3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 General

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the information in the AP1000 Design Control Document (DCD) Tier 2, Section 3.1, "Conformance with Nuclear Regulatory Commission General Design Criteria," to verify that the AP1000 design meets the relevant General Design Criteria (GDC) of Appendix A to Title 10, Part 50, of the Code of Federal Regulations (10 CFR Part 50).

The staff's review of structures, components, equipment, and systems relies, in part, on industry codes and standards that represent accepted industry practices.

Sections 3.7 and 3.8 of this report contain a significant portion of material quoted from the AP1000 DCD. The format used to cite the DCD in these two sections has been revised to be consistent with the nomenclature used throughout this report. Each section below identifies applicable codes and standards and discusses their basis for acceptability.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

In 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena," the NRC requires, in part, that nuclear power plant structures, systems, and components (SSCs) important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Some of these functions are safety related and necessary to ensure the following:

- integrity of the reactor coolant pressure boundary (RCPB),
- capability to shut down the reactor and maintain it in a safe-shutdown condition
- capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures that are comparable to the requirements in 10 CFR 50.34(a)(1)

The earthquake for which these safety-related plant features are designed is defined as the safe-shutdown earthquake (SSE) in Appendix S to 10 CFR Part 50. The SSE is based on an evaluation of the maximum earthquake potential; the SSCs are designed to remain functional through an earthquake which produces the maximum vibratory ground motion. Those plant features that are designed to remain functional, if an SSE occurs, are designated seismic Category I in Revision 3 of Regulatory Guide (RG) 1.29, "Seismic Design Classification." In addition, in Regulatory Position C.1 of 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 AP1000 DCD 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.

Design of Structures, Components, Equipment, and Systems

The safety-related SSCs and the equipment of the AP1000 standard plant that are classified as seismic Category I can be identified by comparing information in DCD Tier 2, Sections 3.2.1.2, "Classifications," and 3.2.4, "Application of AP1000 Safety-Related Equipment and Seismic Classification System," DCD Tier 2, Tables 3.2-1, 3.2-2, and 3.2-3, and applicable piping and instrumentation drawings (P&IDs) found in DCD Tier 2. DCD Tier 2, Table 3.2-3, "AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment," includes seismic classifications for fluid systems, as well as some components in these systems. However, this table does not explicitly include piping and piping supports. The P&IDs in the DCD identify the interconnecting piping and valves, as well as the interface between the safety-related and non-safety-related portions of each system. According to DCD Tier 2, Section 3.2.1.2, these interfaces are synonymous with the interface between seismic Category I and the nonseismic portions of each system. DCD Tier 2, Section 3.2.4 states that the supports for piping and components have the same seismic and safety classifications as the component or piping supported. Based on its review of DCD Tier 2, Sections 3.2.1.2 and 3.2.4, DCD Tier 2, Tables 3.2-1, 3.2-1, and 3.2-3, and the P&IDs as discussed above, the staff concludes that the safety-related SSCs in the AP1000 are acceptably classified as seismic Category I, in accordance with Position C.1 of RG 1.29.

In Position C.2 of RG 1.29, the NRC states that those portions of nonseismic 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 an incapacitating injury to occupants of the control room, should be designed and constructed so that an SSE could not cause such failure. In DCD Tier 2, Section 3.2.1.1.2, "Seismic Category II (C-II)," the applicant classified such SSCs as seismic Category II. DCD Tier 2, Section 3.7, "Seismic Design," discusses the design criteria for seismic Category II SSCs. In Position C.3 of RG 1.29, the NRC recommends guidelines for designing interfaces between seismic Category I and nonseismic SSCs. DCD Tier 2, Section 3.7.3.13, "Interaction of Other Systems with Seismic Category I Systems," provides the AP1000 information relative to Positions C.2 and C.3; Sections 3.7.2 and 3.12.3.7, respectively, of this report discuss the staff's evaluations of this information for structures and piping.

In Positions C.1 and C.4 of RG 1.29, the NRC states that the pertinent 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 that are covered under Positions C.2 and C.3 of RG 1.29. DCD Tier 2, Sections 3.2.2.3, "Equipment Class A"; 3.2.2.4, "Equipment Class B"; and 3.2.2.5, "Equipment Class C"; and Table 3.2-1 state that 10 CFR Part 50, Appendix B applies to all AP1000 Equipment Class A, B, and C (American Society of Mechanical Engineers (ASME) Class 1, 2, and 3) SSCs that are all classified as seismic Category I. Because all seismic Category 1 SSCs are covered, the staff concludes that this is an acceptable commitment to item (1) above. To satisfy Position C.4 of RG 1.29, the pertinent QA requirements of Appendix B to 10 CFR Part 50 should apply to all seismic Category II SSCs. In DCD Tier 2, Section 3.2.1.1.2, the applicant stated that pertinent portions of 10 CFR Part 50, Appendix B are applicable to the AP1000 seismic Category II SSCs. Accordingly, the staff concludes that this represents an acceptable commitment to item (2) above and is consistent with Position C.4 of RG 1.29.

DCD Tier 2, Table 3.2-3, properly identifies the new and spent fuel storage racks as seismic Category I. Although these items are also classified as AP1000 Class D, the staff's position is

that new and spent fuel storage racks are important to safety and, at a minimum, should meet the applicable QA requirements of Appendix B to 10 CFR Part 50, in addition to being classified as seismic Category I. In DCD Tier 2, Section 3.2.2.6, "Equipment Class D," the applicant stated that the requirements of 10 CFR Part 50, Appendix B apply to AP1000 Class D SSCs classified as seismic Category I. The staff concludes that this commitment is consistent with the guidelines in RG 1.29 and, therefore, is acceptable.

3.2.1.1 Conclusions

On the basis of its review of DCD Tier 2, Tables 3.2-1, 3.2-2, and 3.2-3, the applicable P&IDs, and other supporting information in DCD Tier 2, the staff concludes that the AP1000 safety-related SSCs, including their supports, are properly classified as seismic Category I, in accordance with Position C.1 of RG 1.29. In addition, the staff finds that DCD Tier 2 includes acceptable commitments to Positions C.2, C.3, and C.4 of RG 1.29. This constitutes an acceptable basis for satisfying, in part, the portion of GDC 2 which requires that all SSCs important to safety be designed to withstand the effects of natural phenomena, including earthquakes.

3.2.2 Quality Group Classification

In 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records," the NRC requires, in part, 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. These SSCs will be relied upon for the following functions:

• 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

The staff reviewed the AP1000 DCD in accordance with Section 3.2.2 of the SRP, which references Revision 3 of RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." The details of this review are discussed below.

In addition to the seismic classifications discussed in Section 3.2.1 of this report, DCD Tier 2, Tables 3.2-1 and 3.2-3 identify the AP1000 safety classification, the NRC quality group (QG) classification, and the QA requirements necessary to satisfy the requirements of GDC 1 for all safety-related SSCs and equipment. Applicable P&IDs identify the classification boundaries of interconnecting piping and valves. The staff reviewed DCD Tier 2, Tables 3.2-1 and 3.2-3 and the P&IDs in accordance with Section 3.2.2 of the SRP. Section 3.2.2 of the SRP references Revision 3 of RG 1.26 as the principal document used by the staff to identify, on a functional basis, the pressure-retaining components of those systems important to safety as NRC QG A, B, C, or D. Section 5.2.1.1 of this report discusses the conformance of ASME Class 1 RCPB

Design of Structures, Components, Equipment, and Systems

components to the requirements of 10 CFR 50.55a. 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 classified as QG D. Section 5.2.1.1 of this report further discusses the basis for this alternative QG classification.

In 10 CFR 50.55a, the NRC requires that safety-related equipment be designed and fabricated to the requirements of the ASME Code, Section III. In DCD Tier 2, Table 3.2-3, the applicant proposed to use the rules of the ASME Code, Section VIII, Appendix 22 for the design and construction of the air gas storage tanks in the main control room (MCR) emergency habitability system. The NRC staff reviewed the proposal and, for the reasons set forth below, concludes that the requirements of Appendix 22 to ASME Code, Section VIII provides an acceptable alternative to the use of ASME Code, Section III for the design and construction of the air storage tanks. This conclusion is based on the following justification:

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

- 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.

- The applicant specified that 10 CFR Part 50, Appendix B and 10 CFR Part 21 will apply to the manufacture of the air storage tanks.

- The tank material is specified to be Charpy V-notch tested, per supplement S3 of material specification SA-372, and is required to exhibit an average of 20 to 30 mils of lateral expansion at the lowest anticipated service temperature. This value meets 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 pursuant to 10 CFR 50.55a(a)(3).

In DCD Tier 2, Section 3.2.2, "AP1000 Classification System," the applicant described the AP1000 safety classification system. Safety-related SSCs are classified as AP1000 Equipment Class A, B, or C. In DCD Tier 2, Table 3.2-1, the applicant provided a correlation among the three methods of classification, (1) AP1000 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 among the three methods of classification defined in the DCD is shown below.

NRC QG	AP1000 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 AP1000 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 used herein, is an all-inclusive term encompassing the design, materials, 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 DCD Tier 2, Section 3.2.2.6. The staff concludes that the above table acceptably defines the relationship among the three methods of classification.

Based on its review of the information in DCD Tier 2, Section 3.2.2, Tables 3.2-1 and 3.2-3, and the applicable P&IDs, the staff concludes that the QG classifications for the AP1000 SSCs are consistent with the guidelines in RG 1.26, and are in conformance with GDC 1, and, therefore, are acceptable. However, during its review, the staff noted one exception to the classification guidelines in RG 1.26, the safety classification of the passive core cooling system (PXS), which is discussed below.

DCD Tier 2, Section 3.2.2.5, Table 3.2-3, and the P&IDs detailed in DCD Tier 2, Figures 6.3-1 and 6.3-2 collectively identify the following portions of the PXS as AP1000 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). The staff finds that these systems and components have been classified as QG C for the following reasons:

- 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, radioactive releases are contained.

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

- Continuous water level monitoring of the accumulators and the IRWST is performed to detect leaks.

The staff concludes that the QG C classification of the PXS and components identified above can satisfy the guidelines in RG 1.26, if the applicant makes a commitment that portions of

Design of Structures, Components, Equipment, and Systems

these systems will be inspected during construction using rules that are similar to those of ASME Class 2 (QG 2). 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. In addition, the measures described above are sufficient to allow the less stringent ISI rules of ASME Class 3 (QG C).

The staff position further indicates that the weld quality for the emergency core cooling system (ECCS) should be consistent with the system's safety functions, and that the butt welds in the ECCS piping should be examined in accordance with the ASME Code, Section III, ND-5222, using the full radiography option. Therefore, the staff's position is that in order to provide reasonable assurance that the affected systems will perform their safety function when required, the ECCS welds in the systems that are listed in DCD Tier 2, Section 3.2.2.5 should be subjected to this enhanced examination during construction. In DCD Tier 2, Section 3.2.2.5, the applicant states that full radiography in accordance with ASME Code, Section III, ND-5222 will be conducted on the piping butt welds during construction for systems that provide emergency core cooling functions. This conforms to the staff's position on this issue, and is, therefore, acceptable.

3.2.2.1 Conclusions

On the basis of its review of the applicable information in the DCD, and the above discussion, the staff concludes that the QG classifications of the pressure-retaining and non-pressure-retaining SSCs important to safety, as identified in DCD Tier 2, Tables 3.2-1 and 3.2-3, and related P&IDs in the DCD, are in conformance with RG 1.26 and, therefore, are acceptable. These tables and P&IDs identify major components in fluid systems (i.e., pressure vessels, heat exchangers, storage tanks, piping, pumps, valves, and applicable supports) and in mechanical systems (i.e., cranes, fuel handling machines, and other miscellaneous handling equipment). In addition, P&IDs in the DCD 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 as described above, and applicable ASME Codes and industry standards provides assurance that component quality will be commensurate with the importance of the safety functions 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 applicant discussed the design wind velocity and the corresponding applied forces for the AP1000 standard design in DCD Tier 2, Sections 3.3.1.1, "Design Wind Velocity," and 3.3.1.2, "Determination of Applied Forces." The applicant used the American Society of Civil Engineers (ASCE) Standard ASCE 7-98, "Minimum Design Loads for Buildings and Other Structures." The applicant used a basic wind speed of 233 kilometers per hour (kph) (145 miles per hour (mph)), and a 3-second wind gust speed at 10 meters (33 feet) above the ground in open terrain with a mean recurrence interval of 50 years. This basic wind speed is to be scaled by an importance factor (as defined in ASCE 7-98) of 1.0 and 1.15 for non-safety-related and safety-related structures, respectively. It should be noted that the NRC has not reviewed ASCE 7-98 in its entirety regarding applicability of its recommendations with respect to all other loads.

(A detailed review of ASCE 7-98 will be necessary before its acceptance as a general reference). Using ASCE 7-98, with its updated recommendations for higher basic wind velocities and other associated design factors for wind loading, adds conservatism to the design. Therefore, the staff finds its application in the design of the AP1000 to be acceptable.

The importance factor, I, is a multiplier for basic wind speeds shown in the maps of ASCE 7-98. The end product is a wind speed with an appropriate recurrence interval. The basic wind speed values of the maps in ASCE 7-98 are for a 50-year mean recurrence interval (annual probability of 0.02). The commentary, Section C6.5.5 of ASCE 7-98, explains that an importance factor of 1.15 is associated with a mean recurrence interval of 100 years, and is to be used to adjust the structural reliability of a building or other structures to be consistent with building classification. ASCE 7-98 describes four categories of structures, with Category IV including structures that are designed with the highest level of reliability. Category IV is applicable to hospitals, emergency shelters, power generating stations, and other vital facilities having critical national defense functions. The applicant has designated all seismic Category I structures for the AP1000 as Category IV with an importance factor of 1.15. The use of an importance factor of 1.15 is conservative.

Pressure generated from the design wind velocity is further dependent on exposure and gust response factors corresponding to the exposure categories. The applicant used exposure Category C, which is consistent with open shoreline and flat open country exposure. Category C exposure is suitable for most sites in the Eastern United States; however, it is not suitable for sites near open inland waterways; the Great Lakes; and the coastal areas of California, Oregon, Washington, and Alaska. The wind load design for the AP1000 makes it unsuitable for sites that fall under exposure Category D. Seismic Category I structures for the AP1000 are robust, and their lateral load resistance is generally governed by seismic and tornado loading. It may be feasible to demonstrate that the AP1000 wind design is adequate for exposure Category D. Without such a demonstration, the use of the appropriate wind exposure category is an open issue. This was Open Item 3.3.1-1 in the draft safety evaluation report (DSER).

After reviewing the applicant's initial response to this open item, submitted September 23, 2003, the staff found that the applicant had not provided a sufficient basis to support the use of exposure Category C in ASCE 7-98 to calculate the design wind loads. In a teleconference on August 22, 2003, both the staff and the applicant agreed to revisit this issue during the design audit based on careful examination of the requirements in ASCE 7-98.

During the audit on October 6–9, 2003, the staff reviewed Revision 1 to the open item response. The applicant indicated that the AP1000 is adequate for a maximum basic wind speed of 209 kph (130 mph) for exposure Category D, based on a comparison to the design wind loads, which are based on a 233 kph (145 mph) basic wind speed and exposure Category C. The applicant indicated that all the exposure Category D locations have specified basic wind speeds less than 209 kph (130 mph), and concluded that the AP1000 wind design is adequate for all exposure Category D locations. Applicable revisions to DCD Tier 2, Section 2.3, "Meteorology," and DCD Tier 2, Section 3.3.1.1 were also identified. The staff finds that the applicant has provided sufficient additional information to address exposure Category D. The staff confirmed that the identified revisions to DCD Tier 2, Section 2.3 and

Design of Structures, Components, Equipment, and Systems

DCD Tier 2, Section 3.3.1.1 were incorporated in the DCD. On this basis, Open Item 3.3.1-1 is resolved.

The applicant used ASCE 7-98 to calculate the pressure loadings on structures for the design tornado wind velocity and the associated vertical distribution of wind pressures and gust factors. The shape coefficients for the shield building, however, were calculated using ASCE Paper 3269, "Wind Forces on Structures." ASCE Paper 3269 is referenced in Section 3.3.1 of the SRP. It is not clear why the applicant used the latest ASCE standard for the basic wind velocity, importance category, and exposure category, but did not use the recommendations of ASCE 7-98 for the velocity pressure and the corresponding pressure and force coefficients. Structures in the AP1000 design are dynamically rigid, and the use of pressure coefficients different from those recommended in ASCE 7-98 is not likely to produce an unacceptable design because the lateral strength of the AP1000 structures is likely to be governed by seismic and tornado loads. Nevertheless, the staff requested that the applicant clarify its inconsistent use of the recommendations for wind load design in ASCE 7-98. This was Open Item 3.3.1-2 in the DSER.

The staff reviewed the initial response to this open item submitted by letter dated September 23, 2003. Based on its review, during a teleconference on August 22, 2003, the staff asked the applicant to compare the total horizontal load from ASCE 7-98 to that calculated using the circumferential pressure distribution from ASCE Paper 3269.

During the audit on October 6–9, 2003, the staff reviewed Revision 1 to the open item response. The applicant justified the applicability of the circumferential pressure distribution found in ASCE Paper 3269, based on a comparison to the pressure distribution obtained from AP600 wind tunnel tests, as documented in Appendix C to Westinghouse Commercial Atomic Power (WCAP)-13294-P, "Phase I Wind Tunnel Testing for the AP600 Reactor." The applicant also identified a corresponding revision to DCD Tier 2, Section 3.3.1.2 , and added the above WCAP report as Reference 6 in DCD Tier 2, Section 3.3.4. Because the pressure distribution obtained from ASCE Paper 3269 is consistent with the AP600 wind tunnel test results, the staff finds the use of ASCE Paper 3269 to be acceptable. The applicability of AP600 wind tunnel test results to the AP1000 design is discussed in the resolution of Open Item 3.3.2-3 in Section 3.3.2.3 of this report. The staff confirmed that the applicant incorporated the identified revisions into the DCD. On this basis, Open Item 3.3.1-2 is resolved.

3.3.1.1 Conclusions

For the reasons summarized below, the staff concludes that the analysis methodology and the procedures used by the applicant for the wind load design of the AP1000 seismic Category I structures are appropriate and acceptable for protecting public health and safety.

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 DCD Tier 2, Table 2-1 for future sites

- appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena

- the importance of the safety function to be performed

The applicant is addressing these considerations through the use of ASCE 7-98, in the calculation of effective pressure on structures from the design wind velocity, and in the selection of pressure coefficients corresponding to the structural geometry and physical configuration.

The design of all AP1000 safety-related structures for wind loads using acceptable procedures meets the requirements of GDC 2. The procedures, therefore, provide reasonable assurance that, together with other engineering design considerations (e.g., the combination of wind load with other loads as indicated in Section 3.8.4 of this report), the structures will withstand such environmental forces. The use of these procedures provides reasonable assurance that in the event of design-basis winds, the integrity of the plant structures within the scope of the standard design will not be impaired. Consequently, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions, if needed.

3.3.2 Tornado Loading

3.3.2.1 Tornado Loads on Exterior Structures

The applicant provided the tornado wind speed in DCD Tier 2, Table 2-1, "Site Parameters," and more detailed tornado design parameters in DCD Tier 2, Section 3.3.2.1, "Applicable Design Parameters."

The staff's position with regard to design-basis tornados was previously derived from two documents published in 1974, WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," and RG 1.76, "Design Basis Tornado for Nuclear Power Plants." 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 tornado wind speed as 579 kph (360 mph) for the contiguous United States.

The staff reevaluated the regulatory positions in RG 1.76 for the standard design of advanced light-water reactors (ALWR) using tornado data which became available since the RG was developed. NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," discusses this reevaluation. The staff's interim position ("ALWR Design Basis Tornado") regarding RG 1.76 was issued on March 25, 1988. In this interim position, the staff concluded that the maximum tornado wind speed of 531 kph (330 mph) is acceptable. However, in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," the staff recommended that the Commission approve its position that a design-basis tornado with a maximum tornado wind speed of 483 kph (300 mph) be adopted for the design of evolutionary and passive ALWRs, since the 483 kph (300 mph) tornado is suitable for most U.S. sites. In its staff requirements memorandum (SRM) dated July 21, 1993, the Commission approved the staff's position.

In a recent SRM (SRM-SECY-03-027 - Review Standard RS-002, "Processing Applications for Early Site Permits"), the Commission instructed the staff to update the review guidance, including RG 1.76, to reflect the more recent tornado wind speed data that is available. This

Design of Structures, Components, Equipment, and Systems

does not impact the AP1000 review because an applicant, using either the 10 CFR Part 52 or the Part 50 process, would have to meet the site interface requirements and justify any parameter, including tornado wind speed, which exceeds the parameters of the AP1000 design.

The applicant indicated in DCD Tier 2, Section 3.3.2, "Tornado Loadings," that all seismic Category I structures are designed to resist tornado loads without exceeding the allowable stresses defined in DCD Tier 2, Section 3.8.4, "Other Category I Structures." In addition, the seismic Category I structures are designed to remain functional when subjected to tornado-generated missiles, as discussed in DCD Tier 2, Section 3.5.1.4, "Missiles Generated by Natural Phenomenon." The design tornado wind speed for the AP1000 is 483 kph (300 mph) and is one of the site parameters postulated for the design. The values of tornado design parameters for the AP1000 meet the tornado design speeds approved for advanced reactor design per SECY-93-087, as approved in the July 21, 1993, SRM. Therefore, the staff finds the AP1000 design-basis tornado to be acceptable.

The procedures used to calculate pressure loads from the tornado wind velocity are the same as those used for wind, as discussed in Section 3.3.1 of this report. DCD Tier 2, Section 3.5, "Missile Protection," discusses the procedures used to determine the tornado missile effects; Section 3.5 of this report discusses the acceptability of these procedures. Tornado loading includes tornado wind pressure, internal pressure by tornado-created atmospheric pressure drop, and forces generated by the impact of tornado missiles. These loads are combined with other loads, as described in DCD Tier 2, Section 3.8.4. Section 3.8.4 of this report discusses the acceptability of these loads and load combinations. The applicant indicated that a maximum pressure drop of 13.8 kPa (kilopascals) (2 pounds per square inch (psi)) is used for nonvented structures, unless a lower value is justified by a detailed analysis using the provisions of ASCE 7-98 for partially vented structures. However, the applicant initially did not identify any structure within the scope of the AP1000 standard design for which a lower pressure drop had been used. The applicant was asked to identify all the structures for which it has used a pressure drop lower than 13.8 kPa (2 psi). Therefore, the use of a tornado pressure drop of less than 13.8 kPa (2 psi) for vented structures was identified as Open Item 3.3.2-1 in the DSER.

The staff reviewed the initial open item response, submitted by letter dated September 23, 2003, which identified one region that was assumed to be vented. The staff did not consider that the applicant had provided a sufficient technical basis for assuming zero differential pressure in the shield building annulus under design-tornado conditions. During a teleconference on August 22, 2003, the staff asked the applicant if the AP600 wind tunnel tests demonstrate this assumption, and if the use of zero differential pressure can be substantiated.

During the audit on October 6–9, 2003, the staff reviewed Revision 1 to the open item response. The applicant justified the assumption of full venting for the portion of the shield building surrounding the upper annulus due to the large area of the air inlets and discharge stack. The applicant also identified a corresponding revision to DCD Tier 2, Section 3.3.2.2, "Determination of Forces on Structures." Although there are no test data cited to support the assumption, the staff finds it acceptable on the basis that the openings have sufficient area to minimize any differential pressure between the inside and outside of this specific region of the shield building.

The staff confirmed that the DCD was revised to reflect these changes. On this basis, Open Item 3.3.2-1 is resolved.

3.3.2.2 <u>Effect of Failure of Structures or Components Not Designed for Tornado Loads</u>

The applicant stated in DCD Tier 2, Section 3.3.2.3, "Effect of Failure of Structures or Components not Designed for Tornado Loads," 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 because the applicants:

• designed the adjacent non-safety-related structure to the design-basis tornado loading

• investigated the effect of failure of adjacent structures on seismic Category I SSCs to determine that no impairment of safety function will result

• designed a structural barrier to protect seismic Category I SSCs from adjacent structural failure

The applicant stated in DCD Tier 2, Section 3.3.3, that combined license (COL) applicants referencing the AP1000 certified design will address site interface criteria for wind and tornado. These site interface criteria do not make it clear that the COL applicant needs to follow the three acceptable criteria described in DCD Tier 2, Section 3.3.2.3, to ensure that structures outside the scope of the certified design do not compromise the function of safety-related structures or systems of the AP1000 plant. Although DCD Tier 2, Table 1.8-2, mentions DCD Tier 2, Section 3.3.3, for the wind and tornado site interface criteria, neither DCD Tier 2, Section 3.3, nor DCD Tier 2, Table 1.8-2, clearly specifies that the COL applicant must ensure that a tornado-initiated failure of structures and components within the COL scope will not compromise the safety of the AP1000 safety-related structures and components. Identification of wind and tornado site interface criteria was identified as Open Item 3.3.2-2 in the DSER.

The staff reviewed the initial open item response. During a teleconference on August 22, 2003, the staff requested the applicant to revise DCD Tier 2, Section 3.3.3 and Table 1.8-2, to reference the three approaches described in DCD Tier 2, Section 3.3.2.3. This would clearly identify to COL applicants what options are available if they locate structures or components not normally designed for tornado loads in close proximity to the AP1000 nuclear island.

During the audit on October 6–9, 2003, the staff reviewed Revision 1 to the open item response. The applicant identified a revision to DCD Tier 2, Section 3.3.3, that clearly states that it is the COL applicant's responsibility to address site interface criteria for wind and tornado. On the basis that the applicant revised the DCD as requested by the staff, the staff finds these revisions to be acceptable. This is COL Action Item 3.3.2.2-1.

The staff confirmed that the DCD was revised to reflect these changes. On this basis, Open Item 3.3.2-2 is resolved.

Design of Structures, Components, Equipment, and Systems

3.3.2.3 Tornado Loads on Containment Shell and Air Baffle

The AP1000 containment structure is surrounded by the shield building which is open at the top with an air baffle located within the annulus between the steel containment structure and the shield building. The air baffle separates downward flowing air, which enters at the air intake openings at the top of the cylindrical part of the shield building, from upward flowing air that cools the containment vessel. The tornado pressure drop is calculated assuming that the center of the tornado is located at the top of the containment center. The applicant used the following wind tunnel test reports to derive the wind pressure profile for the air baffle from design wind, as well as tornado wind:

- WCAP-13323-P and WCAP-13324-NP, "Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor," dated June 1992

- WCAP-14068-P, "Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor," dated May 1994

- WCAP-14169-P, "Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report," dated September 1994

The shield building and air baffle arrangement produce a reduction in pressure inside the annulus. Consequently, this has the effect of increasing the internal pressure of the containment structure by about 6.9 kPa (1 psig).

The wind condition also creates a lateral pressure on the containment. Pressure loads for the shield building and air baffle arrangement were developed from wind tunnel tests. The arrangement of a structure inside another structure, as in the case of the shield building and the free standing steel containment structure, is an unusual configuration for which wind pressure coefficients are not readily available from any industry code or standard. For this reason, the applicant had conducted wind tunnel tests to determine the wind pressure distribution for the AP600 configuration. The AP600 configuration is identical to the AP1000 design in terms of cross-section and arrangement for wind flow path. Therefore, the NRC staff has determined that the wind pressure loading for the AP600 design is applicable to the AP1000 design for the wind load calculation because the determination of wind and tornado loads is in conformance with the considerations of GDC 2. On this basis, the staff finds that the wind load calculation for the AP1000 containment structure to be acceptable.

The staff's evaluation documented in this section concentrates on the differences between the AP1000 and the AP600 design, with the understanding that the AP600 wind tunnel test reports were found to be acceptable for the AP600 design in accordance with the staff's evaluation documented in Chapter 21 of NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design." The staff's basis for concluding that the AP600 wind tunnel test reports are applicable to the AP1000 design is discussed below. This was Open Item 3.3.2-3 in the DSER.

In its Revision 1 response to request for additional information (RAI) 220.00, the applicant described the technical basis for the applicability of three specific AP600 tests that were used to define the design loads for the AP1000 structures. The three tests utilized are (1) AP600

passive containment cooling system (PCS) water distribution test, (2) AP600 automatic depressurization system (ADS) hydraulic tests, and (3) AP600 wind tunnel tests. Based on similarities of design and operational parameters between the AP600 and the AP1000, as well as an assessment that any differences would have only a small influence on the test results, the applicant concluded that the AP600 test results are applicable to the AP1000. This is also discussed in Section 3.8 of this report.

During the April 2–5, 2003, audit, the staff reviewed WCAP-15613, "AP1000 PIRT and Scaling Assessment," which documents the applicant's technical bases for concluding that the AP600 test results are applicable to the AP1000. Based on its review of WCAP-15613, as clarified by the information provided in the RAI response, the staff accepted the applicability of these three AP600 tests to the design load definition for the AP1000 structures. On this basis, Open Item 3.3.2-3 is resolved.

3.3.2.4 <u>Conclusions</u>

For the reasons summarized below, the staff concludes that the analysis methodology and the procedures used by the applicant are appropriate and acceptable for tornado design of the AP1000 seismic Category I structures.

The AP1000 standard design meets the requirements of GDC 2 and the guidelines of Section 3.3.2 of the SRP with respect to its capacity to withstand design tornado wind loading and tornado missiles. The AP1000 design reflects the following:

- appropriate consideration of a design-basis tornado consistent with NRC policy

- appropriate combinations of the effects of 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 considerations are addressed by using criteria specified in SECY-93-087 and the methods of calculating the effective pressure on structures from wind velocity, as described in DCD Tier 2, Section 3.3.1.

By using design loads and load combinations to meet the guidelines of Section 3.8 of the SRP, the plant structures are designed with a margin sufficient to prevent the failure of structures during severe tornado loads (item 1 above). In addition, the design of seismic Category I structures includes the use of loads and load combinations of severe tornado loads and loads resulting from normal plant operation (item 2 above).

The use of procedures, as discussed above, gives reasonable assurance that, in the event of a design-basis tornado, the structural integrity of all seismic Category I structures will be maintained. Consequently, safety-related systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed (item 3 above).

Design of Structures, Components, Equipment, and Systems

3.4 External and Internal Flooding

3.4.1 Flood Protection

The staff reviewed the AP1000 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, 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), as well as the guidelines of RG 1.102, "Flood Protection for Nuclear Power Plants," with regard to the means used for protecting safety-related SSCs from the effects of the PMF and PMP. The staff's 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.

Sections 2.5.1 and 2.5.2 of this report evaluate the requirements of 10 CFR 100.23, "Geological and seismic siting criteria," Section (c) "Geological and seismological, and engineering characteristics," and Section (d) "Geological and seismic siting factors," as they relate to flooding.

In DCD Tier 2, Section 3.4.1, "Flood Protection," the applicant discussed the flood protection measures that are applicable to the AP1000 design for postulated external flooding resulting from natural phenomena, as well as internal flooding from system and component failures. The seismic Category I SSCs identified in DCD Tier 2, Section 3.2, "Classifications of Structures, Components, and Systems," 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 based on flooding considerations. As a result, non-safety-related SSCs are not important in the mitigation of flood events, and are not required by GDC 2 to be protected from either internal or external flooding.

Based on this information, the staff concludes that the applicant has identified the SSCs that require protection from external and internal 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 manmade events, along with seismic and wind effects. In DCD Tier 2, Section 2.4, "Hydrologic Engineering," the applicant stated that the AP1000 is designed for a normal ground water elevation up to 29.9 meters (m) (98 feet (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 design will fall within the values of these site parameters. This is COL Action Item 3.4.1.1-1.

External flooding does not occur from PMP. The roofs of the AP1000 buildings 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. DCD Tier 2, Table 2-1, defines PMP as 49.3 centimeters per hour (cm/hr) (19.4 inches per hour (in./hr)) and the maximum static roof load due to snow and ice buildup as 3.6 kPa (75 pounds per square foot (lb/ft-sq)). The roofs of the auxiliary and shield buildings are designed for snow loads in accordance with ASCE 7-98.

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

- two fire water tanks with a capacity of 1230 and 1514 kiloliters (kL) (325,000 and 400,000 gallons (g)), respectively, located near the turbine building (DCD Tier 2, Section 9.5.1, "Fire Protection System")

- the condensate storage tank with a capacity of 1836 kL (485,000 g), located near the turbine building (DCD Tier 2, Section 9.2.4, "Demineralized Water Transfer and Storage System")

- the demineralized water storage tank with a capacity of 378 kL (100,000 g), located near the annex building (DCD Tier 2, Section 9.2.4)

- the boric acid storage tank with a capacity of 265 kL (70,000 g), located next to the demineralized water storage tank (DCD Tier 2, Section 9.3.6, "Chemical and Volume Control System")

- two diesel fuel oil tanks, each with a capacity of 379 kL (100,000 g), which are not located near structures housing safety-related equipment and include dikes to retain leaks and spills (DCD Tier 2, Section 9.5.4, "Standby Diesel and Auxiliary Boiler Fuel Oil System")

- the passive containment cooling ancillary water storage tank with a capacity of 2953 kL (789,000 g), located at the west side of the auxiliary building (DCD Tier 2, Section 6.2.2.3, PCS Safety Evaluation)

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.

The AP1000 safety-related systems and components are housed exclusively in seismic Category I structures (i.e., the containment and auxiliary buildings). Seismic Category I structures are located such that the land slopes away from the structures. This ensures 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

Design of Structures, Components, Equipment, and Systems

material is applied to both vertical and horizontal exterior surfaces below grade. Waterstops are installed in exterior construction joints below grade.

The AP1000 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 raceways 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, shield, and auxiliary buildings.

The basemat and exterior walls of seismic Category I structures are designed to withstand the maximum lateral and buoyancy forces associated with the PMF and the highest postulated ground water level. Hydrodynamic forces were not considered in the structural design because the PMF and the highest postulated ground water 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 C.1 of RG 1.59, the NRC states, in part, that the conditions resulting from the worst-probable, site-related flood at a nuclear power plant, with attendant wind-generated wave activity, should constitute the design-basis flood condition from which safety-related SSCs must be protected. The AP1000 safety-related SSCs are designed to withstand the effects of external flooding in accordance with the above-stated criteria of Position C.1 of RG 1.59.

Based on its review, and for the reasons set forth above, the staff concludes that the applicant has properly identified the design-basis flood assumed for the AP1000 design, and has 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 AP1000 design conforms to the guidelines of Position C.1 of RG 1.59.

In Position C.2 of RG 1.59, the NRC provides alternative guidance for flood protection when the "hardened protection" method is not used. The hardened protection method provides that passive structural provisions be incorporated into the plant design to protect safety-related SSCs from the static and dynamic effects of floods. The AP1000 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 such acceptable method 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 incorporate 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. As described above, the AP1000 flood design incorporates 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 specifications 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 AP1000 flood design, which incorporates hardened protection. Consequently, Position C.2 of RG 1.102 is not applicable.

Based on the evaluation of the information provided in the DCD, and for the reasons set forth above, the staff concludes that the applicant has adequately characterized the PMP and PMF for the AP1000 flood design and provided design features to protect safety-related equipment from external flood effects associated with the PMP, PMF, ground water 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. The design also meets the guidelines of RG 1.102 with regard to acceptable external flood protection methods.

The AP1000 design can be used at either single-unit or multiple-unit sites. If more than one unit is built on the same site, the COL applicant should verify that the site-specific flood conditions are within the site parameters assumed in the AP1000 design.

3.4.1.2 Internal Flooding

Safety-related systems and components for the AP1000 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 include the following:

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

In the DCD, the applicant included the results of internal flooding analysis which described the consequences of compartment flooding for various postulated component failures. The analysis included the following elements:

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

The applicant identified the following flood sources considered in the analysis:

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

Design of Structures, Components, Equipment, and Systems

• flow from upper elevations and adjacent areas

Based on the above, and information provided in DCD Tier 2, Section 3.6.1, "Postulated Piping Failures Inside and Outside Containment," the staff did not identify other internal flood sources. The staff, however, requested that the applicant clarify its treatment of nonseismically supported, moderate-energy piping with respect to possible pipe ruptures as a result of a seismic event. In its response to RAI 410.001, the applicant revised the above listing to delete "moderate-energy piping (through-wall cracks)" as an identified flood source, and added the following two flood sources:

• through-wall cracks in seismically supported, moderate-energy piping
• breaks and through-wall cracks in nonseismically supported, moderate-energy piping

The staff did not identify any other internal flood sources. Based on these changes to the DCD, which clearly identify moderate-energy piping flood sources, the staff concludes that the applicant has adequately identified all the internal flood sources for the AP1000 design.

The criteria discussed in DCD Tier 2, Section 3.6, "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 tank rupture was postulated. Except for floor drains, no credit was taken for non-safety-related equipment to mitigate a flooding event.

Because the PMF for the AP1000 design is below 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.

DCD Tier 2, Section 7.4, "Systems Required for Safe Shutdown," identifies safety-related systems and components needed for safe shutdown. The safe shutdown systems and components located in containment are associated with the PXS, the ADS, and the containment isolation valves (CIVs).

In the DCD, the applicant identified seven compartments in containment that are subject to full or partial flooding. These include the reactor vessel cavity, two steam generator (SG) compartments, a vertical access tunnel, the chemical and volume control system (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 107'-2"). The reactor coolant system (RCS) cavity and the two SG compartments are connected 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.

As discussed below, 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 will flood the reactor and SG compartments to a level above the RCS piping during a loss-of-coolant accident (LOCA). The available flooding sources are the RCS, two accumulators, two core makeup tanks (CMTs), the IRWST, and makeup from the cask loading pit and boric acid tank. The flood water would cover the break location and allow backflow either through the break or via the PXS recirculation system flow path. In the event that the source of the containment flooding cannot be terminated, the resulting maximum flood level in containment is at Elevation 108'-10".

The fire protection system (FPS) and demineralized water transfer and storage system (DWS) are open-cycle systems that enter the containment. However, they are limited source systems. These systems are isolated during plant operation and are not a potential flooding source. They also have containment isolation valves which are redundant, such that two failures are needed to fail pipes with water.

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 residual heat removal (RNS) line, the 20.3 cm (8 in.) accumulator injection line, and the 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 reactor vessel cavity and the adjoining equipment room are located at the lowest level of the containment (Elevation 71'-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. Flow through each drain line, as well as total flow from all drain lines, is monitored in the 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 (DCD Tier 2, Section 5.2.5, "Detection of Leakage Through Reactor Coolant Pressure Boundary"). The containment flood-up level instrumentation consists of redundant, Class 1E sensor racks that 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

Design of Structures, Components, Equipment, and Systems

level (Elevation 107'-2"). Curbs are provided around the openings that penetrate the maintenance floor to prevent flooding of the compartments during a LOCA event until the water level on the maintenance floor reaches the top of the curbs. The elevations at the top of the curbs are 108'-10" and 108'-9" for PXS-A and PXS-B, respectively. The curb for the CVS compartment is lower than the PXS compartment curbs to preferentially allow flooding of the CVS compartment first.

Inside the PXS compartments, automatically actuated CIVs include one normally closed CIV for the spent fuel pit cooling system in PXS-A and three normally closed CIVs for the RNS 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 need repositioning during flooded conditions.

In addition, each PXS compartment contains four PXS containment recirculation subsystem isolation valves. A normally closed, explosively actuated valve is located in each of two parallel flow paths. One of the lines includes a check valve in series with the explosively actuated valve. The other line includes a normally closed, motor-operated valve in series with the explosively actuated valve. The explosively actuated and motor-operated valves are opened on a low IRWST-level signal to provide a redundant flow path from the flooded reactor/SG compartments to the reactor vessel. One set of these redundant containment recirculation subsystem isolation valves is required to open to provide a redundant recirculation flow path to the reactor vessel. In the unlikely event that one of the two PXS compartments were to be flooded, the set of recirculation valves in the other, unflooded compartment could be opened. Thus, a redundant, parallel flow path to the PXS system containment recirculation subsystem is provided.

The auxiliary building upper annulus provides the air flow path for the PCS. The annulus floor has a curb on the outside with a flexible seal 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 safety-related 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 level is not high enough to affect the operation of PCS air cooling, and no other safety-related equipment can be affected by this event.

The PCS valve room contains three redundant safety-related trains for the PCS. A through-wall crack of the PCS piping is the only flooding source for this room. Leakage will flow down to the landing at Elevation 264'-6"; the water will then flow through floor drains or under doors to the upper annulus and be discharged through redundant drains to the storm drain. A negligible amount of water will accumulate in the valve room. The PCS isolation valves are located above the maximum flood level in the valve room, so they remain operable. Level sensors in the valve room drain sump alarm in the MCR to alert operators to take corrective action if an abnormal water level in the valve room is detected. No safety-related equipment is affected by the worst-case flood scenario.

Based on its review, as set forth above, the staff concludes that the applicant properly identified safety-related equipment and flood hazards in containment and provided an adequate means of protecting safety-related equipment from the identified flood hazards in containment.

In the DCD, the applicant identified the safety-related equipment in the auxiliary building which requires 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 that are 0.61 to 0.91 m (2 to 3 ft) thick. These structures are designed to prevent floods which may occur in one area from propagating to another. 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 NRCAs are divided into mechanical equipment areas and electrical equipment areas. The electrical equipment areas are further divided into areas housing Class 1E electrical equipment and non-Class 1E electrical equipment.

The safe-shutdown equipment located in the NRCAs is associated with the protection and safety monitoring system (instrument and control (I&C) cabinets on Level 3), the Class 1E direct current (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), two main steam isolation valve (MSIV) 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 MSIV 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 AP1000 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 30.45 cm (12 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 (with two pumps operating) approximately 946.4 liters per minute (L/min) (250 gallons per minute (gpm)), which is equivalent to the maximum flow associated with the operation of two fire hose stations.

Design of Structures, Components, Equipment, and Systems

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

In DCD Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," the applicant stated that in the event of potential flooding/wetting, one of the following criteria is applied for protecting equipment for service in such an environment:

• Equipment will be qualified for submergence due to flooding/wetting.
• Equipment will be protected from wetting due to spray.
• Equipment will be evaluated to show that failure of the equipment due to flooding/wetting is acceptable because its safety-related function is not required or has otherwise been accomplished.

In the NRCAs, 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 and feedwater (MS & FW) isolation valves and the MS & 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 NRCAs of the auxiliary building are separated by 3-hour-rated fire barriers. Portions of these fire barriers also serve as flood barriers. With the exception of the heating, ventilation, and air conditioning (HVAC) ducts that penetrate these barriers and are below the maximum flood level, none of the wall penetrations in Class 1E electrical areas will need to be watertight because they are located above the maximum flood level. The HVAC ducts that penetrate these barriers, and are below the maximum flood level, are designed to be watertight. Floor penetrations between rooms of the same division need not be watertight.

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

Based on the evaluation of the DCD information set forth above, the staff concludes that the applicant properly identified safety-related equipment and flood hazards in the NRCAs and provided adequate means of protecting safety-related equipment from the identified flood hazards in the NRCAs of the auxiliary building.

Flood sources in the RCAs include the component cooling water system (CCS), central chilled water, hot water, spent fuel pit cooling, RNS, FPS, DWS, CVS, and various tanks. Flood water that results from component failures in the RCAs is directed to the Level 1 drain collection sump via the vertical pipe chase, floor gratings, floor drains, stairwells, and elevator shafts. Little water accumulates in the RCAs at higher levels inside the building. The safe-shutdown

equipment located in the RCAs is primarily CIVs that are located on Level 2 near the containment vessel and above the maximum flood level for the area. In addition, these CIVs either close or remain closed during safe-shutdown operations, thus, they are not affected by flooding in the auxiliary building. There is no safe-shutdown equipment on Level 1. The HVAC duct penetrations in the walls in these areas are above the maximum flood levels. Therefore, safety-related systems and equipment in the RCAs of the auxiliary building are protected from the effects of flooding.

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 watertight. 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 two buildings. In addition, there are no tunnels connecting either of these buildings to any other building.

Based on the evaluation of the DCD information set forth above, the staff concludes that the applicant properly identified safety-related equipment and flood hazards in the RCAs and provided an adequate means of protecting safety-related equipment from the identified flood hazards in the RCAs of the auxiliary building.

The turbine building is subject to flooding from a variety of potential sources, including the circulating water, service water, condensate/feedwater, component cooling water, turbine building cooling water, demineralized water, and fire protection systems, as well as the deaerator storage tank. However, no safety-related equipment exists in the turbine building. The applicant performed flooding analysis and determined that the bounding flooding source for the turbine building is a break in the circulating water piping that would result in flooding of the Elevation 100'-0" floor. Flow from this break runs out of the building to the yard through a relief panel in the turbine building west wall. This limits the maximum flood level to less than 6 inches. The component cooling water and service water components on Elevation 100'-0" that provide support for the RNS are expected to remain functional following a flooding event in the turbine building because the pump motors and valve operators of the component cooling water and service water systems are above the expected flood level.

The waste water system (WWS) sump pumps located in the NRCA of the auxiliary building discharge to the turbine building drain tank. The discharge line into the drain tank is provided with a standpipe to prevent siphoning back to the auxiliary building NRCA sump.

Based on its review as described above, the staff concludes that the applicant has adequately evaluated flooding events in the turbine building and concurs with the applicant's conclusion that safety-related equipment will not be affected.

Based on the evaluation of the DCD information set forth above, the staff concludes that the applicant provided adequate features in the AP1000 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 they relate to protecting safety-related SSCs from the effects of floods.

Design of Structures, Components, Equipment, and Systems

The staff's 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 its 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 determined for the reasons set forth above that the capability of the design to protect safety-related SSCs from the effects of floods are in accordance with the following criteria:

- Position C.1 of RG 1.59 regarding the design of safety-related SSCs to withstand the worst-probable, site-related flood

- Position C.1 of RG 1.102 regarding the type of flood protection provided

Therefore, the staff concludes that the AP1000 design meets the applicable guidelines of Section 3.4.1 of the SRP. Accordingly, the staff concludes that the AP1000 design for flood protection conforms to the applicable regulations set forth in GDC 2 and is acceptable.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

The staff reviewed the AP1000 design for protecting SSCs important to safety against internally generated missiles (outside containment), in accordance with Section 3.5.1.1 of the SRP. The acceptance criteria in SRP Section 3.5.1.1 specify that acceptance is based, in part, on the staff's verification that the applicant's SSCs important to safety will be protected from internally generated missiles by location in individual, missile-proof structures or by special localized protective shields or barriers. Conformance with the acceptance criteria of the SRP forms the basis for concluding that the design of the facility will provide protection against internally generated missiles and satisfies the requirements of GDC 4, "Environmental and Dynamic Effects Design Bases," as they relate to protecting SSCs outside containment against the effects of missiles outside containment that may result from equipment failures. This review considered those missiles generated outside containment by rotating or pressurized (high-energy fluid system) equipment. Section 3.5.1.3 of this report discusses the adequacy of the facility design to protect against low-trajectory turbine missiles, including conformance to RG 1.115, "Protection Against Low-Trajectory Turbine Missiles."

In accordance with the review procedures of SRP Section 3.5.1.1, the staff considered the following in its review of missile protection:

- plant design features for protecting SSCs important to safety outside containment against internally generated missiles

- equipment design features that could reduce missile sources

- physical separation or orientation of missile sources such that the expected missile path is in a direction that is away from safety-related SSCs

- protective shielding and barriers that could confine potential internally generated missiles

- hardening of safety-related equipment and components to withstand missile impact if a missile strike cannot be reasonably avoided

The AP1000 design credits only safety-related systems to establish and maintain safe-shutdown conditions. The safety-related systems and components needed to bring the plant to safe shutdown, including the MCR and the RSW, are located inside the containment shield building and the auxiliary building. Both buildings are seismic Category I nuclear island structures having thick structural concrete walls that provide internal and external missile protection. No non-safety-related systems or components that require protection from missiles are housed in these buildings.

In DCD Tier 2, Section 3.5.1.1.2.4, "Credible Sources of Internally Generated Missiles (Outside Containment)," the applicant stated that the only credible missile sources that can affect safety-related SSCs are a few rotating components (e.g., pumps and fans) inside the auxiliary building and a few pressurized components in high-energy systems (e.g., the CVS).

The staff reviewed the credible internally generated missiles from rotating equipment, such as motor-driven pumps and fans. Protection against potential turbine-generator missiles is addressed in DCD Tier 2, Section 3.5.1.3, "Turbine Missiles," and evaluated in Section 3.5.1.3 of this report. The rotating components are not considered credible missile sources for one or more of the following reasons:

- The rotating equipment has a housing or an enclosure that would contain the fragments from a postulated failure or fracture of the rotating element.

- The rotating equipment (e.g., pumps, motors for valve operators, and mechanical handling equipment, etc.) is in use less than 2 percent of the time because of the limited risk for missile generation.

- The rotating equipment is in a compartment surrounded by structural concrete walls with no safety-related systems or components inside the compartment.

In reviewing the missiles generated by pressurized components of high-energy fluid systems, the applicant indicated that in addition to the design features, these missile sources are not considered credible for other reasons, including the following:

- The pressurized components of high-energy systems inside the auxiliary building are constructed to ASME Code, Section III, requirements.

- The high-pressure gas storage cylinders, and attached piping and valves, inside the auxiliary building are constructed to ASME Code, Section VIII (for the gas storage cylinders), and Section III (for the attached piping and valves) requirements.

Design of Structures, Components, Equipment, and Systems

- 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.

- 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, "Proceedings of the Workshop on the Impact of Hydrogen on Water Reactor Safety." 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 ASME Code, Section III, NB/NC/ND-3000 and NB/NC/ND-3500.

- The yoke attached to the valve body is not considered to be a pressure-retaining part. Bolts and nuts do not become missiles unless they break, and the stored energy in nuts, bolts, and nut/bolt combinations is not sufficient to generate a credible missile.

The staff reviewed the above-stated reasons to eliminate certain missile sources, in addition to the detailed supporting information in DCD Tier 2, Section 3.5.1.3. These missile sources either do not have sufficient energy to generate a credible missile or are protected with structures and away from the nuclear island that they cannot cause damage to SSCs. As a result of that review, the staff agrees with the applicant that the above-listed reasons are adequate to eliminate the subject missile sources.

The applicant addressed the possibility of safety-relief valves becoming internally generated missiles (outside containment). These bolted bonnet designs are constructed in accordance with ASME Code, Section III. They are prevented from becoming missiles by limiting stresses in the bonnet to body bolting material in accordance with the ASME Code, and by designing the flanges in accordance with the applicable Code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing simultaneous complete failure is not very credible. This conclusion is consistent with operating experience that demonstrates a low incidence of complete failure. The applicant stated that safety-relief valves in high-energy systems use a bolt bonnet design that will preclude missile generation.

The piping and tubing that connects 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. The welded connections essentially eliminate the instrument as a missile source because the completed joint has a greater design strength than the parent metal. Threaded connections, which could result in a missile source, are not used to connect thermowells and similar fittings to high-energy systems or components. With respect to instrumentation, such as pressure, level, or flow transmitters, the quantity of high-energy fluid in these instruments is limited (i.e., low potential energy) and will not result in missile generation.

In DCD Tier 2, Section 3.5.1.1.2.4, the applicant addressed potential gravitational missiles outside containment. 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 the movement of a heavy load or a nonseismically designed SSC during a seismic event. DCD Tier 2, Section 9.1.5, "Overhead Heavy Load Handling Systems," addresses the protection of safety-related SSCs from missiles during movement of heavy loads. Safety-related SSCs are either protected from nonseismically designed SSCs, or the interaction is evaluated. The design provides physical separation between the safety-related equipment and nonseismic SSCs to the maximum extent practical.

On the basis of its review as described above, 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 AP1000 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. The acceptance criteria for SRP Section 3.5.1.2 specify that the design of SSCs is acceptable if the integrated design affords missile protection in accordance with GDC 4, as it relates to the ability of SSCs important to safety to withstand the effects of internally generated missiles. It is acceptable to protect SSCs by locating the systems or components in individual missile-proof structures, physically separated redundant systems or components of the system, or providing special, localized protective shields or barriers.

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 they relate 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 applicant stated that credible missile sources inside containment that can adversely affect safety-related SSCs are limited to a few rotating components. The safety-related systems and components needed to bring the plant to a safe shutdown are inside the containment shield building and auxiliary building. Both buildings have thick structural concrete exterior walls that provide protection from internal missiles generated in other portions of the plant.

In DCD Tier 2, Section 3.5.1.2.1.1, "Missiles Not Considered Credible," the applicant listed the following potential sources of internally generated missiles:

- any failure of the rotating parts of the reactor coolant pump

- catastrophic failure of rotating equipment, such as pumps, fans, and compressors, leading to the generation of missiles (e.g., reactor cavity supply fans)

Design of Structures, Components, Equipment, and Systems

- 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

The applicant stated that the above-mentioned potential missile sources are not considered credible because insufficient energy exists to produce a missile, or by design, the probability of creating a missile is negligible. The applicant evaluated the potential failure of the rotating components in a shaft-seal reactor coolant pump and indicated that the mass of the structure surrounding the impeller and the nonrotating elements of the pump motor is sufficient to contain any missiles generated by the rotating parts.

In DCD Tier 2, Section 3.5.1.2.1.4, "Evaluation of Internally Generated Missiles (Inside Containment)," regarding internally-generated missiles inside containment due to the failure of other rotating components, the applicant stated that the rotating equipment in containment has been eliminated as a missile source for one or more of the following reasons:

- The rotating equipment has a housing or an enclosure that would contain the fragments of a postulated impeller failure and is not considered a credible missile source.

- Non-safety-related rotating equipment that is not separated from safety-related systems or components has a housing or an enclosure to retain fragments from postulated failure of the rotating element.

- Equipment in use less than 2 percent of the time (e.g., reactor coolant drain pumps, containment sump pumps, motors for valve operators, mechanical handling equipment and pumps) is not considered a missile source.

The failure of the reactor vessel, SG, pressurizer, CMTs, accumulators, reactor coolant pump castings, passive residual heat exchangers, and piping leading to the generation of missiles is not deemed credible. 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 is also not considered credible. The applicant does not consider these events a credible source of missile generation because the material characteristics, preservice and inservice inspections, quality control, conservative design, and prudent operation prevent the generation of missiles from these components. The applicant also determined that the non-safety-related rotating equipment inside containment is not considered to be a credible missile source. The staff reviewed the applicant's bases for eliminating the above missile sources and concluded that they were acceptable. The staff therefore agrees with the applicant's conclusions regarding the elimination of the above components as credible missile sources. DCD Tier 2, Section 3.5.1.2, "Internally Generated Missiles Inside Containment," includes additional bases for eliminating these sources.

In DCD Tier 2, Section 3.5.1.2.1.4, the applicant stated that falling objects heavy enough to generate a secondary missile are postulated as a result of the movement of a heavy load or a nonseismically designed SSC during a seismic event. These falling objects are potential gravitational missiles and may generate secondary missiles when they strike a high-energy system. The internal energy of the damaged high-energy components may generate secondary missiles. The applicant stated that striking a component with a falling object will not generate a secondary missile due to pressurization of the component because of retention features in the design of the component. Movement of heavy loads inside containment is allowed only during shutdown when most of the high-energy systems are depressurized. Non-safety-related equipment that could fall and damage safety-related equipment during an earthquake is designed as seismic Category II and is designed to preclude such failure. Design and operational procedures of the polar crane inside containment preclude dropping a heavy load, as discussed in DCD Tier 2, Section 9.1.5 and evaluated in Section 9.1.5 of this report.

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 DCD Tier 2, Section 3.5.1.2.1.4, the applicant stated that no high-pressure gas storage cylinders are located inside the containment shield building. The staff, therefore, concludes that gas storage cylinders inside containment do not present a potential missile source.

Missiles can be generated by a hydrogen explosion inside containment. Hydrogen is supplied by the CVS. In DCD Tier 2, Section 3.5.1.2.1.2, "Explosions," the applicant stated that the quantity of hydrogen 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 staff concludes that the amount of hydrogen that could be released to the containment would not lead to an explosion.

On the basis of its review as described above, the staff concludes that the AP1000 design meets the guidelines of Section 3.5.1.2 of the SRP. Therefore, the staff concludes that the AP1000 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

GDC 4 requires that SSCs important to safety be protected against the effects of missiles that might result from equipment failures. The steam turbine is considered to be a component important to safety because if its massive rotor fails at a high rotating speed during normal operating conditions of a nuclear unit, it could generate high-energy missiles that have the potential to damage safety-related SSCs.

RG 1.115 and SRP Section 10.2, "Turbine Generator," SRP Section 10.2.3, "Turbine Disk Integrity," and SRP Section 3.5.1.3, "Turbine Missiles," guide the evaluation of the effect of turbine missiles on public health and safety. As specified in SRP Section 3.5.1.3, the probability of unacceptable damage from turbine missiles is expressed as the product of (1) the probability of turbine missile generation resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing, (P_1); (2) the probability of ejected missiles perforating intervening barriers and striking safety-related SSCs, (P_2); and (3) the probability of

Design of Structures, Components, Equipment, and Systems

impacted SSCs failing to perform their safety functions, (P_3). In view of the operating experience of turbines and the NRC safety objectives, the NRC staff shifted its emphasis in the review of turbine missile issues from missile generation, strike, and damage probability, $P_1 x P_2 x P_3$, to the missile generation probability, P_1. The minimum reliability values (i.e., P_1) for loading the turbine and bringing the system on line were established in 1986. These minimum recommended reliability values are P_1 less than 10^{-4} per reactor-year for favorably oriented turbines, and P_1 less than 10^{-5} per reactor-year for unfavorably oriented turbines. These values are derived from (1) simple estimates for a variety of plant layouts which show that $P_2 x P_3$ can be reasonably taken to fall within the range of 10^{-4} to 10^{-3} per year for favorably oriented turbines, and (2) the NRC criterion of 10^{-7} per year for $P_1 x P_2 x P_3$, as stated in RG 1.115. The P_1 calculation is related to maintenance and inspection of turbine rotors and valves, operating experience of similar equipment, and inspection results.

3.5.1.3.1 Summary of Technical Information

The applicant relied on the turbine missile methodology and analytical results documented in WCAP-15783, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines," and WCAP-15785, "Probabilistic Evaluation of Turbine Valve Test Frequency," to demonstrate that its evaluation of the AP1000 full integral nuclear low-pressure turbines meets the NRC guidelines provided in RG 1.115 and SRP Sections 10.2, 10.2.3, and 3.5.1.3, and that the missile generation probability, P_1, does not exceed the NRC criterion of 10^{-4} for favorably oriented turbines.

WCAP-15783 assesses the potential for rotor bursting by evaluating four failure mechanisms, (1) ductile burst from destructive overspeed, (2) fracture from high cycle fatigue (HCF), (3) fracture from low cycle fatigue (LCF) cracking, and (4) fracture from stress-corrosion cracking (SCC). Except for the ductile burst from destructive overspeed, which only includes the deterministic analysis, WCAP-15783 provides complete analyses for HCF, LCF, and SCC. The HCF analysis shows that adequate safety factors exist for initiation and propagation of a turbine disk crack. Further, the turbine missile analysis considering LCF along shows that the probability of missile generation is several orders-of-magnitude lower than the NRC criterion. Hence, SCC is the dominant mechanism for determining the probability for missile generation. In addition, the analyses show that the probability of turbine missile generation does not exceed 10^{-5} per reactor-year, even after a running time between inspections of several times longer than 10 years.

Using detailed nuclear turbine failure data, WCAP-15785 assesses the total risk of turbine missile ejection at destructive overspeed and at lower overspeeds as a function of valve test interval. The evaluation shows that the probability of turbine missile generation, with quarterly valve tests, is less than the NRC evaluation criteria.

3.5.1.3.2 Staff Evaluation

The staff utilized the guidelines of SRP Section 3.5.1.3 to review and evaluate the information submitted by the applicant to ensure a low probability of turbine rotor failure. The evaluation of P_1 relies, in part, on the evaluation of the materials, inspection, and maintenance of the turbine rotors discussed in Section 10.2.8, "Turbine Rotor Integrity," of this report. With the use of proper turbine rotor design, proper materials (i.e., those properly heat treated and tested to

determine that material properties meet specified criteria), and meaningful preservice and inservice non-destructive examination (NDE) methods and acceptance criteria, the probability of turbine missile generation, P_1, is expected to have an acceptable value. The probability of turbine missile generation should be kept to the values stated above (i.e., no greater than 10^{-5} per reactor-year for an unfavorably oriented turbine, and no greater than 10^{-4} for a favorably oriented turbine).

The AP1000 will utilize a favorable turbine generator placement and orientation, and the applicant is committed to meet RG 1.115, which should ensure an acceptably low probability of unacceptable damage to safety-related SCCs. DCD Tier 2, Section 3.5.1.3, discusses turbine missiles in general terms, with detailed information provided in DCD Tier 2, Section 10.2.3. The technical elements discussed below also apply to the safety evaluation of DCD Tier 2, Section 10.2.3, which is provided in Section 10.2.8 of this report.

The methodology and analytical results of the probability of turbine missile generation are contained in the applicant's submittals, including WCAP-15783 and WCAP-15785. The NRC staff requested information in RAI 251.001 about the modifications made to the current turbine missile methodology from methodologies previously approved by the staff. In its response to RAI 251.001, the applicant did not directly provide the information requested regarding changes from previously approved methodologies. However, the staff performed a detailed review to identify and evaluate the modifications made to the previously approved turbine missile methodologies; therefore, RAI 251.001 is considered to be closed. Staff evaluation of these modifications is discussed below.

WCAP-15783 evaluated four potential failure mechanisms, (1) ductile burst from destructive overspeed; (2) fracture from HCF cracking; (3) fracture from LCF cracking; and (4) fracture from SCC. WCAP-15783 concludes that ductile burst will not occur before destructive overspeed is reached (the probability of reaching destructive overspeed is discussed in WCAP-15785). Also, the applicant concluded that the effect due to HCF cracking and LCF cracking can be ignored because of their extremely low probabilities of generating turbine missiles.

Notwithstanding the applicant's probability argument, the staff reviewed this information and determined that the applicant's evaluation methodology and results for the ductile failure from overspeed and HCF cracking are consistent with approved methodologies, and is, therefore, acceptable. The evaluation methodology for fracture from LCF cracking is similar to that previously reviewed in approved methodologies. The NRC staff evaluated the two parameters, C_o and n, in the Paris fatigue crack growth rate equation, $da/dN = C_o(\Delta K)^n$, which the applicant used in the LCF analysis. The staff found these parameters acceptable because they were derived from applicable test data and actual plant data. Although the data set is limited, it is acceptable to the staff because the evaluation used a very conservative fracture toughness for the disk material. The NRC staff also examined the failure equation and determined that it is based on fracture mechanics using the acceptable Paris fatigue crack growth rate discussed above. Further, except for the maximum undetectable crack size, the values of all remaining deterministic parameters, such as flaw shape factor, critical crack depth, and cyclic stress range, are conservative because (1) the flaw shape factor corresponds to a more conservatively assumed flaw shape than industry data reveals, (2) the critical crack depth corresponds to a very conservative K_{IC} value, and (3) the cyclic stress range corresponds to

Design of Structures, Components, Equipment, and Systems

stresses at running speeds and design overspeeds of 120 percent, which are the peak stresses during a startup cycle. Therefore, these values are acceptable to the NRC staff.

The NRC staff requested in RAI 251.002 that the applicant justify the use of the specified value for the maximum undetectable crack size. In its response to RAI 251.002, the applicant provided a length of 1.5 millimeters (mm) and an aspect ratio of 4 to 1 for the maximum undetected flaw size, but did not provide a basis for the maximum undetected flaw size. This was Open Item 3.5.1.3-1 in the DSER.

By letter dated April 4, 2003, the applicant provided Revision 1 to its response to RAI 251.001. The applicant indicated that the ultrasonic testing (UT) device used to inspect the turbine rotors is calibrated by test specimens of the same material with a 1.6 mm (0.063 in.) diameter hole. Based on this calibration, UT performed on the rotor outer surface and center bore is able to distinguish an artificial indication length of 1.5 mm (0.059 in.) from the reflected pulse. Magnetic particle testing (MT) is performed on the rotor outer surface and the center bore. Considering the sensitivity of MT in detecting flaws on the order of 0.5 mm (0.02 in.) in the major dimension, as reported by Alex Vary's survey paper, "Nondestructive Evaluation Technique Guidebook," the staff concludes that the combination of MT and UT will detect a flaw of 1.5 mm (0.059 in.) in length. Regarding the information in the applicant's response on the aspect ratio of the maximum undetectable crack size, the staff concludes that the aspect ratio for the assumed flaw is consistent with industry service data and is, therefore, appropriate. Therefore, Open Item 3.5.1.3-1 is resolved.

Regarding failure by SCC, the applicant responded to a staff question in RAI 251.002 on the interaction of the LCF and SCC failure mechanisms. The response indicated that crack initiation begins at different locations for SCC and LCF and the interaction between them is not considered in the analysis. The staff finds it acceptable that the applicant did not consider this interaction in the analysis. However, the applicant's response to RAI 251.002 regarding the SCC growth rate is not satisfactory. The SCC growth rate reported in WCAP-15783 is based on a statistical analysis of 12 data points. Since the data set is considerably smaller than the data set relied upon in past analyses, the applicant needed to expand the current database by including available data on the same material from other sources.

Further, the staff was concerned that specified values or units for the coefficients in the SCC growth equation may not be correct. Using an SCC growth rate of 4.5×10^{-4} mm/h (1.77×10^{-5} in./h) from Table 4-5 of WCAP-15783, and the specified values for the coefficients (a, b, and c) for the SCC growth calculation, would give a negative value for the yield strength, which is not realistic. This was Open Item 3.5.1.3-2 in the DSER.

In its letter of July 3, 2003, the applicant provided a response to this open item by (1) clarifying the nature of the 12 data points upon which the SCC growth rate is based, and (2) revising Section 4.4.2 of WCAP-15783 by correcting the units for some key parameters in the SCC growth rate equation. The new information indicates that these 12 "data points" should be considered "data sources." Each source was established using approximately 30 test specimens. Hence, the database is actually 30 times larger than originally believed and is, therefore, acceptable.

As to the apparent mistake in the units for some key parameters in the SCC growth rate equation which gave a negative value for the yield strength, Westinghouse provided WCAP-15783, Revision 2, with appropriate corrections to these units. The staff verified that using the parameters with the revised units would produce a reasonable SCC growth rate. Based on the above discussion, Open Item 3.5.1.3-2 is resolved.

WCAP-15785 assesses the total risk of turbine missile ejection at destructive overspeed and at lower overspeeds as a function of valve test interval. Section 10.2.8 of this report provides the staff's evaluation and acceptance of WCAP-15785.

3.5.1.3.3 Conclusions

The applicant performed analyses for the determination of P_1 using the distribution of crack growth rates and critical crack sizes and reported the results in WCAP-15783, to demonstrate the probability of a rotor bursting at the design overspeed as a function of the inspection interval. Section 10.2.8 of this report evaluates the adequacy of the inservice inspection (ISI) and valve test intervals. The results indicate that the probability of missile generation is less than 10^{-5} per year for an inspection interval several times longer than 10 years, although the actual turbine ISI intervals discussed in Section 10.2.8 of this report are 10 years or less. The staff concludes that the risk posed by turbine missiles for the proposed plant design is acceptable and meets the relevant requirements of GDC 4. This conclusion is based on the applicant having sufficiently demonstrated to the staff, in accordance with the guidance of RG 1.115, that the overall probability of turbine missile damage to SSCs 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 AP1000 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 RG 1.76, 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 missiles generated by natural phenomena that are of concern are those resulting from tornados. The tornado missile spectrum used by the applicant is Spectrum I, as identified in SRP Section 3.5.1.4. The utility requirements document (URD) of the Electric Power Research Institute (EPRI) for the ALWR passive plant requires that the selection of a tornado missile spectrum be in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS) 2.3, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites," and that it meets the intent of current SRP criteria.

In DCD Tier 2, Section 3.3.2.1, the applicant provides the following design parameters for the design-basis tornado:

Design of Structures, Components, Equipment, and Systems

- a maximum wind speed of 483 km/hr (300 mph)
- a maximum rotational speed of 386 km/hr (240 mph)
- a maximum translational speed of 97 km/hr (60 mph)
- a radius of maximum rotational wind from center of tornado of 46 m (150 ft)
- an atmospheric pressure drop of 14 kPa (2 psia)
- a rate of pressure change of 8 kPa/sec (1.2 psia/sec)

The applicant selected these design parameters 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." The applicant 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 conformance with RG 1.117, is discussed separately in Section 3.5.2 of this report. Section 3.5.3 of this report provides an evaluation of the design of missile barriers and protective structures to withstand the effects of the identified tornado missiles.

On the basis of its review, and for the reasons set forth above, the staff concludes that the AP1000 design meets the requirements of GDC 2 and 4 with respect to protection against natural phenomena and missiles. The design also 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, so long as the reference site meets the guidelines in Section 3.5.1.4 of the SRP.

3.5.1.5 Missiles Generated by an Event near the Site

In DCD Tier 2, Section 3.5.1.5, "Missiles Generated by Events Near the Site," the applicant stated that the site interface is established to address site-specific missiles in the COL application. The AP1000 missile interface criteria are based on the tornado missiles described in DCD Tier 2, Section 3.5.1.4. Additional analyses are needed to evaluate other site-specific missiles. Each COL applicant referencing the AP1000 will provide analyses of accidents external to the nuclear plant. This is COL Action Item 3.5.1.5-1.

The determination of the probability of occurrence of potential accidents that have severe consequences is based on the analyses of available statistical data on the occurrence of an accident involving 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 AP1000 plant will be identified in the COL application. In DCD Tier 2, Section 2.2, "Nearby Industrial, Transportation, and Military Facilities," the applicant specified 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, including explosions, flammable vapor clouds, toxic chemicals, fires, and airplane crashes. Based on the SRM dated June 26, 1990, responding to SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," 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, the staff finds this to be acceptable.

3.5.1.6 Aircraft Hazards

In DCD Tier 2, Section 3.5.1.6, the applicant established the site interface to address aircraft hazards in the COL application. The AP1000 missile interface criteria are based on the tornado missiles described in DCD Tier 2, Section 3.5.1.4. Additional analyses are needed to evaluate other site-specific missiles. Each COL applicant referencing the AP1000 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 the analyses of available statistical data on the occurrence of an accident involving the plant's safety-related structures and components. This is COL Action Item 3.5.1.5-1.

SRP Section 3.5.1.6 of NUREG-0800 describes acceptable methods of evaluating site-specific aircraft hazards. If an accident is identified for which the probability of exceeding 10 CFR Part 100 dose guidelines is unacceptable, specific changes to the AP1000 will be identified in the COL application. In DCD Tier 2, Section 2.2, the applicant specified the threshold of the total annual frequency of occurrence as 10^{-6} per year for all external event-induced accidents leading to severe consequences, including airplane crashes leading to missile impact or fire in the vicinity of the plant. This conforms to the acceptance criteria described in SRP Section 2.2.3, "Evaluation of Potential Accidents," wherein the rate of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines is estimated not to exceed the NRC staff objective of approximately 10^{-7} per year. The SRP acceptance criteria states that 10^{-6} is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower. Since the 10^{-7} criterion is for each postulated type of accident or event, and with an expected total frequency of 10^{-6} per year for all external events, the staff concluded that the realistic probability for each type of accident was lower.

In addition, in the SRM dated June 26, 1990, 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. On the basis of the above, the staff finds the applicant's approach to addressing site-specific aircraft hazards to be acceptable.

3.5.2 Protection From Externally Generated Missiles

The staff reviewed the AP1000 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 this acceptance is based on meeting the guidelines of RG 1.13, "Spent Fuel Facility Design Basis"; RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants" (concerning tornado missile protection for safety-related SSCs, including stored spent fuel and the ultimate heat sink); RG 1.115 (with respect to protection against turbine missiles); and RG 1.117 (with respect to the protection of SSCs important to safety from the effects of tornado missiles). Section 3.5.1.4 of this report discusses the tornado missile spectrum for the AP1000 design. As set forth in that section, the design is based on an acceptable tornado missile spectrum. The staff's review of externally generated missiles does not include turbine missiles; Section 3.5.1.3 of this report evaluates these types of missiles.

Design of Structures, Components, Equipment, and Systems

In Section 3.5.2 of the SRP, the staff states that the SSCs required for safe shutdown of the reactor should be identified. The identification of SSCs to be protected from externally generated missiles is acceptable, if it is in accordance with the requirements of GDC 2 and 4. These SSCs are identified in DCD Tier 2, Section 7.4. The structural design requirements for the shield building and auxiliary building are outlined in DCD Tier 2, Section 3.8.4. Openings 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. As set forth in DCD Tier 2, Section 3.5.2, "Protection from Externally Generated Missiles," the COL applicant will evaluate site-specific hazards for external events that may produce missiles more energetic than tornado missiles. This is COL Action Item 3.5.1.5-1.

The spent fuel pool meets Regulatory Position C.2 of RG 1.13 because it is protected from externally generated missiles by the reinforced concrete walls and roof of the auxiliary building. Therefore, the AP1000 conforms with the guidelines of RG 1.13 with respect to protection of spent fuel from externally generated missiles.

Positions C.2 and C.3 of RG 1.27 address the use of water sources as heat sinks. The AP1000 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. The shield building, which has reinforced concrete walls 0.61 m (2 ft) thick, provides protection from externally generated missiles. Therefore, the applicant has met the guidance of RG 1.27 with regard to protection of the ultimate heat sink from externally generated missiles.

In DCD Tier 2, Appendix 1A, "Conformance with Regulatory Guides," the applicant indicated that the AP1000 design conforms to Positions C.1, C.2, and C.3 of RG 1.117 with respect to the protection of SSCs important to safety from the effects of tornado missiles. The staff concurs with the applicant in this regard because all the SSCs of the nuclear island that are important to safety, including those necessary for maintaining the integrity of the reactor coolant pressure boundary, and for achieving and maintaining safe plant shutdown, are located within structures having walls that are 61 cm (24 in.) thick and roofs that are 38 cm (15 in.) thick. These features will offer protection from tornado-generated missiles. Section 3.8 of this report provides further information regarding these structures.

On the basis of its review, and for the reasons set forth above, the staff concludes that the AP1000 design for protecting SSCs against externally generated missiles is in accordance with the guidelines of RGs 1.13, 1.27, and 1.117 with respect to the protection of SSCs important to safety from the effects of tornado missiles, including stored spent fuel and the ultimate heat sink. Therefore, the staff concludes that the AP1000 design complies with the requirements of GDC 2 and 4 with respect to missile and environmental effects. As discussed in Section 3.5.1.3 of this report, the design is also in accordance with the guidelines of RG 1.115 and the requirements of GDC 4 with respect to protection from low-trajectory turbine missiles. Therefore, the staff concludes that the AP1000 design meets the guidelines of Section 3.5.2 of the SRP and the requirements of GDC 2 and 4 for providing protection from externally generated missiles.

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 and to satisfy the requirements of GDC 2 and 4 with respect to the capability of structures to withstand the effects of missiles and to provide protection against their dynamic effects. The staff reviewed the design of seismic Category I SSCs, using the guidance of Section 3.5.3 of the SRP, to determine if they are shielded from, or designed to withstand, various postulated missiles. DCD Tier 2, Section 3.5.3, "Barrier Design Procedures," 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 include both local damage and overall damage.

For the prediction of local damage from missiles, the applicant provided information on the procedures used in the design of concrete and steel structures. The applicant applied the modified National Defense Research Committee (NDRC) formula, as shown in DCD Tier 2, Section 3.5.3, analytically for missile protection in concrete. To prevent missile perforation, the applicant used the minimum thickness needed 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 in Table 1 of SRP Section 3.5.3, will result in sufficient concrete barrier thickness to prevent barrier perforation and, when necessary, prevent spalling or scabbing. For missile penetration in steel, the applicant used either the Ballistic Research Laboratory (BRL) or Stanford formulae for missile perforation in steel. As discussed in Section 3.5.3 of the SRP, the staff finds the use of either formula to be acceptable.

Appendix A to SRP Section 3.5.3 states that, in the evaluation of overall responses of reinforced concrete and steel structural elements (i.e., missile barriers, columns, slabs) subjected to impactive or impulsive loads (e.g., impacts due to missiles), assumption of a nonlinear response (i.e., ductility ratios greater than unity) of the structural elements is generally acceptable, provided that the intended safety functions of the structural elements and those of the safety-related systems and components supported or protected by the elements are maintained.

For the prediction of overall damage, the applicant stated, in DCD Tier 2, Section 3.5.3, that structural members required 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 conform to applicable codes and RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)." The staff finds the use of RG 1.142 for concrete to determine the overall damage prediction to be acceptable. The applicant also provided, in DCD Tier 2, Section 3.5.3.1, "Ductility Factors for Steel Structures," the limits on ductility of steel structures. These ductility limits meet the guidance of Appendix A to Section 3.5.3 of the SRP and, therefore, are acceptable.

The staff also finds that the procedures used for determining the effects and loadings on seismic Category I structures, as well as missile shields and barriers induced by design-basis missiles selected for the plant, are acceptable because they provide a conservative basis for

Design of Structures, Components, Equipment, and Systems

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 a seismic Category I structure 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, the staff finds that the applicant used acceptable procedures in its barrier design. The staff also finds that the barrier design procedures meet the guidelines of Section 3.5.3 of the SRP, as well as GDC 2 and 4, with respect to the capabilities of the structures, shields, and barriers to provide sufficient protection to the safety-related SSCs. These procedures are, therefore, acceptable.

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 AP1000 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. Satisfaction of the SRP acceptance criteria ensures that the design meets the requirements of GDC 4, as it relates to accommodating the dynamic effects of postulated pipe rupture, including the effects of pipe whipping and discharging fluids. The AP1000 design is in compliance with GDC 4 if it conforms with Branch Technical Position (BTP) Plant System Branch (SPLB) 3-1 (formerly Auxiliary Systems Branch (ASB) 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 specified 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 DCD Tier 2, Section 3.6.1, the applicant provided the design basis and criteria for the analysis needed to demonstrate that safety-related systems are protected from pipe ruptures. This DCD section enumerates the high- and moderate-energy systems which are potential sources of the dynamic effects associated with pipe ruptures. It also defines separation criteria. In DCD Tier 2, Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," the applicant provided criteria for postulated

pipe rupture location and configuration. Non-safety-related systems for the AP1000 plant are not required to be protected from the dynamic and environmental effects associated with pipe rupture.

By design, nonseismic piping is not routed near safety-related piping or equipment. If there is nonseismic, 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 relied upon for safe shutdown are not expected to be adversely affected by the dynamic effects of postulated pipe ruptures in nonseismic, moderate-energy piping.

Evaluations of dynamic effects for postulated pipe breaks which meet the mechanistic pipe break or leak-before-break (LBB) criteria are eliminated from pipe failure analysis for the AP1000 design. Such evaluations include the reactor coolant loop branch piping, the main steam (MS) piping within containment, and other primary piping inside containment which is equal to or greater than 15.2 cm (6 in.) nominal pipe size (NPS). The piping systems that meet the LBB criteria are not subject to the dynamic effects associated with a pipe failure.

The AP1000 design, as it relates to the mechanistic pipe break (or LBB), is addressed in DCD Tier 2, Section 3.6.3, "Leak-Before-Break Evaluation Procedures." High-energy fluid system piping that meets the LBB criteria is evaluated for the effects of leakage cracks. High-energy fluid system piping that does not meet the LBB criteria is 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. Section 3.6.3 of this report provides the details of the staff's evaluation and conclusions regarding the acceptability of the applicant's LBB analysis.

In DCD Tier 2, Section 3.6.1, the applicant identified those safety-related systems that 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 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 DCD Tier 2, Section 3.6.1, the applicant also provided 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. High-energy systems are defined as those systems or portions of systems containing fluid with a maximum normal operating temperature exceeding 93.3 °C (200 °F) and/or a maximum normal operating pressure exceeding 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. In DCD Tier 2, Table 3.6-1, the applicant identified the high- and moderate-energy fluid systems in the AP1000 design based

Design of Structures, Components, Equipment, and Systems

on the above definitions. These definitions and identified fluid systems are acceptable because they are in accordance with BTP SPLB 3-1.

In accordance with the appropriate criteria, pipe failure evaluations are made based on circumferential or longitudinal pipe breaks, through-wall cracks, or leakage cracks. Through-wall cracks are postulated in both high- and moderate-energy piping, and are assumed to be a circular opening with an area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width, as specified in BTP EMEB 3-1. A leakage crack is the crack size that results in leakages that are assumed in the LBB analysis. 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 main feedwater (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 DCD Tier 2, Sections 3.6.2 and 3.6.3.

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

- Offsite power is not needed for actuation of the passive safety systems. Only the Class 1E dc and UPS electrical systems need 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 an SACF).

- A whipping pipe can break pipes of smaller diameter, regardless of pipe-wall thickness, and can cause a through-wall crack in a 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 are assumed not to 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; and safety-related instrumentation, electric power supplies, components, and controls from performing their safety functions.

In DCD Tier 2, Section 3.6.1, the applicant stated 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, as well as 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 7.62 cm (3 in.) in diameter, the pressurization analysis and the evaluation of venting provisions are based on a 7.62 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 an 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 MSIV subcompartment (part of the 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 MSIV subcompartment is evaluated for the effects of flooding, spray wetting, and pressurization resulting from a break of 0.093 m^2 (1 ft^2) in the MS or MFW line. The subcompartment wall closest to the MCR is also evaluated for the jet impingement resulting from a longitudinal break of 0.093 m^2 (1 ft^2) 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 RSW is not subject to adverse effects from high-energy pipe breaks.

In DCD Tier 2, Section 3.6.1, the applicant provided the measures used in the AP1000 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.

Design of Structures, Components, Equipment, and Systems

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

Based on its review as described above, the staff concludes that the AP1000 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 and the guidelines of SRP Section 3.6.1, including BTP SPLB 3-1, with respect to accommodating the effects of postulated pipe failures. Therefore, the staff finds this aspect of the design to be 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 accommodate the effects of, the environmental conditions resulting from postulated accidents, including LOCAs. GDC 4 also requires that SSCs 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 these GDC 4 requirements, the applicant described the following items in DCD Tier 2, Section 3.6.2:

- 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 DCD Tier 2, Section 3.6.2, and, as discussed below 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 EMEB 3-1, satisfies GDC 4 with respect to dynamic effects, and, therefore, is acceptable.

In one of the guidelines in BTP EMEB 3-1, the staff states that the analyses of the maximum stresses, stress ranges, and usage factors used to determine postulated high- and moderate-energy pipe break and crack locations should be based on loads that include the operating-basis earthquake (OBE). In SECY-93-087, the staff recommended the elimination of the OBE in the design process because 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 applicant incorporated revised high-energy break criteria without

OBE consideration into DCD Tier 2, Sections 3.6.1 and 3.6.2. Therefore, the staff's evaluation of these sections of the DCD is based on the Commission-approved staff recommendations.

3.6.2.1 High- and Moderate-Energy Piping Systems

As discussed in Section 3.6.1 of this report, the staff finds that the criteria in DCD Tier 2, Section 3.6.1.1, "Design Basis," Item A, and DCD Tier 2, Appendix 3E, "High-Energy Piping in the Nuclear Island," regarding the definition of high- and moderate-energy piping systems are consistent with the criteria in Appendix A to BTP SPLB 3-1 in SRP Section 3.6.1, and, therefore, are acceptable. The staff also finds that the applicant 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 1 percent of the plant operation time, are considered moderate-energy systems. The staff notes that the 1 percent or less of plant operating time criterion does not appear to be completely consistent with the BTP EMEB 3-1 criteria in Section 3.6.2 of the SRP for fluid systems that qualify as moderate-energy systems on the basis that they operate as high-energy fluid systems for only a short operational period, but as moderate-energy fluid systems for the major portion of the operational period. In Footnote 5 to BTP EMEB 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 2 percent of the time that the system operates as a moderate-energy fluid system.

The staff concluded that no matter which definition of short operational period is used (1 or 2 percent), the resulting time from either definition is short enough that the likelihood of a break occurring during either period is small. On this basis, the staff concludes that the definitions of high- and moderate-energy systems are consistent with that of the SRP and BTP EMEB 3-1, as well as the definition for a short operational period in DCD Tier 2, Section 3.6.1.1, Item A, and, therefore, are acceptable for the AP1000.

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

In SRP Section 3.6.2, BTP EMEB 3-1, the staff states that breaks need not be postulated in those portions of the high-energy fluid system piping located in the containment penetration area both inside and outside the containment, provided they are designed to meet Article NE-1120 of Section III of the ASME Code and the additional guidelines specified in BTP EMEB 3-1. The staff evaluated the information in DCD Tier 2, Section 3.6.2, to determine if the applicant provided acceptable commitments to these guidelines in the AP1000 design. In DCD Tier 2, Section 3.6.2.1.1.4, "High Energy Piping in Containment Penetration Areas," the applicant identified those portions of AP1000 piping systems that qualify for break exclusion. In DCD Tier 2, Section 3.6.2.1.1.4, the applicant also provided the bases for these break exclusion areas which meet the guidelines in SRP Section 3.6.2, BTP EMEB 3-1, and are consistent with staff-approved break exclusion areas in several of the most recently licensed PWRs. Therefore, these break exclusion areas are acceptable. One exception to this list of break exclusion areas is due to the staff's position of postulating breaks in the MSIV compartment, which is discussed below. The staff's evaluation of several issues related to the break exclusion areas are also discussed below.

One important guideline is the implementation of an augmented ISI program for those portions of piping within the break exclusion zone. In DCD Tier 2, Section 3.6.2.1.1.4, the applicant

Design of Structures, Components, Equipment, and Systems

provided a commitment to such a program for all piping in the break exclusion zone that is 7.62 cm (3 in.) in diameter or larger. This commitment meets the applicable guidelines of SRP Section 3.6.2 and, therefore, is acceptable.

