CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES (Continued)

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 CONFORMANCE WITH NUCLEAR REGULATORY COMMISSION GENERAL DESIGN CRITERIA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.2.1 SEISMIC CLASSIFICATION

Add the following information to the end of DCD Subsection 3.2.1:

STD SUP 3.2-1 There are no safety-related structures, systems, or components outside the scope of the DCD. The nonsafety-related structures, systems and components outside the scope of the DCD are classified as non-seismic (NS).

3.2.2 AP1000 CLASSIFICATION SYSTEM

Add the following information to the end of DCD Subsection 3.2.2:

STD SUP 3.2-1 There are no safety-related structures, systems, or components outside the scope of the DCD.

3.3 WIND AND TORNADO LOADINGS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.3.1.1 Design Wind Velocity

Add the following information to the end of DCD Subsection 3.3.1.1:

WLS COL 3.3-1 The wind velocity characteristics for Lee Nuclear Station Units 1 and 2, are given in Subsection 2.3.1.2.8. These values are bounded by the design wind velocity values given in DCD Subsection 3.3.1.1 for the AP1000 plant.

3.3.2.1 Applicable Design Parameters

Add the following information to the end of DCD Subsection 3.3.2.1:

WLS COL 3.3-1 The tornado characteristics for Lee Nuclear Station Units 1 and 2 are given in Subsection 2.3.1.2.2. These values are bounded by the tornado design parameters given in DCD Subsection 3.3.2.1 for the AP1000 plant.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

Add the following information to the end of DCD Subsection 3.3.2.3:

STD COL 3.3-1 Consideration of the effects of wind and tornado due to failures in an adjacent AP1000 plant are bounded by the evaluation of the buildings and structures in a single unit.

3.3.3 COMBINED LICENSE INFORMATION

Add the following information to the end of DCD Subsection 3.3.3:

WLS COL 3.3-1 The Lee Nuclear Station site satisfies the site interface criteria for wind and tornado (see Subsections 3.3.1.1, 3.3.2.1, and 3.3.2.3) and does not have a wind or tornado-initiated failure of structures and components within the applicant's scope that compromises the safety of AP1000 safety-related structures and components (see also Subsection 3.5.4).

3.4 WATER LEVEL (FLOOD) DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.4.1.3 Permanent Dewatering System

Add the following information to the end of DCD Subsection 3.4.1.3:

WLS COL 3.4-1 No permanent dewatering system is required because site groundwater levels are two feet or more below site grade level as described in Subsection 2.4.12.5.

3.4.3 COMBINED LICENSE INFORMATION

Replace the first paragraph of DCD Subsection 3.4.3 with the following information:

WLS COL 3.4-1 The site-specific water levels given in Section 2.4 satisfy the interface requirements identified in DCD Section 2.4.

3.5 MISSILE PROTECTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.5.1.3 Turbine Missiles

Add the following information to the end of DCD Subsection 3.5.1.3:

STD SUP 3.5-1 The potential for a turbine missile from another AP1000 plant in close proximity has been considered. As noted in DCD Subsection 10.2.2, the probability of generation of a turbine missile (or P1 as identified in SRP 3.5.1.3) is less than 1 x 10^{-5} per year. This missile generation probability (P1) combined with an unfavorable orientation P2xP3 conservative product value of 10^{-2} (from SRP 3.5.1.3) results in a probability of unacceptable damage from turbine missiles (or P4 value) of less than 10^{-7} per year per plant which meets the SRP 3.5.1.3 acceptance criterion and the guidance of Regulatory Guide 1.115. Thus, neither the orientation of the side-by-side AP1000 turbines nor the separation distance is pertinent to meeting the turbine missile generation acceptance criterion. In addition, the shield building and auxiliary building walls, roofs, and floors provide further conservative, inherent protection of the safety-related SSCs from a turbine missile.

STD SUP 3.5-2 The turbine system maintenance and inspection program is discussed in Subsection 10.2.3.6.

3.5.1.5 Missiles Generated by Events Near the Site

Add the following information to the end of DCD Subsection 3.5.1.5:

WLS COL 3.5-1 The gate house, administrative building, security control building, warehouse and shops, water service building, diesel-driven fire pump / enclosure, and miscellaneous structures are common structures that are at a nuclear power plant. They are of similar design and construction to those that are typical at nuclear power plants. Therefore, any missiles resulting from a tornado-initiated failure are not more energetic than the tornado missiles postulated for design of the AP1000.

The missiles generated by events near the site are discussed and evaluated in Subsection 2.2.3. The effects of external events on the safety-related components of the plant are insignificant. The pressure effect of potential explosions in the vicinity of the plant site are estimated to result in less than 1 psi overpressure on plant structures. Therefore, these potential explosions do not represent design basis events per Regulatory Guide 1.91.

3.5.1.6 Aircraft Hazards

Add the following information to the end of DCD Subsection 3.5.1.6:

WLS COL 3.5-1 The approach and methodology outlined in NUREG-0800 Standard Review Plan (SRP) 3.5.1.6, "Aircraft Hazards," have been used in the calculation of the probability of an aircraft crash into the effective plant areas of the safety related structures on the site. In accordance with SRP 3.5.1.6, if the plant-to-airport distance (D) is between five and ten statute miles, and the projected annual number of operations is less than $500D^2$, or the plant-to-airport distance is greater than ten statute miles, and the projected annual number of operations is less than $1000D^2$, the aircraft hazard probability does not need to be calculated because it is considered to be less than an order of magnitude of 10^{-7} per year. If the plant is at least two statute miles beyond the nearest edge of a Federal airway, holding pattern, or approach pattern, the order of magnitude is considered 10^{-7} per year according to SRP 3.5.1.6, and the aircraft hazard probability does not need to be calculated. The aircraft handling facilities and air routes are described in Subsection 2.2.2.7. The aircraft hazard probability developed from the total probability of an aircraft crash into the effective areas of the plant does not constitute a design basis event. The probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR Part 100 exposure guidelines is based on the following:

- Charlotte/Douglas International Airport is located about 34.4 miles from Lee Nuclear Station. The average number of operations is approximately 502,152 operations per year, which is less than the acceptable projected annual number of operations of 1,183,360. Based on forecast for terminal area by Federal Aviation Administration (FAA), the number of Charlotte/Douglas International Airport operations for year 2025 is 767,691 operations per year. Assuming annual compound growth rate of two percent after year 2025, the acceptable projected annual number of operations of 1,183,360 will be reached at year 2046. This increases the number of annual operations over 236 percent from year 2007. Significant expansion of the existing airport facility or construction of a new airport will be required to accommodate this large an increase of air traffic. Thus, the aircraft hazard from this airport to the site is acceptable based on the maximum aircraft activity expected without significant changes to the airport facility.

- One federal airway passes within four miles of the plant site. Low altitude Airway V54 runs between Spartanburg Downtown Memorial Airport, South

Carolina (SPA) located 26.1 miles from Lee Nuclear Station and Charlotte/ Douglas International Airport, North Carolina (CLT) located 34.4 miles from Lee Nuclear Station. The average annual number of flights using Airway V54 is approximately 15 to 25 percent of the total airport operation. The FAA forecast number of Spartanburg Downtown Memorial Airport operation for year 2025 is approximately 73,000 operations per year. Based on annual compound growth rate of one percent from year 2025 to year 2060 for Spartanburg Downtown Memorial airport, the projected annual number of operations at year 2060 is approximately 103,412. The average annual number of flights for Airway V54 is assumed to be 25 percent of the total airport operation. Therefore, the annual number of flights for Airway V54 is assumed to be 25,853.

- No airports having more than $500D^2$ movements per year are located within 10 miles of the site and no airports having more than $1000D^2$ movements per year are located beyond 10 miles of the site.

- There are no military training routes within 10 miles of the site.

There is one private-use heliport, one private-use ultra-light aircraft airport, and one private-use single-engine airport within a twenty mile range from the site. Because these privately-owned heliport and airports are used for small aircraft, which are low weight, low airspeeds, and low penetration capability, these helicopters and light aircraft are not considered a significant hazard to the nuclear plant.

The analysis conservatively shows that the total probability of aircraft accidents that hit safety-related structures is less than 1.8×10^{-7} per year. This result meets the NRC staff objective of an order of magnitude of 10^{-7} per year, as stated in SRP 3.5.1.6 for meeting the requirements of 10 CFR Part 100 exposure guidelines. In addition, if the expected rate of exposure is an order of magnitude of 10^{-6} per year, and it can be shown with rigorous analysis, using realistic assumptions and reasonable arguments that the estimated probability could be lower, then, in accordance with SRP 2.2.3, it is acceptable.

The following conservatisms used in the analysis are summarized below:

- The only safety-related structures of the AP1000 design are the containment and the auxiliary building. The effective area of these structures is determined using a conservative model for each structure; these areas are added together. The containment was modeled as a rectangle with length and width equal to the diameter of the containment. This assumption will result in diagonal length of the containment greater than the actual diameter of the containment. The area and the diagonal length of the auxiliary building assume that the building is rectangular and does not take credit that some of the area is containment. Credit is not taken for the overlap in these structures.

- The above total aircraft hazard probability at the site is obtained from aircraft crashing from the low altitude federal Airway V54 into the site. This low altitude route is primarily flown by small, light general aviation aircraft. Light general aviation aircraft are not considered a significant hazard to nuclear power stations because of their low airspeeds, short distance landing capability, high maneuverability and low penetration capability. In addition, the nuclear plant site is not an attractive emergency landing area. Plant protective features against tornado missiles, the inherent strength of the safety-related systems and structures such as containment and auxiliary building, as well as the diversity and redundancy of plant systems reduce the potential hazards to the facility from light aircraft operations to acceptably low levels.

- The heading of the crashing aircraft with respect to the facility is assumed to be the worst case perpendicular to the diagonal of the bounding rectangle regardless of direction of actual flights.

