CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF FIGURES

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.

3.2.2 AP1000 CLASSIFICATION SYSTEM

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

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

3.3 WIND AND TORNADO LOADINGS

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

3.3.1.1 Design Wind Velocity

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

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

3.3.2.1 Applicable Design Parameters

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

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

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

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

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

3.3.3 COMBINED LICENSE INFORMATION

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

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

3.4 WATER LEVEL (FLOOD) DESIGN

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

3.4.1.3 Permanent Dewatering System

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

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

3.4.3 COMBINED LICENSE INFORMATION

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

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

3.5 MISSILE PROTECTION

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

3.5.1.3 Turbine Missiles

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

STD SUP 3.5-1 The potential for a turbine missile from another AP1000 plant in close proximity has been considered. As noted in DCD Subsection 10.2.2, the probability of generation of a turbine missile is less than 1 x 10^{-5} per year. Given this generation probability and the protection provided by the reinforced concrete shield building and auxiliary building walls, roofs, and floors, the guidance of Regulatory Guide 1.115 is satisfied for two AP1000 plants side by side.

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 2.2.2.6. The aircraft hazard probability developed from the total probability of an aircraft crash into the effective areas of the plant does not constitute a design basis event. The probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR Part 100 exposure guidelines is based on the following:

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

- One federal airway passes within four miles of the plant site. Low altitude Airway V54 runs between Spartanburg Downtown Memorial Airport, South Carolina (SPA) located 26.1 miles from Lee Nuclear Station and Charlotte/Douglas International Airport, North Carolina (CLT) located 34.4 miles from Lee Nuclear Station. The average annual number of flights using Airway V54 is approximately 15 to 25 percent of the total airport operation. The FAA forecast number of Spartanburg Downtown Memorial Airport operation for year 2025 is approximately 73,000 operations per year.

Based on annual compound growth rate of one percent from year 2025 to year 2060 for Spartanburg Downtown Memorial airport, the projected annual number of operations at year 2060 is approximately 103,412. The average annual number of flights for Airway V54 is assumed to be 25 percent of the total airport operation. Therefore, the annual number of flights for Airway V54 is assumed to be 25,853.

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

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

There is one private-use heliport, one private-use ultra-light aircraft airport, and one private-use single-engine airport within a twenty mile range from the site. Because these privately-owned heliport and airports are used for small aircraft, which are low weight, low airspeeds, and low penetration capability, these helicopters and light aircrafts 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 A pipe rupture hazard analysis is part of the piping design. It is used to identify postulated break locations and layout changes, support design, whip restraint design, and jet shield design. The final design for these activities will be completed prior to fabrication and installation of the piping and connected components. The as-built reconciliation of the pipe break hazards analysis in accordance with the criteria outlined in DCD Subsection 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load.

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

---

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

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

3.7 SEISMIC DESIGN

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

Add Subsection 3.7.1.1.1 as follows:

3.7.1.1.1 Design Ground Motion Response Spectra

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

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

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

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

As shown on Figure 3.7-201, the horizontal GMRS and Unit 1 FIRS exceed the horizontal CSDRS at frequencies of about 20 to 75 hertz and 20 to 85 hertz, respectively. PGA at 100 hertz of the GMRS and Unit 1 FIRS is 0.21 g and 0.24 g, respectively. As shown on Figure 3.7-202, the vertical GMRS and Unit 1 FIRS exceed the vertical CSDRS at frequencies between about 25 to 70 hertz.

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

3.7.2.8.1 Annex Building

Add the following text to the end of DCD Subsection 3.7.2.8.1.

STD SUP 3.7-4 The annex building is designed so that it will not collapse and damage the safety related auxiliary and shield building.

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.

3.7.4.1 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 within the protected area 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.

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.

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

3.7.5.5 Free Field Acceleration Sensor

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

3.7.6 REFERENCES

201. Westinghouse Evaluation Report of William S. Lee Site-Specific Spectra, APC/WLG 0010, December 2007.

3.8 DESIGN OF CATEGORY I STRUCTURES

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

3.8.5.1 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.

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

---

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. 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.

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 The program 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.

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 as part of the testing program after the piping analysis has been completed.

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 design specifications. Acceptance criteria for compliance with ASME Section III Subsection NF are cited, and applicable codes and standards are referenced. The following test requirements are included:

- Snubbers are subjected to force or displacement versus time loading at frequencies within the range of significant modes of the piping system.

- Dynamic cyclic load tests are conducted for hydraulic snubbers to determine the operational characteristics of the snubber control valve.

- Displacements are measured to determine the performance characteristics specified.

- Tests are conducted at various temperatures to verify operability over the specified range.

- Peak test loads in both tension and compression are equal to or higher than the rated load requirements.

- 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.

3. Safety-related components which utilize snubbers in their support systems are identified including the following:

   - identification of systems and components
   - number of snubbers utilized in each system and on that component
   - snubber type (s) – (hydraulic or mechanical)
   - constructed to ASME Code Section III, Subsection NF or other
   - snubber use such as shock, vibration, or dual purpose
   - those snubbers identified as dual purpose or vibration arrestor type, indication of fatigue strength evaluation for both snubber and component

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.

c. Snubber Preservice and Inservice Examination and Testing

The pre-service 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 pre-service 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 pre-service examination and initial system preoperational tests exceeds 6 months, reexamination of Items i, iv, and v 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.