In the AP1000 design, the east wall of the east MSIV compartment, which houses the MS & 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 & FW piping are in a break exclusion area, the staff states, in Section B.1a(1) of SRP Section 3.6.1, BTP SPLB 3-1, that essential equipment in this area must be protected from the environmental effects of a postulated break which has a cross-sectional area of at least 0.093 m^2 (1 ft^2). The staff also cautions designers to avoid concentrating essential equipment in the break exclusion zone. In addition, in Section B.1.a(2) of BTP SPLB 3-1, the staff states that the MS & FW lines should not be routed around or in the vicinity of the control room. In DCD Tier 2, Sections 3.6.1 and 3.6.1.2.2, "Main Control Room Habitability," the applicant stated this compartment is evaluated for the effects of flooding, spray wetting, and subcompartment pressurization from a 0.093 m^2 (1 ft^2) break from either the MS or FW line within the respective break exclusion areas. 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 SPLB 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 has neither MS & FW lines routed in the vicinity of the MCR, nor safety-related equipment nearby. Therefore, DCD Tier 2, Sections 3.6.1 and 3.6.1.2.2 state that the wall between the MSIV compartment and the MCR, and the floor slab between the MSIV compartment and the safety-related electrical equipment room, are also evaluated for pipe whip and jet impingement loads for worst-case breaks in either the MS line or the main FW line. The staff finds that this approach meets the guidelines of SRP Section 3.6.2 and, therefore, is acceptable.

Another important guideline given in SRP Section 3.6.2, BTP EMEB 3-1, is that the number of circumferential and longitudinal piping welds and branch connections should be minimized. Where guard pipes are used in high-energy piping in containment penetration areas, the enclosed portion of the fluid system piping should be seamless and without 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 ISI 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. The applicant addresses this issue in DCD Tier 2, Section 3.6.2.1.1.4, which states that there are no circumferential or longitudinal welds in the piping enclosed within the guard pipe, thereby obviating the need for augmented ISI in this area. This is consistent with Section 3.6.2 of the SRP, and, therefore, is acceptable.

SRP Section 3.6.2, BTP EMEB 3-1, also provides additional design and test guidance for guard pipes which are part of the containment boundary and are designed in accordance with the rules of Class MC, Subsection NE of the ASME Code, Section III. In DCD Tier 2, Section 3.6.2.1.1.4, the applicant committed to all of these SRP Section 3.6.2 criteria. In DCD Tier 2, Sections 3.6.2.4 and 3.6.2.4.2, the applicant discussed and clarified the difference between guard pipes in break exclusion zones and auxiliary guard pipes. Guard pipes in the break exclusion zones provide additional confidence that pipes will not leak into the annulus between the containment vessel and the shield building. These guard pipes are designed to the criteria found in DCD Tier 2, Section 3.6.2.1.1.4. Other auxiliary guard pipes are 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 therefore acceptable.

The staff concludes that, on the basis of the above discussion, the criteria in DCD Tier 2, Section 3.6.2.1.1.4, are consistent with SRP Section 3.6.2, BTP EMEB 3-1 and meet GDC 4 with respect to dynamic effects. The staff also notes that the criteria in the DCD are consistent with approved break exclusion areas in the AP600 design and other recently licensed PWRs. Therefore, the AP1000 pipe break exclusion areas identified in the DCD are acceptable.

3.6.2.3 Pipe Rupture Criteria Outside the Containment Penetration Area

Section 3.6.2 of the SRP, including BTP EMEB 3-1, provides the staff guidance for the review of information on the determination of pipe rupture locations and the dynamic effects associated with the postulated rupture of piping.

Section B.1.c of BTP EMEB 3-1 delineates the staff's position for postulating pipe breaks in high-energy piping systems outside the containment penetration area. For moderate-energy piping systems outside the containment penetration area, BTP EMEB 3-1, Section B.2.c, provides the staff guidance for postulating leakage cracks based on analytically calculated piping values and other factors. Section B.3 of EMEB 3-1 provides staff guidance on defining the types (circumferential or longitudinal) and configurations of breaks and leakage cracks. In DCD Tier 2, Section 3.6.2.1, the applicant provided the AP1000 criteria for defining the high- and moderate-energy break and crack locations, types, and configurations for ASME Class 1, 2, and 3 piping, as well as for seismically and nonseismically analyzed ASME B31.1 piping. The staff reviewed this information and determined that the AP1000 criteria are identical to the staff-approved criteria for the AP600, consistent with the staff position described in BTP EMEB 3-1, and are therefore acceptable.

Section 3.6.2.III.3 of the SRP provides staff-acceptable procedures and assumptions for defining the jet impingement loadings on nearby safety-related SSCs due to a postulated pipe break. Section 3.6.2.III.2 of the SRP provides acceptable dynamic analysis criteria, modeling and analysis methods, and assumptions for calculating piping and restraint system responses due to the jet thrust that develops after a postulated rupture, including the definition of the time-dependent jet thrust force. In DCD Tier 2, Sections 3.6.2.2, "Analytical Methods to Define Jet Thrust Forcing Functions and Response Models"; 3.6.2.3, "Dynamic Analysis Methods to Verify Integrity and Operability"; and 3.6.2.4, "Protective Assembly Design Criteria," the applicant described its analytical methods for defining jet thrust forces and jet impingement forces, evaluating dynamic response, and analyzing and designing pipe whip restraints and other protective devices. The staff reviewed this information and determined that the AP1000 criteria and analysis methods are consistent with the staff position described in SRP Section 3.6.2, and are therefore acceptable.

In DCD Tier 2, Section 3.6.2.5, "Evaluation of Dynamic Effects of Pipe Rupture," the applicant provided a description of the pipe break hazards analysis activities, including the following:

- preparation of a stress summary
- identification of pipe break locations in high-energy piping
- identification of through-wall crack locations in high- and moderate-energy piping

Design of Structures, Components, Equipment, and Systems

- identification and location of essential structures, systems, and components
- evaluation of the consequences of pipe whip and jet impingement
- evaluation of the consequences of flooding, environment, and compartment pressurization
- design and location of protective hardware
- reconciliation of as-built condition

In DCD Tier 2, Section 3.6.2.5, the applicant stated that to support design certification, the pipe rupture hazard analysis is complete except for the final piping stress analyses, pipe whip restraint design, and the as-built reconciliation. In this respect, Westinghouse is not relying on the use of piping design acceptance criteria (DAC) for its high- and moderate-energy pipe break design. Instead, Westinghouse has relied on the completeness of its AP600 piping design to establish preliminary pipe break locations and their consequences for the AP1000 standard plants. The staff reviewed the comparison between the AP600 and AP1000 pipe break locations and consequences and its evaluation is discussed in conjunction with RAI 210.035 later in this report. The final piping stress analyses, pipe whip restraint design, and the as-built reconciliation will be addressed by the COL applicant. The as-built reconciliation includes evaluation of the ASME Code fatigue analysis, pipe break dynamic loads, the reconciliation to the certified design floor response spectra (FRS), confirmation of the reactor coolant loop-time history seismic analysis, changes in support locations, and the construction deviations. This is COL Action Item 3.6.2.3-1.

In RAI 210.036, the staff requested the applicant to provide a clear definition for "LBB criteria" and "LBB evaluation report," and discuss how bounding curves, as described in DCD Tier 2, Appendix 3B, "Leak-Before-Break Evaluation of the AP1000 Piping," would be considered by piping analysts in the design stage without completing the piping analysis or the LBB demonstration evaluation to ensure conformance with LBB criteria. This issue was discussed with the NRC staff during the September 9–11, 2002, design review meeting at the Westinghouse office in Monroeville, Pennsylvania. During the meeting, the applicant explained how the LBB bounding curves were developed. The applicant also demonstrated how the bounding curves were incorporated into the piping design criteria. All piping analysts working on the AP1000 were trained in meeting the LBB bounding curve criteria, in addition to the ASME Code, Section III design requirements. The "LBB evaluation report" will document the LBB criteria and evaluation for the applicable piping systems at the COL stage.

The discussion extended to define the "pipe break evaluation report," as both reports are referenced in the "acceptance criteria" of DCD Tier 1, Table 2.2.3-4. The pipe break evaluation report will document the results of the high-energy line break (HELB) design and analysis completed during the construction phase of the COL application to ensure adequate protection against the dynamic effects of postulated HELBs, including as-built piping configuration and piping materials. The applicant revised DCD Tier 2, Sections 3B.3.3.4, "Bounding Analysis Curve Comparison–LBB Criteria," and 3.6.3.4, "Documentation of Leak-Before-Break Evaluations," to reflect these clarifications. The staff finds the approach and the resolution of this issue acceptable for the design of the AP1000 high-energy line piping subjected to pipe rupture and LBB application.

The staff found the description of the pipe break hazard activities consistent with the guidance of Section 3.6.2 of the SRP. However, the staff found that the pipe rupture hazard analysis could be considered complete because a number of activities, including pipe stress and fatigue

analysis, were not complete and could not be completed as part of the AP1000 design certification. In RAI 210.035, the staff requested that the applicant provide additional information to clarify the status of the pipe rupture analysis. If the design is based on pipe rupture protection for the AP600, the applicant was asked to explain how the differences between the AP600 and AP1000 designs were considered in the evaluation.

In its response, the applicant stated that the AP1000 pipe rupture hazard evaluation makes extensive use of the work performed for the AP600 evaluation. The LBB bounding curves are made as part of the piping design requirements and, thus, the final stress analysis results will satisfy the LBB criteria for those lines identified for LBB evaluation. In addition, for high-energy lines which are not LBB candidates, the final fatigue analysis will confirm that the cumulative usage factor does not generate intermediate breaks different from those identified in AP600 Class 1 high-energy lines. This is based on the verification of the same LBB and fatigue analysis assumptions, the similarity in the AP600 and AP1000 piping and support configurations, and the fact that the majority of the postulated breaks are postulated at terminal ends. The physical coordinates of breaks may be shifted due to relocated equipment nozzles. However, the safety-related components for the AP1000 are in the same relative positions and in the same rooms as those for the AP600. Based upon the applicable piping isometrics for the AP600 and AP1000 designs, break locations were compared, as well as the need for a pipe whip restraint to protect any nearby safety-related equipment. The whip restraints for the AP600 and the AP1000 have not yet been designed and will be the responsibility of the COL applicant.

In its response, the applicant also stated that the AP1000 piping design used some larger pipe sizes at a higher power rating than those used in the AP600 design. The higher power rating has no direct impact on the pipe rupture hazard evaluation unless it translates into a higher normal system operating temperature and/or pressure. The pressures and temperatures for the AP1000 piping systems changed by insignificant percentages from that of the AP600 and therefore do not impact the evaluation. There are several piping systems in the AP1000 design where the pipe diameter has increased over that used for the AP600. All of these systems, except for the main feedwater system (MFWS), are part of the LBB evaluation and are expected to meet the LBB criteria, as discussed in the paragraph above. The breaks associated with the MFWS are mitigated by pipe whip restraints for both the AP600 and AP1000 designs. The whip restraints associated with the larger AP1000 feedwater pipe (50.8 cm (20 in.) vs. 40.6 cm (16 in.)) will be designed for loading adjusted to the AP1000 pipe sizes.

The applicant stated that the COL applicant is ultimately responsible for verifying several assumptions associated with LBB loading, the final fatigue analysis, and other analysis and layout (separation/shielding/protection) requirements defined in the pipe break hazard evaluation using inspection, test, analysis, and acceptance criteria (ITAAC).

During a meeting held at the Westinghouse office on September 9–11, 2002, the applicant provided two sample pipe rupture calculations and drawings for staff review. The sample calculations compared the AP1000 FW line and the CVS system letdown charging line with the corresponding AP600 lines. During the course of the staff review, the applicant pointed out that although seismic loads may be higher for some AP1000 piping systems, as compared to corresponding AP600 piping systems, this should not affect the selection of intermediate break

Design of Structures, Components, Equipment, and Systems

locations because of the elimination of the OBE. However, the staff noted that for Class 1 piping systems, the one-third SSE load is, in fact, included in the calculation of Equation 10 stresses (from ASME Code, Section III, NB-3650) and fatigue usage factors that form the basis for selection of intermediate break locations. Upon further discussion, the staff and the applicant agreed that based on the staff position on single earthquake design, the inclusion of one-third SSE load is not needed in the calculation of stresses and usage factors for the selection of intermediate break locations. The inclusion of this smaller earthquake, while not required, is conservative.

The applicant documented this position in its response to RAI 210.047 regarding piping design. In this RAI, the staff had requested clarification of the fatigue evaluation of Class 1 piping. As part of its response, the applicant stated that although the reduced range seismic event is considered in the evaluation of ASME Code Equations 10, 11, and 13, it does not need to be included in the identification of intermediate pipe break locations. DCD Tier 2, Section 3.6.2.1.1.1, "ASME Code, Section III, Division 1—Class Piping," was revised to include this clarification. The staff finds this position consistent with its position on single earthquake design and is, therefore, acceptable.

Based on its review of the two sample calculations and drawings, the staff concurs with the reasonableness of the applicant's approach for determining that the break locations for the AP1000 lines are the same as the break locations for the corresponding AP600 lines based on similarities in layouts and routing through rooms containing the same safety-related equipment. As such, the staff finds that the criteria for establishing postulated preliminary break locations in the AP1000 plant meet the pipe break criteria in SRP Section 3.6.2 and are therefore acceptable. The final pipe break locations will be verified by the COL applicant.

In DCD Tier 2, Section 3.6.4.1, "Pipe Break Hazard Analysis," the applicant stated that COL applicants referencing the AP1000 certified design will complete the final pipe whip restraint design and address as-built reconciliation of the pipe break hazards analysis, in accordance with the criteria outlined in the DCD. The as-built pipe rupture hazard analysis will be documented in an as-built pipe rupture hazards analysis report. In addition, the verification of the pipe break hazards analysis report is performed as a part of the ITAAC specified in DCD Tier 1, Table 3.3-6. The staff concludes that the information and commitments discussed above are consistent with the applicable guidelines in Section 3.6.2 of the SRP and, therefore, are acceptable.

In DCD Tier 2, Section 3.6.2.3.3, "Internal Systems Depressurization," the applicant discussed its design considerations for internal system depressurization loads resulting from a pipe break. It states that the loading has a short duration of approximately 0.5 seconds and arises from rapidly traveling pressure waves in piping systems connected to the broken piping system. Two types of configurations are possible—systems without check valves and systems with check valves. For systems with check valves, valve closure can increase the duration and magnitude of these loads. For piping systems without closing check valves, the high-frequency depressurization loadings contain little energy, and, therefore, these loadings are not considered in the piping and support analysis. The applicant stated that test results reported in the draft EPRI report entitled, "Piping and Fitting Dynamic Reliability Program, Volume I," confirm that this type of loading does not cause collapse of the piping system.

Upon reviewing this section of the DCD, the staff noted that the referenced EPRI report describes a program of tests that were performed to simulate seismic and system loading to failure. It was not clear how the applicant could use the results of the test report to justify not considering loadings generated from the internal system depressurization. In RAI 210.034, the staff requested that the applicant provide clarification and identify the section of the report that supports this conclusion. This was discussed during the September 9–11, 2002, meeting at the Westinghouse office. The applicant indicated that its conclusion is primarily based on past experience with these types of loadings in systems without closing check valves. On this basis, the staff and the applicant agreed that there was no need to reference the EPRI report. In a letter dated October 2, 2002, the applicant responded to this RAI. Subsequently, DCD Tier 2, Section 3.6.2.3.3, was revised by removing the reference to the EPRI report. The staff finds this revision to be acceptable.

In DCD Tier 2, Appendix 3E, the applicant identified and provided figures of the high-energy piping in the nuclear island with a diameter larger than 2.54 cm (1 in.). This appendix includes a statement that, in addition to the high-energy pipe identified in the figures, the hot water heating system (VYS) includes a limited amount of high-energy piping in the auxiliary building. The piping is identified as 7.62-cm (3-in.) diameter piping. The applicant indicated that no breaks are postulated in these lines because there are no anchors or fittings.

In reviewing this information, the staff noted that "no anchors or fittings" is an inadequate reason for not postulating breaks in a high-energy line. In RAI 210.057, the staff requested that the applicant provide additional justification for its conclusion that the approach is consistent with the SRP criteria. This was discussed during the September 9–11, 2002, meeting at the Westinghouse office. The applicant agreed with the staff's position and explained that although the hot water heating system lines are identified as 7.62-cm (3-in.) NPS, the high-energy portions, located in the auxiliary building subcompartments that include seismic Category 1 systems or components, are restricted to 2.54 cm (1 in.) and smaller NPS. Pipe breaks are not postulated in piping runs of a nominal diameter equal to or less than 2.54 cm (1 in.). The applicant responded to this RAI by providing this clarification. The applicant revised DCD Tier 2, Appendix 3E accordingly. The staff finds that the applicant's justification for not postulating breaks in a 2.54-cm (1-in.) diameter portion of the line meets the guidance in BTP MEB 3-1, Position B.3.A, which states that circumferential breaks should be postulated in fluid system piping exceeding a nominal pipe size of 1 inch, and is, therefore, acceptable.

In DCD Tier 2, Sections 3.6.2.6, "Evaluation of Flooding Effects from Pipe Failures," and 3.6.2.7, "Evaluation of Spray Effects from High- and Moderate-Energy Through Wall Cracks," the applicant described its procedures for evaluating flooding effects from pipe failures and spray effects from high- and moderate-energy through-wall cracks. The effects of flooding from high- and moderate-energy pipe failures on essential systems and components are described in DCD Tier 2, Section 3.4 and evaluated in Section 3.4 of this report. As set forth in Section 3.4 of this report, the staff reviewed the procedures and assumptions for the evaluation of spray effects and found them to be consistent with the guidance of SRP Section 3.6.2. Therefore, the applicant's procedures and assumptions are acceptable.

Design of Structures, Components, Equipment, and Systems

3.6.2.4 Conclusions

Based on its review, as documented above, 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 are generally consistent with the guidelines of Section 3.6.2 of the SRP. In addition, the criteria and methodology meet the requirements of GDC 4 as they relate to pipe rupture locations. Therefore, they are acceptable for ensuring that the AP1000 design is adequately protected against the effects of postulated HELB. Deviations from the SRP criteria, as discussed above, have been found to be acceptable alternatives to the SRP criteria. The staff's conclusion is based on the following reasons:

•   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 the applicant, thus providing adequate assurance that upon completion of the HELB analyses, the ability of safety-related SSCs to perform their safety functions will not be impaired by the postulated pipe ruptures. The as-built inspections of the high-energy mitigation devices will be performed as a part of the ITAAC required by the regulations.

•   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 provide adequate assurance that design-basis LOCAs will not be aggravated by the sequential failures of safety-related piping. In addition, the performance of the ECCS will not be degraded as a result of these dynamic effects. These provisions further 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-energy piping system or a postulated crack in a moderate-energy piping system inside and outside containment.

3.6.3 Leak-Before-Break

Under the broad-scope revision to 10 CFR Part 50, Appendix A, GDC 4 (see Volume 52 of the Federal Register (FR), pages 41288–95, October 27, 1987), the NRC allows the use of analyses to exclude from the design basis, consideration of the dynamic effects of pipe ruptures in nuclear power plants, provided the applicant can demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. The demonstration of low probability of pipe rupture utilizes deterministic fracture mechanics and leakage analyses that evaluate the stability of a postulated, circumferential, through-wall flaw in piping, as well as the ability to detect leakage through this postulated flaw, long before the flaw could grow to an unstable size and break the pipe. The evaluation procedures for the analyses are described in Section 3.6.3, "Leak-Before-Break Evaluation Procedures," in the draft SRP. Additional guidance on the fracture mechanics evaluation can be found in NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee—Evaluation of Potential for Pipe Breaks." The concept underlying such analyses is referred to as LBB.

The broad-scope rule excludes the dynamic effects of postulated pipe ruptures from the design basis of a plant with an NRC-approved LBB application. In a request for public comment on this issue, published on April 6, 1988 (53 FR 11311), the NRC staff subsequently clarified that effects resulting from postulated pipe breaks can be generally divided into local dynamic effects and global effects. The dynamic effects mentioned in GDC 4 are local dynamic effects. The local dynamic effects of a pipe break 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. GDC 4 permits the elimination of the local dynamic effects of postulated high-energy pipe breaks from the design basis of ALWRs using fracture mechanics analyses (LBB approach). DCD Tier 2, Section 3.6.3, is intended to incorporate LBB criteria and guidelines into the AP1000 design process to maximize the benefits of applying LBB.

The applicant identified 26 AP1000 piping subsystems for LBB application. The methods and criteria used by the applicant for these LBB evaluations are consistent with the guidance in NUREG-1061 and draft SRP Section 3.6.3 for the following reasons:

- The stability analysis of the LBB uses either a fracture mechanics analysis for brittle materials or a limit load analysis for ductile materials to determine a critical crack size for a postulated circumferential, through-wall crack under normal and seismic loads, combined through an absolute summation with a multiplying factor of 1.0, or 1.4 for algebraic summation (the first margin).

- The leakage analysis of the LBB determines a leakage crack size corresponding to 18.9 L/min (5 gpm), which is 10 times (the second margin) the leakage rate of 1.89 L/min (0.5 gpm) that could be detected by the leakage detection system of the AP1000 plant.

- The critical crack size is demonstrated to be two times (the third margin) the size of the leakage crack size.

Although the applicant addresses all three margins, the NRC staff considers the AP1000 LBB approach to be nonconventional, as set forth below.

Without detailed information on loading, the applicant performed the LBB analysis by creating bounding analysis curves (BACs) for all AP1000 LBB candidate piping subsystems. These BACs are located in DCD Tier 2, Appendix 3B. For a point on a BAC curve, the x-axis intercept represents the normal stress under the normal load condition (pressure + deadweight + thermal), and the y-axis intercept represents the maximum stress under the maximum load condition (normal + safe-shutdown earthquake) such that the three LBB margins mentioned above are satisfied. Conducting LBB evaluation in this way, the applicant has transformed conventional LBB analyses into the creation of BACs, which are used later in the piping systems design as additional design criteria. COL applicants will verify that the as-designed piping satisfies the LBB criteria by demonstrating that the calculated normal and maximum stresses using the as-designed loading for the piping would be below the BACs presented in DCD Tier 2, Figures 3B-1 to 3B-21.

Design of Structures, Components, Equipment, and Systems

3.6.3.1 Leak-Before-Break Acceptance Criteria

In GDC 4, the NRC states, in part, the following:

> …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 GDC 4 should be based on specific data, such as piping geometry, materials, and piping loads. For past generic and plant-specific LBB reviews, the staff reviewed the LBB analyses for piping systems with specific piping designs. However, applicants seeking design certification for ALWRs under 10 CFR Part 52 are allowed to incorporate preliminary stress analysis results in their LBB analyses, provided bounding limits (both upper and lower bound) are determined 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 phase based on actual material properties and final, as-built piping analysis as part of the ITAAC associated with 10 CFR Part 52 prior to fuel loading. The COL applicant will also review the certified material test reports to verify that the manufacturers complied with strength and Charpy toughness requirements of the ASME Code Section III. The preceding staff position on LBB application is stated in SECY-93-087 and was approved by the Commission in its SRM dated July 21, 1993. This is COL Action Item 3.6.3.1-1.

According to the draft SRP Section 3.6.3, the LBB acceptance criteria are specified in terms of three safety factors, or margins. A margin of $\sqrt{2}$ (1.0 is acceptable if loads are combined by the absolute sum method) on loads is necessary to assure that leakage-size flaws are stable at the normal load plus the SSE load. A margin of 10 on leakage is necessary so that leakage from the postulated flaw size is ensured of detection when the pipe is subjected to normal operational loads. A factor of 2 between the leakage flaw size (postulated under normal loads) and the critical flaw size (calculated under normal plus SSE loads) is necessary to ensure an adequate stability margin for the leakage flaw. The analysis should be performed for an entire pipe run from anchor to anchor. In addition to the specified margins, the draft SRP Section 3.6.3 also stipulates that an upper-bound stress limit be established for normal plus SSE loading in the crack stability analysis by using a lower-bound, stress-strain curve for base metal (regardless of whether the weld or base metal is limiting), and by using a lower-bound toughness for weld metal or base metal.

DCD Tier 2, Appendix 3B3.3, "Evaluation of Piping System Using Bounding Analysis Curves," provides procedures for COL applicants to calculate the normal stress by the algebraic summation of the normal operating loads, and to calculate the maximum stress by the absolute summation of the normal and SSE loads. The applicant's load combination satisfies the margin on loads, as specified in the draft SRP Section 3.6.3 stated above, and is acceptable to the staff. DCD Tier 2, Appendices 3B3.1.3, "Low Normal Stress Case (Case 1)," and 3B3.1.4, "High Normal Stress Case (Case 2)," provide a step-by-step approach for constructing the BACs using the leakage margin of 10 and the flaw size ratio margin of 2. Consequently, any

AP1000 LBB candidate piping subsystem with a BAC would satisfy these two remaining LBB margins, provided that, at the COL phase, the calculated normal and maximum stresses for the subsystem, based on the as-designed piping stress analysis, are below the corresponding BAC curve. DCD Tier 2, Section 3.6.4, "Combined License (COL) Information," specifies that COL applicants demonstrating that LBB analyses meet these criteria during the COL phase for an ALWR should compare the results of the as-designed piping stress analysis with the appropriate BACs documented in DCD Tier 2, Appendix 3B. DCD Tier 2, Section 3.6.3.2, "Design Criteria for Leak-Before-Break," further clarifies that "the highest stressed point (critical location) determined from the piping stress analysis is compared to the bounding analysis curve and has to fall on or under the curve." This is COL Action Item 3.6.3.1-2.

DCD Tier 2, Section 3.6.3.2, also specifies that for stainless steel piping, the material properties used are based on the ASME Code minimum values; for ferritic steel piping, the material properties are based on test results. The draft SRP guideline on material properties specifies the use of a lower-bound, stress-strain curve for base metal, and a lower-bound toughness for weld metal or base metal in the crack stability analysis. It should be noted that J-R curves for nonstainless steel materials are not available in the ASME Code, therefore, the use of actual material properties based on test results is appropriate and acceptable to the staff.

3.6.3.2 Leak-Before-Break Limitations

The staff has recommended 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. The draft SRP Section 3.6.3 provides that the NRC staff should review direct pipe failure mechanisms in LBB applications, including water hammer, creep damage, erosion, corrosion, fatigue, and other effects of environmental conditions. Volume 3 of NUREG-1061 also discusses the limitations for LBB used by the NRC staff. The LBB approach should not be applied to piping that can fail in service from these degradation mechanisms. 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 crack morphology may not be bounded by the postulated through-wall flaw used in LBB analysis. Adhering to the defense-in-depth principle, piping susceptible to failure from these potential degradation mechanisms is excluded from LBB applications.

DCD Tier 2, Appendix 3B.2, "Potential Failure Mechanisms for AP1000 Piping," addresses all degradation mechanisms mentioned above. In addition, certain degradation mechanisms, which have become routine review items for LBB applications since the issuance of the draft SRP Section 3.6.3 in 1987, are also addressed by the applicant in DCD Tier 2, Appendix 3B.2. These degradation mechanisms include SCC, thermal aging, and thermal stratification. DCD Tier 2, Appendix 3B.2 evaluates the susceptibility of the candidate piping to various degradation mechanisms to demonstrate that the candidate piping is not susceptible to failure from these degradation mechanisms. The discussion is focused on potential degradation mechanisms for AP1000 piping materials due to factors such as water chemistry, flow velocity, operating temperature, and steam quality, as well as their effects on plant operating procedures, operating temperature limits, water chemistry control, experience of past operating events, precaution measures, and design improvements to minimize undesirable occurrences. The NRC staff reviewed this information, in addition to operating history and the measures available

Design of Structures, Components, Equipment, and Systems

to prevent or mitigate these mechanisms, and determined that, with the possible exception of primary water stress-corrosion cracking (PWSCC) which is discussed in Section 3.6.3.4 of this report, the AP1000 LBB candidate piping subsystems are not susceptible to the identified degradation mechanisms. Section 3.6.3.4 of this report discusses a small number of LBB candidate piping subsystems that require additional attention in the areas of fatigue crack growth and thermal stratification.

3.6.3.2.1 Reactor Coolant System and Main Steam Line Leakage Detection Capability

As discussed in Section 3.6.3.1 of this report, the leakage analysis of LBB for piping is based, in part, on the capability of the plant's leakage detection system to detect leakage from a crack size corresponding to 18.9 L/min (5 gpm). LBB analyses incorporate a margin of 10 between the calculated leakage and the capability of the leakage detection system credited in the analysis. Therefore, for the leakage detection system to be acceptable to the staff, it has to be capable of timely detection of 1.89 L/min (0.5 gpm) leakage. The applicant's qualitative-quantitative assessment of all piping systems dated September 23, 2003, indicates that a system capable of detecting 1.89 L/min (0.5 gpm) would be acceptable for most of the AP1000 LBB candidate piping systems. However, two reactor coolant subsystems and the main steamline (MSL) inside containment may need a leakage detection system capability of 0.945 L/min (0.25 gpm) to qualify for LBB. Leakage detection for the proposed LBB candidate piping subsystems connected to the RCPB is evaluated in Section 5.2.5 of this report. In that evaluation the staff concluded that the RCPB leakage detection system for the AP1000 design is capable of detecting leakage of 0.95 L/min (0.25 gpm) and conforms to the guidelines of SRP Section 5.2.5, and is, therefore, acceptable. The leakage detection for the MSL inside the containment is evaluated below.

Draft SRP 3.6.3 specifies that leakage detection systems for LBB applications be sufficiently reliable, redundant, and sensitive. It further specifies that leakage detection systems for LBB applications be equivalent to the criteria set forth in RG 1.45 for piping inside the containment. RG 1.45 specifies a time frame of 1 hour for leakage detection. This time frame ensures that plant operators have timely information about unidentified leakage. Leakage detection for LBB purposes does not require the same degree of timeliness. This is discussed further below.

In its response to Open Item 3.6.3.4-2 dated March 22, 2004, the applicant provided information on detection instrumentations, procedures, and Technical Specifications (TS) to address leakage detection for MSL LBB. The primary method to detect MSL leakage inside containment is the containment sump level, which has the sensitivity to detect 0.945 L/min (0.25 gpm) as discussed in Section 5.2.5 of this report. As part of resolution of Open Item 3.6.3.4-2, the applicant modified the design of the leakage detection instruments by adding a third containment sump level instrument and by requiring two containment sump level instruments to be available at all times, instead of only one. This change was made to follow RG 1.45 regarding reliability and redundancy. The containment sump level instruments have indicators and alarms in the MCR.

In addition to the containment sump level sensors, the AP1000 provides three containment water level sensors as an additional method to detect MSL leakage. These sensors use a different level measurement process and, therefore, provide a diverse leakage detection method to the containment sump level sensors. As indicated in DCD Tier 2, Table 7.5-1, the

containment water level instrumentation is seismic qualified with 1E power supply, and provides both alarm and indications in the control room. If the containment sump level instruments were to fail, the containment water level sensors would identify a 1.89 L/min (0.5 gpm) leak within 3.5 days (or 0.945 L/min (0.25 gpm) within 7 days). This time lag of 3.5 days or 7 days is acceptable in the context of LBB because, on top of the margin of 10 on the leakage calculation, there is a margin of 2 between the critical flaw size and the leakage flaw size. Even in the event of a safe-shutdown earthquake, it would be extremely unlikely for the leakage flaw to grow to the critical flaw size in 3.5 or even in 7 days. In DCD Tier 2, Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," the applicant revised the procedures to identify the leakage source upon a change in the unidentified leakage into the sump by requiring an RCS inventory balance, which allows identification of MSL leakage when the RCS inventory balance indicates no RCS leakage.

DCD Tier 2, Section 16.1, "Technical Specifications," includes the TS for the AP1000 design. Several TS limiting conditions for operation (LCOs) are applicable to the MSL detection issue with respect to LBB. TS LCO 3.4.7 limits the unidentified RCS leakage to 1.89 L/min (0.5 gpm), and TS LCO 3.7.8 limits the MSL leakage inside containment to 1.89 L/min (0.5 gpm). TS LCO 3.4.9 specifies two redundant containment sump level channels to be operable for both RCS and MSL leakage detection. Consistent with RG 1.45, for RCS leak detection, one containment atmosphere radioactivity monitor (gaseous N13/F18) is also required to be operable for diversity. For MSL leakage detection, TS LCO 3.3.3 specifies two channels of containment water level instrument to be operable to backup the sump level channels.

In DCD Tier 2, Section 3.6.4.2, the applicant states that COL applicants referencing the AP1000 certified design will complete the LBB evaluation, which may necessitate lowering the detection limit for unidentified leakage in containment from 1.89 L/min (0.5 gpm) to 0.945 L/min (0.25 gpm). If so, the COL applicant shall provide a leak detection system capable of detecting a 0.945 L/min (0.25 gpm) leak within 1 hour and shall modify appropriate portions of the DCD including DCD Tier 2, Sections 5.2.5, 3.6.3.3, and 11.2.4.1, "Sump Level Instrument Testing." In addition, the applicant will also need to modify portions of DCD Tier 2, Chapter 16, "Technical Specifications." TS 3.4.7 (and Bases), TS Bases B3.4.9 and TS 3.7.8 (and Bases), TS Bases B3.4.9 and TS 3.7.8 (and Bases). This is COL Action Item 3.6.3.1-1.

Based on the review of the leak detection instruments, procedures, and AP1000 TS for detection of MSL leakage, the staff finds the leakage detection systems for MSL inside containment to be acceptable because these systems are consistent with the staff's review guidance described above.

3.6.3.3 Leak-Before-Break Candidate Piping Systems

In DCD Tier 2, Appendix 3E, the applicant indicated that the LBB methodology is intended to be applied to the following candidate high-energy piping in the nuclear island (NPS provided in parentheses):

- 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 safety injection line (15.2 cm (6 in.))
- ADS lines (35.6 cm, 20.3 cm, and 15.2 cm (14 in., 8 in., and 6 in.))

Design of Structures, Components, Equipment, and Systems

- RNS lines (50.8 cm, 30.5 cm, and 25.4 cm (20 in., 12 in., and 10 in.))
- PXS lines (45.7 cm, 35.6 cm, and 30.5 cm (18 in., 14 in., and 12 in.))
- CMT line (20.3 cm (8 in.))
- direct vessel injection line (20.3 cm (8 in.))
- main steamline (96.52 cm (38 in.))
- accumulator to isolation valve line (20.3 cm (8 in.))
- balance line from isolation valve to CMT (20.3 cm (8 in.))
- accumulator line after isolation valve (20.3 cm (8 in.))
- balance line from cold leg to CMT isolation valve (20.3 cm (8 in.))
- RNS discharge line (15.2 cm (6 in.))

DCD Tier 2, Section 3.6.3.2, indicates that Class 2 and 3 piping of Section III of the ASME Code are included within the LBB scope. Regarding the different ISI requirements of Section XI of the ASME Code for these two classes of piping, the applicant indicated that the weld and welder qualification and the weld inspection requirements applied to Class 3 LBB lines will be 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 a high-stress location. The staff reviewed these additional requirements and concludes that they provide additional and acceptable assurance of LBB integrity for Class 3 piping, consistent with the requirements for LBB applications for Class 1 and 2 piping.

3.6.3.4 Westinghouse Leak-Before-Break Evaluation Approach

The application of the LBB approach to ALWRs for which an applicant seeks 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 by the COL holder performing the appropriate ITAAC. The applicant established the above-mentioned bounding limits through BACs. The LBB analysis was performed for each applicable AP1000 LBB candidate piping subsystem listed in DCD Tier 2, Table 3B-1, with different combinations of piping material type, pipe size, pressure, and temperature. Curves satisfying the LBB criteria were developed and will be used by COL applicants to verify that the as-designed piping satisfies the LBB criteria. Based on the information provided in DCD Tier 2, Appendix 3B and the applicant's responses to RAI 251.004 through RAI 251.010, the staff finds this BAC approach acceptable, subject to resolution of the RAIs discussed below.

RAI 251.009 sought additional clarification on the construction of BACs, including the meaning of the horizontal part of the BACs. The applicant's response showed that for the leftmost point of the horizontal segment of a BAC, its critical flaw size was obtained using a flow stress as the maximum stress. Corresponding normal stress was determined using a leakage flaw size of one-half the critical flaw size. Further, a stress point to the right of the leftmost point of the horizontal segment will provide higher LBB margin because the leakage flaw size will be smaller with a higher normal stress. The NRC staff accepted this interpretation and determined that using a horizontal segment for the right portion of BACs is conservative. RAI 251.009 also requested that the applicant construct design curves considering all ASME Code requirements on piping stresses, and perform traditional LBB analyses for lines whose design curves exceed their corresponding BACs by 25 percent. The applicant's response to RAI 251.009 states that due to the difference in loading combinations and acceptance criteria between ASME piping

qualification and the LBB BAC, it is difficult to construct such a design curve. In addition, the applicant made the following point in its response:

> [T]hat's why for all thirteen AP1000 candidate Leak-Before-Break piping systems…, both ASME stress criteria and LBB stress criteria need to be satisfied as defined in the appropriate AP1000 Piping Analysis Criteria documents. The corresponding AP600 piping systems have all been evaluated for both ASME criteria and LBB criteria and found to be acceptable.

The staff agrees with the applicant's response in principle. However, the AP600 experience is not a guarantee that all AP1000 LBB candidate piping subsystems will meet their respective BACs. Since this issue is encompassed by the staff's concerns regarding validation of the BACs under Open Item 3.6.3.4-2, the staff considers RAI 251.009 resolved.

As discussed in Section 3.6.3.2 of this report, the staff determined that potential degradation mechanisms, such as erosion-corrosion-induced wall thinning, water hammer, fatigue, thermal aging, and thermal stratification, have been appropriately addressed. Since the V.C. Summer main coolant loop weld cracking event involving Alloy 82/182 weld material, the staff has considered the effect of PWSCC on Alloy 82/182 piping welds as an operating-plant issue, potentially affecting piping with or without approved LBB applications.

In RAI 251.004, the staff requested that the applicant (1) clarify whether Alloy 600 material, which is susceptible to PWSCC as indicated by the V.C. Summer primary loop leakage, will be used in any of the AP1000 LBB candidate piping subsystems, (2) provide test and plant operational data demonstrating that the proposed weld material, Alloy 52/152, is not susceptible to PWSCC, and (3) provide an inspection plan licensees would perform to address additional inspection techniques for detecting tight flaws that might exist in LBB piping welds.

The applicant's response to RAI 251.004 makes the following three points:

- Alloy 600 will not be used for any of the AP1000 LBB candidate piping subsystems.

- Alloy 52/152 weld material (for Alloy 690 base material) has been used in various applications, such as SG welds and safe end-nozzle welds, for 9 plants (7 years in one application) without any reported instances of environmental degradation. Further, although laboratory data for Alloy 52/152 in simulated primary water is limited, they indicate that environmentally related crack propagation was not observed for periods up to 4122 hours.

- Since Alloy 52/152 weld material has better crack resistance than Alloy 82/182, augmented ISI using eddy current testing (ET) to supplement the UT required by the ASME Code should not be necessary for the AP1000 applications.

The staff considers the information provided for item 1 above to be complete; no further information is required. Regarding item 2, although the chrome content of Alloy 52/152 is approximately twice the chrome content of Alloy 82/182, making Alloy 52/152 more resistant to PWSCC, the test and plant operational data for Alloy 52/152 are for periods less than 7 years.

Design of Structures, Components, Equipment, and Systems

This is not long enough to resolve the question of PWSCC for Alloy 52/152 material in the AP1000 LBB candidate piping, considering the licensing period for AP1000 facilities.

To address this issue for currently operating plants, the industry has undertaken an initiative to (1) develop overall inspection and evaluation guidance, (2) assess the current inspection technology, and (3) assess the current repair and mitigation technology. An interim industry report, "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for U.S. PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," was published in April 2001 to justify the continued operation of PWRs while the industry completes the development of the final report. The final industry report on this issue has not yet been published. Subsequent to staff review and evaluation of the final report and receipt of additional Inconel UT inspection data from the industry, the staff will determine if additional regulatory actions will need to be imposed to address the potential for PWSCC to occur in lines with currently approved LBB analyses in operating plants. To address this issue for the AP1000 application, the applicant was requested to modify DCD Tier 2, Section 3.6.4, to indicate that COL applicants should implement inspection plans, evaluation criteria, and other types of measures imposed on or adopted by operating PWRs with currently approved LBB applications as part of the resolution of concerns regarding the potential for PWSCC in those units. This was Open Item 3.6.3.4-1 in the DSER.

The applicant responded to this open item by revising DCD Tier 2, Section 3.6.4.4. This revision states that COL applicants will develop an inspection program for piping subsystems qualified for LBB, including "augmented inspection plans and evaluation criteria consistent with those measures imposed on or adopted by operating PWRs as part of the ongoing resolution of concerns regarding the potential for PWSCC in operating plants." This is COL Action Item 3.6.3.4-1.

Since the applicant has committed to requiring COL applicants to develop an inspection program consistent with the staff's position for operating plants stated above, Open Item 3.6.3.4-1 is resolved.

In RAI 251.005, the staff requested that the applicant provide values of crack morphology parameters (e.g., surface roughness, number of 45-degree and 90-degree turns, etc.), that were used in generating the BACs for LBB. The NRC staff also asked for a comparative study using the values of crack morphology parameters associated with transgranular stress-corrosion cracking (TGSCC). This information and the study were requested to evaluate the BACs and to understand the sensitivity of the AP1000 LBB analyses to a crack morphology similar to PWSCC. In its response to RAI 251.005, the applicant provided the values of crack morphology parameters used in generating the BACs. However, since chlorides will be controlled to minimum levels in the water environment of the AP1000 LBB candidate piping subsystems, and the hydrogen overpressure will keep the oxygen levels to near zero, the applicant discounted the possibility of TGSCC and considered the comparative study using the crack morphology parameters associated with TGSCC unnecessary. The applicant's argument does not address the intent of RAI 251.005.

The information provided by the applicant was not sufficient to address the staff position in SECY-93-087, as discussed in Section 3.6.3.1 of the DSER, regarding the need to demonstrate that adequate margins on leakage, loads, and flaw sizes are available for AP1000 LBB

candidate piping subsystems. In addition, the information provided was not sufficient to understand the degree to which PWSCC may affect LBB margins. Therefore, the staff's review was inconclusive. Further analyses were requested from the applicant to resolve these issues. This was Open Item 3.6.3.4-2 in the DSER.

Since the issuance of the DSER, the staff further determined that the applicant should provide additional information in two areas. The first area was a sensitivity study on an LBB candidate subsystem using crack morphology parameters associated with TGSCC as a surrogate for PWSCC. This assessed the reduced LBB margins (leakage rate margin and flaw size margin) for this subsystem under the postulated degradation mechanism. The second area was an LBB evaluation for a candidate AP1000 LBB piping subsystem using AP1000 piping stress analysis results to demonstrate that the calculated LBB stress results are below the BAC for that particular line. Further, an assessment containing quantitative and qualitative features (quantitative-qualitative assessment) of other LBB candidate subsystems was performed to provide reasonable assurance that all other AP1000 LBB candidate subsystems will be within their respective BACs when the COL applicant completes the final piping design and stress analyses.

In its letter of August 13, 2003, the applicant provided additional information regarding results from a sensitivity study that was performed for the AP1000 direct vessel injection (DVI) subsystem piping using the hypothetical IGSCC crack morphology parameters found in NUREG/CR-3600, "Refinement and Evaluation of Crack-Opening Analyses for Short Circumferential Through-Wall Cracks in Pipes." Although the staff requested an analysis using TGSCC parameters, IGSCC is a stress-corrosion cracking mechanism and is also an acceptable surrogate for PWSCC to assess the reduction in LBB margins. This study considers five sets of IGSCC crack morphology parameters with a surface roughness value approximately 3 times larger, and the number of 90-degree turns approximately an order of magnitude larger, than those normally used in LBB applications for operating plants. The results of this analysis indicate that the LBB margins for the DVI line A (DVI-A) are reduced below the LBB margins discussed above, but are consistent with the staff's LBB approach that it be demonstrated with margins such that leakage from a stable crack would be detected in service. These analysis results are consistent with staff safety evaluations of the effects of potential PWSCC in recently approved LBB applications for operating plants, and are therefore acceptable. Hence, the concern associated with the postulated piping degradation mechanism, including crack morphology parameters, that the staff considers essential in resolving DSER Open Item 3.6.3.4-2, has been addressed.

In its letter of September 23, 2003, the applicant provided an LBB evaluation for the DVI-A subsystem using AP1000 piping stress analysis results and a quantitative-qualitative assessment of all other AP1000 LBB candidate subsystems to demonstrate the feasibility of having all candidate subsystems meet the BAC requirements. The staff's review of the LBB evaluation for the DVI-A subsystem using AP1000 piping stress analysis results indicates that the calculated maximum stresses for the various line segments included in the subsystem are all below the BACs, demonstrating the feasibility that the DVI-A piping subsystem can be qualified for LBB at the COL phase.

The applicant's assessment of the feasibility of successfully qualifying the AP1000 LBB piping that had not yet been analyzed was performed by (1) applying correction factors to the piping

Design of Structures, Components, Equipment, and Systems

analysis results for the AP600 plant, (2) adjusting for changes in load and pipe geometry (discussed further below), (3) using flow stress based on statistical evaluation of applicable material samples, and (4) using leak detection capability for a 0.945 L/min (0.25 gpm) leak for the MS line subsystem, the CMT supply subsystem, and the pressurizer safety subsystem.

The applicant's assessment indicates that with these correction factors all other LBB candidate subsystems satisfy the LBB requirements except for the 6-inch pressurizer safety subsystem (DCD Tier 2, Figure 3B-19). For this exception, the applicant identified locations for installing pipe whip restraints if the stress analysis results from the final piping design exceed the BAC limits at the COL phase. The applicant also indicated that no other structural redesign or reanalysis would be required for the 6-inch pressurizer safety subsystem.

The AP1000 piping geometry is generally similar to the AP600 plant piping, but there are some differences between the two designs in terms of the elevations of the piping and equipment supports and the piping diameters. Similarly, the AP1000 seismic FRS differ from that of the AP600 design. However, the AP600 piping systems were analyzed using the same analysis methods and techniques described in Section 3.12 of this report. Therefore, the extrapolation of the AP600 results to the AP1000 design is a valid approach when appropriate consideration is given to adjustments in input loading and pipe geometry. The staff's review of the feasibility assessment of qualifying the AP1000 LBB piping not yet analyzed resulted in concerns about that part of the applied correction factor calculation which is based on scaling the seismic response spectral accelerations between the AP600 and AP1000 plant designs. Further, the staff had concerns about whether the design of the piping most important to the AP1000 thermal-hydraulic passive safety features could be qualified for LBB without major geometrical changes.

In its letter of October 13, 2003, the applicant provided a revised response to Open Item 3.6.3.4-2 which indicated that the AP1000 employs passive safety systems that are critical to emergency core cooling. Of these passive systems, the DVI subsystem and the fourth stage ADS are considered to be the most important portions of the passive safety system features for mitigating LOCAs. Therefore, it is desirable that the layout of these piping subsystems not be significantly changed to accommodate qualification of the piping for LBB requirements. In order to provide further confidence of the feasibility of these lines to be qualified for LBB, the applicant indicated that additional evaluations had been performed. Regarding the DVI-A subsystem, the applicant had previously performed a complete stress analysis of this safety system using the final AP1000 seismic response spectra included in the DCD. As discussed above, the analysis results indicate that the calculated maximum stresses for the various line segments included in the DVI-A piping design are all below the BACs, demonstrating the feasibility that the DVI-A subsystem can be qualified for LBB at the COL phase.

The feasibility assessment for the fourth stage ADS was revised to calculate a bounding seismic correction factor for the applicable seismic response spectra based on a comparison of seismic accelerations for each corresponding frequency of the response spectra curve. The revised evaluation results indicate that the calculated stresses for the 35.6 cm (14 in.) diameter and 45.7 cm (18 in.) diameter, fourth stage ADS piping included in the subsystem are all below the BAC stress requirements. This bounding approach is more conservative than the feasibility assessment methodology used for the other piping, and provides an additional level of confidence regarding the feasibility of qualifying these critical subsystems for LBB. Both the

original and bounding seismic correction factors, adjusted for pipe geometry difference, represent extrapolation from the piping analysis results for the AP600 piping systems. This extrapolation, which considered important parameters affecting pipe stresses and applied appropriate engineering principles in the assessment, is acceptable to the staff.

Two other measures were used in the quantitative-qualitative assessment (i.e., using flow stress based on statistical evaluation of applicable material samples and using leak detection capability of 0.945 L/min (0.25 gpm)). The staff considers the use of specified minimum material properties to be acceptable because such materials are easily obtainable and could be specified as requirements during the materials procurement. Section 3.6.3.2.1 of this report discusses the acceptability of using a leak detection capability of 0.945 L/min (0.25 gpm).

Based on the review of the above information, the staff finds that the applicant has adequately demonstrated the feasibility of meeting the BACs for the LBB candidate piping subsystems with one possible exception for which postulated breaks can be mitigated by the installation of pipe whip restraints if required. The successful result of this assessment provides high confidence that the ITAAC specified for LBB piping design will be satisfied at the COL stage. Therefore, Open Item 3.6.3.4-2 is resolved.

In RAI 251.006, the staff asked the applicant to confirm whether it would apply certain commitments made in the AP600 LBB review related to the evaluation of fatigue crack growth analyses and thermal stratification to the AP1000 design. The applicant confirmed in its response to RAI 251.006 that (1) it will perform fatigue crack growth analyses for AP1000 ASME Code Class 2 and 3 LBB candidate piping subsystems, and (2) it will perform system reviews similar to the calculations performed for the AP600 for the following AP1000 candidate piping subsystems which the AP600 review considered to be susceptible to thermal stratification:

- cold-leg piping in the loop with passive RHR (during long-term PRHR operation)
- pressurizer surge line
- automatic depressurization system stage 4 lines
- normal residual heat removal suction line
- passive residual heat removal return line

The applicant will include the resulting thermal loadings in the piping design analyses. The applicant's approach for the above-mentioned AP1000 LBB candidate piping subsystems includes the effects of fatigue crack growth and thermal stratification in the LBB evaluations and, thereby, goes beyond the explicit guidance of the draft SRP Section 3.6.3. The staff accepts this practice of extending the NRC guidance to address additional degradation mechanisms as an appropriate approach.

DCD Tier 2, Section 3.6.3.3, "Analysis Methods and Criteria," and DCD Tier 2, Appendix 3B.3, "Leak-Before-Break Bounding Analysis," present a detailed description of LBB analysis methodology and acceptance criteria, as well as procedures for performing LBB bounding analyses and establishing LBB bounding curves. The applicant supplemented this description in its responses to RAI 251.007 through RAI 251.010. In summary, the BAC curves are based on a minimum of two points, one corresponding to a low normal stress case (Case 1) and the other to a high normal stress case (Case 2). These two points were determined through a

Design of Structures, Components, Equipment, and Systems

conventional LBB analysis conducted in a reverse order. Instead of using actual piping stresses as input, this approach used an assumed low bending stress (which could be as low as zero), and the stress due to normal operating pressure to determine the normal stress (the x-coordinate for Point 1) for Case 1. The loads associated with this assumed normal stress were used to determine the leakage flaw size corresponding to a leakage rate equal to 10 times the leak detection capability. The maximum stress (the y-coordinate for Point 1) for Case 1 was then determined by increasing the assumed bending stress until the ratio of the critical flaw size, which was determined by the flaw stability analysis, to the leakage flaw size was reduced to the LBB criterion of 2.

This procedure was repeated for Case 2, in which the normal stress (the x-coordinate for Point 2) was calculated using an assumed high bending stress that would raise the maximum combined pressure and bending stress to the flow stress of the material. 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 is generated for each piping subsystem to be qualified for LBB for different pipe sizes and operating conditions. The applicant indicated that minimum wall thicknesses and material properties are assumed in these calculations. 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. The staff found these descriptions to be comprehensive, complete, and in accordance with the draft SRP Section 3.6.3.

In RAI 251.007, the staff expressed a concern about the possibility of having a piping system which satisfies all ASME Code requirements but fails to meet the requirements on BACs. The NRC staff discussed this RAI with the applicant in a meeting held at the Westinghouse office on September 9–11, 2002, in which the NRC staff reviewed several piping analysis calculations for LBB piping systems to confirm how the applicant applies and documents LBB criteria, as defined by the applicable criteria documents. Since both ASME stress criteria and LBB stress criteria are defined in the appropriate AP1000 piping analysis criteria documents, the NRC staff determined that the applicant's document control is appropriate for the evaluation and implementation of both LBB BAC and ASME Code requirements as they relate to AP1000 LBB candidate piping subsystems.

In RAI 251.008, the staff requested the applicant to provide its definition of the flow stress for the piping materials, and to provide axial stress, bending stress, leakage flaw size, and critical flaw size for a typical point on a sample BAC. In its response to RAI 251.008, the applicant confirmed that its flow stress was defined as one-half of the ultimate strength and yield strength of the piping material. The flow stress thus defined has been verified to be less than three times the ASME Code specified stress intensity, S_m, for all AP1000 candidate piping subsystems. The NRC staff recognized that the applicant was conservative in using the smaller value from these two definitions for the flow stress in this application. The purpose of this RAI was to assist the NRC staff in validating the BACs, should a need arise. As mentioned above, such a need was identified when a concern related to RAI 251.005 arose that the BACs might not be easily met by the most limiting piping. The NRC staff resolved its concerns regarding validation of the BACs in its evaluation of Open Item 3.6.3.4-2. Therefore, the staff considers RAI 251.008 closed.

In RAI 251.010, the staff requested the applicant to address the appropriateness of using a linear BAC to cover a wide range of normal stresses. The applicant responded by citing the calculations it performed for the intermediate points on the BAC during the AP600 review. This study showed that actual intermediate points are above the linear segment, indicating that the linear line approach is conservative. Based on a review of this study, the NRC staff accepts the applicant's conclusion that it is not necessary to generate an intermediate point for the AP1000 BACs.

3.6.3.5 Conclusions

For the reasons set forth above, the staff concludes that dynamic effects associated with postulated pipe ruptures for the AP1000 LBB candidate piping subsystems may be excluded from their design bases because the LBB analyses meet the relevant requirements of 10 CFR Part 50, Appendix A, GDC 4. This conclusion is based upon the applicant's demonstration that its LBB evaluation procedures meet the provisions of draft SRP Section 3.6.3 with respect to the areas of review, acceptance criteria, and review procedures. The NRC staff has also reviewed the applicant's additional requirements for Class 3 piping regarding preservice and inservice inspection. The staff concludes that these requirements provide additional and acceptable assurance of LBB integrity for Class 3 piping, consistent with the provisions for LBB application to Class 1 and 2 piping. In conclusion, the LBB design criteria, analyses, and inspections provide reasonable assurance that the probability of piping ruptures for the AP1000 LBB candidate piping subsystems is extremely low under conditions consistent with the design bases for these piping subsystems.

3.7 Seismic Design

The AP1000 maintains the AP600 design configuration with similar components. In WCAP-15612, "AP1000 Plant Description and Analysis Report," the applicant summarized all the design changes made to convert the AP600 standard plant to the AP1000 standard plant. The design parameters, including structural dimensions that will affect the seismic analysis and design of SSCs, are summarized below:

Steel Containment Vessel

- height of containment vessel: 65.63 m (215'-4")

- elevation at the top of containment vessel: 33.97 m (281'-10")

- containment free volume: 58,572 m^3 (2,068,467 ft^3)

- vessel thickness (both cylindrical shell and dome): 4.44 cm (1.75 in.)

- vessel material: SA738 Grade B steel

- design pressure: 406.8 kPa (59 psig)

- design code: ASME Code, 2001 Edition with 2002 Addenda for the containment vessel design

Design of Structures, Components, Equipment, and Systems

- inside diameter of main equipment hatch: 4.88 m (16 ft)

- center line elevation of main equipment hatch: 42.98 m (141 ft)

- bridge girder capacity for polar crane system: 725.5 metric tons (800 tons)

- top of girder rail elevation for polar crane system: 69.8 m (229 ft)

- top of trolley elevation for polar crane system: 73.15 m (240 ft)

Major Components

- overall height of reactor vessel: 28.89 m (94'-9.31")
- overall length of lower reactor vessel internals: 9.83 m (32'-3")
- lower support plate thickness for reactor vessel: 38.1 cm (15 in.)
- number of rod control clusters in reactor vessel: 53
- overall height of steam generator: 22.54 m (73'-11.23")
- upper shell I.D./O.D. of steam generator: 561.3 cm/584.2 cm (221 in./230 in.)
- lower shell I.D./O.D. of steam generator: 422.4 cm/441.4 cm (166.3 in./173.8 in.)
- overall height of reactor coolant pump: 3.87 m (18'-8.33")
- height of pressurizer (excluding support): 15.42 m (50'-7.11")
- height of pressurizer (including support): 17.73 m (58'-2")

Shield Building/Auxiliary Building/Passive Cooling Storage Tank Complex

- height of shield building: 101.7 m (333'-9")
- outer diameter of passive cooling water storage tank: 27.13 m (89 ft)
- height of passive cooling water storage tank: 10.67 m (35 ft)
- capacity of passive cooling water storage tank: 3,028,000 liters (800,000 gallons)
- size of the PCS air inlets: 3.66 m x 1.98 m (12 ft x 6.5 ft)

Containment Internal Structures

- height of steam generator compartment walls: 6.02 m (19'-9")
- height of pressurizer compartment walls: 10.29 m (33'-9")
- minimum water height of the IRWST: 8.71 m (28'-7")
- minimum water volume of the IRWST: 2,234 m^3 (78,900 ft^3)

As stated in DCD Tier 1, Table 5.0-1, and DCD Tier 2, Section 3.7.2, "Seismic System Analysis," the applicant proposes to construct the nuclear island (NI) on a hard rock site with a shear wave velocity of the foundation material equal to or greater than 2438.4 m/sec (8,000 ft/sec).

Based on Sections 3.7.1 through 3.7.3 of the SRP, the staff reviewed the adequacy of the seismic analysis and design of the applicant's AP1000 standard plant, including DCD Tier 2, Section 3.7, in addition to the clarifying information provided in the applicant's responses to the staff's RAIs. The staff also conducted two design calculation review meetings on

November 12–15, 2002, and April 2–5, 2003, at the office of Westinghouse Electric Company in Monroeville, Pennsylvania. The purpose of these two design review meetings was to discuss unresolved issues associated with certain RAIs, review design calculations, and confirm that issues identified by the staff from the DCD review were adequately resolved and DCD commitments were properly implemented.

As stated in DCD Tier 2, Section 3.7, the AP1000 SSCs are placed, depending on their function, into three seismic categories—seismic Category I (SC-I), seismic Category II (SC-II) and nonseismic (NS). DCD Tier 2, Sections 3.2.1, "Seismic Classification," and 3.7.2.8, "Interaction of Seismic Category II and Nonseismic Structures with Seismic Category I Structures, Systems, or Components," define these three categories and provide standards for the seismic analysis and design of items classified into these categories. The applicant defined the categories as follows:

SC-I: Seismic Category I, in general, applies to all safety-related SSCs, as well as those SSCs required to support or protect safety-related SSCs. These SSCs are required to be designed to withstand the seismic loads due to the SSE, as discussed in DCD Tier 2, Section 3.7, and other applicable loads without loss of structural integrity and functional capability.

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

NS: NS SSCs are those that are not classified as SC-I or SC-II. The criteria used for the design of these SSCs are described in DCD Tier 2, Section 3.7.2.

Based on the definition of the seismic categories above, the applicant placed all nuclear island structures, including the basemat, in SC-I; the annex building between Columns A and D in SC-II; and the remainder of the buildings as NS.

Section 3.2 of this report discusses the staff's evaluation of the classification of seismic categories for the AP1000 SSCs and the seismic design criteria for SC-I and SC-II structures used by the applicant.

In DCD Tier 2, Section 3.7.1, "Seismic Input," the applicant detailed the design criteria related to the seismic input ground motion, critical damping values, and supporting ground media for the NI structures. In DCD Tier 2, Section 3.7.2, the applicant described the seismic analysis methods, modeling techniques, and design criteria for all SC-I, SC-II, and NS SSCs. DCD Tier 2, Section 3.7.3, "Seismic Subsystem Analysis," provides the seismic design criteria for the subsystems.

Design of Structures, Components, Equipment, and Systems

3.7.1 Seismic Input

SRP Section 3.7.1 provides guidelines for the staff to use in reviewing issues related to the development of seismic input ground motions (design ground motion response spectra and ground motion time histories), percentage of critical damping values, and supporting media for seismic Category I structures.

3.7.1.1 Design Ground Response Spectra

As described in DCD Tier 2, Section 3.7.1.1, "Design Response Spectra," the input seismic design ground motion response spectra for the SSE are specified at the foundation level in the free field. The horizontal and vertical design ground motion response spectra for the AP1000 were developed based on the response spectra in Revision 1 of RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," with consideration of high-frequency amplification effects. DCD Tier 2, Table 3.7.1-3, provides the relative values of the spectral amplification factors for the design response spectra. The horizontal and vertical ground motion response spectra corresponding to 2, 3, 4, 5, and 7 percent of the critical damping are depicted in DCD Tier 2, Figures 3.7.1-1 and 3.7.1-2, respectively. The peak ground acceleration (PGA) for both horizontal and vertical components of the SSE is 0.3 g, where "g" is the acceleration due to gravity.

For the AP1000, the applicant employed the SSE ground motions applied at the foundation level to calculate the seismic responses (peak floor accelerations, member forces, and FRS) of the SC-I SSCs. Specifying the design ground motion at the foundation level in the free field meets the guidance provided in SRP Section 3.7.1 and is, therefore, acceptable. The staff's review of DCD Tier 2, Sections 2.5 and 3.7.1, also found that the applicant proposed to employ the same SSE ground motion (ground response spectra and peak ground acceleration) used in the AP600 design for the AP1000. Regarding the adequacy of the design response spectra, the staff raised a concern that recent developments in ground motion assessment demonstrate that spectral shapes applicable to the rock sites located in the central region of the Eastern United States are rich in the high frequency range. These shapes of ground response spectra indicate that the peaks at frequencies of 10 Hertz (Hz) and above are higher than those used for the AP1000 design. In its response to RAI 230.005, the applicant indicated that the additional high-frequency energy is not expected to be damaging to the structures and equipment used in nuclear power plants such as the AP1000. The applicant further stated that the COL applicant will demonstrate that the future potential site conditions will fall within the parameters for which the AP1000 is designed. These parameters include the ground motion design response spectra. A COL applicant will need to demonstrate that the site response spectra are equal to or less than those specified for the AP1000 design for both horizontal and vertical motions. Based on the justification provided by the applicant and the staff's evaluation discussed in Section 2.5.2 of this report, the staff finds the ground motion response spectra proposed for the AP1000 design to be acceptable.

3.7.1.2 Design Ground Motion Time History

In DCD Tier 2, Section 3.7.1, the applicant 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 AP1000 SC-I SSCs. Specifically, the applicant generated these

three components of the ground motion time history by modifying a set of actual recorded time histories from the Taft recordings (obtained from the Kern County earthquake recorded at Taft, California). These time histories were digitized to have a total duration of 20 seconds, with a corresponding strong motion duration (defined as the time for the Arias Intensity to rise from 5 percent to 75 percent of its final value) greater than 6 seconds. The applicant then adjusted the amplitude and frequency content of the synthetic motions using the recorded motions as seed motions to yield response spectra for 2, 3, 4, 5, and 7 percent of the critical damping, which equal or envelop the SSE design ground response spectra at a sufficient number of frequency points, as recommended in Section 3.7.1 of the SRP.

In DCD Tier 2, Section 3.7.1.2, the applicant also stated that since the only site condition for the AP1000 design certification application is the hard rock site, and a fixed base foundation model is to be used for the design of the NI structures, the three components of the ground motion time history were generated with a time step size of 0.005 second for applications in the seismic analyses. The maximum frequency (cut-off frequency) of interest in the horizontal and vertical seismic analyses of the NI for the hard rock site is 33 Hz. Modes corresponding to frequencies higher than 33 Hz are included in the analysis so that the mass effects in these higher modes are included in the seismic response calculations.

In order to demonstrate the adequacy of using the 0.005 second time step for the seismic analyses of structures with a relatively high fundamental frequency founded on a rigid foundation media, such as the NI structures, the applicant, in its response to RAI 230.012, presented the results of a parametric study of the time-history analysis of a single degree of freedom (SDOF) having a frequency of 100 Hz. Three integration time steps, 0.01 second, 0.005 second, and 0.00025 second, were used in this study. The results showed that the differences of the relative displacements and absolute accelerations obtained from these three analyses are negligible. Also, the use of the time step size of 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 conventional nuclear power plants and other advanced reactors. Therefore, the time step size proposed by the applicant is acceptable. In addition, the maximum frequency of 33 Hz used in the seismic analyses meets the SRP guidelines. On this basis, the staff concludes that the maximum frequency used by the applicant is 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 no significant frequency gaps exist in the two horizontal synthetic input motions. The applicant did not generate a target PSDF for the vertical component of the ground motion time history. Instead, the applicant 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 guidance of the SRP. Since the vertical acceleration response spectrum exceeds the horizontal spectrum by only a small amount between about 3 Hz and 9 Hz, and significant conservatism exists in the computed PSDF as compared to the target PSDF in this frequency range, the use of the horizontal target PSDF for the vertical target PSDF is acceptable to the staff. In addition, the applicant confirmed that the three components of synthetic time history are statistically independent from each other by demonstrating that the cross-correlation of coefficients at zero time lag between

Design of Structures, Components, Equipment, and Systems

these three earthquake components is less than 0.16, as referenced in Revision 1 of RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".

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 DCD Tier 2, Section 3.7.1, meets the guidelines of SRP Section 3.7.1 and RG 1.60 and is, therefore, acceptable. Consequently, it is also acceptable to define the design ground motion in accordance with the SRP guidelines at the foundation level in the free field, and to calculate the seismic responses for the AP1000 nuclear island structures founded on a hard rock site.

3.7.1.3 Critical Damping Values

As set forth below, the damping ratios used in the analysis of the AP1000 SC-I structures (4 percent for welded and friction-bolted steel structures and equipment, 7 percent for bearing bolted steel structures and equipment, and 7 percent for reinforced concrete structures, as documented in DCD Tier 2, Table 3.7.1-1) are the same as the SSE damping ratios specified in RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," and, therefore, are acceptable. In DCD Tier 2, Table 3.7.1-1, the applicant proposed to use the following damping values, 5 percent for concrete-filled steel plate structures, 10 percent for analyzing full cable trays and related supports, 7 percent for empty cable trays and related supports, 7 percent for the HVAC duct systems, 4 percent for HVAC-welded ductwork, and 20 percent for fuel assemblies. Because welded aluminum structures and prestressed concrete structures are not to be used in the AP1000 standard plant design, the applicant decided to delete the damping ratios assigned to these two types of structures from DCD Tier 2, Table 3.7.1-1. As a result of its review, the staff finds that the damping ratios specified by the applicant for the seismic analysis and design of concrete-filled steel plate structures, cable trays and supports, HVAC ductwork and supports, and fuel assemblies are based on test results performed by the industry, and are consistent with those accepted by the staff in the review of other advanced reactors, such as the advanced boiling-water reactor (ABWR), System 80+, and AP600. The use of the damping ratios documented in DCD Tier 2, Table 3.7.1-1, meets the guidelines prescribed in RG 1.61 and/or common industry practice. On this basis, the staff concludes that the damping ratios proposed by the applicant are acceptable.

3.7.1.4 Key Dimensions of Nuclear Island Structures

As described in DCD Tier 2, Section 3.7.1, the four AP1000 SC-I structures (reinforced concrete shield building, including the passive cooling water storage tank, steel containment vessel, modular constructed containment internal structures, and reinforced concrete auxiliary building) are supported on a common basemat and form the NI. The NI foundation, while not precisely rectangular, is approximately 77.4 m (254 ft) long and 49.1 m (161.0 ft) wide. The foundation embedment depth (measurement from finished grade at Elevation 100'-0" to the bottom of the basemat at Elevation 60'-6") is 12.04 m (39.5 ft) and the thickness of the basemat is 1.83 m (6.0 ft). The applicant also provided other key dimensions (such as actual dimensions of basemat, radius of shield building, geometry of shield building roof, thickness of walls, and floor elevations) in DCD Tier 2, Figures 3.7.1-14 and 3.7.2-12. The staff finds that the applicant has provided sufficient structural dimensions to develop the seismic analysis model for the NI structures.

3.7.1.5 Site Interface Parameters

In DCD Tier 2, Chapter 2.0, "Site Characteristics," and DCD Tier 2, Table 2-1, the applicant specified that the COL applicant will use the following design site parameters to confirm the adequacy of the AP1000 seismic design for a specific site:

- The site-specific ground motion response spectra, defined at the foundation level, are bounded by the proposed design response spectra (the modified RG 1.60 ground response spectra) anchored to 0.3 g as shown in DCD Tier 2, 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 average allowable static bearing capacity is greater or equal to 412 kPa (8600 pounds per square feet (psf)) over the foot print of the NI at its excavation depth. The allowable bearing capacity under static plus dynamic loads exceeds 5,746 kPa (120,000 psf).

- The minimum shear wave velocity of the rock foundation is equal to or greater than 8000 ft/sec.

Based on its review experience of other advanced reactors such as ABWR, System 80+, and AP600, 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 AP1000 seismic design, with the exception of the definition for the average allowable static bearing capacity for a hard rock site.

The staff requested the applicant to clarify whether this term refers to allowable strength or allowable displacement of the foundation. In its response to RAI 241.001, the applicant stated that the design will be acceptable for a hard rock site that has an allowable bearing capacity of 450 kips per square foot. The staff's review experience indicates that this is an extremely high value of "allow bearing capacity," and would be difficult for the COL applicant to substantiate. In addition, the applicant's response did not clarify whether this definition refers to strength or displacement considerations. Further, the review of the civil/structural criteria document performed by the staff during the November 12–15, 2002, audit indicated that hard crystalline bedrock should have an allowable bearing capacity of 4 kips per square foot. The definition of allowable bearing capacity for a hard rock site should also account for the influence of bedding direction, level of cracking, and other discontinuities in the rock material which can serve to limit bearing capacity. The applicant was requested to clarify these discrepancies. This was Open Item 3.7.1.5-1 in the DSER.

In its response to this open item, the applicant referred to its response to RAI 240.001, Revision 1, dated March 23, 2003. The staff discussed this issue with the applicant during the October 6–10, 2003, audit. As a result, the applicant provided a revised definition of the bearing capacity in DCD, Section 2.5.4.2, "Bearing Capacity," and stated that the evaluation of the allowable capacity of the bedrock should be based on the properties of the underlying

Design of Structures, Components, Equipment, and Systems

material, including appropriate laboratory test data to evaluate the strength, and should consider local site effects, such as fracture spacing, variability in properties, and evidence of shear zones. The allowable bearing capacity should provide a factor of safety appropriate for the design load combination, including SSE loads. Also, the applicant revised the DCD to provide the basis for increasing the maximum allowable dynamic bearing capacity from 85,000 lb/ft^2 (415,006 kg/m^2) to 120,000 lb/ft^2 (585,891 kg/m^2). The staff's review found that the definition provided by the applicant is consistent with common industry practice and is acceptable to the staff. On this basis, Open Item 3.7.1.5-1 is resolved.

3.7.1.6 Conclusions

The staff concludes, for the reasons set forth above, that the applicant meets the relevant requirements of GDC 2 and Appendix A to 10 CFR Part 100. Specifically, the staff finds that the applicant has given appropriate consideration to the most severe SSE to which the AP1000 SC-I SSCs might be subjected. In particular, the applicant specified the following for analysis of the design:

•   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 conform to 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 AP1000 SC-I SSCs that conform to the guidelines of RG 1.61 and industry practice

These factors ensure that the seismic inputs are adequately defined to form a reasonable basis for the design of the AP1000 SC-I SSCs to withstand seismic loadings.

3.7.2 Seismic System Analysis

SRP Section 3.7.2 provides guidelines for the staff to review issues related to seismic system analysis areas. These review areas include seismic analysis methods, natural frequencies and response loads, procedures used for analytical modeling, soil-structure interaction, development of FRS, effects due to three components of earthquake motion, combination of modal responses, interaction of non-Category I structures with Category I structures, effects used to account torsional effects, comparison of responses, analysis procedure for damping, and determination of Category I structure overturning moments.

The review scope of the seismic system analysis for the AP1000 considers the seismic analysis methods and acceptance criteria for all SC-I SSCs. It includes a review of basic assumptions, procedures for modeling, seismic analysis methods, development of in-structure response spectrum envelopes, consideration of torsional effects, evaluation of overturning and sliding of SC-I structures, and determination of composite damping. The effects of soil-structure interaction (SSI) to the seismic responses of the NI structures are not included in the review scope because the applicant only considered a hard rock site for the AP1000 design certification application and the SSI effects for structures founded on hard rock sites are negligible. The review also covered design criteria and procedures for evaluating the

interaction of non-SC-I structures with SC-I structures and the effects of parameter variations on FRS.

As discussed in Section 3.7 of this report, AP1000 SSCs have been classified in accordance with RG 1.29. As stated, non-SC-I SSCs are further classified into SC-II and nonseismic categories. Sections 3.2 and 3.7 of this report discuss the staff's evaluation of the seismic classification of SSCs. In DCD Tier 2, Section 3.7.2, the applicant stated that the AP1000 SC-I building structures consist of the steel containment vessel, containment internal structures, and coupled shield and auxiliary buildings. These structures are supported on a common basemat and form the NI structures. The NI basemat is also classified as an SC-I structure. All other building structures are classified as either SC-II or nonseismic.

As described in DCD Tier 2, Section 3.7.2, SC-I structures are analyzed and designed for the SSE specified in DCD Tier 2, Section 3.7.1, and the criteria described in DCD Tier 2, Section 3.7.2. SC-II building structures are designed for the SSE using the same methods prescribed for SC-I structures. Nonseismic structures are analyzed and designed for seismic loads according to the Uniform Building Code (UBC) standards for Zone 2A. The following sections discuss the staff's review of the analysis and design results.

3.7.2.1 Seismic Analysis Methods

SRP Section 3.7.2.II.1 states that the seismic analysis of all Category I SSCs should use either a suitable dynamic analysis method or an equivalent static load method, if justified. The SRP criteria generally deal with linear elastic analysis coupled with allowable stresses near the elastic limits of the structures. However, for certain special cases (e.g., evaluation of as-built structures), the staff has accepted the concept of limited inelastic/nonlinear behavior when appropriate. The actual analysis, incorporating inelastic/nonlinear considerations, is reviewed on a case-by-case basis.

As discussed in Section 3.7 of this report, SC-I structures of the AP1000 include the coupled shield and auxiliary building, steel containment vessel, and containment internal structures. These three SC-I structures are supported by a 1.83 m (6 ft) thick SC-I basemat which is founded on a hard rock site. The coupled upper structures and the basemat are designated as the NI structures. In DCD Tier 2, Section 3.7.2.1, "Seismic Analysis Methods," and DCD Tier 2, Table 3.7.2-14, the applicant stated that the modal superposition time-history method and the equivalent static acceleration method are the two primary analysis methods to be used to perform seismic analyses of SC-I structures. The response spectrum analysis method may be used to perform an analysis of a particular structure or portion of the structure. The purposes of these analyses are (1) the generation of seismic acceleration profiles (peak acceleration versus the height of structures) of NI structures, (2) the determination of seismic loads (member forces and bending moments) for the design of NI building structural components, and (3) the development of in-structure response spectra (or FRS), which will be used as input motions for the analysis and design of subsystems (piping systems and components).

In DCD Tier 2, Sections 2.5 and 3.7.1, the applicant proposed to found the NI structures on a hard rock site with an embedment of (12.04 m) 39'-6". The staff's review identified a question regarding how lateral soil pressures due to embedment were calculated for use in the design of the exterior walls of the NI. In its responses to RAI 230.002 dated October 4, 2002, and

Design of Structures, Components, Equipment, and Systems

January 21, 2003, the applicant stated that the exterior walls of the NI were designed for
(1) lateral earth pressure equal to the sum of static earth pressure plus the dynamic earth
pressure, and (2) lateral earth pressure equal to the passive earth pressure. However, the
applicant did not perform calculations of total earth pressures for the various load cases to
ensure that the load case will lead to the maximum wall moments and shears. This was Open
Item 3.7.2.1-1 in the DSER.

In its response dated June 23, 2003, the applicant referred to Revision 1 of the response to
RAI 230.002 and stated that the exterior walls of the NI structures were designed for (1) lateral
earth pressure equal to the sum of the static soil pressure plus the dynamic earth pressure, and
(2) lateral earth pressure equal to the passive earth pressure from side soils. Design of the
exterior walls based on these two sets of soil pressures is consistent with the industry practice
and is acceptable to the staff. During the October 5–10, 2003, audit, the staff confirmed that
the design shear forces and bending moments for Wall 1 were calculated based on the above
soil pressure cases in Calculation APP-1200-CCC-106. On this basis, Open Item 3.7.2.1-1 is
resolved.

Modal superposition time-history analyses, using the computer program ANSYS, are performed
to calculate (1) floor time histories for generating FRS needed for the analysis and design of
SC-I and SC-II subsystems (substructures, piping systems and components) housed in the NI
structures, (2) maximum absolute nodal accelerations and maximum nodal displacements
relative to the basemat, (3) maximum member forces and moments for certain portion of
structures, and (4) overturning moments and sliding forces at the bottom of the basemat for
evaluation of dynamic stability (sliding, overturning, and floatation) of the NI structures. During
meetings conducted on November 12, 2002, and April 2, 2003, the staff found that when
analyses were performed to calculate seismic responses of the NI structures in the two
horizontal directions (north-south and east-west), a three-dimensional (3D), fixed-base (fixed at
the bottom of the basemat, Elevation 60'-6"), lumped-mass stick model of the coupled shield
and auxiliary buildings, steel containment vessel, containment internal structures, reactor
coolant loop, and basemat was used to represent the NI structures.

The applicant also performed a vertical time-history analysis of a 3D fixed-base (fixed at the
middle of the basemat, Elevation 63'-6"), finite element (FE) model consisting of a model of the
coupled auxiliary/shield building (ASB) and a simplified FE model of the containment internal
structure (CIS) to represent the NI structures. The purpose for using the FE model is to
calculate seismic responses in the vertical direction because the plant has a large plan
dimension and the floors cannot be properly modeled as rigid lumped-mass nodes. In addition,
floor flexibilities are expected to be significant, and only the use of an FE model can reasonably
account for the local amplification due to floor flexibilities. (The adequacy of using a fixed-base
model to represent the hard rock foundation is discussed in Section 3.7.2.3 of this report.)

The three components (two horizontal and one vertical) of the ground motion-time history, with
0.005 second time steps, are applied simultaneously in the analyses of both the 3D lumped-
mass stick model and the 3D finite element model. The algebraic sum technique was used to
combine the responses (such as floor acceleration time histories) due to the three components
of the ground motion-time history. The horizontal (north-south and east-west) and vertical
responses from the 3D lumped-mass stick model are to be used for the design of the NI
structures with the exception that the vertical responses from the finite element model are to be

used for the design of the auxiliary building, as indicated in the DCD. In both the 3D lumped-mass stick model and the 3D finite element model, the lateral support due to soil or hard rock below grade is not considered in the analyses. The basis for not considering the effect of lateral supports due to soil or hard rock in the analyses is that the seismic model analyzed without the embedment effects will result in a higher response than when the model is analyzed with full lateral support below grade. The effect of the lateral soil or rock support on in-structure response spectra is small when compared to the ±15 percent peak spectra spreading used in generating design FRS.

In the equivalent static acceleration analyses, using the ANSYS or STRUDL computer code for the NI structures in the two horizontal directions, the applicant, as described in DCD Tier 2, Section 3.7.2.1.1, "Equivalent Static Acceleration Analysis," used a 3D fixed-base (fixed at the middle of the basemat, Elevation 63'-6") finite element model of the coupled ASB including the shield building roof structures and the basemat. The effects of the CIS are considered in the finite element model by coupling the lumped-mass stick model of the CIS to the FE model of the ASB with rigid links in the radial direction. According to the applicant, the mass effect of the steel containment vessel is not considered in the analyses because the ratio of the steel containment vessel mass to the mass of the coupled ASB complex is negligible.

The acceleration profiles generated from the modal time-history analyses of a 3D lumped-mass stick model are used as input motions to these equivalent static analyses. The equivalent static analyses are also performed for the 3D fixed-base finite element model of the CIS and the shell of revolution model of the steel containment vessel. The analysis for each earthquake component is performed by applying equivalent static loads to the structural model at each finite element nodal point. The static load at each nodal mass point is equal to the corresponding mass times the maximum absolute acceleration value (from the acceleration profiles) for the earthquake ground motion component being evaluated. The results due to each of the three components of the earthquake ground motion are combined by either the square root of the sum of squares (SRSS) method or the 100 percent, 40 percent, 40 percent method. In the latter method, each of the member forces due to the three earthquake components are combined directly, using the assumption that when the peak response from one component occurs, the responses from the other two components are 40 percent of the peak response in that direction.

The purpose of these equivalent static acceleration analyses is to generate (1) the in-plane and out-of-plane forces for the design of floors and walls of the ASB and the CIS, (2) the design-bearing reaction and member forces in the basemat, (3) the design-member forces for the shield building roof structures, and (4) stresses for the containment vessel design. The use of equivalent static acceleration analyses to calculate seismic design forces meets the guideline of SRP Section 3.7.2, and is, therefore, acceptable.

The staff identified an issue regarding the potential effect on the calculated maximum seismic responses as a result of arbitrarily applying two simultaneous horizontal components of the ground motion in two different directions. In its response to RAI 230.018, the applicant stated that the significance of the input direction was investigated by changing the sign of one of the ground motion components and repeating the time-history analyses. The comparison of analysis results (the maximum absolute accelerations, member forces, bending moments, and torsional moments) indicates that the responses for the two cases are similar, with differences

Design of Structures, Components, Equipment, and Systems

of less than 10 percent in most responses. However, the torsional moment at the elevation of the PCS tank on the shield building roof increased by 16 percent. The applicant presented its calculation and demonstrated to the staff, during the meeting conducted on April 2, 2003, that the 16 percent increase in the torsional moment at the PCS tank has no effect on the final design of the roof structures, and that the FRS obtained from the two sets of analysis are comparable. On the basis that the 16 percent increase in the torsional moment has no effect on the final design, the staff considers this issue resolved.

In DCD Tier 2, Section 3.7.2, the response spectrum analysis method is not considered as a primary analysis method for the analyses and design of the NI structures. It may be used to perform an analysis of a particular structure or a portion of a structure. Section 3.7.3 of this report discusses the details of the staff's review of this analysis method for analyzing substructures and components.

As a result of its review of DCD Tier 2, Section 3.7.2.1, the applicant's response to the RAIs, and the discussion conducted with the applicant during the design review meetings as discussed above, the staff finds that the use of the equivalent static acceleration method, modal-time history analysis method, and response spectrum analysis method to seismically analyze the NI structures and calculate the responses meets the guideline prescribed in Section 3.7.2 of the SRP and is consistent with the common industry practice. On this basis, the staff concludes that the seismic analysis methods proposed by the applicant are acceptable. The following open and confirmatory items were addressed by the applicant as discussed below.

- The procedures, as described above, for performing modal time-history analyses for generating seismic responses in both horizontal directions and vertical direction met the SRP Section 3.7.2 guidelines and are acceptable. However, it was unclear to the staff how the analyses were performed and how the results will be used for the design of subsystems and components. The applicant agreed to provide a description of the analysis procedures presented in the April 2–5, 2003, meeting in the DCD, including how the resulting seismic responses are to be utilized for the equivalent static analyses and the design of the subsystems and components. This was Confirmatory Item 3.7.2.1-1.

   The staff reviewed the applicant's revision to DCD Tier 2, Section 3.7.2. The applicant provided a sufficient description of the analysis procedures, including how the resulting seismic responses are to be used as input to the equivalent static analyses and the design of SSCs. In addition, the applicant made several corrections to the DCD that were discussed during the August 22, 2003, telephone conference and October 6–9, 2003, audit. Specifically, the applicant (1) replaced the term "STRUDL" with the term "GTSTRUDL" in DCD Tier 2, Tables 3.7.2-14 and 3.7.2-16, (2) added the phrase "coupled aux/shield building shell model, with superelement of containment internal structures" in the sixth row of the first column of DCD Tier 2, Table 3.7.2-14, and (3) replaced the term "stick models" in the second row of the first column of DCD Tier 2, Table 3.7.2-16 with the term "superelement."

   On the basis of the discussion above, Confirmatory Item 3.7.2.1-1 is resolved.

Design of Structures, Components, Equipment, and Systems

- As discussed above, DCD Tier 2, Section 3.7.2.1, states that the base of the lumped-mass stick model is fixed at the bottom of the basemat at Elevation 60'-6", while the base of the finite element model is fixed at the middle of the basemat at Elevation 63'-6". At the April 2–5, 2003, meeting, the applicant explained that the plate bending elements of the basemat in the auxiliary building section of the FE model had to be placed at the centerline of the basemat to appropriately represent bending of the basemat. In the shield building area, however, the FE model contained solid elements to represent the base of the shield building. For conservatism, the lumped-mass stick model was then fixed at the bottom of the basemat to maximize the computed overturning moments. The 3-foot difference of the fixed-base location was judged to have a minor effect on the computed response of the NI structures. The approach taken by the applicant to locate the fixed-base of models meets the SRP Section 3.7.2 guideline, which states that for structures supported on rock or rock-like material, a fixed base assumption is acceptable. On this basis, the fixed-base developed by the applicant is acceptable. The applicant agreed to include the basis for placing the fixed base of the FE model at the middle of the basemat in the future DCD revision. This was Confirmatory Item 3.7.2.1-2 in the DSER.

  This issue was discussed during the October 6–9, 2003, audit. The applicant indicated that when the finite element model was used, an extra moment equal to the base shear multiplied by one-half of the basemat thickness (3 feet) was added to calculate the final overturning moment. Also, the applicant added a phrase to the end of the fourth paragraph of DCD Tier 2, Section 3.7.2.1.2, "Time-History Analysis," which states that the 3-foot difference in elevation of the fixed-base location is not significant because the concrete between Elevations 60'-6" and 63'-6", below the auxiliary building, is nearly rigid. On this basis, Confirmatory Item 3.7.2.1-2 is resolved.