- Credit is not taken for nearby cooling towers, building structures, transmission lines, natural terrain features, etc. that would reduce the effective area of the safety related structures and prevent many disabled aircraft from reaching the critical structures.

As a result of the above conservatisms in the analysis, the aircraft crash hazard probability calculated for Lee Nuclear Station can qualitatively shown to be much lower than the calculated value. Therefore, the aircraft hazards at Lee Nuclear Station pose no undue risk to the health and safety of the public.

3.5.4 COMBINED LICENSE INFORMATION

WLS COL 3.5-1 This COL Item is addressed in Subsections 3.5.1.5 and 3.5.1.6.

3.6 PROTECTION AGAINST THE DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.6.4.1 Pipe Break Hazard Analysis

Replace the last paragraph in DCD Subsection 3.6.4.1 with the following information:

STD COL 3.6-1 The as-designed pipe rupture hazards evaluation is made available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets protected by associated mitigation features (reference is DCD Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate.

A pipe rupture hazards analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The report will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design.

The pipe whip restraint and jet shield design includes the properties and characteristics of procured components connected to the piping, components, and walls at identified break and target locations. The design will be completed prior to installation of the piping and connected components.

The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load (in accordance with DCD Tier 1 Table 3.3-6, Item 8).

This COL item is also addressed in Subsection 14.3.3.

3.6.4.4 Primary System Inspection Program for Leak-before-Break Piping

Replace the first paragraph of DCD Subsection 3.6.4.4 with the following information:

STD COL 3.6-4 Alloy 690 is not used in leak-before-break piping. No additional or augmented inspections are required beyond the inservice inspection program for leak-before-break piping. An as-built verification of the leak-before-break piping is required to verify that no change was introduced that would invalidate the conclusion reached in this subsection.

3.7 SEISMIC DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add Subsection 3.7.1.1.1 as follows:

3.7.1.1.1 Design Ground Motion Response Spectra

WLS SUP 3.7-3 Design ground motion response spectra for Lee Nuclear Station Unit 1 and Unit 2 nuclear islands are presented in this subsection. The foundation conditions at Lee Nuclear Station are unique in that the Unit 1 nuclear island foundation is supported on new and previously placed concrete materials placed directly over continuous rock. In contrast, the Unit 2 nuclear island foundation is configured more conventionally with the nuclear island founded directly over continuous rock. Based on these foundation conditions, individual design ground motion response spectra are provided for the certified design portion of the plant at Units 1 and 2.

Measured shear wave velocities for continuous rock underlying the Units 1 and 2 nuclear islands range from between 9000 to 10,000 fps, as described in Subsection 2.5.4.7. The stability of subsurface materials including foundation conditions are described in Subsection 2.5.4.

Figures 3.7-201 and 3.7-202 compare the Units 1 and 2 horizontal and vertical site-specific design ground motion response spectra to the certified seismic design response spectrum (CSDRS) and the AP1000 generic hard rock spectrum (WEC). For Unit 1, the Foundation Input Response Spectrum (FIRS) defines the site response foundation input motion for the nuclear island foundation placed on concrete over continuous rock. Unit 1 FIRS represents the nuclear island centerline foundation input motion and is based on the GMRS developed at the top of a hypothetical outcrop (e.g. continuous rock) fixed at 530 feet (NAVD) transferred up through previously placed and new concrete materials to the basemat foundation level at 550.5 feet (NAVD). For Unit 2, the GMRS defines the site response foundation input motion developed at the top of a hypothetical outcrop of competent material (e.g. continuous rock) fixed at the basemat foundation level at 550.5 feet (NAVD).

Detailed discussions of the methods used to calculate the horizontal and vertical GMRS and FIRS are described in Subsections 2.5.2.6, Ground Motion Response Spectra, and 2.5.2.7, Development of Foundation Response Spectra (FIRS). Variations in the Unit 1 FIRS and GMRS horizontal and vertical spectrum shown on Figures 3.7-201 and 3.7-202 are attributed to the independent calculation methodologies used to estimate the site-specific design ground motion response spectra.

As shown on Figure 3.7-201, the horizontal GMRS and Unit 1 FIRS exceed the horizontal CSDRS at frequencies of about 20 to 75 hertz and 20 to 85 hertz, respectively. PGA at 100 hertz of the GMRS and Unit 1 FIRS is 0.21 g and 0.22 g, respectively. As shown on Figure 3.7-202, the vertical GMRS and Unit 1 FIRS exceed the vertical CSDRS at frequencies between about 25 to 70 hertz.

Similar high-frequency exceedances were evaluated by Westinghouse in DCD Appendix 3I using a standard hard rock spectrum (shown as WEC generic hard rock spectrum in Figures 3.7-201 and 3.7-202). In Figures 3.7-201 and 3.7-202, it can be seen that the horizontal and vertical GMRS and Unit 1 FIRS are below the corresponding horizontal and vertical WEC generic hard rock spectrum for all frequencies. As described in DCD Appendix 3I, generic hard rock spectrum high frequency exceedances are within the seismic design margin of the AP1000 and will not adversely affect the systems, structures, or components of the plant.

Subsection 2.5.4.7.4.1, including Figures 2.5.4-245, 2.5.4-246, and 2.5.4-264, describe and illustrate a localized area of non-uniform foundation conditions associated with the Unit 1 Northwest Corner, and the area adjacent to and outside the nuclear island foundation footprint. Westinghouse has evaluated (Reference 201) the potential effect of this condition on the dynamic response of the nuclear island, and has concluded that its effect on in-structure response spectra is small. At the six key nuclear island locations described in AP1000 DCD Appendix 3G, significant margin exists between the site-specific in-structure response spectra and that resulting from the AP1000 CSDRS.
Subsection 3.7.2.15 describes the site-specific analyses of the nuclear island to demonstrate compliance with the AP1000 DCD.

3.7.2.1.2 Time-History Analysis and Complex Frequency Response Analysis

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Add the following information to the end of DCD Subsection 3.7.2.1.2:

FSAR Subsection 3.7.1.1.1 describes site-specific analyses of Lee Nuclear Station Unit 1 to confirm that the effects of localized areas of non-uniform foundation conditions do not result in unacceptable in-structure responses. For use in these site-specific analyses of the nuclear island structures, artificial time histories (two horizontal and one vertical) were developed to be compatible with the Lee Nuclear Station Unit 1 FIRS spectrum (FSAR Figures 3.7-201 and 3.7-202), and to satisfy the requirements of Standard Review Plan (SRP) 3.7.1. The methodology used in the development of these time histories is summarized in the following four steps:

1. Select a real 3-component ground-motion record to use as a starting point. All components should be broad-banded and should have reasonable durations consistent with the magnitude and distance of the earthquake. The NSK record from the 1999 Chi-Chi, Taiwan earthquake (**M** 7.6, distance 64.5 km) was selected for this purpose. This record is part of the NRC Time History Library. and belongs to the WUS Rock, **M**>7, D=50-100 km bin.

2. Modify the time history for each component using spectral-matching software. This software is not QA controlled but the resulting time history is qualified because the matching requirements are checked (in step 3) using QA software, and visual examination of the results confirms that the resulting motions are realistic.

3. Check that each spectrally-matched component meets the requirements for Approach 2 in SRP 3.7.1, and generate other results required by SRP 3.7.1. A scaling factor may be used in this step to make minor adjustments.

4. Calculate the cross-correlation coefficients between the three components of acceleration and check that they do not exceed the criterion of |cross-correlation| <0.16.

Additionally, the following criteria are also applied:

a. Time step interval shall be no more than 0.005 seconds.

b. Total duration of the motion shall be no less than 30 seconds.

c. The strong motion duration shall be consistent with the magnitude and distance of interest.

d. The time histories of the three components shall be statistically independent. The cross-correlation shall not exceed 0.16.

Attributes of the resulting time histories are shown in FSAR Table 3.7-201. FSAR Figure 3.7-203 illustrates a representative horizontal component time history.

3.7.2.8.4 Seismic Modeling and Analysis of Seismic Category II Building Structures

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Add the following information to the end of DCD Subsection 3.7.2.8.4:

WLS SUP 3.7-4 The foundation conditions beneath most of the Unit 1 Annex Building are very uniform, and are in fact similar to those described in the AP1000 DCD, except that the fill material supporting the Annex Building is a few feet thicker. In the northernmost end of the Unit 1 Annex Building, the top-of-continuous-rock slopes away, but the overall character of the building support remains quite uniform. This is illustrated in FSAR Figures 2.5.4-246, 2.5.4-260, and 2.5.4-264. Since the entire Seismic Category II portion of the Annex Building is on a common base mat and will behave as a unit, these localized differences in the support conditions will not significantly affect overall response of the Unit 1 Annex Building, or the potential for interaction with the nuclear island.

FSAR Figure 2.5.4-260 also illustrates the support conditions beneath the Unit 2 Annex Building. Though final excavation profiles to support construction of Unit 2 have not been established, the foundation support provided by the existing rock excavation provides uniform support at a depth about ten feet less than the configuration described in the AP1000 DCD.

The foundation conditions beneath the Seismic Category II portion of the Unit 1 and Unit 2 Turbine Buildings are also very uniform and are in fact similar to those described in the AP1000 DCD, except that the supporting rock or fill concrete will be a few feet above the level considered for the standard design.

As shown in FSAR Subsection 3.7.1.1.1, the Lee GMRS and Unit 1 FIRS are enveloped by the AP1000 HRHF response spectrum. The properties of the granular fill material that will be placed above continuous rock, presented in FSAR Table 2.5.4-211 and FSAR Tables 2.5.4-224A through 2.5.4-224F, are consistent with those used by Westinghouse in developing design criteria for adjacent Seismic Category II structures and include having a shear wave velocity greater than 500 fps.

