Levy Nuclear Plant Units 1 and 2
COL Application
Part 2, Final Safety Analysis Report

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

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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 text to the end of DCD Subsection 3.2.1.

LNP SUP 3.2-1

There are no safety-related structures, systems, or components outside the scope of the DCD, except for roller compacted concrete (RCC) which is classified as a seismic Category I, safety-related structure. See Table 3.2-2R. Refer to Subsections 2.5.4.5 and 2.5.4.12 for a discussion of safety-related RCC.

3.2.1.3 Classification of Building Structures

Add the following text to the end of DCD Subsection 3.2.1.3.

LNP SUP 3.2-2

The seismic classification of the makeup water pump house (See Figure 1.1-201, Sheet 2), Unit 1 freshwater raw water pump house, Unit 2 freshwater raw water pump house, Unit 1 potable water pump house, and Unit 2 potable water pump house are provided in Table 3.2-2R.

3.2.2 AP1000 CLASSIFICATION SYSTEM

Add the following text to the end of DCD Subsection 3.2.2.

LNP SUP 3.2-1

There are no safety-related structures, systems, or components outside the scope of the DCD, except for roller compacted concrete (RCC) which is classified as a seismic Category I, safety-related structure. See Table 3.2-2R. Refer to Subsections 2.5.4.5 and 2.5.4.12 for a discussion of safety-related RCC.

Table 3.2-2R
Seismic Classification of Building Structures

	Structure	Category
DCD	Nuclear Island Basemat Containment Interior Shield Building Auxiliary Building Containment Air Baffle	C-I
	Containment Vessel	C-I
	Plant Vent and Stair Structure	C-II
	Turbine Building	NS
	Annex Building Area Outlined by Columns A-D and 8-13 Area Outlined by Columns A-G and 13-16	NS
	Annex Building Area Outlined by columns E – I.1 and 2-13	C-II
	Radwaste Building	NS
	Diesel-Generator Building	NS
	Circulating Water Pumphouse and Towers	NS
LNP SUP 3.2-2	Unit 1 Freshwater Raw Water Pump House	NS
	Unit 2 Freshwater Raw Water Pump House	NS
	Makeup Water Pump House	NS
	Unit 1 Potable Water Pump House	NS
	Unit 2 Potable Water Pump House	NS
LNP SUP 3.2-1	Roller Compacted Concrete	C-I

DCD

C-I – seismic Category I
C-II – seismic Category II
NS – Non-seismic

Note:

1. Within the broad definition of seismic Category I and II structures, these buildings contain members and structural subsystems the failure of which would not impair the capability for safe shutdown. Examples of such systems

would not impair the capability for safe shutdown. Examples of such systems would be elevators, stairwells not required for access in the event of a postulated earthquake, and nonstructural partitions in nonsafety-related areas. These substructures are classified as non-seismic.

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 text to the end of DCD Subsection 3.3.1.1.

LNP COL 3.3-1
LNP COL 3.5-1

The wind velocity characteristics for the Levy Nuclear Plant, Units 1 and 2 (LNP 1 and 2) are given in Subsection 2.3.1.2.2. 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 text to the end of DCD Subsection 3.3.2.1.

LNP COL 3.3-1
LNP COL 3.5-1

The tornado characteristics for the LNP 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 text to the end of DCD Subsection 3.3.2.3.

STD COL 3.3-1
LNP COL 3.5-1

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 text to the end of DCD Subsection 3.3.3.

LNP COL 3.3-1

The LNP 1 and 2 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 will not have a tornado-initiated failure of structures and components that compromises the safety of AP1000 safety-related structures and components (see also Subsection 3.5.4).

Subsection 1.2.2 discusses differences between the plant specific site plan (see Figure 1.1-201) and the AP1000 typical site plan shown in DCD Figure 1.2-2.

There are no other structures adjacent to the nuclear island other than as described and evaluated in the DCD.

Missiles caused by external events separate from the tornado are addressed in Subsections 2.2 through 2.2.3, 3.5.1.5, and 3.5.1.6.

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 text to the end of DCD Subsection 3.4.1.3.

LNP 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 text.

LNP 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 text 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 text to the end of DCD Subsection 3.5.1.5.

LNP COL 3.3-1
LNP 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.

3.5.1.6 Aircraft Hazards

Add the following text to the end of DCD Subsection 3.5.1.6.

LNP COL 3.3-1
LNP COL 3.5-1

The LNP 1 and 2 are remote from federal airways, airport approaches, military installations or airspace usage and; therefore, an aircraft hazards analysis is not required. Specifically:

- No federal airways, holding patterns, or approaches pass within 3.22 kilometers (2 miles) of the nuclear facility (Subsection 2.2.2.7 and Figure 2.2.1-204),
- No airports are located within 8.05 kilometers (5 miles) of the site (Figure 2.2.1-204),
- There are no airports with projected operations greater than $193d^2$ ($500\ d^2$) movements per year located within 16.10 kilometers (10 statute miles) and greater than $386d^2$ ($1000\ d^2$) outside 16.10 kilometers (10 statute miles) where d is the distance in kilometers (statute miles) from the site (Subsection 2.2.2.7 and Table 2.2.2-203),
- There are no military installations or any airspace usage that might present a hazard to the site within 32.19 kilometers (20 miles) of the site (Section 2.2).

3.5.4 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 3.5.4.

LNP COL 3.5-1

The LNP 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 will not have a 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 3.3.3).

Subsection 1.2.2 discusses differences between the plant specific site plan (see Figure 1.1-201) and the AP1000 typical site plan shown in DCD Figure 1.2-2.

