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)

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

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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 (CLT) 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 CLT 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 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 SPA 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 SPA, 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 Foundation Response Spectra

WLS SUP 3.7-3 Design foundation 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, except for the eastern edge of the Unit 2 nuclear island, which will require approximately 20 ft. of fill concrete to build up the support zone to the base of the nuclear island. The Lee Nuclear Station site provides uniform hard-rock support for the nuclear island; foundation conditions and uniformity are described in Subsection 2.5.4.7.4 (Figure 2.5.4-241). 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.

Based on these foundation conditions, individual foundation response spectra are provided for the certified design portion of the plant at Units 1 and 2. The site-specific dynamic velocity profiles developed for the Lee Nuclear Station are described in Subsection 2.5.4.7.5. As described in Subsection 2.5.2.7.4, the site ground motion response spectra (GMRS) defines the input motion (FIRS) at Unit 2, while the FIRS associated with dynamic profile A1 defines the Unit 1 FIRS.

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, associated with Unit 1 FIRS A1 dynamic velocity profile (Figure 2.5.4-252a), 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 North American Vertical Datum (NAVD) transferred up through previously placed and new concrete materials to the basemat foundation level at 553.5 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 553.5 feet (NAVD). The GMRS (Unit 2 FIRS) is associated with dynamic velocity Profile C (Figure 2.5.4-250).

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 (GMRS), and 2.5.2.7, Development of FIRS for Units 1 and 2.

WLS DEP 2.0-1

Both the GMRS and the Unit 1 FIRS exceed the AP1000 CSDRS at higher frequencies. As a conservative simplification, the site-specific foundation input motion for both Units 1 and 2 is represented as the horizontal and vertical envelope of the GMRS (Unit 2) and Unit 1 FIRS. These envelope spectra, considered to be applicable to both units, are illustrated in Figures 3.7-201 and 3.7-202, and are referred to as the nuclear island FIRS (NI FIRS). As shown on Figure 3.7-201, the horizontal NI FIRS exceeds the horizontal AP1000 CSDRS at frequencies above approximately 14 hertz. PGA at 100 hertz of the NI FIRS is 0.352 g. As shown on Figure 3.7-202, the vertical NI FIRS exceeds the vertical AP1000 CSDRS at frequencies above approximately 16 hertz.

As shown on Figure 3.7-201, the horizontal NI FIRS is above the horizontal AP1000 HRHF spectrum for all frequencies above about 3 hertz. As shown on Figure 3.7-202, the vertical NI FIRS is above the vertical AP1000 HRHF for frequencies between about 3 to 55 hertz and 80 to 100 hertz.

WLS SUP 3.7-3

As described in AP1000 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 structures, systems, or components of the plant.

The Lee Nuclear Station site provides uniform hard-rock support for the nuclear island, but the site characteristic NI FIRS exceed the horizontal and vertical AP1000 HRHF. As described in AP1000 DCD Section 2.5.2.1, Paragraph 4b, if the site-specific spectra are not enveloped by the AP1000 HRHF envelope response spectra or the AP1000 CSDRS, the COL applicant may perform site-specific studies to demonstrate high frequency is not damaging. Therefore, a site-specific analysis of the AP1000 has been performed, similar to the analysis described in AP1000 DCD Appendix 3I, to demonstrate that these high frequency spectra exceedances are within the seismic design margin of the AP1000 certified design and will not adversely affect the structures, systems, or components of the plant (Reference 206). Subsection 3.7.2.15 describes confirmatory site-specific analyses of the nuclear island that demonstrate compliance with the AP1000 DCD.

3.7.2.1.2 Time-History Analysis and Complex Frequency Response Analysis

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

WLS SUP 3.7-6 For cases when site-specific analyses of the nuclear island structures may be required, artificial time histories (two horizontal and one vertical) were developed to be compatible with the Lee Nuclear Station foundation input motions, and to satisfy the requirements of Standard Review Plan (SRP) 3.7.1. As a conservative simplification, the foundation input motion for both units was represented as the envelope of the GMRS (Unit 2 FIRS) and the Unit 1 FIRS, referred to as the Lee Nuclear Station NI FIRS (Figures 3.7-201 and 3.7-202). 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 ILA031 record from the 1999 Chi-Chi earthquake in Taiwan (magnitude (**M)** 7.6, closest distance ($R_{closest}$) = 94.7 km) is selected. This record is part of the NRC time history library (Reference 207) 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 until it is compatible with the target response spectrum.