- As discussed above, the applicant performed time-history analyses using the three components of ground motion time history as input, for both the 3D lumped-mass stick model and the 3D FE model to calculate seismic responses in three directions (two horizontal directions and vertical direction) at each nodal point. When the NI SSCs were designed, the applicant used the horizontal responses (peak floor accelerations and FRS) from the 3D lumped-mass stick model and the vertical responses as input. In the review of calculations conducted during the April 2–5, 2003, audit, the applicant presented the staff with a comparison of FRS and the zero period acceleration, (ZPA, or peak floor acceleration) obtained from the 3D lumped-mass stick model and the 3D FE model. The comparison indicated only insignificant differences. On this basis, the staff concludes that the seismic inputs used for the design of the NI SSCs are reasonable and acceptable. To address the staff's concerns, the applicant agreed to provide a detailed description of seismic analysis procedures, as well as procedures for applying these seismic responses to the design of the NI SSCs in a future revision to the DCD. This was Confirmatory Item 3.7.2.1-3 in the DSER.

  The applicant revised DCD Tier 2, Section 3.7.2, to provide a sufficient description of the analysis procedures, including how the resulting seismic responses are to be used as input to the equivalent static analyses and the design of the NI SSCs. Therefore, Confirmatory Item 3.7.2.1-3 is resolved.

Design of Structures, Components, Equipment, and Systems

3.7.2.2 <u>Natural Frequencies and Response Loads</u>

DCD Tier 2, Section 3.7.2.2, "Natural Frequencies and Response Loads," states that when the modal time-history analyses were performed for the lumped-mass stick model of the NI structures, a total of 200 vibration modes, extending up to a frequency of 83.8 Hz, were included to ensure enough percentage of mass participation in the response calculation. According to the applicant, the consideration of 200 vibration modes resulted in more than 80 percent of the total mass of the NI structures participating in the seismic response calculation. DCD Tier 2, Tables 3.7.2-1 through 3.7.2-4 list the modal properties of the lumped-mass stick model representing the NI structures (coupled ASB, steel containment vessel, CIS without reactor coolant loop, reactor coolant loop, and combined model of all structures).

The staff's review found that the modal properties listed in these tables did not include those for modes corresponding to frequencies of 33 Hz and above. Some of the effective masses shown in the tables are much less than 80 percent of the total mass. The applicant was requested to clarify these inconsistencies. As a result of a discussion on April 3, 2003, the applicant agreed to include all modal properties up to 200 vibration modes in DCD Tier 2, Tables 3.7.2-1 through 3.7.2-4. This was Confirmatory Item 3.7.2.2-1 in the DSER.

The applicant revised DCD Tier 2, Table 3.7.2-4, to include all modal properties (modes, modal frequencies, and effective masses) up to a frequency of 83.8 Hz (200 vibration modes) in the modal time-history analyses, and indicated that more than 80 percent of the total horizontal mass and nearly 80 percent of the total vertical mass were considered to calculate seismic responses. On this basis, Confirmatory Item 3.7.2.2-1 is resolved.

With regard to the adequacy of considering the total cumulative mass up to 80 percent of the total mass participating in the seismic response calculation, the staff applied its review experience from the AP600 standard plant design to the AP1000. During the design review of the AP600, the applicant 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. This comparison showed that the difference between the two sets of results is insignificant. In addition, it is the staff's understanding that in the dynamic analysis, the ANSYS computer code calculates the nodal accelerations relative to the ground support first. The final absolute accelerations at nodal points are obtained by adding the ground-motion time history, which is considered as motions of structures in the rigid frequency range (frequencies higher than 33 Hz), to the calculated relative nodal accelerations at different locations. In doing this, the mass effects of high modes to the seismic responses are implicitly included in the analyses.

On the basis of the above discussion, the staff concludes that the applicant has considered sufficient vibration modes and effective mass of the NI structures in the modal time-history analyses. This is acceptable to the staff.

3.7.2.3 <u>Procedure Used for Modeling</u>

SRP Section 3.7.2.II.3 states that a nuclear power plant facility consists of very complex structural systems. To be acceptable, the stiffness, mass, and damping characteristics of the structural systems should be adequately incorporated into the analytical models.

DCD Tier 2, Figures 3.7.1-14 and 3.7.2-12 (sheets 1 through 12) show the general arrangement of structural elements and key dimensions of the NI basemat, building elevations, distance between column lines, location of the containment vessel and reactor vessel centers, and thickness of walls and floor slabs. Based on general plant arrangement information, the applicant developed seismic models for (1) a finite element model of the coupled ASB, including the shield building roof structures, (2) a finite element model of the CIS, and (3) an axisymmetric shell model of the steel containment vessel. These 3D finite element models provide the basis for the development of the lumped-mass stick model of the NI structures. The general arrangement of structural elements and key dimensions is provided in these DCD figures, and the staff finds them sufficient for the development of analytical models. Therefore, they are acceptable. The applicant designated the information in DCD Tier 2, Figures 3.7.1-14 and 3.7.2-12 (sheets 1 through 12), as Tier 2*, and noted that any proposed change to these figures will require NRC approval prior to implementation of the change. The designation of this information as Tier 2* is also acceptable to the staff.

The procedure used to develop analytical models for the seismic analysis of the NI structures is discussed in DCD Tier 2, Section 3.7.2.3, "Procedure Used for Modeling," and other related sections. The staff's review and evaluation of the adequacy of the modeling techniques used by the applicant are summarized below.

In DCD Tier 2, Section 2.5, "Geology, Seismology, and Geotechnical Engineering," and DCD Tier 2, Table 2-1, the applicant proposed to found the AP1000 NI structures on a hard rock site for the design certification application. For a hard rock site, the shear wave velocity of the foundation media is 2438 meters per second (m/s) (8000 feet per second (fps)) or higher. When the combined model of the NI structures and the supporting foundation material was developed, the applicant used a fixed-base model to represent the hard rock foundation because the effects of the SSI are considered to be negligible for this condition. To use a fixed-base model to represent the NI structures founded on a hard rock site without consideration of the SSI effects meets the guidelines of SRP Section 3.7.2.II.4 "Soil-Structure Interaction," which states that "for structures supported on rock or rock-like material, a fixed base assumption is acceptable." Therefore, the use of a fixed-base model is acceptable.

On the basis of the general arrangement drawings and through the collaborative efforts of the applicant's consultants from Spain (Initec), Japan (Obayashi), Italy (Ansaldo), Switzerland (NOK), and a consultant for USA Lapay, three explicit 3D fixed-base finite element models were developed for the NI structures. These models were developed to represent the coupled ASB, the CIS, and the steel containment vessel, respectively. The properties of the concrete-filled structural modules of the CIS were computed using the combined gross concrete section and the transformed steel face plates of the structural modules. The weight density of concrete plus uniformly distributed miscellaneous deadweights were considered by adding surface mass or by adjusting the material mass density of the structural elements. An equivalent tributary slab area load of 244 kilograms per meter-squared (kg/m^2) (50 pounds per square foot (lbs/ft^2) was considered to represent miscellaneous deadweights. In addition, 25 percent of the floor live load or 75 percent of the roof snow load, where applicable, was considered as an addition to the effective mass in the seismic models. The finite element models were used to determine the stiffness and mass properties of the equivalent 3D lumped-mass stick seismic models. These models, using the acceleration profiles obtained from the modal time-history analyses of

Design of Structures, Components, Equipment, and Systems

the 3D lumped-mass model, were also used in the static analyses to calculate bending moments and member forces in the individual structural elements.

The applicant used the 3D finite element models as the basis to develop an equivalent lumped-mass stick model for each of the three SC-I buildings (i.e., coupled ASBs, including the shield building roof structures, steel containment vessel, and CIS). The translational, bending, and rotational stiffnesses of the finite element models were represented by single vertical members of the 3D lumped-mass stick models. For the coupled ABSs and the CIS, the stiffness values of each vertical member of the stick models were determined by applying static forces or moments equivalent to 1.0 g at the top of the corresponding segment of the explicit 3D finite element models. The vertical sectional properties of the stick model for the coupled ASBs were developed to have the fundamental vertical frequency of the stick model matching that of the finite element model. The mass and equivalent stiffness for the fundamental sloshing mode of the water contained in the PCS water tank was included in the shield building roof structure stick model. The masses of the remaining PCS tank water, the water in the IRWST within the CIS, and the water in the spent fuel pool located in the auxiliary building were also included in the corresponding stick models. A simplified stick model for the reactor coolant loop was developed separately and coupled with the stick model of the CIS. The applicant used the 3D lumped-mass stick model of the coupled ASBs for the seismic time-history analyses to calculate the horizontal floor responses, and used the 3D finite element model of the coupled ASBs to calculate the vertical floor responses, including the responses for the flexible floors. As discussed in Section 3.7.2.1 of this report, the use of the 3D lumped-mass stick model and the 3D FE model for generating seismic responses of the coupled ASBs is acceptable.

For the steel containment vessel, members of the lumped-mass stick model representing the cylindrical portion are based on the properties of the actual circular cross section of the containment shell. Members of the stick model representing the bottom head are based on equivalent stiffness calculated from the shell of revolution analysis for a load equivalent to 1.0 g statically applied in the vertical and horizontal directions. For members of the stick model representing the top head of the containment vessel, the shear, bending, and torsional stiffness properties are based on the average of the properties at the successive nodes using actual circular cross sections, and the axial stiffness properties are based on equivalent stiffness calculated from the shell of revolution analyses for a load equivalent to 1.0 g statically applied in the vertical direction. The polar crane was modeled with five masses at the midheight of the bridge and one mass for the trolley. It includes the flexibility of the crane bridge girders and truck assembly, and the local flexibility of the containment shell. The polar crane model is coupled with the stick model of the steel containment vessel at the elevation of the crane girder. This crane model was developed based on a plant operation procedure that requires, during plant operating conditions, the polar crane to be parked in the north-south direction of the plant with the trolley located at one end near the steel containment vessel.

As discussed above, the applicant developed a 3D fixed-base finite element model to represent the coupled ASB founded on a hard rock site. The fixed base is located at the middle thickness of the basemat, Elevation 63'-6". This model is used to perform equivalent static analyses for calculating member design forces and modal time-history analyses to compute the vertical seismic responses. The lumped-mass stick models for the three SC-I structures (coupled ASBs, including the roof structures, containment steel vessel, and CIS) were combined, using rigid links and beams, with the NI basemat to form the 3D fixed-base, lumped-mass stick model

for seismic analyses of the NI structures. The fixed base is located at Elevation 60'-6", which is the elevation at the bottom of the 1.83 m (6 ft) thick basemat.

The staff, using the guidelines prescribed in SRP Section 3.7.2.3, reviewed the methods and procedures employed by the applicant for modeling the NI structures. On the basis of its review of the DCD, the applicant's response to the RAIs related to structural modeling and design calculations; and discussions with the applicant conducted during the November 12–15, 2002, and April 2–4, 2003, audits, the staff finds the modeling techniques used by the applicant, in general, to be acceptable. However, the staff raised a number of technical questions. These questions and their resolutions are discussed below.

- For the development of the seismic stick model of the steel containment vessel, including the polar crane, the staff raised, in RAI 230.015, a concern that the parked polar crane location described in DCD Tier 2, Section 3.7.2.3.2, "Steel Containment Vessel," be consistent with the location during plant operation. The staff suggested that this location be specified as an interface item for the COL applicant. In its response to RAI 230.015, and in DCD Tier 2, Section 3.7.2.3.2, the applicant specified the requirement for the parked location and orientation of the polar crane as Tier 2* information; any proposed change to these figures will require NRC approval prior to implementation of the change. This commitment is sufficient to resolve the staff's concern.

- The staff was concerned that omitting the lateral supports to the NI stick model due to embedment may underestimate the seismic responses. In its response to RAI 230.014, the applicant provided a comparison between seismic structural responses for cases in which the lateral supports below grade were included in one analysis and omitted in the other. The comparison showed that the results are more conservative for the case in which the lateral supports below grade were omitted. The applicant's justification resolved the staff's question.

- During the audits conducted on November 12, 2002, and April 2, 2003, the staff discussed with the applicant the development of the dynamic model of the NI structures, and reviewed the applicant's analysis reports based on both the 3D fixed-base, lumped-mass stick model and the 3D fixed-base FE model. The seismic analysis results from the 3D fixed-base FE model of the coupled ASB shows net tension in the shield building wall. This phenomenon suggests that during the postulated seismic event, parts of the basemat will lift up from the rock surface, resulting in changes in the basemat stresses and reduction in shear wall stiffness due to reinforced wall cracking. As a result of its detailed review of the seismic modeling approach and analysis methods, the staff was concerned that the assumptions of uncracked reinforced concrete walls and a fixed-base foundation may become invalid. With this finding, the applicant was requested to provide justification to show that the current seismic analysis results used for the design of the NI SSCs are reasonable and acceptable.

    In resolving this issue, the staff, during the meeting conducted on April 2, 2003, explained to the applicant its concern regarding the significance of uplift due to seismic excitation of the NI and the effect of a reduction in the stiffness of the shear walls. The staff reached the following conclusions:

Design of Structures, Components, Equipment, and Systems

- The applicant will consider using the east-west, lumped-mass stick model of the NI structures supported on a rigid plate with nonlinear springs that transmit reactions in horizontal and vertical directions to simulate the foundation contact area, and will perform a seismic time-history analysis (the nonlinear springs will be in action only when the rigid plate is in contact with the subgrade). The results of this seismic time-history analysis will be compared to peak accelerations and the FRS at the lumped-mass node points obtained from the current 3D model analysis without the uplift consideration. If the comparison shows differences, the applicant should evaluate the significance of these differences and their effects on the current seismic design.

- With regard to the effect of a reduction in shear wall stiffness (due to shear wall cracking) on the seismic analysis results (natural frequencies, peak floor accelerations, and the FRS), the applicant will consider using a 3D lumped-mass stick model with reduced member stiffness to conduct a time-history seismic analysis. Results from this analysis will be compared against those currently used by the applicant for the design of the NI SSCs. If the comparison shows differences, the applicant should evaluate the significance of these differences and their effects on the current seismic design.

When the final seismic analyses were performed for the NI structures, the applicant incorporated the two above-discussed effects in its final seismic model for calculating seismic responses. These seismic responses were also compared to those currently used for the seismic design. If the comparison showed differences on the order of 10 percent or less, the combined effect of uplifting and shear wall cracking would have been considered insignificant. Otherwise, the seismic loads used for the design would have to be revised accordingly.

Depending on the outcome of the comparisons from the two separate analyses discussed above (one for the uplift effect and the other for stiffness reduction), the design calculations for the certified design would have to be revised. This was Open Item 3.7.2.3-1 in the DSER.

The applicant submitted its response to this open item by letter dated June 24, 2003. The applicant indicated that (1) a nonlinear time-history analysis, using the ANSYS computer code, was performed to address the concern of foundation mat uplifting, and (2) a separate analysis was performed to respond to the concern related to the shear wall stiffness reduction. In the second analysis, the applicant reduced the stiffness properties, based on the recommendations of the Federal Emergency Management Agency (FEMA) in its report, FEMA-356, "Prestandard and Commentary for the Seismic Rehabilitation of Buildings," (Table 6-5), by a factor of 0.8 to more realistically represent the stiffness properties of the ASB and CIS shear walls with cracking due to an earthquake. The staff's review of the applicant's response identified the following issues:

- The applicant should couple and incorporate the two effects (foundation mat uplifting and shear wall stiffness reduction) in the final seismic model of the NI structures instead of analyzing them separately.

- When the effect of foundation mat uplifting was evaluated, the applicant did not consider the slapping (or impact) between the foundation mat and the hard rock foundation.

- The evaluation results for the shear wall stiffness reduction presented in the June 24, 2003, submittal did not include the comparison of structural frequencies nor the floor response spectra.

In its letter dated August 22, 2003, the applicant provided a comparison of the FRS with the effects of shear wall stiffness reduction to those with full shear wall stiffness at critical locations. Also, during an August 22, 2003, conference call, the applicant explained that an analysis done with the combined effects found that the effects of slapping are insignificant. The applicant committed to use the revised seismic analysis results (seismic forces and FRS) in the final design of the NI structures and seismic Category I components. In addition, the applicant indicated that the following three effects were considered in the final analysis:

(1)   NI foundation mat uplifting
(2)   impact between the foundation mat and hard rock foundation
(3)   member stiffness reduction due to cracking of shear walls

This analysis report was made available for a subsequent staff audit.

During the October 6–9, 2003, audit, the staff reviewed Calculations APP-1000-S2C-037, Revision 3; APP-1000-S2C-061, Revision 1; and APP-1000-S2C-064, Revision 1. The staff reached the following conclusions:

- In order to demonstrate the insignificance of the basemat uplifting, the applicant coupled the two effects (foundation mat uplifting and shear wall stiffness reduction) in the simplified two-dimensional (2D), nonlinear time-history analysis, and compared the obtained result with those based on considering the effect of shear wall stiffness reduction only. The staff finds that the applicant's analysis approach is consistent with industry practice and considers the demonstration to be acceptable.

- The applicant has properly used the ANSYS computer code in the nonlinear seismic analyses to address the staff's concerns related to the basemat uplifting and the impact (slapping) between the basemat and the hard rock foundation. The approach used in the analysis is consistent with industry practice and the results, which concluded that the effects of basemat uplifting and impact (slapping) due to an SSE are insignificant, are reasonable and acceptable.

- The use of FEMA recommendations to modify member stiffness of the seismic model of the NI structures is consistent with current industry practice and is reasonable and acceptable.

Design of Structures, Components, Equipment, and Systems

Also, in its revised response to Open Item 3.7.2.9-1 dated October 10, 2003, and a revision to of DCD Tier 2, Section 3.7.2.3, the applicant provided the basis for using 80 percent of the Young's modulus "E" for modeling the reinforced concrete shear wall structures. The applicant also explained how the final seismic design loads (including the FRS for the subsystem design) were calculated. For the reasons discussed above, Open Item 3.7.2.3-1 is resolved.

- As described in Section 3.7 of this report, the layouts of the coupled ASB for the AP1000 and the AP600 remain the same; only the height of the shield building and the size (or volume) of the passive containment cooling water storage tank (PCCWST) were increased. With the above-mentioned design changes, the staff expected, based on its engineering judgment, that the seismic amplification and seismic responses of the AP1000 ASB would be higher than those of the AP600 ASB. However, as a result of its review of the DCD and the analysis reports during the April 2–5, 2003, audit, the staff, in RAI 230.018, found that the dominating frequency of the AP1000 ASB in the vertical direction is 6.07 Hz. This is lower than the dominating frequency of 6.77 Hz for the AP600 ASB. The seismic response accelerations, which were calculated based on the dominating frequency, at the top of the AP1000 ASB are lower than those at the top of the AP600 ASB. A similar observation was identified for the steel containment vessel roof. In the vertical direction, the dominating frequency is 18.1 Hz with the maximum acceleration of 1.49 g for the AP600, while the dominating frequency is 16.85 Hz with the maximum acceleration of 1.40 g for the AP1000.

In its response to RAI 230.18, the applicant provided the following justifications for these results:

(a)   For the shield building roof structures, the maximum vertical absolute acceleration is 0.9 g for the AP600 and 0.89 g for the AP1000 (in the initial analyses). In the most recent AP1000 analyses, the dominating frequency is 5.81 Hz and the maximum absolute acceleration is 0.96 g in the vertical direction. These differences in seismic response are partly due to changes in modal properties but, are also affected by the ground-motion time history which envelops the design ground response spectrum.

(b)   For the steel containment vessel, the maximum vertical absolute acceleration is 1.49 g for the AP600 and 1.40 g for the AP1000 (in the initial analyses). In the most recent AP1000 analyses, the dominating frequency is 16.97 Hz and the maximum absolute acceleration is 1.13 g in the vertical direction. The reduction in the vertical response is associated with better definition of the AP1000 polar crane, and the use of a multimass model of the polar crane instead of the single mass model used in the AP600 analyses and the initial AP1000 analyses. The first frequency (representing the polar crane mode) of the combined model in the vertical direction is 6.415 Hz compared to that of 5.843 Hz in the previous analyses.

The applicant's justification for the results of the shield building roof structures (see (a) above) appears reasonable and is acceptable. As for the steel containment vessel (see (b) above), the staff's review of the DCD and the seismic analysis report of the steel

containment vessel (Calculation APP-1000-S2C-037) during the November 12–15, 2002, audit revealed that the first frequency (polar mode) of the combined model (combined vessel lumped-mass stick model with multimass polar crane model) is 6.42 Hz, which is in the same range as that of the initial analyses. The frequencies (16.97 Hz and 28.2 Hz) and modal masses corresponding to the two dominating vertical modes of the revised steel containment vessel model (combined vessel model with multimass crane model) also remain essentially unchanged in comparison with those of the initial analyses. Because the frequency corresponding to the crane mode is more than 10 Hz apart from the dominating frequencies of the vessel, the staff did not expect that the vertical absolute acceleration at the top of the AP1000 steel containment vessel would be significantly reduced due to the use of the multimass model of the polar crane. Based on the above discussion, the applicant was asked to justify why the vertical acceleration at the containment vessel dome is reduced from 1.40 g to 1.13 g as a result of using a different polar crane model. This was Open Item 3.7.2.3-2 in the DSER.

During the October 6–9, 2003, audit, the staff found, from its review of Calculation APP-1000-S2C-063, Revision 1, and discussion with the applicant, that the use of a multimass polar crane model versus a single-mass polar crane model will not significantly alter the seismic response of the steel containment dome. The major cause for the reduction of the vertical seismic response (acceleration) from 1.40 g to 1.13 g at the top of the steel containment vessel is due to the modification of the original model to more realistically connect the fixed-base of the steel containment vessel lumped-mass stick model to the concrete internal structures instead of the coupled ASBs, and to make dynamic properties of the stick model more closely representative of the steel containment vessel. As a result of this modification, the vertical acceleration at the top of the steel containment vessel changed from 1.13 g to 1.25 g. The applicant documented the model modification and these results in a revision to DCD Tier 2, Section 3.7.2. The applicant also agreed to revise its response to this open item to document the modifications of the steel containment seismic model that was used in the final seismic analyses.

In its submittal dated October 10, 2003, the applicant revised its response to this open item and documented the modifications (including the basis for the modifications) to the steel containment seismic model used in the final seismic analyses. The modifications to the steel containment seismic model were also incorporated in DCD Tier 2, Section 3.7.2. On this basis, Open Item 3.7.2.3-2 is resolved.

- The staff's review of the applicant's Calculation APP-1200-S2C-001 during the November 13–15, 2002, audit identified a contradiction between the text and Figure 6-4 of the calculation regarding the connection between the stick model of the CIS and the FE model of the coupled ASBs. During the April 2–5, 2003, audit, the applicant indicated that this editorial inconsistency was revised in Revision 1 of the calculation. The staff found the revised Figure 6-4 of the calculation to be acceptable and sufficient to resolve this concern.

- The seismic model for the NI structures was based on the uncracked concrete section properties of shear walls. During the teleconference on January 23, 2003, the staff

questioned the applicant's assumption that the calculation of shear wall stiffness does not consider a reduction in stiffness due to cracking. The stiffness reduction would affect the seismic loads for the design of critical sections of the NI structures and the frequency locations of the FRS peaks. The applicant agreed to review the references provided by the staff on the stiffness reduction in shear walls, and provide justification or correction as needed. This was Open Item 3.7.2.3-3 in the DSER.

The applicant referred to its response to Open Item 3.7.2.3-1 in its June 23, 2003, submittal. As discussed in the evaluation of Open Item 3.7.2.3-1 above, the applicant reduced the stiffness properties, based on the recommendations in FEMA-356, Table 6-5, by a factor of 0.8 to more realistically represent the stiffness properties of the ASB and the CIS shear walls with cracking due to an earthquake. Also, the applicant revised DCD Tier 2, Section 3.7.2.3 to update the design of critical sections based on the final seismic loads (seismic forces and FRS). Based on the evaluation of Open Item 3.7.2.3-1 and review of the DCD revisions, Open Item 3.7.2.3-3 resolved.

3.7.2.4 Soil-Structure Interaction

In DCD Tier 2, Section 3.7.2.4, "Soil-Structure Interaction," the applicant stated that the SSI effect is not significant for the NI structures founded on rock with a shear wave velocity greater than 2348 m/s (8000 fps). The basis provided by the applicant for not performing the SSI analyses for the NI structures meets the guidelines of SRP Section 3.7.2.II.4, which state that for structures supported on rock or rock-like material (such materials are defined by a shear wave velocity of 1067 m/s (3500 fps) or greater), a fixed-base assumption (not considering the SSI effects) is acceptable. Therefore, the basis for not performing the SSI analysis is acceptable.

3.7.2.5 Development of Floor Response Spectra

RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," provides guidelines for (1) developing FRS, (2) smoothing FRS and broadening peaks, and (3) developing design FRS. Using the guidelines of RG 1.122 and the floor response time histories calculated from the 3D fixed-base, lumped-mass stick model of the NI structures, the applicant developed the in-structure response spectra (also known as the floor response spectra) at specified locations and elevations in the NI structures for damping values equal to 2, 3, 4, 5, 7, 10, and 20 percent of the critical damping. (Section 3.7.2.1 of this report discusses the adequacy of the calculated floor time histories.) The spectral peaks associated with the structural frequencies are broadened by ±15 percent to account for the variation in the structural frequencies due to uncertainties in parameters such as material and mass properties of structures, damping values, seismic analysis techniques, and modeling techniques. In the auxiliary building where the FRS at a number of nodes (locations) have similar characteristics, a single set of FRS was developed by enveloping the broadened FRS at each of the nodes. The final peak broadened FRS are to be used as the input motion for the design of subsystems (piping systems and components).

From its review of the DCD and discussions with the applicant, the staff finds that the procedures used for the development of in-structure response spectra at different locations, as

well as the method for peak broadening, conform to the guidance provided in Section 3.7.2.II.5 of the SRP and RG 1.122. However, to broaden the peaks by ±15 percent of the corresponding frequencies may not be adequate because of the modeling uncertainties. As discussed in Section 3.7.2.3 of this report, when the 3D seismic model was developed, the applicant did not consider the possible cracking of concrete shear walls. The concern regarding the adequacy of ±15 percent peak broadening of the FRS due to the effect of shear wall cracking is addressed in the resolution of DSER Open Item 3.7.2.3-1.

3.7.2.6 Three Components of Earthquake Motion

In DCD Tier 2, Section 3.7.2.6, "Three Components of Earthquake Motion," and during discussions in the November 12–15, 2002, and April 2–4, 2003, meetings, the applicant stated that the seismic analyses of the NI structures, using the computer code ANSYS, were performed considering the simultaneous occurrences of the two horizontal and the vertical components of the ground motion (ground-motion time history or ground response spectra). However, in the seismic analyses, the three components of earthquake motion were applied either simultaneously (time-history analyses) or separately (modal time-history analysis, equivalent static acceleration analysis, and response spectrum analysis).

In the modal time-history analyses performed for both the 3D lumped-mass stick model and the 3D FE model, the three earthquake components were applied simultaneously and the responses due to the three components of ground motion were combined within the analytical procedure at each time step using the algebraic sum method. The three components of ground motion are statistically independent and applied separately for the modal time-history analyses. The corresponding responses from the three individual analyses were combined algebraically, at each time step, to obtain the acceleration response time history. In some cases, the peak responses from the three individual analyses were combined using either the SRSS technique or the 100 percent, 40 percent, 40 percent direct combination technique. These techniques were also applied to the response spectrum and equivalent static analyses.

Axisymmetric structures, such as the steel containment vessel and the shield building roof structure, were analyzed for either one of the horizontal components and the vertical component of the seismic ground motion. Responses from different components of ground motion were combined by either the SRSS method or by the modified 100 percent, 40 percent, 40 percent method in which one component is taken at 100 percent of its maximum value and the other is taken at 40 percent of its maximum value. DCD Tier 2, Tables 3.7.2-14 and 3.7.2-16 provide a summary of applied combinations of models, analysis methods, and techniques for combining spatial effects of the earthquake ground motion.

The staff finds that the use of the algebraical sum method for the time-history analysis, and the SRSS technique for the response spectrum analysis, to combine the responses attributable to the three earthquake components meets the guidance provided in SRP Section 3.7.2, which provides that the use of the algebraical sum method for the time-history analysis and the SRSS technique for the response spectrum analysis to combine the resources due to the three earthquake components are acceptable. These methods, therefore, are acceptable. As for the suitability of using the 100 percent, 40 percent, 40 percent combination method, the applicant, during the audits performed by the staff, provided calculations to demonstrate that this combination method always gives reasonable results by comparing the results with those from

Design of Structures, Components, Equipment, and Systems

the SRSS combination method. From its review of the design calculations, the staff also finds that the difference between results obtained using the two methods was less than 5 percent which is considered insignificant and, therefore, is acceptable.

3.7.2.7 Combination of Modal Responses

In DCD Tier 2, Section 3.7.2.7, "Combination of Modal Responses," the applicant stated that when the modal superposition time-history analysis method is used to analyze the fixed-base model of the NI structures founded on a hard rock site, the total seismic responses were calculated by superimposing the modal responses (up to 200 modes to cover the high frequency effects, as discussed in Section 3.7.2.2 of this report) within the analytical procedure. Therefore, further combination is not necessary. When the response spectrum analysis method is used for the substructures or a portion of structures, the modal responses are combined using the grouping method described in Section C of Revision 1 of RG 1.92, which provides guidance to combine modal responses. If high frequency effects are significant, the responses corresponding to these high frequencies are considered using the procedure provided in Appendix A to SRP Section 3.7.2, which provides guidance to consider high frequency responses. DCD Tier 2, Table 3.7.2-16 provides a summary of the combination methods for modal responses.

The staff finds that the techniques used by the applicant for the combination of modal responses meet the guidelines of SRP Section 3.7.2 and RG 1.92 and, therefore, are acceptable.

3.7.2.8 Interaction of Seismic Category II and Nonseismic Structures with Seismic Category I Structures

SRP Section 3.7.2 specifies that the interfaces between Category I and non-Category I structures and plant equipment must be designed for the dynamic loads and displacements produced by both the Category I and non-Category I structures and plant equipment.

According to the definition provided in DCD Tier 2, Section 3.2.1, nonseismic Category I structures include seismic Category II and nonseismic structures. In DCD Tier 2, Section 3.7.2.8, the applicant classifies the structures adjacent to the NI structures in seismic categories as follows:

•	annex building: SC-II
•	turbine building: nonseismic
•	radwaste building: nonseismic

In order to ensure that the non-SC-I structures will not affect the safety functions of SC-I SSCs, the following three general approaches or interaction requirements, as stated in DCD Tier 2, Section 3.7.2.8, are used by the applicant:

•	The collapse of the non-seismic structure will not cause the nonseismic structure to strike a seismic Category I structure, system, or component.

- The collapse of the nonseismic structure will not impair the integrity of seismic Category I structures, systems, or components.

- Seismic Category II structures will be analyzed and designed to prevent their collapse under the SSE.

The staff finds that the applicant's interaction requirements specified in the DCD meet the guideline of SRP Section 3.7.2.II.8 and, therefore, are acceptable.

As shown in DCD Tier 2, Figure 1.2-2, the annex building, radwaste building, and turbine building are very close to the NI structures. The annex building is located to the east of the NI, the radwaste building is located to the south, and the turbine building is to the north. With the seismic classification of plant structures and the general approach described above, the applicant evaluated the potential and/or the consequence of the interaction of nonseismic Category I (seismic Category II and nonseismic) structures with the seismic Category I SSCs. The staff's review findings of the applicant's application of the above-mentioned general approach to the structures adjacent to the NI structures are summarized below.

3.7.2.8.1 Annex Building

The annex building is classified as seismic Category II and the minimum clearance between the structural elements of this building above grade and the NI is 10.2 cm (4 in.). As stated in DCD Tier 2, Section 3.2.1.1.2, seismic Category II SSCs are designed to prevent their collapse under the SSE and to preclude their structural failure during an SSE or interaction with SC-I items. Either of these outcomes could degrade the functioning of a safety-related SSC to an unacceptable level, or could result in an incapacitating injury to the occupants of the MCR. The applicant also states in DCD Tier 2, Section 3.7.2 that seismic Category II building structures are designed for the SSE using the same method and allowable limits as those used for seismic Category I structures. In addition, during the review meetings, the staff examined the design calculations and found that the 10.2 cm (4 in.) clearance between the annex building and the NI will prevent any interaction between these two buildings. On the basis of the above discussion, the staff finds that the interaction from the annex building to the NI will not be a concern in the event of an SSE.

3.7.2.8.2 Radwaste Building

In DCD Tier 2, Section 3.7.2.8.2, "Radwaste Building," the applicant stated that the radwaste building is classified as a nonseismic structure, and the minimum clearance between the structural elements of the radwaste building above grade and the NI is 10.2 cm (4 in.). As described in the DCD, this building structure is a small, steel-frame structure designed to the seismic criteria of the UBC, Zone 2A, with an importance factor of 1.25. Three methods were used by the applicant to demonstrate that a potential radwaste building impact on the NI during a seismic event will not impair its structural integrity. The first method demonstrated that the total kinetic energy due to the maximum radwaste building and NI velocities can be absorbed by the NI without any unacceptable damage to the NI. The second method demonstrated that the maximum compressive stress resulting from the impact of the radwaste building on the NI is less than the concrete compressive strength. The third method demonstrated that the kinetic

Design of Structures, Components, Equipment, and Systems

energy of the radwaste building is less than the kinetic energy of tornado missiles for which the NI exterior walls are designed.

The three methods used by the applicant to demonstrate that the collapse of the radwaste building will not cause any damage to the NI structures are derived based on the energy balance theory. This is consistent with standard industry practice and, therefore, is acceptable. Based on the audit conducted on April 2–7, 2003, the staff finds that these three methods were properly applied by the applicant in the evaluation of the potential for impact between the NI and the radwaste building. The staff also finds that the applicant has demonstrated that the impact from the radwaste building in the event of an SSE would not impair the integrity of the NI.

3.7.2.8.3 Turbine Building

According to the DCD, the turbine building is classified as a nonseismic structure. The major structure of the turbine building, a braced steel frame structure, is separated from the NI by approximately 5.5 m (18 ft). The roof and floors (which are also classified as nonseismic) 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 30.5 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 specified clearance and will not jeopardize the 0.61 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 floor beams and roof structure, is based on the criteria of UBC Zone 3 with an importance factor of 1. Also, in DCD Tier 2, Section 3.7.2.8, the applicant stated that for an eccentrically braced structure, the resistance modification factor is 10 using allowable stress design without considering the increase in allowable stresses by one-third for seismic loads. In addition, the design of the lateral bracing system complies with the seismic standards for eccentrically braced steel frames found in Section 9.3 of the American Institute of Steel Construction (AISC), "Seismic Provisions for Structural Steel Buildings." The staff finds that the method and criteria used for the design of the turbine building will prevent the turbine building from jeopardizing the safety function of the NI structures, during an SSE event. Therefore, this method and criteria are acceptable.

On the basis discussed above, the staff concludes that the potential for interaction between the NI and the three adjacent buildings (the annex building, radwaste building, and turbine building) is not a concern in the event of an SSE.

3.7.2.9 The Effects of Parameter Variations on Floor Response Spectra

As described in DCD Tier 2, Section 3.7.2.9, "Effects of Parameter Variations on Floor Response Spectra," the applicant has not explicitly considered the effects of parameter uncertainty. To account for such effects, the applicant, following the guidelines of SRP Section 3.7.2 and RG 1.122, broadened the peaks of the floor spectra by ±15 percent based on the corresponding spectral peak frequency. The staff found this acceptable. However, Open Item 3.7.2.3-3 (see Section 3.7.2.3 of this report) concerning the issue of stiffness reduction due to shear wall concrete cracking was not resolved. This issue was especially significant,

considering the additional uncertainties associated with structural modeling. This was Open Item 3.7.2.9-1 in the DSER.

In its June 23, 2003, submittal, the applicant addressed this open item by referring to its response to Open Item 3.7.2.3-1. As discussed in Section 3.7.2.3 of this report, the applicant reduced the member stiffness by 20 percent (the use of 80 percent of concrete Young's modulus, "E", instead of 100 percent "E") to account for the uncertainties due to shear wall concrete cracking, and to make the seismic model more realistically representative of the NI structures under an earthquake event. Also, in its submittal dated August 22, 2003, and in Revision 1 of its response to Open Item 3.7.2.9-1 dated October 10, 2003, the applicant broadened the peaks of the FRS by ±15 percent of frequency corresponding to the spectral peaks. Broadening the peaks of FRS by ±15 percent of frequency corresponding to the spectral peaks meets the guidelines of RG 1.122 and is, therefore, acceptable. The resulting FRS (with peak broadening) are to be used for the design of substructures, systems, and components. On the basis of the discussion above, Open Item 3.7.2.9-1 is resolved.

3.7.2.10 The Use of Constant Vertical Static Factors

The vertical seismic response was explicitly considered in the fixed-base seismic analyses by applying the vertical component of the ground motion simultaneously with the two horizontal components. Therefore, equivalent vertical static factors were not used to compute seismic design loads of major structures. The staff finds that this issue is not applicable to the AP1000 design.

3.7.2.11 Method Used to Account for Torsional Effects

Seismic responses of structures, such as in-plane shear in structural elements and in-structure response spectra, are typically affected by torsional effects due to eccentricities between the center of mass and center of rigidity of the structure. Based on its review of DCD Tier 2, Section 3.7.2.3, and its review of the design calculations during the November 13–15, 2002, audit, the staff finds that known eccentricities were explicitly represented in the NI lumped-mass stick model used for the seismic analyses. Also, eccentricities in the steel containment vessel that are associated with the two equipment hatches, the two personnel airlocks, and the polar crane trolley (which is to be parked at one end of the polar crane near the containment shell, as described in DCD Tier 2, Section 3.7.2.3.2) were also explicitly included in the lumped-mass stick model of the steel containment vessel.

From its review of DCD Tier 2, Section 3.7.2.11, "Method Used to Account for Torsional Effects," the staff requested that the applicant provide a clear description of the analysis procedures used to determine how the seismic loads obtained from the time-history seismic analysis of the NI stick models were applied to the equivalent static analysis of the FE models for calculating the seismic member forces to be used in the design. In its response to RAI 230.007 and the revised DCD, the applicant provided the analysis procedure and stated that in each given horizontal direction and at each given floor elevation, the maximum seismic floor acceleration and a torsional moment were applied to the FE models of the NI structures for performing the static analyses. The torsional moment applied at a given floor elevation is equal to the product of the maximum floor acceleration, the corresponding lumped mass in the stick model, and the eccentricity (equal to 10 percent of the maximum building dimensions.) One-

Design of Structures, Components, Equipment, and Systems

half of the applied torsional moment is treated to account for the effect of accidental torsion. The other half supplements the seismic torsion effect produced by the applied floor acceleration on the FE model, such that the total seismic torsion acting on the FE model matches or exceeds the seismic torsional moment in the corresponding member of the NI stick model, as determined from the seismic analysis. The structural element forces and moments due to the three components of ground motion are then combined by the SRSS technique or the 100 percent, 40 percent, 40 percent rule. As discussed in Section 3.7.2.6 of this report, the use of the SRSS technique or the 100 percent, 40 percent, 40 percent rule to combine seismic responses due to the three components of ground motion is acceptable to the staff.

On the basis discussed above, the staff concludes that the applicant has adequately included the eccentricities due to the mass and member stiffness, as well as the accidental eccentricities, in the NI lumped-mass stick model used for the seismic time-history analyses of the NI structures.

3.7.2.12 Comparison of Responses

As stated in DCD Tier 2, Section 3.7.2.1, the applicant used the modal time-history analysis method as the primary method to perform seismic analyses for the NI structures. The response spectrum analysis method was used for the analyses of SC-I components and substructures. Therefore, the applicant deleted this topic from the DCD. As discussed in Sections 3.7.2.1 and 3.7.2.2 of this report, the modal time-history analysis method is an acceptable method for the seismic analyses of the NI structures. On this basis, the staff finds the deletion of this topic from the DCD to be acceptable.

3.7.2.13 Methods of Seismic Analysis of Dams

In DCD Tier 2, Section 3.7.2.12, "Methods of Seismic Analysis of Dams," the applicant stated that seismic analysis of dams is a site-specific design. The staff agrees with this DCD statement. The applicant, in DCD Tier 2, Section 3.7.5.1, "Seismic Analysis of Dams," also stated that COL applicants referencing the AP1000 certified design will, using the site-specific SSE, evaluate the safety of existing and new dams whose failure could affect the site flood level specified in DCD Tier 2, Section 2.4.1.2, "Floods." On the basis that dams are site-specific features, as well as the rationale used for liquefaction potential evaluation described in Section 2.5 of this report, the staff finds the use of site-specific SSE for evaluating these dams to be acceptable. This is COL Action Item 3.7.2.13-1.

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 AP1000 SSCs is essentially defined at the foundation level in the free field by an SSE with a peak acceleration of 0.3 g and the ground response spectra shown in DCD Tier 2, Figures 3.7.1-1 and 3.7.1-2. The seismic design of the NI features (structures including basemat, systems, and components) is predicated on constructing the AP1000 at hard rock sites with shear wave velocity equal to 2438 m/sec (8000 fps) or higher. If these design bases are not satisfied (i.e., the site characteristics condition is not bounded by the range of site conditions specified in the DCD), or if the seismic analysis responses used for the design are not enveloped by the results obtained from actual plant site conditions other than the hard rock sites, the basis established for the design certification will no longer apply. The applicant should commit in the DCD that the COL applicants will perform an analysis and an evaluation using the design-basis earthquake ground motion and plant-specific site conditions to confirm the design adequacy of the AP1000 design.

During the audit meetings on November 12–15, 2002, and on April 2–7, 2003, the staff reviewed the seismic analysis summary report concerning the placement of the NI on a hard rock site and the seismic analysis summary report of the structural modeling. Based on these audits, the staff finds that the DCD commitments for the seismic analysis of the NI structures have been properly implemented. On this basis, the staff concluded that it is not necessary for COL applicants to perform the reconciliation analysis for sites with site parameters that bound those specified in DCD Tier 2, Table 2-1. However, the staff requested that the applicant document in the DCD the results from these seismic analysis summary reports to demonstrate that seismic Category I structures are analyzed according to the procedures described in the DCD. This was Open Item 3.7.2.16-1 in the DSER.

In response to this open item, the applicant committed, in DCD Tier 2, Sections 2.5.2.3, "Site with Geoscience Parameters Outside the Certified Design," and 2.5.4.5.6, that if the site bounding criteria are exceeded, the COL applicant will, using the specific rock site conditions (including earthquake ground motion), reanalyze the plant SSCs and reevaluate the design adequacy of these SSC. In the case of plant sites at which the design bases are not satisfied, the COL applicant will, using specific rock site conditions including earthquake ground motion, reanalyze the plant SSCs and develop a comparison to ensure that (1) the FRS from the site-specific evaluations do not exceed the design spectra at specific critical locations, (2) the site-specific seismic design forces (forces, shears, and moments) at critical locations do not exceed the design forces, and (3) the site-specific static and dynamic-bearing pressures against lateral walls and under the basemat do not exceed the design allowable. These are COL Action Items 2.6-2 and 2.6-3.

The staff finds that the applicant's commitments stated in the DCD meet Appendix C to 10 CFR Part 52. On this basis, Open Item 3.7.2.16-1 is resolved.

For sites with site parameters that do not bound those described in DCD Tier 2, Table 2-1, the applicant committed, in DCD Tier 2, Section 2.5.2.3, that site-specific SSI analyses must be performed by the COL applicant; such analyses must demonstrate the acceptability of sites with seismic and foundation characteristics that do not bound the postulated site parameters in DCD Tier 2, Table 2-1. On the basis of the discussion above, the staff's technical concerns are resolved.

Design of Structures, Components, Equipment, and Systems

3.7.2.17 Conclusions

On the basis of the above discussion, the staff concludes that the applicant has met the requirements of GDC 2 and Appendix A to 10 CFR Part 100 with respect to the capability of the AP1000 structures to withstand the effects of earthquakes by satisfying the following three requirements:

(1)   appropriate consideration for the most severe earthquake recorded for sites classified as hard rock 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 effects of natural phenomena

(3)   the importance of the safety functions to be performed (GDC 2) and the use of a suitable dynamic analysis to demonstrate that structures, systems, and components can withstand the seismic and other concurrent loads

The applicant has met the requirements of item 1 listed above by using seismic design parameters that meet the guidelines of Section 3.7.1 of the SRP. With respect to item 2, 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 of this report.

The applicant performed the structural system and subsystem analyses on an elastic and linear basis. The modal time-history analysis method forms the basis of the seismic analysis for the NI structures founded on hard rock sites and for generating in-structure response spectra. To ensure that a sufficient effective mass of structures is included in the modal time-history analyses, the applicant included 200 vibration modes, extending up to 83.8 Hz and including up to 80 percent of the mass of the NI structures. The applicant used the algebraic sum method at each time step in the analyses to account for the three components of the earthquake motion for generating the final floor time histories. The applicant also used the SRSS method to combine the maximum floor responses for the design of subsystems. The applicant's approach to perform modal time-history analyses is in conformance with SRP Section 3.7.2.

The applicant used an equivalent acceleration static analysis method to calculate seismic loads to design structural members. When the equivalent acceleration static analysis method is used, the SRSS method or 100 percent, 40 percent, 40 percent method was used to combine spatial responses in conformance with RG 1.92 and consistent with accepted common industry practice. In-structure response spectra used for the analysis and design of subsystems are generated from the time-history method and are in conformance with RG 1.122. A vertical seismic system dynamic analysis is employed for all SSCs for which the analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning, sliding, and flotation are considered.

The staff concludes that the applicant's use of the seismic structural analysis procedures and criteria delineated above provides an adequate basis for the seismic design, which is in conformance with the requirements of item 3 above.

3.7.3 Seismic Subsystem Analysis

The staff finds that subsystems defined by the applicant 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 DCD Tier 2, Sections 3.7.3.3, "Procedures Used for Modeling," and 3.7.3.1, "Seismic Analysis Methods."

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 AP1000 SC-I subsystems. In particular, this review focused on such subsystems as the miscellaneous steel platforms, steel frame structures, tanks, cable trays and supports, HVAC ductwork and supports, and conduit and supports.

Section 3.12 of this report discusses the staff's review of the applicant's analysis and design criteria for AP1000 piping systems, while Section 3.10 of this report discusses the review of AP1000 electrical and mechanical components. Sections 3.8.3 and 3.8.4 of this report discuss the staff's evaluation regarding the design of components and subsystems other than piping.

3.7.3.1 <u>Seismic Input Motion</u>

Regarding input motions for the dynamic analysis of AP1000 SC-I subsystems and components, the applicant committed to use enveloped in-structure response spectra (or FRS) generated according to the procedures described in DCD Tier 2, Section 3.7.2.5, "Development of Floor Response Spectra," or the response time histories described in DCD Tier 2, Section 3.7.2.1. When the equivalent static load method of analysis is utilized for analyzing subsystems, the applicant uses the peak floor accelerations described in DCD Tier 2, Section 3.7.2.1, as input to calculate the static forces.

The use of the enveloped FRS, the floor response time history, or the peak floor accelerations as input motions for the analysis of subsystems meets the guidelines provided in SRP Section 3.7.2.II.5 and are acceptable.

3.7.3.2 <u>Analysis Methods</u>

SRP Section 3.7.3.II.1 states that the acceptance criteria provided in SRP Section 3.7.2, Subsection II.1, are applicable for seismic analysis methods. Section 3.7.2.1 of this report provides the basis of the staff's review and acceptance of the analysis methods proposed by the applicant.

In DCD Tier 2, Section 3.7.3.1, the applicant stated that the modal response spectrum analysis method, the modal time-history analysis method, and the equivalent static analysis method are to be used for seismic analysis of the AP1000 subsystems. The following is a summary of the staff's evaluation regarding the adequacy of these analysis methods:

•   Based on its review of DCD Tier 2, Section 3.7.3 and DCD Tier 2, Table 3.7.2-16, and the discussion during the April 2–5, 2003, meeting, the staff finds that the applicant

applied the response spectrum analysis method to analyze PCS valve room, miscellaneous steel frame structures, and certain flexible floors and walls. The FRS generated from DCD Tier 2, Section 3.7.2, are used as input motion for the analyses. Use of the modal response spectrum analysis method for the analysis of AP1000 subsystems meets the guidelines prescribed in Section 3.7.2 of the SRP and is, therefore, acceptable to the staff.

• In DCD Tier 2, Section 3.7.3, the applicant stated that the time-history analysis method is used for the analysis of piping systems. The staff's evaluation of the adequacy of using this analysis method for piping system is discussed in Section 3.12 of this report.

• The equivalent static analysis method, described in DCD Tier 2, Section 3.7.3.5, "Equivalent Static Load Method of Analysis," involves the calculation of equivalent horizontal and vertical static forces applied at the center of gravity of various masses. The equivalent force at a mass location is computed as the product of the mass and the seismic acceleration value applicable to that mass location. Loads, stresses, or deflections, obtained using the equivalent static load method, are adjusted to account for the relative motion between points of support when significant.

When equivalent static loads are calculated, the applicant classifies the subsystems into either (1) subsystems that can be modeled as single-mode dominant or rigid systems, and (2) subsystems whose responses are dominated by multiple modes. The staff's evaluation of the applicant's application of the equivalent static analysis method to the AP1000 subsystems is discussed below.

For rigid subsystems, the equivalent seismic loads for the direction of excitation are calculated by multiplying the total mass of subsystems by the peak floor accelerations where these subsystems are located. For the rigid subsystems with flexible supports, the equivalent static loads for the direction of excitation are calculated by multiplying the total mass of subsystems by the spectral accelerations at corresponding frequencies from the applicable FRS. If the frequency of subsystems is not determined, the peak spectral acceleration from the applicable FRS will be used.

For subsystems with distributed mass whose dynamic response is single-mode dominant, the equivalent static seismic loads for the direction of excitation are defined as the product of the total mass of subsystems and the spectral acceleration at the natural frequency from the applicable FRS multiplied by a factor of 1.5. A factor of 1.0 will be used for subsystems that can be represented as uniformly loaded cantilever, simply supported, fixed-simply supported, or fixed-fixed supported beams. If the frequency is not determined, the peak spectral acceleration from the applicable floor response spectrum is to be used. For subsystems whose responses are dominated by multiple modes, such as cable trays and HVAC systems with multiple spans, a static load factor of 1.5 is applied to the peak spectral acceleration of the applicable FRS to obtain the seismic loads for the design.

After reviewing the DCD description and discussion with the applicant during the April 2–5, 2003, meeting, the staff finds that the applicant's use of the equivalent static analysis method to analyze the AP1000 subsystems meets the guidelines described in

Section 3.7.2 of the SRP and is consistent with common industry practice. SRP Section 3.7.2.II.1 states that the use of the equivalent static analysis method is acceptable for the analysis of seismic Category I subsystems. This is considered acceptable, except for the case involving the use of a factor of 1.0 for subsystems that can be represented as uniformly loaded cantilever, simply supported, fixed-simply supported, or fixed-fixed supported beams. The staff's view is that subsystems to be analyzed and designed based on this guideline cannot be properly categorized as single-mode dominant. To address this issue, the applicant agreed to revise DCD Tier 2, Section 3.7.3.5.1, "Single-Mode Dominant or Rigid Structures or Components," by adding the phrase, "…, if the fundamental frequency of the subsystem is higher than the frequency corresponding to the spectral peak," to the end of the appropriate DCD statement. This revision to the DCD is acceptable to the staff because this method will always provide more conservative results when the fundamental frequency of the subsystem is higher than the frequency corresponding to the spectral peak. This was Confirmatory Item 3.7.3.2-1 in the DSER.

The applicant revised DCD Tier 2, Section 3.7.3.5.1, to add the phrase, "… when the fundamental frequency is higher than the peak acceleration frequency associated with the applicable floor response spectrum," to the end of the above-mentioned sentence. On this basis, the staff considers Confirmatory Item 3.7.3.2-1 resolved.

3.7.3.3 Procedure Used for Modeling

Based on its review of the DCD, the staff finds that subsystems defined by the applicant 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). The applicant modeled and analyzed these subsystems using the approach stated in DCD Tier 2, Sections 3.7.3.3 and 3.7.3.1.

For the modeling, the applicant discretized the subsystems by concentrating the mass of the systems at distinct characteristic points or nodes, and interconnecting these masses by a network of elastic springs representing the stiffness properties of the systems. The stiffness properties are calculated either by hand calculations for simple systems, or by FE methods for more complex systems. The location of nodal points are selected in such a way to provide an adequate representation of the mass distribution and high-stress concentration points of the systems. At each nodal point, degrees of freedom (DOFs) corresponding to translations along three orthogonal axes, and the rotations about these axes, are assigned. The number of DOFs may be reduced according to the number of constraints provided. The model size is considered adequate if additional DOFs do not result in more than a 10 percent increase in response, or if the number of DOFs equals or exceeds twice the number of modes with frequencies less than 33 Hz.

When dynamic models of floor and roof slabs and miscellaneous steel platforms and framing were modeled, the applicant included masses equal to 25 percent of floor live load or 75 percent of the roof snow load, whichever is applicable.

Design of Structures, Components, Equipment, and Systems

The staff reviewed DCD Tier 2, Section 3.7.3, and other related DCD Tier 2 sections and finds that the modeling procedure used by the applicant for SC-I subsystems meets the guidelines provided in SRP Section 3.7.2.II.3, which state that the model size is considered acceptable if additional DOFs do not result in more than a 10 percent increase in response, or if the number of DOFs equals or exceeds twice the number of modes with frequencies less than 33 Hertz. On this basis, the modeling procedure is acceptable.

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 DCD Tier 2, Section 3.7.3.16, "Analysis of Seismic Category I Tanks," the applicant stated that AP1000 SC-I tanks include (1) the spent fuel pit, which is a reinforced concrete tank located in the auxiliary building, (2) the IRWST, which is an irregularly shaped, steel structural module located between the steel containment shell and the CIS, and (3) the PCS water tank, which is an axisymmetrical, reinforced concrete structure located at the top of the shield building. The AP1000 standard plant design includes no other SC-I tanks.

In the seismic analysis, the applicant modeled both the spent fuel pit and the PCS water tank, together with the NI structures; the IRWST was modeled with the CIS. Section 3.7.2 of this report discusses the staff's evaluation of the seismic input, modeling procedures, and analysis methods that the applicant applied for these tanks, while Sections 3.8.3 and 3.8.4 of this report discuss the tank design.

Based on the above discussion, the staff concludes that the design approach specified by the applicant for the design of SC-I tanks is acceptable.

3.7.3.6 Conclusions

On the basis of its review and evaluation as discussed above, the staff finds that the input motion, modeling of subsystems, selection of damping, and subsystem analysis methods discussed in DCD Tier 2, Section 3.7.3, meet the guidelines described in Sections 3.7.1 and 3.7.2 of the SRP and, therefore, are acceptable.

3.7.4 Seismic Instrumentation

RG 1.12 provides guidance for installing seismic instrumentation and the method for recording seismic data.

DCD Tier 2, Section 3.7.4.1.1, "Safety Design Basis," states the following:

> The seismic instrumentation serves no safety-related function and therefore has no nuclear safety design basis.

DCD Tier 2, Section 3.7.4.1.2, "Power Generation Design Basis," states the following:

> The seismic instrumentation is designed to provide the following:
>
> - Collection of seismic data in digital format
>
> - Analysis of seismic data after a seismic event
>
> - Operator notification that a seismic event exceeding a preset value has occurred
>
> - Operator notification (after analysis of data) that a predetermined cumulative absolute velocity value has been exceeded

3.7.4.1 Location and Description of Instrumentation

DCD Tier 2, Section 3.7.4.2, "Location and Description of Instrumentation," states the following:

> The following instrumentation and associated equipment are used to measure plant response to earthquake motion. Four triaxial acceleration sensor units, located as stated in DCD Tier 2, Section 3.7.4.2.1, are connected to a time-history analyzer. The time-history analyzer recording and playback system is located in a panel in the nuclear island in a room near the main control room. Seismic event data from these sensors are recorded on a solid-state digital recording system at 200 samples per second per data channel.
>
> This solid-state recording and analysis system has internal batteries and a charger to prevent the loss of data during a power outage, and to allow data collection and analysis in a seismic event coincident with power failure. Normally 120 volt alternating current power is supplied from the non-Class 1E dc and uninterruptible power supply system. The system uses triaxial acceleration sensor input signals to initiate the time-history analyzer recording and main control room alarms.
>
> The system initiation value is adjustable from 0.002 g to 0.02 g. The time-history analyzer starts recording triaxial acceleration data from each of the triaxial acceleration sensors after the initiation value has been exceeded. Pre-event recording time is adjustable from 1.2 to 15.0 seconds, and will be set to record at least 3 seconds of pre-event signal. Post-event run time is adjustable from 10 to 90 seconds. A minimum of 25 minutes of continuous recording is provided. Each recording channel has an associated timing mark record with 2 marks per second, with an accuracy of about 0.02 percent.
>
> The instrumentation components are qualified to [Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1987] ([DCD] Reference 16).
>
> The sensor installation anchors are rigid so that the vibratory transmissibility over the design spectra frequency range is essentially unity.

Design of Structures, Components, Equipment, and Systems

3.7.4.1.1 Triaxial Acceleration Sensors

DCD Tier 2, Section 3.7.4.2.1, "Triaxial Acceleration Sensors," states the following:

> Each sensor unit contains three accelerometers mounted in a mutually
> orthogonal array mounted with one horizontal axis parallel to the major axis
> assumed in the seismic analysis. The triaxial acceleration sensors have a
> dynamic range of 1000 to 1 (0.001 to 1.0 g) and a frequency range of 0.2 to
> 50 hertz. One sensor unit will be located in the free field. Because this location
> is site-specific, the planned location will be determined by the combined license
> applicant. The AP1000 seismic monitoring system will provide for signal input
> from the free field sensor. A second sensor unit is located on the nuclear island
> basemat in the spare battery charger room at Elevation 66'-6" near column
> lines 9 and L. A third sensor unit is located on the shield building structure at
> Elevation 266' near column lines 4-1 and K. The fourth sensor unit is located on
> the containment internal structure on the east wall of the east steam generator
> compartment just above the operating floor at Elevation 138' close to column
> lines 6 and K. Seismic instrumentation is not located on equipment, piping, or
> supports since experience has shown that data obtained at these locations are
> obscured by vibratory motion associated with normal plant operation.

3.7.4.1.2 Time-History Analyzer

DCD Tier 2, Section 3.7.4.2.2, "Time-History Analyzer," states the following:

> The time-history analyzer receives input from the triaxial acceleration sensors
> and, when activated as described in [DCD Tier 2, Section] 3.7.4.3, begins
> recording the triaxial data from each triaxial acceleration sensor and initiates
> audio and visual alarms in the main control room. This recorded data will be
> used to evaluate the seismic acceleration of the structure on which the triaxial
> acceleration sensors are mounted.
>
> The time-history analyzer is a multichannel, digital recording system with the
> capability to automatically download the recorded acceleration data to a
> dedicated computer for data storage, playback, and analysis after a seismic
> event. The time-history analyzer can compute cumulative absolute velocity
> (CAV) and the 5 percent of critical damping response spectrum for frequencies
> between 1 and 10 Hz. The operator may select the analysis of either CAV or the
> response spectrum. Analysis results are printed out on a dedicated graphics
> printer that is part of the system and is located in the same panel as the time-
> history analyzer.

3.7.4.2 Control Room Operator Notification

DCD Tier 2, Section 3.7.4.3, "Control Room Operator Notification," states the following:

> The time-history analyzer provides for initiation of audible and visual alarms in
> the main control room when predetermined seismic acceleration values sensed

by any of the triaxial acceleration sensors are exceeded and when the system is activated to record a seismic event. In addition to alarming when the system is activated, the analyzer portion of the system will provide a second alarm if the predetermined cumulative absolute velocity value has been exceeded by any of the sensors. Alarms are annunciated in the main control room.

3.7.4.3 Comparison of Measured and Predicted Responses

DCD Tier 2, Section 3.7.4.4, "Comparison of Measured and Predicted Responses," states the following:

> The recorded seismic data is used by the COL holder operations and engineering departments to evaluate the effects of the earthquake on the plant structures and equipment. The criterion for initiating a plant shutdown following a seismic event will be exceedance of a specified response spectrum limit or a cumulative absolute velocity limit. The seismic instrumentation system is capable of computing the cumulative absolute velocity as described in EPRI Report NP-5930 ["A Criterion for Determining Exceedance of the Operating Basis Earthquake"] ([DCD] Reference 1) and EPRI Report TR-100082 ["Standardization of the Cumulative Absolute Velocity"] ([DCD] Reference 17).

3.7.4.4 Tests and Inspections

DCD Tier 2, Section 3.7.4.5, "Test and Inspections," states the following:

> Periodic testing of the seismic instrumentation system is accomplished by the functional test feature included in the software of the time-history recording accelerograph. The system is modular and is capable of single-channel testing or single channel maintenance without disabling the remainder of the system.

3.7.4.5 Conclusions

Based on its review of DCD Tier 2, Section 3.7.4, the staff finds that the instrumentation type, locations, specified characteristics, capability for notifying operators, and procedures for testing and inspections described meet the guidelines of RG 1.12, and, therefore, are acceptable.

3.7.5 Other Combined License Action Items

The following action for the COL applicant applies to Section 3.7 of this report.

- DCD Tier 2, Section 3.7.5.4, "Reconciliation of Seismic Analyses of Nuclear Island Structures," states the following:

> The Combined License applicant will reconcile the seismic analyses described in [DCD Tier 2, Section] 3.7.2 for detail design changes at rock sites such as those due to as-procured equipment information. Deviations are acceptable based on an evaluation consistent with the methods and procedure of [DCD Tier 2, Section] 3.7 provided the amplitude of the seismic floor response spectra

Design of Structures, Components, Equipment, and Systems

> including the effect due to these deviations, do not exceed the design basis floor response spectra by more than 10 percent.

This is COL Action Item 3.7.5-1.

The following sections of DCD Tier 2 include combined license information items in which the staff has determined not be applicable to the design certification review. These items are repeated below.

- DCD Tier 2, Section 3.7.5.2, "Post-Earthquake Procedures," states the following:

> Combined License applicants referencing the AP1000 certified design will prepare site-specific procedures for activities following an earthquake. These procedures will be used to accurately determine both the response spectrum and the cumulative absolute velocity of the recorded earthquake ground motion from the seismic instrumentation system. The procedures and the data from the seismic instrumentation system will provide sufficient information to guide the operator on a timely basis to determine if the level of earthquake ground motion requiring shutdown has been exceeded. The procedures will follow the guidance of EPRI Reports NP-5930 ([DCD] Reference 1), TR-100082 ([DCD] Reference 17), and NP-6695 ["Guidelines for Nuclear Plant Response to an Earthquake,"], as modified by the NRC staff [NRC Letter from James T. Wiggins to John J. Taylor, September 13, 1993] ([DCD] Reference 32).

This is COL Action Item 3.7.5-2.

- DCD Tier 2, Section 3.7.5.3, "Seismic Interaction Review," states the following:

> The seismic interaction review will be updated by the Combined License applicant. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition.

This is COL Action Item 3.7.5-3

- DCD Tier 2, Section 3.7.5.5, "Free Field Acceleration Sensor," states the following:

> The Combined License applicant will determine the location for the free-field acceleration sensor as described in [DCD Tier 2, Section] 3.7.4.2.1

This is COL Action Item 3.7.5-4.

3.8 Design of Category I Structures

DCD Tier 2, Section 1.5, "Requirement for Further Technical Information," states the following:

> Tests were conducted during the AP600 Conceptual Design Program (1986 through 1989) to provide input for plant design and to demonstrate the feasibility

of unique design features. Tests for the AP600 design certification and design program were devised to provide input for the final safety analyses, to verify the safety analysis models (computer codes), and to provide data for final design and verification of plant components. An AP1000 specific Phenomena Identification and Ranking Table (PIRT) and scaling analysis ([DCD] Reference 25) and a review of a safety analysis evaluations of AP1000 [Chapter 15 of this DCD] show that AP600 and AP1000 exhibit a similar range of conditions for the events analyzed. This provides justification that the database of test information generated during the AP600 Conceptual Design Program is sufficient to meet the requirements of 10 CFR Part 52 for AP1000. [DCD Tier 2,] Table 1.5-1 is a list of the AP600 tests and AP1000 evaluations with references to test and evaluation documentation. Note that [DCD] Reference 25 reviews each of the AP600 tests described and assesses their applicability to AP1000. The evaluations of [DCD] Reference 25 show that the AP600 tests are sufficient to support AP1000 safety analysis.

Reference 25 in the above quote refers to WCAP-15613. The staff's review of the AP1000 DCD Tier 2 found that the applicant had not provided a sufficient detailed technical bases for the applicability of the AP600 test results to the definition of design loads for the AP1000 steel containment vessel, CIS, and other Category I structures. In RAI 220.001, the staff asked the applicant to provide additional information to address this concern.

In Revision 1 of its response to RAI 220.001, the applicant described the technical basis for the applicability of three specific AP600 tests that were used to define the design loads for the AP1000 structures. The three tests utilized were (1) the AP600 PCS water distribution test, (2) the AP600 automatic depressurization system hydraulic test, and (3) the AP600 wind tunnel test. Based on similarities of design and operational parameters between the AP600 and AP1000, and an assessment that any differences would have only a small influence on the test results, the applicant concluded that these AP600 test results are applicable to AP1000. The staff also reviewed WCAP-15613 during the April 2–5, 2003, audit to verify the applicant's conclusion.

Based on the information provided in the RAI response, which states that the temperature differences between wet and dry regions of the AP1000 during activation of passive containment cooling are bounded by the temperature differences evaluated for the AP600, and the staff's review of WCAP-15613 to verify the validity of the applicant's justification, the staff accepts the applicability of these three AP600 tests to design load definition for the AP1000 structures.

3.8.1 Concrete Containment

This section is not applicable to the AP1000 design.

3.8.2 Steel Containment

SRP Section 3.8.2 provides guidelines for the staff to use in reviewing the technical areas related to the design of a containment vessel based on the requirements of GDC 1, 2, 4, 16, 50, 51, and 53. These technical areas include a description of the containment; applicable

Design of Structures, Components, Equipment, and Systems

codes, standards, and specifications; loads and load combinations; design and analysis procedures; structural acceptance criteria; materials, quality control, and special construction techniques; and testing and inservice surveillance requirements. Using the guidance described in Section 3.8.2 of the SRP, the staff reviewed DCD Tier 2, Section 3.8.2, "Steel Containment." In particular, the review described in this section focused on the analysis and design of the AP1000 steel containment vessel shell structure, including its (1) material, (2) geometry, (3) codes and standards, (4) loadings, and (5) design and analysis procedures.

3.8.2.1 Description of the Containment

3.8.2.1.1 General

DCD Tier 2, Section 3.8.2.1.1, "General," states the following:

> This subsection describes the structural design of the steel containment vessel and its parts and appurtenances. The steel containment vessel is an integral part of the containment system whose function is described in [DCD Tier 2,] Section 6.2. It serves to both limit releases in the event of an accident and to provide the safety-related ultimate heat sink.
>
> The containment vessel is an ASME metal containment. The information contained in this subsection is based on the design specification and preliminary design and analyses of the vessel.

During the April 2–5, 2003, audit, the applicant informed the staff that the final detailed analyses, to be documented in the ASME design report, were not available and will be the responsibility of the COL applicant. The staff expected that the final detailed analyses for the AP1000 steel containment would be submitted for staff review as part of the design certification process. To complete its evaluation of the AP1000 steel containment design, the staff informed the applicant it would be necessary to audit the final detailed analyses. This was Open Item 3.8.2.1-1 in the DSER.

In a letter dated July 8, 2003, in response to this open item, the applicant cited its letter dated May 1, 2003, for identification of the additional detailed analyses to be performed for the containment vessel. The applicant further stated, "These analyses are available for NRC staff review and demonstrate that the AP1000 containment vessel satisfies the acceptance criteria documented in the DCD."

Based on the review of the May 1, 2003, letter, it was not apparent to the staff that the additional analyses would be sufficient to evaluate the adequacy of the steel containment vessel design for the load combinations and acceptance criteria documented in the DCD. The applicant made revisions to the load combinations and acceptance criteria for the containment vessel that represent a departure from those previously accepted by the staff for the AP600. These revisions had not been fully evaluated by the staff. In addition, evaluating only those load combinations that were controlling for the AP600 would need to be technically justified. Due to design, material, and loading changes, it is conceivable that a different load combination may be controlling for the AP1000.

During the October 6–9, 2003, audit, the applicant made a presentation to the staff describing the analyses performed for the AP1000 steel containment vessel and the results obtained. These analysis were performed by the applicant's contractor, Chicago Bridge & Iron Company (CBI). The staff reviewed six design calculations presented by the applicant and found that the analysis methods were consistent with industry practice, and the results were within allowable code limits for the cases considered in the applicant's evaluation. However, the following four issues were identified that needed to be addressed by the applicant before the staff could conclude that the design of the steel containment vessel is acceptable:

(1)   The CBI evaluation does not consider all loadings on the containment shell. Only design internal pressure, dead load, polar crane loads, seismic loads, and design external pressure are considered in the four load combinations evaluated. The contribution of other loads (such as thermal loads) included in the load combinations need to be addressed before the staff can conclude that the design is acceptable for these four load combinations.

(2)   The CBI evaluation does not consider all load combinations. A technical basis is needed to justify that the evaluated load combinations are the limiting load combinations for the steel containment vessel design.

(3)   The CBI evaluation of potential containment shell buckling assumed certain containment vessel dimensions that are not designated as Tier 2*. These include the moment of inertia of the T-section ring stiffeners, the axial spacing between stiffeners, and the axial dimension and minimum moment of inertia of the crane support ring girder. All dimensions that are critical for demonstrating adequate buckling capacity need to be designated Tier 2*.

(4)   The CBI evaluation is based on an earlier set of seismic loads that have been revised, and in most cases increased, for the final AP1000 design calculations. In Calculation APP-1000-S3R-001, Revision 0, entitled "Reconciliation of Critical Sections to Revision 3 Seismic Spectra," the applicant attempted to address this by a simple ratio approach. On the basis that several containment shell design criteria are barely satisfied, and the seismic loads increase by as much as 30 percent, the evaluation approach based on a load scaling factor is not acceptable to the staff. A more detailed quantitative basis is needed to demonstrate design adequacy for the final seismic loads.

In Revision 2 of its response to Open Item 3.8.2.1-1 dated November 17, 2003, the applicant indicated that the four issues identified above were addressed by revising the Containment Vessel Design Specification, revising the design verification calculations, and designating the maximum allowable stiffener spacing as Tier 2* in the DCD.

For the resolution of issue 3 above, the staff reviewed the revision to DCD Tier 2, Section 3.8.2 and verified that the applicant has specified the maximum allowable stiffener spacing as Tier 2*. This issue is resolved.

To ensure that the applicant had properly resolved issues 1, 2, and 4 above, the staff reviewed the revised Containment Vessel Design Specification, the revised design verification

Design of Structures, Components, Equipment, and Systems

calculations, and the following documents and calculations during the December 15–16, 2003, audit:

- Containment Vessel Design Specification, APP-MV50-Z0-001, Revision 1

- Stress Evaluation for Load Combinations, APP-MV50-S2C-006, Revision 1 (prepared by Chicago Bridge & Iron Company)

- Independent Verification of Containment Vessel Stability Analysis, APP-1100-S2C-101, Revision 1

- Reconciliation of Critical Sections to Revision 3 Seismic Spectra, APP-1000-S3R-001, Revision 1

- Containment Vessel Shear Studs, APP-1100-S2C-102, Revision 0

As indicated by the applicant, these documents and the supporting calculations performed constitute the technical basis for the AP1000 containment vessel design.

The staff reviewed the "Containment Vessel Design Specification (APP-MV50-Z0-001, Revision 1)" and "Stress Evaluation for Load Combinations (APP-MV50-S2C-006, Revision 1)" and found that the final seismic loads have been incorporated into the design specification and the stress/buckling calculations. The staff also reviewed the document, "Reconciliation of Critical Sections with Revision 3 Seismic Spectra (APP-1000-S3R-001, Revision 1)," and found that the final seismic loads have been adequately considered. In addition, the staff verified that the stress evaluation has been properly updated, and that the design meets the stress/buckling limits for the load combinations evaluated. This resolves issue 4 above.

For the resolution of issues 1 and 2, the applicant, in the document, "Stress Evaluation for Load Combinations (APP-MV50-S2C-006, Revision 1)," included a new section entitled "Explanation of the Selection of Loads and Loading Combinations for AP1000." In this new section, the applicant discussed a qualitative technical basis for not evaluating all of the loads and load combinations detailed in the AP1000 Design Specification. Based on a comparison of the results obtained from the evaluation for the AP1000 steel containment vessel with the results of the detailed stress calculations for the AP600, the applicant concluded that those loads and load combinations not evaluated for the AP1000 steel containment vessel will not control the design of the containment vessel. The staff finds this to be acceptable for the verification of the containment vessel design adequacy. On this basis, issues 1 and 2 are resolved.

In addition to the four issues discussed above, the staff requested the applicant to provide the technical basis for using Service Level D allowable stress, instead of Service Level C allowable stress, for the load combination of seismic loads plus design external pressure when the evaluation of the containment vessel adequacy was performed. During the audit conducted on October 6–9, 2003, the applicant presented an evaluation based on the load combination, assuming that these two events occur simultaneously. In its submittal dated December 12, 2003 (Revision 3 of the response to Open Item 3.8.2.1-1), the applicant provided a final calculation that justifies the change of design basis from Service Level C to Service Level D.

Based on its review of these documents and the discussion with the applicant, the staff found that the change from Service Level C to Service Level D for the load combination of seismic plus design external pressure is technically justified because of the extremely low sequence frequency (less than 1E-10 per year) leading to containment failure.

Because the applicant has adequately addressed the four issues and provided reasonable bases for the change in service level, the staff considers Open Item 3.8.2.1-1 resolved. According to the applicant, the ASME design report will be completed by the COL applicant and the COL applicant will document the final results for all loads and load combinations specified in the AP1000 Containment Vessel Design Specification. This is COL Action Item 3.8.2.4.1.2-1.

In addition, the applicant presented an evaluation of containment vessel design to ensure that the containment vessel/containment internal structures will not lift off from the NI basemat during the margin level earthquake (0.5 g PGA). As a result, shear studs were added to the external surface of the lower vessel head to provide positive anchoring to the NI basemat. The description of the shear studs and the preliminary design calculations are presented in the document, "Containment Vessel Shear Studs (APP-1100-S2C-102, Revision 0)." From its review of this calculation and the discussion with the applicant during the audit, the staff found that the shear stud size and pattern developed by the applicant was designed to provide adequate resistance against liftoff, while limiting local stresses in the containment shell within the design allowable. Additional evaluation of the dynamic stability of the containment vessel against the margin level earthquake is discussed in Section 19A of this report. On this basis, the staff concludes that the applicant properly demonstrated the design adequacy and the dynamic stability of the steel containment vessel.

DCD Tier 2, Section 3.8.2.1.1, further states:

> The containment arrangement is indicated in the general arrangement figures in [DCD Tier 2,] Section 1.2. The portion of the vessel above Elevation 132'-3" is surrounded by the shield building but is exposed to ambient conditions as a part of the passive cooling flow path. A flexible watertight and airtight seal is provided at Elevation 132'-3" between the containment vessel and the shield building. The portion of the vessel below Elevation 132'-3" is fully enclosed within the shield building.
>
> [DCD Tier 2,] Figure 3.8.2-1 shows the containment vessel outline, including the plate configuration and crane girder. It is a free-standing, cylindrical steel vessel with ellipsoidal upper and lower heads. [*The containment vessel has the following design characteristics*:
>
> *Diameter: 130 ft* [39.62 m]
> *Height: 215 feet 4 in.* [65.63 m]
> *Design Code: ASME III, Div. 1*
> *Material: SA738 Grade B*
> *Design Pressure: 59 psig* [406.8 kPa]
> *Design Temperature: 300 °F* [148.9 °C]
> *Design External Pressure: 2.9 psid* [20 kPa]

Design of Structures, Components, Equipment, and Systems

The wall thickness in most of the cylindrical is [4.44 cm (*1.75 in.*)]. *The wall thickness of the lowest course of the cylindrical shell is increased to* [4.76 cm (*1.875 in.*)] *to provide margins in the event of corrosion in the embedded transition region. The thickness of the heads is* [4.13 cm (*1.625 in.*)].]* The heads are ellipsoidal with a major diameter of [39.62 m (130 ft)] and a height of [11.47 m (37 ft, 7.5 in.)].

The containment vessel includes the shell, hoop stiffeners and crane girder, equipment hatches, personnel airlocks, penetration assemblies, and miscellaneous appurtenances and attachments...

...The polar crane is designed for handling the reactor vessel head during normal refueling. The crane girder and wheel assemblies are designed to support a special trolley to be installed in the event of steam generator replacement.

The containment vessel supports most of the containment air baffle as described in DCD Tier 2, Section 3.8.4. The air baffle is arranged to permit inspection of the exterior surface of the containment vessel. Steel plates are welded to the dome as part of the water distribution system, described in [DCD Tier 2, Section] 6.2.2. The polar crane system is described in [DCD Tier 2, Section] 9.1.5.

The staff's review of the containment shell design identified a concern that the 4.44 cm (1.75 in.) thickness of the cylindrical shell just meets the minimum thickness requirement of 4.4336 cm (1.7455 in.) of the 1998 ASME Code, Section III, Subsection NE, Paragraph NE-3324.3(a), based on a 406.8 kPa (59 psi) design pressure, a 148.9 °C (300 °F) design temperature, allowable stress, S = 182 MPa (26.4 ksi), and a containment vessel radius, R = 1981.2 cm (780 in.). The staff noted that there is no margin in the nominal design thickness for corrosion allowance. Of particular concern is the embedment transition region of the cylinder, which has been prone to corrosion in operating plants. Paragraph NE-3121 specifically requires that the need for a corrosion allowance be evaluated. Consequently, the staff requested the applicant to provide justification for (1) making no provision, in defining the nominal design thickness, for general corrosion of the containment shell over its 60-year design life, and (2) not specifying a corrosion allowance in the embedment transition region. In its response to RAI 220.002 (Revision 1), the applicant submitted the following information to address the corrosion allowance for the AP1000 containment shell:

- The ASME Code of record has been updated to the 2001 Edition including 2002 Addenda. (The applicant has revised the DCD to incorporate this change.) Per the revised Code of record, S = 184.09 MPa (26.7 ksi) and t_{min} = 4.38 cm (1.726 in.), which provides a nominal margin for corrosion of 0.06 cm (0.024 in.).

- The design has been changed to add a corrosion allowance for the embedment transition region, as was provided for the AP600. The nominal thickness of the bottom cylinder section is increased to 4.76225 cm (1.875 in.) and the vertical weld joints in the first course will be post-weld, heat-treated per ASME Code requirements.

- Corrosion protection has been identified as a safety-related function for the containment vessel coating in DCD Tier 2, Section 6.1.2.1.1, "General (Protection Coatings)."   The COL applicant will provide a program to monitor the coatings, as described in DCD Tier 2, Section 6.1.3.2, "Coating Program."

On the basis that enough corrosion allowance and proper corrosion protection were provided, the staff found the applicant's response acceptable, pending (1) incorporation of the design change in the cylinder embedment transition region in a future revision, and (2) designation of the "inhibit corrosion" function as "safety" for coatings on the outside surface of the containment vessel in a future revision of DCD Tier 2, Table 6.1-2. This was Confirmatory Item 3.8.2.1-1 in the DSER.

The applicant incorporated the design change in the cylinder embedment transition region in DCD Tier 2, Section 3.8.2.1.1 and designated the steel containment vessel thickness as Tier 2*. The applicant also designated the "inhibit corrosion" function as "safety" for coatings on the outside surface of the containment vessel in a revision to DCD Tier 2, Table 6.1-2. On this basis, Confirmatory Item 3.8.2.1-1 is resolved.

3.8.2.1.2 Containment Vessel Support

DCD Tier 2, Section 3.8.2.1.2, "Containment Vessel Support," states the following:

> The bottom head is embedded in concrete, with concrete up to Elevation 100' on the outside and to the maintenance floor at Elevation 107'-2" on the inside. The containment vessel is assumed as an independent, free-standing structure above Elevation 100'. The thickness of the lower head is the same as that of the upper head. There is no reduction in shell thickness even though credit could be taken for the concrete encasement of the lower head.

> Vertical and lateral loads on the containment vessel and internal structures are transferred to the basemat below the vessel by friction and bearing. The shear studs are not required for design basis loads. They provide additional margin for earthquakes beyond the safe shutdown earthquake.

> Seals are provided at the top of the concrete on the inside and outside of the vessel to prevent moisture between the vessel and concrete. A typical cross section design of the seal is presented in [DCD Tier 2,] Figure 3.8.2-8, sheets 1 and 2.

3.8.2.1.3 Equipment Hatches

DCD Tier 2, Section 3.8.2.1.3, "Equipment Hatches," states the following:

> Two equipment hatches are provided. One is at the operating floor (Elevation 135'-3") with an inside diameter of [4.87 m (16 ft)]. The other is at floor Elevation 107'-2" to permit grade-level access into the containment, with an inside diameter of [4.87 m (16 ft)]. The hatches, shown in [DCD Tier 2,] Figure 3.8.2-2, consist of a cylindrical sleeve with a pressure seated dished head bolted on the inside of the vessel. The containment internal pressure acts on the

Design of Structures, Components, Equipment, and Systems

convex face of the dished head and the head is in compression. The flanged joint has double O-ring or gum-drop seals with an annular space that may be pressurized for leak testing the seals. Each of the two equipment hatches is provided with an electrically powered hoist and with a set of hardware, tools, equipment and a self-contained power source for moving the hatch from its storage location and installing it in the opening.

3.8.2.1.4 Personnel Airlocks

DCD Tier 2, Section 3.8.2.1.4, "Personnel Airlocks," states the following:

Two personnel airlocks are provided, one located adjacent to each of the equipment hatches. [DCD Tier 2,] Figure 3.8.2-3 shows the typical arrangement. Each personnel airlock has about a [3.3 m (10 ft)] external diameter to accommodate a door opening of width [1.07 m (3 ft - 6 in.)] and a height of [2.03 m (6 ft - 8 in.)]. The airlocks are long enough to provide a clear distance of [2.44 m (8 ft)], which is not impaired by the swing of the doors within the lock. The airlocks extend radially out from the containment vessel through the shield building. They are supported by the containment vessel.

Each airlock has two double-gasketed, pressure-seated doors in series. The doors are mechanically interlocked to prevent simultaneous opening of both doors and to allow one door to be completely closed before the second door can be opened. The interlock can be bypassed by using special tools and procedures.

3.8.2.1.5 Mechanical Penetrations

DCD Tier 2, Section 3.8.2.1.5, "Mechanical Penetrations," states the following:

The mechanical penetrations consist of the fuel transfer penetration and mechanical piping penetrations that are listed in [DCD Tier 2,] Table 6.2.3-4.

[DCD Tier 2,] Figure 3.8.2-4, Sheet 1, shows typical details for the main steam penetration. This includes bellows to minimize piping loads applied to the containment vessel and a guard pipe to protect the bellows and to prevent overpressurization of the containment annulus in a postulated pipe rupture event. Similar details are used for the feedwater penetration.

[DCD Tier 2,] Figure 3.8.2-4, Sheet 2, shows typical details for the startup feedwater penetration. This includes a guard pipe to prevent overpressurization of the containment annulus in a postulated pipe rupture event. Similar details are used for the steam generator blowdown penetration.

[DCD Tier 2,] Figure 3.8.2-4, Sheet 3, shows typical details for the normal residual heat removal penetration. Similar details are used for other penetrations below Elevation 107'-2" where there is concrete inside the containment vessel. The flued head is integral with the process piping and is welded to the

containment sleeve. The welds are accessible for inservice inspection. The containment sleeve is separated from the concrete by compressible material. [DCD Tier 2,] Figure 3.8.2-4, Sheet 4 shows typical details for the other mechanical penetrations. These consist of a sleeve welded to containment with either a flued head welded to the sleeve (detail A), or with the process piping welded directly to the sleeve (detail B). Flued heads are used for stainless piping greater than [5.04 cm (2 in.)] in nominal diameter and for piping with high operating temperatures.

Design criteria for the mechanical penetrations are as follows:

- Design and construction of the process piping follow ASME [Code], Section III, Subsection NC. Design and construction of the remaining portions follow ASME Code, Section III, Subsection NE. The boundary of jurisdiction is according to ASME Code, Section III, Subsection NE.

- Penetrations are designed to maintain containment integrity under design-basis accident conditions, including pressure, temperature, and radiation.

- Guard pipes are designed for pipe ruptures as described in [DCD Tier 2, Section] 3.6.2.1.1.4.

- Bellows are stainless steel or nickel alloy and are designed to accommodate axial and lateral displacements between the piping and the containment vessel. These displacements include thermal growth of the main steam and feedwater piping during plant operation, relative seismic movements, and containment accident and testing conditions. Cover plates are provided to protect the bellows from foreign objects during construction and operation. These cover plates are removable to permit inservice inspection.

The fuel transfer penetration, shown in [DCD Tier 2,] Figure 3.8.2-4, Sheet 5, is provided to transfer fuel between the containment and the fuel handling area of the auxiliary building. The fuel transfer tube is welded to the penetration sleeve. The containment boundary is a double-gasketed blind flange at the refueling canal end. The expansion bellows are not a part of the containment boundary. Rather, they are water seals during refueling operations and accommodate differential movement between the containment vessel, containment internal structures, and the auxiliary building.

3.8.2.1.6 Electrical Penetrations

DCD Tier 2, Section 3.8.2.1.6, "Electrical Penetrations," states the following:

[DCD Tier 2,] Figure 3.8.2-4, Sheet 6, shows a typical [30.5 cm (12 in.)] diameter electrical penetration. The penetration assemblies consist of three modules (or six modules in a similar [45.7 cm (18 in.)] diameter penetration) passing through

Design of Structures, Components, Equipment, and Systems

> a bulkhead attached to the containment nozzle. Electrical design of these
> penetrations is described in [DCD Tier 2, Section] 8.3.1.1.5.
>
> Electrical penetrations are designed to maintain containment integrity under
> design-basis accident conditions, including pressure, temperature, and radiation.
> Double barriers permit testing of each assembly to verify that containment
> integrity is maintained...

