7213-A) shows that stability against xenon-induced spatial oscillations is expected to be equal to, or better than, that of earlier designs for cores of similar size.

From its analyses (calculational and experimental), Westinghouse concludes that the core will be stable in response to both radial and azimuthal xenon oscillations throughout core life. Westinghouse verified this conclusion by measurements on an operating PWR reactor (Rochester Gas and Electric Corporation's R.E. Ginna plant) having a height of 3.66 m (12 ft) and 121 fuel assemblies, as reported in WCAP-7964. On the basis of the similarity of core design between the AP600 reactor and the analyzed plant, the staff concurs with this conclusion.

4.3.2.5 Vessel Irradiation

Section 5.3 of this report presents a complete review of the methods analyses used in determining neutron and gamma ray flux attenuation between the AP600 reactor core and the pressure vessel.

4.3.2.6 Criticality of Fuel Assemblies

Adequate design of the AP600 fuel transfer and storage facilities precludes criticality of fuel assemblies outside the reactor vessel. Section 9.1 of this report discusses the staff's evaluation.

4.3.3 Analytical Methods

Chapter 4 of the SSAR describes the calculational methods used to analyze the nuclear characteristics of the AP600 reactor design. The staff reviewed the examples provided in that description to demonstrate that these methods can predict experimental results. Lattice codes such as PHOENIX-P (WCAP-11596-P-A) and the 3-D depletion code ANC (WCAP-10965-P-A) are used to model advanced fuel designs. These codes address the 3-D features of the fuel. Data provided by Westinghouse in tabular form provides comparisons of empirical versus calculated data. Based on the above, the staff concludes that the examples and information presented in the SSAR adequately demonstrate the ability of these analytical methods to calculate the reactor physics characteristics of the AP600 reactor core.

4.3.4 Summary of Evaluation Findings

To allow for reactivity changes resulting from reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, Westinghouse designed a significant amount of excess reactivity into the core. Westinghouse also provided substantial information about core reactivity balances for the first cycle and showed that the design incorporates to control excess reactivity at all times. Westinghouse also showed that sufficient control rod worth will be available at any time during the cycle to shut down the reactor with at least a 1.6-percent delta-k/k subcritical margin in the hot shutdown condition, with the most reactive RCCA stuck in the fully withdrawn position. The staff concludes that Westinghouse's assessment of reactivity control requirements over the first core cycle is suitably conservative, and the AP600 control system has adequate negative worth to ensure shutdown capability.

In Chapter 4.0 of the SSAR, Westinghouse described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design, and provided examples to demonstrate the ability of these tools to predict experimental results. The information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the AP600 design.

On the basis of the following observations, the staff concludes that the AP600 nuclear design satisfies the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28:

- Westinghouse has satisfied the requirements of GDC 10, 20, and 25 with respect to SAFDLs by demonstrating that the AP600 design meets the following objectives:
    - No fuel damage occurs during normal operation, including the effects of AOOs (GDC 10).
    - Automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of AOOs, and that systems and components important to safety will automatically operate under accident conditions (GDC 20).
    - No single malfunction of the reactivity control system will violate the fuel design limits (GDC 25).
- Westinghouse has satisfied the requirements of GDC 11, with respect to nuclear feedback characteristics, by performing calculations of the negative Doppler and moderator coefficients of reactivity to compensate for a rapid, uncontrolled reactivity excursion. The staff reviewed these reactivity coefficients and concludes that they are suitably conservative.
- Westinghouse has satisfied the requirements of GDC 12 by showing that power oscillations are not possible or can be easily detected and suppressed. The staff reviewed Westinghouse's analysis of these power oscillations and concludes that they are suitably conservative.
- Westinghouse has satisfied the requirements of GDC 13 by providing instrumentation and controls to monitor the following variables and systems that can affect the fission process:
    - the reactor coolant system (RCS)
    - steam and core power conversion systems
    - containment
    - engineered safety systems
    - auxiliary systems
    - reactor power distribution
    - control rod positions and patterns
    - process variables, such as temperatures and pressures
- Westinghouse has satisfied the requirements of GDC 26 by providing two independent reactivity control systems of different design. Specifically, the design includes RCCAs

and gray rod assemblies, as well as a chemical shim (boric acid) which provide the following capabilities:

- Reliably shut down the reactor during normal operation conditions and during AOOs
- Provide adequate boration to establish and maintain safe-shutdown conditions.

- Westinghouse has satisfied the requirements of GDC 27, by providing reactivity control systems, in conjunction with absorber addition by the emergency core cooling system (ECCS), to reliably control reactivity changes under postulated accident conditions as follows:
  - Provide a movable rod control system and a liquid control system.
  - Perform calculations to demonstrate that the core has sufficient shutdown margin with the highest worth RCCA stuck, as discussed in Section 4.3.2 of this report.
- Westinghouse has satisfied the requirements of GDC 28, with respect to postulated reactivity accidents, by following the methodology described in the approved topical report, WCAP-7588, Rev. 1-A. This topical report analyzes the assumptions used in evaluating a control rod ejection accident for PWRs. Moreover, the criteria and results presented in WCAP-7588 are within the criteria and limits prescribed by the NRC's Regulatory Guide (RG) 1.77.

Therefore, the staff concludes that the AP600 nuclear design is acceptable.

4.4 Thermal-Hydraulic Design

Information contained in the SSAR, responses to staff RAIs, and the referenced topical reports represented the basis for the staff's review of the AP600 thermal-hydraulic design. In addition, the staff conducted its review in accordance with the guidelines provided by Section 4.4 of the SRP.

4.4.1 Thermal-Hydraulic Design Bases

Section 4.4 of the SRP sets forth the acceptance criteria used by the staff to evaluate the thermal-hydraulic design of the AP600 reactor core. The principal thermal-hydraulic design basis for the AP600 is the avoidance of thermal-hydraulically induced damage during normal steady-state operation and anticipated operational transients. Westinghouse performed the AP600 thermal-hydraulic design analyses using the Revised Thermal Design Procedure (RTDP) described in WCAP-11397-P-A. This report was approved by the staff in April 1989.

4.4.1.1 Departure From Nucleate Boiling

The RTDP is a statistically based methodology whereby uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are statistically combined to determine DNB uncertainty factors. The departure from nucleate boiling ratio (DNBR) values are determined such that there is at least a 95-percent probability at a 95-percent confidence level that the DNB will not occur on the most limiting fuel rod during normal operation, operational transients, and transient conditions arising from faults of moderate frequency (ANS Conditions I and II events).

For the AP600 fuel, the RTDP design limit DNBR value is 1.23 for a typical cell and 1.22 for the thimble cell. For those transients that use the WESTAR computer program and the modified low-flow WRB-2 correlation (i.e., loss of flow and locked rotor), the RTDP design limits are 1.25 for the typical cell and 1.24 for the thimble cell. Section 4.4.2.2 of this report provides more information on this subject.

To maintain a DNBR margin, and thus offset DNB penalties such as those attributable to fuel rod bow, Westinghouse performed safety analyses using DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis DNBRs results in available DNBR margin.

4.4.1.2 Fuel Temperature

During modes of operation associated with ANS Condition I and Condition II events, there is at least a 95-percent probability at a 95-percent confidence level that the peak KW/ft fuel rods will not exceed the uranium dioxide melting temperature. The melting temperature of unirradiated UO_2 is taken to be 5080 °F, decreasing 58 °F per 10,000 MWD/MTU. By precluding UO_2 melting, the AP600 design preserves the fuel geometry and eliminates possible adverse effects of molten UO_2 on the cladding. In addition, to preclude center melting and as a basis for overpower protection system setpoints, Westinghouse selected a calculated centerline fuel temperature of 4700 °F as the overpower limit.

4.4.1.3 Core Flow Design Basis

This section addresses the minimum coolant flow through the fuel rod regions at the entrance of the reactor vessel. A minimum of 91.0 percent of the thermal flow rate passes the fuel rod region of the core and is effective for fuel rod cooling. Coolant flow through the thimble tubes as well as the leakage from the core barrel-baffle region into the core are not considered effective for heat removal.

Core cooling evaluations are dependent on the thermal flow rate (minimum flow) entering the reactor vessel. A maximum of 9.0 percent of this value is allowed as bypass flow. This includes rod cluster control guide thimble cooling flow, head cooling flow, baffle leakage, and leakage to the vessel outlet nozzle. This bypass assumes that no plugging devices or burnable absorbers are in the rod cluster control guide thimble tubes that do not contain rod cluster control rods.

4.4.1.4 Hydrodynamic Stability

The hydrodynamic stability design basis for the AP600 reactor specifies that modes of operation associated with ANS Condition I and II events do not lead to hydrodynamic instability.

In Sections 4.4.1.4 and 4.4.4.6 of the SSAR, Westinghouse states that the AP600 is thermal-hydraulically stable. In steady-state, two-phase, heated flow in parallel channels, the potential for hydrodynamic instability exists. Boiling flows may be susceptible to thermodynamic instabilities. These instabilities are undesirable in reactors since they may cause a change in thermal-hydraulic conditions that may lead to a reduction in the DNB heat flux, relative to that observed during a steady flow condition or to undesired forced vibrations of core components. Therefore, Westinghouse developed a thermal-hydraulic design criterion that states that modes of operation under Condition I and II events must not lead to thermal-hydrodynamic instabilities.

For the AP600 reactor, two specific types of flow instabilities exist. Specifically, these are the Ledinegg or flow excursion type of static instability, and the density wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flow rate from one steady-state to another. This instability occurs when the slope of the reactor coolant system pressure drop-flow rate curve ($\partial \Delta P/\partial G_{internal}$) becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve ($\partial \Delta P/\partial G_{external}$).

The criterion for stability is thus ($\partial \Delta P/\partial G_{internal} \geq \partial \Delta P/\partial G_{external}$). The Westinghouse pump head curve has a negative slope ($\partial \Delta P/\partial G_{external} < O$), whereas the reactor coolant system pressure drop-flow curve has a positive slope ($\partial \Delta P/\partial G_{internal} > O$) over the Condition I and Condition II operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody (1977). Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single-phase region, and causes quality or void perturbations in the two-phase regions that travel up the channel with the flow. The quality and length perturbations in the two-phase region create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, the two-phase pressure drop perturbation feeds back to the single-phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by Ishii for parallel closed channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability. This method had been used to assess the stability of typical Westinghouse reactor designs under Condition I and II operation. The results indicate that a large margin-to-density-wave instability exists; e.g., increases on the order of 150% of rated reactor power would be required for the predicted inception of this type of instability.

The application of Ishii's method to Westinghouse reactor designs is conservative because of the parallel open channel feature of Westinghouse PWR cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high power density. There is also energy transfer from channels of high power density to channels of lower power density. This coupling with cooler channels has led to the opinion that an open channel configuration is more stable than the above closed channel analysis under the same boundary conditions. Moreover, tests of flow stability have shown that the closed channel systems were *less* stable than when the same channels were cross-connected at several locations. The cross-connections were such that the resistance to channel-to-channel cross flow and enthalpy perturbations would be greater than that which would exist in a PWR core, which has a relatively low resistance to cross flow.

Flow instabilities that have been observed have occurred almost exclusively in closed channel systems operating at low pressure relative to the Westinghouse PWR operating pressures. Kao, Morgan, and Parker analyzed parallel closed channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Data from the rod bundle DNB tests provide additional evidence that flow instabilities do not adversely affect thermal margin. Moreover, many Westinghouse rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data that might indicate flow instabilities in the rod bundle.

In summary, the staff concludes that thermal-hydrodynamic instabilities will not occur under Condition I and II modes of operation for Westinghouse PWR reactor designs. A large power margin, greater than doubling rated power, exists to predicted inception of such instabilities. Analyses have been performed to show that minor plant-to-plant differences in Westinghouse reactor designs, such as fuel assembly arrays, core power to flow ratios, and fuel assembly length, will not result in gross deterioration of the above power margins.

The staff concludes that past operating experience, flow stability experiments, and the inherent thermal-hydraulic characteristics of Westinghouse PWRs provide a basis for accepting the AP600 stability evaluation.

4.4.2 Thermal-Hydraulic Design Methodology

The NRC staff has reviewed the information on thermal-hydraulic design presented by Westinghouse in the AP600 SSAR as described below.

4.4.2.1 Thermal-Hydraulic Analyses Methods

Table 4.4-1 of the AP600 SSAR compares the AP600 design parameters to the Westinghouse four-loop plant using 17x17 fuel assemblies.

4.4.2.2 Departure From Nucleate Boiling

For the AP600 reactor, Westinghouse calculated the DNBRs using the WRB-2 correlation, described in the Westinghouse approved Topical Report WCAP-10444-P-A. Also, Westinghouse used the THINC-IV (approved in WCAP-7956-P-A; WCAP-8054-P-A; and WCAP-12330-P-A) and WESTAR (approved in WCAP-10951-P-A) computer codes to determine the flow distribution in the core and the local conditions in the hot channels for use in the DNB correlation. Finally, Westinghouse has stated that it performed critical heat flux tests which model the AP600 fuel assembly and that the channel hot factor (CHF) characteristics of the AP600 fuel assembly can be adequately described by the WRB-2 correlation with a design criteria of 1.17.

4.4.2.3 CHF/Low Flow Tests

Westinghouse stated that the WRB-2 correlation was developed on the basis of mixing vane data and is only applicable in the heated rod spans above the first mixing vane grid. The WRB-2 correlation is also applied to the analysis of the loss-of-flow event. For the AP600, the flow rates at the time of minimum DNBR are considerably lower than the previously licensed lower limit of G = $0.9X10^6$ lb/ft^2-hr. Consequently, Westinghouse conducted extensive testing at the Columbia University Heat Transfer Research Facility (HTRF) to investigate the extrapolation of the WRB-2 correlation into the low-flow regions. These CHF tests yielded a total of 372 data points from four configurations of 5x5 rodded test bundles. The data collected pertained to inlet pressure, inlet mass velocity, inlet temperature, average bundle heat flux, and identification of the thermocouples which indicated a CHF event.

The four configurations tested differed with respect to cell type, use of intermediate flow mixers (IFMs), and grid rotation. Each configuration consisted of electrically heated rods arranged in 5x5 rectangular arrays. Rod spacing was maintained by Zircaloy mixing vane grids, IFMs, and/or simple support grids. Each array was encased in a ceramic-lined shroud box, which was positioned within the pressure boundary of the test section housing. In operation, the flow entered the test section housing near the bottom and then traveled vertically upward. The rod inner diameters were tapered to produce a non-uniform (cosine) axial flux distribution and a radial power distribution.

Section 5.0 of WCAP-14371 provided the system conditions bounding the low-flow conditions encountered in the AP600 loss-of-flow and locked-rotor accident analyses. Westinghouse also presented the collected test data in Tables 6-1 and 6-2 of WCAP-14371, and in a letter to the NRC dated June 24, 1996.

Test data analysis and comparison of measured CHF with predicted CHF (M/P) data showed that the utilization of the WRB-2 correlation in the low-flow regions (less than 1.0 x 10^6 lbm/hr-ft^2) leads to instabilities and (consequently) unpredictability. Specifically, Westinghouse found that the WRB-2 correlation tends to overpredict CHF at the low-flow conditions. Westinghouse reported data that showed that the magnitude of the overprediction depended greatly on the local mass flux and slightly on the local pressure. As a consequence of these overpredicted results, Westinghouse applied a multiplier (derived from the conducted DNB testing data) to the WRB-2 correlation to account for the CHF overprediction, in Section 8.0 of WCAP-14371. In the low-flow regions, Westinghouse referred to the WRB-2 as the Adjusted WRB-2 correlation.

In the DSER, the staff stated that its conclusions regarding the thermal-hydraulic design of the AP600 would remain open until Westinghouse submitted the results from the DNB and flow-induced vibration tests. This was identified as DSER Open Item 4.4-1. The staff has reviewed the Thermal and Hydraulic sections of Chapter 4 of the AP600 SSAR and the DNB testing data submitted in WCAP-14371 and in the letter dated June 24, 1996. On the basis of a comparison of the measured and calculated data, the staff find the analyses and data acceptable, subject to the following conditions on the application of the WRB-2 correlation:

- If the local mass flux is between 4.4E+06 and 1.8E+07 kg/hr-m^2 (0.9E+06 and 3.7E+06 lbm/hr-ft^2), the WRB-2 correlation is applicable for the DNBR calculation.

- If the local mass flux in the hot channel is outside the range of the WRB-2 correlation as noted above and between 2.34E+06 and 5.08E+06 kg/hr-m^2 (4.8E+05 and 1.04E+06 lbm/hr-ft^2), the adjusted WRB-2 correlation must be used for the DNB Ratio (DNBR) calculation.

Therefore, DSER Open Item 4.4-1 is closed for the AP600 low flow-testing conducted at Columbia University.

4.4.2.4 Effects of Fuel Rod Bow on DNB

The phenomenon of fuel rod bowing as described in WCAP-8691 is accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from the evaluation of DNBR are used to offset the effect of rod bow.

The safety analysis for the AP600 core maintained sufficient margin between the safety analysis limit DNBR (as described in Section 4.4.1.1 above) to accommodate the full- and low-flow rod bow DNBR penalties identified in WCAP-10444-P-A. The amount of fuel rod bow, and its associated DNBR penalties, is predicted to be less than 1.5 percent DNBR for the AP600 fuel. These penalties are accounted for in the design safety analysis and are founded on an assembly average burnup of 24,000 MWd/MTU. At burnup greater than 24,000 MWd/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown, because of the decrease in fissionable isotopes and the buildup of fission product inventory and no additional rod bow penalty is required. This evaluation is based on the use of the NRC-approved scaling factors described in WCAP-8691.

The staff concludes that rod bow penalties have been properly offset by the DNBR margins calculated by Westinghouse.

4.4.3 Instrumentation Requirements

The NRC staff has reviewed the information on reactor instrumentation design presented by Westinghouse in the AP600 SSAR as described below.

4.4.3.1 Incore Instrumentation

The incore instrumentation system consists of incore instrumentation thimble assemblies, housing fixed incore detectors, core exit thermocouple assemblies contained within an inner and outer sheath assembly, and associated signal processing and data processing equipment.

The AP600 design uses incore instrument thimble assemblies, each of which is composed of multiple fixed in-core detectors and one thermocouple. The primary function of the in-core instrumentation system is to provide a three-dimensional (3D) flux map of the reactor core. Flux mapping is necessary to calibrate instrumentation for protection and safety systems, and also to provide information for optimizing core performance.

A secondary function of the incore monitoring system is to provide an indication of inadequate core cooling. This secondary function is the result of a mechanical design that groups the flux mapping detectors in the same thimble as the sensors used for the inadequate core cooling monitor.

All of the data from the incore instrumentation is processed and the results are made available in the main control room. Chapter 7 of this report describes the overtemperature and overpower delta T instrumentation used in the AP600 reactor design.

4.4.3.2 Loose Parts Monitoring Systems

Westinghouse has provided documentation of their loose parts monitoring system, which uses the Westinghouse Digital Metal Impact Monitoring System (DMIMS) previously reviewed and approved by the NRC for the Virgil Summer and Shearon Harris plants. The DMIMS consists of several active instrumentation channels, each comprising a piezoelectric accelerator (sensor), signal conditioning, and diagnostic equipment. Data base channel checks and functional tests are incorporated in the DMIMS designs. The DMIMS is calibrated before plant startup. Capabilities exist for subsequent periodic online channel checks and channel functional tests and for offline channel calibrations at refueling outages. Operators will be trained in the operation and maintenance of the DMIMS before plant startup.

In Section 4.4.6.4 of the AP600 SSAR, Westinghouse states that the overall design of DMIMS is in conformance to Revision 1 to RG 1.133. The staff concurs with that assertion.

4.4.4 Conclusion and Summary

The staff's review of the thermal-hydraulic design of the AP600 reactor core included the design basis and steady-state analysis of the core thermal-hydraulic performance. The acceptance criteria used as the basis for this evaluation are set forth in Section 4.4, "Thermal and Hydraulic Design," of the SRP. The review concentrated on the difference between the proposed design and those designs that the staff has previously reviewed and found acceptable.

On the basis of the discussion above, the staff concludes that the thermal-hydraulic design of the initial AP600 core is acceptable.

4.5 <u>Reactor Materials</u>

Information contained in the SSAR, responses to staff requests for additional information, and the referenced topical reports represented the basis for the staff's review of the AP600 reactor materials selection. In addition, the staff conducted its review in accordance with the guidelines provided by Sections 4.5.1 and 4.5.2 of the SRP

4.5.1 Control Rod Drive System Structural Materials

GDC 1 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 commensurate with the importance of the safety function to be performed. These quality standards shall be identified and evaluated to determine their adequacy to ensure a quality product in keeping with the required safety function. The NRC staff reviewed the AP600 control rod drive (CRD) system to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met as they relate to the selection of materials for the CRD system.

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, rapidly propagating failure, and gross rupture. The staff reviewed the CRD system structural materials to ensure that the relevant requirements of GDC 14 have been met.

GDC 26 requires, in part, that one of the radioactivity control systems shall use control rods (preferably including a positive means for inserting the rods) and shall be capable of reliably controlling reactivity changes so that specified acceptable fuel design limits are not exceeded. The staff reviewed the materials of the CRD system to ensure that the relevant requirements of GDC 26 have been met.

The AP600 CRD system, described in Section 3.9.4.1 of the SSAR, builds upon a proven Westinghouse design that has been used in many operating nuclear power plants. The staff reviewed the structural materials aspect of the CRD as presented in the SSAR in accordance with the guidelines in Section 4.5.1 of the SRP. During the course of its review, the staff transmitted RAIs to Westinghouse concerning the CRD 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.

Section 4.5.1 of the SSAR describes the materials used to fabricate components of the control rod drive mechanism (CRDM) and the control rod driveline. The SSAR also provides information relative to the material specifications, the fabrication and processing of austenitic stainless steel components, the contamination protection and cleaning of austenitic stainless steel, and items concerned with materials other than austenitic stainless steel.

The staff requested that Westinghouse provide information to confirm that the materials selected for the CRDM components that will be exposed to reactor coolant conform to Section III of the ASME Boiler and Pressure Vessel Code. This was identified as DSER Open Item 4.5.1-1. In Revision 5, Westinghouse subsequently revised the SSAR to state that materials for the CRD mechanism and the control rod assemblies are selected for acceptable performance in service, the design goal being to achieve a service life of 9E+06 full-step cycles as a minimum. Pressure-retaining materials will comply with Section III of the ASME Code. Other materials that are not part of the RCPB are not required to conform to Section III of the ASME Code. This response is acceptable to the staff because pressure-retaining materials will meet the requirements of Section III of the ASME Code. Therefore, DSER Open Item 4.5.1-1 is closed.