The inservice examination and testing plan for applicable snubbers is prepared in accordance with the requirements of the ASME OM Code, Subsection ISTD. Snubber maintenance, repairs, replacements and modifications are performed in accordance with the requirements of the ASME OM Code, Subsection ISTD. Details of the inservice examination and testing program, including test schedules and frequencies, are reported in the inservice inspection and testing plan.

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 perservice and inservice test program implementation.

3.9.6.2.2 Valve Testing

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

STD COL 3.9-4 , and for motor-operated valves the JOG MOV PV study and ASME Code Case OMN-1 Revision 1

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

Revise the first sentence of the second paragraph under the paragraph with subheading "Power Operated Valve Operability Tests" in DCD Subsection 3.9.6.2.2 to read as follows:

STD COL 3.9-4 Static and dynamic testing with diagnostic measurements will be performed on these valves as described below.

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 the valve's operability test. The required capability of the MOV is a calculated value. The preservice operability test measures the valve's actual actuator output capability, which is then compared to the valve's required capability. For an MOV, functional margin is that increment by which the MOV's actual capability exceeds the capability required to operate the MOV under design basis conditions. Using the valve functional margin and risk ranking, a periodic verification test interval/ frequency is determined. This determined test frequency is first compared to the valve's historical data to verify that any potential valve degradation during the test period would not reduce the functional margin to less than zero prior to the next scheduled periodic verification test. If the data shows that the functional margin may be reduced to less than zero, the frequency is reduced to perform the next periodic verification test prior to a loss of functional margin. If there is not sufficient data to determine whether there will be a loss of functional margin prior to the next periodic verification test, the test frequency is limited to not exceed two (2) refueling cycles or three (3) years, whichever is longer, for high risk safety significant components, and is limited to not exceed three (3) refueling cycles or five (5) years, whichever is longer, for low risk safety significant components.

A motor operated valve with an adequate functional margin is capable of opening and/or closing under design basis conditions.

Design Basis Verification Test - Prior to power operation a design basis verification (operability) test will be performed on each motor-operated valve so as to verify the capability of each valve to meet it's safety-related design requirements. The test is performed at conditions that are as close to design basis conditions as practicable.

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. Active and passive power-operated valves upon which operability testing may be performed are identified in DCD Table 3.9-16.

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 Preservice test 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.

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." 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 for in ASME OM Code Subsection ISTC. Implementation of the program described will require request for relief, unless Code Case OMN-1, Revision 1 is approved by NRC in Regulatory Guide 1.192, or the case has been incorporated into the OM Code on which the IST program is based, and that code is approved in 10 CFR 50.55a(b).

Normal residual heat removal system containment penetration relief valve (RNS-V021) and containment isolation motor-operated valve (RNS-V023) are subjected to containment leak testing by pressurizing the lines in the reverse direction to the flow of a containment leak via this path. This test method requires a Relief Request in the IST Program.

DCD Table 3.9-16, Note 20 applies to the main steam isolation valves and main feedwater isolation valves (SGS-V040A/B, V057A/B). The valves are not full stroke tested quarterly at power since full valve stroking results in a plant transient during normal power operation. Therefore, these valves are full stroke tested on a cold shutdown frequency basis. The full stroke testing is a full "slow" closure operation. The large size and fast stroking nature of the valve makes it advantageous to limit the number of fast closure operations which the valve experiences. The timed slow closure verifies the valves operability status and that the valve is not mechanically bound but does not fully satisfy Code exercising requirements. This test condition requires a Relief Request in the IST Program.

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 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 after the construction of the piping systems and prior to fuel load.

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 Subsection 3.9.6.

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.

3.9.9 REFERENCES

201. Joint Owners Group (JOG) Motor Operated Valve Periodic Verification Study, MPR 2524-A, ADAMS ML063490199, November 2006.

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.”

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

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

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

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

3.11.5 COMBINED LICENSE INFORMATION ITEM FOR EQUIPMENT QUALIFICATION FILE

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

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

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

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

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

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

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

Table 13.4-201 provides milestones for EQ implementation.

APPENDIX 3A
HVAC DUCTS AND DUCT SUPPORTS

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

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

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

APPENDIX 3C
REACTOR COOLANT LOOP ANALYSIS METHODS

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

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

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

APPENDIX 3E
HIGH-ENERGY PIPING IN THE NUCLEAR ISLAND

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

APPENDIX 3F
CABLE TRAYS AND CABLE TRAY SUPPORTS

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

APPENDIX 3G
NUCLEAR ISLAND SEISMIC ANALYSES

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

APPENDIX 3H
AUXILIARY AND SHIELD BUILDING CRITICAL SECTIONS

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

APPENDIX 3I
EVALUATION FOR HIGH FREQUENCY SEISMIC INPUT

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