3.8.2.1.7 Evaluation

Because the descriptive information and referenced figures in DCD Tier 2, Section 3.8.2.1,
"Description of the Containment," contain sufficient detail to define the primary structural
aspects and elements relied upon for the structure to perform its safety-related function, in
accordance with SRP Section 3.8.2, the staff finds the descriptive information to be acceptable.

3.8.2.2 Applicable Codes, Standards, and Specifications

The containment vessel was initially designed according to the 1998 edition of the ASME Code,
Section III, Subsection NE, "Metal Containment," including the 1999 and 2000 addenda. The
selected shell material is SA738, Grade B, which was not listed for containment vessels in the
2000 addenda. The material was approved for containment vessel applications by Code
Case N655 in February 2002.

During its initial review, the staff raised concerns related to the selection of the containment
material and the allowable stress criteria being applied to the design. The staff requested that
the applicant provide its justification for adopting allowable stress values for SA738 Grade B
material which, at the time, were not included in the ASME Code for Class MC components.

In its response to RAI 220.003 (Revision 1), the applicant (1) identified the 2002 addenda to
ASME Code, Section III, Subsection NE, as the new containment design basis, (2) provided
technical background information in support of the allowable stress criteria adopted in the 2002
addenda to Subsection NE, and (3) noted the acceptance of SA738 Grade B material for
Class MC applications. The staff finds the applicant's justification sufficient to address the
concern identified by the staff because 10 CFR 50.55a incorporates by reference the 2002
addenda to ASME Code, Section III, Subsection NE.

In DCD Tier 2, Section 3.8.2.2, "Applicable Codes, Standards, and Specifications," the applicant
changed the Code of record for the AP1000 containment vessel as follows,

> *[The containment vessel is designed and constructed according to the 2001 edition of
> the ASME Code, Section III, Subsection NE, Metal Containment, including the 2002
> addenda...]**

In addition, in its response to RAI 220.003 (Revision 1), the applicant submitted its technical
basis for concluding that SA738 Grade B material has adequate fracture toughness to meet the
requirements of NE-2000, in the as-welded condition, for thicknesses up to and including
4.44 cm (1.75 in.). After review of the response, the staff concluded that the applicant had not
provided sufficient quantitative data, and had made confusing statements concerning the

effects of post-weld heat treatment (PWHT). The applicant addressed this issue during the April 2–5, 2003, audit by presenting directly applicable test data compiled by CBI. The staff found the information provided by the applicant to be acceptable because the test data indicated that SA738 Grade B steel has adequate fracture toughness to meet the NE-2000 requirements (with the as-welded condition). The applicant committed to revise its response to RAI 220.003 to include the information presented at the audit. This was Confirmatory Item 3.8.2.2-1 in the DSER.

In Revision 2 of the response to RAI 220.003 dated April 23, 2003, the applicant provided the above-mentioned CBI test data for the staff's review. Therefore, Confirmatory Item 3.8.2.2-1 is resolved.

Stability of the AP1000 containment vessel and appurtenances is evaluated using ASME Code Case N-284-1, "Metal Containment Shell Buckling Design Methods," Class MC, Section III, Division 1, as published in the 2001 Code Cases, 2001 edition, July 1, 2001. Since the latest version of Code Case N-284-1 has not been endorsed by the NRC, the staff in RAI 220.004, requested the applicant to "provide its technical justification for the acceptability of this code case by demonstrating an equivalent level of safety when compared to Code Case N-284, Revision 0 plus the supplemental requirements of AP600 DCD Appendix 3G." In its response to RAI 220.004 (Revision 0), the applicant confirmed that the AP1000 criteria are the same as the AP600 criteria previously accepted by the staff. Because the applicant has demonstrated that the criteria of Code Case N-284-1 are consistent with the staff's position for the evaluation of the steel containment buckling documented in Appendix 3G to the AP600 DCD, the staff finds these criteria to be acceptable. However, the applicant did not designate Code Case N-284-1 as Tier 2* material, requiring any proposed change to these criteria to have NRC approval prior to implementation of the change. The staff notes that in the AP600 DCD, Appendix 3G is designated Tier 2*. This was Open Item 3.8.2.2-2 in the DSER.

In DCD Tier 2, Section 3.8.2.2, the applicant designated ASME Code Case N-284-1 as Tier 2* information. On this basis, the staff concludes that Open Item 3.8.2.2-2 is resolved.

3.8.2.3 Loads and Load Combinations

DCD Tier 2, Section 3.8.2.3, "Loads and Load Combinations," states the following:

> [DCD Tier 2,] Table 3.8.2-1 summarizes the design loads, load combinations and ASME Service Levels. They meet the requirements of the ASME Code, Section III, Subsection NE. The containment vessel is designed for the following loads specified during construction, test, normal plant operation and shutdown, and during accident conditions:
>
> D Dead loads or their related internal moments and forces, including any permanent piping and equipment loads
>
> L Live loads or their related internal moments and forces, including crane loads

Design of Structures, Components, Equipment, and Systems

P_o Operating pressure loads during normal operating conditions resulting from pressure variations either inside or outside containment

T_o Thermal effects and loads during normal operating conditions, based on the most critical transient or steady-state condition

R_o Piping and equipment reactions during normal operating conditions, based on the most critical transient or steady-state condition

W   Loads generated by the design wind on the portion of the containment vessel above Elevation 132', as described in [DCD Tier 2, Section] 3.3.1.1

E_s Loads generated by the safe-shutdown earthquake (SSE) as described in [DCD Tier 2,] Section 3.7

W_t Loads generated by the design tornado on the portion of the containment vessel above Elevation 132', as described in [DCD Tier 2, Section] 3.3.2

P_t Test pressure

P_d Containment vessel design pressure that exceeds the pressure load generated by the postulated pipe break accidents and passive cooling function

P_e Containment vessel external pressure

T_a Thermal loads under thermal conditions generated by the postulated break or passive cooling function including T_0. This includes variations around the shell due to the surrounding buildings and unpredicted distribution of the passive containment cooling system water.

R_a Piping and equipment reactions under thermal conditions generated by the postulated break, as described in [DCD Tier 2,] Section 3.6, and including R_o

Y_r Loads generated by the reaction on the broken high-energy pipe during the postulated break, as described in [DCD Tier 2,] Section 3.6

Y_j Jet impingement load on a structure generated by postulated breaks, as described in [DCD Tier 2,] Section 3.6

Y_m Missile impact load on a structure generated by or during postulated breaks, as from pipe whipping, as described in [DCD Tier 2,] Section 3.6

Note that loads associated with flooding of the containment below Elevation 107' are resisted by the concrete structures and not by the containment vessel.

In accordance with the guidelines of SRP Section 3.8.2, including the load combinations recommended in SRP Section 3.8.2.II.3.b, the staff finds that DCD Tier 2, Table 3.8.2-1, lists acceptable load combinations for the containment vessel design. The design loads and load combinations described in DCD Tier 2, Table 3.8.2-1, are consistent with those specified in SRP Section 3.8.2.II.3.

3.8.2.4 Design and Analysis Procedures

SRP Section 3.8.2.II.4 states that design and analysis procedures for steel containment are covered by Article NE-3000 of Subsection NE of ASME Code, Section III, Division 1. The procedures given in the Code, as augmented by the applicable provisions of RG 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor System Components," constitute an acceptable basis for design analysis. The SRP section also provides acceptable criteria for treatment of nonaxisymmetric and localized loads, treatment of buckling effects, computer programs, ultimate capacity of steel containment, structural audit, and design report.

In DCD Tier 2, Section 3.8.2.4, "Design and Analysis Procedures," states the following:

> The design and analysis procedures for the containment vessel are according to the requirements of the ASME Code, Section III, Subsection NE.

> The analyses are summarized in [DCD Tier 2,] Table 3.8.2-4. The detailed analyses will use a series of general-purpose finite element, axisymmetric shell and special purpose computer codes to conduct such analyses. Code development, verification, validation, configuration control, and error reporting and resolution are according to the Quality Assurance requirements of [DCD Tier 2,] Chapter 17.

3.8.2.4.1 Analyses for Design Conditions

3.8.2.4.1.1 Axisymmetric Shell Analyses

DCD Tier 2, Section 3.8.2.4.1.1, "Axisymmetric Shell Analyses," states the following:

> The containment vessel is modeled as an axisymmetric shell and analyzed using the ANSYS computer program. A model used for static analyses is shown in [DCD Tier 2,] Figure 3.8.2-6.

> Dynamic analyses of the axisymmetric model, which is similar to that shown in [DCD Tier 2,] Figure 3.8.2-6, are performed to obtain frequencies and mode shapes. These are used to confirm the adequacy of the containment vessel stick model as described in [DCD Tier 2, Section] 3.7.2.3.2. Static stress analysis is performed for each of the following loads:

> - dead load
> - internal pressure
> - equivalent static seismic accelerations
> - polar crane wheel loads

Design of Structures, Components, Equipment, and Systems

- wind loads
- thermal loads

The equivalent static accelerations applied in the seismic analysis are the maximum acceleration responses based on the results for a hard rock site shown in [DCD Tier 2,] Table 3.7.2-6. These accelerations are applied as separate load cases in the east-west, north-south, and vertical directions. The torsional moments, which include the effects of the eccentric masses, are increased to account for accidental torsion and are evaluated in a separate calculation.

The results of these load cases are factored and combined in accordance with the load combinations identified in [DCD Tier 2,] Table 3.8.2-1. These results are used to evaluate the general shell away from local penetrations and attachments, that is, for areas of the shell represented by the axisymmetric geometry. The results for the polar crane wheel loads are also used to establish local shell stiffness for inclusion in the containment vessel stick model described in [DCD Tier 2, Section] 3.7.2.3. The results of the analyses and evaluations are included in the containment vessel design report.

Design of the containment shell is primarily controlled by the internal pressure of [406.8 kPa (59 psig)]. The meridional and circumferential stresses for the internal pressure case are shown in [DCD Tier 2,] Figure 3.8.2-5. The most highly stressed regions for this load case are the portions of the shell away from the hoop stiffeners and the knuckle region of the top head. In these regions the stress intensity is close to the allowable for the design condition.

Major loads that induce compressive stresses in the containment vessel are internal and external pressure and crane and seismic loads...

- Internal pressure causes compressive stresses in the knuckle region of the top head and in the equipment hatch covers. The evaluation methods are similar to those discussed in [DCD Tier 2, Section] 3.8.2.4.2 for the ultimate capacity.

- Evaluation of external pressure loads is performed in accordance with ASME Code, Section III, Subsection NE, Paragraph NE-3133.

- Crane wheel loads due to crane dead load, live load, and seismic loads result in local compressive stresses in the vicinity of the crane girder. These are evaluated in accordance with ASME Code Case N-284.

- Overall seismic loads result in axial compression and tangential shear stresses at the base of the cylindrical portion. These are evaluated in accordance with ASME Code, Case N-284.

The bottom head is embedded in the concrete base at Elevation 100'. This leads to circumferential compressive stresses at the discontinuity under thermal

loading associated with the design-basis accident. The containment vessel design includes a Service Level A combination in which the vessel above Elevation 107'-2" is specified at the design temperature of [148.9 °C (300 °F)] and the portion of the embedded vessel (and concrete) below Elevation 100' is specified at a temperature of [21.1 °C (70 °F)]. The temperature profile for the vessel is linear between these elevations. Containment shell buckling close to the base is evaluated against the criteria of ASME Code Case N-284...

3.8.2.4.1.2 Local Analyses

DCD Tier 2, Section 3.8.2.4.1.2, "Local Analyses," states the following:

The penetrations and penetration reinforcements are designed in accordance with the rules of ASME III, Subsection NE. The design of the large penetrations for the two equipment hatches and the two airlocks use the results of finite element analyses which consider the effect of the penetration and its dynamic response as follows:

1. The upper airlock and equipment hatch penetrations are modeled in individual finite element models. The lower airlock and equipment hatch are modeled in a combined finite element model ([DCD Tier 2,] Figure 3.8.2-7) including the boundary conditions representing the embedment. The finite element models include a portion of the shell sufficient that the boundary condition do not affect the results of the local analyses.

2. Surface loads are applied for pressure and inertia loads on the shell included in the model. Loads corresponding to the stresses in the unpenetrated vessel at the location of the penetration, obtained from the axisymmetric analyses described in the previous subsection, are applied as boundary conditions for the local finite element models.

3. The out-of-plane stiffness of the containment vessel is determined for unit radial loads and moments at the location of the penetration. The frequency of the local radial and rotational modes are calculated using single degree of freedom models with mass and rotational inertia of the penetration. Seismic response accelerations for the radial and rotational modes are determined from the applicable floor response spectra for the containment vessel. Equivalent static radial loads and moments are calculated from these seismic response accelerations.

4. Radial loads and moments due to the local seismic response and due to external loads on the penetration are applied statically at the location of the penetration. These loads are applied individually corresponding to the three directions of input (radial, tangential and vertical). The three directions of seismic input are combined by the square root sum of the squares method or by the 100%, 40%, 40% method as described in [DCD Tier 2, Section] 3.7.2.6. [The staff's acceptance of using the 100 percent, 40 percent, 40 percent combination method is discussed in Section 3.7.2 of this report.]

5.   Stresses due to local loads on the penetration (step 4) are combined with those from the global vessel analyses (step 2). Stresses are evaluated against the stress intensity criteria of ASME Section III, Subsection NE. Stability is evaluated against ASME Code Case N-284. Local stresses in the regions adjacent to the major penetrations are evaluated in accordance with paragraph 1711 of the code case. Stability is not evaluated in the reinforced penetration neck and insert plate which are substantially stiffer than the adjacent shell.

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. [This is COL Action Item 3.8.2.4.1.2-1.]

The [4.87 m (16 ft)] diameter equipment hatch located at Elevation 112'-6" and the personnel airlock located at Elevation 110'-6" are in proximity to each other and to the concrete embedment. Design of these penetrations uses the finite element model shown in [DCD Tier 2,] Figure 3.8.2-7. Static analyses are performed for dead loads and containment pressure. Response spectrum analyses are performed for seismic loads. Stresses are evaluated as described for the single penetrations in step 5 above.

Finite element analyses are performed to confirm that the design of the penetration in accordance with the ASME Code provides adequate margin against buckling. A finite element ANSYS model, as shown in [DCD Tier 2,] Figure 3.8.2-7, represents the portion of the vessel close to the embedment with the lower equipment hatch and personnel airlock. This is analyzed for external pressure and axial loads and demonstrates that the penetration reinforcement is sufficient and precludes buckling close to the penetrations. The lowest buckling mode occurs in the shell away from the penetrations and embedment.

The design and analysis procedures used in the analyses for design conditions are appropriate and consistent with the guidelines described in SRP Section 3.8.2 for buckling of the shell under pressure loading, and, therefore, are acceptable.

3.8.2.4.2 Evaluation of Ultimate Capacity

In DCD Tier 2, Section 3.8.2.4.2, "Evaluation of Ultimate Capacity," the applicant described the analyses and tests relied on to estimate the capacity of the containment vessel to resist internal pressure for use in the probabilistic risk assessment analyses and severe accident evaluations. As stated in DCD Tier 2, Section 3.8.2.4.2, each element of the containment vessel boundary was evaluated to estimate the maximum pressure corresponding to each of the following two stress and buckling criteria:

•   Deterministic severe accident pressure capacity corresponding to ASME Service Level C limits on stress intensity, ASME Paragraph NE-3222, and ASME Code Case N-284 for buckling of the equipment hatch covers, and 60 percent of the

critical buckling for the top head. The deterministic severe accident pressure capacity corresponds to the approach in SECY-93-087, to maintain a reliable leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges. This approach was approved by the Nuclear Regulatory Commission as outline[d] in the Staff Requirements Memorandum on SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, Dated July 21, 1993.

- Best-estimate capacity corresponding to gross membrane yield at the ASME-specified minimum yield stress (SA738, Grade B, yield stress = [413.7 MPa (60 ksi)], ultimate stress = [586.0 MPa (85 ksi)], and critical buckling for the equipment hatch covers and top head.

The results are shown in DCD Tier 2, Table 3.8.2-2. The evaluation considered the following containment boundary elements:

- cylindrical shell
- top and bottom heads
- equipment hatches and covers
- personnel airlocks
- mechanical and electrical penetrations

In its evaluation, the applicant identified the most likely failure mode to be that associated with gross yield of the cylindrical shell. Loss of containment function would be expected to occur because the large, post-yield deflections would lead to local failures at penetrations, bellows, or other local discontinuities.

During the staff's review of DCD Tier 2, Section 3.8.2.4.2, two apparent errors were identified. In response to a staff question during the April 2–5, 2003, audit, the applicant acknowledged that "SA537, Class 2" in the second paragraph of DCD Tier 2, Section 3.8.2.4.2.6, "Material Properties," should be "SA738, Grade B," and agreed to correct it in the next DCD revision. This was Confirmatory Item 3.8.2.4-1 in the DSER.

The applicant revised DCD Tier 2, Section 3.8.2.4.2.6, by replacing the term, "SA537, Class 2," with the term, "SA738, Grade B." Therefore, Confirmatory Item 3.8.2.4-1 is resolved.

In response to a staff question during the April 2–5, 2003, audit, the applicant acknowledged that the second paragraph of DCD Tier 2, Section 3.8.2.4.2.3, "Equipment Hatches," and the second paragraph of DCD Tier 2, Section 3.8.2.4.2.8, "Summary of Containment Pressure Capacity," contain conflicting pressure capacities for the 4.87 m (16 ft) diameter equipment hatches at 38.7 °C (100 °F), 510.2 kPa (74 psig) vs. 579.2 kPa (84 psig), using ASME Code, Section III, Paragraph NE-3222 Service Level C limits, and 765.3 kPa (111 psig) vs. 868.7 kPa (126 psig), using ASME Code Case N-284. The applicant agreed to correct these items in the next DCD revision. This was Confirmatory Item 3.8.2.4-2 in the DSER.

In DCD Tier 2, Sections 3.8.2.4.2.3 and 3.8.2.4.2.8, the applicant corrected the above errors. Therefore, Confirmatory Item 3.8.2.4-2 is resolved.

Design of Structures, Components, Equipment, and Systems

The applicant summarized the results of the ultimate capacity evaluation in DCD Tier 2, Section 3.8.2.4.2.8, which is repeated below:

> The ultimate pressure capacity for containment function is expected to be associated with leakage caused by excessive radial deflection of the containment cylindrical shell. This radial deflection causes distress to the mechanical penetrations, and leakage would be expected at the expansion bellows for the main steam and feedwater piping. There is high confidence that this failure would not occur before stresses in the shell reach the minimum specified material yield [stress]. This is calculated to occur at a pressure of [1.069 MPa (155 psig)] at ambient temperature and [889.4 kPa (129 psig)] at [204.4 °C (400 °F)]. Failure would be more likely to occur at a pressure about 15 percent higher based on expected actual material properties.

> The deterministic severe accident pressure that can be accommodated according to the ASME Service Level C stress intensity limits and using a factor of safety of 1.67 for buckling of the top head is determined by the capacity of the [4.87 m (16 foot)]-diameter equipment hatch cover and the ellipsoidal head. The maximum capacity of the hatch cover, calculated according to ASME paragraph NE-3222, Service Level C, is [579.2 kPa (84 psig)] at an ambient temperature of [38.7 °C (100 °F)] and [558.5 kPa (81 psig)] at [148.9 °C (300 °F)]. When calculated in accordance with ASME Code Case N-284, Service Level C, the maximum capacity is [868.7 kPa (126 psig)] at an ambient temperature of [38.7 °C (100 °F)] and [834.3 kPa (121 psig)] at [148.9 °C (300 °F)]. The maximum capacity of the ellipsoidal head is [717.1 kPa (104 psig)] at [37.8 °C (100 °F)] and [627.4 kPa (91 psig)] at [148.9 °C (300 °F)].

> The maximum pressure that can be accommodated according to the ASME Service Level C stress intensity limits, excluding evaluation of instability, is determined by yield of the cylinder and is [930.8 kPa (135 psig)] at an ambient temperature of [37.8 °C (100 °F)] and [806.7 kPa (117 psig)] at [148.9 °C (300 °F)]. This limit is used in the evaluations required by 10 CFR 50.34(f).

The staff considers the analysis procedures used in evaluating the ultimate capacity of the AP1000 containment to be consistent with sound engineering practice for such evaluations. On this basis, the staff concludes that the results of the AP1000 ultimate capacity evaluation constitute acceptable input for probabilistic risk assessment analyses and severe accident evaluations. Chapter 19 of this report includes the staff's detailed evaluation of these analyses.

3.8.2.5 Structural Criteria

DCD Tier 2, Section 3.8.2.5, "Structural Criteria," states as follows:

> The containment vessel is designed, fabricated, installed, and tested according to the ASME Code, Section III, Subsection NE, and will receive a code stamp.

> Stress intensity limits are according to ASME Code, Section III, Paragraph NE-3221 and Table NE-3221-1. [*Critical buckling stresses are*

evaluated according to the provisions of ASME Code, Section III, Paragraph NE-3222, or ASME Code, Case N-284].*

The staff accepts ASME Code, Section III, Subsection NE as the governing structural criteria. Section 3.8.2.2 of this report provides the staff's evaluation of codes, standards, and specifications. As set forth in that section, the use of ASME Code, Section III, Subsection NE for the steel containment design is acceptable to the staff.

3.8.2.6 <u>Materials, Quality Control, and Special Construction Techniques</u>

DCD Tier 2, Section 3.8.2.6, "Materials, Quality Control, and Special Construction Techniques," states the following:

> Materials for the containment vessel, including the equipment hatches, personnel locks, penetrations, attachments, and appurtenances meet the requirements of NE-2000 of the ASME Code. The basic containment material is SA738, Grade B, plate. The procurement specifications for the SA738, Grade B, plate includes supplemental requirements S17, Vacuum Carbon-Deoxidized Steel and S20, Maximum Carbon Equivalent for Weldability. This material has been selected to satisfy the lowest service metal temperature requirement of [-26.1 °C (-15 °F)]. This temperature is established by analysis for the portion of the vessel exposed to the environment when the minimum ambient air temperature is [-40 °C (-40 °F)]. Impact test requirements are as specified in NE-2000.

The staff requested that the applicant provide details of the analyses conducted for the AP1000 to establish the minimum service temperature of -26.1 °C (-15 °F). The staff also asked the applicant to indicate whether SA738, Grade B material would meet the impact requirements of NE-2000, if the minimum service temperature requirement is -40 °C (-40 °F). In response to RAI 220.005 (Revision 1), the applicant described the global heat transfer calculation and the local heat transfer calculation to account for the "fin" effect of the air baffles. Both calculations predicted a minimum containment shell temperature greater than -26.1 °C (-15 °F). The applicant further stated, "It is expected that it (SA738, Grade B) could be procured to meet the impact requirements of NE-2000 if the minimum service temperature requirement were -40 °C (-40 °F)."

During the April 2–5, 2003, audit, the staff reviewed the heat transfer calculations referenced in the RAI response. After discussion with the applicant, the calculations were re-done with several parameter changes to test the sensitivity of the results. The predicted minimum temperature was slightly lower, but still greater than -26.1 °C (-15 °F). During the audit, the staff reviewed the calculations for testing the sensitivity of the results and found them to be acceptable. The applicant agreed to formalize the calculations and revise its response to RAI 220.005. This was Confirmatory Action 3.8.2.6-1 in the DSER.

During the October 6–9, 2003, audit, the staff reviewed Calculation APP-PCS-M3C-002, Revision 1, "AP1000 Containment Shell Minimum Service Temperature." Based on its review and discussion with the applicant, the staff concluded that the calculation provides an adequate technical basis to establish -26.1 °C (-15 °F) as the minimum service temperature for the AP1000. On this basis, Confirmatory Action 3.8.2.6-1 is resolved.

Design of Structures, Components, Equipment, and Systems

The issue of fracture toughness of SA738 Grade B material with regard to its ability to satisfy the impact requirements of NE-2000 was reviewed the staff under the resolution of RAI 220.003 (Section 3.8.2.2 of this report).

In its response to RAI 220.003 (Revision 1), the applicant submitted its technical basis for concluding that SA738 Grade B material has adequate fracture toughness to meet the requirements of NE-2000, in the as-welded condition, for thicknesses up to and including 4.44 cm (1.75 in.). After review of the response, the staff concluded that the applicant had not provided sufficient quantitative data, and had made confusing statements concerning the effects of post-weld heat treatment (PWHT). The applicant addressed this issue during the April 2–5, 2003, audit by presenting directly applicable test data compiled by the CBI. The staff finds the information provided by the applicant to be acceptable because the test data indicated that SA738 Grade B steel has adequate fracture toughness to meet the NE-2000 requirements (with the as-welded condition). The applicant committed to revise its response to RAI 220.003 to include the information presented at the audit. This was Confirmatory Item 3.8.2.2-1 in the DSER. From its review of Revision 2 of the response to RAI 220.003 dated April 23, 2003, the staff confirmed that the applicant provided the above-mentioned CBI test data for the staff's review. Therefore, Confirmatory Item 3.8.2.2-1 is resolved.

DCD Tier 2, Section 3.8.2.6, further states:

> The containment vessel is coated with an inorganic zinc coating, except for those portions fully embedded in concrete. The inside of the vessel below the operating floor and up to [2.44 m (8 ft)] above the operating floor also has a phenolic top coat. Below Elevation 100' the vessel is fully embedded in concrete with the exception of the few penetrations at low elevations (see [DCD Tier 2,] Figure 3.8.2-4, sheet 3 of 6, for typical details). Embedding the steel vessel in concrete protects the steel from corrosion...

> ...The exterior of the vessel is embedded at Elevation 100' and concrete is placed against the inside of the vessel up to the maintenance floor at Elevation 107'-2". Above this elevation, the inside and outside of the containment vessel are accessible for inspection of the coating. The vessel is coated with an inorganic zinc primer to a level just below the concrete. Seals are provided at the surface of the concrete inside and outside the vessel so that moisture is not trapped next to the steel vessel just below the top of the concrete. The seal on the inside accommodates radial growth of the vessel due to pressurization and heatup.

The staff has identified that corrosion protection needs to be considered as a safety-related function of the containment coatings. This is addressed in Section 3.8.2.1 of this report.

The staff's review of DCD Tier 2, Section 3.8.2.6, finds that the quality control program related to welding procedures, erection tolerances, and nondestructive examination of shop- and field-fabricated welds conforms the Subsections NE-4000 and NE-5000 of the ASME Code and is, therefore, acceptable. The staff's review also finds that the containment vessel is designed to permit its construction using large subassemblies. These subassemblies consist of the two heads and three ring sections. Each ring section comprises three or four courses of plates and

is approximately 11.58 m (38 ft) to 15.54 m (51 ft) high. These are assembled in an area near the final location, using plates fabricated in a shop facility. The applicant's construction procedures are consistent with industry practice and are also acceptable.

3.8.2.7 Testing and Inservice Inspection Requirements

DCD Tier 2, Section 3.8.2.7, "Testing and Inservice Inspection Requirements," states the following:

> Testing of the containment vessel and the pipe assemblies forming the pressure boundary within the containment vessel will be according to the provisions of NE-6000 and NC-6000, respectively...

> ...Inservice inspection of the containment vessel will be performed according to the ASME Code, Section XI, Subsection IWE, and is the responsibility of the COL applicant.

DCD Tier 2, Section 6.2.5, "Containment Leak Rate Test System," describes leak rate testing of the containment system, including the containment vessel.

The staff finds the commitments to structural integrity testing and ISI to be acceptable because the applicant's approach and acceptance criteria related to these topics meet the ASME Code requirements. The staff's evaluation of containment leak rate testing is in Section 6.2 of this report.

3.8.2.8 Conclusions

For the reasons set forth above, the staff concludes that the design of the AP1000 steel containment vessel meets the relevant requirements of 10 CFR Part 50, GDC 1, 16, 51, and 53. The staff further concludes that satisfaction of the relevant requirements of GDC 2, 4, and 50 will be demonstrated upon completion of the ASME design report by the COL applicant. On this basis, the staff finds the AP1000 steel containment vessel design to be acceptable. In particular, this conclusion is based on the following observations:

- By following the guidelines of RG 1.57 and the ASME Code, Section III, Subsection NE, the applicant 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.

- The applicant has met the requirements of GDC 2 by designing the AP1000 steel containment vessel to withstand a 0.3 g SSE with sufficient margin. The combinations of the effects of normal and accident conditions with the effects of environmental loadings, such as earthquakes and other natural phenomena, will be documented in the ASME design report.

- The applicant has met the requirements of GDC 4 by ensuring that the design of the AP1000 steel containment vessel is capable of withstanding the dynamic effects

Design of Structures, Components, Equipment, and Systems

associated with missiles, pipe whipping, and fluid discharges. Evaluations associated with GDC 4 will be documented in the ASME design report.

- The applicant has met the requirements of GDC 16 by designing the AP1000 steel containment vessel so that it essentially provides a leak-tight barrier to prevent the uncontrolled release of radioactive effluent to the environment.

- The applicant has met the requirements of GDC 50 by designing the AP1000 steel containment vessel to accommodate, with sufficient margin, the design leakage rate, calculated pressure, and temperature conditions resulting from postulated accidents. In meeting these design requirements, the applicant has followed the recommendations of RG 1.57 and ASME Code, Section III, Subsection NE. The applicant has also performed an appropriate analysis to demonstrate that the ultimate capacity of the containment will not be exceeded and that an acceptable margin of safety has been established for the design. The ASME design report will document that the design conditions are not exceeded during the full course of the accident.

- For the portion of the AP1000 containment vessel above Elevation 100', the applicant provided access space to perform any necessary periodic inspection of all important areas, as well as to implement surveillance program inspections and periodic testing. The remainder of the containment vessel is fully embedded in concrete. Therefore, the applicant has ensured that this portion of the containment vessel is leak-tight, and periodic inspection above Elevation 100' would provide the necessary indication of moisture intrusion or evidence of degradation in progress. In addition, as indicated in DCD Tier 2, Section 3.8.2, and Figure 3.8.2-4, the majority of containment penetrations (both mechanical and electrical) are located above Elevation 100'. For those penetrations located below this elevation, the applicant provided access (pockets) for testing and inspection from outside the containment vessel. On the basis stated above, the applicant has met the requirements of GDC 53.

- The AP1000 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 that 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 the 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 AP1000 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 to the requirements of ASME Code, Section III, Subsection NE, which are incorporated by reference into 10 CFR 50.55a.

In addition, the applicant 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 its structural integrity or its safety function of limiting the release of radioactive material.

3.8.3 Concrete and Steel Containment Internal Structures

Using the guidance described in Section 3.8.3 of the SRP, based on the requirements in 10 CFR 50.55a, GDC 1, 2, 4, and 50, the staff reviewed DCD Tier 2, Section 3.8.3. In particular, the review of this section focused on the analysis and design of the AP1000 concrete and steel internal structures of the steel containment vessel, with emphasis on the (1) materials, (2) geometry, (3) codes and standards, (4) loadings, and (5) design and analysis procedures.

3.8.3.1 Description of the Containment Internal Structures

DCD Tier 2, Section 3.8.3.1, "Description of the Containment Internal Structures," provides the following description of the CIS:

> The containment internal structures are those concrete and steel structures inside (not part of) the containment pressure boundary that support the reactor coolant system components and related piping systems and equipment. The concrete and steel structures also provide radiation shielding. The containment internal structures are shown on the general arrangement drawings in [DCD Tier 2,] Section 1.2. The containment internal structures consist of the primary shield wall, reactor cavity, secondary shield walls, in-containment refueling water storage tank (IRWST), refueling cavity walls, operating floor, intermediate floors, and various platforms. The polar crane girders are considered part of the containment vessel. They are described in [DCD Tier 2, Section] 3.8.2.
>
> Component supports are those steel members designed to transmit loads from the reactor coolant system to the load-carrying building structures. The component configurations are described in this subsection including the local building structure backing up the component support. The design and construction of the component supports are described in [DCD Tier 2, Section] 5.4.10 [and evaluated in Section 5.4 of this report].
>
> The containment internal structures are designed using reinforced concrete and structural steel. At the lower elevations conventional concrete and reinforcing steel are used, except that permanent steel forms are used in some areas in lieu of removable forms based on constructibility considerations. These steel form modules (liners) consist of plate reinforced with angle stiffeners and tee sections, as shown in [DCD Tier 2,] Figure 3.8.3-16. The angles and the tee sections are on the concrete side of the plate. Welded studs, or similar embedded steel elements, are attached on the concrete face of the permanent steel form where surface attachments transfer loads into the concrete. Where these surface attachments are seismic Category I, the portion of the steel form module transferring the load into the concrete is classified as seismic Category I.

Design of Structures, Components, Equipment, and Systems

> Walls and floors are concrete filled steel plate structural modules. The walls are supported on the [massive] concrete containment internal structures basemat with the steel surface plate extending down to the concrete floor on each side of the wall...

In RAI 220.006, the staff requested the applicant to clarify a statement in DCD Tier 2, Section 3.8.3.1, that indicates that the steel surface plates of the structural modules provide reinforcement in the concrete and anchor the structural modules to the base concrete.

The staff also requested a clarification of the difference between the above noted statement and a notation in DCD Tier 2, Figure 3.8.3-8, that indicates that the structural modules also require anchoring to the concrete with mechanical connectors/rebars. The staff requested the applicant to specifically explain whether the steel surface plates are sufficient to provide anchorage to the concrete, or if additional mechanical connectors/rebars are also needed. The applicant was also requested to identify where the details are described in the AP1000 DCD, or to provide the details as part of the response. In its response to RAI 220.006, Revision 1, the applicant described the connection details for anchoring the modules to concrete, which were designed in accordance with American Concrete Institute (ACI) 349 Code, and revised DCD Tier 2, Sections 3.8.3.1 and 3.8.3.5.3, accordingly. The staff finds these changes to the DCD to be acceptable.

DCD Tier 2, Section 3.8.3.1, further states the following:

> ...[DCD Tier 2,] Figure 3.8.3-1 shows the location of the structural modules. [DCD Tier 2,] Figures 3.8.3-2 and 3.8.3-15 show the typical structural configuration of the wall modules. A typical floor module, is shown in [DCD Tier 2,] Figure 3.8.3-3 and also in [DCD Tier 2,] Figure 3.8.3-16 combined with the linear module. These structural modules are structural elements built up with welded steel structural shapes and plates. Concrete is used where required for shielding, but reinforcing steel is not normally used.

> Walls and floors exposed to water during normal operation or refueling are constructed using stainless steel plates.

DCD Tier 2, Section 3.8.3.1.1.1, "Reactor Vessel Support System," states that following:

> The reactor vessel is supported by four supports located under the cold-legs, which are spaced 90 degrees apart in the primary shield wall. The supports are designed to provide for radial thermal growth of the reactor coolant system, including the reactor vessel, but they prevent the vessel from lateral and torsional movement. The loads are carried by the reactor vessel supports to embedded steel plates of the CA-04 structural module which forms the inside face of the primary shield concrete. Figure 3.8.3-4 shows the reactor vessel supports. Sheet 4 of [DCD Tier 2,] Figure 3.8.3-14 shows the CA-04 structural module.

DCD Tier 2, Section 3.8.3.1.1.2, "Steam Generator Support System," states the following:

> The steam generator vertical support consists of a single vertical column extending from the steam generator compartment floor to the bottom of the steam generator channel head. The column is constructed of heavy plate sections and is pinned at both ends to permit unrestricted radial displacement of the steam generator during plant heatup and cooldown. The location of this column is such that it will allow full access to the steam generator for routine maintenance activities. It is located a sufficient distance away from the reactor coolant pump motors to permit pump maintenance and inservice inspection.

> The lower steam generator horizontal support is located at the top of the vertical column. It consists of a tension/compression strut oriented approximately perpendicular to the hot-leg. The strut is pinned at both the wall bracket and the vertical column to permit movement of the generator during plant heatup and cooldown.

> The upper steam generator horizontal support in the direction of the hot-leg is located on the upper shell just above the transition cone. It consists of two large hydraulic snubbers oriented parallel with the hot-leg centerline. One snubber is mounted on each side of the generator on top of the steam generator compartment wall. The hydraulic snubbers are valved to permit steam generator movement for thermal transition conditions, and to "lock-up" and act as rigid struts under dynamic loads.

> The upper steam generator horizontal support in the direction normal to the hot-leg is located on the lower shell just below the transition cone. It consists of two rigid tension/compression struts oriented perpendicular to the hot-leg. The two rigid struts are mounted on the steam generator compartment wall at the elevation of the operating deck. The steam generator loads are transferred to the struts and snubbers through trunnions on the generator shell. [DCD Tier 2,] Figure 3.8.3-5 shows the steam generator supports...

DCD Tier 2, Section 3.8.3.1.1.3, "Reactor Coolant Pump Support System," states the following:

> Because the reactor coolant pumps are integrated into the steam generator channel head, they do not have individual supports. They are supported by the steam generators.

DCD Tier 2, Section 3.8.3.1.1.4, "Pressurizer Support System," states the following:

> The pressurizer is supported by four columns mounted from the pressurizer compartment floor. A lateral support is provided at the top of the columns. This lateral support consists of eight struts connecting it to the pressurizer compartment walls. A lateral support is also provided on the upper portion of the pressurizer. This lateral support consists of a ring girder around the pressurizer and eight struts connecting it to the pressurizer compartment walls. [DCD Tier 2,] Figure 3.8.3-6 shows the pressurizer supports.

Design of Structures, Components, Equipment, and Systems

DCD Tier 2, Section 3.8.3.1.2, "Containment Internal Structures Basemat," states the following:

> The containment internal structures basemat is the reinforced concrete structure filling the bottom head of the containment vessel. It extends from the bottom of the containment vessel head at Elevation 66'-6" up to the bottom of the structural modules that start between Elevations 71'-6" and 103'-0". The basemat includes rooms as shown on [DCD Tier 2,] Figure 1.2-5. The primary shield wall and reactor cavity extend from Elevation 71'-6" to Elevation 107'-2". They provide support for the reactor vessel and portions of the secondary shield walls and refueling cavity walls. The general arrangement drawings in [DCD Tier 2,] Section 1.2 show the location and configuration of the primary shield wall and reactor cavity. The walls of the primary shield, the steam generator compartment and the CVS room are structural modules as shown in [DCD Tier 2,] Figure 3.8.3-1. The rest of the basemat is constructed from reinforced concrete.

DCD Tier 2, Section 3.8.3.1.3, "Structural Wall Modules," states the following:

> Structural wall modules are used for the primary shield wall around the reactor vessel, the wall between the vertical access and the CVS room, secondary shield walls around the steam generators and pressurizer, for the east side of the in-containment refueling water storage tank, and for the refueling cavity. The general arrangement drawings in [DCD Tier 2,] Section 1.2 drawings. Locations of the structural modules are shown in [DCD Tier 2,] Figure 3.8.3-1. Isometric views of the structural modules are shown in [DCD Tier 2,] Figure 3.8.3-14. The secondary shield walls are a series of walls that, together with the refueling cavity wall, enclose the steam generators. Each of the two secondary shield wall compartments provides support [to], and houses, a steam generator and reactor coolant loop piping. The [IRWST] is approximately [9.1 m (30 ft)] high. The floor elevation of this tank is Elevation 103'-0". The tank extends up to about Elevation 133'-3", directly below the operating deck. On the west side, along the containment vessel 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, filled with concrete and forming, in part, the refueling cavity, steam generator compartment, and pressurizer compartment walls, compose the east wall. The refueling cavity has two floor elevations. The area around the reactor vessel flange is at Elevation 107'-2". The lower level is at Elevation 98'-1". The upper and lower reactor internals storage is at the lower elevation, as is the fuel transfer tube. The center line of the fuel transfer tube is at Elevation 100'-8.75".
>
> Structural wall modules consist of steel faceplates connected by steel trusses. The primary purpose of the trusses is to stiffen and hold together the faceplates during handling, erection, and concrete placement. The nominal thickness of the steel faceplates is [1.27 cm] 0.5 inch. The nominal spacing of the trusses is [76.2 cm] 30 inches. Shear studs are welded to the inside face of the steel faceplates. Faceplates are welded to adjacent plates with full penetration welds so that the weld is at least as strong as the plate. Plates on each face of the wall

module extend down to the elevation of the adjacent floor. Since the floors in the subcompartments on each side of the wall module are at different elevations [as shown in DCD Tier 2, Figure 3.8.3-1 (sheets 1 through 7)], one of the plates extends further than the other. This portion is designated on [DCD Tier 2,] Figure 3.8.3-1 as "CA Structure Module with Single Surface Plate." A typical configuration is shown on [DCD Tier 2,] Figure 3.8.3-8. The module functions as a wall above the upper floor level (Elevation 103'-0" in [DCD Tier 2,] Figure 3.8.3-8). The single plate below this elevation is designed to transfer the reactions at the base of the wall into the basemat. This plate also acts as face reinforcement for the basemat. Basemat reinforcement dowels are provided at the bottom of the single plate as shown in [DCD Tier 2,] Figure 3.8.3-8. The structural wall modules are anchored to the concrete base by reinforcing steel dowels or other types of connections embedded in the reinforced concrete below. After erection, concrete is placed between the faceplates. Typical details of the structural modules are shown in [DCD Tier 2,] Figures 3.8.3-2, 3.8.3-8, and 3.8.3-17.

DCD Tier 2, Section 3.8.3.1.4, "Structural Floor Modules," states the following:

Structural floor modules are used for the operating floor at Elevation 135'-3" over the in-containment refueling water storage tank and for the 107'-2" floor over the rooms in the containment internal structures basemat. The floors are shown on the general arrangement drawings in [DCD Tier 2,] Section 1.2. The [Elevation] 107'-2" floors and the floor above the in-containment refueling water storage tank consist of steel tee and wide flange sections, welded to horizontal steel bottom plates stiffened by transverse stiffeners. After erection, concrete is placed on top of the horizontal plate and around the structural steel section. The remaining region of the operating floor consists of a concrete slab, placed on Q decking supported by structural steel beams. The operating floor is supported by the in-containment refueling water storage tank walls, refueling cavity walls, the secondary shield walls, and steel columns originating at Elevation 107'-2". Structural details of the operating floor structural module are shown in [DCD Tier 2,] Figure 3.8.3-3.

DCD Tier 2, Section 3.8.3.1.5, "Internal Steel Framing," states the following:

The region of the operating floor away from the in-containment refueling water storage tank consists of a concrete slab, placed on Q decking supported by structural steel beams. The floor at Elevation 118'-6" consists of steel grating supported by structural steel framing. In addition, a number of steel platforms are located above and below the operating floor. These platforms support either grating floors or equipment, such as piping and valves.

On the basis that the descriptive information and referenced figures in DCD Tier 2, Section 3.8.3.1, contain sufficient details to define the primary structural aspects and elements relied upon for the structures to perform their safety-related functions, in accordance with SRP Section 3.8.3, the staff finds the descriptive information to be acceptable.

Design of Structures, Components, Equipment, and Systems

3.8.3.2 Applicable Codes, Standards, and Specifications

DCD Tier 2, Section 3.8.3.2, "Applicable Codes, Standards, and Specifications," states the following:

> The following [Codes and standards] are applicable to the design, materials, fabrication, construction, inspection, or testing of the containment internal structures:
>
> • *[American Concrete Institute (ACI), Code Requirements for Nuclear Safety Related Structures, ACI-349-01]** (refer to [DCD Tier 2, Section] 3.8.4.5 for supplemental requirements)
>
> • American Concrete Institute (ACI), ACI Detailing Manual, 1994
>
> • American Concrete Institute (ACI), "Standard Specifications for Tolerances for Concrete Construction and Materials, ACI-117-90"
>
> • American Concrete Institute (ACI), Guide to Formwork for Concrete, ACI-347-94
>
> • *[American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, AISC-N690-1984]** (refer to [DCD Tier 2, Section] 3.8.4.5 for supplemental requirements)
>
> • American Welding Society (AWS), Structural Welding Code, AWS D 1.1-2000
>
> • American Welding Society (AWS), Reinforcing Steel Welding Code, AWS D 1.4-98
>
> • National Construction Issues Group (NCIG), Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, NCIG-01, Revision 2, May 7, 1985

As indicated in DCD Tier 2, Section 3.8.4.2, "Applicable Codes, Standards, and Specifications," the applicant was requested to designate ACI-349-01 and AISC-N690-1984 as Tier 2*, and note that any proposed change to these documents will require NRC approval prior to implementation of the change. During the design audit on April 2–5, 2003, the applicant committed to incorporate this change in the next DCD revision. This was Confirmatory Item 3.8.3.2-1 in the DSER.

In DCD Tier 2, Section 3.8.3.2, the applicant designated ACI-349-01 and AISC-N690-94 (replaces AISC-N690-1984) as Tier 2*. The correct Tier 2* information is reflected in the Codes and standards listed above. Therefore, Confirmatory Item 3.8.3.2-1 is resolved.

DCD Tier 2, Section 3.8.3.2, further states the following:

> Nationally recognized industry standards, such as American Society for Testing and Materials, American Concrete Institute, and American Iron and Steel Institute, are used to specify material properties, testing procedures, fabrication, and construction methods. [DCD Tier 2,] Section 1.9 describes conformance with the Regulatory Guides.
>
> Welding and inspection activities for seismic Category I structural steel, including building structures, structural modules, cable tray supports, and heating, ventilating and air-conditioning (HVAC) duct supports are accomplished in accordance with written procedures and meet the requirements of the American Institute of Steel Construction (AISC-690). The weld acceptance criteria is as defined in NCIG-01, Revision 2. The welded seams of the plates forming part of the leak-tight boundary of the in-containment refueling water storage tank are examined by liquid penetrant and vacuum box after fabrication to confirm that the boundary does not leak.

The staff finds that the Codes, standards, and specifications identified in the DCD for the design of the CIS are consistent with the guidelines in SRP Section 3.8.3.II.2 and RG 1.142, Revision 2, in which the staff's review guidelines for the applicable Codes, standards, and specifications are provided. DCD Tier 2, Section 3.8.4.2, discusses the use of ACI-349-01 for the design of reinforced concrete structures, and for the reasons set forth in Section 3.8.4.2 of this report, the staff finds it acceptable. Therefore, the staff concludes that the Codes, standards, and specifications used in the AP1000 design are acceptable.

3.8.3.3 Loads and Load Combinations

DCD Tier 2, Section 3.8.3.3, "Loads and Load Combinations," states the following:

> The loads and load combinations for the containment internal structures are the same as for other Category I structures described in [DCD Tier 2,] Section 3.8.4.3 and the associated tables, except for the following modifications:
>
> > Wind loads (W), tornado loads (W_t), and precipitation loads (N) are not applicable to the design of the containment internal structures because of the protection provided by the steel containment...

The staff's evaluation of the loads and load combinations is presented in Section 3.8.4.3 of this report. Staff evaluations of loads described in DCD Tier 2, Sections 3.8.3.3.1, "Passive Core Cooling System Loads," and 3.8.3.3.2, "Concrete Placement Loads," are provided below.

3.8.3.3.1 Passive Core Cooling System Loads

DCD Tier 2, Section 3.8.3.3.1, states the following:

> Structures are evaluated for pressure and thermal transients associated with operation of the passive core cooling system. The effects of temperatures

higher than [37.8 °C (100 °F)] on the modulus of elasticity and yield strength of steel are considered.

The passive core cooling system and the automatic depressurization system (ADS) are described in [DCD Tier 2,] Section 6.3. The automatic depressurization system is in part composed of two spargers that are submerged in the in-containment refueling water storage tank. The spargers provide a controlled distribution of steam flow to prevent imposing excessive dynamic loads on the tank structures. Capped vent pipes are installed in the roof of the tank on the side near the containment wall. These caps prevent debris from entering the tank at the containment operating deck, but they open under slight pressurization of the in-containment refueling water storage tank. This provides a path to vent steam released by the spargers. An overflow is provided from the in-containment refueling water storage tank to the refueling cavity to accommodate volume and mass increases during automatic depressurization system operation. Two sets of loads representing bounding operational or inadvertent transients are considered in the design of the in-containment refueling water storage tank.

- ADS_1—This automatic depressurization system load is associated with blowdown of the primary system through the spargers when the water in the in-containment refueling water storage tank is cold and the tank is at ambient pressure. Dynamic loads on the in-containment refueling water storage tank due to automatic depressurization system operation are determined using the results from the automatic depressurization system hydraulic test as described in [DCD Tier 2, Section] 3.8.3.4.2. The hydrodynamic analysis described in [DCD Tier 2, Section] 3.8.3.4.2 show that member forces in the walls of the in-containment refueling water storage tank are bounded by a case with a uniform pressure of [34.5 kPa (5 psi)] applied to the walls. The in-containment refueling water storage tank is designed for a uniform pressure of [34.5 kPa (5 psi)] applied to the walls. This pressure is taken as both positive and negative due to the oscillatory nature of the hydrodynamic loads. This automatic depressurization system transient is of short duration such that the concrete walls do not heat up significantly. It is combined with ambient thermal conditions. Long-term heating of the tank is bounded by the design for the ADS_2 load.

- ADS_2—This automatic depressurization system transient considers heatup of the water in the in-containment refueling water storage tank. This may be due to prolonged operation of the passive residual heat removal heat exchanger or due to an automatic depressurization system discharge. For structural design, an extreme transient is defined starting at [10 °C (50 °F)] since this maximizes the temperature gradient across the concrete-filled structural module walls. Prolonged operation of the passive residual heat removal heat exchanger raises the water temperature from an ambient temperature of [10 °C (50 °F)] to saturation in about 4 hours, increasing to about [126.7 °C (260 °F)] within about

10 hours. Steaming to the containment atmosphere initiates once the water reaches its saturation temperature. The temperature transient is shown in [DCD Tier 2,] Figure 3.8.3-7. Blowdown of the primary system through the spargers may occur during this transient and occurs prior to 24 hours after the initiation of the event. Since the flow through the sparger cannot fully condense in the saturated conditions, the pressure increases in the in-containment refueling water storage tank and steam is vented through the in-containment refueling water storage tank roof. The in-containment refueling water storage tank is designed for an equivalent static internal pressure of [34.5 kPa (5 psi)] in addition to the hydrostatic pressure occurring at any time up to 24 hours after the initiation of the event.

The ADS_1 and ADS_2 loads are considered as live loads. The dynamic ADS_1 load is combined with the safe-shutdown earthquake by the square root sum of the squares (SRSS). ADS_2 is an equivalent static pressure which is included algebraically with other normal loads and then combined with plus/minus SSE loads.

From its review of DCD Tier 2, Section 3.8.3.3.1, the staff raised a question with respect to some of the water temperature transients for the AP1000 design; it was not clear how these water temperature transients have affected the analysis and design of the structural modules for AP1000. In RAI 220.007, the staff requested the applicant to provide the following information:

- The transient temperature reaches 121.1 °C (250 °F) in 3.5 hours for the AP1000. Provide the basis for this temperature increase, and explain how it was considered in the analysis and design of the modules.

- The extreme transient starting temperature used for the structural design is 21.1 °C (70 °F) for the AP1000. Provide the basis for this starting temperature and explain how this change was considered in the analysis and design of the AP1000 modules.

In its response to RAI 220.007 (Revision 0), the applicant addressed the staff's concerns by revising the thermal transient and providing a new figure showing the IRWST temperature transient (DCD Tier 2, Figure 3.8.3-7). These revisions have subsequently been incorporated into the DCD and are shown above. During the design audit on April 2–5, 2003, the staff reviewed Calculation Nos. APP-1100-S2C-004, Revision 0, and APP-1100-S2C-005, Revision 1, which contain the thermal analyses for the critical structural modules inside containment. As part of the review, discussions were held with the applicant regarding the heat up of the module faceplates during the thermal transient. The mismatch in thermal conductivity between the steel faceplates and the concrete could impose significant thermal loads on the faceplates, studs, and concrete core. The applicant's calculations and explanation demonstrated that the thermal transient imposed on the module wall is sufficiently slow so that the thermal loads caused by the relative expansion of faceplates and concrete is not significant. Based on the above discussion, the staff finds that the thermal analysis and design of the structural modules inside containment are acceptable.

Design of Structures, Components, Equipment, and Systems

3.8.3.3.2 Concrete Placement Loads

DCD Tier 2, Section 3.8.3.3.2, states the following:

> The steel faceplates of the structural wall modules, designed for the hydrostatic pressure of the concrete, act as concrete forms. The concrete placement loads are [50.3 kPa (1050 lbs/ft^2)] determined in accordance with ACI-347. The bending stress in the faceplate due to this hydrostatic pressure of the concrete placements is approximately [89.6 MPa (13 ksi)], based on the assumption of a continuous faceplate, or [137.9 MPa (20 ksi)] based on the assumption of simple spans. The minimum yield strength of material for the faceplates is 36 ksi for A36 steel. The stress is well below the allowable limit, since the plate is designed to limit the out-of-plane deflection. After the concrete has gained strength, these stresses remain in the steel; however, since the average residual stress is zero and since the concrete no longer requires hydrostatic support, the ultimate strength of the composite section is not affected, and the full steel plate is available to carry other loads as described below.
>
> The steel plates and the concrete act as a composite section after the concrete has reached sufficient strength. The composite section resists bending moment by one face resisting tension and the other face resisting compression. The steel plate resists the tension and behaves as reinforcing steel in reinforced concrete. The composite section is under reinforced so that the steel would yield before the concrete reaches its strain limit of 0.003 in/in. As the steel faceplates are strained beyond yield to allow the composite section to attain its ultimate capacity, the modest residual bending stress from concrete placement is relieved, since the stress across the entire faceplate in tension is at yield. The small residual strain induced by the concrete placement loads is secondary and has negligible effect on the ultimate bending capacity of the composite section. The stresses in the faceplates resulting from concrete placement are therefore not combined with the stresses in the post-construction load combinations.

The staff verified that the upper bound calculation (137.9 MPa (20 ksi)) of maximum local bending stress in the module faceplate, due to concrete placement, is less than the AISC N690 allowable stress for plate bending. Therefore, the staff finds this acceptable. The staff also evaluated the applicant's technical basis for excluding local bending stresses due to concrete placement in the post-construction load combinations. The staff concurs that the maximum load carrying capacity of the faceplates, acting as reinforcement for the concrete, is not reduced by the pre-existence of a linearly varying local bending stress through the thickness of the faceplate because the average through-wall stress is zero. While yielding will occur earlier on one side of the plate, it will be retarded on the other side of the plate. The maximum design capacity will still be governed by the assumption of a uniform tensile yield stress through the thickness of the faceplate. In addition, the residual strain induced by the concrete placement loads is considered to be small and secondary, and thus has a negligible effect on the ultimate bending capacity of the composite section. On this basis, the staff finds the applicant's approach to consider concrete placement loads to be acceptable.

3.8.3.4 Analysis Procedures

DCD Tier 2, Section 3.8.3.4, "Analysis Procedures," states the following:

This subsection describes the modeling and overall analyses of the containment internal structures, including the concrete-filled structural modules. Concrete and steel composite structures are used extensively in conventional construction. Applications include concrete slabs on steel beams and concrete-filled steel columns. Testing of concrete-filled structural modules is described in [DCD] References 27 through 29 for in-plane loading and in [DCD] References 30 through 33 for out-of-plane loading. The tests indicate that these composite structures behave in a manner similar to reinforced concrete structures. The initial load deflection behavior is well predicted using the gross properties of the steel and concrete. This is similar to the behavior of reinforced concrete elements where the initial stiffness is predicted by the gross properties. As the load is increased on reinforced concrete members, cracking of the concrete occurs and the stiffness decreases. The behavior of concrete and steel composite structures is similar in its trends to reinforced concrete but has a superior performance. The results of the test program by Akiyama et al. ([DCD] Reference 27) indicate that concrete and steel composites similar to the structural modules have significant advantages over reinforced concrete elements of equivalent thickness and reinforcement ratios:

- Over 50 percent higher ultimate load carrying capacity,
- Three times higher ductility, and
- Less stiffness degradation under peak cyclic loads, 30 percent for concrete and steel composites versus 65 percent for reinforced concrete

Methods of analysis for the structural modules are similar to the methods used for reinforced concrete. [DCD Tier 2,] Table 3.8.3-2 summarizes the finite element analyses of the containment internal structures and identifies the purpose of each analysis and the stiffness assumptions for the concrete filled steel modules. For static loads the analyses use the monolithic (uncracked) stiffness of each concrete element. For thermal and dynamic loads the analyses consider the extent of concrete cracking as described in later subsections. Stiffnesses are established based on analyses of the behavior and review of the test data related to concrete-filled structural modules. The stiffness directly affects the member forces resulting from restraint of thermal growth. The in-plane shear stiffness of the module influences the fundamental horizontal natural frequencies of the containment internal structures in the nuclear island seismic analyses described in [DCD Tier 2, Section] 3.7.2. The out-of-plane flexural stiffness of the module influences the local wall frequencies in the seismic and hydrodynamic analyses of the in-containment refueling water storage tank. Member forces are evaluated against the strength of the section calculated as a reinforced concrete section with zero strength assigned to the concrete in tension.

Design of Structures, Components, Equipment, and Systems

>ACI 349, Section 9.5.2.3 specifies an effective moment of inertia for calculating the deflection of reinforced concrete beams. For loads less than the cracking moment, the moment of inertia is the gross (uncracked) inertia of the section. The cracking moment is specified as the moment corresponding to a maximum flexural tensile stress of $7.5\sqrt{f_c'}$ [in which f_c' is the specified compressive strength of concrete]. For large loads, the moment of inertia is that of the transformed cracked concrete section. The effective moment of inertia provides a transition between these two depending on the ratio of the cracking moment to the maximum moment in the beam at the stage the deflection is to be computed.

>[DCD Tier 2,] Table 3.8.3-1 summarizes in-plane shear and out-of-plane flexural stiffness properties of the [121.9 cm (48 in.)] and [76.2 cm (30 in.)] walls based on a series of different assumptions. The [stiffness is] expressed for unit length and height of each wall. The ratio of the stiffness to the stiffness of the monolithic case is also shown.

DCD Tier 2, Table 3.8.3-1, presents the wall stiffness properties for the following three cases corresponding to the monolithic section, uncracked gross section, and transformed cracked section.

DCD Tier 2, Section 3.8.3.4 further states:

>Case 1 assumes monolithic behavior of the steel plate and uncracked concrete. This stiffness is supported by the test data described in [DCD] References 27 through 33 for loading that does not cause significant cracking. This stiffness value is the basis for the stiffness of the concrete-filled steel module walls in the nuclear island seismic analyses and in the uncracked case for the hydrodynamic analyses.

>Case 2 considers the full thickness of the wall as uncracked concrete. This stiffness value is shown for comparison purposes. It is applicable for loads that do not result in significant cracking of the concrete and is the basis for the stiffness of the reinforced concrete walls in the nuclear island seismic analyses. This stiffness was used in the harmonic analyses of the internal structures described in [DCD Tier 2, Section] 3.8.3.4.2.2.

>Case 3 assumes that the concrete in tension has no stiffness. For the flexural stiffness, this is the conventional stiffness value used in working stress design of reinforced concrete sections. For in-plane shear stiffness, a 45-degree diagonal concrete compression strut is assumed with tensile loads carried only by the steel plate. The in-plane stiffness calculated by these assumptions are lower than the stiffness measured in the tests described in [DCD] References 27 through 29 for loading that causes cracking.

Test data on concrete-filled steel modules ([DCD] Reference 27) demonstrate that structural properties such as ductility, ultimate capacity, and stiffness degradation (due to cracking at higher loads) are comparable or superior to those of reinforced concrete walls. Therefore, the

staff finds the analysis method described above, which treats structural modules as reinforced concrete members, to be acceptable.

3.8.3.4.1 Seismic Analyses

3.8.3.4.1.1 Finite Element Model

DCD Tier 2, Section 3.8.3.4.1.1, "Finite Element Model," states the following:

> The three-dimensional (3D) lumped-mass stick model of the containment internal structure is developed based on the structural properties obtained from a 3D finite element model. The structural modules are simulated within the finite element model using 3D shell elements. Equivalent shell element thickness and modulus of elasticity of the structural modules are computed as shown below. The shell element properties are computed using the combined gross concrete section and the transformed steel faceplates of the structural modules. This representation models the composite behavior of the steel and concrete.
>
> Axial and Shear Stiffness of module:
>
> $$\Sigma \, EA = \, E_c \, (\, L \, t + 2 \, (n\text{-}1) \, L \, t_s \,)$$
>
> Bending Stiffness of module:
>
> $$\Sigma \, EI = \, E_c \, [\, (L/12) \, t^3 + 2 \, (L/12) \, (n\text{-}1) \, t_s^{\,3} + 2 \, (n\text{-}1) \, L \, t_s \, (t/2)^2 \,]$$
>
> where:
>
> E_c = concrete modulus of elasticity
> n = modular ratio of steel to concrete
> L = length of wall module
> t = thickness of wall module
> t_s = thickness of plate on each face of wall module
>
> The equivalent thickness, t_m, and modulus of elasticity of the plate elements, E_m, are calculated from the following formulae:
>
> $$t_m \quad = \quad [\, \{1+3\alpha(n\text{-}1)\} \, / \, \{1+\alpha(n\text{-}1)\} \,]^{\frac{1}{2}} \, t$$
>
> $$E_m \quad = \quad [\, 1+\alpha(n\text{-}1) \,] \, [\, \{1+3\alpha(n\text{-}1)\} \, / \, \{1+\alpha(n\text{-}1)\} \,]^{\frac{1}{2}} \, E_c$$
>
> where $\alpha = 2 \, t_s \, / \, t$ and terms of order α^3 are neglected (for a typical 30-inch thick wall with ½ inch steel plates, $\alpha = 0.033$).

The above method is recognized as a technically accepted approach for calculating equivalent wall stiffnesses of multi-layered materials for use in computerized FE shell models. Since the method has also been verified by test data on similar concrete-filled steel modules, the staff finds this approach acceptable.

Design of Structures, Components, Equipment, and Systems

3.8.3.4.1.2 Stiffness Assumptions for Global Seismic Analyses

DCD Tier 2, Section 3.8.3.4.1.2, "Stiffness Assumptions for Global Seismic Analyses," states the following:

> The monolithic initial stiffness (Case 1 of [DCD Tier 2,] Table 3.8.3-1) is used in the seismic analyses of the containment internal structures and the auxiliary building modules. This stiffness is used since the stresses due to mechanical loads including the safe-shutdown earthquakes are less than the cracking stress. The maximum in-plane concrete shear stresses in the AP600 containment internal structural modules are [668.8 kPa (97 psi)] for the [121.9 cm (48 in.)] wall and [944.6 kPa (137 psi)] for the [76.2 cm (30 in.)] wall due to the safe-shutdown earthquake based on the monolithic section properties...

In RAI 220.008, the staff requested that the applicant provide its technical bases for the following DCD Tier 2, Section 3.8.3.4.1.2, statements and conclusions:

> ...These stresses will increase slightly for the AP1000 due to the increased height of the steam generator and pressurizer compartments and the increased mass of the steam generators and pressurizer. The stresses will still be well below the magnitude causing significant cracking of concrete so the monolithic assumption is also appropriate for the AP1000.

In its response to RAI 220.008, Revision 1, the applicant submitted quantitative data comparing maximum forces and moments for the AP600 and the AP1000 designs. The maximum increase in shear force between the AP600 and the AP1000 is 25 percent. This would raise the maximum predicted shear stress to 1.17 MPa (170 psi). The applicant based this predicted increase in shear stress on the time-history seismic analysis. The applicant asserts that this increase would not cause significant cracking of the concrete, suggesting that the monolithic assumption is still appropriate.

During the design audit on April 2–5, 2003, the staff reviewed Westinghouse Calculation Nos. APP-1100-S2C-002, Revision 1; APP-1100-S2C-006, Revision 1; and APP-1100-S2C-007, Revision 0, which contain the seismic equivalent static analysis and stress analysis for the CIS. This analysis was based on the detailed FE model of the CIS. The maximum acceleration values in each of the three directions from the time-history seismic analysis and accidental torsion were used in the FE model as equivalent static accelerations. An examination of the shear forces and resulting shear stresses demonstrated that the maximum predicted shear stress of 1.17 MPa (170 psi) is appropriate. On the basis that the results from the FE model support the maximum prediction of shear stress equal to 1.17 MPa (170 psi), and that this level of shear stress would not cause significant cracking, the staff finds that the use of monolithic properties is appropriate for the AP1000 global seismic analysis.

DCD Tier 2, Section 3.8.3.4.1.2, further states the following:

> The [peak] broadening of the floor response spectra is sufficient to account for the reduction of structural frequencies due to cracking of those portions of the structural modules that are boundaries of the in-containment refueling water

storage tank exposed to abnormal thermal transients...Both tests and analyses show that this cracking has only a small effect on the in-plane shear stiffness of a panel.

Section 3.7.2 of this report presents the staff evaluation of the effect of concrete cracking on the global seismic analysis.

3.8.3.4.1.3 Stiffness Assumptions for Seismic Analyses of the In-Containment Refueling Water Storage Tank

DCD Tier 2, Section 3.8.3.4.1.3, "Stiffness Assumptions for Seismic Analyses of the In-Containment Refueling Water Storage Tank," states the following:

> The seismic analyses of the in-containment refueling water storage tank address the local response of the walls and water and are performed to verify the structural design of the tank. The lowest significant wall frequency is about 30 hertz using monolithic properties and would not be excited by the seismic input. The local analyses are therefore performed using the cracked section stiffness values based on composite behavior with zero stiffness for the concrete in tension (Case 3 of [DCD Tier 2,] Table 3.8.3.1). The local analyses use the finite element model described in [DCD Tier 2, Section] 3.8.3.4.2.2. Response spectrum analyses are performed using the floor response spectra at the base of the tank.

Since some cracks may develop in the concrete-filled steel modules of the IRWST, the staff finds the applicant's use of the cracked section stiffness values (Case 3 of DCD Tier 2, Table 3.8.3.1) for seismic analyses of the IRWST to be acceptable.

3.8.3.4.1.4 Damping of Structural Modules

DCD Tier 2, Section 3.8.3.4.1.4, "Damping of Structural Modules," states the following:

> Damping of the structural modules is reported in [DCD] Reference 27 based on the cyclic load tests of a containment internal structure model. The equivalent viscous damping at the design load level was 5 percent for the concrete-filled steel member. This was almost constant up to the load level at which the steel plate started yielding. Dynamic analyses are performed using 7 percent damping for the reinforced concrete structures and 5 percent for the structural modules as shown in [DCD Tier 2, Section] 3.7.1.

RG 1.61 recommends 7 percent damping for dynamic analysis of reinforced concrete structures. The use of 7 percent damping for the reinforced concrete structures meets the guidelines of RG 1.61, and the use of 5 percent damping for the structural modules is in accordance with test data reported in DCD Reference 27. On this basis, the damping values specified in DCD Tier 2, Section 3.8.3.4.1.4, for reinforced concrete and structural modules of the CIS are acceptable.

Design of Structures, Components, Equipment, and Systems

3.8.3.4.2 Hydrodynamic Analyses

DCD Tier 2, Section 3.8.3.4.2, "Hydrodynamic Analyses," states the following:

> This subsection describes the hydrodynamic analyses performed for the AP600
> which demonstrated that design of the walls of the in-containment refueling water
> storage tank for [34.5 kPa (5 psi)] as described in [DCD Tier 2, Section] 3.8.3.3.1
> would bound the loads from the time history transient analysis. The analyses
> were performed using the AP600 test results. The peak values from these tests
> are also applicable to the AP1000 design, since they occur at the beginning of the
> transient, and the automatic depressurization system and the initial conditions are
> the same for the two plant designs ([DCD] Reference 52). The structural
> configuration of the tank is identical. The minor differences in the height of the
> steam generator and pressurizer compartment walls and in the mass of the steam
> generators and pressurizer will only have a minor effect on the significant
> structural frequencies. Since the time histories applied in the AP600 analyses
> cover a broad range of frequencies, the response of the AP1000 tank boundary
> will be similar to that of the AP600. The [34.5 kPa (5 psi)] pressure design basis
> for the tank boundary is therefore also applicable to the AP1000.
>
> Hydrodynamic analyses were performed for the AP600 for automatic depressurization
> system discharge into the in-containment refueling water storage tank. This discharge is
> designated as ADS_1 in the load description of [DCD Tier 2, Section] 3.8.3.3.1 and results
> in higher hydrodynamic loading than the automatic depressurization system discharge
> into a hot tank in ADS_2. The first three stages of the ADS valves discharge into the tank
> through spargers under water, producing hydrodynamic loads on the tank walls and
> equipment. Hydrodynamic loads, measured in hydraulic tests of the automatic
> depressurization system sparger in a test tank, are evaluated using the source load
> approach ([DCD] Reference 34). Analyses of the tests define source pressure loads that
> are then used in analyses of the in-containment refueling water storage tank to give the
> dynamic responses of the containment internal structures. The basic analysis approach
> consists of the following steps:
>
> - A pressure source, an impulsive forcing function at the sparger discharge, is
>   selected from the tests using a coupled fluid structure finite element model of the
>   test tank, taking into account fluid compressibility effects. This source
>   development procedure is based on a comparison between analysis and test
>   results, both near the sparger exit and at the boundaries of the test tank.
>
> - The pressure source is applied at each sparger location in a coupled fluid
>   structure finite element model of the in-containment refueling water storage tank
>   structure and of the contained water. The mesh characteristics of the model at
>   the sparger locations and the applied forcing functions correspond to those of the
>   test tank analysis.

The staff raised a concern that, from the information provided by the applicant, it is not evident
that the changes in the structural elements and masses are "minor," as stated in DCD Tier 2,

Section 3.8.3.4.2. Specifically, in RAI 220.009, the staff requested the applicant to address the following:

- the technical basis for concluding that the increase in wall heights and mass of the steam generator and pressurizer will have a minor effect on the structural frequencies

- an explanation of how the range of frequencies considered in the AP600 time-history analyses adequately covers the expected frequency shifts caused by the differences between the AP600 and AP1000 design

- the margin between the maximum wall pressure calculated from the analyses and the 34.5 kPa (5 psi) pressure used as the design basis for the AP600 IRWST boundary

In its response to RAI 220.009, Revision 1, the applicant submitted both proprietary and non-proprietary data to address the concern raised by the staff. In its response, the applicant provided an explanation as to why the increase in wall heights and mass of the steam generator and pressurizer will not greatly affect the structural frequencies of the IRWST walls. The change in the dominant natural frequency (from 29.2 Hz to 29 Hz) associated with the governing wall (south wall of the steam generator compartment) is 1 percent or less. This change in frequency is considered to be insignificant. As part of the RAI response, the applicant also provided the response spectra of the two ADS_1 forcing functions taken from the automatic depressurization hydraulic tests which were used in the IRWST hydrodynamic analyses. From these data, it is evident that a 1 percent shift in the structural frequency will not affect the structural response of the IRWST. To demonstrate that the ±34.5 kPa (±5 psi) used in the design of the IRWST modules bounds the ADS_1 transient loads, the applicant provided tables that compare the member forces obtained from the hydrodynamic analyses to the ±34.5 kPa (±5 psi) static pressure analysis based on the AP600 design. The minimum ratio of moments obtained from the 34.5 kPa (5 psi) static pressure to moments from the hydrodynamic pressure for the two critical walls evaluated is 1.09. Based on the information provided in the RAI response, the staff concluded that the ±34.5 kPa (±5 psi) static pressure analysis will bound the hydrodynamic analysis for AP1000.

During the design audit on April 2–5, 2003, the staff reviewed Westinghouse Calculation No. APP-1100-S2C-003, Revision 1, "Static Analyses - Pressures" which documents the ±34.5 kPa (±5 psi) static pressure analysis for the AP1000 containment internal structures. This audit is documented in a meeting summary dated September 26, 2003. The analysis develops the member forces for hydrodynamic loading on the CIS, which are later combined with other loads for design the critical sections. Based on review of this calculation, the staff finds that the applicant has appropriately evaluated the AP1000 containment internal structures for hydrodynamic loading and the justification is acceptable.

3.8.3.4.2.1 Sparger Source Term Evaluation

DCD Tier 2, Section 3.8.3.4.2.1, "Sparger Source Term Evaluation," states the following:

A series of tests were conducted with discharge conditions representative of one sparger for the AP600 ([DCD] References 35 and 36). Pressure traces measured during the test discharges were investigated, at both sparger exit and tank

Design of Structures, Components, Equipment, and Systems

boundaries, to (1) bound the expected discharge from the automatic depressurization system; (2) characterize the pressure wave transmission through the pool water; (3) determine the maximum pressure amplitudes and the frequency content; and (4) produce reference data for qualification of the analytical procedure. Pressure time histories and power spectrum densities were examined at reference sensors, both for the total duration of the discharge transient (about 50 seconds) and for critical time intervals.

Fluid-structure interaction analyses were performed with the ANSYS computer code ([DCD] Reference 37). The mathematical model consists of a 3D sector finite element model, 15 degrees wide, as shown in [DCD Tier 2,] Figure 3.8.3-9. It uses STIF30 fluid and STIF63 structural ANSYS finite elements, which take into account fluid compressibility and fluid-structure interaction. Rayleigh damping of 4 percent is used for the concrete structure, and the fluid damping is neglected. Direct step-by-step time integration is used. The measured discharge pressures for single time intervals are imposed as uniform forcing functions on the idealized spherical surface of the steam/water interface, and pressures transmitted through the water to the tank boundary are calculated and compared with test measurements. The analyses of the test tank showed satisfactory agreement for the pressures at the tank boundary.

The examination of test results related to the structural design of the in-containment refueling water storage tank under automatic depressurization system hydrodynamic excitation and comparison with the analytical procedure previously described, lead to the following conclusions by the applicant regarding the sparger source term definition:

- The automatic depressurization system discharge into cold water produces the highest hydrodynamic pressures. The tests at higher water temperatures produce significantly lower pressures.

- Two pressure time histories, characterized by different shapes and frequency content, can be selected as representative of the sparger discharge pressures; they are assumed as acting on a spherical bubble centered on the sparger centerline and enveloping the ends of the sparger arms.

- The application of such time histories as forcing functions to an analytical model, simulating the fluid-structure interaction effects in the test tank, has been found to predict the measured tank wall pressures for the two selected reference time intervals.

- The two defined sparger source term pressure time histories can be used as forcing functions for global hydrodynamic analyses of the in-containment refueling water storage tank by developing a comprehensive fluid-structure finite element model and reproducing the test tank mesh pattern in the sparger region.

- The hydrodynamic loads on the vessel head support columns and ADS sparger piping located in the IRWST are developed from the forcing functions using the methodology documented in [DCD] Reference 51.

3-140

As discussed above, an FE model, which includes the test tank wall and fluid elements, was developed for evaluation of the hydrodynamic loading. Pressure traces measured during actual test discharges at the sparger exit location were used in the hydrodynamic analyses to calculate pressures at the tank walls. The calculated wall pressures were compared to the pressure traces measured from the test discharges. Since the analyses of the test tank showed satisfactory agreement with the pressures measured from the test at the tank boundary, the staff concludes that the FE model and analytical approach used by the applicant to predict pressures due to hydrodynamic loads are acceptable.

3.8.3.4.2.2 In-Containment Refueling Water Storage Tank Analyses

DCD Tier 2, Section 3.8.3.4.2.2, "In-Containment Refueling Water Storage Tank Analyses," states the following:

>   The in-containment refueling water storage tank is constructed as an integral part of the containment internal structures as described in [DCD Tier 2, Section] 3.8.3.1.3. It contains two depressurization spargers that are submerged approximately [2.74 m (9 ft)] below the normal water level. Transmission of the hydrodynamic pressures from the sparger discharge to the wetted in-containment refueling water storage tank is evaluated using the coupled fluid-structure interaction method similar to that described for the test tank analysis in the previous [DCD section].
>
>   The 3D ANSYS finite element model includes the in-containment refueling water storage tank boundary, the water within the in-containment refueling water storage tank, the adjacent structural walls of the containment internal structures, and the operating floor. The model of the in-containment refueling water storage tank, shown in [DCD Tier 2,] Figures 3.8.3-10 (sheet 2), 3.8.3-11, and 3.8.3-12, represents the outer steel structures, the inner concrete walls, and the water. The model of the adjacent structural walls and floors is shown in [DCD Tier 2,] Figure 3.8.3-10 (sheet 1). The flexible steel outer wall is represented using beam and shell elements; isotropic plate elements are used to represent the inner structural module walls. The water is modeled as a compressible fluid to provide an acoustic medium to transmit the source pressure. The model has two bubble boundaries representing the spargers. Pressure loads are applied to the solid element faces adjacent to the air bubbles. The forcing functions at the sparger locations are conservatively assumed to be in phase. Rayleigh damping of 5 percent is used for the concrete-filled structural modules and fluid damping is neglected. All degrees of freedom were retained in the step-by-step direct integration solution procedure for the in-containment refueling water storage tank boundary and the water. Degrees of freedom in the adjacent walls and floor were condensed by Guyon reduction.
>
>   Significant structural frequencies of the AP600 containment internal structures were analyzed using the harmonic response option with the ANSYS model of the IRWST and containment internal structures. A harmonic unit pressure is applied at the surface of the spherical bubble representing the automatic depressurization

Design of Structures, Components, Equipment, and Systems

> system spargers. Material properties for the concrete elements are based on the
> uncracked gross concrete section (Case 2 of [DCD Tier 2,] Table 3.8.3-1). The
> results of these harmonic response analyses show the response deflection as a
> function of input frequency at nodes in the containment internal structures. The
> harmonic response analyses show that the largest responses are close to the
> wetted boundary of the in-containment refueling water storage tank and that the
> significant frequencies are from 18 to 50 hertz.

> Two time histories are identified for the structural hydrodynamic analyses; one
> has significant frequencies below 40 hertz while the other has significant
> frequencies in the range of 40 to 60 hertz. Both time-history inputs are used in
> the hydrodynamic analyses with the monolithic uncracked section properties for
> all walls. The lower frequency input is also applied in lower bound analyses using
> the cracked section stiffness values (Case 3 of [DCD Tier 2,] Table 3.8.3-1) for
> the concrete walls that are boundaries of the in-containment refueling water
> storage tank. Monolithic properties are used for the other walls. Results from
> these cases are enveloped, thereby accounting for variabilities in the structural
> analyses.

> The analyses of the AP600 in-containment refueling water storage tank give wall
> pressures, displacements, accelerations, hydrodynamic floor response spectra,
> and member forces due to the automatic depressurization system discharge
> pressure forcing functions. Consideration of pressure wave transmission and
> fluid-structure interaction shows a significant wall pressure attenuation with
> distance from the sparger region and with increasing wall flexibilities, relative to
> the measured sparger pressure forcing function. The member stresses are
> evaluated against the allowable stresses specified in [DCD Tier 2,
> Section] 3.8.3.5 for seismic Category I structures, considering the hydrodynamic
> loads as live loads. The analyses show that the member forces in the walls of the
> in-containment refueling water storage tank are bounded by a case with a uniform
> pressure of [34.5 kPa (5 psi)] applied to the walls.

As discussed above, FE models which included the IRWST walls and fluid elements were
developed for evaluation of the hydrodynamic loadings. These models were used to perform
time-history analyses to verify the adequacy of using ±5 psi equivalent static pressure for
hydrodynamic loadings. The development of these models and analytical approach followed
methods similar to those described in Section 3.8.3.1 of this report for the test tank. The results
of these analyses confirmed that the use of ±34.5 kPa (5 psi) bounds the hydrodynamic
loadings. On this basis, the staff concludes that the hydrodynamic analyses performed are
acceptable.

3.8.3.4.3 Thermal Analyses

DCD Tier 2, Section 3.8.3.4.3, "Thermal Analyses," states the following:

> The in-containment refueling water storage tank water and containment
> atmosphere are subject to temperature transients as described in [DCD Tier 2,
> Section] 3.8.3.3.1. The temperature transients result in a nonlinear temperature

distribution within the wall modules. Temperatures within the concrete wall are calculated in a unidimensional heat flow analysis. The average and equivalent linear gradients are applied to a finite element model of the containment internal structures at selected times during the transient. The effect of concrete cracking is considered in the stiffness properties for the concrete elements subjected to the thermal transient. The finite element model is that described in [DCD Tier 2, Section] 3.8.3.4.2.2 except that the model of the water in the IRWST is not needed...

Thermal transients for the design-basis accidents are described in [DCD Tier 2,] Section 6.3. The analyses for these transients are similar to those described above.

Section 3.8.3.3 of this report provides the staff's evaluation of the thermal analyses of the IRWST walls due to thermal transients generated inside the tank. Section 3.8.4.3 of this report discusses RAI 220.015 as it relates to the staff's evaluation of the thermal analyses of module walls inside containment (other than the IRWST) and locations outside containment. For the reasons set forth in those sections, the staff concludes that the thermal analyses performed are acceptable.

3.8.3.5 <u>Acceptance Criteria</u>

DCD Tier 2, Section 3.8.3.5, "Design Procedures and Acceptance Criteria," states the following:

> The containment internal structures that contain reinforcing steel including most of the areas below Elevation 98', are designed by the strength method, as specified in the ACI Code Requirements for Nuclear Safety Related Structures, ACI-349. This code includes ductility criteria for use in detailing, placing, anchoring, and splicing of the reinforcing steel.

> The internal steel framing is designed according to the AISC Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, AISC-N690, supplemented by the requirements given in [DCD Tier 2, Section] 3.8.4.5.

> The secondary shield walls, in-containment refueling water storage tank, refueling cavity, and operating floor above the in-containment refueling water storage tank are designed using structural modules. Concrete-filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349...Structural floor modules are designed as composite structures in accordance with AISC-N690.

> Methods of analysis used are based on accepted principles of structural mechanics and are consistent with the geometry and boundary conditions of the structures.

> The methods described in [DCD Tier 2, Section] 3.7.2 are employed to obtain the safe-shutdown earthquake loads at various locations in the containment internal structures. The safe-shutdown earthquake loads are derived from the response

Design of Structures, Components, Equipment, and Systems

> spectrum analysis of a three-dimensional, finite element model representing the entire containment internal structures.

In RAI 220.010, the staff noted that there was no discussion of the response spectrum analysis in DCD Tier 2, Section 3.7.2. DCD Tier 2, Section 3.7.2.1.1, discusses the use of equivalent static acceleration analysis for CIS and the coupled ASBs; however, no details of the analysis method were provided. The staff indicated that DCD Tier 2, Table 3.8.3-2, identifies that an equivalent static analysis of the 3D finite element model was utilized to obtain in-plane seismic forces for the design of floors and walls for the CIS fixed at Elevation 82'-6". It was unclear what method was used to obtain out-of-plane seismic forces for design of floors and walls for the CIS.

Therefore, in RAI 222.010, the staff requested the applicant to provide the following information:

- a description of the use of response spectrum analysis and equivalent static analysis in defining the seismic design loads for the containment internal structures, specifically identifying where each of the methods was employed, either singly or in combination, as well as an indication of how the three simultaneous components of seismic input motion were applied in the analyses and design

- a detailed description of how the equivalent static analysis method was implemented for the containment internal structures, the auxiliary building, and the shield building, including (1) how possible seismic amplification due to out-of-plane flexibility of walls and floors was considered, (2) how the equivalent static acceleration was calculated, (3) numerical values for the significant modal frequencies, and (4) numerical values for the equivalent static accelerations used in the analyses

- the technical basis for concluding that an adequate level of safety is achieved for the AP1000 containment internal structures when utilizing equivalent static analysis

In its response to RAI 220.010, Revision 1, the applicant indicated that (1) responses to RAIs 230.006 and 230.007 related to DCD Tier 2, Section 3.7, provide the information to address the concern relating to seismic analysis methods and the techniques for combining spatial effects of the three earthquake components used for the internal structures; (2) DCD Tier 2, Tables 3.7.2-1 through 3.7.2-7, provide numerical values for frequency and accelerations; and (3) adequate safety is maintained because code criteria stress limits are used.

The applicant revised DCD Tier 2, Section 3.8.3.5, to replace "response spectrum" with "equivalent static," and revised DCD Tier 2, Table 3.8.3-2, to clarify the models and methodology utilized for the various analyses of the structural modules. During the design audit on April 2–5, 2003, the staff reviewed Westinghouse Calculation Nos. APP-1000-S2C-034, Revision 1, "Finite Element Solid-Shell Model of Containment Internal Structures," APP-1100-S2C-002, Revision 1, "Static Analysis of Containment Internal Structures—SSE Equivalent Static Accelerations and Pressures," and APP-1200-S2C-001, Revision 0, "Auxiliary Shield building 3-D Finite Element Dead Load, Live Load, and Seismic analysis of the Hard, Rock Fixed Base Case." These analyses show how the equivalent static analysis method was implemented for the CIS and structures outside the containment. The calculations demonstrated that the maximum equivalent static accelerations obtained from the stick model time-history analysis were used as input to the FE models of the plant structures. To combine the structural responses due to the

three components of earthquake motion, the calculations used either the SRSS method or 100 percent, 40 percent, 40 percent method. Accidental torsion was also included in the two horizontal directions. Modal frequencies for the structures were determined and presented in the calculations. Seismic amplification for out-of-plane flexibility of walls and floors was accounted for in most cases by either including the flexibility in the seismic time-history model, or developing an amplification factor (such as the CIS wall modules). One item that arose during the design audit on April 2–5, 2003, was the lack of a technical guidance document that demonstrates how the flexibility of walls and floors other than critical sections will be considered in the seismic analyses. Based on the above discussion and the discussion in Section 3.7.2 of this report, the staff found that the seismic analysis method used by the applicant and the results obtained are acceptable, except for the concern regarding the lack of a documented method for considering out-of-plane wall and floor flexibility. This was identified as Open Item 3.8.3.5-1 in the DSER.

In its submittal dated June 23, 2003, the applicant committed to incorporate specific written guidance into the civil and seismic design criteria for evaluating out-of-plane flexibility. The staff found that proposed guidance meets the SRP guideline and is acceptable. However, verification was needed to confirm that the applicant actually incorporated guidance into its design records for considering the out-of-plane wall and floor flexibility.

