Appendix 3A HVAC Ducts and Duct Supports

This appendix provides the design criteria for seismic Category I and II HVAC ducts and their supports. These design criteria maintain structural integrity for seismic Category I and II ducts and functional capability for seismic Category I duct.

The structural components of a typical HVAC duct system include the sheet metal ducts, stiffeners for the ducts, duct supports, and other inline components such as duct heaters, dampers, etc.

3A.1 Codes and Standards

The design of the HVAC ducts and their supports conform to the following codes and standards:

- ASME N509-1989(R1996), Nuclear Power Plants Air Cleaning Units and Components
- ASME/ANSI AG-1-1997, Code on Nuclear Air and Gas Treatment
- American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, AISC-N690-1994
- American Iron and Steel Institute (AISI), Specification for the Design of Cold Formed Steel Structural Members, 1996 Edition and Supplement No. 1, July 30, 1999
- SMACNA, HVAC Duct Construction Standards, Metal and Flexible, Second Edition 1995.

3A.2 Loads and Load Combinations

3A.2.1 Loads

3A.2.1.1 Dead Load (D)

Dead load includes the weight of the duct sheet, stiffeners and inline components such as duct heaters and dampers. It also includes permanently attached items such as insulation and fireproofing, where applicable, and the weight of the duct supports. Temporary items used during construction or maintenance are removed prior to operation.

3A.2.1.2 Construction Live Load (L)

Live load consists of a load of 250 pounds to be applied only during construction or maintenance on an area of 10 square inches on the duct at a critical location to maximize flexural and shear stresses. This load is not combined with seismic loads.

3A.2.1.3 Pressure (P)

The duct metal thickness and stiffener requirements are based on maximum system design pressures. SMACNA or ASME guidelines, as applicable, are used in the design of duct metal thickness and stiffener requirements.

The pressure loads occur during normal plant operation, including plant start up testing, damper closure and normal airflow. Occasionally, overpressure transient loads such as rapid damper closure may also produce short duration pressure differential.

3A.2.1.4 Safe Shutdown Earthquake (E_s)

Seismic response of the HVAC ductwork and its support system are produced due to seismic excitation of the supports.

3A.2.1.5 Wind Loads (W)

Ductwork within partially or fully vented buildings is subject to wind effects. Design wind loads are discussed in Section 3.3.

3A.2.1.6 Tornado Loads (W_t)

Ductwork within partially or fully vented buildings is subject to tornado differential pressure effects. Tornado loads are discussed in Section 3.3. Seismic Category I HVAC ductwork is protected from impact by tornado missiles.

3A.2.1.7 External Pressure Differential Loads (P_A)

Seismic Category I HVAC ductwork and its supports are designed to withstand dynamic external pressure differential loads resulting from postulated accident conditions. Usually HVAC ducts are routed outside the areas of potential pipe break.

3A.2.1.8 Thermal (T_O/T_A)

Stresses on the supports resulting from the ductwork expansion due to temperature changes are avoided by designing the system to take care of the expansion or by utilizing expansion joints. For ducts of gasketed companion angle construction, thermal loads are negligible. For ducts exposed to higher temperatures during a postulated accident condition, an evaluation is performed on a case by case basis for its effect.

3A.2.2 Load Combinations

The load combinations for various service levels are as follows:

Service Level		Load Combination
A	(Construction / maintenance)	$D + L + P + T_O$
A	(Normal Operating Condition)	$D + P + T_O$
B	(Severe Condition)	$D + W + P + T_O$
C	(Extreme Condition)	$D + E_s + P + T_O$
C	(Extreme Condition)	$D + W_t + P + T_O$
D	(Abnormal Condition)	$D + P + P_A + E_s + T_A$

3A.3 Analysis and Design

The HVAC duct support system is designed to maintain structural integrity of the duct. Function is not required for the seismic Category II ductwork. The stresses are maintained within the allowable limits specified in Subsection 3A.3.4. Section properties and masses are calculated in accordance with SMACNA standard.

The damping values for seismic analysis are as follows:

- Welded HVAC Ductwork 4 percent
- Bolted HVAC Ductwork 7 percent

The duct design due to pressure loads is based on ASME/ANSI AG-1 for seismic Category I ducts and SMACNA for seismic Category II ducts.

The global behavior of the duct is determined from the overall bending of the duct between the supports. It is similar to the beam type bending. The dead load is combined with the seismic inertial load to determine the maximum bending moment. For determining the section modulus, the corners of the duct are considered effective. The corner length in each direction equals 32 times the thickness of the duct (t) for this purpose.

3A.3.1 Response Due to Seismic Loads

The methodology for seismic analysis is provided in Subsection 3.7.3. Seismic loads are determined by either using the equivalent static load method of analysis or by performing dynamic analysis.

Stresses are determined for the seismic excitation in two horizontal and one vertical direction. The stresses in the three directions are combined using the square root of sum of the squares (SRSS) method or the 100-40-40 method as described in Subsection 3.7.3.6.

3A.3.2 Deflection Criteria

Deflections for panels and stiffeners conform to the limits stated in the “Code on Nuclear Air and Gas Treatment.”

3A.3.3 Relative Movement

Clearances are provided for allowing relative movement between equipment, other commodities, and HVAC system.

3A.3.4 Allowable Stresses

The basic stress allowables for the HVAC ducts are in accordance with paragraph SA-4220 of ASME/ANSI AG-1.

The basic stress allowables for duct supports utilizing rolled structural shapes are in accordance with ANSI/AISC N-690 and the supplemental requirements described in Subsection 3.8.4.5.2. The basic stress allowables for supports utilizing light gage cold rolled channel type sections are based on the manufacturer's published catalog values.

Service Level A and B	Basic Allowable
Service Level C and D	1.6 times basic allowable for tension and 1.4 times basic allowable for compression

3A.3.5 Connections

Connections are designed in accordance with the applicable codes and standards listed in Section 3A.1. For connections used with light gage cold rolled channel type sections, design is based on the manufacturer's published catalog values. Supports are attached to the building structure by bolted or welded connections. Fastening of the supports to concrete structures meets the supplemental requirements given in Subsection 3.8.4.5.1.

Appendix 3B Leak-Before-Break Evaluation of the AP1000 Piping

General Design Criterion 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of conditions associated with normal operation, anticipated transients, and postulated accident conditions. However, the dynamic effects associated with pipe rupture may be excluded when analysis demonstrates that the probability of fluid system pipe rupture is extremely low. Dynamic effects are not considered for those segments of piping that are shown mechanistically, with a large margin, not to be susceptible to a pipe rupture.

The dynamic effects associated with pipe rupture include effects such as pipe break reaction loads, jets and jet impingement, subcompartment pressurization loads, and transient pipe rupture depressurization loads on other components.

The use of mechanistic pipe break to eliminate evaluation of dynamic effects of pipe rupture includes material selection, inspection, leak detection, and analysis. Subsection 3.6.3 outlines considerations relative to material selection, inspections, and leak detection. Subsection 5.2.5 describes the leak detection system inside containment. This appendix describes the analysis methods used to support the application of mechanistic pipe break to high-energy piping in the AP1000.

The analysis and criteria to eliminate dynamic effects of pipe breaks are encompassed in a methodology called leak-before-break (LBB). This methodology has been validated by theoretical investigations and test demonstrations sponsored by the industry and the NRC.

The primary regulatory documents for leak-before-break analyses are General Design Criterion No. 4 (GDC-4), Draft Standard Review Plan 3.6.3 (SRP 3.6.3) (Reference 1), and NUREG-1061, Volume 3 (Reference 2). Although SRP 3.6.3 has been issued only as a draft, its provisions are followed as guidelines to leak-before-break analyses.

Leak-before-break methodology has been applied to the reactor coolant loop and high-energy auxiliary line piping in operating nuclear power plants. The leak-before-break analysis used to support the piping design of the AP1000 is an application of the same methodology used in leak-before-beak evaluations previously accepted by the NRC.

In the AP1000, leak-before-break evaluations are performed for the reactor coolant loop, the surge line, selected other branch lines containing reactor coolant down to and including 6-inch diameter nominal pipe size, and portions of the main steam line. Those lines not qualified to the leak-before-break criteria are evaluated using the pipe rupture protection criteria outlined in Subsections 3.6.1 and 3.6.2.

This appendix provides a leak-before-break analysis for the applicable piping systems. Table 3B-1 provides a list of AP1000 leak-before-break piping systems.

3B.1 Leak-before-Break Criteria for AP1000 Piping

The methodology used for leak-before-break analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2). The steps are:

- Evaluate potential failure mechanisms
- Perform bounding analysis

3B.2 Potential Failure Mechanisms for AP1000 Piping

In high-energy piping, there are material degradation mechanisms that could adversely affect the integrity of the system as well as its suitability for leak-before-break analysis. The following lists potential degradation (or "failure") mechanisms:

- Erosion-corrosion induced wall thinning
- Stress corrosion cracking (SCC)
- Water hammer
- Fatigue
- Thermal aging
- Thermal stratification
- Other mechanisms

The stainless steel piping is fabricated of SA312TP316LN or SA312TP304L material. The type 304L material is used in the accumulator discharge lines. The main steam piping is fabricated of SA335 Grade P11. The welds are made by the gas tungsten arc welding (GTAW) method.

The various degradation mechanisms are discussed in the following subsections.

3B.2.1 Erosion-Corrosion Induced Wall Thinning

Primary Loop Piping

Wall thinning by erosion and erosion-corrosion effects does not occur in the primary loop piping because Series 300 austenitic stainless steel material is highly resistant to these effects. The coolant velocity in the AP1000 primary loop is about 76 feet per second. This flow velocity is not expected to create erosion-corrosion effects since stainless steels are considered to be virtually immune (Reference 3). A review of erosion-corrosion in nuclear power systems (Reference 4) reported that "stainless steels are increasingly being used due to their excellent resistance to erosion-corrosion, even at high water velocities, 40 m/s (131 ft/sec)." The bend radii in the AP1000 hot and cold legs are greater than the bend radii used in the crossover legs of operating plants. There is no record of erosion-corrosion induced wall thinning in the primary loops of operating plants.

Auxiliary Stainless Steel Piping

Wall thinning by erosion-corrosion effects does not occur in the auxiliary stainless steel piping because Series 300 austenitic stainless materials are highly resistant to these effects. The coolant velocity in these systems is lower than in comparable systems in operating Westinghouse-designed pressurized water reactors. There is no record of erosion-corrosion induced wall thinning in the stainless steel piping of operating plants.

Main Steam Line

Main steam lines in the AP1000 are fabricated from SA335 Grade P11 Alloy steel. Erosion-corrosion induced wall thinning is not expected in the main steam line. Extensive work has been done investigating erosion-corrosion in carbon steel pipes. The main steam line has low susceptibility to erosion due to the pipe material composition, which has sufficient levels of chromium to preclude erosion-corrosion material loss. Susceptibility is also low due to the relatively high operating temperature and the high quality steam in the main steam line.

Based on the above discussion, erosion-corrosion induced wall thinning does not have an adverse effect on the integrity of the AP1000 leak-before-break piping systems.

3B.2.2 Stress Corrosion Cracking

Stress corrosion cracking is not expected to occur in the AP1000 piping systems because the three conditions necessary for stress corrosion cracking to take place are not present. If any of these three conditions is not present, stress corrosion cracking will not take place. The three conditions are:

- There must be a corrosive environment.
- The material itself must be susceptible.
- Tensile stresses must be present in the material.

Primary Loop Piping

During plant operation, the reactor coolant water chemistry is monitored and maintained within specific limits (see Subsection 5.2.3 for a discussion of reactor coolant chemistry). Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking. The major water chemistry control standards are included in the plant operating procedures as a condition for plant operation.

The key to avoidance of a corrosive environment is control of oxygen. During normal power operation, oxygen concentration in the reactor coolant system is controlled to extremely low levels by controlling charging flow chemistry and maintaining a hydrogen overpressure in the reactor coolant at specified concentrations. Halogen concentration is controlled by maintaining concentrations of chlorides and fluorides within the specified limits. During plant operations, the likelihood of stress corrosion cracking in the primary loop piping systems is very low.

The elements of a water environment known to increase the susceptibility of austenitic stainless steel to stress corrosion are oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (for example, sulfides, sulfites, and thionates). Pipe cleaning standards prior to operation and careful water chemistry control during plant operation are applied to prevent the occurrence of a corrosive environment. Before being placed in service the piping is cleaned. During flushes and preoperational testing, water chemistry is controlled according to written specifications. Standards on chlorides, fluorides, conductivity, and pH are included in the guidelines for water for cleaning the piping.

Series 300 stainless steel materials have been chosen for the AP1000 due to their proven operating experience. These materials have operated in low-oxygen or no-oxygen environments with no incidents for a number of years. The requirements of Regulatory Guide 1.44 will be used to maintain the experiences of the PWR applications for the use of Series 300 stainless steel materials.

Design tensile stresses in the reactor coolant loop are within the ASME Code, Section III allowables. Residual tensile stresses are expected in the welds and such stresses are not considered when designing by the ASME Code, Section III because these stresses are self-equilibrating and do not affect the failure loads. The residual stresses should not be more severe than for the operating Westinghouse pressurized water reactor plants (which have not experienced stress corrosion cracking in the primary loop).

The material used for buttering nozzles at the stainless-to-carbon steel safe ends is a high nickel alloy. The nickel-chromium-iron alloy selected and qualified for this application is not susceptible to primary water stress corrosion cracking.

Auxiliary Stainless Steel Piping

The discussion above regarding the necessary conditions for primary loop piping stress corrosion cracking is also applicable to the other stainless steel piping of the primary system.

Series 300 stainless steel materials have been chosen for the AP1000 due to their proven operating experience. These materials have operated in low-oxygen or no-oxygen environments with no incidents for a number of years. The requirements of Regulatory Guide 1.44 will be used to maintain the experiences of the PWR applications for the use of Series 300 stainless steel materials.

Design tensile stresses in the other stainless steel piping are within the ASME Code, Section III allowables. Residual tensile stresses are expected in the welds; however, the residual stresses should not be more severe than for the operating Westinghouse pressurized water reactor plants (which have not experienced stress corrosion cracking in the auxiliary stainless steel piping).

Main Steam Line

The main steam piping is constructed from ferritic steel. Stress corrosion cracking in ferritic steels commonly result from a caustic environment. A source of a caustic environment in the main steam piping would be moisture carryover from the steam generator. However, the secondary side water treatment utilizes all volatile treatment. All volatile treatment effectively precludes causticity in the steam generator bulk liquid environment. For some operating plants prior to implementing all volatile treatment, the phosphate water treatment caused a caustic chemical imbalance resulting in stress corrosion cracking of steam generator tubing. Under all volatile treatment water treatment conditions, there is no instance of caustic stress corrosion cracking on the ferritic steam lines indicating no significant caustic carryover. The operating secondary side chemistry precludes stress corrosion cracking on the ferritic main steam line.

Based on the above discussion, stress corrosion cracking does not have an adverse effect on the integrity of AP1000 leak-before-break piping systems.

3B.2.3 Water Hammer

Primary Loop Piping

The reactor coolant loop is designed to operate at a pressure greater than the saturation pressure of the coolant, thus precluding the voiding conditions necessary for water hammer to occur. The reactor coolant primary system is designed for Level A, B, C, and D (normal, upset, emergency, and faulted) service condition transients. The design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening have been considered in the system design. Other valve and pump actuations cause relatively slow transients with no significant effect on the system dynamic loads.

To provide dynamic system stability, reactor coolant parameters are controlled. Temperature during normal operation is maintained within a narrow range by control rod positioning. Pressure is controlled within a narrow range for steady-state conditions by pressurizer heaters and pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle. The operating transients of the reactor coolant system primary loop piping are such that significant water hammer loads are not expected to occur.

Auxiliary Stainless Steel Piping

The passive core cooling system and automatic depressurization system are designed to minimize the potential for water hammer induced dynamic loads. Design features include:

- Continuously sloping core makeup tank and passive residual heat exchanger inlet lines to eliminate local high points

- Inlet diffusers in the core makeup tanks to preclude adverse steam and water interactions

- Vacuum breakers in the discharge lines of the automatic depressurization valves connected to the pressurizer

The AP1000 pressurizer spray control valve is similar to what is used in the operating plants. There is no history of water hammer caused by the spray control valve.

The normal residual heat removal system isolation valves are slow closing valves, identical to operating plants, and therefore would not be a source of water hammer.

These features minimize the potential of water hammer in the auxiliary stainless steel piping system.

Main Steam Line

The steam lines are not subject to water hammer by the nature of the fluid transported. The following system design provisions address concerns regarding steam hammer within the main steam line and identify the significant dynamic loads included in the main steam piping design.

- Design features that prevent water slug formations are included in the system design and layout. In the main steam system, these include the use of drain pots and the proper sloping of lines.

- The operating and maintenance procedures that protect against a potential occurrence of steam hammer include system operating procedures that provide for slowly heating up (to avoid condensate formation from hotter steam on colder surfaces), operating procedures that caution against fast closing of the main steam isolation valves except when necessary, and operating and maintenance procedures that emphasize proper draining.

- The stress analyses for the safety-related portion of the main steam system piping and components include the dynamic loads from rapid valve actuations, including actuation of the main steam isolation valves and the safety valves.

Based on the above discussion, water hammer does not have an adverse effect on the integrity of AP1000 leak-before-break piping systems.

3B.2.4 Fatigue

Low-Cycle Fatigue

Low-cycle fatigue due to normal operation and anticipated transients is accounted for in the design of the piping system. The Class 1 piping systems comply with the fatigue usage requirements of the ASME Code, Section III. The Class 2 and 3 piping systems comply with the stress range reduction factors of the ASME Code, Section III.

Due to the nature of operating parameters, main steam line piping (Class 2) and the Class 3 portion of the accumulator piping, are not subjected to any significant transients to cause low-cycle fatigue.

Based on the above discussion, low-cycle fatigue is not a concern of AP1000 leak-before-break piping systems.

High-Cycle Fatigue

High-cycle fatigue loads in the system result primarily from pump vibrations. The steam generator is designed so that flow-induced vibrations in the tubes are avoided (see Subsection 5.4.2). The loads from reactor coolant pump vibrations are minimized by criteria for pump shaft vibrations during hot functional testing and operation. During operation, an alarm signals when the reactor coolant pump vibration is greater than the limits.

With these precautions taken, the likelihood of leakage due to fatigue in piping systems evaluated for leak-before-break is very small.

3B.2.5 Thermal Aging

Stainless Steel Piping

Piping used in the reactor coolant loop and other auxiliary lines are wrought stainless steel materials, rather than cast materials, so that thermal aging concerns are not expected for the AP1000 piping and fittings. The welds used in the assembly of the AP1000 are gas tungsten arc welds (GTAW). These welds are essentially as resistant to the effects of thermal aging as the base metal materials. This is due to the typically low ferrite contact in welds which results in minimal impact from thermal aging. Based on this information, thermal aging of weld materials and piping used in the AP1000 is not an issue.

Main Steam Lines

The main steam piping system does not have cast materials. The welding process used on these lines is also gas tungsten arc weld (GTAW).

There are no thermal aging concerns for the carbon steel piping of the main steam line and the alloy steel of the main feedwater piping.

The material used for the main steam piping system is not susceptible to dynamic strain aging effects.

3B.2.6 Thermal Stratification

Leak-before-break analyses include consideration of the loads and stresses due to thermal stratification.

Thermal stratification occurs only in a pipe that has a susceptible geometry and low flow velocities. A temperature difference between the flowing fluid and stagnant fluid is also a prerequisite.

The design of piping and component nozzles in the AP1000 includes provisions to minimize the potential for and the effects of thermal stratification, cycling, and striping, pursuant to actions requested in several NRC bulletins, as discussed below.

Primary Loop Piping

Thermal stratification in the reactor coolant loops resulting from actuation of passive safety features is evaluated as a design transient. Stratification effects due to both Level B and Level D service conditions are considered. The criteria used in the evaluation of the stress in the loop piping due to stratification is the same as that applicable for other Level B and Level D service conditions.

Auxiliary Stainless Steel Piping

Pursuant to the actions requested in NRC Bulletin 88-11, the pressurizer surge line is analyzed to demonstrate that the applicable requirements of the ASME Code, Section III are met. This analysis includes consideration of plant operation, thermal stratification, and thermal striping using temperature distributions and transients developed from experience on existing plant monitoring programs.

Pursuant to the actions requested in NRC Bulletin 88-08 (cracking in piping connected to reactor coolant systems due to isolation valve leakage), a systems review of the AP1000 piping was performed in accordance with the criteria provided in Subsection 3.9.3.1.2.

The unisolable sections of the following lines which are evaluated for leak-before-break have been reviewed and are not susceptible to adverse stresses as described in NRC Bulletin 88-08:

Passive residual heat removal (PRHR) line from the hot leg, through the passive residual heat removal heat exchanger, and to the steam generator channel head
The potential for leakage through the isolation valves is not a concern for the piping extending from the reactor coolant system hot leg connection to the passive residual heat removal heat exchanger inlet, since hot leakage from the reactor coolant system would be entering a hot section of piping. Leakage exiting the passive residual heat removal heat exchanger would not be a concern since the cooled leakage would be entering a cold section of piping. This leakage would then heat up in the piping directly below the steam generator. Any amount of leakage is expected to be small, since the pressure differential across the isolation valves is about 50 psi (the difference between the hot leg and reactor coolant pump suction pressures). Activation of the passive residual heat removal system following a plant scram is not a concern, since stratification will not occur due to the high flow velocity in the passive residual heat removal return flow line.

Automatic depressurization stage 4 lines from the hot legs to the stage 4 depressurization valves
Leakage is not a concern since the squib valves are leaktight and other potential leakage flow paths have double isolation.

Pressurizer safety line from the pressurizer to the safety valve
This line is steam filled and will not experience stratified loadings.

Automatic depressurization stage 2 and 3 lines from the pressurizer to the depressurization valves
Leakage is not a concern since double isolation exists in all potential leakage flow paths.

Normal residual heat removal suction lines from the hot legs to the isolation valves
Thermal stratification in the normal residual heat removal suction lines, including leakage through the isolation valves, is considered in the ASME pipe stress and fatigue analysis of these lines.

Direct vessel injection lines
Thermal stratification in the direct vessel injection lines, including leakage through the isolation vavles, is considered in the ASME Code pipe stress and fatigue analysis of these lines.

Main Steam Line

The steam lines are not subjected to thermal stratification by the nature of fluid transported.

Based on the above discussion, thermal stratification does not have an adverse effect on the integrity of AP1000 leak-before-break piping systems.

3B.2.7 Other Mechanisms

The pipe evaluated for leak-before-break does not operate at temperature for which creep fatigue must be considered. Creep fatigue is a concern for ferritic steel piping operation at temperatures above 700°F and for austenitic stainless steel operation above 800°F.

Pipe degradation or failure by indirect causes such as fires, missiles, and component support failures is precluded by criteria for design, fabrication, inspection, and separation of potential hazards in the vicinity of the safety-related piping. The structures, larger pipe, and components in the vicinity of pipe evaluated for leak-before-break are safety-related and seismically designed or are seismically supported if nonsafety-related.

Cleavage type failures are not a concern for systems operating temperature and material used in the stainless steel piping systems. The material used in the main steam line is highly ductile and resistant to cleavage type failure at operating temperatures. The resistance to failure have been demonstrated by material fracture toughness tests.

3B.3 Leak-before-Break Bounding Analysis

The methodology used for performing the bounding analysis is consistent with that set forth in GDC-4, SRP 3.6.3 (Reference 1) and NUREG-1061, Volume 3 (Reference 2).

Bounding leak-before-break analysis for the applicable AP1000 piping systems is performed. The analysis criteria and development techniques of the bounding analysis curves (BAC) are described below. The bounding analysis curve allows for the evaluation of the piping system in advance of the final piping analysis, incorporating leak-before-break considerations early in the piping design process. The leak-before-break bounding analysis curve is used to evaluate critical points in the piping system. A minimum of two points are required to develop the bounding analysis curve. One point for the low normal stress case and the other point for the high normal stress case. If variations in pipe size, material, pressure or temperature occur for a specific piping system, an additional bounding analysis curve is generated. These points meet the following margins for leak-before-break analysis: (References 1 and 2).

- Margin of 10 on leak detection capability
- Margin of 2 on flaw size
- Establish margin of 1 on load by using absolute combination method of maximum loads

The calculations to establish the bounding analysis curves use minimum values for wall thickness at the weld counterbore and ASME Code material properties. For the main steam line lower bound material property values determined from tests of the material are used. The use of the minimum values bounds the results of larger values. Since the piping is designed and analyzed using ASME Code minimum material properties, these are used conservatively in a consistent manner for evaluation of leak-before-break evaluations. The as-built material properties are expected to be higher than the ASME Code minimum properties. Using minimum thickness instead of a nominal thickness is conservative for the stability analysis and was also used for leak-before-break in operating plants. The use of one thickness (either nominal or minimum) for both leak rate and stability calculation gives comparable overall margins for typical plant loads. The bounding analysis curves are established using the axial load from internal pressure and neglecting other axial loads. This is an appropriate approximation because experience with leak-before-break calculations has shown that the axial load due to pressure is the dominant axial load.

3B.3.1 Procedure for Stainless Steel Piping

3B.3.1.1 Pipe Geometry, Material and Operating Conditions

The following information is identified for each of the lines:

- Piping materials - 316LN/304L, Type 304L is used for the accumulator discharge line
- Normal operating temperature
- Normal operating pressure
- Pipe outside diameter
- Pipe thickness

The number of bounding analysis curves needed for each analyzable piping system is determined by a review of the combinations of the following parameters:

- Pipe size
- Pipe schedule
- Operating pressures (100 percent power and maximum stress condition)
- Operating temperatures (100 percent power and maximum stress condition)

3B.3.1.2 Pipe Physical Properties

The physical and metallurgical properties for each of the lines are determined in the following manner

- Minimum wall thickness is calculated at the weld counterbore
- The area (A) and section modulus (Z) are calculated using minimum wall thickness
- The yield strength is the ASME Code, Section II (Reference 5) minimum value, at temperature of interest
- The ultimate strength is the ASME Code, Section II (Reference 5) minimum value, at temperature of interest
- The modulus of elasticity is the ASME Code, Section II (Reference 5) at temperature of interest

3B.3.1.3 Low Normal Stress Case (Case 1)

To determine the first point of the bounding analysis curve the following steps are used.

- Calculate axial force Fp (for normal operating pressure)
- Assume a lower magnitude of bending stress. The magnitude selected is a very small number that is lower than the expected minimum bending stress.
- Calculate bending moment = (bending stress) x (section modulus)
- Calculate the leakage flaw size at 100 percent power condition for 10 times the leak detection capability (for 0.5 gpm leak detection capability, this is 10 x 0.5 = 5 gpm)
- Perform the stability analysis using the limit load methodology to obtain the critical flaw size. For AP1000 piping systems, there is no cast material and the weld process is gas tungsten arc welds (Z factor is 1.0 since weld process is gas tungsten arc welds, Reference 1.)
  - Determine the maximum loads for a critical flaw size of twice the leakage flaw size. The margin of 2 on flaw size is satisfied.
- Calculate the low normal stress and corresponding maximum stress by using:

$$\text{Stress} = \frac{\text{Axial Force}}{\text{Area}} + \frac{\text{Bending Moment}}{\text{Section Modulus}} \tag{3B-1}$$

3B.3.1.4 High Normal Stress Case (Case 2)

To determine the other endpoint of the bounding analysis curve the following steps are used.

- Axial force Fp is calculated as above for normal operating pressure
- Assume a higher magnitude of bending stress to get higher bending moment. The magnitude of bending is selected such that the corresponding maximum stress generated is close to the flow stress.
- Calculate bending moment = (bending stress) x (section modulus)
- Repeat leakage flaw size and stability calculations as outlined for the low normal stress case above

Note: For an intermediate point, calculation steps are the same as low normal or the high normal case.

3B.3.1.5 Develop the Bounding Analysis Curve

- For Case 1, normal and maximum stresses are established.
- For Case 2, normal and maximum stresses are established.
- Plot these two points with normal versus maximum stress. The curve is generated by joining these two points in a straight line. More than two points may be used if desired, to obtain a smooth curve fit between the calculated points. A typical curve is shown in Figure 3B-1.

3B.3.2 Procedure for Non-stainless Steel Piping

The procedure to develop the bounding analysis curve for the carbon steel for main steam lines is similar to that for the stainless steel and is described below.

3B.3.2.1 Pipe Geometry, Material and Operating Conditions

The following information is identified for each of the lines:

- Piping materials
- Normal operating temperature
- Normal operating pressure
- Pipe outside diameter
- Piping thickness

The number of bounding analysis curves needed for each analyzable piping system is determined by a review of the combinations of the following parameters:

- – Pipe size
- – Pipe schedule
- – Operating pressures (100 percent power and maximum stress condition)
- – Operating temperatures (100 percent power and maximum stress condition)

3B.3.2.2 Calculations Steps

- The minimum wall thickness is calculated at the weld counterbore
- The area (A) and section modulus (Z) are calculated using minimum wall thickness
- The material yield strength, ultimate strength, modulus of elasticity, stress-strain curves, and J-R curves are determined from the material tests

3B.3.2.3 Low Normal Stress Case (Case 1)

To determine the first point of the bounding analysis curve the following steps are used.

- Calculate axial force Fp (for normal operating pressure)
- Assume a lower magnitude of bending stress
- Calculate bending moment = (bending stress) x (section modulus)
- Calculate the leakage flaw size at 100 percent power condition for 10 times the leak detection capability (for 0.5 gpm leak detection capability, this is 10 x 0.5 = 5 gpm)
- Stability analysis
  - Perform J-integral analysis
  - Determine the maximum loads for a critical flaw size of twice the leakage flaw size by satisfying the stability criteria. The margin of 2 on flaw size is satisfied.
- Stability criteria
  - Japplied ≤ JIC
  - If Japplied > JIC, then Japplied < Jmax and Tapplied < Tmat
- Calculate the low normal stress and corresponding maximum stress by using:

$$\text{Stress} = \frac{\text{Axial Force}}{\text{Area}} + \frac{\text{Bending Moment}}{\text{Section Modulus}}$$

3B.3.2.4 High Normal Stress Case (Case 2)

To determine the other endpoint of the bounding analysis curve the following steps are used.

- Axial force Fp is calculated above (for normal operating pressure)
- Assume a higher magnitude of bending stress to get higher bending moment
- Calculate bending moment = (bending stress) x (section modulus)

- Repeat leakage flaw size and stability calculations as outlined for the low normal stress case above

Note: For an intermediate point, calculation steps are the same as low normal or the high normal case.

3B.3.2.5 Develop the Bounding Analysis Curve

Follow steps as outlined for the stainless steel case in Subsection 3B.3.1.5.

3B.3.3 Evaluation of Piping System Using Bounding Analysis Curves

To evaluate the applicability of leak-before-break, the results of the pipe stress analysis are compared to the bounding analysis curve. The critical location is the location of highest maximum stress as determined by the pipe stress results. A comparison is made with the applicable bounding analysis curves for the analyzable piping systems. As outlined in 3B.3.1.1 and 3B.3.2.1, bounding analysis curves are calculated for different combinations of pipe size, pipe schedule, operating pressures, operating temperatures.

The bounding analysis curves are used during the layout and design of the piping systems to provide a design that satisfies leak-before-break criteria. In addition, the results of the as-built piping analysis reconciliation to the bounding analysis curves to verify that the fabricated piping systems satisfy leak-before-break criteria. See Subsection 3.6.4 for the Combined License information item associated with this verification.

At the critical location, the load combination for the maximum stress calculation uses the absolute sum method. The load combination is as follows:

(1) |Pressure| + |Deadweight| + |Thermal (100% Power)*| + |Safe Shutdown Earthquake|

The normal stress is calculated using the algebraic sum method at critical location and the following load combination.

(1) Pressure + Deadweight + Thermal (100% Power*)

* Includes applicable stratification loads.

3B.3.3.1 Calculation of Stresses

The stresses due to axial loads and moments are calculated by the following equation:

where:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad \text{(3B-2)}$$

σ = stress

F = axial load

M = moment

A = cross-sectional area

Z = section modulus

The moments for the desired loading combinations are calculated by the following equation:

$$M = \sqrt{M_X^2 + M_Y^2 + M_Z^2} \tag{3B-3}$$

where,

M = moment for required loading

M_X = torsional moment

M_Y = Y component of bending moment

M_Z = Z component of bending moment

The Y and Z-axes are lateral axes to the X-axis which is the axial axis

The axial load and moments for the normal case and maximum case are computed by the methods shown below.

3B.3.3.2 Normal Loads

The normal operating loads are calculated by the following equations:

$$F = F_{DW} + F_{Th} + F_P \tag{3B-4}$$

$$M_X = (M_X)_{DW} + (M_X)_{Th} \tag{3B-5}$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{Th} \tag{3B-6}$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{Th} \tag{3B-7}$$

The subscripts of the above equations represent the following load cases:

DW = deadweight

Th = normal thermal expansion (100 percent power, including applicable stratification loads)

P = load due to internal pressure

The method of combining loads is often referred to as the algebraic sum method.

Calculate the normal stress at the critical location.

3B.3.3.3 Maximum Loads

For the maximum case, the absolute summation method of load combination is applied which results in higher magnitude of the combined loads. Since stability is demonstrated using these loads, the leak-before-break margin on loads is satisfied. An example of the absolute summation expressions are shown below:

$$F = |F_{DW}| + |F_{Th}| + |F_P| + |F_{SSEINERTIA}| + |F_{SSEAM}| \tag{3B-8}$$

$$M_X = |(M_X)_{DW}| + |(M_X)_{Th}| + |(M_X)_{SSEINERTIA}| + |(M_X)_{SSEAM}| \tag{3B-9}$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{Th}| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \quad (3B\text{-}10)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{Th}| + |(M_Z)_{SSEINERTIA}| + |(M_Z)_{SSEAM}| \quad (3B\text{-}11)$$

where subscripts SSE, Inertia and AM mean safe shutdown earthquake, inertia and anchor motion respectively.

3B.3.3.4 Bounding Analysis Curve Comparison – LBB Criteria

To compare the stress results with the bounding analysis curve the following process is followed. The normal and maximum stress at the critical location are calculated by using the loads defined in Subsection 3B.3.3. Plot the normal stress versus maximum stress on the bounding analysis curve for the specified system. If the point is on or below the bounding analysis curve, the leak-before-break analysis and margins are satisfied. If the point falls above the bounding analysis curve, the leak-before-break analysis criteria are not satisfied and the pipe layout or support configuration needs to be revised to meet the leak-before-break bounding analysis. Figure 3B-1 shows a typical bounding analysis curve.

3B.3.4 Bounding Analysis Results

Table 3B-1 shows a summary of piping systems and corresponding bounding analysis figures. Figures 3B-1 to 3B-22 show the bounding analysis curves. The curves satisfy the margins as indicated in Section 3B.3.

3B.4 Differences in Leak-before-Break Analysis for Stainless Steel and Ferritic Steel Pipe

The significant difference between leak-before-break analysis performed for the stainless steel (Class 1 and Class 3) systems and the ferritic steel in the Class 2 systems is in the stability analysis. In the case of stainless steel systems, stability analyses are performed by limit load approach. In the ferritic steel systems, stability analyses are performed by J-integral approach.

3B.5 Differences in Inspection Criteria for Class 1, 2, and 3 Systems

Class 1, 2 and 3 systems are subjected to in-service inspection requirements from ASME Code, Section XI. For Class 1 piping, terminal ends and dissimilar metal welds are volumetrically inspected, along with other locations, to total 25 percent of the welds. For Class 2 piping, the requirement is to volumetrically inspect the terminal ends and other locations to total 7.5 percent of the welds. For Class 3 systems (the only Class 3 piping is in the accumulator line which is always at room temperature), the system receives periodic visual examinations in conjunction with pressure testing. These requirements were developed by ASME Code, Section XI consistent with the different safety classes of these systems.

The leak-before-break evaluations are based on the ability to detect a potential leaking crack; not the ability to find cracks by inservice inspections. The criteria or methods of the leak-before-break evaluations are the same for ASME Code Class 1, 2, and 3.

3B.6 Differences in Fabrication Requirements of ASME Class 1, Class 2, and Class 3 Piping

The significant difference among Class 1, 2 and 3 seamless pipe occurs in the nondestructive examination requirements. The Class 1 seamless pipe examination requirements include an ultrasonic testing examination, whereas Class 2 and 3 do not. In addition, the Class 1 examination requirements for a circumferential butt welded joint include radioagraphic testing and magnetic

particle or liquid penetrant examination where Class 2 does not. The examination requirements for Class 2 pipe require radiographic examination of the welds and normally Class 3 pipe does not. As noted in Subsection 3.2.2.5, for Class 3 lines required for emergency core cooling functions, radiography will be conducted on a random sample of welds. The Class 3 leak-before-break lines are included in the lines that are radiographed. In addition see Subsection 3.6.3.2 for augmented inspection of Class 3 leak-before-break lines.

For the fabrication of welds in the Class 1, Class 2 and Class 3 pipes there is no significant differences.

The differences in fabrication and nondestructive examination requirements do not affect the leak-before-break analyses assumptions, criteria, or methods.

3B.7 Sensitivity Study for the Constraint Effect on LBB

Westinghouse performed a sensitivity study on a 6-inch diameter pipe to demonstrate that the leak-before-break evaluation margins are not significantly affected when constraint effects of pressure induced bending are included. The analysis used a finite element model of a 6-inch diameter pipe welded to a nozzle with a fixed end condition. This conservatively represents the bounding conditions for AP1000 piping. The normal and maximum stresses were used from a representative AP600 6-inch line bounding analysis curve. The material properties for the base metal and TIG weld were considered in the analysis. The stability analysis was performed using the J-integral method. This analysis was developed in consultation with the NRC.

The conclusion of this sensitivity study is that the leak-before-break margins for 6-inch and larger piping on AP1000 are not significantly affected by the constraint effect and application of leak-before-break to such piping is acceptable.

3B.8 References

1. Standard Review Plan 3.6.3, "Leak Before Break Evaluation Procedures," Federal Register, Volume 52, Number 167, Friday, August 28, 1987; Notice (Public Comment Solicited), pp. 32626-32633.

2. NUREG-1061, "Evaluation of Potential for Pipe Breaks, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volume 3, (prepared by the Pipe Break Task Group), November 1984.

3. "Erosion-Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Program Guidelines," EPRI NP-3944, April 1985.

4. G. Cragnolino, "Erosion-Corrosion in Nuclear Power Systems-An Overview," Corrosion '87, Paper No. 86, March 1987.

5. ASME Boiler and Pressure Vessel Code, Section II, "Materials," 1998 Edition through 2000 Addenda.

Table 3B-1 (Sheet 1 of 2)
AP1000 Leak-Before-Break Bounding Analysis Systems and Parameters

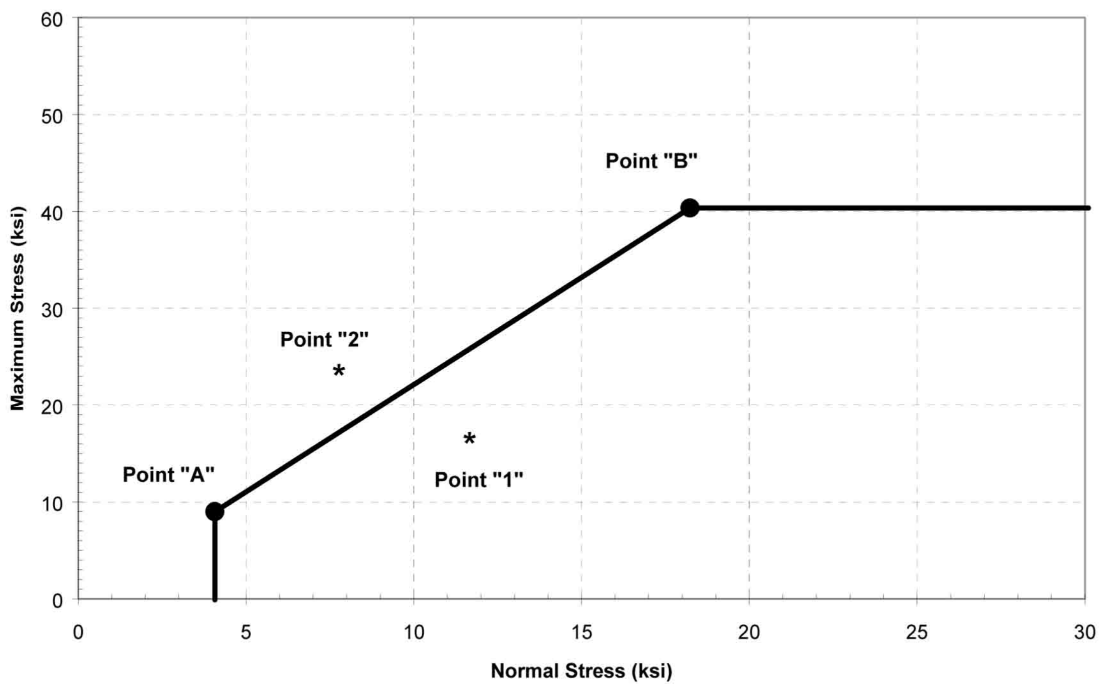
System	Subsystem	Line No(s).	Nominal Diameter (Inches)	Material	Temp (°F)	Pressure (psig)	Figure No.
RCS	Primary Loop Hot Leg	L001A, B	31 (ID)(1)	SA-376 TP316LN	610.0	2248	3B-2
RCS	Primary Loop Cold Leg	L002A, B, C, D	22 (ID)(1)	SA-376 TP316LN	537.2	2310	3B-3
SGS	Main Steam Line	L006A, B	38	SA-335 GR P11	523.0	821	3B-4
RCS	Normal Residual Heat Removal	L139	20	SA-312 TP316LN	610.0	2248	3B-5
RCS	Surge Line	L003	18	SA-312 TP316LN	653.0	2248	3B-6 (Sheet 1)
RCS	Surge Line	L003	18	SA-312 TP316LN	455.0	430	3B-6 (Sheet 2)
RCS	Passive Residual Heat Removal Supply/ ADS 4	L135A,B; L136A,B	18	SA-312 TP316LN	610.0	2248	3B-7
RCS	Passive Removal Heat Removal Supply/ ADS 4	L133A, B; L137A, B; L134	14	SA-312 TP316LN	610.0	2248	3B-8
PXS	Passive Residual Heat Removal Supply to Cold Trap and Vent Line	L102, L107	14	SA-312 TP316LN	610.0	2248	3B-8
PXS	Passive Residual Heat Removal Supply after Cold Trap to PRHR HX	L102	14	SA-312 TP316LN	120.0	2248	3B-9
PXS	Return – PRHR HX to Isolation Valve	L103; L104A, B	14	SA-312 TP316LN	120.0	2248	3B-9
RCS	Automatic Depressurization System Stage 2, 3	L004A,B; L006A,B; L020A,B; L030A, B; L131	14	SA-312 TP316LN	653.0	2235	3B-10
PXS	Passive Residual Heat Removal Return – after Isolation Valve	L104A, B; L105	14	SA-312 TP316LN	537.0	2190	3B-11

Table 3B-1 (Sheet 2 of 2)
AP1000 Leak-Before-Break Bounding Analysis Systems and Parameters

System	Subsystem	Line No(s).	Nominal Diameter (Inches)	Material	Temp (°F)	Pressure (psig)	Figure No.
RCS	Passive Residual Heat Removal Return	L113	14	SA-312 TP316LN	537.0	2190	3B-11
PXS	Passive Residual Heat Removal Vent Line	L107	12	SA-312 TP316LN	610.0	2248	3B-12 (Not Used)
PXS	Accumulator to Isolation Valve	L029A, B	8	SA-312 TP304L	120.0	700	3B-13
RCS	Balance Line from Cold Leg to CMT Isolation Valve	L118A, B	8	SA-312 TP316LN	537.0	2310	3B-14
PXS	Balance Line from CMT Isolation Valve to CMT	L007A, B; L070A, B	8	SA-312 TP316LN	537.0	2310	3B-14
PXS	Direct Vessel Injection Line to RV	L021A, B; L125A, B	8	SA-312 TP316LN	537.0	2310	3B-14
PXS	Core Makeup Tank (Injection Line, RV Side of Isolation Valve, Core Makeup Tank Side of Isolation Valve), Direct Vessel Injection (Accumulator Connection to Cold Trap), IWRST Injection	L015, L016, L017, L018, L020, L021, L025, L127 (All A, B)	8	SA-312 TP316LN	120.0	2310	3B-15
RCS	Automatic Depressurization System Stage 2, 3	L021A,B; L031A,B	8	SA-312 TP316LN	653.0	2235	3B-16 (Not Used)
PXS	Accumulator after Isolation Valve	L027A, B	8	SA-312 TP304L	120.0	700	3B-17
PXS	RNS Discharge	L019A, B	6	SA-312 TP316LN	120.0	2310	3B-18
RCS	Automatic Depressurization System Header to RCS Safety Valve	L005A, B	6	SA-312 TP316LN	653.0	2235	3B-19
RCS	Normal Residual Heat Removal	L140	12	SA-312 TP316LN	610.0	2248	3B-20
RNS	Normal Residual Heat Removal	L001, L002A, B	10	SA-312 TP316LN	610.0	2248	3B-21
RCS	Automatic Depressurization System Stage 2, 3 (Cold Trap)	L021A, B; L031A, B	8	SA-312TP316LN	250	2235	3B-22

Note:
1. ID = Inside diameter

AP1000
Typical Bounding Analysis Curve (BAC)
Line Number(s):



Pipe Designator:
System:
Nominal Diameter: inch
Pipe Schedule:
Outside Diameter: inch
Pipe Material:
Minimum Weld Thickness:

Normal Operating Pressure: psig
Normal Operating Temperature: F
Critical Flaw Size = 2 x Leakage Flaw Size
Load Margin = 1.0
Leak Rate Margin = 10 (Typical for All Curves)

Notes for Typical Bounding Analysis Curve:
Point "A" - for low normal case to generate BAC.
Point "B" - for high normal case to generate BAC.
Point "A" and Point "B" are joined by a straight line.
Point "1" - analyzed critical point which meets LBB criteria.
Point "2" - analyzed critical point which fails LBB criteria.

Figure 3B-1
Typical Bounding Analysis Curve (BAC)

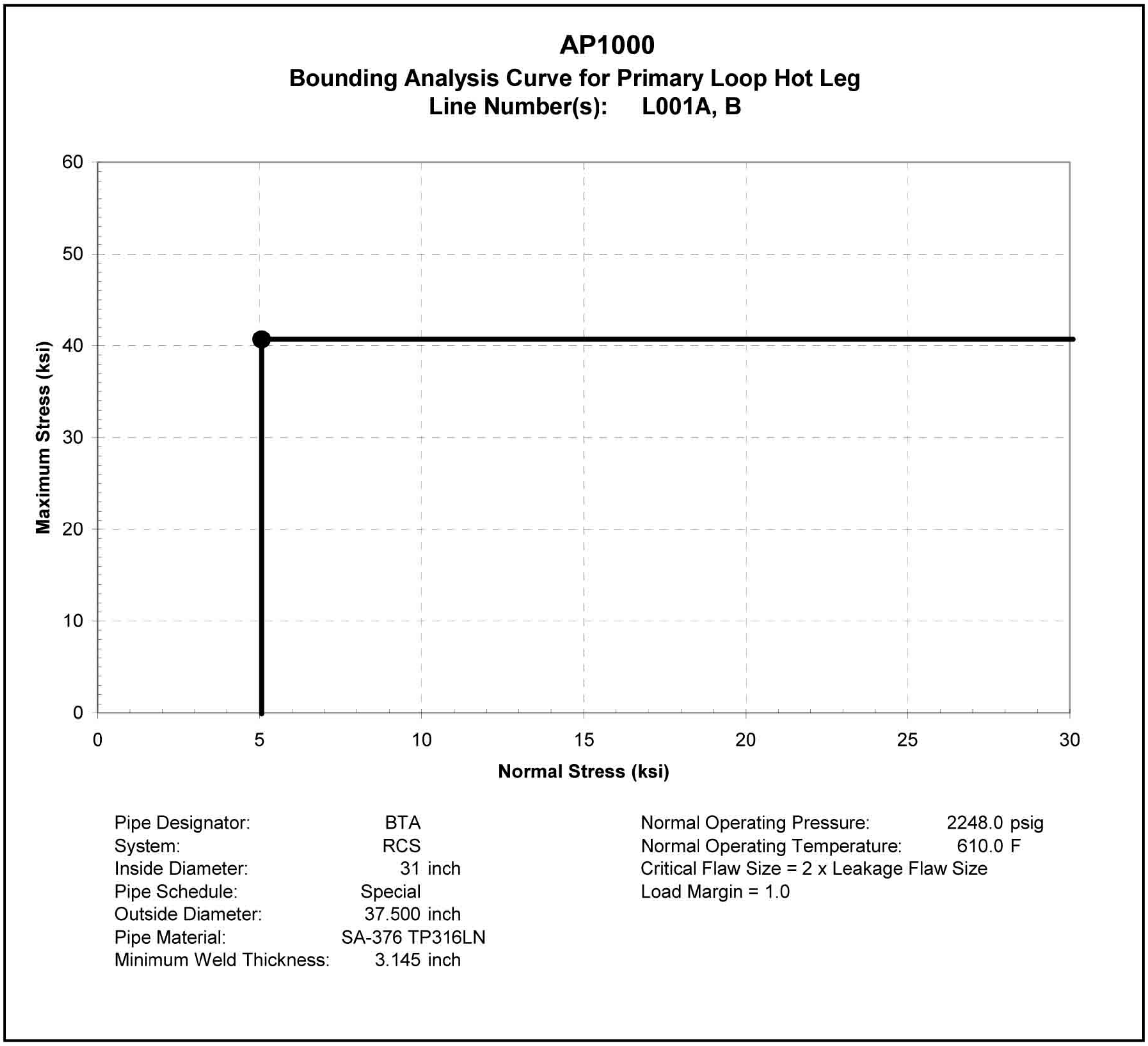


Figure 3B-2
Bounding Analysis Curve for Primary Loop Hot Leg

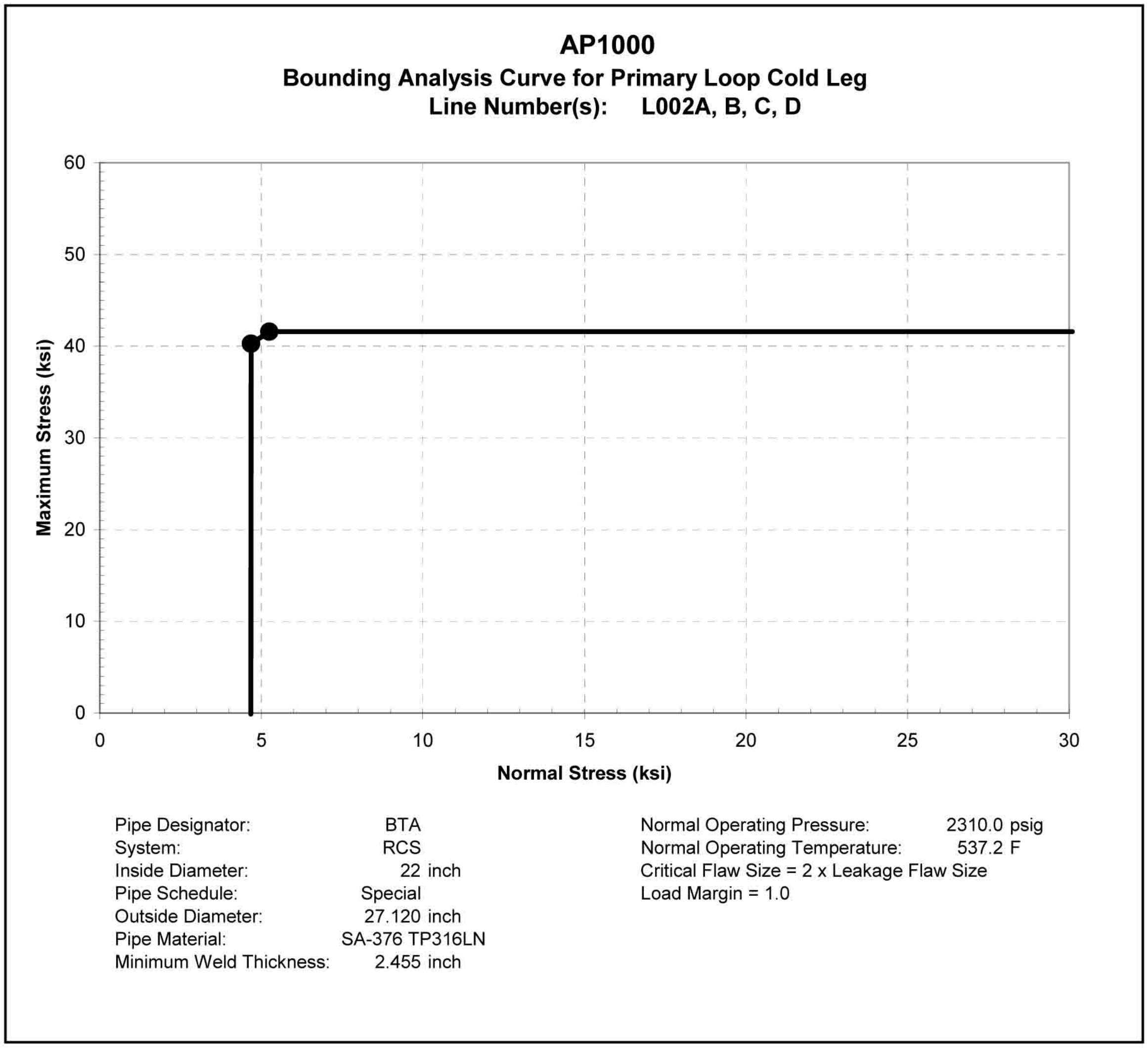


Figure 3B-3
Bounding Analysis Curve for Primary Loop Cold Leg

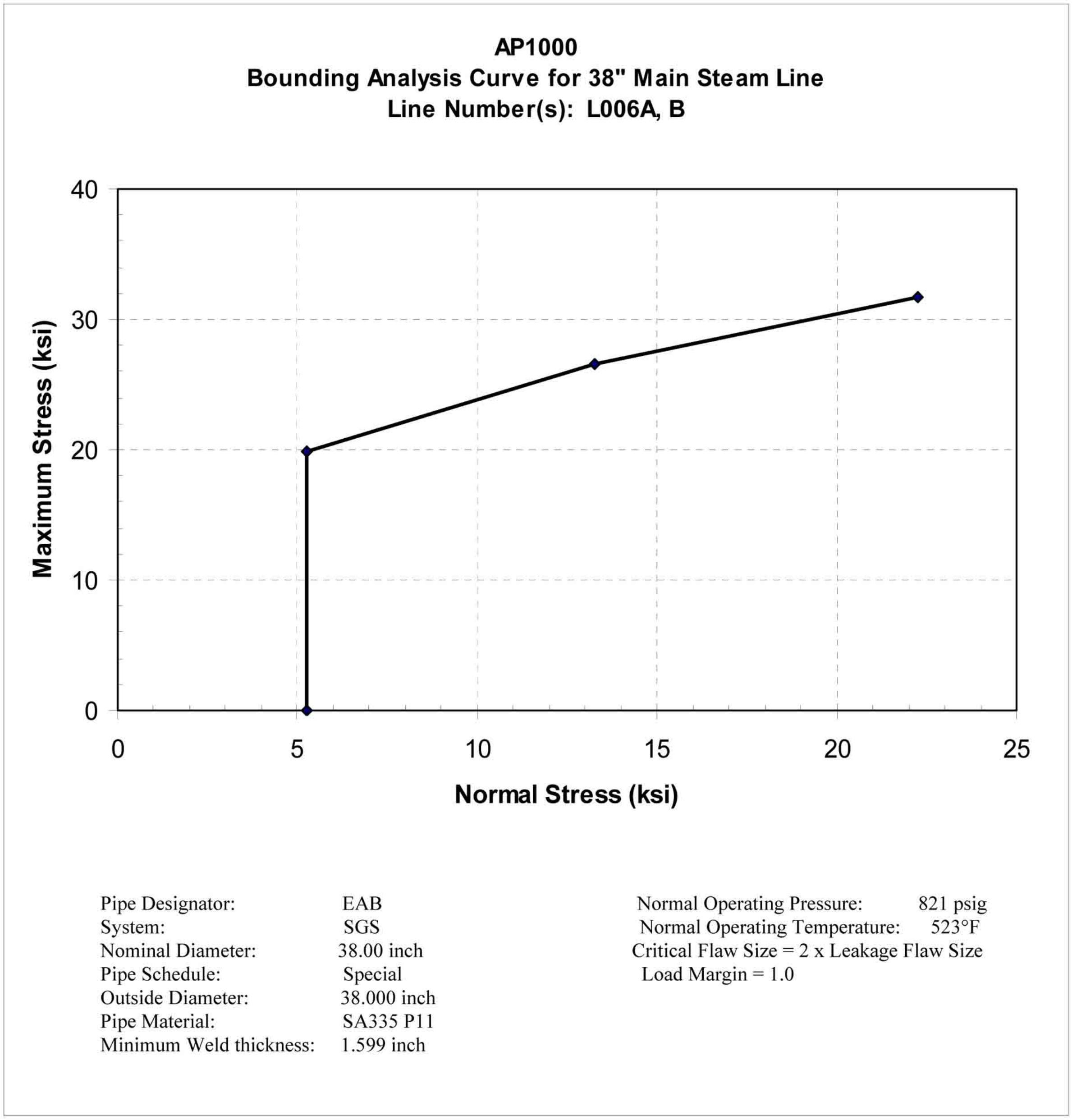


Figure 3B-4
Bounding Analysis Curve for 38″ Main Steam Line

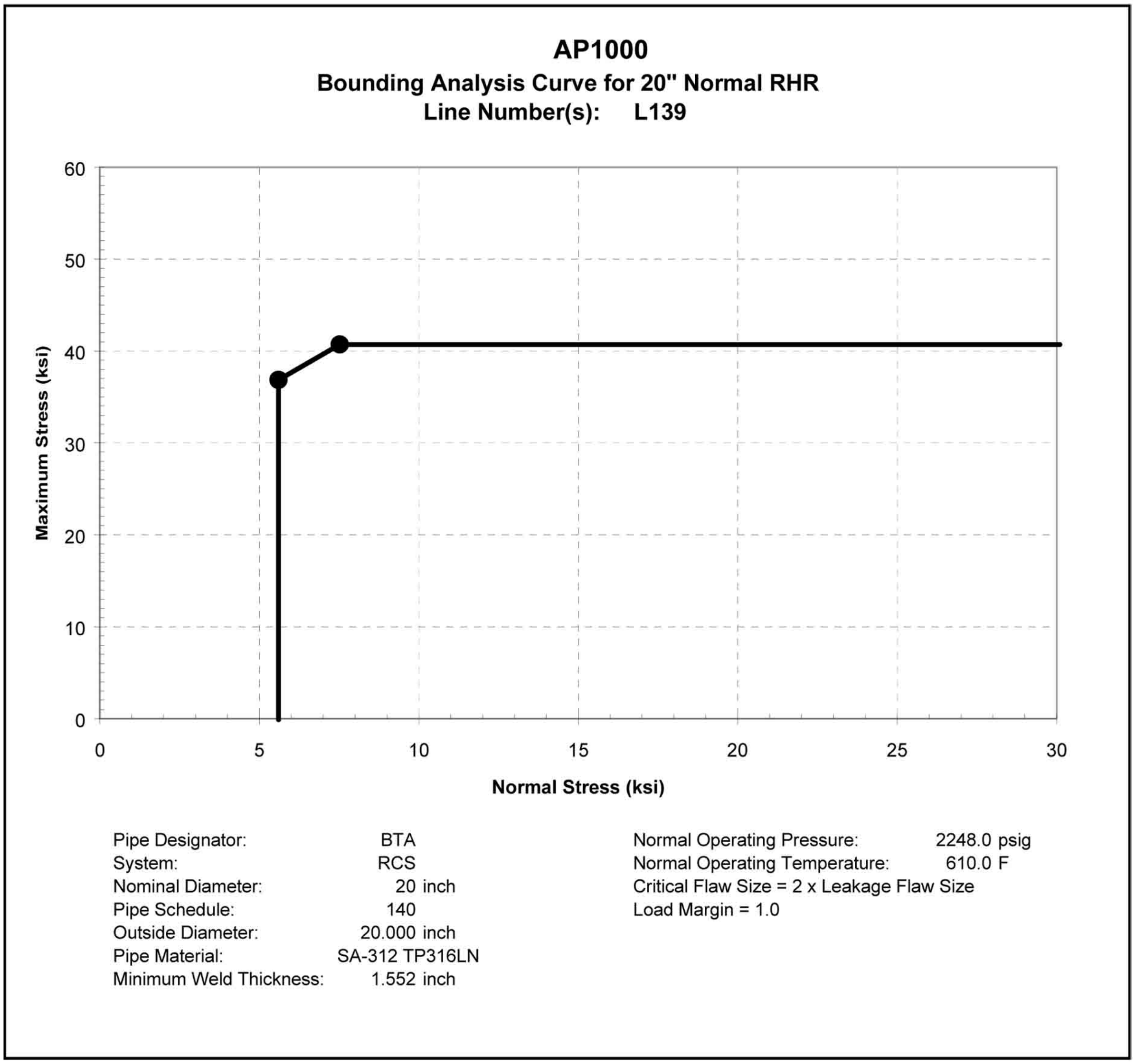
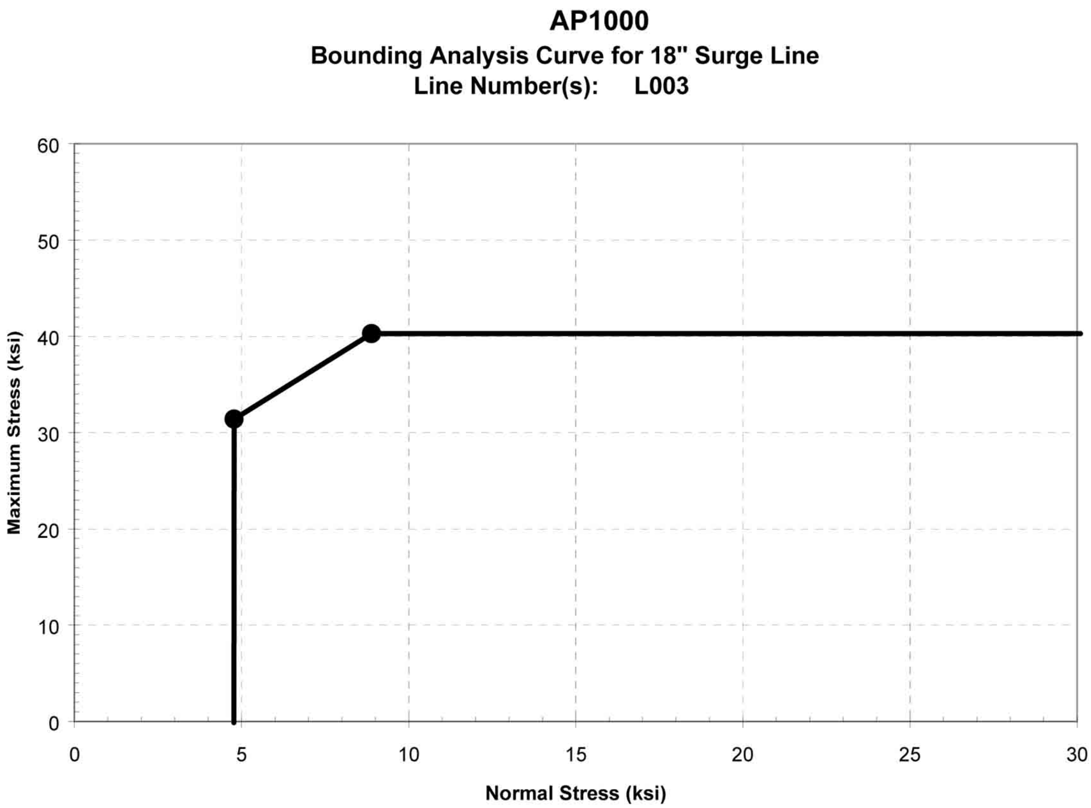