Revision 0 of the SSAR did not provide information on the specifications, types, grades, heat treatments, and properties used for the materials of the CRD mechanism components. Such information is essential for evaluating the equivalency of such specifications to those given in Section III of the ASME Code; Parts A, B, and C of Section II of the ASME Code; or RG 1.85, "Materials Code Case Acceptability - ASME Section III, Division 1." Accordingly, the staff requested that Westinghouse provide the needed information. This was identified as DSER Open Items 4.5.1-2 and 4.5.1-17. In Revision 3, Westinghouse revised the SSAR to state that the CRD components that are part of the RCPB, the latch housing and the rod travel housing, are fabricated from austenitic stainless steel (SA-336, Types 316LN, and 304LN). These materials comply with Section III of the ASME Code. The internal latch assembly components are fabricated from a variety of materials, including Type 410 stainless steel (magnetic pole pieces), Alloy X-750 (springs), a cobalt alloy (link pins), and Type 304 stainless steel. Resistance to wear of load-bearing surfaces is provided by hard chrome plate and cobalt-based hardfacing. The drive rod assembly includes a Type 403 stainless steel drive rod coupling, a Type 410 stainless steel drive rod, Alloy X-750 springs, a cobalt-based alloy locking button, and some Type 304 stainless steel parts. No ASME Code specifications are provided for the materials of either the latch assembly or the drive rod assembly, with the exception of Type 403 stainless steel (included in a response to an RAI) and Alloy X-750. However, the latch assembly and the drive rod assembly do not constitute part of the RCPB. Component parts of these assemblies do not have to be designed to the requirements of the ASME Code, nor do the materials have to be procured to ASME specifications or their equivalents. The staff concludes that the selected materials are acceptable because they have provided many years of successful operation in existing nuclear power plants. Therefore, DSER Open Items 4.5.1-2 and 4.5.1-17 are closed.

Westinghouse proposes to use Alloy X-750 for springs in the CRD system. In a response to RAI 252.33, dated March 7, 1995, Westinghouse indicated that the specifications that will define the chemistry, thermal-mechanical processing, and mechanical processing of this alloy had not yet been selected. The staff requested that Westinghouse submit a copy of the specification(s) to be used for this material in the CRD system, and identify and discuss the differences between the selected specification(s) and the recommendations in EPRI NP-7032. This was identified as DSER Open Item 4.5.1-13. In Revision 5, the SSAR was revised to state that the springs in the CRD mechanism are made from nickel-chromium-iron alloy (Alloy X-750), ordered to AMS 5698E or AMS 5699E with additional restrictions on prohibited materials. Operating experience with Alloy X-750 springs has shown that they are not susceptible to stress-corrosion cracking in pressurized water reactor primary water environments, the prime concern behind the recommendations contained in EPRI NP-7032. The staff finds this response acceptable and, therefore, DSER Open Item 4.5.1-13 is closed.

The staff also requested that Westinghouse identify in the SSAR the specific weld metals used in fabricating the CRD system. This was identified as DSER Open Item 4.5.1-4. In Revision 3, Westinghouse revised the SSAR to identify the weld materials to be used specifically in fabricating those components of the CRDM that constitute the RCPB. The materials to be used in welding other (non-RCPB) CRDM components are not explicitly identified as such, but are provided in a general context applicable to all non-RCPB components. The revised SSAR thus identifies all of the weld metals that will be used in fabricating the CRD system. The staff finds this response acceptable and, therefore, DSER Open Item 4.5.1-4 is closed.

The staff requested that Westinghouse provide cutaway sketches of the CRDM in the SSAR, to simplify identification of components and materials used in the CRD, and to facilitate review of the AP600 design. This was identified as DSER Open Item 4.5.1-3. In Revision 4 of the SSAR, Westinghouse provided a cutaway drawing (Figure 3.9-4) of the CRDM that identifies the component parts of the mechanism. Table 5.2-1 in the revised SSAR supplies information on materials for those components that are RCPB. Section 4.5.1.1 of Revision 5 to the SSAR identifies the materials to be used for other components. Thus, Westinghouse has provided the requested information in the SSAR. The staff finds this response acceptable and, therefore, DSER Open Item 4.5.1-3 is closed.

The staff noted that no definitive statement existed in the SSAR or in responses to RAIs about the specific applications of the various nickel-chromium-iron (Ni-Cr-Fe) alloys in the CRD system. The staff therefore requested that Westinghouse identify, in the SSAR, the nickel alloy(s) to be used, as well as their specification, type, grade, and heat treatment. This was identified as DSER Open Item 4.5.1-5. In addition, the staff asked Westinghouse to justify each application of nickel-based alloys and their weld metals in the AP600, except for the reactor coolant pump (RCP) flywheel enclosure, which is discussed in Chapter 5 of this report. The justifications were to address the reason for the choice of one nickel-based alloy over others. This was identified as DSER Open Item 4.5.1-6.

In Revision 3, Westinghouse modified Section 4.5.1.1 of the SSAR to indicate that a Ni-Cr-Fe alloy will be used in fabricating the reactor vessel head penetrations. (Table 5.2-1 identifies this material as Alloy 690.) The appropriate specification for Alloy 690 is included in the revised SSAR, which also states that the material will be in the thermally treated condition. The SSAR also justifies the use of Ni-Cr-Fe alloys in RCPB applications in the AP600 and Westinghouse addressed the superior performance of Alloy 690 in PWR primary water environments in their response to an RAI. Alloy 600 will be used only for cladding or buttering applications. The staff approves of the choice of Alloy 690 (and its equivalent weld metals, Types 52 and 152) as the preferred nickel-based alloy because of its superior corrosion resistance to the reactor coolant environment. Therefore, DSER Open Items 4.5.1-5 and 4.5.1-6 are closed.

In the SSAR, and in responses to RAIs 252.31 and 252.77, dated January 8, 1993, Westinghouse stated that cobalt-based alloys may be used for a few applications where wear resistance is important. These materials will be exposed to primary coolant water and efforts are underway to identify alternative cobalt-free or low cobalt materials, in order to reduce the radioactivity level in the coolant. However, Westinghouse did not make any explicit statements regarding applications where such substitutions were to take place or the extent to which cobalt had been eliminated from the AP600 design. The staff therefore asked Westinghouse to indicate, in the SSAR, the base materials and/or surfacing materials and processes that are to be used in lieu of cobalt-based alloys. Information should also be provided on test programs to qualify such materials and the results of such programs and Westinghouse should present data to ensure a 60-year design life. This was identified as DSER Open Item 4.5.1-7.

In a related matter, the SSAR stated that materials used in the CRD system had been selected for their compatibility with the reactor coolant, on the basis of successful past experience. However, Westinghouse did not provide any data to support this statement for the new materials under consideration as substitutes for cobalt-containing alloys. The staff therefore requested that Westinghouse provide such information. This was identified as DSER Open Item 4.5.1-16.

In Revision 5, Westinghouse revised the SSAR to state that where hardfacing material is used in the latch assembly, a cobalt-based alloy equivalent to Stellite-6, or qualified low- or zero-cobalt substitute is used. Low- or zero-cobalt alloys used for hardfacing or other applications where cobalt alloys have been previously used are qualified using wear and corrosion tests. The corrosion tests qualify the corrosion resistance of the alloy in reactor coolant. Cobalt-free wear-resistant alloys considered for this application include those developed and qualified in industry programs. Westinghouse is committed to continuing efforts to eliminate the use of cobalt-based alloys in the CRD system and to providing the information needed to justify the selection of any alternative materials before incorporating them in the AP600 design. The staff finds the AP600 commitment to reduce the use of cobalt-based alloy acceptable. The limited use of cobalt-based alloys for wear-resistant applications in the baseline RCPB design is also acceptable, given the adequate performance of such materials in similar applications in current nuclear power plants. Therefore, DSER Open Items 4.5.1-7 and 4.5.1-16 are closed.

In its response to RAI 252.33 dated March 7, 1995, regarding the heat treatment of Types 403 and 410 martensitic stainless steel, Westinghouse specified a minimum "annealing" temperature of 607.2°C (1125°F). The staff subsequently asked Westinghouse to replace the word "annealing" with the more appropriate term "tempering." The staff also requested that Westinghouse specify a temperature range for the tempering process and discuss hardness controls. This was identified as DSER Open Item 4.5.1-8. Westinghouse's response to the staff's RAI incorporated the change in terminology and provided information on the tempering temperature range and control of hardness. The tempering of the material determines the hardness of the material and it is controlled by the applicable material specifications. The staff finds this response acceptable and, therefore, DSER Open Item 4.5.1-8 is closed.

Section 4.5.1.4 of the SSAR did not contain any provision to limit tools for power brushing and grinding operations of austenitic stainless steels to use only on stainless steels. Without such provisions, ferritic carbon steel particles could become embedded in the austenitic stainless steel and cause pitting corrosion when exposed to moist atmospheres. The staff therefore requested that Westinghouse indicate, in the SSAR, its plans to control the use of tools for power brushing and grinding of austenitic stainless steels. This was identified as DSER Open Item 4.5.1-9. In Revision 3, Westinghouse modified the SSAR to state that such provisions will be in effect during abrasive work operations on austenitic stainless steels. The staff finds this response acceptable and, therefore, DSER Open Item 4.5.1-9 is closed.

The SSAR contained the statement that pressure boundary parts and components made of stainless steel do not have yield strengths greater than 620.5 MPa (90,000 psi). The staff requested that Westinghouse explain the means used to control this requirement. This was identified as DSER Open Item 4.5.1-14. In a related matter, the staff requested that Westinghouse address how the amount of cold work in austenitic stainless steels will be controlled. This was identified as DSER Open Item 4.5.1-15. In Revision 5, Westinghouse modified the SSAR to address the subject of cold work in austenitic stainless steels, including how the amount will be monitored and controlled in pressure boundary applications. The methods described, which include hardness checks on raw material and process control of bending and similar deformation operations during fabrication, will ensure control over the amount of cold work. The staff finds this response acceptable and, therefore, DSER Open Items 4.5.1-14 and 4.5.1-15 are closed.

Section 4.5.1.2 of the SSAR refers to Section 5.2.3.4 for discussion of the fabrication and processing of austenitic stainless steels and compliance with the recommendations contained in RGs 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and 1.44, "Control of the Use of Sensitized Stainless Steel." The controls imposed on the austenitic stainless steel of the CRDMs regarding control of the use of sensitized stainless steel conform to the recommendations of RG 1.44. Welding of austenitic stainless steel components of the CRD system follow the recommendations of RG 1.31. Section 5.2.3 of this report presents the staff's discussion on this matter.

To minimize the susceptibility of austenitic stainless steel components to stress corrosion cracking, the Advanced Light Water Reactor Utility Requirements Document (URD) developed by the Electric Power Research Institute (EPRI) provided additional recommendations in Paragraph 5.3.1.1. In NUREG-1242, the staff accepted these recommendations. These additional recommendations deal with 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 austenitic stainless steels pertaining to cold work that had not been addressed in the SSAR, and identify those positions which differ from those contained in the EPRI URD. This was identified as DSER Open Item 4.5.1-10. The subject of cold work in austenitic stainless steels is addressed in Section 5.2.3 of Revision 3 to the SSAR, and discussed in Section 5.2.3 of this report. The comments provided therein are equally applicable to the austenitic stainless steel components in the CRD system. The fabrication and processing of austenitic stainless steel parts essentially conform with all the recommendations contained in the EPRI URD, and there are no staff positions that differ from those adopted in the EPRI URD. In Section 5.2.3 of this report, the staff finds Westinghouse's response acceptable and, therefore, DSER Open Item 4.5.1-10 is closed.

The staff also requested that Westinghouse provide information needed to evaluate the compatibility of the CRD system materials with the reactor coolant, as described in Subarticles NB-2160 and NB-3120 of the ASME Code. This was identified as DSER Open Item 4.5.1-11. Westinghouse indicated that the RCPB materials used in the CRD system are compatible with the reactor coolant and, thus, comply with Subarticles NB-2160 and NB-3120 of the ASME Code. Further, the materials selected for the CRD system are currently in use in nuclear power plants and have been proven to perform satisfactorily under the environmental conditions found in these plants. Current experience indicates that they represent the best available selection. Materials being considered as substitutes will be fully qualified before being incorporated in the AP600 design. The staff finds this acceptable and, therefore, DSER Open Item 4.5.1-11 is closed.

Subsection II.3 of SRP 4.5.1 recommends that all materials selected for use in the system be reviewed for their compatibility with the reactor coolant. Section 4.5.1.1 of the SSAR indicates that some non-metallic materials will be used in the CRD system. Section 5.2.8 of the EPRI URD specifies, and the staff has accepted (in NUREG-1242), that the impurity levels of non-metallic materials used within the nuclear steam supply system and associated systems shall be controlled within certain specified limits. Therefore, the staff requested that Westinghouse discuss, in its SSAR, the chemical content controls for non-metallic materials to protect RCPB components, and identify those positions related to chemical content control that differ from the EPRI URD. This was identified as DSER Open Item 4.5.1-12.

In this instance the EPRI URD requirements relate 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 the requirements include such materials as cutting fluids, lubricants, abrasive adhesives, and tape. In Revision 5, Westinghouse modified the SSAR to state that CRDMs are cleaned before delivery in accordance with the guidance provided in NQA-2, "Quality Assurance Requirements for Nuclear Power Plants" Part 2.2. Tools used in abrasive work operations (such as grinding or wire brushing) on stainless steel do not contain ferritic or other materials that could contribute to intergranular stress corrosion cracking. Other non-metallic materials cited in Section 4.5.1.1 of the SSAR are associated with the coil assembly and do not fall within this category of materials. In addition, they are not exposed to the primary coolant and are not subject to non-metallic impurity limits. The staff finds that the AP600 design includes controls in the fabrication, processing, handling, packaging, and shipping of austenitic stainless steel components to ensure contamination protection and cleanliness. Therefore, DSER Open Item 4.5.1-12 is closed.

The cleaning and cleanliness controls included in the design of the CRD system are in accordance with ANSI/ASME NQA-2-1983, "Quality Assurance Requirements for Nuclear Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." These cleaning and cleanliness controls will adequately control contamination of components during fabrication, shipment, and storage.

In reviewing the contents of the initial SSAR submittal (Revision 0) and the responses to RAIs submitted by Westinghouse, the staff found that most of the information available with regard to the CRD system structural materials was of a conclusional or summational nature, rather than a detailed technical presentation of facts that can be evaluated. Thus, the staff could not reach a determination regarding the acceptability of these materials in the DSER. This was identified as DSER Open Item 4.5.1-18. With the information subsequently provided, particularly that included in Revision 3 of the SSAR providing more detailed information on materials, their specifications, and processing parameters, and the bases for their selection, Westinghouse has adequately resolved this concern. Therefore, DSER Open Item 4.5.1-18 is closed.

The staff concludes that the structural materials selected for the CRD system are acceptable and meet the safety requirements of GDCs 1, 14, and 26, as well as 10 CFR 50.55a(a)(1). The staff based this conclusion on the following observations:

- The structural materials selected for components of the control rod drive system have been identified by specification, are in conformance with the requirements of the ASME Code, and are approved for use by ASME Code cases. The selected materials thus meet the requirements of GDC 1 and 10 CFR 50.55a(a)(1) with respect to providing adequate assurance of a quality product commensurate with the importance of the safety function.

- The controls imposed on the components fabricated of austenitic stainless steel satisfy the recommendations of RG 1.31 regarding control of ferrite content in the weld metal, as well as the guidelines of RG 1.44 regarding control of the use of sensitized stainless steel. Aspects related to cold work in austenitic stainless steels conform to the recommendations of the EPRI URD. These controls provide added assurance that stress corrosion cracking will not occur during the design life of the components. Thus, they meet the requirements of GDC 1 and 10 CFR 50.55a(a)(1) with respect to providing adequate assurance of a quality product, as well as the requirements of GDC 14 relative to the prevention of leakage and failure of the RCPB, and the requirements of GDC 26 relative to the capability of reliably controlling reactivity changes.

- Because of their proven satisfactory performance in service, the specified materials are deemed to be compatible with the expected environment and corrosion is expected to be negligible. Thus, the selected materials satisfy the criteria of Subarticles NB- 2160 and NB-3120 of the ASME Code. They also meet the requirements of GDC 14 relative to the prevention of leakage and failure of the RCPB, and the requirements of GDC 26 relative to the capability to reliably control reactivity changes.

- The cleaning and cleanliness controls are in accordance with the recommendations of ANSI/ASME NQA-2-1983 and RG 1.37, and will ensure adequate control of contamination of components during fabrication, shipment, and storage. Conformance with these guidelines will fulfill (in part) the requirements of GDC 1 and 10 CFR 50.55a(a)(1), with respect to providing adequate assurance of a quality product and those of GDC 14 relative to the prevention of leakage and failure of the RCPB.

4.5.2 Reactor Internal and Core Support Materials

GDC 1 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 commensurate with the importance of the safety function to be performed. These quality standards shall be identified and evaluated to determine their adequacy to ensure a quality product in keeping with the required safety function. The NRC staff reviewed the AP600 reactor internal and support materials in accordance with SRP Section 4.5.2 to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met as they relate to the selection of materials for the reactor internals and core support structure.

In the AP600 design, the major material used in fabricating the core support structure and other reactor internals (other than the fuel and control assemblies) is Type 304LN stainless steel. The threaded fasteners are to be fabricated from strain-hardened Type 316 stainless steel. Other relatively minor components are made from a variety of materials according to specialized requirements. Such materials include Stellite 6 hardfacing for wear-resistant surfaces, Type 403 stainless steel for holddown springs, and Type 302 stainless steel for specimen springs.

In reviewing these materials, the staff requested that Westinghouse revise the SSAR to identify the materials used by specification, type, grade, and heat treatment, and to provide sketches of the reactor internals so that components and materials used can be identified. This was identified as DSER Open Item 4.5.2-2. In Revision 3, Westinghouse modified the SSAR to identify all of the materials to be used for the reactor internals, including the parts for which they have been designated. The revision also contained cutaway drawings to identify the main components comprising the AP600 reactor internals.

As noted above, the major core support material is Type 304LN stainless steel, and the threaded structural fasteners are of strain-hardened Type 316 stainless steel. The core support structure and threaded structural fastener materials are specified in the ASME Boiler and Pressure Vessel Code, Section III, Appendix I, as supplemented by Code Cases N-60 and N-4 which are referenced in RG 1.85, "Materials Code Case Acceptability," ASME Section III, Division 1. The Type 304LN stainless steel will be procured to one of three specifications (SA-182, SA-240, or SA-479) depending on the material form required. Remaining parts not fabricated from Type 304LN include keys, inserts, pins, and springs which are non-structural items. The materials for those parts are not required to conform to Section III of the ASME Code. The staff found the information presented in the revised SSAR for these comments acceptable. Therefore, DSER Open Item 4.5.2-2 is closed.

In addition, the staff requested that Westinghouse provide information to confirm that the materials, including the surfacing procedures and processes, selected for all of the reactor internals exposed to the reactor coolant, conform to Subarticles NG-2160 and NG-3120 of Section III of the ASME Code. This was identified as DSER Open Item 4.5.2-1. Westinghouse indicated that the materials selected for the reactor internals and core support structures have demonstrated satisfactory performance in current nuclear power plants, and their selection is consistent with current practices. Corrosion is expected to be negligible on the basis of inservice observations of operating nuclear plants and the results of extensive test programs. Therefore, the materials satisfy the requirements of Subarticle NG-2160 and NG-3120. This is acceptable to the staff. Therefore, DSER Open Item 4.5.2-1 is closed.

Next, the staff requested that Westinghouse identify in the SSAR the specific weld metals used in fabricating the reactor internals. This was identified as DSER Open Item 4.5.2-3. In Revision 5, Westinghouse modified the SSAR to specify the controls to be imposed on welding austenitic stainless steel, including the requirements related to the welding materials themselves. (Westinghouse identified specific weld materials by ASME Code weld analysis designation, A-8, Type 308, 308L, 316, or 316L.) This is acceptable to the staff. Therefore, DSER Open Item 4.5.2-3 is closed.

In its response to RAI 252.44, Westinghouse indicated that cobalt-free and low-cobalt alloys were being considered as substitutes for the cobalt-based hardfacing alloy, and efforts were underway to develop other materials or low-cobalt content alloys. However, Westinghouse did not make any firm statements regarding the specific applications for which such substitutions were being considered, or the extent to which cobalt might be eliminated from the design. Therefore, the staff asked Westinghouse to indicate, in the SSAR, the base materials and/or surfacing (materials and processes) to be used in lieu of cobalt-based alloys. The staff also asked Westinghouse to describe the test programs, the results of such programs, and the extent to which radiation will be decreased by reducing or eliminating cobalt in the reactor vessel and its contents. This was identified as DSER Open Item 4.5.2-4. In Revision 3, Westinghouse modified the SSAR to indicate that the qualification of cobalt-free, wear-resistant alloys for reactor internals applications would be addressed in a fashion similar to that in place for CRD system applications. (See Section 4.5.1 of this report.) Revision 5 of the SSAR stated that, where hardfacing material is used, a cobalt-based alloy equivalent to Stellite-6 or qualified low- or zero-cobalt substitute is also used. Low- or zero-cobalt alloys used for hardfacing or other applications where cobalt alloys have previously been used are qualified using wear and corrosion tests. The corrosion tests qualify the corrosion resistance of the alloy in reactor coolant. Cobalt-free, wear-resistant alloys considered for this application include those developed and qualified in industry programs. Westinghouse is committed to continuing efforts to eliminate the use of cobalt-based alloys in the AP600 design. The staff finds the AP600 commitment to reduce the use of cobalt-based alloy acceptable. The limited use of cobalt-based alloys for wear-resistant applications in the baseline AP600 design is also acceptable because of the adequate performance of such materials in similar applications in current nuclear power plants. Therefore, DSER Open Item 4.5.2-4 is closed.

The staff also requested that Westinghouse revise the SSAR to demonstrate the adequacy of the materials, surfacing processes, and procedures for the 60-year life of the design. This was identified as DSER Open Item 4.5.2-5. The materials selected for the AP600 reactor internals, as given in Revision 3 of the SSAR, are among those currently used in nuclear power plants and have performed satisfactorily in similar applications. The staff therefore concludes that they represent the best available selections in light of current experience. Those materials considered as substitutes will be fully qualified before being incorporated in the design. This is acceptable to the staff. Therefore, DSER Open Item 4.5.2-5 is closed.