During the October 6–9, 2003, audit, the staff reviewed Revision 1 of Westinghouse AP1000 Document No. APP-GW-S1-008, "Design Guide for Reinforcement in Walls and Floor Slabs." Section 2.1 of this document provides guidance for the designer to evaluate out-of-plane wall and floor flexibility. This document requires the designer to document that the structure is rigid or that flexibility of the structure has been accounted for in the seismic analysis as part of the design calculation. If the structure is flexible, three methods are presented that can be used to account for the floor or wall flexibility—(1) including the out-of-plane flexibility in the model that was used to develop the seismic loading, (2) calculating the fundamental frequency of the out-of-plane wall or floor slab and amplifying the seismic loads based on the applicable floor response spectrum, and (3) performing a dynamic analysis of a more detailed model that includes the flexible wall or slab out-of-plane response. All of these methods to account for out-of-plane flexibility of walls and floors meet the SRP Section 3.7.2 guideline and, therefore, are acceptable. The staff also reviewed Calculation APP-1000-S3R-001, Revision 0, "Reconciliations of Critical Sections to Revision 3 Seismic Spectra." Section 7.1, "Reconciliation of Containment Internal Structures," of this calculation indicates that only two walls have potential seismic amplification due to their flexibility (fundamental frequency below 33 Hz). The flexibility of these two walls was taken into account by calculating the seismic inertia loads based on the in-structure spectral acceleration corresponding to the fundamental frequency of the wall. This approach meets the SRP Section 3.7.2 guideline and is acceptable to the staff. On the basis discussed above, Open Item 3.8.3.5-1 is resolved.

DCD Tier 2, Section 3.8.3.5, further states:

> The determination of pressure and temperature loads due to pipe breaks is described in [DCD Tier 2, Sections] 3.6.1 and 6.2.1.2. Subcompartments inside containment containing high energy piping are designed for a pressurization load of [34.5 kPa (5 psi)].

Design of Structures, Components, Equipment, and Systems

Verification of the 34.5 kPa (5 psi) pressurization load and the temperature loads, for subcompartments inside containment containing high energy piping, is addressed by the applicant's response to RAI 220.015, which is discussed in Section 3.8.4. of this report.

DCD Tier 2, Section 3.8.3.5, further states:

> ...The pipe tunnel in the CVS room (room 11209, [DCD Tier 2,] Figure 1.2-6) is designed for a pressurization load of [51.7 kPa (7.5 psi). These subcompartment design pressures bound the pressurization effects due to postulated breaks in high-energy piping. The design for the effects of postulated pipe breaks is performed as described in [DCD Tier 2, Section] 3.6.2. Determination of pressure loads resulting from actuation of the automatic depressurization system is described in [DCD Tier 2, Section] 3.8.3.3.1.

> Determination of reactor coolant loop support loads is described in [DCD Tier 2, Section] 3.9.3. Design of the reactor coolant loop supports within the jurisdiction of ASME Code, Section III, Division 1, Subsection NF is described in [DCD Tier 2, Sections] 3.9.3 and 5.4.10.

> Computer codes used are general purpose codes. The code development, verification, validation, configuration control, and error reporting and resolution are according to the Quality Assurance requirements of [DCD Tier 2,] Chapter 17.

These matters are discussed in detail below in Sections 3.8.3.5.1–3.8.3.5.8.

3.8.3.5.1 Reactor Coolant Loop Supports

3.8.3.5.1.1 Reactor Vessel Support System

DCD Tier 2, Section 3.8.3.5.1.1, "Reactor Vessel Support System," states the following:

> The embedded portions of the reactor vessel supports, which are outside the ASME jurisdictional boundary, are designed by elastic methods of analysis. They are analyzed and designed to resist the applicable loads and load combinations given in [DCD Tier 2, Section] 3.8.4.3. The design is according to AISC-N690 and ACI-349. [DCD Tier 2,] Figure 3.8.3-4 shows the jurisdictional boundaries.

3.8.3.5.1.2 Steam Generator Support System

DCD Tier 2, Section 3.8.3.5.1.2, "Steam Generator Support System," states the following:

> The embedded portions of the steam generator supports, which are outside the ASME jurisdictional boundary, are designed using elastic methods of analysis. They are analyzed and designed to resist the applicable loads and load combinations given in [DCD Tier 2, Section] 3.8.4.3. The design is according to AISC-N690 and ACI-349. [DCD Tier 2,] Figure 3.8.3-5 shows the jurisdictional boundaries.

3.8.3.5.1.3 Reactor Coolant Pump Support System

DCD Tier 2, Section 3.8.3.5.1.3, "Reactor Coolant Pump Support System," states the following:

> The reactor coolant pumps are integrated into the steam generator channel head and consequently do not have a separate support system.

3.8.3.5.1.4 Pressurizer Support System

DCD Tier 2, Section 3.8.3.5.1.4, "Pressurizer Support System," states the following:

> The embedded portions of the pressurizer supports, which are outside the ASME jurisdictional boundary, are designed by elastic methods of analysis. They are analyzed and designed to resist the applicable loads and load combinations given in [DCD Tier 2, Section] 3.8.4.3. The design is according to AISC-N690 and ACI-349. [DCD Tier 2,] Figure 3.8.3-6 shows the jurisdictional boundaries.

3.8.3.5.2 Containment Internal Structures Basemat

DCD Tier 2, Section 3.8.3.5.2, "Containment Internal Structures Basemat," states the following:

> The containment internal structures basemat including the primary shield wall and reactor cavity are designed for dead, live, thermal, pressure, and safe-shutdown earthquake loads. The structural modules are designed as described in [DCD Tier 2, Section] 3.8.3.5.3.

> The reinforced concrete forming the base of the containment internal structures is designed according to ACI-349.

3.8.3.5.3 Structural Wall Modules

DCD Tier 2, Section 3.8.3.5.3, "Structural Wall Modules," states the following:

> Structural wall modules without concrete fill, such as the west wall of the in-containment refueling water storage tank, are designed as steel structures, according to the requirements of AISC-N690. This code is applicable since the module is constructed entirely out of structural steel plates and shapes. In local areas stresses due to restraint of thermal growth may exceed yield and the allowable stress intensity is 3 S_{m1}. This allowable is based on the allowable stress intensity for Service Level A loads given in ASME Code, Section III, Subsection NE, Paragraph NE-3221.4.

> The concrete-filled steel module walls are designed for dead, live, thermal, pressure, safe-shutdown earthquake, and loads due to postulated pipe breaks. The in-containment refueling water storage tank walls are also designed for the hydrostatic head due to the water in the tank and the hydrodynamic pressure effects of the water due to the safe-shutdown earthquake, and automatic depressurization system pressure loads due to sparger operation. The walls of

Design of Structures, Components, Equipment, and Systems

the refueling cavity are also designed for the hydrostatic head due to the water in the refueling cavity and the hydrodynamic pressure effects of the water due to the safe-shutdown earthquake.

[DCD Tier 2,] Figure 3.8.3-8 shows the typical design details of the structural modules, typical configuration of the wall modules, typical anchorages of the wall modules to the reinforced base concrete, and connections between adjacent modules. Concrete-filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349...The faceplates are considered as the reinforcing steel, bonded to the concrete by headed studs. The application of ACI-349 and the supplemental requirements are supported by the behavior studies described in [DCD Tier 2, Section] 3.8.3.4.1. The steel plate modules are anchored to the reinforced concrete basemat by mechanical connections welded to the steel plate or by lap splices where the reinforcement overlaps shear studs on the steel plate. The design of critical sections is described in [DCD Tier 2, Section] 3.8.3.5.8.

3.8.3.5.3.1 Design for Axial Loads and Bending

DCD Tier 2, Section 3.8.3.5.3.1, "Design for Axial Loads and Bending," states the following:

Design for axial load (tension and compression), in-plane bending, and out-of-plane bending is performed in accordance with the requirements of ACI-349, Chapters 10 and 14.

3.8.3.5.3.2 Design for In-Plane Shear

DCD Tier 2, Section 3.8.3.5.3.2, "Design for In-Plane Shear," states the following:

Design for in-plane shear is performed in accordance with the requirements of ACI-349, Chapters 11 and 14. The steel faceplates are treated as reinforcing steel in accordance with Section 11.10 of ACI-349.

3.8.3.5.3.3 Design for Out-of-Plane Shear

DCD Tier 2, Section 3.8.3.5.3.3, "Design for Out-of-Plane Shear," states the following:

Design for out-of-plane shear is performed in accordance with the requirements of ACI-349, Chapter 11.

3.8.3.5.3.4 Evaluation for Thermal Loads

DCD Tier 2, Section 3.8.3.5.3.4, "Evaluation for Thermal Loads," states the following:

The effect of thermal loads on the structural wall modules, with and without concrete fill, is evaluated by using the working stress design method for load combination 3 of [DCD Tier 2,] Tables 3.8.4-1 and 3.8.4-2. This evaluation is in addition to the evaluation using the strength design method of ACI-349 for the

load combination without the thermal load...Acceptance for the load combination with normal thermal loads, which includes the thermal transients described in [DCD Tier 2, Section] 3.8.3.3.1, is that the stress in general areas of the steel plate be less than yield. In local areas where the stress may exceed yield the total stress intensity range is less than twice the yield stress. This evaluation of thermal loads is based on the ASME Code philosophy for Service Level A loads given in ASME Code, Section III, Subsection NE, Paragraphs NE-3213.13 and 3221.4.

3.8.3.5.3.5 Design of Trusses

DCD Tier 2, Section 3.8.3.5.3.5, "Design of Trusses," states the following:

The trusses provide a structural framework for the modules, maintain the separation between the faceplates, support the modules during transportation and erection, and act as "form ties" between the faceplates when concrete is being placed. After the concrete has cured, the trusses are not required to contribute to the strength or stiffness of the completed modules. However, they do provide additional shear capacity between the steel plates and concrete as well as additional strength similar to that provided by stirrups in reinforced concrete. The trusses are designed according to the requirements of AISC-N690.

3.8.3.5.3.6 Design of Shear Studs

DCD Tier 2, Section 3.8.3.5.3.6, "Design of Shear Studs," states the following:

The wall structural modules are designed as reinforced concrete elements, with the faceplates serving as reinforcing steel. Since the faceplates do not have deformation patterns typical of reinforcing steel, shear studs are provided to transfer the forces between the concrete and the steel faceplates. The shear studs make the concrete and steel faceplates behave compositely. In addition, the shear studs permit anchorage for piping and other items attached to the walls.

The size and spacing of the shear studs is based on Section Q1.11.4 of AISC-N690 to develop full composite action between the concrete and the steel faceplates.

3.8.3.5.4 Structural Floor Modules

DCD Tier 2, Section 3.8.3.5.4, "Structural Floor Modules," states the following:

[DCD Tier 2,] Figure 3.8.3-3 shows the typical design details of the floor modules. The operating floor is designed for dead, live, thermal, safe-shutdown earthquake, and pressure due to automatic depressurization system operation or due to postulated pipe break loads. The operating floor region above the in-containment refueling water storage tank is a series of structural modules. The remaining floor is designed as a composite structure of concrete slab and steel beams in accordance with AISC-N690.

Design of Structures, Components, Equipment, and Systems

For vertical downward loads, the floor modules are designed as a composite section, according to the requirements of Section Q1.11 of AISC-N690. Composite action of the steel section and concrete fill is assumed based on meeting the intent of Section Q1.11.1 for beams totally encased in concrete. Although the bottom flange of the steel section is not encased within concrete, the design configuration of the floor module provides complete concrete confinement to prevent spalling. It also provides a better natural bonding than the code-required configuration.

For vertical upward loads, no credit is taken for composite action. The steel members are relied upon to provide load-carrying capacity. Concrete, together with the embedded angle stiffeners, is assumed to provide stability to the plates.

Floor modules are designed using the following basic assumptions and related requirements:

- Concrete provides restraint against buckling of steel plates. The buckling unbraced length of the steel plate, therefore, is assumed to equal the span length between the fully embedded steel plates and shapes.

- Although the floor modules forming the top (ceiling) of the in-containment refueling water storage tank are not in contact with water, stainless steel plates are used for the tank boundary.

- The floor modules are designed as simply supported beams.

3.8.3.5.4.1 Design for Vertical Downward Loads

DCD Tier 2, Section 3.8.3.5.4.1, "Design for Vertical Downward Loads," states the following:

The floor modules are designed as a one-way composite concrete slab and steel beam system in supporting the vertical downward loads. The effective width of the concrete slab is determined according to Section Q1.11.1 of AISC-N690. The effective concrete compression area is extended to the neutral axis of the composite section. The concrete compression area is treated as an equivalent steel area based on the modular ratio between steel and concrete material. [DCD Tier 2,] Figure 3.8.3-13 shows the effective composite sections. The steel section is proportioned to support the dead load and construction loads existing prior to hardening of the concrete. The allowable stresses are provided in [DCD Tier 2,] Table 3.8.4-1.

3.8.3.5.4.2 Design for Vertical Upward Loads

DCD Tier 2, Section 3.8.3.5.4.2, "Design for Vertical Upward Loads," states the following:

For vertical upward loads, the floor modules are designed as noncomposite steel structures. The effective width, b_e, of the faceplate in compression is based on post-buckling strength of steel plates and is determined from Equation (4.16) of

[DCD] Reference 44. The faceplates of the structural floor modules are stiffened and supported by embedded horizontal angles. Hence, the buckling unbraced length of the faceplates is equal to the span length between the horizontal angles. Since concrete provides restraint against buckling of the steel plates, a value of 0.65 is used for k [the effective length factor for perismatic members] when calculating the effective length of the steel plates and stiffeners whenever the plate or stiffener is continuous. The buckling stress, f_{cr}, of the faceplates is determined from Sections 9.2 and 9.3 of [DCD] Reference 45. The effective width of the faceplates of the structural floor modules in compression is shown in [DCD Tier 2,] Figure 3.8.3-13. The allowable stresses are provided in [DCD Tier 2,] Table 3.8.4-1.

3.8.3.5.4.3 Design for In-Plane Loads

DCD Tier 2, Section 3.8.3.5.4.3, "Design for In-Plane Loads," states the following:

In-plane shear loads acting on the floor modules are assumed to be resisted only by the steel faceplate without reliance on the concrete for strength. The stresses in the faceplate due to the in-plane loads are combined with those due to vertical loads. The critical stress locations of the floor faceplate are evaluated for the combined normal and shear stress, based on the von Mises yield criterion:

For the particular case of a two-dimension stress condition the equation is:

$$(\sigma_1)^2 - \sigma_1 \, \sigma_2 + (\sigma_2)^2 = (f_y)^2$$

where σ_1 and σ_2 are the principal stresses and f_y is the uniaxial yield stress.

For the faceplate where normal, σ, and shear, τ, stresses are calculated, the principal stresses can be expressed as follows:

$$\sigma_1 \ = \ (\sigma \, / \, 2) \ + \ (\sigma^2/4 + \ \tau^2)^{\frac{1}{2}}$$

$$\sigma_2 \ = \ (\sigma \, / \, 2) \ - \ (\sigma^2/4 + \ \tau^2)^{\frac{1}{2}}$$

Therefore, the condition at yield becomes:

$$\sigma^2 \ + \ 3 \, \tau^2 \ = \ (f_y)^2$$

For the design of the structural floor module faceplate, the allowable stresses for the various loading conditions are as follows:

Normal condition:

$$\sigma^2 \ + 3\tau^2 \ \leq \ (0.6 \, f_y)^2$$

Design of Structures, Components, Equipment, and Systems

Severe condition:

$$\sigma^2 + 3\tau^2 \leq (0.6\, f_y)^2$$

Extreme/abnormal condition:

$$\sigma^2 + 3\,\tau^2 \leq (0.96\, f_y)^2$$

Thermal stresses in the faceplates result from restraint of growth during the thermal transients described in [DCD Tier 2, Section] 3.8.3.3.1. Evaluation for thermal stresses is the same as discussed in [DCD Tier 2, Section] 3.8.3.5.3.4 for the wall modules.

3.8.3.5.5 Internal Steel Framing

DCD Tier 2, Section 3.8.3.5.5, "Internal Steel Framing," states the following:

Internal steel framing is analyzed and designed according to AISC-N690. Seismic analysis methods are described in [DCD Tier 2, Section] 3.7.3.

3.8.3.5.6 Steel Form Modules

DCD Tier 2, Section 3.8.3.5.6, "Steel Form Modules," states the following:

The steel form modules consist of plate reinforced with angle stiffeners and tee sections as shown in [DCD Tier 2,] Figure 3.8.3-16. The steel form modules are designed for concrete placement loads defined in [DCD Tier 2, Section] 3.8.3.3.2.

The steel form modules are designed as steel structures according to the requirements of AISC-N690. This code is applicable since the form modules are constructed entirely out of structural steel plates and shapes and the applied loads are resisted by the steel elements.

The staff reviewed the design procedures and acceptance criteria discussed in DCD Tier 2, Section 3.8.3.5.1, "Reactor Coolant Loop Supports," Section 3.8.3.5.2, "Containment Internal Structures Basemat," Section 3.8.3.5.3, "Structural Wall Modules," Section 3.8.3.5.4, "Structural Floor Modules," Section 3.8.3.5.5, "Internal Steel Framing," and Section 3.8.3.5.6, "Steel Form Modules." The procedures and acceptance criteria are consistent with the guidelines provided in Sections 3.8.3.II.4 and 3.8.3.II.5 of the SRP, and are in accordance with applicable sections of AISC-N690 and ACI-349. The load combinations, design approach, and allowable limits for the concrete and steel components are in agreement with those defined in the referenced SRP sections, AISC-N690, and ACI-349. Therefore, the staff concludes that the design procedures and acceptance criteria are acceptable.

3.8.3.5.7 Design Summary Report

As described in DCD Tier 2, Section 3.8.3.5.7, "Design Summary Report," a design summary report was prepared for the CIS documenting that the structures meet the acceptance criteria

specified in DCD Tier 2, Section 3.8.3.5. During the April 2–5, 2003, audit, the applicant provided the preliminary containment internal structures summary report for review. Since the design summary report had not been completed, the staff could not perform its review of the report in accordance with SRP Section 3.8.3. As indicated in SRP Sections 3.8.3.I.4 and 3.8.3.II.4, the design summary report is reviewed and considered acceptable if it satisfies the guidelines of Appendix C to SRP Section 3.8.4. Review of the design summary report was Open Item 3.8.3.5-2 in the DSER.

During the October 6–9, 2003, audit, the staff reviewed the final design summary report, APP-1100-S3R-001, "Design Summary Report of Containment Internal Structures," dated May 2003. This document is a final summary report of the CIS which contains a description of the CIS; governing codes and regulations; list of design calculations, drawings, and references; materials; design loads and load combinations; analysis and design of the structural modules; and a summary of the results. Based on its review of selected sections of this report during the October 6–9, 2003 audit, the staff concluded the following issues still needed to be addressed by the applicant:

- The columns supporting the operating floor, which is identified as one of the critical sections, need to be evaluated for the final seismic loads.

- Revisions to the design summary report and the DCD were needed to reflect consistent design parameters (i.e., required plate thicknesses for the concrete-filled steel modules).

- An explanation was needed as to why the envelope of the interaction values for the design of the steel wall IRWST columns is less than the representative value given in Table 8.4-4 of the design summary report. This applies to the tables in the design summary report and the DCD.

In Revision 1 of its response to this open item dated October 21, 2003, the applicant indicated that (1) the CIS design summary report had been revised, (2) the design of the critical sections had been reconciled to the final seismic loads, and (3) the proposed revisions and clarifications, which are to be incorporated in the DCD, were included in the revision to the open item response.

The staff's review of this submittal found that the evaluation of the columns supporting the operating floor, which is identified as one of the critical sections, had been reconciled to the final seismic loads. The proposed modifications to the DCD presented design parameters which are consistent with the design summary report. Also, a footnote to DCD Tier 2, Table 3.8.3-7 in the proposed revision to the DCD explains why the envelope of the interaction values for the design of the steel IRWST columns is less than the representative value given in the table. On the basis discussed above, the staff concluded that this open item would be resolved, pending review of the CIS summary report, reconciliation calculation, and modification of the DCD.

The staff verified that revisions to the DCD were included as proposed by the applicant in Revision 1 of its response to Open Item 3.8.3.5-2. During the December 15–16, 2003, audit, the staff reviewed APP-1100-S3R-001, Revision 1, and APP-1000-S3R-001, Revision 1. The review indicated that the design of the critical sections had been reconciled to the final seismic loadings;

Design of Structures, Components, Equipment, and Systems

the results presented in the design summary report and the revised DCD are consistent with the results presented in the reconciliation calculation.

During the review of the design summary report (APP-1100-S3R-001, Revision 1), the staff identified a footnote in Table 8.4-4 containing a summary of the design loads and results, that excludes two locations from the results presented in the table. At these two locations, the maximum stress predicted for the mechanical plus thermal load combination for the IRWST walls exceeded the applicant's acceptance criteria under the case of primary plus secondary stress. During discussions at the December 15–16, 2003, audit, the applicant indicated that the thermal transient used in this load combination is an accident condition which may occur 10 to 25 times over a 60-year period of operation of the plant. Based on the relatively few occurrences of this thermal loading condition, the staff concluded that these secondary thermal stresses (which are self-relieving in nature) will not affect the structural integrity of the IRWST steel walls. The staff's conclusion is consistent with the design philosophy of the ASME Code for Service Level C (Accident Conditions) which does not require consideration of secondary stresses. On the basis of the assessments discussed above, Open Item 3.8.3.5-2 is resolved.

DCD Tier 2, Section 3.8.3.5.7, also states that deviations from the design due to as-procured or as-built conditions are to be evaluated to ensure that: (1) the structural design meets the acceptance criteria specified in DCD Tier 2, Section 3.8; and (2) the seismic floor response spectra meet the acceptance criteria specified in DCD Tier 2, Section 3.7.5.4. The COL applicant must document the evaluation of deviations in an as-built summary report. The staff concurs with the applicant's treatment of deviations from the design. Section 3.8.6 of this report identifies the preparation of the as-built summary report as COL Action Item 3.8.6-2.

3.8.3.5.8 Design Summary of Critical Sections of Internal Structures

DCD Tier 2, Section 3.8.3.5.8, "Design Summary of Critical Sections," provides a description of the critical sections of the CIS. The five selected critical sections include the following:

• southwest wall of the refueling cavity (1.219 m (4 ft) thick)
• south wall of west steam generator cavity (0.762 m (2' 6") thick)
• northeast wall of in-containment refueling water storage tank (0.762 m (2' 6") thick)
• IRWST steel wall
• column supporting operating floor

During the April 2–5, 2003, audit, the applicant provided the following calculations, applicable to the analysis and design of the five critical sections, for staff review:

• APP-1000-S2C-034, Revision 1, "Finite Element Solid-Shell Model of Containment Internal Structures"

• APP-1100-S2C-001, Revision 0, "Static Analysis of CIS—Dead Load and Live Load"

• APP-1100-S2C-002, Revision 1, "Static Analysis of Containment Internal Structures—SSE Equivalent Static Accelerations and Pressures"

- APP-1100-S2C-003, Revision 1, "Static Analysis of Containment Internal Structures—Pressures"

- APP-1100-S2C-004, Revision 0, "Temperature Distribution Through Wall of Steel Plate Module"

- APP-1100-S2C-005, Revision 1, "Static Analysis of Containment Internal Structures—Thermal Analyses"

- APP-1100-S2C-006, Revision 1, "Static Analysis of Containment Internal Structures—Load Combinations"

- APP-1100-S2C-007, Revision 0, "Static Analysis of Containment Internal Structures—Required Steel Area Calculations of Main IRWST Concrete Filled Module Walls"

- APP-1100-S2C-008, Revision 0, "Static Analysis of Containment Internal Structures—IRWST Steel Wall and Main Operating Floor Columns Verifications"

- APP-1000-S2C-009, Revision 0, "IRWST Seismic Sloshing and Wall Flexibility (IRWST)"

These calculations have been identified as final approved calculations. Calculation Nos. APP-1000-S2C-034, Revision 1; APP-1100-S2C-001, Revision 0; APP-1100-S2C-002, Revision 1; APP-1100-S2C-003, Revision 1; APP-1100-S2C-004, Revision 0; APP-1100-S2C-005, Revision 1; and APP-1100-S2C-006, Revision 1, contain the analysis of the CIS which provide the loadings for design of the critical sections. Calculation No. APP-1100-S2C-007, Revision 0 contains the design of the structural wall modules, and Calculation No. APP-1100-S2C-008, Revision 0 contains the design of the IRWST steel wall modules and the columns supporting the main operating floor.

The staff's review included a sampling of the applicant's calculations listed below. The staff's evaluation of these calculations follows and is divided into separate discussions regarding (1) structural wall modules, (2) in-containment refueling water storage tank steel wall, and (3) column supporting operating floor.

- APP-1000-S2C-034, Revision 1
- APP-1100-S2C-001, Revision 0
- APP-1100-S2C-002, Revision 1
- APP-1100-S2C-004, Revision 0
- APP-1100-S2C-005, Revision 1
- APP-1100-S2C-006, Revision 1
- APP-1100-S2C-007, Revision 0
- APP-1100-S2C-008, Revision 0

3.8.3.5.8.1 Structural Wall Modules

DCD Tier 2, Section 3.8.3.5.8.1, "Structural Wall Modules," states the following:

Design of Structures, Components, Equipment, and Systems

[*The subsection summarizes the design of the following critical sections:*

- *Southwest wall of the refueling cavity ([1.219 m (4 ft)] thick)*
- *South wall of west steam generator cavity ([0.762 m (2 ft 6 in.)] thick)*
- *Northeast wall of IRWST ([0.762 m (2 ft 6 in.)] thick)*]*

[*The thicknesses and locations of these walls which are part of the boundary of the in-containment refueling water storage tank are shown in [DCD Tier 2,] Table 3.8.3-3 and [DCD Tier 2,] Figure 3.8.3-18. They are the portions of the structural wall modules experiencing the largest demand. The structural configuration and typical details are shown in [DCD Tier 2,] Figures 3.8.3-1, 3.8.3-2, 3.8.3-8, 3.8.3-14, 3.8.3-15, and 3.8.3-17.*]* The structural analyses are described in [DCD Tier 2, Section] 3.8.3.4 [and] summarized in [DCD Tier 2,] Table 3.8.3-2. The design procedures are described in [DCD Tier 2,] Section 3.8.3.5.3.

[*The three walls extend from the floor of the in-containment refueling water storage tank at Elevation 103'-0" to the operating floor at Elevation 135'-3". The southwest wall is also a boundary of the refueling cavity and has stainless steel plate on both faces. The other walls have stainless steel on one face and carbon steel on the other. For each wall design information is summarized in [DCD Tier 2,] Tables 3.8.3-4, 3.8.3-5 and 3.8.3-6 at three locations. Results are shown at the middle of the wall (mid span at mid height), at the base of the wall at its midpoint of wall base (mid span at base) and at the base of the wall at the end experiencing greater demand (corner at base). The first part of each table shows the member forces due to individual loading. The lower part of the table shows governing load combinations. The steel plate thickness required to resist mechanical loads is shown at the bottom of the table as well as the thickness provided. The maximum principal stress for the load combination including thermal is also tabulated. If this value exceeds the yield stress at temperature, a supplemental evaluation is performed*]* as described in [DCD Tier 2, Section] 3.8.3.5.3.4. [*For these cases, the maximum stress intensity range is shown together with the allowable stress intensity range, which is twice the yield stress at temperature.*]*

The staff noted that DCD Tier 2, Figure 3.8.3-8 (sheet 1 of 3), and Figure 3.8.3-15 (sheets 1 and 2), contained some information that is illegible. DCD Tier 2, Figure 3.8.3-8, also contained unrecognizable symbols for the weld information. The applicant was requested to submit legible copies of these figures to permit review of the technical information contained on these figures. During the April 2–5, 2003, audit, the applicant agreed to revise these figures in the DCD to address the items identified above. This was Confirmatory Item 3.8.3.5-1 in the DSER.

The applicant revised the DCD to correct the identified illegibility and unrecognizable symbols. Therefore, Confirmatory Item 3.8.3.5-1 is resolved.

The staff reviewed Calculation No. APP-1000-S2C-034, Revision 1, which created the 3D FE shell model of the CIS. This model was used to develop an equivalent lumped-mass model and to perform detailed analyses (equivalent static acceleration) for use in the design of the

structural members. The staff examined selected portions of the model development section, including the various structural elements and masses, and reviewed some of the results from the modal analysis and calculation of dead load plus 0.25 times the live load. The staff concludes that the modeling approach appears to be reasonable and in accordance with the description presented in DCD Tier 2, Section 3.8.3.4, and is, therefore, acceptable. Section 3.8.3.4 of this report discusses the basis for the acceptance of the modeling approach.

The staff reviewed Calculation No. APP-1100-S2C-001, Revision 0, and finds that the applicant properly selected the results from Calculation No. APP-1000-S2C-034, Revision 1, to calculate separate database loadings for dead load, live load, and normal operating reactions. Thus, the staff concludes that Calculation No. APP-1100-S2C-001, Revision 0, is acceptable.

The staff's review of Calculation No. APP-1100-S2C-002, Revision 1, confirmed that the applicant correctly selected the seismic equivalent static accelerations that were applied to the 3D finite element model. The maximum equivalent static accelerations obtained from the seismic stick model time-history analysis at varying elevations were used as input to this calculation. Accelerations in the three global directions, as well as accidental torsional moment for eccentricity in the two horizontal directions, were included. In addition, wall pressures were applied to the IRWST walls to account for water sloshing and seismic load amplification due to local wall flexibility. The portions of calculations reviewed were performed in accordance with DCD Tier 2, Section 3.8.3.4, and, therefore, are acceptable. Section 3.8.4.5 of this report discusses the basis for the acceptance of the analysis and design approaches.

Westinghouse Calculation Nos. APP-1100-S2C-004, Revision 0, and APP-1100-S2C-005, Revision 1, contain the thermal analyses for determining the temperature distribution through the wall and the thermal analysis of the structural module walls. The staff reviewed the methodology used to determine the equivalent linear thermal profile and the resulting temperatures input into the thermal analysis. Section 3.8.3.3 of this report contains further staff evaluation of these thermal calculations as part of RAI 220.007. As a result of clarifications provided in RAI 220.007, as discussed in Section 3.8.3.3 of this report, the staff finds Calculation Nos. APP-1100-S2C-004, Revision 0, and APP-1100-S2C-005, Revision 1, acceptable.

The staff's review of Calculation No. APP-1100-S2C-006, Revision 1, confirmed that the applicant used the correct load combinations for the design of the CIS, consistent with the load combinations specified in DCD Tier 2, Section 3.8.3.3. The load combinations are acceptable for the reasons set forth in Section 3.8.3.3 of this report. Therefore, this calculation is acceptable.

Westinghouse Calculation No. APP-1100-S2C-007, Revision 0, contains the design of the IRWST concrete-filled steel module walls. The staff reviewed the approach used to calculate the needed steel area of the structural walls. The calculation determined the necessary steel reinforcement area at various locations in each of the critical walls using the methodology contained in Westinghouse guidance document APP-GW-S1-008, Revision 0, "Design Guide for Reinforcement in Walls and Floor Slabs." During the audit, the applicant indicated that boundary elements are not needed for walls that frame into other walls because the other walls act as boundary elements. The staff found that the applicant's approach for the analysis and design did not meet the criteria of Chapter 21.6, "Structural Walls, Diaphragms, and Trusses," of ACI-349-01 which specifies the criteria for using boundary elements. A similar issue is

Design of Structures, Components, Equipment, and Systems

presented in Section 3.8.4.2 of this report under Open Item 3.8.4.2-1. This was Open Item 3.8.3.5-3 in the DSER.

In its submittal dated June 23, 2003, the applicant provided its response to this open item and indicated that the open item on boundary elements for reinforced concrete walls is addressed in the response to Open Item 3.8.4.2-1. The information presented for the structural modules inside containment referred to some corner details described in the DCD, provided some descriptive information about the steel module configuration, and stated that the calculated compressive stresses for the governing load combination are low in comparison with the code allowable. The response also indicated that the corner details and the low stresses of the structural modules inside containment result in a design that satisfies the intent of Chapter 21.6 of ACI-349-01. With only this descriptive information, the staff was unable to conclude that the applicant has demonstrated that it had satisfied the requirements of Chapter 21.6 of ACI-349-01 relating to boundary elements for structural modules inside the containment.

During the October 6–9, 2003, audit, the applicant justified that boundary elements are not required for the critical sections inside the containment because the maximum compressive stress in the CIS is less than 0.2 f'_c, in accordance with Chapter 21.6 of ACI-349-01. The applicant agreed to revise the response to this open item to document this statement and to provide the maximum compressive stress in the critical sections inside containment. On this basis, the staff considered this open item resolved, pending the staff's review of the applicant's revised response to Open Item 3.8.3.5-3.

In a letter dated October 10, 2003, the applicant submitted Revision 1 to Open Item 3.8.3.5-3 which indicated that the maximum compressive stress in the module walls occurs at the south wall of the steam generator compartment (DCD Tier 2, Table 3.8.3.5, Sheet 3). The compressive stress at this location was calculated to be 4.25 MPa (616 psi), which is less than the threshold of 0.2 f'_c, as specified in ACI 349-01. On this basis, Open Item 3.8.3.5-3 is resolved.

3.8.3.5.8.2 In-Containment Refueling Water Storage Tank Steel Wall

The IRWST steel wall is identified as one of the critical sections. DCD Tier 2, Section 3.8.3.5.8.2, "In-Containment Refueling Water Storage Tank Steel Wall," states the following:

> [*The in-containment refueling water storage tank steel wall is the circular boundary of the in-containment refueling water storage tank. The structural configuration and typical details are shown in sheet 3 of [DCD Tier 2,] Figure 3.8.3-8.*]* The structural analyses are described in [DCD Tier 2, Section] 3.8.3.4 and summarized in [DCD Tier 2,] Table 3.8.3-2. The design procedures are described in [DCD Tier 2, Section] 3.8.3.5.3. [*The steel wall extends from the floor of the in-containment refueling water storage tank at Elevation 103'-0" to the operating floor at Elevation 135'-3". The wall is a [*1.58 cm *(5/8")] thick stainless steel plate. It has internal vertical stainless steel T-section columns spaced* 1.42 m *(4'-8") apart and external hoop carbon steel (L-section) angles spaced [*45.17 cm to 60.96 cm *(18" to 24")] apart. The wall is*

fixed to the adjacent modules and floor except for the top of columns which are
free to slide radially and to rotate around the hoop direction...]*

The structural evaluation is performed for the central and end regions. The central region envelopes results for the wall except for the last four columns at each end. The end region envelopes results for the four columns at each end.

DCD Tier 2, Section 3.8.3.5.8.2, further states:

[*...The wall is evaluated as vertical and horizontal beams. The vertical beams comprise*
the T-section columns plus the effective width of the plate. The horizontal beams
comprise the L-section angles plus the effective width of the plate. When thermal effects
result in stresses above yield, the evaluation is in accordance with the supplemental
criteria...]* as described in [DCD Tier 2, Section] 3.8.3.5.3.4.

The AP1000 evaluations are summarized in DCD Tier 2, Table 3.8.3-7. Design loads and load combinations are shown on sheet 1. Sheet 2 shows the ratio of the design stresses to the allowable stresses.

The staff reviewed selected sections of Calculation No. APP-1000-S2C-008, Revision 0, "Static Analysis of Containment Internal Structures—IRWST Steel Wall and Main Operating Floor Columns Verification," which addresses the design of the steel wall of the IRWST. The loads used as input for the IRWST design were developed in a series of analysis calculations described earlier and evaluated by the staff in Section 3.8.3.5.8.1 of this report and, for the reasons set forth in that section, are acceptable. The staff also reviewed the configuration details of the steel tank wall and stiffeners included in Calculation No. APP-1000-S2C-008, Revision 0, and finds that the design results and configuration details were properly summarized in DCD Tier 2, Figure 3.8.3-8 (sheet 3 of 3). In addition, the calculation followed the design approach described in DCD Tier 2, Section 3.8.3.5.3, and therefore, the staff concludes this calculation is acceptable. Section 3.8.3.5 of this report discusses the basis for the acceptance of the design approach.

3.8.3.5.8.3 Column Supporting Operating Floor

DCD Tier 2, Section 3.8.3.5.8.3, states the following:

[*This subsection summarizes the design of the most heavily loaded column in the*
containment internal structures. The column extends from Elevation 107'-2" to
the underside of the operating floor at Elevation 135'-3". In addition to supporting
the operating floor, it also supports a steel grating floor at Elevation 118'-0".

The load combinations in [DCD Tier 2, Table] 3.8.4-1 were used to assess the
adequacy of the column. For mechanical load combinations, the maximum
interaction factor due to biaxial bending and axial load is 0.59. For load
combinations with thermal loads the interaction factor is 0.94. Since the
interaction factors are less than 1, the column is adequate for all the applied
loads.]*

Design of Structures, Components, Equipment, and Systems

DCD Tier 2, Sections 3.8.3.5.8.1, 3.8.3.5.8.2, and 3.8.3.5.8.3, as well as the referenced tables for the critical sections of the structural wall modules, IRWST steel wall, and the columns supporting the operating floor, did not contain the final AP1000 design results. The applicant agreed to update this information in the DCD. This was Confirmatory Item 3.8.3.5-2 in the DSER.

The applicant updated the DCD sections and related tables to include the final AP1000 design results for the CIS. Therefore, Confirmatory Item 3.8.3.5-2 is resolved.

3.8.3.6 Materials, Quality Control, and Construction Techniques

In DCD Tier 2, Section 3.8.3.6, the applicant states that DCD Tier 2, Section 3.8.4.6, describes the materials and quality control program used in the construction of the containment internal structures. The staff evaluation for materials and quality control program is in Section 3.8.4.6 of this report.

In DCD Tier 2, Section 3.8.4.6.3 "Special Construction Techniques", the applicant states that the construction techniques for the structural modules are the same as special construction techniques for the containment internal structures, discussed in DCD Tier 2, Section 3.8.3.6.1. The staff evaluation for special construction techniques is in Section 3.8.3.6 of this report.

DCD Tier 2, Section 3.8.3.6, "Materials, Quality Control, and Special Construction Techniques," states the following:

> [DCD Tier 2, Section] 3.8.4.6 describes the materials and quality control program used in the construction of the containment internal structures. The structural steel modules are constructed using A36 plates and shapes. Nitronic 33 (American Society for Testing and Materials 240, designation S24000, Type XM-29) stainless steel plates are used on the surfaces of the modules in contact with water during normal operation or refueling. The structural wall and floor modules are fabricated and erected in accordance with AISC-N690. Loads during fabrication and erection due to handling and shipping are considered as normal loads as described in DCD Tier 2, Section 3.8.4.3.1.1. 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 AISC-N690).

3.8.3.6.1 Fabrication, Erection, and Construction of Structural Modules

DCD Tier 2, Section 3.8.3.6.1, "Fabrication, Erection, and Construction of Structural Modules," states the following:

> Modular construction techniques are used extensively in the containment internal structures ([DCD Tier 2,] Figure 3.8.3-1). Subassemblies, sized for commercial rail shipment, are assembled offsite and transported to the site. Onsite fabrication consists of combining the subassemblies in structural modules, which are then installed in the plant. A typical modular construction technique is described in the following paragraphs for Module CA01, which is the main structural module in the containment internal structures.

The CA01 module is a multicompartmented structure which, in its final form, comprises the central walls of the containment internal structures. The vertical walls of the module house the refueling cavity, the reactor vessel compartment, and the two steam generator compartments. The module ([DCD Tier 2,] Figure 3.8.3-14) is in the form of a "T" and is approximately [26.8 m (88 ft)] long, [29 m (95 ft)] wide and [26.2 m (86 ft)] high. The module is assembled from about 40 prefabricated wall sections called structural submodules ([DCD Tier 2,] Figure 3.8.3-15). The submodules are designed for railroad transport from the fabricator's shop to the plant site with sizes up to [3.66 m by 3.66 m by 24.4 m (12 ft by 12 ft by 80 ft)] long, weighing up to [72.6 metric tons (80 tons)]. A typical submodule weighs between [8.16 and 9.98 metric tons (9 and 11 tons)]. The submodules are assembled outside the nuclear island with full penetration welds between the faceplates of adjacent subunits. The completed CA01 module is lifted to its final location within the containment vessel by the heavy lift construction crane. Following placement of the CA01 module within the containment building, the hollow wall structures are filled with concrete, forming a portion of the structural walls of the containment internal structures.

Tolerances for fabrication, assembly and erection of the structural modules conform to the requirements of Section 4 of ACI-117, Sections 3.3 and 3.4 of AWS D1.1, and Sections Q1.23 and Q1.25 of AISC-N690.

3.8.3.6.2 Nondestructive Examination

DCD Tier 2, Section 3.8.3.6.2, "Nondestructive Examination," states the following:

Nondestructive examination of the submodules and module is performed according to AISC-N690 and AWS D1.1. Welds are visually examined for 100 percent of their length. Full penetration welds are inspected by ultrasonic or radiographic examination for 10 percent of their length. Partial penetration welds are inspected by magnetic particle or liquid penetrant examination for 10 percent of their length.

3.8.3.6.3 Concrete Placement

DCD Tier 2, Section 3.8.3.6.3, "Concrete Placement," states the following:

After installation of the CA01 module in the containment, the hollow walls are filled with concrete. Concrete is placed in each wall continuously from bottom to top. The concrete is placed through multiple delivery trunks located along the top of the wall. It is placed in incremental layers with the placement rate based on the pressure of the wet concrete and its setting time. During concrete placement, workers and inspectors have access to the inside of the modules. The arrangement of the module internal trusses provides a means for the free flow of concrete and movement of personnel.

The special construction techniques to be employed for AP1000 are identical to the special construction techniques that were reviewed during the AP600 design certification process, and

Design of Structures, Components, Equipment, and Systems

documented in Section 3.8.3.6 of the AP600 FSER. These techniques are consistent with SRP Section 3.8.3.II.6 and applicable sections of AISC-N690; ACI-349; ACI-117; American Welding Society (AWS) D1.1; Nuclear Quality Assurance (NQA)-2, Part 2.2; and the American Society for Testing and Materials (ASTM) (for Nitronic 33 material.) On the basis that the staff has already accepted these for AP600, the staff also finds them acceptable for AP1000.

3.8.3.7 Conclusions

For the reasons set forth above in Sections 3.8.3.1 through 3.8.3.6 of this report, the staff concludes that the design of the CIS is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDC 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 NRC RGs and industry standards, the applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the CIS are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with their safety functions.

• The applicant has met the requirements of GDC 2 by designing the AP1000 CIS to withstand the 0.3 g SSE and appropriate combinations of the effects of normal and accident conditions, including the effects of environmental loadings such as earthquakes and other natural phenomena, with sufficient margin for limitations in site data.

• The applicant has met the requirements of GDC 4 by ensuring that the design of the CIS is capable of withstanding the dynamic effects associated with missiles, pipe whip, and fluid discharges (excluding dynamic effects associated with pipe ruptures, the probability of which is extremely low under conditions consistent with the design basis for the piping).

• The applicant has met the requirements of GDC 50 by designing the CIS to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from postulated accidents. In meeting these design requirements, the applicant has followed the recommendations of the RGs and industry standards. The applicant 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 AP1000 CIS, 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 to established codes, standards, and specifications acceptable to the staff, including RGs 1.57, 1.94, and 1.142, as well as 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 and Pressure Vessel Code, Section III, Subsection NE

- AISC-N690-1994, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities"

In addition, the applicant has used these criteria (as defined by applicable codes, standards, and specifications regarding loads and load combinations), design and analysis procedures, structural acceptance criteria, materials, quality control programs, special construction techniques, and testing and inservice surveillance requirements. Together, these considerations provide reasonable assurance that the CIS will withstand the specified design conditions without losing their structural integrity or the capability to perform their safety functions in the event of earthquakes and various postulated accidents.

Furthermore, the staff's conclusion regarding the design of the CIS is based on its review of a sample of design calculations for the critical sections of the internal structures described in DCD Tier 2, Section 3.8.3.5.8, which are designated as Tier 2*.

3.8.4 Other Seismic Category I Structures

SRP Section 3.8.4 provides guidelines for the staff to use in performing its review of issues related to all seismic Category I structures and other safety-related structures, except for the containment structure and foundation mat. These review areas include description of structures; applicable codes, standards, and specifications; loads and loading combinations; design and analysis procedures; structural acceptance criteria; and material, quality control, special construction techniques, and quality assurance. This SRP section also states that the RGs and industry standards identified in SRP Section 3.8.4.II.2 provide information, recommendations, and guidance, and in general, describe a basis acceptable to the staff that may be used to implement the requirements of 10 CFR 50.55a, GDC 1, 2, 4, and 5, and Appendix B to 10 CFR Part 50.

Using the guidance described in Section 3.8.4 of the SRP and related RGs, the staff reviewed DCD Tier 2, Section 3.8.4. In particular, the review of this section focused on the analysis and design of the other seismic Category I structures included in the AP1000 design, with emphasis on the (1) material, (2) geometry, (3) codes and standards, (4) loadings, and (5) design and analysis procedures.

The applicant defined the other seismic Category I structures as the shield building, the auxiliary building, the containment air baffle, Category I cable tray supports, and Category I HVAC supports. New fuel and spent fuel racks are described in DCD Tier 2, Section 9.1, "Fuel Storage and Handling."

3.8.4.1 Description of Other Seismic Category I Structures

SRP Section 3.8.4.II.1 states that the descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the minimum standards set forth in Section 3.8.4.1 of RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." In this case, the SAR is the same as the DCD for the AP1000. New or unique design features that are not specifically covered in RG 1.70 may require a more detailed review. The reviewer

Design of Structures, Components, Equipment, and Systems

determines the additional information that may be required to accomplish a meaningful review of the structural aspects of such new or unique features.

3.8.4.1.1 Shield Building

DCD Tier 2, Section 3.8.4.1.1, "Shield Building," states the following:

>The shield building is the shield building structure and annulus area that surrounds the containment building. It shares a common basemat with the containment building and the auxiliary building. The shield building is a reinforced concrete structure. The figures in [DCD Tier 2,] Section 1.2 show the layout of the shield building and its interface with the other buildings of the nuclear island.

>The following are the significant features and the principal systems and components of the shield building:

>- shield building cylindrical structure
>- shield building roof structure
>- lower annulus area
>- middle annulus area
>- upper annulus area
>- passive containment cooling system air inlet
>- passive containment cooling system water storage tank
>- passive containment cooling system air diffuser
>- passive containment cooling system air baffle
>- passive containment cooling system air inlet plenum

>The cylindrical section of the shield building provides a radiation shielding function, a missile barrier function, and a passive containment cooling function. Additionally, the cylindrical section structurally supports the roof structure with the passive containment cooling system water storage tank and serves as a major structural member for the nuclear island. The floor slabs and structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

>The shield building roof structure is a reinforced concrete conical shell supporting the passive containment cooling system tank and air diffuser. Air intakes are located at the top of the cylindrical portion of the shield building. The conical roof supports the passive containment cooling system tank as shown in [DCD Tier 2,] Figure 3.8.4-2. The air diffuser is located in the center of the roof and discharges containment cooling air upwards.

During the November 11–15, 2002, audit meeting, the staff raised a question as to whether the applicant may have revised the AP600 DCD Tier 2, Figure 3.8.4-7, to reflect the AP1000 shield building roof, tank dimensions, and elevations in DCD Tier 2, Figure 3.8.4-2. While the structures shown in AP600 DCD Tier 2, Figure 3.8.4-7, were designated as Tier 2*, there is no such designation in AP1000 DCD Tier 2, Figure 3.8.4-2. During the design audit on April 2–5,

2003, the applicant acknowledged the need to address this finding and will revise the DCD to designate this information as Tier 2*. This was Confirmatory Item 3.8.4.1-1 in the DSER.

The staff verified that the applicant designated DCD Tier 2, Figure 3.8.4-2 as Tier 2*, which resolved Confirmatory Item 3.8.4.1-1.

DCD Tier 2, Section 3.8.4.1.1, further states the following:

> The passive containment cooling system tank has a stainless steel liner which provides a leak-tight barrier on the inside surfaces of the tank. The wall liner consists of a plate with stiffeners on the concrete side of the plate. The floor liner is welded to steel plates embedded in the surface of the concrete. The liner is welded and inspected during construction to assure its leak-tightness. Leak chase channels are provided over the liner welds. This permits monitoring for leakage and also prevents degradation of the reinforced concrete wall due to freezing and thawing of leakage. The exterior face of the reinforced concrete boundary of the PCS tank is designed to control cracking in accordance with Section 10.6.4 of ACI-349 as the reinforcing steel is maintained at a stress limit based on sustained loads including thermal effects.

> The upper annulus of the shield building is the volume of the annulus between Elevation 132'-3" and the bottom of the air diffuser. The middle annulus area, the volume of annulus between Elevation 100'-0" and Elevation 132'-3", contains the majority of the containment vessel penetrations. The area below Elevation 100'-0" is the lower annulus of the shield building. There is a concrete floor slab in the annulus at Elevation 132'-3", which is supported by the stiffeners that are attached to the containment vessel.

> A permanent flexible watertight and airtight seal is provided between the concrete floor slab at Elevation 132'-3" and the shield building to provide an environmental barrier between the upper and middle annulus sections. The flexible watertight seal is utilized to seal against water leakage from the upper annulus into the middle annulus. The seal is designated as non-safety-related and nonseismic and is not relied upon to mitigate design-basis events. The seal is able to accommodate events resulting in containment temperature and pressure excursions that result in lateral shell movement inward or outward.

3.8.4.1.2 Auxiliary Building

DCD Tier 2, Section 3.8.4.1.2, "Auxiliary Building," states the following:

> The auxiliary building is a reinforced concrete and structural steel structure. Three floors are above grade and two are located below grade. It is one of the three buildings that make up the nuclear island and shares a common basemat with the containment building and the shield building.

> The auxiliary building is a C-shaped section of the nuclear island that wraps around approximately 50 percent of the circumference of the shield building. The

floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

The figures in [DCD Tier 2,] Section 1.2 show the layout of the auxiliary building and its interface with the other buildings of the nuclear island. The following are the significant features and the principal systems and components of the auxiliary building:

- main control room
- remote shutdown area
- Class 1E dc switchgear
- Class 1E batteries
- reactor trip switchgear
- reactor coolant pump trip switchgear
- main steam and feedwater piping
- main control room heating, ventilating, and air conditioning (HVAC)
- Class 1E switchgear rooms heating, ventilating, and air conditioning
- spent fuel pool
- fuel transfer canal
- cask loading and washdown pits
- new fuel storage area
- cask handling crane
- fuel handling machine
- chemical and volume control system (CVS) makeup pumps
- normal residual heat removal system (RNS) pumps and heat exchangers
- liquid radwaste tanks and components
- spent fuel cooling system
- gaseous radwaste processing system
- mechanical and electrical containment penetrations

Structural modules are used for part of the south side of the auxiliary building. These structural modules are structural elements built up with welded steel structural shapes and plates. Concrete is used where required for shielding, but reinforcing steel is not normally used. These modules include the spent fuel pool, fuel transfer canal, and cask loading and cask washdown pits. The configuration and typical details of the structural modules are the same as for the structural modules described in [DCD Tier 2, Section] 3.8.3.1 for the containment internal structures. [DCD Tier 2,] Figure 3.8.4-4 shows the location of the structural modules. The thickness of the structural wall modules ranges from [0.762 m (2'-6") to 1.524 m (5-0")]. The structural modules extend from Elevation 66'-6" to Elevation 135'-3". The minimum thickness of the faceplates is [1.27 cm (0.5 in.)].

The ceiling of the main control room (floor at Elevation 135'-3"), and the instrumentation and control rooms (floor at Elevation 117'-6") are designed as finned floor modules ([DCD Tier 2,] Figure 3H.5-9). A finned floor consists of a [60.96 cm (24 in.)] thick concrete slab poured over a stiffened steel plate ceiling. The fins are rectangular plates welded perpendicular to the faceplate of the floor module. Shear studs are welded on the other side of the steel plate, and the

steel and concrete act as a composite section. The fins are exposed to the environment of the room, and enhance the heat-absorbing capacity of the ceiling (see [DCD Tier 2, Section] 6.4.2.2). Several shop-fabricated steel panels, placed side by side, are used to construct the stiffened plate ceiling in a modularized fashion. The stiffened plate is designed to withstand construction loads prior to concrete hardening.

The new fuel storage area is a separate reinforced concrete pit providing temporary dry storage for the new fuel assemblies.

A cask handling crane travels in the east-west direction. The location and travel of this crane prevents the crane from carrying loads over the spent fuel pool, thus precluding them from falling into the spent fuel pool.

3.8.4.1.3 Containment Air Baffle

DCD Tier 2, Section 3.8.4.1.3, "Containment Air Baffle," states the following:

The containment air baffle is located within the upper annulus of the shield building, providing an air flow path for the passive containment cooling system. The air baffle separates the downward air flow entering at the air inlets from the upward air flow that cools the containment vessel and flows out of the discharge stack. The upper portion is supported from the shield building roof and the remainder is supported from the containment vessel. The air baffle is a seismic Category I structure designed to withstand the wind and tornado loads defined in [DCD Tier 2,] Section 3.3. The air baffle structural configuration is depicted in [DCD Tier 2,] Figures 1.2-14 and 3.8.4-1. The baffle includes the following sections:

- A wall supported off the shield building roof (see [DCD Tier 2,] Figure 1.2-14)

- A series of panels attached to the containment vessel cylindrical wall and the knuckle region of the dome

- A sliding plate for closing the gap between the wall and the panels fixed to the containment vessel, designed to accommodate the differential movements between the containment vessel and shield building

- Flow guides attached at the bottom of the air baffle to minimize pressure drop

The air baffle is designed to meet the following functional requirements:

- The baffle and its supports are configured to minimize pressure losses as air flows through the system

- The baffle and its supports have a design objective of 60 years

- The baffle and its supports are configured to permit visual inspection and maintenance of the air baffle as well as the containment vessel to verify the condition of the coatings

- The baffle is designed to maintain its function during postulated design-basis accidents

- The baffle is designed to maintain its function under specified external events including earthquakes, hurricanes and tornadoes

The design of the containment air baffle is shown in [DCD Tier 2,] Figure 3.8.4-1. The portion of the air baffle attached to the containment cylinder comprises 60 panels circumferentially in each of seven rows vertically, with each panel subtending an arc of six degrees (approximately [2.11 m (6 feet 11 inches)] wide). Each panel is supported by horizontal beams spaced approximately [4.16 m (13 feet 8 inches)] apart. These horizontal beams span the six-degree arc and are bolted to U-shaped attachments welded to the containment vessel. The attachment locations are established considering the containment vessel plate and ring assemblies, as shown in [DCD Tier 2,] Figure 3.8.2-1. The lowest attachments are at the bottom of the middle containment ring subassembly. The upper attachments are on the head. The attachments can be installed in the subassembly area and, therefore, should not interfere with the containment vessel erection welds. The only penetrations through the containment vessel above the operating deck at Elevation 135'-3" are the main equipment hatch and personnel airlock. Five air baffle panels are deleted to provide openings at the equipment hatch and two flow guides at the personnel airlock.

Two rows of panels are attached to the containment vessel above the cylindrical portion. The panels are curved to follow the curvature of the knuckle region of the head and then become flat forming a conical baffle that provides a transitional flow region into the upper shield building. A vertical sliding plate is provided between this upper row of panels and the air baffle that is attached directly to the shield building roof as shown in sheet 4 of [DCD Tier 2,] Figure 3.8.4-1. This sliding plate rests on the [30.48 cm (12 in.)] wide horizontal top surface of the upper row of panels. At ambient conditions the vertical sliding plate is approximately centered on the horizontal plate. The sliding plate is set at ambient conditions to permit relative movements from minus [5.08 cm (2 in.) to plus 7.62 cm(3 in.)] radially and minus [2.54 cm (1 in.)] to plus [10.16 cm (4 in.)] vertically. This accommodates the differential movement between the containment vessel and the shield building, based on the absolute sum of the containment pressure, temperature deflections, and the seismic deflections, such that the integrity of the air baffle is maintained.

The panels accommodate displacements between each panel due to containment pressure and thermal growth. Radial and circumferential growth of the containment vessel are accommodated by slip at the bolts between the horizontal beams and the U-shaped attachment resulting in small gaps between adjacent

panels. Vertical growth is accommodated by slip between the panel and the horizontal beam supporting the top of the panel. Cover plates between the panels limit leakage during and after occurrence of these differential displacements.

3.8.4.1.4 Seismic Category I Cable Tray Supports

DCD Tier 2, Section 3.8.4.1.4, "Seismic Category I Cable Tray Supports," states the following:

Electric cables are routed in horizontal and vertical steel trays supported by channel type struts made out of cold rolled channel type sections. Spacing of the supports is determined from allowable loads in the trays and stresses in the supports. The supports are attached to the walls, floors, and ceiling of the structures as required by the arrangement of the cable trays. Longitudinal and transverse bracing is provided where required.

3.8.4.1.5 Seismic Category I Heating, Ventilating, and Air Conditioning Duct Supports

DCD Tier 2, Section 3.8.4.1.5, "Seismic Category I Heating, Ventilating, and Air Conditioning Duct Supports," states the following:

Heating, ventilating, and air conditioning duct supports consist of structural steel members or cold rolled channel type sections attached to the walls, floors, and ceiling of the structures as required by the arrangement of the duct. Spacing of the supports is determined by allowable stresses in the duct work and supports. Longitudinal and transverse bracing is provided where required.

3.8.4.1.6 Conclusions

Since the descriptive information and referenced figures in DCD Tier 2, Section 3.8.4.1, contain sufficient detail to define the primary structural aspects and elements relied upon for each structure to perform its safety-related functions, in accordance with SRP Section 3.8.4, the staff finds the descriptive information acceptable.

3.8.4.2 Applicable Codes, Standards, and Specifications

In DCD Tier 2, Section 3.8.4.2, the applicant identified the following standards as applicable to the design, materials, fabrication, construction, inspection, or testing of the AP1000. DCD Tier 2, Section 1.9, describes conformance with applicable RGs.

- [*American Concrete Institute (ACI), Code Requirements for Nuclear Safety Related Structures, ACI-349-01*]* (refer to [DCD Tier 2, Section] 3.8.4.5, for supplemental requirements)

- American Concrete Institute (ACI), ACI Detailing Manual, 1994

- [*American Institute of Steel Construction (AISC), Specification for the Design, Fabrication, and Erection of Steel in Safety Related Structures for Nuclear*

Design of Structures, Components, Equipment, and Systems

> *Facilities, AISC-N690-1994*]* (refer to [DCD Tier 2, Section] 3.8.4.5, for
> supplemental requirements)

- American Iron and Steel Institute (AISI), Specification for the Design of Cold
  Formed Steel Structural Members, Parts 1 and 2, 1996 Edition and 2000
  Supplement

- American Welding Society (AWS), Structural Welding Code, AWS D 1.1-2000

- American Welding Society (AWS), Reinforcing Steel Welding Code,
  AWS D 1.4-98

- National Construction Issues Group (NCIG), Visual Weld Acceptance Criteria for
  Structural Welding at Nuclear Power Plants, NCIG-01, Revision 2, May 7, 1985

DCD Tier 2, Section 3.8.4.2, further states:

> Welding and inspection activities for seismic Category I structural steel, including
> building structures, structural modules, cable tray supports and heating,
> ventilating, and air conditioning duct supports are accomplished in accordance
> with written procedures and meet the requirements of the American Institute of
> Steel Construction (AISC-N690). The weld acceptance criteria is as defined in
> NCIG-01, Revision 2. The welded seam of the plates forming part of the leaktight
> boundary of the spent fuel pool and fuel transfer canal are examined by liquid
> penetrant and vacuum box after fabrication to confirm that the boundary does not
> leak.

DCD Tier 2, Section 3.8.4.5.1, "Supplemental Requirements for Concrete Structures," states that
"supplemental requirements for ACI-349 are given in the position on Regulatory Guide 1.142 in
DCD [Tier 2] Appendix 1A." According to SRP Section 3.8.4.II.2, the use of ACI-349 Code for
the design of seismic Category I structures other than the containment building is acceptable to
the staff. In DCD Tier 2, Appendix 1A, "Conformance with Regulatory Guides," the applicant
indicated that the AP1000 position "conforms" to all applicable Positions C.1 through C.15 of
RG 1.142, Revision 2. The staff notes a general exception because this RG endorses
ACI-349-97, rather than ACI-349-01. The applicant stated that "[t]he AP1000 uses the latest
version of industry standards as of October 2001." Since the staff has not formally reviewed and
endorsed ACI-349-01 at this time, the staff requested, in RAI 220.013, the applicant to
specifically identify all deviations between ACI-349-97/RG 1.142 and ACI-349-01/Westinghouse
position (identified in DCD Tier 2, Appendix 1A) that affect the AP1000 design. The applicant
was also requested to provide the technical basis for ensuring that a comparable level of safety
is achieved for each such deviation. In its response to RAI 220.013 (Revision 1), the applicant
indicated that no differences exist between ACI-349-97/RG 1.142 and
ACI-349-01/Westinghouse position that affect the AP1000 design. The staff finds the applicant's
response acceptable.

In RAI 220.014, the staff also noted that DCD Tier 2, Section 3.8.4.5.2, identifies the exact same
supplemental criteria for AISC-N690-94 that the staff had previously accepted for AISC-N690-84
during its review of other advanced reactors, including the AP600 review. However,

AISC-N690-94 has not been formally reviewed and accepted by the staff at this time. Therefore, the staff requested the applicant to identify all deviations between AISC-N690-84 (with NRC-accepted supplemental requirements) and AISC-N690-94 (with identical supplemental requirements) that affect the AP1000 design. The staff also asked the applicant to discuss the technical basis for ensuring that a comparable level of safety is achieved for each such deviation. In its response to RAI 220.014 (Revision 0), the applicant indicated that no differences exist between AISC-N690-84 and AISC-N690-94 that would affect the AP1000 design; consequently, the structural steel design criteria for the AP1000 and the staff's position developed for the use of AISC-N690-84 are the same. The staff finds this to be acceptable.

During the design audit on April 2–5, 2003, the staff reviewed the applicant's approach to the design of boundary elements that might be needed to reinforce boundaries and edges around openings of structural walls. In accordance with Chapter 21.6 of ACI-349-01, if the vertical compressive stress at the opening does not exceed 0.2 f'_c, then a boundary element is not needed. The applicant contended that this compressive stress limit is not applicable when seismic member forces are based on elastic analysis and no ductility reduction factor is applied. According to the applicant, under such conditions, a stress limit of 1.0 f'_c may be used as the stress threshold for boundary elements. The staff disagreed with the approach proposed by the applicant, and pointed out that the stress prediction at an opening is highly dependent on the finite element mesh refinement. In addition, the staff's review of Westinghouse Calculation APP-1200-CCC-102, "Auxiliary Building Wall 7.3 Reinforcement Calculation," indicated that boundary element evaluations were not considered at the intersection of reinforced concrete walls. The staff took the position that the need for boundary elements around openings and at intersections of reinforced concrete walls should be evaluated in accordance with Chapter 21.6 of ACI-349-01. The applicant agreed to consider the staff's position and to develop criteria to implement the provisions of Chapter 21.6 of ACI-349-01. This was Open Item 3.8.4.2-1 in the DSER.

In its submittal dated June 23, 2003, the applicant presented its technical position related to the implementation of the provisions of Chapter 21.6 of ACI-349-01. The information provided by the applicant did not include criteria to implement the provisions of Chapter 21.6 of ACI 349-01 Code for boundary elements. The applicant also did not substantiate the following statement made in the response: "The requirement in the code for boundary elements is applicable to a single wall and is based on the assumption that the boundary element may have to carry all compressive forces at the critical section at the time when maximum lateral forces are acting on the structural wall." For the design of the shield building columns between the air inlet openings, the response refers to DCD Tier 2, Section 3H.5.6.2, "Column (Shear Wall) Between Air Inlets." The staff's review of this section found that this portion of the shield building is designed as a shear wall and the compressive stress exceeds the 0.2 f'_c threshold, which would require boundary elements. The staff was unable to ascertain whether the boundary element requirements of the ACI code are utilized for design. The justifications provided in the applicant's response only addressed the boundary elements for the shield building and Wall 7.3 of the auxiliary building. The applicant did not address boundary element requirements for other reinforced concrete walls.

In a telephone conference on August 22, 2003, the applicant indicated that only the intersection of the shield building wall and Wall 7.3 of the auxiliary building required evaluation for boundary elements; the evaluation is documented in detailed critical section calculations. During the

Design of Structures, Components, Equipment, and Systems

October 6–9, 2003, audit, the applicant presented the detailed calculations for the conditions in which the need for boundary elements had been considered in the design, and described how boundary elements are to be evaluated for certain walls, such as the shield building wall region at the vent openings. The applicant also clarified why other walls outside containment do not require boundary elements. At the conclusion of the discussion during the audit, the applicant committed to document the results of these discussions in the updated response to this open item.

In its submittal dated October 21, 2003, the applicant provided Revision 1 of the response to this open item and explained why the boundary element requirements presented in Section 21.6 of the ACI-349-01 Code were not implemented for structural walls. The revised response also explained that Paragraph 21.6.1 in the earlier 1997 Code states that, "For shear walls with h_w/l_w of less than 2.0, provisions of 21.6.5 can be waived." In the 1997 Code, Section 21.6.5 provided the boundary element requirements for structural walls. When the 2001 Code renumbered Section 21.6.5 to 21.6.6, Paragraph 21.6.1 was not revised accordingly to reference Section 21.6.6. Commentary Paragraph R21.6 in ACI-349R-01 also confirms this. This commentary paragraph to the ACI-349-01 Code explains that boundary elements are essentially required to provide adequate deformability and confinement for flexure and, therefore, the Code does not require the consideration of boundary element for shear walls with aspect ratios of less than 2.0. The staff's review of the 1997 ACI-349 Code and the 2001 Code, including the commentary to the Code, validated the applicant's justification.

The revised response also states that Wall 7.3 has the maximum height-to-length ratio of all walls in the auxiliary building. The height-to-length ratio for this wall is equal to 2.2. Considering that this wall is structurally connected to the shield building and its overall response is controlled by the shield building, which has a height-to-length ratio well below 2.0, the applicant concluded that all walls are controlled by shear, and boundary elements are not required. The staff agrees that the horizontal response in the plane of Wall 7.3 is controlled by the shield building. In addition, the height-to-length ratio of the shield building, as stated in the response, is between 1.3 and 1.8, depending on whether the cylindrical portion or the entire height of the shield building is used. On this basis, the staff concurs that boundary element requirements contained in Section 21.6.6 of ACI-349-01 do not need to be considered for the design of critical walls outside containment. Therefore, this issue is technically resolved.

However, in the revised response, the applicant also indicated its plans to revise DCD Tier 2, Section 3.8.4.5.1, to include the following statement:

> The last sentence in 21.6.1 of ACI-349-01 is revised to state, "For shear walls with h_w/l_w of less than 2.0, provisions of 21.6.6 can be waived." This is an editorial correction consistent with the requirements of ACI-349-97 endorsed by Regulatory Guide 1.142.

The staff did not accept the wording proposed by the applicant since it implies that the DCD can be used as a basis to revise ACI-349-01. The proposed DCD revision remained unresolved.

In Revision 2 of the response to Open Item 3.8.4.2-1 dated November 18, 2003, the applicant revised the above wording to state, "The errata for ACI-349-01 are being updated to include this

3-172

correction." This wording was incorporated in DCD Tier 2, Section 3.8.4.5.1 and is acceptable to the staff. On this basis, Open Item 3.8.4.2-1 is resolved.

The staff finds the referenced codes and standards to be acceptable because they are consistent with the acceptance criteria of SRP Section 3.8.4. The applicant confirmed, in its responses to RAIs 220.013 and 220.014, that the use of ACI-349-01 instead of ACI-349-97, and the use of N690-94 instead of N690-84, do not introduce any deviations from the current staff positions documented in RG 1.142.

3.8.4.3 <u>Loads and Load Combinations</u>

In DCD Tier 2, Section 3.8.4.3, "Loads and Load Combinations," the applicant defined the normal, severe environmental, extreme environmental, and abnormal loads, in addition to load combinations used in the design of other seismic Category I structures.

3.8.4.3.1 Normal Loads

DCD Tier 2, Section 3.8.4.3.1.1, "Normal Loads," states the following:

> Normal loads are those loads to be encountered, as specified, during initial construction stages, during test conditions, and later, during normal plant operation and shutdown:

$D =$ Dead loads or their related internal moments and forces, including any permanent piping and equipment loads.

$F =$ Lateral and vertical pressure of liquids or their related internal moments and forces.

$L =$ Live loads or their related internal moments and forces, including any movable equipment loads and other loads that vary with intensity and occurrence.

$H =$ Static earth pressure or its related internal moments and forces.

$T_o =$ Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

$R_o =$ Piping and equipment reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

3.8.4.3.2 Severe Environmental Loads

DCD Tier 2, Section 3.8.4.3.1.2, "Severe Environmental Loads," states the following:

Design of Structures, Components, Equipment, and Systems

The severe environmental load is:

W = Loads generated by the design wind specified for the plant in [DCD Tier 2, Section] 3.3.1.1.

3.8.4.3.3 Extreme Environmental Loads

DCD Tier 2, Section 3.8.4.3.1.3, "Extreme Environmental Loads," states the following:

Extreme environmental loads are:

E_s = Loads generated by the safe-shutdown earthquake specified for the plant, including the associated hydrodynamic and dynamic incremental soil pressure. Loads generated by the safe-shutdown earthquake are specified in [DCD Tier 2,] Section 3.7.

W_t = Loads generated by the design tornado specified for the plant in [DCD Tier 2, Section] 3.3.2, including loads due to tornado wind pressure, differential pressure, and tornado-generated missiles.

N = Loads generated by the probable maximum precipitation (provided previously in [DCD Tier 2,] Table 2.0-1).

3.8.4.3.4 Abnormal Loads

DCD Tier 2, Section 3.8.4.3.1.4, "Abnormal Loads," states the following:

Abnormal loads are those loads generated by a postulated high-energy pipe break accident for pipes not qualified for leak-before-break. Abnormal loads include the following:

P_a = Pressure load within or across a compartment generated by the postulated break. The main steam isolation valve (MSIV) and steam generator blowdown valve compartments are designed for a pressurization load of [41.37 kPa (6 psi)]. The subcompartment design pressure bounds the pressurization effects due to postulated breaks in high energy pipe. Determination of subcompartment pressure loads is discussed in [DCD Tier 2, Section] 6.2.1.2.

T_a = Thermal loads under thermal conditions generated by the postulated break and including T_o. Determination of subcompartment temperatures is discussed in [DCD Tier 2, Section] 6.2.1.2.

R_a = Piping and equipment reactions under thermal conditions generated by the postulated break and including R_o.

Determination of pipe reactions generated by postulated breaks is discussed in [DCD Tier 2,] Section 3.6.

Y_r =   Load on the structure generated by the reaction on the broken high-energy pipe during the postulated break. Determination of the loads is discussed in [DCD Tier 2,] Section 3.6.

Y_j =   Jet impingement load on the structure generated by the postulated break. Determination of the loads is discussed in [DCD Tier 2,] Section 3.6.

Y_m =   Missile impact load on the structure generated by or during the postulated break, as from pipe whipping. Determination of the loads is discussed in [DCD Tier 2,] Section 3.6.

The definitions of the normal, severe environmental, extreme environmental, and abnormal loads discussed above in Sections 3.8.4.3.1, 3.8.4.3.2, 3.8.4.3.3, and 3.8.4.3.4 of this report are consistent with SRP Section 3.8.4.II.3, although F and H, as defined in Section 3.8.4.3.1 of this report, are not explicitly defined in the SRP. However, F and H are defined in ACI-349-01, Chapter 9. As discussed in Section 3.8.4.2 of this report, the staff accepts ACI-349-01 as the design basis for AP1000 concrete structures. Therefore, the staff finds the applicant's definition of loads to be acceptable.

The staff reviewed DCD Tier 2, Section 6.2.1.2, "Containment Subcompartments," but did not identify any quantitative data on subcompartment pressures and temperatures. In RAI 220.015, the staff requested that the applicant provide quantitative pressure and temperature results from the AP1000 subcompartment analyses, for both high- and medium-energy line breaks, for all subcompartments inside and outside containment in which a significant line break has been postulated. In addition, the applicant was requested to demonstrate that the quantitative results validate the use of a uniform 34.5 kPa (5 psi) subcompartment design pressure, and to describe the methodology used to evaluate the effects of temperature transients resulting from the postulated line breaks.

In its response to RAI 220.015 (Revision 0), the applicant indicated that the design pressure for the AP1000 MSIV rooms had been increased to 41.37 kPa (6 psi), to "accommodate its higher short-term mass and energy release." Corresponding revisions to DCD Tier 2, Section 3.8.4.3.1.4 and Tables 6.2.1.2-1 and 6.2.1.3-4, were identified and are reflected in Section 3.8.4.3.4 of this report. DCD Tier 2, Table 6.2.1.2-1, lists compartment pressures, and forms the basis for the selection of bounding pressures for structural design. Revised Footnote (2) to DCD Tier 2, Table 6.2.1.2-1 states, "Structures are designed to a pressurization load of [34.5 kPa (5.0 psig)] except…the CVS room pipe tunnel is designed to a pressurization load of [51.71 kPa (7.5 psig)]…the MSIV rooms are designed to a pressurization load of [41.37 kPa (6 psig)… ."

The staff finds that the applicant provided the requested data related to subcompartment pressurization. The actual subcompartment pressures predicted by the applicant are evaluated by the staff in Chapter 6 of this report.

Design of Structures, Components, Equipment, and Systems

Also in its response to RAI 220.015, the applicant described its methodology for evaluating the effects of temperature transients. The applicant indicated that similar to the approach for the AP600, the design of subcompartments inside containment use the results of the bounding global temperature analysis provided in DCD Tier 2, Chapter 6. The applicant further indicated that the MSIV subcompartment temperatures are shown in DCD Tier 2, Figure 3D.5-9. The applicant clarified that subsequent to the initial blowdown, the compartment atmosphere cools and the compartment walls are evaluated for a long-term gradient across the walls. The staff found this response lacked sufficient detail to address the staff's question.

During the April 2–5, 2003, audit, the applicant provided a number of calculations related to thermal analysis of plant structures, including subcompartments. The staff reviewed selected portions of the following calculations:

- APP-1100-S2C-004, Revision 0

- APP-1100-S2C-005, Revision 1

- APP-1200-S2C-002, Revision 0, "ASB Exterior Walls Thermal and Earth Pressure Analyses"

- APP-GW-S1-009, Revision 0, "Design Guide for Thermal Effects on Concrete Structures"

- APP-SSAR-GSC-529, Revision 0, "APP1000 MSIV Compartment Temperature Response Following MSLB in Support of the Equipment Qualification"

The applicant referenced DCD Tier 2, Figure 3D.5-9, in its response to RAI 220.015; however, based on review of the selected calculations, it does not appear that the applicant considered the potential effect of a rapid increase in subcompartment temperature on the stresses in the faceplates and concrete of the structural wall modules. Rapid heat-up of the steel plate of the structural wall modules must be considered in the analysis and design of the structural wall module. The mismatch in thermal conductivity between the steel faceplate and the concrete may result in significant thermal stresses on the faceplate, studs, and concrete core, potentially causing degradation of the faceplate/concrete bond and invalidating the assumption of composite behavior. For subcompartment locations inside containment (except for the IRWST) and outside containment, the staff found that the applicant needed to define any rapid thermal transients that can occur, and demonstrate that no unacceptable degradation would result from differential thermal expansion of the steel and concrete throughout the entire transient. This was Open Item 3.8.4.3-1 in the DSER.

In its submittal dated June 23, 2003, the applicant presented the results of the heat transfer and structural analyses performed to address this issue, and concluded that yielding of the faceplate and cracking of concrete will occur under the combined load conditions. The details presented are applicable to a carbon steel faceplate. No detailed results were presented for a stainless steel faceplate. Depending on the thermal conductivity, coefficient of thermal expansion, modulus of elasticity, and yield stress of the stainless steel material, compared to the carbon steel material, the worst case may not have been analyzed. The applicant needed to demonstrate that the worst case has been considered in the design. A key element of this open item was to demonstrate that the unacceptable degradation will not occur. The applicant also

needed to provide its technical basis for the conclusion statement, "The cracking of the concrete does not cause degradation of the structural integrity of the wall." In addition, the applicant, in its response, did not address the staff's concern that the degradation may invalidate the assumption of composite behavior. The staff discussed these concerns with the applicant during a telephone conference on August 22, 2003.

In response to the staff's concerns, the applicant, in its submittal dated September 23, 2003, revised the response to this open item to include an assessment of the stainless steel faceplates, and provided a description of the structural evaluation for the effects of the rapid thermal transients, which will be incorporated in the future revision of DCD Tier 2, Section 3.8.3.4.3. During the October 6–9, 2003, audit, the applicant also provided a description of the basis for its conclusion that the cracking of the concrete does not cause degradation of the structural integrity of the wall. The analysis models of the CIS utilized concrete-cracked properties, including the effects of concrete cracking due to rapid thermal transients, for thermal analyses, as described in DCD Tier 2, Section 3.8.3.4.3 and identified in DCD Tier 2, Table 3.8.3-2. These analyses, along with the prior AP600 studies, were used to address potential cracking of the concrete modules. The audit performed by the staff revealed that the analysis approach and method used by the applicant are consistent with industry practice. In addition, DCD Tier 2, Section 3.8.4.3 was revised to provide a description of the structural evaluation of modules for the effects of rapid transients.

Based on its review of the revised response to this open item and the revised DCD, Open Item 3.8.4.3-1 is resolved.

Section 3.8.3.3 of this report discusses the analysis approach for the thermal transient inside the IRWST. The staff finds it to be acceptable because the heatup of the water in the IRWST occurs slowly, allowing sufficient time for the steel faceplate and concrete core of the module wall to heat up without development of a significant temperature differential between them.

3.8.4.3.5 Dynamic Effects of Abnormal Loads

DCD Tier 2, Section 3.8.4.3.1.5, "Dynamic Effects of Abnormal Loads," states the following:

the dynamic effects from the impulsive and impactive loads caused by P_a, R_a, Y_r, Y_j, Y_m, and tornado missiles are considered by one of the following methods:

- applying an appropriate dynamic load factor to the peak value of the transient load

- using impulse, momentum, and energy balance techniques

- performing a time-history dynamic analysis

The applicant indicated that dynamic increase factors appropriate for the strain rates involved may be applied to static material strengths of steel and concrete for the purpose of determining section strength. In performing the evaluation for abnormal loads, elastoplastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Design of Structures, Components, Equipment, and Systems

The staff considers the methods described by the applicant to be potentially applicable for considering the dynamic effects of abnormal loads. When appropriately applied, the staff has accepted these methods in the past. The staff evaluated the applicant's implementation of any of these methods in the final critical section design calculations. The resolution of Open Item 3.8.4.5-1 in Section 3.8.4.5 of this report addresses the issue of completion and documentation of the final design calculations for the critical sections.

3.8.4.3.6 Load Combinations—Steel Structures

DCD Tier 2, Section 3.8.4.3.2.1, "Steel Structures," states the following:

> The steel structures and components are designed according to the elastic working stress design methods of the AISC-N690 specification using the load combinations specified in [DCD Tier 2,] Table 3.8.4-1.

The staff finds that the tabulated load combinations are consistent with the acceptance criteria of SRP Section 3.8.4 for the working stress design method for steel structures and, therefore, acceptable.

3.8.4.3.7 Load Combinations—Concrete Structures

DCD Tier 2, Section 3.8.4.3.2.2, "Concrete Structures," states the following:

> The concrete structures and components are designed according to the strength design methods of ACI-349 Code, using the load combinations specified in [DCD Tier 2,] Table 3.8.4-2.

The staff finds that the tabulated load combinations are consistent with the acceptance criteria of SRP Section 3.8.4 for the strength design method for concrete structures and, therefore, acceptable.

3.8.4.3.8 Live Load for Seismic Design

DCD Tier 2, Section 3.8.4.3.2.3, "Live Load for Seismic Design," states the following:

> Floor live loads, based on requirements during plant construction and maintenance activities, are specified to vary from [2.39 kPa (50 lb/ft^2)] to [11.97 kPa (250 lb/ft^2)] (with the exception of the containment operating deck which is designed for [38.30 kPa (800 lb/ft^2) specified for plant maintenance condition).
>
> For the local design of members, such as the floors and beams, seismic loads include the response due to masses equal to 25 percent of the specified floor live loads or 75 percent of the roof snow load, whichever is applicable. These seismic loads are combined with 100 percent of these 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 plant operating condition. The mass of equipment and distributed systems is included in both the dead and seismic loads.

As discussed in Section 3.7.2 of this report, the staff finds the approach for considering the effects of live load mass in the design-basis seismic analyses to be acceptable.

3.8.4.4 Design and Analysis Procedures

SRP Section 3.8.4 provides acceptance criteria based on the requirements of 10 CFR 50.55a, GDC 2, 4,and 6, and 10 CFR Part 50, Appendix B.

3.8.4.4.1 Seismic Category I Structures

DCD Tier 2, Section 3.8.4.4.1, "Seismic Category I Structures," states the following:

> [*The design and analysis procedures for the seismic Category I structures (other than the containment vessel and containment internal structures), including assumptions on boundary conditions and expected behavior under loads, are in accordance with ACI-349 for concrete structures, with AISC-N690 for steel structures, and AISI for cold formed steel structures.*]* The structural modules in the auxiliary building are designed using the same procedures as the structural modules in the containment internal structures described in [DCD Tier 2, Section] 3.8.3.

> [*The criteria of ACI-349, Chapter 12, are applied in the design of splicing for reinforcing steel. The ductility criteria of ACI--349, Chapter 21, are applied in detailing and anchoring of the reinforcing steel.*

> *The application of Chapter 21 detailing is demonstrated in the reinforcement details of critical sections*]* in [DCD Tier 2, Section] 3.8.5 and [DCD Tier 2,] Appendix 3H.

The staff notes that the final design of critical sections for other seismic Category I structures is incomplete at this time. Open Item 3.8.4.5-1 in Section 3.8.4.5 of this report addressed this issue.

DCD Tier 2, Section 3.8.4.4.1 further states:

> [*Sections 21.2 through 21.5 of Chapter 21 of ACI-349 are applicable to frame members that resist earthquake effects, and are considered by the applicant in detailing structural elements subjected to significant flexure and out-of-plane shear...*]*

> [*Sections 21.2 and 21.6 of Chapter 21 of ACI-349 are applicable to walls, diaphragms, and trusses serving as parts of the earthquake force-resisting systems, as well as to diaphragms, struts, ties, chords and collector elements, and are considered by the applicant in the detailing of reinforcement in the walls*

and floors of the auxiliary building and in the shield building cylindrical wall and roof.]*...

...The bases of design for the tornado, pipe breaks, and seismic effects are discussed in [DCD Tier 2, Sections] 3.3, 3.6, and 3.7, respectively. The foundation design is described in [DCD Tier 2, Section] 3.8.5.

The seismic Category I structures are reinforced concrete and structural module shear wall structures consisting of vertical shear/bearing walls and horizontal slabs supported by structural steel framing. Seismic forces are obtained from the equivalent static analysis of the three dimensional finite element models described in [DCD Tier 2,] Table 3.7.2-14. The out-of-plane bending and shear loads for flexible floors and walls are analyzed using the methodology described in [DCD Tier 2, Sections] 3.7.2.6 and 3.7.3. These results are modified to account for accidental torsion as described in [DCD Tier 2, Section] 3.7.2.11. Where the refinement of these finite element models is insufficient for design of the reinforcement, (e.g., in walls with a large number of openings), detailed finite element models are used. Also evaluated and considered in the shear wall and floor slab design are out-of-plane bending and shear loads, such as live load, dead load, seismic, lateral earth pressure, hydrostatic, hydrodynamic, and wind pressure. These out-of-plane bending and shear loads are obtained from the equivalent static analyses supplemented by hand calculations.

The exterior walls of the seismic Category I structures below the grade are designed to resist the worst case lateral earth pressure loads (static and dynamic), soil surcharge loads, and loads due to external flooding as described in [DCD Tier 2,] Section 3.4. The lateral earth pressure loads are evaluated for two cases:

- Lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with ASCE 4-98 ([DCD] Reference 3), Section 3.5.3, Figure 3.5-1, "Variation of Normal Dynamic Soil Pressures for the Elastic Solution."

- Lateral earth pressure equal to the passive earth pressure.

The shield building roof and the passive containment cooling water storage tank are analyzed using three-dimensional finite element models with the GTSTRUDL computer codes. The model is shown in [DCD Tier 2,] Figure 3.8.4-3. It represents one quarter of the roof with symmetric or asymmetric boundary conditions dependent on the applied load. Loads and load combinations are given in [DCD Tier 2, Section] 3.8.4.3 and include construction, dead, live, thermal, wind and seismic loads. Seismic loads are applied as equivalent static accelerations. The seismic response of the water in the tank is analyzed in a separate finite element response spectrum analysis with seismic input defined by the floor response spectrum...

During the April 2–5, 2003, audit, the staff noted that DCD Tier 2, Figure 3.8.4-3, which identifies the GTSTRUDL model of the AP1000 shield building roof and the PCCWST, appears to be a revision of the AP600 model shown in Figure 3.8.4-9 of the AP600 DCD. Based on a comparison of the two figures, the AP1000 model appears to contain geometry errors in the vicinity of the openings, as evidenced by the irregularities in the finite element mesh. The staff requested the applicant to either (1) provide a corrected DCD Tier 2, Figure 3.8.4-3, or (2) justify why the model is correct as shown.

During the design audit on April 2–5, 2003, the applicant explained that the GTSTRUDL computer code generated the model plot, and while the image was distorted, the nodal coordinates were located correctly. Because this explains the apparent irregularities in the finite element mesh, the staff finds the applicant's explanation to be acceptable.

DCD Tier 2, Section 3.8.4.4.1, further states the following:

> The liner for the passive containment cooling water storage system tank is analyzed by hand calculation. The design considers construction loads during concrete placement, loads due to handling and shipping, normal loads including thermal, and the safe-shutdown earthquake. Buckling of the liner is prevented by anchoring the liner using the embedded stiffeners and welded studs. The liner is designed as a seismic Category I steel structure in accordance with AISC-N690 with the supplemental requirements given in [DCD Tier 2, Section] 3.8.4.