Figure 3B-5
Bounding Analysis Curve for 20″ Normal RHR



Pipe Designator:	BTA
System:	RCS
Nominal Diameter:	18 inch
Pipe Schedule:	160
Outside Diameter:	18.000 inch
Pipe Material:	SA-312 TP316LN
Minimum Weld Thickness:	1.579 inch

Normal Operating Pressure:	2248.0 psig
Normal Operating Temperature:	653.0 F
Critical Flaw Size = 2 x Leakage Flaw Size	
Load Margin = 1.0	

Figure 3B-6 (Sheet 1 of 2)
Bounding Analysis Curve for 18″ Surge Line

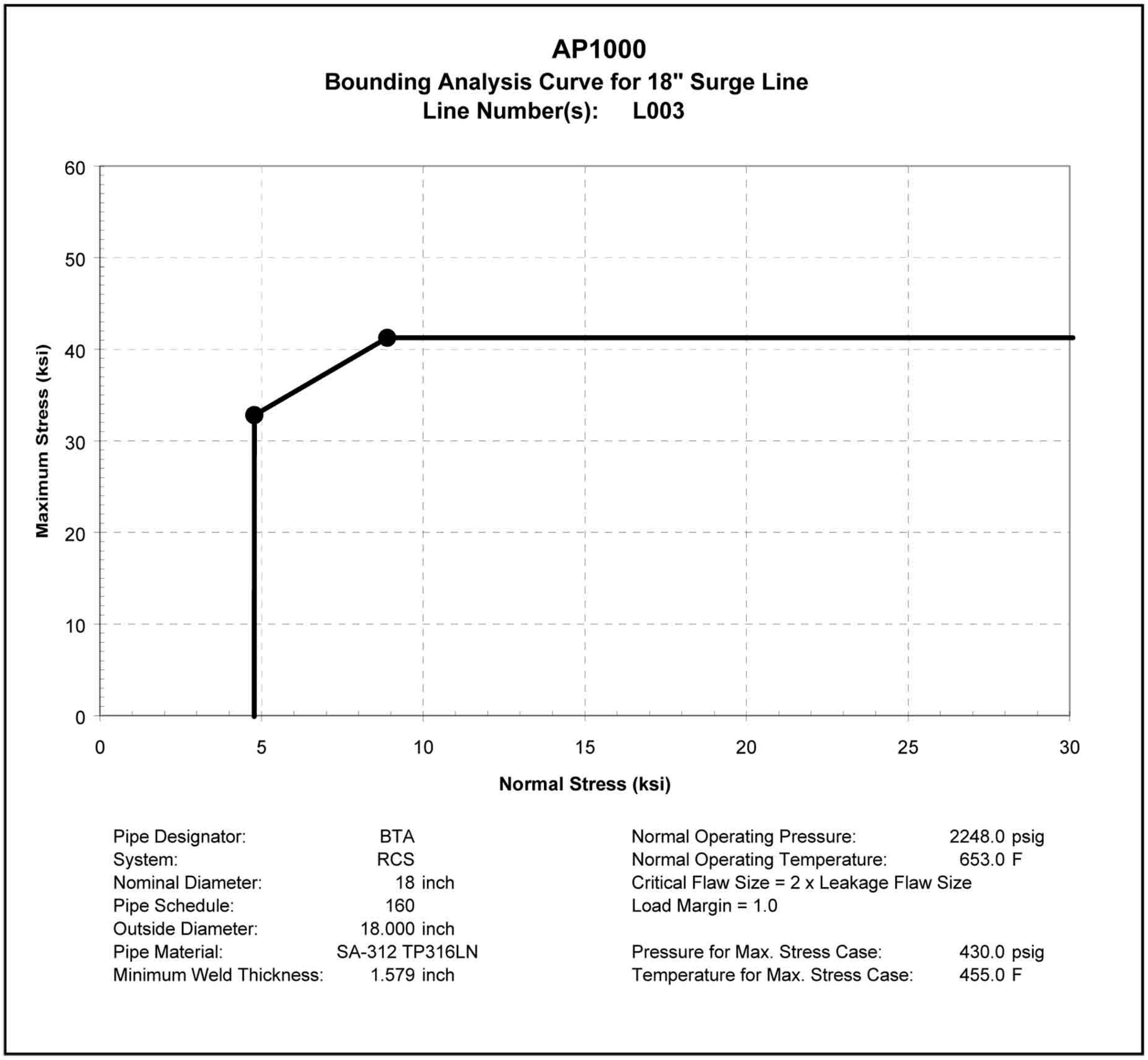


Figure 3B-6 (Sheet 2 of 2)
Bounding Analysis Curve for 18″ Surge Line

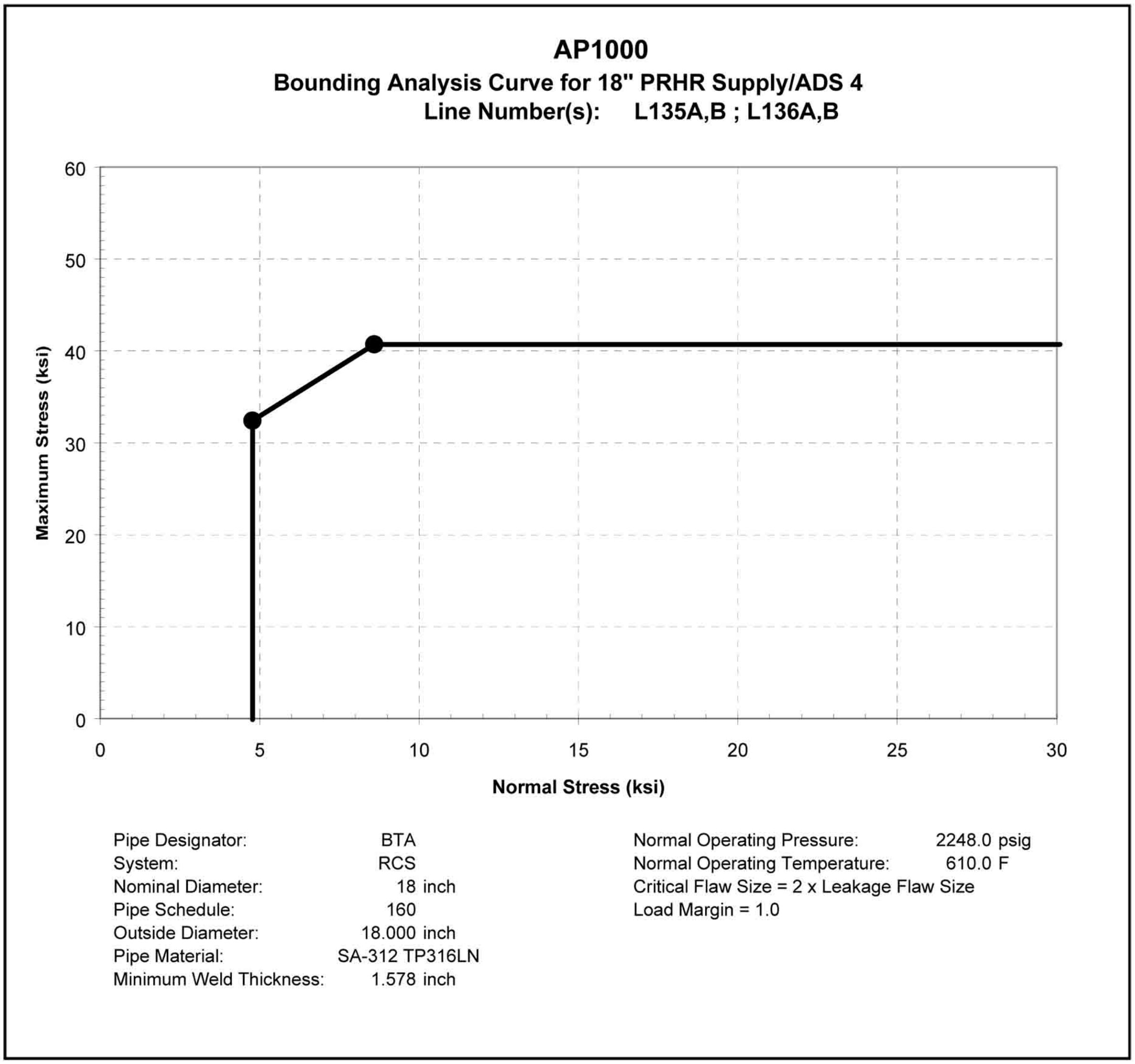


Figure 3B-7
Bounding Analysis Curve for 18″ PRHR Supply/ADS 4

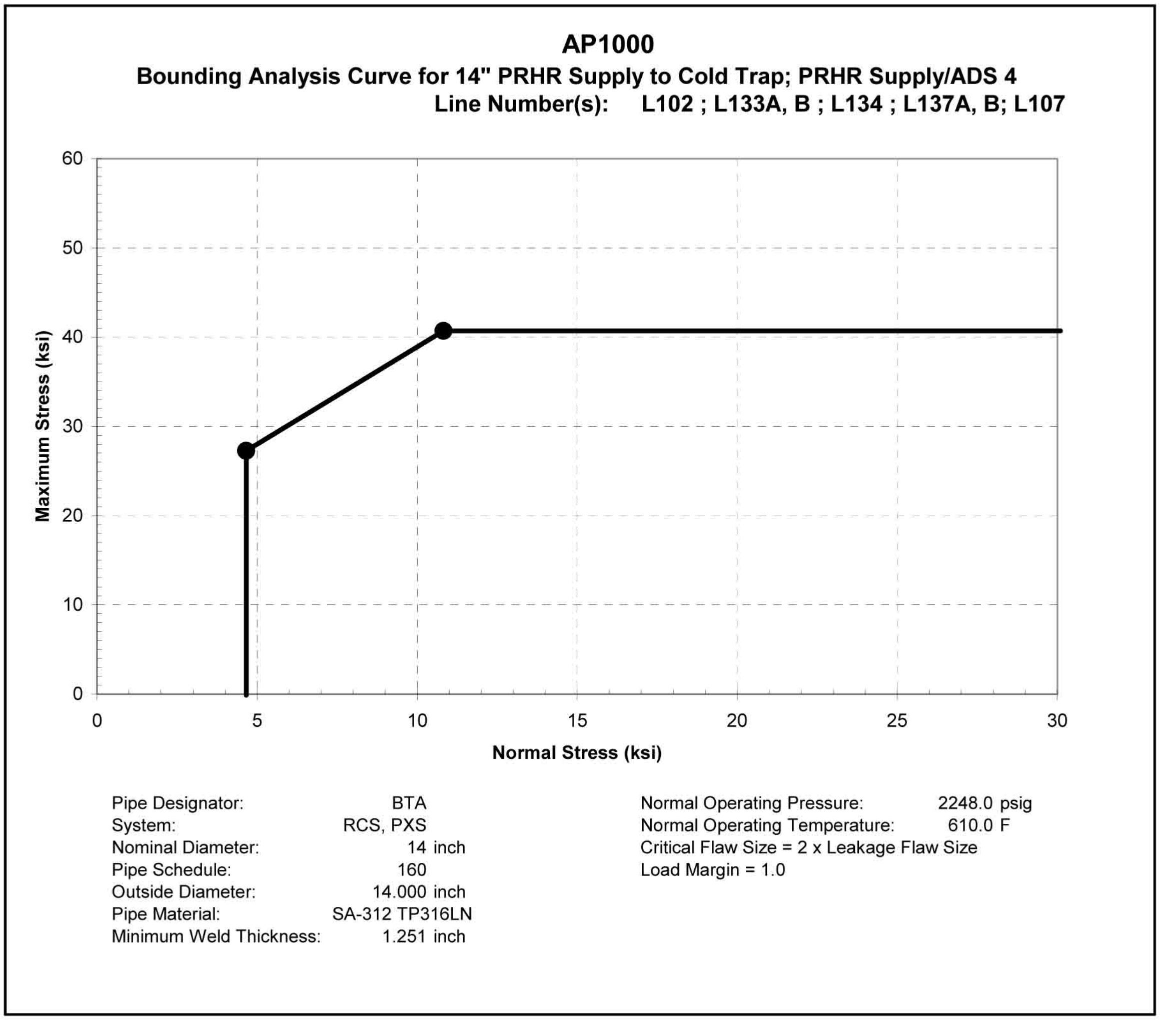


Figure 3B-8
Bounding Analysis Curve for 14″ PRHR Supply to Cold Trap, PRHR Supply/ADS4

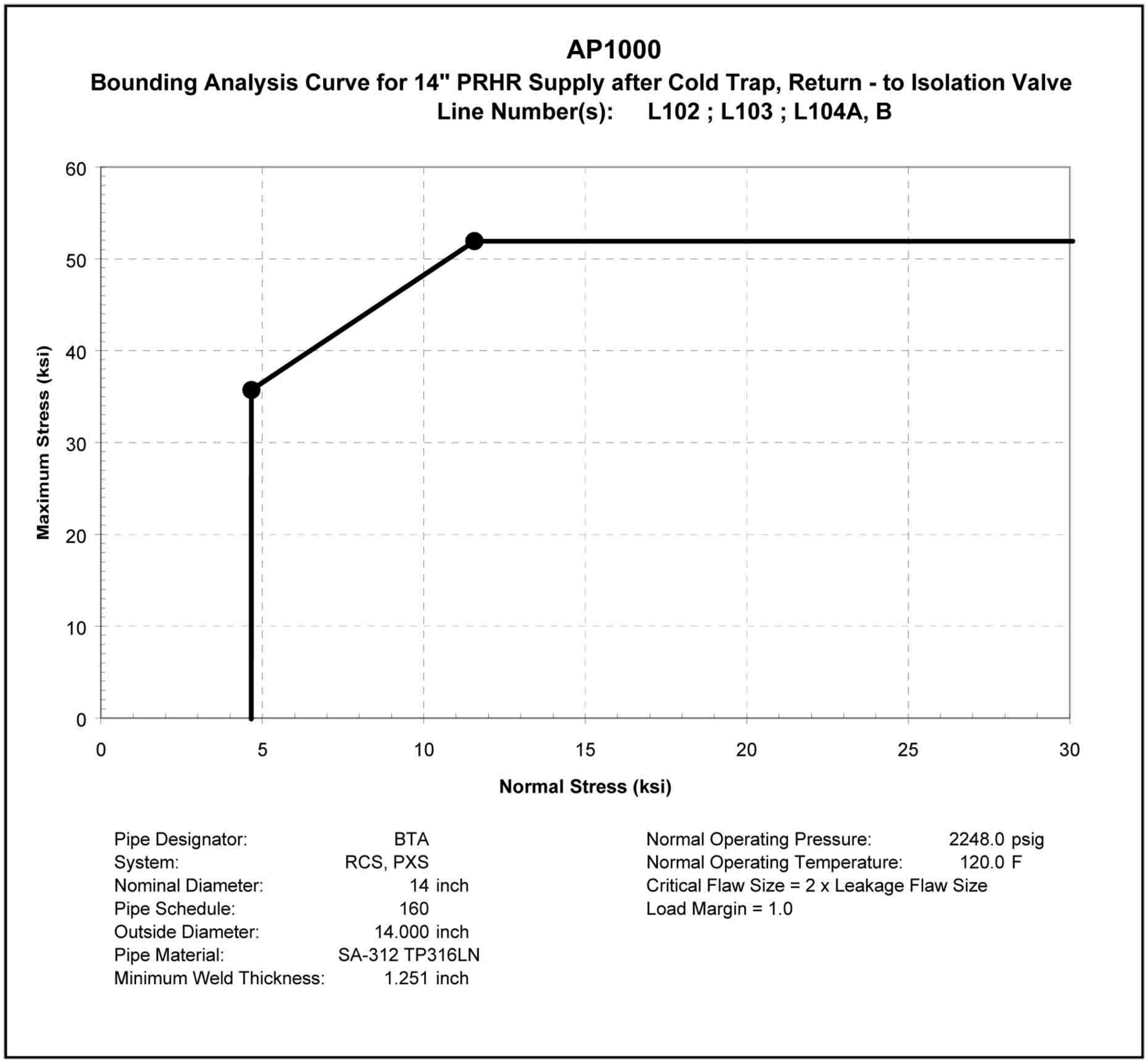


Figure 3B-9
Bounding Analysis Curve for 14″ PRHR Supply after Cold Trap, Return – to Isolation Valve

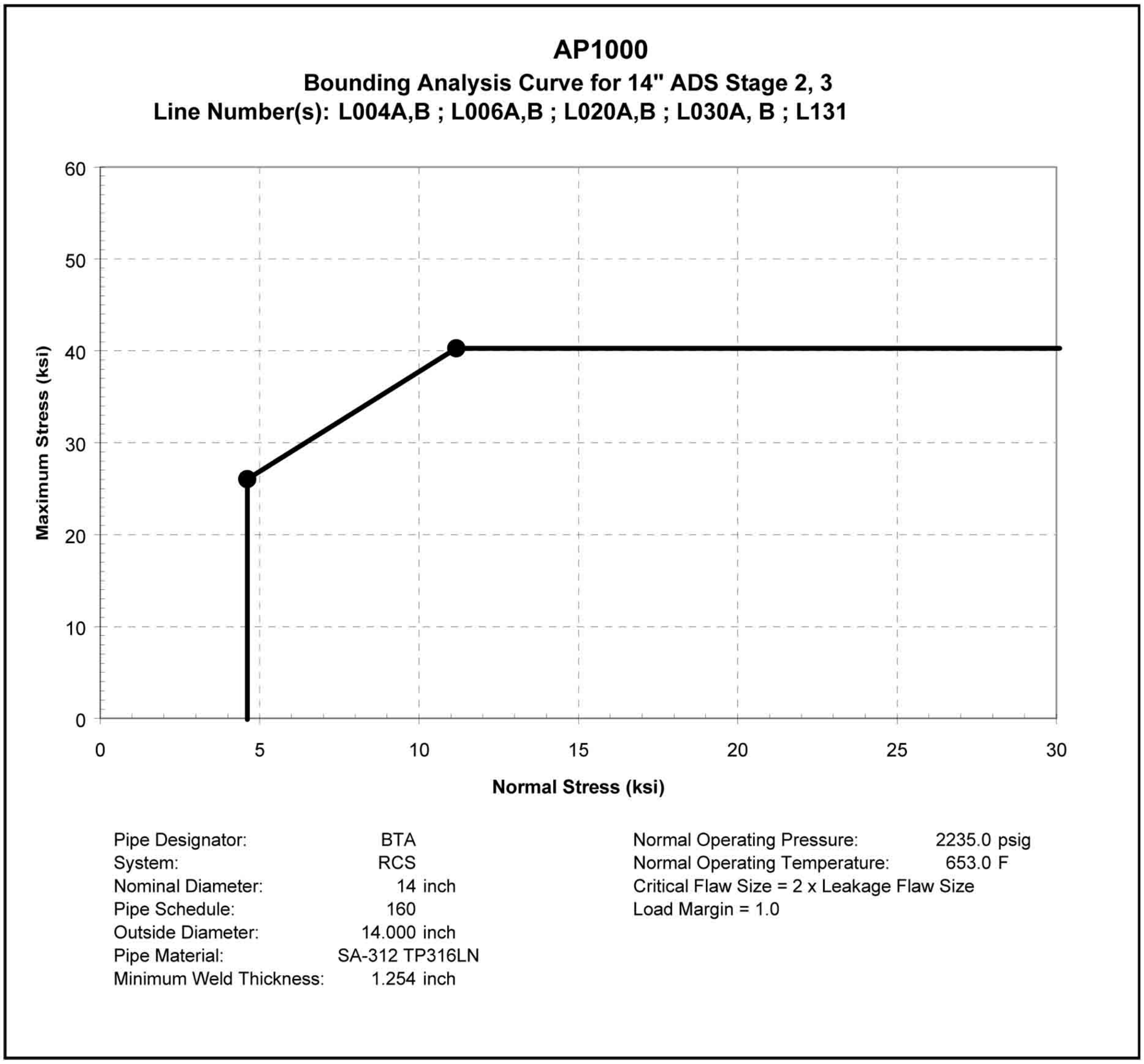


Figure 3B-10
Bounding Analysis Curve for 14″ ADS Stage 2, 3

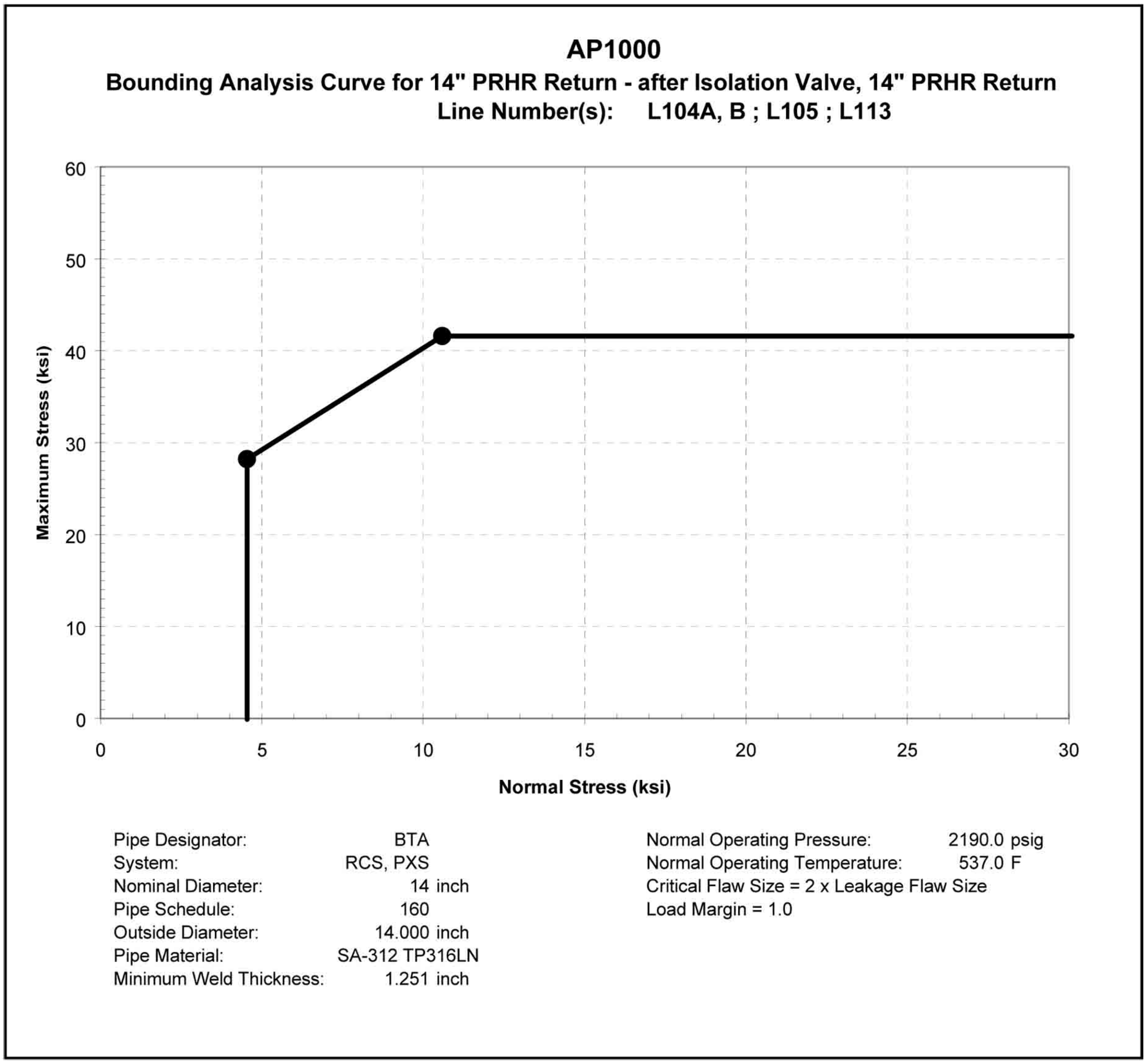


Figure 3B-11
Bounding Analysis Curve for 14″ PRHR Return – after Isolation Valve, 14″ PRHR Return

Figure 3B-12
Not Used.

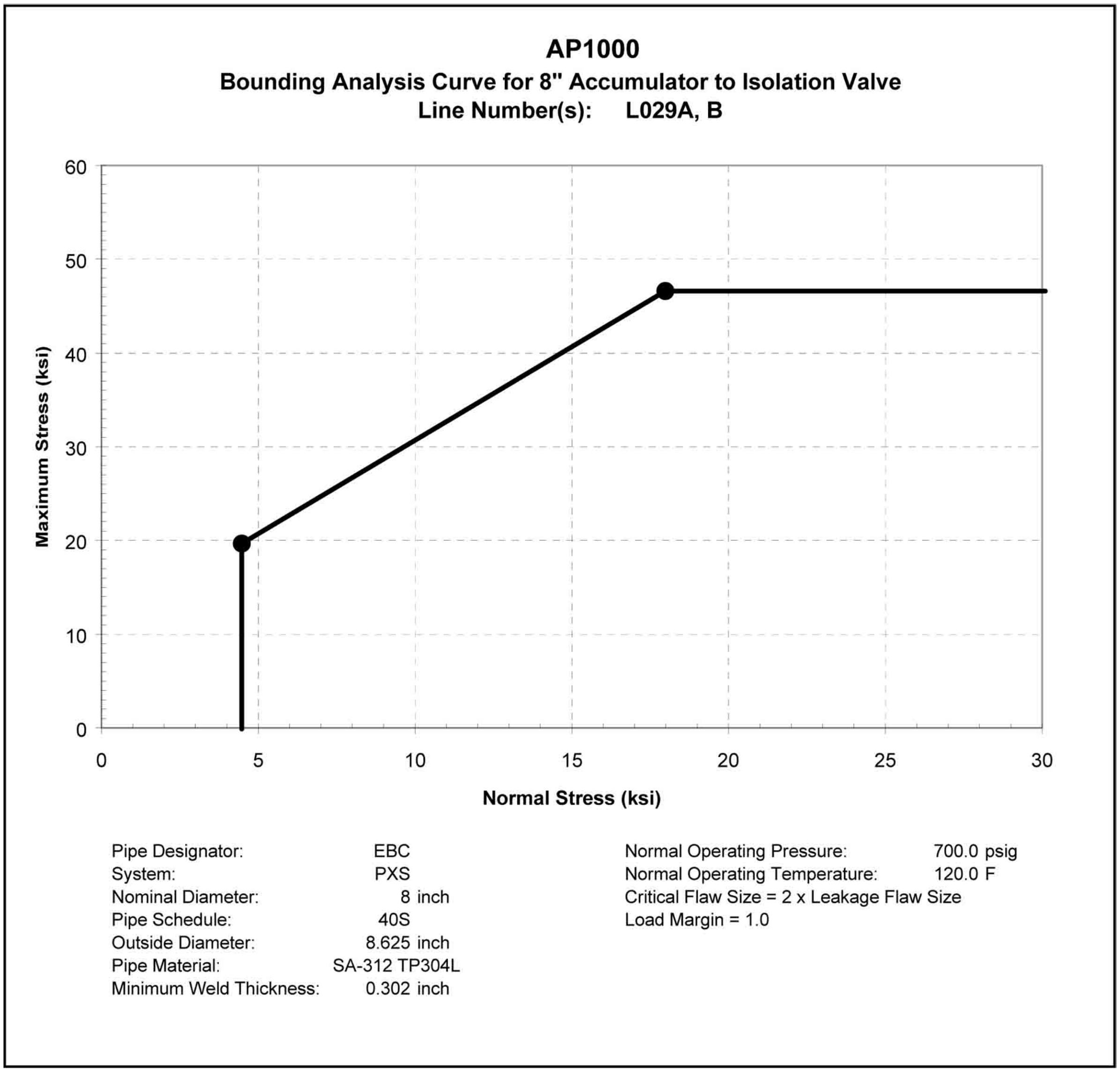


Figure 3B-13
Bounding Analysis Curve for 8″ Accumulator to Isolation Valve

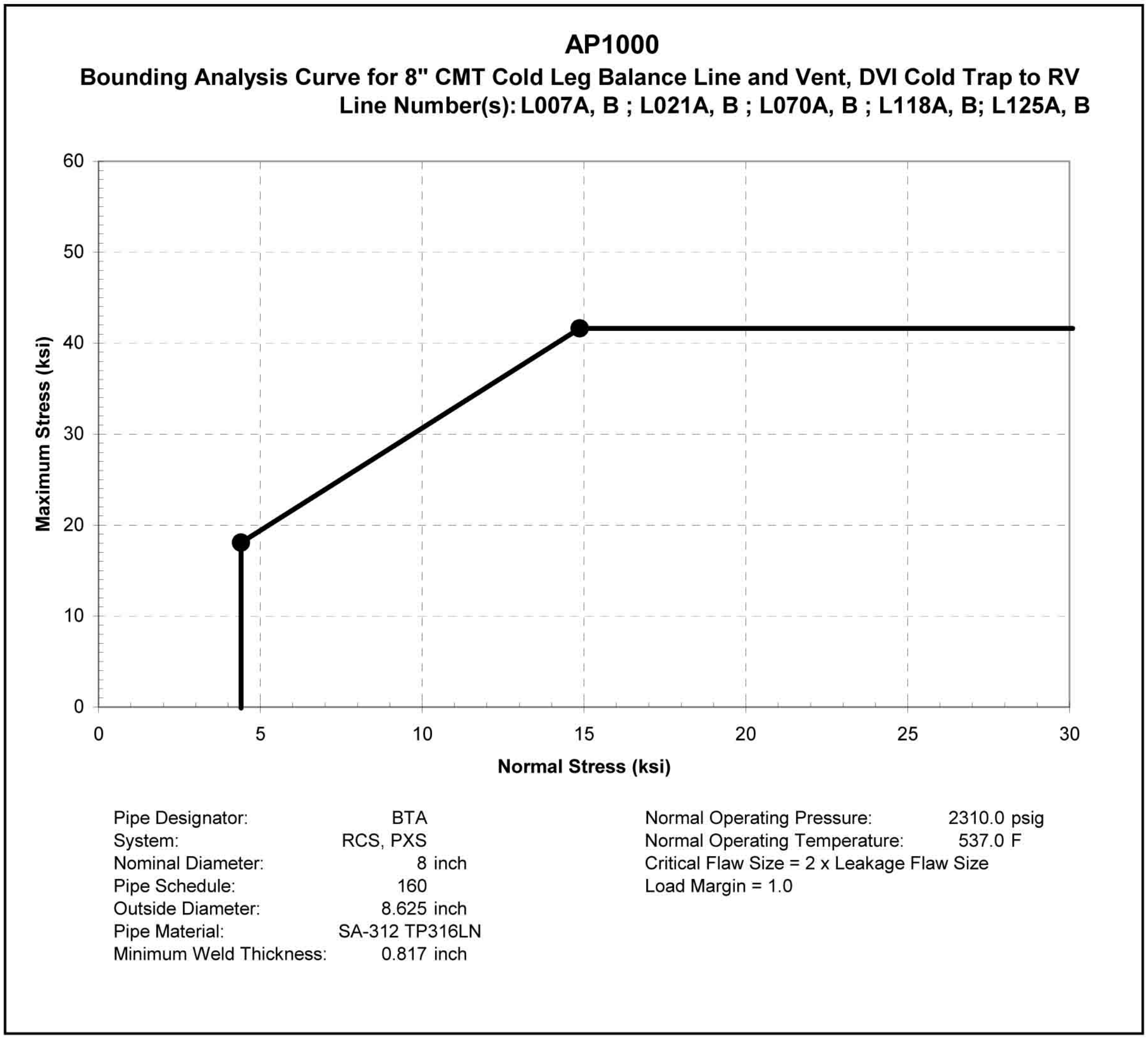


Figure 3B-14
Bounding Analysis Curve for 8″ CMT Cold Leg Balance Line and Vent, DVI Cold Trap to RV

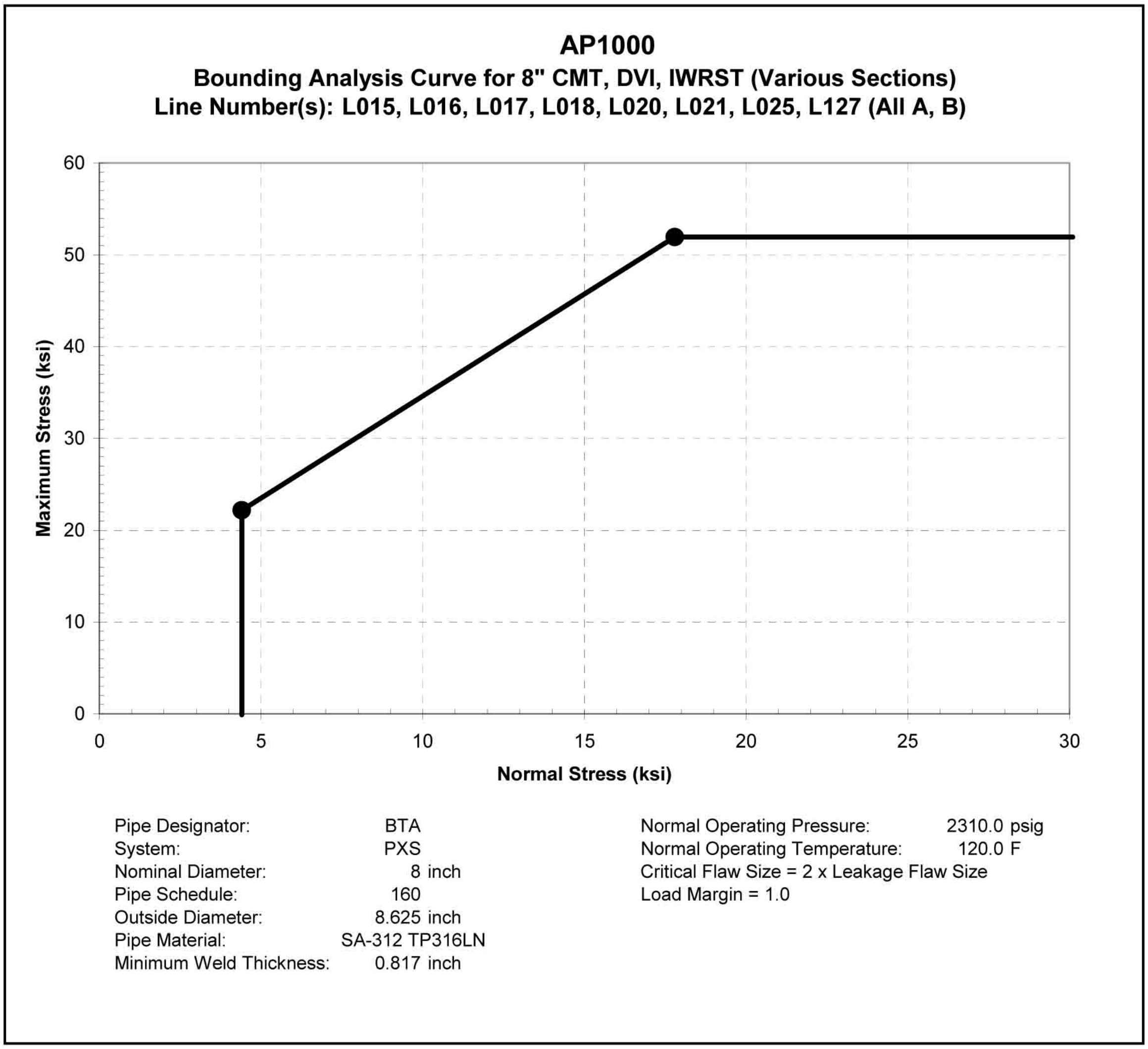
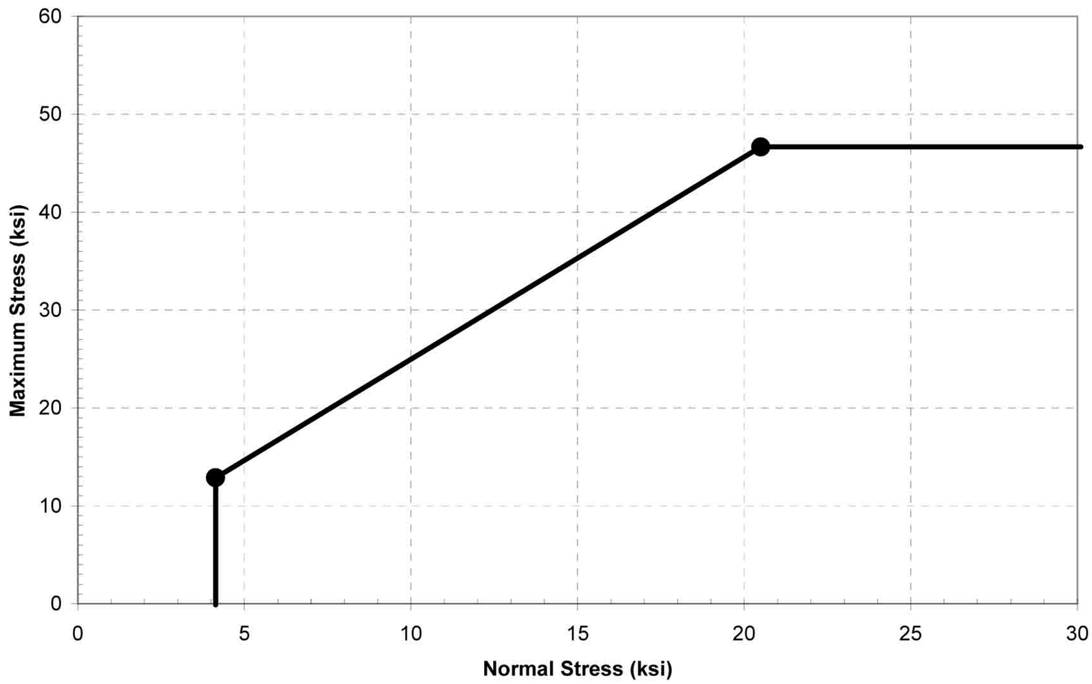


Figure 3B-15
Bounding Analysis Curve for 8″ CMT, DVI IWRST (Various Sections)

Figure 3B-16
Not Used.

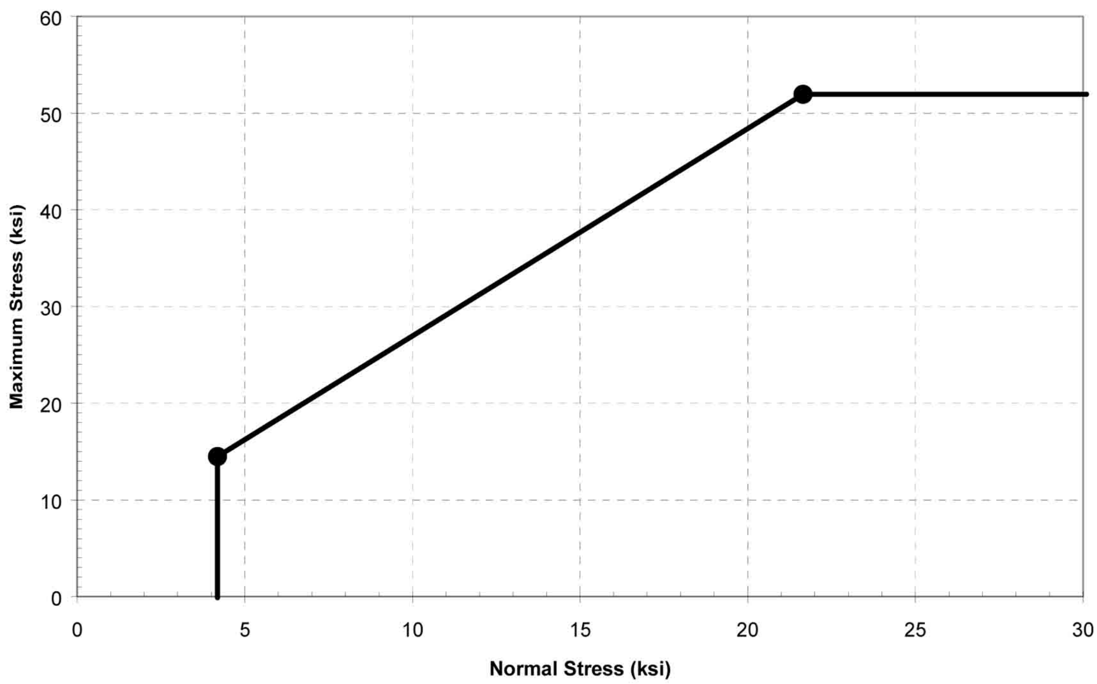
AP1000
Bounding Analysis Curve for Accumulator after Isolation Valve
Line Number(s): L027A, B



Pipe Designator:	BBC	Normal Operating Pressure:	700.0 psig
System:	PXS	Normal Operating Temperature:	120.0 F
Nominal Diameter:	8 inch	Critical Flaw Size = 2 x Leakage Flaw Size	
Pipe Schedule:	160	Load Margin = 1.0	
Outside Diameter:	8.625 inch		
Pipe Material:	SA-312 TP304L		
Minimum Weld Thickness:	0.817 inch		

Figure 3B-17
Bounding Analysis Curve for Accumulator after Isolation Valve

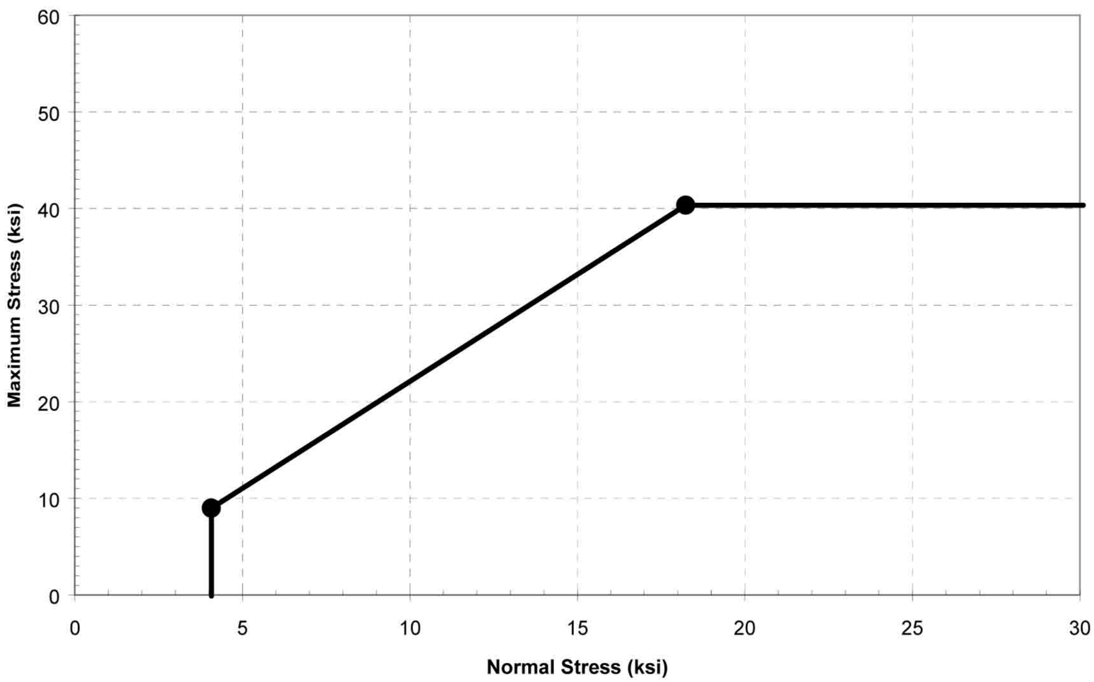
AP1000
Bounding Analysis Curve for RNS Discharge
Line Number(s): L019A, B



Pipe Designator:	BTA	Normal Operating Pressure:	2310.0 psig
System:	PXS	Normal Operating Temperature:	120.0 F
Nominal Diameter:	6 inch	Critical Flaw Size = 2 x Leakage Flaw Size	
Pipe Schedule:	160	Load Margin = 1.0	
Outside Diameter:	6.625 inch		
Pipe Material:	SA-312 TP316LN		
Minimum Weld Thickness:	0.651 inch		

Figure 3B-18
Bounding Analysis Curve for RNS Discharge

AP1000
Bounding Analysis Curve for ADS Header to RCS Safety Valve
Line Number(s): L005A, B



Pipe Designator:	BTA	Normal Operating Pressure:	2235.0 psig
System:	RCS	Normal Operating Temperature:	653.0 F
Nominal Diameter:	6 inch	Critical Flaw Size = 2 x Leakage Flaw Size	
Pipe Schedule:	160	Load Margin = 1.0	
Outside Diameter:	6.625 inch		
Pipe Material:	SA-312 TP316LN		
Minimum Weld Thickness:	0.651 inch		

Figure 3B-19
Bounding Analysis Curve for ADS Header to RCS Safety Valve

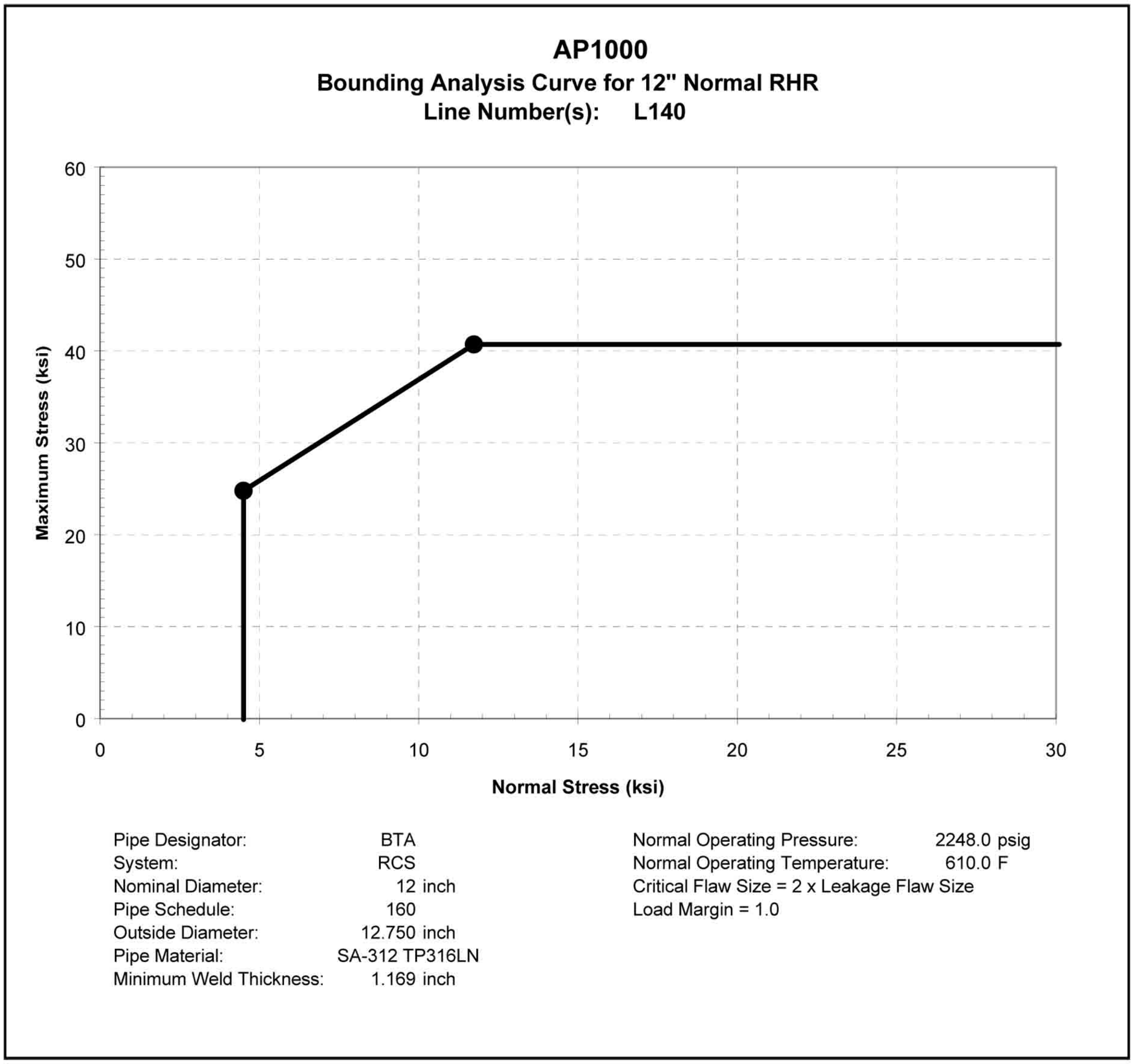
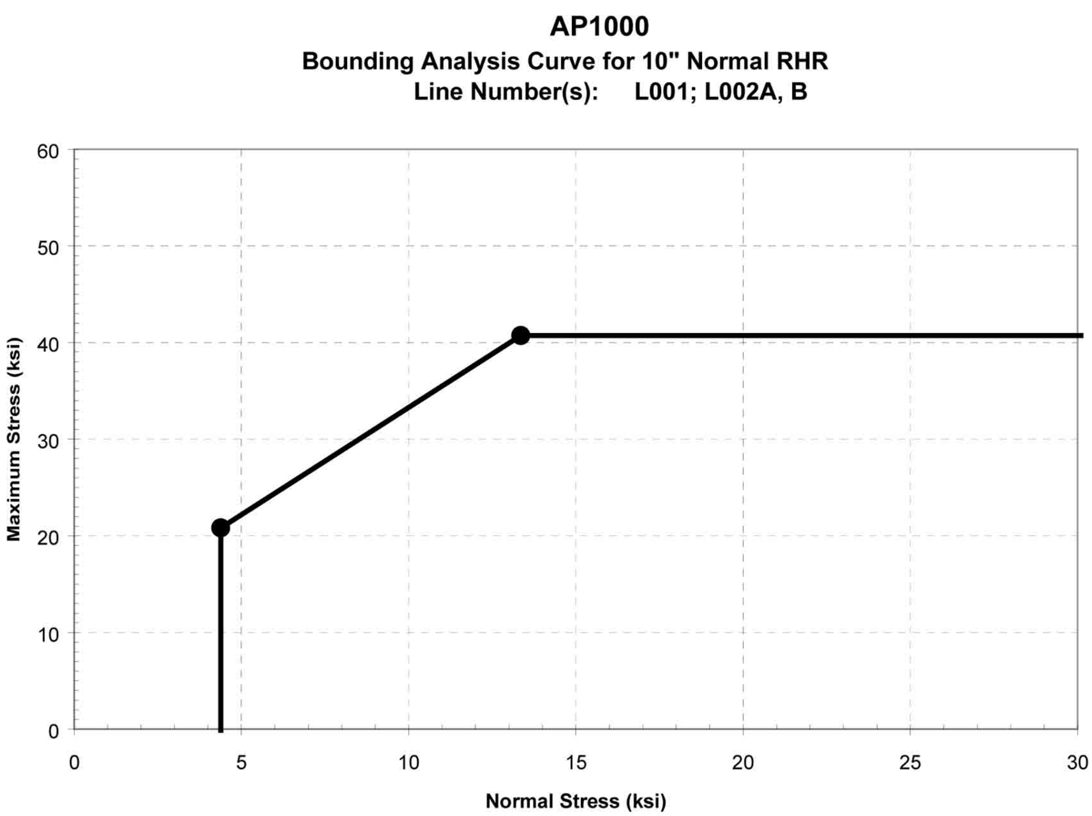


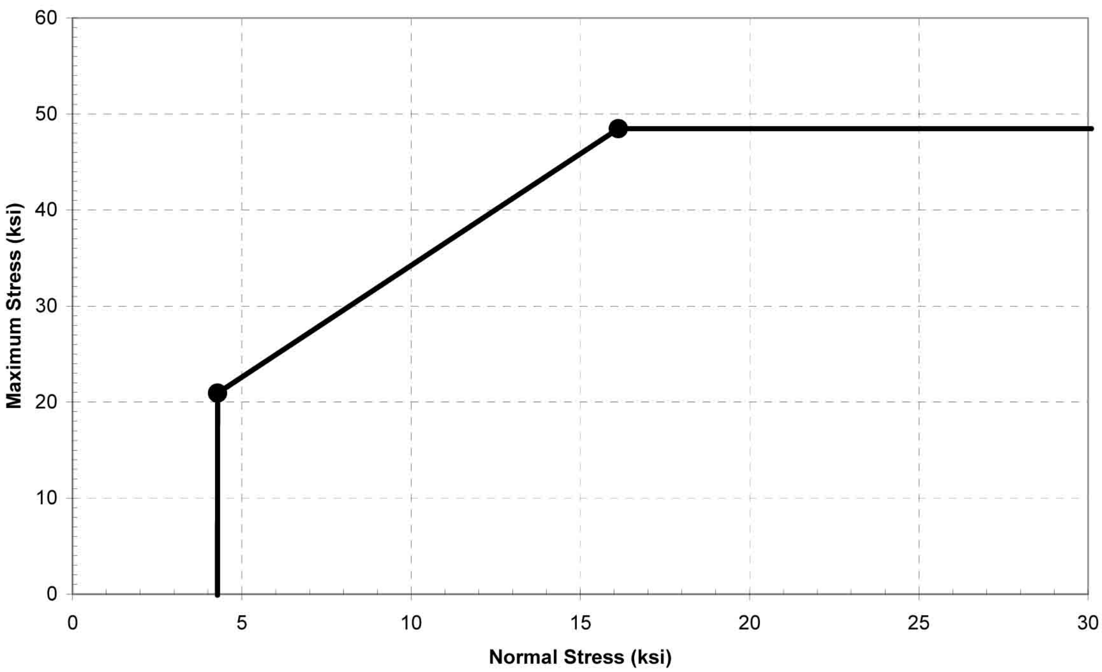
Figure 3B-20
Bounding Analysis Curve for 12″ Normal RHR



Pipe Designator:	BTA	Normal Operating Pressure:	2248.0 psig
System:	RNS	Normal Operating Temperature:	610.0 F
Nominal Diameter:	10 inch	Critical Flaw Size = 2 x Leakage Flaw Size	
Pipe Schedule:	160	Load Margin = 1.0	
Outside Diameter:	10.750 inch		
Pipe Material:	SA-312 TP316LN		
Minimum Weld Thickness:	1.005 inch		

Figure 3B-21
Bounding Analysis Curve for 10″ Normal RHR

AP1000
Bounding Analysis Curve for 8" ADS Stage 2, 3
Line Number(s): L021A,B ; L031A,B



Pipe Designator:	BTA	Normal Operating Pressure:	2235.0 psig
System:	RCS	Normal Operating Temperature:	250.0 F
Nominal Diameter:	8 inch	Critical Flaw Size = 2 x Leakage Flaw Size	
Pipe Schedule:	160	Load Margin = 1.0	
Outside Diameter:	8.625 inch		
Pipe Material:	SA-312 TP316LN		
Minimum Weld Thickness:	0.817 inch		

Figure 3B-22
Bounding Analysis Curve for 8″ ADS Stage 2, 3

Appendix 3C Reactor Coolant Loop Analysis Methods

The AP1000 reactor coolant loop (RCL) model consists of three-dimensional finite elements such as pipes, beams, elbows, masses, and springs. The structural model is subjected to internal pressure, thermal expansion, weight, seismic, and pipe break loadings with imposed boundary conditions. The finite element displacement method is used for the analysis. The stiffness matrix for each element is assembled into a system of simultaneous linear equations for the entire structure. This set of equations is then solved by a variation of the Gaussian elimination method, known as the wave-front technique. This technique makes it possible to solve systems of equations with a large number of degrees of freedom using a minimum amount of computer memory.