3. Use visual examination of the results to confirm that the resulting time histories are realistic and independent calculations to confirm that they meet the requirements of SRP 3.7.1 as well as the requirements defined in Item 4 below. 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 (5%-75% Arias intensity) 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 representing the NI FIRS are shown in Table 3.7-201. Figures 3.7-203a through 203c illustrate the three component time histories.

3.7.2.8.4 Seismic Modeling and Analysis of Seismic Category II Building Structures

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

WLS SUP 3.7-4 FSAR Subsection 2.5.4.5.2 describes how areas in the foundation support zones of Seismic Category II buildings (the Annex Building and Turbine Building first bay) will be excavated to expose concrete or rock, and fill concrete will be used to build up to the base level of the nuclear island. If rock within the foundation support zone of these Seismic Category II structures is higher than the base of the nuclear island, the rock will be removed to the elevation of the base of the nuclear island. In areas where the pre-existing concrete and/or rock within the foundation support zone of these Seismic Category II structures are at a lower elevation than the base of the nuclear island, fill concrete will be used to build up to the base level of the nuclear island. This configuration is illustrated in FSAR Figures 2.5.4-245 and 2.5.4-260 through 2.5.4-265. These measures ensure that the Lee Nuclear Station site provides uniform support for the Seismic Category II structures in a configuration identical to that considered in the AP1000 DCD designs.

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.

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.

The Lee site-specific bearing capacity for the granular fill material supporting the Seismic Category II structures (shown in FSAR Table 2.5.4-228) is greater than the generic AP1000 bearing demand for these structures.

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 information above 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; the configuration of the granular fill supporting the Seismic Category II buildings is consistent with that described in the AP1000 DCD; and the bearing capacity of the supporting granular fill is greater than the bearing demand. Therefore, the Lee Nuclear Station site complies explicitly with the requirements of DCD
WLS DEP 2.0-1 Subsection 3.7.2.8.4 for a hard rock site, with the exception that the NI FIRS is not bounded by the AP1000 HRHF spectra.

WLS SUP 3.7-4 As permitted by DCD Subsection 3.7.2.8.4, site-specific analyses of the Lee Nuclear Station Seismic Category II adjacent buildings were performed. Site-specific performance-based surface response spectra (PBSRS) at plant grade were developed for each Seismic Category II building, using the same analytical methods used in calculating the Unit 1 FIRS. Hazard-consistent, strain-compatible properties were also developed for the granular fill material supporting the Seismic Category II adjacent buildings. These site-specific characteristics are used as inputs to the site-specific soil-structure interaction analyses described in Reference 205, which are performed using analyses consistent with those supporting the AP1000 DCD.

The analyses presented in Reference 205 confirm that the calculated site-specific relative displacements of the Seismic Category II adjacent buildings are much less than the building separation provided, so there is no contact between the nuclear island and the Seismic Category II adjacent buildings. The maximum site-specific bearing demand (approximately 24.5 ksf for the Annex Building and 5.3 ksf for the Turbine Building) is significantly less than the site-specific allowable bearing pressure shown in FSAR Table 2.5.4-228 (approximately 33.55 ksf for the Annex Building and 45.03 ksf for the Turbine Building), demonstrating that the granular fill material selected is adequate for supporting those structures.

As required by AP1000 DCD Subsection 3.7.2.8.4, the Lee site-specific Seismic Category II foundation seismic response spectra are compared to the corresponding AP1000 annex building and turbine building first bay generic design envelope response spectra. This comparison is shown in Figures 3.7-213a and 3.7-213b for the annex building and Figures 3.7-214a and 3.7-214b for the first bay of the turbine building. These foundation response spectra are computed in 2-D analyses, and have been adjusted for 3-D effects, as required by the AP1000 DCD.

Figure 3.7-213a demonstrates that the site-specific annex building horizontal foundation response spectrum falls beneath the generic design envelope that is used in designing the AP1000 standard annex building. Figure 3.7-214a illustrates the comparable horizontal foundation response spectrum for the turbine building first bay, which is generally less than the generic design envelope, but exhibits a minor exceedance between 3 Hz and 5 Hz for one soil case. These horizontal foundation response spectra are of primary importance in assessing the potential interactions of Seismic Category II adjacent buildings with the nuclear island. The comparison provides high confidence that the lateral force resisting system for the AP1000 standard Seismic Category II adjacent structures is also adequate for the Lee site-specific seismic requirements.