Sections 4.5.2.2 and 4.5.2.4 of the SSAR refer to Sections 5.2.3.4 and 1.9 for discussion of the fabrication and processing of austenitic stainless steels and compliance to the guidelines of

RGs 1.31, 1.34, 1.44, and 1.71. The staff has reviewed and accepted these sections. (See Section 5.2.3 of this report for details.) In Paragraph 5.3.1.1, the EPRI URD specifies additional recommendations applicable to the fabrication and processing of austenitic stainless steels. These recommendations address the control of cold work, limitations on its use, and the surface grinding of cold-worked material, and are applicable to those reactor internals and core support components fabricated from austenitic stainless steel. Section 5.2.3.4 of Revision 3 of the SSAR specifically addresses the area of cold work in austenitic stainless steels, indicating that it is in conformance with the recommendations contained in the EPRI URD. In NUREG-1242, the staff concluded that the EPRI URD criteria are acceptable for fabricating cold-worked austenitic stainless steel parts. Thus, the positions adopted in the AP600 design are appropriate.

Recently, revisions to 10 CFR 50.55a(g)(6)(ii)(A) imposed augmented inservice inspection requirements for the reactor vessel upon licensees. In essence, licensees must volumetrically examine 100% of all shell welds in the reactor vessel. (In the past, the designs of the core support structure and reactor internals have limited such examinations.) The staff therefore requested that Westinghouse describe in the SSAR the extent of inspectability of all shell welds. Where core support and reactor internal structures inhibit volumetric examination of the shell welds, the staff further requested that Westinghouse justify the use of such welds. This was identified as DSER Open Item 4.5.2-6. SSAR Section 5.2.4.2, "Arrangement and Inspectability," addresses the inspectability of nuclear power components including reactor shell welds. In Revision 3, Westinghouse modified this discussion to address the inspectability of nuclear power components. Section 5.2.4.2 of this report presents the staff's evaluation of this area. In particular, Westinghouse stated that all items within the Class 1 boundary are designed to provide access for the examinations required by Section XI, IWB-2500, of the ASME Code. Conformance with the requirements of IWB-2500 ensures that 100% of the shell welds in question can be volumetrically examined. This is acceptable to the staff. Therefore, DSER Open Item 4.5.2-6 is closed.

In reviewing the initial SSAR submittal (Revision 0) and the responses to RAIs submitted by Westinghouse, the staff found that most of the information available on the reactor internals and core support materials was of a conclusional or summational nature, rather than a detailed technical presentation of facts that can be evaluated. Thus, the staff could not, in the DSER, reach a conclusion regarding the acceptability of these materials. This was identified as DSER Open Item 4.5.2-7. With the information provided subsequently, particularly that included in Revision 3 of the SSAR, Westinghouse has adequately resolved this concern. Therefore, DSER Open Item 4.5.2-7 is closed.

The staff concludes that the design, fabrication, and testing of materials selected for use in the construction of the reactor internals and core support structure are acceptable and meet the applicable requirements of GDC 1 and 10 CFR 50.55a with respect to providing adequate assurance of a quality product commensurate with the importance of the safety function. The staff based its conclusion on the following observations:

- The materials selected for components of the reactor internals and core support structure have been identified by specification, are in conformance with the requirements of the ASME Code, and are approved for use by ASME Code cases. Moreover, the proven satisfactory performance of the specified materials in service demonstrates that they are compatible with the expected environment, and corrosion is expected to be negligible.

- The controls imposed upon the components fabricated of austenitic stainless steel satisfy the recommendations of RG 1.31 regarding control of ferrite content in the weld metal, as well as the guidelines of RG 1.44 regarding control of the use of sensitized stainless steel. Aspects related to cold work in austenitic stainless steels conform to the recommendations of the EPRI URD that were reviewed and accepted by the NRC staff.

As a result, the staff concludes that Westinghouse has provided reasonable assurance that the materials used for the reactor internals and core support structure will be in a metallurgical condition to preclude inservice deterioration. Compliance with the requirements of the ASME Code and with the recommendations of the RGs constitutes an acceptable basis for meeting the relevant requirements of GDC 1 and 10 CFR 50.55a.

4.6 Functional Design of Reactivity Control Systems

The staff reviewed the SSAR to confirm that the design of the AP600 reactivity control systems has the capability to satisfy the following reactivity control conditions for all modes of plant operations:

- Vary power level from full power to hot shutdown and have power distributions within acceptable limits at any power level.
- Shut down the reactor to mitigate the effects of postulated events discussed in Chapter 15 of this report.

The reactivity control systems for the facility are the CRDs, the reactor trip system, and the passive core cooling system. No credit is taken for the boration capabilities of the chemical and volume control system (CVS).

The CRD system contains a magnetically-operated jack (magjack). When electrical power is removed from the coils of the magjack, armature springs automatically disengage holding latches from the magjack's drive shaft, allowing insertion of the control rod and the gray rods by gravity. There are 45 full-strength RCCAs and 16 gray rod cluster control assemblies. The regulating CRD system may be used to compensate for changes in reactivity associated with power-level changes and power distribution, variations in moderator temperature, or changes in boron concentration. The gray rods, which have lower worth than the full-strength control rods, control reactivity and axial power shape during power operations.

The CVS is a non-safety-grade system designed to control slow or long-term reactivity changes, such as those caused by fuel burnup and variations in coolant temperature and xenon concentration. The CVS controls reactivity by adjusting the dissolved boron concentration in the RCS. The boron concentration is controlled to obtain optimum RCCA positioning, to compensate for reactivity changes during startup, load-following (changes in reactor power level based on electrical demand), and shutdown, and to provide shutdown margin for maintenance and refueling operations. The boric acid concentration in the RCS is controlled by the charging and letdown portions of the CVS.

The CVS can be used to maintain reactivity within the required bounds by means of the automatic makeup system, which replaces minor coolant leakage without significantly changing the boron concentration in the RCS system. Dilution of the RCS boron concentration is required to compensate for reactivity losses from fuel depletion. Dilution is accomplished by manual operation of the CVS. The CVS is discussed further in Section 9.3.6 of the SSAR, as well as Section 9.3.6 of this report.

The CRD system is the primary shutdown mechanism for normal operation, accidents, and transients. Using this system, control rods automatically insert in accident and transient conditions. In the event of a LOCA, steamline break, loss of normal feedwater flow, steam generator tube rupture, or control rod ejection, the passive reactor cooling system provides concentrated boric acid solution to the RCS. (Chapter 15 of the SSAR describes the design-basis analyses of these events.)

The operability of the CRD system is tested by changing the position of the CRDMs. These tests verify that the CRDMs meets design requirements for reactor trip time. The trip time requirement is confirmed for each CRDM before initial reactor operation, and at periodic intervals after initial reactor operation, as required by the TS. At every refueling outage, the CRD system is stepped over the entire range of movement, and the RCCAs are drop-tested to demonstrate trip time capability.

The CRD system is designed such that a single failure will not result in loss of the protection system, and removing a channel or component from service will not result in a loss of redundancy. These matters are discussed further in Section 7.2 of this report. Periodic testing and operability of the CRD system is verified by AP600 technical specifications. The provisions for periodic testing, reliability, and redundancy conform to the requirements of GDC 21, "Protection System Reliability and Testability."

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The schematic and layout of the Westinghouse AP600 reactor coolant system (RCS) and its principal auxiliary systems are shown in Figures 5.1-1 through 5.1-3 of the standard safety analysis report (SSAR). The RCS consists of two heat transfer circuits (loops), each with a U-tube steam generator, two reactor coolant pumps, and a single hot leg pipe and two cold leg pipes for circulation of reactor coolant. The RCS also includes the pressurizer, interconnecting piping, valves, and instrumentation for operational control, actuation, and monitoring of plant safety systems. All RCS equipment is located in the reactor containment.

The reactor coolant pressure boundary (RCPB) provides a barrier against the release of radioactivity generated within the reactor. It is designed to provide a high degree of integrity throughout operation of the plant.

5.1.1 Design Bases

In Section 5.1.1 of the SSAR, Westinghouse lists the following design bases for the RCS and its major components:

- The RCS transfers to the steam and power conversion system the heat produced during power operation, as well as the heat produced when the reactor is subcritical (including the initial phase of plant cooldown).

- The RCS transfers to the normal residual heat removal system (RNS) the heat produced during the subsequent phase of plant cooldown and cold shutdown.

- During power operation and normal operational transients (including the transition from forced to natural circulation), the RCS removes heat and maintains fuel condition within the operating bounds permitted by the reactor control and protection systems.

- The RCS provides the water used as the core neutron moderator and reflector, supplementing the metal radial reflector located outside the core, in conserving thermal neutrons and improving neutron economy. It also provides the water used as a solvent for the neutron absorber used in chemical shim reactivity control.

- The RCS maintains the homogeneity of the soluble neutron poison concentration and the rate of change of the coolant temperature so that uncontrolled reactivity changes do not occur.

- The RCS pressure boundary accommodates the temperatures and pressures associated with operational transients.

- The reactor vessel supports the reactor core and control rod drive mechanisms.
- The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, the pressurizer accommodates volume changes in the reactor coolant.
- The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.
- The steam generators provide high-quality steam to the turbine. The tubes and tubesheet boundary prevent the transfer of radioactivity generated within the core to the secondary system.
- The RCS piping contains the coolant under operating temperature and pressure conditions and limits leakage (and activity release) to the containment atmosphere. The RCS piping contains demineralized and borated water that is circulated at the flow rate and temperature consistent with achieving the reactor core thermal and hydraulic performance.
- The RCS is monitored for loose parts, as described in Section 4.4.6 of the SSAR.
- Applicable industry standards and equipment classifications of RCS components are identified in Tables 3.2-1 and 3.2-3 of Section 3.2.2 of the SSAR.
- The reactor vessel head is equipped with suitable provisions for connecting the head vent system, which meets the requirements of 10 CFR 50.34(f)(2)(vi) (Three Mile Island [TMI] Action Item II.B.1).
- The pressurizer surge line and each loop spray line connected with the RCS are instrumented with resistance temperature detectors (RTDs) attached to the pipe to detect thermal stratification.

5.1.2 Design Description

The following components are included in the AP600 RCS:

- the reactor vessel, including control rod drive mechanism housings
- the reactor coolant pumps, comprised of four canned motor pumps, which transfer fluid through the entire reactor coolant and reactor systems
- the primary portion of the steam generators containing reactor coolant, including the channel head, tubesheet, and tubes
- the pressurizer, which is attached by the surge line to one of the reactor coolant hot legs
- the pressurizer safety valves and automatic depressurization system valves

- the reactor vessel head vent isolation valves
- the interconnecting piping and fittings between the system components
- the piping, fittings, and valves leading to connecting auxiliary or support systems

The principal system pressures, temperatures, flow rates, the system design and operating parameters, and the thermal-hydraulic parameters of the RCS are specified in Tables 5.1-1 through 5.1-3 of the AP600 SSAR.

During operation, the reactor coolant pumps circulate pressurized water through the reactor vessel and then, through the steam generators. The water, which serves as coolant, moderator, and solvent for boric acid (chemical shim control), is heated as it passes through the reactor core. Heat is removed from the water and transferred to the main steam system in the steam generators. The water is then returned to the reactor vessel by the reactor coolant pumps to repeat the heat removal cycle.

RCS pressure is controlled by operation of the pressurizer, where water and steam are maintained in equilibrium by the activation of electrical heaters, or a water spray, or both. Steam is formed by the heaters or condensed by the water spray to control pressure variations resulting from expansion and contraction of the reactor coolant.

Spring-loaded safety valves are connected to the pressurizer to provide overpressure protection for the RCS. These valves discharge into the containment atmosphere. Also attached to the pressurizer are two redundant sets of RCS automatic depressurization system (ADS) valves. These valves discharge steam and water (in three stages of operation) through spargers located in the in-containment refueling water storage tank (IRWST). The IRWST is part of the AP600's passive core cooling system.

Two fourth-stage automatic depressurization valves are connected by two redundant paths to the RCS's hot legs. These valves discharge directly to the containment atmosphere.

The RCS is also served by a number of auxiliary systems:

- the chemical and volume control system (CVS)
- the passive core cooling system (PXS)
- the normal residual heat removal system (RNS)
- the steam generator system (SGS)
- the primary sampling system (PSS)
- the liquid radwaste system (WLS)
- the component cooling water system (CCS)

5.1.3 System Components

In Section 5.1.3 of the SSAR, Westinghouse describes the major components of the RCS, as follows.

5.1.3.1 Reactor Vessel

The reactor vessel is cylindrical, with a hemispherical bottom head and a removable, flanged, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The vessel interfaces with the reactor internals, the integrated head package, and reactor coolant loop piping. It is supported by the containment building concrete structure.

The design of the AP600 reactor vessel closely matches the existing vessel designs of Westinghouse's three-loop plants. New features for the AP600 have been incorporated without departing from the proven features of existing vessel designs.

The reactor vessel has inlet and outlet nozzles positioned in two horizontal planes between the upper head flange and the top of the core. The nozzles are located in this configuration to provide an acceptable cross-flow velocity in the vessel outlet region, and to facilitate optimum layout of the RCS equipment. The inlet and outlet nozzles are offset, with the inlet positioned above the outlet, to allow mid-loop operation for removal of a main coolant pump without discharge of the core.

Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

5.1.3.2 Steam Generators

Each steam generator (SG) is a vertical shell and U-tube evaporator with integral moisture separating equipment. The basic SG design and features are similar to previous Westinghouse SGs, including replacement SG designs.

The SSAR describes several design enhancements to the AP600 SGs. These include nickel-chromium-iron Alloy 690 thermally treated tubes on a triangular pitch, broached tube support plates, improved anti-vibration bars, single-tier separators, enhanced maintenance features, and a primary-side channel head design that allows easy access and maintenance by robotic tooling. All tubes in the SG are accessible for sleeving, if necessary.

The basic function of the AP600 SG is to transfer heat from the single-phase reactor coolant water through the U-shaped heat exchanger tubes to the boiling, two-phase steam mixture in the secondary side of the SG. The SG separates dry, saturated steam from the boiling mixture, and delivers the steam to a nozzle from which it is delivered to the turbine. Water from the feedwater system replenishes the SG water inventory by entering the SG through a feedwater inlet nozzle and feedring.

In addition to its steady-state performance function, the SG secondary side provides a water inventory that is continuously available as a heat sink to absorb primary side high-temperature transients.

5.1.3.3 Reactor Coolant Pumps

Each reactor coolant pump (RCP) is a high-inertia, high-reliability, low-maintenance, hermetically sealed canned motor pump that circulates reactor coolant through the reactor vessel, loop piping, and SGs. The AP600 design uses four RCPs. Two pumps are coupled with each SG. The pumps are integrated into the SG channel head.

The integration of the pump suction into the bottom of the SG channel head eliminates the cross-over leg of coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support system for the SG, pumps, and piping; and reduces the potential for uncovering the core by eliminating the need to clear the loop seal during a small loss-of-coolant accident (LOCA).

Each AP600 is a vertical, single-stage centrifugal pump designed to pump large volumes of coolant at high pressures and temperatures. The pump impeller attaches to the rotor shaft of the driving motor, which is an electric induction motor. Both the stator and rotor are encased in corrosion-resistant cans constructed and supported to withstand full system pressure. Because of the RCP's canned design, shaft seals are eliminated in the AP600 design. To provide the rotating inertia needed for flow coast-down, a uranium alloy flywheel is attached to the pump shaft.

5.1.3.4 Primary Coolant Piping

RCS piping is configured with two identical main coolant loops, each of which employs a single 78.34 cm (31 in.) inside diameter hot leg pipe to transport reactor coolant to a SG. The two reactor coolant pump suction nozzles are welded directly to the outlet nozzles on the bottom of the SG channel head. Two 55.88 cm (22 in.) inside diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the reactor vessel to complete the circuit. The loop configuration and material have been selected such that pipe stresses are sufficiently low for the primary loop and large auxiliary lines to meet the requirements to demonstrate "leak-before-break" (LBB). Thus, pipe rupture restraints are not required, and the loop is analyzed for pipe ruptures only for small auxiliary lines that do not meet the LBB requirements.

5.1.3.5 Pressurizer

The pressurizer is the principal component of the RCS pressure control system. This is a vertical, cylindrical vessel with hemispherical top and bottom heads, where liquid and vapor are maintained in equilibrium, saturated conditions.

A 10.16 cm (4 in.) spray nozzle and two 35.56 cm (14 in.) nozzles for connecting the safety and depressurization valve inlet headers are located in the top head. Electrical heaters are installed through the bottom head. The heaters are removable for replacement. The bottom head contains the nozzle for attaching the surge line. This line, which connects the pressurizer to a hot leg, provides for the flow of reactor coolant into and out of the pressurizer during RCS thermal expansions and contractions.

5.1.3.6 Pressurizer Safety Valves

The two pressurizer safety valves are spring-loaded and self-actuated with back-pressure compensation. Valve set pressure is 17.23 MPa (2485 psig). Their combined capacity is determined by the requirement to not exceed maximum RCS pressure limit during the Level B service condition loss-of-load transient., i.e., 110 percent of the RCS design pressure of 17.23 MPa (2485 psig), in compliance with the ASME Code, Section III.

5.1.3.7 Automatic Depressurization Valves

Several of the passive safety features of the AP600 design are dependent on depressurization of the RCS. This is accomplished by the ADS valves located above the pressurizer (Stages 1 to 3) and attached to the RCS hot legs (Stage 4). The Stage 1 to 3 valves are arranged in six parallel sets (two valves in series) opening in three stages. The Stage 4 ADS valves consist of four paths, each path having two valves in series. To mitigate the consequences of the various accident scenarios, the ADS valves are arranged to open in a prescribed sequence determined by core makeup tank level and a sequence timer. A more detailed description of the ADS valves is included in Sections 5.4.6 and 6.3 of the AP600 SSAR.

5.1.4 System Performance Characteristics

Section 5.1.4 of the SSAR discusses the thermal-hydraulic parameters, system performance parameters and supporting design procedures used to establish the performance characteristics of the AP600 RCS. The detailed design procedure establishes a best-estimate flow and conservatively high and low flows for the applicable mechanical and thermal design considerations. In establishing the range of design flows, the procedure accounts for uncertainties in the component flow resistances and in pump head-flow capability. The procedure also accounts for the uncertainties in the technique used to measure flow in the operating plant. Section 5.1.4 of the SSAR also defines the four reactor coolant flows that are applied in plant design considerations, which are described as follows.

5.1.4.1 Best Estimate Flow

The best-estimate flow is the most likely value for the normal full-power operating condition. This flow value is determined by the best estimate of fuel, reactor vessel, SG, and piping flow resistances, and on the best estimate of the RCP head and flow capability. No uncertainties are assigned to either the system flow resistance or the pump head. The best-estimate flow provides the basis for the other design flows required for the system and component design. The best-estimate flow and head also define the performance requirement for the RCP. Table 5.1-3 of the SSAR lists system pressure losses on the basis of best-estimate flow.

Although the best-estimate flow is the most likely value to be expected in operation, more conservative flow rates (such as thermal design flow rate and mechanical design flow rate) are applied in the thermal and mechanical designs.

5.1.4.2 Minimum Measured Flow

The minimum measured flow is specified in the technical specifications (TS) as the flow that must be confirmed or exceeded by the flow measurements obtained during plant startup. This is the flow used in reactor core departure from nucleate boiling (DNB) analysis for the AP600 thermal design procedure. In the thermal design procedure methodology for DNB analysis, flow measurement uncertainties are combined statistically with fuel design and manufacturing uncertainties. The measured reactor coolant flow will most likely differ from the best-estimate flow because of uncertainties in the hydraulics analysis and inaccuracies in the instrumentation used to measure flow. The measured flow is expected to fall within a range around the best-estimate flow. The magnitude of the expected range is established by statistically combining the system hydraulics uncertainty with the total flow rate within the expected range, less any excess flow margin that may be provided to account for future changes in the hydraulics of the RCS.

5.1.4.3 Thermal Design Flow

The thermal design flow is the conservatively low value used for thermal-hydraulic analyses where the design and measurement uncertainties are not combined statistically. Additional flow margin must therefore be explicitly included. The thermal design flow is derived by subtracting the plant flow measurement uncertainty from the minimum measured flow. The thermal design flow is approximately 4.5 percent less than the best-estimate flow. The thermal design flow is confirmed when the plant is placed in operation. Table 5.1-3 of the SSAR presents important design parameters founded on the thermal design flow.

5.1.4.4 Mechanical Design Flow

Mechanical design flow is the conservatively high flow used as the basis for the mechanical design of the reactor vessel internals, fuel assemblies, and other system components. Mechanical design flow is established at 104 percent of best-estimate flow.

5.2 Integrity of Reactor Coolant Pressure Boundary

Section 5.2 discusses the measures to provide and maintain the integrity of the RCPB during power operation. 10 CFR 50.2 defines the RCPB as vessels, piping, pumps, and valves that are part of the RCS or that are connected to the RCS, up to and including the following:

- the outermost containment isolation valve in the system piping that penetrates the containment
- the second of the two valves closed during normal operation in the system piping that does not penetrate the containment
- the RCS overpressure protection valves

5.2.1 Compliance With Code and Code Cases

GDC 1 requires that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement is applicable to both pressure-retaining and non-pressure-retaining SSCs that are part of the RCPB and other systems important to safety. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability.

The staff reviewed the measures used to provide and maintain the integrity of the RCPB and other pressure-retaining components and their supports that are important to safety for the design lifetime of the plant.

5.2.1.1 Compliance With 10 CFR 50.55a

According to 10 CFR 50.55a, components important to safety are subject to the following requirements:

(1) RCPB components must meet the requirements for American Society of Mechanical engineers (ASME) Class 1 (Quality Group (QG) A components specified in ASME Boiler and Pressure Vessel Code, Section III, except for those components that meet the exceptions of 10 CFR 50.55a(c)(2). Those RCPB components that meet these exceptions may be classified as Class 2 (QG B), or Class 3 (QG C).

(2) In accordance with 10 CFR 50.55a(d) and (e), components classified as QG B and C must meet the requirements for Class 2 and 3 components, respectively, as specified in ASME Code, Section III.