3C.1 Reactor Coolant Loop Model Description

The piping model of the reactor coolant loop consists of a number of elements of given dimensions, sizes, and physical properties that mathematically simulate the structural response of the physical system. The system model contains the reactor pressure vessel (RPV), two steam generators (SGs), four reactor coolant pumps (RCPs), the reactor coolant loop piping, and the primary equipment supports. A two-loop model is developed for the AP1000 reactor coolant loop system.

The stiffness and mass effects of branch piping connected to the primary loop piping are considered when significant (Subsection 3.7.3.8.1).

3C.1.1 Steam Generator Model

3C.1.1.1 Steam Generator Mass and Geometrical Model

The steam generator is represented by discrete masses. The geometry of the steam generator vessel is used to determine the properties of the equivalent piping elements that join the steam generator masses for sections of the steam generator above the tubesheet. For the steam generator channel head, a super element is used to represent the stiffness characteristics that link the steam generator lower shell with the steam generator supports and nozzles. The modulus of elasticity and coefficient of thermal expansion corresponding to the thermal conditions are applied to the steam generator equivalent piping elements.

3C.1.1.2 Steam Generator Supports

The values of the steam generator support stiffnesses and locations of the supports are determined from the finite element models of the support members. The stiffness of the upper lateral supports include the steam generator shell flexibility. The local concrete building flexibility is included in the support stiffness.

3C.1.2 Reactor Coolant Pump Model

3C.1.2.1 Static Model

The reactor coolant pump is represented by a super element to represent the mass and stiffness characteristics of the pump. For a thermal expansion analysis, rigid links are modeled in parallel with a super element with the thermal expansion coefficient incorporated.

3C.1.2.2 Seismic Model

The reactor coolant pump is represented by a super element to represent the mass and stiffness characteristics of the pump. The reactor coolant pump model is a detailed model similar to that used to qualify the pump.

3C.1.2.3 Reactor Coolant Pump Supports

There are no reactor coolant pump supports. Two reactor coolant pumps are attached to the steam generator channel head in each of the reactor coolant loops.

3C.1.3 Reactor Pressure Vessel Model

3C.1.3.1 Mass and Geometrical Model

The reactor pressure vessel model consists of equivalent pipe, stiffness, and mass elements. The elements represent the vessel shell, the vessel core barrel, the fuel assemblies, and the integrated head lift package.

The reactor pressure vessel is modeled with equivalent pipe elements and connecting stiffnesses. The equivalent pipe element properties of the vessel and barrel are those of the cylindrical structures. The beam properties of the reactor internals are adjusted to simulate their fundamental frequency. The appropriate modulus of elasticity and coefficient of thermal expansion are used for the equivalent pipe elements representing the reactor pressure vessel.

3C.1.3.2 Reactor Pressure Vessel Supports

The reactor pressure vessel is supported at the four reactor pressure vessel inlet nozzles. Each support consists of a vertical stiffness and a lateral tangential stiffness. The support is represented by a stiffness matrix. The reactor pressure vessel supports are active for the analyzed loading conditions. The reactor pressure vessel model includes the effects of the vessel shell flexibility at the inlet and outlet nozzles. The local concrete building flexibility is included in the support stiffness.

3C.1.4 Containment Interior Building Structure Model

A containment interior building structure finite element model is not required because the seismic inputs to the reactor coolant loop model are provided at all of the building attachments to the reactor coolant loop.

3C.1.5 Reactor Coolant Loop Piping Model

The reactor coolant loop piping model consists of piping elements and bends. Each reactor coolant loop has two cold legs and one hot leg. The straight runs and bends of the cold leg and hot leg are input with the nominal dimensions. Each reactor coolant loop branch connection is represented by a node point. The reactor coolant loop piping model contains distributed masses of the hot and cold leg piping for static deadweight analysis and lumped masses representing the hot and cold leg piping for dynamic analysis.

3C.2 Design Requirements

The reactor coolant piping is qualified to the requirements of the ASME Code, Section III, Subsection NB, 1989 Edition with 1989 Addenda.

The loadings for ASME Code, Section III, Class 1 components are defined in Subsection 3.9.3. The following loadings are considered in the reactor coolant loop piping analysis:

- Design pressure (P)
- Weight (DW)

- Thermal expansion during normal operating condition
- Thermal expansion during other transient conditions (not part of this appendix)
- Safe shutdown earthquake (SSE)
- Design basis pipe break (DBPB)
- Building motions due to automatic depressurization system sparger discharge into the IRWST
- Thermal stratification during transient conditions

In addition to the analyses of these loads, the reactor coolant piping is analyzed for the effect of cyclic fatigue due to the design transients and earthquakes smaller than SSE.

3C.3 Static Analyses

3C.3.1 Deadweight Analysis

The reactor coolant loop piping system is analyzed for the effect of deadweight. The deadweight analysis is performed without considering the dry weight of the directly supported equipment. The effects of the auxiliary branch piping on the reactor coolant loop are generally negligible by the design of the auxiliary supports. A deadweight analysis is performed to include the total weight of the reactor coolant loop piping and the water weight in the components.

The reactor coolant loop deadweight model includes the corresponding active reactor coolant loop supports - reactor pressure vessel supports, and the steam generator column and lower and intermediate lateral strut supports. The steam generator upper lateral snubber supports are considered as inactive.

3C.3.2 Internal Pressure Analysis

The effects of the internal primary coolant pipe pressure are used in the calculations of forces and moments for both the reactor coolant loop piping and equipment supports. The moment stress due to pressure is considered negligible for the ASME Code pipe stress equations.

3C.3.3 Thermal Expansion Analysis

The reactor coolant loop piping is analyzed for the effects of thermal expansion. The thermal expansion analysis model considers the expansion of the reactor coolant loop piping, reactor pressure vessel, steam generator, reactor coolant pump, and the equipment supports. The stiffness effects of the auxiliary piping on the reactor coolant loop expansion are generally negligible by the design of the auxiliary lines supports.

3C.4 Seismic Analyses

The reactor coolant loop piping is analyzed for the dynamic effects of a safe shutdown earthquake (SSE).

The model used in the static analysis is modified for the dynamic analysis by including the lumped mass characteristics of the piping and equipment. The effect of the equipment motion on the reactor coolant loop piping and support system is obtained by modeling the mass and stiffness characteristics of the equipment in the overall system model. The reactor coolant loop seismic

analysis is performed at normal full-power operation. This operating condition is considered based on the lower probability of occurrence of the earthquake at reactor coolant loop temperatures below full power.

The time history integration method of analysis is used for the reactor coolant loops. The seismic input considers the soil profiles described in Subsection 3.7.1. This input is obtained from the nuclear island seismic analysis with time history input generated from the enveloped basemat response spectra of the soil cases described in Subsection 3.7.1. The duration of the input is between 12 to 20 seconds, depending on the duration needed to envelop the design response spectra. Three runs were performed based on the envelope of the soil profiles, the building model at nominal stiffness, and at stiffness varied by + or - 30 percent to account for uncertainties. The reactor coolant loop uses separate time history displacement input from the building analysis at the primary support locations. Full direct integration is used with Rayleigh damping for loop components at 4 percent of critical damping. The steam generator snubbers have different stiffnesses in tension and compression. The mean value of the tension and compression stiffness is used in order to keep the model linear. The reactor pressure vessel vertical supports are acting downward only and are preloaded by deadweight, pressure, and thermal expansion loadings. The time history analysis is performed to evaluate the effect of lift-off of the vessel at the location of these supports.

3C.5 Reactor Coolant Loop Piping Stresses

To prevent gross rupture of the reactor coolant loop piping system, the general and local primary membrane stress criteria must be satisfied. This is accomplished by satisfying Equation (9) in paragraph NB-3652 of the ASME Code, Section III. The secondary stress caused by thermal expansion is qualified by satisfying Equation (12) in paragraph NB-3653 of the ASME Code, Section III.

3C.6 Description of Computer Programs

This section provides a list of computer codes used for the AP1000 reactor coolant loop system analysis. Brief descriptions of the functions of each computer code are the following:

ANSYS – Performs Structural Analysis Using Finite Element Analysis Method. Displacements and loads are calculated at the pipe elements, supports, and equipment nozzles for pressure, deadweight, thermal, and seismic loadings.

Appendix 3D Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment

Safety-related electrical equipment is tested under the environmental conditions expected to occur in the event of a design basis event. This testing provides a high degree of confidence in the safety-related system performance under the limiting environmental conditions. Qualification criteria were revised by IEEE 323-1974 (Reference 1) and by Regulatory Guide 1.89, which endorses this IEEE standard. The concept of aging was highlighted in IEEE 323-1974, and interpretation of the scope of aging and implementation methods were subsequently developed. 10 CFR 50.49 provides the NRC requirements for qualification of equipment located in potentially harsh environments. Therefore, the guidance provided by IEEE 323-1974 is the evolutionary root of requirements, recommended methods, and qualification procedures described in this appendix.

Specific treatment of seismic qualification, part of the qualification test sequence recommended in IEEE 323-1974, is addressed in IEEE 344-1987 (Reference 2). This appendix bases technical guidance, recommendations, and requirements for seismic qualification on IEEE 344-1987.

The AP1000 Equipment Qualification methodology addresses the expanded scope of IEEE 627-1980 (Reference 3), which encompasses the qualification of Class 1E electrical and safety-related mechanical equipment. IEEE 627 generalizes the principles and technical guidance of IEEE 323 and 344. Compliance with the IEEE 323-1974 and 344-1987 is the specific means of compliance with the intent of IEEE 627-1980 for safety-related electrical and mechanical equipment.

Safety-related electrical and mechanical equipment is typically qualified using analysis, testing, or a combination of these methods. The specific method or methods used depend on the safety-related function of the equipment type to be qualified. Safety-related mechanical equipment, such as tanks and valves, is typically qualified by analysis, with supplementary functional testing when functional operability is demonstrated only through testing, as is the case for active valves. Either testing or testing combined with analysis is the method used for environmental and seismic qualification of safety-related (Class 1E) electrical equipment.

The technical discussions of this appendix follow the format headings of the equipment qualification data packages (EQDPs) to be issued as specific qualification program documentation. This formatting (see Section 3D.7) permits easy cross-reference between the methodology defined in this report and the detailed plans contained in the equipment qualification data packages. Attachment A of this appendix is the format used for the equipment qualification data package.

Attachment B of this appendix, "Aging Evaluation Program," describes methods for addressing potential age-related, common-mode failure mechanisms used in AP1000 equipment qualification programs. The approach conforms with current industry positions and makes maximum use of available data and experience in the evaluation, test, and analysis of aging mechanisms.

Attachment C, "Effects of Gamma Radiation Doses Below 10^4 rads on the Mechanical Properties of Materials," provides the basis that radiation aging below 10^4 rads is not a significant factor in the ability of the equipment to perform properly during a seismic event. For some devices, electrical properties are degraded above 10^3 rads. Radiation aging for safety-related equipment which is subject to lifetime doses of less than 10^4 rads (10^3 rads for certain electrical components) and not subject to a high-energy line break environment is not required to be addressed in AP1000 qualification programs.

Attachment D, "Accelerated Thermal Aging Parameters," describes the methodology employed in calculating the accelerated thermal aging parameters used in this program.

Attachment E, "Seismic Qualification Techniques," discusses available methods for establishing a seismic qualification basis, by either test or analysis, and its application to the qualification of safety-related equipment for the AP1000.

3D.1 Purpose

The basic objectives of qualification of safety-related electrical and mechanical equipment follow:

- To reduce the potential for common cause failures due to specified environmental and seismic events
- To demonstrate that safety-related electrical and mechanical equipment is capable of performing its designated safety-related functions.

This appendix describes the methodology that has been adopted to qualify equipment according to IEEE 627-1980, "IEEE Standard for Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations." The two standards primarily used to demonstrate compliance with this standard are IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

3D.2 Scope

The qualification criteria, methods, and environmental conditions described constitute the methodology that is adopted to comply with the standards for the AP1000. This methodology applies to safety-related, seismic Category I electrical and mechanical equipment and is also used for certain monitoring equipment. Seismic Category II equipment is also within the scope of this program. The criteria used for the design of seismic Category II structures, systems, and components are discussed in Section 3.7.

Performance during abnormal environmental conditions, while not specifically designated as an industry or a regulatory qualification requirement, is also addressed by this appendix. Performance during normal service conditions is demonstrated by tests and inspections addressed by the equipment specification. Electromagnetic interference (EMI) testing or analysis is not included in the qualification process and is addressed on an individual equipment basis, as necessary.

3D.3 Introduction

This appendix identifies qualification methods used for the AP1000 to demonstrate the performance of safety-related electrical and mechanical equipment when subjected to abnormal and accident environmental conditions including loss of ventilation systems, feedline, steam line and main coolant system breaks, and seismic events. This appendix provides the expected conditions for various locations in the AP1000. General requirements for the development of plans/procedures/reports are also provided. Section 3D.4 identifies the various industry and regulatory criteria upon which the program is based. Section 3D.5 defines the design specifications and applicable test environments. Section 3D.6 defines the basis for the qualification method selection. Section 3D.7 outlines the documentation requirements.

3D.4 Qualification Criteria

The environmental requirements considered in the design of safety-related equipment are embodied in GDC 2, "Design Bases for Protection Against Natural Phenomena"; GDC 4, "Environmental and Missile Design Bases"; and GDC 23, "Protection System Failure Modes." GDC 1, "Quality Standards and Records," and Criterion III, "Design Control," Criterion XI, "Test Control," and Criterion XVII,

"Quality Assurance Record" of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to require that the environmental design of safety-related equipment is verified, documented, and controlled.

The qualification methods described in this appendix are used to verify the environmental design basis and capability of the safety-related electrical and mechanical equipment supplied for the AP1000. The results of the verification, as well as the design basis for each equipment, is documented in an equipment qualification data package. (See Attachment A for sample format.) Design control, test control, and quality assurance record keeping is performed through the AP1000 Quality Assurance Program. (See Chapter 17.)

3D.4.1 Qualification Guides

IEEE 323-1974 and 344-1987 serve as the basis upon which the AP1000 equipment qualification methodology demonstrates compliance with IEEE 627-1980. NRC regulations stated in 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," and NRC guidance provided in Regulatory Guide 1.89, and Regulatory Guide 1.100, endorse IEEE 323-1974 and IEEE 344-1987, respectively. The intent of the more general IEEE 627-1980 is addressed through conformance with IEEE 323 and 344.

3D.4.1.1 IEEE Standards

The following lists additional standards and guides used in developing the methodology:

- IEEE 98-1984, "IEEE Standard for the Preparation of Test Procedures for the Thermal Evaluation of Solid Electrical Insulating Materials"
- IEEE 100-1996, "IEEE Standard Dictionary of Electrical and Electronic Terms"
- IEEE 308-1991, "IEEE Standard Criteria for Class 1E Power System for Nuclear Power Generating Stations"
- IEEE 317-1983, "IEEE Standard for Electric Penetration Assemblies in Containment Structure for Nuclear Power Generating Stations"
- IEEE 381-1977, "IEEE Standard Criteria for Type Tests of Class 1E Modules Used in Nuclear Power Generating Stations"
- IEEE 382-1996, "IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Generating Stations"
- IEEE 383-1974, "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations"
- IEEE 420-1982, "IEEE Standard Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Powered Generating Stations"
- IEEE 494-1974, "IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Stations"
- IEEE 535-1986, "IEEE Standard for Qualifying Class 1E Lead Storage Batteries for Nuclear Power Generating Stations"

- IEEE 572-1985, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations"
- IEEE 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"
- IEEE 649-1991, "IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations"
- IEEE 650-1990, "IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations"
- IEEE-741-1997, "IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations"
- ANSI/IEEE C37.98-1987, "IEEE Standard for Seismic Testing of Relays."

3D.4.1.2 NRC Regulatory Guides

In the area of seismic and environmental qualification of safety-related electrical and mechanical equipment, the NRC has issued the following Regulatory Guides:

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" – The guide endorses ANS and ANSI standards for quality assurance programs, but is considered here specifically for guidance in determining documentation adequacy. Appendix A of the guide, Item 9, "Procedures for Performing Maintenance," addresses procedural and documentation requirements for maintenance of safety-related equipment, preventive maintenance, repair, and replacement. This guide is a source in the development of qualification in the on-going qualification programs discussed in Subsection 3D.6.4.

Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Plants" – The guide prescribes acceptable values of damping used in elastic modal dynamic seismic analysis of seismic Category I structures, systems, and components. The AP1000 equipment qualification program is based on Regulatory Guide 1.61 and on values considered to be acceptable based on past NRC acceptances. The safe shutdown earthquake (SSE) damping values used for the qualification of mechanical and electrical equipment are listed in Table 3.7.1-1 of Chapter 3.

Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants" – The guide endorses, with certain qualifications, IEEE 317-1983. External circuit protection of electric penetration assemblies should meet the provisions of Section 5.4 of IEEE 741-1986, "Criteria for Protection of Class 1E Power Systems and Equipment in Nuclear Generating Stations, as these are beyond the of scope IEEE 317. The AP1000 design complies with IEEE 741-1997. The AP1000 equipment qualification program employs the recommendations of Regulatory Guide 1.63, Revision 3, in specifying qualification plans as a means of supplementing the guidance of IEEE 317 and 323.

Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants" – The guide endorses, with certain qualifications, IEEE 382-1972. The AP1000 equipment qualification program employs recommendations of Regulatory Guide 1.73, but gives preference to the guidance of IEEE 382-1996, where it is necessary to supplement the guidance of IEEE 323 or 344 in specifying qualification plans for electric valve operators.

Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants" – The guide provides guidance for conformance with 10 CFR 50.49, and endorses the procedures of IEEE 323-1974 as an acceptable means for qualifying Class 1E equipment. Implicit in the endorsement of IEEE 323 is the reference to seismic qualification methods of IEEE 344 as a part of the qualification test sequence. (See Regulatory Guide 1.100 later in this discussion.) The AP1000 equipment qualification methodology addresses the recommendations of Regulatory Guide 1.89 by the following:

- The recommendations of IEEE 323-1974 are met by the methods discussed in this appendix
- The radiation source terms used in qualification differ from those of Regulatory Guide 1.89, and are described in Section 3D.5 of this appendix
- The seismic qualification requirements employ the recommendations of IEEE 344-1987 as described in Attachment E of this appendix.

Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis" – The guide describes methods and procedures for the following:

- Combining the values of the response of individual modes in a response spectrum modal dynamic analysis to find the representative maximum value of a particular response of interest for each of the three orthogonal seismic spatial components
- Combining the maximum values (or representative maximum values) of the responses for a given element of a system or item of equipment, determined for each of the three orthogonal spatial components.

The AP1000 equipment qualification program employs methods consistent with the recommendations of Regulatory Guide 1.92 when combining individual modal response values or the response of three independent spatial components in seismic analyses.

Regulatory Guide 1.97, Revision 3, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident.” The guide describes a method acceptable to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant. The AP1000 program, identified as the post-accident monitoring instrumentation system (PAMS), provides the capability to monitor plant variables and systems operating status during and following an accident. PAMS includes those instruments provided to indicate system operating status and furnish information regarding the release of radioactive materials.

Regulatory Guide 1.100, "Seismic Qualification of Electrical Equipment for Nuclear Power Plants" – The guide endorses IEEE 344-1987. Regulatory Guide 1.100 particularly notes that IEEE 344-1987 is applied in the qualification of safety-related mechanical equipment, as well as Class 1E electrical equipment. The AP1000 equipment qualification methodology employs the recommendations of Regulatory Guide 1.100, as described in Attachment E of this appendix.

Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components" – The guide describes specific methods for developing floor (and other equipment mounting locations) response spectra. Included are specific criteria for the broadening frequency amplitude peaks and smoothing of the frequency amplitude spectrum to incorporate conservatism in the seismic requirements. This is to compensate for other uncertainties of analysis. The AP1000 equipment qualification program employs methods consistent with the recommendations of Regulatory Guide 1.122.

Regulatory Guide 1.131, "Qualification Tests of Electrical Cables, Field Splices, and Connections for Light-Water Cooled Nuclear Power Plants" – The guide endorses IEEE 383-1974. The AP1000 equipment qualification program employs the recommendations of Regulatory Guide 1.131 in specifying the qualification program plans where this guide supplements the guidance of IEEE 383 and to further demonstrate conformance with the guidance of IEEE 323. As neither IEEE 383 nor Regulatory Guide 1.131 specifically addresses considerations for cable field splices and connections, guidance for their qualification is taken from IEEE 572 and Regulatory Guide 1.156.

Regulatory Guide 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants" – The guide endorses IEEE 572-1985. The AP1000 equipment qualification program employs the recommendations of Regulatory Guide 1.156 in specifying the qualification program plans where this guide supplements the guidance of IEEE 572 to demonstrate conformance with the guidance of IEEE 323.

Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants" – The guide endorses IEEE 535-1986. The AP1000 equipment qualification program employs the recommendations of Regulatory Guide 1.158 in specifying the qualification program plans where this guide supplements the guidance of IEEE 535 to demonstrate conformance with the guidance of IEEE 323.

Regulatory Guide 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems.” Regulatory Guide 1.180 provides guidance to evaluate electromagnetic and radio-frequency interference in safety-related instrumentation and control systems. The AP1000 equipment qualification program employs methods consistent with the recommendations of Regulatory Guide 1.180, where applicable.

Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactor.” The radiation dose rates and integrated doses applicable for AP1000 following a design basis accident are determined based on the criteria of NUREG-1465 and this regulatory guide.

3D.4.2 Definitions

Definitions of terms used in this appendix are contained in the referenced standards and IEEE 100, "The Authoritative Dictionary of IEEE Standard Terms, Seventh Edition." Subsection 3D.4.5 clarifies the definitions of "life" (that is, design, shelf, and qualified life) as used in this methodology. The terms "design life" and "qualified life" have the meanings set forth in IEEE 323 and are used in the context of that standard.

3D.4.3 Mild Versus Harsh Environments

Qualification requirements differ for equipment located in mild and harsh environments.

IEEE 323 defines a mild environment as an environment expected as a result of normal service conditions and the extremes of abnormal service conditions where a safe shutdown earthquake is the only design basis event of consequence or conditions where thresholds of material degradation are reached. The following limits are established as the delimiting environmental parameter values for mild and harsh environments.

Typically a mild environment conforms with the environmental parameter limits of Table 3D.4-1, though others may apply to specific equipment applications or locations.

The scope of 10 CFR 50.49 is limited exclusively to equipment located in a harsh environment. The AP1000 equipment qualification program conforms with the requirements of 10 CFR 50.49 for the

qualification of harsh environment equipment. The "radiation-harsh" environment is a significant subset of the harsh environment category. A radiation-harsh environment is defined for equipment designed to operate above certain radiation thresholds where other environmental parameters remain bounded by normal or abnormal conditions. Any equipment that is above 10^4 rads gamma (10^3 for electronics) will be evaluated to determine if a sequential test which includes aging, radiation, and the applicable seismic event is required or if sufficient documentation exists to preclude such a test.

3D.4.4 Test Sequence

Where the test sequence deviates from that recommended by IEEE 323-1974, the deviation is justified. The test sequence employed for a given hardware item is specified in the equipment qualification data package Sections 2.1 and 3.6 (see Attachment A for example). Note that for this reference and subsequent references to Attachment A the information in Attachment A will be completed in accordance with Subsection 3.11.5. Clarifications to the IEEE 323-1974 recommended test sequence are discussed in the following:

1. Burn-In Test

   For electronic equipment, a burn-in test is completed, before operational testing of the equipment, to eliminate infant failures. The test consists of energizing the equipment for a minimum of 50 hours at nominal voltage and frequency under ambient temperature conditions. Any malfunction observed during these tests are repaired, and the 50-hour burn-in test is repeated for the repaired portion of the equipment.

2. Performance Extremes Test

   For equipment where seismic testing has previously been completed employing the recommended methods of IEEE 344-1987, seismic testing is not repeated. Testing of the equipment to demonstrate qualification at performance extremes is separately performed as permitted by IEEE 323-1974, Subsection 6.3.2(3). Additional discussion is provided in Subsection 3D.6.5.1.

3. Aging Simulation and Testing

   For equipment located in a mild environment, aging is addressed as described in Subsections 3D.6.3, 3D.6.4, and Attachment B. If there are no known aging mechanisms that significantly degrades the equipment during its service life, it is acceptable to perform seismic testing of unaged equipment. Separate testing or analysis (or both) is provided to demonstrate that the aging of components is not significant during the projected service or qualified life of the equipment.

4. Synergistic Effects

   An important consideration in the aging of equipment for harsh environment service is the possible existence of synergistic effects when multiple stress environments are applied simultaneously. This potential is addressed by conservatism inherent in the determination and use of the worst-case aging sequence and conservative accelerated aging parameters.

   The combination of effects from pressure, temperatures, humidity, and chemistry are addressed by the high-energy line break (HELB) tests. Since the test item is not exposed to radiation during this test, the effects of this parameter are conservatively addressed by subjecting the test items to the required total integrated dose before the high-energy line break. Specifically for instruments,

the summing of errors for the irradiation and high-energy line break portions of the test sequence is a means of achieving conservatism.

5. Visual Inspections/Disassembly

   The results of post-test visual inspections are not necessarily documented unless problems are discovered. Disassembly is performed only when test results or visual inspections require further investigation.

3D.4.5 Aging

3D.4.5.1 Design Life

The AP1000 equipment qualification program relies on the IEEE 323 definition for design life, particularly its distinction with respect to qualified life.

Instead of determining a qualified life for mild environment equipment for which the seismic event is the exclusive design basis event to be addressed, a design life is determined. Design lives offered in manufacturers' literature are accepted cautiously, particularly where the equipment is typically used for applications outside the nuclear industry.

An application of the design life is substantiated by sound bases in reliability theory and relevant industry standards, or experience data sources within the nuclear industry. Analyses treat the applicability and similarity of the equipment and conditions relevant for the AP1000 safety-related application. These analyses, and documentation of such, conform with guidelines of IEEE standards, as applicable, and with Sections 3D.6 and 3D.7 of this appendix.

3D.4.5.2 Shelf Life

Based on recommended storage environments, the shelf life of an equipment item is not typically a significant portion of the defined qualified life. For example, ambient temperatures during storage are typically less than the operating temperatures assumed for aging calculations. Therefore, as long as equipment is in storage and is not energized (not experiencing self-heating), a reduction in qualified life is not appropriate. However, if storage conditions differ significantly from those recommended or the storage time becomes dramatically extended, the impact to the qualified life is determined by application of the Arrhenius time-temperature relationship.

3D.4.5.3 Qualified Life

A qualified life is established for each item of safety-related equipment that is exposed to a harsh environment based on the conditions postulated at the equipment location with consideration of the equipment operability requirements.

The determination of qualified life considers potential aging mechanisms resulting from significant in-service thermal, radiation, and vibration sources, and the effects of operational cycling (mechanical or electrical or both). Generally, all aging mechanisms do not apply to each item of equipment. Relevant aging mechanisms addressed or simulated are determined jointly with the identification of the equipment's critical components, functional modes, and material characteristics, and the assessment of tolerable limits in degradation of the components. An a priori consideration in selecting equipment to qualify is the evaluation of the equipment's inherent capability to survive and operate under the conditions for which it is qualified.

Since past qualification tests have provided a substantial basis for this assessment (indeed, some may provide sufficient basis to preclude any new testing as part of the AP1000 program) specific

guidance on each equipment type is not provided here. Application of the lessons of past tests, insights provided in generic industry communications (for example, technical bulletins, NRC Information Notices), and sound judgment in the development of test plans and analysis procedures are addressed in the documentation of qualification for each equipment type, as applicable.

Qualified life is established by the most limiting of the five aging mechanisms. Qualified life may be limited by the tolerable degradation of a single component or material critical to the equipment's capability to perform its safety function. Aging is subject to the requirement for margin. See Subsection 3D.4.8 of this appendix.

For some equipment, qualified life is established on the basis of periodic replacement of certain short-lived, age-sensitive components. The user complies with the mandatory replacement practices documented in the equipment qualification data packages (see Subsection 3D.7.2.5 and Attachment A, Subsections 3.9.3 and 6.1) to affirm the equipment qualified life.

The objective of thermal and irradiation qualified life testing is to simulate, according to the available empirical material data, the degradation effects such that the equipment is in its end-of-life condition before the application of the design basis event conditions testing.

Thermal qualified life is evaluated using the Arrhenius time-temperature relationship. (See more detailed discussions in Attachments B and D of this appendix.) The activation energy is the exclusive material-dependent parameter input into the Arrhenius time-temperature relationship. The activation energy is an empirically determined parameter indicative of the thermal degradation of a physical property of a material (for example, elasticity of silicone rubbers or insulation resistance of cross-linked polyethylene cable insulation). Each material may have more than one physical property that may be subject to thermal degradation over time. Consequently, it may have different activation energies with respect to each property. Thus, the selection of activation energy considers the material property most germane to the safety-related function of the material or component. (Also see Subsection 3D.4.5.4.)

Common practice for the evaluation of irradiation-induced degradation is to consider the sum of estimated life and the accident radiation doses before design basis event testing. When testing, the total dose is applied during the radiation aging simulation portion of the qualification test sequences. This is considered conservative because the equipment has accumulated an exposure, or total integrated dose, before the initiation of the seismic and accident environment testing. Further bases for test dose determination are provided in Subsection 3D.5.1.2. Sufficient margin must be included in test parameters (see Subsection 3D.4.8). The same margins are applied in an analysis of radiation life or design basis event radiation dosage.

The simulation of age also includes the effects of operational cycling, both electrical and mechanical. Generally, these considerations are applied specifically to electromechanical equipment such as valve operators, limit switches, motors, relays, switches, and circuit breakers. Furthermore, the simulation of these effects is waived where existing data demonstrates equipment durability greatly in excess of estimated number of operating cycles for Class 1E service. Analysis or justification is provided for any case where operational cycling is omitted in the test sequence.

It is not practicable to simultaneously simulate the aspects of aging. Development of each test plan considers known synergies and sequences the simulation of the various applicable aging mechanisms with regard for conservatism of the overall effect on the test specimens.

3D.4.5.4 Qualified Life Reevaluation

It may be possible to extend the qualified life of a particular piece of equipment at some future date by comparing the actual in-plant environments and conditions during the equipment's installed life to

the values assumed for the AP1000 in establishing the qualified life. For example, the thermal qualified life might be extended by performing an analysis of actual internal or external temperatures (or both) experienced. Continuous temperature monitoring or use of sample devices for testing and trending materials aging may be used. These efforts reveal the conservatism of the original thermal life calculation, which assumes that the maximum value specified for the normal plant operating environment endured at all times.

Although a strict Arrhenius calculation may yield an extended qualified life, care is taken in using this extrapolation because of uncertainties in the methodology. The Arrhenius time-temperature relationship relies on empirically determined activation energies of materials. This parameter has been determined for a number of materials to at least a good approximation for small temperature extrapolations. Extrapolation of the Arrhenius model to time periods of temperature beyond the range of materials test data is questionable and may result in large errors.

Calculated qualified lives based on this methodology should be limited to 20 years unless sound technical bases can be cited. This position is consistent with industry guidelines such as IEEE 98-1984, NUREG/CR-3156 (Reference 4), and EPRI NP-1558 (Reference 5).

3D.4.6 Operability Time

The post-accident operability times specified in Subsection 1.7.1 of each equipment qualification data package (see Attachment A) are conservatively established based on the safety-related function performed by that equipment for the spectrum of design basis event conditions. These include the following:

- Trip and/or monitoring functions of sensors and instruments
- Operability requirements for electromechanical equipment
- Duration of required operability for active valves.

This evaluation also considers what consequences the failure of the device has on the operator's action or decisions and the mitigation of the event. Table 3D.4-2 lists and explains typical operability times.

For monitoring functions, simulated aging techniques are employed to shorten the test time following a high-energy line break. These also comply with the margin guidelines of Subsection 3D.4.8.

Margins for trip function requirements are contained in the high-energy line break envelopes that encompass a full spectrum of break sizes. The defined margins are also justified by the fact that the signal generated by the sensor is locked in by the protection system and does not reset should the sensor fail after completion of its designated trip time requirement.

3D.4.7 Performance Criterion

The basic performance criterion is that the qualification test program demonstrates the capability of the equipment to meet the safety-related performance requirements defined in the equipment qualification data package, Section 1.7, while subjected to the environmental conditions specified in the equipment qualification data package, Section 1.8. Where three or more specimens are tested, failure of one of three may be considered a random failure, subject to an investigation concluding that the observed failure is not indicative of a common-mode occurrence.

For equipment for which aging is addressed by evaluation of appropriate mechanism(s) through a review of available material and component information, the basic acceptance criterion is that the evaluation of test data demonstrates that the effect of aging is minor and does not affect the capability of the aged equipment to perform specified functions.

3D.4.8 Margin

IEEE 323 (Subsection 6.3.1.5) recommends that margin be applied to the most severe specified service conditions in order to establish the conditions for qualification. This margin is provided in order to account for normal variations in commercial production of equipment and for reasonable errors in defining satisfactory performance. Further guidance for determining the acceptability of margin with respect to application-specific or location-specific requirements is provided by the NRC in NUREG-0588 and Regulatory Guide 1.89, Revision 1. Margins are included in addition to conservatisms applied during the derivation of the local environmental conditions of the equipment, unless the conservatism is quantified and specifically shown to meet or exceed the guidance of IEEE 323, NUREG-0588, and Regulatory Guide 1.97.

Consistent with IEEE 323, margin is incorporated into the specification of the generic qualification parameters by either increasing the test levels, number of test cycles, test duration, or a combination of these options as appropriate. The AP1000 generic qualification parameters are selected to envelop a range of loss of coolant accident and high-energy line break sizes, and equipment locations. Margin in seismic conditions for test and analysis are addressed in Subsection 3D.4.8.4. The margins available for a specific application may be larger than the generic equipment qualification test objective for seismic events and some events outside containment and are verified on an application-specific basis.

In defining qualification parameters, the AP1000 equipment qualification program incorporates margin as described in the following subsections. Table 3D.4-3 lists margin requirements applied.

For generic testing, margin is applied at the time of testing to cover known safety-related applications of the equipment. Generally, this results in a worst-case test that provides substantial margin for applications where lesser environments apply. Application of margin for seismic qualification addresses several cases unique to the qualification approach. (See Subsection 3D.4.8.4.)

3D.4.8.1 Normal and Abnormal Extremes

As indicated in Section 7 of IEEE 323, the application of margin is directed at specifying adequate qualification requirements for the most severe service conditions represented by the design basis events (that is, high-energy line break accidents and seismic events). Consequently, the AP1000 equipment qualification methodology does not apply any systematic margin to the normal and abnormal environment parameters in defining the qualification conditions.

For electronic equipment not required to operate in a high-energy line break environment, additional margin is included by requiring that the equipment operate through the conservative normal and abnormal service conditions indicated in Figure 3D.5-1. The environmental parameters at least equal the specified range of service condition parameters. An exception occurs for transmitters where a performance verification is completed at 130°F on each transmitter to encompass the specified maximum abnormal conditions. For equipment to be qualified to operate in a high-energy line break environment, qualification to the severe high-energy line break conditions demonstrates ample margin for acceptable performance under certain specified normal and abnormal service conditions.

3D.4.8.2 Aging

No specific margin is applied to the time component in deriving appropriate aging parameters, if margin is included in deriving the accelerated aging parameters employed for simulating each applicable aging mechanism.

Margin may be addressed by demonstrating the adequacy of the aging simulated by test through the calculation of time-temperature equivalence (See Attachment B of this appendix) or the comparison

of simulated parameters with those applicable to the intended service of the equipment. The installed life of equipment must not exceed the thermal qualified life demonstrated by this calculation. Additionally, the selection and use of the thermal aging parameters both for test and subsequent calculations are subject to criteria, including the following:

- Test temperature must endure for at least 100 hours
- Test temperature must exceed any application temperature (that is, the normal or abnormal environment in which the equipment is to be used, and for which the life is calculated)
- Test temperature must be less than state-change temperature for materials critical to the equipment safety-related function or capability to endure the subsequent design basis event testing
- A conservative activation energy is used. Activation energies for materials critical to the equipment safety-related function or capability to endure the subsequent design basis event testing are considered. Materials may have several activation energies, each for a different material property. Relevant material properties are considered.

If margin is not demonstrated through conservatism in the aging parameters or calculation, then a +10 percent time margin is included.

A margin of 10 percent in the other parameters (for example, irradiation, operational cycling) applies to both the aging simulation and the post-accident simulated aging, with few exceptions.

For equipment required by design to perform its safety-related function within a short time period into the design basis event (that is, within seconds or minutes), and having completed its function, subsequent failure is shown not to be detrimental to plant safety, margin by percentage of additional time or equivalent time-temperature is not applied. Margins for trip function requirements are contained in the worst-case high-energy line break envelope. Test parameters are simulated on a real-time basis with the transient condition margins listed in Table 3D.4-3. Trip signals, once generated by the sensors, are locked in by the protection system and do not reset in the event of subsequent sensor failure.

3D.4.8.3 Radiation

An additional 10 percent is added to the calculated total integrated dose in specifying the test requirements.

3D.4.8.4 Seismic Conditions

Required response spectra included in Subsection 3.7.2 or other AP1000 program specifications are the conditions to be enveloped. No amplitude margin is added to these conditions. Peak broadening is also discussed in Subsection 3.7.2. Seismic qualification by analysis addresses margin requirements by other methods of conservatism while using the same sets of requirements - no amplitude margin is included. For qualification tests, the test facility increases the amplitude of seismic profiles by 10 percent to incorporate margin.

For most applications, considerable margin exists with respect to the acceleration levels employed and the width of the response spectra. Further details are addressed in Attachment E.

3D.4.8.5 High-Energy Line Break Conditions

The envelopes specified for high-energy line breaks are selected to encompass the transients resulting from a spectra of loss of coolant accidents and high-energy line break sizes and locations, and various nodes in the containment. As a consequence, these design envelopes already contain significant margin with respect to any transient corresponding to a single break.

The AP1000 equipment qualification methodology requires that the qualification envelopes be derived with a margin of 15°F and 10 psi with respect to the design envelopes in Figures 3D.5-2 and 3D.5-3. The margin on dose is identified by comparing the location specific dose requirements and the AP1000 equipment qualification parameters.

The alkalinity of the chemistry is increased by 10 percent with respect to the peak value determined for the AP1000 containment sump conditions.

3D.4.9 Treatment of Failures

The primary purpose of equipment qualification is to reduce the potential for common mode failures due to anticipated environmental and seismic conditions. The redundancy, diversity, and periodic testing of nuclear power plant safety-related equipment are designed to accommodate random failures of individual components.

Where an adequate test sample is available, the failure of one component or device together with a successful test of two identical components or devices indicates a random failure mechanism, subject to an investigation concluding that the observed failure is not common mode. Where insufficient test samples prevent such a conclusion, any failures are investigated to ascertain whether the failure mechanism is of common mode origin. Should a common mode failure mechanism be identified as causing the failure, either a design change is implemented to eliminate the problem or a repeat test completed to demonstrate compliance with the criteria.

For those mild environment equipment items that, through a review of available documentation, are subject to failure during a seismic event due to significant aging mechanisms, the material or component is replaced or monitored through a maintenance/surveillance program.

3D.4.10 Traceability

A system of baseline design documentation is instituted to control the design, procurement, and manufacturing of safety-related products. As part of this quality control program, critical parts are identified and assigned a level of control to reflect the estimate of potential qualification or procurement problems. In addition, levels of quality inspection are also assigned to each part. The baseline design documentation describes the equipment in sufficient detail (drawing number, part number, manufacturer) to establish traceability between equipment shipped and equipment tested in the qualification program.

3D.4.10.1 Auditable Link Document

The purchaser of equipment referencing this program requires an auditable link document that provides a tie between the specific equipment and documentation of qualification reviewed for acceptance under this program. This auditable link document includes one or more of the following sections, as applicable.

3D.4.10.1.1 Equipment Link

This documentation certifies that the plant specific equipment is covered by the applicable equipment test reports. This link reflects a comparison of the as-built drawings, baseline design document or other documentation of the tested equipment to the specific equipment.

3D.4.10.1.2 Component Link

This documentation certifies that the components (for example, replacement parts) used in the specific equipment are represented in the applicable test reports or via analysis under a component aging program, such as that described in Attachment B (Subprogram B). This link applies only to equipment whose equipment qualification data package references a component testing program. This link reflects a comparison of the as-built drawings, baseline design document, or other documentation of the specific equipment to the component program listing.

3D.4.10.1.3 Material Link

This documentation certifies that the materials used in the equipment are represented in a materials aging analysis, such as that described in Attachment B, (Subprogram B). This link applies only to equipment whose equipment qualification data package references the materials aging analysis and reflects a comparison of the as-built drawings, baseline design document, or other documentation of the plant specific equipment to the materials aging analysis listing.

3D.4.10.2 Similarity

Where differences exist between items of equipment, analysis may be employed to demonstrate that the test results obtained for one piece of equipment are applicable to a similar piece of equipment. Documentation of this analysis conforms with guidelines in IEEE 323 and 627, and Subsection 3D.6.2.1 and Section 3D.7 of this appendix.

3D.5 Design Specifications

The conditions and parameters considered in the environmental and seismic qualification of AP1000 safety-related equipment are separated into three categories: normal, abnormal, and design basis event. Normal conditions are those sets and ranges of plant conditions that are expected to occur regularly and for which plant equipment is expected to perform its safety-related function, as required, on a continuous, steady-state basis. Abnormal conditions refer to the extreme ranges of normal plant conditions for which the equipment is designed to operate for a period of time without any special calibration or maintenance effort. Design basis event conditions refers to environmental parameters to which the equipment may be subjected without impairment of its defined operating characteristics for those conditions.

The following subsections define the basis for the normal, abnormal, design basis event, and post-design basis event environmental conditions specified for the qualification of safety-related equipment in the AP1000 equipment qualification program. (These are cited in Section 1.7 of each equipment qualification data package; See Attachment A.)

The service conditions simulated by the test plan are identified in equipment qualification data package Section 3.7. (See Subsection 3D.7.4.6 and Attachment A.) In general, the parameters employed are selected to be equal to (normal and abnormal) or have margin (design basis event and post-design basis event) with respect to the specified service conditions of equipment qualification data package, Section 1.7, as recommended by IEEE 323. These conditions are conservatively derived to allow for possible alternative locations of equipment within the plant.

3D.5.1 Normal Operating Conditions

Equipment not subject to high-energy line break environments is qualified for normal and abnormal conditions, as applicable, employing a cyclic test sequence of environmental and electrical extremes. A typical test profile, including voltage and frequency cycling, is shown in Figure 3D.5-1.

3D.5.1.1 Pressure, Temperature, Humidity

The calculated values for temperature, pressure, and humidity during normal operation are specified in Table 3D.5-1 as a function of in-plant location.

3D.5.1.2 Radiation Dose

The normal operating dose rates and consequent 60-year design expectation doses at various locations inside containment are specified in Table 3D.5-2. These values have been derived from theoretical calculations assuming an expected 60 years of continuous operation with a reactor power of 3468 MWth (including 2-percent power uncertainty) and steady-state operating conditions. Equivalent data at various locations outside containment are also specified in Table 3D.5-2.

The total integrated dose employed for testing is a combination of normal and accident doses (where applicable), and is defined to equal or exceed the maximum radiation dose contained in the equipment qualification data package. (See Section 3D.7 and Attachment A.) A margin of 10 percent is included in defining the total integrated doses for testing. Normal operating and accident gamma doses are simulated using a cobalt-60 or spent fuel source. The test dose is applied at a rate approximate to the maximum accident dose rate. Irradiation dose rates less than the maximum are considered where there is significant shielding (greater than two mm of steel) or where the peak in-containment design basis event dose rate is not expected to affect the equipment's electrical performance.

Low radiation dose rates encountered during normal operation for most equipment are not considered critical parameters because of the resultant low total integrated dose (10^4 to 10^5 rads) achieved. For equipment not required post-accident, material can be selected based on previous test results. Another test on the completed assembly is not required.

If equipment is located in an environment where the normal total integrated dose exceeds the threshold for radiation damage, then testing is required. For equipment required post-accident, the dose received during normal operation is usually an insignificant part of the total integrated dose, including accident conditions effects. The supposition that a concern over low dose rate effects diminishes as the total integrated dose decreases is supported by Sandia National Laboratories tests (References 6 and 7) on selected materials over a range of dose rates. These studies indicate that reduction in original properties is about the same (and not significant) for dose rates up to a total integrated dose in the megarad range. Although these tests were not performed at dose rates as low as those expected in a nuclear power plant and electrical properties were not evaluated, they do give some indication of the effect of varying the rate.

Based on results of research programs to date and low total integrated dose reached during normal operation, the AP1000 equipment qualification program does not consider degradation due to low dose rate effects to be a significant concern. Therefore, the program does not include any action other than inspecting organic material degradation in the plant through normal maintenance.

3D.5.2 Abnormal Operating Conditions

Abnormal environments are defined to recognize possible plant service abnormalities that lead to short-term changes in environments at various equipment locations.

For equipment located inside containment, several abnormal environment types are considered in Subsection 3D.5.2.1. Equipment located outside containment is addressed in Subsection 3D.5.2.2.

3D.5.2.1 Abnormal Environments Inside Containment

In the AP1000 equipment qualification program there are multiple events postulated at least once over the 60 year design expectation which cause abnormal environmental conditions in the containment. These are divided into two groups of events, based on peak containment temperatures expected.

Group 1: 150°F Events

- Loss of a fan cooler
- Loss of all ac for up to 2 hours
- Pressurizer safety valve open/close during reactor coolant system transient.

Group 2: 250°F Events

- Spurious automatic depressurization system (ADS) actuation
- Passive residual heat removal (PRHR) system use (long-term)
- Reactor coolant system depressurization via pressurizer safety valve
- Small loss of coolant accident.

Table 3D.5-3 presents the conditions associated with each of these abnormal environment events. Plant recovery occurs after each event with varying degrees of time and maintenance efforts. Thus, the conditions resulting from these events are considered in the development of aging test parameters. Event frequency, conditions, and duration are accounted for within the context of the qualified life objective of each equipment type test program.

3D.5.2.2 Abnormal Environments Outside Containment

Figure 3D.5-1 represents the assumptions made in defining potential abnormal environments due to loss of air-conditioning or ventilation systems.

Table 3D.5-4 defines the abnormal environments as a function of equipment location. The assumed duration of the abnormal conditions specified in Table 3D.5-4 are consistent with operating practices and technical specification limits. For certain plant applications, qualification for abnormal environments is not necessary when equipment is located in environmental zones that do not exceed manufacturer's design limits for equipment operation.

3D.5.3 Seismic Events

See Attachment E.

3D.5.4 Containment Test Environment

Regulatory Guide 1.18 specifies that containment integrity is demonstrated at 1.15 times design pressure. The design pressure of the AP1000 containment is 59 psig. Consequently, the maximum pressure specified for the containment test is 59 x 1.15 = 67.85 psig. Other environmental parameters (such as temperature and humidity) of the containment test are adequately enveloped by the parameters specified for normal or abnormal plant conditions.

3D.5.5 Design Basis Event Conditions

Performance requirements are specified for those design basis events for which the equipment performs a safety-related function and which have a potential for changing the equipment environment due to increased temperature, pressure, humidity, radiation, or seismic effects. The environmental conditions for each applicable design basis event are summarized in Table 3D.5-5 and are defined in the equipment qualification data package (see Section 1.8 of Attachment A) based on considerations and assumptions described in the following subsections.

3D.5.5.1 High-Energy Line Break Accidents Inside Containment

3D.5.5.1.1 Radiation Environment – Loss of Coolant Accident

The radiation dose rates and integrated doses following a design basis loss-of-coolant accident (LOCA) are determined based on the criteria and guidance provided in NUREG 1465, "Accident Source Terms for Light-Water Nuclear Power Plants – Final Report" (Reference 8) and Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 9).

The radiation exposure inside the containment is conservatively estimated by considering the dose in the middle of the AP1000 containment. Radioactive sources are assumed to be uniformly distributed throughout the containment atmosphere, and plate out of non-gaseous activity on containment surfaces is considered. No credit is taken for the shielding provided by internal structures and equipment.

Sources are based on the emergency safeguards system core thermal power rating and the following analytical assumptions:

- Power Level (including 2-percent power uncertainty) .... 3,468 MWt
- Fraction of total core inventory released to the containment atmosphere:

  Noble Gases (Xe, Kr) 1.0

  Halogens (I, Br) 0.40

  Alkali Metals (Cs, Rb) 0.30

  Tellurium Group (Te, Sb, Se) 0.05

  Barium, Strontium (Ba, Sr) 0.02

  Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) 0.0025

  Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) 0.0002

  Cerium Group (Ce, Pu, Np) 0.0005

The radionuclide groups and elemental release fractions listed above are consistent with the accident source term information presented in NUREG-1465 and Regulatory Guide 1.183.

The timing of the releases are based on NUREG-1465 assumptions. The release scenario assumed in the calculations is described below.

An initial release of activity from the gaps of a number of failed fuel rods at 10 minutes into the accident is considered. The release of 3 percent of the core inventory of the volatile species (defined as noble gases, halogens, and alkali metals) is assumed. An additional release period occurs over

the next 30 minutes, that is, from 10 to 40 minutes into the accident. At this point, 5 percent of the total core inventory of volatile species has been considered to be released.

Over the next 1.3 hours, releases associated with an early in-vessel release period are assumed to occur, that is, from 40 minutes to 1.97 hours into the accident. This source term is a time-varying release in which the release rate is assumed to be constant during the duration time. Additional releases during the early in-vessel release period include 95 percent of the noble gases, 35 percent of the halogens, and 25 percent of the alkali metals, as well as the fractions of the tellurium group, barium and strontium, noble metals, lanthanides, and cerium group as listed above.

There is no additional release of activity to the containment atmosphere after the in-vessel release phase. Activity removal by natural mechanisms as described in Chapter 15 Subsection 15.6.5.3.2 and Appendix B are considered only during the first 24 hours following the accident.

The above source terms are consistent with the guidance provided by the NRC in Regulatory Guide 1.183 for design basis accident (DBA) loss-of-coolant accident (LOCA) evaluations.

Based on these assumptions the instantaneous and integrated gamma and beta doses for the containment atmosphere following a loss of coolant accident are shown in Figures 3D.5-2 and 3D.5-3, respectively.

The total integrated dose of radiation employed for testing is a combination of normal and design basis event dose, as applicable. It is defined to equal or exceed the maximum radiation dose contained in the specification (Attachment A, Subsection 1.8.4.). A margin of 10 percent is included in defining the total integrated dose for testing. Normal operating and design basis event gamma doses are simulated using a cobalt-60 source. The test dose is applied at a rate approximate to the initial phase of the design basis event dose rate shown in Figure 3D.5-2 as modified by shielding effects (typically 0.2 to 0.25 Mr/hr).

Where exposed organic material is evaluated by test for the effect of (accident) beta radiation, a beta source is employed. Or a cobalt-60 or spent fuel source is used to impart the same dose using gamma radiation. When doing beta equivalent testing, the total integrated dose using gamma is conservatively equal to the beta total integrated dose, or the resulting bremsstrahlung is calculated and the test item is exposed to an equivalent gamma dose.

Radiation conditions for loss of coolant accident envelop other scenarios, such as rod ejection.

3D.5.5.1.2 Radiation Environment – Steam Line Break Accident

Sources associated with a steam line break accident are based on the release of reactor coolant system activity, assuming operation with the design basis fuel defect level of 0.25 percent. It is further assumed that an “event-initiated” iodine activity spike occurs, which increases the reactor coolant activity during the accident based on a rate of increase that is 500 times the normal activity appearance rate in the reactor coolant.