> The structural steel framing is used primarily to support the concrete slabs and roofs. Metal decking, supported by the steel framing, is used as form work for the concrete slabs and roofs. The structural steel framing is designed for vertical loads. [DCD Tier 2,] Appendix 3H shows typical structural steel framing in the auxiliary building.

> Computer codes used are general purpose computer codes. The code development, verification, validation, configuration control, and error reporting and resolution are according to the quality assurance requirements of [DCD Tier 2,] Chapter 17.

> [*The finned floors for the main control room and the instrumentation and control room ceilings are designed as reinforced concrete slabs in accordance with ACI-349. The steel panels are designed and constructed in accordance with AISC-N690. For positive bending, the steel plate is in tension and the steel plate with fin stiffeners serves as the bottom reinforcement. For negative bending, compression is resisted by the stiffened plate and tension by top reinforcement in the concrete.*]*

The staff finds the design and analysis procedures for seismic Category I structures to be acceptable because they are based on recognized codes and standards and the analytical methods used meet the guidelines of SRP Section 3.8.4. Section 3.8.4.2 of this report discusses the staff's review and acceptance of AISC-N690. Section 3.8.4.2 of this report discusses implementation of ACI-349-01, Section 21.6, relating to boundary elements.

Design of Structures, Components, Equipment, and Systems

3.8.4.4.2 Cable Trays and Cable Tray Supports

DCD Tier 2, Appendix 3F, "Cable Trays and Cable Tray Supports," describes the design and analysis procedures for seismic Category I HVAC cable trays and their supports and addresses codes and standards, loads and load combinations, and analysis and design. The cable tray systems (including supports) are to be designed for the combined load conditions of dead load, live load, SSE, and thermal load. When seismic analysis is performed, either the equivalent static analysis method or dynamic analysis methods are to be used. As shown in DCD Tier 2, Table 3.7.1-1, damping values of 10 percent and 7 percent were assigned for the full tray and empty tray systems, respectively. The stresses in the three directions (two horizontal directions and one vertical direction) were combined using the SRSS method. For supports utilizing rolled structural shapes, basic stress allowables were in accordance with AISC-N690-94 and the supplemental requirements of DCD Tier 2, Section 3.8.4.5.2, "Supplemental Requirements for Steel Structures." The manufacturer's published catalog values provide the basic stress allowables for supports utilizing light gauge, cold rolled channel type sections.

The staff finds that the design criteria and the procedures of analysis and design described in DCD Tier 2, Appendix 3F, provide an adequate design basis for the cable tray systems, including supports, because they are based on recognized codes and standards and reflect common engineering practice. Section 3.8.4.2 of this report discusses the staff's review and acceptance of AISC-N690, which is used for the design of cable trays and supports.

3.8.4.4.3 HVAC Ducts and Duct Supports

DCD Tier 2, Appendix 3A, "HVAC Ducts and Duct Supports," to describes the design and analysis procedures for seismic Category I HVAC ducts and their supports and addresses codes and standards, loads and load combinations, and analysis and design. The HVAC systems are to be designed for the combined load conditions of live load, pressure load, SSE, wind load, tornado load, external pressure differential load, and thermal load. When seismic analysis is performed, either the equivalent static analysis method or dynamic analysis methods are to be used. Damping values of 4 percent and 7 percent were assigned for the welded and bolted HVAC ductworks, respectively. The stresses in the three directions (two horizontal directions and one vertical direction) were combined using the SRSS method. For supports utilizing rolled structural shapes, basic stress allowables were in accordance with AISC-N690-94 and the supplemental requirements of DCD Tier 2, Section 3.8.4.5.2. The manufacturer's published catalog values provide the basis for the basic stress allowables for supports utilizing light gauge cold rolled channel type sections.

The staff finds that the design criteria and the procedures of analysis and design described in DCD Tier 2, Appendix 3A, provide an adequate design basis for HVAC duct supports because they are based on recognized codes and standards and reflect common engineering practice.

3.8.4.5 Structural Criteria

DCD Tier 2, Section 3.8.4.5, "Structural Criteria," states the following:

> [*The analysis and design of concrete conform to ACI-349. The analysis and design of structural steel conform to AISC-N690. The analysis and design of*

cold-formed steel structures conform to AISI. The margins of structural safety are as specified by those codes.]*

As discussed in SRP Section 3.8.4.II.2, the staff finds the use of ACI-349 for the design of seismic Category I structures other than the containment building to be acceptable.

3.8.4.5.1 Supplemental Criteria for Concrete Structures

DCD Tier 2, Section 3.8.4.5.1, states the following:

> [*Supplemental requirements for ACI-349-01 are given in the position on Regulatory Guide 1.142 in [DCD Tier 2,] Appendix 1A. The structural design meets the supplemental requirements identified in Regulatory Positions 2 through 8, 10 through 13, and 15.*]*...

> ...[*Design of fasteners to concrete is in accordance with ACI-349-01, Appendix B.*]*

The staff finds the specified supplemental criteria described above for the design of concrete structures to be acceptable. Section 3.8.4.2 of this report discusses the details of the staff's evaluation of applicable codes, standards, and specifications.

3.8.4.5.2 Supplemental Criteria for Steel Structures

The supplemental criteria for use of AISC-N690 for the AP1000 steel design are identical to the supplementary criteria in the staff position developed for AP600. Section 3.8.4.2 of this report discusses the staff's review and acceptance of applicable codes, standards, and specifications used in the AP1000 design.

3.8.4.5.3 Design Summary Report

DCD Tier 2, Section 3.8.4.5.3, "Design Summary Report," states the following:

> A design summary report is prepared for seismic Category I structures documenting that the design of structures meets the acceptance criteria specified in [DCD Tier 2, Section] 3.8.4.5.

> Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of [DCD Tier 2,] Section 3.7 and 3.8, provided the following acceptance criteria are met:

> •   the structural design meets the acceptance criteria specified in [DCD Tier 2,] Section 3.8.

> •   the seismic floor response spectra meet the acceptance criteria specified in [DCD Tier 2, Section] 3.7.5.4.

Design of Structures, Components, Equipment, and Systems

> Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report by the combined license applicant.

During the design audit on April 2–5, 2003, the design summary report for other Category I structures was not available for staff review. This was Open Item 3.8.4.5-1 in the DSER.

During the October 6–9, 2003, audit, the staff reviewed Calculation AP-1200-S3R-001, "Auxiliary Building Structures Design Summary Report," dated September 2003. This report summarizes the final design of the critical sections of other structures outside containment, which include shear walls, composite structures (floors and roof), reinforced concrete slabs (tagging room ceiling), concrete finned floors (control room ceiling), structural modules (west wall of spent fuel pool), and shield building roof (tension ring and column shear wall between air inlets). The report also describes the governing codes and standards, loads and load combinations, acceptance criteria, materials, seismic analyses, design of the critical structural sections, and a summary of the results. Based on its review of selected sections of this report, the staff found that the design approach and acceptance criteria used meet the guidelines of SRP Section 3.8.4 and, therefore, are acceptable. However, the staff identified some minor differences between the results presented in the report and the information in the DCD. The applicant agreed to correct these differences.

During the design audit on December 15–16, 2003, the staff reviewed the Revision 1 of changes to APP-1200-S3R-001. The staff also reviewed the DCD, and found that the results presented in Revision 1 to the design summary report and the results presented in the DCD are consistent. On this basis, Open Item 3.8.4.5-1 is resolved.

3.8.4.5.4 Design of Critical Sections

In order to demonstrate the design adequacy of other seismic Category I structures, the applicant completed the design of the critical sections listed below. The design details of these critical sections are summarized in DCD Tier 2, Appendix 3H, "Auxiliary and Shield Building Critical Sections." These critical sections re designated as Tier 2* in DCD Tier 2, Section 3.8.4.5.4, "Design Summary of Critical Sections."

- [*South wall of the auxiliary building (column line 1), Elevation 66'-6" to Elevation 180'-0"*

- *Interior wall of the auxiliary building (column line 7.3), Elevation 66'-6" to Elevation 160'-6"*

- *West wall of the main control room in the auxiliary building (column line L), Elevation 117'-6" to Elevation 153'-0"*

- *North wall of the MSIV east compartment (column line 11), Elevation 117'-6" to Elevation 153'-0"*

- *Shield building cylinder, Elevation 160'-6" to Elevation 200'-0"*

- *Roof slab at Elevation 180'-0" adjacent to the shield building cylinder*

- *Floor slab on metal decking at Elevation 135'-3"*

- *2'-0" slab in the auxiliary building (tagging room ceiling) at Elevation 135'-3"*

- *Finned floor in the main control room at Elevation 135'-3"*

- *Shield building roof, exterior wall of the PCCS water storage tank*

- *The shield building roof, tension ring and columns between air inlets, Elevation 265'-0" to Elevation 276'-0"*

- *Divider wall between the spent fuel pool and the fuel transfer canal]**

The staff reviewed DCD Tier 2, Appendix 3H, and found that the critical section design summaries had not been updated. In addition, the staff identified a note in the summary tables of Appendix 3H, which states that "[t]he results shown are representative for the AP1000 and may be updated when structural calculations are completed." The applicant committed to update the design summaries of critical sections in DCD Tier 2, Appendix 3H, in the next DCD revision. This was Confirmatory Item 3.8.4.5-1 in the DSER.

The staff verified that the applicant updated the summary tables in DCD Tier 2, Appendix 3H and removed the note in the tables. Therefore, Confirmatory Item 3.8.4.5-1 is resolved.

During the design audit on April 2–5, 2003, the applicant provided its analysis and design calculations of the 12 critical sections for the staff's review. According to the applicant, these calculations were identified as final approved calculations, recognizing that they are subject to revision based on the results of additional analyses to be performed to address Open Item 3.7.2.3-1. After resolution of Open Item 3.7.2.3-1, these calculations formed the basis for updating DCD Tier 2, Appendix 3H. During this audit, the staff selected the following analyses and design calculations for a detailed review:

- APP-1200-S2C-001, Revision 0

- APP-1200-S2C-106, Revision 1, "Auxiliary Building Wall 1 Dead Load, Live Load, and Seismic Member Forces"

- APP-1200-CCC-106, Revision 0, "Auxiliary Building—Wall 1 Design Report"

- APP-1200-CCC-102, Revision 0

- APP-1277-S3C-006, Revision 2, "Shield Building Roof Structural Analysis and Reinforced Concrete Section Design"

During the course of its review of the Wall 7.3 design calculation, the staff noted that the applicant had previously identified and corrected an error in the equation used by INITEC to

Design of Structures, Components, Equipment, and Systems

calculate the necessary positive reinforcement for a section subjected to both a bending moment and an axial load. The staff could not conclude during the audit that the corrected equation accurately calculated the necessary positive reinforcement. Therefore, the staff requested the applicant to submit the derivation of the equation currently used to calculate the necessary reinforcement. The staff also requested the applicant to submit a sample verification calculation for the computer algorithm and verify that the corrected equation had been utilized in all calculations. This was Open Item 3.8.4.5-2 in the DSER.

In the submittal dated July 7, 2003, the applicant responded to this open item. The staff's review identified several potential shortcomings that may limit the scope of applicability of the calculation as follows:

(1)   The formulation does not apply to conditions involving large compressive loads. At locations where large compressive loads are present, the ultimate strength must be estimated based on the interaction of the axial compression and bending moment.

(2)   As stated in the applicant's response, the formulation presented is applicable when the strength of the section is controlled by yielding of the tension steel, and both tension and compression steel, if any, are at yield. However, there is no indication in the description of the approach that an assessment was made to ensure that the compression steel does yield.

(3)   When calculating the required reinforcement for the case where $M_u > M_{75}$, it is not evident that the limits on the percentage of reinforcement required by the ACI-code are evaluated.

(4)   In its response, the applicant states, "The corrected equation as developed herein has been used in all calculations of reinforcement using the ANSYS post processors and EXCEL macros." If this corrected equation has been used in all calculations, then it may have been applied to concrete sections that are outside the range of applicability, as described in items 1, 2, and 3 above.

In addressing the staff's concerns, the applicant, in its submittal dated September 23, 2003, revised its response to this open item to explain the applicability of the computer algorithm used to design reinforcement in walls and slabs. For sections which fall outside the range of applicability of the computer algorithm, additional guidance is provided to the engineer to properly design the concrete section. When the axial compressive forces exceed φP_b (axial load strength corresponding to balanced strain conditions), guidance is provided to consider the axial load-moment interaction diagram. During the October 6–9, 2003 audit, the staff reviewed APP-1000-CCC-002, Revision 0, "Guidance on Checking Results of Design Macro Calculation." This document provides design procedures for checking the adequacy of design reinforcement, including the case in which the design computer algorithm does not apply (i.e., larger compressive forces). The staff requested the applicant to demonstrate that it evaluated all concrete sections using this guidance.

In a letter dated October 24, 2003, the applicant provided Revision 2 of the response to Open Item 3.8.4.5-2, which includes a summary of results from the evaluation of axial member forces in each of the critical sections. The applicant also indicated that a revision to the reconciliation

report reviewed during the meeting documents the evaluation of each critical wall section and floor. This open item was considered resolved, pending staff review and acceptance of the revised reconciliation report.

During the design audit on December 15–16, 2003, the staff reviewed APP-1000-S3R-001, Revision 1. Section 7.1 of this calculation explains that the revised "macro" (computer algorithm) and the additional guidance (APP-1000-CCC-002, Revision 0, "Guidance on Checking Results of Design Macro Calculations") do not affect the concrete modules inside containment because compression reinforcement is not required and axial stresses are less than 0.25 f_c'. Section 7.12 of the calculation indicates that axial forces in the basemat are well below 0.25 f_c'. Therefore, the staff finds the use of the "macro" in the nuclear island basemat to be acceptable. Section 7.12.1 of the calculation describes how the design calculations for the other critical sections of the auxiliary and shield buildings use the "macro" and the additional guidance. The staff concludes that the information presented in these sections demonstrated that the applicant appropriately utilized the revised "macro" and the additional guidance document for the design of the critical sections. On this basis, Open Item 3.8.4.5-2 is resolved.

During the course of reviewing the shield building roof design calculation, the staff found that this calculation (performed by ANSALDO) had not been finalized. The calculation contained some outdated references, and the seismic loads applied did not reflect the applicant's most recent input. Based on its review of the analysis method used for the shield building roof design, the staff finds that the technical bases used for the design meet the SRP Section 3.8.4 guidelines and are acceptable. The applicant agreed to finalize this calculation to document the final design of the shield building roof. This was Confirmatory Item 3.8.4.5-2 in the DSER.

During the audit from October 6–9, 2003, the staff reviewed APP-1000-S3R-001, Revision 0. Section 7.3 of the calculation addresses the shield building and roof. The calculation provides the updated references and evaluates the effect of changes to the seismic loads. The staff finds the information in Section 7.3 acceptable in addressing the two identified issues. On this basis, Confirmatory Item 3.8.4.5-2 is resolved.

The staff reviewed the design calculation of Wall 1, as well as the analysis results used as input to the final design of Wall 1. Wall 1 is a reinforced concrete wall. The staff finds that the approach for generating design loads meets the SRP Section 3.8.4 guidelines and that the reinforcement meets the design criteria. Therefore, the staff considers the design calculation for Wall 1 acceptable.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

DCD Tier 2, Section 3.8.4.6, "Materials, Quality Control, and Special Construction Techniques," contains information relating to the materials, quality control program, and special construction techniques used in the construction of the other seismic Category I structures. According to the applicant, the edition of the referenced specifications applicable at the start of construction will be used. DCD Tier 2, Section 3.8.4.6, also describes the materials and quality control program used in the construction of the CIS and the NI structures foundation mat.

Design of Structures, Components, Equipment, and Systems

3.8.4.6.1 Materials

DCD Tier 2, Section 3.8.4.6.1.1, "Concrete," states the following:

> The compressive strength of concrete used in the seismic Category I structures and containment internal structures is f_c' = [27.58 MPa (4000 psi)]. The test age of concrete containing pozzolan is 90 days. The test age of concrete without pozzolan is the normal 28 days. Concrete is batched and placed according to [ASTM C 94, ACI-304], and ACI-349.
>
> Portland cement [indicated for construction of the nuclear island] conforms to [ASTM C 150], Type II, with the sum of tricalcium silicate and tricalcium aluminate limited to no more than 58 percent. It is also limited to no more than 0.60 percent by weight of alkalies calculated as Na_2O plus 0.658 K_2O. Certified copies of mill test reports showing that the chemical composition and physical properties conform to the specification are obtained for each cement delivery.
>
> Aggregates conform to [ASTM C 33]. The fineness modulus of fine aggregate (sand) is not less than 2.5, nor more than 3.1. In at least four of five successive test samples, such modulus is not allowed to vary more than 0.20 from the moving average established by the last five tests. Coarse aggregates may be rejected if the loss from the Los Angeles abrasion test, [ASTM C 131], using Grading A or [ASTM C 535], exceeds 40 percent by weight at 500 revolutions. Acceptance of source and aggregates is based on the tests specified in [DCD Tier 2,] Table 3.8.4-3.
>
> Water and ice used in the mixing of concrete do not contain more than 250 parts per million of chlorides (as Cl) as determined in accordance with [ASTM D 512]. They do not contain more than 2000 parts per million of total solids as determined in accordance with [ASTM D 1888]. Water meets the criteria in [DCD Tier 2,] Table 3.8.4-4 in regard to the effects of the proposed mixing water on hardened cement pastes and mortars compared with distilled water.
>
> The concrete contains a pozzolan, an air entraining admixture, and a water-reducing admixture. Admixtures, except pozzolan, are stored in liquid solution. Admixtures do not contain added chlorides except as contained in potable drinking water used for manufacture of the admixtures. The chloride content is stated in the manufacturer's material certification.
>
> Pozzolan conforms to [ASTM C 618], except that the ignition loss does not exceed 6 percent.
>
> Pozzolan is sampled and tested in accordance with [ASTM C 311] for source approval.
>
> Air entraining admixture conforms to [ASTM C 260] and is the vinsol resin type. Water-reducing admixture conforms to [ASTM C 494] of types A and D. Use of types A and D as limited by concrete placing temperature, least dimension of member sizes, and type of placement is as shown in [DCD Tier 2,] Table 3.8.4-5.

Manufacturer's certification for the air entraining admixture is required demonstrating compliance with [ASTM C 260], Section 4 requirements.

Manufacturer's certification for the water-reducing admixture is required demonstrating compliance with [ASTM C 494], Section 5 requirements.

Manufacturer's test reports are required for each delivery of pozzolan showing the chemical composition and physical properties and to certify that the pozzolan complies with the specification.

Proportioning of the concrete mix is in accordance with [ACI-211.1] and Option B of [ASTM C 94], except that in lieu of the requirements of [ASTM C 94], Paragraph 5.3.1.2, the concrete has a specified slump of [7.62 cm (3 in.)]. A testing laboratory designs and tests the concrete mixes. Only mixes meeting the design requirements specified for concrete are used.

The forms for concrete are designed as recommended in ACI-347.

DCD Tier 2, Section 3.8.4.6.1.2, "Reinforcing Steel," states the following:

Reinforcing bars for concrete are deformed bars according to [ASTM A 615], Grade 60, and [ASTM A 706]. Certified material test reports are provided by the supplier for each heat of reinforcing steel delivered showing physical (both tensile and bend test results) and chemical analysis. In addition, a minimum of one tensile test is performed for each [454 metric tons (50 tons)] of each bar size produced from each heat of steel.

In areas where reinforcing steel splices are necessary and lap splices are not practical, mechanical connections (e.g., threaded splices, swaged sleeves or cadwelds) are used. Headed reinforcement meeting the requirements of [ASTM A970] is used where mechanical anchorage is required, such as for shear reinforcement in the nuclear island basemat and in the exterior walls below grade.

As stated in [DCD Tier 2, Section] 3.4.1.1.1, seismic Category I structures that are located below grade elevation are protected against flooding by a waterproofing system and waterstops. This, in conjunction with the [5.08 cm (2 in.)] of concrete cover for the reinforcing steel, provides sufficient protection for the reinforcing steel. Therefore, the use of coated reinforcing steel is not planned.

DCD Tier 2, Section 3.8.4.6.1.3, "Structural Steel," states the following:

Basic materials used in the structural and miscellaneous steel construction conform to the ASTM standards listed in [DCD Tier 2,] Table 3.8.4-6.

ACI-349-01 and AISC-N690-94 reference many of the standards described above. Since the staff has accepted these two codes as the design basis for other seismic Category I structures of the AP1000, the staff finds the use of the referenced standards to be acceptable. The staff

Design of Structures, Components, Equipment, and Systems

reviewed the other standards that ACI-349-01 or AISC-N690-94 do not directly reference. The staff did not identify any technical concerns relating to these standards and finds them to be acceptable.

3.8.4.6.2 Quality Control

DCD Tier 2, Chapter 17, describes the quality assurance program. DCD Tier 2, Section 1.9, describes conformance to RG 1.94 and is, therefore, acceptable.

3.8.4.6.3 Special Construction Techniques

The applicant discusses special construction techniques for structural modules in DCD Tier 2, Section 3.8.3.6. In Section 3.8.3.6 of this report, the staff evaluates the special construction techniques for containment internal structures modules, which also apply to the auxiliary building structural modules.

3.8.4.7 Conclusions

For the reasons set forth above, the staff concludes that the design of safety-related structures other than the containment vessel and the CIS is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDC 1, 2, and 4. In particular, the staff reached the following conclusions based on its observations:

- By following the guidelines of the relevant RGs and industry standards (indicated below), the applicant has met the requirements of 10 CFR 50.55a and GDC 1 for ensuring that the safety-related structures other than the containment vessel and the CIS are designed, fabricated, erected, and constructed to quality standards commensurate with their safety functions.

- The applicant has met the requirements of GDC 2 by designing the safety-related structures other than the containment vessel and CIS to withstand the 0.3 g SSE and appropriate combinations of the effects of normal and accident conditions, including the effects of environmental loadings, such as earthquakes and other natural phenomena with sufficient margin for limitations in site data.

- The applicant has met the requirements of GDC 4 by ensuring that the design of the safety-related structures can withstand the dynamic effects associated with missiles, pipe whipping, and discharging fluids, excluding dynamic effects associated with pipe ruptures, the probability of which is extremely low under conditions consistent with the design basis for the piping.

The criteria used in the analysis, design, and construction of the plant's seismic Category I structures other than the containment vessel and CIS 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 SRP Section 3.8.4; RGs 1.69, 1.115, 1.142, and 1.143; ACI-349-01; AISC-N690-1994; and supplemental criteria as defined in DCD Tier 2, Section 3.8.4.5.

In addition, the applicant has used these criteria, as defined by codes, standards, and specifications applicable to the loads and loading combinations, design and analysis procedures, structural acceptance criteria, materials, quality control programs, special construction techniques, and testing and inservice surveillance requirements. Together, these considerations provide reasonable assurance that the structures will withstand the specified design conditions without losing their structural integrity or the capability to perform their safety functions in the event of winds, tornados, earthquakes, and various postulated accidents.

Furthermore, the staff based its conclusion regarding the design of the ASB (including the PCCWST structures, the shield building roof structure, and the structural modules outside the containment) on its review of samples of design calculations for the critical sections in these structures, as described in DCD Tier 2, Section 3.8.4.5.4, and Appendix 3H. Any proposed change to the text of DCD Tier 2, Section 3.8.4.5.4 and Appendix 3H will require NRC review and approval before implementation of the change because the information contained in these sections is designated as Tier 2*.

3.8.5 Foundations

SRP Section 3.8.5 provides guidelines for the staff to perform its review of issues related to the foundation mat. These review areas include a description of the foundation; applicable codes, standards, and specifications; loads and loading combinations; design and analysis procedures; structural acceptance criteria; material; quality control; special construction techniques; and testing and inservice surveillance requirements. This SRP section also states that the RGs and industry standards identified in SRP Section 3.8.4.II.2 provide information, recommendations, and guidance. In general, SRP Section 3.8.5 describes a basis acceptable to the staff that may be used to implement the requirements of 10 CFR 50.55a and GDC 1, 2, 4, and 5.

The following seismic and geotechnical design bases described in DCD Tier 2, Sections 2.5 and 3.8.5, and responses to the related RAIs provide the basis for the AP1000 standard design:

- The foundation material for siting the plant is hard rock to ensure that the analyses using a fixed-base model can reasonably predict the dynamic behavior of the nuclear island during a seismic excitation.

- For the purpose of characterizing the hard rock, the material has a shear wave velocity equal to or higher than 2438 m/sec (8000 ft/sec), based on low strain properties, over the entire footprint of the nuclear island at its foundation level.

- The average allowable bearing capacity of the foundation material is indicated in the DCD to be 412 kPa (8600 psf) under static loading.

- As indicated in the DCD, the maximum allowable dynamic bearing capacity for the normal plus seismic load (SSE) conditions is 5.75 MPa (120,000 psf).

Using the guidance described in Section 3.8.5 of the SRP, the staff reviewed the analysis and design of the NI foundation mat, relating to the (1) material, (2) geometry, (3) codes and standards, (4) loadings, and (5) analysis and design procedures.

Design of Structures, Components, Equipment, and Systems

3.8.5.1 Description of the Foundations

SRP Section 3.8.5.II.1 states that the descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the minimum requirements set forth in Section 3.8.5.1 of RG 1.70.

DCD Tier 2, Section 3.8.5.1, "Description of the Foundations," states the following:

> The nuclear island structures, consisting of the containment building, shield building, and auxiliary building are founded on a common [1.829 m (6 ft)] thick, cast-in-place, reinforced concrete basemat foundation. The top of the foundation is at Elevation 66'-6"...

The bottom of the foundation is at Elevation 60'-6". Relative to the reference elevation at the free grade surface Elevation (100'-0"), the foundation is embedded to a depth of 12.04 m (39'-6"). As shown in DCD Tier 2, Figure 3.7.1-14, the maximum plan dimensions of the basemat foundation are 78 m (256 ft) in the north-south (N-S) direction by 49.1 m (161 ft) in the east-west (E-W) direction.

DCD Tier 2, Section 3.8.5.1, further states the following:

> Adjoining buildings, such as the radwaste building, turbine building, and annex building, are structurally separated from the nuclear island structures by a [5.08 cm (2 in.)] gap at and below the grade. A [10.2 cm (4 in.)] minimum gap is provided above grade. This provides space to prevent interaction between the nuclear island structures and the adjacent structures during a seismic event. [DCD Tier 2,] Figure 3.8.5-1 shows the foundations for the nuclear island structures and the adjoining structures...

In DCD Tier 2, Section 3.8.5.1, the applicant stated that the foundation is built on a mudmat, for ease of construction. The mudmat consists of lean, nonstructural concrete and rests upon the load-bearing rock. DCD Tier 2, Section 3.4.1.1.1, "Protection from External Flooding," describes waterproofing standards. In RAI 230.23, the staff raised a concern that the nonstructural concrete mudmat may not withstand the very high toe pressure predicted in the applicant's seismic analysis. This pressure might crush the nonstructural concrete mudmat and potentially affect the structural integrity of the NI foundation mat under the design-basis combination of loads. This issue was Open Item 3.8.5.1-1 in the DSER.

In addressing this open item, the applicant, in a letter dated June 23, 2003, referred to its response to RAI 230.023 dated May 13, 2003. In this response, the applicant explained that the mudmat is a thin layer of lean, nonstructural concrete sandwiched between the rock and the underside of the basemat. The applicant justified that the lean concrete in this confined condition could withstand the high toe pressures conservatively predicted in its liftoff analysis. As a result of its review, the staff, in a telephone conference conducted on August 22, 2003, stated that the applicant needs to include a discussion of the design criteria for the mudmat in the DCD to ensure its integrity under the design-basis earthquake.

During the design audit on October 6–9, 2003, the staff reviewed the proposed revision to the DCD which states that the specified concrete compressive strength for the mudmat design must be no less than 17.24 MPa (2500 psi). The use of a minimum compressive strength of 17.24 MPa (2500 psi) is consistent with industry practice and is acceptable to the staff. The staff also confirmed that DCD Tier 2, Section 2.5.4.5.3, incorporated these design criteria. On the basis discussed above, the staff concludes that Open Item 3.8.5.1-1 is resolved.

In DCD Tier 2, Section 3.8.5.1, the applicant stated that passive soil pressure and friction between the basemat and the rock foundation provides resistance to sliding of the concrete basemat foundation. The applicant concluded that such behavior provides the required safety factor against lateral movement under the most stringent loading conditions.

Since the design certification applies only to the case of the NI founded on a hard rock site, the staff, in RAI 220.016, requested the applicant to provide specific clarifications relating to this limitation in the following areas:

- Describe the construction techniques and sequence to ensure that the surrounding soil or rock (embedment) will provide enough passive pressure to prevent the NI from sliding and overturning.

- Clarify the applicability of the words "soil friction" to the AP1000 design.

- Indicate how passive lateral pressures and base rock friction components can be properly estimated, considering consistent lateral displacements for both forces.

In its response to RAI 220.016, Revision 1, the applicant submitted the following information:

- The excavation technique for the AP1000, described in DCD Tier 2, Section 2.5.4.1, "Excavation," is the same as that for the AP600. It may vary depending on the depth of soil over the rock. Passive pressure is calculated using a 35° internal friction angle for the surrounding soil. DCD Tier 2, Section 3.8.5.5, "Structural Criteria," describes the method used to calculate the safety factor against sliding.

- The term "soil friction" is used in the global soil mechanics meaning. The AP1000 uses basic soil mechanics formulas. The hard rock and concrete interface has a coefficient of friction of 0.55, as defined in DCD Tier 2, Section 3.8.5.5.3, "Sliding."

- Passive soil pressure provides sliding resistance, while active soil pressure reduces sliding resistance. The applicant documented classical formulas and the corresponding references for calculating active and passive soil pressures.

As part of its response, the applicant identified a revision to DCD Tier 2, Section 2.5.4.5.2, to address site-specific criteria related to properties of materials adjacent to NI exterior walls. The staff verified that the DCD was revised to incorporate this change. On the basis of the applicant's response and revision to the DCD, and the additional information provided in the RAI response, the staff considers this issue to be resolved.

Design of Structures, Components, Equipment, and Systems

Evaluation

The descriptive information and referenced figures in DCD Tier 2, Section 3.8.5.1, contain sufficient detail to define the primary structural aspects and elements relied upon for the foundation mat to perform its safety-related function, as described in SRP Section 3.8.5.

3.8.5.2 Applicable Codes, Standards, and Specifications

In DCD Tier 2, Section 3.8.5.2, "Applicable Codes, Standards, and Specifications," the applicant indicated that DCD Tier 2, Section 3.8.4.2 describes the applicable codes, standards, and specifications for the foundation. Section 3.8.4.2 of this report contains the staff's evaluation of referenced codes, standards, and specifications.

3.8.5.3 Loads and Load Combinations

DCD Tier 2, Section 3.8.5.3, "Loads and Load Combinations," states the following:

> Loads and load combinations are described in [DCD Tier 2, Section] 3.8.4.3. As described in [DCD Tier 2, Section] 3.8.2.1.2, the bottom head of the steel containment vessel is the same as the upper head and is capable of resisting the containment internal pressure without benefit of the nuclear island basemat. However, containment pressure loads affect the nuclear island basemat since the concrete is stiffer than the steel head. The containment design pressure is included in the design of the nuclear island basemat as an accident pressure in load combinations 5, 6, and 7 of [DCD Tier 2,] Table 3.8.4-2. In addition to the load combinations described in [DCD Tier 2, Section] 3.8.4.3, the nuclear island is checked for resistance against sliding and overturning due to the safe-shutdown earthquake, winds and tornados, and against flotation due to floods and groundwater, according to the load combinations presented in [DCD Tier 2,] Table 3.8.5-1.

Based on its review of the loads and load combinations described in DCD Tier 2, Section 3.8.4.3, and the additional load combinations and acceptance criteria for the basemat foundation delineated in DCD Tier 2, Table 3.8.5-1, the staff concludes that the loads and load combinations identified for the foundation are consistent with those described in SRP Section 3.8.5. On this basis, the staff finds them to be acceptable.

3.8.5.4 Design and Analysis Procedures

DCD Tier 2, Section 3.8.5.4, "Design and Analysis Procedures," states the following:

> The seismic Category I structures are concrete, shear-wall structures consisting of vertical shear/bearing walls and horizontal floor slabs. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs. The walls then transmit the loads to the basemat. The walls also provide stiffness to the basemat and distribute the foundation loads between them.

The design of the basemat consists primarily of applying the design loads to the structures, calculating shears and moments in the basemat, and determining the required reinforcement. For a site with hard rock below the underside of the basemat, the applicant asserts that vertical loads are transmitted directly through the basemat into the rock, and horizontal loads due to seismic are distributed on the underside of the basemat, resulting primarily in small membrane forces in the mat. The [1.83 m (6 ft)] thick basemat is designed for the upward hydrostatic pressure due to groundwater reduced by the downward deadweight of the mat.

The staff finds that the applicant's technical approach to design of the basemat is consistent with industry practice and SRP Section 3.8.5.

3.8.5.4.1 Analyses for Loads During Operation

The applicant uses a 3D FE model together with the computer program ANSYS (DCD Reference 21) for the analysis of the basemat. The model considers the interaction of the basemat with the overlying structures and with the soil. The model considers two possible uplifts—(1) the uplift of the containment internal structures from the lower basemat, and (2) the uplift of the basemat from the rock foundation.

In its review of DCD Tier 2, Section 3.8.5.4, the staff determined that the potential uplift and slapping back of the containment internal structures foundation on the basemat through the steel containment vessel during a seismic event could affect both the seismic design loads and in-structure response spectra for all SSCs associated with the CIS, as well as the seismic response of the steel containment shell. In RAI 220.021, the staff requested the applicant to perform additional analyses to demonstrate how the uplifting effect will be addressed and to discuss how it will use the uplifting effect on the seismic analysis results in the design of the containment and CIS. This was Open Item 3.8.5.4-1 in the DSER.

In response to this open item, the applicant, in a letter dated June 23, 2003, referenced its response to RAI 230.022 dated May 21, 2003. In its response to RAI 230.022, the applicant indicated that the stability evaluation showed a factor of safety against overturning of about 2.5. Since the seismic load has not overcome the deadweight of the NI structures, no basemat uplifting or slapping (or impact) between the basemat and hard rock foundation is expected to occur. The applicant further stated, "Therefore, it is not necessary to modify the analysis methods from those that were accepted by the NRC for the AP600 plant."

In a telephone conference on August 22, 2003, the applicant indicated that its response to Open Item 19A.2-8 more completely discussed the issue related to basemat uplifting and slapping, specifically for the margin-level earthquake of 0.5 g. At the design-level earthquake (SSE) of 0.3 g, the calculated total uplift is about 0.25 cm (0.1 in.), which is insignificant to the design.

During the audit from October 6–9, 2003, the staff reviewed additional analyses performed by the applicant to evaluate the potential uplifting of both the basemat and the containment shell/containment internal structures. The staff found that the analysis approach and method used by the applicant are consistent with industry practice, and the effect of uplift at the SSE level is negligible. However, the effect of uplift is potentially significant at the margin-level earthquake. Section 19A (Open Item 19A.2-8) of this report discusses the issue related to the

Design of Structures, Components, Equipment, and Systems

margin-level earthquake and concludes that the applicant has appropriately implemented the conservative deterministic failure margin method to estimate conservative high confidence, low probability of failures capacity, in accordance with standard industry practices. For these reasons, the staff concludes that applicant's evaluation results for the design of the basemat are acceptable. Therefore, Open Item 3.8.5.4-1 resolved.

The applicant states in DCD Tier 2, Section 3.8.5, that the 3D FE model of the basemat includes a portion of the structures above the basemat sufficient to consider the effect of openings in the shear walls on the distribution of loads on the basemat. Some of the shear walls on the north side are modeled to their full height. Shell-type elements simulate the basemat, walls, and slabs. The vertical stiffness of the rock is represented by the subgrade modulus that are directly incorporated in the FE model of the foundation slab. The subgrade modulus represents a rock foundation with a shear wave velocity of 1067 m/sec (3500 fps). Horizontal springs attached to some of the nodes on the foundation represent the horizontal stiffness. The horizontal springs are uniformly distributed. Horizontal bearing reactions on the side walls below grade are neglected.

The staff notes that the specified shear wave velocity for a hard rock site condition should be 2438 m/sec (8000 fps), instead of 1067 m/sec (3500 fps). The applicant was requested to verify that the subgrade modulus used in the analyses represents a rock foundation with shear wave velocity equal to 2438 m/sec (8000 fps). This was Open Item 3.8.5.4-2 in the DSER.

During the October 6–9, 2003, audit, the staff reviewed Appendix D to the basemat design summary report (APP-1000-S2C-064, "Effects of Basemat Lift-off on Seismic Response," Revision 1) and verified that the modulus of the subgrade material used in the analyses and design represents a rock foundation with shear wave velocity equal to 2438 m/sec (8000 fps). On this basis, Open Item 3.8.5.4-2 is resolved.

The applicant uses tetrahedral to elements simulate the CIS, which connect to the basemat with spring elements normal to the theoretical surface of the containment vessel. DCD Tier 2, Figure 3.8.5-2, shows some representative features of the model.

The analysis considers normal and extreme environmental loads and containment pressure loads. The normal loads include dead loads and live loads. Extreme environmental loads include the SSE.

The dead, live, and SSE loads for those portions of the structure not included explicitly in the model are applied as concentrated loads on the nodes of the supporting walls and as distributed loads on the top edge of the supporting walls. For portions of the structure that are explicitly modeled, these loads are applied as inertia forces and uniformly distributed loads.

The SSE loads are applied using the assumption that while maximum response from one direction occurs, the responses from the other two directions are 40 percent of the maximum. The analysis considers combinations of seismic responses due to the three orthogonal components of the SSE.

The applicant stated that the analysis is an iterative process because basemat liftoff occurs in many of the load combination cases evaluated. The elastic foundation stiffness capability

included in the basemat elements is designed to support both tension and compression loads. Based on the results from each load combination, in the next iteration, the tension capability is removed for those springs that are in tension. Similarly, the next iteration removed those springs connecting the internal structures with the basemat that showed tension. This iterative process continues until there are no more reactions or springs in tension. As noted in Section 3.7.2.3 of this report, the staff identified Open Item 3.7.2.3-1 relating to the liftoff analysis.

The iterative process is performed for the most critical load combinations. These load conditions are selected based on the results from linear analysis which include all design load combinations. The results from the analysis include forces, shears, and moments in the basemat, bearing pressures under the basemat, and the uplifted area of the basemat. Reinforcing steel areas are calculated from the member forces for each load combination.

The necessary reinforcing steel under the shield building is determined by considering both the reinforcement envelope for the linear analyses which do not consider liftoff and the reinforcement envelope for the full nonlinear iteration of the most critical load combinations.

The necessary reinforcing steel for the portion of the basemat under the auxiliary building is calculated from shears and bending moments in the slab obtained from separate calculations. Beam strip models of the slab segments are loaded with the bearing pressures under the basemat from the 3D finite element analyses. DCD Tier 2, Figure 3.8.5-3, depicts the basemat reinforcement. Section 3.8.5.4.3 of this report discusses the staff's review and acceptance of the basemat design.

3.8.5.4.2 Design Summary Report

DCD Tier 2, Section 3.8.5.4.2, "Design Summary Report," states the following:

> A design summary report is prepared for the basemat, documenting that the structures meet the acceptance criteria specified in [DCD Tier 2, Section] 3.8.5.5.
>
> Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of [DCD Tier 2,] Section 3.7 and 3.8 provided the following acceptance criteria are met:
>
> • The structural design meets the acceptance criteria specified in [DCD Tier 2,] Section 3.8
>
> • The seismic floor response spectra meet the acceptance criteria specified in [DCD Tier 2, Section] 3.7.5.4
>
> Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report by the combined license applicant.

Design of Structures, Components, Equipment, and Systems

SRP Section 3.8.5 prescribes the preparation of a design report containing the information listed in Appendix C to SRP Section 3.8.4. During the April 2–5, 2003, audit, the design summary report for the basemat foundation was not available for review by the staff. Completion of the design summary report and review by the staff was Open Item 3.8.5.4-3 in the DSER.

During the October 6–9, 2003, audit, the staff reviewed Calculation APP-1010-CCC-001, Revision 2, "AP1000 Basemat Design Report." This document describes the governing design codes and regulations, materials, loads and load combinations, structural analysis methodology, and results. The design of the critical sections of the AP1000 basemat is based on the AP600 design. The required additional reinforcement for those instances in which increased loadings exist in the AP1000 critical sections, is identified and incorporated in the final design. Based on the review of selected sections of this report, the staff concluded that the applicant's design of the basemat was in accordance with the design criteria presented in DCD Tier 2, Section 3.8.5.4.2 that were found to be acceptable. Therefore, Open Item 3.8.5.4-3 is resolved.

3.8.5.4.3 Design of Critical Sections

DCD Tier 2, Section 3.8.5.4.3, "Design of Critical Sections," states the following:

> The basemat is designed to meet the acceptance criteria specified in [DCD Tier 2, Section] 3.8.4.5. Two critical portions of the basemat are identified, together with a summary of their design. The boundaries are defined by the walls and column lines which are shown in [DCD Tier 2,] Figure 3.7.2-12 (sheet 1 of 12). [DCD Tier 2,] Table 3.8.5-3 shows the reinforcement required and the reinforcement provided for the critical sections.
>
> [*Basemat between column lines 9.1 and 11 and column lines K and L*
>
> *This portion of the basemat is designed as a one way slab spanning a distance of [7.16 m (23'-6")] between the walls on column lines K and L. The slab is continuous with the adjacent slabs to the east and west. The critical loading is the bearing pressure on the underside of the slab due to dead and seismic loads. This establishes the demand for the top flexural reinforcement at mid span and for the bottom flexural and shear reinforcement at the walls. The basemat is designed for the bearing pressures and membrane forces from the analyses*]* described in [DCD Tier 2, Section] 3.8.5.4.1. [*Negative moments are redistributed as permitted by ACI-349.*
>
> *The top and bottom reinforcement in the east-west direction of span are equal. The reinforcement provided is shown in sheets 1, 2 and 5 of [DCD Tier 2,] Figure 3.8.5-3. Typical reinforcement details showing use of headed reinforcement for shear reinforcement are shown in [DCD Tier 2,] Figure 3H.5-3.*]*
>
> [*Basemat between column lines 1 and 2 and column lines K-2 and N*
>
> *This portion of the basemat is designed as a one way slab spanning a distance of [6.70 m (22'-0")] between the walls on column lines 1 and 2. The slab is continuous with the adjacent slabs to the north and with the exterior wall to the*

south. The critical loading is the bearing pressure on the underside of the slab due to dead and seismic loads. This establishes the demand for the top flexural reinforcement at mid span and for the bottom flexural and shear reinforcement at wall. The basemat is designed for the bearing pressures and membrane forces from the analyses based on uniform soil springs]* described in [DCD Tier 2, Section] 3.8.5.4.1. [The reinforcement provided is shown in sheets 1, 2 and 5 of [DCD Tier 2,] Figure 3.8.5-3. Typical reinforcement details showing use of headed reinforcement for shear reinforcement are shown in [DCD Tier 2,] Figure 3H.5-3.]*

The staff note that DCD Tier 2, Table 3.8.5-3, indicated that, "The results shown are representative for the AP1000 and may be updated when structural calculations are completed." The applicant, at that time, planned to update Table 3.8.5-3 to document the final design of the critical sections. This was Confirmatory Item 3.8.5.4-1 in the DSER.

The staff confirmed that the applicant has updated Table 3.8.5-3. On this basis, Confirmatory Item 3.8.5.4-1 is resolved.

In RAI 220.017, the staff noted an apparent inconsistency in DCD Tier 2, Section 3.8.5.4.3, related to the designation of Tier 2* material. The text of DCD Tier 2, Section 3.8.5.4.3 and Table 3.8.5-3, designates the design of the critical sections as Tier 2*. DCD Tier 2, Figure 3.8.5-3, only designates sheets 1, 2, and 5 as Tier 2* and not sheets 3 and 4.

The staff requested the applicant to identify what is and is not Tier 2*, as well as the technical basis for the proposed designation. In its response to RAI 220.017, Revision 0, the applicant indicated that the NRC staff decides what material needs to be designated Tier 2*, and that the designation in AP1000 DCD Tier 2, Figure 3.8.5-3, is the same as in AP600 DCD Tier 2, Figure 3.8.5-3. The applicant further proposed that only DCD Tier 2, Table 3.8.5-3, be designated Tier 2* on the basis that the table summarizes the design of the critical sections, while the figure contains significantly more information than is covered by the critical sections.

During the April 2–5, 2003, audit, the staff indicated that Tier 2* information contained in DCD Tier 2, Figure 3.8.5-3, sheets 1–5, needs to be designated as such. The applicant agreed to identify pertinent Tier 2* information on each sheet of the figure. This was Confirmatory Item 3.8.5.4-2 in the DSER.

The staff confirmed that the applicant has made the subject revisions in the DCD. On this basis, Confirmatory Item 3.8.5.4-2 is resolved.

3.8.5.5 Structural Criteria

In DCD Tier 2, Section 3.8.5.5, the applicant stated that the analysis and design of the basemat for the NI structures are in accordance with ACI-349, with margins of structural safety as specified within the basemat. DCD Tier 2, Section 2.5, describes the limiting conditions for the foundation medium, together with a comparison of actual capacity and estimated structure loads. The staff raised several technical issues related to the foundation medium in Sections 2.5 and 3.7 of this report. During the December 15 - 16, 2003 audit, the staff reviewed the calculations presented by the applicant and found that the applicant has incorporated, as discussed in

Design of Structures, Components, Equipment, and Systems

Sections 2.5 and 3.7 of this report, the resolution of the associated open items in the analysis and design, and demonstrated that the limiting conditions for the foundation medium on the design of the basemat foundation are appropriately implemented.

DCD Tier 2, Table 3.8.5-1, sets forth the minimum factor-of-safety against sliding, overturning, and flotation for the NI structures. DCD Tier 2, Table 3.8.5-2, lists the calculated factor-of-safety for flotation, sliding, and overturning of the NI basemat, applicable to the hard rock condition. DCD Tier 2, Sections 3.8.5.5.2, "Flotation," 3.8.5.5.3, and 3.8.5.5.4, "Overturning," list the equations used to calculate the factor-of-safety for flotation, sliding, and overturning, respectively.

Since there was no indication that the AP1000 factor-of-safety were subject to change, the staff initially assumed that these factor-of-safety were based on the actual AP1000 basemat loads due to deadweight, flood, ground water, wind, tornado, and earthquake loads. In RAI 220.018, the staff requested the applicant to provide the numerical values of the basemat loads used in the factor-of-safety calculations, for both the AP1000 and the AP600, and to describe any basemat design changes between the AP600 and the AP1000 that were necessary to meet the minimum factor-of-safety listed in DCD Tier 2, Table 3.8.5-1.

In its response to RAI 220.018, Revision 0, the applicant submitted a significant amount of quantitative data comparing basemat loads and factor-of-safety for the AP1000 and the AP600 and indicated that the basemat design is the same for the AP1000 and the AP600. The applicant also revised DCD Tier 2, Table 3.8.5-2, to correct two factor-of-safety entries.

At the November 2002 meeting at Westinghouse, the staff raised additional technical issues concerning the methodology used to predict the maximum seismic-induced loads on the basemat and how the effects of potential liftoff of the basemat are considered. In its Revision 1 response to this RAI, the applicant submitted additional quantitative data in Tables 220.018-1 through 220.018-3, comparing maximum seismic accelerations and forces using different combinations of seismic load directions. The applicant also indicated that its Revision 1 response to RAI 241.001 considered liftoff of the basemat.

The safety margins against rigid body overturning and sliding should meet the acceptance criteria provided in SRP Section 3.8.5. The applicant performed evaluations of the dynamic stability of the NI against overturning and sliding using the moment balance method. The use of the moment balance method for the foundation mat dynamic stability evaluation meets the guidelines of SRP Section 3.8.5 and is acceptable to the staff. This approach defines the factor-of-safety against overturning as the ratio of the restoring moment to the overturning moment caused either by wind or an SSE. According to the applicant, the effects of potential buoyant forces from a high-water table or from flooding conditions have been included in these evaluations.

In response to RAI 220.018, the applicant compared changes in seismic load conditions from the AP600 to the AP1000 case. The applicant indicated that the total deadweight of the NI increased by 10 percent, while the elevation of the center of gravity of the NI increased by 6.6 percent. Considering these two increases from the AP600 to the AP1000 design, the increase in equivalent static overturning moment (used for the uplift analyses) is about 17 percent. All other resisting lateral forces (lateral soil pressures) remain the same as for the

AP600 analyses. In the table below, which was summarized in the response to RAI 220.018, the applicant provided a comparison of seismic design loads for the AP1000 and AP600.

Seismic Reactions		AP600	AP1000	Ratio: AP1000/AP600
Vertical	Kg	40,475,862	46,037,358	1.137
	Kips	89,234	101,495	
E-W Shear	Kg	41,670,624	42,418,144	1.018
	Kips	91,868	93,516	
E-W Overturing Moment	Kg-m	1.0668×10^9	1.1818×10^9	1.108
	Kip-Ft	7,715,877	8,547,756	
N-S Shear	Kg	41,443,828	44,821,276	1.081
	Kips	91,368	98,814	
N-S Overturning Moment	Kg-m	1.034×10^9	1.4742×10^9	1.428
	Kip-Ft	7,478,927	10,662,959	

The N-S overturning moment (moment about the E-W or long axis of the basemat) increases by 42.8 percent, while the E-W overturning moment (moment about the N-S or short axis of the basemat) increases by 10.8 percent. The E-W base shear increases by only 1.8 percent, while the N-S base shear increases by 8.1 percent. The reported increases are not consistent with the 9.9 percent increase in the NI mass or the 17 percent increase in simple equivalent static overturning moment. In addition, the applicant indicated in its response to RAI 220.018 that the safety factor against overturning for the N-S earthquake increases for the AP1000 (compared to the AP600) even though the overturning moment increases by 42.8 percent. Resolution of these apparent discrepancies was Open Item 3.8.5.5-1 in the DSER.

In a letter dated June 23, 2003, the applicant submitted its response to this open item. In Table 3.8.5.5-1 of the response, the applicant provided a comparison of the overturning moments and factor of safety against overturning for the AP1000 and AP600 design. Based on a review of this table, the staff concluded that the applicant's analytical approach meets the guidelines of SRP Section 3.8.5. On this basis, the staff found the applicant's dynamic stability evaluation for the NI structures (including the basemat) acceptable, pending the audit of the final calculations that support the data in Table 3.8.5.5-1 of the open item response.

During the October 6–9, 2003, audit, the staff reviewed two calculations that evaluate the stability of the containment vessel and of the NI. The staff concluded that the applicant's evaluation of overturning follows the SRP guidelines and industry practice and is, therefore, acceptable. On the basis discussed above, Open Item 3.8.5.5-1 is resolved.

Design of Structures, Components, Equipment, and Systems

In DCD Tier 2, Section 3.8.5.5, includes the following Tier 2* information:

[*The basemat below the auxiliary building is designed for shear in accordance with the provisions for continuous deep flexural members in paragraph 11.8.3 of ACI-349-01. As permitted by paragraph 11.5.5.1 of ACI-349-01, shear reinforcement is not provided when the factored shear force, V_u, is less than one half of the shear strength provided by the concrete, φV_c.*]*

In RAI 220.019, the staff noted that the AP1000 criteria for design of shear reinforcement for the basemat below the auxiliary building departs significantly from the comparable criteria presented in the AP600 DCD and previously accepted by the staff. Therefore, the staff requested the applicant to provide (1) a detailed explanation of the differences between the new AP1000 criteria and the accepted AP600 criteria, and (2) the technical justification that a comparable level of safety will be achieved.

In its response to RAI 220.019, Revision 1, the applicant indicated that for the AP1000 design for a hard rock site, bearing reactions are transmitted primarily below the walls of the auxiliary building and that the design shear forces in the 6 foot thick basemat are much lower than those for the basemat of conventional nuclear power plants. For such cases, the applicant proposed to apply paragraph 11.5.5.1 of ACI-349, which does not provide for minimum shear reinforcement when the factored shear force is less than one-half the shear strength provided by concrete, φV_c. The applicant also indicated that the second paragraph of DCD Tier 2, Section 3.8.5.5, would be revised as follows, "As permitted by paragraph 11.5.5.1 of ACI-349-01, shear reinforcement is not provided when the factored shear force, V_u, is less than one-half of the shear strength provided by the concrete, φV_c." Section 3.8.4.2 of this report discusses the acceptance of ACI-349-01.

The staff finds the applicant's response to be acceptable on the basis that ACI-349-01 permits this relaxation on minimum shear reinforcement. However, the applicant incorrectly incorporated this change into the DCD. The phrase "factored shear strength, V_u" should have been "factored shear force, V_u." The applicant was asked to correct this error in the next revision to the DCD. This was Confirmatory Item 3.8.5.5-3 in the DSER.

The staff confirmed that the applicant corrected the error discussed above in the DCD. This correction is reflected in the Tier 2* information quoted from DCD Tier 2, Section 3.8.5.5, above. On this basis, Confirmatory Item 3.8.5.5-3 is resolved.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

In DCD Tier 2, Section 3.8.5.6, the applicant stated that DCD Tier 2, Section 3.8.4.6, describes the materials and quality control program used in the construction of the NI structures foundation. Section 3.8.4.6 of this report contains the staff evaluation of this section.

The applicant also indicated that the construction of the NI structures foundation used no special construction techniques. The staff notes that special techniques are utilized for the foundation excavation. The COL applicant should establish feasibility or propose alternatives that are subject to NRC review and approval. Section 2.5 of this report discusses the staff evaluation of this issue. This is COL Action Item 2.5.4-1.

3.8.5.7 Conclusions

The staff concludes that the design of the seismic Category I foundation is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDC 1, 2, and 4. The following factors, in addition to the reasons set forth above, provide the basis for this conclusion:

•   By meeting the guidelines of the RGs and industry standards indicated below, the applicant has met the requirements of 10 CFR 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 the importance of the safety functions to be performed.

•   The applicant has met the requirements of GDC 2 by designing the NI foundation mat to withstand the 0.3 g SSE and appropriate combinations of the effects of normal and accident conditions, including the effects of environmental loadings, such as earthquakes and other natural phenomena with sufficient margin for limitations in site data.

•   The applicant has met the requirements of GDC 4 by ensuring that the design of the NI foundation mat can withstand the dynamic effects associated with missiles, pipe whipping, and discharging fluids, excluding dynamic effects associated with pipe ruptures, the probability of which is extremely low under conditions consistent with the design basis for the piping.

The criteria used in the analysis and design, as well as 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, conform with established criteria, codes, standards, and specifications that are acceptable to the staff. This includes ACI-349-01, as supplemented and/or modified in DCD Tier 2, Sections 3.8.4 and 3.8.5. The use of these criteria, as defined by applicable codes, standards, and specifications; the specified loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials; quality control; special construction techniques; and the testing and ISI requirements, provides 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 its structural integrity and stability or the capability to perform its safety functions.

In addition, the staff bases its conclusions regarding the design of the NI foundation mat on its review of samples of design calculations for the critical sections of the foundation mat, described in DCD Tier 2, Section 3.8.5.4.3, "Design Summary of Critical Sections." Therefore, any proposed change to the text of DCD Tier 2, Section 3.8.5.4.3, will require NRC review and approval before implementation.

3.8.6   Other Combined License Action Items

The following sections of DCD Tier 2 include combined license information items in which the staff has determined to be acceptable. These items are repeated below.

•   DCD Tier 2, Section 3.8.6.2, "Passive Containment Cooling System Water Storage Tank Examination," states the following:

> The Combined License applicant should examine the structures supporting the passive containment cooling storage tank on the shield building roof during initial tank filling as described in [DCD Tier 2, Section] 3.8.4.7.

This is COL Action Item 3.8.6-1.

- DCD Tier 2, Section 3.8.6.3, "As-Built Summary Report," states the following:

> The Combined License applicant will evaluate deviations from the design due to as-procured or as-built conditions and will summarize the results of the evaluation in an as-built summary report as described in [DCD Tier 2, Sections] 3.8.3.5.7, 3.8.4.5.3 and 3.8.5.4.2.

This is COL Action Item 3.8.6-2.

- DCD Tier 2, Section 3.8.6.4, "In-Service Inspection of Containment Vessel," states the following:

> The Combined License applicant will perform in-service inspection of the containment according to the ASME Code Section XI, Subsection IWE, as described in [DCD Tier 2, Section] 3.8.2.7.

This is COL Action Item 3.8.6-3.

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 extends to other components such as those portions of the control rod drive mechanisms which are not considered part of the RCPB, 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, preoperational 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 DCD Tier 2, Section 3.9.1, "Special Topics for Mechanical Components," related to the design transients used in the design and fatigue evaluations for ASME Class 1 and core support (CS) components, as well as the 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.

The following requirements provided the basis for the acceptance criteria for the staff's review:

- GDC 1, as it relates to the design, fabrication, erection, construction, testing and inspection of components important to safety in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety function to be performed

- GDC 2, as it relates to the design of mechanical components important to safety to withstand the effects of earthquakes without loss of capability to perform their safety function

- GDC 14, as it relates to the design of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture

- GDC 15, as it relates to the design of mechanical components of the RCS with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences

- 10 CFR Part 50, Appendix B, as it relates to design quality control

- 10 CFR Part 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics

To meet the requirements of the regulations identified above, the DCD must include the following information:

- a complete list of transients to be used in the design and fatigue analysis of ASME Code, Section III, Code Class 1 and Class CS components within the RCPB.

- a list of computer programs that will be used for determination of the structural and functional integrity of seismic Category I mechanical components, including a description of the methods used for computer program qualification.

- if experimental stress analysis methods are used in lieu of analytical methods for any seismic Category I mechanical components, sufficient information to allow the staff to determine its acceptability when compared to the requirements of the ASME Code, Section III, Appendix II.

- if inelastic analysis methods, including ASME Code, Section III, Service Level D limits, are used for any seismic Category I mechanical components, conformance of the analytical methodology used to calculate stresses and deformations to the methods specified in the ASME Code, Section III, Appendix F.

Design of Structures, Components, Equipment, and Systems

3.9.1.1 Design Transients

In DCD Tier 2, Table 3.9-1, the applicant listed the fluid system design transients for five operating conditions and the number of cycles for each transient 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
- test conditions

DCD Tier 2, Section 3.9.1.1, "Design Transients," discusses the basis for the number of cycles for the transients in DCD Tier 2, Table 3.9.1. The number of cycles is 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, adjusted for a 60-year AP1000 plant life. The table does not include the effects of seismic events because the table only addresses fluid system transients. However, in DCD Tier 2, Section 3.9.1.1, the applicant stated that in addition to the cycles due to fluid system transients, the fatigue analyses mentioned above consider the effect of earthquake cycles. DCD Tier 2, Section 3.9.3, discusses the seismic loading conditions included in these analyses. Sections 3.9.3 and 3.12 of this report discuss the staff's evaluation of these conditions. On the basis of the above discussion and the evaluations in Sections 3.9.3 and 3.12 of this report, the staff concludes 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 AP1000 design transients in DCD Tier 2, Section 3.9.1.1, is consistent with the applicable guidelines in Section 3.9.1 of the SRP and is, therefore, acceptable.

3.9.1.2 Computer Programs

The applicant used computer codes to analyze mechanical components. Appendix B to 10 CFR Part 50 requires design control measures to verify the adequacy of the design of safety-related components. In Section 3.9.1 of the SRP, the staff provides guidelines sufficient to meet Appendix B. DCD Tier 2, Table 3.9-15 lists 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, DCD Tier 2, Section 3.9.1.2, "Computer Programs Used in Analysis," includes a description of the method used to verify these programs. The staff's review of this information concludes that the computer code qualification methods described are consistent with the requirements of SRP Section 3.9.1 and, therefore, are acceptable.

3.9.1.3 Experimental Stress Analysis

In DCD Tier 2, Section 3.9.1.3, "Experimental Stress Analysis," the applicant stated that the only experimental stress analysis used for the AP1000 is performed in conjunction with the preoperational flow-induced vibration testing of reactor internals. Section 3.9.2.3 of this report

discusses the staff's evaluation of this issue. For the reasons set forth in Section 3.9.3.2 of this report, the staff concludes that the determination of the vibratory effects of fluid flow on the reactor internals structure uses acceptable methods of experimental stress analysis.

3.9.1.4 Inelastic Analyses

Section 3.12.3.5 of this report discusses the staff's evaluation of the inelastic analysis methodology.

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 S to 10 CFR Part 50, and the guidelines in Section 3.9.1 of the SRP and, therefore, are acceptable.

The applicant met the requirements of GDC 2 and Appendix S to 10 CFR Part 50 by including seismic events in the design transients that serve as the design basis for withstanding the effects of natural phenomena.

To meet the requirements of Appendix B to 10 CFR Part 50 and GDC 1, the applicant demonstrated 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, as well as those not covered by the Code, within the present state-of-the-art limits. The applicant also demonstrated design control measures 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 than those used by the applicant for the design of any safety-related item, with the exception of piping systems, such programs should meet the guidelines of Section 3.9.1 of the SRP. Section 3.12 of this report includes the staff's review of the piping systems.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The staff reviewed the methodology, testing procedures, and dynamic analyses that the applicant used to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loadings. The following requirements provide the basis for the acceptance criteria for the staff's review:

- GDC 14 and 15, by conducting piping vibration, thermal expansion, and dynamic effects testing to ensure structural integrity of the RCPB piping

- GDC 2, by reviewing the seismic subsystem analysis methods

- GDC 1 and 4, by committing to the testing of the dynamic responses of structural components in the reactor caused by steady-state and operational flow transient conditions

Design of Structures, Components, Equipment, and Systems

- GDC 1 and 4, by committing to the flow-induced vibration testing of reactor internals to be conducted during the preoperational 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 Preoperational Vibration and Dynamic Effects Testing

Piping vibration, thermal expansion, and dynamic effects testing should be conducted on all AP1000 plants during the preoperational testing program. These tests confirm that (1) the applicable piping systems, restraints, components, and supports have been adequately designed, fabricated, and installed to withstand flow-induced dynamic loadings under steady-state and operational transient conditions, and (2) the piping system can expand thermally in a manner consistent with the design.

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 DCD Tier 2, Section 3.9.2.1, the applicant included the piping systems discussed above in the AP1000 preoperational vibration and dynamics effects testing programs. In addition, DCD Tier 2, Sections 3.9.2.1 and 3.9.2.1.1 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.

As mentioned above, during the plant's preoperational and startup testing program, all AP1000 license holders will 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 $1E^{10}$ cycles of stress in the pipe during the 60-year design life of the plant. For this reason, stresses associated with steady-state vibration should be minimized and limited to acceptable levels. The test program should consist of a mixture of instrument measurements and visual observations by qualified personnel. DCD Tier 2, Section 3.9.2.1.1, states that piping vibration testing and assessment will be performed in accordance with ANSI/ASME OM-1995, "Operation and Maintenance of Nuclear Power Plants," Part 3. The staff finds that ASME OM Code, Part 3, provides adequate guidance for vibration startup testing of piping systems.

DCD Tier 2, Sections 3.9.2.1.2, "Piping Thermal Expansion Program," 14.2.9.1.7, "Expansion, Vibration, and Dynamic Effects Testing," and 14.2.10.4.25, "Thermal Expansion," state that detailed test specifications for thermal expansion testing of piping systems during preoperational and startup testing are in accordance with the ANSI/ASME OM-1995 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 needed during all plant conditions by verifying the following:

• Piping system restraints accommodate expected expansion.
• Unintentional restraints do not obstruct movement.
• Responses fall within design tolerances.

The standard 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 found this standard to be acceptable for thermal expansion testing of piping systems.

3.9.2.1.1 Conclusions

On the basis of the above evaluation, the staff concludes that the AP1000 piping preoperational vibration, thermal expansion, and dynamic effects test program described in the DCD meets the relevant requirements of GDC 14 and 15 with regard to the design and testing of the RCPB. This provides reasonable assurance of a low probability of rapidly propagating failure and 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 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. This test program conforms to Section 3.9.2 of the SRP and is, therefore, acceptable.

3.9.2.2 Seismic Subsystem Analysis

In DCD Tier 2, Section 3.7.3, the applicant identified those items that are categorized as seismic subsystems. Sections 3.9, 3.10, and 3.12 of this report address the staff's evaluation of the criteria and methodology used for seismic analyses of the mechanical and piping systems and supports and the instrumentation lines and supports of those items listed in DCD Tier 2, Section 3.7.3. Section 3.12 of this report contains a detailed discussion of the piping design. Sections 3.7.3 and 3.10 contain the staff's evaluations of the remainder of the seismic structural subsystems (e.g., cable trays).

Design of Structures, Components, Equipment, and Systems

On the basis of the applicable evaluations in Sections 3.9, 3.10, and 3.12 of this report, the staff concludes that the AP1000 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 <u>Preoperational Flow-Induced Vibration Analysis and Testing of Reactor Internals</u>

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 relate to structural type and location of reactor internal components and reactor operational flow conditions.

The regulatory requirements and guidelines applicable to the reactor internals include the following SRP sections, RGs, and ASME Code requirements. SRP Sections 3.9.5 and 3.9.2 contain design review criteria for vibration loading conditions and dynamic testing and analysis of reactor vessel internals. SRP Section 3.9.5, in part, recommends that reactor vessel internals be designed in conformance with to the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG. ASME Code, Section III, Subsection NG, in Articles NG-3000, paragraphs NG-3111(i), "Loading Conditions," and NG-3112.3(c), "Design Mechanical Loads," require that the reactor vessel internals design take vibratory loads into account.

Section 3.9.2 of the SRP, indicates, in part, that dynamic responses of structural components within the reactor vessel caused by steady-state flow and operational transients should be analyzed for prototype reactors (first of a design type). The SRP indicates that, for nonprototypes, this analysis is not necessary, except that segments of an analysis may be necessary if they deviate substantially from the prototype designs. The SRP further provides guidelines in addition to RG 1.20 that apply to analytical solutions to predict vibrations of reactor internals. The SRP indicates that one acceptable method for formulating forcing functions for vibration predictions is by analyses and test methods based on data from scale model and in-plant testing.

Section 3.9.2 of the SRP further indicates that the preoperational vibration test program for the internals prototype should conform to the guidelines specified in RG 1.20, including vibration prediction, vibration monitoring, descriptions of monitoring instruments and their locations and functions, testing duration of at least $1E^6$ cycles for critical internals components, testing all flow modes of operation and upset transients, data reduction, and pre- and post-hot functional testing inspections.

RG 1.20 presents a method acceptable to the NRC staff for implementing the requirements with respect to the reactor internals during preoperational and initial startup testing. RG 1.20 defines these requirements as Criteria 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50. This criterion requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. RG 1.20 guidelines include the use of a comprehensive vibration assessment program prior to plant operation to verify the structural integrity of the reactor internals by analytical methods, and for the use of the component vibrational measurement program to confirm the analyses.

RG 1.20 characterizes reactor internals designs by the categories of prototype, valid prototype, and nonprototype and indicates the analyses, vibration measurement testing, and inspection programs needed prior to plant operation for each individual category. In DCD Tier 2, Section 3.9.2.4, "Pre-Operational Flow-Induced Vibration Testing of Reactor Internals," the applicant stated that the first AP1000 plant is classified as a prototype, as defined in RG 1.20. Regulatory Positions C.2.1, C.2.2, and C.2.3 of the guideline indicate that the vibration assessment program for a reactor internals design designated a prototype should consist of the following activities:

- Conduct a vibration analysis to predict the internal response to those steady-state and anticipated transient conditions that correspond to preoperational and initial startup test conditions and normal operating conditions.

- Monitor vibrations of reactor internal components during preoperational flow testing with sufficient instrumentation to confirm consistent responses with acceptable safety margins.

- Perform visual inspections of reactor internals prior to and following preoperational startup testing to ensure no indications of structural degradation.

The applicant's initial presentation of the dynamic response analysis program for the AP1000 reactor internals, provided in DCD Tier 2, Section 3.9.2.4, proposes that the COL applicant would perform the entire vibration assessment program. During a meeting with the applicant on July 17, 2002, the NRC staff expressed concern with this proposal because the requirements of 10 CFR 52.47(a)(2) specify that an application for a standard design certification must contain a level of design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design before granting the certification. Delaying the entire vibration assessment program to the COL application stage does not provide the staff with a level of technical information sufficient to reach a final conclusion prior to issuing the design certification on the adequacy of the RPV internals preliminary design.

Following the July 17, 2002, meeting, the staff requested that the applicant address the concerns raised about the RG 1.20, Regulatory Position C.2.1, predictive analysis phase of the internals response program (see RAIs 210.001 through 210.004, 210.006 through 210.008, 210.010, 210.013, and 210.014). In requesting this information, the staff sought to obtain an analytical basis for the qualitative conclusions given in the DCD for the adequacy of the vibratory response of the reactor internals preliminary design. The staff recognizes that the final predictive analysis phase of the RG 1.20 internals vibration assessment program will not be completed until the final design of the reactors internals is fully developed, and that this will not occur until the COL application stage. However, in order to meet the design certification requirements of 10 CFR 52.47(a)(2), the staff requested technical data for review which represent an analytical evaluation of the existing preliminary design of the reactor internals, including an estimation of the fatigue stress margins of safety for the major critical components of the internals structure.

In response to these RAIs, the applicant provided a technical summary report of the reactor internals vibration analysis program documented in WCAP-15949-P, "AP1000 Reactor Internals Flow-Induced Vibration Assessment Program," Revision 0, dated November 2002. Following

Design of Structures, Components, Equipment, and Systems

this report, WCAP-15949-P, Revision 1, and WCAP-15949-NP, Revision 1, dated April 2003, included a revised response to RAI 210.001 by incorporating staff review technical comments on WCAP-15949-P, Revision 0. This report presents the analyses and methodology used to determine the structural adequacy of the AP1000 reactor internals design with regard to the vibrational effects of flow-induced hydrodynamic loadings, probable vortex generation, and pump-induced excitation. The program presented is similar to the program submitted for the design certification of the AP600 reactor vessel internals. Scale model tests, results of instrumented tests on prototype reactors, studied behavior of previous internals designs of operating reactors, and hydrodynamics and structural analyses provide the basis for developing the estimates of flow-induced vibration levels and forces.

In its review of WCAP-15949-P, the staff evaluated the major elements of the AP1000 vibration assessment program, including the predictive analysis of internals vibration response, the proposed methods for both measurement of actual vibration response during preoperational testing, and for inspections of the reactor internals prior to plant commercial operation. Based on its review of this information, the staff finds that the analyses and methodology presented in the report are consistent with the reactor vessel internals vibratory assessment guidance provided in SRP Sections 3.9.5 and 3.9.2, RG 1.20, and the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG. The program defined in WCAP-15949-P establishes an analytical basis for the structural integrity of the conceptual design of the reactor internals subject to flow-induced vibrations, including vibrational measurements, which will be used to confirm the validity of the analytical phase of the program. Furthermore, the staff finds that the vibratory behavior of the reactor vessel internals is adequately characterized because the vibration amplitudes and forces are demonstrated to be sufficiently low with acceptable margins for structural adequacy of the components.

The first AP1000 reactor internals design is classified as a prototype, as defined in RG 1.20. However, in its report, the applicant states that it does not consider the AP1000 reactor vessel internals a first-of-a-kind or unique design based on the general arrangement, design, or size of the internals, or operating experience. Several operating units collectively have similar reactor vessel internals design features and configurations that have successfully completed vibration assessment programs, including vibration measurements, and are referenced in support of the AP1000 reactor vessel internals design. These units, described below, have demonstrated extended satisfactory inservice operation.

The applicant's program to demonstrate the internals components structural adequacy with regard to flow-induced vibration includes the following:

• studies of behavior of evolutionary variations of the reactor internal designs
• scale model tests
• prototype reactor vibration test measurements
• examinations of reactor internals before and after hot functional testing
• analyses for determination of applied forcing functions, component natural frequencies, mode shapes, vibration amplitudes, and stresses

The development of the report data and conclusions include studies of one-, two-, three-, and four-loop reactors which have shown similar vibratory behavior in all four sizes of internals designs. The AP1000 reactor vessel internals represent the evolution of internals design

improvements contained in referenced plants, which include Indian Point, H.B. Robinson, R.E. Ginna, Trojan, Sequoyah, Doel, and Paluel. The subject report addresses the effects of successive reactor vessel internals hardware improvements. The vibration assessment approach for the AP1000 is similar to that applied on previous Westinghouse plants, and is also similar to the program proposed for the AP600 (see WCAP-14761, "AP600 Reactor Internals Flow-Induced Vibration Assessment Program," Revision 3). The results of scale model tests and in-plant vibration measurement programs performed on the reactor vessel internals of the referenced plants were used to develop and support the AP1000 reactor internals vibration assessment program.

The AP1000 reactor vessel internals are most generally similar to the three-loop (3XL) Doel 3 and Doel 4 units, which have incorporated the evolutionary design improvements and which have completed in-plant instrumented measurement test programs. The dimensions of the AP1000, Doel 3, and Doel 4 core barrel diameter, barrel wall thickness, and the annulus between the reactor vessel and core barrel are similar, and the guide tubes and support column designs are identical. The AP1000, Doel 3, and Doel 4 reactor vessel internals flow velocities are similar with the exception of flow through the inlet nozzles and annulus between the core barrel and pressure vessel wall. The AP1000 flow velocity is 18 percent higher in the inlet nozzles and 13 percent lower in the barrel/vessel annulus. The AP1000 higher flow rate, elimination of the thermal shield and neutron pads in the annulus, and the effects of top-mounted in-core instrumentation resulting in simplification of the structures in the lower plenum, in part, cause these variations.

The primary changes in the AP1000 design relative to the Doel designs are the addition of a vortex suppression plate beneath the lower core plate, the replacement of the baffle assembly with a core shroud within the core barrel, and the simplification of the lower support structure resulting from the change from bottom-mounted to top-mounted in-core instrumentation. The overall height of the AP1000 core barrel is slightly larger. The height of the core barrel is 29.2 cm (11.5 in.) longer than the Doel 3 and 4 core barrel, which represents a relatively small dimensional difference of less than 5 percent.

Design modifications incorporated in both the AP1000 and AP600 include elimination of the thermal shield and neutron pads, the replacement of the baffle assembly with a radial reflector/core shroud within the core barrel, change from bottom- to top-mounted in-core instrumentation, and the addition of a vortex suppression plate and support columns. The vortex suppression plate suppresses the formation of standing vortices in the core inlet plenum in laboratory testing performed for the AP600 design.

The applicant has developed detailed computational fluid dynamic (CFD) and FE structural models of the 3XL (Doel) and the AP1000 reactor vessel internals designs. The 3XL FE model calculated vibratory-induced deflections, which were compared to applicable plant test data taken during the Doel hot functional testing. The FE modeling techniques are refined to accurately replicate the Doel test results, and these modeling techniques are then applied to the AP1000 model. The CFD model determined the steady-state flow loads on the upper internals components.

The analyses used to estimate the vibratory forces include broadband flow turbulence, turbulent excitation of the reactor coolant loop fundamental acoustic mode, reactor coolant pump induced

Design of Structures, Components, Equipment, and Systems

excitation, and postulated vortex shedding. Although test data for these structures have not shown vortex shedding, sinusoidal vortex forces were applied to the vortex suppression plate and secondary core support structures in the analytical model for conservatism. The broadband flow turbulent forces are the predominant vibration excitation sources. The CFD model of the 3XL plants was used with power spectral density (PSD) test data to define the broadband turbulent forces and distributions in the inlet nozzle, core barrel/vessel annulus, and lower plenum flow regions. The turbulent forces and distributions were applied to the 3XL FE model, and adjustments and refinements were made to produce conservative estimates of 3XL tests results. These modeling techniques, developed to generate the turbulent forces and distributions, were then applied to the AP1000 finite element structural model to determine its flow-induced vibration behavior in these flow regions.

The preoperational test program for the first AP1000 reactor vessel internals will include an instrumented vibration measuring program and internals inspections conducted before and after preoperational hot functional testing. The changes that are unique to the AP1000 reactor internals will be instrumented for the preoperational vibration measurement program. The internals instrumented measurement program will be similar to plant tests previously used, and will include predicted responses at specific instrumented component locations for comparison with test results.

WCAP-15949-P indicates that the responses measured during hot functional testing will be conservative with respect to normal operating conditions. Evidence is provided to demonstrate that the reactor internals will be subject to higher flow loads during hot functional testing than during normal operation with the fuel assemblies installed. The vibration levels and flow rates are higher during the hot functional testing. Modal analyses performed indicate similar frequency responses and mode shapes with and without the core, and two referenced plants demonstrated this behavior during hot functional testing and initial startup testing.

The applicant has developed and used analytical models of reactor internals designs, similar to the AP1000, which have successfully completed reactor internals vibration measurement test programs, in addition to scale model testing. The applicant has benchmarked the analytical models against these past tests and has applied these refined analytical techniques to the AP1000 analyses. The AP1000 analyses, benchmarked to scale model tests and the instrumented plant tests, demonstrate that the internals vibration levels are low and are acceptable with adequate margins of safety. The development of the reactor internals flow-induced vibration assessment program is consistent with the acceptable method, as stated in SRP Section 3.9.2, for formulating forcing functions for vibration predictions by analyses and test methods based on data from scale model and in-plant testing.

WCAP-15949-P concludes that the AP1000 reactor internals design and vibration assessment program adequately ensure structural integrity against flow-induced vibration. The report satisfies the recommendations of RG 1.20 by specifying the confirmatory pre- and post-hot functional visual and nondestructive examinations, and an instrumented measurement program on the first AP1000 prototype plant. The vibration assessment report includes the descriptions of the flow-induced vibration measurement program and pre- and post-internals inspections to be used for the preoperational tests planned for the first AP1000 plant. The descriptions include the number, location, and sensitivity of accelerometers; strain gauges; pressure and displacement transducers; data acquisition equipment; calibration; data reduction; test

conditions; and provisions for comparisons of predicted responses to expected test results. The reactor vessel internals flow-induced vibration assessment program report, with regard to the first AP1000 prototype plant, is consistent with RG 1.20 recommendations, including predictive analysis, in-plant vibration measurements, and inspections before and after hot functional testing. The report indicates that all subsequent AP1000 plants will satisfy the RG 1.20 guidelines for a nonprototype Category IV plant.

Extensive, demonstrated knowledge and experience with flow-induced vibration analyses, scale model flow testing, in-plant vibration measurements, pre- and post-hot functional testing inspection programs, analyses benchmarked to scale model tests, and studies that characterize and provide insight to the vibrational behavior of previous internals designs support the strength and validity of the applicant's reactor vessel internals vibration assessment program. Based on its review of the analytical data and on the proposed programs for vibration monitoring and physical inspections provided in the report, the staff finds that the reactor internals vibration assessment program presented in WCAP-15949-P, Revision 1, is adequate and consistent with the reactor vessel internals vibratory assessment guidelines provided in RG 1.20, SRP Sections 3.9.5 and 3.9.2, and the ASME Code, Section III, Subsection NG provisions. The approach used in the report is acceptable based on the similarity of the calculated vibrational responses with previously referenced prototype Westinghouse internals designs and their variations, and the provisions for preoperational testing and required reactor vessel internals inspections for the first AP1000 RG 1.20 prototype category plant. The COL applicant will demonstrate and verify confirmation of the reactor internals vibration levels by comparing the predicted responses included in the report with the AP1000 prototype plant preoperational instrumented hot functional program test and inspection results using ITAAC. Any significant anomalies between the predicted vibration assessment responses and the measured vibration test results and internals inspections should be identified to the NRC and reconciled prior to plant operation. This is COL Action Item 3.9.2.3-1.

Based on its review of the analytical methodology and the design margins of safety reported in WCAP-15949-P for the major internals components, as discussed above, the staff concludes that WCAP-15949-P provides an adequate predictive analysis of the effects of flow-induced vibration on the AP1000 reactor internals and provides adequate justification, for purposes of design certification, of the structural integrity of the conceptual design of the AP1000 reactor internals when subjected to operational flow transients. The staff also concludes that the vibration measurement and internals inspection programs specified in WCAP-15949-P and the ITAAC in DCD Tier 1, Table 2.1.3-2, Item 7, adequately address the confirmation of structural integrity of the reactor internals prior to commercial operation. These programs include the COL applicant's preoperational vibration measurement program and pre- and post-hot functional inspection program, as indicated in DCD Tier 2, Section 3.9.8, "Combined License Information."

The staff concludes that the applicant meets GDC 1 and 4 with regard to designing and testing the reactor internals to quality standards commensurate with the importance of the safety functions being performed. The AP1000 design is also appropriately protected against dynamic effects (1) by meeting RG 1.20 for the conduct of preoperational vibration analyses and tests and (2) by having a preoperational 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, pretest inspections, vibration measurement tests, and

Design of Structures, Components, Equipment, and Systems

posttest 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 in-core instrumentation system to ensure safe operation and shutdown of the reactor.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

For its review of this subject, the staff focused on the structural integrity of the reactor internals under the combined effects of a postulated LOCA and an SSE. The following requirements provide the basis for the acceptance criteria for the staff's review:

- GDC 2, as it relates to the design of mechanical systems and components important to safety to withstand appropriate combinations of the effects of normal and accident conditions with the effects of the SSE

- GDC 4, as it relates to the appropriate protection of mechanical systems and components important to safety against the dynamic effects of discharging fluids

In DCD Tier 2, Section 3.9.2.5, "Dynamic System Analysis of the Reactor Intervals Under Faulted Conditions," the applicant stated that reactor internals analysis for ASME Level D service condition events considers the effects of a simultaneous combination of a postulated LOCA and SSE. 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. The applicant described forcing functions, analysis methodology, and modeling techniques. Establishment of the design limitations on deflections and stability of internals components, in addition to stress criteria, controls deformation of the reactor internals structure to ensure adequate core cooling and safe reactor shutdown capability. DCD Tier 2, Table 3.9-14, lists the maximum deflections allowed for the reactor internals support structures. The use of mechanistic pipe break criteria, evaluated in further detail in Section 3.6.2 of this report, determines the AP1000 postulated pipe rupture conditions. The application of LBB analysis criteria to AP1000 high-energy piping qualifies nominal pipe sizes of 15.2 cm (6 in.) and larger for elimination of postrupture dynamic analysis requirements, as discussed in Section 3.6.3 of this report. As a result of the LBB analysis, the 10.2 cm (4 in.) nominal diameter pressurizer spray line and first stage automatic depressurization line are identified as the limiting design-basis high-energy pipe breaks for LOCA pipe rupture analysis.

DCD Tier 2, Sections 3.9.5.2, "Design Loading Conditions," and 3.9.5.3, "Design Bases," specify the design criteria for the reactor internals, including loading combinations and acceptance criteria for Service Level D faulted conditions. The applicant states that the core barrel, core support plates, support columns, core plate and fuel alignment pins, and radial key supports are considered core support structures and are designed to the standards of Subsection NG of Section III of the ASME Code. Other internal structures are designed and fabricated using the ASME Code as a guide, in accordance with ASME Code, Section III, Subsection NG-1122. For ensuring control rod insertion, the applicant indicated in DCD Tier 2, Section 3.9.2.5.3, "Control Rod Insertion," that the guide tubes are evaluated for the limiting high-energy pipe break size combined with SSE loading. The design of the guide tubes permits control rod insertion at each control assembly position under faulted conditions.

The staff reviewed DCD Tier 2, Sections 3.9.2.5, 3.9.3, 3.9.5.2, and 3.9.5.3, and the applicant's responses to RAIs 210.011, 210.012, 210.018, and 210.019. Descriptions of the findings follow.

The Service Level D condition analysis of the reactor internals specified the use of mechanistic pipe rupture criteria as the basis for determining the postulated pipe breaks. DCD Tier 2, Section 3.9.2.5, discusses the application of LBB criteria as the analytical means of eliminating certain piping sizes from consideration of postrupture dynamic effects. However, the DCD discussions did not specifically identify the resulting piping systems and the limiting pipe break sizes used in the design-basis faulted condition analysis of the internal structures. In response to RAIs 210.011 and 210.018, the applicant stated that, for the AP1000, nominal pipe sizes of 15.2 cm (6 in.) and larger are qualified for elimination of pipe rupture postulation for purposes of evaluating the dynamic effects of postrupture piping behavior. Therefore, the largest pipe break analyzed to determine the dynamic response of the AP1000 reactor internals is that of a 10.2 cm (4 in.) nominal diameter pipe connected to the reactor coolant system components or loop piping. The specific lines analyzed for maximum response of the reactor internal structures are the 10.2 cm (4 in.) diameter pressurizer spray lines and the 10.2 cm (4 in.) diameter first stage ADS lines. The staff considers this response to provide an adequate definition of the design-basis postulated pipe rupture conditions to be combined with the SSE for the faulted condition evaluation of the structural integrity of the reactor internals components.

DCD Tier 2, Section 3.9.2.5.2, "Analytical Methods," describes the analytical methods used to calculate stresses and deflections in the reactor internals subjected to Service Level D faulted conditions. The discussion concludes that the reactor internals components are within acceptable stress and deflection limits without providing or referencing any supporting analytical data. The staff requested further technical justification, including a summary of analysis results, in support of these conclusions of adequacy. In response to RAIs 210.012 and 210.019, Westinghouse provided a summary of the margins of safety for Service Level D allowable stresses for the major components of the AP1000 reactor core support structure conceptual design. For this conceptual design phase of the AP1000, the approach taken for demonstration of the adequacy of the core support structures is based on estimating stress margins for the core support components resulting from a comparison of the differences in applied loadings, design configurations, and dimensions between Westinghouse reference plants and the AP1000. The evolution of the AP600, the 3XL plant, and other previously licensed Westinghouse plants provide the basis for the design of the AP1000 reactor internals. The AP1000 design is structurally similar to those designs. The Standard 3XL design serves as the reference plant used for this comparison because of the close similarity between the reactor vessel internals configuration and the dimensions with the AP1000 design.