Lee site-specific performance-based surface response spectra at the ground surface of the granular fill supporting the adjacent Seismic Category II buildings have been developed, considering the effects of the different thicknesses of granular fill material beneath the adjacent buildings. For frequencies above 5 Hz, these site-specific spectra are generally lower than the AP1000 generic plant-grade spectra for a Hard Rock High Frequency site that were considered in developing design criteria for the Seismic Category II buildings and any isolated exceedances of the AP1000 generic plant-grade spectra would not result in any change to the standard design criteria for the AP1000 Seismic Category II buildings. For frequencies below 5 Hz, the design of Seismic Category II structures is governed by the CSDRS.

From the candidate granular fill materials described in FSAR Subsection 2.5.4, Duke Energy has determined that Macadam Base Course material provides properties appropriate for precluding interaction of Seismic Category II buildings with the nuclear island. Duke Energy has selected the static and dynamic properties described in FSAR Subsection 2.5.4 as well-graded gravel (GW) to represent that Macadam Base Course material.

Westinghouse has performed a site-specific analysis of Seismic Category II structures supported by granular fill material with the static and dynamic properties associated with well-graded gravel (GW), and has concluded that all DCD criteria have been met. This analysis is presented in Reference 205. The calculated site-specific relative displacements of adjacent buildings are less than the building separation, so there is no contact between the nuclear island and adjacent buildings. The calculated foundation input response spectra at the base of the Annex Building and at the base of the first bay of the Turbine Building are less than those considered in the AP1000 standard design of those structures. The maximum site-specific bearing demand (approximately 13.06 ksf for the Annex Building and 7.75 ksf for the Turbine Building) is significantly less than the site-specific allowable bearing pressure shown in FSAR Table 2.5.4-228 (approximately 32.05 ksf for the Annex Building and 43.74 ksf for the Turbine Building). The base shears and moments for those two structures are also significantly less than those considered in the AP1000 standard design of the Seismic Category II structures for the CSDRS.

As described in FSAR Subsection 2.5.4.5.1, the source for the granular fill material (Macadam Base Course) supporting the Seismic Category II buildings has not yet been identified. Once a source for the granular fill material has been selected, the static and dynamic properties of the material supporting Seismic Category II buildings will be verified as compatible with Lee Nuclear Station site response analyses.

The site-specific analysis presented in Reference 205 demonstrates that the Lee site provides uniform support for the Seismic Category II buildings; site-specific fill material is consistent with that considered in establishing generic AP1000 design criteria for these buildings; and the site-specific seismic demands on the Seismic Category II buildings are less than those considered in the AP1000 standard design.

3.7.2.12 Methods for Seismic Analysis of Dams

Add the following information to the end of DCD Subsection 3.7.2.12:

WLS COL 3.7-1 The evaluation of existing and new dams whose failure could affect the site interface flood level specified in DCD Subsection 2.4.1.2 is included in Subsection 2.4.4.

3.7.2.15 Site-Specific Analyses of Nuclear Island Seismic Category I Structures

Add the following information to the end of DCD Subsection 3.7.2:

WLS COL 2.5-3 To fully document the acceptability of the WLS site, Westinghouse has performed site-specific analyses of the nuclear island Seismic Category I structures. These analyses were initially documented in Revision 1 of Reference 201, and were subsequently updated in Revision 2 of Reference 201 to address AP1000 modeling updates during the Design Certification Amendment, revisions to the WLS Unit 1 Foundation Input Response Spectrum (FIRS) and the associated time-histories, and the decision to use granular fill material adjacent to the WLS nuclear island structures.

The site-specific analyses included a combination of two-dimensional (2D) and three-dimensional (3D) analyses. The 2D analyses were conducted to:

- Address the revised Unit 1 FIRS and associated time-history;
- Evaluate the extent of subsurface characterization, site response and surface motions;
- Analyze the site-specific dynamic profile and foundation medium underlying the Units 1 and 2 NI footprints;
- Assess the effect of the Unit 1 Northwest corner site-specific conditions on the seismic response; and
- Compare the 2D SSI results of the various site subsurface and foundation conditions, and determine the controlling conditions to be used in subsequent three-dimensional (3D) SSI analyses.

The 3D analyses updated the WLS 3D SSI model and analysis to include the AP1000 NI20r 3D Model, incorporated the results and parameters established in the 2D parametric studies and performed twenty-five (25) 3D SASSI incoherent simulation analyses.

The WLS 3D horizontal and vertical in-structure floor response spectra (FRS) were compared to the AP1000 3D Certified Seismic Design Response Spectra (CSDRS) and Hard Rock High Frequency (HRHF) FRS envelopes at six (6) key AP1000 NI locations.

The 2D SSI analyses were performed using the computer code SASSI2000 and post-processed using ACS SASSI. 3D Incoherent SSI analyses were performed using ACS SASSI. All SASSI SSI analyses performed used the SASSI Direct method for computing in-structure FRS.

Site-specific SSI analyses were performed using the AP1000 NI20r finite element model and the site-specific Foundation Input Response Spectra (FIRS) A1 time history inputs described in Subsection 3.7.2.1.2.

As described below, the 3D SSI analyses results show that the in-structure FRS of an AP1000 plant at the WLS Unit 1 & 2 sites are enveloped by the FRS from AP1000 CSDRS and HRHF analyses at the six key AP1000 NI locations described in DCD Subsection 3.7.2 and DCD Table 3G.4-1.

3.7.2.15.1 Site Characteristics

Foundation conditions for the two (2) WLS units are described in Subsection 2.5.4.7.4. The Unit 1 NI basemat is founded at El. 550.5 ft. msl (AP1000 El. 60.5 ft.), predominately on fill concrete over hard rock, and the Unit 2 NI basemat is founded at El. 550.5 ft. msl. on hard rock. The final grade level for both units is at El 590.0 ft. msl (AP1000 El. 100.0 ft.).

Unit 1 will overlie portions of the former Cherokee Nuclear Station fill concrete and legacy structural slabs and native rock. Similarly, Lee Unit 2 will occupy portions of the former Cherokee Unit 3 area, and will overlie native rock. Both nuclear island (NI) structures will require some additional minor excavation and replacement with fill concrete. In the Northwest corner of Unit 1, engineered fill will be placed adjacent to the fill concrete which will extend below the elevation of the AP1000 basemat.

The foundation conditions and geologic profiles vary between Units 1 and 2, and locally at the Northwest corner of Unit 1. A total of three (3) site-specific SSI models were developed with corresponding site dynamic profiles to represent the varied conditions and backfill beneath the Units 1 and Unit 2 NIs. Three cross-sections were modeled:

- Unit 1 Centerline Cross-Section B-B' (Figure 2.5.4-260);
- Unit 2 Centerline Cross-Section B-B' (Figure 2.5.4-260); and
- Unit 1 Northwest Corner Cross-Sections Y-Y' and U-U' (Figures 2.5.4-264 and 2.5.4-245).

Three dynamic profiles were developed to represent the conditions at each plant basemat, corresponding to:

- Unit 1 Centerline – Base Case A1 (Figure 2.5.4-252);
- Unit 2 Centerline – Profile C (Figure 2.5.4-250); and
- Unit 1 Northwest Corner – Profile B (Figure 2.5.4-249).

As shown in the Unit 1 Northwest Corner Cross-Sections U-U' (Figure 2.5.4-245) and Y-Y' (Figure 2.5.4-260), up to approximately 30 feet of engineered fill is required adjacent to the fill concrete beneath the NI (below El. 550.5), which replaces excavated lower shear wave velocity weathered rock down to continuous rock at the Northwest corner. Strain compatible dynamic soil properties were calculated for granular fill materials in three (3) representative profiles located within the Unit 1 Northwest corner. Calculation of these properties considered the three candidate engineered fill material types (GP, GW, and SW) described in Subsection 2.5.4, and a range of ground water conditions. The calculation results included 16th, median, and 84th percentile values for the dynamic soil properties. A range of average dynamic properties was determined, parametrically evaluated in the 2D SSI analysis of the Unit 1 Northwest corner, and the results (in-structure FRS) were enveloped.

Cross-Section B-B' (Figure 2.5.4-260) shows bedrock conditions on an East-West centerline of Unit 1 and Unit 2. The new Unit 1 NI basemat will be constructed over approximately five (5) feet of new fill concrete overlying an average of about 15 feet of existing fill concrete, structural basemat concrete and native rock from the former Cherokee foundation. The Unit 2 NI basemat is founded on native hard rock. The Unit 1 NI centerline rock shear wave velocity (Vs) ranges from about

7,500 feet per second (fps) (fill concrete) to about 9,600 fps (continuous rock) as shown in the Unit 1 Base Case A1 (Figure 2.5.4-252). The Unit 2 Centerline continuous rock Vs ranges from about 8,400 fps to about 9,600 fps as shown in the Unit 2 Profile C (Figure 2.5.4-250).

Cross-Sections Y-Y' and U-U' (Figures 2.5.4-264 and 2.5.4-245) represents bedrock conditions at the Northwest corner of the Unit 1 NI. In this area, the NI overlies a localized zone of weathered and fractured rock, extending approximately 15 to 25 feet deep below the Unit 1 basemat elevation (El. 550.5 ft.). This localized zone of weathered rock exhibits lower Vs velocities, ranging from approximately 4500 to 6000 fps, than the underlying and adjacent sound rock with Vs of approximately 9200 fps. Excavation of this isolated lower velocity material to continuous rock at the Northwest corner of Unit 1 NI will be replaced with fill concrete beneath the basemat. Engineered backfill will be placed and compacted adjacent to the fill concrete beneath the NI (and beneath the northern end of the Annex Building) approximately 20 to 30 feet below the NI basemat elevation. The Unit 1 rock shear wave velocity at the Northwest corner ranges from about 5,300 fps to about 9,200.

Because the rock and fill concrete materials were found to behave linearly in the development of site response spectra, a material damping value of 0.005 was used for rock and for fill concrete in all profiles. For the granular fill materials adjacent to the Northwest corner, damping values were determined as one of the strain-compatible material properties, and varied between approximately 0.05 and 0.10, depending on material type and depth.