Figure 3.7-213b and Figure 3.7-214b compare the site-specific vertical foundation response spectra to the comparable generic design envelope for the AP1000 standard annex building and turbine building first bay, respectively. For both the annex building and the turbine building first bay, vertical spectra exceedances are noted between about 6 Hz and 25 Hz. These exceedances are likely associated with vertical resonance of the granular fill column in this frequency range. These vertical foundation response spectra are of less importance in assessing potential interactions with the nuclear island, but are important for design of individual building elements such as floor slabs and roofs within the Seismic Category II adjacent structures.

As required by DCD Subsection 3.7.2.8, Seismic Category II adjacent structures must be designed to prevent their collapse when subjected to their design earthquake. Therefore, the detailed design of the building elements making up the AP1000 standard Seismic Category II adjacent structures will be reviewed to confirm that they satisfy the acceptance criteria specified in AP1000 DCD Subsection 3.7.2 when subjected to the forces resulting from the site-specific foundation response spectra. Should any building element not meet those criteria, appropriate design changes will be implemented to increase its capacity. In this manner, Duke Energy will confirm that the Seismic Category II adjacent buildings are designed not only for the site-specific seismic requirements, but also for the AP1000 Seismic Category II adjacent buildings generic design envelope. This review will be conducted when the source of the actual granular fill material supporting the Seismic Category II adjacent buildings is selected, as part of verifying the compatibility of that material with the facility seismic design. This approach to the non-safety related Seismic Category II adjacent buildings is similar to that used for an item of equipment for which detailed fabrication design has not been completed, but for which all safety-related performance requirements have been identified. The review and any required design changes will be completed prior to start of construction of the Seismic Category II adjacent buildings at Lee Nuclear Station.

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 DEP 2.0-1
WLS SUP 3.7-5 As described in FSAR Subsection 3.7.1.1.1, the Lee Nuclear Station site provides uniform hard-rock support but the updated site characteristic GMRS and Unit 1 FIRS are not bounded by the HRHF spectra. To address the exceedances of the design basis AP1000 CSDRS and alternate AP1000 HRHF spectra described above, confirmatory site-specific analyses of the nuclear island Seismic Category I structures were performed using an envelope of the updated GMRS (Unit 2 FIRS) and Unit 1 FIRS, referred to as NI FIRS, to confirm that site-specific seismic demands will not adversely affect the structures, systems, or components of the Lee Nuclear Station (Reference 206). This site-specific evaluation uses the same methodology described in AP1000 DCD Appendix 3I to evaluate and qualify the AP1000 HRHF spectra.

WLS SUP 3.7-5 These site-specific analyses described in Reference 206 include three-dimensional incoherent SSI analysis based on the NI FIRS. The nuclear island analytical model used in these analyses is an updated version of the NI20 model referred to as NI20u. The updated NI20u model includes detailed design changes identified since AP1000 DCD Revision 19. The model also includes refinements to provide a better match to the more detailed NI10 model used in the AP1000 DCD analyses, while continuing to be a conservative representation of the NI10 model. All AP1000 DCD descriptions of the NI20 model remain applicable to the NI20u model, and the changes incorporated do not impact or require an update to the licensing basis as defined in the AP1000 DCD Revision 19.

A screening criteria to identify a representative sample of AP1000 structures, systems, and components (SSCs) to be evaluated to demonstrate acceptability of the AP1000 certified design for the Lee Nuclear Station NI FIRS (Reference 206). To better understand the significance of the evaluations required, the in-structure response spectra for the AP1000 CSDRS, HRHF spectra and NI FIRS were compared at the six key locations identified in AP1000 DCD Table 3G.4-1 as listed below.