SSAR Tables 3.2-1 and 3.2-3 and applicable piping and instrumentation diagrams (P&IDs) collectively classify the mechanical and pressure-retaining components of the RCPB that do not meet the exclusion requirements discussed in (1) above, as ASME Code, Section III, Class 1 components. These Class 1 components are designated QG A in conformance with Regulatory Guide (RG) 1.26, Quality Group Classifications and Standards for Water-, Steam-, and *Radioactive-Waste-Containing Components of Nuclear Power Plants*, Revision 3. The staff has reviewed the SSAR tables and P&IDs mentioned above, supplemented by the evaluation of the quality group classification discussed in Section 3.2.2 of this report, and concludes that AP600 mechanical and pressure-retaining components in the RCPB have been acceptably classified as ASME Class 1 (QG A) in accordance with 10 CFR 50.55a, and are consistent with applicable portions of the NRC Standard Review Plan (SRP) Section 5.2.1.1.

In addition to the QG A components of the RCPB, certain lines that will perform a safety function and that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified as QG B or C in accordance with Positions C.1 or C.2 of RG 1.26, Revision 3, and will be constructed as ASME Code, Section III, Class 2 or Class 3 components.

As discussed in SSAR, Revision 22, Sections 5.2.1.1 and 5.2.1.3, the portion of the CVS inside containment that is defined as part of the RCPB uses an alternate quality group classification to that discussed above. This portion of the CVS is classified as non-safety, Class D. The safety-related classification of the RCPB ends at the third isolation valve between the RCS and the CVS (Ref. SSAR Fig. 9.3.6-1). This is considered to be an alternate to the usual classification of the RCPB. 10 CFR 50.55a(a)(3) allows alternatives to 10 CFR 50.55a(c) requirements if the proposed alternative design provides an acceptable level of quality and safety. The applicant has provided the following design enhancements to the Class D portion of the CVS as an alternate design to meet an acceptable level of quality and safety:

- The isolation valves between the RCS and the CVS are ASME Class 1 valves designed and qualified for design conditions that include closing against blowdown full flow with full system differential pressure. In addition, although these valves are not classified as pressure isolation valves, SSAR Table 3.9.16, Revision 22 provides a commitment that at each refueling outage, these valves will be leak tested to the same leak rate criteria that is specified in the AP600 technical specifications for pressure isolation valves.

Implementation of these additional leak rate tests will provide redundant leak tight barriers, when required, in each of the lines that connect the RCS and CVS.

- The AP600 design also contains a third valve in each of the lines that connect the RCS and CVS. These third valves are in addition to the Class 1 valves discussed in the above design enhancement, and they will provide additional assurance that the RCS will be isolated in the event of a CVS failure.

- Although the Class D portions of the CVS are non-seismic, those portions inside containment will be analyzed to the same seismic design criteria as that accepted by the staff for Seismic Category II piping. The staff's acceptance of this criteria is discussed in Section 3.12.3.7 of this report. The seismic Category II analyses will provide adequate assurance that the loads resulting from an SSE will not result in a loss of structural integrity of the CVS piping.

- All of the Class D portion of the CVS is constructed of or clad with corrosion-resistant material such as Type 304 or Type 316 stainless steel that is compatible with the reactor coolant. In addition, this portion of the CVS is designed to a design pressure of 21.4 MPa (3100 psi), which exceeds the RCS design pressure.

Based on the above design enhancements that have been added to the Class D portion of the CVS, the staff considers that the alternative design provides an acceptable level of quality and safety and is, therefore, acceptable.

In SSAR Section 5.2.1.1, Westinghouse states that the baseline code used to support the AP600 SSAR is the ASME Code, Section III, 1989 Edition, 1989 Addenda. In the draft safety evaluation report (DSER), the staff stated that, currently 10 CFR 50.55a(b)(1) only endorses ASME Section III through the 1989 Edition. In request for additional information (RAI) 210.112, the staff requested that Westinghouse identify in the SSAR the specific portions of the 1989 Addenda that are being used in the AP600 design and analysis. This was DSER Open Item 5.2.1.1-1. The response to RAI 210.112 did not provide the requested information. In a letter to Westinghouse dated August 25, 1997, the staff identified its specific concerns relative to the 1989 Addenda, and requested that Westinghouse either remove the reference to the 1989 Addenda of ASME Section III from the SSAR, or include a statement in the SSAR that the AP600 design does not rely on the portion of the 1989 Addenda that the staff finds unacceptable. In response to this request, Revision 17 to the SSAR revised Section 5.2.1.1 to state that for the AP600 design, the ASME Code, Section III, 1989 Edition, 1989 Addenda will be used with the exception that for fillet welded or socket welded joints, the rules in the 1989 Edition will be used rather than those in the 1989 Addenda. This is consistent with the staff's position on the 1989 Addenda that was documented in the August 25, 1997 letter, and is acceptable. Therefore, DSER Open Item 5.2.1.1-1 is closed. Furthermore, any proposed change to the use of the ASME code editions or addenda, by a COL applicant or licensee, will require NRC approval prior to implementation.

The ASME Code is Tier 1 information and the specific edition and addenda are designated Tier 2* because of the continually evolving design and construction practices (including inspection and examination techniques) of the Code. Establishing a specific edition and addenda during the design certification stage might result in inconsistencies between design and construction practices during the detailed design and construction stages. The ASME Code involves a consensus process to reflect the evolving design and construction practices of the industry. Although reference to a specific edition of the Code for the design of ASME Code class components and their supports is necessary to reach a safety finding during the design certification stage, it is also important that the construction practices and examination methods of an updated Code be consistent with the design practices established at the design certification stage. To avoid this potential inconsistency for the AP600 pressure-retaining components and their supports, proposed changes to the specific edition and addenda require NRC approval at the COL stage before implementation. This provides the COL applicant with the option to revise or supplement the referenced Code edition with portions of the later Code editions and addenda to ensure consistency between the design and construction practices. However, the staff finds that there might be a need to establish certain design parameters from a specific Code edition or addenda during its design certification review, particularly when that information is important for establishing a significant aspect of the design or is used by the staff to reach its final safety determination. Such considerations, if necessary, are reflected in the various sections of this report. Therefore, all ASME Code Class 1, 2, and 3 pressure-retaining components and their supports shall be designed in accordance with the requirements of ASME Code, Section III, using the specific edition and addenda given in the SSAR. The COL applicant should ensure that the design is consistent with the construction practices (including inspection and examination methods) of the ASME Code edition and addenda, as endorsed in 10 CFR 50.55a. This was COL Action Item 5.2.1.1-1 and Open Item 5.2.1.1-2 in the DSER. SSAR Section 5.2.6.1, "ASME Code and Addenda," contains a commitment that the COL applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code edition and addenda. This is an acceptable commitment. Therefore, DSER Open Item 5.2.1.1-2 is closed.

The COL applicant should identify in its application the portions of the later code editions and addenda that it requests approval to adopt. This was COL Action Item 5.2.1.1-2 and Open Item 5.2.1.1-3 in the DSER. In Revision 3 to the SSAR, Section 5.2.6.1 was added to include an acceptable COL commitment to provide this information for staff review. Therefore, DSER Open Item 5.2.1.1-3 is closed.

Conclusion

On the basis of the above evaluations, the staff concludes that the construction of all ASME Code, Class 1, 2, and 3 components and their supports for the AP600 plant will conform to the appropriate ASME Code editions and addenda and the Commission's regulations, and that component quality will be commensurate with the importance of the safety function of all such components and their supports. This constitutes an acceptable basis for satisfying GDC 1 and is acceptable.

5.2.1.2 Applicable Code Cases

In the DSER, the staff stated that it was still reviewing a new SSAR Table 5.2-3, which was proposed by Westinghouse in response to Q210.109. This table identifies specific ASME Section III Code cases that will be applied in the construction of pressure-retaining ASME Code, Section III, Class 1, 2, and 3 components in the AP600 plant. This was identified as DSER Open Item 5.2.1.2-1. The staff's review of this table is founded on the guidelines in RG 1.84, "Design and Fabrication Code Case Acceptability - ASME Section III, Division 1," and RG 1.85, "Materials Code Case Acceptability - ASME Section III, Division 1." All ASME Section III Code cases that have been either conditionally or unconditionally endorsed by the staff are discussed in one of these RGs, as applicable. The staff transmitted the results of its review to Westinghouse in a letter dated May 10, 1995. In this letter, the staff requested several changes to the table. These changes were necessary because some of the requested Code cases are recent revisions to existing cases. Although these revisions have not yet been accepted in either of the above RGs, the previous versions of these cases have been conditionally endorsed in applicable RGs. 10 CFR 50.55a, Footnote 6, allows the use of ASME Code cases that have not been endorsed by either RG 1.84 or RG 1.85 if 10 CFR 50.55a(a)(3) is satisfied. 10 CFR 50.55a(a)(3) allows alternatives to 10 CFR 50.55a(c), (d), and (e) if the proposed alternative provides an acceptable level of quality and safety. The staff's evaluation determined that, because these recent revisions contain no safety-related concerns for the AP600, a commitment to the applicable RG conditions of endorsement for the previous versions of each Code case provides an acceptable level of quality and safety for the AP600 design. In Revision 3 to the SSAR, Westinghouse revised Table 5.2-3 to provide commitments to meet the conditions of endorsement in RGS 1.84 or 1.85 for the previous versions of each applicable Code case. Therefore, with the exception of Code Cases 2142 and 2143 discussed below, SSAR Table 5.2-3, through Revision 15, contains ASME Section III Code cases that are either endorsed in RGs 1.84 or 1.85, or have been determined to be acceptable to the staff as discussed above. Therefore, DSER Open Item 5.2.1.2-1 is closed.

Code Case (CC) 2142, "F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section IX," and CC 2143, "F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode, Section IX" are also listed in the proposed Table 5.2-3. These cases will not be included in RG 1.85 because they are not ASME Section III Code Cases. However, these cases are acceptable because they include weld metal to be used in the welding of Ni-Cr-Fe Alloy 690, which the staff endorsed and accepted for use in its SER for the Electric Power Research Institute (EPRI) advanced light water reactor Utility Requirements Document (URD), Volume III.

The only acceptable ASME Code cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the AP600 standard plant are those either conditionally or unconditionally approved in RGs 1.84 and 1.85 in effect at the time of design certification, or determined to be conditionally acceptable as discussed above. However, the COL applicant may submit, with its COL application, future code cases that are endorsed in RGs 1.84 and 1.85 at the time of the application provided they do not alter the staff's safety findings on the AP600 certified design. In addition, the COL applicant should submit those Code cases which are in effect at the time of the COL application that are applicable to RG 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI, Division 1."

Conclusion

On the basis of the above evaluation, the staff concludes that all of the Code cases in SSAR Table 5.2-3 either meet the guidelines of RG 1.84 or 1.85 or have been reviewed and endorsed by the staff and are acceptable for use on the AP600 design. Compliance with the requirements of these Code Cases will result in a component quality that is commensurate with the importance of the safety functions of these components, constitutes the basis for satisfying GDC 1, and is acceptable.

5.2.2 Overpressure Protection

In the AP600 design, overpressure protection for the RCS and steam system pressure boundaries is provided by the pressurizer safety valves (PSVs) and the SG safety valves (SGSVs) during normal operation, and a relief valve in the suction line of the RNS during low temperature operation, in conjunction with the action of the reactor protection system. There are two PSVs, six SGSVs with three valves located in the safety-related portion of each main steam piping upstream of the main steam isolation valve, and one relief valve in the suction line of the RNS in the AP600 design. Combinations of these systems provide compliance with the overpressure protection requirements of the ASME Code, Section III, Paragraphs NB-7300 and NC-7300, for pressurized-water reactor (PWR) systems. The ASME code requires the total relieving capacity be sufficient to prevent a pressure rise of more than 10 percent above the design pressure of the RCS and steam generators under any expected system pressurization transient conditions. The RNS suction relief valve for low-temperature over pressure protection prevents the RCS from exceeding the pressure-temperature limits determined from the ASME Code, Appendix G analyses.

The staff's review of the AP600 overpressure protection was performed in accordance with Section 5.2.2 of the SRP, including Branch Technical Position (BTP) RSB 5-2. The staff reviewed the following sections of the SSAR:

- 5.2.2, Overpressure Protection
- 5.4.5, Pressurizer
- 5.4.7, Normal Residual Heat Removal System
- 5.4.9, RCS Pressure Relief Devices
- 5.4.11, Pressurizer Relief Discharge System
- 10.3.2.2.2, Main Steam Safety Valves

5.2.2.1 Overpressure Protection During Power Operation

During power operation, overpressure protection for the RCS is provided by the two PSVs, six SGSVs, and the reactor protection system to maintain the primary and secondary pressures within 110 percent of their respective design pressures. The details of the SGSV design are discussed in Section 10.3 of the SSAR with design data, including set pressures and relieving capacities, listed in SSAR Table 10.3.2-2. The design parameters of the PSVs are specified in Table 5.4-17 of the SSAR. The discharge of the PSV is routed through a rupture disk to containment atmosphere. The rupture disk, which has a pressure rating substantially less than the set pressure of the PSV, is to contain leakage past the PSV.

The PSVs are sized as determined by the analysis of a complete loss of steam flow to the turbine, with the reactor operating at 102 percent of rated power. This design-basis event bounds other events that could lead to overpressure of the RCS if adequate overpressure protection were not provided. Such overpressure events include loss of electrical load and/or turbine trip, uncontrolled rod withdrawal at power, loss of reactor coolant flow, loss of normal feedwater, and loss of offsite power to the station auxiliaries. The total PSV capacity is required to be at least as large as the maximum surge rate into the pressurizer during this transient. In this analysis, feedwater flow is also assumed to be lost, and steam relief through the SGSVs is considered when the secondary side pressure reaches 103 percent of the SG shell design pressure. No credit is taken for operation of the pressurizer level control system, pressurizer spray system, rod control system, steam dump system, or steamline power-operated relief valve. The reactor is maintained at full power with no credit taken for reactor trip or reactivity feedback during the transient. A 3-percent set pressure accumulation is also considered for the PSV relief. These assumptions meet the acceptance criteria of II.A of Section 5.2.2 of the SRP. With these assumptions, the results of Westinghouse's analysis (Westinghouse submittal, "Revised RAI 440.75 Response for Pressurizer Safety Valve Sizing," NSD-NRC-97-5146, June 2, 1997) indicate that a PSV capacity of 355,454 kg/hr (782,000 lbm/hr) at the RCS pressure of 17.75 MPa (2575 psia) is sufficient to carry the maximum pressurizer volumetric insurge flow following a complete loss of load and feedwater from 102 percent of rated power. The rated relieving capacity of the PSVs is at least 181,818 kg/hr (400,000 lbm/hr) per valve, which is in excess of the capacity required to prevent exceeding 110 percent of system design pressure. Design-basis safety analyses of the overpressure events, described in Sections 15.2, 15.3, and 15.4 of the SSAR, demonstrate that the capacities and setpoints of the PSVs and SGSVs are sufficient to ensure that the pressures of the RCS and the SGs remain below 110 percent of their design pressures. The PSV and SGSV setpoints and relieving capacities are, therefore, acceptable.

The PSV set pressure of 17.24 MPa (2485 psig) is specified in the limiting condition for operation (LCO) for AP600 technical specification 3.4.7, with a tolerance of plus/minus one percent. The PSVs are part of the RCPB and ASME B&PV Code Class 1 components. These valves are tested and analyzed using the design transients, loading conditions, seismic considerations, and stress limits for Class 1 components discussed in Sections 3.9.1, 3.9.2, and 3.9.3 of the SSAR. The staff evaluation of these sections are discussed in the corresponding sections of this report. In addition, the PSVs are subjected to the verification program established by EPRI to address the requirements of 10 CFR 50.34(f)(2)(x) to qualify their operation for all fluid conditions expected under operating conditions, transients and accidents. This is addressed in Item II.D.1, "Performance Testing of PWR Safety and Relief Valves," in Chapter 20 of this report. The PSVs are also subject to the surveillance specification of technical specification 3.4.7.1 and the inservice testing program requirements specified in SSAR Table 3.9-16.

As discussed above, the overpressure protection design for the AP600, at power operating conditions, complies with the guidelines of Section 5.2.2 of the SRP and the requirement of GDC 15, which specifies 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 conditions of normal operation, including anticipated operational occurrences. The design-basis analysis of a complete loss of steam flow to the turbine is performed with the LOFTRAN code. In the DSER, the staff identified Open Item 5.2.2.1-1, stating that the application of LOFTRAN to the AP600 design was still under staff review, and therefore, the acceptability of the overpressure protection design was still under review. As discussed in Section 21.6.1 of this report, the staff has reviewed and approved the application of LOFTRAN to the AP600 design. Therefore, DSER Open Item 5.2.2.1-1 is closed.

5.2.2.2 Overpressure Protection During Low-Temperature Operation

In Section 5.2.2 of the SRP, the staff specifies that the low-temperature overpressure protection (LTOP) system be designed in accordance with the guidance of BTP RSB 5-2. In BTP RSB 5-2, the staff specifies that the LTOP system be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to prevent exceeding the applicable technical specifications and Appendix G limits for the RCS while operating at low temperatures. In BTP RSB 5-2, the staff also specifies that the LTOP system meet the ASME Code Section III requirements, as well as RGs 1.26 and 1.29 regarding quality group and seismic design classifications. In addition, Section 5.2.2 of the SRP specifies that the LTOP system must be operable during startup and shutdown conditions below the enable temperature defined in BTP RSB 5-2. The enable temperature is defined as the water temperature corresponding to a metal temperature of at least the reference nil-ductility temperature plus 50 °C (90 °F) at the beltline location.

The LTOP system for the AP600 is provided by the relief valve in the suction line of the RNS, which discharges to the containment sump. Administrative controls and plant procedures aid in controlling RCS pressure during low-temperature operation. Normal plant operating procedures maximize the use of a steam or gas bubble in the pressurizer during periods of low-pressure, low-temperature operation. For those low-temperature modes when operation with a water-solid pressurizer is possible, the RNS relief valve provides LTOP for the RCS. As discussed in 5.4.7 of the SSAR, the RNS relief valve and associated piping are safety-related. Table 3.2-3 of the SSAR specifies that the RNS suction pressure relief valve (RNS-PL-V021) is an AP600 Class B component, seismic Category I and meeting the ASME Code, Section III, Class 2 requirements. Because the relief valve is connected to the piping between the containment isolation valves for the system, it also provides a containment boundary function and is subjected to the containment isolation requirements discussed in Section 6.2.3 of the SSAR. Also, the relief valve is subject to inservice test requirements as described in SSAR Table 3.9-16. In addition, TS LCO 3.4.15 specifies operability of the RNS suction relief valve for low temperature overpressure protection. The relief valve will automatically open for overpressure protection when the RCS pressure exceeds the RNS relief valve setpoint.

The sizing and setpressure of the RNS relief valve for LTOP are founded on sizing analysis performed to prevent the RCS pressure from exceeding the reactor vessel pressure/temperature limits described in Section 5.3.3 of SSAR for the following two types of events:

(1) the mass addition transient caused by a makeup/letdown mismatch
(2) the heat addition transient caused by an inadvertent start of one inactive RCP

These events result in bounding mass and energy input conditions relative to other credible events, such as inadvertent actuation of the pressurizer heaters, loss of residual heat removal with RCS heatup as a result of decay heat and pump heat, and inadvertent hydrogen addition. The design-basis analyses for the sizing of the RNS relief valve for LTOP protection (Westinghouse submittal, "AP600 Design Certification, Response to Open Item 2275," NSD-NRC-97-5229, July 10, 1997) assumes the transients occur while the pressurizer is in water-solid condition. The makeup/letdown mismatch case is postulated to occur over a range of reactor coolant temperatures between 37.8 and 176.7 °C (100 and 350 °F) with both CVS makeup pumps in operation at the maximum makeup water flow to the RCS and the letdown isolated. The case of inadvertent restart of one reactor coolant pump is postulated to occur over a range of reactor coolant temperatures between 37.8 and 93.3 °C (100 and 200 °F) and with the water in the SG secondary side 27.8 °C (50 °F) hotter than the primary side water. The assumption of a 27.8 °C (50 °F) temperature difference as the initial condition for the energy input transient conservatively bounds the cooldown operation controlled by the procedure. To prevent the possibility of a heat input transient, and thereby limit the required flow rate of the

RNS suction relief valve, an administrative requirement is imposed in TS LCO 3.4.15 for the LTOP protection system that does not allow an RC pump to be started with the pressurizer level above 92 percent and the RCS temperature above 93.3 °C (200 °F). The results of the analyses of the mass addition and heat addition transients showed that the mass addition transient is limiting. The minimum RNS relief valve capacity is calculated at an RCS pressure equivalent to the valve setpoint of 3.98 MPa (563 psig) plus 10 percent accumulation, i.e. 4.37 MPa (619 psig). With this setpoint, the relief valve would mitigate the limiting LTOP transient while maintaining the RCS pressure less than the Pressure/Temperature (P/T) limit of 4.38 MPa (621 psig). The minimum RNS relief valve capacity required is 126 m^3/h (555 gpm), which is the maximum makeup water flow at 4.37 MPa (619 psig) RCS pressure. The RNS relief valve design parameters with the nominal set pressure of 3.98 MPa (563 psig) and the relieving capacity of 126 m^3/h (555 gpm) are provided in Table 5.4.17 of the SSAR.

The RCS P/T limit of 4.38 MPa (621 psig), on which the RNS relief valve setpoint of 3.98 MPa (563 psig) was derived, was obtained from the bounding P/T heatup and cooldown curves specified in Figures 5.3-2 and 5.3-3 of the SSAR, which are generic limiting curves for AP600 reactor vessel design on the basis of the copper and nickel material composition of SSAR Table 5.3-1 and 54 effective full power years (EFPY). Therefore, the RNS relief valve setpoint must be reevaluated if the specific AP600 P/T curves are not bounded by the curves of SSAR Figures 5.3-2 and 5.3-3, either due to different reactor vessel material composition, or plant operation greater than 54 EFPY. Because the nil-ductility reference temperature of the reactor vessel material increases as exposure to neutron fluence increases as a result of neutron embrittlement effect, the operating P/T limit curves need to be periodically adjusted to accommodate the actual shift in the nil-ductility temperature. The RCS P/T limit curves are specified in the Pressure-Temperature Limits Report (PTLR) as required in the AP600 technical specification LCO 3.4.3. The basis for AP600 technical specification 3.4.15 notes that each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RNS suction relief valve, or the depressurized and vented RCS condition. In Section 5.3.6.1 of the SSAR, Westinghouse requires the COL applicant to address the use of plant-specific P/T limit curves relative to the reactor vessel material composition during procurement of the reactor vessel, as well as the evaluation of the LTOP system, including evaluating the setpoint pressure for the RNS relief valve as noted in the basis of TS 3.4.15. The staff concludes that the appropriate set pressure will be used for the RNS relief valve to ensure the P/T limits are not exceeded, and therefore, the AP600 LTOP system is acceptable.