3.7.2.15.2 Seismic Inputs

The horizontal and vertical site GMRS and Unit 1 Foundation Input Response Spectra (FIRS) are described in Subsection 3.7.1.1.1. Subsection 3.7.2.1.2 describes the development of artificial time histories to represent the Unit 1 FIRS, consistent with the guidance in Standard Review Plan 3.7.1. Since the Unit 1 FIRS bounds the site GMRS, which is the Unit 2 base motion, the Unit 1 FIRS time histories are conservatively used for the analysis of both Unit 1 and Unit 2.

Analysis of the AP1000 for the Certified Seismic Design Response Spectra (CSDRS) envelope is provided in Reference 202, and for the hard rock high frequency (HRHF) FRS envelope in Reference 203. The WLS 3D SSI in-structure FRS are compared to the AP1000 CSDRS and HRHF envelopes at the six key locations identified in DCD Subsection 3.7.2 and DCD Table 3G.4-1.

3.7.2.15.3 Two-Dimensional SASSI Parametric Studies

Two-dimensional (2D) parametric SSI analyses were performed using SASSI to compare the 2D SSI results of the various site subsurface and foundation conditions, and determine the controlling conditions to be evaluated in greater detail in subsequent 3D SSI analyses. The 2D East-West (EW) model typically yields a higher response than the north-south model as described in Section 6.2 of Westinghouse Technical Report TR03 (APP-GW-S2R-010, “Extension of

Nuclear Island Seismic Analysis to Soil Site", Reference 202). Therefore, the 2D EW model is used for these parametric studies.

The models consist of a 2D SASSI stick model of the AP1000 nuclear island that is used with three site-specific 2D SASSI finite element models representing three (3) cross-sections of interest for Units 1 and 2. The AP1000 Nuclear Island model includes three stick models representing the the Auxiliary Shield Building (ASB), the Steel Containment Vessel (SCV), and the Containment Internal Structure (CIS). The three (3) east-west cross-sections modeled are the Unit 1 NI centerline, the Unit 2 NI centerline, and the Unit 1 nuclear island Northwest corner. The SASSI Direct method is used to compute in-structure FRS. From the analyses using the 2D models, the important modes of the structure and seismic interaction between the NI structures and supporting media are obtained to evaluate the response of the three cross-sections.

The Unit 1 centerline east-west 2D SASSI finite element model includes the supporting medium up to the bottom of the basemat of the nuclear island. Consistent with DCD analyses of hard-rock conditions, the model does not include backfill material adjacent to the nuclear island above that level. The 2D model of the supporting medium has properties assigned to represent areas of continuous rock, legacy fill concrete and structural concrete remaining from the Cherokee construction, and new fill concrete to be used to bring the site to the level of the bottom of the nuclear island basemat. The Unit 2 centerline east-west 2D SASSI finite element model is constructed similarly, but the finite element properties are selected to represent the continuous hard rock supporting the Unit 2 nuclear island.

The Unit 1 Northwest corner east-west 2D SASSI model is constructed similarly, but is extended laterally so that the SASSI finite elements can represent not only the material types in the Unit 1 centerline model, but also materials and configurations that are unique to the Northwest corner. These include areas of continuous rock with lower shear wave velocity, thicker fill concrete layers, the irregular surface of the continuous rock, and the presence of weathered rock and granular fill outside the support zone of the nuclear island, but adjacent to the nuclear island and below the level of the bottom of the basemat.

The configuration of each of the site-specific models is selected with the objective that each material layer should have a passing frequency of approximately 50 Hz based on the material properties. The 2D SASSI analysis uses a 50 Hz cut-off frequency. Time-history seismic analyses of the three (3) east-west 2D SASSI models were performed considering simultaneous occurrences of one horizontal and one vertical component. The Unit 1 FIRS time history was input at the basemat bottom elevation. The 2D in-structure FRS results were combined algebraically using each directional analysis FRS to produce site-specific in-structure 5% damped horizontal and vertical spectra at each of the six (6) key locations identified in DCD Subsection 3.7.2 and DCD Table 3G.4-1. These six (6) locations are shown below. (Note that Lee North corresponds to AP1000 South.)

- 2D Node 4041, CIS at Reactor Vessel Support Elevation
- 2D Node 4061, ASB SW Corner at Control Room Floor
- 2D Node 4535, CIS at Operating Deck
- 2D Node 4120, ASB Corner of Fuel Building Roof at Shield Building
- 2D Node 4412, SCV Near Polar Crane
- 2D Node 4310, ASB Shield Building Roof Area

For the Northwest corner 2D SASSI analyses, individual analyses were conducted for the three (3) candidate granular fill materials (GP, GW, and SW), for a range of groundwater levels, and for the 16th, median and 84th percentile values of the strain-compatible dynamic properties, and the results enveloped for each granular fill type. The resulting site-specific in-structure FRS are shown in Figures 3.7-204a through 3.7-205c for these six (6) key locations.

It is important to note that the HRHF broad curve (envelope) is based on SASSI 3D analyses and includes seismic motion incoherency effects. The WLS 2D FRS does not include in the SSI analyses coherency functions. The purpose of the 2D SSI analyses was to evaluate the various cases for subsequent 3D SSI analyses and to assess potential FRS impacts from the NW corner subsurface conditions. Subsequent 3D analyses compare the WLS 3D FRS results with incoherency to the AP1000 HRHF envelope, also including incoherency. The following observations can be made from the 2D SASSI parametric analyses results:

- Consideration of the Unit 1 Northwest Corner configuration and materials results in a relatively small change in the calculated in-structure FRS compared to the Unit 1 Centerline model. Likewise, the selection of engineered fill (GP, GW or SW) to be used adjacent to the nuclear island also has a relatively small effect on the calculated in-structure FRS for the nuclear island.

- Only minor spectral acceleration differences are observed between the Unit 1 NI Centerline 2D FRS and the Unit 2 NI Centerline 2D FRS across the entire frequency spectrum in both the horizontal and vertical directions. The slight variation of the dynamic properties of Unit 1 situated on fill concrete versus Unit 2 founded on sound rock do not result in a appreciable difference in each respective model FRS; and

- Comparing the Unit 1 and Unit 2 NI Centerline 2D FRS to the AP1000 2D CSDRS and HRHF FRS envelopes suggests that above 20 Hz, the Unit 1 and 2 in-structure FRS exceed the AP1000 envelope FRS. As previously discussed, coherency functions were not applied to the WLS 2D parametric analyses. As demonstrated in the 3D analyses below, consideration of incoherency effects reduces the calculated in-structure FRS above 20 Hz.

Based on the results of the 2D parametric SSI analyses, subsequent 3D incoherent SSI analyses were performed using both the Unit 1 and Unit 2 NI Centerline cross-section models and the corresponding Base Case A1 and Profile C site dynamic profiles, respectively. In-structure FRS from 3D incoherent SSI analyses are compared to the AP1000 and HRHF 3D envelope spectra.

3.7.2.15.4 Three-Dimensional SASSI SSI Analyses

3D SASSI analyses were performed to demonstrate that the in-structure FRS of an AP1000 plant at the WLS site is enveloped by the AP1000 CSDRS and HRHF 3D envelopes at the six (6) NI key locations shown below. (Note that Lee North corresponds to AP1000 South.)

- 3D Node 1761, CIS at Reactor Vessel Support Elevation
- 3D Node 2078, ASB SW Corner at Control Room Floor
- 3D Node 2199, CIS at Operating Deck
- 3D Node 2675, ASB Corner of Fuel Building Roof at Shield Building
- 3D Node 2788, SCV Near Polar Crane
- 3D Node 3329, ASB Shield Building Roof Area

Since WLS is a hard-rock site, the Unit 1 FIRS spectra shown in Figures 3.7-201 and 3.7-202 exhibit a shape similar to the HRHF response spectra documented in Westinghouse Technical Report TR115 (Reference 204). Therefore, the same incoherent analysis methodology was used in the WLS site-specific 3D SASSI SSI analyses.

The 3D SASSI incoherent analyses were performed with the Unit 1 and Unit 2 NI Centerline soil profiles and the corresponding Unit 1 FIRS time history. The 3D SASSI model used is the AP1000 NI20r surface model that is described in Westinghouse Technical Report TR03 (APP-GW-S2R-010, Rev. 5, Reference 202). Consistent with AP1000 DCD analyses, reinforced concrete elements are assigned 7% damping, structural steel elements are assigned 4% damping, and concrete-filled steel plate (SC) structures are assigned 5% damping. The benchmarking of AP1000 NI10 and NI20r was documented in Westinghouse Technical Report TR03 Appendix C – Comparison of NI10 and NI20r Responses. Structural damping of 7% is used in the development of HRHF in-structure response spectra (ISRS).

The 3D incoherent analyses include performing 25 simulations of the Unit 1 and Unit 2 NI20r surface models with outcrop input time history. The coherency functions employed and the methods of analysis are consistent with COL/DC-ISG-1 (Reference 204) and are also consistent with those used in DCD-supporting analyses. The 3D incoherent analyses includes 25 simulations using the NI20r surface model, and were performed using the Unit 1 and Unit 2 NI

Centerline Base Case A1 and C site dynamic profiles, respectively, Unit 1 FIRS time history input at AP1000 El. 60.5 ft., and SASSI Direct method.

Similar to the 2D SASSI analyses, the configuration of each 3D SASSI layer was selected considering the SASSI wavelength criteria for 50 Hz. Since the Unit 1 and Unit 2 centerline profiles are comprised of fill concrete and hard rock, with Vs>7500 fps, this criteria is easily met. The 3D SASSI analyses use a 50 Hz cut-off frequency, and are consistent with the guidance in COL/DC-ISG-1.

The site-specific NI 3D SASSI results are shown in Figures 3.7-206a through 3.7-208c. The WLS Units 1 and 2 NI20r surface models were run through 25 simulations of incoherent 3D analysis using the three predefined direction-based Unit 1 FIRS time histories. The calculated 5% damping in-structure FRS at the six (6) key locations are enveloped by the AP1000 3D CSDRS and HRHF SSI envelopes with significant margin.