- Containment internal structures (CISs) at elevation of reactor vessel support – Node 1761
- Auxiliary building northeast corner at elevation 116’-6” – Node 2078
- Containment operating floor – Node 2199
- Shield building at fuel building roof – Node 2675
- Steel containment vessel (SCV) at polar crane support – Node 2788
- Shield building roof – Node 3329

The resulting site-specific in-structure FRS are shown in Figures 3.7-209a through 3.7-211c for these six (6) key locations. These figures compare the in-structure spectra resulting from the envelope of the AP1000 CSDRS cases (bold black curve) to that resulting from the AP1000 HRHF analyses (solid red curve) to that resulting from the site-specific NI FIRS (dashed blue curve). It should be noted that these spectra are at the same locations, though the node numbers in the site-specific NI20u model are different than in the NI20 model used for AP1000 DCD Revision 19. It is important to note that the AP1000 HRHF broad curve (envelope) is based on SASSI 3D analyses and includes seismic motion incoherency effects. The 3D analyses compare the Lee Nuclear Station 3D FRS results with incoherency to the AP1000 HRHF envelope, also including incoherency.

WLS DEP 2.0-1 At these six key locations, minor exceedances of the comparable AP1000 DCD in-structure spectra resulting from the AP1000 CSDRS or HRHF spectra were noted. More detailed evaluations are therefore presented in Reference 206 to justify those exceedances and to demonstrate that they are acceptable.

In-structure floor response spectra were investigated in additional locations, consistent with the evaluations supporting Appendix 3I of the DCD. These locations are shown below.

- Auxiliary and Shield Building at Grade
- Auxiliary and Shield Building at Main Control Room Floor
- Auxiliary and Shield Building at Elevation 135'
- Auxiliary and Shield Building at Elevations 160', 180' and 230'
- Auxiliary and Shield Building at Elevations 267', 289' and 327'
- Containment Internal Structures at Grade Elevation 99'

- Containment Internal Structures at Elevation 135'
- Containment Internal Structures at Elevations 153' East, 153' West, and 160'
- Hot Legs and Pressurizer Bottom

Consistent with the criteria established in DCD Section 3.7.5.4, in-structure spectra are determined to be acceptable if they are within ten percent of the corresponding CSDRS and HRHF spectra. Using this criterion, three instances of exceedances were noted. The first instance is above 60 Hz in the vertical direction for the Auxiliary and Shield Building at Main Control Room Floor. The second instance is above 60 Hz in the vertical direction for the Containment Internal Structures at Elevation 153'. Figure 3.7-212 shows the in-structure vertical spectra for these two locations. The only equipment potentially affected by such high frequency exceedance is tested to levels higher than those imposed by these in-structure spectra, as described below. The third instance of exceedance was at lower horizontal frequencies at the reactor coolant loop hot legs and pressurizer bottom. This exceedance was justified by detailed comparative analyses of representative piping attached at these locations, as described below.

Evaluations of representative portions of the building structures (three locations in the Auxiliary Building, eight locations in the Shield Building and three areas in the CIS) confirmed that the seismic loads associated with the design basis AP1000 CSDRS envelope those from the Lee site-specific spectra in all cases (Reference 206).

The reactor vessel and internals were selected for evaluation as representative of major equipment. The analyses described in Reference 206 demonstrate that the AP1000 CSDRS results in higher loads and stresses than those from the Lee site-specific spectra. Likewise, the design of the primary component supports and the reactor coolant loop primary equipment nozzles were found to be controlled by the CSDRS rather than the Lee site-specific spectra.

As described in DCD Appendix 3I, ASME Class 1, Class 2, and Class 3 piping is designed for both CSDRS and HRHF spectra. As described in Reference 206, forty piping packages and the associated floor response spectra were reviewed for susceptibility to high frequency input motion. Three piping packages were selected for detailed review as the most susceptible to the effects of high frequency inputs, and to any differences between the site-specific spectra and CSDRS or HRHF spectra. Two of those packages, a portion of the Automatic Depressurization System and the Pressurizer Surge Line, attach to the reactor coolant loop hot legs and to the pressurizer bottom, where in-structure spectra exceedances of the Lee site-specific spectra compared to CSDRS or HRHF spectra were noted. The third package, a portion of the Spent Fuel System in the Auxiliary Building, was selected based on its potential susceptibility to high-frequency input motion.