5.2.3 Pressure Boundary Materials

GDC 1 of Appendix A of 10 CFR Part 50 and 10 CFR 50.55a(a)(1) of 10 CFR Part 50 require that structures, systems, and components 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 pressure boundary materials to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met as they relate to the selection of materials for the RCPB to determine their adequacy to assure a quality product commensurate with the importance of the safety function to be performed.

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. The staff reviewed the RCPB 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 environmental conditions.

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 RCPB materials to ensure that the relevant requirements of GDC 14 have been met as they relate to extremely low probability of rapidly propagating fracture and gross rupture of the RCPB.

GDC 30 requires that components that are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. The staff reviewed the pressure boundary materials to ensure that the relevant requirements of GDC 30 have been met as they relate to the quality standards for design, fabrication, erection, and testing.

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 pressure boundary materials to ensure that the relevant requirements of GDC 31 have been met as they relate to behavior in a non-brittle manner and an extremely low probability of rapidly propagating fracture.

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 RCPB materials to ensure that the requirements of Appendix B have been met as they relate to the establishment of measures to control the handling, storing, shipping, cleaning, and preservation of material and equipment in accordance with work and inspection instructions, to prevent damage or deterioration.

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviewed the RCPB materials as they related to the materials testing and acceptance criteria for fracture toughness contained in Appendix G.

The RCPB comprises parts of many components, including the reactor vessel, the SG, the pressurizer, the reactor coolant pump, the control rod drive mechanism, the core makeup tank, the passive residual heat removal heat exchanger, and various valves and piping. The integrity of the RCPB is addressed in SSAR Section 5.2. The staff reviewed the RCPB materials, as presented in Section 5.2.3, in accordance with Section 5.2.3 of the SRP. The staff reviewed the materials specifications, compatibility of materials with reactor coolant, fabrication and processing of ferritic materials, and the fabrication and processing of stainless steels. In the course of its review, the staff transmitted to Westinghouse RAIs concerning the RCPB materials, and received from Westinghouse responses to these RAIs. In addition, the staff and Westinghouse held several discussions to help clarify and resolve outstanding issues.

The major structural materials used in the fabrication of the component parts of the RCPB include carbon and low-alloy steels, austenitic stainless steels, and nickel-chromium-iron (Ni-Cr-Fe) alloys. The staff requested that Westinghouse identify in the SSAR all the materials used in the RCPB, including penetrations in the pressure vessel, and list them in Table 5.2-1. This was Open Item 5.2.3-2. In association with this, the staff requested clarification concerning all of the weld metals, including their specifications, class, type, grade, and any special requirements or relaxations used in the fabrication of the RCPB. This was Open Item 5.2.3-1. The SSAR was subsequently revised (Revision 5) and Table 5.2-1 of the SSAR was expanded to include the class, grade, or type for all the materials that will be used for the fabrication of the component parts of the RCPB. The listing includes the specifications for all the structural materials, the cladding and buttering materials, and the welding materials employed in the fabrication of the control rod drive mechanism. The specifications for all the other welding materials are given in the text of Section 5.2.3.1 of the SSAR. These include welds between ferritic materials, between austenitic stainless steel parts, between Ni-Cr-Fe alloys, and in dissimilar material combinations. The staff finds that the AP600 design has adequately identified materials for the RCPB and is acceptable. Therefore, Open Items 5.2.3-2 and 5.2.3-1 are closed.

The staff requested that Westinghouse revise Table 5.2-1 in the SSAR to reflect the choice of a given material for particular components if one material is not used for all similar components. The justification for the choice of one material over another should be defined. This was Open Item 5.2.3-4. The SSAR was subsequently revised (Revision 5) and the requested information was provided. Westinghouse proposes using Types 304LN and 316LN austenitic stainless steel in the RCPB, in preference to other types of austenitic stainless steels. These materials are not susceptible to intergranular stress corrosion cracking when the oxygen content of the reactor coolant exceeds 0.010 ppm at temperatures above 93 °C (200 °F) during normal operations. During startup and operation, these temperature and chemical conditions are avoided through water chemistry controls specified in the AP600 SSAR. Westinghouse has thus taken an alternative mitigating approach, as provided in RG 1.44, "Control of Sensitized Stainless Steel," dated May 1973. This approach will provide reasonable assurance that the integrity of austenitic stainless steel components in contact with reactor coolant will be maintained and is, therefore, acceptable. In addition, Alloy 690 will be used in the RCPB in preference to other nickel-based alloys. The reasons for this selection are discussed under Open Item 5.2.3-5. The staff finds these materials acceptable, therefore, Open Item 5.2.3-4 is closed.

The staff requested that Westinghouse identify in the SSAR the nickel-based alloy(s) and the weld metals to be used in the RCPB and identify them by specification, type, grade, and heat treatment. This was Open Item 5.2.3-3. The staff also requested that information be provided concerning the use of these materials, including justification for each application of nickel-based alloys and their weld metals (except for the reactor coolant pump flywheel enclosure). This was Open Item 5.2.3-5. Revision 3 of the SSAR identifies Alloy 690 as the material in question and provides an appropriate specification. Alloy 690 is the only nickel-based alloy to be used for structural components in the RCPB, a decision made on the basis of its superior performance in the primary water environments experienced in pressurized water reactors. Alloy 600 will be used only for cladding or buttering applications, in line with current practice. The staff approves of the choice of Alloy 690 as the preferred nickel-based alloy because of its resistance to stress corrosion cracking. Therefore, Open Items 5.2.3-3 and 5.2.3-5 are closed.

The SSAR and the response to a RAI 252.77 state that cobalt-based alloys will be used for various applications, but that also efforts are underway to develop and use other materials or low-cobalt content alloys in these applications. No definitive statements were made concerning the applications where such substitutions were to take place, or the extent to which cobalt had been eliminated from the design. The staff requested that Westinghouse indicate in the SSAR the base materials and/or surfacing (materials and processes) that are to be used in lieu of cobalt-based alloys, the test programs and the results of such programs, and to present data that assured a 60-year design life. This was Open Item 5.2.3-6. The SSAR was subsequently revised (Revision 5) and the requested information was provided. Revision 5 of the SSAR states that hardfacing material in contact with reactor coolant is primarily qualified low or zero cobalt alloys equivalent to Stellite-6. The use of cobalt base alloy is minimized. Low or zero cobalt alloys used for hardfacing or other applications where cobalt alloys have been previously used are qualified using wear and corrosion tests. The corrosion tests qualify the corrosion resistance of the alloy in reactor coolant. Cobalt-free wear-resistant alloys considered for this application include those developed and qualified in nuclear industry programs. The staff finds the AP600 commitment to minimize the use of cobalt-based alloys acceptable. Further, the minimal use of low cobalt alloys such as Stellite-6 for wear-resistant applications in the baseline RCPB design is acceptable on the basis of the adequate performance of such materials in similar applications in current nuclear power plants. Therefore, Open Item 5.2.3-6 is closed.

Some RCPB parts may be fabricated from cast austenitic stainless steel (specifically, SA-351, type CF3A). Thermal aging of cast austenitic stainless steel at reactor operating temperatures can lead to the transformation of delta ferrite into the brittle sigma phase and an associated reduction in the fracture toughness of the material. The staff asked Westinghouse to address the effect of thermal embrittlement of these castings over the 60-year plant design life. This was Open Item 5.2.3-7. Revision 3 of the SSAR indicates that the cast austenitic stainless steel used in the RCPB parts will have a ferrite content (defined by a ferrite number, FN) of not more than 30. The EPRI URD specifies (in Section 5.3.1.4) that the ferrite content should be controlled between FNs of 8 and 30, and should not exceed an FN of 30. The staff considers these requirements to be acceptable, therefore, Open Item 5.2.3-7 is closed.

Cold work of austenitic stainless steel has been identified as a potential cause of failures. The EPRI URD specifies (in Section 5.3.1.1), and the staff has accepted (in NUREG-1242), certain recommendations 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 austenitic stainless steels pertaining to cold work that had not been addressed in the SSAR, and to identify those positions that differed from the EPRI URD. This was Open Item 5.2.3-10. Revision 3 of the SSAR specifically discusses cold work in austenitic stainless steels. Cold-worked material will only be used for small parts, such as pins and fasteners, where such material has been used successfully in similar applications. Austenitic stainless steel used in pressure boundary applications will be in the solution annealed or thermally treated condition and will not have yield strengths greater than 620.5 MPa (90,000 psi). Control of cold work in austenitic stainless steel for pressure boundary applications will be provided by limiting the hardness of the raw material and controlling it during fabrication through process control of bending, straightening, and other similar operations. Grinding of material will be controlled by procedures, and cold work imparted during the grinding operations will be removed in the surface finishing operations. The procedures to be adopted in the fabrication of austenitic stainless steel components of the AP600 RCPB are in conformance with all the recommendations contained in the EPRI URD and no NRC staff positions exist that differ from those adopted in the EPRI URD. Open Item 5.2.3-10 is closed.

Welding of austenitic stainless steel components in the RCPB must be performed in accordance with the requirements of Section III of the ASME Code, one of which specifies that the ferrite content of the weld metal shall be a minimum FN of 5. The guidance provided in the EPRI URD stipulates that the average ferrite content should be in the FN range 5 to 13. The staff requested that Westinghouse address in the SSAR the welding of austenitic stainless steels of the RCPB with specific regard to the ferrite content of the weld metal, and discuss its position if different from the EPRI URD guidelines. This was Open Item 5.2.3-11. Revision 3 of the SSAR indicates that the weld filler metal to be used will be capable of providing weld deposits with a ferrite number in the range 5 to 13, in compliance with the EPRI guidelines. The staff has concluded (in NUREG-1242) that the guidance contained in the EPRI URD are acceptable and thus the position adopted in the AP600 regarding weld metal ferrite content design is acceptable. The guidance provided in the EPRI URD is compatible with the recommendation in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revision 3, dated April 1978, that the ferrite content in weld filler metal as depicted by a FN be between 5 and 20. Open Item 5.2.3-11 is closed.

Welding of ferritic steels will be performed in accordance with the requirements of Sections III and IX of the ASME Code. In addition, the welding procedures and practices will follow the recommendations contained in RGs 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel Components," and 1.71, "Welder Qualification for Areas of Limited Accessibility," dated December 1973. There is one exception relative to RG 1.71. Westinghouse has proposed that, for shop welds, the welder's position will be controlled and the joints of limited accessibility are repetitive because multiple production of similar components. Such welding is closely supervised. 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 nondestructive examination requirements of Section III of the ASME Code. These precautions should provide adequate assurance of the acceptability of joints welded under conditions of limited accessibility. On this basis, Westinghouse has provided an acceptable alternative to the recommendations in RG 1.71. This practice will ensure that proper requalification of welders will be required in accordance with the welding conditions.

The design of the AP600 RCPB does not include the use of electroslag welds. Thus, no requirement exists to impose any of the controls recommended by RG 1.34, "Control of Electroslag Weld Properties."

General corrosion of all materials is expected to be negligible. There will not be any unclad carbon and low-alloy steel in contact with reactor coolant during normal operations. However, the lack of commitment to consider the use of materials and processes for replacing cobalt-containing materials for the RCS was not acceptable to the staff. This was Open Item 5.2.3-12. As was indicated in the discussion associated with Open Item 5.2.3-6, Westinghouse revised its AP600 SSAR to state that the use of cobalt-based alloys will be minimized. Further, the elimination of cobalt-based alloys from the primary coolant system is related to as low as reasonably achievable (ALARA) concerns and, although highly desirable, has no direct effect on the safe reactor operation or shutdown. The staff concluded that

Westinghouse is committed to minimize the use of cobalt-based alloys in the design of the AP600 RCS. Therefore, Open Item 5.2.3-12 is closed.

The thermal insulation used on the AP600 RCPB is of the reflective stainless steel type, or fibrous insulation enclosed in stainless steel cans, or made of compounded materials that yield low leachable chloride and fluoride concentrations. The compounded materials, in the form of blocks, boards, cloths, tapes, adhesives, cements, and so forth, are silicated to protect the austenitic stainless steel components against stress corrosion cracking that may result from accidental wetting of the insulation by spillage, minor leakage, or other contamination from the surrounding environment. The SSAR commits the thermal insulation used on the components of the AP600 RCPB to meet the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels," dated February 1973. RG 1.36 provides acceptable guidance concerning the use of nonmetallic thermal insulation for austenitic stainless steels. Therefore, the staff finds this commitment acceptable.

Section 5.2.3 of the SRP specifies that all materials selected for use in the construction of the RCPB be reviewed for their compatibility with the reactor coolant. Inherent in this is that consideration should be taken of any nonmetallic materials that might be used, to ensure that their presence will not lead to a potential loss of integrity of the RCPB. The major concern is that the presence of certain nonmetallics can lead to enhanced potential for corrosion and stress corrosion cracking. The EPRI URD (in Section 5.2.8) specifies, and the staff has accepted (in NUREG-1242), 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 revise the SSAR to include discussion of its chemical content controls for nonmetallic materials to protect RCPB components and to identify those positions related to chemical content control that differ from the EPRI URD. This was Open Item 5.2.3-9. The concern expressed in the EPRI URD in this instance relates 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 fluids, lubricants, abrasive adhesives, and tape. The revised SSAR indicates that appropriate measures will be taken to avoid such contamination in the handling, storing, and cleaning of the austenitic stainless steels during the fabrication, installation and testing phases. In addition, the lubricants to be used on the threaded fasteners that maintain the integrity of the RCPB will be selected on the basis of satisfactory experience and test data that show them to be effective but not to cause or accelerate corrosion of the fastener. The lubricants will be specified in the design specifications and field selection of thread lubricants will not be permitted. Also, lubricants containing molybdenum disulfide will not be used in the AP600 plant. Similar restrictions will apply to the selection of leak sealants. The position adopted by Westinghouse related to control of nonmetallic materials is in conformance with the recommendations contained in the EPRI URD and is therefore acceptable. Open Item 5.2.3-9 is closed.

The integrity of the AP600 RCPB is further assured by the adoption of cleaning and cleanliness controls in accordance with American National Standards Institute (ANSI)/ASME NQA-2-1983, "Quality Assurance Requirements for Nuclear Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," dated March 1973. The staff has previously reviewed ANSI/ASME NQA-2-1983 and finds it acceptable. The cleaning and cleanliness controls specified will adequately control contamination of components during fabrication, shipment, and storage.

As discussed in Sections 5.2.4 and 6.6 of this report, 10 CFR 50.55a(g) requires that ASME Code Class 1, 2, and 3 components be designed to enable the performance of inservice examination in accordance with Section XI of the ASME Code. Further, 10 CFR 50.55a(g) requires the performance of a preservice inspection in accordance with Section XI of the ASME Code. This regulation has no relief provisions for preservice inspection. A material such as cast austenitic stainless steel that is difficult to inspect using ultrasonic techniques may not be able to conform to 10 CFR 50.55a(g). The staff's position is that inspectability is a basic consideration of design. Thus, relief from Code-required inservice inspection (ISI) requirements will not be granted on the basis that it is difficult to perform a meaningful examination on a cast stainless steel component because such inspection difficulty was well known at the time of design and thus could have been avoided. The staff requested that Westinghouse specify in the SSAR where the AP600 design will not meet the ASME Code, Section XI, 1989 Edition with Appendix VIII of the 1989 Addenda requirements. This was Open Item 5.2.3-8. Revision 3 of the SSAR addresses this concern in Section 5.2.4. Relief from the inspection requirements of Section XI of the ASME Code for Class 1 pressure-retaining components in the AP600 design should not be necessary. However, it is conceivable that future unanticipated changes in the ASME Code, Section XI requirements could necessitate relief requests. At such times, relief from the requirements will be requested when full compliance is not practical according to the requirements of 10 CFR 50.55a(g)(5)(iv). In these cases, specific information will be provided to identify the applicable Code requirements, the justification for the relief request, and the inspection method proposed as an alternative. Therefore, Open Item 5.2.3-8 is closed.

The staff concludes that the RCPB materials are acceptable and will meet GDCs 1, 4, 14, 30, and 31; Appendices B and G to 10 CFR Part 50; and 10 CFR 50.55a of 10 CFR Part 50. This conclusion is made on the basis of the following observations:

- The materials to be used for the construction of components of the RCPB have been identified by specification and found to be in conformance with Section III of the ASME Code. Such compliance satisfies the quality standards requirements of GDCs 1 and 30, and the Codes and Standards considerations of 10 CFR 50.55a.

- The RCPB materials identified are all compatible with the primary coolant water, which itself is chemically controlled in accordance with appropriate technical specifications. This compatibility has been proven by extensive testing and satisfactory inservice performance. This includes, for the austenitic stainless steels, conformance with the recommendations of RG 1.44. General corrosion of all materials are expected to be negligible and there will not be any unclad carbon and low-alloy steel in contact with reactor coolant during normal operations. The above evidence of compatibility with the reactor coolant satisfies the requirements of GDC 4, as they relate to the compatibility of components with environmental conditions.

- The materials to be used for the construction for the RCPB are compatible with the thermal insulation used in these areas and conform to the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Conformance with the recommendations of RG 1.36 satisfies the requirements of GDC 14 and GDC 31, as they relate to the prevention of failure of the RCPB.

- The AP600 RCPB materials will meet the fracture toughness tests required by the ASME Code and augmented by Appendix G to 10 CFR Part 50. This provides reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the RCPB. The use of Appendix G to Section III of the ASME Code and the results of fracture toughness tests performed in accordance with the ASME Code and NRC regulations in establishing safe operating procedures provide adequate safety margins during operations, testing, maintenance, and postulated accident conditions. This satisfies the requirements of GDC 31 and Appendix G of 10 CFR Part 50 regarding the prevention of the fracture of the RCPB.

- The AP600 controls imposed on welding preheat temperatures for welding ferritic steels are in conformance with the recommendations of RG 1.50. These controls offer reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment, and satisfy the quality standards requirements of GDC 1 and GDC 30, and 10 CFR 50.55a.

- The controls imposed on the welding of ferritic steels under conditions of limited accessibility are in accordance with the recommendations of RG 1.71, except for the welder performance qualifications. In this instance, Westinghouse proposed an acceptable alternative to the recommendations in RG 1.71, which should provide adequate assurance of the acceptability of joints welded under conditions of limited accessibility. These controls also satisfy the quality standards requirements of GDCs 1 and 30 and 10 CFR 50.55a.

- The controls imposed during the welding of austenitic stainless steel components in the RCPB are in accordance with the guidance provided in the EPRI URD and the recommendations of RG 1.31 concerning control of the ferrite content in stainless steel weld metal, and provide reasonable assurance that these welds will have high structural integrity. The controls thus meet the quality standards requirements of GDCs 1 and 30, and 10 CFR 50.55a, and satisfy the requirements of GDC 14 relative to the prevention of leakage and failure of the RCPB.

- The controls to avoid stress corrosion cracking in RCPB components constructed of austenitic stainless steels limit the yield strength of cold-worked austenitic stainless steels to 620.5 MPa (90,000 psi) maximum, and conform to the recommendations of RGs 1.37 and 1.44. Implementation of these controls provides reasonable assurance that these components will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. Thus, they meet the requirements of GDC 4 relative to the compatibility of components with environmental conditions, and those of GDC 14 relative to the prevention of leakage and failure of the RCPB.

- The specified controls on cleaning and cleanliness are in accordance with the recommendations of ANSI/ASME NQA-2-1983 and RG 1.37, and should assure adequate control of contamination of components during fabrication, shipment, and storage. Implementation of these controls will satisfy the requirements of Appendix B of 10 CFR Part 50 as they relate to the handling, storing, shipping, cleaning, and preservation of material and equipment.

5.2.4 RCS Pressure Boundary Inservice Inspection and Testing

10 CFR 50.55a(g)(2) requires, in part, that ASME Code Class 1 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 ASME Code applied to the construction of the particular component. The NRC staff reviewed the designs of the components comprising the AP600 RCPB as Class 1 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.

10 CFR Part 50, Appendix A, GDC 32 requires, in part, that components that are part of the RCPB shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The NRC staff reviewed the inservice inspection and testing program for components of the AP600 RCPB to ensure that the requirements of GDC 32 have been met as they relate to periodic testing and inspection.

The NRC staff reviewed the preservice and inservice inspection and testing program of the AP600 RCPB as presented in the SSAR in accordance with Section 5.2.4 of the SRP. The staff conducted the review to ascertain that inspection of selected welds and weld heat-affected zones before plant startup and periodically throughout the life of the plant will be such as to ensure that no deleterious defects develop during service. The areas evaluated included the inspection requirements for the RCPB, accessibility of the welds, examination categories and methods, inspection intervals, evaluation of examination results, system leakage and hydrostatic pressure tests, ASME Code exemptions, and relief requests, as appropriate. In the course of its review, the staff transmitted to Westinghouse RAIs concerning these procedures, and received from Westinghouse responses to these RAIs. In addition, the staff and Westinghouse held several discussions to help clarify and resolve outstanding issues.

The information contained in the SSAR describes the provisions for access for examination of the major components for RCPB, including the reactor vessel, closure head, reactor pressure vessel studs, nuts and washers, reactor vessel support skirt, piping, pumps, valves, and component supports. Westinghouse stated that all items within the Class 1 boundary are designed to provide access for the examinations required by Section XI, IWB-2500, of the ASME Code. In Table 5.2-3 of the SSAR, Westinghouse lists the applicable ASME Code cases for the major RCPB components.

The staff noted that the SSAR did not have a specific commitment that the AP600 will be designed such that the inservice inspection requirements of Section XI of the 1989 Edition of the ASME Code can be adequately performed. The staff requested that Westinghouse revise the SSAR to identify such a commitment. This was Open Item 5.2.4-2. The staff also noted that Section XI of the ASME Code contains two appendices related to ultrasonic examination, Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." The staff requested that the SSAR be revised to commit to the ultrasonic examination requirements of Appendix VIII of Section XI of the ASME Code. This was Open Item 5.2.4-1. The SSAR was subsequently revised (Revision 3) to indicate, in Section 5.2.4.2, that the ASME Class 1 components are designed so that access is provided in the installed condition for visual, surface, and volumetric examinations specified by Section XI of the baseline

ASME Code (1989 Edition, 1989 Addenda) and its mandatory appendices. The mandatory appendices associated with the baseline ASME Code include Appendix VII and Appendix VIII. Therefore, Open Items 5.2.4-2 and 5.2.4-1 are closed.