3.7.2.15.5 Site-Specific Analyses Conclusions

The site-specific analyses of the WLS nuclear islands led to the following conclusions:

- Consideration of the Unit 1 Northwest Corner configuration and materials results in a relatively small change in the calculated in-structure FRS compared to the Unit 1 Centerline model. Likewise, the selection of engineered fill (GP, GW or SW) to be used adjacent to the nuclear island also has a relatively small effect on the calculated in-structure FRS for the nuclear island.

- Only minor differences exist between the Unit 1 NI Centerline 2D FRS and the Unit 2 NI Centerline 2D FRS across the frequency spectrum in both the horizontal and vertical directions;

- The site-specific WLS 3D incoherent SSI analyses results for Unit 1 and Unit 2, which incorporate the AP1000 NI20r 3D model and revised Unit 1 FIRS time history, indicate that the 5% damping in-structure FRS at six (6) key NI locations are enveloped by the AP1000 CSDRS and HRHF SSI envelopes.

3.7.4.1 Comparison with Regulatory Guide 1.12

Add the following information to the end of DCD Subsection 3.7.4.1:

STD SUP 3.7-1 Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operation and shutdown in accordance with Regulatory Guide 1.12.

3.7.4.2.1 Triaxial Acceleration Sensors

Add the following information to the end of DCD Subsection 3.7.4.2.1:

STD COL 3.7-5 A free-field sensor will be located and installed to record the ground surface motion representative of the site. It will be located such that the effects associated with surface features, buildings, and components on the recorded ground motion will be insignificant. The trigger value is initially set at 0.01g.

3.7.4.4 Comparison of Measured and Predicted Responses

Add the following information to the end of DCD Subsection 3.7.4.4:

STD COL 3.7-2 Post-earthquake operating procedures utilize the guidance of EPRI Reports NP-5930, TR-100082, and NP-6695, as modified and endorsed by the NRC in Regulatory Guides 1.166 and 1.167. A response spectrum check up to 10Hz will be based on the foundation instrument. The cumulative absolute velocity will be calculated based on the recorded motions at the free field instrument. If the operating basis earthquake ground motion is exceeded or significant plant damage occurs, the plant must be shutdown in an orderly manner.

In addition, the procedures address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit, between the individual spent fuel racks, and from the spent fuel racks to the spent fuel pool walls, and provide for appropriate corrective actions to be taken if needed (such as repositioning the racks or analysis of the as-found condition).

3.7.4.5 Tests and Inspections

Add the following information to the end of DCD Subsection 3.7.4.5:

STD SUP 3.7-2 Installation and acceptance testing of the triaxial acceleration sensors described in DCD Subsection 3.7.4.2.1 is completed prior to initial startup. Installation and acceptance testing of the time-history analyzer described in DCD Subsection 3.7.4.2.2 is completed prior to initial startup.

3.7.5 COMBINED LICENSE INFORMATION

3.7.5.1 Seismic Analysis of Dams

WLS COL 3.7-1 This COL Item is addressed in Subsection 3.7.2.12.

3.7.5.2 Post-Earthquake Procedures

STD COL 3.7-2 This COL Item is addressed in Subsection 3.7.4.4.

3.7.5.3 Seismic Interaction Review

Replace DCD Subsection 3.7.5.3 with the following information:

STD COL 3.7-3 The seismic interaction review will be updated for as-built information. 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. The as-built seismic interaction review is completed prior to fuel load.

3.7.5.4 Reconciliation of Seismic Analyses of Nuclear Island Structures

Replace DCD Subsection 3.7.5.4 with the following information:

STD COL 3.7-4 The seismic analyses described in DCD Subsection 3.7.2 will be reconciled for detailed design changes, such as those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information. Deviations are acceptable based on an evaluation consistent with the methods and procedure of DCD Section 3.7 provided the amplitude of the seismic floor response spectra, including the effect due to these deviations, does not exceed the design basis floor response spectra by more than 10 percent. This reconciliation will be completed prior to fuel load.

3.7.5.5 Free Field Acceleration Sensor

STD COL 3.7-5 This COL Item is addressed in Subsection 3.7.4.2.1.

3.7.6 REFERENCES

201. Westinghouse Electric Company Report WLG-1000-S2R-802, Revision 2, William S. Lee Site Specific Seismic Evaluation Report, March 15, 2012.

202. Westinghouse Electric Company Report APP-GW-S2R-010, TR03 “Extension of Nuclear Island Seismic Analyses to Soil Sites,” Rev. 5, February 2011.

203. Westinghouse Electric Company Report APP-GW-GLR-115, TR115 “Effect of High Frequency Seismic Content on SSCs,” Rev.3, January 2011.

204. COL/DC-ISG-1, “Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications,” May 2008.

205. Westinghouse Electric Company Report WLG-1000-S2R-804, Revision 2, William S. Lee Site Specific Adjacent Building Seismic Evaluation Report, July 2012.

TABLE 3.7-201
SUMMARY OF CHARACTERISTICS OF ARTIFICIAL TIME HISTORIES REPRESENTING UNIT 1 FIRS

Parameter	Horizontal 1	Horizontal 2	Vertical
Duration (5-75%; sec)	12.9	12.4	15.5
PGA (g)	0.23	0.23	0.17
PGV (cm/sec)	8.8	8.7	7.4
PGD (cm)	7.6	7.0	4.9
PGD/PGA (cm/g)	33	30	29
PGV/PGA (cm/sec/g)	38	38	44
$PGA*PGD/PGV^2$	22	21	15
Correlation with Horizontal 1	--	0.074	-0.017
Correlation with Horizontal 2	--	--	-0.091

3.8 DESIGN OF CATEGORY I STRUCTURES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.8.3.7 In-Service Testing and Inspection Requirements

Replace the existing DCD statement with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.4.7 Testing and In-Service Inspection Requirements

Replace the existing DCD final statement of the subsection with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.5.1 Description of the Foundations

Add the following information after paragraph one of DCD Subsection 3.8.5.1:

STD SUP 3.8-1 The depth of overburden and depth of embedment are given in Subsection 2.5.4.

Add the following information to the end of DCD Subsection 3.8.5.1 as a new paragraph:

WLS COL 2.5-17 The Lee Nuclear Station site-specific waterproofing approach has not yet been selected. However, the waterproof membrane or waterproofing system for the Seismic Category I structures below grade will be selected from one of the acceptable approaches described in DCD Subsection 3.4.1.1.1.1. Duke Energy will notify NRC within 60 days of selecting the waterproofing system to be used, including the qualification methods planned to demonstrate the required performance characteristics. Duke Energy will also notify NRC not less than 90 days prior to site-specific qualification testing to demonstrate that the selected waterproofing system complies with DCD requirements. Both selection and testing milestones will be added to the detailed construction schedule to ensure tracking and closure of ITAAC 14.3.3.1.

3.8.5.7 In-Service Testing and Inspection Requirements

Replace the existing DCD first statement with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.6.5 Structures Inspection Program

STD COL 3.8-5 This item is addressed in Subsections 3.8.3.7, 3.8.4.7, 3.8.5.7, and 17.6.

3.8.6.6 Construction Procedures Program

Add the following to the end of DCD Subsection 3.8.6.6:

STD COL 3.8-6 Construction and inspection procedures for concrete filled steel plate modules address activities before and after concrete placement, use of construction mock-ups, and inspection of modules before and after concrete placement as discussed in DCD Subsection 3.8.4.8. The procedures will be made available to NRC inspectors prior to use.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.9.3.1.2 Loads for Class 1 Components, Core Support, and Component Supports

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STD COL 3.9-5 Add the following after the last paragraph under DCD subheading Request 3) and prior to DCD subheading Other Applications.

PRESSURIZER SURGE LINE MONITORING

<u>General</u>

The pressurizer surge line is monitored at the first AP1000 plant to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters. This monitoring occurs during the hot functional testing and first fuel cycle. The resulting monitoring data is evaluated to verify that the pressurizer surge line is within the bounds of the analytical temperature distributions and displacements.

Subsequent AP1000 plants (after the first AP1000 plant) confirm that the heatup and cooldown procedures are consistent with the pertinent attributes of the first AP1000 plant surge line monitoring. In addition, changes to the heatup and cooldown procedures consider the potential impact on stress and fatigue analyses consistent with the concerns of NRC Bulletin 88-11.

The pressurizer surge line monitoring activities include the following methodology and requirements:

<u>Monitoring Method</u>

The pressurizer surge line pipe wall is instrumented with outside mounted temperature and displacement sensors. The data from this instrumentation is supplemented by plant computer data from related process and control parameters.

<u>Locations to be Monitored</u>

In addition to the existing permanent plant temperature instrumentation, temperature and displacement monitoring will be included at critical locations on the surge line. The additional locations utilized for monitoring during the hot functional testing and the first fuel cycle (see Subsection 14.2.9.2.22) are selected based on the capability to provide effective monitoring.

Data Evaluation

Data evaluation is performed at the completion of the monitoring period (one fuel cycle). The evaluation includes a comparison of the data evaluation results with the thermal profiles and transient loadings defined for the pressurizer surge line, accounting for expected pipe outside wall temperatures. Interim evaluations of the data are performed during the hot functional testing period, up to the start of normal power operation, and again once three months worth of normal operating data has been collected, to identify any unexpected conditions in the pressurizer surge line.

3.9.3.4.4 Inspection, Testing, Repair, and/or Replacement of Snubbers

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Add the following text after the last paragraph of DCD Subsection 3.9.3.4.4:

STD COL 3.9-3 a. Snubber Design and Testing

1. A list of snubbers on systems which experience sufficient thermal movement to measure cold to hot position is included in Table 3.9-201.