The activity inventory is instantaneously released into the containment atmosphere. The dose is conservatively estimated by considering the dose rate in the middle of the containment, with no credit for the shielding provided by the internal structures, components, and equipment. The instantaneous and integrated gamma and beta doses for the containment atmosphere following a steam line break are shown in Figures 3D.5-4 and 3D.5-5, respectively.

3D.5.5.1.3 Radiation Environment – Feedline Break

For convenience and simplicity, it is conservatively assumed that the radiation doses resulting from a feedline break are equal to the values specified in Figures 3D.5-4 and 3D.5-5 for a steam line break.

3D.5.5.1.4 Total Integrated Dose Specification

The applicable accident doses specified in equipment qualification data package Subsection 1.7.4 of Attachment A, have been derived based upon the time required to perform the specified safety function in the accident environment (Attachment A, Subsection 1.6.1) and the dose calculations described previously, subject to the following modifications:

- For equipment only required to provide trip or activation functions after accidents involving no release of radioactive material for at least one hour, the radiation dose is based on the normal dose rates (Table 3D.5-2).

3D.5.5.1.5 Temperature/Pressure Environments

The design basis events addressed are the loss of coolant accident, steam line break and feedwater line break. The WGOTHIC code is utilized to calculate the temperature and pressure conditions resulting from these breaks. To retain the option of qualifying equipment for each of these high-energy line break conditions, as applicable, separate environmental containment envelopes are specified for the higher irradiation/lower saturated temperature conditions of the loss of coolant accident as against the lower irradiation/short-term superheated temperature conditions associated with the steam line break. To limit the number of basic envelopes, this latter envelope is conservatively employed to define the containment environmental envelope following a feedline break.

Additionally, to facilitate AP1000 generic qualification and testing, the environmental envelopes have been combined to a single high-energy line break profile depicted in Figure 3D.5-8. This combined profile encompasses all locations inside containment on the basis of the containment analyses for the AP1000 design. The profile is used to qualify equipment for any application or location for the AP1000 consistent with the NRC requirements in 10 CFR 50.49 and IEEE 308, 323, 603, and 627 when margin is added and via conformance with IEEE 323 guidelines.

Qualification tests to high-energy line break conditions are designed to address the applicable specified environment(s) with a margin of 15°F and 10 psi. Separate envelopes with margin are employed, or a combined loss of coolant accident/steam line break/feedwater line break envelope (Figures 3D.5-8 and 3D.5-9) may be employed for in-containment equipment qualification tests. Figures 3D.5-8 and 3D.5-9 do not include margin from IEEE 323-1974, which will be incorporated in the environmental qualification programs. The simulated post-design basis event aging time-temperature profile (Figures 3D.5-8 and 3D.5-9 from 24 hours to test conclusion) is defined consistent with the smallest value of activation energy applicable to the thermal aging sensitive components composing the test equipment or by a demonstrably conservative activation energy, as described in Attachment D.

3D.5.5.1.6 Chemical Environment

The high-energy line break test will include chemical injection during the first 24 hours of the test, to simulate the reactor coolant system fluid. Initial pH is from 4 to 4.5, with the solution consisting primarily of boric acid.

Since there is no caustic containment spray in the AP1000, subsequent adjustments in pH may not be necessary for all tests. Sump solution chemistry is adjusted by release of alkaline chemistry,

which will rise to 7.0 to 9.5 within a few hours of containment flooding. These conditions are simulated for submerged equipment.

Margin in low pH value is not included, but is addressed by the continued injection through the first 24 hours. Margin in alkaline pH, where adjustment is necessary, is incorporated by a 10 percent increase in alkalinity.

3D.5.5.1.7 Submergence

Performance of equipment in a submerged condition is verified by a test that replicates the actual conditions with appropriate margin.

3D.5.5.2 High-Energy Line Break Accidents Outside Containment

For the majority of equipment located outside containment, the normal operating environment remains unchanged by a high-energy line break accident. As a consequence, qualification for such events is covered by qualification for normal conditions.

A limited amount of equipment located outside containment, near high-energy lines, could be subject to local hostile environmental conditions because of a high-energy line break outside containment. In this case, the equipment is qualified for the conditions resulting from events affecting its location and for which it is required to operate. Figure 3D.5-9 shows the design conditions for equipment that is required to perform throughout postulated events. Figure 3D.5-9 does not include margin from IEEE 323-1974, which will be incorporated in the environmental qualification programs. The maximum pressure for any event outside containment is 6 psig.

3D.6 Qualification Methods

The recognized methods available for qualifying safety-related electrical equipment are established in IEEE 323. These are type testing, analysis, on-going qualification, or a combination of these methods. The choice of qualification method for a particular item of equipment is based upon many factors. These factors include practicability, size and complexity of equipment, economics, and availability of previous qualification to earlier standards.

The qualification method employed for each equipment type included under the AP1000 equipment qualification program is identified in the individual equipment qualification data packages whether by test (Attachment A, Section 3.0), analysis (Attachment A, Section 4.0), or by a combination of these methods. The AP1000 equipment qualification program may employ on-going qualification through the use of maintenance and surveillance. Guidance for such an approach is not included in this appendix.

3D.6.1 Type Test

The preferred method of environmental and seismic qualification of safety-related electrical and electromechanical equipment for the AP1000 equipment qualification program is type testing according to the guidelines and requirements of IEEE 323-1974 and 344-1987. Development of type test requirements are discussed in Section 3D.5. Documentation requirements and test plan development are addressed in Section 3D.7.

Additionally, qualification based on type tests performed according to IEEE 323 and 344, but not specifically for the AP1000, may be used as a qualification basis. Subsection 3D.6.5 of this appendix discusses the combination of qualification methods as they apply to the AP1000 equipment qualification program. (See Subsection 3D.6.5.1.)

3D.6.2 Analysis

The AP1000 equipment qualification program uses analysis for seismic qualification of equipment if the primary requirement is the demonstration of structural integrity during a seismic event. For equipment that performs an active or dynamic function, seismic qualification by analysis may also be used. (See Section E.3 of Attachment E.) However, the similarity between a qualified test unit and an as-supplied unit must be demonstrated unless otherwise justified. Subsection 3.9.2.2 describes the qualification requirements for safety-related mechanical equipment where a fluid pressure boundary is involved. For those mechanical components that are not pressure boundaries, analysis is performed in compliance with the applicable industry design standard. Where age-sensitive materials, such as gaskets and packing, are used in the assembly of mechanical equipment, the aging of these materials is normally evaluated based on an item-by-item review of the aging characteristics of the material. (See Subsection 3D.6.2.3.)

Requirements for documentation of the analysis are further treated in Section 3D.7.

3D.6.2.1 Similarity

Similarities among manufacturer's models provides several options for extending qualification to equipment without the need for a complete qualification test program.

A model series, such as that for a solenoid valve design, consists of numerous models that are identical in materials of construction and manufacturing process, but have minor variance in size, functional mode, operating voltage, electrical termination type, and mechanical interface sizing. Such variances in most cases have no impact on or relevance to the capability of the various models to perform acceptably under environmental or seismic (or both) qualification test conditions. Furthermore, the design basis document may apply equally to each member of the model series. In such cases, all members of the model series can be qualified by reference to the same testing or analysis.

There may be sufficient similarities between different model series to justify the case for similarity. A documented comparison addressing differences in the design for each, or apparent physical differences between members of each model series, may be sufficient to preclude the testing of one model series based on the testing of the other.

Similarly, different models of a manufacturer's transmitters may be identical in some respects but different in others. The justification of similarity addresses the degree of similarity for critical characteristics. Differences that are not significant to qualification are also addressed for completeness. The mechanical and electrical functional modes and configurations must be the same. The materials of construction may be different, but must demonstrate equivalent performance. Other means of assuring accuracy may be necessary. When the devices are sufficiently similar in all attributes affecting qualification, qualification testing of one item can adequately cover another.

3D.6.2.2 Substitution

The objectives are to establish a degree of similarity and equivalence of performance for parts and materials that are different and, ultimately, to preclude the need for testing. For example, a gasket material is changed or a new type of capacitor is used because the original is no longer available, economical, or inadequate. Substitution of parts and materials is acceptable if comparison or analysis supports the conclusion that equipment performance is the same or better as a result. Consideration is given to characteristics of materials and the relative degree to which each is affected (or degraded) by the environmental parameters of qualification.

3D.6.2.3 Analysis of Safety-Related Mechanical Equipment

Environmental qualification of safety-related mechanical equipment is required to preclude common mode failures due to environmental effects of a design basis accident. Requirements are based on GDC 4 and 10 CFR 50, Appendix B. These criteria mandate that safety-related structures, systems, and components be designed to accommodate both normal and accident environmental effects.

3D.6.2.3.1 Equipment Identification

Safety-related mechanical equipment to be qualified is identified through the review of design basis documentation or the requirements of each safety-related fluid system. Only nonmetallic parts or subcomponents within the safety-related mechanical equipment are addressed for the effects of the postulated environments. The principal scope is typically valve "soft parts" that are critical to the valve safety-related function or pressure boundary integrity.

The types of components most frequently encountered in the mechanical equipment evaluations are discussed in Subsection 3D.6.2.3.3. Properties of materials that are assessed to provide confidence in safety-related function performance are also identified.

3D.6.2.3.2 Safety-Related Function

Safety-related functions and performance criteria are identified based on system and component classification. Structure, system, and component design basis documentation is reviewed to determine the specific safety functions. Components and subcomponents not involved in the equipment's safety-related function(s) are excluded from the qualification process if it is shown that their failures have no effect on the safety-related functions.

3D.6.2.3.3 Performance Criteria

Comprehensive performance criteria are established to satisfy the fundamental qualification requirements. The criterion for qualification is that the property of the nonmetallic material with regard to its application is not degraded during the specified qualified life to the point that the component is unable to perform its intended safety-related function. Properties for the component types listed in Table 3D.6-1 are discussed as examples.

Gaskets and O-Rings

The capability of gaskets and O-rings to keep their shapes determines their ability to maintain pressure boundaries. When an O-ring or gasket loses its dimensional memory, it does not exert the necessary force on the confining surfaces. This could result in leakage. Compression set and elongation are good indicators of the dimensional memory of a material. They also reflect the extent of thermal aging and radiation-induced cross-linking. A compression set of 50 percent is chosen as a conservative end-of-life criterion even though leakage is unlikely to occur until the component takes a compression set of greater than 75 percent. When compression set data is not available for a gasket or O-ring, elongation at break is the material property evaluated because like compression set, it is an indication of dimensional memory and cross-link.

Diaphragms

Diaphragms must remain flexible yet maintain their dimensional memory throughout the estimated mechanical cycles. Retention of elongation or tensile strength is evaluated for radiation and thermal aging.

Diaphragm Support Sheets

The diaphragm support sheet prevents puncture and tearing of the diaphragm. It is not considered critical to the operability of diaphragm valves. The best indication of radiation damage and thermal aging to diaphragm support materials is retention of elongation.

Lubricants

One of the primary functions of oils and greases is to maintain a thin film barrier between moving parts to reduce friction and wear. Irradiation reduces the capability of a lubricant to perform this function by decreasing viscosity in oils and increasing penetration in greases and finally converting lubricants to hard, brittle solids if exposure is severe.

Worm Gears

Worm gears must be capable of transmitting forces without excessive deformation. Flexural strength is the material property chosen to evaluate radiation and thermal aging resistance of worm gears.

3D.6.2.3.4 Identification of Service Conditions

Service conditions are identified for the normal and accident conditions. The general design of equipment permits exemption of environmental parameters such as pressure and humidity. Where critical parts are totally enclosed by metal and not directly exposed to potentially harsh environments, the effects of humidity and chemical spray are not addressed. The degradation of mechanical equipment due to thermal and radiation aging is typically more severe than the possible degradation due to other environments. Since most mechanical equipment interfaces with process fluid, the effect of the fluid on the environmental conditions (temperature, radiation, and chemical) is considered.

3D.6.2.3.5 Description of Potential Failure

Where applicable, potential failure modes are identified and assessed for the equipment. Assessment of equipment aging mechanisms is essential to determine if aging has a significant effect on operability. This assessment provides confidence that significant aging mechanisms are unlikely to contribute to common-mode failures adverse to the safety-related function of equipment.

3D.6.2.3.6 Qualification Procedure

The nonmetallic materials identified are evaluated to the normal and accident environmental parameters. The evaluation procedure includes the following steps:

- Identification of the environmental effect on the material properties
- Performance of a thermal aging analysis
- Determination of the environmental effects on the equipment safety-related function.

These are detailed in the equipment qualification data package of Attachment A, Section 4.Y.

3D.6.2.3.7 Performance Criteria

The nonmetallic subcomponents of the mechanical equipment:

a. are acceptable for the plant environment by exhibiting threshold radiation values above the postulated environmental condition, and

b. are acceptable for the plant environment by exhibiting a maximum service temperature above the maximum postulated environmental, and

c. does exhibit a service life sufficient to survive the accident duration, or

d. instead of a, b, and c, are acceptable for the plant environment by analysis that demonstrates that the safety-related function of the component is not compromised.

The mechanical equipment is considered qualified if subcomponents important to the safety function are acceptable.

Nonmetallic subcomponents not meeting the criteria must have a replacement interval specified to maintain the qualification of the affected equipment. The replacement interval is determined by analysis and documented.

3D.6.2.3.8 Equipment Qualification Maintenance Requirements

The maintenance requirements resulting from the activities described herein are identified. The qualification maintenance requirements are based on the following:

- Qualification evaluation results (for example, periodic replacement of age-susceptible parts before the end of their qualified lives)
- Equipment qualification-related maintenance activities derived from the qualification report(s)
- Vendor recommended equipment qualification maintenance. Vendor recommended maintenance is included if it is required in order to maintain qualification.

3D.6.2.3.9 Qualification Documentation

The qualification of the mechanical equipment to the postulated environments is documented in an auditable form. See Section 3D.7.

3D.6.3 Operating Experience

Qualification by experience is not employed in the AP1000 equipment qualification program as a method of qualification.

3D.6.4 On-Going Qualification

The AP1000 equipment qualification program may employ on-going qualification through special maintenance and surveillance activities. However, this method of qualification is not suitable as a sole means for qualifying equipment for design basis event conditions. On-going qualification, as a method, is used exclusively for safety-related equipment located in a mild environment area. Such use requires supplementary test, or analysis to address equipment operability and performance during and after a seismic design basis event.

Documentation requirements for qualification that includes on-going qualification as a method are developed to conform with NRC guidance provided in Regulatory Guide 1.33, Revision 2.

3D.6.5 Combinations of Methods

Qualification by a combination of the preceding methods may be used under the AP1000 equipment qualification program.

3D.6.5.1 Use of Existing Qualification Reports

Pre-existing qualification programs and documents are used only if the seismic test program satisfies the guidelines of IEEE 344-1987 and the environmental qualification program satisfies the guidelines of IEEE 323-1974.

Qualification test and analysis reports conforming to those IEEE Standards, but not specifically performed to the AP1000 equipment qualification program parameters, may be acceptable as qualification bases. In such cases, supplementary qualification efforts described in Subsections 3D.6.2, 3D.6.3, and 3D.6.4 of this appendix may be required to validate acceptability under the AP1000 equipment qualification program. Justifications are documented as analyses, and appear in equipment qualification data package, Section 4.0. (See Attachment A.)

3D.6.5.1.1 Aging

Past qualification tests may provide sufficient basis to preclude new aging simulation testing as part of the AP1000 program. Also, simulation of both electrical and mechanical operational cycling may be waived where existing data demonstrates equipment durability greatly in the excess of the estimated number of operating cycles for Class 1E service. Application of past qualification and other tests is considered in the development of test plans and analysis procedures. The bases and justification is provided in qualification documentation for cases where applicable aging parameters are omitted from the test sequence.

3D.6.5.1.2 Seismic

Seismic qualification generally relies on analyses and justification to verify the adequacy or applicability of generic testing to a particular installed configuration of similar equipment. Analytical methods and documentation guidelines of IEEE 344-1987, as supplemented by Regulatory Guide 1.100, Revision 2, address these needs. Attachment E of this appendix provides the AP1000 equipment qualification program requirements regarding seismic qualification.

3D.6.5.1.3 High-Energy Line Break Conditions

Typically, existing qualification tests address conditions of high-energy line break environments occurring inside containment. These are used where it is demonstrated that the qualification envelops the applicable requirements.

3D.7 Documentation

The AP1000 equipment qualification program documentation consists principally of three types of documents:

- "Methodology for Qualifying AP1000 Safety-Related Electrical and Mechanical Equipment" is the generic program "parent" document. It describes the methods and practices employed in the AP1000 equipment qualification program.

- Equipment qualification data packages are "daughter" documents to the methodology. Each is a summary of the qualification program for a specific equipment type (for example, a particular model or design series of a manufacturer, an as-provided system, or a family of equipment tested as a set). The equipment qualification data package defines the qualification program objectives, methods, applicable equipment performance specifications, and the qualification plan. It provides a summary of the results.

- Equipment Qualification Test Reports (EQTRs) are the reports that present specific methods used during the qualification process and the results of that process.

The equipment qualification data packages are developed separate from the parent document. Similarly, the equipment qualification test reports are developed separate from the equipment qualification data packages. Equipment qualification test reports used in the AP1000 equipment qualification program may include existing reports of testing or analysis that comply with the relevant aspects of this methodology. Information necessary to demonstrate the equipment's capability to perform its intended safety-related function(s) while exposed to normal, abnormal, accident, and post-accident environments is provided in or referenced by the equipment qualification data package. If maintenance, refurbishment, or replacement of the equipment is necessary to provide confidence in the equipment's capability to perform its safety function, this information is also included in the equipment qualification data package. Data, in raw form, cited in the equipment qualification data packages or equipment qualification test reports is available for audit for the life of the plant.

3D.7.1 Equipment Qualification Data Package

Attachment A contains sample of the equipment qualification data package format. Each equipment qualification data package consists of the following elements:

Section 1.0 – Specifications
Section 2.0 – Qualification Program
Section 3.0 – Qualification by Test
Section 4.0 – Qualification by Analysis
Section 5.0 – Qualification by Experience (Not Used)
Section 6.0 – Qualification Program Conclusions
Table 1 – Qualification Summary

The following paragraphs discuss the six sections in the equipment qualification data packages.

3D.7.2 Specifications

Section 1.0 of the equipment qualification data packages (Attachment A) contains the performance specification of the equipment. This specification establishes the necessary parameters for which qualification is demonstrated. The basic criterion for qualification is that the safety-related functional requirements defined in Section 1.0 are successfully demonstrated, with margin, under the specified environmental conditions.

The following sections define the bases on which the parameters contained in Section 1.0 are selected.

3D.7.2.1 Equipment Identification

Equipment is identified in Section 1.1 of Attachment A by manufacturer, model or model series, and reference to other documents describing or depicting its construction, configuration, and modifications that are uniquely necessary after manufacture to its application in the AP1000 plant design. Model series (for example, a limit switch design family) and other pertinent details on items making up the equipment type qualified are compiled as a table and referenced from this section.

3D.7.2.2 Installation Requirements

So that the qualification represents the in-plant condition, the method of installation, as specified in Section 1.2 of Attachment A, is in accordance with the supplier's installation instructions. Differences

unique to safety-related applications in the AP1000 design are included, with appropriate reference to drawings, technical manual supplements, or mandatory modification packages.

3D.7.2.3 Electrical Requirements

The pertinent electrical requirements are specified (for example, voltage, frequency, load) in this section. Also included is any variation in the defined parameters for which the equipment is to perform its specified functions (Section 1.3 of Attachment A).

3D.7.2.4 Auxiliary Devices

Sometimes the equipment qualified relies upon the operation of auxiliary devices in order to perform the specified safety-related functions. These devices are identified in Section 1.4 of Attachment A. Auxiliary devices include items such as electrical conductor seal assemblies that, in service, become part of the qualified equipment's pressure boundary. The applicable equipment qualification data package for the auxiliary device(s) is specified, if known.

3D.7.2.5 Preventive Maintenance

Preventive maintenance (Section 1.5 of Attachment A) to be performed includes maintenance or periodic activities assumed as part of the qualification program or necessary to support qualification. Only those activities that are required in order to support qualification or the qualified life are specified. The manufacturer's recommended maintenance activities are considered to determine that there is no adverse impact to qualification or the maintenance of qualified life. Likewise, manufacturer's recommendations for maintenance or surveillance activities necessary to support operability are identified, or reference is made to the appropriate technical manual or supplements.

"None" means that maintenance is not essential to qualification or the qualified life of the equipment. However, this should not preclude development of a preventive maintenance program designed to enhance equipment performance and to identify unanticipated equipment degradation as long as such a program does not compromise the qualification status of the equipment. Surveillance activities may also be considered to support a basis for and a possible extension of the qualified life.

3D.7.2.6 Performance Requirements

Section 1.7 of Attachment A contains a tabulation of performance requirements for each safety-related function for which the equipment is qualified. Several such sections or tables may be necessary when the equipment is qualified for applications where the performance requirements vary. Performance requirements are stated regarding the normal and abnormal environmental conditions applicable at the location where the equipment is installed. Similarly, each design basis event and the subsequent post-event period is included in the table.

Margin is not included in the performance requirements except by conservatism in their determination.

3D.7.2.7 Environmental Conditions

Within each set of performance requirements, a set of environmental parameters is specified in section 1.8 of Attachment A, also in tabular form. Parameters are based on the equipment location and function and include those addressed in other sections of this appendix.

Margin is not included in the environmental parameters except by conservatism in their determination. The objective is to provide the baseline reference onto which margin is added.

3D.7.3 Qualification Program

An overview of the qualification program and its objective is presented in narrative form in Section 2.0. Attachment A includes a table to be completed as a graphic reference. As it is assumed that tests, analyses, or some combination of the two are the principal methods of qualification, columns are included for each. Other methods, when used, are summarized in brief notes appended to the table.

References to reports of testing, analysis, or other information considered in support of the qualification program are compiled in Section 2.2 of Attachment A. This includes any technical manuals, drawings, and supporting material cited or referenced by text throughout the equipment qualification data package.

3D.7.4 Qualification by Test

Qualification by test is selected as the primary method of qualification for complex equipment not readily amenable to analysis or for equipment required to perform a safety-related function in a high-energy line break environment. The proposed test plan is identified in Section 3.0 of Attachment A. Where supportive analysis is claimed as an integral part of the qualification program, cross reference is provided to Attachment A, Section 4.0 for those aspects of the qualification not covered by the test plan. The following sections establish the basis on which the information specified in Section 3.0 is selected.

3D.7.4.1 Specimen Description

The equipment qualified is identified, including the baseline design document number/reference, where applicable, the equipment type, manufacturer and model number, in Section 3.1 of Attachment A. When testing a model series (or equipment families), the representative items tested are clearly identified. The basis of their representation should be included.

Section 3.1 is primarily intended to identify test specimens used in a test supporting the qualification program. But it also discusses the specimens considered for other methods used in the qualification program.

3D.7.4.2 Number Tested

The test program is based upon selectively testing a representative number of components according to type, size, or other appropriate classification, on a prototype basis. The number of items of equipment representative of the equipment type that are tested is defined in Section 3.2 of Attachment A.

3D.7.4.3 Mounting

The method of mounting the equipment for the test is identified in Section 3.3 of Attachment A. The in-plant installation requirements, as specified by the supplier under Section 1.2 of Attachment A, are fully represented.

3D.7.4.4 Connections

The equipment connections necessary to demonstrate safety-related functional operability during testing are identified in Section 3.4 of Attachment A. This includes items that are part of the installed configuration, but are not part of the test apparatus.

Particularly important are items that are included by "practice of good workmanship," such as pipe thread sealant. Another example is the use of electrical connection sealing materials. Where these items are included in the testing, they become factors in the performance of the equipment, especially under aggressive or adverse environmental conditions. Their thermal degradation and sensitivity to irradiation and chemistry environments are considered in the qualification program, both for impact to equipment performance under harsh conditions and for their contribution to equipment qualified life.

3D.7.4.5 Test Sequence

The preferred test sequence specified in Attachment A, Section 3.5 is the one recommended by IEEE 323-1974. The qualification test sequence used is specified in Section 3.6 of Attachment A. Justification for departures or additions to the preferred test sequence are included. Also, any portion of the test sequence that is supplemented by analysis or other methods is identified for completeness.

3D.7.4.6 Simulated Service Conditions

The service conditions simulated by the test plan are identified in Attachment A, Section 3.7. In general, the parameters employed are selected to be equal to (normal and abnormal) or have margin (accident and post-accident) with respect to the specified service conditions of Attachment A, Section 1.8. Criteria for margin is detailed in Subsection 3D.4.8.

3D.7.4.7 Measured Variables

The parameters measured during the specified test sequence in order to demonstrate qualification for the performance specification (Attachment A, Section 1.0) are individually listed in Attachment A, Section 3.8 of Attachment A. This section is formatted to include parameters relevant to the test environment and the electrical and mechanical characteristics of equipment operation. Other characteristics unique to a particular test or equipment type are included, when applicable.

3D.7.4.8 Type Test Summary

Section 3.9 of Attachment A provides a narrative summary of the qualification tests and results. The applicable test reports are provided as references in Attachment A, Section 2.2. Test data is available for audit throughout the operation of the plant.

Each test report referenced by the equipment qualification data package should contain information cited in the preceding section, as well as the following:

- The test facility, location, and a description of the test equipment used. Monitoring equipment should have current calibration traceable to the National Bureau of Standards.
- Test setup and specimen installation details.
- Description of the mounting conditions simulated during the test program and any difference between them and the mounting details shown on the equipment drawings, with qualification of any differences found.
- Description of limitations on the use and mounting of the qualified equipment found as a result of the qualification test program.
- Description of the test method and the justification that the method meets the specification test requirements.

- Description of operational settings used to demonstrate functional operability and any limitation imposed on them.

- Test records (for example, test response spectra, time history; accident transient parameters - temperature, pressure). This includes performance and operability test results, inspection results, and the monitored test and specimen and calibration records of instruments used.

- Record of compliance of test results with the seismic qualification criteria.

- Description of anomalies found during the test program, and their resolution(s).

Potential aging mechanisms resulting from significant in-service thermal, electrical, mechanical, radiation, and vibration sources are identified in Subsection 3.9.3 of Attachment A. When aging is addressed as part of the test sequence, the method employed for aging the equipment is indicated and is chosen to conservatively simulate the potential aging effects resulting from the operating cycles and environmental conditions specified in Attachment A, Section 1.0. The methods employed to address each of the potential aging mechanisms are discussed.

3D.7.5 Qualification by Analysis

Qualification by this methodology does not rely solely on analyses. Generally, analysis is permitted to support qualification testing or to establish that testing of other sufficiently similar equipment can be cited to establish or extend the qualification of equipment covered by the equipment qualification data package.

The sample format for Section 4.0 of Attachment A is formatted to conform with the recommendations of IEEE 323-1974. Each subsection addresses a particular analysis if more than one is performed to support qualification. Not all subsections identified in the sample format apply to any particular analysis. Documentation of analyses demonstrating or supporting seismic qualification conforms with the guidelines of Attachment E and the recommendations of IEEE 344-1987.

3D.7.6 Qualification by Experience

This method of qualification is not used.

3D.7.7 Qualification Program Conclusions

Section 6.0 of Attachment A summarizes the conclusions of the qualification program, including and addressing methods employed and conditions upon which qualification of the equipment is based. Details regarding each aspect of simulated aging are addressed distinctly, with conclusions as to the life-limiting aspects clearly stated.

Conclusions for each design basis event are summarized. Generally, these are combined as either design basis event seismic and design basis event environmental.

3D.7.8 Combined License Information

Not used.

3D.8 References

1. IEEE-323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

2. IEEE-344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

3. IEEE-627-1980, "IEEE Standard for Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations."

4. NUREG/CR-3156, "A Survey of the State-of-the-Art in Aging of Electronics with Application to Nuclear Plant Instrumentation."

5. EPRI NP-1558, Project 890-1, "A Review of Equipment Aging Theory and Technology."

6. NUREG/CR 2156, "Radiation Thermal Degradation of PE and PVC: Mechanism of Synergism and Dose Rate Effects," Clough and Gillen, June 1981.

7. NUREG/CR 2157, "Occurrence and Implication of Radiation Dose Rate Effects for Material Aging Studies," Clough and Gillen, June 1981.

8. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants – Final Report," L. Soffer, et al., February 1995.

9. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

Note: See Subsection 3D.4.1.1 for other IEEE references.

Table 3D.4-1
Typical Mild Environment Parameter Limits

Parameter	Limit	Notes
Temperature	≤ 120°F	
Pressure	Atmospheric	(Nominal)
Humidity	30 – 65% ≤ 95%	(Typical) (Abnormal)
Radiation	≤ 10^4 rads gamma ≤ 10^3 rads gamma	 (IC electronics and microprocessors)
Chemistry	None	
Submergence	None	

Table 3D.4-2
Equipment Post-Accident Operability Times

Equipment	Required Post-Accident Operability	
Equipment necessary to perform trip functions	5 minutes	(Envelops trip time requirements)
Equipment located outside containment, is accessible, and can be repaired, replaced, or recalibrated	2 weeks	
Equipment located inside containment that is inaccessible and is required for post-accident monitoring	4 months	(This number is based on an acceptable amount of time to be repaired, replaced, or recalibrated, or for an equivalent indication to be obtained.)
Equipment located inside containment, is inaccessible, or cannot be repaired, replaced, recalibrated or equivalent indication cannot be obtained	1 year	
Equipment in a location that will have a mild environment following an accident or equipment that does not provide information for a Type A, B, or C primary post-accident monitoring parameter	Various	(Specific as to function, maximum of 1 year)

Table 3D.4-3
AP1000 EQ Program Margin Requirements

Condition	Parameter	Required Margin	Notes
NORMAL:	Aging	+10%	+10% time margin, +10% radiation and/or selection of conservative test parameters. Comply with guidance of Subsection 3D.4.8.2
ABNORMAL:	Temperature/ Humidity		Margin is in "time" at abnormal test extremes.
	Pressure	None	Nominally atmospheric.
	Radiation	+10%	Include in aging doses, if applicable.
	Voltage & Frequency	+/- 10%	Simulated during temperature/humidity test.
ACCIDENT:	Transient Temperature and Pressure		Temperature (+15°F) and pressure (+10 psig peak) margins added to transient profile.
	Chemical effects	+10%	In alkalinity of adjusted sump pH. Not applicable outside containment.
	Radiation	+10%	Added to calculated total integrated dose.
	Submergence	Note 1	Generally, precluded by design.
	Seismic/ Vibration	+10%	Of acceleration at equipment mounting point for either SSE or line-mounted equipment vibration. (See Subsection 3D.4.8.4.)
	Post-accident Aging	+10%	In time demonstrated via Arrhenius time/temperature relationship calculation.

Note:
1. Margin in submergence conditions is achieved by increases in the post-accident time duration (+10%) and chemistry (+10% in alkalinity of adjusted sump pH).

Table 3D.5-1 (Sheet 1 of 3)
Normal Operating Environments

(Notes 1 and 2)

Location/Parameter	Normal Range	Notes
Zone 1 – Containment (Room numbers: 11000 through 11999)		
Temperature	50° - 120°F	
Pressure	-0.2 - +1.0 psig	
Humidity	0 - 100%	
Radiation	see Table 3D.5-2	
Chemistry	None	
Zone 2 - Auxiliary Building - Non-Radiological - I&C, DC Equipment, RCP Switchgear & Battery rooms, etc. (Room numbers: 12101, 12102, 12103, 12104, 12105, 12111, 12112, 12113, 12201, 12202, 12203, 12204, 12205, 12207, 12211, 12212, 12213, 12301, 12302, 12303, 12304, 12305, 12311, 12312, 12313, 12405, 12411, 12412, 12501, and 12505)		
Temperature	67 - 77°F (All rooms except 12405 and 12505) 50 - 85°F (Rooms 12405 and 12505)	
Pressure	Slightly positive to slightly negative	
Humidity	10 - 60%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	
Zone 3 - Auxiliary Building - Non-Radiological - Main Control Room (Room number: 12400, 12401)		
Temperature	67 - 78°F	
Pressure	Slightly positive	
Humidity	25 - 60%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	
Zone 4 - Auxiliary Building - Non-Radiological - Accessible (Room numbers: 12321, 12421, 12422, 12423)		
Temperature	50 - 105°F	
Pressure	Slightly positive	
Humidity	10 - 60%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	

Table 3D.5-1 (Sheet 2 of 3)
Normal Operating Environments

(Notes 1 and 2)

Location/Parameter	Normal Range	Notes
Zone 5 - Auxiliary Building - Non-Radiological - MSIV Compartments (Room numbers: 12404, 12406, 12504, 12506)		
Temperature	50 - 130°F	
Pressure	Atmospheric	
Humidity	10 - 100%	
Radiation	$<10^4$ rads gamma	
Chemistry	None	
Zone 6 - Auxiliary Building - Radiological - Inaccessible (Room numbers: 12154, 12158, 12162, 12163, 12166, 12167, 12171, 12172, 12254, 12255, 12256, 12258, 12262, 12264, 12265, 12354, 12362, 12363, 12365, 12371, 12372, 12373, 12374, 12454, 12462, 12463)		
Temperature	50 - 130°F	
Pressure	Slightly negative to atmospheric	
Humidity	10 - 100%	
Radiation	See Table 3D.5-2	
Chemistry	None	
Zone 7 - Auxiliary Building - Radiological - Accessible (Room numbers: 12151, 12152, 12153, 12155, 12156, 12161, 12169, 12241, 12242, 12244, 12251, 12252, 12261, 12268, 12271, 12272, 12273, 12274, 12275, 12341, 12351, 12352, 12361, 12451, 12452, 12461, 12553, 12554, 12555, 12561)		
Temperature	50 - 104°F	
Pressure	Atmospheric	
Humidity	10 - 100%	
Radiation	See Table 3D.5-2	
Chemistry	None	
Zone 8 - Turbine Building (Room numbers: 20300 through 20799)		
Temperature	50 - 105°F	
Pressure	Atmospheric	
Humidity	10 - 100%	
Radiation	$<10^3$ rads gamma	
Chemistry	None	

Table 3D.5-1 (Sheet 3 of 3)
Normal Operating Environments

(Notes 1 and 2)

Location/Parameter	Normal Range	Notes
Zone 9 - Auxiliary Building - PCS Valve Room (Room number: 12541, 12701)		
Temperature	50 - 120°F	
Pressure	Atmospheric	
Humidity	10 - 100%	
Radiation	See Table 3D.5-2	
Chemistry	None	
Zone 10 - Auxiliary Building - Non-Radiological - Valve/Piping Penetration Room with SG Blowdown (Room number: 12306)		
Temperature	50 - 105°F	
Pressure	Slightly positive	
Humidity	10 - 60%	
Radiation	<10^3 rads gamma	
Chemistry	None	
Zone 11 - Auxiliary Building - Radiological - Fuel Handling Area (Room numbers: 12562, 12563, 12564)		
Temperature	50 - 105°F	
Pressure	Slightly negative	
Humidity	10 - 100%	
Radiation	See Table 3D.5-2	
Chemistry	None	

Notes:
1. Room numbers - see Section 1.2, General Arrangement drawings.
2. Relative humidity is not controlled except in the main control room.

Table 3D.5-2
60-Year Normal Operating Doses

Location	Gamma Dose Rate (Rad air hour)	60-Year Gamma Dose (Rads air)
Inside Containment:		
RCS Pipe - Center	$1.9x10^3$	$1.0x10^9$
RCS Pipe - ID	$1.1x10^3$	$5.7x10^8$
RCS Pipe - OD (contact)	$7.8x10^1$	$4.1x10^7$
RCS Pipe - General Area[b]	$4.0x10^1$	$2.1x10^7$
Outside Loop/Compartment Wall	<0.1	$<5x10^4$
Outside CA01 Excluding Rooms 11104 and 11204	<0.45	$<2.4x10^5$
Adjacent to Reactor Vessel Wall	$\leq 3.6x10^4$	$\leq 1.9x10^{10}$[a]
Outside Containment:		
Penetration Area	--	$<2x10^7$
Pump Cubicles	--	$<2x10^7$
Radioactive Waste Area	--	$<2x10^7$
Radwaste Tank Cubicles	--	$<5x10^7$
Other General Areas Not Under Radiation Control	--	$<1x10^4$

Notes:
a. 60-year neutron fluence for E>1 MeV is $4.6x10^{18}$ n/cm^2
b. 12 inches from RCS pipe OD

Table 3D.5-3
Abnormal Operating Environments
Inside Containment

Conditions/Parameter	Abnormal Extreme	Duration	Notes
Group 1 (150°F) Abnormal Events			
Temperature	150°F	4 hours	Note 1
Pressure	Atmospheric		
Humidity	100%	4 hours	Note 1
Radiation	Same as normal		
Chemistry	None		
Submergence	None		
Group 2 (250°F) Abnormal Events			
Temperature	250°F	30 days	Note 1
Pressure	15 psig	30 days	Note 1
Humidity	100%	30 days	Note 1
Radiation			Note 2
Chemistry	None		
Submergence	None		

Notes:
1. Parameter value is not maximum for full duration.
2. Minor increase over normal radiation conditions expected.

Table 3D.5-4 (Sheet 1 of 2)
Abnormal Operating Environments
Outside Containment

Conditions/Parameter	Abnormal Extreme	Duration	Notes
Zone 2 – Loss of AC Power			
Temperature	Figure 3D.5-1 (Sheet 2)	7 days	Note 3
Pressure	Atmospheric		
Humidity	40 – 95%		Note 2
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 3 – Loss of HVAC			
Temperature	Figure 3D.5-1 (Sheet 1)	7 days	
Pressure	Atmospheric		Note 1
Humidity	60 – 95%		Note 2
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 4 – Loss of AC Power			
Temperature	120°F max	10x4 hrs	
Pressure	Atmospheric		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 5 – Loss of AC Power			
Temperature	150°F max	10x4 hrs	
Pressure	Atmospheric		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 6 – Loss of AC Power			
Temperature	140°F max	10x4 hrs	
Pressure	Atmospheric		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		

Table 3D.5-4 (Sheet 2 of 2)
Abnormal Operating Environments
Outside Containment

Conditions/Parameter	Abnormal Extreme	Duration	Notes
Zone 7 – Loss of AC Power			
Temperature	114°F max	10x4 hrs	
Pressure	Atmospheric		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Zones 8, 9, 10			
Temperature	Same as normal		
Pressure	Same as normal		
Humidity	Same as normal		
Radiation	Same as normal		
Chemistry/Submergence	None		
Zone 11 – Loss of AC Power (Fuel Handling Area)			
Temperature	212°F max	7 days	
Pressure	Atmospheric		Note 4
Humidity	100%		
Radiation	Same as normal		
Chemistry/Submergence	None		

Notes:

1. Main control room air pressure is maintained above a nominal value of atmospheric during accident conditions to prevent radioactive contaminant entry.
2. Figure 3D.5-1 Sheets 1 and 2 have two curves post-72 hours. The high curve represents the introduction of outside air that is high temperature, low humidity. The low curve represents the introduction of outside air that is low temperature, high humidity. The EQ Programs will include both of these extremes.
3. Test environments resulting from rooms with equipment supplied by 24- and 72-hour batteries are shown on Sheet 2 for the dc equipment rooms 12203 and 12207 and for the I&C rooms 12302 and 12304. The 24-hour battery is disconnected at 24 hours. The 72-hour battery is not disconnected. Environments resulting from rooms with equipment supplied by 24-hour batteries only, – that is, dc equipment rooms 12201 and 12205 and I&C rooms 12301 and 12305 – are enveloped by the environments shown on Sheet 2.
4. A relief panel is designed to open when the fuel handling area temperature exceeds 165°F.

Table 3D.5-5
Accident Environments

(See Table 3D.5-1 for environmental zones)

Zone 1 - Inside Containment	
Temperature and pressure	See Figure 3D.5-8.
Submergence as applicable up to elevation 110′-6″	
Radiation	See Figures 3D.5-2 through 3D.5-5.
Zones 2, 3, 4, 6, 7, 8, 9, 11	
(Same as abnormal – see Table 3D.5-4.)	
Zones 5 and 10 - Outside Containment	
MSIV Compartments	
Temperature	See Figure 3D.5-9.
Radiation	See Figures 3D.5-4 and 3D.5-5.

Table 3D.6-1
Mechanical Equipment Components Requiring Environmental Qualification

Component	Material Property
Gaskets	Compression set/elongation
O-rings	Compression set/elongation
Diaphragms	Elongation/tensile strength
Diaphragm support sheets	Tensile strength/elongation
Lubricant	Viscosity/penetration
Worm gear	Flexural strength

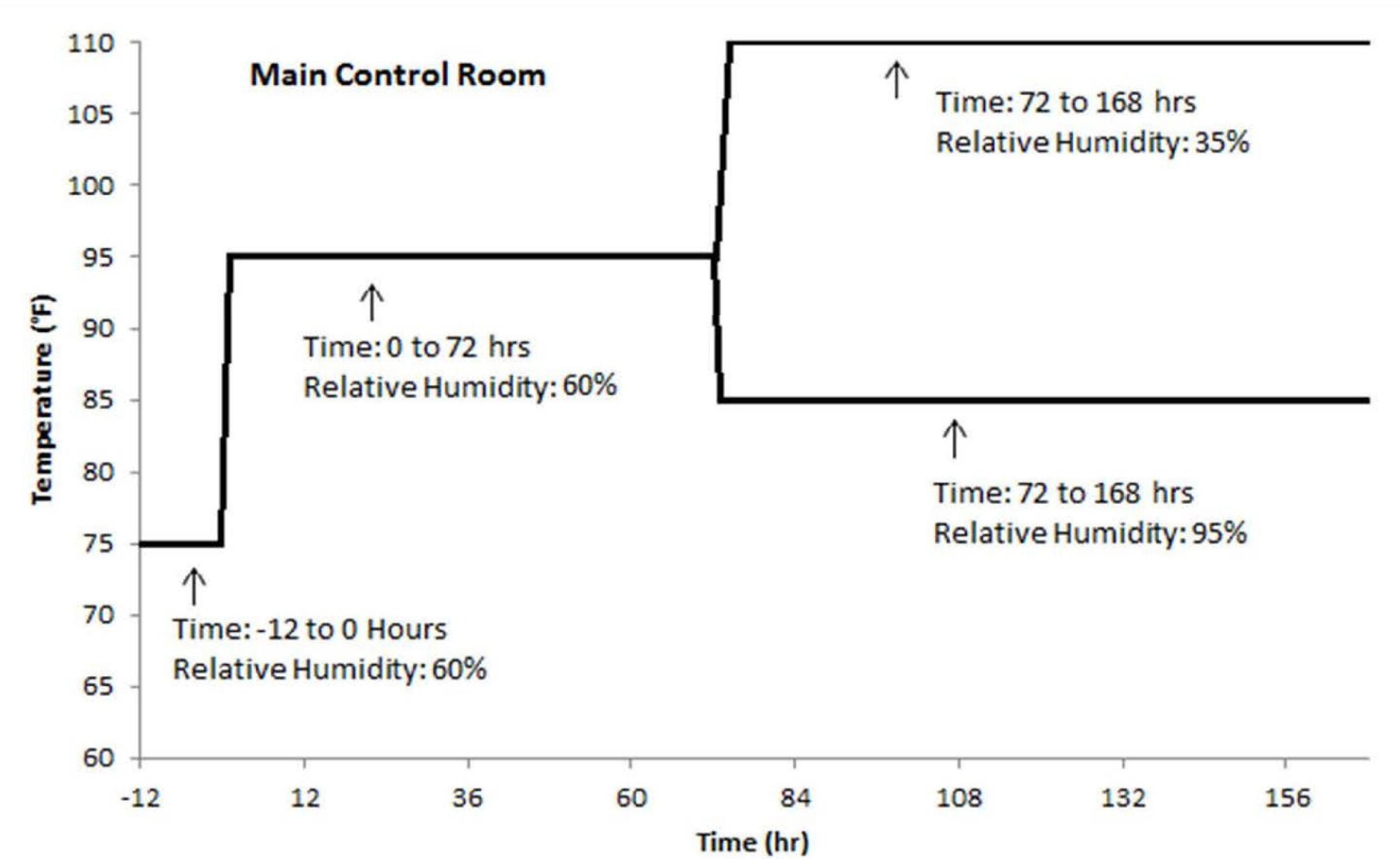


Figure 3D.5-1 (Sheet 1 of 3)
Typical Abnormal Environmental Test Profile: Main Control Room

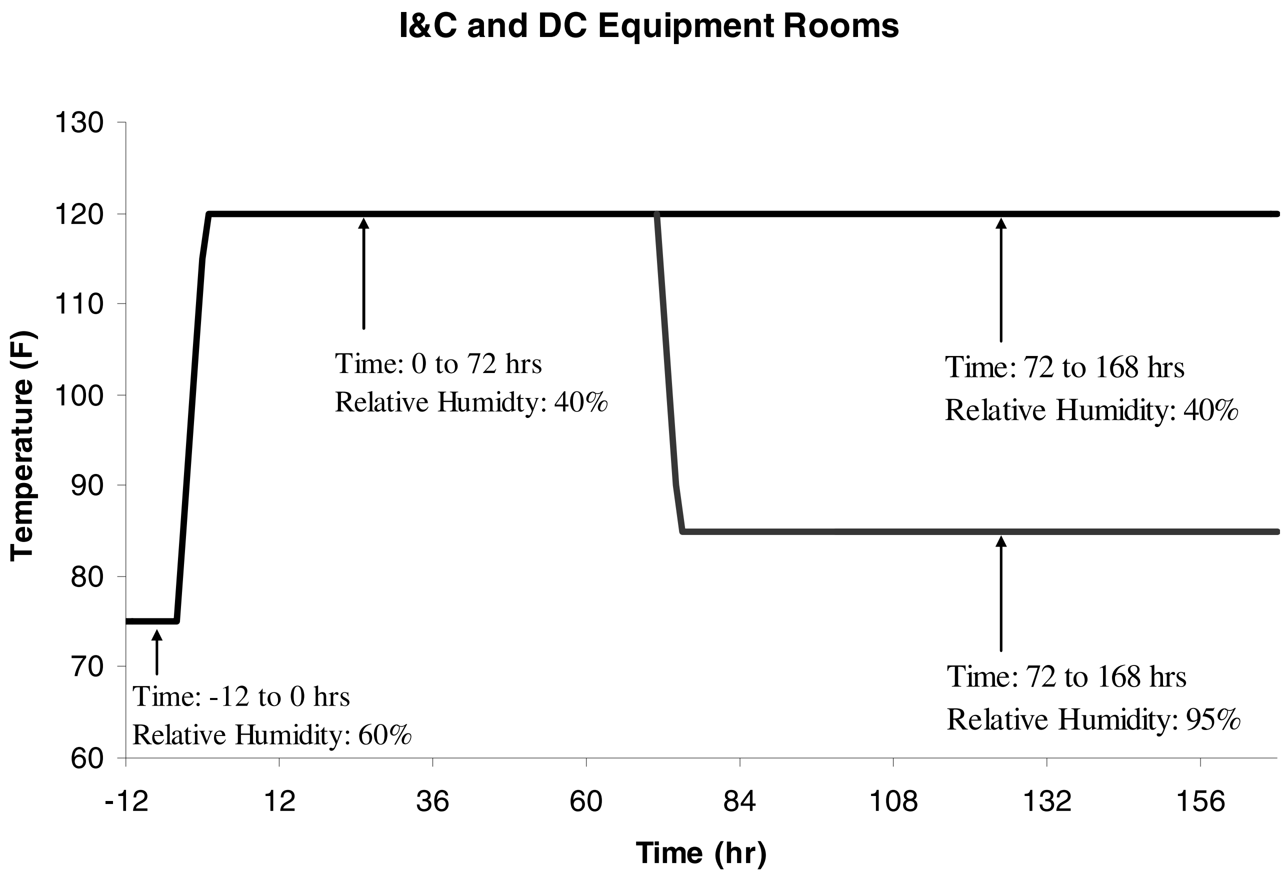


Figure 3D.5-1 (Sheet 2 of 3)
Typical Abnormal Environmental Test Profile: I&C and DC Equipment Rooms

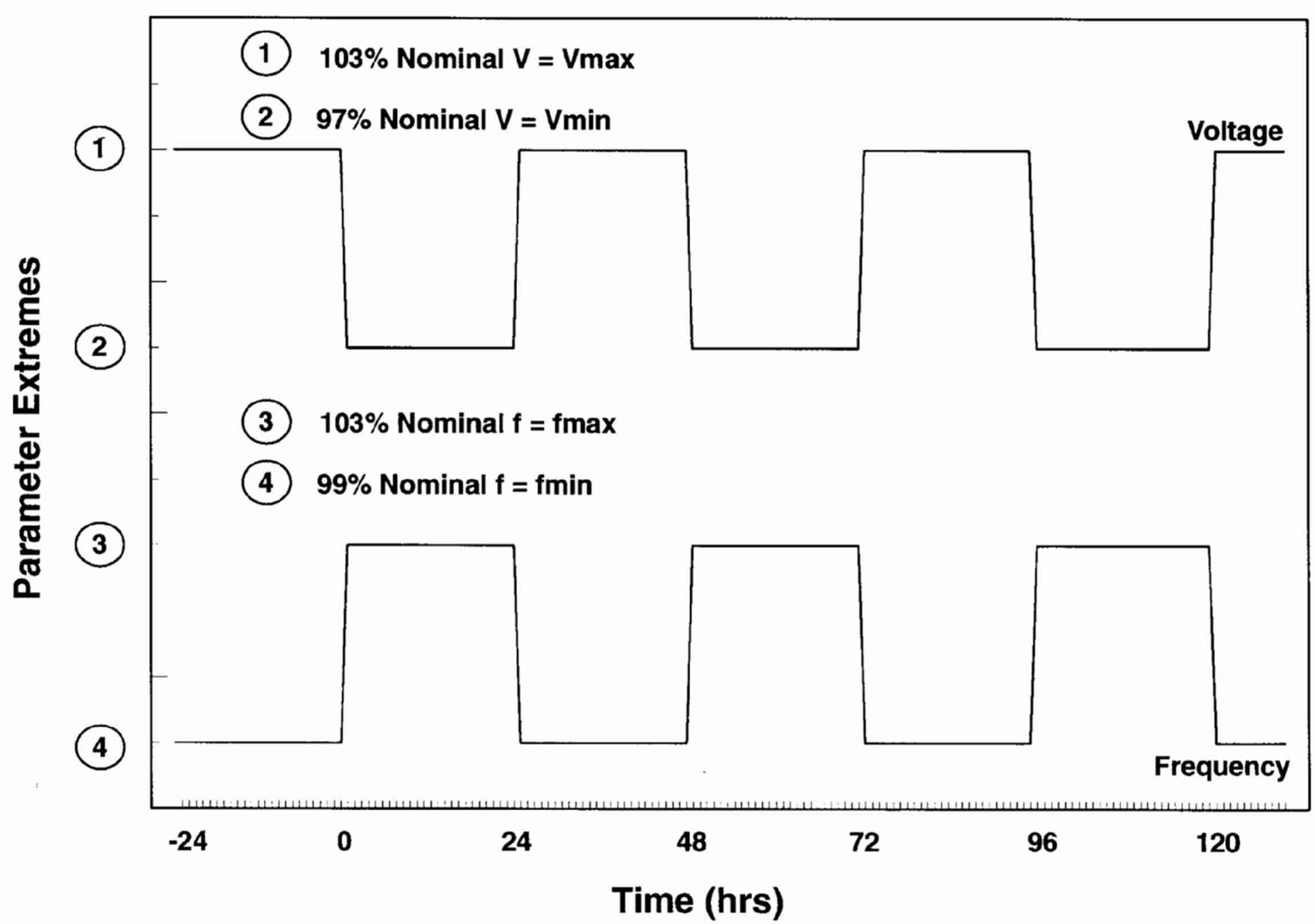


Figure 3D.5-1 (Sheet 3 of 3)
Typical Abnormal Environmental Test Profile: Voltage and Frequency Variations