The stress analysis results for these three piping systems (Reference 206) indicate that the piping stresses resulting from the site-specific spectra are less than those resulting from HRHF spectra except for the Pressurizer Surge Line, where the site-specific stresses are slightly higher. Nevertheless, it was demonstrated that the stresses resulting from the CSDRS control the design over both the site-specific spectra and HRHF spectra, except for one point where the site-specific stresses are only approximately 3% higher than those due to CSDRS. Based on the selection of these piping packages, these results are representative of all safety class piping for the plant. It is therefore concluded that stresses resulting from site-specific high frequency input are bounded by AP1000 design basis analysis results, and the effect of site-specific high frequency input on piping is non-damaging.

Reference 206 describes a review of current AP1000 equipment qualification test methods and requirements, and a comparison of those requirements to the Lee site-specific requirements. Some of the site-specific in-structure spectra exhibit minor exceedances of the comparable standard AP1000 qualification required response spectra (RRS) envelopes. Nevertheless, in all cases the actual test response spectra (TRS) used in completed testing exceed the site-specific demands by a significant margin. It is therefore concluded that those qualification tests are also applicable for the Lee site-specific requirements, and that there is high confidence that future tests will also be applicable to Lee site-specific requirements. Duke Energy will ensure that all seismic qualification testing for safety-related equipment required per DCD Appendix 3I appropriately envelopes the Lee site-specific requirements, in addition to the CSDRS and HRHF RRS.

As described above, a site-specific analysis of the AP1000 using the Lee site-specific criteria has been performed, comparing the resulting in-structure spectra at six key locations to the spectra resulting from the design basis AP1000 CSDRS envelope. Exceedances of those in-structure standard plant spectra have been evaluated and justified (Reference 206) by detailed site-specific analyses of representative portions of building structures, primary equipment and piping systems, and by a review of standard AP1000 seismic qualification testing practices.

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. Removed

202. Removed

203. Removed

204. Removed

205. Westinghouse Electric Company Report WLG-1000-S2R-804, Revision 3, William S. Lee Site Specific Adjacent Buildings Seismic Evaluation Report, February 2014.

206. Westinghouse Electric Company, LLC, "Effect of William S. Lee Site Specific Seismic Requirements on AP1000 SSCs," WLG-GW-GLR-815, Revision 0, January 17, 2014.

207. McGuire, R.K., Silva, W.J., and Constantino, C.J. (2001). Technical basis for revision of regulatory guidance on design ground motions: hazard- and risk-consistent ground motion spectra guidelines, U.S. Nuclear Regulatory Commission Report, NUREG/CR-6728, October, 2001.

WLS SUP 3.7-6

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

Parameter	Horizontal 1	Horizontal 2	Vertical
Duration (5-75%; sec)	13.6	14.4	11.3
PGA (g)	0.36	0.36	0.33
PGV (cm/sec)	11	15	8.7
PGD (cm)	7.4	7.7	5.5
PGD/PGA (cm/g)	21	21	17
PGV/PGA (cm/sec/g)	30	42	27
PGA*PGD/PGV^2	22	12	23
Correlation with Horizontal 1	--	0.044	0.15
Correlation with Horizontal 2	--	--	0.077

Note: Individual time histories are denoted as follows:
Horizontal 1 = ILA031-N
Horizontal 2 = ILA031-W
Vertical = ILA031-V

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

General

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:

Monitoring Method

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.

Locations to be Monitored

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

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 (IST). 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 core damage frequency 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.