2. The snubbers are tested to verify they can perform as required during the seismic events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. Production and qualification test programs for both hydraulic and mechanical snubbers are carried out by the snubber vendors in accordance with the snubber installation instruction manual required to be furnished by the snubber supplier. Acceptance criteria for compliance with ASME Section III Subsection NF, and other applicable codes, standards and requirements, are as follows:

   - Snubber production and qualification test programs are carried out by strict adherence to the manufacturer’s snubber installation and instruction manual. This manual is prepared by the snubber manufacturer and subjected to review for compliance with the applicable provisions of the ASME Pressure Vessel and Piping Code of record. The test program is periodically audited during implementation for compliance.

   - Snubbers are inspected and tested for compliance with the design drawings and functional requirements of the procurement specifications.

- Snubbers are inspected and qualification tested. No sampling methods are used in the qualification tests.

- Snubbers are load rated by testing in accordance with the snubber manufacturer's testing program and in compliance with the applicable sections of ASME QME-1-2007, Subsection QDR and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Subsection ISTD.

- Design compliance of the snubbers per ASME Section III Paragraph NF-3128, and Subparagraphs NF-3411.3 and NF-3412.4.

- The snubbers are tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test. The functional parameters cited in Subparagraph NF-3412.4 are included in the snubber qualification and testing program. Other parameters in accordance with applicable ASME QME-1-2007 and the ASME OM Code will be incorporated.

- The codes and standards used for snubber qualification and production testing are as follows:

  – ASME B&PV Code Section III (Code of Record date) and Subsection NF.

  – ASME QME-1-2007, Subsection QDR and ASME OM Code, Subsection ISTD.

- Large bore hydraulic snubbers are full Service Level D load tested, including verifying bleed rates, control valve closure within the specified velocity ranges and drag forces/ breakaway forces are acceptable in accordance with ASME, QME-1-2007 and ASME OM Codes.

3. Safety-related snubbers are identified in Table 3.9-201, including the snubber identification and the associated system or component, e.g., line number. The snubbers on the list are hydraulic and constructed to ASME Section III, Subsection NF. The snubbers are used for shock loading only. None of the snubbers are dual purpose or vibration arrestor type snubbers.

b. Snubber Installation Requirements

Installation instructions contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing that contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

STD COL 3.9-3 The description of the snubber preservice and inservice testing programs in this section is based on the ASME OM Code 2001 Edition through 2003 Addenda. The initial inservice testing program incorporates the latest edition and addenda of the ASME OM Code approved in 10 CFR 50.55a(f) on the date 12 months before initial fuel load. Limitations and modifications set forth in 10 CFR 50.55a are incorporated.

c. Snubber Preservice Examination and Testing

The preservice examination plan for applicable snubbers is prepared in accordance with the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Subsection ISTD, and the additional requirements of this Section. This examination is made after snubber installation but not more than 6 months prior to initial system preoperational testing. The preservice examination verifies the following:

1. There are no visible signs of damage or impaired operational readiness as a result of storage, handling, or installation.

2. The snubber load rating, location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

3. Snubbers are not seized, frozen or jammed.

4. Adequate swing clearance is provided to allow snubber movements.

5. If applicable, fluid is to the recommended level and is not to be leaking from the snubber system.

6. Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial preservice examination and initial system preoperational tests exceeds 6 months, reexamination of Items 1, 4, and 5 is performed. Snubbers, which are installed incorrectly or otherwise fail to meet the above requirements, are repaired or replaced and re-examined in accordance with the above criteria.

A preservice thermal movement examination is also performed, during initial system heatup and cooldown. For systems whose design operating temperature exceeds 250°F (121°C), snubber thermal movement is verified.

Additionally, preservice operational readiness testing is performed on snubbers. The operational readiness test is performed to verify the parameters of ISTD-5120. Snubbers that fail the preservice operational readiness test are evaluated to determine the cause of failure, and are retested following completion of corrective action(s).

Snubbers that are installed incorrectly or otherwise fail preservice testing requirements are re-installed correctly, adjusted, modified, repaired or replaced, as required. Preservice examination and testing is re-performed on installation-corrected, adjusted, modified, repaired or replaced snubbers as required.

d. Snubber Inservice Examination and Testing

Inservice examination and testing of safety-related snubbers is conducted in accordance with the requirements of the ASME OM Code, Subsection ISTD. Inservice examination is initially performed not less than two months after attaining 5% reactor power operation and is completed within 12 calendar months after attaining 5% reactor power. Subsequent examinations are performed at intervals defined by ISTD-4252 and Table ISTD-4252-1. Examination intervals, subsequent to the third interval, are adjusted based on the number of unacceptable snubbers identified in the current interval.

An inservice visual examination is performed on the snubbers to identify physical damage, leakage, corrosion, degradation, indication of binding, misalignment or deformation and potential defects generic to a particular design. Snubbers that do not meet visual examination requirements are evaluated to determine the root cause of the unacceptability, and appropriate corrective actions (e.g., snubber is adjusted, repaired, modified or replaced) are taken. Snubbers evaluated as unacceptable during visual examination may be accepted for continued service by successful completion of an operational readiness test.

Snubbers are tested inservice to determine operational readiness during each fuel cycle, beginning no sooner than 60 days before the start of the refueling outage. Snubber operational readiness tests are conducted with the snubber in the as-found condition, to the extent practicable, either in place or on a test bench, to verify the test parameters of ISTD-5210. When an in-place test or bench test cannot be performed, snubber subcomponents that control the parameters to be verified are examined and tested. Preservice examinations are performed on snubbers after reinstallation when bench testing is used (ISTD-5224), or on snubbers where individual subcomponents are reinstalled after examination (ISTD-5225).

Defined test plan groups (DTPG) are established and the snubbers of each DTPG are tested each fuel cycle according to an established sampling plan. Sample plan size and composition is determined as required for the selected sample plan, with additional sampling as may be required for that sample plan based on test failures and failure modes identified. Snubbers that do not meet test requirements are evaluated to determine root cause of the failure, and are assigned to failure mode groups (FMG) based on the evaluation, unless the failure is considered unexplained or isolated. The number of unexplained snubber failures, not assigned to a FMG, determines the additional testing sample. Isolated failures do not require additional testing. For unacceptable snubbers, additional testing is conducted for the DTPG or FMG until the appropriate sample plan completion criteria are satisfied.

Unacceptable snubbers are adjusted, repaired, modified or replaced. Replacement snubbers meet the requirements of ISTD-1600. Post-maintenance examination and testing, and examination and testing of repaired snubbers, is done to verify as acceptable the test parameters that may have been affected by the repair or maintenance activity.

Service life for snubbers is established, monitored and adjusted as required by ISTD-6000 and the guidance of ASME OM Code Nonmandatory Appendix F.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

Revise the third sentence of the third paragraph of DCD Subsection 3.9.6, and add information between the third and fourth sentences as follows:

STD COL 3.9-4 The edition and addenda to be used for the inservice testing program are administratively controlled; the description of the inservice testing program in this section is based on the ASME OM Code 2001 Edition through 2003 Addenda. The initial inservice testing program incorporates the latest edition and addenda of the ASME OM Code approved in 10 CFR 50.55a(f) on the date 12 months before initial fuel load. Limitations and modifications set forth in 10 CFR 50.55a are incorporated.

Revise the fifth sentence of the sixth paragraph of DCD Subsection 3.9.6 as follows:

STD COL 3.9-4 Alternate means of performing these tests and inspections that provide equivalent demonstration may be developed in the inservice test program ~~as described in subsection 3.9.8~~.

Revise the first two sentences of the final paragraph of DCD Subsection 3.9.6 to read as follows:

STD COL 3.9-4 A preservice test program, which identifies the required functional testing, is to be submitted to the NRC prior to performing the tests and following the start of construction. The inservice test program, which identifies requirements for functional testing, is to be submitted to the NRC prior to the anticipated date of commercial operation as described above.

Add the following text after the last paragraph of DCD Subsection 3.9.6:

Table 13.4-201 provides milestones for preservice and inservice test program implementation.

3.9.6.2.2 Valve Testing

Add the following prior the initial paragraph of DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Valve testing uses reference values determined from the results of preservice testing (PST) or inservice testing. These tests that establish reference and IST values are performed under conditions as near as practicable to those expected during the IST. Reference values are established only when a valve is known to be operating acceptably.

Pre-conditioning of valves or their associated actuators or controls prior to IST testing undermines the purpose of IST testing and is not allowed. Pre-conditioning includes manipulation, pre-testing, maintenance, lubrication, cleaning, exercising, stroking, operating, or disturbing the valve to be tested in any way, except as may occur in an unscheduled, unplanned, and unanticipated manner during normal operation.

Add the following sentence to the end of the fourth paragraph under the heading “Manual/Power-Operated Valve Tests”:

STD COL 3.9-4 Stroke time is measured and compared to the reference value, except for valves classified as fast-acting (e.g., solenoid-operated valves with stroke time less than 2 seconds), for which a stroke time limit of 2 seconds is assigned.

Add the following paragraph after the fifth paragraph under the heading "Manual/Power-Operated Valve Tests":

STD COL 3.9-4 During valve exercise tests, the necessary valve obturator movement is verified while observing an appropriate direct indicator, such as indicating lights that signal the required changes of obturator position, or by observing other evidence or positive means, such as changes in system pressure, flow, level, or temperature that reflects change of obturator position.

STD COL 3.9-4 Insert new second sentence of the paragraph containing the subheading "Power-Operated Valve Operability Tests" in DCD Subsection 3.9.6.2.2 (immediately following the first sentence of the DCD paragraph) to read:

The POVs include the motor-operated valves.

Add the following sentence as the last sentence of the paragraph containing the subheading "Power-Operated Valve Operability Tests" in DCD Subsection 3.9.6.2.2:

Table 13.4-201 provides milestones for the MOV program implementation.

Insert the following as the last sentence in the paragraph under the bulleted item titled "Risk Ranking" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Guidance for this process is outlined in the JOG MOV PV Study, MPR-2524-A.