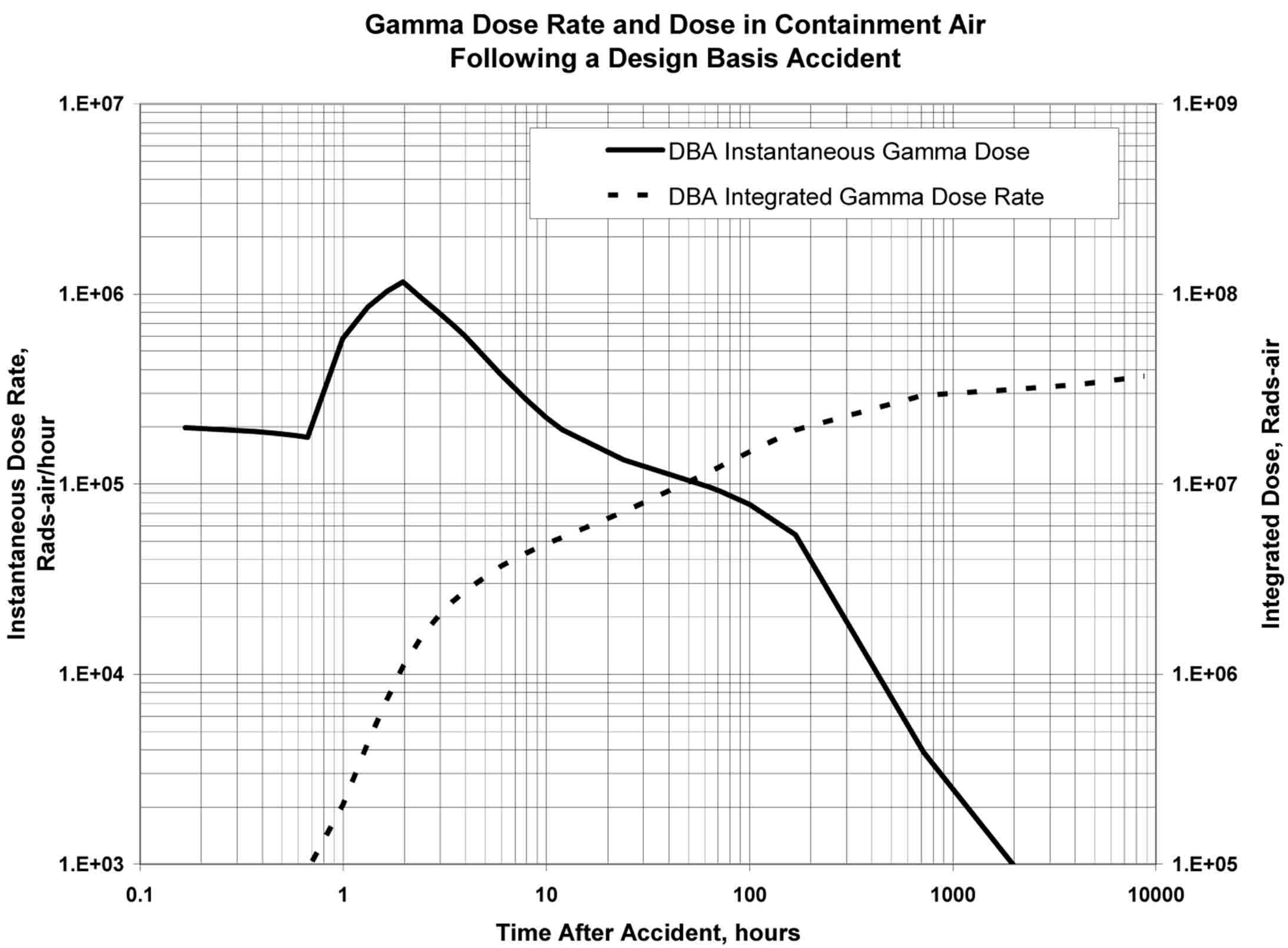


Figure 3D.5-2
Gamma Dose and Dose Rate Inside Containment After a LOCA

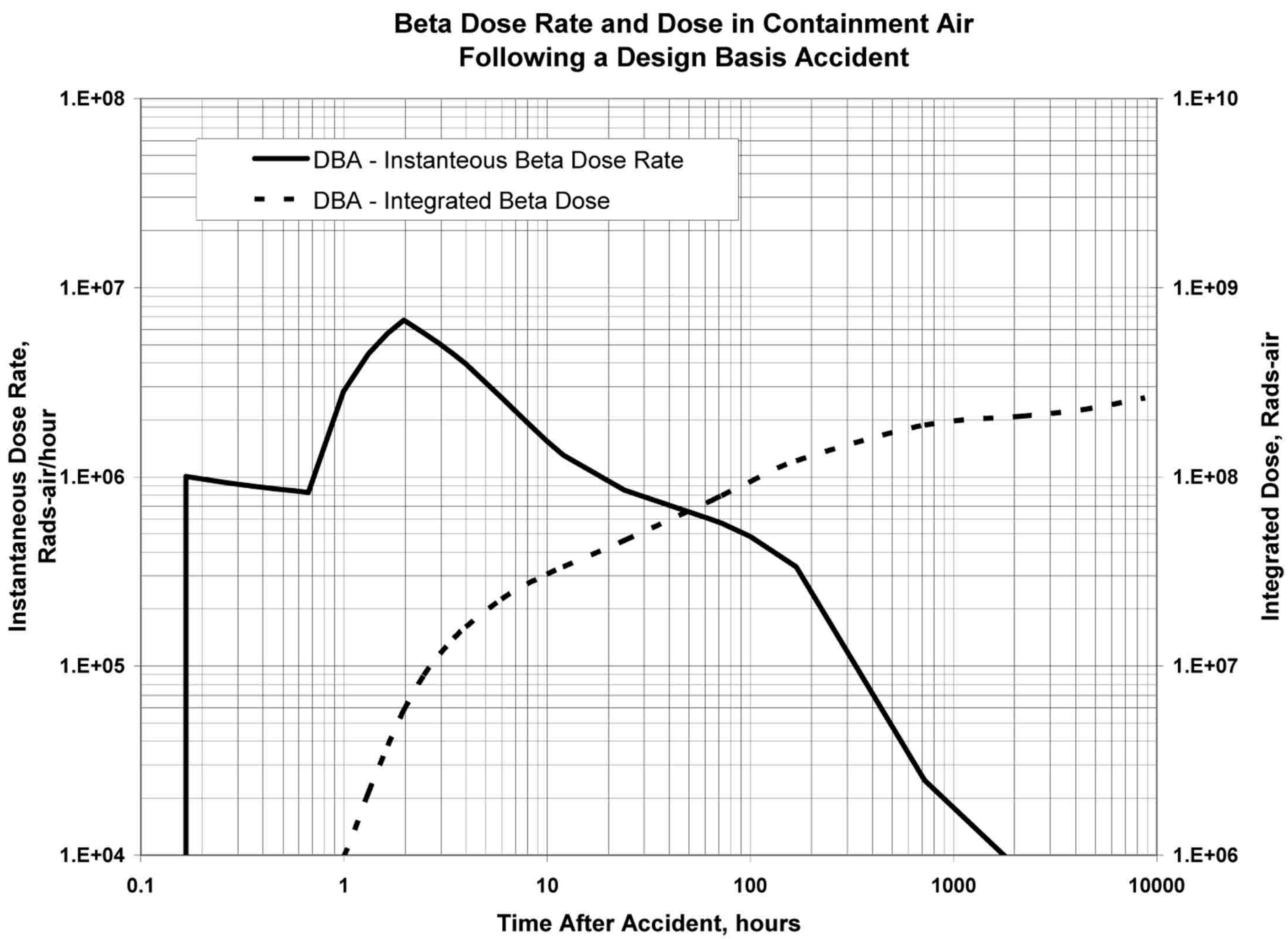


Figure 3D.5-3
Beta Dose and Dose Rate Inside Containment After a LOCA

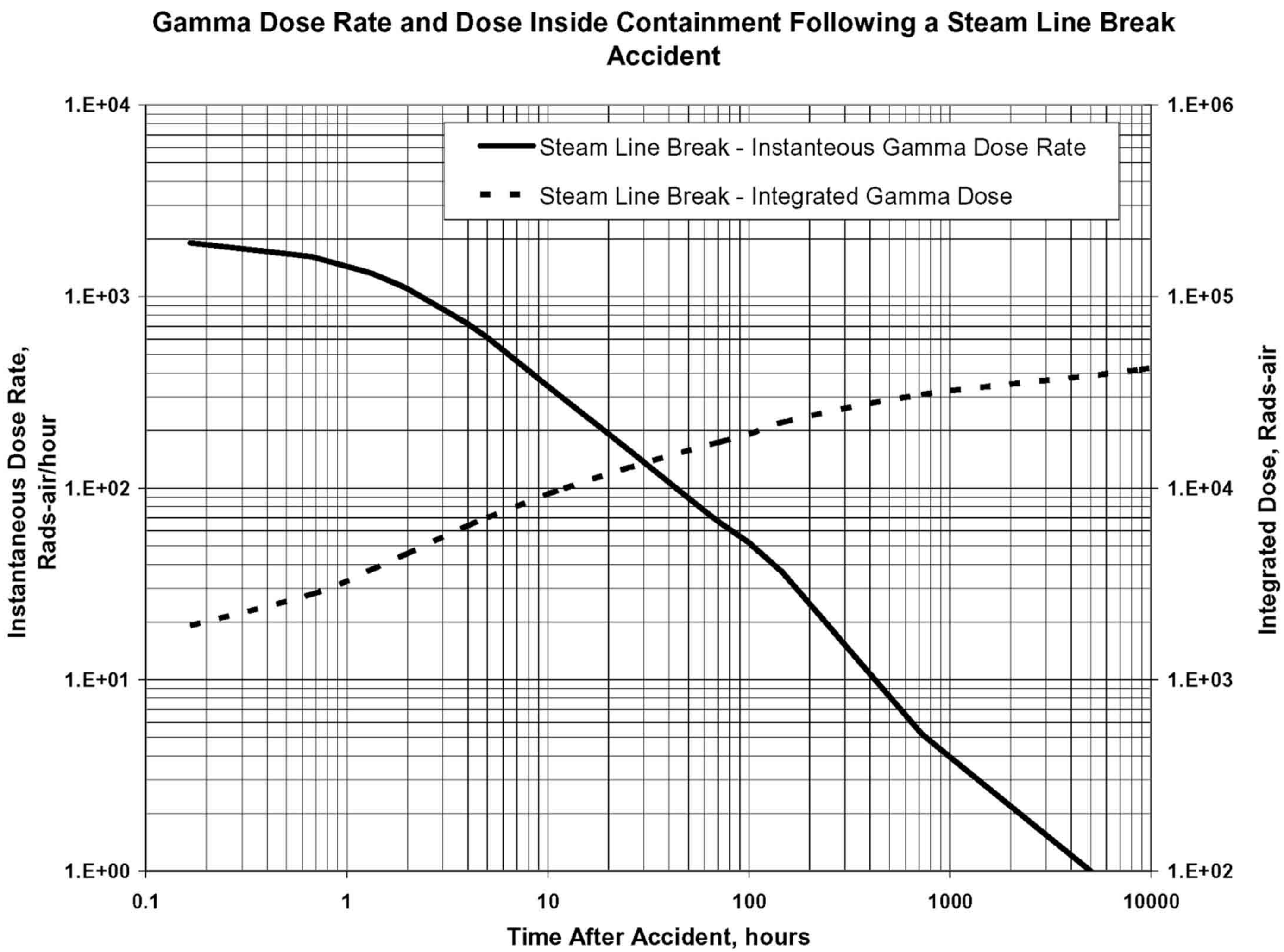


Figure 3D.5-4
Gamma Dose and Dose Rate Inside Containment After a Steam Line Break

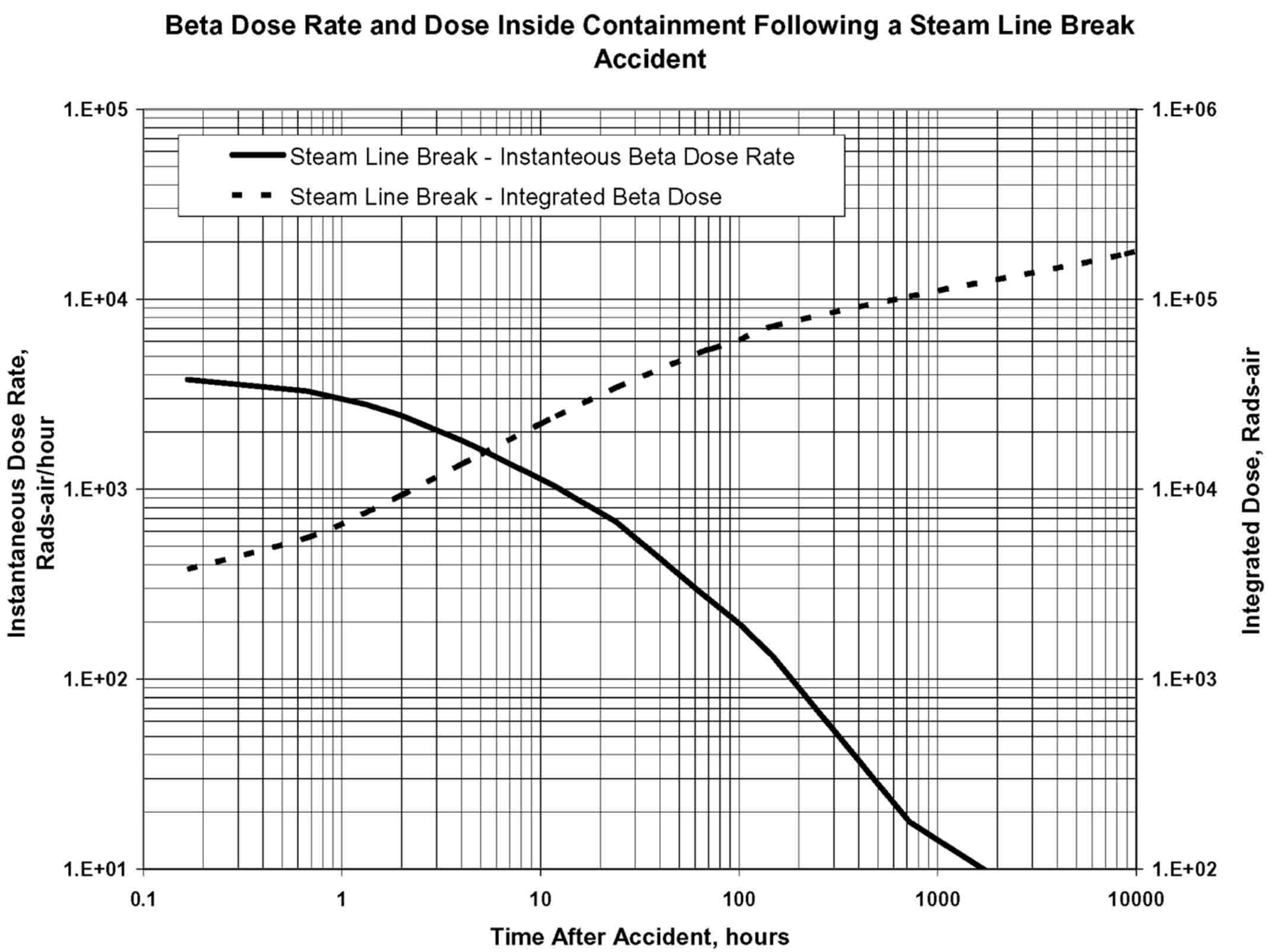


Figure 3D.5-5
Beta Dose and Dose Rate Inside Containment After a Steam Line Break

Figure 3D.5-6
Not Used.

Figure 3D.5-7
Not Used.

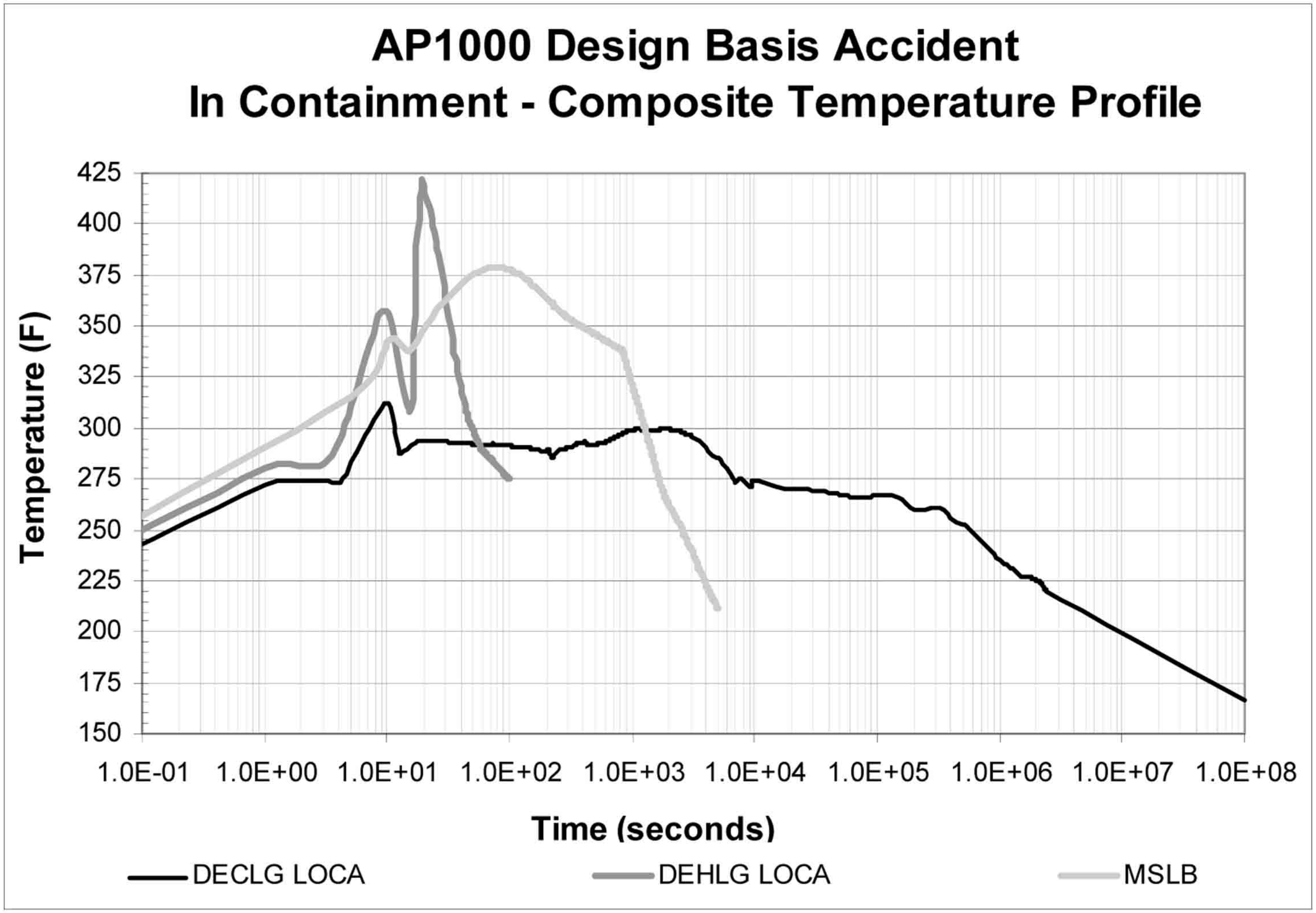


Figure 3D.5-8 (Sheet 1 of 2)
Typical Combined LOCA/SLB/FLB Inside Containment Temperature

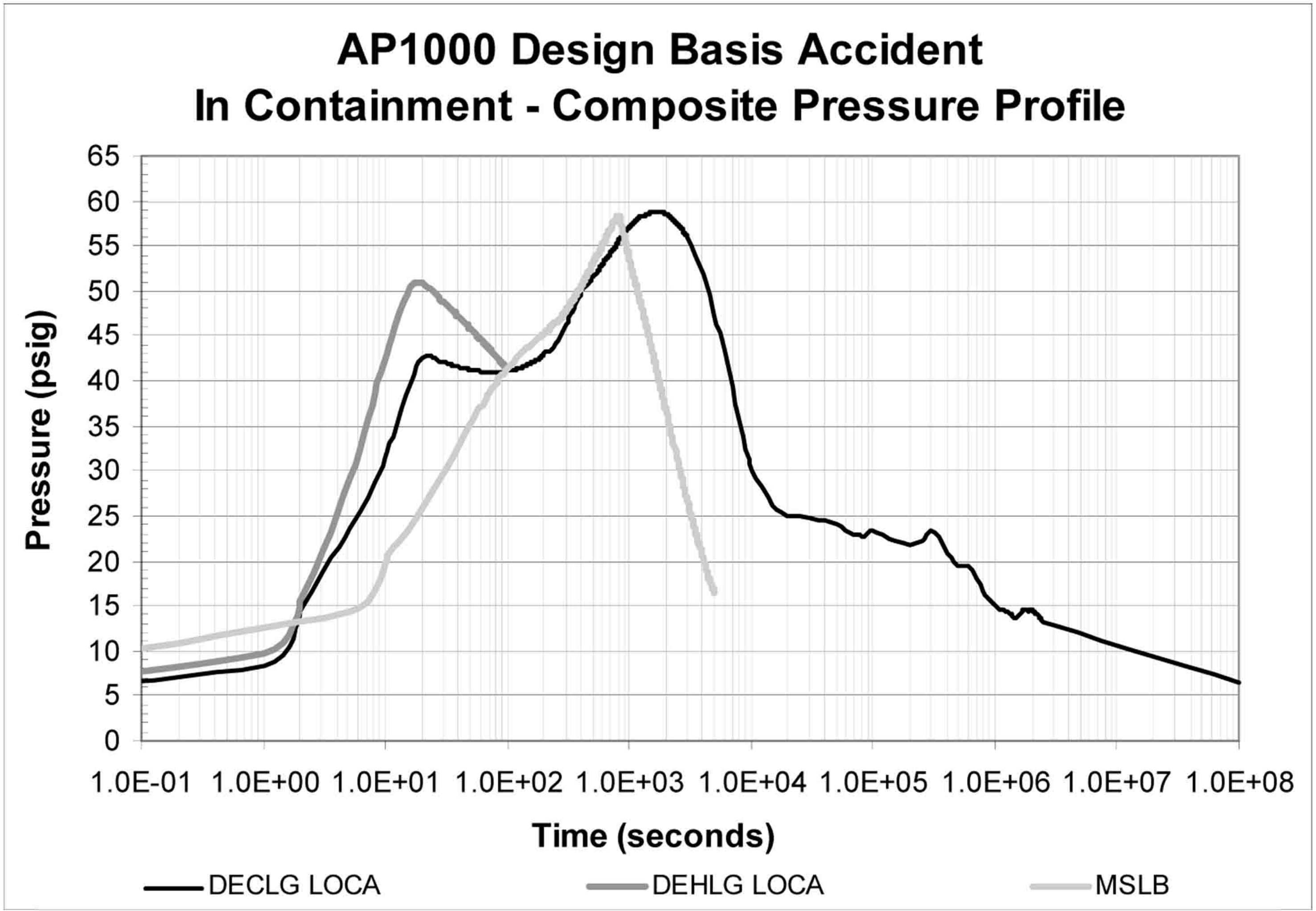
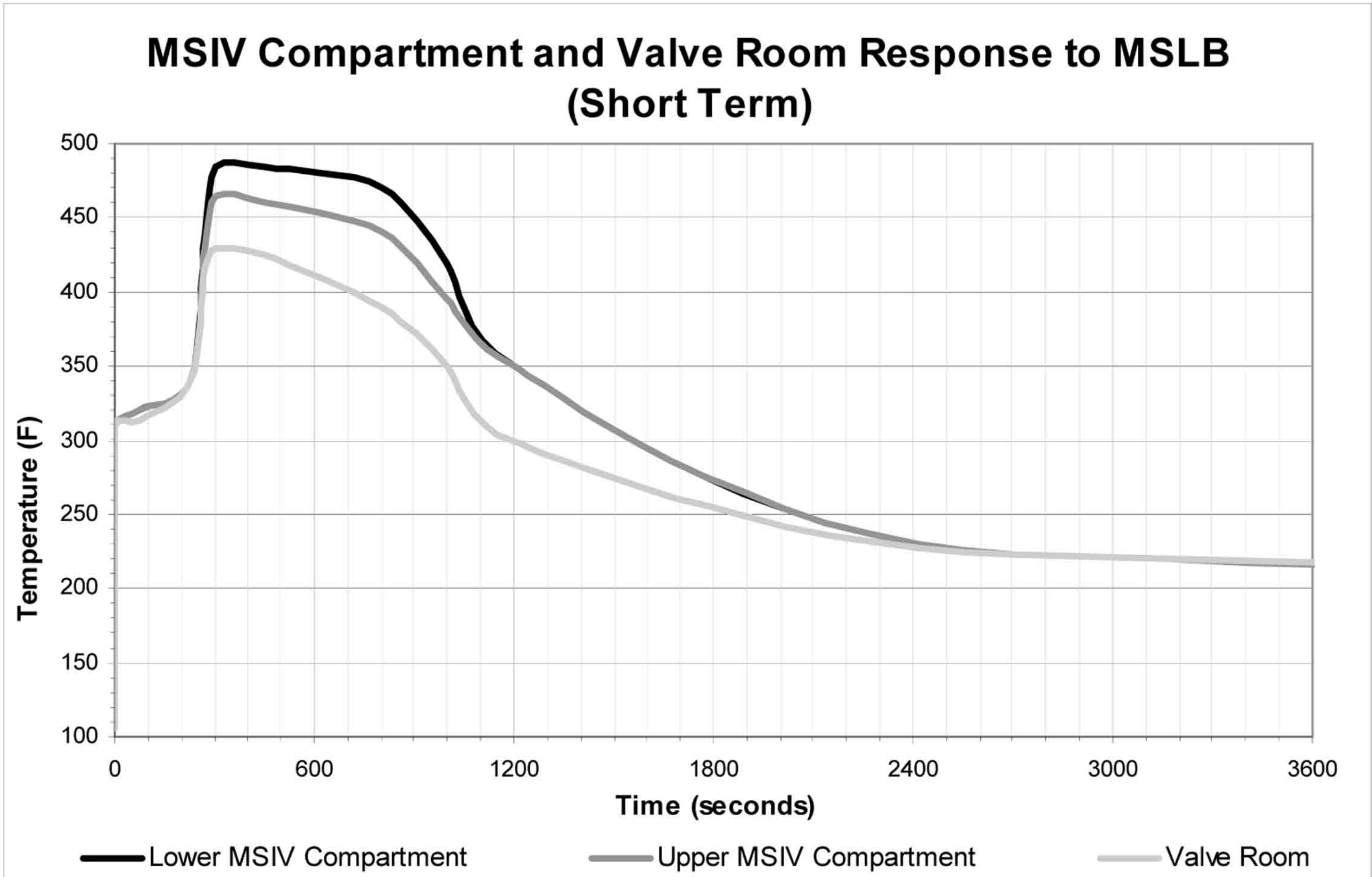


Figure 3D.5-8 (Sheet 2 of 2)
Typical Combined LOCA/SLB/FLB Inside Containment Pressure



Note: This figure represents bounding curves of varying break sizes and power levels.

Figure 3D.5-9 (Sheet 1 of 2)
MSIV Compartment Response to MSLB (Short Term)

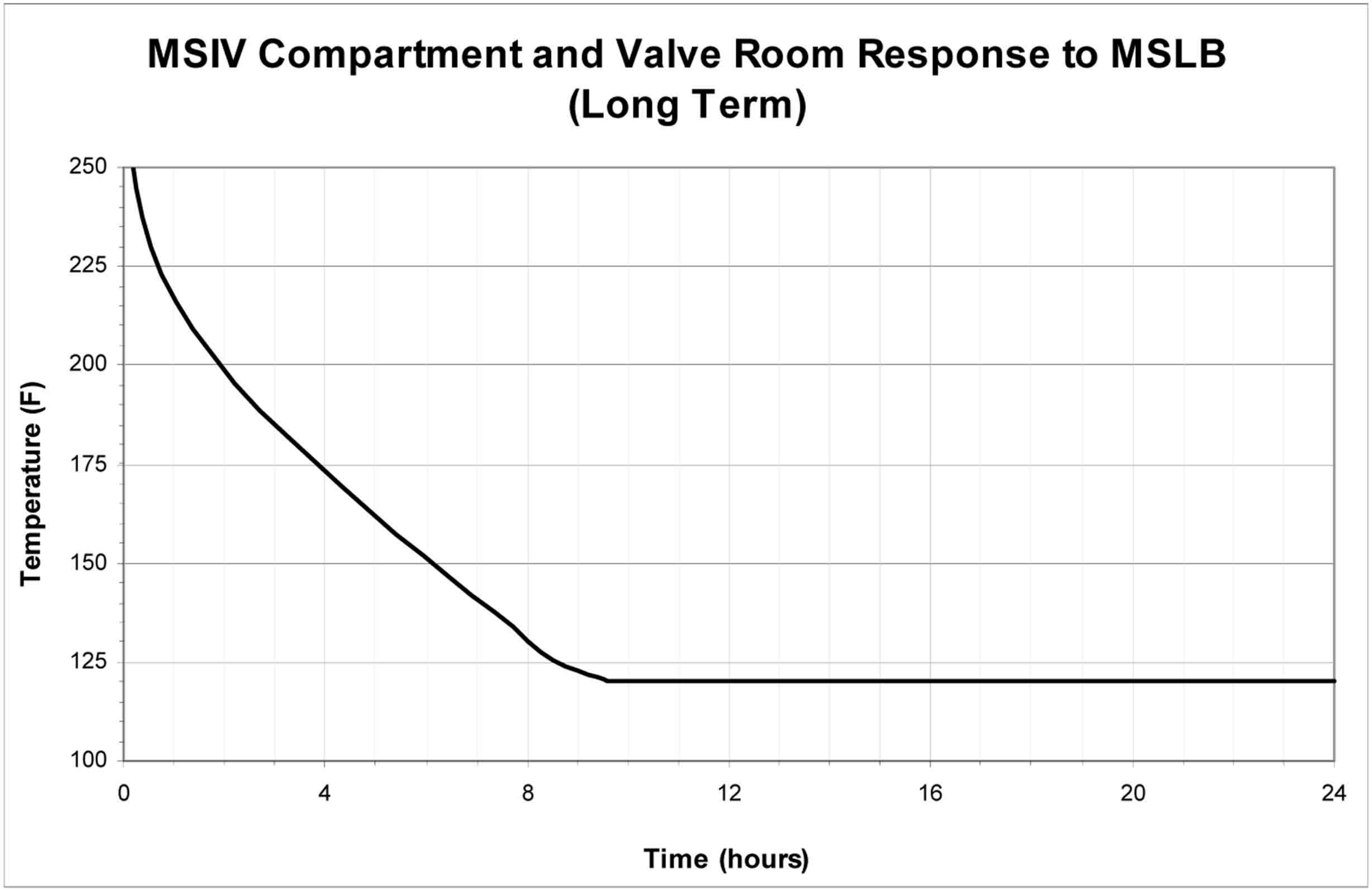


Figure 3D.5-9 (Sheet 2 of 2)
MSIV Compartment Response to MSLB (Long Term)

Figure 3D-201
Not Used.

ATTACHMENT A

SAMPLE EQUIPMENT QUALIFICATION DATA PACKAGE (EQDP)

The equipment qualification data package consists of the following elements:

Section 1.0–Specifications
Section 2.0–Qualification Program
Section 3.0–Qualification by Test
Section 4.0–Qualification by Analysis
Section 5.0–Qualification by Experience
Section 6.0–Qualification Program Conclusions
Table 1–Qualification Summary

EQDP-_______
Rev. _______
{date issued}

EQUIPMENT QUALIFICATION DATA PACKAGE

Equipment ____

Manufacturer ____

Model ____

Application ____

Environment: ____ Harsh ____ Mild

Prepared by: ____
{name}

Reviewed by: ____
{name}

Approved by: ____
{name}

This document provides or summarizes the seismic and environmental qualification of the equipment identified above in accordance with the AP1000 EQ Program Methodology.

1.0 SPECIFICATIONS

1.1 EQUIPMENT IDENTIFICATION: {create table(s) for details if a model series is to be qualified.}

Manufacturer	____________________
Model	____________________
Technical Manual	____________________
Drawings	____________________
Specification No.	____________________
Modifications	____________________

1.2 INSTALLATION REQUIREMENTS: {Cite vendor technical manual; details of mounting used for seismic test specimen(s); include any special requirements unique to Class 1E service}

1.3 ELECTRICAL REQUIREMENTS

1.3.1	Voltage:	__________	
1.3.2	Frequency:	__________	{if powered by AC}
1.3.3	Load:	__________	{as applicable}
1.3.4	Other:	__________	{identify and address as needed}

1.4 AUXILIARY DEVICES: {These are devices required to be interfaced with the subject equipment to provide qualification or operability but not specifically included or addressed in this document.}

1.5 PREVENTATIVE MAINTENANCE: {Identify manufacturer recommended maintenance activities required as part of the qualification program. Identify activities that are required to support qualification or the qualified life. "None" shall mean that maintenance is not essential to qualification or the qualified life. The following statement may be used in cases where qualification is not contingent upon maintenance or surveillance activities:

"No preventive maintenance is required to support the equipment qualified life. This does not preclude development of a preventive maintenance program designed to enhance equipment performance and identify unanticipated equipment degradation as long as this program does not compromise the qualification status of the equipment. Surveillance activities may also be considered to support the basis for, and a possible extension, of the qualified life."}

1.6 SAFETY FUNCTIONS

{Specify known safety functions for which qualification is intended to apply.}

1.7 PERFORMANCE REQUIREMENTS[a] for: {RCS Loop RTDs}

			Containment	DBE[b] Conditions	
	Normal	Abnormal	Test		
Parameter	Conditions	Conditions	Abnormal	Seismic	LOCA
1.7.1 Time requirement					
1.7.2 Performance					

1.8 ENVIRONMENTAL CONDITIONS[a] for Same Function

1.8.1 Temperature (°F)

1.8.2 Pressure (psig)

1.8.3 Humidity (%RH)

1.8.4 Radiation (Rads)

1.8.5 Chemicals

1.8.6 Vibration

1.8.7 Acceleration (g)

Notes: a: Test margin is not included in the parameters of this section.
b: DBE is the Design Basis Event.

{If more than one set of performance requirements and/or associated environmental conditions are to be specified, replicate these sections in pairs as "1.8 Performance ..." and "1.9 Environment ...", etc.}

2.0 QUALIFICATION PROGRAM

2.1 PROGRAM OBJECTIVE

The objective of this qualification program is to demonstrate, employing the recommended practices of Regulatory Guides 1.89 and 1.100 and IEEE 323-1974, 344-1987, {cite others as applicable} capability of the {Equipment description} to perform its/their safety related function(s) described in EQDP Section 1.7 while exposed to the applicable conditions and events defined in EQDP Section 1.8.

{Narrative should introduce an outline of the program plan. Table below to be completed as graphic reference. Table shall not be abbreviated; items must appear and be addressed by direct response.}

2.2 REFERENCES

{List test report(s) and information sources cited in this document}

	Qualification Method(s)		
CONDITION	**TEST**	**ANALYSIS**	**OTHER**
Aging:			
Thermal	________	________	________
Radiation	________	________	________
Vibrational	________	________	________
Operational Cycling	________	________	________
Electrical	________	________	________
Mechanical	________	________	________
Abnormal Environment	________	________	________
Inadvertent ADS Actuation	________	________	________
Seismic	________	________	________
LOCA	________	________	________
HELB Inside Containment	________	________	________
HELB Outside Containment	________	________	________
Post-accident Aging	________	________	________

NOTES:

{All spaces above to be noted as "Yes," "No," or "Note #." Notes will be appended to the table. Notes will also include items "Not Applicable" with terse explanation and/or forwarding reference.}

3.0 QUALIFICATION BY TEST (TEST PLAN AND SUMMARY)

3.1 SPECIMEN DESCRIPTION

{Identify the item or items to be tested}

3.2 NUMBER TESTED

{If more than one type is to be tested, identify how many of each. Subsequent Sections should clarify specifics for each.}

3.3 MOUNTING

{Identify specific seismic mounting details, referencing applicable drawings, instructions, documents. Note existence of differences from manufacturer recommendations}

3.4 CONNECTIONS

{Identify interfaces, both electrical and mechanical, identify any connectors or sealing assemblies used which are not provided with the equipment, or are not covered by this qualification.

3.5 TEST SEQUENCE PREFERRED

This section identifies the preferred test sequences as specified in IEEE 323-1974.

3.5.1 Inspection of Test Item
3.5.2 Operation (Normal Condition)
3.5.3 Operation (Performance Specifications Extremes: Section 1)
3.5.4 Simulated Aging
3.5.5 Vibration/Seismic
3.5.6 Operation (Simulated High Energy Line Break Conditions)
3.5.7 Operation (Simulated Post-HELB Conditions)
3.5.8 Inspection

3.6 TEST SEQUENCE ACTUAL

This section identifies the actual test sequence which constitutes the qualification program for this equipment. A justification for anything other than the preferred sequence is provided.

Test Sequence (from Section 3.5):

{List and explain; provide forwarding references to subsequent subsections as necessary}

3.7 **SERVICE CONDITIONS TO BE SIMULATED BY TEST**[(1)]

	Normal	Abnormal	Seismic	HELB	Post-HELB
3.7.1 Temperature (°F)					
3.7.2 Pressure (psig)					
3.7.3 Humidity (% RH)					
3.7.4 Radiation (Rads)					
3.7.5 Chemicals					
3.7.6 Vibration					
3.7.7 Seismic (g)					

(1) Test parameter margins are included for the worst-case known requirements applicable to the equipment type. Margin for a specific parameter is dependent on the requirements of each application or location for the equipment; these may vary.

(2) Post-accident operability addressed through simulated thermal aging. Temperature and other parameters are selected to envelop the requirements.

3.8 MEASURED VARIABLES

This section tabulates the variables and parameters required to be measured during each of the following tests in the qualification test sequence.

Tests: {example}
A: Thermal Aging
B: Mechanical Cycling
C: Irradiation
D: Seismic Test
E: HELB Test

	Required	Not Required
3.8.1 Category I – Environment		
3.8.1.1 Temperature	...	...
3.8.1.2 Pressure	...	...
3.8.1.3 Moisture	...	...
3.8.1.4 Gas Composition	...	...
3.8.1.5 Vibration	...	...
3.8.1.6 Time	...	...
3.8.2 Category II – Input Electrical Characteristics		
3.8.2.1 Voltage	...	...
3.8.2.2 Current	...	...
3.8.2.3 Frequency	...	...
3.8.2.4 Power	...	...
3.8.2.5 Other	...	...
3.8.3 Category III – Fluid Characteristics		
3.8.3.1 Chemical Composition	...	...
3.8.3.2 Flowrate	...	...
3.8.3.3 Spray	...	...
3.8.3.4 Temperature	...	...
3.8.4 Category IV – Radiological Features		
3.8.4.1 Energy Type	...	...
3.8.4.2 Energy Level	...	...
3.8.4.3 Dose Rate	...	...
3.8.4.4 Integrated Dose	...	...
3.8.5 Category V – Electrical Characteristics		
3.8.5.1 Insulation Resistance	...	...
3.8.5.2 Output Voltage	...	...
3.8.5.3 Output Current	...	...
3.8.5.4 Output Power	...	...
3.8.5.5 Response Time	...	...
3.8.5.6 Frequency Characteristics	...	...
3.8.5.7 Simulated Load	...	...

	Required	Not Required
3.8.6 Category VI – Mechanical Characteristics		
3.8.6.1 Thrust	...	...
3.8.6.2 Torque	...	...
3.8.6.3 Time	...	...
3.8.6.4 Load Profile	...	...
3.8.7 Category VII – Auxiliary Equipment		
3.8.7.1 {as applicable, also see Section 1.4 of EQDP}	...	...

3.9 TYPE TEST SUMMARY

3.9.1 Normal Environment Testing

Operation of the {equipment} under normal conditions is demonstrated by {discuss test, checks, et. al. which provide baseline performance data} ... , as reported in Reference _.

3.9.2 Abnormal Environment Testing

Operation of the {equipment} under abnormal conditions is demonstrated by {discuss test, checks, et. al. which provide baseline performance data} ... , as reported in Reference _.

3.9.3 Aging Simulation Procedure

{Describe the aging mechanisms simulated and the sequence, including justifications as necessary.}

The test units were pre-conditioned to simulate an aged condition prior to subjecting them to the Design Basis Event (DBE) seismic and environmental conditions/simulation. The aged condition was achieved by separate phases of {accelerated thermal aging, thermal cycling, and radiation exposure to a total integrated gamma dose equivalent to a twenty-year normal dose plus the design basis accident dose, and accelerated flow induced and pipe vibration simulation}. Through all the pre-conditioning phases, the {equipment, performance} were monitored to verify {continuous operation}.

3.9.3.1 Design Life: {Also, justification of the bases for a design life goal should be provided, when used in mild-environment programs. Generally inapplicable to harsh-environment programs.}

3.9.3.2 Shelf Life: {Though not typically applicable, state any limitation in life, as well as conditions which may be detrimental if known.}

3.9.3.3 Thermal Aging: The qualified life is __ years based on an ambient temperature of {__°C (__°F) and a __°C temperature rise due to _______}. Calculations are based on a test temperature of __, test duration of ___ hours, and an activation of ___ eV (See References x, et al.)}.

3.9.3.4 Radiation Aging: The qualified life is limited by the expected radiation during the __-year life and the Design Basis Event. {Subtract accident TID from qualified TID; account for margin, remainder is to be compared to normal/abnormal radiation requirements to yield life limits.}

3.9.3.5 Operating Cycles: {Expected number of electrical and/or mechanical cycles, or numbers of actuations, as applicable. Estimated on the basis of the expected for the design, qualified, installed life of the equipment. Specification may be on a per annum or a per fuel cycle basis. Compare to cycle life data from test.}

3.9.3.6 Vibration Aging: {present bases; refer to test profile and/or Subsection 3.9.4}.

3.9.4 Seismic Tests

The seismic testing reported in Reference x was completed on aged equipment employing {method(s)} in accordance with Regulatory Guide 1.100 and IEEE 344-1987. ... {Summarize equipment condition and/or performance versus the acceptance criteria.} ... Actual margin should be determined for each application/location throughout the plant and verified to meet or exceed the margin requirements.

{Discuss or reference discussion of test anomalies.}

3.9.5 High Energy Line Break/Post HELB Simulation

The {equipment} were subjected to the HELB simulation temperature/pressure profile of Figure x. Following the __°F temperature peak, the temperature gradually declines to __°F and is held at saturated steam conditions for _ days, simulating a _____ period of Post-HELB operation. The test data and activation energy specified in Subsection 3.9.3.3 can be used to determine margin in post-accident aging for each application/location of the equipment.

{Summarize equipment condition and/or performance versus the criteria}

{Discuss or reference discussion of test anomalies.}

4.0 <u>QUALIFICATION BY ANALYSIS</u>

The AP1000 EQ Program does permit qualification solely on the basis of analyses for equipment outside the scope of 10CFR50.49. The following subsections discuss each of the analyses preformed, its test basis and justification, and summarizes conclusions documented in References x; et. al., which provided detailed accounts of each analysis.

{Each subsection will address a particular analysis, if more than one is performed to support qualification.}

4.x (EXAMPLE)

{The purpose and objective will be identified here. Subsections will provide necessary details per the following format.}

4.x.1 {Equipment, Characteristic or Aspect} Analyzed

{A general description of the equipment and its function based on applicable equipment and mounting drawings, and purchase orders.}

4.x.2 Equipment Specification(s)

{The applicable design standards shall be documented including any limitations imposed by the equipment specification. Installation detail considered or represented are to be included.}

4.x.3 Methods and Codes

{Description of analytical methods or techniques, computer program, mathematical model(s) used, and the method(s) of verification}

4.x.4 Acceptance Criteria

{The specific safety function(s), postulated failure modes, or the failure effects to be demonstrated by analysis.}

4.x.5 Model

{Description of mathematical model of equipment or feature analyzed.}

4.x.6 Assumptions and Justifications

{EXAMPLES: Description of the loading conditions to be used. Summary of stresses to be considered.}

4.x.7 Impact to Safety Function

{Summarize analytically established performance characteristics and their acceptability. Discussion and summary of the analytical results which demonstrate equipment structural integrity and, where appropriate, operability. Particular to cabinets, critical deflections should be determined and included in mounting requirements for spacing with respect to other equipment and structures.}

4.x.8 Conclusions

{Descriptive summary, including any conditions imposed on qualification or use; qualified life, limitations, surveillance/maintenance requirements, et. al.} Further discussion of this analysis is presented in Reference x.

4.Y ENVIRONMENTAL QUALIFICATION ANALYSIS FOR {VALVE SOFT PARTS}

{purpose and objective}

4.Y.1 Equipment Identification

{Per Subsection 6.2.3.1}

4.Y.2 Component Identification

{Per Subsection 6.2.3.1}

4.Y.3 Safety Related Functions

{Per Subsection 6.2.3.2}

4.Y.4 Component Acceptance Criteria

{Per Subsection 6.2.3.3}

4.Y.5 Service Conditions

{Per Subsection 6.2.3.4}

4.Y.6 Potential Failure Modes

{Per Subsection 6.2.3.5}

4.Y.7 Identify the Environmental Effects on Material Properties

Each non-metallic, including lubricants, is evaluated to determine the effect of the environmental conditions on the material properties. For each non-metallic, a radiation threshold level and maximum service temperature is identified.

The radiation threshold level and the maximum service temperature are identified using materials handbooks, textbooks, government and industry reports, and laboratory data. If the evaluation indicates that the lowest levels may be exceeded for certain equipment, higher levels are identified at which varying degrees of material degradation may occur.

Mechanical equipment is highly resistive to degradation due to elevated humidity levels: therefore, relative humidity is not included as a parameter to be evaluated for environmental qualification. Pressure can be discounted for most equipment types, as there are no foreseen failures due to elevated pressure levels for most mechanical equipment. However, pressure must be addressed in the evaluation.

The susceptibility of the non-metallic material to the chemicals due to the design basis accident and exposure to the process fluid is evaluated. The material information in the chemical handbooks is an acceptable source of qualification documentation.

4.Y.7.1 Perform Thermal Aging Analysis

Aging analysis is performed for organic materials. Mineral-based subcomponents are not considered to be sensitive to thermal aging during the design life of a plant and, therefore, are not analyzed.

Aging in mechanical components is associated with corrosion, erosion, particle deposits and embrittlement. In new construction, corrosion and erosion are considered by providing additional material thickness as a corrosion or erosion allowance above the required design. The other aging phenomena are considered during inservice inspections of operating components in accordance with ASME Code, Section XI. Aging qualification of metallic parts of equipment except for corrosion and erosion is in compliance with ASME Code, Section XI, therefore aging effects on metallic components are not addressed herein.

The non-metallic material analysis for determining the expected qualified thermal life is performed using Arrhenius methodology. The thermal input during the operating time, as explained below, is deducted from the tested thermal aging of the material at service temperature to obtain the qualified life.

The component is evaluated for the specified post-accident operating time. The thermal input from the postulated accident profile (i.e., LOCA/MSLB) for the duration of the specified operating time is compared to the material thermal aging data. The Arrhenius model is used to perform this comparison. The component is evaluated for the maximum post-accident operating time unless a system analysis is performed to justify shorter operating times.

Analysis of the non-metallics should also take into account any degradation of the part due to its use in dynamic modes (i.e., moving part).

4.Y.7.2 Evaluate the Environmental Effects on Equipment Safety-Related Function

A conservative initial screening of the non-metallic subcomponents is made by comparison of the material capabilities (threshold radiation level and maximum service temperature) with the maximum postulated environmental conditions. If the threshold radiation values and the maximum service temperatures are above the maximum postulated environmental conditions, and if the material aging analysis demonstrates a service life sufficient to survive the accident duration, then the material is considered acceptable.

Those items which are not shown to be acceptable based on the above comparison are evaluated in further detail regarding:

- extent of material degradation
- material properties affected
- equipment/subcomponent function
- extent of equipment functional degradation
- location-specific environmental conditions

4.Y.8 Conclusions

{Per subsection 3D.6.2.3.7}

4.Y.9 EQ Maintenance Requirements

{Per subsection 3D.6.2.3.8}

5.0 QUALIFICATION BY EXPERIENCE

This method of qualification is not used.

6.0 QUALIFICATION PROGRAM CONCLUSIONS

6.1 AGING

{Discuss specifics and state on limitations or requirements; specifics with respect to:

- Design Life Goal
- Thermal Aging
- Radiation Aging
- Operating Cycles
- Vibration Aging}

6.2 DBE QUALIFICATIONS

6.3 PROGRAM CONCLUSIONS

The qualification of the {equipment} is demonstrated by the completion of the simulated aging and Design Basis Event testing described herein and reported in Reference {1}.

{State any conditions imposed on qualification or qualified life, cite any lessons learned which necessitate future user actions to preserve continued qualification}

{Refer to Table 1}

Table 1

QUALIFICATION SUMMARY

SYSTEM	{RPS}
CATEGORY	Category[1] {a}
LOCATION	{Containment bldg.}
STRUCTURE/AREA	{Zone Number}
EQUIPMENT TYPE	{pressure transmitter }
MANUFACTURER	{___________}
MODEL	{___________}

PARAMETER	QUAL METHOD[2]	ENVIRONMENTAL EXTREMES QUALIFIED[3]	ENVIRONMENTAL EXTREMES SPECIFIED[4]	NOTES
NORMAL				
ABNORMAL				
QUALIFIED LIFE				{5}
SEISMIC	{Both}	Figure x	{Ref; Fig.}	
ACCIDENT		Figure x	{Ref; Fig.}	
Temperature	{Test}	___°F		
Pressure	{Test}	___ psig		
Rel. humidity	{Test}	___%		
Radiation	{Both}	___E+06 R(γ)		
	{Both}	___E+06 R(β)		
Chemistry	{Test}	{Note 6}		
Operability	{Both}			
Accuracy	{Test}			

NOTES:

1. Equipment category as per NUREG-0588, Appendix E, Section 2.
2. Qual. Methods are: Test, Analysis, Both (Test & Anal.), or Other.
3. Qualified values are test extremes which include margin.
4. Environmental parameters for the plant location are to be inserted. If more than one applicable, most extreme are to be cited
5. Qualified life estimated on basis of maximum normal temperature of ___°C (__°F) and a temperature rise of ___°C (____°F).
6. Chemistry Conditions: {pH and composition}.

ATTACHMENT B

AGING EVALUATION PROGRAM

B.1 Introduction

As stated in IEEE 323, aging of Class 1E equipment during normal service is considered as an integral part of the qualification program. The objective is not to address random age-induced failures that occur in-service and are detected by periodic testing and maintenance programs. The objective is to address the concern that some aging mechanisms, when considered in conjunction with the specified design basis events (DBE), may have the potential for common mode failure.

The AP1000 equipment qualification program addresses the aging concern and makes maximum use of available data and experience on aging mechanisms. This approach places primary emphasis on common mode failures due to enveloping design basis events. For example, reasonable assurance against common mode failures being induced because of a loss of heating, ventilation, and air conditioning (HVAC) is provided by adequate design, normal maintenance, and calibration procedures.

B.2 Objectives

The objectives of the aging evaluation program follow:

- To establish, where possible, the effects of the degradation due to aging mechanisms that occur before the occurrence of an accident, when safety-related equipment is called upon to function

- To provide increased confidence that safety-related equipment performs its safety-related function under the specified service condition.

B.3 Basic Approach

The general approach to addressing aging allocates equipment to one of two subprograms (A or B).

- Subprogram A includes electrical equipment required to perform a safety-related function in a high-energy line break (HELB) environment. For this equipment an aging simulation is included as part of the equipment qualification test sequence. The equipment is energized during the aging simulation.

- Subprogram B includes equipment required to mitigate high-energy line breaks but which, due to its location, is isolated from any adverse external environment resulting from the accident. For equipment in Subprogram B the single design basis event capable of producing an adverse environment at the equipment location is the seismic event. Aging, for Subprogram B, is not included in the equipment qualification test sequence. Significant aging mechanisms are determined by evaluation of available test data. Generally, this data is from separate programs conducted to demonstrate that aged components continue to meet manufacturer's performance specifications under applicable seismic design basis event conditions and that seismic testing of unaged equipment is not invalidated by anticipated aging mechanisms.

B.4 Subprogram A

Electrical equipment required to perform a safety-related function in a high-energy line break (such as a loss of coolant accident, feed line break, or steam line break) environment is included in

Subprogram A. This subprogram provides for an aging simulation to be included in the equipment's qualification test sequence.

B.4.1 Scope

The typical equipment scope and aging mechanisms applied under Subprogram A are shown in Tables 3D.B-1 and 3D.B-2, respectively. The equipment selected is that Class 1E equipment qualified to operate in a high-energy line break environment. The aging mechanisms discussed next are those to which the equipment may be potentially sensitive in its installed location.

B.4.2 Aging Mechanisms

The aging mechanisms that could potentially affect electrical equipment in Subprogram A are discussed under the following headings:

Time, in conjunction with:

- Operational stresses (current, voltage, operating cycles, Joulean self-heating)
- (External stresses (thermal, vibration, radiation, humidity, seismic).

The aging mechanisms considered potentially significant and to be simulated are identified in Table 3D.B-2 for each item of equipment in Subprogram A. Where applied, the aging mechanisms are simulated as described in the following discussions.

B.4.3 Time

For equipment subject to high-energy line break conditions, the most significant in-service aging mechanisms (that is, radiation and thermal) come into effect during reactor operation. Consequently, it can be assumed that the "aging clock" starts on plant startup.

B.4.4 Operational Stresses

<u>Electrical Cycling</u>

Electrical supplies to safety-related equipment are, in general, highly stable. So aging effects due to supply cycling during service are not anticipated. Where the equipment is anticipated to experience multiple startup and shutdown cycles, the equipment is electrically cycled to simulate the number of anticipated startup and shutdown cycles plus 10 percent.

<u>Mechanical Cycling</u>

Aging effects resulting from anticipated mechanical cycling of the equipment are simulated by applying, as a minimum, the number of cycles estimated to occur during the target qualified life plus 10 percent. Mechanical cycling covers such operations as switching and relay actuation.

<u>Joulean Self-Heating</u>

Where the equipment is not aged in a live condition, the aging effects resulting from Joulean self-heating are recognized by employing the equipment operating temperature as the datum temperature for assessing the accelerated thermal aging parameters to be employed.

B.4.5 External Stresses

Thermal Effects

Thermal effects are considered one of the most significant aging mechanisms to address. The equipment is thermally aged to simulate an end-of-qualified-life condition using the Arrhenius model to establish the appropriate conditioning period at elevated temperature. Where data is not available to establish the model parameters for the materials employed, a verifiably conservative value of 0.5 eV is used for activation energy (Attachment D).

For each piece of equipment an appropriate normal and abnormal operating temperature and an associated time history are determined for inclusion in the Arrhenius model. The equipment temperature is determined by the addition of an appropriate equipment specific ΔT to the external ambient temperature. Attachment D also provides information concerning the determination of appropriate ambient temperatures and time-temperature histories for use in thermal aging evaluation of equipment. Post-accident thermal aging is included by recognizing the higher post-accident ambient temperatures in determining the parameters employed for the post-accident accelerated thermal aging simulation.

In-Service Vibration

The majority of safety-related electrical equipment has a proven history of in-plant service. Thus, it is unlikely that a significant, undetected, failure mechanism exists because of low-level, in-plant vibration. In addition, a simulation of earthquakes smaller than the safe shutdown earthquake (SSE) employed during equipment and component seismic testing give added confidence that this potential aging mechanism is covered (See Attachment E, Section E.4.4). For line-mounted equipment, in-service pipe and flow induced vibration may be significant. As a consequence, an additional vibration aging step is included in the aging sequence as indicated for certain items of equipment in Table 3D.B-2. (See Attachment E, Subsection E.5.2.4.)

Radiation

Radiation during normal operation is not considered an aging mechanism for equipment subject to in-service integrated doses less than 10^4 rads. Research has established that no aging mechanisms are measurable below 10^4 rads (Attachment C) for materials and most components supplied in safety-related electrical equipment. Some devices may have performance limitations below 10^4 rads. For radiation doses in excess of 10^4 rads, the equipment is irradiated using a gamma (γ) source to a dose equivalent to the estimated dose to be incurred during normal operation for the target qualified life. The estimated doses employed are specified in the equipment qualification data package, Subsection 1.8.4, and are based on a 100 percent load factor, including appropriate margin. For Subprogram A equipment, the equivalent accident dose is usually applied before design basis event testing.

Humidity

The use of materials significantly affected by humidity is avoided. For equipment subject to high energy line break environments, the aging effects due to humidity during normal operation are judged to be insignificant compared to the effects of the high-temperature steam accident simulation. Therefore, no additional humidity aging simulation is required.

Seismic Aging

The potential aging effects of low-level seismic activity and some low-level, in-plant vibration are addressed by employing a simulation of five earthquakes of 50 percent of the magnitude of a safe shutdown earthquake before seismic testing of the aged equipment.

B.4.6 Synergism

An important consideration in aging is the possible existence of synergistic effects when multiple stress environments are applied simultaneously. The potential for significant synergistic effects is addressed by the conservatisms inherent in using the "worst-case" aging sequence, conservative accelerated aging parameters and conservative, design basis event test levels which provide confidence that any synergistic effects are enveloped.

B.4.7 Design Basis Event Testing

Design basis event testing subsequent to equipment aging is discussed in Appendix 3D as to guidelines for defining high-energy line break environments and seismic conditions. Testing for equipment specific test environments and seismic parameters is discussed in Attachment A, Section 3.0.

B.4.8 Aging Sequence

The aging mechanisms applied to equipment subject to high-energy line break environments are determined by definition of the aging environments at the equipment location and by a subsequent evaluation of the sensitivity of the equipment to these environments. If the sensitivity of the equipment is not known, aging mechanisms are simulated by conservative methods as previously described. Those aging mechanisms that are simulated for typical equipment subject to high-energy line break environments are shown in Table 3D.B-2.

The order in which each of the aging mechanisms is applied is as shown in Table 3D.B-2. This order is considered to be conservative, as no aging mechanism is anticipated to be capable of reducing the impact of the previously applied mechanisms. As an example, thermal aging is applied before radiation aging to preclude the annealing out of radiation-induced defects. Similarly, the effects of mechanical aging are considered more significant when applied to equipment that has already been preaged to address thermal and radiation phenomena.

B.4.9 Performance Criterion

The basic acceptance criterion is that the qualification tests demonstrate the capability of the aged equipment to perform prespecified, safety-related functions consistent with meeting the performance specification of Attachment A, Section 1.7 of the applicable equipment qualification data packages while exposed to the associated environmental conditions defined in Attachment A, Section 1.8.

B.4.10 Failure Treatment

When thermal aging is simulated at an equipment level, a conservative value for the activation energy is assumed for the components composing the equipment. As a consequence, many components are grossly overaged, and failure of some of the components is expected during the aging simulation. When three test units are preaged, in the event of such failure(s), one of the following options is selected.

- when a particular component fails in one of the three test units, the failure is considered random. The failed component is replaced by a new component, and the test is continued

- when a particular component fails in more than one of the three test units, either:

    1. the failed components are replaced by new identical components and the aging simulation continued. The claimed qualified life of the unit is consistent with the minimum aging period simulated by at least two of the three units; or

2. the failed components is replaced by identical components specifically aged to the qualified life by assuming for thermal aging a less conservative activation energy specifically determined for the component, or
3. the failed components are replaced by a different type of component which is aged for a period equal to the test units.

When less than three test samples prevent such a conclusion from being reached, any failures are investigated to ascertain whether the failure mechanism is of common mode origin. Should a common mode failure mechanism be identified as having caused the failure, a design change is implemented to eliminate the problem. Supplemental or repeat tests will be completed to demonstrate compliance with the acceptance criteria.

B.5 Subprogram B

Subprogram B includes Class 1E equipment not required to perform a safety related function in a high-energy line break environment. It involves a review of available information to demonstrate the absence of significant in-service aging mechanisms. For equipment allocated to this subprogram, the single design basis event capable of producing an adverse environment at the equipment location is the seismic event. Seismic testing completed on unaged equipment is verified as valid by demonstrating via this subprogram that no available information suggests that aged materials and components would not continue to meet their design specification during a seismic event.

B.5.1 Scope

Subprogram B includes both a review of material analysis and the results of a component testing program for equipment not required to perform a safety-related function in a high-energy line break environment. Equipment is included that is required to mitigate high-energy line breaks but which, because of the equipment location, is isolated from the adverse environment resulting from the accident. Typical equipment allocated to Subprogram B is identified in Table 3D.B-1.

B.5.2 Performance Criteria

Available Material Analysis – For equipment and components for which aging is addressed by evaluation of appropriate mechanisms, the basic performance criterion is that the evaluation of test data demonstrates the effect of aging is minor and does not affect the capability of the aged equipment to perform prespecified functions. This is consistent with meeting the performance specification of Attachment A, Section 1.7 of the applicable equipment qualification data package while exposed to the associated environmental conditions defined in Attachment A, Section 1.8.

Available Component Aging Data – Random component failure or unacceptable performance due to aging is detected by routine maintenance and equipment calibration during service. The objective of Subprogram B is to provide reasonable assurance that a seismic event does not constitute a common mode failure mechanism capable of inducing unacceptable performance characteristics in aged components. Consequently, the single performance criterion for the aging portion of the qualification sequence requires that the component not fail to perform its general function, not that the component meets the original design and procurement specifications.

For the seismic event simulation, the component is considered acceptable if, during and after the simulation, it does not exhibit any temporary or permanent step change in performance characteristics. Failure of one of three components tested is considered a random failure, subject to an investigation concluding the observed failure is not common mode.

B.5.3 Failure Treatment

In the event of failure to demonstrate conformance to criteria, the following options are available for resolution of qualification with respect to age:

- Establish a maintenance and surveillance program

- Replace the materials or components with those constructed of materials of known acceptable characteristics.