TABLE 3.11-201 (Sheet 14 of 51)
ENVIRONMENTALLY QUALIFIED ELECTRICAL AND
MECHANICAL EQUIPMENT

	Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
	SG1 Wide Range Level	SGS-JE-LT011	1	ESF PAMS	5 min 2 wks	E *
	SG1 Wide Range Level	SGS-JE-LT012	1	ESF PAMS	5 min 2 wks	E *
	SG1 Wide Range Level	SGS-JE-LT015	1	ESF PAMS	5 min 2 wks	E * E *
	SG1 Wide Range Level	SGS-JE-LT016	1	ESF PAMS	5 min 2 wks	E * E *
	SG2 Wide Range Level	SGS-JE-LT013	1	ESF PAMS	5 min 2 wks	E *
	SG2 Wide Range Level	SGS-JE-LT014	1	ESF PAMS	5 min 2 wks	E *
	SG2 Wide Range Level	SGS-JE-LT017	1	ESF PAMS	5 min 2 wks	E *
	SG2 Wide Range Level	SGS-JE-LT018	1	ESF PAMS	5 min 2 wks	E *
WLS DEP 3.11-1	Spent Fuel Pool Level	SFS-JE-LT019A	6	PAMS	2 wks	E **
WLS DEP 3.11-1	Spent Fuel Pool Level	SFS-JE-LT019B	7	PAMS	2 wks	E **
WLS DEP 3.11-1	Spent Fuel Pool Level	SFS-JE-LT019C	6	PAMS	2 wks	E **
	Air Storage Tank Pressure - A	VES-JE-PT001A	7	PAMS	2 wks	E+
	Air Storage Tank Pressure - B	VES-JE-PT001B	7	PAMS	2 wks	E+
	Containment Pressure Normal Range	PCS-JE-PT005	7	ESF PAMS	5 min 4 mos	E *
	Containment Pressure Normal Range	PCS-JE-PT006	7	ESF PAMS	5 min 4 mos	E *
	Containment Pressure Normal Range	PCS-JE-PT007	7	ESF PAMS	5 min 4 mos	E *
	Containment Pressure Normal Range	PCS-JE-PT008	7	ESF PAMS	5 min 4 mos	E *
	Containment Pressure Extended Range	PCS-JE-PT012	7	PAMS	4 mos	E *
	Containment Pressure Extended Range	PCS-JE-PT013	7	PAMS	4 mos	E *
	Containment Pressure Extended Range	PCS-JE-PT014	7	PAMS	4 mos	E *

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 the following departures and/or supplements.

WLS DEP 2.0-1 3I.1 INTRODUCTION

Add the following information to the end of DCD Subsection 3I.1

The nuclear island foundation input response spectra (NI FIRS) for Lee Nuclear Station, the envelope of the GMRS (Unit 2 FIRS) and the Unit 1 FIRS (Subsection 3.7.1.1.1), are slightly above the AP1000 HRHF spectra, but the spectra are very similar. Figures 3I.1-201 and 3I.1 202 compare the NI FIRS to the AP1000 CSDRS and the AP1000 HRHF spectra for the horizontal and vertical directions for 5% damping. The NI FIRS exceeds the AP1000 CSDRS for frequencies above approximately 14 Hz and the AP1000 HRHF spectra above approximately 3 Hz.

Because the NI FIRS are not enveloped by the AP1000 HRHF spectra, a site-specific analysis is performed to evaluate and justify exceedances. Technical report WLG-GW-GLR-815 (Reference 201) provides a summary of those evaluations and results. This report presents in-structure response spectra throughout the Nuclear Island resulting from the site-specific input. These in-structure response spectra are less than or within ten percent of those for the AP1000 CSDRS and the AP1000 HRHF spectra at most locations/elevations and several minor exceedances are justified by further evaluations.

WLS DEP 2.0-1 3I.2 HIGH FREQUENCY SEISMIC INPUT

Add the following information to the end of DCD Subsection 3I.2

Figures 3I.1-201 and 3I.1-202 present a comparison of the horizontal and vertical (respectively) Lee Nuclear Station NI FIRS to the AP1000 CSDRS and the AP1000 HRHF. The NI FIRS are calculated at foundation level (39.5' below grade), at the upper most competent material and treated as an outcrop for calculation purposes.

For each direction, the NI FIRS exceeds the CSDRS in higher frequencies (greater than 14 Hz horizontal and 16 Hz vertical) and the AP1000 HRHF spectra at frequencies greater than 3 Hz in both the horizontal and vertical directions.

WLS DEP 2.0-1 3I.3 NI MODELS USED TO DEVELOP HIGH FREQUENCY RESPONSE

Add the following information to the end of DCD Subsection 3I.3

The NI20u nuclear island model (Reference 201) is analyzed in ACS SASSI using the Lee Nuclear Station NI FIRS time histories (Subsection 3.7.2.1.2) applied at foundation level to obtain the motion at the base.

The NI20u model used in the Lee Nuclear Station site-specific analysis was updated to incorporate design changes from detailed design finalization of the AP1000 standard plant (no impact from design changes to licensing basis as defined in AP1000 DCD Rev 19) and to improve the match between the NI20u model and the more realistic NI10 model used to design and qualify the AP1000 standard plant for the CSDRS.

Evaluation of incoherent NI FIRS has been performed. In-structure response spectra for the AP1000 CSDRS, incoherent HRHF spectra and the incoherent NI FIRS were compared at a number of locations/elevations in the Nuclear Island. Several minor exceedances were noted that are addressed as part of the sampling evaluation outlined in DCD Subsection 3I.6.