Insert the following text after the last paragraph under the sub-heading of "Power-Operated Valve Operability Tests" and before the sub-heading "Check Valve Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 **Active MOV Test Frequency Determination** - The ability of a valve to meet its design basis functional requirements (i.e. required capability) is verified during valve qualification testing as required by procurement specifications. Valve qualification testing measures valve actuator actual output capability. The actuator output capability is compared to the valve's required capability defined in procurement specifications, establishing functional margin; that is, that increment by which the MOV's actual output capability exceeds the capability required to operate the MOV under design basis conditions. DCD Subsection 5.4.8 discusses valve functional design and qualification requirements. The initial inservice test frequency is determined as required by ASME OM Code Case OMN-1, Revision 1 (Reference 202). The design basis capability testing of MOVs utilizes guidance from Generic Letter 96-05 and the JOG MOV Periodic Verification PV Program. Valve functional margin is evaluated following subsequent periodic testing to address potential time-related performance degradation, accounting for applicable uncertainties in the analysis. If the evaluation shows that the functional margin will be reduced to less than established acceptance criteria within the established test interval, the test interval is decreased to less than the time for the functional margin to decrease below acceptance criteria. If there is not sufficient data to determine test frequency as described above, the test frequency is limited to not exceed two (2) refueling cycles or three (3) years, whichever is longer, until sufficient data exist to extend the test frequency. Appropriate justification is provided for any increased test interval, and the maximum test interval shall not exceed 10 years. This is to ensure that each MOV in the IST program will have adequate margin (including consideration for aging-related degradation, degraded voltage, control switch repeatability, and load-sensitive MOV behavior) to remain operable until the next scheduled test, regardless of its risk categorization or safety significance. Uncertainties associated with performance of these periodic verification tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.

Maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) are established so as not to exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.

Solenoid-operated valves (SOVs) are tested to confirm the valve moves to its energized position and is maintained in that position, and to confirm that the valve moves to the appropriate failure mode position when de-energized.

Other Power-Operated Valve Operability Tests - Power-Operated valves other than active MOVs are exercised quarterly in accordance with ASME OM ISTC, unless justification is provided in the inservice testing program for testing these valves at other than Code mandated frequencies.

Although the design basis capability of power-operated valves is verified as part of the design and qualification process, power-operated valves that perform an active safety function are tested again after installation in the plant, as required, to ensure valve setup is acceptable to perform their required functions, consistent with valve qualification. These tests, which are typically performed under static (no flow or pressure) conditions, also document the "baseline" performance of the valves to support maintenance and trending programs. During the testing, critical parameters needed to ensure proper valve setup are measured. Depending on the valve and actuator type, these parameters may include seat load, running torque or thrust, valve travel, actuator spring rate, bench set and regulator supply pressure. Uncertainties associated with performance of these tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.

Additional testing is performed as part of the air-operated valve (AOV) program, which includes the key elements for an AOV Program as identified in the JOG AOV program document, Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000 (References 203 and 204). The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air- and other power-operated valves included in the IST program. For example, key lessons learned addressed in the AOV program include:

- Valves are categorized according to their safety significance and risk ranking.

- Setpoints for AOVs are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).

- Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, to identify potential degradation, unless those valves are periodically cycled during normal plant operation, under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If required based on valve qualification or operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform its required functions.

- Sufficient diagnostics are used to collect relevant data (e.g., valve stem thrust and torque, fluid pressure and temperature, stroke time, operating and/or control air pressure, etc.) to verify the valve meets the functional requirements of the qualification specification.

- Test frequency is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with References 203 and 204, with a minimum of 5 years (or

3 refueling cycles) of data collected and evaluated before extending test intervals.

- Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when maintenance on the valve, repair or replacement, have the potential to affect valve functional performance.

- Guidance is included to address lessons learned from other valve programs specific to the AOV program.

- Documentation from AOV testing, including maintenance records and records from the corrective action program are retained and periodically evaluated as a part of the AOV program.

Insert the following two paragraphs as the final paragraphs under the sub-heading of “Power-Operated Valve Operability Tests” in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 The attributes of the AOV testing program described above, to the extent that they apply to and can be implemented on other safety-related power-operated valves, such as electro-hydraulic valves, are applied to those other power-operated valves.

Successful completion of the preservice and IST of MOVs, in addition to MOV testing as required by 10 CFR 50.55a, demonstrates that the following criteria are met for each valve tested: (i) valve fully opens and/or closes as required by its safety function; (ii) adequate margin exists and includes consideration of diagnostic equipment inaccuracies, degraded voltage, control switch repeatability, load-sensitive MOV behavior, and margin for degradation; and (iii) maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) does not exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.

Add the following new paragraph under the heading "Check Valve Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Preoperational testing is performed during the initial test program (refer to DCD Section 14.2) to verify that valves are installed in a configuration that allows correct operation, testing, and maintenance. Preoperational testing verifies that piping design features accommodate check valve testing requirements. Tests also verify disk movement to and from the seat and determine, without disassembly, that the valve disk positions correctly, fully opens or fully closes as expected, and remains stable in the open position under the full spectrum of system design-basis fluid flow conditions.

Add the following new last paragraphs under the subheading "Check Valve Exercise Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Acceptance criteria for this testing consider the specific system design and valve application. For example, a valve's safety function may require obturator movement in both open and closed directions. A mechanical exerciser may be used to operate a check valve for testing. Where a mechanical exerciser is used, acceptance criteria are provided for the force or torque required to move the check valve's obturator. Exercise tests also detect missing, sticking, or binding obturators.

When operating conditions, valve design, valve location, or other considerations prevent direct observation or measurements by use of conventional methods to determine adequate check valve function, diagnostic equipment and nonintrusive techniques are used to monitor internal conditions. Nonintrusive tests used are dependent on system and valve configuration, valve design and materials, and include methods such as ultrasonic (acoustic), magnetic, radiography, and use of accelerometers to measure system and valve operating parameters (e.g., fluid flow, disk position, disk movement, disk impact, and the presence or absence of cavitation and back-tapping). Nonintrusive techniques also detect valve degradation. Diagnostic equipment and techniques used for valve operability determinations are verified as effective and accurate under the PST program.

Testing is performed, to the extent practicable, under normal operation, cold shutdown, or refueling conditions applicable to each check valve. Testing includes effects created by sudden starting and stopping of pumps, if applicable, or other conditions, such as flow reversal. When maintenance that could affect valve performance is performed on a valve in the IST program, post-maintenance testing is conducted prior to returning the valve to service.

STD COL 3.9-4 Add the following new paragraph under the heading "Other Valve Inservice Tests" following the Explosively Actuated Valves paragraph in DCD Subsection 3.9.6.2.2:

Industry and regulatory guidance is considered in development of the IST program for squib valves. In addition, the IST program for squib valves incorporates lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions.

3.9.6.2.3 Valve Disassembly and Inspection

Add the following paragraph as the new second paragraph of DCD Subsection 3.9.6.2.3:

STD COL 3.9-4 During the disassembly process, the full-stroke motion of the obturator is verified. Nondestructive examination is performed on the hinge pin to assess wear, and seat contact surfaces are examined to verify adequate contact. Full-stroke motion of the obturator is re-verified immediately prior to completing reassembly. At least one valve from each group is disassembled and examined at each refueling outage, and all the valves in each group are disassembled and examined at least once every eight years. Before being returned to service, valves disassembled for examination or valves that received maintenance that could affect their performance are exercised with a full- or part-stroke. Details and bases of the sampling program are documented and recorded in the test plan.

Add Subsections 3.9.6.2.4 and 3.9.6.2.5 following the last paragraph of DCD Subsection 3.9.6.2.3:

3.9.6.2.4 Valve Preservice Tests

STD COL 3.9-4 Each valve subject to inservice testing is also tested during the preservice test period. Preservice tests are conducted under conditions as near as practicable to those expected during subsequent inservice testing. Valves (or the control system) that have undergone maintenance that could affect performance, and valves that have been repaired or replaced, are re-tested to verify performance parameters that could have been affected are within acceptable limits. Safety and relief valves and nonreclosing pressure relief devices are preservice tested in accordance with the requirements of the ASME OM Code, Mandatory Appendix I.

Preservice tests for valves are performed in accordance with ASME OM, ISTC-3100.

3.9.6.2.5 Valve Replacement, Repair, and Maintenance

STD COL 3.9-4 Testing in accordance with ASME OM, ISTC-3310 is performed after a valve is replaced, repaired, or undergoes maintenance. When a valve or its control system has been replaced, repaired, or has undergone maintenance that could affect valve performance, a new reference value is determined, or the previous value is reconfirmed by an inservice test. This test is performed before the valve is returned to service, or immediately if the valve is not removed from service. Deviations between the previous and new reference values are identified and analyzed. Verification that the new values represent acceptable operation is documented.

3.9.6.3 Relief Requests

Insert the following text after the first paragraph in DCD Subsection 3.9.6.3:

STD COL 3.9-4 The IST Program described herein utilizes Code Case OMN-1, Revision 1, "Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants" (Reference 202). Code Case OMN-1 establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor operated valves, in lieu of the requirements set forth in ASME OM Code Subsection ISTC.

STD COL 3.9-4 OMN-1, Alternative Rules for the Preservice and Inservice Testing of Certain MOVs

Code Case OMN-1, Revision 1, "Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in Light Water Reactor Power Plants," establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor-operated valves in lieu of the requirements set forth in OM Code Subsection ISTC. However, Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," June 2003, has not yet endorsed OMN-1, Revision 1.

Code Case OMN-1, Revision 0, has been determined by the NRC to provide an acceptable level of quality and safety when implemented in conjunction with the conditions imposed in Regulatory Guide 1.192. NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," recommends the implementation of OMN-1 by all licensees. Revision 1 to OMN-1 represents an improvement over Revision 0, as published in the ASME OM-2004 Code. OMN-1 Revision 1 incorporates the guidance on risk-informed testing of MOVs from OMN-11, "Risk-Informed Testing of Motor-Operated Valves," and provides additional guidance on design basis verification testing and functional margin, which eliminates the need for the figures on functional margin and test intervals in Code Case OMN-1.