Table 3D.B-1
Typical Class 1E Equipment Scope and Subprogram Allocation

Aging Method	Equipment
Subprogram A	Valve Motor Operators Solenoid Valves Externally Mounted Limit Switches Pressure Transmitter (Group A) Differential Pressure Transmitter (Group A) Resistance Temperature Detectors Neutron Detectors Pressure Sensor Batteries*
Subprogram B	Pressure Transmitter (Group B) Differential Pressure Transmitter (Group B) Main Control Board Switch Modules Recorders (Post-Accident Monitoring) Indicators (Post-Accident Monitoring) Instrument Bus Distribution Panels Instrument Bus Power Supply (Static Inverter) Motor Control Centers Integrated Protection Cabinets (IPC) Engineered Safety Features Actuation Cabinets (ESFAC) Logic Cabinets Reactor Trip Switchgear Reactor Coolant Pump Switchgear

Note:
* To comply with R.G. 1.158

Table 3D.B-2
Aging Mechanism Sequence

Equipment	Location	Subprogram	Burn-in	Aging Mechanisms						DBE	
				Thermal	Radiation	Mechanical	Vibration	Electrical	Seismic	Seismic	HELB
Safety-related Valve Motor Operators	I/C O/C	A A		X X	X X	X X	X X		X X	X X	X
Safety-related Solenoid Valves	I/C O/C	A A		X X	X X	X X	X X		X X	X X	X
Safety-related Externally Mounted Limit Switches	I/C O/C	A A		X X	X X	X X	X X		X X	X X	X
Pressure Transmitters	I/C&OC	A	X	X	X				X	X	X
Differential Pressure Transmitters	I/C&OC	A	X	X	X				X	X	X
Resistance Temperature Detectors: Well Mounted	I/C	A		X	X		X		X	X	X
Excore Neutron Detectors	I/C	A		X	X				X	X	X
Pressure Sensor	I/C	A							X	X	X

ATTACHMENT C

EFFECTS OF GAMMA RADIATION DOSES BELOW 10^4 RADS ON THE MECHANICAL PROPERTIES OF MATERIALS

C.1 Introduction

One potential common-mode failure mechanism to consider in the qualification of safety-related equipment is gamma radiation. As part of a qualification program, the effect of gamma radiation dose is considered for two purposes: as a component of the high-energy line break environment and as a potential aging mechanism that could reduce the capability of safety-related equipment to perform safety-related functions under design basis event conditions (seismic or high-energy line break).

The scope of this attachment is limited to consideration of the effect of radiation for that substantial portion of equipment that does not experience an adverse change in external environment as a result of a high-energy line break, and for which, therefore, the only gamma radiation concern is an in-service aging mechanism.

This attachment assumes that the equipment contains devices that have been selected for performance through the total integrated dose expected in service. For example, devices such as integrated circuits may have a limit of 1000 rads established, in which case the following discussion applies for its installed life. The information in this attachment is not adequate to be applied to equipment that must perform its function in a high-energy line break.

The primary purpose of equipment qualification is to reduce the potential for common-cause failures due to environmental effects during the qualified life. Random failures that inevitably occur inservice are accommodated by the redundancy and diversity of the design of safety-related systems. Furthermore, in-service maintenance and testing programs are designed to detect such random failures. The chances of two identical components that perform identical functions failing during the same limited time period in between routine tests considered insignificant because of the following:

- General low failure rate of components used in nuclear equipment
- Minor differences in component material or geometric tolerances or both
- Minor differences in operating environment.

Therefore, failures that are induced in components by normal background gamma radiation below 10^4 rads (10^3 rads for some devices) alone are considered to be random. Thus, the only gamma radiation concern addressed for equipment not subject to an adverse high-energy line break environment is the potential for an aging mechanism resulting in a deterioration in component properties such that, when subject to seismic stress, a common-cause failure results. When considering such a failure mode, the aging mechanism of concern is not one that affects the electrical properties of components but one that reduces the mechanical strength and flexibility of components.

C.2 Scope

This report summarizes available information concerning the effects of gamma radiation on material mechanical properties. It justifies that for a gamma dose of less than 10^4 rads there are no observable radiation effects that impact material mechanical properties. Of the materials investigated, only Teflon TFE is subject to an alteration of mechanical properties for a gamma dose of less than 10^5 rads. Information is drawn from several sources listed as references in Section C.5. They include various texts concerning radiation effects and damage and pertinent reports.

C.3 Discussion

The primary effects of gamma photons on materials are ionization, material heating (primarily at high dose rates, which is of negligible significance here), and some displacement damage caused by high-energy photons. Some other types of radiation have effects similar to those induced by gamma radiation. This allows the use of data obtained from exposure of material to an alternate radiation to provide limited information concerning the effects of exposure to gamma radiation.

For example, the primary consequence of fast-neutron bombardment of material is atom displacement. Therefore, if the effect of radiation on a material property is primarily dependent on atom displacement, it is inferred that for an equivalent dose (rads) of gamma and fast-neutron radiation, data obtained from neutron irradiation provides a conservative estimate of the effect of gamma irradiation in producing displacements.

The same type of inference is drawn for the ionization effect of charged particle (for example, electron, proton, alpha particle) irradiation. Charged particles do not have the penetration capability that gamma or neutron radiations exhibit as a result of extensive interaction between charged particles and atomic charge centers.

Table 3D.C-1 summarizes information derived from the listed references. The information relates to the effect of gamma radiation on material mechanical properties. Table 3D.C-1 presents either the threshold dose (that dose at which an effect on any mechanical property can first be detected) or, the dose that results in the identified effect. This provides a general indication of the susceptibility of material mechanical properties to gamma radiation.

An evaluation of the information available on inorganic materials summarized in Table 3D.C-1 shows that the mechanical damage threshold for gamma radiation is many orders of magnitude greater than 10^4 rads. For the organic materials listed in Table 3D.C-1, a histogram comparing threshold dose level and frequency of material susceptibility is provided. In instances for which a material threshold dose is not indicated in Table 3D.C-1, a threshold value is assumed which is one order of magnitude lower than the indicated damage dose. Where information is available, referenced documents indicate that the difference between threshold dose and 25 percent damage dose is about a factor of three. Thus, a factor of 10 supplies substantial margin in estimating the threshold dose level. Figure 3D.C-1 shows that any indications of mechanical property damage thresholds below 10^4 rads would be extremely unusual.

The references listed do not identify the existence of materials whose mechanical properties are deteriorated when exposed to a gamma radiation dose up to 10^4 rads. So it can be concluded that common-cause failures do not occur in electrical equipment during or after a seismic event as a result of radiation-induced degradation up to 10^4 rads.

This is supported by NRC documentation available as an attachment to "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," which provides further justification for the use of 10^4 rads as a threshold for mechanical damage. The NRC information appears to be consistent with the information provided in Table 3D.C-1.

C.4 Conclusions

For Class 1E equipment subject to a lifetime gamma dose of up to 10^4 rads, it is not necessary to address radiation aging for qualification purposes provided that the equipment is not required to perform a safety-related function in a high-energy line break environment.

As previously noted, this appendix does not apply to electrical properties of components in safety-related equipment.

C.5 References

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Table 3D.C-1 (Sheet 1 of 2)
Radiation-Induced Degradation
of Material Mechanical Properties

Material	Mechanical Damage	Threshold Dose for Comments
Structural Metals	10^{19} n/cm^2 (fast neutron spectrum)	Similar to cold work (10^{10} rads)
Inorganic Materials	~10^{17} n/cm^2 (fast neutron spectrum)	Borated materials have lower threshold values for neutron irradiation.
Elastomers		
Natural Rubber	$2x10^6$ rads(C)	
Polyurethane Rubber	$9x10^5$ rads(C)	
Styrene-Butadiene Rubber	$2x10^6$ rads(C)	
Nitrile Rubber	$7x10^6$ rads(C)	Compression set is 25% degraded
Neoprene Rubber	$7x10^6$ rads(C)	
Hypalon	~10^7 rads(C)	Variable
Acrylic Rubber	$9x10^7$ rads(C)	Variable
Silicone Rubber	10^7 rads(C)	~25% damage
Fluorocarbon Rubber	$9x10^7$ rads(C)	~25% hardness, 80% elongation
Polysulfate Rubber	10^8 rads(C)	
Butyl Rubber	10^7 rads(C)	~25% damage
One rad (C) is the field of radiation that will produce 100 ergs/gm in carbon.		
Plastic		
Teflon TFE	$1.7x10^4$ rads(C)	
Kel-F	$1.3x10^6$ rads(C)	
Polyethylene	$\geq 10^7$ rads(C)	
Polystyrene	10^8 rads	
Mylar	10^6 rads(C)	Conservative
Polyamide (Nylon)	$8.6x10^5$ rads(C)	
Diallyl Phthalate	10^8 rads(C)	
Polypropylene	10^7 rads(C)	
Polyurethane	$7x10^8$ rads(C)	

Table 3D.C-1 (Sheet 2 of 2)
Radiation-Induced Degradation
of Material Mechanical Properties

Material	Mechanical Damage	Threshold Dose for Comments
Plastic (Continued)		
Kynar (400)	10^7 rads(C)	
Acrylics	8.2×10^5 rads	
Amino Resins	10^6 rads	
Aromatic Amide-Imide	10^7 rads	
Resins	10^7 rads	
Cellulose Derivatives	3×10^7 rads	25% damage
Polyester, Glass Filled	8.7×10^8 rads	
Phenolics	3×10^8 rads(C)	25% damage
Silicones	10^8 rads(C)	
Polycarbonate Resins	5×10^7 rads	25% damage to elongation
Polyesters	~ 10^5 - 10^6 rads	
Styrene Polymers	4×10^7 rads(C)	
Styrene Copolymers	4×10^7 rads(C)	25% damage
Vinyl Polymers	1.4×10^6 – 8.8×10^7 rads(C)	
Vinyl Copolymers	1.4×10^6 – 8.8×10^7 rads(C)	
Encapsulating Compounds		
RTV 501	2×10^6 rads	
Sylgard 182	2×10^6 rads	
Sylgard 1383	2×10^6 rads	
Polyurethane Foam	2×10^6 rads	
Epoxies	10^9 rads	

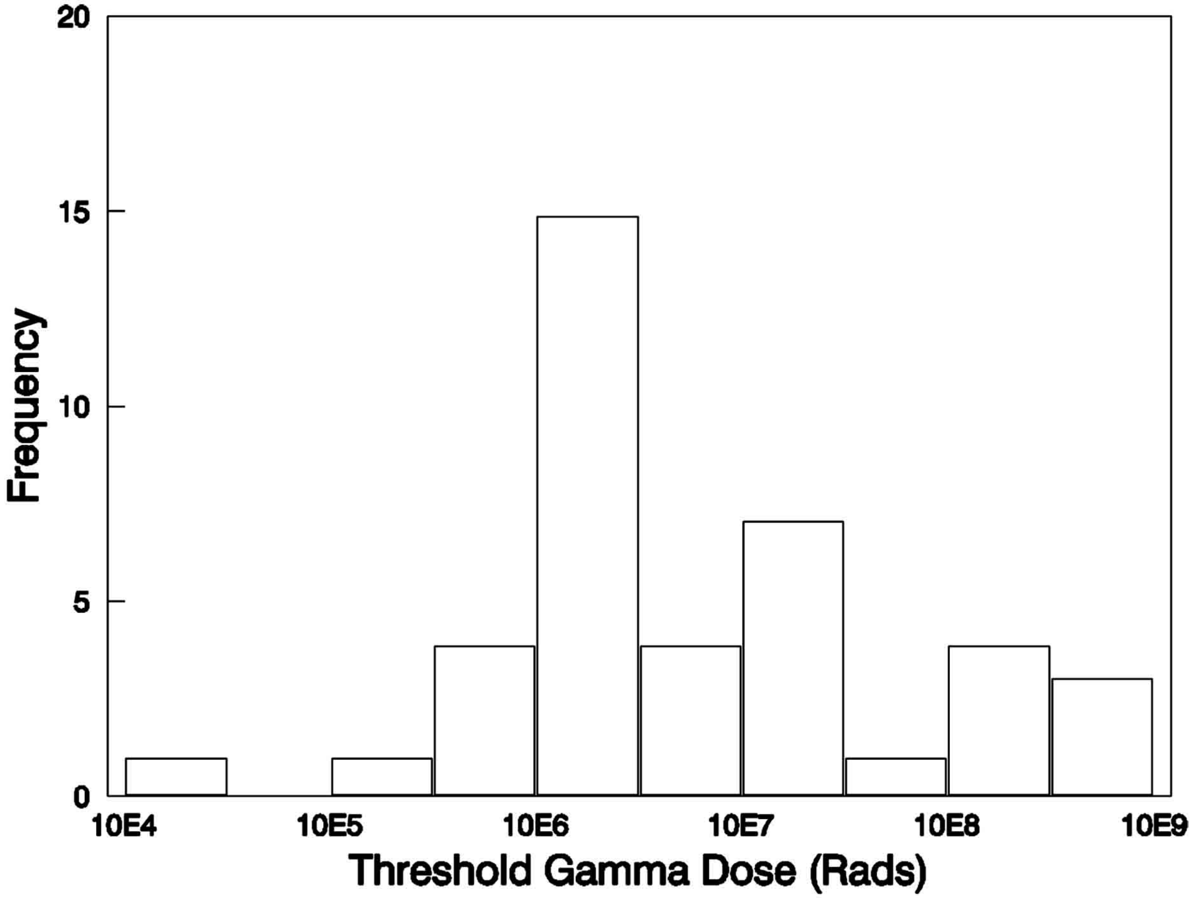


Figure 3D.C-1
Histogram of Threshold Gamma Dose for Mechanical Damage to Elastomers, Plastics, and Encapsulation Compounds

ATTACHMENT D

ACCELERATED THERMAL AGING PARAMETERS

D.1 Introduction

Attachment B describes the approach employed in the AP1000 equipment qualification program to address the aging requirement of IEEE 323. For equipment required to perform a safety-related function in a high-energy line break environment, the AP1000 equipment qualification program includes an aging simulation as part of its qualification test sequence (Subprogram A of Attachment B).

For equipment not required to perform a safety related function in a high-energy line break environment, the single design basis event considered is a seismic event. Aging, in this case (Subprogram B of Attachment B) is not usually included in the test sequence. Aging, where significant, is addressed by separate qualification of aged components, using conservative testing under applicable seismic design basis event conditions.

Thermal effects are one of the primary aging mechanisms addressed by the AP1000 equipment qualification program described in Attachment B for equipment containing nonmetallic or nonceramic materials. When thermal aging effects are established as potentially significant to the capability of the component or equipment to perform its safety-related function under design basis event conditions, or in the absence of evidence to the contrary, the component or equipment is thermally aged to simulate an end-of-qualified-life condition before design basis event testing. Equipment required to operate in a high-energy line break environment is also thermally aged to simulate the post-accident conditions consistent with its established functional requirements.

This attachment defines the appropriate thermal environments considered for each item of equipment in the AP1000 equipment qualification program and establishes consequent accelerated thermal aging parameters for use in the qualification programs.

D.2 Arrhenius Model

If an aging mechanism is governed by a single chemical reaction, the rate of which is dependent on temperature alone, the Arrhenius equation can be used as the basis for establishing the accelerated aging parameters:

$$\frac{dR}{dt} = Ae^{\frac{-E}{kT}} \tag{1}$$

where:

E	=	activation energy (eV)
k	=	Boltzmann's constant (8.617×10^{-5} eV/K)
A	=	constant factor
T	=	material temperature (K)
$\frac{dR}{dt}$	=	reaction rate = aging rate

Integration gives:

$$\Delta R = Be^{\frac{-E}{kT}} \Delta t \tag{2}$$

where:

ΔR	=	change in measured property due to aging
Δt	=	time for aging effect ΔR to occur
B	=	constant factor

If the accelerated aging process employed correctly simulates the change in properties due to aging under normal operating or post-accident temperature conditions, then:

$$\Delta R_1 = \Delta R_0 \tag{3}$$

and

$$B\, t_1\, e^{\frac{-E}{kT_1}} = B\, t_o\, e^{\frac{-E}{kT_0}}$$

and

$$\mathrm{Ln}\, t_1 = \frac{-E}{k} \frac{T_1 - T_0}{T_1 T_0} + \mathrm{Ln}\, t_o$$

where:

T_1	=	accelerated aging material temperature (K)
t_1	=	time at temperature T_1
T_0	=	material temperature under normal operating or post-accident conditions (K)
t_0	=	time at temperature T_0

From Equation 3, given an activation energy (E) for the material, the time required at any selected elevated temperature can be calculated to simulate the ambient aging effects.

This model has been verified to represent the thermal aging characteristics of nonmetallic and non-ceramic materials and is employed in the AP1000 equipment qualification program to derive accelerated thermal aging parameters. The only material dependent parameter input into this model, when establishing the accelerated aging parameters, is the activation energy. This parameter is a direct measure of the chemical reaction rate governing the thermal degradation of the material.

D.3 Activation Energy

A single material may have more than one physical property that thermally degrades (for example, dielectric strength, flexural strength.) As a consequence, the material exhibits different activation energies with respect to each property. The activation energy selected is the one that reflects the physical property most significant to the safety-related function performed or the stresses applied to the material by the design basis fault(s) considered.

In actual practice, however, rarely is the choice so simple. Electrical components are invariably made up of more than one material. In many cases either the materials employed are not known in any chemical detail but just by a general organic or industrial trade name, or the appropriate activation energy is not known.

Where an activation energy is not available that reflects the material or component as well as the physical property of interest, a single conservative activation energy is used.

A distribution of activation energies (Figure 3D.D-1) was produced by EPRI (Reference 1) based on 170 materials. An independent review of materials used in Westinghouse-supplied equipment is summarized in Table 3D.D-1 and plotted in similar form in Figure 3D.D-2. A statistical analysis indicates that 95 percent of the activation energies exceed about 0.4 eV from the EPRI data and 0.6 eV from the Westinghouse data. Based on this information, a value of 0.5 eV is selected for use throughout this program whenever specific activation energies are not available. Employing a low value of activation energy in deriving the accelerated aging parameters causes materials having a high activation energy to be overaged with respect to the simulated conditions.

D.4 Thermal Aging (Normal/Abnormal Operating Conditions)

This section establishes the methodology employed and derives a typical set of accelerated aging parameters for equipment in various plant locations.

D.4.1 Normal Operation Temperature (T_0)

In determining the ambient operating temperature (T_0) of the component/material/equipment under investigation, the following is considered:

- External ambient temperature (T_a)
- Temperature rise in cabinet/enclosure (T_r)
- Self-heating effects (T_j)

where $T_o = T_a + T_r + T_j$

D.4.1.1 External Ambient Temperature (T_a)

a) For equipment located in areas supplied by an air-conditioning system, a typical value assumed for (T_a) throughout the qualified life is 68°F (20°C). For air-conditioning systems, two excursions per year to 91°F (33.3°C), each lasting 72 hours, has a negligible additional aging effect.

b) For equipment located in areas supplied by a ventilation system, a typical value assumed (T_a) throughout the qualified life is 77°F (25°C). Two excursions per year to 122°F (50°C), each lasting 72 hours, has a negligible additional aging effect.

D.4.1.2 Temperature Rise in Enclosure (T_r)

This temperature rise is estimated based on the heat generated (radiative and conductive) by equipment inside or attached to the enclosure. For example, limit switches may be affected by process heat through the valve. Temperatures measured during test runs may be available. A typical value for temperature rise inside an electronics cabinet is 10°C.

D.4.1.3 Self-Heating Effects (T_j)

For equipment that is energized during most of its life, a self-heating effect is measured or established. If the equipment is energized only for short durations, this effect may be determined to be negligible. Temperature effects due to the solenoid of an energized valve may be significant (over 40°C). In determining junction temperatures of semiconductor devices, known operating parameters along with the thermal impedance are used. If the power dissipation is not known, a 50 percent operating stress is assumed.

D.4.2 Accelerated Aging Temperature (T_i)

Temperatures used for actual accelerated thermal aging tests are determined based on the equipment or component specifications in an attempt to prevent damage from high temperature alone and second-order (non-Arrhenius) effects such as the glass transition temperature of plastics. A maximum of 130°C is typically used for electronic component aging, but this is evaluated on a case basis. If the device is energized during the accelerated aging process, the self-heating effect as determined in the preceding section is added to the oven temperature to determine the total aging temperature (T_i).

D.4.3 Examples of Arrhenius Calculations

D.4.3.1 For a Normally Energized Component Aged Energized – The Self-Heating Effect is Added to Both (T_o) and (T_i):

Conditions: T_a = 25°C, T_r = 10°C

T_j = 25°C, eV = 0.5,

Aging time = t_i

Oven temperature = 130°C

Qualified life goal = 10 years

Therefore T_o = 25 + 10 + 25 = 60°C = 333K

T_i = 130 + 25 + 155°C = 428K

$t_1 = 10e \; \frac{-0.5}{K} \frac{(428-333)}{(428 \times 333)}$ = 1831 hours

D.4.3.2 For a Normally De-energized Component Aged Energized – the Self-heating Effect is Added Only to T_i:

Conditions: T_a = 25°C, T_r = 10°C

T_j = 25°C, eV = 0.5, Aging time = t_1

Oven temperature = 130°C

Qualified life goal = 10 years

Therefore T_o = 25 + 10 = 35°C = 308K

T_i = 130 + 25 + 155°C = 428K

$$t_1 = 10e^{\frac{-0.5}{K}\frac{(428-308)}{(428\times 308)}} = 445 \text{ hours}$$

D.5 Post-Accident Thermal Aging

Most cases, some safety-related postaccident performance capability is specified by the functional requirements. As a consequence, to qualify equipment to IEEE 323, the effects of post-accident thermal aging must be simulated after the high-energy line break test. This section establishes the accelerated thermal aging parameters employed in performing this simulation.

D.5.1 Post-Accident Operating Temperatures

Assuming continuous operation of containment safeguards systems following an accident, the containment environment temperature is reduced to the external ambient temperature well within one year for any postulated high-energy line break. However, to allow for possible variations in plant operations following an accident, the limiting design high-energy line break envelope extending to one year is indicated by Figure 3D.5-8.

For safety-related equipment located inside containment, either the self-heating effects of the operating unit, under post-accident conditions, may be insignificant compared to the heat input from the external environment (transmitters, RTDs), or the unit may not be in continuous operation during this phase (valve operators). So it may not be necessary to add a specific temperature increment to account for self-heating of these devices following an accident. The portion of Figure 3D.5-8 that is not addressed by DBA testing is then input at T_0 into the Arrhenius equation to calculate appropriate accelerated aging parameters for post-accident conditions. However, as noted in Section D.4, if the equipment is energized during the aging simulation period, the self-heating effect is added to both T_o and T_i.

D.5.2 Accelerated Thermal Aging Parameters for Post-Accident Conditions

The aging temperature most often used for post-accident simulation is 250°F (121°C). This temperature is selected as a maximum for electronic components and is generally used for tests. Using this value and the conservative activation energy of 0.5 eV, the Arrhenius equation is applied to the curve in Figure 3D.5-8 from one day to four months or to one year in small increments of time. The required aging times to simulate these small increments are then summed to yield a total test time of 42 days to simulate four months and about 67 days to simulate one year post-accident operation. Including appropriate margin adds four and seven days respectively to the total test time.

If an activation energy of 0.8 eV is justified, the Arrhenius equation yields 19 days to simulate four months and 26 days to simulate one year with two days and three days margin to be included in the total test time.

D.6 References

1. EPRI NP-1558, Project 890-1, "A Review of Equipment Aging Theory and Technology," September 1980.

Table 3D.D-1 (Sheet 1 of 2)
Activation Energies From Westinghouse Reports

Material	Electron Volts
Melamine-Glass, G5	0.29
Epoxy B-725	0.48
Ester-Glass, GPO-3	0.57
RTV Silicone	0.60
Phenolic-Asbestos, A	0.61
Nylon 33 GF	0.70
Acetal	0.73
Mineral Phenolic	0.74
Silicone Varnish	0.74
Polypropylene	0.81
Polysulfone	0.83
Phenolic-Cotton, C	0.84
Formvar	0.85
Epoxy	0.88
Epoxy Adhesive	0.89
Nylon	0.90
Pressboard	0.91
Kapton	0.93
Silicone	0.94
Phenolic-Asbestos, A	0.94
Cast Epoxy	0.98
Urethane-Nylon	0.99
Phenolic-Glass, G-3	1.01
Polycarbonate	1.01
Phenolic-Paper, X	1.02
Epoxy Wire	1.05
Epoxy-Glass, FR-4	1.05
Varnish Cotton	1.06
PVC	1.08
Ester-Glass, GPO-1	1.09
Cellulose Phenolic	1.10
X-Link Ethylene	1.11
Urethane	1.12
Ester-Glass, GPO-2	1.13
Ester-Nylon	1.14

Table 3D.D-1 (Sheet 2 of 2)
Activation Energies From Westinghouse Reports

Material	Electron Volts
Ester-Glass, GPO-1	1.16
32102BK Varnish	1.16
Vulcanized Fiber	1.16
Cellulose Mineral Phenolic	1.17
Mylar	1.18
Cast Epoxy	1.18
32101EV Varnish	1.18
Epoxy	1.18
Silicone	1.18
Phenolic-Paper, XX	1.20
Vulanized Fiber	1.21
Cellulose Phenolic	1.24
Phenolic-Glass, G-3	1.24
Kraft Phenolic	1.25
Neoprene	1.26
Amide-Imide Varnish	1.31
Loctite 75	1.38
Acetyl. Cotton	1.39
Silicone-Asbestos	1.41
Epoxy-Glass, FR-4	1.50
Mylar	1.58
Nomex	1.59
Omega Varnish	1.59
Epoxy-Glass, G-11	1.64
Polythermaleze	1.64
Kraft Paper	1.67
Valox 310SE-0	1.75
Varnished Kraft	1.86
Nomex	1.91
Ester-Glass, GPO-3	2.03
Phenolic-Cotton, C	2.12
Melamine-Glass, G-5	2.18

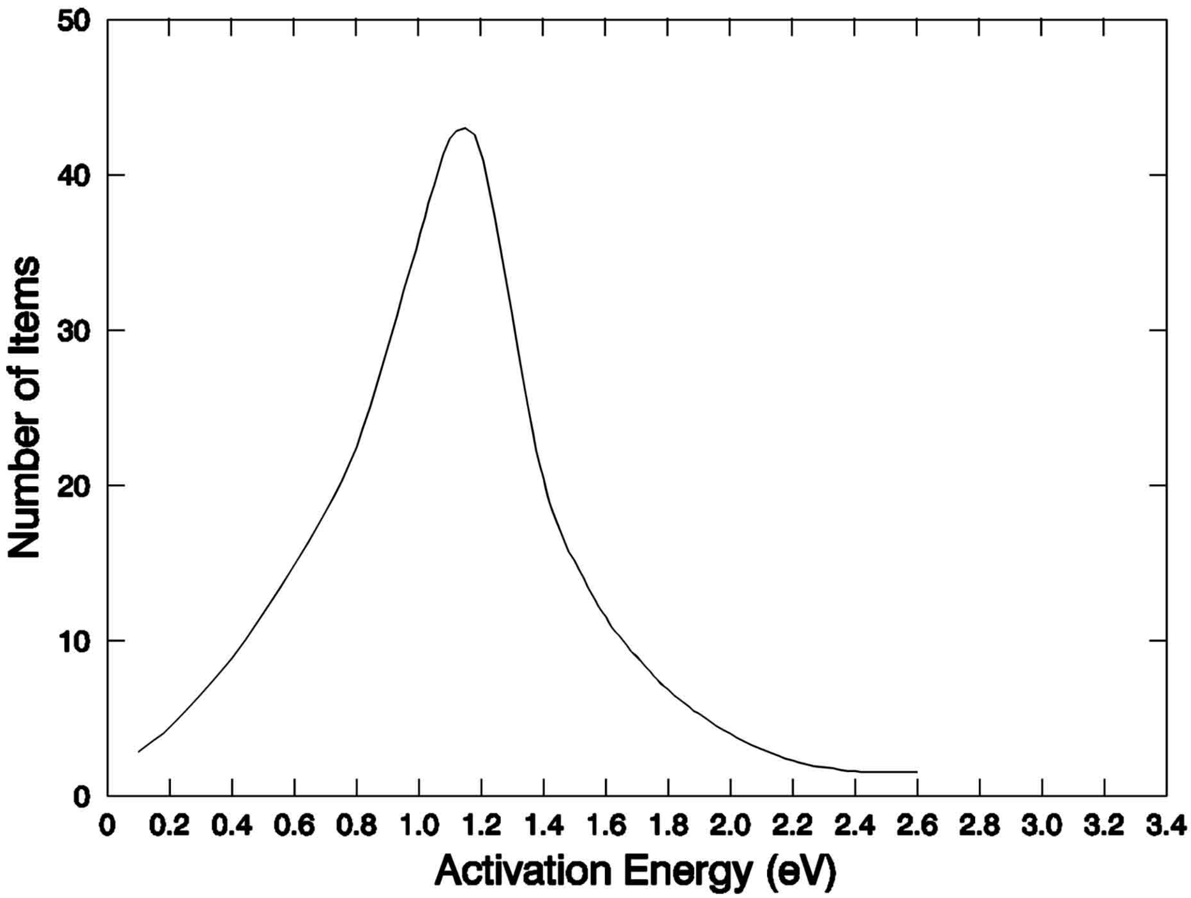


Figure 3D.D-1
Frequency Distribution of Activation Energies of Various Components/Materials (EPRI Data)

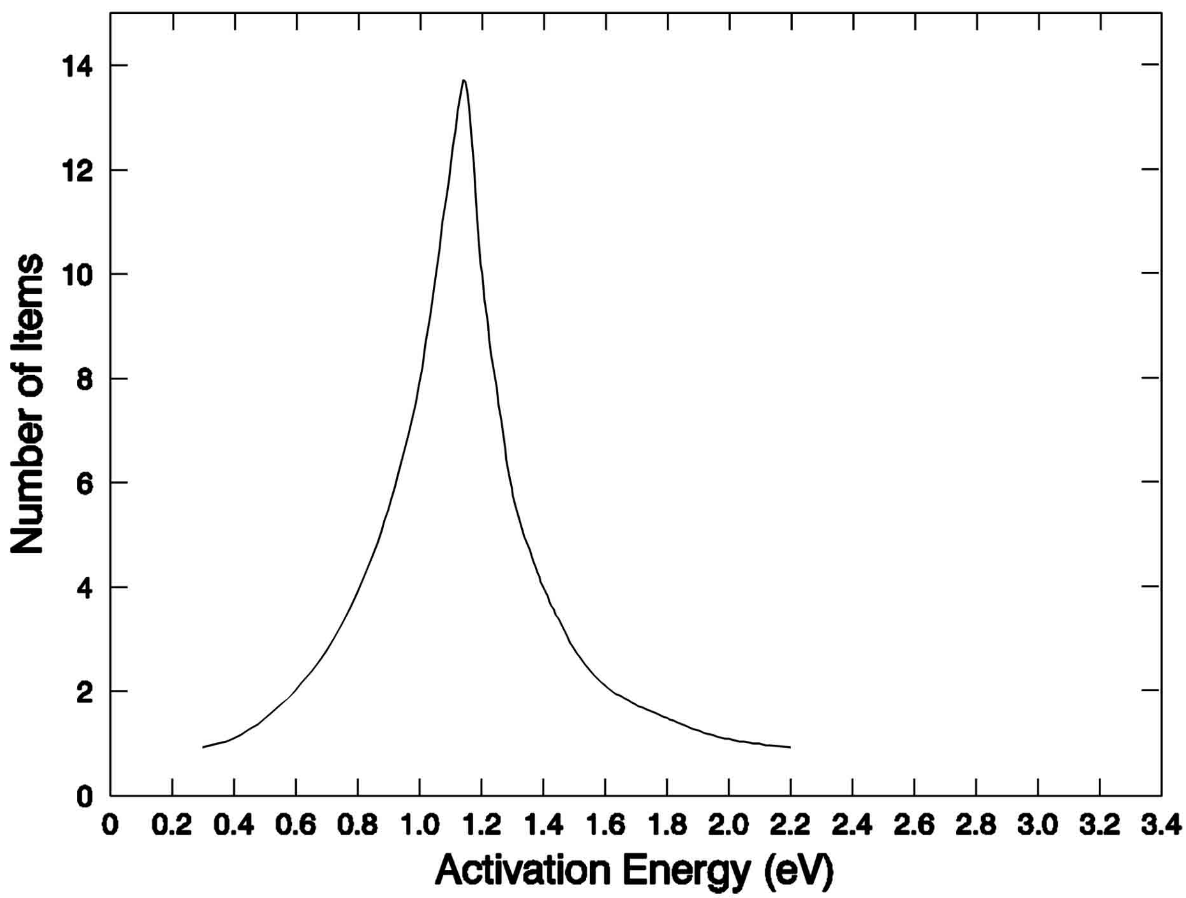


Figure 3D.D-2
Frequency Distribution of Activation Energies of Various Components/Materials
(Westinghouse Data)

Figure 3D.D-3
Not Used.

ATTACHMENT E

SEISMIC QUALIFICATION TECHNIQUES

E.1 Purpose

The following is the methodology used to seismically qualify seismic Category I mechanical and electrical equipment for the AP1000 equipment qualification program. Qualification work covered by this appendix meets the applicable requirements of IEEE 344-1987 and 382-1996.

E.2 Definitions

The following are definitions of terms unique to or distinct from common industry usage. (See Section E.4.2.)

E.2.1 1/2 Safe Shutdown Earthquake

The 1/2 safe shutdown earth (SSE) is the earthquake level used during seismic testing to seismically age safety-related equipment before performing safe shutdown earthquake testing.

E.2.2 Seismic Category I Equipment

Seismic Category 1 equipment consists of structures, systems, and components required to withstand the effects of the safe shutdown earthquake and remain structurally intact, leak-tight (in case of pressurized systems), and functional to the extent required to perform their safety-related function.

E.2.3 Seismic Category II Equipment

Seismic Category II equipment is that equipment whose continued function is not required, but whose failure could reduce the functioning of seismic Category I structures, systems, and components to an unacceptable level. Seismic Category II equipment must be capable of maintaining structural integrity so that a seismic event up to and including an SSE would not cause such a failure.

E.2.4 Non-seismic Equipment

Equipment designated as non-seismic does not require seismic qualification.

E.2.5 Active Equipment

Equipment that must perform a mechanical or electrical operation during or after (or both) the safe shutdown earthquake in order to accomplish its safety-related function.

E.2.6 Passive Equipment

Equipment where maintenance of structural or pressure integrity is the only requirement necessary for accomplishing its safety-related function.

E.3 Qualification Methods

This section presents a general description of the seismic qualification methods used by AP1000 for the seismic qualification of seismic Category I safety-related mechanical and electrical equipment. Three methods are used: test, analysis, and a combination of the two. The approaches for qualification by testing and by analysis are discussed in Section E.5 and Section E.6, respectively. The following discussion covers the conditions under which each approach is used and the general

requirements applicable to the use of the methods. The qualification sequence is defined in Appendix 3D.

E.3.1 Use of Qualification by Testing

The preferred method for seismic qualification of safety-related Class 1E electrical and electromechanical equipment is seismic testing. The nature of the seismic and vibrational input used depends on where the equipment is used. For equipment mounted so that the seismic environment includes frequency content between 1 and 33 hertz (hard mounted), the seismic test input is multifrequency. For equipment mounted so that seismic ground motion is filtered to contain one predominant structural mode (line mounted), single frequency testing is appropriate. This is the case for equipment mounted on piping systems, ductwork, or cable trays.

E.3.2 Use of Qualification by Analysis

Analysis is used for seismic qualification when one of the following conditions is met:

- The equipment is too large or the interface support conditions cannot adequately be simulated on the test table.

- The only requirement is to maintain structural integrity during a postulated seismic event.

- The equipment represents a linear system, or the nonlinearities can conservatively be accounted for in the analysis. This approach is also applicable to the development of the seismic environment, required response spectrum curve, at the mounting location of a component attached to a larger structure when the device is seismically qualified by separate component testing.

- The analysis is used to document the seismic similarity of the equipment provided and that previously qualified by testing.

Seismic qualification of safety-related electrical equipment by analysis alone is not recommended for complex equipment that cannot be modeled to adequately predict its response. Analysis without testing may be acceptable provided structural integrity alone can ensure the design-intended function.

E.4 Requirements

E.4.1 Damping

Damping level of a component or system describes its capability to dissipate vibrational energy during a seismic event. The damping level used defines the response magnitude of an ideal single degree of freedom linear oscillator when subjected to the specified input as documented by the required response spectrum (RRS) curve. The significance of the damping value used depends on whether qualification is by testing or analysis.

E.4.1.1 Testing

Equipment qualification by testing involves subjecting the base of the equipment to a representative seismic acceleration time history. The response characteristics of the equipment are a function of the inherent damping present in the equipment. In this case the damping value used (typically five percent) serves as a convenient means of showing the compliance of the test response spectrum (TRS) with the required response spectrum.

E.4.1.2 Analysis

In the case of qualification by analysis, the damping level used is representative of the damping actually present in the equipment. Unless other documented equipment damping data is available, the values specified in Table 3.7.1-1 of Chapter 3 are used.

E.4.2 Interface Requirements

As part of the seismic qualification program, consideration is given to the definition of the clearances needed around the equipment mounted in the plant to permit the equipment to move during a postulated seismic event without causing impact between adjacent pieces of safety-related equipment. This is done as part of seismic testing by measuring the maximum dynamic relative displacement of the top and bottom of the equipment.

When performing qualification by analysis, the relative motion is obtained as part of the analytical results. These motions are reported in the qualification report and are used to determine the required clearance between adjacent pieces of equipment.

In addition, the qualification program takes into account the restraining effect of other interfaces, such as cables and conduits attached to the equipment, which may change the dynamic response characteristic of the equipment.

E.4.3 Mounting Simulation

The mounting conditions simulated by analysis or during seismic test are representative of the equipment as-installed mounting conditions used for the AP1000 equipment. When an interfacing structure exists between the safety-related equipment being qualified and the floor or wall at which the equipment mounting required response spectrum is specified, its flexibility is simulated as part of the qualification program. If this is not done, justification must be provided, demonstrating that the deviations in mounting conditions do not affect the applicability of qualification program.

E.4.4 1/2 Safe Shutdown Earthquake

The AP1000 makes use of a small earthquake having the intensity of one-half of the safe shutdown earthquake at the safety-related equipment mounting location to simulate the fatigue effects of smaller earthquakes that may occur before the postulated safe shutdown earthquake. These small earthquakes correspond to the operating basis earthquakes (OBEs) referenced in IEEE 344-1987. When qualification by testing is used, five of these small earthquakes are used to vibrationally age the equipment before the safe shutdown earthquake. When qualification by analysis is used, two safe shutdown earthquake events are used to simulate the fatigue aging effects. Each event contains 10 peak cycles. These stress cycles are used to verify that the equipment is not subject to failure due to low cycle fatigue.

E.4.5 Safe Shutdown Earthquake

The safe shutdown earthquake required response spectrum curve defines the seismic qualification basis for each piece of safety-related equipment. The seismic level varies according to the mounting location of the equipment. When equipment qualification is based on testing, an additional 10 percent test acceleration margin is added as specified in IEEE 323-1974.

E.4.6 Other Dynamic Loads

Hydrodynamic loads are considered as part of the qualification program, where applicable.

E.5 Qualification by Test

Seismic qualification testing is the preferred method for electrical, mechanical, and electromechanical equipment. Seismic testing shall be performed and input generated as specified in IEEE 344-1987. The nature of the test input used depends on whether the equipment is hard mounted or line mounted. The test program consists of the following elements, as applicable: environmental aging, mechanical aging, vibrational aging, and safe shutdown earthquake testing. For those cases where the equipment is also subject to a loss of coolant or a high-energy line break accident, these accidents are simulated on the same qualification specimen after completion of the testing previously discussed. (See Subsections 3D.4.4 and 3D.7.4.)

The characteristics of the required seismic and dynamic input motions should be specified by the response spectrum or time history methods. These characteristics, derived from the structures or systems seismic and dynamic analyses, should be representative of the input motions at the equipment mounting locations.

For seismic and dynamic loads, the actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (that is, the test response spectrum should closely resemble and envelop the required response spectrum over the critical frequency range).

Since seismic and the dynamic load excitation generally have a broad frequency content, multi-frequency vibration input motion should be used. However, single frequency input motion, such as sine beats, is acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (for example, by structural filtering effects), or that the anticipated response of the equipment is adequately represented by one mode, or in the case of structural integrity assurance, that the input has sufficient intensity and duration to produce sufficiently high levels of stress for such assurance. Components that have been previously tested to IEEE-344-1971 should be reevaluated or retested to justify the appropriateness of the input motion used, and requalified if necessary.

For the seismic and dynamic portion of the loads, the test input motion should be applied to one vertical axis and one principal axis (or two orthogonal axes) simultaneously unless it can be demonstrated that the equipment response motion in the horizontal direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

E.5.1 Qualification of Hard-Mounted Equipment

Hard-mounted equipment is seismically tested mounted on a test table capable of producing multifrequency, multiaxis inputs. The waveform characteristics of the input are random and scaled in such a way that the test response spectrum equals or exceeds the required response spectrum (including margin). The input signal meets the requirements of Subsection 7.6.3 of IEEE 344-1987.

Furthermore, the test input simulates the multidirectional nature of the earthquake. The preferred method for meeting this requirement is to the use a triaxial test table capable of producing three statistically independent, orthogonal input motions. In this case the seismic testing consists of five 1/2 safe shutdown earthquake tests and one safe shutdown earthquake test in one orientation.

Using a biaxial test table is acceptable if it is justified that the horizontal and vertical test inputs conservatively simulate the three-dimensional nature of the seismic event. One acceptable approach

is to mount the equipment on the test table with its front-to-back axis oriented at 45 degrees to the horizontal drive axis and scale the horizontal component of the input by a factor of the square root of two. Statistically independent inputs are preferred and, if used, the test can be performed in two stages, with the equipment rotated 90 degrees about the vertical axis. In this case, the five 1/2 safe shutdown earthquake inputs need to be applied only in the first orientation.

If a dependent biaxial test table is used, the test is performed in four stages. The first stage involves five 1/2 safe shutdown earthquake tests and one safe shutdown earthquake test in the first orientation. The second, third, and fourth orientations are obtained by successively rotating the equipment 90 degrees clockwise from its previous position. One safe shutdown earthquake test is performed in each of the last three orientations.

Each multifrequency test has a minimum of 15 seconds of strong motion input. The strong motion portion is preceded and followed by a period of testing where the test input is ramped up and ramped down, respectively, so that the equipment is not subjected to impact loading. The adequacy of each test run is evaluated using the criteria set forth in Subsection 7.6.3.1 of IEEE 344-1987.

E.5.2 Qualification of Line-Mounted Equipment

Line-mounted equipment, because of the dynamic filtering characteristics of its mounting, is effectively subject to single frequency input. This condition is common for valves and sensors supported by piping systems, cable trays, and duct systems. This equipment is qualified consistent with the requirements of IEEE 382-1996.

In some cases this equipment may also be used in the hard-mounted condition. In this case multifrequency, multiaxis testing is also required unless justification is provided that the previous single frequency tests demonstrate the capability of the equipment to operate under the hard-mounted seismic conditions. Because of the large size of typical valves, it may be necessary to perform separate testing of the operators and valve assembly.

E.5.2.1 Seismic Qualification Test Sequence

The seismic qualification process is broken down into the following steps:

1. Mount the equipment on a rigid test fixture and perform a resonant search test to demonstrate that the equipment is structurally rigid (fundamental frequency greater than 33 hertz) and does not amplify the seismic motions acting at the equipment mounting interface.

2. Perform single frequency testing on the line-mounted equipment.

3. Perform multifrequency, multiaxis testing on the equipment, if appropriate.

4. If an active valve assembly is to be seismically qualified, additional testing is needed as follows:

    a. Perform a static pull test on the valve.

    b. Perform a static seismic analysis using a verified model of the valve and its extended structure to demonstrate that the valve has adequate structural strength to perform its safety-related function without exceeding the design allowable stresses specified in ASME Code, Section III, Subsection NB, NC, or ND for pressure-retaining parts, as

appropriate, and Subsection NF for non-pressure-retaining boundary parts. Limiting extended structure stress to material yield strength minimizes deflections, which could interfere with valve stroke function.

E.5.2.2 Line Vibration Aging

Line-mounted equipment may be subject to operational vibrations resulting from normal plant operations. The potential fatiguing effect of this vibrational aging is simulated as part of the qualification program. This requirement is satisfied by subjecting the equipment to a sine sweep from 5 to 100 to 5 hertz at an acceleration level of 0.75g or such reduced acceleration at low frequencies to limit the double amplitude to 0.025 inch as specified in Section 5.3.a, Part III of IEEE 382-1996.

E.5.2.3 Single Frequency Testing

The single frequency testing acceleration waveform is either sine beat or sine dwell applied at one-third octave frequency intervals as specified in IEEE 382-1996. Each dwell has a time length adequate to permit performance of functional testing, with a minimum time of 15 seconds. To account for the three-dimensional nature of the seismic event, the test input level is taken as the square root of two times the required input motion (RIM) level specified in IEEE 382. The level includes the 10 percent test margin. Each test series is performed using single axis input. The test series is performed successively in each of three orthogonal axes.

E.5.2.4 Seismic Aging

The aging effect of the five 1/2 safe shutdown earthquake earthquakes can be simulated by exposing the equipment to two sinusoidal sweeps at one-half of the safe shutdown earthquake required input motion level in each orthogonal axis. Each sweep shall go from 2 to 35 hertz to 2 hertz at a rate not to exceed one octave per minute. One sweep is performed with the equipment in its inactive mode, and the other with the equipment in its safety-related operational mode.

E.5.2.5 Static Deflection Testing of Active Valves

The seismic testing just discussed is normally performed only on the valve operator and the attached appurtenances. If the valve assembly is rigid, the operability of the valve assembly during a postulated seismic event may be demonstrated by performing a static pull test using a peak acceleration value equivalent to a triaxial acceleration of 6g. If the valve assembly is determined to be flexible, a supplemental analysis of the seismic response of the flexible valve and its supporting piping is performed to determine the actual acceleration level present at the center of gravity of the valve assembly.

The valve is placed in a suitable test fixture with the operator and appurtenances mounted and oriented as in the normal valve assembly installation. The valve is mounted so that the extended structure is freestanding and supported only by the valve nozzles. The valve is positioned so that the horizontal and vertical load components simulating the three-dimensional nature of the seismic event produce a worst-case stress condition in the valve extended structure.

During testing, the valve shall be internally pressurized and nozzle loads applied. Static loads simulating dead weight and seismic loads are applied to the extended structure. The tests are normally performed at ambient temperature. These loads simulate to the extent feasible the load distribution acting on critical parts of the valve assembly. The valve is actuated using the actuator system seismically qualified according to IEEE 382-1996. The valve assembly is cycled from its normal to the desired safety-related position within the time limits defined in the equipment specification. Leakage measurements are made, where required, and compared to the allowable values specified in the valve design specification.

E.5.3 Operational Conditions

When equipment being qualified performs a safety-related function during the safe shutdown earthquake, the equipment is operated and monitored to demonstrate that the equipment functions properly before, during, and after the seismic event. If the test time is not long enough to complete the required functional tests, the length of the strong motion test time is increased to permit completion of the required functional testing.

Where functional testing is dependent on external electrical supply, the testing is performed using the worst-case electrical supply conditions.

E.5.4 Resonant Search Testing

Resonant search testing is performed to provide data on the natural frequency and dynamic response characteristics of the equipment qualified. For hard-mounted equipment being qualified by seismic testing, resonant search testing is done to provide additional information but is not required for qualification of the equipment. This is an important consideration because frequency testing for hard-mounted equipment is normally performed with the equipment mounted on the test table, where dynamic interaction of the table and the equipment has a significant effect on the measured natural frequency.

For qualification of line-mounted valve assemblies, it is necessary that the assemblies be rigid. To meet this requirement, the assembly mounted to a rigid test fixture so that the frequencies measured are indeed representative of the valve assembly. If it is not feasible to provide a rigid fixture, as is likely the case when testing such very large valves, as the main steam and feedwater isolation valves, additional tests and analyses may be required to determine if the apparent flexibility measured is due to the test fixture or to the characteristic of the valve assembly itself.

If the resonant search test data is being generated to verify the accuracy of an analytical modeling technique, the test specimen mounting details must accurately simulate the boundary conditions used in the analytical model.

E.6 Qualification by Analysis

Section E.3.2 defines the limits on the use of analysis to demonstrate seismic qualification of safety-related equipment. The following sections describe the analytical methods to be employed for qualification of equipment. There are two techniques, static and dynamic, used to qualify equipment. The success of either method depends on the ability of the analytical model to describe the response of the system to seismic loads. Alternative methods of analysis are accepted if their conservatism is documented.

The analysis is used to demonstrate the structural adequacy of the equipment being qualified. This is done by showing that the calculated stresses do not exceed the design allowable stresses specified in ASME Code, Section III, Subsection NB, NC, or ND for pressure-retaining equipment and Subsection NF for nonpressure-retaining equipment.

E.6.1 Modeling

Analysis may be performed by hand calculations, finite element, or mathematical models that adequately represent the mass and stiffness characteristics of the equipment. The model contains enough degrees of freedom to adequately represent the dynamic behavior over the frequency range of interest. It includes the essential features of the equipment.

Dynamic properties reflect the in-service operating conditions, such as structural coupling, dynamic effects of contained liquids, and externally applied restraints (where appropriate). Where the modeled equipment exhibits some nonlinear behavior, this nonlinearity is modeled unless justification is provided that it is insignificant or that the linear model provides conservative results. The adequacy of the model or of the modeling techniques is shown by comparing the predicted responses to the responses predicted by benchmark problems or modal testing. Acceptable benchmark problems include hand calculations, analysis of the same problem using a comparable verified public-domain program, empirical data, or information from the technical literature.

In addition to documenting the modeling technique, a quality assurance program is in place that defines the requirements for the control, verification, and documentation for the computer programs used for qualification of safety-related equipment. The computer programs used in the qualification process are verified on the same computer on which the qualification analysis is performed.

E.6.2 Qualification by Static Analysis

For rigid equipment, the seismic forces resulting from one seismic input direction are calculated for each node point by multiplying the nodal mass in that direction by the appropriate zero period acceleration (ZPA) floor acceleration. The combined system response of the equipment to the simultaneous loads acting in all three directions is calculated by combining the three components, using the square root sum of the squares (SRSS) method. The square root sum of the squares method is used to account for the statistical independence of the individual orthogonal seismic components.

E.6.3 Qualification by Dynamic Analysis

If the lowest natural frequency of the equipment lies below the cutoff frequency, the response of the equipment to the seismic event in each orthogonal direction will be dynamically amplified and the equipment is said to be flexible. The analysis is performed in compliance with the guidelines set forth in the SSAR and in Regulatory Guides 1.92, 1.100, and 1.122.

The preferred method of analysis is the response spectrum method. In this method the responses in each equipment mode are calculated separately and combined by the square root sum of the squares method, provided the modes are not closely spaced. (Consecutive modes are said to be closely spaced if their frequencies differ from that of the first mode in the group by less than 10 percent.) The responses for each mode in a group are combined absolutely. The group response is then combined with the remaining modal responses using the square root sum of the squares method. The responses for each of the three orthogonal seismic components can then be combined as discussed in Section E.6.2. The applicable damping levels are noted in Table 3.7.1-1 of Chapter 3.

E.6.3.1 Response Spectrum Analysis

Modes up to and including the cutoff frequency are included in this summation. In some cases, the structure is basically rigid, with some of the flexible mode representing local effects. This situation is evaluated by reviewing the modal masses applicable to a given seismic input direction. If the sum of the effective modal masses used in the response spectrum analysis is greater than 0.9 times the total equipment mass, the model is assumed to adequately represent the total equipment mass. If this criterion is not satisfied, it means that a significant part of the equipment seismic response is due to the static seismic response of the higher equipment modes (above the cutoff frequency). If this situation occurs, the analyst determines the component of the response due to the higher modes and combines it with the flexible response component by square root sum of the squares. (This requirement is discussed in the SSAR, Subsection 3.7.2.)

E.6.3.2 Static Coefficient Method

As an alternative to the response spectrum method, the static coefficient method of analysis may be used. In this method the frequencies of the equipment are not determined, but a static analysis is performed, assuming that a peak acceleration equal to 1.5 times the peak spectral acceleration given in the applicable required response spectrum acts on the structure as described in Section E.6.2.

The static coefficient of 1.5 takes into account the combined effects of multifrequency excitation and multimode response for equipment and structures that can be represented by a simple model. A lower static coefficient may be used when it can be demonstrated that it will yield conservative results.

E.6.3.3 Time History Analysis

The time-history method of analysis is the preferred method of analysis when the equipment exhibits significant nonlinear behavior or when it is necessary to generate response spectra for specific component mounting locations in the equipment. The acceptable methods that are used to develop the seismic time histories are discussed in Regulatory Guide 1.122, ASME Code, Section III, Appendix N, and in Section 6.2 of IEEE 344-1987. Other analytical methods may be used to generate in-equipment response spectra provided that they are verified to produce accurate and/or conservative results.

E.7 Qualification by Test Experience

This method of qualification is not used.

E.8 Performance Criteria

E.8.1 Equipment Qualification by Test

The performance criterion for qualification of equipment is that the equipment successfully perform its safety-related function during and after the postulated seismic event. Acceptance requires, as a minimum, that:

- No spurious or unwanted outputs occur in the circuits that could impair the safety-related functional operability of the equipment;
- No gross structural damage of the equipment occur during the seismic event that could lead to the equipment or any part thereof becoming a missile. Local inelastic deformation of the equipment is permitted; and,
- Satisfactory completion of specified baseline tests are demonstrated before, during, and after the seismic test sequence.

E.8.2 Equipment Qualification by Analysis

E.8.2.1 Structural Integrity

The analysis verifies that the equipment, when subjected to the worst case combination of operating and seismic loads, maintains its structural integrity. In addition the analysis shows that the equipment is not subject to low cycle fatigue failure when subject to postulated seismic loading. Finally the analysis verifies that seismically induced equipment motion does not lead to impacting with other nearby equipment.

E.8.2.2 Operability

Analysis can be used to demonstrate equipment operability for those pieces of equipment where structural integrity or limitation of deformation guarantees operability. As an example the analysis of active equipment verifies that the combination of operating and postulated seismic loads do not produce stress levels or deformations that exceed established functional limits. The rationale for use of these limits is justified.

Appendix 3E High-Energy Piping in the Nuclear Island

This appendix identifies high-energy piping in the nuclear island with a diameter larger than 1 inch. Candidate leak-before-break piping is identified in Figures 3E-1 through 3E-5 along with other piping for which high-energy pipe failures are postulated. These figures also identify piping in the break exclusion zones inside and outside containment. These figures do not include piping of 1 inch size and smaller. Instrumentation and instrumentation lines are not included.

The selection of the failure type is based on whether the system is high or moderate energy during normal operating conditions of the system. High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig. Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high energy are considered moderate energy. Piping systems that exceed 200°F or 275 psig for 2 percent or less of the time during which the system is in operation or that experience high-energy pressures or temperatures for less than 1 percent of the plant operation time are considered moderate energy. In piping whose nominal diameter is greater than 1 inch but less than 4 inches, only circumferential breaks are postulated at each selected location. No breaks are postulated for piping whose nominal diameter is 1 inch or less.

The three-letter code included in the line numbering identifies the pipe specification. The letters define the pressure class, material specification, and AP1000 equipment classification, respectively. The symbols used in Figures 3E-1 through 3E-5 are the same as the P&ID figures. See Figure 1.7-2 for additional information on the drawing legend and for the key for the pipe specification. Section 3.2 includes additional information on the AP1000 equipment classification.