The applicant's RAI response provided the following discussion of the development of LOCA loads for the faulted condition analysis of the reactor internals. Service Level D LOCA loads for the Standard 3XL plant resulted from a postulated pipe break flow area of 0.093 m^2 (1 ft^2) or larger. For the AP1000, the requirements of pipe break dynamic analysis by the application of LBB criteria exclude nominal pipe sizes of 15.2 cm (6 in.) and larger. Therefore, as discussed previously, the limiting design-basis pipe size used to determine the pipe rupture dynamic response of the AP1000 internals is a 10.2 cm (4 in.) nominal diameter pipe, with a postulated pipe break flow area of less than 64.5 cm^2 (10 in.^{2}) (pressurizer spray line and the first stage ADS line). The Standard 3XL plant LOCA loads and stresses resulting from a 0.093 m^2 (1 ft^2) break area were adjusted where appropriate to account for the small dimensional differences

Design of Structures, Components, Equipment, and Systems

between the 3XL and the AP1000. The resulting enveloping loads and stresses were used to evaluate the stress margins for the AP1000 core support structures. A comparison of the responses of the AP600 core support structure model to forcing functions from these two break area sizes quantified the conservatism of this approach. The loads resulting from a 0.093 m^2 (1 ft^2) LOCA compared with those from a 64.5 m^2 (10 in.^{2}) break (10.2 cm (4 in.) schedule 160 pipe), show that the internals response reactions to the 0.093 m^2 (1 ft^2) break are approximately an order of magnitude higher. Therefore, the enveloping LOCA loads from the 3XL design used for the AP1000 evaluation are conservative. Upon review of this approach for estimating the enveloping LOCA loads and their contribution to the margins of safety for the reactor internals, the staff finds that it is an acceptable method for establishing a basis for structural integrity of the preliminary design of the AP1000 under faulted condition loading, because the actual LOCA loading expected for the AP1000 will be much lower than the LOCA loads resulting from the 0.093 m^2 (1 ft^2) break size used in the above evaluation. The ASME Code, Section III, design reports developed by the COL applicant will include the final stress analyses calculating the actual Service Level D margins of safety used to verify the structural integrity of the reactor core supports and internals components.

The applicant's RAI response provided the following discussion of the development of seismic loads and their combination with LOCA loads for the faulted condition analysis of the reactor internals. The preliminary seismic analysis of the AP1000 reactor internals resulted in SSE loads significantly less (by a factor of at least 5) than the corresponding LOCA loads. These SSE loads were combined with the enveloping LOCA loads, developed from the 3XL plant, to evaluate the stress margins in the AP1000 core support structures. For Service Level D conditions, the SSE and LOCA loads are combined as the square root of the sum of the square of each load component, as specified in the load combinations listed in DCD Tier 2, Table 3.9-5. Revision 1 of the applicant's response to RAI 210.012 provides a table of the margins to allowable stresses for the AP1000 core support structures for Service Level D conditions. These results show positive margins of safety for all AP1000 core support structures, even though the assumed enveloping LOCA loads are very conservative. Accordingly, the staff concludes that this evaluation and the resulting margins of safety provide an adequate justification of the structural integrity of the reactor internals design for faulted condition loading. Furthermore, DCD Tier 2, Section 3.9.8.2, "Design Specifications and Reports," provides the commitment that COL applicants referencing the AP1000 design will have available for NRC audit the design specifications and final design reports prepared for ASME Code, Section III, components. The design reports conforming to ASME Code, Section III, Subsection NG provisions will include the final stress analyses for the reactor vessel internals. This is COL Action Item 3.9.2.4-1. Not withstanding that this matter is labeled as a COL Action Item, it is also Tier 2* information. A COL applicant who wishes to change or depart from this Tier 2* information must seek prior NRC approval in accordance with applicable requirements.

The staff, based on the foregoing review of the above issues, including the additional technical information provided by the applicant in response to staff questions, concludes that the applicant has completed an adequate evaluation of the AP1000 reactor internals response to Service Level D conditions. Therefore, the staff concludes that the AP1000 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 by appropriate combinations of the effects of postulated accident conditions with the effects of the SSE in a dynamic system analysis, an

acceptable basis is provided for establishing the structural design adequacy of the reactor internals to withstand the combined dynamic effects of a postulated LOCA and SSE. The analysis provides adequate assurance that the combined stresses and strains in the components of the 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. Therefore, the staff finds the methods used for analysis of the reactor internals under the combined effects of a LOCA and SSE are adequate, and provide an acceptable basis for the structural integrity of the reactor internals for ASME Service Level D conditions.

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 following requirements provide the basis for the acceptance criteria for the staff's review:

- 10 CFR 50.55a and GDC 1, as they relate to design, fabrication, erection, construction, testing, and inspecting of structures and components to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, as it relates to the design of structures and components important to safety to withstand the effects of earthquakes combined with the effects of normal or accident conditions

- GDC 4, as it relates to the design of structures and components important to safety to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions

- GDC 14, as it relates to the design, fabrication, erection, and testing of reactor coolant pressure boundary to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture

- GDC 15, as it relates to the design of the RCS with sufficient margin to assure that the design conditions are not exceeded

3.9.3.1 Loading Combinations and Stress Limits

The staff review of this subject focused on the design and service loading combinations specified for ASME Code, Section III, components designated as Code Class 1, 2, 3, and Class CS structures. In accordance with SRP Section 3.9.3, this review determines whether appropriate design and service limits have been designated for all loading combinations, and

Design of Structures, Components, Equipment, and Systems

whether the stress limits and deformation criteria comply with the applicable limits specified in the ASME Code, Section III.

Westinghouse evaluated 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 DCD Tier 2, Sections 3.9.3.1 and 3.9.3.2. As discussed in more detail in Section 3.2.2 of this report, safety-related items classified as AP1000 equipment Class A, B, or C are constructed to applicable rules of Section III of the ASME Code. The staff's review of DCD Tier 2, Sections 3.9.3.1, "Load Combinations, Design Transients, and Stress Limits," and 3.9.3.2, "Pump and Valve Operability Assurance," resulted in the following evaluations.

3.9.3.1.1 Loads, Loading Combinations, and Stress Limits

DCD Tier 2, Table 3.9-3, identifies loadings for ASME Code Class 1, 2, 3, and CS systems and components, including piping and supports. This table defines the sustained and transient loading components applicable to ASME Code, Section III, system and component design, including the definition for design-basis pipe break (DBPB) loading that includes both LOCA and non-LOCA transient loads. Additionally, DCD Tier 2, Tables 3.9-5 and 3.9-8 identify loading combinations for ASME Code Class 1, 2, 3, and CS components, and for piping systems and supports, respectively. The staff review of these load definitions and loading combinations finds that they are consistent with the guidance in SRP Section 3.9.3 and, therefore, are acceptable. Section 3.12 of this report includes additional review of loadings and load combinations given in DCD Tier 2, Tables 3.9-3, 3.9-5, 3.9-6, 3.9-7, and 3.9-8 applicable to ASME Code, Section III, piping systems and pipe supports. Sections 3.12.5.3, 3.12.5.14, and 3.12.6.3 of this report further discuss the staff's evaluations of these tables with respect to piping design criteria. Section 3.12.5.12 of this report evaluates DCD Tier 2, Table 3.9-11, "Piping Functional Capability—ASME Class 1, 2, and 3."

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. The AP1000 design does not rely on any active pumps to perform a safety-related function. In addition to testing active valves to demonstrate operability when the valves are subjected to loads up to and including Service Level D, the calculated maximum stress in the valves under these conditions is held to a low value (i.e., only slightly above the allowable yield strength (S_y) of the material). This will help to ensure that the deformations resulting from these loads will be small enough such that the operability of the valve will not be adversely affected. As in the AP600 design, DCD Tier 2, Tables 3.9-9 and 3.9-10 include a note to each table which states that for active valves, pressure integrity verification will be based on 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. For example, for Class 1 valves, the allowable stress will be approximately 1.2 S_y, and for Class 2 and 3 valves the allowable stress will be approximately 1.12 S_y. The staff concludes that these allowable stresses will not result in excessive deformations, because the allowable stress limits result in essentially elastic behavior of the materials of construction and will help to ensure the operability of Class 1, 2, and 3 active valves. Therefore, the allowable stresses are acceptable.

On the basis of the above evaluation and the evaluation in Section 3.12 of this report, the staff finds that the criteria in DCD Tier 2, Tables 3.9-5 through 3.9-10, for loads, loading combinations, and stress limits used in the design of AP1000 ASME Class 1, 2, 3, and CS systems, components, and supports are consistent with the guidelines in SRP Section 3.9.3. Therefore, the criteria are acceptable.

3.9.3.1.2 Environmental Effects on ASME Fatigue Design Curves

Section III of the ASME Code requires evaluation of the cumulative damage resulting from fatigue for all ASME Code Class 1 SSCs. The cumulative fatigue usage factor must take into consideration all cyclic effects caused by plant operating transients, as listed in DCD Tier 2, Table 3.9-1, plus additional cycles induced by seismic events. As the applicant stated in DCD Tier 2, Section 1.2.1.1.2, the AP1000 design life objective is 60 years. Test data to address fatigue concerns indicate that the effects of the reactor environment could significantly reduce the fatigue resistance of certain materials. A comparison of the test data with the ASME 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 operating life. Section 3.12.5.7 of this report discusses the staff's evaluation of this issue for piping. The evaluation also applies to all ASME Code Class 1 SSCs, and any Class 2 and 3 SSCs subject to the discussion below.

3.9.3.1.3 Design of Certain ASME Class 2 and 3 Components for Fatigue

Design of the AP1000 for a 60-year operating life results in ASME Code Class 2 and 3 SSCs that are subjected to loadings that could result in thermal or dynamic fatigue so severe that the required code calculations cannot ensure the 60-year design life. As in the AP600 design, the AP1000 Class 2 and 3 SSCs subjected to such loadings are the nozzles on the secondary side of the steam generators. In DCD Tier 2, Section 5.4.2.1, "Design Bases," the applicant stated that although the secondary side of the steam generator is classified as ASME Code Class 2, all pressure-retaining parts of both the primary and secondary pressure boundaries are designed to satisfy the criteria specified in ASME Code, Section III, Subsection NB for Class 1 components. Since ASME Subsection NB contains acceptable rules for evaluating fatigue in Class 1 components, the staff also finds this approach to be an acceptable method for establishing the basis for a 60-year operating life for Class 2 components.

3.9.3.1.4 Thermal Cycling and Thermal Stratification in Piping Systems

Sections 3.12.5.9 and 3.12.5.10 of this report discuss the staff's evaluations of these issues in detail. For the reasons set forth in those sections, the staff concludes that these issues have been adequately addressed in the design criteria for AP1000 piping systems.

3.9.3.1.5 Intersystem Loss-of-Coolant Accident Design for Piping Systems

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff recommended that the Commission approve the staff's resolution of the intersystem loss-of-coolant accident (ISLOCA) issue for ALWR plants by providing that the 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,

Design of Structures, Components, Equipment, and Systems

1990, the Commission approved the staff's recommendation in SECY-90-016, provided that all elements of the low-pressure system are considered.

The standard design should 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 the applicant's approach for implementing the ISLOCA resolution for the AP1000, in terms of the practicality for systems, components, and equipment. In the following subsections, 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 served 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) which 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.

3.9.3.1.5.1 Low-Pressure Piping Design

To achieve these objectives, the staff evaluated, 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. Table 3.9-1 of this report depicts the results of the staff's evaluation for typical carbon steel (SA-106 Grade B) and stainless steel (SA-312 Types 304 and 316) materials. The report also discusses the results 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 will 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.

3.9.3.1.5.2 Piping Integrity at P_d/P_v = ½ (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 for SA-312, Type 304 and 1.65 for SA-312, Type 316 materials). 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.

3.9.3.1.5.3 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). Some yielding in carbon steel piping at a lower temperature may occur, although to a lesser extent. As a result of significant pipe yielding (without bursting), 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.

3.9.3.1.5.4 Piping Integrity at P_d/P_v = 1/4

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 becomes 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 to the NRC's Office of Nuclear Regulatory Research.

The INEL calculations led to results similar to the qualitative conclusions discussed above. The calculations of the following survival probabilities used a temperature of 177 °C (350 °F). 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 equation 3 in Subarticle NC/ND-3640 of Section III of the ASME Code. The wall thickness

Design of Structures, Components, Equipment, and Systems

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 materials, 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 the same as the thickness designated for standard weight piping (for piping with a diameter of 35.6 cm (14 in.) and less), or is a minimum of Schedule 40 (for piping with a diameter of 40.6 cm (16 in.) and greater), the 90-percent survival probability goal will be attained. The minimum survival 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 Section 3.9.3.1.5.7 of this report, DCD Tier 2, Section 5.4.7.2.2, "Design Features Addressing Intersystem LOCA," 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, therefore, acceptable.

3.9.3.1.5.5 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 1.5x153 MPa = 230 MPa (1.5x2220 psig = 3330 psig).

The Class 900 valve tested to a pressure of 230 MPa (3330 psig) would be expected to withstand an RCS normal operating pressure of 15.4 MPa (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.

3.9.3.1.5.6 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 provisions for shop hydrotests, the method used 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 failure for these remaining components are at least equivalent to that of the piping at its minimum wall thickness because these components typically have wall thicknesses greater than that of the pipe minimum wall thickness.

3.9.3.1.5.7 AP1000 Design Criteria for Intersystem Loss-of-Coolant Accident

The applicant has identified the low-pressure portion of the normal RNS as the only system in the AP1000 plant that carries reactor coolant outside containment that could fail because of overpressurization. In DCD Tier 2, Section 5.4.7.2.2, the applicant 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 AP1000 Class C piping outside containment is 80S.

•   The American National Standard Class for the valves, flanges, and fittings in the AP1000 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 the normal residual heat removal system and component design pressure to the RCS normal operating pressure is 16.21 MPa (900 psig) to 15.4 MPa (2235 psig), or 40 percent.

The staff concludes that these AP1000 ISLOCA design criteria are consistent with the staff's positions relative to piping, valves, and other components in low-pressure systems discussed above is and, therefore, are acceptable. Section 20.3 of this report discusses the implementation of these criteria under Generic Issue 105.

3.9.3.1.5.8 Intersystem Loss-of-Coolant Accident Conclusion

On the basis of the above evaluation, the staff finds that for the AP1000 low-pressure piping systems that interface with the RCS pressure boundary, 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 a minimum wall thickness of the low-pressure piping of schedule 80S, provide an adequate basis for ensuring that these systems can withstand full reactor pressure and thus meet the Commission-approved staff recommendations in SECY-90-016 for designing against

Design of Structures, Components, Equipment, and Systems

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 may 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.

3.9.3.1.6 Design Criteria for Heating, Ventilation, and Air Conditioning Ductwork

The design of HVAC ductwork and ductwork supports is in accordance with design codes and industry standards specified in DCD Tier 2, Appendix 3A. Section 3.8.4.4.3 of this report includes the staff's evaluation of this issue.

3.9.3.1.7 Conclusions

On the basis of the evaluations in Section 3.9.3.1 of this report, the staff concludes that the applicant 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 DCD Tier 2, Section 3.9.3.3, 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. DCD Tier 2, Sections 3.9.3.3, "Design and Installation Criteria of

Class 1, 2, and 3 Pressure-Relief Devices," and 10.3.2.2.2, "Main Steam Safety Valves," state that the design of pressure relieving valves complies with the requirements of Appendix O to Section III of the ASME Code. In addition, the DCD describes supplemental design criteria consistent with Section 3.9.3.II.2 of the SRP. On the basis of the above information, the staff concludes that the criteria in the DCD for design and installation of pressure-relief devices are consistent with applicable guidelines in Section 3.9.3 of the SRP and, therefore, are acceptable.

On the basis of the above evaluation, which states that the criteria in DCD Tier 2, Section 3.9.3.3, 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 the applicant 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 the effect of postulated events such as LOCAs and the dynamic effects resulting from the SSE. DCD Tier 2, Section 3.9.3.3, also meets the requirements of 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 the applicant 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 to ensure that they will withstand these loads without loss of structural integrity or impairment of the overpressure-protection function.

In accordance with 10 CFR 50.34(f)(2)(x), PWR and BWR licensees and applicants must conduct testing to qualify the RCS SRVs and associated piping and supports under expected operating conditions for design-basis transients and accidents (Three Mile Island Action Item II.D.1). DCD Tier 2, Section 1.9.3, "Three Mile Island Issues," paragraph (2)(x), states that the AP1000 reactor coolant system design does not include power-operated relief valves and their associated block valves. However, the safety valve and discharge piping used in the AP1000 design will either be of similar design as those items tested by EPRI and documented in EPRI NP-2770-LD, or will be tested in accordance with the guidelines in Item II.D.1 of NUREG-0737. Section 20.4 of this report discusses the staff's evaluation of this response. The applicant's commitment is consistent with the acceptance criteria used by the staff in its evaluations of Issue II.D.1 for operating plants, and is therefore, acceptable for the AP1000 design.

3.9.3.3 Component Supports

The staff reviewed DCD Tier 2, Section 3.9.3.4, "Component and Piping Systems," 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 criteria, analysis methods, and loading combinations used in establishing a basis for structural integrity of the supports. It addressed plate and shell, linear, and component standard types of supports. The staff conducted its review in accordance with the guidelines in SRP Section 3.9.3, Subsection III.3.

In DCD Tier 2, Section 3.9.3.4, the applicant stated that all ASME Code Class 1, 2, and 3 component supports for the AP1000 design, including piping supports, are constructed in

Design of Structures, Components, Equipment, and Systems

accordance with the ASME Code, Section III, Subsection NF. The rules in Subsection NF provide the basis for the jurisdictional boundary between the Subsection NF supports and the building structure. The staff considers ASME Code, Section III, Subsection NF an acceptable code for the construction of all safety-related component and piping supports. In addition, Subsection NF sufficiently defines the jurisdictional boundaries to ensure a clear division between the component or pipe support and the structural steel. Therefore, the boundaries are acceptable. Section 3.9.3.1 of this report discusses the AP1000 design criteria for loadings and loading combinations for supports.

DCD Tier 2, Section 3.9.3.4 and DCD Tier 2, Tables 3.9-9 and 3.9-10 state that the maximum allowable stress for supports of active components will be held to ASME Code, Section III, Level C stress limits. In DCD Tier 2, Appendix 1A, pursuant to RG 1.130, the applicant states that for Service Level C and D loading conditions, the design of supports for active components uses Service Level C limits. In Revision 0 to WCAP-15799, the applicant's position on SRP Section 3.9.3, Subsection II.3.a, states that in addition to imposing Service Level C stress limits on the design of supports for active components, the evaluation of component operability considers any significant deformation that might occur. These design criteria also apply to snubbers used as supports for active components. The staff concludes that limiting the allowable stress of supports designed to the rules of the ASME Code, Section III, Subsection NF to Service Level C limits, and the additional consideration of support deformation on component operability as stated in the applicant's position on conformance to SRP Section 3.9.3, Subsection II.3.a, will result in control of support deflections such that operability of active components will not be impaired. The staff finds these criteria to be consistent with the guidelines for component supports in SRP Section 3.9.3 and, therefore, acceptable.

In its review of DCD Tier 2, Section 3.9.3.4, the staff also requested additional details of the programs which address surveillance and testing of snubbers used as component supports in the AP1000 design, including specific reference to the industry standards used in the development of these testing plans. In response to RAI 210.068, the applicant provided the following information. ASME Code, Section XI, governs the testing of dynamic restraints (snubbers). DCD Tier 2, Section 3.9.3.4.3, "Snubbers Used as Component and Piping Supports," discusses requirements for the production and qualification testing of hydraulic snubbers. Two types of tests will be performed on the snubbers to verify proper operation:

• production tests, including dynamic testing, on every unit to verify operability
• qualification tests of randomly selected production models to demonstrate the required load performance (load rating)

The production operability tests for large hydraulic snubbers (i.e., those with load capacities of 222.4 kilonewtons (kN) (50 kips) or greater) include (1) a full Service Level D load test to verify sufficient load capacity, (2) testing at full load capacity to verify proper bleed with the control valve closed, (3) testing to verify that the control valve closes within the specified velocity range, and (4) testing to demonstrate that breakaway and drag loads are within the design limits. The applicant revised DCD Tier 2, Section 3.9.3.4.3, to identify the design code governing snubber testing. The ASME OM Code is used to develop the preservice testing plan for snubbers in the AP1000 as part of design certification. Inservice testing (IST) is performed in accordance with Section XI of the ASME Code and applicable addenda (which reference the ASME OM Code, Part 4), as required by 10 CFR 50.55a at the time of COL issuance. The requirements of

10 CFR 50.55a(b)(3) permit the use of ASME OM Code, Subsection ISTD, in lieu of ASME Code, Section XI, for IST of snubbers. Furthermore, DCD Tier 2, Section 3.9.8.3, requires that the COL applicant develop a program to verify the operability of snubbers used in the AP1000 design. The changes to DCD Tier 2, Section 3.9.3.4.3, include a specific reference to the 1995 edition and 1996 addenda of the ASME OM Code for use in developing the IST plan for the AP1000 design certification. The staff's review of this specific reference to the 1995 edition of the ASME OM Code concludes that this ASME OM Code edition may be used as the basis for the AP1000 IST program. While the use of this edition of the ASME OM Code is acceptable in developing a preservice testing plan for design certification, the ASME Code edition and addenda applicable, by reference in 10 CFR 50.55a, to the IST of AP1000 components will be established at the time of COL issuance.

The staff, based on the review described previously, finds the above information to be consistent with the requirements of 10 CFR 50.55a and the applicable guidelines of SRP Section 3.9.3. Therefore, it is acceptable. Commitments to inservice inspection found in DCD Tier 2, Section 6.6, "Inservice Inspection of Class 2 and 3 Components," are in accordance with the ASME Code, Section XI, and further ensure snubber operability during plant operation. Based on its review of the information provided in the DCD and clarifications provided in the applicant's responses discussed above, the staff concludes that these provisions will provide an acceptable program for ensuring the operability of snubbers used as ASME Code Class 1, 2, and 3 component supports. The provisions are consistent with the requirements of 10 CFR 50.55a and, therefore, are acceptable.

On the basis of its evaluation of DCD Tier 2, Section 3.9.3.3, supplemented by the evaluations in applicable portions of Section 3.12.6 of this report, the staff concludes that the applicant 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 an 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 Section 3.9.3, and therefore, 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 ensure that support components can withstand the most adverse combination of loading events without loss of structural integrity.

Based on its review of the information discussed above, the staff concludes that the applicant has established an acceptable basis for the structural integrity and functional capability of the AP1000 pressure-retaining component support structures that are designed in accordance with ASME Code, Section III, Class 1, 2, and 3 requirements. Section 3.9.5 of this report provides the staff's evaluation of core support structures designed to ASME Code, Section III, Class CS standards.

Design of Structures, Components, Equipment, and Systems

3.9.4 Control Rod Drive Systems

The staff's review under SRP Section 3.9.4 included the control rod drive system (CRDS) up to its interface with the control rod cluster assemblies. Those components of the CRDS that are part of the primary pressure boundary are classified as Safety Class 1, Quality Group A, and are designed according to ASME Code, Section III, Class 1 requirements and are in accordance with the quality assurance requirements of Appendix B to 10 CFR Part 50. The staff reviewed the CRDS to ensure that it will reliably control reactivity changes under conditions of anticipated normal plant operational occurrences and under postulated accident conditions. The staff reviewed the information in DCD Tier 2, Section 3.9.4, "Control Rod Drive System (CRDS)," related to the criteria used to ensure the structural integrity of the CRDS during normal operation and under postulated accident conditions. The staff reviewed the criteria for conformance to the acceptance criteria in Section 3.9.4 of the SRP. Section 3.9.3 of this report discusses loading combinations for the CRDS.

Section 3.9.7 of this report discusses the evaluation of the structural integrity of the seismic restraints for the CRDM. Section 4.6 of this report includes additional evaluations related to the functional design and testing of the CRDS.

The staff based its review of the design and acceptance criteria for the CRDS on the following requirements and guidance:

- GDC 1 and 10 CFR 50.55a, requiring, in part, that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, requiring, in part, that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions

- GDC 14, requiring, in part, that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture

- GDC 29, requiring, in part, that the CRDS be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences

- SRP 3.9.4 guidelines for quality group classification, design, construction, and operability assurance for pressurized and nonpressurized equipment in accordance with appropriate codes and standards utilized by the nuclear industry

- SRP 3.9.3 guidelines for service loading combinations and allowable stress limits for those portions of the CRDS classified as Quality Group A reactor coolant pressure boundary components

DCD Tier 2, Section 3.9.4, presents the technical information supporting the design basis for the CRDM. The primary functions of the CRDM are to insert or withdraw the rod cluster control assemblies and the gray rod control assemblies from the reactor core to control average core temperature and to control changes in reactivity during reactor startup and shutdown. The AP1000 CRDM is a magnetically operated jack consisting of an arrangement of three

electromagnets energized in a controlled sequence to insert or withdraw the rod control assemblies in the reactor core in discrete steps. The CRDM is designed to release the rod control assemblies during any part of the power cycle sequencing in the event that electrical power to the electromagnets is interrupted. When released from the CRDM, the rod control assemblies fall by gravity into a fully inserted position within the reactor core. The pressure housing subassembly of the CRDM forms a part of the RCPB. The CRDM pressure housing is constructed in conformance with the requirements of 10 CFR 50.55a, including design, analysis, materials, fabrication, and quality assurance requirements for Class 1 components specified in Section III of the ASME Code.

DCD Tier 2, Section 3.9.4.2.3, "Internal Component Requirements," states that design of the CRDM non-pressure-boundary components is based on the material specification mechanical property requirements. However, material property specifications typically specify minimum required yield stress and minimum required tensile ultimate stress but do not include allowable stress criteria for design purposes. The staff requested, in RAI 210.062, more specific identification of the allowable stress criteria used for the design of CRDM non-pressure-boundary components, including design considerations given to the establishment of a 60-year design life for the moving components of the CRDM. In response to RAI 210.062, the applicant stated that the CRDM non-pressure-boundary components, including the latch assembly and coil stack assembly, do not come under the jurisdiction of the ASME Boiler and Pressure Vessel Code. However, as a conservative approach, the applicant used minimum material properties, where available for the non-pressure-boundary components, as provided by ASME Code, Section III. Material properties specified by the ASTM are used for those materials not covered by the ASME Code. Although maintenance of structural integrity under the design loading conditions is the only governing design requirement for these components, ASME Code, Section III, criteria serve as guidelines for the allowable stresses.

The CRDM pressure boundary components are designed for a 60-year life. The moving, non-pressure-boundary components, such as the latch assembly, are not designed to have a 60-year operating life. The CRDM design includes provisions for replacement of these components as needed during the life of the plant.

The staff finds this response to be acceptable because it provides for a conservative design approach by using ASME Code allowable stress criteria for non-pressure-boundary components which do not fall within ASME Code jurisdiction.

The discussion in DCD Tier 2, Section 3.9.4.3, concerning the analysis to ensure the functional capability of the CRDM, does not clearly define the acceptance criteria. In RAI 210.063, the staff requested additional details of the analysis, including definition of the ASME Code, Section III, service level stress limits used to define the allowable bending moments and the allowable deformations of the CRDM. Additionally, the staff requested a justification of the use of ASME Code, Section III, stress criteria (normally used for establishing pressure boundary structural integrity) for the purpose of ensuring component functional capability. In response to RAI 210.063, the applicant provided a summary of the design criteria applicable to the CRDM pressure housing that forms part of the reactor vessel pressure boundary. The design of the pressure housing is in accordance with the requirements of Section III, Subsection NB of the ASME Code. Results of stress analyses of the CRDM pressure housing are compared to the ASME Code stress limits for design conditions and for service limits associated with ASME

Design of Structures, Components, Equipment, and Systems

Service Levels A, B, C, and D. DCD Tier 2, Table 3.9-5, gives the load combinations which must be considered for each of the design and service level conditions. The ASME Code, Section III, subsections indicated below give the stress limits for each service condition:

- Design Conditions NB-3221
- Normal (Service Level A) NB-3222
- Upset (Service Level B) NB-3223
- Emergency (Service Level C) NB-3224
- Faulted Conditions (Service Level D) NB-3225/Appendix F

A standard Westinghouse design utilized in currently operating reactors provides the basis for the AP1000 CRDM. Tests and analyses of this design have shown that when the bending moments on the CRDM are limited to those that produce stress levels in the CRDM pressure housing less than ASME Code stress limits during anticipated transient conditions, the functional capability of the CRDM is assured. The design process thus accounts for CRDM functional capability because meeting the ASME stress limits for the pressure housing limits bending moment deformations to the extent that the CRDM drive rods do not bind during insertion of the control rod assemblies. On this basis, the staff finds this application of ASME service limits and stress criteria for establishment of deformation limits to be an acceptable means of designing the CRDM for functional capability.

DCD Tier 2, Section 3.9.4.4, "Control Rod Drive Mechanism Performance Assurance Program," discusses functional test programs that have been conducted to confirm the operational capability of the CRDM. In RAI 210.064, the staff requested additional details of these tests, including the criteria used for demonstration of CRDM operational capability following exposure to the combined effects of a LOCA and an SSE. In response to RAI 210.064, the applicant cited tests of CRDM designs similar to the AP1000 conducted to ensure the ability of the CRDM to function under postulated faulted condition loading. Two reports, WCAP-8446 (Proprietary) and WCAP-8449 (Nonproprietary), "17x17 Drive Line Components Tests—Phase 1B 11, 111 D-Loop Drop and Deflection," document these tests. The NRC staff has previously reviewed and accepted these documents. The results of these tests were used to establish a deflection limit for the top of the CRDM rod travel housing.

The CRDM seismic support provided by the integrated head package (DCD Tier 2, Section 3.9.7, "Integrated Head Package") limits the deflection of the top of the CRDM rod travel housing. This deflection limit restricts the bending moments on the CRDM to values that result in stress levels in the CRDM pressure boundary that are less than the ASME Code stress limits for anticipated transient or postulated accident conditions. As discussed above, this limitation on bending deflection ensures that any distortion of the rod travel housing does not interfere with the movement of the CRDM drive rods during insertion of the control rod assemblies. The stress analysis evaluates load combinations that include the combined effects of postulated pipe rupture and SSE loadings as faulted (Level D) events. Subsection NB-3225 and Appendix F to ASME Code, Section III, specify the stress limits for this analysis.

Based on its review of DCD Tier 2, Section 3.9.4, the additional supporting technical information provided by the applicant, and for the reasons set forth above, the staff concludes that the design of the CRDS for the AP1000 meets GDC 1, 2, 14, and 29 and 10 CFR 50.55a and is thus acceptable. By designing the CRDS, up to its interface with the control rod cluster assemblies,

to acceptable loading combinations of normal operation and accident conditions using ASME Code Class 1 and the requirements of Appendix B to 10 CFR Part 50, the applicant has ensured the structural integrity of the CRDS. Therefore, the applicant 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, the applicant meets GDC 2 and GDC 14 with regard to designing the CRDS to withstand the effects of a postulated earthquake 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 applicant meets the requirements of GDC 29 for CRDS operability under anticipated normal operational occurrences by imposing a performance assurance program consisting of production tests of each CRDM prior to installation, trip time testing during preoperational and startup testing, and operational testing at periodic intervals after initial reactor operation, as required by the technical specifications.

The staff's review of the design configuration and description of operation of the CRDM confirms that the applicant has based the AP1000 CRDM on a proven Westinghouse electromagnetic jack design that has been used successfully in many operating nuclear reactors. Sections 3.9.1 and 3.9.3.1 of this report discuss the staff's evaluations of the specified design transients, design and service loadings, and combinations of loads. 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.

Based on review of the design information provided in DCD Tier 2, Section 3.9.4, and on the additional information included in the applicant's responses to the RAIs, the staff concludes that the design of the CRDS is structurally adequate and provides a reliable means of movement of the control rod assemblies within the reactor core under conditions of normal plant transients or under postulated accident conditions.

3.9.5 Reactor Pressure Vessel Internals

In accordance with Section 3.9.5 of the SRP, the staff reviewed DCD Tier 2, Section 3.9.5, relative to the specified design codes, load combinations, allowable stress and deformation limits, and other criteria used in the design of the AP1000 reactor internals. The staff based its review of acceptance criteria 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 their 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 accidents, including LOCAs

- 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

Design of Structures, Components, Equipment, and Systems

The staff's review of this subject, as set forth in SRP Section 3.9.5, focuses on the design and construction of the reactor core support structures to ensure that the design conforms to the provisions of the ASME Code, Section III, Subsection NG. Furthermore, the design criteria and loading conditions that provide the basis for the design of reactor internals other than core support structures should meet the applicable guidelines of the ASME Code, Section III, Subsection NG, and be designed so as not to adversely affect the integrity of the core support structures as outlined in ASME Code, Section III, Subsection NG, paragraph NG-1122.

In DCD Tier 2, Section 3.9.5.3, "Design Basis," the applicant included a general discussion of the design bases for the reactor internals. However, the information provided did not specifically identify the design codes and acceptance criteria applicable to the design, analysis, fabrication, and nondestructive examination of the internals components. Furthermore, the DCD did not identify those internals components designated as core support structures, those designated as internal structures, and the implications of this designation on applicable design criteria. In the response to RAIs 210.020 and 210.021, the applicant provided specific clarification of the design criteria. The applicant revised DCD Tier 2, Section 3.9.5.3, to state that reactor vessel internals components designated as ASME Code, Section III, Class CS core support structures are designed, fabricated, and examined in accordance with the provisions of Subsection NG of Section III of the ASME Code. The design documentation for these Class CS core support structures includes a certified design specification and certified design report conforming to the provisions of Subsection NCA of ASME Code, Section III. In accordance with Subsection NG-1100, this means that the construction and installation of the AP1000 core support structures are in accordance with the Subsection NG rules for materials, design, fabrication, examination, and preparation of reports. For design of components, this means that Service Level A, B, C, and D conditions should meet requirements shown in Figures NG-3221-1, NG-3224-1, NG-3232-1, and Appendix F to Section III of the ASME Code. The staff's review finds that this conforms to Section 3.9.5 of the SRP and is, therefore, acceptable. In addition, the applicant's responses to RAIs 210.020 and 210.021 specifically identified those reactor internals components designated as ASME Code, Section III, Class CS core support structures, as follows:

- core barrel assembly (flange and cylindrical shell)
- lower core support plate and fuel alignment pins
- lower radial restraint system (keys and clevis inserts)
- upper support assembly (flange, plate, and cylindrical skirt)
- upper support columns
- upper core plate and fuel alignment pins
- upper core plate alignment pins and clevis inserts

Those reactor internals components not designated as ASME Code, Section III, Class CS core support structures are designated as internal structures in accordance with ASME Code, Section III, Subsection NG-1122. As provided in paragraph NG-1122, Westinghouse, as an ASME certificate holder, defines the design criteria for these internal structures. As provided by ASME Code, Section III, Subsection NG-1122(c), these internal structures components should be constructed so as not to adversely affect the integrity of the core support structures. The staff, based on the above review of this additional information, finds that these criteria conform to Section 3.9.5 of the SRP and are acceptable.

In RAI 210.023, the staff requested clarification of allowable deflection criteria presented in DCD Tier 2, Table 3.9-14. The staff also requested more specific identification of the corresponding components (listed in DCD Tier 2, Table 3.9-14) in the reactor vessel interface arrangement drawing presented in DCD Tier 2, Figure 3.9-8. In its response to RAI 210.023, the applicant explained the bases for deflection limits and also provided clarifying configuration details in the revisions to DCD Tier 2, Table 3.9-14 and Figure 3.9-8.

The applicant clarified the description of the deflection limits by providing the following information. The upper core barrel radial inward deflection limit is necessary to prevent contact between the core barrel and the peripheral upper guide tubes during a LOCA event, such that insertion of the control rods will not be impaired. The upper core barrel 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 with control rod insertion to maintain acceptable control rod drop times. The staff, based on the above review of this revised information, finds that the deflection 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 effects of a postulated combination of loads from a LOCA and SSE. Therefore, the deflection limits are acceptable.

The staff also, questioned in RAI 210.017, whether the design and analysis of the reactor internals structure accounts for the potential effects of thermal stratification. In its response to RAI 210.017, the applicant indicated that the design of the AP1000 reactor vessel and reactor internals addresses thermal stratification during those few operating conditions when stratification could occur. The AP1000 TS require reactor coolant pump operation, and thus forced circulation flow through the reactor vessel, when the plant operates in Modes 1 through 5. In Modes 1 and 2, all four reactor coolant pumps must operate (DCD Tier 2, Section 16.1, TS LCO 3.4.9). In Modes 3, 4, and 5, a TS (DCD Tier 2, Section 16.1, LCO 3.4.4) requires maintaining a minimum flow through the core, with at least one reactor coolant pump operating. During these five modes of operation, forced flow conditions exist, and fluid velocities in the reactor vessel are sufficiently high to preclude thermal stratification. The lowest fluid velocities occur in the upper reactor vessel head. Studies have been performed, and measurements have been taken, which indicate that the flow pattern in the upper head regions of plants with spray nozzles, like the AP1000, is a momentum-dominated circulation rather than a buoyancy-driven stratification. Because the AP1000 has higher spray nozzle flow rates than those of typical plants and has drain holes in the upper support plate, this momentum-dominated circulation pattern characterizes the behavior of the AP1000, as well. Therefore, during operation in Modes 1 through 5, thermal stratification is not an issue for the reactor vessel components.

Thermal stratification could occur in the reactor vessel during passive core cooling system operation and natural circulation cooldown. Thermal stratification in the reactor vessel is considered by performing a thermal/flow analysis which provides temperature distribution data used to evaluate thermal stresses and cyclic loading effects on fatigue life. Stress analysis results show that the reactor vessel complies with the stress and fatigue limits of ASME Code, Section III. The staff finds that this response adequately ensures that fatigue evaluations for the design and analysis of the AP1000 reactor vessel components consider the cyclic effects of thermal stratification, and the response is, therefore, acceptable.

Design of Structures, Components, Equipment, and Systems

In RAI 210.027, the staff requested clarification of the Service Level A and B conditions defined in DCD Tier 2, Section 3.9.5.2.1, "Level A and B Service Conditions," and Table 3.9-5, specifically with respect to the inclusion of earthquake loading in these service conditions. In response to RAI 210.027, the applicant clarified the "earthquake" terminology used. In DCD Tier 2, Section 3.9.5.2.1, the "earthquake" listed is a transient seismic event that is included only in the fatigue evaluation of the reactor internals structure. The evaluation considers five seismic events, each with an amplitude equal to one-third of the SSE response. Each of the one-third SSE transient events has 63 high-stress cycles. Service Levels A or B in DCD Tier 2, Table 3.9-5, do not include earthquake loading because this "earthquake" is considered only for fatigue evaluation by the inclusion of the additional cycles, as defined above, to the thermal transient cycles given in DCD Tier 2, Table 3.9-1. The applicant designed the AP1000 reactor internals for one occurrence of mechanical loadings due to SSE excitation, which is evaluated as a Service Level D condition and is included in DCD Tier 2, Table 3.9-5 for design-loading combinations for ASME Class CS systems and components. The staff review verified that the applicant revised DCD Tier 2, Section 3.9.5.2.1. The staff concludes that the criteria presented in the applicant response are consistent with seismic analysis guidance in 10 CFR Part 50, Appendix S, and SRP Section 3.7.3 and, therefore, provide an acceptable basis for design of the AP1000 reactor internals for the effects of transient seismic events.

To ensure the safety function of the reactor internals under Service Level D faulted conditions, the staff requested that the applicant provide analytical data justifying the adequacy of the response of the reactor internals to faulted condition loading. The applicant's response to RAIs 210.012 and 210.019 indicated that, to ensure the safety function of the reactor internals, the applicant analyzes and compares the stresses on the internals to the allowable stresses of ASME Code, Section III, Appendix F, while also meeting the deflection limits in DCD Tier 2, Table 3.9-14 and Section 3.9.5.3.2, "Allowable Deflections." As discussed in the staff's evaluation of the faulted condition analysis of the reactor internals design in Section 3.9.2.4 of this report, these criteria are acceptable because they provide adequate margins to maintain the geometry of the reactor internals components for control rod insertion and for core flow passage.

As a result of its review of DCD Tier 2, Table 3.9-5, the staff requested justification for not including DBPB loading in the Service Level C loading combinations, since DBPB loading is included in the Level C loadings suggested by SRP Section 3.9.3. In response to RAI 210.028, the applicant provided the following justification. The applicant defines the Service Level C pipe break to be a maximum of 2.54-cm (1-in.) nominal diameter pipe size in a Class 1 branch line (DCD Tier 2, Section 3.9.1.1.3.1, "Small Loss-of-Coolant Accident"). This is somewhat larger than the DBPB identified in SRP Section C.1.3.3 for Appendix A to SRP Section 3.9.3, which is equivalent to a 0.952-cm (0.375-in.) nominal diameter break (i.e., the break size in a Class 1 branch line that results in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system). Postulated breaks in 2.54-cm (1-in.) nominal diameter piping and smaller piping, in accordance with guidance in SRP Section 3.6.2, do not require the analysis of the dynamic mechanical loadings from the ruptured pipe on reactor coolant system components and therefore are not included in DCD Tier 2, Table 3.9-5, which gives the loading combinations for mechanical loads.

A break in a 2.54-cm (1-in.) diameter Class 1 branch line results in reactor coolant system temperature and pressure transient conditions and is thus included in the reactor coolant system design transients given in DCD Tier 2, Table 3.9-1. The applicant analyzed the reactor vessel

internals for each of the design transients, either individually or by using conservative enveloping transients to show that the internals meet the appropriate ASME Code, Section III, stress limits. Pipe breaks in lines larger than 2.54-cm (1-in.) nominal diameter are LOCAs, which are treated as faulted condition events. DCD Tier 2, Table 3.9-5, includes the mechanical loadings resulting from a LOCA under Service Level D conditions. The staff, based on the above review and the applicant revision of DCD Tier 2, Section 3.5.1.1.3.1, concludes that this information provides an adequate justification for the Service Level C load combinations in DCD Tier 2, Table 3.9-5, is consistent with the guidance in Appendix A to SRP Section 3.9.3 and is, therefore, acceptable.

On the basis of the evaluation and resolution of the staff's questions discussed in this section, the staff reached the conclusions presented below relative to the design of the AP1000 reactor internals.

In accordance with DCD Tier 2, Table 3.2-3, the core support structures and safety-related reactor internals are designed as Safety Class 3 components to the quality assurance requirements of 10 CFR Part 50, Appendix B. In addition, as discussed in Sections 3.9.1.1, 3.9.2.4, and 3.9.3.1 of this report, the DCD contains acceptable criteria for the design of reactor internals under ASME Code, Section III, Service Level A (normal), B (upset), C (emergency), and D (faulted) condition loading.

On this basis, the staff finds that the applicant 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 the evaluation related to designing 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, the staff finds that the applicant 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 the reactor internals maintain their capability to perform their safety functions and the specified fuel design limits are not exceeded.

The application of the criteria discussed above to the design of the reactor internals 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 postulated to occur during their service lifetime without loss of structural integrity or impairment of function.

The staff concludes that the design of reactor internals for the AP1000 meets GDC 1, 2, 4, and 10 and 10 CFR 50.55a and is, therefore, acceptable.

Design of Structures, Components, Equipment, and Systems

3.9.6 Testing of Pumps and Valves

In DCD Tier 2, Section 3.9.6, "Inservice Testing of Pumps and Valves," the applicant discussed IST of certain safety-related pumps and valves typically designated as ASME Code Class 1, 2, or 3. The staff based its review of DCD Tier 2, Section 3.9.6, and its acceptance criteria on the following requirements:

• GDC 1, as related to designing, fabricating, erecting, and testing SSCs important to safety to quality standards commensurate with the importance of the safety functions to be performed

• GDC 2, as related to designing SSCs important to safety to withstand the effects of natural phenomena without loss of capability to perform their safety functions

• GDC 4, as related to designing SSCs important to safety to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents

• 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 components

• 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 systems and components classified as ASME Code Class 1, 2, and 3 are designed and provided with access to enable the performance of inservice 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 AP1000 design. The design uses no safety-related pumps, except for the reactor coolant pump whose only safety function is its coast-down function. The load combinations and stress limits used in the design of valves ensure maintenance of the integrity of the component pressure boundary. In addition, a licensee will periodically test the performance and measure performance parameters of safety-related valves in accordance with the ASME Code for the Operation and Maintenance of Nuclear Power Plants (OM Code), as required by 10 CFR

50.55a(f). The Code requires comparison of periodic measurements of various parameters to baseline measurements to detect long-term degradation of valve performance. The tests, measurements, and comparisons will ensure the operational readiness of these valves. However, as discussed in SECY-90-016, the staff determined that the requirements of the ASME Code at that time (1990) might not ensure the necessary level of component operability desired for ALWR designs. Accordingly, in SECY-90-016, as supplemented by the staff's April 27, 1990, response to comments of the Advisory Committee on Reactor Safeguards (ACRS), the staff recommended to the Commission criteria to supplement those in the ASME Code. In its SRM to SECY-90-016, dated June 26, 1990, the Commission approved the staff's recommendations. The staff proposed the following criteria for pump and valve testing:

- Piping design should incorporate provisions for full-flow testing at maximum design flow of pumps and check valves.

- Check valve testing should incorporate the use of advanced nonintrusive techniques to address degradation and performance characteristics.

- Provisions should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation not detectable through the use of advanced nonintrusive techniques.

- Provisions should be incorporated to test motor-operated valves (MOVs) under design-basis differential pressure.

Since issuance of the above policy papers and the design certification of the AP600, the ASME Code requirements and regulations that diminish or preclude the need for additional criteria on IST have changed. The most significant change is the transfer of IST requirements from ASME Code, Section XI, to the OM Code and the subsequent incorporation by reference of the OM Code into the regulations in 10 CFR 50.55a. As a result, the staff based its review of the AP1000 on the adequacy of the design in providing access to enable IST of pumps and valves using the OM Code rather than the ASME Code, Section XI. In addition, the ASME OM Code and the regulations have subsequently addressed the concerns associated with the proposed IST criteria. Therefore, the staff based its evaluation of DCD Tier 2, Section 3.9.6 on meeting the requirements of the ASME OM Code, as well as on the IST criteria proposed above as they apply today to the AP1000. The following sections discuss the staff's evaluation.

3.9.6.1 Testing of Pumps

In DCD Tier 2, Section 3.9.6.1, "Inservice Testing of Pumps," the applicant stated that the only safety-related function performed by an AP1000 pump is the coastdown of the RCP. As a result, the AP1000 IST plan does not include any pumps. Because the coastdown of the RCP is not a testable function to establish operational readiness under the IST program, the staff finds that its exclusion from the IST program is acceptable. Accordingly, the staff concludes that there are no safety-related pumps that must be included in the AP1000 IST program.

Design of Structures, Components, Equipment, and Systems

3.9.6.2 <u>Testing of Safety-Related Valves</u>

In accordance with the regulatory requirements and supplementary criteria discussed in Section 3.9.6 of this report, the AP1000 design must ensure the capability of MOVs to perform their safety functions under design-basis conditions. The design-basis capability should be verified before installation of the MOVs, before startup, and periodically throughout plant life. To address the concerns and issues identified in Generic Letter (GL) 89-10, its supplements, and GL 96-05, 10 CFR 50.55a(b)(3) specifies a requirement to supplement the ASME OM Code provisions for MOV testing. COL applicants must address those concerns and issues before plant startup. In DCD Tier 2, Section 5.4.8.1.2, "Motor-Operated Valves Design and Qualification," the applicant describes its criteria for sizing motor operators for valves and commits to design and qualify the MOVs for a range of conditions up to their design conditions. The applicant states that MOVs are designed to change their position from an improper position (mispositioned) either before or during accidents. The applicant provides, where possible, for in situ testing of MOVs under a range of conditions up to maximum design-basis operating conditions for valve opening or closing in the appropriate direction for the safety-related function of the valve. In DCD Tier 2, Section 5.4.8.5, "Preoperational Testing," the applicant specifies provisions for preoperational testing before startup that COL applicants may use to demonstrate that the results of testing under in situ or installed conditions can confirm the capacity of MOVs to operate under design conditions. In DCD Tier 2, Section 5.4.8.5.2, "Motor-Operated Valves," the applicant stated that during preoperational testing, active safety-related MOVs are tested to verify that the valves open and close under static and safety-related conditions. Where the safety-related design conditions cannot be achieved, the applicant stated that testing is performed at the maximum achievable dynamic conditions. Preoperational testing and evaluation demonstrate the acceptability of the MOV's functional performance. The NRC regulations in 10 CFR 50.55a(b)(3) will require the applicant to develop an IST program for MOVs consistent with staff positions and criteria as identified in GL 89-10, its supplements, and GL 96-05 to demonstrate the design-basis capability of the MOVs throughout the plant life. The staff has reviewed the submitted information and finds that it meets the staff's positions and criteria and is, therefore, acceptable.

In DCD Tier 2, Section 5.4.8.1.3, "Other Power-Operated Valves Including Explosively Actuated Valves Design and Qualification," the applicant stated that safety-related power-operated valves (POVs) other than MOVs in the AP1000 are designed to operate at design operating conditions. Further, functional qualification is performed to demonstrate the ability of POVs to operate under design conditions. A program similar to that recommended for MOVs will likely verify the design-basis capability of these POVs before installation, before startup, and periodically thereafter. In DCD Tier 2, Section 5.4.8.5, the applicant specifies provisions for preoperational testing before startup that COL applicants should use to demonstrate that the results of testing under in situ or installed conditions confirm the capacity of POVs to operate under design conditions. In DCD Tier 2, Section 5.4.8.5.3, "Power-Operated Valves," the applicant stated that active safety-related POVs are tested in preoperation to verify that they open and close under static and dynamic conditions. When it cannot achieve design conditions, preoperational testing occurs at maximum achievable dynamic conditions. Preoperational testing verifies that the valves open and close as applicable at a range of conditions up to the design conditions to perform their safety functions. The staff has reviewed this information and finds that it meets the applicable staff positions and is, thus, acceptable. The design of the AP1000 system will incorporate provisions to permit all critical check valves to be tested for performance in both the

forward and reverse flow directions. The ASME OM Code (1996 Addenda) includes this requirement for bi-directional testing of check valves. The staff finds that meeting these ASME OM Code requirements by the AP1000 system design is acceptable.

According to the staff's position on the use of nonintrusive diagnostic techniques as stated in SECY-90-016, IST should incorporate the use of advanced nonintrusive techniques to periodically assess degradation and performance characteristics of the check valves. The system and component design should accommodate nonintrusive diagnostic methods. In DCD Tier 2, Section 3.9.8.4, the applicant stated that the IST program will include provisions for nonintrusive check valve testing methods. In DCD Tier 2, Section 5.4.8.1.1, "Check Valve Design and Qualification," the applicant discusses the use of nonintrusive test methods for check valves. This is consistent with the applicable staff positions and is thus acceptable.

With regard to flow testing of check valves, the applicant stated in DCD Tier 2, Section 5.4.8.1.1, that all active safety-related check valves include the capability to verify valve obturator movement by a direct indication or by using nonintrusive test methods. The applicant stated in DCD Tier 2, Section 5.4.8.5.1, "Check Valves," that in most cases, the system design permits full-flow testing of check valves during applicable plant modes or sufficient flow to fully open the check valve to demonstrate valve operability under design conditions. Where this testing is not possible, an alternative method of demonstrating operability is developed and justified. The applicant stated in DCD Tier 2, Section 3.9.6.2.2, that check valves will be tested under sufficient flow to fully open the valve unless the maximum accident flow is not sufficient to fully open the valve. It is acceptable to exercise check valves with sufficient flow to fully open the valve, provided the valve's full-open position can be positively confirmed.

As discussed in Section 3.9.6 of this report, the AP1000 design provides for periodic disassembly and inspection of safety-related valves to check for indications of unacceptable corrosion or degradation not detectable through the use of advanced nonintrusive techniques. The staff believes that information derived from IST alone is not adequate to assess valve condition and to determine necessary maintenance. The frequency of inspection and the extent of disassembly may vary depending upon the service conditions for the valve. The staff expects, as a minimum, a commitment from 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. In DCD Tier 2, Section 3.9.6.2.3, "Valve Disassembly and Inspection," the applicant stated that the COL applicant is responsible for developing a program for periodic valve disassembly and inspection. The applicant describes factors to be evaluated for the identification of the valves to be disassembled and inspected and the frequency of the inspections. The applicant stated in DCD Tier 2, Section 3.9.8.4, "Valve Inservice Testing," that the COL applicant will develop a program for valve disassembly and inspection outlined in DCD Tier 2, Section 3.9.6.2.3. This is COL Action Item 3.9.6.4-1. The staff finds this acceptable.

3.9.6.3 Relief Requests

In DCD Tier 2, Section 3.9.6.3, "Relief Requests," the applicant stated that the COL applicant will request relief from the testing requirements of the ASME OM Code when full compliance with the OM Code is not practical. The applicant further states that, in such cases, the COL applicant will provide specific information that identifies (1) the applicable code requirements,

Design of Structures, Components, Equipment, and Systems

(2) justification for the relief request, and (3) the testing method to be used as an alternative. The staff finds that submittal of relief requests by the COL applicant when it establishes the Code edition and addenda to be used for the AP1000 IST program is appropriate. The applicant has not identified any OM Code requirements that are impractical for the AP1000 plant using its baseline Code (1995 edition and 1996 addenda to the OM Code).

3.9.6.4 Valve IST Program

In DCD Tier 2, Table 3.9-16, the applicant submitted valve IST requirements for safety-related valves as related to information needed for design certification. The development of a complete plant-specific IST program falls outside the scope of design certification and remains the responsibility of the COL applicant. However, at the design certification stage, it is necessary to establish a baseline Code edition and addenda to ensure that IST requirements of the baseline Code can be performed without exception, and that the design of the AP1000 systems and components provides access to permit the performance of testing pursuant to 10 CFR 50.55a(f)(3). As a result of its review, the staff finds that the AP1000 has not taken exceptions to any OM Code requirements as established in the 1995 edition and 1996 addenda.

The staff, based on its review of the valve IST program, finds that, at the COL stage, the COL applicant should develop a comprehensive plant-specific IST program which should include the following items:

- tests performed on each component and the Code requirement met by each test
- test parameters and frequency of the tests
- normal, safety, and fail-safe position on each valve
- component type for each component
- P&ID coordinates for each component

In addition, the COL applicant should submit any requests for relief, and the NRC staff will review these requests on the basis of the applicable ASME Code edition and addenda incorporated by reference in 10 CFR 50.55a(b), together with applicable limitations and modifications specified in the regulations at the time a COL might be issued, and the state-of-the-art IST methods available at the time of the COL application. 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 OM Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license, subject to the limitations and modifications listed in the regulations. In DCD Tier 2, Section 3.9.8.4, the applicant stated that the COL applicant will develop an IST program in conformance with the valve IST requirements outlined in DCD Tier 2, Section 3.9.6 and Table 3.9-16. The staff finds that this commitment adequately addresses the need for plants referencing the AP1000 DCD to develop a comprehensive IST program using the ASME OM Code at the COL stage and, therefore, is acceptable. This is COL Action Item 3.9.6.4-1.

3.9.7 Integrated Head Package

DCD Tier 2, Section 3.9.7, describes the integrated head package (IHP). 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 DCD Tier 2, Figure 3.9-7, the applicant provided an illustration of the IHP. The following discussion addresses the structural integrity and deflection limits of the seismic restraints and shroud assembly.

The CRDM seismic restraint structure interfaces with the shroud assembly to transfer seismic loads from the mechanisms to the reactor vessel head. The seismic restraint structure and the shroud are both classified as seismic Category 1. DCD Tier 2, Sections 3.9.7.1, "Design Basis," 3.9.7.3, "Design Evaluation," and Table 3.2-3 (sheet 35) state that both of these items are classified as AP1000 Class C (ASME Code Class 3) and constructed in accordance with the rules in Subsection NF of Section III of the ASME Code.

In DCD Tier 2, Section 3.9.7.3, the applicant stated that, under design-basis loads, the deflection at the top of the CRDM rod travel housing is limited to ensure that the rod travel housing does not bend to the extent that the CRDM drive rod binds during insertion of the control rods. The deflection limits for the IHP are based on limiting deflections of the CRDM housing to ensure rod cluster control assembly insertion following a seismic event combined with the effects of a postulated pipe break. The loads from postulated branch line breaks in reactor coolant loop piping are limited to breaks in 10.16-cm (4-in.) nominal diameter piping as a result of the application of LBB criteria. These pipe break loadings have no adverse effects on the functions of the CRDM and the IHP. Section 3.9.4 of this report discusses the staff's evaluations of the operational adequacy of the CRDM under LOCA and SSE loads.

On the basis of its review of DCD Tier 2, Section 3.9.7, as described above, the staff reached the conclusions about the design of the AP1000 IHP described below.

In accordance with DCD Tier 2, Table 3.2-3, the IHP 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 DCD contains acceptable criteria for the design of the IHP components which provide support for the CRDM and safety-related piping and instrumentation when subjected to ASME Service Level A, B, C, and D loading conditions.

On the basis of this evaluation, the staff concludes that the design of the IHP structures conforms to the following requirements:

•   ASME Code, Section III, Safety Class 3
•   quality assurance requirements of 10 CFR Part 50, Appendix B
•   design rules of Section III, Subsection NF of the ASME Code

On the basis of the above evaluation, the staff concludes that application of the criteria discussed above to the design of the structures of the reactor IHP 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 AP1000 design meets GDC 1 and 10 CFR 50.55a with regard to designing the IHP 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 the IHP structures and components

Design of Structures, Components, Equipment, and Systems

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.

3.9.8   Other Combined License Action Items

The following section of DCD Tier 2 includes a combined license information item in which the staff has determined to be acceptable. These item is repeated below.

- DCD Tier 2, Section 3.9.8.3, "Snubber Operability Testing," states the following:

    Combined License applicants referencing the AP1000 design will develop a program to verify operability of essential snubbers as outlined in [DCD Tier 2, Section] 3.9.3.4.3.

    This is COL Action Item 3.9.8-1.

3.10   Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

DCD Tier 2, Section 3.10, "Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment," provides information on the criteria, procedures, and methods for seismic and dynamic qualification of seismic Category I electrical equipment, instrumentation, and mechanical components (other than piping), including the following equipment types:

- safety-related instrumentation and electrical equipment and certain postaccident monitoring equipment

- safety-related active mechanical equipment that performs a mechanical motion while accomplishing a system safety-related function

- safety-related, nonactive mechanical equipment whose mechanical motion is not required while accomplishing a system safety-related function, but whose structural integrity must be maintained in order to fulfill its design safety-related function

The staff based its review of the acceptance criteria on meeting the following requirements and guidelines:

- 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 S to 10 CFR Part 50, as related to qualifying equipment to withstand the effects of natural phenomena such as earthquakes

- GDC 4, as related to qualifying equipment 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

- SRP 3.10 guidelines for static and dynamic tests and analyses, including complete and auditable records, for confirmation of the operability of mechanical and electrical equipment during and after an earthquake of magnitude up to and including the SSE, and for all static and dynamic loads from normal, transient, and accident conditions.

- SRP 3.9.3 guidelines for loading combinations and methods for combining dynamic responses for equipment subject to seismic qualification.

The seismic qualification methodology described in DCD Tier 2, Section 3.10, will be used for both mechanical and electrical equipment. The applicant designated this information as Tier 2*. This program conforms to IEEE Std 323 and RG 1.89. The program also meets the criteria in IEEE Std 344 as modified by RG 1.100, Revision 2. The applicant's seismic qualification methodology, documented in WCAP-15799, complies with the guidelines of Section 3.10 of the SRP. The following discussion gives the details of the applicant's seismic qualification methodology and any alternatives to staff guidelines.

The seismic qualification criteria presented in DCD Tier 2, Section 3.10.1, "Seismic and Dynamic Qualification Criteria," state that seismic testing is the preferred method for equipment qualification. Both dynamic and static test approaches are used to demonstrate structural integrity and operability of mechanical and electrical equipment. The seismic test environment includes site-specific earthquake loadings equivalent to an SSE preceded by five earthquake cycles of a magnitude equal to 50 percent of the calculated SSE. For seismic Category I instrumentation and electrical equipment, seismic testing is performed in accordance with the guidelines of IEEE Std 344-1987. Where dynamic testing is used, the procedures defined in DCD Tier 2, Appendix 3D, "Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment," develop multifrequency, multiaxis inputs. The test results, documented in the individual equipment qualification data package file, demonstrate that the applied seismic test response spectrum envelops the response spectrum defined in the equipment qualification data package. Alternative methods, such as single-frequency, single-axis test inputs for in-line mounted equipment may be used in selected cases as permitted by IEEE Std 344-1987 and RG 1.100.

Analytical evaluations may also be used for seismic qualification of AP1000 equipment. Analysis using mathematical modeling techniques correlated to tests performed on similar equipment or structures provide a basis for equipment seismic qualification. A combination of testing and analysis can also qualify equipment.

The analytical approach for seismic qualification without testing is used under certain circumstances, including the following:

- if only maintaining structural integrity is required for the safety-related function

Design of Structures, Components, Equipment, and Systems

- if the equipment is too large or heavy to obtain a representative test input at existing test facilities (in these cases, the essential control and electrical devices of large equipment are tested separately)

- if the response of the equipment is essentially linear or the equipment has simple nonlinear behavior predictable by conservative analytical methods

In DCD Tier 2, Section 3.10.2, "Methods and Procedures for Qualifying Electrical Equipment, Instrumentation, and Mechanical Components," the applicant stated 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 Std 344-1987 on a case-by-case basis. As used in IEEE Std 344, experience data include both seismic experience and previous qualifications. Although the staff has not accepted the generic usage of seismic experience data from either evolutionary or passive plants, in accordance with RG 1.100, Revision 2, the COL applicant may use this method of qualification, subject to review by the staff on a case-by-case basis. DCD Tier 2, Section 3.10.2, further states that when using seismic experience data, the COL applicant will properly document all aspects of the methodology, qualification basis, and supporting data. In the equipment qualification file, the COL applicant will include 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. DCD Tier 2, Section 3.10.6, "Combined License Information on Experienced-Based Qualification," provides 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 procedure is consistent with RG 1.100, Revision 2, which, as stated above, conditions its endorsement of IEEE Std 344-1987 by stating that the NRC staff will evaluate the use of experience data for qualification of equipment on a case-by-case basis. The staff, based on the preceding review, finds the use of seismic experience data on a case-by-case basis, subject to NRC review, to be an acceptable alternative method for seismic qualification of equipment. This is COL Action Item 3.10-1. Not withstanding that this matter is labeled as a COL Action Item, it is also Tier 2* information. A COL applicant who wishes to change or depart from this Tier 2* information must seek prior NRC approval in accordance with applicable requirements.

The applicant presents general performance standards and procedures for equipment seismic qualification in DCD Tier 2, Sections 3.10.1 and 3.10.2. An equipment qualification data package (EQDP) is developed for every item of instrumentation and electrical equipment classified as seismic Category I. DCD Tier 2, Appendix 3D, Attachment A, provides the format and content of the individual EQDP. DCD Tier 2, Table 3.11-1, identifies the seismic Category I instrumentation and electrical equipment supplied for the AP1000. Each EQDP contains a section specifying performance requirements. This specification establishes the safety-related functional standards of the equipment to be demonstrated during and after a seismic event. This EQDP performance specification includes the site-specific seismic test response spectrum used for AP1000 equipment seismic qualification. In DCD Tier 2, Section 3.10.2, the applicant also stated that the effects of hydrodynamic and vibratory loads, in addition to seismic loads, will be considered in the qualification of electrical equipment, where applicable.

The EQDP defines the performance standards for active seismic Category I mechanical components in appropriate design and equipment procurement specifications. Active mechanical equipment is qualified for both structural integrity and operability for its intended service conditions by a combination of testing and analysis which, in addition to seismic loading, addresses nonseismic loads, if applicable. Safety-related active valves, listed in DCD Tier 2, Table 3.11-1, undergo a series of tests before operational service. These tests include valve body hydrostatic testing to ASME Code, Section III requirements, seat leakage tests, disc hydrostatic tests, and operational tests to verify that the valve opens and closes as required. The analysis of active valves includes consideration of the effects of seismic excitation of extended structures by application of the static equivalent SSE loadings at the center of gravity of the extended structure and comparison of the resulting stresses to ASME Code Class 1, 2, and 3 stress limits, as applicable. Active valve motor operators, position sensors, and solenoid valves are seismically qualified according to the criteria of IEEE Std 382-1996. Active valve discs are analyzed for maximum design piping pressure and maximum differential pressure resulting from plant operating, transient, and accident conditions. The valve design specification includes valve operation conditions, which are used to evaluate the structural integrity of the valve disk. Feedwater piping valves may be subject to significant impact from faulted condition dynamic loads from a postulated LOCA. Feedwater line valve disks are evaluated, using appropriate ASME Code, Section III limits, for 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.

DCD Tier 2, Section 3.10.2.2, states that the qualification program for valves that are a part of the reactor coolant pressure boundary shall include testing or analysis that demonstrates that these valves will not experience leakage beyond the design criteria when subjected to design-basis loading conditions. DCD Tier 2, Section 3.9.3.2, also contains the following information related to operability qualification for active valves (note that the AP1000 design does not rely upon pumps to perform a safety-related function):

• rationale used to determine if seismic tests, analyses, or combinations of both will be performed

• criteria used to define the seismic and other relevant dynamic load input motions

• performance criteria demonstrating the adequacy of the qualification program

Section 3.9 of this report provides additional staff review of information pertaining to the structural integrity of pressure-retaining components, their supports, and reactor core support structures.

Based on the staff's review of these qualification standards, performance requirements, and procedures for equipment seismic qualification as set forth above, the staff concludes that these commitments are consistent with the guidelines of Section 3.10 of the SRP and, therefore, are acceptable.

The applicant, in DCD Tier 2, Section 3.10.4 and Table 1.8-2, specifies that the COL applicant will establish and maintain the equipment qualification file, including the individual EQDP and

Design of Structures, Components, Equipment, and Systems

seismic test reports, during the equipment selection and procurement phase. On the basis of its review of DCD Tier 2, Sections 3.10.4 and 3.10.6, the staff finds that the applicant's program for seismic qualification of equipment described in DCD Tier 2, Section 3.10, provides qualification criteria and methodology for design certification only. The section contains no plant-specific information. Therefore, each COL applicant using this methodology should ensure that specific environmental parameters, along with seismic and dynamic input response spectra, are properly defined and included in the methodology for its specific plant and implemented in its equipment qualification program. In DCD Tier 2, Section 3.10.4, the applicant committed that the COL applicant will maintain equipment qualification records in a permanent file readily available for staff audit. This is COL Action Item 3.11.2-1.

The staff concludes that the above commitments satisfy applicable portions of Section 3.10 of the SRP and are therefore acceptable. DCD Tier 2, Sections 3.10.4 and 3.10.6 add commitments that the COL applicant is responsible for maintaining the equipment qualification file, including the individual EQDP for seismic Category I electrical and mechanical equipment, during the equipment selection and procurement stage. As stated above, this is consistent with Section 3.10 of the SRP and is acceptable.

3.10.1 Conclusions

On the basis of its review of DCD Tier 2, Sections 3.10, 3.9.2.2, and 3.9.3.2, WCAP-15799, and the complementary information in DCD Tier 2, Sections 3D.4.1.2, 3D.6.2, 3D.7.1, and 3D Attachments A, E.3.2, E.4.3, and E.5, the staff concludes that these provisions collectively provide commitments regarding the operability and seismic qualification of electrical and mechanical equipment consistent with the guidelines of Section 3.10 of the SRP and, therefore, are acceptable.

On the basis of the above evaluation, the staff concludes that the applicant has defined appropriate criteria and methodology for a seismic and dynamic qualification program for mechanical and electrical equipment that meets the guidelines in SRP Section 3.10. This program also meets applicable portions of GDC 1, 2, 4, 14, and 30 and Appendix B to 10 CFR Part 50, and, therefore, are acceptable. This conclusion is based on the information presented below.

In DCD Tier 2, Table 3.2-3, the applicant identified all AP1000 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 DCD Tier 2, Table 3.2-3, is acceptable. On the basis of its evaluations, the staff concludes that the criteria and commitments in the DCD meet GDC 1 and 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 staff finds that the criteria and commitments in DCD Tier 2, Section 3.10 meet GDC 4 for this seismic qualification program. The staff evaluations include analysis of the protection from external missiles and internally-generated missiles (Section 3.5 of this report) and review of analyses of the dynamic effects of postulated pipe breaks (Section 3.6.2 of this report).

The criteria and commitments for the seismic qualification program meet GDC 2 and 14, as they relate to qualifying equipment to withstand the effects of natural phenomena such as earthquakes, and demonstrate that equipment associated with the reactor coolant pressure boundary has a low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

The staff concludes that the AP1000 seismic qualification program, which will be implemented for mechanical, instrumentation, and electrical equipment, is consistent with the recommendations of IEEE Std 344-1987, the regulatory positions of RGs 1.61, 1.89, 1.92, and 1.100, and Section 3.9.3 of the SRP. The staff, based on its review of the information in DCD Tier 2, Section 3.10 and other related sections discussed above, concludes that the AP1000 design provides adequate assurance that AP1000 seismic Category I equipment will function properly under the effects of design and service loads, including the loadings imposed by the SSE, postulated accidents, and LOCAs.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

3.11.1 Introduction

Equipment 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 the equipment's installed life for the time it is required to operate. This requirement, embodied in GDC 1 and 4 and Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50, applies 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 Std 323; and various RGs and industry standards, including RG 1.89, Revision 1.

3.11.2 Regulatory Evaluation

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 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 appropriate methods for qualifying equipment in different areas of the plant
- margin, aging, and documentation

A final rule on environmental qualification 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

Design of Structures, Components, Equipment, and Systems

in harsh environments. Each item of electrical 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 analysis 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. In accordance with 10 CFR 50.49(d), the COL applicant shall prepare a list of electrical equipment important to safety covered by the qualification requirements. In addition, the COL applicant shall include the following information for electrical equipment important to safety in a qualification file:

(1)   the performance specifications under conditions existing during and following DBAs

(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

Pursuant to 10 CFR 50.49(j), 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 electrical equipment important to safety (1) is qualified for its application, and (2) meets is specified performance requirements. To conform with 10 CFR 50.49, electrical equipment for PWRs referencing the AP1000 design should be qualified according to the criteria in Category I of NUREG-0588 and Revision 1 of RG 1.89. This is COL Action Item 3.11.2-1.

Appendices A and B to 10 CFR Part 50 contain the principal qualification requirements for mechanical equipment. The qualification methods defined in NUREG-0588 also apply to mechanical equipment.

DCD Tier 2, Section 3.11, and Appendix 3D document the degree to which the EQ program for the AP1000 design complies with the EQ requirements and criteria.

3.11.3 Technical Evaluation

The staff limited its evaluation of the EQ program for the AP1000 design to a review of the applicant's submittals on its approach to selecting and identifying equipment required to be environmentally qualified for the AP1000 design, qualification methods proposed, and completeness of information in DCD Tier 2, Appendix 3D. Guidance for the staff's evaluation appears 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 AP1000 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 to ensure that certain functions are accomplished)

- (b)(2)—non-safety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory performance of the safety functions of 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 DCD Tier 2, Table 3.11-1, the applicant provided 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 that list and found it reasonably complete. Accordingly, the staff concludes that it is acceptable.

In DCD Tier 2, Appendix 3D, the applicant did not include Figures 3D.5.6 and 3D.5.7, "Containment Temperature Design Conditions: LOCA," and "Containment Temperature Design Conditions: Steam Line and Feedwater Line Breaks," respectively. However, the applicant agreed to include these figures in a subsequent revision to the DCD. Consequently, this was Confirmatory Item 3.11.3-1 in the DSER.

The NRC staff reviewed the applicant's revision to DCD Tier 2, Appendix 3D, and determined that this appendix now includes Figures 3D.5.6 and 3D.5.7. Therefore, Confirmatory Item 3.11.3-1 is resolved.

The radiation qualification for individual safety-related components should be developed on the basis of the following two conditions:

Design of Structures, Components, Equipment, and Systems

(1)   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

(2)   the limiting DBA for which the component provides a safety function

Chapter 15 of this report discusses these DBA conditions.

For the LOCA source term, the AP1000 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

3.11.3.2.1 Electrical Equipment in a Harsh Environment

NUREG-0588 and RG 1.89 define detailed procedures for qualifying safety-related electrical equipment located in a harsh environment. The criteria in these documents also apply to other equipment important to safety defined in 10 CFR 50.49.

In reviewing the DCD, the staff determined that the methodology used by the applicant for the AP1000 relied primarily on IEEE Std 323-1974. As indicated in the footnote to 10 CFR 50.49 and stated in NUREG-0588 and RG 1.89, the guidance in IEEE Std 323-1974 is acceptable to the NRC staff for qualifying equipment within the scope of 10 CFR 50.49.

In addition, for current-generation operating reactors, the staff's definition of a mild radiation environment for electronic components, such as semiconductors, or any electronic component containing organic materials, differs from the definition of a mild radiation environment for other equipment. The staff defines a mild radiation environment for such electronic equipment as a total integrated dose of less than 10 gray (Gy) ($1E^3$ rad). For other equipment, it is less than $10E^2$ Gy ($1E^4$ Rad). With the expected significant increase in the quantity and variety of electronic components in newer generation plants, the staff has increasing concern about the ability of these components to be environmentally qualified.

In DCD Tier 2, Appendix 3D.4.3, the applicant discussed mild versus harsh environments. In this discussion, the applicant states the following:

> 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 [$1E^2$ Gy ($1E^4$ rads)] gamma [10 Gy ($1E^3$ rads)] 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 its own position and finds it acceptable.

3.11.3.2.2 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 of 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 have the following objectives:

• Identify safety-related mechanical equipment located in harsh environment areas, including required operating time.

• Identify nonmetallic subcomponents of this equipment.

• Identify the environmental conditions for which this equipment must be qualified. (Mechanical equipment will experience the same environmental conditions as those defined in 10 CFR 50.49 for electrical equipment, and such conditions should be used in qualifying mechanical equipment.)

• Identify nonmetallic material capabilities.

• Evaluate environmental effects.

DCD Tier 2, Table 3.11-1, includes both electrical and mechanical equipment, with a clear distinction between the two classes of equipment. The applicant clearly identified which items of equipment are classified as electrical and separated them from those that are classified as mechanical. The staff finds this acceptable.

3.11.3.3 Conclusions

On the basis of its review of the DCD Tier 2 and the NRC staff policies and practices, and for the reasons set forth above, the staff concludes that the program proposed by the applicant for environmentally qualifying electrical equipment important to safety and safety-related mechanical equipment for the AP1000 design complies with the requirements of 10 CFR 50.49

Design of Structures, Components, Equipment, and Systems

and other relevant requirements and criteria 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 the applicant's design of piping systems for the AP1000 design certification, which comprise the seismic Category I, Category II, and non-safety systems piping. The staff used the SRP Section 3.9.3 guidelines to evaluate the piping design information in the DCD. The staff's evaluation considered the adequacy of the structural integrity as well as the functional capability of piping systems. The review was not limited to only 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 nonseismic Category I piping with seismic Category I piping.

After completing its initial review of the piping design information in the DCD, the staff sent the applicant several RAIs. In a public meeting between the staff and the applicant on July 17, 2002, the applicant indicated that although it had not performed piping and pipe support design and analysis and might not complete such an analysis as part of the design certification, preliminary piping layout drawings were available. In RAI 210.033, the staff asked the applicant to provide the piping layout drawings for review. The NRC staff and the applicant met at Westinghouse offices on September 9–11, 2002, to discuss the staff RAIs and to review additional supporting information. During this meeting, the staff reviewed sample layout drawings and noted the similarities between the AP600 and AP1000 piping designs. In RAI 210.032, the staff indicated that the applicant did not provide any information in the DCD that described the completion status of the piping design or its plans for using design acceptance criteria (DAC) for piping design. In response to RAI 210.032, the applicant stated that it would revise DCD Introduction Table 1-1, "Index of AP1000 Tier 2, Information Requiring NRC Approval for Change," to include piping DAC. In addition, the applicant stated that it would include a new DCD Introduction Table 1-2, "Piping Design Acceptance Criteria." This was submitted in a revision to the DCD, dated February 6, 2003. This table lists the proposed piping DAC commitments and identifies the DCD Tier 2 sections which describe them. The staff has verified that the applicant added DCD Introduction, Tables 1-1 and 1-2. The staff finds that the new information in the DCD adequately describes the use of piping DAC for AP1000 and is thus acceptable. This section describes the staff evaluation of the substance of the DAC.

The AP1000 piping design analysis methods, design procedures, and acceptance criteria that are to be used for completion of the AP1000 piping design comprise the piping DAC. The staff's evaluation in this section addresses the following areas of piping DAC:

- applicable codes and standards
- analysis methods to be used for completing the piping design
- modeling techniques
- pipe stress analyses criteria
- pipe support design criteria

Sections 3.6.2 and 3.6.3 of this report address the staff's evaluation of the piping design areas involving high- and moderate-energy line break analyses and LBB evaluation, respectively. The applicant completes these areas of piping design to a preliminary stage, and the COL applicant completes the final design.

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. The regulation specifies the use of the latest edition and addenda endorsed by the NRC and any limitations discussed in the regulations. In RG 1.84 the staff lists acceptable ASME Code cases for design and materials acceptability (Section III) and any conditions that apply to them.

3.12.2.1 ASME Boiler and Pressure Vessel Code

For the AP1000 design certification, the applicant 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. DCD Tier 2, Section 5.2.1.1, "Compliance with 10 CFR 50.55a," specifies the 1998 edition, up to and including the 2000 addenda, of the ASME Code for the AP1000 design. However, for piping design, the applicant indicated that it used the 1989 edition, including the 1989 addenda, for Articles NB-3200, NB-3600, NC-3600, and ND-3600 in lieu of later editions and addenda. Section 5.2.1.1 of this report discusses the staff's evaluation of the ASME Code edition and addenda and the process for changing ASME Code editions and addenda. For the reasons set forth in that section, the staff concludes that the ASME Code Class 1, 2, and 3 piping will conform to the appropriate ASME Code editions and addenda and the Commission's regulations.

3.12.2.2 ASME Code Cases

The only ASME Code cases that may be used for the design of ASME Code Class 1, 2, and 3 piping systems in the AP1000 design are those either conditionally or unconditionally approved in RG 1.84 in effect at the time of design certification. These cases are listed below. However, the COL applicant may submit with its COL application for staff review and approval future Code cases that are endorsed in RG 1.84 at the time of the COL application, provided the cases do not alter the staff's safety findings on the AP1000 certified design.

DCD Tier 2, Table 5.2-3, lists all ASME Code cases to be used in the AP1000 design. In RAI 210.030, the staff requested the applicant to identify the applicable Code cases that will be used in the design and analysis of piping systems, including piping components and associated supports. The applicant responded that the following ASME Code cases apply to the AP1000 piping and support design:

Design of Structures, Components, Equipment, and Systems

- ASME Code Case N-122-2, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 1 Piping, Section III, Division 1"

- ASME Code Case N-318-5, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1"

- ASME Code Case N-319-3, "Alternate Procedure for Evaluation of Stresses in Butt Welding Elbows in Class 1 Piping, Section III, Division 1"

- ASME Code Case N-391-2, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1"

- ASME Code Case N-392-3, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping, Section III, Division 1"

In Revision 32 of RG 1.84 dated June 2003, the staff endorsed ASME Code Cases N-122-2 and N-318-5, as well as N-391-2 and N-392-3. On this basis the staff finds that the ASME Code cases proposed by the applicant for the AP1000 piping design are acceptable (see also Section 5.2.1.2 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 inputs. The Code also requires a design report for ASME Code Class 1, 2, and 3 piping and components.

DCD Tier 2, Section 3.9.3, states that the COL applicant or its agent will complete the design specifications and design reports. It also states that design specifications for ASME Class 1, 2, and 3 components and piping are prepared utilizing procedures that meet the ASME Code. In RAI 210.031, the staff requested the applicant to provide these procedures for staff review and identify any differences between the AP600 and AP1000 procedures. In its response, the applicant indicated that the requirements and procedures for preparation of design specifications for ASME Class 1, 2, and 3 components and piping are the same for both the AP600 and the AP1000. The staff received the procedures for review at a meeting held on September 9–11, 2002, at the Westinghouse office. The staff's review of the procedures finds that they will result in AP1000 design specifications that will comply with ASME Code, Section III, and, therefore, are acceptable.

3.12.2.4 Conclusions

The staff concludes that in DCD Tier 2, Sections 3.9.3, 5.2.1.1, and 5.2.1.2, the applicant 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 its importance to safety.

The staff reviewed the information in DCD Tier 2, Sections 3.7.3, 3.9.1, and 3.9.3, 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 components as discussed in DCD Tier 2, Section 3.9.1.1. The staff also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports.

As indicated in DCD Tier 2, Section 3.7.3.1, the methods used for seismic analysis of subsystems include modal response spectrum analysis, time-history analysis, and equivalent static analysis. The designer selects a particular method, based on its appropriateness for the specific piping system. The following sections discuss the staff's evaluation of the methods applied to piping analysis.

3.12.3 Piping Analysis Methods

The staff based its review of the piping DAC, in part, on SRP Section 3.7.3. SRP Section 3.7.3 addresses seismic subsystem analysis related to seismic analysis methods, number of earthquake cycles, procedures used for analytical modeling, selection of frequencies, damping criteria, combination of modal responses, equivalent static factors, and interaction of Category I systems with other systems. However, because the piping DAC addresses many technical issues other than seismic issues and those criteria in the SRP, the staff also based its review on common industry practices and practical engineering considerations proven through extensive experience.

3.12.3.1 <u>Experimental Stress Analysis</u>

In DCD Tier 2, Section 3.9.1.3, the applicant stated that, except for the validation of reactor internals vibration analysis by prototype and scale model testing, the AP1000 piping design will use no other experimental stress analysis. If a COL applicant wishes to use this method in any AP1000 piping design, the applicant must submit the details of the method, as well as the scope and extent of its application, to the staff for approval before its use. The staff's position is that experimental stress analysis methods shall comply with Appendix II to ASME Section III, Division 1. Section 3.9.2.3 of this report discusses the acceptability of experimental stress analysis for reactor internals vibration analysis.

3.12.3.2 <u>Modal Response Spectrum Method</u>

SRP Section 3.7.3 provides acceptance criteria for combining three components of earthquake motion for the response spectra method and for combining modal responses.

DCD Tier 2, Section 3.7.3.1, describes modal response spectrum analysis methods. The applicant specifies both the envelope and independent support motion response spectrum methods as modal response spectrum analysis options. In either method, a mathematical model is first constructed. DCD Tier 2, Section 3.7.3.8, "Analytical Procedure for Piping," describes in detail the development of the mathematical model to reflect the dynamic characteristics of the piping system.

Design of Structures, Components, Equipment, and Systems

DCD Tier 2, Section 3.7.3.9, "Combination of Support Responses," describes the analysis procedure for the enveloped response spectrum analysis method. 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 cutoff 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 analyst calculates the modal inertia response forces, moments, displacements, and accelerations. For each direction, the modal responses are combined in accordance with one of the procedures described in DCD Tier 2, Section 3.7.3.7.2, "Combination of Low-Frequency Modes." The high-frequency mode responses are determined and combined with the low-frequency mode responses through one of the methods described in DCD Tier 2, Section 3.7.3.7.1, "Combination of High-Frequency Modes." The SRSS method combines the total seismic responses for all three earthquake directions. DCD Tier 2, Section 3.7.3.9, describes the use of static analysis to calculate the response resulting from differential seismic anchor motions. The results of the seismic inertia analysis are combined with the results of the seismic anchor motion analysis by the absolute sum method. These methods and criteria are consistent with those specified in SRP Section 3.7.3. Therefore, the staff finds the enveloped response spectrum methodology described in the DCD to be acceptable. 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 provide detailed descriptions and staff evaluations of specific elements of the analytical procedure.

3.12.3.3 Independent Support Motion Method

DCD Tier 2, Section 3.7.3.9, describes the independent support motion response spectrum method. The applicant stated that this method may be used when there is more than one supporting structure. When this methodology is applied, 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. This approach results in a conservative, but more precise method than the response spectra method, and is consistent with standard industry practice. DCD Tier 2, Section 3.7.3.15, "Analysis Procedure for Damping," and Table 3.7.1-1, give the damping criteria. For independent support motion analysis, piping systems less than or equal to 30.5 cm (12 in.) in diameter use 2-percent damping, piping systems greater than 30.5 cm (12 in.) in diameter use 3-percent damping, and the primary coolant loop piping uses 4-percent damping. This method is consistent with RG 1.61 and staff-approved WCAP-7921-AR, and, therefore, is acceptable.

3.12.3.4 Time-History Method

In DCD Tier 2, Section 3.7.3.17, "Time-History Analysis of Piping Systems," the applicant stated that time-history dynamic analysis is an alternative seismic analysis method that may be used with time-history seismic input. It may also be used for dynamic analyses of piping systems subjected to hydraulic transient loadings or forcing functions induced by postulated pipe breaks. Direct integration or the modal superposition method is used to solve the equations of motion.

The AP1000 piping analyses use the computer programs GAPPIPE, PIPESTRESS, ANSYS, and WECAN.

The modal superposition method is based on the equations of motion which can be decoupled as long as the piping system is within its elastic limit. The modal responses are obtained from integrating the decoupled equations. The total responses are obtained by the algebraic sum of the modal responses at each time step. The frequency content of the input forcing function and the highest significant frequency of the piping system provide the basis for selecting the cutoff frequency. The integration time step is no larger than 10 percent of the period of the cutoff frequency.

In RAI 210.043, the staff noted that the DCD does not include any description of the direct integration time-history analysis methodology. The applicant responded that all time-history analyses performed for AP600 piping systems utilized modal superposition methods (not direct integration methods). The analysis of AP1000 piping systems will use this same approach.

In DCD Tier 2, Section 3.7.3.6, the applicant stated that one set of three mutually orthogonal artificial time histories is used in time-history analyses. The components of earthquake motion specified in the three directions are statistically independent and applied simultaneously. In this method, the responses from each of the three components of motion are combined algebraically at each time step. The staff finds that this approach meets the guidelines of SRP Section 3.7.3, which recommends that the responses from each of the three components of earthquakes be combined algebraically at each time step. Therefore, this approach is acceptable.