There are no other structures adjacent to the nuclear island other than as described and evaluated in the DCD.

Missiles caused by external events separate from the tornado are addressed in Subsections 2.2 through 2.2.3, 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 text.

STD COL 3.6-1 The pipe whip restraint design and an as-designed pipe break hazards analysis will be completed in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. The as-designed pipe rupture hazard analysis including break locations based on as-designed pipe analysis will be documented in an as-designed Pipe Rupture Hazards Analysis Report. The design, analysis, and the report will be completed prior to fuel load.

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 Subsections 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

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Replace the first paragraph of DCD Subsection 3.6.4.4 with the following text.

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:

LNP SUP 3.7-3

3.7.1.1.1 Design Ground Motion Response Spectra

Figure 2.5.2-296 shows the comparison of the horizontal and vertical site-specific ground motion response spectra (GMRS) to the AP1000 certified design seismic design response spectra (CSDRS). The horizontal and vertical response spectra were developed as free-field outcrop motions on the uppermost in-situ competent material as described in Subsection 2.5.2.6. Site response analyses were conducted to evaluate the effect of 1300 meters (4300 feet) of Cretaceous and Cenozoic limestone and dolomite on the generic CEUS hard rock ground motions as described in Subsection 2.5.2.5. GMRS are enveloped by CSDRS by a factor of > 2 for both the horizontal and vertical GMRS in the entire frequency range of interest.

The horizontal GMRS peak ground acceleration at 100 hertz is 0.069g and the vertical GMRS peak ground acceleration at 100 hertz is 0.051g as noted in Table 2.5.2-226. Where design requires site specific seismic analysis, horizontal and vertical GMRS scaled to 0.1g and 0.074g respectively at 100 Hz shall be used.

For the purposes of shutdown criteria, the lower of the GMRS spectra scaled to 0.1g or one third of the spectra shown in DCD Figure 3.7-1 shall be used.

Add Subsection 3.7.1.1.2 as follows:

3.7.1.1.2 Foundation Input Response Spectra

The nuclear island is supported on 10.7 meters (35 feet) of roller compacted concrete over rock formations at the site as described in Subsection 2.5.4.5. The seismic Category II Annex Building and other adjacent non-seismic structures are supported on drilled shafts. The upper most in-situ competent rock occurs approximately 10.7 meters (35 feet) below the nuclear island foundation elevation. Foundation input response spectra (FIRS) were developed for the nuclear island at the top of the competent rock per Subsection 2.5.2.5. Figure 3.7-201 shows a comparison of the horizontal and vertical nuclear island FIRS to the CSDRS. The FIRS is enveloped by CSDRS by a factor of > 3 for both the horizontal and vertical FIRS in the entire frequency range of interest. These margins are large enough to accommodate any variation in Soil Structure Interaction (SSI) effects due to adjacent drilled shaft supported structures, when compared to the generic SSI analyses documented in the DCD Subsections 3.7.1.4 and 3.7.2.8.1.

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 seismic Category I auxiliary building and shield building.

LNP SUP 3.7-5 Peak foundation elevation displacement resulting from a GMRS scaled to 0.1g is conservatively computed to be less than 2.5 cm (1 in.). Considering that 5 cm (2 in.) seismic gaps are installed between the Annex Building foundation and the Auxiliary Building, no seismic interaction at the Annex Building foundation elevation is expected.

3.7.2.8.2 Radwaste Building

Add the following text after the first paragraph of DCD Subsection 3.7.2.8.2.

Peak foundation elevation displacement resulting from a GMRS scaled to 0.1g is conservatively computed to be less than 2.5 cm (1 in.). Considering that 5 cm (2 in.) seismic gaps are installed between the Radwaste Building foundation and the Auxiliary Building, no seismic interaction at the Radwaste Building foundation elevation is expected.

3.7.2.8.3 Turbine Building

Add the following text to the end of DCD Subsection 3.7.2.8.3.

Peak foundation elevation displacement resulting from a GMRS scaled to 0.1g is conservatively computed to be less than 2.5 cm (1 in.). Considering that 5 cm (2 in.) seismic gaps are installed between the Turbine Building foundation and the Auxiliary Building, no seismic interaction at the Turbine Building foundation elevation is expected.

3.7.2.12 Methods for Seismic Analysis of Dams

Add the following text to the end of DCD Subsection 3.7.2.12.

LNP COL 3.7-1 There are no existing dams that can affect the site interface flood level as specified in DCD Subsection 2.4.1.2 and discussed in FSAR Subsection 2.4.4.

3.7.4.1 Comparison with Regulatory Guide 1.12

Add the following text 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 text 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.

3.7.4.4 Comparison of Measured and Predicted Responses

Add the following text 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 text 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

LNP 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 text.

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 Analysis of Nuclear Island Structures

Replace DCD Subsection 3.7.5.4 with the following text.

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.

3.8 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 text 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

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Add the following after the last paragraph under DCD subheading Request 3) and prior to DCD subheading Other Applications.

STD COL 3.9-5 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, and/or Repair 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 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.

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 the 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 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 (Reference 201) and ASME Code Case OMN-1 Revision 1 (Reference 202)

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 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:

STD COL 3.9-4

3.9.6.2.4 Valve Preservice Tests

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

3.9.6.2.5 Valve Replacement, Repair, and Maintenance

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 forth 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 the 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

The design specifications and as-designed design reports prepared for major ASME Section III components and ASME Code, Section III piping will be available for NRC audit prior to fuel load.

The design specifications prepared for ASME Section III auxiliary components and valves will be available for NRC audit prior to fuel load.

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 ML 063490199, November 2006.

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

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 text 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.