In a related matter, the staff observed that, although the COL applicant has the overall responsibility for the completion of the preservice inspection (PSI) and ISI of the RCPB throughout the service life of the plant, the plant should be designed to permit the accomplishment of these inspections and, in particular, reflect the requirements of Appendix VIII of the ASME Code, Section XI. The staff requested that Westinghouse modify the SSAR to indicate that the AP600 design will provide for PSI and ISI examinations to meet the requirements of Appendix VIII. The staff requested that Westinghouse provide some discussion of the means of accomplishing this commitment during the design phase. This was Open Item 5.2.4-3. As noted in the previous paragraph, the AP600 has been designed taking into account the requirements of Section XI of the ASME Code (1989 Edition including the 1989 Addenda), including the mandatory appendices. The means by which this commitment is accomplished is through a design-for-inspectability program. Therefore, Open Item 5.2.4-3 is closed.

The staff emphasized that during the design certification stage, relief from the requirements of the Section XI of the ASME Code will not be granted for reasons involving materials of construction, geometry, design, or access. Therefore, the AP600 design must be carried out to implement the criteria of the 1989 Edition, including the 1989 Addenda, of the ASME Code Section XI. Also, the PSI requirements are established and known at the time each component is ordered. Therefore, full adherence to the 1989 Edition, including the 1989 Addenda, of the ASME Code Section XI is required. The approach used for PSI requirements is also applicable to ISI requirements in the design phase. Each part, component, assembly, system and support should be designed so that it can be inspected. Accordingly, the staff requested that Westinghouse provide a detailed discussion of its methodology for achieving inspectability of parts, components, assemblies, systems, and supports, to provide assurance that relief from ISI requirements will not be sought for problems arising from the above-mentioned causes. This was Open Item 5.2.4-4. In a related matter, it was noted that 10 CFR 50.55a(g)(6)(ii)(A) requires augmented examination of reactor vessels. The staff's review of the vessel drawing in Figure 5.3-1 of the SSAR found that it could not assure that essentially 100 percent of the ASME Code-required reactor vessel examinations can be accomplished. The staff requested that Westinghouse modify the SSAR to provide a commitment that this can be accomplished and to demonstrate that there will be no need of relief from any ASME Code-required inspections at the construction stage. This was Open Item 5.2.4-6. In response, Westinghouse revised the SSAR by adding a description of the AP600 inspectability program. The goal of the program, as described in Revision 3 of the SSAR, is to provide for the inspectability access and conformance of component design with available inspection equipment and techniques. Factors taken into account in evaluating component designs include examination requirements and techniques, accessibility, component geometry, and materials selection. Other factors facilitating accessibility for inspection include removable insulation; shielding; hangers and pipe whip restraints; and the provision of working platforms, scaffolding and ladders. Modules fabricated offsite will be designed and engineered to provide access for ISI and maintenance activities. These factors will ensure that sufficient clearances for personnel and equipment, maximum examination surface distances, two-sided access, elimination of geometrical interferences and proper weld surface preparation are provided during PSIs and ISIs. Therefore, Open Items 5.2.4-4 and 5.2.4-6 are closed.

The staff recognizes that there are situations where ISI of a component may not be meaningful, and that there may be very high costs in terms of radiation exposure or risk of damage to components. An example of such a situation is the volumetric examination of welds and surfaces of complex geometry, heavy section, stainless steel castings after having been in operation (Section XI, Table IWB-2500-1, Items No. B12.10, B12.20, B12.40 and B12.50). The staff requested that Westinghouse define those situations where ISI of a component is not meaningful, or where there are excessive costs in terms of radiation exposure or risk of damage to components and, if necessary, propose alternative requirements to the ASME Code to address these situations on a case-by-case basis. This was Open Item 5.2.4-5. In response, Westinghouse stated that no such situations had been identified and thus no revision of the SSAR was necessary to address such problems. Therefore, Open Item 5.2.4-5 is closed.

Westinghouse stated that Section XI of the ASME Code has provisions to use certain shop and field examinations in lieu of the onsite preservice examination. The AP600 design is committed to meet all access requirements of the regulations. The COL applicant must submit the complete plant-specific PSI and ISI programs to the NRC, including references to the Edition and Addenda of the ASME Code, Section XI, that will be used in selecting components subject to examination; a description of the components exempt from examination by the applicable Code; and isometric drawings used for the examination. The COL applicant should verify that its PSI and ISI programs will incorporate the requirements of Appendices VII and VIII and Subsection IWH of Section XI of the ASME Code. This was COL Action Item 5.2.4-1. The staff requested that Westinghouse include COL Action Item 5.2.4-1, related to the plant-specific PSI and ISI programs, in the SSAR. This was Open Item 5.2.4-7. In response to this request, Westinghouse included a new section (Section 5.2.6.2) in the SSAR that addresses the provision of a plant-specific inspection program by the COL applicant was included in Revision 3 of the SSAR. Therefore, COL Action Item 5.2.4-1 and Open Item 5.2.4.7 are closed.

In the SSAR, Westinghouse states that the periodic inspections and leakage and hydrostatic testing of pressure-retaining components of the AP600 RCPB will be performed in accordance with the requirements of Section XI of the ASME Code. This will provide reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of the component is compromised.

Acceptable PSI and ISI programs must meet the inspection and testing requirements of GDC 32 and 10 CFR 50.55a. The staff concludes that, with the satisfactory resolution of the open items listed above, acceptable PSI and ISI programs can be accomplished by a COL applicant referencing the AP600 Standard Design. The COL applicant must meet the requirements of Section XI of the ASME Code, as reviewed by the staff and determined to be appropriate for this application. This was COL Action Item 5.2.4-2. The staff requested that Westinghouse include COL Action Item 5.2.4-2, related to the requirements of the ASME Code, in the SSAR. This was Open Item 5.2.4-8. In response to this request, Westinghouse included in the SSAR a new section (Section 5.2.6.1) that addresses the responsibility of the COL applicant in meeting the relevant requirements of the ASME Code. Therefore, COL Action Item 5.2.4-2 and Open Item 5.2.4.8 are closed.

The staff concludes that PSI and ISI programs incorporated in the design of the AP600 are acceptable and in accordance with the relevant requirements of 10 CFR 50.55a, as they relate to the preservice and inservice inspectability of RCPB components, and in accordance with the requirements of GDC 32, as they relate to periodic testing and inspection. This conclusion is made on the basis of the applicant's meeting the requirements of Section XI of the ASME Code. Selected welds and weld heat-affected zones will be inspected before plant startup and periodically throughout the life of the AP600 to ensure that no deleterious defects develop during service. Westinghouse has stated that the AP600 inspection program will comply with the rules of 10 CFR 50.55a of 10 CFR Part 50 and that the design of the RCS has provisions for access for ISI in accordance with Section XI of the 1989 Edition of the ASME Code, including the 1989 Addenda. The periodic inspections, leakage testing, and hydrostatic testing of pressure-retaining components of the AP600 RCPB, to be performed in accordance with the requirements of Section XI of the ASME Code, will provide reasonable assurance that structural degradation or loss of leaktight-integrity during service will be detected in time to permit corrective action before the safety function of the component is compromised. Compliance with the ISIs required by the ASME Code constitutes an acceptable basis for satisfying the requirements of GDC 32 as they relate to periodic inspection and testing.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The staff reviewed the AP600 design as it relates to its capability to detect and, to the extent practical, identify the source of RCPB leakage. The staff reviewed the RCPB leakage detection design in accordance with the guidelines provided in SRP Section 5.2.5. Staff acceptance of the leakage detection design is on the basis of the design meeting the requirements of GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake, and on the design meeting the requirements of GDC 30, "Quality of Reactor Coolant Pressure Boundary," as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage. Conformance with GDC 2 is on the basis of the leakage detection design meeting the guidelines of RG 1.29, "Seismic Design Classification," Positions C.1 and C.2. Conformance with GDC 30 is on the basis of the leakage detection design meeting the guidelines of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Positions C.1 through C.9. Leakage detection monitoring is also maintained in support of LBB criteria for high-energy fluid piping in containment. SSAR Section 3.6.3 addresses the application of LBB criteria.

The staff also reviewed the RCPB leakage detection design for compliance with the requirements of the TMI issue designated by 10 CFR 50.34(f)(2)(xxvi). With respect to this issue, the NRC states that applicants should provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident.

RCPB leakage detection is accomplished using instrumentation and other components of several systems. Diverse measurement methods including level, flow, and radioactivity measurements are used for leakage detection. The equipment classification for each of the systems and components used for leakage detection is generally determined by the requirements and functions of the system in which it is located. There is no requirement that leakage detection and monitoring equipment be safety-related.

RCPB leakage is classified as either identified or unidentified leakage. Identified leakage includes (1) leakage from closed systems such as reactor vessel seal or valve leakage that is captured and conducted to a collecting tank, and (2) intersystem leakage into auxiliary systems and secondary systems. (Intersystem leakage must be considered in the evaluation of the reactor coolant inventory balance.) Other leakage is unidentified leakage.

5.2.5.1 Identified Leakage Detection

Sources of identified leakage in containment include leaks from the reactor vessel head flange, pressurizer safety relief valves, and automatic depressurization valves. In the course of plant operations, various minor leaks of the RCPB may be detected by operating personnel. If these leaks can be subsequently observed, quantified, and routed to the containment sump, this leakage will be considered identified leakage.

Identified leakage other than intersystem leakage is collected in a closed reactor coolant drain tank (RCDT) located in the reactor cavity in containment. The RCDT vent is piped to the gaseous radwaste system to prevent release of radioactive gas to the containment atmosphere. Leakage detection alarms and indications are provided in the main control room (MCR). The RCDT, pumps, and sensors are part of the liquid radwaste system.

5.2.5.2 Intersystem Leakage Detection

In Section 5.2.5.2 of the SSAR, Westinghouse states that possible intersystem leakage points across passive barriers or valves and their detection methods were considered. Auxiliary systems connected to the RCPB incorporate design and administrative provisions that limit leakage. Such leakage is detected by increasing auxiliary system level, temperature, flow, or pressure; by lifting relief valves; or increasing values of monitored radiation in the auxiliary system. The RNS and the CVS have the potential for intersystem leakage past closed valves.

An important potentially identifiable leakage path for reactor coolant is through the SG tubes into the secondary side of the SG. Identified leakage from the SG primary side is detected by one or a combination of the following methods:

- the condenser air removal radiation monitor
- the SG blowdown radiation monitor
- the main steamline radiation monitor
- the laboratory analysis of condensate

In addition, leakage from the RCS to the CCS is detected by the CCS radiation monitor, by increasing surge tank level, by high flow downstream of selected components, or by some combination of the preceding.

5.2.5.3 Unidentified Leakage Detection

In Section 5.2.5.3 of the SSAR, Westinghouse states that to detect unidentified leakage in containment, three diverse methods may be utilized to quantify and assist in locating the leakage, including the following:

(1) containment sump level
(2) RCS inventory balance
(3) containment atmosphere radiation

In addition, other supplemental methods utilize containment atmosphere pressure, temperature, humidity, and visual inspection.

In Position C.1 of RG 1.29, the NRC states that the SSCs listed in the RG, including their foundations and supports, should be designated as seismic Category I to ensure that they can withstand the effects of a safe-shutdown earthquake (SSE) and remain functional. In Section 5.2.5.4 of the SSAR, Westinghouse states that the containment sump level monitor and the containment atmosphere radiation monitor are classified as seismic Category I.

In Position C.2 of RG 1.29, the NRC states that those parts of SSCs, whose continued function is not required but whose failure could reduce the functioning of any plant feature (identified in Position C.1) to an unacceptable safety level, or could result in an incapacitating injury to occupants of the MCR, should be designed and constructed so that an SSE would not cause such a failure. In Section 5.2.5 of the SSAR, Westinghouse states that equipment classification for each of the systems and components used for leakage detection is generally determined by the requirements and functions of the system in which it is located. There is no requirement that leakage detection and monitoring equipment be safety-related.

In DSER Section 5.2.5.3, the staff stated that Westinghouse should provide complete information regarding the seismic and safety classifications of the leakage detection system (LDS). This was DSER Open Item 5.2.5.3-1. Westinghouse provided the applicable information as discussed above. Therefore, DSER Open Item 5.2.5.3-1 is closed.

On the basis of the above, the staff concludes that the design of systems and components used for leakage detection meets the guidelines of RG 1.29, Positions C.1 and C.2. Therefore, the design meets the requirements of GDC 2, as it relates to the capability of the systems and components to maintain and perform their safety function following an earthquake.

In Position C.1 of RG 1.45, the NRC states that leakage to containment from identified sources should be collected or isolated so that flow rates are monitored separately from unidentified leakage and so that the total flow rate can be established and monitored. As stated above, identified leakage is monitored separately for the reactor vessel head flange, pressurizer safety relief valves, and automatic depressurization valves.

In Position C.2 of RG 1.45, the NRC states that leakage to containment from unidentified sources should be collected and the flow rate monitored with an accuracy of 1 gpm (3.78 L/min) or better. In Section 5.2.5.3 of the SSAR, Westinghouse states that the sensitivity of leakage detection monitoring is such that the monitoring can detect a change of 0.5 gpm (1.89 L/min) in one hour.

In Position C.3 of RG 1.45, the NRC states that at least three separate methods should be used for leakage detection. Two of these methods should include (1) sump level and flow monitoring and, (2) airborne particulate radioactivity monitoring. The third method may be selected from monitoring either (1) condensate flow from the containment air coolers or, (2) containment airborne gaseous activity. In Section 5.2.5.3 of the SSAR, Westinghouse states that containment sump level monitoring, containment atmosphere radiation monitoring, and RCS inventory balance are utilized in the AP600 design to detect and monitor leakage in containment. In particular, Westinghouse selected the gaseous N13/F18 monitor for containment atmosphere radiation monitoring. In Section 5.2.5.3 of the SSAR, Westinghouse also states that humidity, temperature, and pressure monitoring are also used for alarms and indirect indication of possible leakage in containment.

In Position C.4 of RG 1.45, the NRC states that provisions should be made to monitor the systems connected to the RCPB for indications of intersystem leakage. Methods should include radioactivity monitoring and indicators to show abnormal water levels or flow in the affected systems. In Section 5.2.5.2 of the SSAR, Westinghouse states that associated systems and components connected to the RCS have intersystem leakage monitoring devices. SG tube leakage is detected by the condenser air removal radiation monitor, the SG blowdown radiation monitor, the main steamline radiation monitor, or laboratory analysis of condensate. Leakage from the RCS to the CCS is detected by CCS radiation monitors, by increasing surge tank level, by high flow downstream of selected components, or by some combination of the preceding.

In Position C.5 of RG 1.45, the NRC states that the sensitivity and response time of each method used to detect and monitor unidentified leakage in containment should be a minimum of 1 gpm (3.78 L/min) in less than one hour. However, in DSER Section 5.2.5.3, the staff considered the SSAR information incomplete and requested Westinghouse to provide additional information regarding the sensitivity and response times for all methods of leakage detection and monitoring used in the AP600 design. This was DSER Open Item 5.2.5.3-2. Westinghouse provided additional information regarding the sensitivity and response times for methods of leakage detection and monitoring used in the AP600 design. In Section 5.2.5.3 of the SSAR, Westinghouse states that the equipment used to detect and monitor unidentified leakage has a sensitivity and response time of 0.5 gpm (1.89 L/min) in less than one hour. Therefore, DSER Open Item 5.2.5.3-2 is closed.

In Position C.6 of RG 1.45, the NRC states that the LDSs should be capable of performing their functions during and following an SSE. In Section 5.2.5.4 of the SSAR, Westinghouse states that the containment sump level monitor and the containment atmosphere radiation monitor are classified as seismic Category I. Containment activity is monitored by the containment high-range radiation monitor, which is seismically qualified.

In Position C.7 of RG 1.45, the NRC states that indicators and alarms for each LDS should be provided in the MCR. In addition, procedures for converting indications to a common leakage equivalent should be available to the operators. In Section 5.2.5.6 of the SSAR, Westinghouse lists the alarms and/or indications for RCPB leakage provided in the MCR. The plant instrumentation system is a microprocessor-based system that accepts inputs from all RCPB leakage detection sensors and monitors. The containment sump level, containment atmosphere radioactivity, RCS inventory balance, and the flow measurements are provided as gallon per minute leakage equivalent.

In Position C.8 of RG 1.45, the NRC states that the LDSs should be equipped with provisions for operability testing and calibration during plant operation. In Section 5.2.5.5 of the SSAR, Westinghouse states that periodic testing of the leakage detection monitors verifies the operability and sensitivity of detection equipment. These tests include calibrations and alignments during installation, periodic channel calibrations, functional tests, and channel checks. The instrumentation for RCPB leakage detection can be tested for operability during plant operation.

In Position C.9 of RG 1.45, the NRC states that the TS should include limits for both identified and unidentified leakage, and should address the availability of various instruments to assure coverage at all times. In Chapter 16 of the SSAR TS, Westinghouse defines the operability requirements for the RCS leakage detection instrumentation. The instrumentation is designed to verify its operability at all times. Should a detector fail (e.g., signal outside the calibrated range or self-monitored trouble is detected), the plant instrumentation system will initiate a trouble alarm in the MCR, indicating that the readout of a specific monitor is questionable. On the basis of this information, the staff concludes that the AP600 design provides adequate assurance that the instruments used to detect and monitor RCPB leakage are available at all times.

The staff compared AP600 TS 3.4.8, "RCS Operational Leakage," and 3.4.10, "RCS Leakage Detection System," with the Westinghouse Operating Group Standard TS (WOG STS) 3.4.13 and 3.4.15, respectively. WOG STS were used in development of the AP600 TS.

During the development of the AP600 TS, Westinghouse retained the TS addressing pressure isolation valve leakage testing, which is also included in the WOG STS. AP600 TS, "RCS Pressure Isolation Valve (PIV) Integrity," requires the leakage testing of any PIV in accordance with American Society for Testing and Materials (ASTM) Section XI (OM-10).

On the basis of the information provided by Westinghouse and evaluated above, the staff concludes that the RCPB leakage detection design conforms to the guidelines of RG 1.45, Positions C.1 through C.9. Therefore, the design meets the requirements of GDC 30 as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage.

The TMI issue designated by 10 CFR 50.34(f)(2)(xxvi) (Item III.D.1.1 of NUREG-0737) states that applicants should provide for leakage control and detection in the design of systems outside of containment that contain (or might contain) total integrated dose (TID)-14844 source term radioactive materials following an accident. Applicants will submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and the public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. Westinghouse has addressed this TMI issue in Section 1.9.3 of the SSAR. In the SSAR, Westinghouse states that the safety-related passive systems do not recirculate radioactive fluids outside containment following an accident. A non-safety-related system can be used to recirculate coolant outside of containment following an accident, but this systems is not operated when high containment radiation levels exist. This satisfies the requirements of 10 CFR 50.34(f)(2)(xxvi).

During the staff's review of RCPB leakage detection for the AP600 design, several additional issues were identified and documented in the DSER, as follows:

- Westinghouse should incorporate RCPB leakage detection RAI responses into the SSAR (Open Item 5.2.5.3-3). As agreed upon with the staff, Westinghouse incorporated the RAI responses into Section 5.2.5 of the SSAR.

- Westinghouse should correct discrepancies noted by the staff in the SSAR (Open Item 5.2.5.3-4). Westinghouse corrected discrepancies identified by the staff in SSAR Section 5.2.5 and its related technical specifications, TS 3.4.8 and TS 3.4.10.

- Westinghouse should discuss its commitment to identify leakage requirements (Open Item 5.2.5.3-5). With regard to identified leakage detection and monitoring, Westinghouse addressed how the requirements of GDC 2 and GDC 30 were met by complying with the guidelines of RGs 1.29 and 1.45, respectively, in Section 5.2.5 of the SSAR.

- Westinghouse should commit to compliance with the guidance of RG 1.45 (Open Item 5.2.5.3-6). Westinghouse addressed its compliance with Positions C.1 through C.9 of RG 1.45 in Section 5.2.5 of the SSAR.

- Westinghouse should supply additional information related to RCPB LDS details (Open Item 5.2.5.3-7). As agreed upon with the staff, Westinghouse provided additional information related to RCPB leakage detection and monitoring in SSAR Section 5.2.5 and its related technical specifications, TSs 3.4.8 and 3.4.10.

Based on the above discussion, DSER Open Items 5.2.5.3-3 through 5.2.5.3-7 are closed.

Systems and components utilized for RCPB leakage detection provide reasonable assurance that structural degradation, which may develop in pressure-retaining equipment of the RCPB and result in coolant leakage during service, will be detected on a timely basis. Thus, corrective actions may be taken before such degradation can become sufficiently severe to jeopardize the safety of the equipment, or before the leakage can increase to a level exceeding the capability of the makeup system to replenish the coolant loss.

On the basis of its review of information provided in the SSAR and the responses to RAIs, the staff concludes that the design of the systems and components for RCPB leakage detection is acceptable. The design meets the requirements of GDC 2 with respect to the capability of systems and components to maintain and perform their safety functions in the event of an earthquake, and meets the requirements of GDC 30 with respect to the detection, identification, and monitoring of the source of reactor coolant leakage. This conclusion is made on the basis of the following:

- Westinghouse has met the requirements of GDC 2 with respect to the capability of systems and components to perform and maintain their safety functions in the event of an earthquake by meeting the guidelines of RG 1.29, Positions C.1 and C.2.

- Westinghouse has met the requirements of GDC 30 with respect to the detection, identification, and monitoring of the source of reactor coolant leakage by meeting the guidelines of RG 1.45, Positions C.1 through C.9.

- Westinghouse has met the requirements of 10 CFR 50.34(f)(2)(xxvi) with respect to minimizing leakage from systems outside containment that contain (or might contain) radioactive materials following an accident.

Therefore, the staff concludes that RCPB leakage detection for the AP600 design conforms to the guidelines of SRP Section 5.2.5, and is acceptable.

5.3 Reactor Vessel

The AP600 reactor vessel is described in Section 5.1.3.1 of the SSAR. The reactor vessel is cylindrical, with a hemispherical bottom head and a removable, flanged, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The vessel interfaces with the reactor internals, the integrated head package, and reactor coolant loop piping, and is supported on the containment building concrete structure.

5.3.1 Reactor Vessel Design

The design of the AP600 reactor vessel closely matches the existing vessel designs of Westinghouse three-loop plants. New features for the AP600 have been incorporated without departing from the proven features of existing vessel designs. The reactor vessel has inlet and outlet nozzles positioned in two horizontal planes between the upper head flange and the top of the core. The nozzles are located in this configuration to provide an acceptable crossflow velocity in the vessel outlet region and to facilitate optimum layout of the RCS equipment. The inlet and outlet nozzles are offset, with the inlet positioned above the outlet, to allow midloop operation for removal of a main coolant pump without discharge of the core.