The IST Program implements Code Case OMN-1, Revision 1, in lieu of the stroke-time provisions specified in ISTC-5120 for MOVs, consistent with the guidelines provided in NUREG-1482, Revision 1, Section 4.2.5.

Regulatory Guide 1.192 states that licensees may use Code Case OMN-1, Revision 0, in lieu of the provisions for stroke-time testing in Subsection ISTC of the 1995 Edition up to and including the 2000 Addenda of the ASME OM Code when applied in conjunction with the provisions for leakage rate testing in ISTC-3600 (1998 Edition with the 1999 and 2000 Addenda). Licensees who choose to apply OMN-1 are required to apply all of its provisions. The IST program incorporates the following provisions from Regulatory Guide 1.192:

(1) The adequacy of the diagnostic test interval for each motor-operated valve (MOV) is evaluated and adjusted as necessary, but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of OMN-1.

(2) The potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(3) Risk insights are applied using MOV risk ranking methodologies accepted by the NRC on a plant-specific or industry-wide basis, consistent with the conditions in the applicable safety evaluations.

(4) Consistent with the provisions specified for Code Case OMN-11 the potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

Compliance with the above items is addressed in Subsection 3.9.6.2.2. Code Case OMN-1, Revision 1, is considered acceptable for use with OM Code-2001 Edition with 2003 Addenda. Finally, consistent with Regulatory Guide 1.192, the benefits of performing any particular test are balanced against the potential adverse effects placed on the valves or systems caused by this testing.

3.9.8 COMBINED LICENSE INFORMATION

3.9.8.2 Design Specifications and Reports

Add the following text after the second paragraph in DCD Subsection 3.9.8.2.

STD COL 3.9-2 Design specifications and design reports for ASME Section III piping are made available for NRC review. Reconciliation of the as-built piping (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in DCD Subsection 3.9.3.1.2) is completed by the COL holder after the construction of the piping systems and prior to fuel load (in accordance with DCD Tier 1 Section 2 ITAAC line items for the applicable systems).

3.9.8.3 Snubber Operability Testing

STD COL 3.9-3 This COL Item is addressed in Subsection 3.9.3.4.4.

3.9.8.4 Valve Inservice Testing

STD COL 3.9-4 This COL Item is addressed in Subsections 3.9.6, 3.9.6.2.2, 3.9.6.2.3, 3.9.6.2.4, 3.9.6.2.5, and 3.9.6.3.

3.9.8.5 Surge Line Thermal Monitoring

STD COL 3.9-5 This COL item is addressed in Subsection 3.9.3.1.2 and Subsection 14.2.9.2.22.

3.9.8.7 As-Designed Piping Analysis

Add the following text at the end of DCD Subsection 3.9.8.7.

STD COL 3.9-7 The as-designed piping analysis is provided for the piping lines chosen to demonstrate all aspects of the piping design. A design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class I piping using the methods and criteria outlined in DCD Table 3.9-19 is made available for NRC review.

This COL item is also addressed in Subsection 14.3.3.

3.9.9 REFERENCES

201. Not used.

202. ASME Code Case OMN-1, Revision 1, “Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants.”

203. Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000.

204. USNRC, Eugene V. Imbro, letter to Mr. David J. Modeen, Nuclear Energy Institute, Comments On Joint Owners' Group Air Operated Valve Program Document, dated October 8, 1999.

STD COL 3.9-3

TABLE 3.9-201
SAFETY RELATED SNUBBERS

System	Snubber (Hanger) No.	Line #	System	Snubber (Hanger) No.	Line #
CVS	APP-CVS-PH-11Y0164	L001	RNS	APP-RNS-PH-12Y2060	L006
PXS	APP-PXS-PH-11Y0020	L021A	SGS	APP-SGS-PH-11Y0001	L003B
RCS	APP-RCS-PH-11Y0039	L215	SGS	APP-SGS-PH-11Y0002	L003B
RCS	APP-RCS-PH-11Y0067	L005B	SGS	APP-SGS-PH-11Y0004	L003B
RCS	APP-RCS-PH-11Y0080	L112	SGS	APP-SGS-PH-11Y0057	L003A
RCS	APP-RCS-PH-11Y0081	L215	SGS	APP-SGS-PH-11Y0058	L004B
RCS	APP-RCS-PH-11Y0082	L112	SGS	APP-SGS-PH-11Y0063	L003A
RCS	APP-RCS-PH-11Y0090	L118A	SGS	APP-SGS-PH-11Y0065	L005B
RCS	APP-RCS-PH-11Y0099	L022B	SGS	APP-SGS-PH-12Y0136	L015C
RCS	APP-RCS-PH-11Y0103	L003	SGS	APP-SGS-PH-12Y0137	L015C
RCS	APP-RCS-PH-11Y0105	L003	SGS	APP-SGS-PH-11Y0470	L006B
RCS	APP-RCS-PH-11Y0112	L032A	SGS	APP-SGS-PH-11Y2002	L006A
RCS	APP-RCS-PH-11Y0429	L225B	SGS	APP-SGS-PH-11Y2021	L006A
RCS	APP-RCS-PH-11Y0528	L005A	SGS	APP-SGS-PH-11Y3101	L006B
RCS	APP-RCS-PH-11Y0539	L225C	SGS	APP-SGS-PH-11Y3102	L006B
RCS	APP-RCS-PH-11Y0550	L011B	SGS	APP-SGS-PH-11Y3121	L006B
RCS	APP-RCS-PH-11Y0551	L011A	SGS	APP-SGS-PH-11Y0463	L006A
RCS	APP-RCS-PH-11Y0553	L153B	SGS	APP-SGS-PH-11Y0464	L006A
RCS	APP-RCS-PH-11Y0555	L153A	SGS	SG 1 Snubber A (1A)	(1)
RCS	APP-RCS-PH-11Y2005	L022A	SGS	SG 1 Snubber B (1B)	(1)
RCS	APP-RCS-PH-11Y2101	L032B	SGS	SG 2 Snubber A (2A)	(1)
RCS	APP-RCS-PH-11Y2117	L225A	SGS	SG 2 Snubber B (2B)	(1)

(1) These snubbers are on the upper lateral support assembly of the steam generators.

3.10 SEISMIC AND DYNAMIC QUALIFICATION OF SEISMIC CATEGORY I MECHANICAL AND ELECTRICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.11.5 COMBINED LICENSE INFORMATION ITEM FOR EQUIPMENT QUALIFICATION FILE

Add the following information to the end of DCD Subsection 3.11.5:

STD COL 3.11-1 The COL holder is responsible for the maintenance of the equipment qualification file upon receipt from the reactor vendor. The documentation necessary to support the continued qualification of the equipment installed in the plant that is within the environmental qualification (EQ) program scope is available in accordance with 10 CFR Part 50 Appendix A, General Design Criterion 1.

EQ files developed by the reactor vendor are maintained as applicable for equipment and certain post-accident monitoring devices that are subject to a harsh environment. The contents of the qualification files are discussed in DCD Section 3D.7. The files are maintained for the operational life of the plant.

For equipment not located in a harsh environment, design specifications received from the reactor vendor are retained. Any plant modifications that impact the equipment use the original specifications for modification or procurement. This process is governed by applicable plant design control or configuration control procedures.

Central to the EQ program is the EQ master equipment list (EQMEL). This EQMEL identifies the electrical and mechanical equipment or components that must be environmentally qualified for use in a harsh environment. The EQMEL consists of 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 significant release of radioactive material to the environment. This list is developed from the equipment list provided in AP1000 DCD Table 3.11-1. The EQMEL and a summary of equipment qualification results are maintained as part of the equipment qualification file for the operational life of the plant.

Administrative programs are in place to control revision to the EQ files and the EQMEL. When adding or modifying components in the EQ Program, EQ files are generated or revised to support qualification. The EQMEL is revised to reflect these new components. To delete a component from the EQ Program, a deletion justification is prepared that demonstrates why the component can be deleted. This justification consists of an analysis of the component, an associated circuit review if appropriate, and a safety evaluation. The justification is released and/or referenced on an appropriate change document. For changes to the EQMEL, supporting documentation is completed and approved prior to issuing the changes. This documentation includes safety reviews and new or revised EQ files. Plant modifications and design basis changes are subject to change process reviews, e.g. reviews in accordance with 10 CFR 50.59 or Section VIII of Appendix D to 10 CFR Part 52, in accordance with appropriate plant procedures. These reviews address EQ issues associated with the activity. Any changes to the EQMEL that are not the result of a modification or design basis change are subject to a separate review that is accomplished and documented in accordance with plant procedures.

Engineering change documents or maintenance documents generated to document work performed on an EQ component, which may not have an impact on the EQ file, are reviewed against the current revision of the EQ files for potential impact. Changes to EQ documentation may be due to, but not limited to, plant modifications, calculations, corrective maintenance, or other EQ concerns.

Table 13.4-201 provides milestones for EQ implementation.

APPENDIX 3A
HVAC DUCTS AND DUCT SUPPORTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3B
LEAK-BEFORE-BREAK EVALUATION OF THE AP1000 PIPING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3C
REACTOR COOLANT LOOP ANALYSIS METHODS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3D
METHODOLOGY FOR QUALIFYING AP1000 SAFETY-RELATED ELECTRICAL AND MECHANICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3E
HIGH-ENERGY PIPING IN THE NUCLEAR ISLAND

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3F
CABLE TRAYS AND CABLE TRAY SUPPORTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3G
NUCLEAR ISLAND SEISMIC ANALYSES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3H
AUXILIARY AND SHIELD BUILDING CRITICAL SECTIONS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3I
EVALUATION FOR HIGH FREQUENCY SEISMIC INPUT

This section of the referenced DCD is incorporated by reference with no departures or supplements.