WLS DEP 2.0-1

3I.6 EVALUATION

Add the following information to the end of DCD Subsection 3I.6

As described in Lee Nuclear Station site-specific Technical Report WLG-GW-GLR-815 (Reference 201), the in-structure response spectra resulting from the Lee Nuclear Station NI FIRS input are less than or within ten percent of those generated for the AP1000 CSDRS and the AP1000 HRHF spectra at most locations/elevations and several minor exceedances are justified by further evaluation. Therefore, the sample of structures, systems and components selected for evaluation remains unchanged.

WLS DEP 2.0-1

3I.6.1 Building Structures

Add the following information to the end of DCD Subsection 3I.6.1

Load comparisons for the building structures evaluated show that the seismic loads resulting from the CSDRS input motion are greater than the seismic loads generated from the NI FIRS (Reference 201).

WLS DEP 2.0-1

3I.6.2 Primary Coolant Loop

Add the following information to the end of DCD Subsection 3I.6.2

Load comparisons for the primary component supports and nozzles evaluated show that the seismic loads resulting from the CSDRS input motion are greater than the seismic loads generated from the NI FIRS (Reference 201).

WLS DEP 2.0-1

3I.6.3 Piping Systems

Add the following information to the end of DCD Subsection 3I.6.3

ASME Class 1, 2, and 3 piping packages were reviewed along with local input seismic response spectra for susceptibility to excitation from high frequency seismic input motion. Since the in-structure floor response spectra (FRS) generated from the Lee Nuclear Station NI FIRS are enveloped completely by either by the FRS generated from the CSDRS or HRHF spectra in most locations, all of the piping analyses do not need to be redone for the NI FIRS.

Three piping packages, ADS 4th Stage East Compartment and Passive RHR Supply, Pressurizer Surge Line, and SFS from Auxiliary Building Area 4 SCV to Auxiliary Building Area 6 SFS Pumps were chosen for evaluation (Reference 201). These packages are representative of all safety class piping in Lee Nuclear Station because they are the most susceptible to excitation from high frequency seismic input motion.

The stress results of the sample piping analysis packages show that the AP1000 HRHF stresses were greater than the NI FIRS stresses for all nodes in the ADS 4th Stage and SFS from Auxiliary Building from 4 to 6 piping packages and only slightly less in the Pressurizer Surge Line piping package. Stress comparison results show that AP1000 CSDRS stresses are greater than the NI FIRS stresses at all nodes in all three piping packages except for one node in the SFS from Auxiliary Building 4 to 6 piping package where there was a slight NI FIRS exceedance of less than three percent.

The stresses due to the Lee Nuclear Station NI FIRS input are bounded by design basis analysis results. The same applies to all of the analyzed piping supports. As a result, the effect of the NI FIRS input on safety class piping is found to be non-damaging (Reference 201).

WLS DEP 2.0-1

3I.6.4 Electrical and Electro-Mechanical Equipment

Add the following information to the end of DCD Subsection 3I.6.4

To demonstrate acceptability, the test response spectra (TRS) for high frequency sensitive equipment procured for Lee Nuclear Station will have to bound the required response spectra (RRS) of the AP1000 CSDRS, AP1000 HRHF spectra, and the NI FIRS generated in-structure response spectra. As shown in the Lee Nuclear Station site-specific Technical Report WLG-GW-GLR-815 (Reference 201), very little if any of the AP1000 equipment will need to be re-qualified for the Lee Nuclear Station high frequency seismic motion considering margins in the TRS currently being used to qualify AP1000 high frequency sensitive equipment. However, per the licensing commitment in Subsection 3.7.2.15, Duke Energy will ensure that all seismic qualification testing for safety-related equipment required per this Appendix appropriately envelopes the Lee Nuclear Station site-specific seismic requirements, in addition to the CSDRS and HRHF RRS.

WLS DEP 2.0-1

3I.7 REFERENCES

Add the following information to the end of DCD Subsection 3I.7

201. Westinghouse Electric Company, LLC, "Effect of William S. Lee Site Specific Seismic Requirements on AP1000 SSCs," WLG-GW-GLR-815, Revision 0, January 17, 2014.