In DCD Tier 2, Section 3.7.3.6, the applicant also stated that, as an alternative, the time-history seismic analysis of a subsystem may be performed by simultaneously applying the displacements and rotations at the interface points between the subsystem and system. These displacements and rotations are obtained from a model of a larger subsystem or a system that includes a simplified representation of the subsystem. The time-history SSE analysis of the system is performed by applying three mutually orthogonal and statistically independent artificial time histories.

In DCD Tier 2, Section 3.7.3.17, the applicant discussed the methodology for consideration of high-frequency modes in a modal superposition time-history analysis. When the PIPESTRESS program is used, the response from high-frequency modes above the cutoff frequency is calculated based on the static response to the left-out-forces. The response is combined with the response from the low-frequency modes by algebraic sum at each time step.
Section 3.12.5.6 of this report discusses the staff evaluation of the left-out-force method. As an alternative, the number of modes used in the modal analysis is selected to account for the principal vibration modes of the piping system, and the total number of modes used is chosen so that the results of the analysis using the chosen number of modes are within 10 percent of the results of an analysis based on the next higher number of modes. The number of modes analyzed is selected to account for the principal vibration modes of the piping system. The modes are combined by algebraic sum. This approach is consistent with the regulatory position stated in SRP Section 3.7.3 and is, therefore, acceptable.

In DCD Tier 2, Section 3.7.3.17, the applicant indicated that the PIPESTRESS, ANSYS, or WECAN programs use composite modal damping. DCD Tier 2, Table 3.7.1-1, lists the damping

Design of Structures, Components, Equipment, and Systems

values for individual components. For piping, this is 2 percent for diameters less than or equal to 30.5 cm (12 in.), 3 percent for diameters greater than 30.5 cm (12 in.), and 4 percent for the primary coolant loop. The damping values are acceptable as they are consistent with the guidelines of RG 1.61 and the staff-approved WCAP-7921-AR.

In DCD Tier 2, Section 3.7.3.17, the applicant also discussed how a time-history analysis accounts for modeling uncertainties. For the AP1000, the seismic ground motion is based on a hard rock site. For dynamic time-history analysis of piping systems, including seismic analysis at a hard rock site, three separate analyses are performed for each loading case. The three analyses correspond to three different time scales—normal time, time expanded by 15 percent, and time compressed by 15 percent. Alternatively, when the results prove acceptable based on comparison with test data, the performance of one time-history analysis uses normal time. For time-history analysis of piping system models that include a dynamic model of the supporting concrete building, either the building stiffness varies by + or - 30 percent, or the time scale shifts by + or - 15 percent. Alternately, when uniform enveloping time-history analysis is performed, the spreading included in the broadened response spectra accounts for modeling uncertainties. In RAI 210.042, the staff requested the applicant to clarify and justify the types of loadings and conditions for which a single time-history analysis using normal time would be considered acceptable. In addition, the staff noted that the references to time scale variations of "+ or - 15 percent" and stiffness variations of "+ or - 30 percent" should be corrected to "+ and - 15 percent" and to "+ and - 30 percent," respectively, since both variations should be analyzed. The applicant acknowledged its agreement with this position. DCD Tier 2, Section 3.7.3.17, was revised to reflect this position, and since the seismic criteria do not provide for the alternative method utilizing the test data, the reference to test data was also removed. The staff finds this revision acceptable.

3.12.3.5 Inelastic Analysis Method

In DCD Tier 2, Section 3.9.3.1.5, the applicant stated that inelastic analysis methods are not used. If a COL applicant wishes to use this method in any AP1000 piping design, the COL applicant should identify the specific systems, provide a detailed description of the methodology, and provide acceptance criteria consistent with the guidelines of Section 3.9.1 of the SRP. The COL applicant should submit this information to the staff for review and approval before using the method.

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 DCD Tier 2, Section 3.7.3.8.2.2, "Small-Diameter Auxiliary Piping," the applicant specified two analysis options for the seismic analysis of these systems. The options include the response spectrum method and the equivalent static load method. The applicant stated that the COL applicant will complete the final design of the small-bore piping and address the as-built reconciliation in accordance with the criteria provided in DCD Tier 2, Sections 3.9.3 and 3.9.8.2.

The response spectrum method is an acceptable seismic analysis methodology for the analysis of both small and large bore piping. Section 3.12.3.2 of this report presents staff comments regarding the use of the response spectrum in the AP1000.

In DCD Tier 2, Section 3.7.3.5, the applicant describes the equivalent static load method. In this 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. DCD Tier 2, Section 3.7.3.5.1, provides procedures for defining equivalent static loads for rigid systems or for cases where the response can be classified as single-mode dominant including pipelines that are modeled on a span-by-span basis. For a piping system with a single-mode dominant dynamic response, the equivalent static load for the direction of excitation is the product of the piping mass and the seismic acceleration value at the component natural frequency from the applicable floor response spectrum times a factor of 1.5. If the frequency is not determined, the peak acceleration from the floor spectrum is used. DCD Tier 2, Section 3.7.3.5.2, "Multiple Mode Dominant Response," provides the procedure for defining equivalent static loads for systems with multiple-mode response. This includes piping systems that are multiple-span models. For these systems, a static load factor of 1.5 is applied to the peak accelerations of the applicable floor response spectra. For runs with axial supports which are rigid in the axial direction (fundamental frequency $\geq$ 33 Hz), the acceleration value of the mass of piping in its axial direction may be reduced to 1.0 times its calculated spectral acceleration value. The spectral acceleration value is based on the frequency of the piping system along the axial direction. For all cases, the loads, stresses, or deflections obtained using the equivalent static load method are adjusted to account for relative motion between points of support when significant. The staff finds that these static load factors meet the equivalent static load factor of 1.5, as recommended in Section 3.9.2.II.2.a(2)(c) of the SRP, and, therefore, are acceptable.

In RAI 210.039, the staff requested the applicant to provide the small-bore piping design and analysis procedures and criteria for staff review. In its response, the applicant indicated that the AP1000 piping analysis design criteria document applies to both large- and small-bore piping. The staff reviewed the document during the September 9–11, 2002, meeting held at the Westinghouse office. The staff found that the document provides an appropriate method to design small-bore piping using the same analysis method as for large-bore piping designs. Thus, the staff finds the small-bore piping design and analysis procedures acceptable for the AP1000.

3.12.3.7 Nonseismic/Seismic Interaction (II/I)

The applicant provided the criteria for protection against nonseismic/seismic interaction in DCD Tier 2, Section 3.7.3.13. Separation (with physical barriers) or segregation (by routing) of seismic Category I piping from nonseismic SSCs is the preferred method of eliminating the possibility of seismic interaction. As an alternative, an impact analysis may be performed to demonstrate that a potential nonseismic SSCs 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 ensure that the SSE will not cause unacceptable structural failure of or interaction with seismic Category I piping. DCD Tier 2, Sections 3.7.3.13.1, "Separation and Segregation," 3.7.3.13.2, "Impact Analysis," 3.7.3.13.3, "Seismic Category II Supports," and 3.7.3.13.4, "Interaction of Piping with Seismic Category I Piping Systems, Structures, and

Design of Structures, Components, Equipment, and Systems

Components," describe the design criteria and guidelines for performing the review for seismic interactions. The 3D computer model and composites developed for the NI are used during the design process of the systems and components in the NI, to aid in evaluating and documenting the review. The applicant stated that the COL applicant will update the seismic interaction review. That review will be based on as-procured data, as well as the as-constructed condition. This is COL Action Item 3.7.5-3. The staff finds that this approach is consistent with the staff guidelines for addressing interaction of other piping with Category I piping provided in Section 3.9.2.II.2.k of the SRP and is, therefore, acceptable.

For nonseismic Category I piping systems attached to seismic Category I piping systems, SRP Section 3.9.2.II.2.k states that the dynamic effects of the nonseismic Category I systems should be considered in the analysis of the seismic Category I piping. In addition, SRP Section 3.9.2.II.2.k states that the nonseismic 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. DCD Tier 2, Section 3.7.3.13.4.2, "Seismic Category II Piping," describes the methods and criteria for piping that is connected to seismic Category I piping. The applicant stated that the interaction of seismic Category I piping with connected nonseismic piping 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 complete run of the nonseismic Category I system. The additional length considered, which is classified as seismic Category II, terminates at either (1) the first anchor beyond the interface, (2) the interface rigid support (last seismic support) which follows a six-way anchor, or (3) the last seismic support of a rigidly supported region of the piping system. The rigid region is typically defined as either four bilateral supports around an elbow or six bilateral supports around a tee. The frequency of the piping system in the rigid region is greater than or equal to 33 Hz. The seismic Category II portion of the line is analyzed according to equation 9 of the ASME Code, Section III, Class 3, with a stress limit equal to the smaller of 4.5 S_h and 3.0 S_y. Although these stress limits exceed those of the ASME Code for Class 3 piping and may not ensure the functional capability of the piping, the staff finds that the functional capability of Category II piping is not needed. The stress limits provide reasonable confidence that the Category II piping will not collapse under seismic loadings in a manner that would adversely affect safety-related equipment below it. The anchor or seismic Category I supports are designed for loads from the nonseismic piping. This includes three plastic moment components in each of the three local coordinate directions. 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 staff finds this approach acceptable, as it is consistent with the acceptance criteria of SRP Section 3.9.2.II.2.k.

3.12.3.8 Seismic Category I Buried Piping

In DCD Tier 2, Section 3.7.3.12, the applicant stated that there are no seismic Category I buried piping systems or tunnels in the AP1000 design.

3.12.3.9 ASME Code, Section III, Appendix N

The NRC has not endorsed the use of Appendix N to Section III of the ASME Code. The applicant has not referenced Appendix N as the basis for the AP1000 piping design methodology.

3.12.3.10 Conclusions

On the basis of the above evaluation, the staff concludes that the analysis methods to be used for all seismic Category I piping systems, as well as nonseismic 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, are consistent with guidelines contained in the SRP, 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 Modeling Techniques

The staff has evaluated the piping methodology used in the design of the AP1000, as presented in the DCD, as described below.

3.12.4.1 Computer Codes

DCD Tier 2, Section 3.9.1.2, provides information on the computer programs used in the AP1000 analysis. DCD Tier 2, Table 3.9-15, lists a number of computer programs used in the dynamic and static analysis of mechanical loads, stresses, and deformations, and in the hydraulic transient load analyses, of seismic Category 1 components and supports. For piping design, the COL applicant will implement the NRC benchmark program, using problems specific to the AP1000, if it uses a piping design program other than those employed in design certification (PIPESTRESS, GAPPIPE, WECAN, and ANSYS). Appendix B to 10 CFR Part 50 requires design control measures to verify the adequacy of the design of safety-related components. SRP Section 3.9.1.II.5 provides acceptance criteria that are based on meeting the relevant requirements of 10 CFR Part 50, Appendix B, as it relates to design quality control including the use of analytical calculations, tests, or benchmark problems. In DCD Tier 2, Section 3.9.1.2, the applicant stated that the development process, verification, configuration control, error reporting, and resolution for computer programs used in these analyses for the AP1000 are completed in compliance with an established quality assurance program described in DCD Tier 2, Chapter 17. The verification conforms to at least one of the following methods:

• hand calculations
• alternative verified calculational methods
• results of other verified programs
• results obtained from experiments and tests
• known solutions for similar or standard problems
• measured and documented plant data
• confirmed published data and correlations
• results of standard programs and benchmarks
• parametric sensitivity analysis
• reference to a verification and validation that have been reviewed and accepted by an independent third party

The staff finds that the above verification methods are consistent with the methods recommended in Section 3.9.1 of the SRP and are acceptable. Section 3.12.4.3 of this report presents the staff evaluation of the COL applicant commitment on the implementation of the NRC benchmark program.

Design of Structures, Components, Equipment, and Systems

In RAI 210.045, the staff requested the applicant to provide additional clarifying information on the computer programs that will be used in AP1000 piping design. DCD Tier 2, Section 3.9.1.2, identified four programs that could be used for piping design—PIPESTRESS, GAPPIPE, WECAN, and ANSYS. The staff asked the applicant to clarify the specific application for each program. The applicant responded that the analysis codes WECAN and ANSYS are used to perform analysis of the reactor coolant loop piping and Class 1 components. GAPPIPE is used for applications to replace snubbers with limit stops, as needed, if it is determined that snubbers are not desirable for the plant. PIPESTRESS will be used for initial qualification of all auxiliary lines. The staff had previously reviewed and accepted the WECAN, ANSYS, and GAPPIPE programs for other Westinghouse plants, including the AP600, and the staff considers these programs appropriate and adequate for the piping systems in the AP1000 plant. These programs were approved using benchmarking. However, the staff noted that for the AP600, the applicant used the PS+CAEPIPE program for auxiliary lines instead of PIPESTRESS. The staff also noted that DCD Tier 2, Table 3.9-15, listed PS+CAEPIPE and CAEPIPE as piping analysis programs but did not list PIPESTRESS.

As part of RAI 210.045, the staff requested the applicant to clarify this discrepancy. In its response, the applicant stated that PS+CAEPIPE refers to a family of computer codes that was provided to the applicant by SST Systems, Inc. The "PS" stands for PIPESTRESS, which is the main processor for this family of codes and performs the applicable stress analyses. PIPESTRESS is a nuclear quality-assured computer program provided by DST Computer Services and distributed by SST Systems. The PC-based program CAEPIPE was not qualified in accordance with WCAP-8370, "Westinghouse Energy Systems Business Unit/Power Generation Business Unit Quality Assurance Plan." The applicant did not use CAEPIPE for any piping analysis work associated with the AP1000. The applicant does not currently have a license agreement for PIPESTRESS with SST Systems. DST Computer Services now distributes the PIPESTRESS program directly. The applicant agreed that the information in DCD Tier 2, Table 3.9-15, should be corrected, and stated that it would remove any reference to the CAEPIPE program from the DCD and change any reference to the PS+CAEPIPE program to PIPESTRESS. Based on this explanation, the staff understands that PIPESTRESS is the same computer program as PS+CAEPIPE, which was used for the AP600 piping analysis. On the basis that the PIPESTRESS program has been appropriately verified and benchmarked, as well as, the corrections the applicant made to DCD Tier 2, Table 3.9-15 and Sections 3.9.1.2 and 3.9.8.6, and the addition of Reference 20 of the DCD, the staff finds the use of the PIPESTRESS program acceptable for AP1000 piping design. Section 3.12.4.3 of this report provides additional information and staff evaluation of the verification and benchmarking of the PIPESTRESS program.

3.12.4.2 Dynamic Piping Model

DCD Tier 2, Section 3.7.3.8, describes the modeling methods used to develop an idealized mathematical model of a piping system for dynamic analysis. The piping system is modeled as beam elements with lumped masses connected by a network of elastic springs representing the stiffness properties of the piping system. Concentrated weights such as valves or flanges are also modeled as lumped masses. The dynamic piping model accounts for the effects of torsion (including eccentric masses), bending, shear, and axial deformations, and effects due to the changes in stiffness values of curved members.

The lump masses are selected so that the maximum spacing does not exceed the length that would produce a natural frequency equal to the lowest zero period acceleration (ZPA) frequency of the seismic input when calculated based on a simply supported beam. As a minimum, the number of degrees of freedom is equal to twice the number of modes with frequencies less than the ZPA frequency.

The piping system analysis model includes the effect of piping support mass when the contributory mass of the support exceeds 10 percent of the total mass of the affected piping spans. The contributory mass of the support is the portion of the support mass that is attached to the piping, such as clamps, bolts, trunnions, struts, and snubbers. Supports that are not directly attached to the piping, such as box frames, need not be considered for mass effects. The mass of the applicable support will not affect the response of the system in the supported direction; therefore, only the model needs to consider the unsupported direction. Based on this reasoning, the mass of full anchors can be neglected. The total mass of each affected piping span includes the mass of the piping, fluid contents, insulation, and any concentrated masses (for example, valves or flanges) between the adjacent supports in each unrestrained direction on both sides of the applicable support. For example, the contributory mass of an X direction support must be compared to the mass of the piping spans in the unrestrained Y and Z directions. A contributory support mass that is less than 10 percent of the masses of the affected spans will have insignificant effects on the response of the piping system and can be neglected.

The stiffness matrix of the piping system is calculated based on the stiffness values of the pipe elements and support elements. DCD Tier 2, Sections 3.9.3.1.5, "ASME Classes 1, 2, and 3 Piping," and 3.9.3.4, "Component and Piping Supports," describe the use of minimum rigid or calculated support stiffness values. When the support deflections are limited to 0.318 cm (0.125 in.) in the combined faulted condition, minimum rigid support stiffness values are used. If the combined faulted condition deflection for any support exceeds 0.318 cm (0.125 in.), the piping system uses calculated support stiffness values.

Valves, equipment, and piping modules are considered to be rigid if the natural frequencies are greater than 33 Hz. The piping system model includes valves with lower frequencies. DCD Tier 2, Sections 3.7.3.8.2.1, "Large Diameter Auxiliary Piping," and 3.7.3.8.3, "Piping Systems on Modules," respectively, provide additional standards for piping supported by flexible equipment or flexible modules.

The staff found the modeling methods for AP1000 piping identical to the methods approved for AP600. The staff reviewed these requirements and found them acceptable because they conform to SRP Section 3.7.3 and good engineering practice. DCD Tier 2, Sections 3.7.3.8.1, 3.7.3.8.2, and 3.9.3.4, include additional information on the stiffness, mass, and decoupling requirements for supporting systems, including supplementary steel, equipment, and other piping systems. Sections 3.12.4.4 and 3.12.6.7 of this report discuss the staff evaluation of these criteria.

Design of Structures, Components, Equipment, and Systems

3.12.4.3 Piping Benchmark Program

Final piping and pipe support stress analyses cannot be completed before design certification because their completion depends on as-built or as-procured information. However, the staff has accepted an alternative approach using DAC in its evaluation of evolutionary and advanced reactor standard designs. The DAC are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC staff relies in making a final safety determination to support a design certification. As part of the DAC approach in the evaluation of piping and pipe support design, the staff requests COL applicants who will complete the piping analysis and finalize the piping designs to verify their computer programs in accordance with the NRC benchmark program specific to the standardized plant design. Under the benchmark program, the COL applicant applies its computer stress analysis program to construct a series of selected piping system mathematical models that are representative of the standard plant piping designs. The models are analyzed using approved dynamic analysis methods and representative loads. The results of the analyses are compared with the results of independent benchmark problem analyses developed by the staff. For each benchmark problem, modal frequencies, maximum pipe moments, maximum support loads, maximum equipment nozzle loads, and maximum deflections should meet the range of acceptable values specified in the NRC benchmark program report. The COL applicant should 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 piping analyses. The benchmark program ensures that the computer program used to complete the piping design and analyses produces results that are consistent with results considered acceptable to the NRC staff.

"Piping Benchmark Problems for the Applicant AP600," NUREG/CR-6414, dated August 1996, describes the NRC piping benchmark problems, analysis results, and acceptance criteria for the AP600. The three benchmark problems included (1) a seismic analysis of the pressurizer surge line using the enveloped response spectrum method, (2) a seismic analysis of the pressurizer surge line using the independent support motion response spectrum method, and (3) a fluid transient dynamic analysis of the main steam line using the modal superposition time-history analysis method. Based on discussions between the NRC staff and the applicant, these problems were selected as representative AP600 piping designs and analytical methods. During the staff's review of the AP600, the applicant provided its PS+CAEPIPE program mathematical models and analysis results for these three sample problems to the staff. The staff performed its own confirmatory piping stress analyses for the three problems using an independently developed and verified computer program. A comparison of the two sets of analysis results showed the values to be within an acceptable range of agreement. The NRC benchmark program report for the AP600 documented the NRC-developed mathematical models and input loads, analysis results, and range of acceptable values. In the AP600 DCD, the applicant made a commitment providing that the COL applicant would implement the NRC benchmark program using AP600-specific problems if it employs a piping analysis computer program other than those used for design certification.

In DCD Tier 2, Sections 3.9.1.2 and 3.9.8.6, the applicant stated that the COL applicant will implement the NRC benchmark program using AP1000-specific problems if it employs a piping analysis computer program other than those used for design certification. As part of RAI 210.045, the staff requested the applicant to identify and provide three sample AP1000-specific problems for benchmarking. The applicant responded that the three benchmark

problems used for the AP600 are also representative piping analysis problems for the AP1000. The applicant and staff also discussed this subject during the NRC review meeting at the Westinghouse office on September 9–11, 2002. The applicant stated that the AP1000 pressurizer surge line is identical to the AP600 surge line. The main steam line has a larger diameter but a similar layout. As discussed in Section 3.12.4.1 of this report, the PIPESTRESS program used for the AP1000 is the same program as PS+CAEPIPE which the applicant used for the AP600 and benchmarked against the NRC AP600 benchmark problems. In addition, the applicant indicated that DST Computer Services had incorporated the three AP600 benchmark problems into the formal library of verification test set problems for PIPESTRESS, as documented in the user's manual. DST maintains these problems as part of its formal quality assurance program. The applicant revised DCD Tier 2, Sections 3.9.1.2 and 3.9.8.6, to reference the AP600 NRC benchmark program report and to indicate that the AP600 benchmark problems are representative for the AP1000 and can be used for the AP1000 benchmark program.

On the basis of the additional information on the application and verification of the PIPESTRESS computer program and because the piping systems selected for the AP600 benchmark problems are representative of those in the AP1000, the staff finds the Westinghouse approach to the piping benchmark program acceptable.

3.12.4.4 Decoupling Criteria

In developing mathematical models of piping systems, it is generally desirable to limit the size of the model by decoupling small branch lines from larger run lines. DCD Tier 2, Sections 3.7.3.8.1, "Supporting Systems," and 3.7.3.8.2, "Supported Systems," define the considerations and the acceptance criteria for the decoupling of branch line piping.

DCD Tier 2, Section 3.7.3.8.1, addresses the analysis of piping systems that provide support to other piping systems. It states that the supported branch piping may be excluded from the analysis of the supporting run 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 3. In addition, the mass and stiffness effects of the decoupled branch line on the analysis of the run line must be considered. The stiffness effect is considered significant when the distance from the run pipe outside diameter to the first rigid or seismic support on the decoupled branch pipe is less than or equal to half the deadweight span of the branch pipe. The mass effect is significant when the weight of half the span of the branch line (in each direction) exceeds 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 decoupling point for the run pipe analysis. If the stiffness and/or mass effects are considered significant, the piping analysis model for the run pipe analysis includes the branch piping. The portion of the branch line piping considered in the analysis should adequately represent the behavior of the run pipe and branch pipe. The branch line model ends at either the (1) first six-way anchor, (2) four rigid/seismic supports in each of the three perpendicular directions, or (3) rigidly supported zone as described in DCD Tier 2, Section 3.7.3.13.4.2.

DCD Tier 2, Section 3.7.3.8.2, addresses the analysis of piping systems that are supported by other piping systems or equipment. When the supporting system is a piping system, the

Design of Structures, Components, Equipment, and Systems

supported (branch) pipe can be decoupled from the supporting pipe (run) if the ratio of the moment of inertia 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 3. During the analysis of the branch piping, resulting values of tee anchor reactions are checked against the capabilities of the tee. When significant, the branch line analysis considers the seismic inertia effects of equipment and piping that provide support to supported (branch) piping systems. When the frequency of the supporting equipment is less than 33 Hz, then either a coupled dynamic model of the piping and equipment is used, or the amplified response spectra at the equipment connection point is used with a decoupled model of the supported piping. When supported piping is supported by larger piping, either a coupled dynamic model of the supported piping and 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 above criteria are consistent with the criteria for decoupling branch lines from run pipe as recommended in NUREG/CR-1980, dated March 1981. The staff review concludes that the decoupling criteria are based on sound engineering practice and are acceptable to ensure that the AP1000 piping design properly considers dynamic coupling of the supporting and supported piping.

3.12.4.5 Conclusions

The staff concludes that the applicant has 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 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 AP1000 design are not required to use the applicant's computer programs, the computer programs used by the COL applicant or licensee to complete its analyses of AP1000 piping systems will be validated using the piping benchmark program discussed herein. This is COL Action Item 3.12.4.5-1.

3.12.5 Piping Stress Analysis Criteria

The staff has evaluated the pipe stress analysis criteria described in the DCD for the AP1000, as discussed below.

3.12.5.1 Seismic Input

The AP1000 is designed for an SSE ground motion defined by an RG 1.60 response spectrum that is enhanced in the high-frequency range and anchored to a peak ground acceleration of 0.3 g. Amplified building response spectra are generated for the AP1000 design for a hard rock site.

In DCD Tier 2, Section 3.7.2.5, "Development of Floor Response Spectra," the applicant described the development of floor response spectra for the AP1000 design. It stated that the design floor response spectra are generated according to RG 1.122. The seismic floor spectra are computed using time-history responses determined from the NI seismic analysis. An

ANSYS modal superposition time-history analysis determines the time-history responses for the hard rock condition. Floor response spectra are computed at the specified locations for damping values of 2, 3, 4, 5, 7, 10, and 20 percent of critical damping. For the design of subsystems and components, the floor spectra are generated by enveloping the nodal response spectra determined for the hard rock site. The enveloped floor response spectra are smoothed, and the spectral peaks associated with the structural frequencies are broadened by ±15 percent to account for the variation in the structural frequencies due to the uncertainties in parameters, such as material and mass properties of the structure and soil, damping values, seismic analysis technique, and the seismic modeling technique. In RAI 210.037, the staff requested the applicant to clarify the need for enveloping the floor response spectra since only one analysis will be performed for a hard rock site. The applicant responded that no enveloping is needed since there is only one analysis on hard rock. The applicant revised DCD Tier 2, Section 3.7.2.5, to delete the references to enveloping the response spectra.

In DCD Tier 2, Section 3.7.3.9, the applicant described an alternative seismic analysis method for subsystems and components, including piping systems. It stated that the peak shifting method, which employs the unbroadened floor response spectra, may be used in place of the broadened response spectrum method. When this method is applied, the unbroadened response spectrum peaks are shifted to match all natural frequencies within a ±15 percent interval of the peak. Separate analyses are then performed using the original unbroadened spectrum, the modified spectra with peaks at the natural frequencies, and the modified spectra with peaks shifted ±15 percent. The results of each analysis are enveloped to obtain the final results. The peak shifting is performed independently in each of the three directions.

With regard to the adequacy of broadening response spectrum peaks ±15 percent, the staff identified an issue as a part of its seismic design audit that could potentially affect the extent to which the response spectrum peaks need to be broadened. Section 3.7.3 of this report discusses this issue in more detail. A proposed resolution might have involved varying the amount of peak broadening to -20/+10 percent. Broadening the response spectra peaks by -20/+10 percent would have had a significant impact on the staff's conclusions in this report on piping seismic analyses and LBB evaluations, as well as on the need to develop a piping benchmark program specific to the AP1000 using the new seismic response spectra. The potential impact of changing the peak broadening of the seismic response spectra on the AP1000 piping design was Open Item 3.7.2-1 in the DSER.

DSER Open Item 3.7.2-1 is directly related to the resolution of Open Item 3.7.2.3-1 in the DSER. The resolution of Open Item 3.7.2.3-1, discussed in Section 3.7.2.3, concludes that the seismic response spectral difference is insignificant and therefore has no effect on piping seismic analyses and LBB evaluations. Therefore, Open Item 3.7.2-1 is resolved.

3.12.5.2 Design Transients

In DCD Tier 2, Section 3.9.1.1, the applicant discussed the design transients for ASME Code Class 1 components and supports. In DCD Tier 2, Table 3.9-1 lists the design transients for five groups of plant operating conditions and the number of cycles for each event within the group that are normally used for fatigue evaluation of components including ASME Code Class 1 piping systems.

Design of Structures, Components, Equipment, and 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

The applicant stated that the number of events or cycles resulting from each of the listed design transients was defined to be consistent with a 60-year design objective. In RAI 210.054, the staff noted that an AP1000 reactor will be licensed for an initial term of only 40 years. In light of this fact, the staff requested the applicant to clarify the fatigue life considered in the design. The applicant responded that the AP1000 is actually designed for a 60-year design life. Therefore, the number of normal and upset transients used in fatigue evaluations performed for the design of SSCs is based on a 60-year design life. Due to regulatory requirements, the COL granted for a standard plant has a 40-year duration. A COL holder seeking to operate an AP1000 beyond 40 years would need to apply for license renewal with the NRC. The staff finds this clarification consistent with the rules contained in NUREG-1800 and the SRP for review of license renewal applications for nuclear power plants, and thus acceptable.

The staff finds the description of the design transients for fatigue evaluation of ASME Code Class 1 piping systems consistent with ASME Code requirements and acceptable. Section 3.9.1.1 of this report contains a more detailed discussion of this issue.

3.12.5.3 Loadings and Load Combinations

The staff reviewed DCD Tier 2, Section 3.9.3.1, "Loading Combinations, Design Transients, and Stress Limits," in accordance with SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Components Supports, and Core Support Structures." The loadings and load combinations should be sufficiently defined to provide the basis for design Code 1, 2, and 3 components and Class CS core support structures for all conditions. The acceptability is also based on comparisons with positions in Appendix A to SRP Section 3.9.3 and with appropriate standards acceptable to the staff.

DCD Tier 2, Section 3.9.3.1, describes the loads, loading combinations, design transients, and stress limits for AP1000 ASME Code Class 1, 2, and 3 components. DCD Tier 2, Section 3.9.3.1.1, "Seismic Loads and Combinations Including Seismic Loads," discusses seismic loads and combinations, including seismic loads for ASME Code Class 1, 2, and 3 components. DCD Tier 2, Section 3.9.3.1.2, "Loads for Class 1 Components, Core Support, and Component Supports," discusses all other loads used in the analysis of Class 1 components and supports. DCD Tier 2, Table 3.9-3, provides a summary listing and description of loading for ASME Class 1, 2, and 3 components and supports including piping. Detailed discussions of loading combinations and stress limits are provided in DCD Tier 2, Section 3.9.3.1.3, "ASME Code Class 1 Components and Supports and Class CS Core Support Loading Combinations and Stress Limits," for Class 1 components; DCD Tier 2, Section 3.9.3.1.7, "ASME Code Class 2 and 3 Components," for Class 2 and 3 components; and DCD Tier 2, Section 3.9.3.1.5, for Class 1, 2 and 3 piping. DCD Tier 2, Section 3.9.3.1.5, states that piping systems are designed and analyzed for Levels A, B, C, and D service condition requirements according to the rules of

the ASME Code, Section III. DCD Tier 2, Table 3.9-5, provides the minimum design loading combinations for ASME components for design and service level conditions. These combinations include loads that induce primary stresses as defined in the ASME Code. DCD Tier 2, Tables 3.9-6 and 3.9-7 list additional load combinations and stress limits for Class 1 piping and for Class 2 and 3 piping, respectively. These combinations include thermal expansion, anchor motions, and other loads that induce secondary stresses and peak stresses. Stress limits for primary stress load combinations for design and service level conditions are included in DCD Tier 2, Table 3.9-9 for Class 1 components and DCD Tier 2, Table 3.9-10 for Class 2 and 3 components. DCD Tier 2, Table 3.9-11, gives the functional capability requirements for ASME piping systems that maintain an adequate fluid flow path to mitigate a Level C or Level D plant event.

The staff reviewed the proposed loads, load combinations, and stress limits given in the DCD sections and tables discussed above, including the clarifications and proposed DCD revisions provided in the RAI responses. The loads, load combinations, and stress limits are identical to those approved for AP600. The staff finds them to be consistent with the guidelines provided in Section 3.9.3 of the SRP and the staff position on single earthquake design, and therefore they are acceptable.

3.12.5.4 <u>Damping Values</u>

DCD Tier 2, Sections 3.7.3.15 and 3.7.1.3, discuss the damping values specified by the applicant for use in the AP1000 design. DCD Tier 2, Table 3.7.1-1 lists these values. DCD Tier 2, Appendix 3C.4 discusses damping for the reactor coolant loop seismic analysis. The damping values depend on the characteristics of the piping system and on the analysis method as summarized in DCD Tier 2, Table 3.7.1-1. For the primary coolant loop, a damping value of 4 percent is used. For piping systems with rigid valves analyzed by the uniform envelope response spectrum method, a damping value of 5 percent is used. However, this value is not used in piping systems susceptible to stress-corrosion cracking. For the auxiliary piping systems analyzed by the independent support motion response spectrum method or the time-history analysis method, damping values of 2 percent or 3 percent are used, as specified in RG 1.61. DCD Tier 2, Section 3.7.3.15, states that for subsystems that are composed of different material types, the composite modal damping approach with the weighted stiffness method is used to determine the composite modal damping value. Alternately, the minimum damping value may be used for these systems. DCD Tier 2, Appendix 3C.4 states that composite modal damping is applied in the reactor coolant system seismic analysis using 5 percent for the building structures and 4 percent for the RCS components. This method may also be used for coupled models of building and piping systems attached to the primary coolant loop system and the interior concrete building using the appropriate piping and structure damping values. Composite modal damping may also be used for piping systems that are coupled to flexible equipment or flexible valves, and to coupled models of piping and nonsimple module steel frames.

The staff had reviewed and accepted for the AP600 the use of 5-percent damping for piping systems for ALWR plants on the basis that ALWR plants must be designed to a minimum 0.3 ZPA for the SSE. This high seismic acceleration provides assurance that piping systems will experience higher damping valves. Its acceptance was also subject to certain limitations specified in RG 1.84 for ASME Code Case N-411-1. The limitations applicable to design include (1) limiting the building filtered responses to 33 Hz and below, (2) using damping values only in

Design of Structures, Components, Equipment, and Systems

those analyses in which current seismic spectra are used, (3) not allowing the use of damping values when using supports to dissipate energy by yielding, and (4) not allowing their use where stress-corrosion cracking is a concern. In RAI 210.040, the staff requested the applicant to verify that these limitations will apply to AP1000 piping. The applicant's response confirmed the staff's assumptions. The applicant stated that the 5-percent damping value will be used consistently for all piping system seismic analyses utilizing enveloped response spectrum methods. The enveloped response spectra are developed in accordance with RG 1.122, as described in the DCD. The design of the AP1000 piping systems does not include supports designed to dissipate energy by yielding, and the piping systems analyzed are not susceptible to stress-corrosion cracking. The staff concurs that these limitations conform to the RG 1.84 limitations.

Based on its review of the information in the DCD, the staff concludes the following:

- The application of the composite modal damping approach with the weighted stiffness method as described in the DCD is consistent with one of the two techniques for determining an equivalent modal damping matrix or composite damping matrix given in SRP Section 3.7.2 and, therefore, is acceptable.

- The use of 2-percent damping for pipes less than or equal to 30.5-cm (12-in.) diameter and 3 percent damping for pipes greater than 30.5-cm (12-in.) diameter is consistent with RG 1.61 and acceptable for independent support motion response spectrum analysis and for time-history analysis.

- The use of 4-percent damping for the primary coolant loop is based on a Westinghouse study documented in staff-approved WCAP-7921-AR. The report included an NRC staff evaluation letter which stated that this damping value is acceptable for the seismic analysis of the primary coolant loop without restriction. On this basis, the staff finds it acceptable for the AP1000.

- The staff had reviewed and accepted the use of 5-percent damping for auxiliary systems piping analyzed by the uniform envelope response spectrum method for ALWR plants. The use of 5-percent damping is acceptable for piping systems, subject to the limitations as specified in RG 1.84 for ASME Code Case N-411-1. As discussed above, the applicant has confirmed that these limitations will be met for the AP1000. Based on meeting these limitations, the staff finds the use of 5-percent damping acceptable for the AP1000.

3.12.5.5 Combination of Modal Responses

DCD Tier 2, Section 3.7.3.7.2, discusses the methods for combining modal responses in seismic response spectrum analysis. The applicant stated that the total unidirectional seismic response for subsystems is obtained by combining the individual modal responses using the SRSS method. For subsystems having modes with closely spaced frequencies, this method is modified by one of the alternative grouping options provided in DCD Tier 2, Section 3.7.3.7.2, to include possible interaction effects of these modes. All these grouping methods are comparable to those recommended in RG 1.92. Because the staff position allows the use of any of the

methods described in RG 1.92, the staff finds the modal combination method for AP1000 piping systems acceptable.

3.12.5.6 High-Frequency Modes

DCD Tier 2, Section 3.7.3.7.1, describes alternative methods of accounting for high-frequency modes (generally greater than 33 Hz) in seismic response spectrum analysis. These include the left-out-force (or missing mass correction) method and a method similar to that described in Appendix A to Section 3.7.2 of the SRP that combines responses associated with high-frequency modes when the lower frequency modes do not adequately define the mass content of the structure. The applicant also stated that high-frequency modes can be excluded from the response calculation if the change in response is less than or equal to 10 percent. The criterion for exclusion of high-frequency modes based on change of response less than or equal to 10 percent is consistent with SRP Section 3.7.2, Appendix A, and is acceptable to the staff.

As described in DCD Tier 2, Section 3.7.3.7.1.1, "Left-Out-Force Method or Missing Mass Correction for High-Frequency Modes," the left-out-force method is based on the left-out-force theorem, which states that for every time-history load there is a frequency, f_r, called the "rigid mode cutoff frequency" above which the response in modes with natural frequencies above f_r will closely resemble the applied force at each instant in time. These modes are called "rigid modes." The applicant provided the equations defining the left-out-force vector in this section of the DCD. The left-out-force vectors are used to generate left-out-force solutions which are multiplied by the ZPA of the response spectrum for the corresponding direction. The SRSS method then combines low-frequency mode responses with these high-frequency responses (rigid modes). The PIPESTRESS computer program incorporates the left-out-force method. The staff evaluated the left-out-force method and confirmed its adequacy on the basis of a comparison between the applicant's AP600 piping benchmark problem results (which are also applicable to and representative of AP1000 piping) and the results of the staff's independent confirmatory analyses as discussed in Section 3.12.4.3 of this report. On this basis, the staff finds the left-out-force method acceptable for use in the AP1000.

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 consider all cyclic effects caused by the plant operating transients for a 60-year design life. However, current test data indicate that the ASME Code, Section III design fatigue curves may not be conservative for nuclear power plant primary system environments. The ASME is currently assessing the effects of the environment on the Section III design fatigue curves. In RAI 210.055, the staff requested the applicant to describe the method that will be used to account for the effects of the environment on the fatigue design of RCPB components in the AP1000 plant.

The applicant responded that in SECY-95-245, "Completion of the Fatigue Action Plan," the NRC staff concluded that based on component sample evaluations including fatigue environmental effects, the fatigue limit would not be exceeded for most components, and that a fatigue failure of piping is not a significant contributor to core-melt frequency. Therefore, no further evaluation of fatigue environmental effects on operating plants was needed. The evaluations were based on typical component designs of plants with a 40-year design life. The

Design of Structures, Components, Equipment, and Systems

current focus of industry efforts to address fatigue environmental effects for license renewal is in the EPRI Materials Reliability Program Integrated Task Group on Fatigue Issues. This group has proposed methods to address environmental effects in fatigue evaluations. The Pressure Valve Research Committee (PVRC) Steering Committee on Cyclic Life and Environmental Effects has proposed and discussed similar methods, as described in the PVRC draft report, "Assessment of Environmental Effects on Fatigue Life in LWR Nuclear Applications." The applicant stated that these methods, based on industry data, will be used to evaluate the effect of environment on the fatigue design of components.

The staff is in the process of conducting a generic review of these industry data and the proposed methodology to address the issue of environmental fatigue. The NRC staff has funded studies of this issue, and NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," provide environmental fatigue test data. New test data are being published and modified methods to utilize these data are being proposed. However, the staff has not yet established a final position on this issue to be imposed unilaterally for all new plant designs pursuant to 10 CFR 50.55a or backfitted onto existing plants pursuant to 10 CFR 50.109. In view of this context, the staff finds that the AP1000 design meets current ASME Code, Section III, fatigue requirements for Class 1 piping. In the future, holders of licenses for AP1000 plants will have to address the environmental effect on fatigue design if they seek to renew those licenses and will use the criteria acceptable to the staff at that time. On this basis, the staff finds the Class 1 piping fatigue design acceptable for design certification.

3.12.5.8 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

During its review of the AP600, the staff raised a concern that the current ASME Code Class 2 and 3 rules for fatigue may be inadequate to ensure a design life of 60 years for some piping system components. The staff subsequently determined that the only Class 2 or 3 components subjected to severe fatigue loadings are the nozzles on the secondary side of the steam generators. The applicant stated that, although those components are classified as ASME Class 2, they are designed to satisfy the criteria specified in Section III of the ASME Code for Class 1 components which include fatigue evaluation. In DCD Tier 2, Section 5.4.2.1, the applicant provided the same commitment for its AP1000 steam generator nozzles. The staff finds that the application of Class 1 fatigue design rules to Class 2 and 3 piping is conservative, and, therefore, is acceptable. The staff notes that any Class 1 fatigue evaluations for Class 2 or 3 components should treat the effects of environment in the same manner as discussed in Section 3.12.5.7 of this report. Section 3.9.3.1 of this report also discusses this issue.

3.12.5.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System (NRC Bulletin 88-08)

The NRC issued Bulletin (BL) 88-08 following the discovery of cracks in unisolable piping connected to the RCS at several nuclear power plants. These cracks were attributed to thermal stresses resulting from unanalyzed temperature oscillation that could be induced by isolation valve leakage. The bulletin recommended that licensees review systems connected to the RCS to determine whether any unisolable sections of this piping can be subjected to temperature oscillations induced by leaking valves resulting in cyclic thermal stresses that normally were not

evaluated in the piping design analysis. It also recommended nondestructive examinations of potentially affected pipes to assure that no flaws exist, as well as the development and implementation of a program to provide continuing assurance of piping integrity. Ways to provide this assurance include designing the system to withstand the cycles and stresses from valve leakage, instrumenting the piping to detect adverse temperature distributions and establishing appropriate limits, and providing a means to monitor pressure differentials that may lead to valve leakage.

DCD Tier 2, Section 3.9.3.1.2, describes the applicant's approach to the BL 88-08 issues in the design of piping for the AP1000. The applicant stated that, for adverse stresses from leakage to occur, the necessary conditions are (1) a component with potential leakage (usually a valve) must exist, (2) a pressure differential capable of forcing leakage must exist, and (3) a temperature differential between the unisolable piping section and the leakage source sufficient to produce significant stresses in the event of leakage must exist. The applicant provided a list of assumptions that were made in evaluating the systems for susceptibility to adverse stresses from valve leakage. Based on its evaluation, the applicant determined that unisolable portions of 10 lines connected to the reactor coolant system were not susceptible. However, the applicant determined that unisolable portions of two lines, the passive residual heat removal line and the automatic depressurization Stage 4 lines, were susceptible to thermal stratification, cycling, or striping. For these lines, piping stress analyses are performed to demonstrate that they meet the applicable requirements of ASME Code, Section III. These piping stress analyses include consideration of plant operation and thermal stratification using temperature distributions which are developed from finite element fluid flow and heat transfer analyses. For the AP600, the staff notes that the applicant stated that it would use the methods from the EPRI report, TR-103581, "Thermal Stratification, Cycling and Striping (TASCS)," to define isolation valve leakage transients, locate sites of thermal penetration cycling, determine number of leakage cycles, and calculate the thermal striping fatigue usage factors. The staff did not find this statement in the AP1000 DCD, but requested that the applicant include it. Documentation of the calculation method of the thermal striping fatigue usage factor was identified as Confirmatory Item 3.12.5-1 in the DSER.

The staff reviewed the revised DCD and concluded that the calculation method of the thermal striping fatigue usage factor was acceptably documented. Therefore, Confirmatory Item 3.12.5-1 is resolved.

During its previous review of the AP600 design certification application, the staff conducted a design review of the applicant's methodology for the identification and evaluation of piping systems susceptible to thermal stratification, cycling, and striping. This included a review of the detailed calculations that documented the susceptibility evaluation methods and results and a review of the results of the pipe stress analyses performed for systems found to be susceptible. The staff found the AP600 methodology adequately addressed the staff's concerns identified in BL 88-08. In its review of DCD Tier 2, Section 3.9.3.1.2, the staff found the description of the applicant's evaluation and its results to be identical to those for the AP600. Because the piping systems and layout for the AP600 and AP1000 are similar, the AP600 methodology for identifying and evaluating piping systems susceptible to thermal stratification, cycling, and striping may be used for the AP1000 as well. However, the DCD did not describe how differences in design parameters between the two plants (such as fluid temperature, pressures, or flow rates) were considered. In RAI 210.049, the staff asked the applicant to explain how it

Design of Structures, Components, Equipment, and Systems

considered differences in design parameters that may affect thermal cycling and thermal stratification loadings between the AP1000 and the AP600 in its evaluation. During the September 9–11, 2002, meeting at the Westinghouse office, the applicant provided and explained its AP600 calculation which identified lines susceptible to thermal stratification, cycling, and striping (TASCS). The staff noted that it was the same calculation that it had reviewed during the AP600 design certification review. In its RAI response, the applicant provided a comparison of the critical parameters affecting TASCS loadings. The applicant identified the following five systems, including the two systems identified in the preceding paragraph, that are susceptible to adverse stresses resulting from TASCS loadings:

(1)   cold-leg piping in the loop with passive RHR (during long-term PRHR operation)
(2)   pressurizer surge line
(3)   automatic depressurization system Stage 4 lines
(4)   RNS suction line
(5)   PRHR return line

The applicant provided a table of parameters relevant to TASCS comparing the AP600 and AP1000 designs for these five systems. Relevant parameters included pipe size, pipe routing, temperature, and flow rate. These five lines include lines susceptible to thermal cycling (BL 88-08) as well as lines susceptible to thermal stratification. The applicant stated that the AP1000 physical design for these piping systems is similar to the AP600 design in most respects, except for some pipe size increases. Temperature changes are minor. Flow rates have increased for the reactor coolant loop for normal power operation. In its evaluation of parameter changes, the applicant noted that AP1000 analyses for these lines should demonstrate similar results for all lines, except the ADS Stage 4 lines which may be affected by the higher hot-leg flow rates. The applicant stated that as part of the detailed piping design for the AP1000, it will perform system reviews of the AP1000 piping similar to the calculations performed for the AP600. The applicant will evaluate the design differences with respect to the effects on TASCS loading and perform additional calculations, including CFD, if necessary. The piping design analyses will include resulting thermal loadings.

The applicant provided this information in its revised response to RAI 210.049. The response stated that the evaluation of design differences provided in the original RAI response establishes a basis for concluding that the TASCS loadings for the AP1000 piping systems are not significantly different than those for the AP600. DCD Tier 2, Section 3.9.8.2, notes that the COL applicant will complete the final stress analysis of the ASME components and piping systems as part of the COL application. This COL commitment is amended to specify that the TASCS loadings will be evaluated, and additional calculations, if necessary, will be performed as part of the COL application. DCD Tier 2, Sections 3.9.3.1.2 and 3.9.8.2 adequately reflect these commitments. This is COL Action Item 3.9.2.4-1. The issue identified in RAI 210.049 is therefore resolved.

3.12.5.10 Thermal Stratification

The phenomenon of thermal stratification 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. NRC BL 79-13 discusses the effects of thermal stratification in operating reactors in feedwater lines, and NRC BL 88-11 discusses these effects in pressurizer surge lines.

DCD Tier 2, Section 3.9.3.1.2, discusses the applicant's approach to thermal stratification issues in the design of AP1000 piping systems. The applicant stated that the design of piping and component nozzles in the AP1000 includes provisions to minimize the potential for and the effects of thermal stratification and cycling. Piping and component supports are designed and evaluated for the thermal expansion of the piping resulting from potential stratification modes. The evaluation considers the information on thermal cycling and thermal stratification included in applicable NRC bulletins and other design standards.

The NRC issued BL 79-13 as a result of a feedwater line cracking incident at D.C. Cook, Unit 2. This bulletin recommended that operating plant feedwater lines be inspected. This resulted in the discovery of cracks in the feedwater lines of several plants. To provide a uniform approach to address this issue, a Feedwater Line Cracking Owners Group was established. This program evaluated the thermal, hydraulic, structural, and environmental conditions which could individually or collectively contribute to feedwater line crack initiation and growth. The results of the program indicated that the primary cause of cracking was thermal fatigue loading induced by thermal stratification and high-cycle thermal striping during low-flow auxiliary feedwater injection. The program concluded that the mode of failure was corrosion fatigue.

In DCD Tier 2, Section 3.9.3.1.2, the applicant stated that the AP1000 steam generators are equipped with separate nozzles for the main feedwater and the startup feedwater lines. Analyses of the main feedwater nozzles are performed to demonstrate that the applicable requirements of the ASME Code, Section III are met. Thermal stratification is prevented in the main feedwater line based on the flow rate limitations within the main feedwater line and the flow control stability for feedwater control. The startup feedwater line provides low feedwater flow duty, while the main feedwater line provides and controls higher feedwater flow rates. The switchover from the startup to the main feedwater line occurs above a minimum flow rate to prevent thermal stratification for limiting temperature deviations. Main feedwater control valve positioning during normal operation is the function of the plant control system. The control scheme enhances steam generator level stability and thus reduces potential feedwater thermal stratification resulting from temporary low-flow transients.

In DCD Tier 2, Section 3.9.3.1.5, the applicant stated that DCD Tier 2, Section 3.9.3.1.2 identifies a monitoring program for the feedwater line at the first AP1000. Upon review, however, the staff did not find any information on a feedwater line monitoring program in this subsection. In RAI 210.052, the staff requested the applicant explain this discrepancy. In a letter dated October 2, 2002, the applicant responded that feedwater line monitoring was not required for the AP600 and is not implemented in the AP1000. Feedwater line monitoring was part of the AP600 justification for applying LBB methodology to the feedwater line piping. The applicant deleted these provisions for the AP600 when the NRC did not approve LBB for the feedwater line but did not make the appropriate update in DCD Tier 2, Section 3.9.3.1.5. The applicant subsequently updated the AP1000 DCD to delete the reference to feedwater line monitoring. The staff finds that because LBB is not applied to the AP1000 feedwater line, the deletion is appropriate and acceptable for the AP1000.

Design of Structures, Components, Equipment, and Systems

The NRC issued BL 88-11 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 recommended that all PWR licensees 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).

In DCD Tier 2, Section 3.9.3.1.2, the applicant described the actions recommended in BL 88-11 and the manner in which it will address these actions for the AP1000. The analysis of the AP1000 surge line considers thermal stratification and thermal striping and demonstrates that the surge line meets applicable ASME Code, Section III requirements for the licensed life of the plant. Hot functional testing requirements for the AP1000 ensure that piping thermal deflections result in no adverse consequences. As part of the Westinghouse Owners Group (WOG) program on surge line thermal stratification, the applicant collected surge line physical design and plant operational data for all domestic Westinghouse PWRs. In addition, the applicant collected surge line monitoring data from approximately 30 plants. The applicant used this experience in the development of the AP1000 thermal stratification loadings, and, therefore, monitoring of the AP1000 surge line is not necessary.

The COL holder will implement a monitoring program at the first AP1000 plant to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters, such as pressurizer temperature and level, hot-leg temperature, and reactor coolant pump status. Monitoring will be performed during hot functional testing and during the first fuel cycle. The resulting monitoring data will be evaluated to show that it is within the bounds of the analytical temperature distributions and displacements. Procedures for the AP1000 provide for documentation and maintenance of records in accordance with 10 CFR Part 50, Appendix B. In RAI 210.048, the staff noted that the monitoring program provisions were not clear and requested that the applicant clarify them. In a letter dated October 2, 2002, the applicant stated that the monitoring program described in the DCD applies only to the first AP1000 plant. The applicant revised DCD Tier 2, Section 3.9.3.1.2, to indicate that a monitoring program is not needed for plants subsequent to the first AP1000 plant. The staff reviewed the information in the DCD and determined that the methodology for confirming pressurizer surge line integrity in view of the occurrence of thermal stratification is acceptable as it satisfies the recommended actions in NRC BL 88-11. This is COL Action Item 3.12.5.10-1.

In RAI 210.050, the staff requested the applicant to explain whether it had considered the differences between the AP1000 and the AP600 with regard to the potential for stratification between the pressurizer and the hot leg. Specifically, the staff expressed a concern that the pressurizer could be stratified and the heatup and cooldown rate could exceed the defined limit with a large surge flow rate. The staff asked the applicant to describe the control of the heatup and cooldown procedure such that the ΔT between the pressurizer and the RCS hot leg will be less than acceptable values, and pressurizer stratification will not cause stress and fatigue concerns. The applicant responded that the design of the AP1000 surge line is identical to that of the AP600 surge line. The AP1000 design will have a slightly lower susceptibility to surge line

stratification during normal operation due to the increased AP1000 operating temperature. Specifically, surge line stratification can develop due to the temperature difference between the pressurizer and the hot leg. The normal operating ΔT between the pressurizer and hot leg is 29.4 °C (53 °F) for the AP600, while the normal operating ΔT for the AP1000 is 23.9 °C (43 °F). The surge line is designed to accommodate a temperature difference of 177.8 °C (320 °F), which can occur during shutdown operations. The staff reviewed the RAI response and concluded that it did not adequately address the staff's concern. The critical ΔT occurs during heatup and cooldown. The applicant did not provide a comparison of the AP600 and AP1000 ΔTs during these critical operating conditions. In addition, the applicant did not address the staff's specific concern regarding the procedural controls to prevent exceeding the heatup and cooldown rate limits to ensure that pressurizer stresses do not exceed fatigue stress limits.

The applicant provided additional information in response to RAI 210.050, Revision 1. A WOG program evaluated the presence of insurge/outsurge transients in operating plants that could result in stratification of the pressurizer. This program recommended modified plant operational procedures to protect the pressurizer lower head from these transients during heatup and cooldown operations. System operation described in DCD Tier 2, Section 5.4.5.2.3, "Operation," is consistent with the recommended operational procedures from the WOG program.

Energizing all of the backup pressurizer heater groups, demonstrated to reduce the number of insurge transients at several operating plants, maintains continuous pressurizer outsurge flow during plant heatup and cooldown. The maximum ΔT between the pressurizer and hot leg in the AP1000 is reduced because of the use of canned motor RCPs. The WOG program also recommended that the design provide for inadvertent or potentially unavoidable transient events. These transients, defined to envelop a fast moving hot/cold stratified interface, may result in pressurizer ΔT exceeding the overall rate of 55.6 °C/hour (100 °F/hour) heatup and 111.1 °C/hour (200 °F/hour) cooldown, and are included in the fatigue analysis. The applicant committed to include a description of AP1000 operations to minimize pressurizer stratification during plant heatup and cooldown in DCD Tier 2, Section 5.4.5.2.3. The staff finds the above clarification and the recommended resolution acceptable, contingent upon the revision of AP1000 DCD Section 5.4.5.2.3. This was Confirmatory Item 3.12.5-2 in the DSER.

Westinghouse revised DCD Tier 2, Section 5.4.5.2.3 and after the staff reviewed the revision concluded that the description of AP1000 operations to minimize pressurizer stratification during plant heatup and cooldown in the DCD was acceptable. Therefore, Confirmatory Item 3.12.5-2 was resolved.

In DCD Tier 2, Section 3.9.3.1.2, the applicant also addressed other applications. It stated that thermal stratification in the reactor coolant loops resulting from actuation of passive safety features is evaluated as a design transient. Stratification effects due to all levels of service conditions are considered. The criteria used in the evaluation of the stress in the loop piping due to stratification are the same. The staff had reviewed the methodology and found it acceptable. As discussed in Section 3.12.5.9 of this report, the COL applicant will perform an evaluation for the AP1000. The applicant will evaluate the design differences between the AP600 and the AP1000 with respect to the effects on TASCS loading and will perform additional calculations if necessary. The piping design analysis will include resulting thermal loadings. Based on stratification effects considering all levels of service conditions, the staff finds that thermal stratification loadings are conservatively assessed and are thus acceptable.

Design of Structures, Components, Equipment, and Systems

3.12.5.11 Safety Relief Valve Design, Installation, and Testing

DCD Tier 2, Section 3.9.3.3, "Design and Installation Criteria of Class 1, 2, and 3 Pressure Relieving Devices," contains the design and installation criteria applicable to the mounting of pressure relief devices used for the overpressure protection of ASME Code Class 1, 2, and 3 components. The applicant stated that the design of pressure-relieving valves complies with the requirements of ASME Code, Section III, Appendix O, "Rules for the Design of Safety Valve Installations." When there is more than one valve on the same run of pipe, the sequence of valve openings is based on the anticipated sequence of valve opening. The set point pressures or control system logic determine this sequence. The valve opening generates transient thrust forces at each change in flow direction or area. These forces are applied to the piping system to generate structural responses. The applicable stress limits are satisfied for the components in the piping run and connecting systems including supports. The applied forces and moments are based on the static application of transient thrust forces multiplied by a dynamic load factor of 2.0 unless a dynamic structural analysis is performed to calculate these forces and moments.

DCD Tier 2, Section 3.9.3.3.1, "Pressure Relief Devices and Automatic Depressurization Valves Connected to the Pressurizer," provides the design, operation, and analysis criteria for the pressure relief devices and automatic depressurization valves connected to the pressurizer. The pressurizer safety relief valves provide overpressure protection for the reactor coolant system and are the only ASME Code, Section III, Class 1 pressure relief valves in the AP1000. The automatic depressurization system valves provide a means to reduce reactor coolant system pressure to allow the passive core cooling system to fully function and are not designed to provide overpressure protection. The safety valves and the first three stages of the automatic depressurization valves are mounted in and supported by the pressurizer safety and relief valve (PSARV) module located above the pressurizer. The valve opening generates transient thrust forces at each change in flow direction or area. The analysis of the piping system and support considers the transient forces associated with valve opening.

Pressure-relieving devices for ASME Code, Section III, Class 2 systems include the safety valves and power-operated relief valves on the steam line and the relief valve on the containment isolation portion of the normal RNS, as discussed in DCD Tier 2, Section 3.9.3.3.2, "Pressure Relief Devices for Class 2 Systems and Components." DCD Tier 2, Section 3.9.3.3.3, "Design and Analysis Requirements for Pressure Relief Devices," discusses additional design and analysis considerations and criteria for open discharge and closed discharge valve stations.

The staff reviewed the information provided in DCD Tier 2, Section 3.9.3.3, in accordance with Section 3.9.3 of the SRP. This review included 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. On the basis that they meet the acceptance criteria in SRP Section 3.9.3.3, the staff concludes that the applicant's criteria for the design and evaluation of safety relief valves in the AP1000 plant are acceptable. In RAI 210.056, the staff requested the applicant to explain how it can ensure, without performing structural dynamic and thermal fatigue analysis, that the AP1000 plant-specific PSARV piping configuration can be designed to withstand the combined action of transient thrust forces and thermal gradients caused by the valve opening.

The applicant responded that the piping systems for the AP1000 ADS valves and the pressurizer safety valves are mounted in and supported by a module that is essentially the same as the module designed for the AP600. Based on its experience in the design of this piping system for the AP600, the applicant has high confidence that the piping configuration can be designed to withstand the combined action of transient thrust forces and thermal gradients caused by the valve opening. The applicant noted that, because of specific design features incorporated in both the AP600 and AP1000, the effects of the thrust forces on this module have been significantly reduced compared to PSARV systems used in conventional Westinghouse PWRs. Specifically, the ADS valves are slow opening compared to PORVs. This significantly reduces the thrust loads associated with opening of these valves at high pressure. In addition, the pressurizer safety valves are designed without a water loop seal. This also significantly reduces the thrust loads associated with the opening of these valves at high pressure. The pressurizer safety valve discharge piping is also significantly reduced in length and discharges to the containment atmosphere. This eliminates the amount of piping that experiences loads due to safety valve discharge. During the September 9–11, 2002, meeting at the Westinghouse office, the staff reviewed piping layout drawings for the AP600 and AP1000 PSARV modules. The review confirmed (1) the similarities between the AP600 and AP1000 designs, (2) that the safety valves do not contain any water loop seals upstream of the valves, and (3) that the length of piping downstream of the safety valves is minimized. Based on the above information, the staff concurs with the applicant conclusion that there is high confidence that the piping configuration can be designed to withstand the combined action of transient thrust forces and thermal gradients caused by the valve opening. Upon completing the final piping design analysis, the COL applicant will verify this conclusion using the DAC process.

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 as required by GDC 2. The staff accepts designs meeting the recommendations in NUREG-1367, "Functional Capability of Piping Systems," as satisfying the functional capability requirements.

In DCD Tier 2, Section 3.9.3.1.5, the applicant stated that DCD Tier 2, Table 3.9-11 shows the functional capability criteria for ASME piping systems that maintain an adequate fluid flow path to mitigate a Level C or Level D plant event. These criteria are based on NUREG-1367. The stress limits for Service Level D, equation 9, are the smaller of 2.0 S_y or 3.0 S_m for Class 1 piping, or the smaller of 2.0 S_y or 3.0 S_h for Class 2 and 3 piping. These limits are consistent with the 1989 edition of the ASME Code and NUREG-1367.

In addition, to ensure piping functional capability, the following conditions should be 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 S_y.

Design of Structures, Components, Equipment, and Systems

- D_o/t does not exceed 50.

- External pressure does not exceed internal pressure.

For load combinations that include slug-flow water hammer loads combined with pressure, weight, and other sustained mechanical loads, the equation 9 stresses are limited to the smaller of 1.8 S_y or 2.25 S_m for Class 1 piping, or the smaller of 1.8 S_y or 2.25 S_h for Class 2 and 3 piping. DCD Tier 2, Table 3.9-11, also provides stress limits for thermal expansion and steel containment vessel anchor motions under Service Level C and D events for piping systems that maintain an adequate fluid flow path.

During the review of the AP600 design certification, the staff examined the piping functional capability criteria in detail and found them to be consistent with the staff position. The staff finds that the proposed AP1000 piping functional capability criteria in DCD Tier 2, Section 3.9.3.1.7, meet the guidelines in NUREG-1367 and are identical to those approved for the AP600 and are therefore acceptable.

3.12.5.13 Combination of Inertial and Seismic Anchor Motion Effects

Piping analyses should include the effects of relative building movements at supports and anchors (seismic anchor motion), as well as the effects of 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 should be considered by imposing the maximum support displacements in the most unfavorable combination. This can be performed using a static analysis procedure. The analysis should include relative displacements of equipment supports (e.g., pumps or tanks), along with the building support movements. When needed 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.

DCD Tier 2, Section 3.7.3.9, describes the methods for combining the responses from the individual support or attachment points that connect the supported system or subsystem to the supporting system or subsystem. The response due to differential seismic anchor motions is calculated using static analysis (without including a dynamic load factor). In this analysis, the static model is identical to the dynamic model used to compute the seismic response due to inertial loading. In particular, the structural system supports in the static model are identical to those in the dynamic model.

The effect of relative anchor displacements is obtained using the worst combination of the peak displacements or by proper representation of the relative phasing characteristics associated with different support inputs. For components supported by a single concrete building, the seismic motions at all elevations above the basemat are taken to be in phase. When the component supports are in the same structure, the relative seismic anchor motions are small, and the effects are neglected. Supplemental steel frames that are flexible can have significant seismic anchor motions which are considered. When the component supports are in different structures, the relative seismic anchor motion between the structures is taken to be out-of-phase, and the

effects are considered. The results of the modal spectra analysis are combined with the results from the seismic anchor motion analysis by the absolute sum method.

DCD Tier 2, Table 3.9-8, provides the loading combinations for ASME Class 1, 2, and 3 piping and component supports. For Service Level D load combinations that include earthquake loads, the table specifies that the responses due to seismic inertia, seismic anchor motions, and seismic self-weight excitation are combined by the absolute sum method.

The staff reviewed the DCD sections discussed above and determined that the AP1000 methodology for combining seismic inertia and seismic anchor motion effects is consistent with the guidance of SRP Section 3.9.2 and, therefore, is acceptable.

3.12.5.14 Operating Basis Earthquake as a Design Load

Appendix S to 10 CFR Part 50, "Earthquake Engineering Criteria for Nuclear Power Plants," allows the use of single-earthquake design by providing the applicant an option to use an operating basis earthquake (OBE) value of one-third the maximum vibratory ground acceleration of the SSE, and to eliminate the requirement to perform explicit response analyses for the OBE. During its review of the AP600 design, the staff issued a document, included in a letter dated April 29, 1994, entitled, "Staff Position on the Use of a Single-Earthquake Design for Systems, Structures, and Components in the AP600 Standard Plant." It contained the staff's position on the types of analyses and information required in the DCD for the staff to approve the design of safety-related SSCs when the OBE is eliminated as a design-basis event. This document included specific supplemental criteria for fatigue, seismic anchor motion, and piping stress limits. The criteria applicable to ASME Code Class 1, 2, and 3 piping design are summarized below.

An acceptable cyclic load basis for fatigue evaluation is two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). Alternately, a number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D to IEEE Std 344-1987.

The effects of displacement-limited seismic anchor motions (SAM) on ASME Code components and supports should be evaluated to ensure their functionality during and following an SSE. For piping systems, the effects of SAM due to an SSE should be combined with the effects of other normal operational loadings that might occur concurrently as specified in the supplemental criteria of the staff position document as described below.

The design requirements in the 1989 edition including the 1989 addenda of the ASME Code, Section III, Subsections NB, NC, and ND will be met for Class 1, 2, and 3 piping. In addition, the following changes and additions to paragraphs NB-3650, NC-3650, and ND-3650 are necessary and will be satisfied for piping systems when OBE is eliminated from the design.

For the Class 1 primary stress evaluation (NB-3654.2), seismic loads are not required for consideration of Level B service limits for equation 9. For satisfaction of primary plus secondary stress intensity range limits in equation 10, M_i shall be either (1) the resultant range of all loads considering one-half the range of the SSE, or (2) the resultant range of moment due to the full

Design of Structures, Components, Equipment, and Systems

range of the SSE, whichever is greater. A reduced range (with an equivalent number of fractional vibratory peak cycles) may be used for consideration of Level B service limits (but with a range not less than one-third of the maximum SSE moment range). These load sets should also be used in calculating the peak stress intensity and alternating stress intensity for evaluating the fatigue effects and cumulative damage (NB-3653.2). In addition, the stress due to 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 under occasional loads (NC/ND-3653.1) in satisfying the Level B service limits for equation 9. Seismic anchor motion stresses are not required for consideration of thermal expansion or secondary stresses (NC/ND-3653.2) in equation 10. However, the stresses due to the combination of the range of moments caused by thermal expansion and SSE anchor motions must not exceed 3.0 S_h.

In DCD Tier 2, Section 3.7, the applicant stated that it has eliminated the OBE as a design requirement for the AP1000. As discussed in Section 3.12.5.3 of this report, DCD Tier 2, Section 3.9.3.1, describes loads, loading combinations, and stress limits for ASME Class 1, 2, and 3 piping and components. In DCD Tier 2, Tables 3.9-3, 3.9-5, 3.9-6, 3.9-7, 3.9-9, 3.9-10, and 3.9-11 define load cases, load combinations, stress criteria, and allowable limits. The staff reviewed these tables and found them to be consistent with the staff position on single-earthquake design described above.

In addition, DCD Tier 2, Sections 3.7.3.2 and 3.9.3.1.1 state that the fatigue evaluation will be based on five seismic events with an amplitude equal to one-third the SSE response. Each event has 63 high-stress cycles. Additionally, during the meetings held on September 9–11, 2002, at the Westinghouse office, the applicant and the NRC staff discussed the elimination of OBE from the design basis. Although OBE is not considered for the ASME Code primary stress evaluations, a reduced range seismic event, as identified in DCD Tier 2, Sections 3.7.3.2 and 3.9.3.1.1, is required for the ASME Code secondary stress and fatigue evaluations of Class 1 piping components. DCD Tier 2, Section 3.6.2.1.1.1 discusses the criteria for postulating intermediate breaks in Class 1 piping. Specifically, intermediate pipe break locations are postulated based on conditions excluding seismic loading. The staff finds these criteria acceptable and concludes that the applicant's criteria are consistent with the staff position on OBE elimination and are thus acceptable.

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 DCD does not include a description of the analysis methods and criteria for the design of welded attachments to piping for the AP1000. However, DCD Tier 2, Table 5.2-3 provides a listing of the ASME Code cases that would be used for this design purpose. As discussed in Section 3.12.2.2 of this report, in response to RAI 210.030, the applicant identified five ASME Code cases that are applicable to AP1000 piping and pipe support design. The staff reviewed the Code cases and determined that four are pertinent to the design of welded attachments. They include ASME Code Cases N-122-2, N-318-5, N-391-2, and N-392-3. Section 3.12.2.2 of this report presents the staff evaluation and acceptability of these Code cases.

3.12.5.16 Modal Damping for Composite Structures

In DCD Tier 2, Section 3.7.3.15, the applicant discussed methods and various applications for composite modal damping. For subsystems that are composed of different material types, the composite modal damping approach with the weighted stiffness method is used to determine the composite modal damping value. Composite modal damping for coupled building and piping systems is used for piping systems that are coupled to the primary coolant loop system and the interior concrete building. Composite modal damping is used for piping systems that are coupled to flexible equipment or flexible valves. DCD Tier 2, Section 3.7.1.3 describes the use of composite damping when piping systems and nonsimple module steel frames are in a single coupled model.

As discussed in Section 3.12.5.4 of this report, the staff noted an inconsistency between DCD Tier 2, Sections 3.7.3.15 and 3.7.1.3, in regard to the method for determining the composite modal damping value. DCD Tier 2, Section 3.7.1.3, indicates that the strain energy method is used, while DCD Tier 2, Section 3.7.3.15 indicates that the weighted stiffness method is used. In response to RAI 210.041, the applicant agreed to revise DCD Tier 2, Section 3.7.1.3, to indicate that the stiffness-weighted method is used. As noted in Section 3.12.5.4 of this report, the staff finds the application of the composite modal damping approach with the weighted stiffness method as described in the DCD consistent with the acceptance criteria given in SRP Section 3.7.2 and, therefore, are acceptable.

3.12.5.17 Minimum Temperature for Thermal Analyses

DCD Tier 2, Section 3.9.3.1.5, discusses the analysis of ASME Class 1, 2, and 3 piping systems including loads, load combinations, and stress limits including piping thermal expansion loadings. It states that thermal expansion analysis is needed to obtain the stresses and loadings above the stress-free reference temperature. The stress-free reference temperature for a piping system is defined as a temperature of 21.1 °C (70 °F). This value is consistent with industry practice and is acceptable to the staff. If the piping system operating temperature is 65.6 °C (150 °F) or less, no thermal expansion analysis is necessary. If the piping system does not contain at least one 90-degree bend, then thermal expansion analysis is needed. This type of layout is avoided when practical. The thermal anchor displacements are also considered as negligible if they are 0.158 cm (0.0625 in.) or less. This is consistent with the industry practice illustrated in EPRI NCIG-05 and Welding Research Council (WRC) Bulletin 353 that permits a 0.158 cm (0.0625 in.) gap at a pipe support.

The staff reviewed the criteria described above and noted that the value of minimum temperature for thermal expansion analysis is consistent with industry practice and is acceptable to the staff.

3.12.5.18 Intersystem Loss-of-Coolant Accident

Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. In SECY-90-016, dated January 12, 1990, the NRC staff discussed the resolution of the ISLOCA issue for ALWR plants by recommending that low-pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its SRM

Design of Structures, Components, Equipment, and Systems

dated June 26, 1990, the Commission approved these staff recommendations provided that all elements of the low-pressure systems are considered.

DCD Tier 2, Section 1.9.5.1.7, "Intersystem LOCA," addresses the ISLOCA issue. It states that the AP1000 has incorporated various design features to address ISLOCA challenges. These design features result in very low AP1000 core damage frequency for ISLOCAs compared with operating nuclear power plants. The design features, discussed in Section 3 of WCAP-14425 as well as in DCD Tier 2, Section 5.4.7, "Normal Residual Heat Removal System," are primarily associated with the RNS. The applicant prepared WCAP-14425 to document the evaluation of the AP600 for conformance to the ISLOCA regulatory criteria identified in various NRC documents. The AP1000 has a fluid system design similar to that of the AP600; therefore, the conclusions of WCAP-14425 also apply to the AP1000. As a result of the evaluation documented in WCAP-14425, the AP600 also incorporated additional design features. The AP1000 includes these design features, which are documented in other portions of the DCD.

Based on a review of the information in DCD Tier 2, Sections 1.9.5.1.7 and 5.4.7, the staff concludes that the piping design for ISLOCA meets regulatory guidance and is adequately considered for the AP1000. Section 3.9.3.1 of this report presents a more detailed discussion.

3.12.5.19 Conclusions

The staff finds that the AP1000 DCD adequately addresses the piping issues identified above and reflects the staff's position as indicated. Therefore, the staff concludes that the applicant 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 functions to be performed and with appropriate quality control

• GDC 2 and 10 CFR Part 50, Appendix S 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 RCPB 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 Design Criteria

The staff reviewed the methodology used in the design of ASME Code Class 1, 2, and 3 pipe supports as described in DCD Tier 2, Section 3.9.3.4, "Component and Piping Supports." The following sections summarize the staff's evaluation of the pipe support design methods, procedures, and criteria.

3.12.6.1 Applicable Codes

In DCD Tier 2, Section 3.9.3.4, the applicant stated that for the AP1000, ASME Code, Section III, Class 1, 2, and 3 component supports including pipe supports satisfy the requirements of Subsection NF of Section III of the ASME Code. The criteria of Appendix F to the ASME Code, Section III, are used for the evaluation of Service Level D conditions. In addition, the welded connections of ASTM A500 Grade B tube steel members met the standards of the Structural Welding Code, ANSI/AWS D1.1, Section 10. Because these codes are commonly used in pipe support design in nuclear power plants and the staff has found them acceptable for the design of pipe supports in previous applications, they are acceptable for the AP1000.

3.12.6.2 Jurisdictional Boundaries

In DCD Tier 2, Section 3.9.3.4, the applicant defined 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, therefore, are acceptable.

3.12.6.3 Loads and Load Combinations

In DCD Tier 2, Section 3.9.3.4, the applicant provided the loading conditions and combinations for the design of piping supports in DCD Tier 2, Tables 3.9-3 and 3.9-8, respectively. DCD Tier 2, Tables 3.9-9 and 3.9-10 present the stress limits for the various ASME Code service levels. The stress limits for pipe supports are in accordance with Subsection NF of Section III of the ASME Code. The criteria of Appendix F to Section III of the ASME Code are used for the evaluation of Service Level D conditions. When supports for components not built to ASME Code, Section III, criteria are evaluated for the effect of Level D service conditions, the allowable stress levels are based on tests or accepted industry standards comparable to those in Appendix F to ASME Code, Section III. In order to provide for operability of active equipment, including valves, the supports of these items meet ASME limits for Service Level C loadings.

The staff reviewed the loads, loading combinations, and stress criteria for piping supports in the DCD sections and tables referenced above and determined that they provide adequate margins of safety under all combinations of loading. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service limit for each loading combination meets the criteria in the ASME Code, Section III, Subsection NF and is, thus, acceptable for AP1000.

Design of Structures, Components, Equipment, and Systems

3.12.6.4 Pipe Support Baseplate and Anchor Bolt Design

NRC BL 79-02, Revision 2, dated November 8, 1979, describes the staff position on pipe support baseplate and anchor bolt design. This document provides the factor-of-safety for anchor bolts and states that baseplate flexibility should be accounted for in the calculation of concrete anchor bolt loads. The factor-of-safety apply to all types of expansion anchor bolts (including undercut type anchor bolts), unless justification for alternative safety factors is provided.

In DCD Tier 2, Section 3.9.3.4, the applicant indicated that the AP1000 minimizes the use of baseplates with concrete expansion anchors. However, pipe supports may use concrete expansion anchors. For these pipe support baseplate designs, accounting for the baseplate flexibility in the calculation of anchor bolt loads meets the baseplate flexibility recommendations of Inspection and Enforcement (IE) BL 79-02, Revision 2, dated November 8, 1979. DCD Tier 2, Section 3.8.4.5.1, outlines supplemental criteria for fastening anchor bolts to concrete.

The staff reviewed the information in DCD Tier 2, Section 3.9.3.4, and finds that the applicant committed to account for the baseplate flexibility in the calculation of anchor bolt loads. Accordingly, the staff concludes that these loads are calculated acceptably. In reviewing DCD Tier 2, Section 3.8.4.5.1, the staff noted that the applicant specifies the design of fasteners to concrete to be in accordance with ACI-349-01, Appendix B. Section 3.8.4.2 of this report contains the staff's evaluation of these criteria.

3.12.6.5 Use of Energy Absorbers and Limit Stops

In DCD Tier 2, Section 3.7.3.8.4, "Piping Systems With Gapped Supports," the applicant discussed the use of rigid gapped supports (limit stops) in AP1000 piping systems. These supports may be used to minimize the number of pipe support snubbers and the corresponding inservice testing and maintenance activities. The analysis consists of an iterative response spectra analysis of the piping and support system. The iterations establish calculated piping displacements that are compatible with the stiffness and gap of the rigid gapped supports. The applicant indicates that test data support the results of the computer program GAPPIPE, which uses this methodology. The basis of the concept is to find an equivalent linear spring with a response-dependent stiffness for each nonlinear rigid gapped support, or limit stop, in the mathematical model of the piping system. The equivalent linearized stiffness minimizes the mean difference in force in the support between the equivalent spring and the corresponding original gapped support. The mean difference is estimated by an averaging process in the time domain, that is, across the response duration, using the concept of random vibration.

The staff had previously reviewed the GAPPIPE program and prepared 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. The staff reviewed the methodology, as well as the conditions for applicability for use, proposed by the applicant and determined that it is consistent with the staff position and is appropriate for the AP1000. Therefore, the staff concludes that the GAPPIPE methodology that will be used in the design and analysis of gapped supports (limit stops) is acceptable for the AP1000.

3.12.6.6 <u>Use of Snubbers</u>

In DCD Tier 2, Section 3.9.3.4.3, the applicant summarized the considerations for snubbers used as piping supports, including design criteria and analytical considerations, modeling techniques, operational and performance testing, and maintenance standards, as follows:

> The location and size of the snubbers are determined by stress analysis. Access for the testing, inspection, and maintenance of snubbers is considered in the AP1000 layout. The location and line of action of a snubber are selected based on the necessity of limiting seismic stresses in the piping and nozzle loads on equipment. Snubbers are chosen in lieu of rigid supports where restricting thermal growth would induce excessive thermal stresses in the piping or nozzle loads on equipment. Snubbers that are designed to lock up at a given velocity are specified with lock-up velocities sufficiently large to envelope the highest thermal growth rates of the pipe or equipment for design thermal transients. The snubbers are constructed to ASME Code, Section III, Subsection NF standards.

> [*In the piping system seismic structural analysis, the snubbers are modeled as stiffness elements. The stiffness value is based on vendor stiffness data for the snubber, snubber extension, and pipe clamp assembly.*]* Supports for active valves are included in the overall design and qualification of the valve...

> ...Design specifications for snubbers include:

> - Seismic requirements
> - Normal environmental parameters
> - Accident/postaccident environmental parameters
> - Full-scale performance test to measure pertinent performance requirements
> - Instructions for periodic maintenance (in technical manuals)

> Two types of tests will be performed on the snubbers to verify proper operation:

> - Production tests, including dynamic testing, on every unit to verify proper operability

> - Qualification tests on randomly selected production models to demonstrate the necessary load performance (load rating)

> The production operability tests for large hydraulic snubbers (that is, those with capacities of [344.7 MPa (50 Kips)] or greater) include 1) a full Level D load test to verify sufficient load capacity; 2) testing at full load to verify proper bleed with the control valve closed; 3) testing to verify the control valve closes within the specified velocity range; and 4) testing to demonstrate that breakaway and drag loads are within the design limits.

> The operability of essential snubbers is verified by the COL applicant by verifying the proper installation of the snubbers, and performing visual inspections and measurements of the cold and hot positions of the snubbers as needed during plant heatup to verify the snubbers are performing as intended...

Design of Structures, Components, Equipment, and Systems

The staff finds that the requirements described above are consistent with applicable portions of Section 3.9.3 of the SRP and are acceptable for the AP1000.

3.12.6.7 Pipe Support Stiffnesses

In DCD Tier 2, Section 3.9.3.4, the applicant discussed pipe support stiffness values and support deflection limits used in the piping analyses and support designs. The stiffness of the pipe support miscellaneous steel is controlled so that support deformation does not adversely affect component nozzle loads. Pipe support miscellaneous steel deflections are limited, for dynamic loading, to 0.3175 cm (0.125 in.) in each restrained direction. The dynamic loading combinations considered are those associated with the Level D service limits given in DCD Tier 2, Table 3.9-8. These deflections are defined with respect to the structure to which the miscellaneous steel is attached. In this case, a generic stiffness value, one for each pipe size, represents the pipe support and miscellaneous steel in the piping system analysis. This represents the minimum rigid stiffness values and is used for fabricated supports, and vendor stiffness values are used for standard supports such as snubbers and rigid gapped supports. Alternately, if the deflection for dynamic loading exceeds 0.3175 cm (0.125 in.), the pipe support and miscellaneous steel are represented by elements with calculated stiffness values in the piping system analysis.

The staff reviewed the standards described above and determined that they are reasonable, consistent with industry practices for establishing appropriate stiffness values for pipe supports as presented in Welding Research Council (WRC) Bulletin 353. On this basis, the staff concludes that these standards are acceptable for the AP1000.

3.12.6.8 Seismic Self-Weight Excitation

In DCD Tier 2, Section 3.9.3.4, the applicant stated 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. DCD Tier 2, Table 3.9-8, indicates that for Level D load combinations, the SSE self-weight excitation is combined with SSE inertia and anchor motion by the absolute sum method. The staff notes that this is the same method that was approved for the AP600. The staff concludes that this method results in consideration of service loading combinations resulting from postulated events and the designation of appropriate service limits for pipe support seismic loads and is consistent with SRP Section 3.9.3, and is, therefore, acceptable.

3.12.6.9 Design of Supplementary Steel

In DCD Tier 2, Section 3.9.3.4, the applicant stated that pipe supports are designed in accordance with Subsection NF of Section III of the ASME Code. This includes supplementary steel within the jurisdictional boundary of Subsection NF. Subsection NF is an appropriate standard developed by a professional society and voluntary consensus standards organization. As it provides adequate guidelines for the design of structural steel for use as pipe supports, the staff finds it acceptable. In addition, as Section 3.12.6.1 of this report discusses, for ASTM A500 Grade B tube steel members, the weld criteria of AWS D1.1, "Structural Welding Code," will supplement the NF requirements. The staff finds that the use of these criteria for the design of

AP1000 supplementary steel provides reasonable assurance of the structural integrity of the supports and is thus acceptable.

3.12.6.10 Consideration of Friction Forces

In DCD Tier 2, Section 3.9.3.4, the applicant addressed the consideration of friction forces in the AP1000 pipe support design. It stated that friction loads induced by the pipe on the support should be considered in the analysis of sliding type supports, such as guides or box supports, when the resultant unrestrained thermal motion is greater than 0.159 cm (0.0625 in.). The friction force is equal to the coefficient of friction times the pipe support load, and it acts in the direction of pipe movement. A coefficient of friction of 0.35 for steel-on-steel sliding surfaces shall be used. If a self-lubricated bearing plate is used, a 0.15 coefficient of friction will be used. The pipe support load from which the friction force is developed includes only deadweight and thermal loads. The friction force cannot be greater than the product of the pipe movement and the stiffness of the pipe support in the direction of movement. The staff notes that the coefficients of friction are reasonable values commonly used in the nuclear industry. They have been validated through years of design experience and therefore are acceptable for use in the AP1000 design.

3.12.6.11 Pipe Support Gaps and Clearances

DCD Tier 2, Section 3.9.3.4, provides information on pipe support gaps and clearances. The applicant stated that small gaps are provided for frame-type supports built around the pipe. The gaps allow for radial thermal expansion of the pipe and for pipe rotation. The minimum gap (total of opposing sides) between the pipe and the support equals the diametrical expansion of the pipe due to temperature and pressure. The maximum gap equals the diametrical expansion of the pipe due to temperature and pressure plus 0.159 cm (0.0625 in.). The staff notes that this practice is consistent with standard industry practice that the staff has found to be acceptable on a generic basis and therefore acceptable for use in the design of the AP1000 pipe support.

3.12.6.12 Instrumentation Line Support Criteria

In DCD Tier 2, Section 3.9.3.5, "Instrument Line Supports," the applicant provided design criteria for instrumentation line supports. It states that the design loads, load combinations, and acceptance criteria for safety-related instrumentation supports are similar to those for pipe supports. Design loads include deadweight, thermal, and seismic loads (as appropriate). The acceptance criteria are based on the design rules in the ASME Code Section III, Subsection NF. The staff notes that the use of pipe support design criteria for instrumentation line supports provides a conservative design and utilizes standards developed by a professional society and voluntary consensus standards organization, which are acceptable to the staff. Therefore, these criteria are acceptable for use in the design of the AP1000 instrumentation line supports.

3.12.6.13 Pipe Deflection Limits

DCD Tier 2, Section 3.9.3.4, provides pipe deflection limits for standard component pipe supports. The applicant stated that for standard component pipe supports, all manufacturers' functional limitations (for example, travel limits and sway angles) should be followed. This criterion applies to limit stops, snubbers, rods, hangers, and sway struts. Snubber settings

Design of Structures, Components, Equipment, and Systems

should be chosen such that pipe movement occurs over the midrange of the snubber travel. Some margin should be provided between the expected pipe movement and the maximum or minimum snubber-stroke to accommodate construction tolerance. The staff finds these requirements acceptable, because they assume that the component movement will remain within intended design limits of the component supports, thus ensuring the functionality of supports.

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 that the applicant satisfies the following:

•   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.

•   the requirements of GDC 2 and 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.

•   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 assure adequate design of all safety-related piping and pipe supports in the AP600 for their safety functions.

•   10 CFR Part 52 requirements by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design. Through the performance of the ITAAC, the COL holder will verify the implementation of these preapproved methods and satisfaction of the acceptance criteria. This will assure that the as-constructed piping systems conform to the certified design for their safety functions.

•   10 CFR Part 50, Appendix S, requirements by designing the safety-related piping systems with a reasonable assurance that they will 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 = $1.15 \times S_y \times (P_d/P_v)/S$