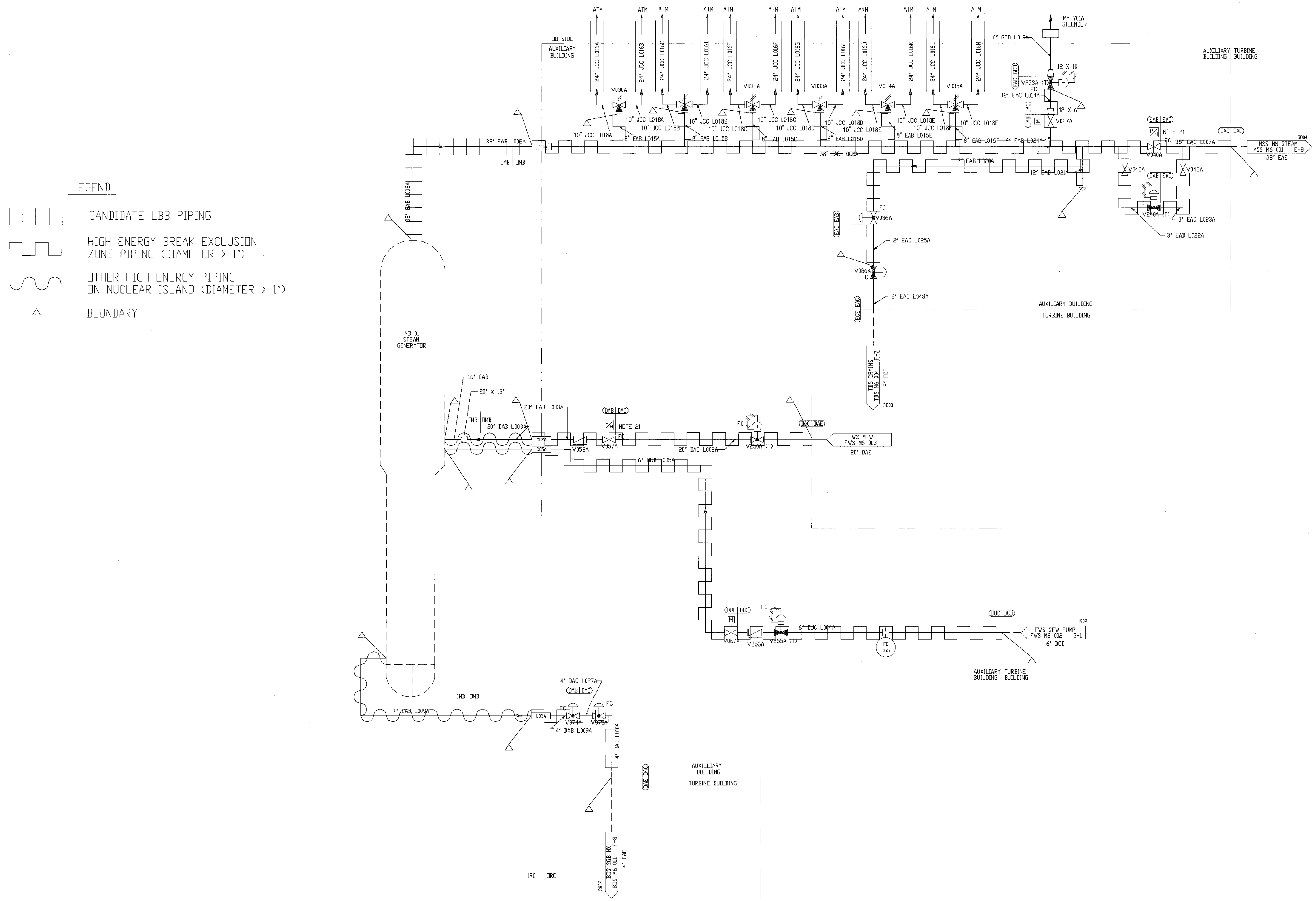


Figure 3E-1 (Sheet 1 of 2)
High Energy Piping – Steam Generator System

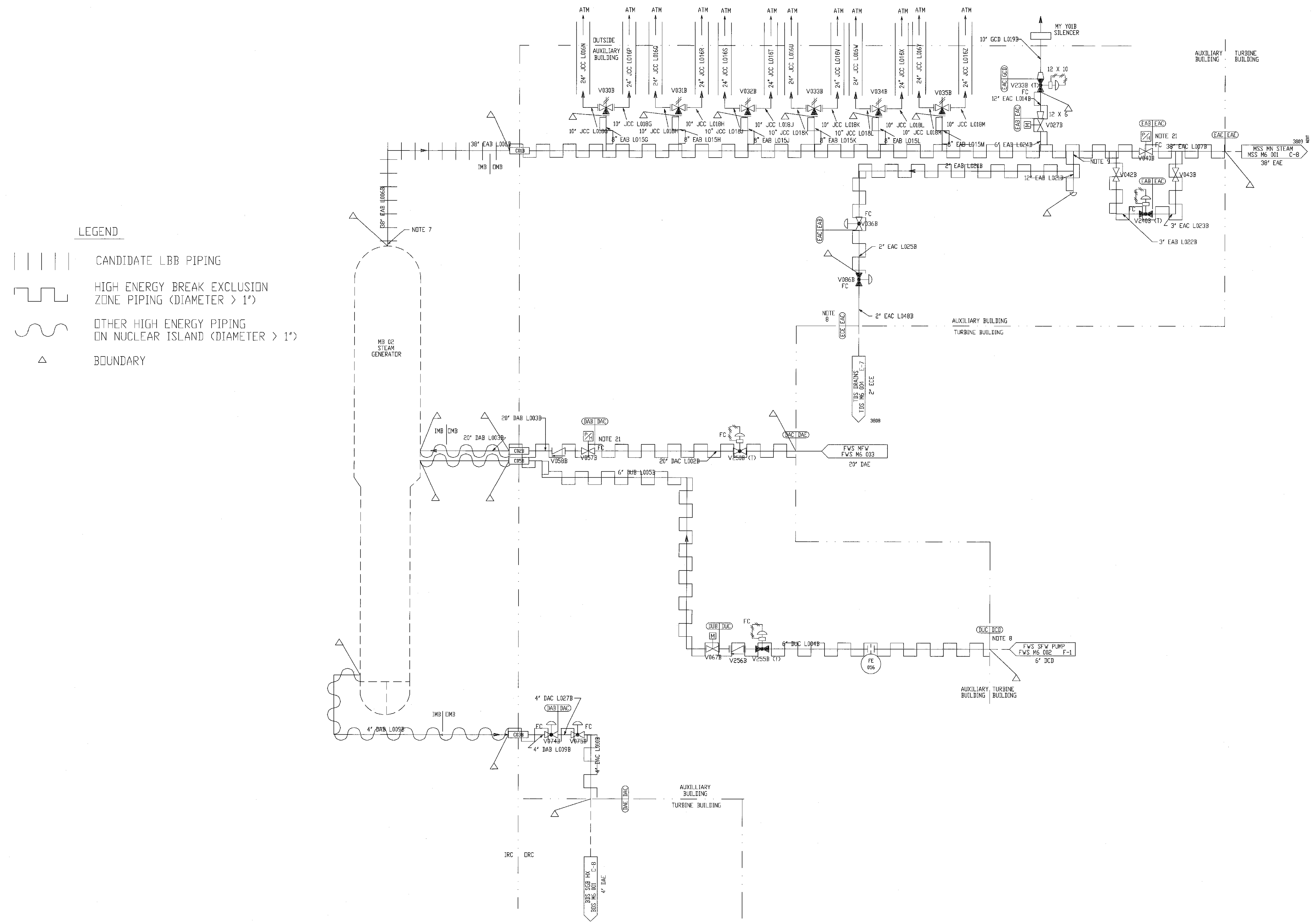


Figure 3E-1 (Sheet 2 of 2)
High Energy Piping – Steam Generator System

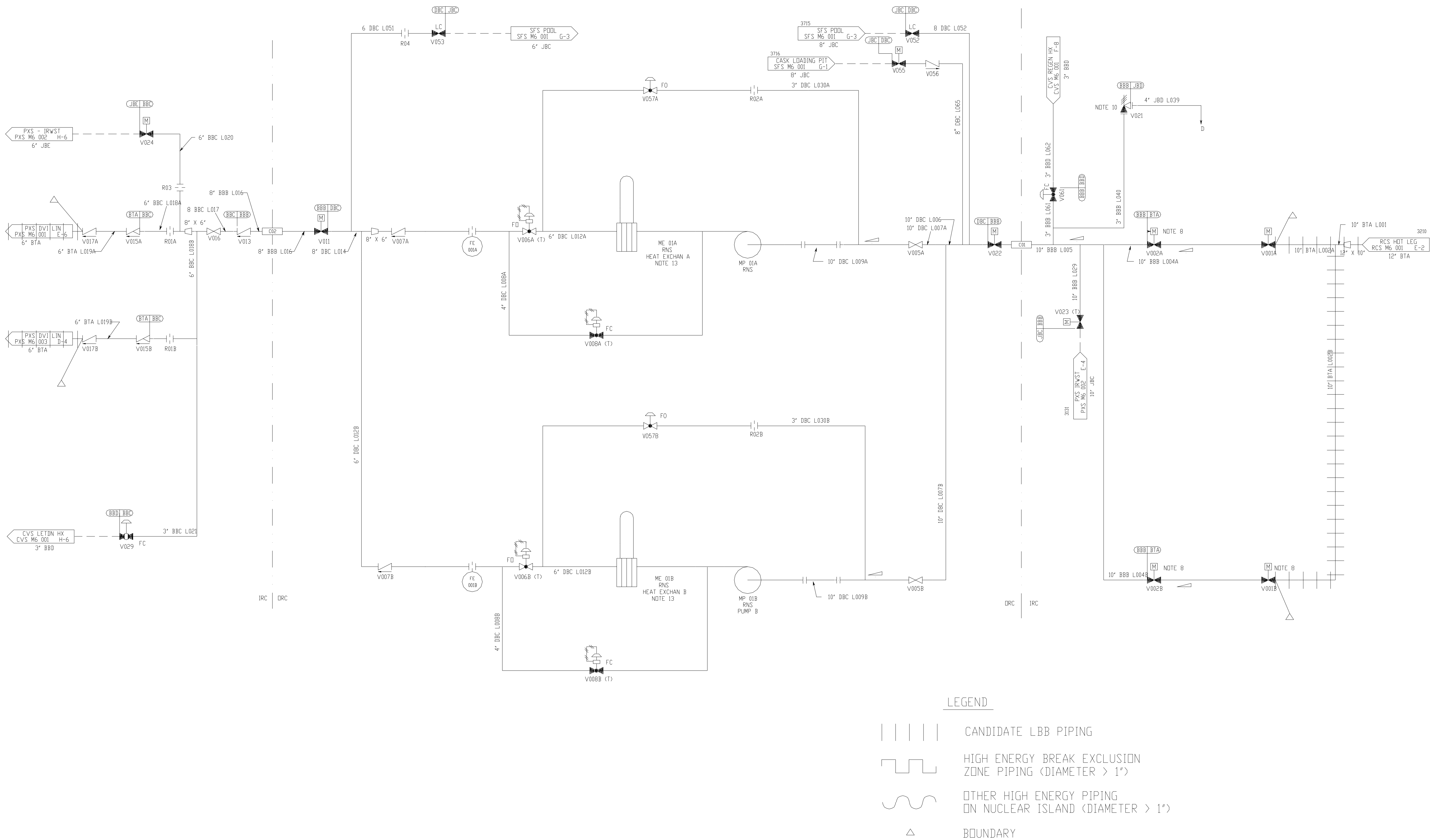


Figure 3E-2
High Energy Piping – Normal Residual Heat Removal System

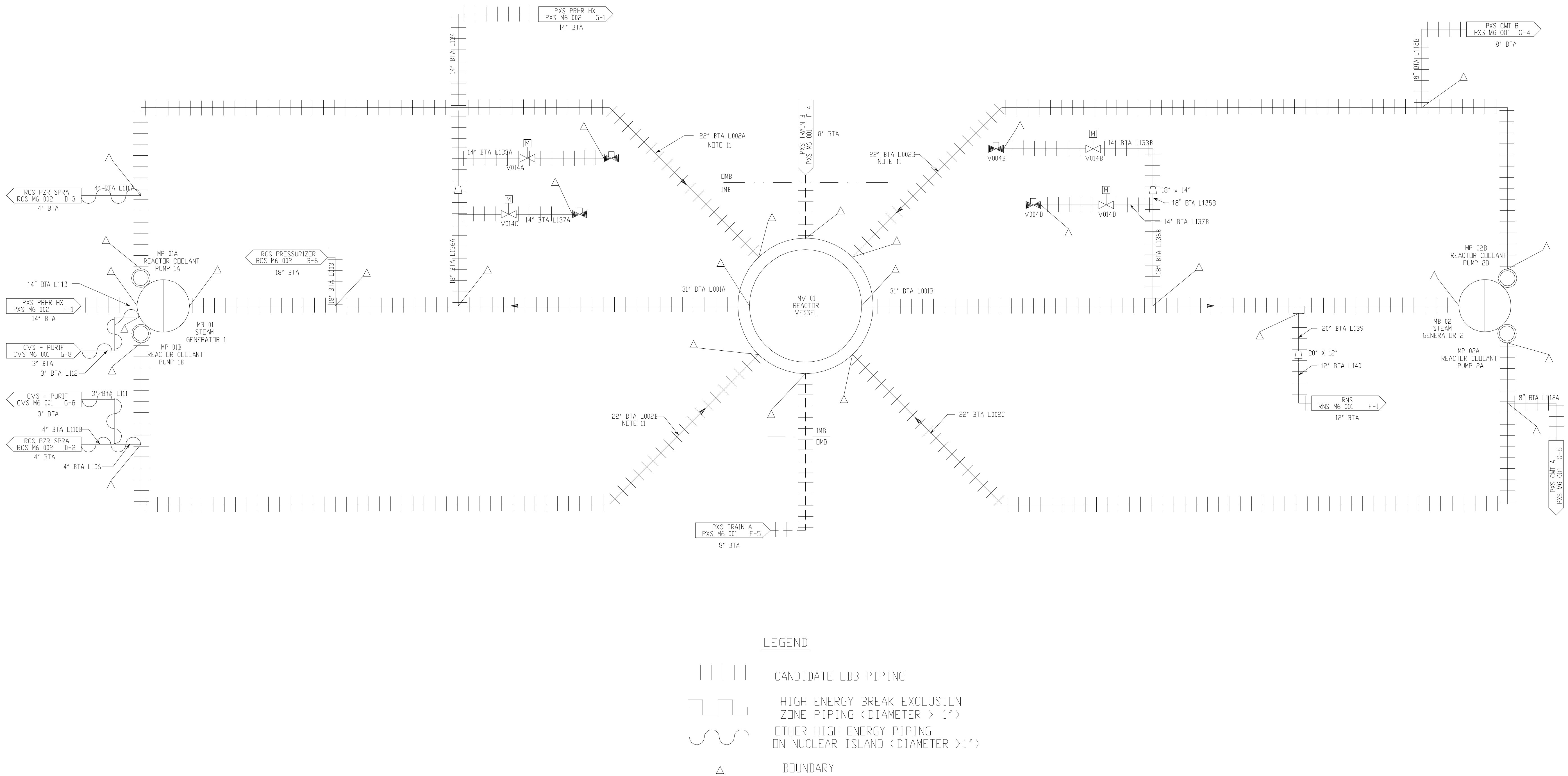


Figure 3E-3 (Sheet 1 of 2)
High Energy Piping – Reactor Coolant System

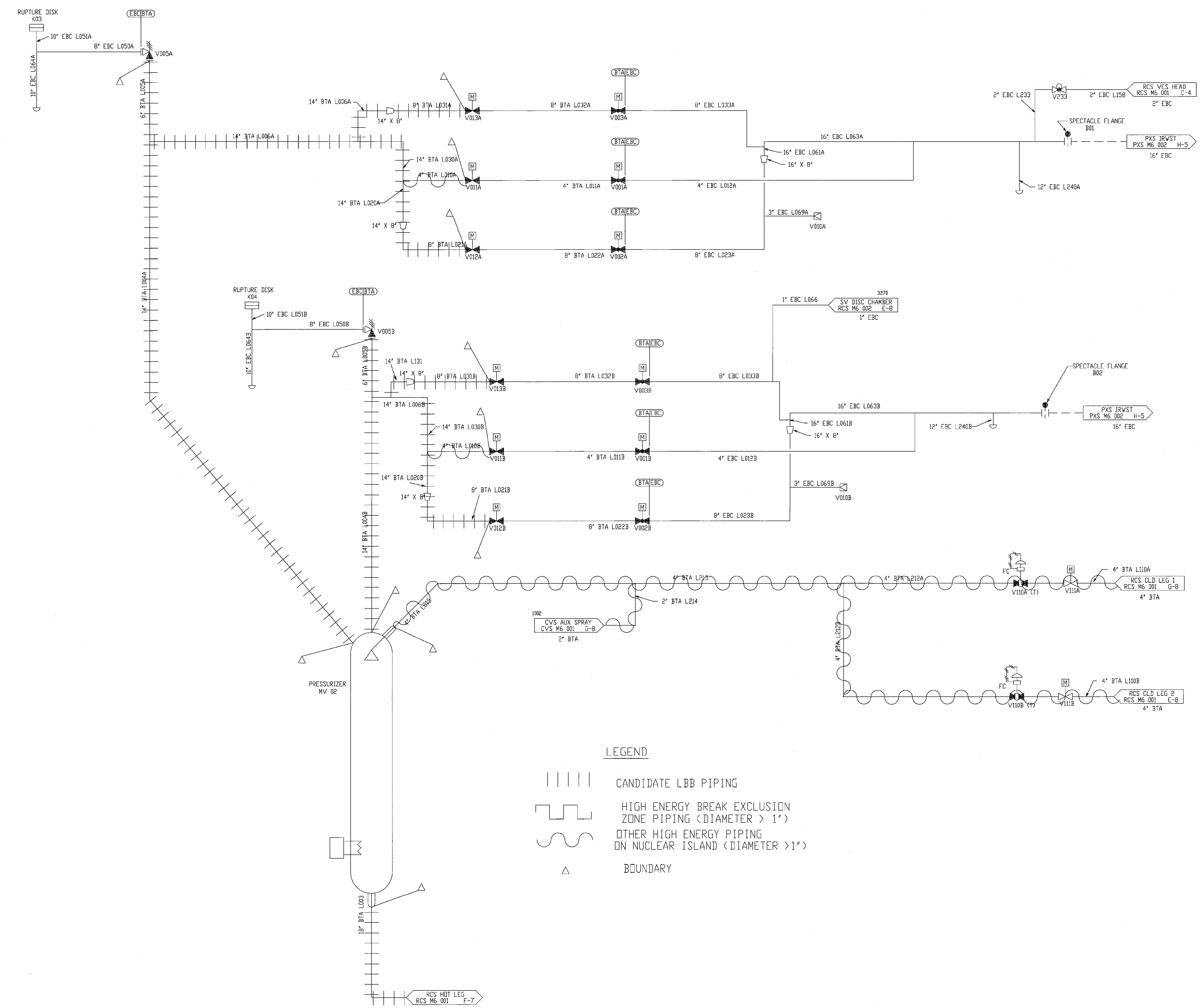


Figure 3E-3 (Sheet 2 of 2)
High Energy Piping – Reactor Coolant System

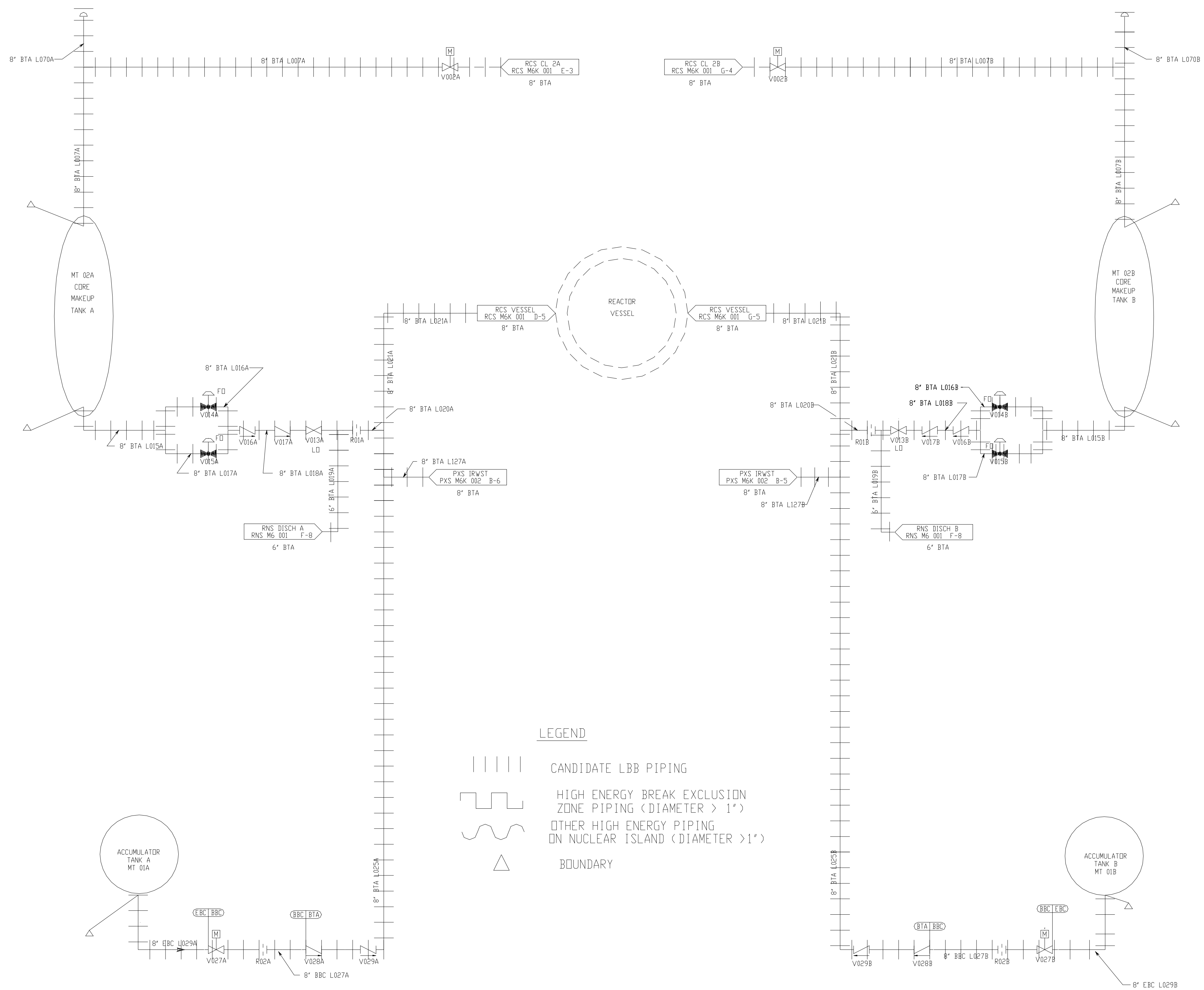


Figure 3E-4 (Sheet 1 of 2)
High Energy Piping – Passive Core Cooling System

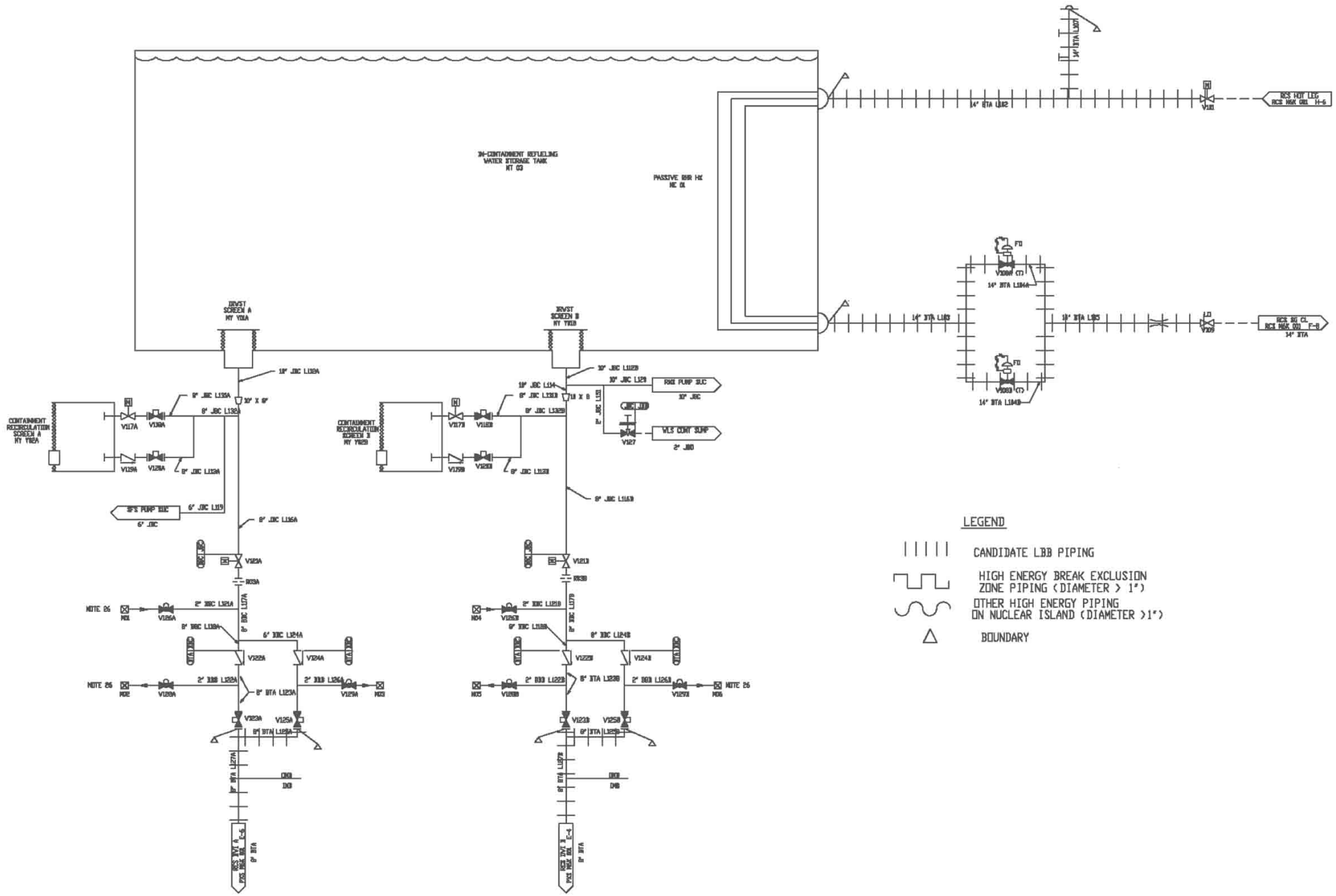


Figure 3E-4 (Sheet 2 of 2)
High Energy Piping – Passive Core Cooling System

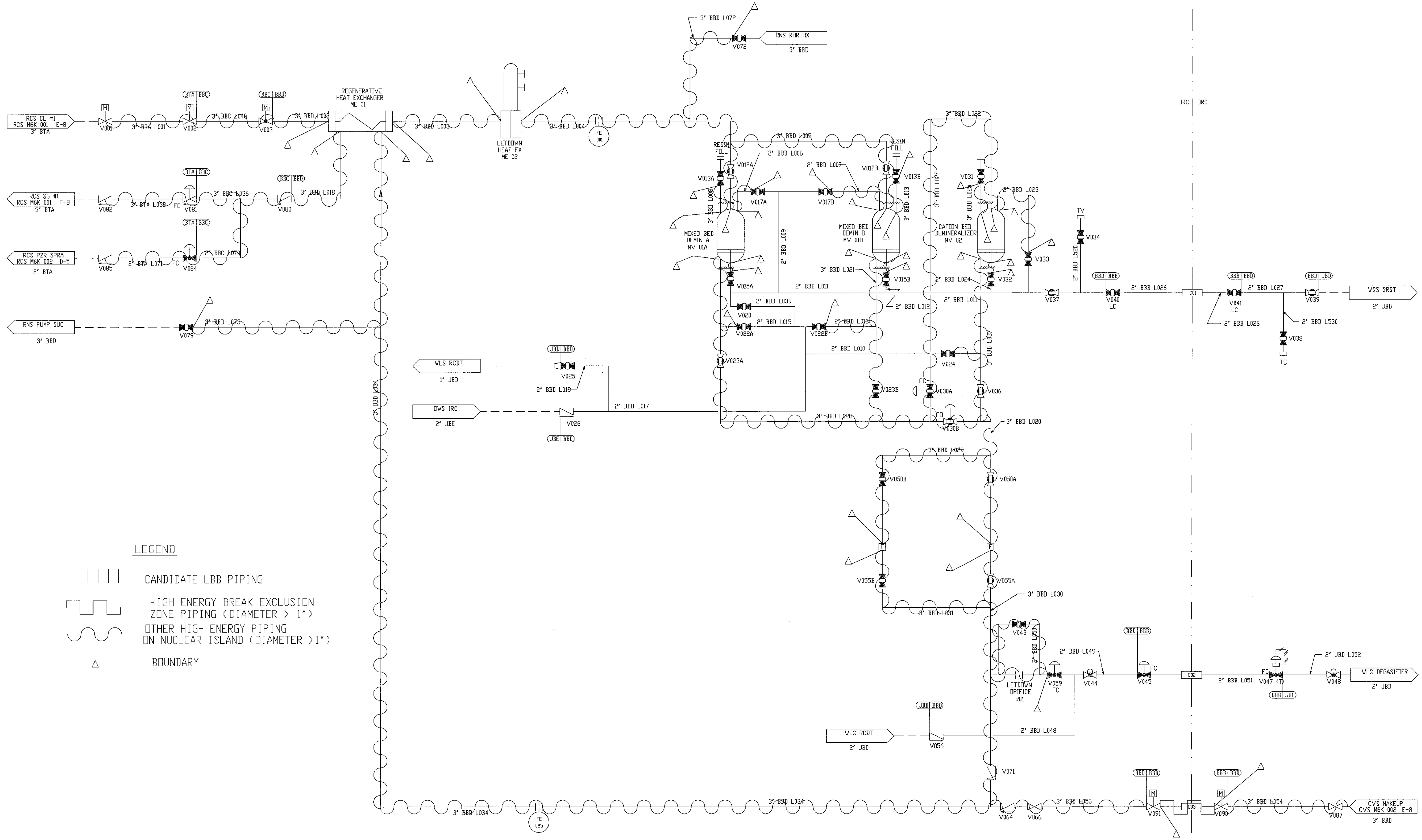


Figure 3E-5 (Sheet 1 of 2)
High Energy Piping – Chemical and Volume Control System

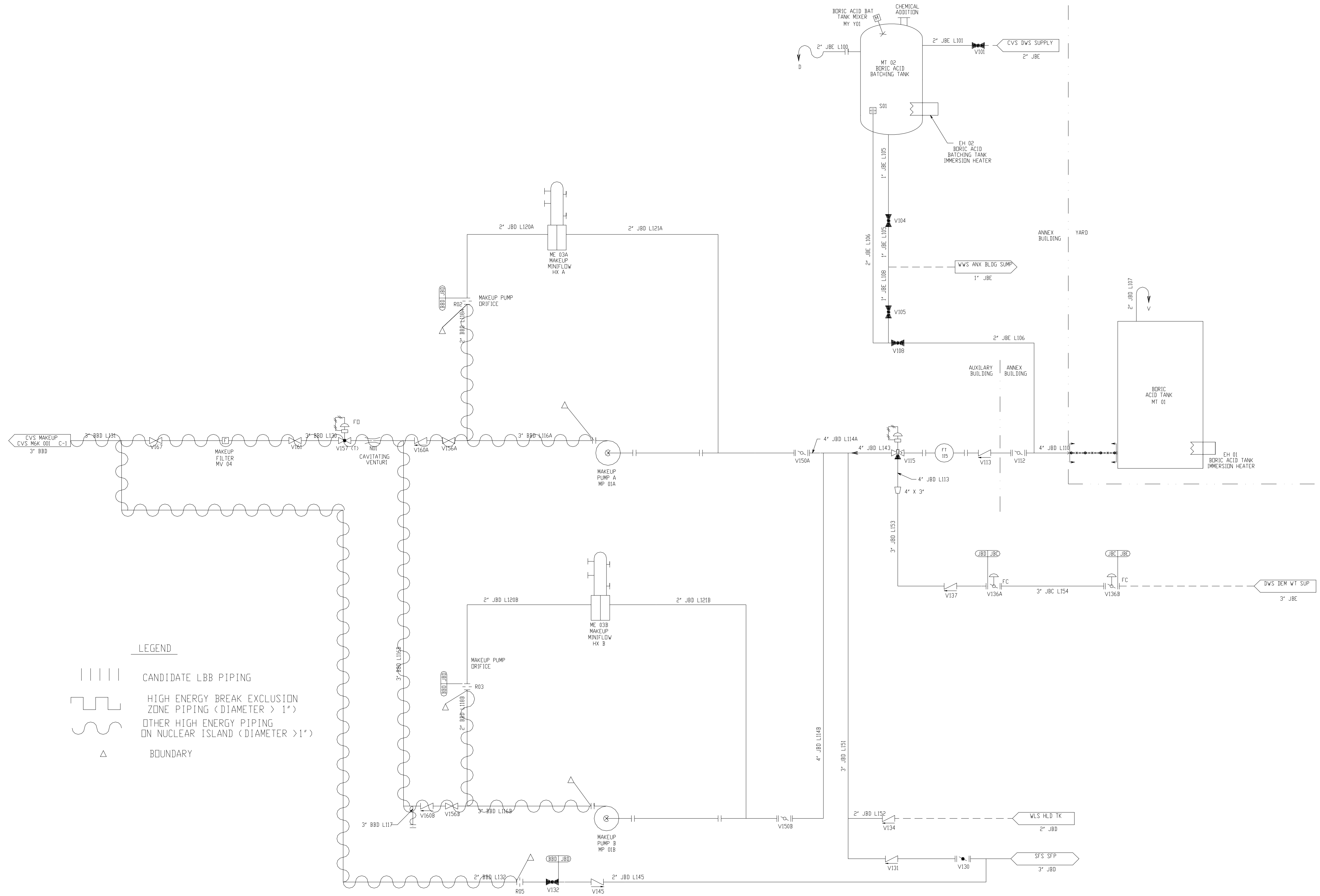


Figure 3E-5 (Sheet 2 of 2)
High Energy Piping – Chemical and Volume Control System

Appendix 3F Cable Trays and Cable Tray Supports

This appendix provides the design criteria for seismic Category I cable trays and their supports. Seismic Category II cable trays and their supports are also designed utilizing the design criteria of this appendix.

3F.1 Codes and Standards

The design of cable trays and their supports conform to the following codes and standards:

- American Iron and Steel Institute (AISI), Specification for the Design of Cold Formed Steel Structural Members, 1996 Edition and Supplement No. 1, July 30, 1999

- American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, AISC-N690-1994

- Institute of Electrical and Electronic Engineers (IEEE), Standard 344-1987, IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

- National Electrical Manufacturers Association (NEMA), Standard Publication No. VE 1-1998, Metallic Cable Tray Systems

3F.2 Loads and Load Combinations

3F.2.1 Loads

3F.2.1.1 Dead Load (D)

Dead load includes the weight of the cable trays, their supports and the cables inside the trays and any permanently attached items. Temporary items used during construction or maintenance are removed prior to operation.

It also includes the weight of

- Cable tray covers and
- Other components and fittings

3F.2.1.2 Construction Live Load (L)

Live load consists of a load of 250 pounds to be applied only during construction on the tray at a critical location to maximize flexural and shear stresses. This load is not combined with seismic loads.

3F.2.1.3 Safe Shutdown Earthquake (E_s)

Seismic response of the cable trays and their supports are produced due to seismic excitation of the supports.

3F.2.1.4 Thermal Load

These loads are usually not considered and trays are provided with expansion joints in accordance with NEMA.

3F.2.2 Load Combinations

The following load combinations are used for designing the cable trays and their supports:

(a) D + L

(b) D + E_s

3F.3 Analysis and Design

Cable trays and their supports are designed to maintain structural integrity. The stresses are maintained within the allowable limits as specified in Subsection 3F.3.3. Section properties and weights of the trays are obtained from manufacturer's data.

3F.3.1 Damping

The maximum damping ratio is 10 percent unless the configuration is demonstrated to be similar to that of the tests described in (Reference 19) of Subsection 3.7.6.

As stated in Subsection 3.7.1.3, the damping ratio used for the AP1000 cable tray systems may be based on test results presented in Reference 19 (Subsection 3.7.6). The cable tray test program conducted by ANCO Engineers Inc. included more than 2000 dynamic tests of representative cable tray system design and construction. The test configurations included items such as various tray types on rigid supports, various tray hanger systems, effects of tray types, effects of strut connections and effects of bracing spacing, unbraced and braced tray systems. Cable ties were also used during the test program. Based on observations during the tests, the high damping values within the cable tray system are provided mainly by the movement, sliding or bouncing of the cables within the tray. The tests show that, for unloaded trays, the damping ratio closely approximates the 7 percent used for bolted structures, and a minimum damping value of 20 percent is maintained with cable ties at spacing greater than or equal to four feet. The tests show that for loaded trays, the damping ratio increases with increased cable loading, reaching a value of 30 percent at cable fill ratio of 50 percent to 100 percent. The major factors which affect the damping ratio of the cable tray systems are the input acceleration level, cable fill ratio, and the ability of the cables to move within the trays during a safe shutdown earthquake.

The AP1000 cable tray system design requires no sprayed-on material for fire protection. Cable ties are provided at spacing greater than 4 feet, thereby permitting cable movement within the trays. The damping ratio used for the cable tray system is dependent on the level of seismic input and the amount of cable fill within the trays. As shown in Figure 3.7.1-13, the 20 percent constant damping ratio may be used for trays loaded to more than 50 percent and subjected to input floor acceleration greater than 0.35g. For cable trays loaded to less than 50 percent and lower than 0.35g input floor acceleration, linearly interpolated lower damping values may be used.

3F.3.2 Seismic Analysis

The methodology for seismic analysis is provided in Subsection 3.7.3. Seismic loads are determined by either using the equivalent static load method of analysis or by performing dynamic analysis.

Stresses are determined for the seismic excitation in two horizontal and one vertical direction. The stresses in the three directions are combined using the square root of the sum of the squares (SRSS) method or the 100-40-40 method as described in Subsection 3.7.3.6.

3F.3.3 Allowable Stresses

The basic stress allowables for the cable trays are based on the American Iron and Steel Institute specification. The basic stress allowables for cable tray supports utilizing light gage cold rolled channel type sections are based on the manufacturer's published catalog values. The basic stress allowables for cable tray supports utilizing rolled structural shapes are in accordance with ANSI/AISC N-690 and the supplemental requirements described in Subsection 3.8.4.5.2.

The allowable stresses for the load combinations are as follows:

D + L	Basic Allowable
D + E_s	1.6 times basic allowable for tension and 1.4 times basic allowable for compression

3F.3.4 Connections

Connections are designed in accordance with the applicable codes and standards listed in Section 3F.1. For connections used with light gage cold rolled channel type sections, design is based on the manufacturer's published catalog values. Supports are attached to the building structure by bolted or welded connections. Fastening of the supports to concrete structures meets the supplemental requirements given in Subsection 3.8.4.5.1.

Appendix 3G Nuclear Island Seismic Analyses

3G.1 Introduction

This appendix summarizes the seismic analyses of the nuclear island building structures performed to support the AP1000 design certification extension from just hard rock sites, to sites ranging from soft soils to hard rock. The seismic Category I building structures consist of the containment building (the steel containment vessel [SCV] and the containment internal structures [CIS]), the shield building, and the auxiliary building. These structures are founded on a common basemat and are collectively known as the nuclear island or nuclear island structures. Key dimensions of the seismic Category I building structures, such as thickness of the basemat, floor slabs, roofs and walls, are shown in Figures 3.7.1-14 and 3.7.2-12.

Analyses were performed in accordance with the criteria and methods described in Section 3.7. Section 3G.2 describes the development of the finite element models. Section 3G.3 describes the soil structure interaction analyses of a range of site parameters and the selection of the parameters used in the design analyses. Section 3G.4 describes the fixed base and soil structure interaction dynamic analyses and provides typical results from these dynamic analyses. References 3 and 6 provide a summary of dynamic and seismic analysis results (i.e., modal model properties, accelerations, displacements, response spectra) and the nuclear island liftoff analyses. The seismic analyses of the nuclear island are summarized in a seismic analysis summary report. Deviations from the design due to as-procured or as-built conditions are acceptable based on an evaluation consistent with the methods and procedures of Sections 3.7 and 3.8 provided the following acceptance criteria are met:

- The structural design meets the acceptance criteria specified in Section 3.8.
- The seismic floor response spectra (FRS) meet the acceptance criteria specified in Subsection 3.7.5.4.

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgment to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report by the Combined License applicant.

Table 3G.1-1 and Figure 3G.1-1 summarize the types of models and analysis methods that are used in the seismic analyses of the nuclear island, as well as the type of results that are obtained and where they are used in the design. Table 3G.1-2 summarizes the dynamic analyses performed and the methods used for combination of modal responses and directional input.

3G.2 Nuclear Island Finite Element Models

The AP1000 nuclear island consists of three distinct seismic Category I structures founded on a common basemat. The three building structures that make up the nuclear island are the coupled auxiliary and shield building (ASB), the SCV, and the CIS. The shield building and the auxiliary building are monolithically constructed with reinforced concrete and, therefore, considered one structure. The nuclear island is embedded approximately 40 feet with the bottom of basemat at elevation 60′-6″ and plant grade located at elevation 100′-0″. The CSV is described in Subsection 3.8.2, the CIS in Subsection 3.8.3, the ASB in Subsection 3.8.4, and the nuclear island basemat in Subsection 3.8.5.

Seismic systems are defined, according to SRP 3.7.2 (Reference 1), Section II.3.a, as the seismic Category I structures that are considered in conjunction with their foundation and supporting media to form a soil-structure interaction model. Fixed base seismic analyses are performed for the nuclear island at a rock site. Soil-structure interaction analyses are performed for soil sites. The analyses generate a set of in-structure responses (design member forces, nodal accelerations, nodal

displacements, and floor response spectra), which are used in the design and analysis of seismic Category I structures, components, and seismic subsystems. Concrete structures are modeled with linear elastic uncracked properties. However, the modulus of elasticity is reduced to 80% of the ACI code value to reduce stiffness to simulate cracking.

A seismic response spectrum analysis is performed to develop the seismic design loads for the design of the auxiliary building, shield building, and containment internal structure, and the loads generated include the amplified load due to flexibility and the distribution of this load to the surrounding structures. Equivalent static analyses are used to design the shield building roof and radial roof beams, tension ring, air inlet structure, and PCS tank.

3G.2.1 Individual Building and Equipment Models

3G.2.1.1 Coupled Auxiliary and Shield Building

The finite element shell dynamic model of the coupled ASB is a finite element model using primarily shell elements. The portion of the model up to the elevation of the auxiliary building roof is developed using the solid model features of ANSYS, which allow definition of the geometry and structural properties. The nominal element size in the auxiliary building model is about 9 feet so that each wall has two elements for the wall height of about 18 feet between floors. This mesh size, which is the same as that of the solid model, has sufficient refinement for global seismic behavior. It is combined with a finite element model of the shield building roof and cylinder above the elevation of the auxiliary building roof. This model is shown in Figure 3G.2-1. This finite element shell dynamic model is part of the NI10 model.

Since the water in the passive containment cooling system tank responds at a very low frequency (sloshing) and does not affect building response, the passive containment cooling system tank water mass is reduced to exclude the low frequency water sloshing mass. The wall thickness of the bottom portion of the shield building (elevation 63.5′ to 81.5′) is modeled as one half (1.5′) since the CIS model is connected to this portion and extends out to the mid-radius of the shield building cylindrical wall. Local portions of the ASB floors and walls are modeled with sufficient detail to give the response of the flexible areas.

3G.2.1.2 Containment Internal Structures

The finite element shell model of the containment internal structures is a finite element model using primarily shell elements for the walls and floors and solid elements for the mass concrete. It is developed using the solid model features of ANSYS, which allow definition of the geometry and structural properties. This model is used in both static and dynamic analyses. It models the inner and outer mass concrete basemats embedding the lower portion of the containment vessel, and the concrete structures above the mass concrete inside the containment vessel. The walls and basemat inside containment for this model are shown in Figure 3G.2-2. The basemat (dish) outside the containment vessel is shown in Figure 3G.2-3. This finite element shell dynamic model is part of the NI10 model. Static analyses are also performed on the model to obtain member forces in the walls. This model is also used in the 3D finite element basemat model (see Subsection 3.8.5.4.1).

3G.2.1.3 Containment Vessel

The SCV is a freestanding, cylindrical, steel shell structure with ellipsoidal upper and lower steel domes. The finite element model of the containment vessel is an axisymmetric model fixed at elevation 100′. Static analyses are performed with this model to obtain shell stresses as described in Subsection 3.8.2.4.1.1. The model is also used to develop modal properties (frequencies and mode shapes). The three-dimensional, lumped-mass stick model of the SCV is developed based on the

axisymmetric shell model. Figure 3G.2-4 presents the SCV stick model. In the stick model, the properties are calculated as follows:

- Members representing the cylindrical portion are based on the properties of the actual circular cross section of the containment vessel.
- Members representing the bottom head are based on equivalent stiffnesses calculated from the shell of revolution analyses for static 1.0g in vertical and horizontal directions.
- Shear, bending and torsional properties for members representing the top head are based on the average of the properties at the successive nodes, using the actual circular cross section. These are the properties that affect the horizontal modes. Axial properties, which affect the vertical modes, are based on equivalent stiffnesses calculated from the shell of revolution analyses for static 1.0g in the vertical direction.

The equivalent static acceleration analyses of the containment vessel use a finite element shell model with a refined mesh in the area adjacent to the large penetrations. Comparison of this with a time history analysis for the regions immediately surrounding the large penetrations verifies that the loads from equivalent static analysis are conservative to time history using a representative study.

The stick model is combined with the polar crane stick model as shown in Figure 3G.2-4. Modal properties of the containment vessel with and without the polar crane are shown in Table 3G.2-1. It is connected to nodes on the dish model. NI10 node numbers are shown in red and NI20 node numbers are shown in black.

The method used to construct a stick model from the axisymmetric shell model of the containment vessel is verified by comparison of the natural frequencies determined from the stick model and the shell of revolution model as shown in Table 3G.2-2. The shell of revolution vertical model (n = 0 harmonic) has a series of local shell modes of the top head above elevation 265′ between 23 and 30 hertz. These modes are predominantly in a direction normal to the shell surface and cannot be represented by a stick model. These local modes have small contribution to the total response to a vertical earthquake as they are at a high frequency where seismic excitation is small. The only seismic Category I components attached to this portion of the top head are the water distribution weirs of the passive containment cooling system. These weirs are designed such that their fundamental frequencies are outside the 23 to 30 hertz range of the local shell modes.

An evaluation was made of the connection of the bottom of the steel containment vessel stick model to the CIS finite element model. Comparisons were made between the unconstrained fully symmetric, radially constrained fully symmetric, and original asymmetric connectivity models. The response spectra at the elevation of the polar crane girder for the first two models are almost identical, and the third model had only minor differences. Based on this comparison, the unconstrained fully symmetric connectivity model is used.

3G.2.1.4 Polar Crane

The polar crane is supported on a ring girder, which is an integral part of the SCV at elevation 228′-0″, as shown in Figure 3.8.2-1. It is modeled as a multi-degree of freedom system attached to the steel containment shell at elevation 224′ (midpoint of ring girder) as shown in Figure 3G.2-4. The polar crane is modeled using a simplified and detailed model. The simplified model has five masses at the mid-height of the bridge at elevation 233′-6″ and one mass for the trolley, as shown in Figure 3G.2-5A. The polar crane model includes the flexibility of the crane bridge girders and truck assembly, and the containment shell's local flexibility. When fixed at the center of containment, the model shows fundamental frequencies of 3.3 hertz transverse to the bridge, 7.0 hertz vertically, and 6.4 hertz along the bridge. The Detailed Model of the polar crane consists of 28 nodes is defined

having 96 dynamic degrees of freedom. It is used to verify the accuracy of the simplified model. This model is shown in Figure 3G.2-5B.

Nodes 1 to 4 represent the Trucks with elevation at top of rails (TOR). There are four nodes that are coincident with nodes 1 to 4 and used to add the local SCV stiffnesses (nodes 465 to 468, not shown in Figure).

1. Nodes 9 to 12 represent the trolley. The trolley is connected to the centerline of the polar crane girders at nodes 9 and 10.

2. Nodes 13 to 26 are located on the polar crane girders. The end nodes (13, 19, 20 and 26) are used to connect the cross beams to the girders; these nodes are also attached to the trucks (nodes 1 to 4) by rigid links.

3. Node 470 is at the center of containment at the top of rail elevation. Nodes 465 to 468 are attached to node 470 using rigid links.

4. Node 29, not shown in Figure, is located on the SCV. It is attached to 470 by a rigid link.

3G.2.1.5 Major Equipment and Structures Using Stick Models

The major equipment supported by the CIS is represented by stick models connected to the CIS. These stick models are the reactor coolant loop model shown in Figure 3G.2-6, the pressurizer model shown in Figure 3G.2-7, and the core makeup tank model shown in Figure 3G.2-8. The core makeup tank model is used only in the nuclear island fine (NI10) model; the core makeup tank is represented by mass in the nuclear island coarse model (NI20).

3G.2.2 Nuclear Island Dynamic Models

Finite element shell models (3D) of the nuclear island concrete structures are used for the time history seismic analyses. Stick models are coupled to the shell models of the concrete structures for the containment vessel, polar crane, the reactor coolant loop and pressurizer. Two models are used. The fine (NI10) model is used to define the seismic response for the hard rock site. The coarse (NI20) model is used for the soil structure interaction (SSI) analyses. It is similar to the NI10 model with the exception that the mesh size for the ASB and CIS is approximately 20 feet instead of 10 feet. This model is set up in both ANSYS and SASSI. The NI05 model is used to develop amplified seismic response for the envelope of soil profiles presented in Subsection 3.7.1.4 for flexible regions not captured by the coarser NI20 model. The NI05 model is also used in response spectrum analysis of the nuclear island to develop design seismic member forces and moments. The NI10, NI20, and NI05 models are described in the subsections below.

3G.2.2.1 NI10 Model

The large solid-shell finite element model of the AP1000 nuclear island shown in Figure 3G.2-9 combines the ASB solid-shell model described in Subsection 3G.2.1.1, and the CIS solid-shell model described in Subsection 3G.2.1.2. The containment vessel and major equipment that are supported by the CIS are represented by stick models and are connected to the CIS. These stick models are the SCV and the polar crane models, the reactor coolant loop model, core makeup tank models, and the pressurizer model. The stick models are described in Subsections 3G.2.1.3 and 3G.2.1.4. The CIS and attached sticks are shown in Figure 3G.2-10. This AP1000 nuclear island model is referred to as the NI10 or fine model. The ASB portion of this model has a mesh size of approximately 10 feet.

The SCV is connected to the CIS model using constraint equations. The SCV at the bottom of the stick at elevation 100′ (node 130401) is connected to CIS nodes at the same elevation. Figure 3G.2-4

shows the SCV stick model with the constraint equation nodes. The nodes are defined using a cylindrical coordinate system whose origin coincides with the center of containment (node 130401). The CIS vertical displacement is tied rigidly (constrained) to the vertical displacement and RX and RY rotations of node 130401. The CIS tangential displacement is tied rigidly (constrained) to the horizontal displacement and RZ rotation of node 130401.

3G.2.2.2 NI20 Model

The NI20 coarse model has fewer nodes and elements than the NI10 model. It captures the essential features of the nuclear island configuration. The nominal shell and solid element dimension is about 20 feet. It is used in the soil-structure interaction analyses of the nuclear island performed using the program SASSI. The stick models are the same as used for the NI10 model except that the core makeup tank is not included. This model is shown in Figures 3G.2-11 and 3G.2-12. Results of fixed base analyses of the NI20 model were compared to those of the NI10 model to confirm the adequacy of the NI20 model for use in the soil-structure-interaction analyses.

3G.2.2.3 Nuclear Island Stick Model

The nuclear island lumped-mass stick model consists of the stick models of the individual buildings interconnected by rigid links. Each individual stick model is developed to match the modal properties of the finite element models described in Subsections 3G.2.1.1 and 3G.2.1.2 above. Modal analyses and seismic time history analyses were performed using this model for the hard rock design certification.

The nuclear island lumped-mass stick model has been replaced in the design analyses described in this appendix by the NI10 and NI20 finite element shell dynamic models of the nuclear island described in Subsections 3G.2.2.1 and 3G.2.2.2 above. A 2D stick model is used in the soil sensitivity analyses described in Section 3G.3.

3G.2.2.4 NI05 Model

The NI05 solid-shell finite element model of the AP1000 nuclear island is shown in Figures 3G.2-13 to 3G.2-15. The NI05 model is used for response spectrum analysis of the nuclear island auxiliary and shield building structures. The NI05 model is also used for the mode superposition time history analysis of the nuclear island for the amplified response at flexible floors. The NI05 model is used for the static analysis of the nuclear island for the basemat design. The NI05 model is a refined version of the NI10 model where the auxiliary and shield building mesh size is reduced from approximately 10 feet by 10 feet tetrahedral mesh to approximately 5 feet by 5 feet. The major equipment stick models supported by the CIS are the same as used for the NI10 model. The steel containment vessel stick model and connections are also the same as the NI10 model. The only difference between the NI05 CIS and NI10 CIS is the basemat (bowl) and dish region as shown in Figure 3G.2-15. The model is validated by a comparison of the mass participation by frequency of the fundamental modes to those of the NI10 model.

3G.2.2.5 Seismic Stability Model

The sliding stability of the nuclear island basemat is evaluated using a non-linear 2D East-West (EW) stick model of the nuclear island structures using the ANSYS program. Three concentric sticks represent ASB, CIS, and SCV, respectively. The reactor coolant loop is included as mass only. The basemat is modeled as a rigid beam, which is free in translation along the EW and vertical directions. The nuclear island combined sticks are attached to the rigid basemat at the nuclear island mass center.

Each node of the rigid basemat is connected with two spring elements in the horizontal and vertical directions, respectively. The spring elements only model the foundation media (rock or soil) damping, not stiffness. A layer of contact elements is added along the rigid basemat bottom to simulate the friction forces between basemat bottom and foundation media as well as foundation media stiffnesses. The friction coefficient between the basemat bottom and the soil media is set at 0.55. Figure 3G.2-19 shows the schematic of this non-linear 2D EW nuclear island stick model. The contact elements are free to uplift when the upward force (normal force) is larger than the associated dead load component. When the tangential force is larger than the friction force, sliding occurs.

3G.2.3 Static Models

Member forces in the ASB are obtained from analyses of a model that is more refined than the finite element model described in Subsection 3G.2.1.1. This model is developed by meshing one area of the solid model with four finite elements. The nominal element size in this auxiliary building model is about 4.5 feet so that each wall has four elements for the wall height of about 18 feet between floors. This finite element shell model is referred to as the NI05 model. This refinement is used to calculate the design member forces and moments using response spectra analysis of the nuclear island models with seismic input enveloping all soil conditions. The finite element shell model of the containment internal structures described in Subsection 3G.2.1.2, which includes the basemat within the shield building and the containment vessel stick model, is also included.

Finite element solid/shell models were used for the equivalent static seismic analysis. For the detailed design of the shield building roof, a finite element model of one quadrant of the roof is used as described in Subsection 3G.2.3.1. For the detailed design of the steel containment vessel, a shell mesh finite element model with a much finer mesh in the areas surrounding the major penetrations is used as described in Subsection 3G.2.3.2. For the static analysis of the containment vessel, an axisymmetric model is used as described in Subsection 3G.2.3.3. The nuclear island basemat is evaluated using the NI05 finite element model described in Subsection 3G.2.2.4.

3G.2.3.1 Quadrant Model of Shield Building Roof

The one quadrant model of the shield building roof is shown in Figure 3G.2-16. The model is constructed with solid and shell elements and contains structures from the exposed shield wall through the top of the shield building roof. The quadrant model is used for the equivalent static analysis of the shield building roof. The results from the more detailed analysis are used in the design of the shield building roof and radial roof beams, tension ring, air inlet structure, and PCS tank.

3G.2.3.2 Containment Vessel 3D Finite Element Model

The 3D finite element model of the steel containment vessel is shown in Figure 3G.2-17. The finite element model for the steel containment vessel is used for the stress analysis of the large penetrations (personnel locks and equipment hatches) of the containment vessel.

3G.2.3.3 Containment Vessel Axisymmetric Model

The axisymmetric finite element model of the steel containment vessel is shown in Figure 3G.2-18. The axisymmetric model is a two-dimensional model with added mass for the stiffeners, crane girder, equipment hatches, and air locks.

3G.3 2D SASSI Analyses

This section describes the soil structure interaction analyses performed using 2D models in SASSI to select the design soil cases for the AP1000. The AP1000 footprint, or interface to the soil medium, is identical to the AP600. The AP1000 containment and shield building are 25′ 6″ and 20′ 6″

(Reference 4) respectively taller than AP600. Results and conclusions from the AP600 soil studies (Reference 2) are considered in establishing the design soil profiles for the AP1000.

