

Bellefonte Nuclear Plant, Units 3 & 4
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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.

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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.

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 text 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.

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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.

BLN COL 3.3-1 The wind velocity characteristics for the Bellefonte Nuclear Plant, Units 3 and 4
BLN COL 3.5-1 (BLN), are given in **Subsection 2.3.1.5**. 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.

BLN COL 3.3-1 The tornado characteristics for the BLN are given in **Subsection 2.3.1.4**. These
BLN COL 3.5-1 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 text 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
BLN COL 3.5-1 AP1000 plant is bounded by the evaluation of the buildings and structures in a
single unit.

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3.3.3 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 3.3.3.

BLN COL 3.3-1 The BLN 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 3.5.4**).

Subsection 1.2.2 discusses differences between the plant specific site plan (see **Figure 1.1-202**) 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**.

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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.

BLN 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.

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

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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 (or P1 as identified in SRP 3.5.1.3) is less than 1×10^{-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 reinforced concrete 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 text to the end of DCD Subsection 3.5.1.5.

BLN COL 3.5-1 The gate house, administrative building, security control building, warehouse and
BLN COL 3.3-1 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.

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The missiles generated by events near the site are discussed and evaluated in [Subsection 2.2.3](#). With the exception of a potential barge explosion, the effects of external events on the safety-related components of the plant are insignificant. The probability of a missile generating barge explosion is determined to be less than 10^{-7} events per year. Based on Regulatory Guide 1.91, this does not represent a design basis event. This also meets the criteria of 10^{-6} occurrences per year in the [DCD Section 2.2](#) for not requiring changes to the AP1000 design for an external accident leading to severe consequences.

3.5.1.6 Aircraft Hazards

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

BLN COL 3.5-1 The approach and methodology outlined in NUREG-0800 Standard Review Plan
BLN COL 3.3-1 (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 5 and 10 statute miles, and the projected annual number of operations is less than $500D^2$, or the plant-to-airport distance is greater than 10 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 2 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:

- One federal airway passes within two miles of the plant site. High altitude airway J73 runs between Lagrange-Callaway Airport, Georgia and Nashville International Airport, Tennessee. It is primarily used by commercial aviation. The total number of flights that use the section of J73 that is within two miles of the Bellefonte site is conservatively estimated to be 21,960 annually. Applying an annual compound growth rate of 2.84 percent, the number of flights increases to 96,877 annually in the year 2060.
- One airport is located within five miles of the plant center. The Scottsboro Municipal Airport is approximately 4.9 miles from the plant center. The

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annual number of operations in year 2060 is projected to be 10,972, based on an annual compound growth rate of 1 percent from 2025, the last year for which FAA projected data is available. Although the projected number of flights does not meet the 500D² movements per year criterion, the probability of an aircraft crash from this airport was determined, because it is less than five miles from the site.

- No airports having more than 500D² movements per year are located within 10 miles of the site and no airports having more than 1,000D² movements per year are located beyond 10 miles of the site.
- There are no military training routes within 10 miles of the site.

There are three private use heliports, one private-use single-engine airport, and one public-use airport within a ten to twenty mile range from the site. Because these public and privately-owned heliports and airports are used for small aircrafts, 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.

Utilizing the above data, the total probability of an aircraft crash into the plant was determined to be 8.8×10^{-7} per year. 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 Scottsboro Municipal Airport is primarily used by single-engine aircraft. There are one-multi-engine, 22 single-engine and six ultra lights aircraft based at the field. All aircraft are piston type engine. The weight of single-engine aircraft is approximately 2400 lbs and 3600 lbs for multi-engine aircraft. The weight of ultra light aircraft is approximately 300 lbs. Light general aviation aircraft are not considered a significant hazard to nuclear power stations because of their low weight, low airspeeds, short distance landing capability, high maneuverability and low penetration capability. Thus, the types of aircraft that use the Scottsboro Municipal Airport are not likely to be able to damage a vital structure to the extent that would cause exposures in excess of 10 CFR Part 100 criteria.
- 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.