Reactor Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

5.3.2 Reactor Vessel Materials

The following requirements apply to the reactor vessel materials:

GDC 1 and 10 CFR 50.55a(a)(1) require 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 reactor vessel (RV) materials to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met as they relate to the selection of materials for the reactor vessel to determine their adequacy to assure a quality product commensurate with the importance of the safety function to be performed.

GDC 4 requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The staff reviewed the RV materials to ensure that the relevant requirements of GDC 4 have been met as they relate to the compatibility of SSCs with environmental conditions.

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 RV materials to ensure that the relevant requirements of GDC 14 have been met as they relate to the prevention of rapidly propagating failure of the RCPB.

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 RV materials to ensure that the relevant requirements of GDC 31 have been met as they relate to the materials fracture toughness.

GDC 32 requires, in part, that the RCPB components shall be designed to permit an appropriate material surveillance program for the reactor pressure vessel. The staff reviewed the RV materials to ensure that the relevant requirements of GDC 32 have been met as they relate to the provision of a materials surveillance program.

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 RV materials to ensure that the requirements of Appendix B have been met as they relate to the establishment of controls for the onsite cleaning of materials and components.

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviewed the RV materials as they relate to the materials testing and acceptance criteria for fracture toughness contained in Appendix G.

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in fracture toughness properties of materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with ASTM E 185, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels." Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the reactor vessel. The staff reviewed the RV materials to determine that they meet the relevant requirements of Appendix H as they relate to the determination and monitoring of fracture toughness.

The staff reviewed the materials of construction of the reactor vessel in accordance with Section 5.3.1 of the SRP. In the course of its review, the staff transmitted to Westinghouse RAIs concerning the RCPB materials and received from Westinghouse responses to these RAIs. In addition, the staff and Westinghouse held several discussions to help clarify and resolve outstanding issues.

The AP600 reactor vessel is described in Section 5.1.3.1 of the SSAR. The vessel is cylindrical, with a hemispherical bottom head and a removable, flanged, hemispherical upper head. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The vessel interfaces with the reactor internals, the integrated head package and the reactor coolant loop piping, and is supported on the containment building concrete structure. The cylindrical section consists of an upper and a lower shell fabricated from low-alloy steel forgings and clad with austenitic stainless steel. The shells are joined by a circumferential weld; the structure contains no longitudinal welds. The hemispherical bottom is made of similar materials and welded to the lower shell. The removable flanged hemispherical upper head consists of the closure head flange (a low-alloy steel forging) and the closure head dome (a dished low-alloy steel plate). Both the flange and the dome are clad with austenitic stainless steel.

The materials used in the reactor vessel and its appurtenances are listed in Table 5.2-1 of the SSAR. Resolution of the open items associated with materials issues are addressed in Section 5.2.3 of this report. The major structural components, such as the shells, the closure flange and the head plates, will be made of low-alloy ferritic steels; austenitic stainless steels and Alloy 690 are used for various appurtenances. All material specifications are in accordance with ASME Code, Section III requirements.

The maximum limits for the elements in the materials of the reactor vessel beltline are listed in Table 5.3-1 in the SSAR. The sulfur and phosphorous content of welds and forgings are limited to a maximum of 0.01 percent. Nickel is limited to 0.85 percent, copper to 0.03 percent and vanadium to 0.05 percent. Data compiled in EPRI report NP-933, "Nuclear Pressure Vessel Steel Data Base," indicate that this control on the level of material elements will provide the fracture toughness required to ensure the structural integrity of the reactor vessel as specified by Appendix G of 10 CFR Part 50. The staff accepts this position.

The staff requested that Westinghouse further define the controls to be used during all stages of welding to prevent contamination as described in Section 5.2.3 of this report. This was Open Item 5.3.2-2. Westinghouse subsequently revised Section 5.3.2 of the SSAR (Revision 3) and made a cross-reference to Section 5.2.3 regarding the special controls to be used during all stages of welding. The controls identified provide reasonable assurance that contamination will not occur during welding and thus are acceptable to the staff. Therefore, Open Item 5.3.2-2 is closed.

The staff requested that the SSAR be amended to address how the AP600 design satisfies the recommendations of Revision 0 of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." These recommendations include specification of the bolting material, nondestructive examination of the bolts, and ISI. This was Open Item 5.3.2-4. Revision 3 of the SSAR indicates that the closure studs will be fabricated from ASME SA-5, Grade B23 or B24 (specified in Table 5.2-1), and that the material will meet the fracture toughness requirements of Section III of the ASME Code and Appendix G of 10 CFR Part 50. Nondestructive examination of the studs will be performed according to Section III of the ASME Code, Subarticle NB-2580. ISI will be performed according to Section XI of the Code, supplemented by paragraphs NB-2545 or NB-2546 of Section III of the Code. A summation of the AP600 position with respect to the criteria of RG 1.65 is provided in Section 1.9 of the SSAR. The staff finds that Westinghouse has adequately revised the SSAR to address how the AP600 design satisfies the recommendations of RG 1.65 and, therefore, Open Item 5.3.2-4 is closed.

Westinghouse's response to a RAI 252.66 indicated that lubricants containing molybdenum disulfide will be prohibited from use in all areas of the AP600 plant. These restrictions will be reflected in the plant and equipment specifications. The staff requested that Westinghouse modify the SSAR to indicate this commitment. This was Open Item 5.3.2-3. Revision 3 of the SSAR subsequently contained that restriction in Section 5.2.3 and reference to this section is contained in Section 5.3.2.7, "Reactor Vessel Fasteners." On the basis of the inclusion of this restriction in the SSAR, Open Item 5.3.2-3 is closed.

The design of a reactor vessel must take into account the potential embrittlement of RV materials as a consequence of neutron irradiation and the thermal environment. GDC 32 requires, in part, that the RCPB components shall be designed to permit an appropriate material surveillance program for the reactor pressure vessel. The requirements for such a program are defined in Appendix H of 10 CFR Part 50.

To meet the requirements of GDC 32, the AP600 design incorporates a material surveillance program to monitor the changes in fracture toughness of the RV beltline materials as a consequence of exposure to neutron irradiation and thermal environment. Appendix H to 10 CFR Part 50 requires that the surveillance program for the AP600 meet the recommendations of ASTM E 185. ASTM E 185 was prepared to be applicable to plants designed for a 40-year life, whereas the design life of the AP600 is 60 years. The recommended minimum number of surveillance capsules in ASTM E 185 for a reactor vessel with an end-of-life (EOL) shift between 38 °C (100 °F) and 93 °C (200 °F) is four. The AP600 surveillance program includes eight capsules, with archive materials available for at least two additional complete replacement capsules. These capsules can be installed in the reactor at any time, should circumstances indicate that an additional capsule is required, and when there is an available holder location. This exceeds the recommendations in ASTM E 185 and is consistent with the increase in design life associated with the AP600. The use of eight surveillance capsules is therefore acceptable to the staff.

Following response to a RAI 252.93, the staff requested that Westinghouse include information on the surveillance program schedule in the SSAR. This was Open Item 5.3.2-1. Revision 3 of the SSAR provides the requested description of the recommended capsule withdrawal schedule. The withdrawal schedule indicates that, of the eight capsules available, four will be withdrawn before the end of the 60-year plant life; this is in excess of the three specified in ASTM E 185. A fifth capsule will be withdrawn at the end of plant life, with the remaining three available on standby. The withdrawal schedule thus exceeds the requirements of Appendix H and ASTM E 185, consistent with the increase in design life, and is acceptable to the staff. Therefore, Open Item 5.3.2-1 is closed.

A surveillance program plan for RV materials should be founded on a reasonably conservative estimate of the reference temperature shift calculated for these materials. Westinghouse predicted shifts in the reference temperature for the AP600 RV materials using the methodology of Revision 2 of RG 1.99, "Radiation Damage to Reactor Vessel Materials." This methodology provides reasonably accurate and conservative predictions of adjusted reference temperatures for RV beltline materials, including low copper base and weld metals with phosphorus impurities controlled to low levels. Westinghouse calculated that the predicted adjusted reference temperature at EOL for the beltline weld and forgings at the inside of the vessel to be 7.8 °C (46 °F) and 6.7 °C (44 °F), respectively, and at the 1/4T location, 3.3 °C (38 °F) and 4.4 °C (40 °F), respectively. The margins used for the highest adjusted reference temperatures for the welds and forgings were 23.3 °C (42 °F) and 22.8 °C (41 °F), respectively. As discussed in Section 5.3.3 of this report, a more appropriate value of margin for forgings is 28.9 °C (52 °F). Although understating the adjusted reference temperature at EOL, the maximum understated difference was only 6.1 °C (11 °F), the difference between 11.1 °C (52 °F) and 5 °C (41 °F), which is insignificant. Actual data from a given vessel will be used to establish heat-up and cool-down curves for that vessel. The EOL upper shelf energy is calculated to exceed 68 J (50 ft-lb). These requirements will ensure that the surveillance program will generate sufficient information to determine the conditions under which the reactor vessel will be operated throughout its 60-year service lifetime. Westinghouse's calculated adjusted reference temperature meets the requirements of Appendix G to 10 CFR Part 50, and is therefore acceptable.

The materials surveillance program will be used to monitor changes in the fracture toughness properties of ferritic materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment as required by GDC 32. The AP600 surveillance program must comply with Appendix H to 10 CFR Part 50 and ASTM E 185. This standard requires the testing of welds, heat-affected zones, and base metals. The SSAR requires the testing of base and weld metals, but does not require that fracture toughness data be obtained from material specimens that are representative of the heat-affected-zone materials in the beltline region. The staff requested that Westinghouse provide adequate technical justification for not including heat-affected-zone materials in the AP600 materials surveillance program or to revise the SSAR to reflect the inclusion of such material in the program. This was Open Item 5.3.2-5. Westinghouse complied with this request and Revision 3 of the SSAR indicates that the surveillance program now includes material from the heat-affected zone. Therefore, Open Item 5.3.2-5 is closed.

There was no indication in the SSAR that the locations of the surveillance capsules had been established and, accordingly, the lead factor (the ratio of the neutron flux at the capsule to the flux at the location of peak exposure on the pressure vessel inner diameter) had not been determined. The staff requested that this information be provided in the SSAR. This was Open Item 5.3.2-6. Subsequently, Westinghouse revised Section 5.3.2.6.2 in the SSAR to indicate that the lead factor will be approximately 2.5. The staff considers that a factor of this magnitude is sufficient to permit mitigating or other actions to be taken if the results of fracture toughness testing of the surveillance specimens indicate that they are warranted. Therefore, Open Item 5.3.2-6 is closed.

A national program exists in which material from a standard reference heat of material (ASME SA-533, Type B, Class 1) is used as a monitor for Charpy impact tests, to facilitate comparisons between irradiation effects from different power and test reactors. The staff requested that Westinghouse describe any plans to include specimens from this reference heat of material in the surveillance capsules. This was Open Item 5.3.2-7. Westinghouse observed that the inclusion of such material in a surveillance program is a voluntary activity; no requirements exist to indicate that such material should be included in a surveillance program.. The baseline design of the AP600 has no provisions to incorporate the reference material in its surveillance program. The staff accepts that the national program is of a purely voluntary nature and notes that any action on this matter should be reserved to the COL applicant. However, because there are no direct safety implications associated with the inclusion of material from the reference heat in the AP600 surveillance program, no COL Action Item is required. Therefore, Open Item 5.3.2-7 is closed.

A COL applicant referencing the AP600 standard design should provide details of its reactor vessel materials surveillance program, including the specific materials in each surveillance capsule, the capsule lead factors, the withdrawal schedule for each capsule, the neutron fluence to be received by each capsule at the time of its withdrawal, and the vessel EOL peak neutron fluence. This was COL Action Item 5.3.2-1. The staff requested that Westinghouse include COL Action Item 5.3.2-1, related to the reactor vessel materials surveillance program, in the

SSAR. This was Open Item 5.3.2-8. The SSAR was subsequently revised to include a statement that the COL applicant will address a reactor vessel material surveillance program founded on Section 5.3.2.6 of the SSAR. The inclusion of this commitment in the SSAR satisfies COL Action Item 5.3.2-1, therefore, Open Item 5.3.2-8 is closed.

The information contained in the SSAR, combined with the satisfactory resolution of the open items listed above, enables the staff to conclude that the AP600 reactor vessel materials are acceptable and meet the applicable requirements of GDCs 1, 4, 14, 30, 31 and 32 of Appendix A of 10 CFR 50; the material testing and monitoring requirements of Appendices B, G and H of 10 CFR 50; and the requirements of 10 CFR 50.55a of 10 CFR Part 50. The staff's conclusion is on the basis of the following:

- The controls imposed upon austenitic stainless steel are either in accordance with the recommendations of RG 1.44, "Control of Sensitized Stainless Steel," Revision 0, dated May 1973, or, if they are not in accordance with this RG, the positions and actions taken have been accepted by the staff. These controls will provide reasonable assurance that welded components will not be excessively sensitized before or during the welding process, thus satisfying the quality standards requirements of GDCs 1 and 30, and 10 CFR 50.55a. These controls also satisfy the requirement of GDC 4 relative to material compatibility.

- Ordinary processes will be used for the manufacture, fabrication, welding, and nondestructive examination (NDE) of the reactor vessel and its appurtenances. Fabrication processes and NDEs will be performed in accordance with the requirements specified in the ASME Code, Section III. Compliance with these ASME Code provisions meets the quality standards requirements of GDCs 1 and 30, and 10 CFR 50.55a, and the requirements of Appendix B of 10 CFR Part 50, as they relate to onsite material cleaning control.

- Welding of ferritic steel components will be performed in accordance with the requirements of the ASME Code, Section III. The Code controls will be supplemented by conformance with the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," Revision 0. The AP600 controls imposed on welding preheat temperatures for welding ferritic steels are in conformance with the recommendations of RG 1.50 or provide acceptable alternative approaches to the RG's guidelines. These controls offer reasonable assurance that components made from low-alloy steels will not crack during fabrication and minimize the possibility of subsequent cracking due to residual stresses in the weldment. These controls also satisfy the quality standards requirements of GDCs 1 and 30, and 10 CFR 50.55a.

- Adherence to the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," is not necessary because the reactor vessel specifications require that all low-alloy steels are produced to the fine grain practice. This will provide reasonable assurance that underclad cracking will not occur during the weld cladding process. These controls satisfy the requirements of GDCs 1 and 30, and 10 CFR 50.55a.

- The fracture toughness of the materials of the AP600 reactor vessel and its appurtenances is controlled by conformance with Appendix G of 10 CFR Part 50, of which the ASME Code forms the basis. The fracture toughness tests required by the ASME Code and Appendix G to 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB. This methodology will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with the provisions of Appendix G of 10 CFR Part 50 satisfies the requirements of GDCs 14 and 31 and 10 CFR 50.55a regarding prevention of fracture of the RCPB.

- The design of the AP600 includes provisions to monitor changes in the fracture toughness, caused by exposure to neutron radiation, of RV beltline materials via use of a materials surveillance program. The program is in compliance with the requirements and intent of Appendix H of 10 CFR Part 50 and ASTM E 185. The latter recommends four materials surveillance capsules be installed in the reactor vessel beltline for a vessel with EOL shift between 38 °C (100.4 °F) and 93 °C (199.4 °F), the basis of a design life of 40 years. However, because the AP600 has a design life of 60 years, the AP600 surveillance program includes eight specimen capsules instead of four capsules. Thus the design of the AP600 includes a surveillance program founded on the extended design life and consequently exceeds the minimum requirements of ASTM E 185. Compliance with the materials surveillance requirements of Appendix H of 10 CFR Part 50 and ASTM E 185 satisfies the requirements of GDC 32 regarding an appropriate surveillance program for the reactor vessel.

- The integrity of the AP600 RV closure studs is assured by conformance with the recommendations of RG 1.65, thus satisfying the quality standards requirements of GDCs 1 and 30, and 10 CFR 50.55a. Compliance with the recommendations of RG 1.65 also satisfies the prevention of fracture of the RCPB requirement of GDC 31, and the requirements of Appendix G of 10 CFR Part 50 as detailed in the provisions of Sections II and III of the ASME Code.

5.3.3 Pressure Temperature Limits

GDC 1 of Appendix A of 10 CFR Part 50 and 10 CFR Part 50, 10 CFR 50.55a(a)(1), require 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 P/T limits imposed on the AP600 RV materials to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met as they relate to the selection of materials for the RV and their ability to assure adequate safety margins for the structural integrity of the RCPB ferritic components.

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 P/T limits imposed on the RV to ensure that the materials selected for the RV meet the relevant requirements of GDC 14, in that they possess adequate fracture toughness properties to resist rapidly propagating failure and act in a nonbrittle manner.

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 P/T limits imposed on the RV materials to ensure that the relevant requirements of GDC 31 have been met as they relate to behavior in a non-brittle manner and an extremely low probability of rapidly propagating fracture.

GDC 32 requires, in part, that the RCPB components shall be designed to permit an appropriate material surveillance program for the RV beltline region. The staff reviewed the P/T limits imposed on the RV materials to ensure that the relevant requirements of GDC 32 have been met as they relate to a materials surveillance program that monitors the change in the fracture toughness properties of the ferritic materials used in the RV.

Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 specifies the fracture toughness requirements for ferritic materials of the pressure-retaining components of the RCPB. The staff reviewed the P/T limits as they relate to the prediction of the effects of neutron radiation on the reference temperature and upper shelf energy of the RV beltline material required by Appendix G.

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 presents the requirements for a materials surveillance program to monitor the changes in fracture toughness properties of materials in the RV beltline region resulting from exposure to neutron irradiation and the thermal environment. These requirements include conformance with ASTM E 185. Compliance with Appendix H satisfies the requirements of GDC 32 regarding the provision of an appropriate materials surveillance program for the reactor vessel. The staff reviewed the P/T limits imposed on the RV materials to determine that the relevant requirements of Appendix H have been met regarding the establishment of a materials surveillance program to monitor changes in the reference temperature and upper shelf energy of the RV beltline material.

The staff reviewed the P/T limits for the AP600 in accordance with Section 5.3.2 of the SRP to assure adequate safety margins of structural integrity for the ferritic components of the RCPB. In the course of its review, the staff transmitted to Westinghouse RAIs concerning the RCPB materials and the P/T limits, and received from Westinghouse responses to these RAIs. In addition the staff and Westinghouse held several discussions to help clarify and resolve outstanding issues.

The staff reviewed the P/T limits that will be imposed on the RCPB during the operations and tests listed below to ensure that there will be adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by GDC 31:

- preservice hydrostatic tests
- inservice leak and hydrostatic tests
- heatup and cooldown operations
- core operation - criticality.

The fracture toughness requirements for ferritic materials in the pressure-retaining components of the RCPB are specified in Appendix G of 10 CFR Part 50. Changes in the fracture toughness properties of materials in the RV beltline region, resulting from exposure to neutron irradiation and the thermal environment, are monitored during the reactor lifetime by a surveillance program in compliance to the requirements of Appendix H of 10 CFR Part 50. The effect of neutron fluence on the shift in the nil-ductility temperature (NDT) of the pressure vessel steel is predicted by using Revision 2 of RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

Appendices G and H to 10 CFR Part 50 require the applicant to predict the amount of increase in the reference temperature, RT_{NDT}, resulting from neutron irradiation. This increase in RT_{NDT} is then added to the initial RT_{NDT} and the margin (the quantity added to obtain conservative upper-bound values) to establish the adjusted reference temperature. The staff's recommended method for calculating the increase in RT_{NDT} resulting from neutron irradiation is contained in RG 1.99, Revision 2. The relationships contained in the guide were derived by statistical analysis of 216 material data points that were reported from the testing of irradiated materials. These materials were contained in surveillance capsules and had been irradiated inside U.S. commercial nuclear reactor vessels. The relationship between the increase in RT_{NDT} and neutron fluence is empirically derived from the analysis of this surveillance data.

The limits on residual elements for the AP600 RV material are included in Table 5.3-1 of the SSAR. On the basis of previous assessment of the effects of impurity elements on the irradiation response of materials, the existing controls on residual elements are concluded to be sufficient. The limits on copper, phosphorus and other residual elements will minimize the extent of radiation damage to the RV beltline materials. The test results from the surveillance program should provide data to determine more accurately the rate of embrittlement. The radiation-induced shifts in reference temperatures for these materials can be predicted with reasonable accuracy and conservatism using the methodology of RG 1.99, Revision 2. Because of these controls, it is anticipated that embrittlement will proceed at a low rate.

The P/T curves for the AP600 are shown in Figures 5.3-2 and 5.3-3 of the SSAR. These curves are generic and are determined, in part, by the maximum copper and nickel contents specified in the SSAR. A COL applicant, referencing the AP600 standard design, should submit actual P/T curves for staff review. This was COL Action Item 5.3.3-1. The staff requested that Westinghouse include COL Action Item 5.3.3-1, related to actual pressure-temperature limit curves, in the SSAR. This was Open Item 5.3.3-1. Westinghouse subsequently revised the SSAR (Revision 3) and Section 5.3.5.1 indicates that the COL applicant will address plant-specific curves during procurement of the reactor vessel. If the plant-specific curves fall within the acceptable region of the generic curves, then they will be deemed acceptable. This is acceptable to the staff. Therefore, Open Item 5.3.3-1 and COL Action Item 5.3.3-1 are closed.

Westinghouse calculated predicted shifts in the reference temperature for the RV materials using the methodology of RG 1.99, Revision 2. This RG provides reasonably accurate and conservative predictions of adjusted reference temperatures for RV beltline materials that are produced domestically. Westinghouse's approach is therefore acceptable for domestically produced steels. The staff has postulated that steels from nondomestic sources could have different characteristic responses to radiation embrittlement, particularly those steels with high phosphorus and sulfur contents. Conceivably, the methodology adopted in RG 1.99, Revision 2 might cease to be appropriate. Thus, COL applicants that propose to use non-domestic steel for RV beltline applications should consider the need to estimate neutron irradiation embrittlement with a methodology more relevant than that defined in RG 1.99, Revision 2. This was COL

Action Item 5.3.3-2. The staff requested that Westinghouse include COL Action Item 5.3.3-2, related to the use of nondomestic steel for beltline materials, in the revised SSAR. This was Open Item 5.3.3-2. In response, Westinghouse emphasized that, irrespective of the source, the RV beltline material would be procured to ASME Code specifications. Additionally, Table 5.3-1 in Revision 3 of the SSAR shows that restrictive maximum content limits would be imposed on critical residual elements (copper, nickel, phosphorus, sulfur and vanadium); these limits are consistent with the guidelines for new plants contained in RG 1.99, Revision 2. The staff accepts that the chemical content controls imposed on the RV materials meet the guidelines for new plants specified in RG 1.99, Revision 2, and compliance with the requirements of the ASME Code specifications should ensure a quality product. Further, Revision 3 of the SSAR requires the COL applicant to address verification of plant-specific RV beltline material properties. Consequently, the staff has determined that the need for the COL applicant to address technical aspects specifically related to nondomestic steel is not necessary at this time. Therefore, Open Item 5.3.3-2 and COL Action Item 5.3.3-2 are closed.