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- Scottsboro Municipal Airport is approximately 4.9 miles west-southwest of the site. The runways for this airport are not aligned with the Bellefonte site and flight paths that fly over the site are not considered to be normal or likely. For conservatism in the calculation, 20 percent of the flight trajectories approaching or departing this airport are assumed to fly directly over the Bellefonte site.
- The FAA does not project growth at the Scottsboro Municipal Airport subsequent to 2025, the last year for which projections are available. The number of flights for the year 2060 is determined assuming a compound growth rate of 1 percent per year after 2025. The FAA projections show 7745 operations per year from 2001 to 2025, which exceeds the current number of operations of 7686. Assuming annual compound traffic growth rate of 1 percent from year 2025 to year 2060, the projected annual number of operation at year 2060 is approximately 10,972. This future airport operation is still less than the acceptable annual operation number of 12,005. The difference between the plant-to-airport distance and the acceptable distance of 5 miles to meet the proximity criteria in SRP 3.5.1.6 is approximately 0.1 miles. The Scottsboro Municipal Airport nearly meets the proximity criterion without further detailed analysis for the airport.
- 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 of these structures.
- 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 can be qualitatively shown to be much lower than the calculated value. Therefore, the aircraft hazards pose no undue risk to the health and safety of the public.

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3.5.4 COMBINED LICENSE INFORMATION

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

BLN COL 3.5-1 The BLN 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 1.1-202**) 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**.

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**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 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

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.

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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

BLN SUP 3.7-3 **Figures 3.7-201** and **3.7-202** show a comparison of the horizontal and vertical site-specific ground motion response spectra (GMRS) to the certified seismic design response spectra (CSDRS), respectively. The horizontal and vertical response spectra were developed at the top of a hypothetical outcrop of competent material at the elevation of the AP1000 basemat as described in **Section 2.5.2.4.4**. Bedrock at 588.6 ft. (NAVD 88), the elevation of the AP1000 basemat, has a shear wave velocity that exceeds 9,200 feet per second as described in **Section 2.5.4.7**; thus no site response analysis was required to develop the GMRS.

As shown on **Figure 3.7-201**, the horizontal GMRS exceeds the CSDRS at frequencies of about 15 to 80 hertz. Peak ground acceleration at 100 hertz is about 0.24g. As shown on **Figure 3.7-202**, the vertical GMRS exceeds the CSDRS at frequencies of about 20 to 85 hertz.

Similar high-frequency exceedances were evaluated by Westinghouse in **DCD Appendix 3I** using a 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 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 (and therefore the site specific exceedances) will not adversely affect the systems, structures, or components of the plant.

3.7.2.12 Methods for Seismic Analysis of Dams

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

BLN 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**.

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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

BLN SUP 3.7-4 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 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.

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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

BLN 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.

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3.7.5.4 Reconciliation of Seismic Analyses 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**.

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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**.

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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.

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.

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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 SUP 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.

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- 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.

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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

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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 percent reactor power operation and is completed within 12 calendar months after attaining 5 percent 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 practical, 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

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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 according to an established sampling plan each fuel cycle. 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.

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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 or inservice testing. These tests that establish reference and IST values are performed under conditions as near as practicable to those expected during the IST. Reference values are established only when a valve is known to be operating acceptably.

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

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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:

Power-Operated Valve Operability Tests - The safety-related, power-operated valves (POVs) are required by the procurement specifications to have the capabilities to perform diagnostic testing to verify the capability of the valves to perform their design basis safety functions. 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.
-

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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.

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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 ([Reference 203](#) and [Reference 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:

- 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 |

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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 **Reference 203** and **Reference 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 high risk 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.

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 Valves 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 Subsection 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

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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

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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

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

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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.

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

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

- (1) The adequacy of the diagnostic test interval for each motor-operated valve (MOV) is evaluated and adjusted as necessary, but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of OMN-1.
- (2) The potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
- (3) Risk insights are applied using MOV risk ranking methodologies accepted by the NRC on a plant-specific or industry-wide basis, consistent with the conditions in the applicable safety evaluations.
- (4) Consistent with the provisions specified for Code Case OMN-11 the potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

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

3.9.8 COMBINED LICENSE INFORMATION

3.9.8.2 Design Specifications and Reports

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

STD COL 3.9-2 Reconciliation of the as-built piping (verification of the thermal cycling and stratification loading considered in the stress analysis discussed in **DCD**

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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** and **Subsection 14.2.9.2.22.**

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.
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STD SUP 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	005B
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.

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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.

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**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.

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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.

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APPENDIX 3A
HVAC DUCTS AND DUCT SUPPORTS

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

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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.

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APPENDIX 3C
REACTOR COOLANT LOOP ANALYSIS METHODS

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

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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.

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APPENDIX 3E
HIGH-ENERGY PIPING IN THE NUCLEAR ISLAND

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

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APPENDIX 3F
CABLE TRAYS AND CABLE TRAY SUPPORTS

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

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APPENDIX 3G
NUCLEAR ISLAND SEISMIC ANALYSES

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

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APPENDIX 3H
AUXILIARY AND SHIELD BUILDING CRITICAL SECTIONS

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

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APPENDIX 3I
EVALUATION FOR HIGH FREQUENCY SEISMIC INPUT

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