The staff concludes that the P/T limits imposed on the RCS for operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure are in conformance with the fracture toughness criteria of Appendix G to 10 CFR Part 50. The change in fracture toughness properties of the RV beltline materials during operation will be determined through a surveillance program in conformance with Appendix H to 10 CFR Part 50. The use of operating limits, determined by the criteria defined in Section 5.3.2 of the SRP, provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the requirements of 10 CFR 50.55a of 10 CFR Part 50 and of GDCs 1, 14, 31, and 32 of Appendix A to 10 CFR Part 50.

5.3.4 Reactor Vessel Integrity

Although the staff reviewed most areas separately in accordance with its review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity was warranted. The staff reviewed the fracture toughness of the ferritic materials for the reactor vessel and the RCPB, the P/T limits for the operation of the reactor vessel, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references that are the bases for this evaluation are provided in Section 5.3.3 of the SRP.

The staff reviewed the information in each area to ensure that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed and the sections of this report in which they are discussed are given below.

- pressure boundary materials (Section 5.2.3)
- ISI and testing of the RCPB (Section 5.2.4)
- reactor vessel materials fabrication methods (Section 5.3.2)
- pressure-temperature limits and operating conditions (Section 5.3.3)

The staff concludes that the structural integrity of the AP600 reactor vessel meets the requirements of GDCs 1, 4, 14, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Appendices B, G, and H to 10 CFR Part 50; and 10 CFR 50.55a; and is therefore acceptable. The basis for this conclusion is that the design, materials, fabrication, inspection, and quality assurance requirements for the AP600 plant will conform to the applicable NRC regulations and RGs set forth above, and the rules of the ASME Code, Section III. The stringent fracture toughness requirements of the regulations and the ASME Code, Section III, will be met, including requirements for surveillance of vessel material properties throughout service life, in accordance with Appendix H to 10 CFR Part 50. Also, operating limitations on temperature and pressure will be established for the plant in accordance with Appendix G, "Protection Against Nonductile Failure," of ASME Code Section III, Appendix G, 10 CFR Part 50.

The integrity of the reactor vessel is assured because:

- The RV will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and the pertinent Code Cases discussed in Section 5.2.1.2.

- The RV will be made from material of controlled and demonstrated quality.

- The RV will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrications deficiencies.

- The RV will operate under conditions, procedures, and protective devices that provide assurance that the vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated transients.

- The RV will be subjected to periodic inspection to demonstrate that the high initial quality of the RV has not deteriorated significantly under service conditions.

- The RV will be subjected to surveillance to account for neutron irradiation damage so that the operating limitation may be adjusted.

5.3.5 Pressurized Thermal Shock

In Section 50.61 of 10 CFR Part 50, the NRC defines the fracture toughness requirements for protection against pressurized thermal shock (PTS) events. Section 50.61 establishes the PTS screening criteria, below which no additional action is required for protection from PTS events. The screening criteria are given in terms of reference temperature (RT_{PTS}). These criteria are 148.9 °C (300 °F) for circumferential welds and 132.2 °C (270 °F) for plates, forgings, and axial welds.

The NRC staff reviewed the SSAR to ensure that the AP600 design meets the requirements of 10 CFR 50.61 in that the fracture toughness properties of the RV beltline materials will be substantially below the PTS screening criteria after 60 years of operation. In the course of its review, the staff transmitted to Westinghouse RAIs concerning the RV beltline materials and received from Westinghouse responses to these RAIs. In addition, the staff and Westinghouse held discussions to help clarify and resolve outstanding issues.

PTS events are system transients in a pressurized-water RV that can cause severe overcooling of the vessel wall, followed by immediate repressurization to a high level. The thermal stresses, caused when the inside surface of the RV cools rapidly, combined with the high pressure stresses will increase the potential for fracture if a flaw is present in low-toughness material. The materials most susceptible to PTS are the materials in the RV beltline where neutron radiation gradually embrittles the material over time.

The methodology to be used in calculating the RT_{PTS} value is specified in 10 CFR 50.61(b)(2) and includes use of the equation:

$$RT_{PTS} = I + M + \Delta RT_{PTS}$$

where I is the initial reference temperature (RT_{NDT}), M is the margin used to cover uncertainties in the various parameters and the calculational method, and ΔRT_{PTS} is the mean value of the adjustment in reference temperature caused by irradiation. ΔRT_{PTS} is calculated from knowledge of the copper and nickel contents of the material and the neutron fluence.

In its response to an RAI 252.86, Westinghouse demonstrated that the AP600 design meets the PTS screening criterion. The AP600 reactor beltline design consists of two forgings and one circumferential weld. The AP600's beltline forging material and weld metal will contain a maximum of 0.03 wt. percent copper and 0.85 wt. percent nickel. The initial RT_{NDT} both for the forgings and the weld metal will be less than or equal to -28.9 °C (-20 °F). The maximum assumed neutron fluence is 1.632E+19 n/cm^2 for the forgings and 2.1E+18 n/cm^2 for the circumferential weld at end-of-life (60 years). The margins, defined in 10 CFR 50.61, are 18.9 °C (34 °F) for the forgings and 31.1 °C (56 °F) for the circumferential weld.

Using the above values, the staff has determined that after 60 years of operation the RT_{PTS} values for the forgings and circumferential weld will be 2.2 °C (36 °F) and 15.6 °C (60 °F), respectively, well below the PTS screening criteria.

As part of the staff's ongoing review of operating plants, it analyzes the plate, forging and weld metal surveillance data from these plants. The surveillance data are maintained in a power reactor embrittlement database. Recent evaluation of these surveillance data indicates that, for forgings, a standard deviation for the increase in reference temperature of 14.4 °C (26 °F) and a margin of 28.9 °C (52 °F) may be more appropriate. The staff is currently reviewing the database to determine whether 10 CFR 50.61 should be revised. If the RT_{PTS} value is determined using the margin value of 28.9 °C (52 °F), higher than that specified in 10 CFR 50.61, the projected RT_{PTS} value for the AP600 forgings would be 12.2 °C (54 °F). This value is still well below the PTS screening criterion for forgings of 132.2 °C (270 °F).

A COL applicant, referencing the AP600 standard design, should verify that plant-specific material properties and end-of-life fluence (60 years) are within the limits assumed in the AP600 analysis. This was COL Action Item 5.3.5-1. The staff requested that Westinghouse include COL Action Item 5.3.5-1, related to plant-specific material properties, in the SSAR. This was Open Item 5.3.5-1. Revision 3 of the SSAR incorporates a section stating that the COL applicant will address verification of plant-specific beltline material properties. On the basis of the inclusion of the requested commitment in the SSAR, Open Item 5.3.5-1 and COL Action Item 5.3.5-1 are closed.

The staff concludes that the AP600 RV meets the relevant requirements of 10 CFR 50.61. The staff's conclusion is on the basis of the finding that the RV beltline materials will be substantially below the PTS screening criteria after 60 years of operation.

5.4 Component and Subsystem Design

In Section 5.4 of the SSAR, Westinghouse describes the design of RCS components and subsystems for the AP600.

5.4.1 Reactor Coolant Pump Assembly

The AP600 RCPs are single-stage, hermetically sealed, high-inertia, centrifugal, canned-motor pumps. There are a total of four RCPs, two in each SG. Two pumps, rotating in opposite directions, are directly connected to the two outlet nozzles on the SG channel heads. The RCPs are designed to pump large volumes of reactor coolant at high pressures and temperature. High volumetric flow rates are needed to ensure adequate core heat transfer so as to maintain a departure from nucleate boiling ratio (DNBR) greater than the acceptable limit established in the safety analysis. Rotational inertia of a flywheel and other rotating parts in the pump assembly results in continuous coastdown flow after an RCP trip.

The RCP is an integral part of the RCPB. A canned motor pump contains the motor and all rotating components inside a pressure vessel. The pressure vessel consists of the pump casing, thermal barrier, stator shell, and stator cap, which are designed for full RCS pressure. The stator and rotor are encased in corrosion-resistant cans that prevent contact of the rotor bars and stator windings with the reactor coolant. Because the shaft for the impeller and rotor is contained within the pressure boundary, seals are not required to restrict leakage out of the pump into containment.

The RCP driving motor is a vertical, water-cooled, squirrel-cage induction motor with a canned rotor and a canned stator. It is designed for removal from the casing for inspection, maintenance, and replacement, if required. The motor is cooled by component cooling water circulating through a cooling jacket on the outside of the motor housing and through a thermal barrier between the pump casing and the rest of the motor internals. Inside the cooling jacket are coils filled with circulating rotor cavity coolant. This rotor cavity coolant is a controlled volume of reactor coolant that circulates inside the rotor cavity.

5.4.1.1 Pump Performance

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. For PWR designs, SRP Section 4.4 specifies the criterion necessary to meet GDC 10 as that the hot rod in the core does not experience a departure from nucleate boiling, or the DNBR limit is not violated, during normal operation or anticipated operational occurrences.

The RCP is sized to deliver a flow rate that equals or exceeds that required to ensure adequate thermal performance under normal and anticipated transient conditions. Adequacy of the RCP design capacity of delivering the forced reactor coolant flow and coastdown flow rates after a RCP trip is verified through the safety analyses of the design basis transients to ensure that the DNBR limit is not violated during the transients. In the DSER, the staff identified Open Item 5.4.1.1-1, stating that Westinghouse should ensure that the RCP flow rate used in the Chapter 15 transient and accident analyses are conservative when the final head-capacity curve for the AP600 RCPs is provided. Open Item 5.4.1.1-1 is resolved based on the following discussion. The RCP design parameters with the design flow rate of 11,582 m^3/hr (51,000 gpm) per pump, the developed head of 73.15 m (240 ft), and the synchronous speed of 1800 rpm are specified in SSAR Table 5.4-1. SSAR Table 4.4-1 provided the thermal and hydraulic data for the AP600 design with the vessel minimum measured flow rate of 43,876 m^3/hr (193,200 gpm), and the vessel thermal design flow rate of 43,058 m^3/hr (189,600 gpm), representing the design and measurement flow uncertainties of 1.8 percent. SSAR Table 15.0-3 lists the nominal values of pertinent plant parameters utilized in the accident analyses. With the assumption of 10 percent SG tube plugging, the minimum measured and thermal design flow rates of 43,873 m^3/hr (193,200 gpm) and 43,058 m^3/hr (189,600 gpm), respectively, are used in Chapter 15 safety analyses with or without the revised thermal design procedure. AP600 TS LCO 3.4.1 requires the RCS flow to be greater than or equal to the minimum measured flow rate of 43,876 m^3/hr (193,200 gpm) for Mode 1 power operation, with a surveillance verification every 24 hours per TS surveillance requirement SR 3.4.1.3. This will ensure that the RCS flow rate used in the Chapter 15 transient and accident analyses are conservative with respect to the actual RCS flow rate delivered by the RCPs. The staff has reviewed the safety analyses of the design-basis events described in Chapter 15 of the SSAR. With the minimum measured flow rate of the reactor coolant as the initial condition, and the flow coastdown after the reactor trip, the DNBR limit is not violated for all the anticipated transients analyzed. therefore, the staff concludes that the RCP design flow capacity is acceptable. The total delivery capability of the four RCPs will be verified per inspection, test, analysis, and acceptance criteria (ITAAC) Table 2.1.2-4, Item 9.a.

Section 14.2.8.1.40 of the SSAR originally required preoperational tests to be performed for the RCPs to measure the system and operating parameters of the RCPs. In the DSER, the staff identified COL Action Item 5.4.1.1-1, stating that a COL applicant should submit its planned preoperational test program to be performed for the RCPs, and Open Item 5.4.1.1-2, requiring this COL action item be included in the SSAR. Chapter 14, Initial Test Program, of the SSAR has since been revised. The startup testing of the AP600 initial test program requires the verification of adequacy of the RCS flow rate by (1) measurement prior to initial criticality, per Item 14.2.10.1.17, to verify adequacy of the RCS flow rate for power operation, and (2) measurement at approximately 100-percent rated thermal power condition, per Item 14.2.10.4.11, to verify that the RCS flow equals or exceeds the minimum value required by the plant technical specifications. The COL applicant is required by SSAR 14.4.2, Test Specification and Procedure, to provide test specifications and test procedures for the pre-operational and startup tests for review by NRC. Therefore, COL Action Item 5.4.1.1-1 and Open Item 5.4.1.1-2 are closed.

To provide operational integrity and to minimize the potential for cavitation, ample margin is provided between the available net positive suction head (NPSH) and the required NPSH by conservative pump design and operation. In response to RAI 440.124, Westinghouse made the following assertions:

- There are no special requirements and/or restrictions related to the design and operation of the canned-motor pumps to achieve the NPSH margins during normal operation.

- The required NPSH is well within the operating RCS pressure during heatup, cooldown, and power operation with four pumps running.

- It may be necessary to restrict the operation of certain combinations of pumps when running at low reactor coolant pressures approaching the cut-in pressure of the normal residual heat removal system.

In the DSER, the staff identified Open Item 5.4.1.1-3, stating that Westinghouse did not provide these pump restrictions or operation guidelines. Section 5.4.1.3.1 of the SSAR states that the required NPSH of the RCPs is provided with ample margin to provide operational integrity and minimize the potential for cavitation, and that the AP600 does not require RCP operation to achieve safe shutdown, and minimum NPSH requirements are not required to provide safe operation of the AP600. The setpressure of the relief valve in the RNS suction line for LTOP protection is higher than the lowest permissible set pressure to satisfy the required NPSH of the RCPs. The staff concludes that there is no need for the RCP operational restriction or guidelines when running at low reactor coolant pressure approaching the cut-in pressure of the RNS. Therefore, Open Item 5.4.1.1-3 is closed.

5.4.1.2 Coastdown Capability

For reactor fuel protection, each RCP has a high-density flywheel and high-inertia rotor. These provide rotating inertia to increase the pump's coastdown time following a pump trip and loss of electrical power. Continued coastdown flow of reactor coolant is important in ensuring that the fuel's DNBR limit will not be violated in the event of a partial or complete loss of the forced reactor coolant flow analyzed in Chapter 15.3 of the SSAR. The adequacy of the RCP flywheel-rotor design to provide for sufficient rotating inertia, and thus flow coastdown capability following and RCP trip, is verified through the safety analyses of the loss of flow transients to demonstrate that the minimum DNBR limit is not violated. In the DSER, the staff identified Open Item 5.4.1.2-1, stating that the staff's evaluation of these events was not complete at the time, and that Westinghouse should ensure that it has provided correct or conservative RCP coastdown flow rates for use in the Chapter 15 design-basis analyses. The staff has reviewed the safety analyses of the design-basis transients of partial and complete loss of forced reactor flow described in Sections 15.3.1 and 15.3.2 of the SSAR, respectively. The RCP coastdown flow rate is calculated on the basis of an RCP rotating moment of inertia of 210.7 kg-m^2 (5,000 lb-ft^2), which is specified in SSAR Table 5.4-1, using the LOFTRAN computer code, which has been approved for the AP600 transient analyses as discussed in Section 21.6.1 of this report. The analysis results of partial and complete loss of forced reactor coolant flow demonstrate that, with coastdown of the affected pumps, the DNBR does not decrease below the design basis limit value at any time during the transients. Therefore, the staff concludes that the RCP flywheel design provides adequate flow coastdown capability, and Open Item 5.4.1.2-1 is closed.

In the DSER, the staff also identified Open Item 5.4.1.2-2, stating that Westinghouse should ensure that RCP coastdown flow be included in the ITAAC. Westinghouse has revised the SSAR to require that the rotating moment of inertia of the RCPs be verified by the ITAAC, per Table 2.1.2-4, Item 8c. Therefore, Open Item 5.4.1.2-2 is closed.

5.4.1.3 Rotor Seizure

In Section 5.4.1.3.6.2 of the SSAR, Westinghouse states that the design of the AP600 RCP (and motor) precludes the instantaneous stopping of any rotating component. However, a design-basis analysis of a postulated RCP rotor seizure is presented in Section 15.3.3 of the

SSAR. The analysis of thermal and hydraulic effects of the locked rotor event uses a nonmechanistic, instantaneous stop of the impeller. This conservative assumption bounds any slower stop. The transient analysis considers the effect of the locked rotor on the reactor core and RCS pressure to demonstrate that acceptable RV pressure boundary and radiological consequence limits are not exceeded. The staff reviewed the analysis of the pump rotor seizure event as part of the Chapter 15 design-basis analysis and found the result to be acceptable as discussed in Section 15.3.3 of this report.

5.4.1.4 Reactor Coolant Pump Flywheel Integrity

GDC 1 and 10 CFR 50.55a(a)(1) require that SSCs important to safety shall be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The NRC staff reviewed the AP600 RCP flywheel design to ensure that the relevant requirements of GDC 1 and 10 CFR 50.55a(a)(1) have been met as they relate to the pump flywheel design, materials selection, fracture toughness, preservice and ISI requirements, and overspeed test procedures to determine their adequacy to assure a quality product commensurate with the importance of the safety function to be performed.

GDC 4 requires that SSCs of nuclear power plants important to safety be protected against the effects of missiles that might result from equipment failures. Because flywheels have large masses and rotate at speeds of 900 to 1800 rpm during normal reactor operation, a loss of flywheel integrity could result in high energy missiles and excessive vibration of the RCP. The safety consequences could be significant because of possible damage to the RCS, the containment, or the engineered safety features. The staff reviewed the flywheel design to the relevant requirements of GDC 4 as it relates to protecting safety-related SSCs from the effects of missiles that might result from RCP failure.

The staff reviewed the RCP flywheel integrity in accordance with Section 5.4.1.1 of the SRP. In the course of its review, the staff held several meetings with Westinghouse and transmitted RAIs to Westinghouse concerning the flywheel design. The staff met with Westinghouse in November 1993 and March 1994, and numerous RAIs and RAI responses have been transmitted. A second round of 29 RAIs was transmitted to Westinghouse on September 2, 1994.

The AP600 RCP flywheel assembly is fabricated from a high-quality, depleted uranium alloy casting and is enclosed within the pump casing. Thus it is located within the RCPB and the RCPB will capture the fragments of any postulated flywheel fracture. The RCP's stator shell and flange serve to shield the balance of the RCPB from theoretical worst-case flywheel failures. The purpose of this design is to obviate the need for an ISI program.

The RCP flywheel and stator shell were analyzed to demonstrate that a fractured flywheel cannot breach the RCS pressure boundary (stator shell and flange) and impair the operation of nearby safety-related systems or components. The analysis of the flywheel capture is predicated on energy calculations. The analysis of theoretical worst-case flywheel failures is similar to the approach taken with theoretical worst-case turbine disc failures analysis. The highest amount of energy available from a RCP flywheel failure is a small fraction (approximately 9 percent) of that necessary to breach the boundary.

The staff considers that the precautions taken to prevent RCPB breaching, the advantages in providing the necessary inertia to fulfill plant safety functions, and the redundancy of four RCPs allow this flywheel design to be considered acceptable for this application. Further, because analysis has demonstrated that flywheel failure is unlikely to lead to RCPB breaching and will not result in a missile that could have adverse effects on the plant safety functions, the requirement for an ISI program to preclude such failures is unnecessary from a safety standpoint.

The flywheel are enclosed in an Alloy 600 can. The nickel can is welded utilizing a flex-foot design. This aspect of the design has been used on other motor rotor designs and no failures have occurred. However, the flex-foot design of the flywheel can is subjected to high stresses and this may have an impact on meeting the goal of a 60-year life. The staff requested that Westinghouse provide additional information to demonstrate the adequacy of this particular enclosure. This was Open Item 5.4.1.4-1. In a letter dated February 7, 1995, Westinghouse provided more details on the weld design, including the implementation of ASME Code stress limits and reduction factors. The staff reviewed the additional information and found that appropriate design efforts had been made in an attempt to meet the goal of a 60-year design life. Therefore, Open Item 5.4.1.1 is closed.

The uranium flywheel castings are made by a centrifugal casting process that minimizes casting defects. The flywheel is subjected to volumetric and surface examinations. There is, however, a lack of knowledge related to the fracture toughness of this flywheel material. Westinghouse proposed using Charpy V-notch energy as the criterion to control fracture toughness as a quality assurance method of production parts. The staff does not believe that Charpy V-notch energy of uranium alloys is a viable measure of fracture toughness and recommended that fracture mechanics data such as J_{Ic} or K_{Ic} should be used. Otherwise, Westinghouse must demonstrate the validity of a correlation between the Charpy V-notch and J_c or K_{Ic} values for the uranium flywheel material. This was Open Item 5.4.1.4-2. In a letter dated March 7, 1995, Westinghouse agreed with this assessment and emphasized that, because of the limited applicability of Charpy V-notch data, it will only use such data when direct fracture toughness data are available for comparison. The lack of fracture toughness data for uranium alloys diminishes the reliability aspect of the design and may lead to future economic penalties should flywheel fracture occur. However, the staff finds that the potential for diminished reliability does not have an adverse effect on the plant safety because Westinghouse has demonstrated that the highest amount of energy available from flywheel failure is less than 10 percent of that necessary to breach the RCPB. Therefore, Open Item 5.4.1.2 is closed.

The staff concludes that the measures taken to assure the integrity of the RCP flywheels are acceptable and meet the safety requirements of GDCs 1 and 4, and 10 CFR 50.55a(a)(1). The staff's conclusion is on the basis of the following findings:

- The applicant's selection of materials, fracture toughness tests, design procedures, and overspeed spin testing program for the RCP flywheels have been reviewed and found to meet the requirements of GDC 1 and 10 CFR 50.55a(a)(1) with respect to providing adequate assurance of a quality product commensurate with the importance of the safety function.

- The applicant has met the requirements of GDC 4 by ensuring that failures of the flywheel will not result in a missile that could adversely impact SSCs important to safety.