Analyses were performed using 2D stick models of the AP1000 for horizontal seismic input with and without adjacent structures. The soil profiles included a hard rock site, a firm rock site, a soft rock site, a soft-to-medium soil site, an upper bound soft-to-medium site, and a soft soil site. Analyses were also performed without adjacent structures for a hard rock site, a firm rock site, a soft rock site, a soft-to-medium soil site, an upper bound soft-to-medium site, and a soft soil site. The soil damping and degradation curves are described in Subsection 3.7.1.4. The soil profiles selected for the AP1000 use the same parameters on depth to bedrock, depth to water table, and variation of shear wave velocity with depth as those used in the AP600 design analyses. The Poisson's ratio is 0.25 for rock sites (hard and firm rock) and 0.35 for soil sites (soft-to-medium soil, and upper bound soft-to-medium soil). For all the soil profiles defined, the base rock has been taken to be at 120 feet below grade level. The soil profiles are shown in Figure 3G.3-1. The shear wave velocity profiles and related governing parameters are as follows:

- For the hard rock site, an upper bound case for rock sites using a shear wave velocity of 8000 feet per second.
- For the firm rock site, a shear wave velocity of 3500 feet per second to a depth of 120 feet, and base rock at the depth of 120 feet.
- For the soft rock site, a shear wave velocity of 2400 feet per second at the ground surface, increasing linearly to 3200 feet per second at a depth of 240 feet, and base rock at the depth of 120 feet.
- For the upper bound soft-to-medium soil site, a shear wave velocity of 1414 feet per second at ground surface, increasing parabolically to 3394 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water at grade level. The initial soil shear modulus profile is twice that of the soft-to-medium soil site.
- For the soft-to-medium soil site, a shear wave velocity of 1000 feet per second at ground surface, increasing parabolically to 2400 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water is assumed at grade level.
- For the soft soil site, a shear wave velocity of 1000 feet per second at ground surface, increasing linearly to 1200 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water is assumed at grade level.

The analyses with and without adjacent structures demonstrated that the effect of adjacent buildings on the nuclear island response is small. Based on this, the 3D SASSI analyses of the AP1000 nuclear island can be performed without adjacent buildings similar to those performed for the AP600.

The maximum acceleration values obtained from the AP1000 analyses without adjacent structures are given in Table 3G.3-1. The soil cases giving the maximum response are shown in bold. Floor response spectra associated with nodes 41, 120, 310, 411, and 535 for the six AP1000 soil cases are shown in Figures 3G.3-2 to 3G.3-11.

Based on review of the above results, five soil conditions were selected for 3D SASSI analyses in addition to the hard rock condition evaluated in the existing AP1000 Design Certification. Thus, the following five soil and rock cases identified in Subsection 3.7.1.4 are considered: hard rock, firm rock, soft rock, soft-to-medium soil, upper bound soft-to-medium, and soft soil.

3G.4 Nuclear Island Dynamic Analyses

3G.4.1 ANSYS Fixed Base Analysis

The NI10 model described in Subsection 3G.2.2.1 was analyzed by time history modal superposition. To perform the time history analysis of this large model, the ANSYS superelement (substructuring) techniques were applied. Substructuring is a procedure that condenses a group of finite elements into one element represented as a matrix. The reasons for substructuring are to reduce computer time of subsequent evaluations. Two sets of analyses were performed. To obtain the time history response of the ASB, the ASB finite element model was merged with the superelement of the CIS and its major components. To obtain the time history response of the CIS, the CIS finite element model was merged with the superelement of the ASB.

Deflection time history responses were obtained at selected representative locations. These locations included major wall and floor intersections and nodes at the cardinal orientations at key elevations of the shield building. Nodes were also selected at mid-span on flexible walls and floors. Typical locations are shown for the ASB at elevation 135′ on Figures 3G.4-1 and 3G.4-2. Figure 3G.4-1 shows the "rigid" locations, and Figure 3G.4-2 shows the "flexible" locations.

ANSYS is used to calculate the maximum relative deflection to the nuclear island for the envelope case that considers all of the soil and hard rock site cases. Synthesized displacement time histories are developed using the envelope seismic response spectra from the six site conditions (hard rock, firm rock, soft rock, upper-bound-soft-to-medium, soft-to-medium, and soft soil). Seismic response spectra at nine locations are used (four edge locations, one center location, and four corner locations). It is not necessary to adjust for drift since relative deflections to the basemat are calculated and the drift would be subtracted from the results.

3G.4.2 3D SASSI Analyses

The computer program SASSI 2000 is used to perform Soil-Structure Interaction analysis with the NI20 Coarse Finite Element Model. The SASSI Soil-Structure Interaction analyses are performed for the five soil conditions established from the AP1000 2D SASSI analyses. These soil conditions are firm rock, soft rock, soft-to-medium soil, upper bound soft-to-medium, and soft soil. The model includes a surrounding layer of excavated soil and the existing soil media as shown in Figures 3G.4-3 and 3G.4-4. Acceleration time histories and floor response spectra are obtained. Adjacent structures have a negligible effect on the nuclear island structures and, thus, are not considered in the 3D SASSI analyses.

Westinghouse has adopted the approach that calculates displacements internally within the ACS SASSI program based on an analytical complex frequency domain approach that uses inverse Fast-Fourier Transforms (FFT) to compute relative displacement histories instead of double numerical integration in the time domain that computes absolute displacement time histories from absolute acceleration time histories.

The relative displacement time history is calculated using ACS SASSI RELDISP module. The complex acceleration transfer functions (TF) are computed for reference and all selected output nodes. The relative acceleration transfer function is calculated by subtracting the reference node TF from the output node TF. The relative displacement transfer function is obtained by dividing the circular frequency square (ω^2) for each frequency data point. The relative displacement time history is obtained by taking the inverse FFT.

Relative displacements are calculated between adjacent buildings and the nuclear island using soft springs between the buildings. The spring stiffness is very small so that it does not affect the dynamic

response. These calculations are performed using 2D models and SASSI 2000. The relative deflection is calculated using the maximum compressive spring force and the stiffness value.

In these analyses, the three components of ground motions (N-S, E-W, and vertical direction) are input separately. Each design acceleration time history (N-S, E-W, and vertical) is applied separately, and the time history responses are calculated at the required nodes. The resulting co-linear time history responses at a node due to the three earthquake components are then combined algebraically.

3G.4.3 Seismic Analysis

3G.4.3.1 Response Spectrum Analysis

The response spectrum methodology used in the AP1000 design employs the Complete Quadratic Combination (CQC, Section 1.1 of Reference 5) grouping method for closely spaced modes with the Der Kiureghian Correlation Coefficient (Section 1.1.3 of Reference 5) used for correlation between modes. The Lindley-Yow (Section 1.3.2, Reference 5) spectra analysis methodology is employed for modes with both periodic and rigid response components. The modal analysis performed to develop composite modal participation is used to develop input for the response spectrum analysis. Modes ranging from 0 to 33 Hz or higher are considered. For modes above the cutoff frequency, the Lindley-Yow is used. The Static ZPA Method (Section 1.4.2, Reference 5) is employed for the residual rigid response component for each mode as outlined in NRC Regulatory Guide 1.92 (Reference 5). The complete solution is developed via Combination Method B (Section 1.5.2, Reference 5). The combined effects, considering three spatial components of an earthquake (N-S, E-W, and Vertical), are combined by square root sum of the squares method (Section 2.1, Reference 5).

In Subsection 3.7.2.6, “Three Components of Earthquake Motion,” the combination of three components of earthquake motion is discussed.

3G.4.3.2 Absolute Accelerations

The seismic analyses results, which include the new shield building configuration described in Section 3.8, are given in Reference 3.

3G.4.3.3 Seismic Response Spectra

The AP1000 plant floor response spectra for the six key locations are provided in Figure 3G.4-5X to 3G.4-10Z. The key locations are defined in Table 3G.4-1. The design seismic response spectra are conservatively adjusted in the low frequency range in anticipation of future sites having a slightly higher response at the lower frequency.

The in-structure response spectra at six key locations, as defined below, are used if a site-specific 3D dynamic analysis evaluation as outlined in Subsection 2.5.2 is required. The site is acceptable if the floor response spectra from the site-specific evaluation do not exceed the AP1000 spectra for each of the locations identified below or the exceedances are justified.

[*FRS Location*	*Figure Numbers*
Containment internal structures at elevation of reactor vessel support	*Figure 3G.4-5X to 3G.4-5Z*
Containment operating floor	*Figure 3G.4-6X to 3G.4-6Z*
Auxiliary building NE corner at elevation 116'-6"	*Figure 3G.4-7X to 3G.4-7Z*
Shield building at fuel building roof	*Figure 3G.4-8X to 3G.4-8Z*

*NRC Staff approval is required prior to implementing a change in this information.

Shield building roof	*Figure 3G.4-9X to 3G.4-9Z*
Steel containment vessel at polar crane support	*Figure 3G.4-10X to 3G.4-10Z*]*

Note:
See Table 3G.4-1 for locations of six key locations.

3G.4.3.4 Bearing Pressure Demand

Bearing pressure demand was calculated using both 2D and 3D analyses. Both linear and non-linear analyses are performed with the 2D nuclear island model. The maximum bearing pressures calculated include the effect of dead, live, and seismic loading.

The 2D model was used to evaluate the effect of liftoff on the bearing pressure. Since the largest bearing pressure will result from the east-west seismic excitation because of the smaller width of the basemat in this direction, liftoff was evaluated using an east-west stick model of the nuclear island structures, supported on a rigid basemat with non-linear springs. Direct integration time history analyses were performed. The bearing pressures calculated from these analyses are summarized in Table 3G.4-2. The pressures are at the edge of the basemat. Results are given for the three cases that result in the highest bearing pressure (hard rock [HR], upper bound soft to medium [UBSM] soil, and soft to medium [SM] soil). The linear results show maximum bearing pressures on the west side of 31 to 33 ksf. Liftoff increases the subgrade pressure close to the west edge by 4 percent to 6 percent with insignificant effect beneath most of the basemat.

The SASSI soil-structure interaction analyses are performed based on the nuclear island 3D SASSI model for the hard rock and five soil conditions established from the AP1000 2D SASSI analyses. The SASSI model of the nuclear island is based on the NI20 finite element model. The bearing pressures from the 3D SASSI analyses have been obtained by combining the time history results from the north-south, east-west, and vertical earthquakes. The maximum soil-bearing pressure demand is obtained from the hard rock (HR) case equal to 35 ksf. It is noted that a maximum localized peak is obtained on the west edge of 38 ksf; a limit of 35 ksf for maximum bearing seismic demand is obtained by averaging the soil pressure over 335 ft^2 of the west edge of the shield building where the maximum stress occurs.

3G.5 References

1. NUREG-800, Review of Safety Analysis Reports for Nuclear Power Plants, Section 3.7.2, Seismic System Analysis, Revision 2.

2. GW-GL-700, AP600 Design Control Document, Appendices 2A and 2B, Revision 4.

3. APP-GW-S2R-010, “Extension of Nuclear Island Seismic Analyses to Soil Sites,” Westinghouse Electric Company LLC.

4. APP-GW-GLN-112, “Structural Verification for Enhanced Shield Building,” Westinghouse Electric Company LLC.

5. U.S. NRC Regulatory 1.92, Revision 2, “Combining Modal Responses and Spatial Components in Seismic Analysis.”

6. APP-GW-GLR-044, “Nuclear Island Basemat and Foundation,” Westinghouse Electric Company LLC.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3G.1-1 (Sheet 1 of 4)
Summary of Models and Analysis Methods

Model	Analysis Method	Program	Type of Dynamic Response/Purpose
3D (ASB) solid-shell model	-	ANSYS	Creates the finite element mesh for the ASB finite element model.
3D (CIS) solid-shell model	-	ANSYS	Creates the finite element mesh for the CIS finite element model.
3D finite element model including shield building roof (ASB10)	-	ANSYS	ASB portion of NI10.
3D finite element model including dish below containment vessel	Response spectrum analysis	ANSYS	CIS portion of NI10.
3D finite element shell model of nuclear island [NI10] (coupled auxiliary and shield building shell model, containment internal structures, steel containment vessel, polar crane, RCL, pressurizer, and CMTs)	Mode superposition time history analysis	ANSYS	Performed for hard rock profile for ASB with CIS as superelement and for CIS with ASB as superelement. To develop time histories for generating plant design floor response spectra for nuclear island structures. To obtain maximum absolute nodal accelerations (ZPA) to be used in equivalent static analyses. To obtain maximum displacements relative to basemat.
3D finite element coarse shell model of auxiliary and shield building and containment internal structures [NI20] (including steel containment vessel, polar crane, RCL, and pressurizer)	Mode superposition time history analysis	ANSYS	Performed for hard rock profile for comparisons against more detailed NI10 model.

Table 3G.1-1 (Sheet 2 of 4)
Summary of Models and Analysis Methods

Model	Analysis Method	Program	Type of Dynamic Response/Purpose
Finite element lumped-mass stick model of nuclear island	Time history analysis	SASSI	Performed 2D parametric soil studies to help establish the bounding generic soil conditions and to develop adjustment factors to reflect all generic site conditions for seismic stability evaluation.
Finite element lumped-mass stick model of nuclear island	Direct integration time history analysis	ANSYS	Performed 2D linear and non-linear seismic analyses to evaluate effect of liftoff on Floor Response Spectra and bearing.
3D finite element coarse shell model of auxiliary and shield building and containment internal structures [NI20] (including steel containment vessel, polar crane, RCL, and pressurizer)	Time history analysis Complex frequency response analysis	SASSI	Performed for the five soil profiles of firm rock, soft rock, upper bound soft-to-medium soil, soft-to-medium soil, and soft soil. To develop time histories for generating plant design floor response spectra for nuclear island structures. To obtain maximum absolute nodal accelerations (ZPA) to be used in equivalent static analyses. To obtain maximum displacements relative to basemat. To obtain SSE bearing pressures for all generic soil cases. To obtain maximum member forces and moments in selected elements for comparison to equivalent static results.
3D shell model of auxiliary and shield building and containment internal structures [NI20] (including steel containment vessel)	Mode superposition time history analysis	ANSYS	Performed to develop loads for seismic stability evaluation.

Table 3G.1-1 (Sheet 3 of 4)
Summary of Models and Analysis Methods

Model	Analysis Method	Program	Type of Dynamic Response/Purpose
3D shell of revolution model of steel containment vessel	Modal analysis; equivalent static analysis using accelerations from time history analyses	ANSYS	To obtain dynamic properties. To obtain SSE stresses for the containment vessel.
3D lumped-mass stick model of the SCV	-	ANSYS	Used in the NI10 and NI20 models.
3D lumped-mass stick model of the RCL	-	ANSYS	Used in the NI10 and NI20 models.
3D lumped-mass stick model of the pressurizer	-	ANSYS	Used in the NI10 and NI20 models.
3D lumped-mass stick model of the CMT	-	ANSYS	Used in the NI10 model.
3D lumped mass detailed model of the polar crane	Modal analysis	ANSYS	To obtain dynamic properties. Used with 3D finite element shell model of the containment vessel.
3D lumped mass simplified (single beam) model of the polar crane	-	ANSYS	Used in the NI10 and NI20 models.
3D finite element shell model of containment vessel	Mode superposition time history analysis Equivalent static analysis	ANSYS	Used with detailed polar crane model to obtain acceleration response of equipment hatch and airlocks. To obtain shell stresses in vicinity of the large penetrations of the containment vessel.

Table 3G.1-1 (Sheet 4 of 4)
Summary of Models and Analysis Methods

Model	Analysis Method	Program	Type of Dynamic Response/Purpose
3D finite element refined shell model of nuclear island (NI05)	Equivalent static non-linear analysis using accelerations from time history analyses Mode superposition time history analysis for the wall and floor flexibility using synthetic time histories developed to match spectral envelopes applied at the base Response spectrum analysis with seismic input enveloping all soils cases	ANSYS	To obtain SSE member forces for the nuclear island basemat. To obtain floor and wall flexibility response characteristics. To obtain maximum displacements relative to basemat. To obtain SSE member forces for the auxiliary and shield building and the containment internal structures.
3D finite element coarse shell model of auxiliary and shield building and containment internal structures [NI20] (including steel containment vessel, polar crane, RCL, and pressurizer)	Mode superposition time history analysis with seismic input enveloping all soil cases	ANSYS	To obtain total basemat reactions for comparison to reactions in equivalent static linear analyses using NI05 model.
Quadrant model of shield building roof (See Subsection 3.8.4.4.1 for information on use of the quadrant model.)	Equivalent static analysis The PCS tank is designed using the maximum accelerations at the applicable elevation resulting from time history dynamic analyses of the nuclear island. The tension ring and air inlet use maximum accelerations that are increased based on results of response spectrum analysis.	ANSYS	To obtain member forces for shield building roof and radial roof beams, air inlet structure, tension ring, and PCS tank.

Table 3G.1-2
Summary of Dynamic Analyses and Combination Techniques

Model	Analysis Method	Program	Three Components Combination	Modal Combination
3D finite element, fixed base models, coupled auxiliary and shield building shell model, with superelement of containment internal structures (NI10 and NI20)	Mode superposition time history analysis	ANSYS	Algebraic Sum	n/a
3D finite element nuclear island model (NI20)	Complex frequency response analysis	SASSI	Algebraic Sum	n/a
3D finite element, fixed base models, coupled auxiliary and shield building and containment internal structures including shield building roof (NI05)	Response spectrum analysis	ANSYS	SRSS or 100%, 40%, 40%	Lindley-Yow
3D finite element model of the nuclear island basemat (NI05)	Equivalent static analysis using nodal accelerations from shell model	ANSYS	100%, 40%, 40%	n/a
3D shell of revolution model of steel containment vessel	Equivalent static analysis using nodal accelerations from 3D stick model	ANSYS	SRSS or 100%, 40%, 40%	n/a
PCS valve room and miscellaneous steel frame structures, miscellaneous flexible walls, and floors	Response spectrum analysis	ANSYS	SRSS or 100%, 40%, 40%	Grouping or Lindley-Yow
2D stick model analyses with liftoff	Direct integration time history	ANSYS	Algebraic Sum	n/a

Table 3G.2-1 (Sheet 1 of 2)
Steel Containment
Vessel Lumped-Mass Stick Model (Without Polar Crane) Modal Properties

Mode	Frequency	Effective Mass		
		X Direction	Y Direction	Z Direction
1	6.309	2.380	159.153	0.005
2	6.311	159.290	2.382	0.000
3	12.942	0.018	0.000	0.000
4	16.970	0.000	0.006	171.030
5	18.960	0.102	40.263	0.002
6	18.970	40.161	0.102	0.000
7	28.201	0.000	0.000	28.073
8	31.898	0.054	2.636	0.000
9	31.999	2.789	0.057	0.000
10	37.990	0.909	0.007	0.000
11	38.634	0.022	4.846	0.009
12	38.877	3.758	0.014	0.000
13	47.387	0.000	0.000	5.066
14	54.039	4.649	0.633	0.000
15	54.065	0.624	4.693	0.002
16	60.628	0.002	0.042	3.389
17	62.734	0.147	0.001	0.018
18	63.180	0.000	0.050	7.069
19	63.613	0.002	0.001	0.003
20	65.994	0.022	0.659	0.041
Sum of Effective Masses		214.929	215.545	214.706

Notes:
1. Fixed at Elevation 100′.
2. The total mass of the containment vessel is 225.697 kip-sec^2/ft.

Table 3G.2-1 (Sheet 2 of 2)
Steel Containment Vessel Lumped-Mass Stick Model (With Polar Crane)
Modal Properties

Mode	Frequency	Effective Mass		
		X Direction	Y Direction	Z Direction
1	3.619	0.000	41.959	0.000
2	5.387	175.274	0.000	0.175
3	6.192	0.000	148.385	0.005
4	6.415	3.321	0.000	24.074
5	9.422	0.002	1.017	0.000
6	9.674	10.510	0.000	0.532
7	12.811	0.015	0.001	0.000
8	15.757	0.004	0.320	0.010
9	16.367	3.103	0.003	159.153
10	17.495	28.537	0.001	19.546
11	18.944	0.000	40.053	0.001
12	21.043	10.724	0.000	0.426
13	22.102	0.000	0.005	0.000
14	27.340	0.054	0.000	18.661
15	30.387	2.978	0.001	1.559
16	31.577	0.002	3.526	0.004
17	35.033	0.194	0.006	3.895
18	35.535	0.211	0.027	0.399
19	35.646	0.000	1.451	0.019
20	37.599	0.325	0.426	0.007
Sum of Effective Masses		235.254	237.181	228.465

Notes:
1. Fixed at Elevation 100′.
2. The total mass of the containment vessel with the polar crane is 255.85 kip-sec^2/ft.

Table 3G.2-2 Comparison of Frequencies for Containment Vessel Seismic Model

Mode No.	Vertical Model		Horizontal Model	
	Shell of Revolution Model	Stick Model	Shell of Revolution Model	Stick Model
1	16.51 hertz	16.97 hertz	6.20 hertz	6.31 hertz
2	23.26 hertz	28.20 hertz	18.58 hertz	18.96 hertz

Note:
1. Fixed at elevation 100′.

Table 3G.3-1
AP1000 ZPA for 2D SASSI Cases

	North-South		Hard Rock ZPA [g]	Firm Rock ZPA [g]	Soft Rock ZPA [g]	UBSM ZPA [g]	SM ZPA [g]	Soft Soil ZPA [g]
	Node	**El. feet**						
ASB	21	81.5	0.326	0.326	0.345	**0.358**	0.306	0.249
	41	99	0.348	0.327	0.347	**0.361**	0.308	0.227
	120	179.6	**0.571**	0.501	0.469	0.498	0.529	0.247
	150	242.5	0.803	0.795	0.816	**0.819**	0.787	0.29
	310	333.1	1.449	1.561	**1.567**	1.524	1.226	0.453
SCV	407	138.6	0.405	**0.424**	0.408	0.387	0.407	0.232
	411	200	0.82	**0.916**	0.672	0.541	0.484	0.263
	417	281.9	1.396	**1.465**	1.031	0.723	0.598	0.372
CIS	535	134.3	**0.548**	0.45	0.347	0.368	0.355	0.229
	538	169	**1.517**	0.874	0.45	0.441	0.397	0.317
	East-West		**Hard Rock ZPA [g]**	**Firm Rock ZPA [g]**	**Soft Rock ZPA [g]**	**UBSM ZPA [g]**	**SM ZPA [g]**	**Soft Soil ZPA [g]**
	Node	**El. feet**						
ASB	21	81.5	0.309	0.318	0.359	**0.376**	0.311	0.235
	41	99	0.318	0.336	0.367	**0.385**	0.317	0.237
	120	179.6	**0.607**	0.561	0.546	0.549	0.605	0.295
	150	242.5	0.84	0.823	0.854	0.912	**0.962**	0.557
	310	333.1	1.449	1.536	1.624	**1.74**	1.506	0.891
SCV	407	138.6	0.528	0.529	**0.535**	0.513	0.38	0.247
	411	200	0.817	**0.95**	0.816	0.741	0.515	0.429
	417	281.9	1.251	**1.503**	1.136	0.985	0.716	0.675
CIS	535	134.3	**0.52**	0.404	0.391	0.404	0.365	0.259
	538	169	**1.679**	1.052	0.755	0.553	0.526	0.441

Table 3G.4-1
Key Nodes at Location

Location	General Area	Elevation (feet)
CIS at Reactor Vessel Support Elevation	SCV Center	100.00
CIS at Operating Deck	SG West Compartment, NE	134.25
ASB NE Corner at Control Room Floor	NE Corner	116.50
ASB Corner of Fuel Building Roof at Shield Building	NW Corner of Fuel Bldg	179.19
ASB Shield Building Roof Area	South Side of Shield Bldg	327.41
SCV Near Polar Crane	SCV Stick Model	224.00

Table 3G.4-2
Maximum Bearing Pressure from 2D Time History Analyses

Soil Case	Analysis	East Edge (ksf)	West Edge (ksf)
Hard Rock	Linear	17.18	32.77
	Liftoff	17.38	34.85
Upper-bound Soft to Medium	Linear	19.46	31.69
	Liftoff	18.42	33.51
Soft to Medium	Linear	15.84	30.82
	Liftoff	17.06	32.18

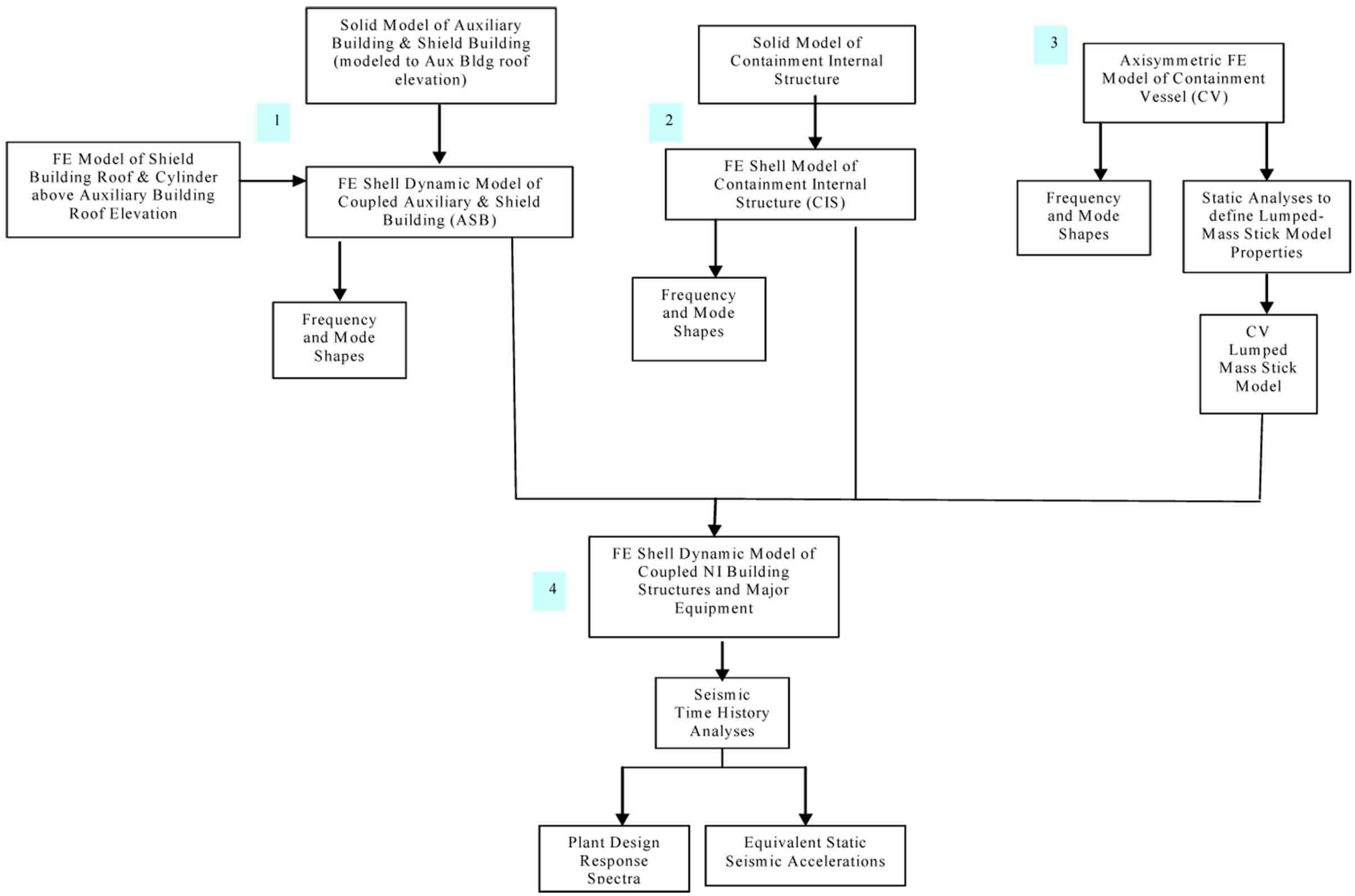


Figure 3G.1-1
Nuclear Island Seismic Analysis Models

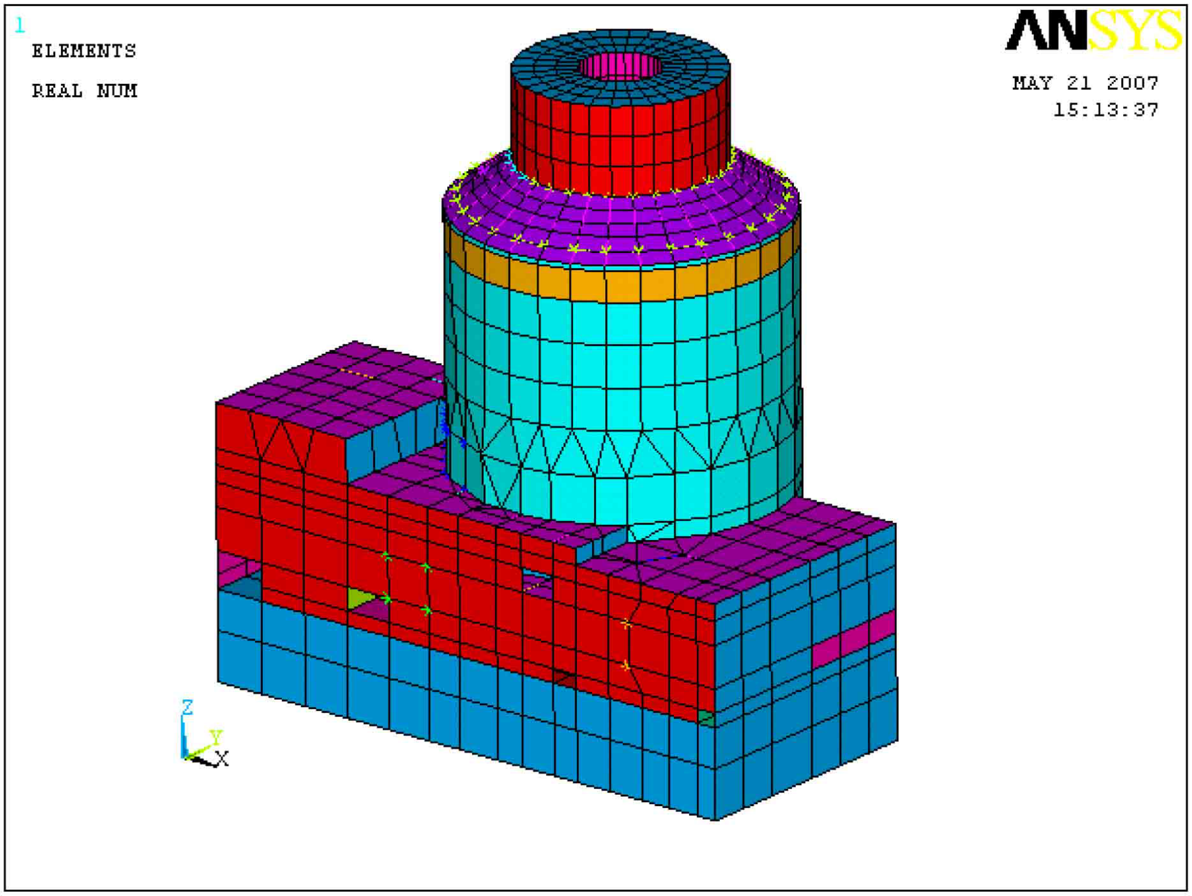
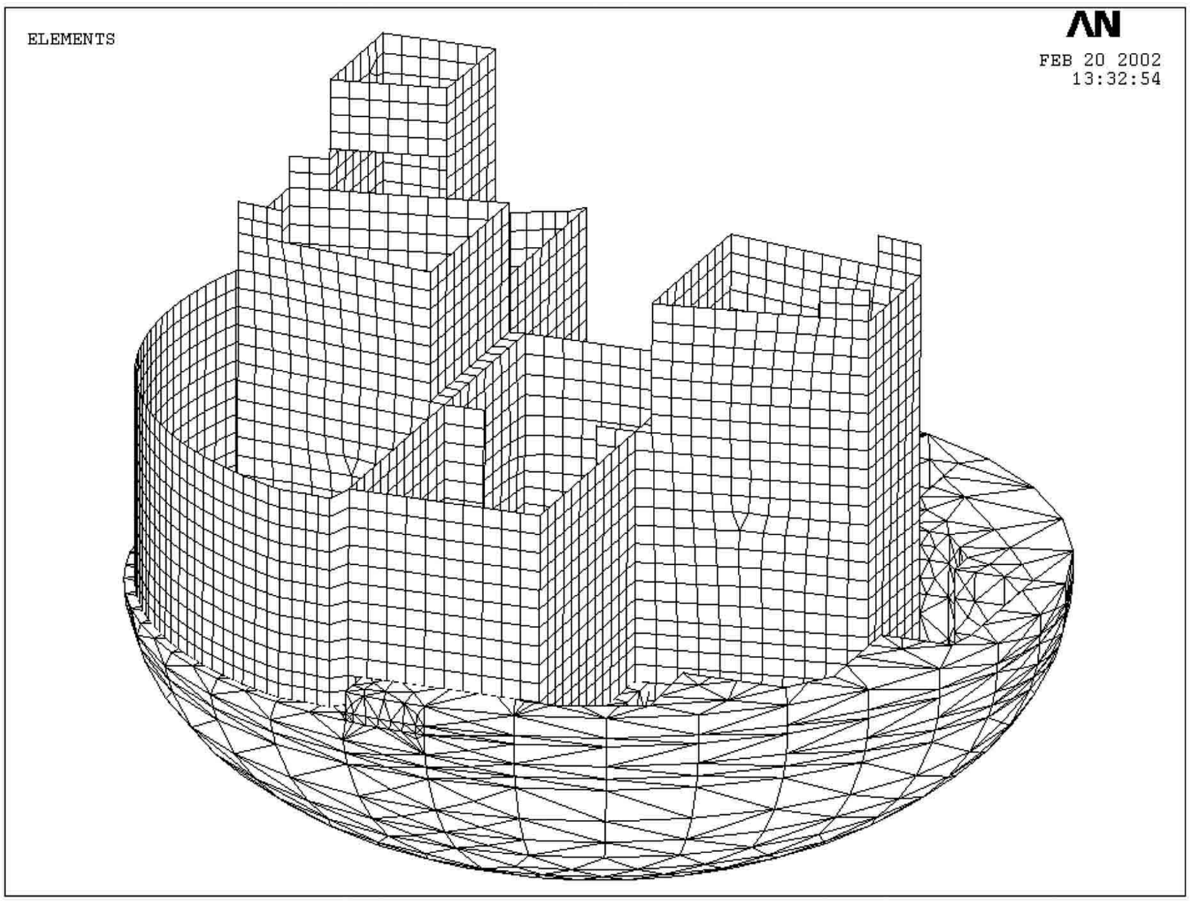


Figure 3G.2-1
3D Finite Element Model of Coupled Shield and Auxiliary Building



Note: This figure shows the finite element model of walls and basemat inside containment.
Floors are not shown.

Figure 3G.2-2
3D Finite Element Model of Containment Internal Structures

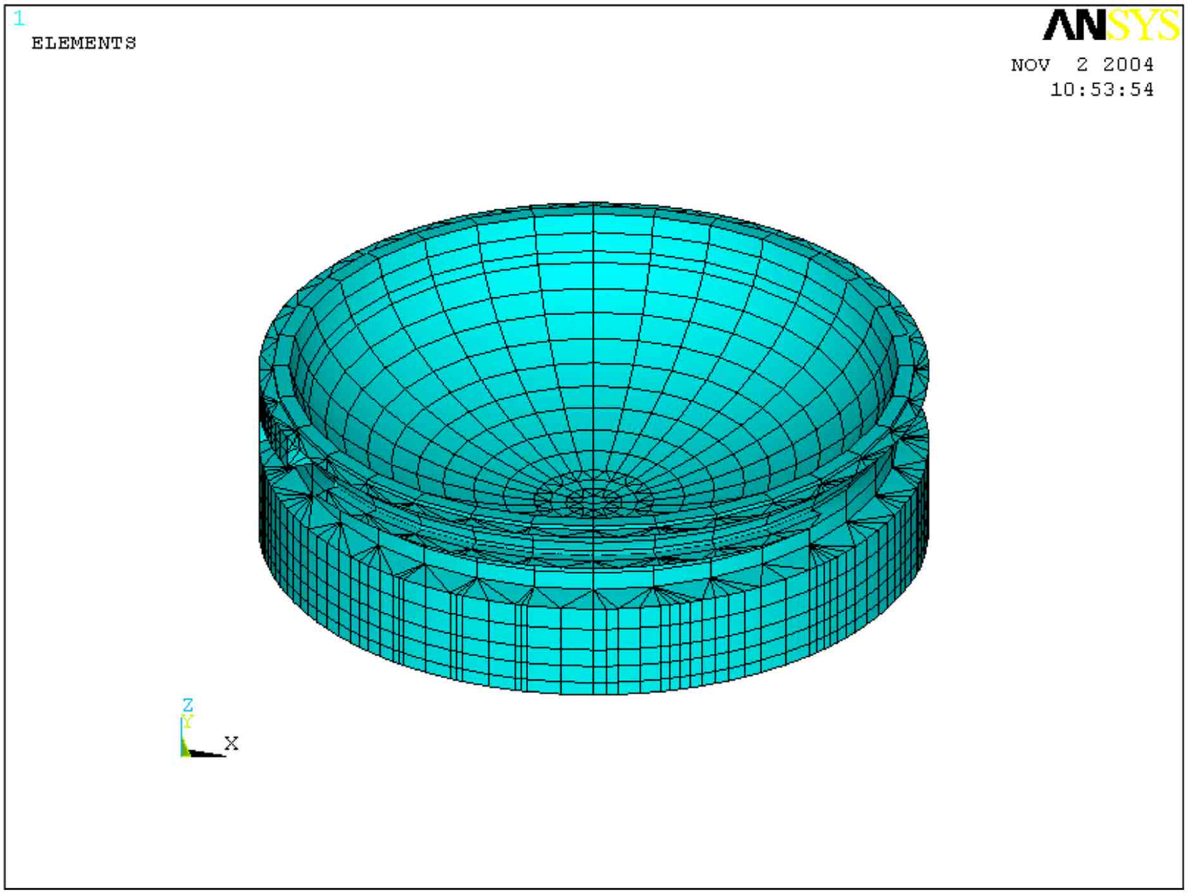


Figure 3G.2-3
3D Finite Element Model of Containment Outer Basemat (Dish)

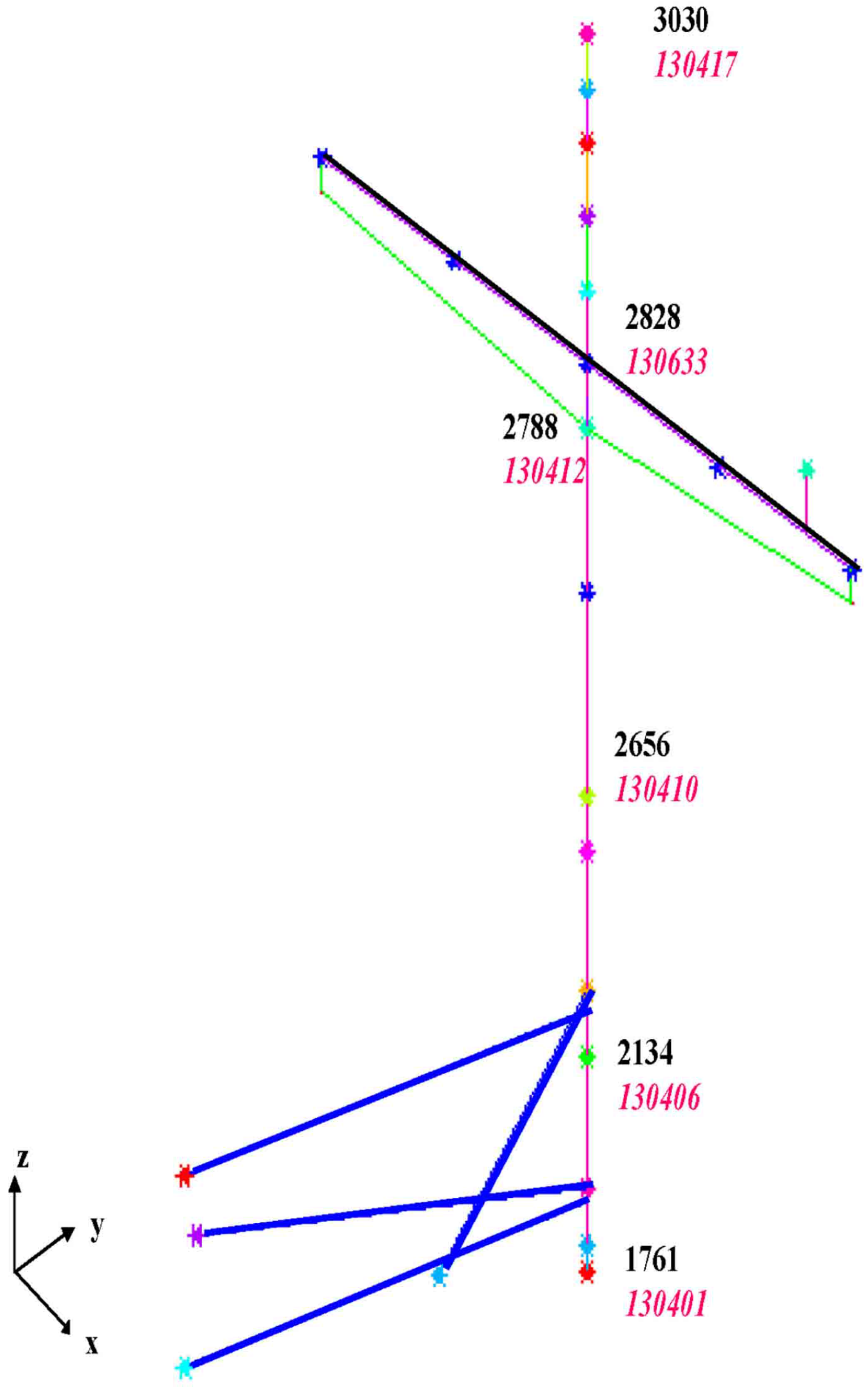
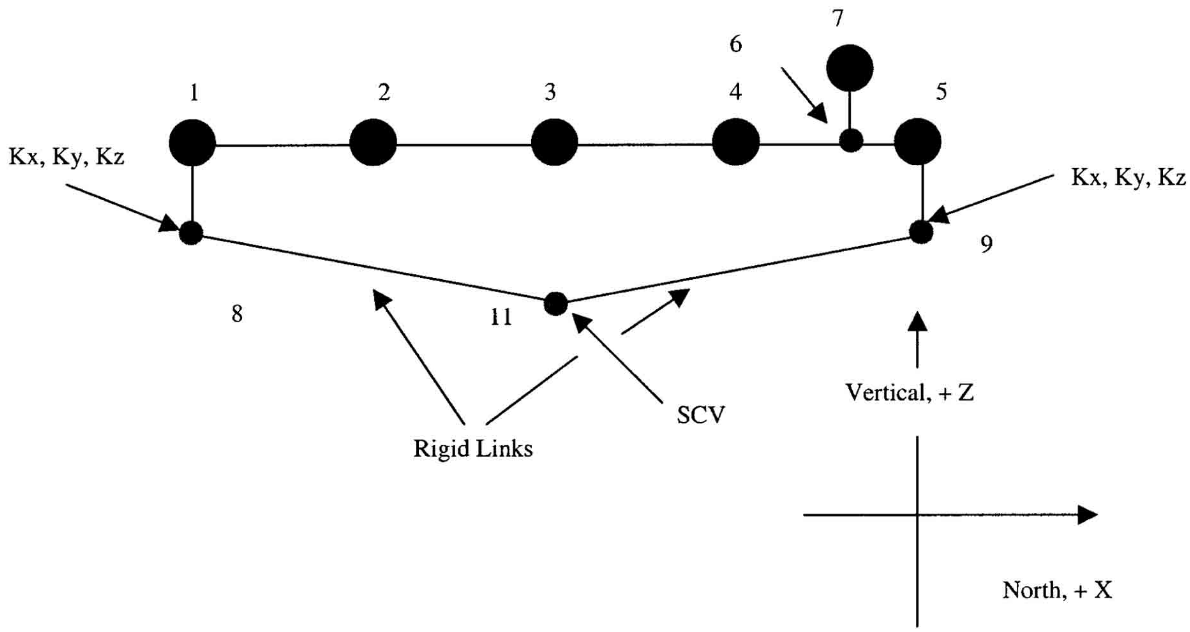


Figure 3G.2-4
Steel Containment Vessel and Polar Crane Models



Local SCV Stiffness are Kx, Ky, Kz

Dynamic Degrees of Freedom

- Masses at nodes 1, 2, 3, 4, 5, and 7
- All Mass nodes have DOFs in X, Y, and Z directions

Comments:

1. Cross Beams between girders are represented by rotation spring constants Kxx and Kzz
2. Cross Beam rotational spring constant Kyy is negligible compared to girder stiffness

Figure 3G.2-5A
Polar Crane Model Simplified Model

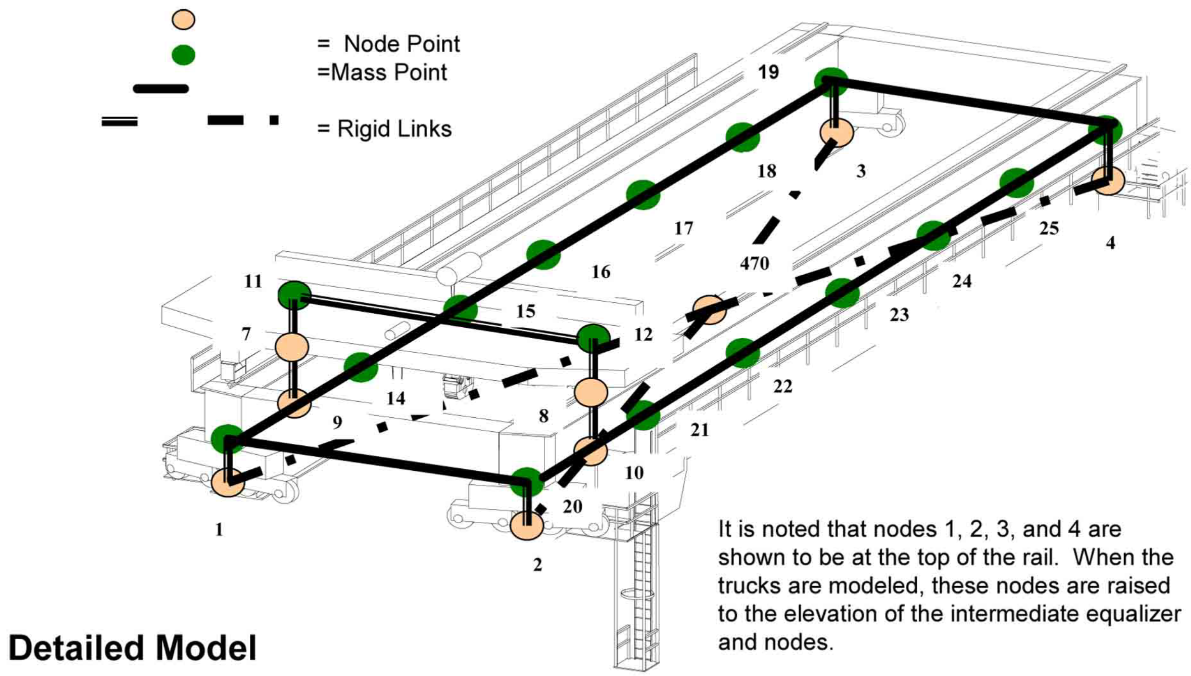


Figure 3G.2-5B
Polar Crane Model Detailed Model

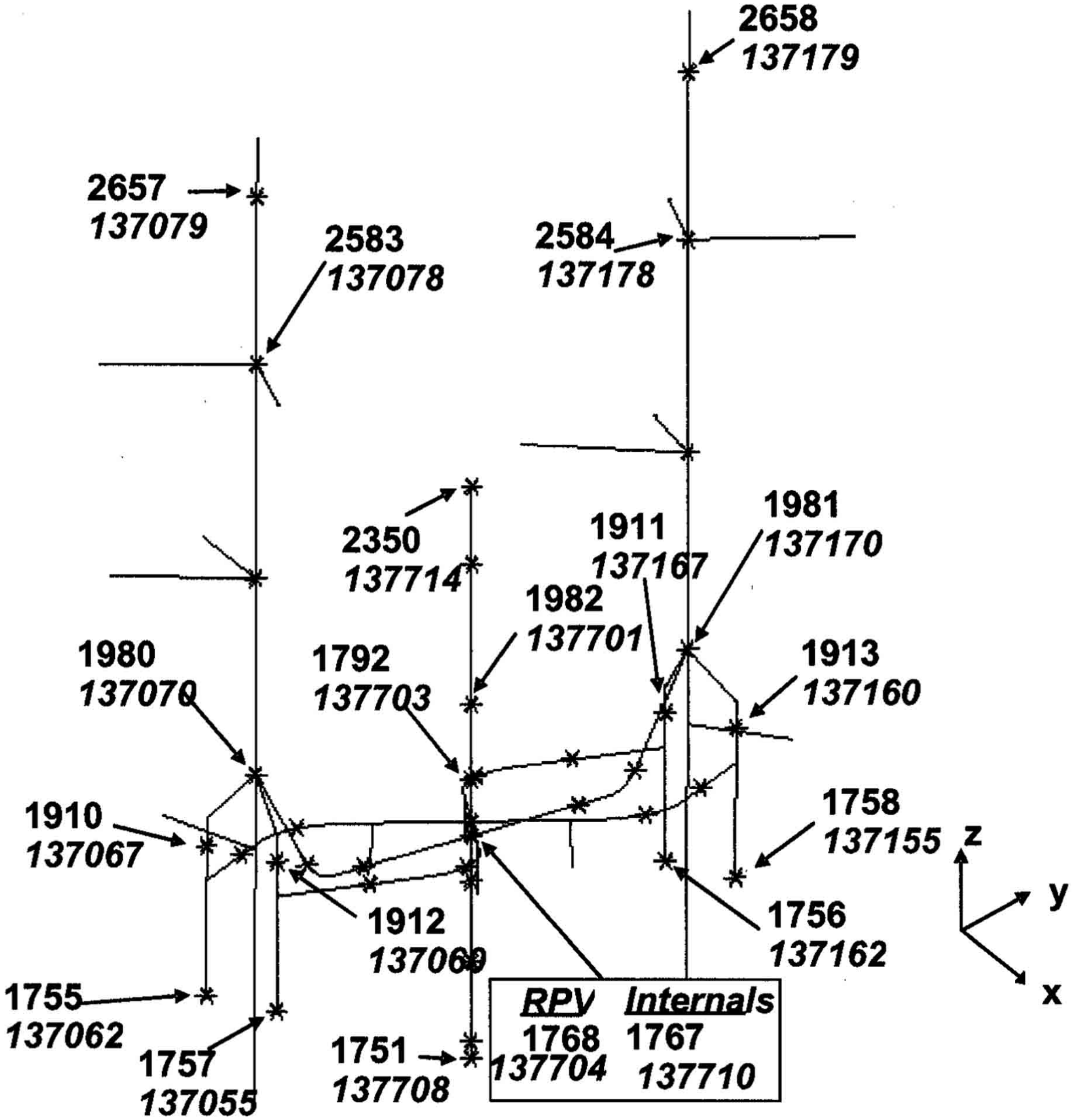


Figure 3G.2-6
Reactor Coolant Loop Lumped-Mass Stick Model

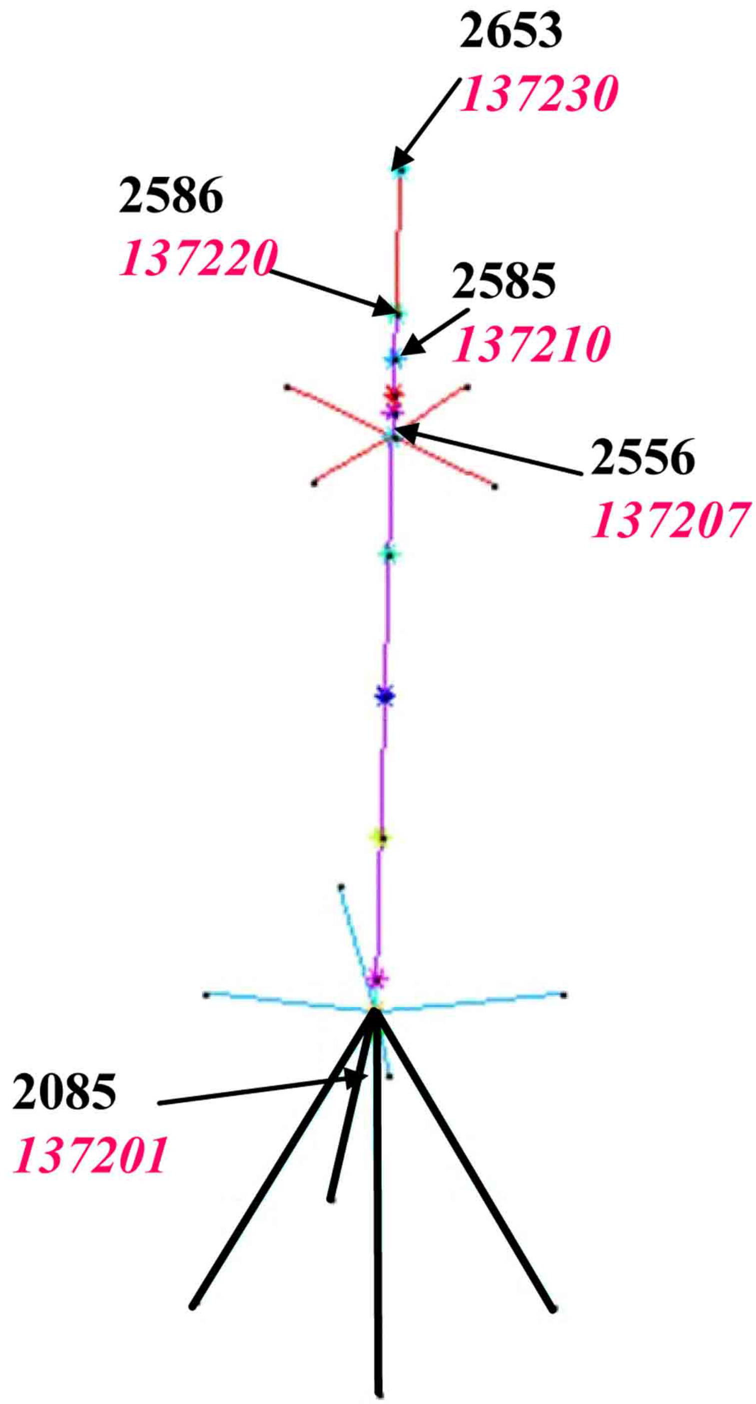


Figure 3G.2-7
Pressurizer Model

Figure 3G.2-8
Core Makeup Tank Models

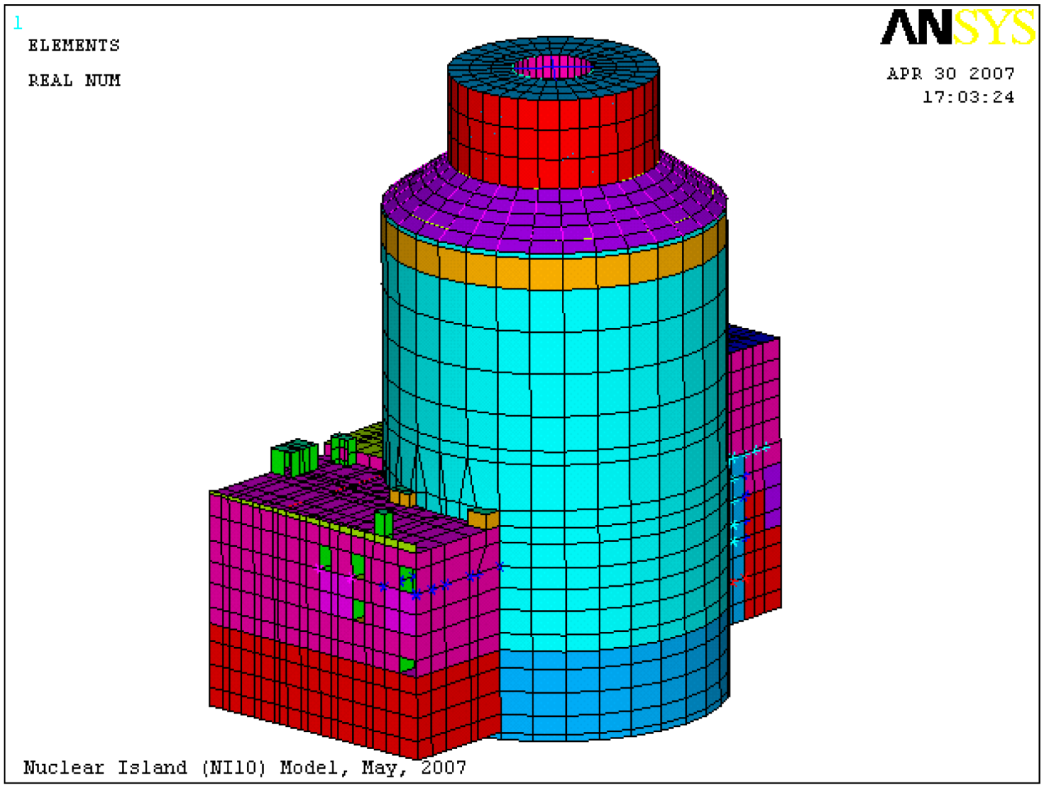


Figure 3G.2-9
AP1000 Nuclear Island Solid-Shell Model (NI10)

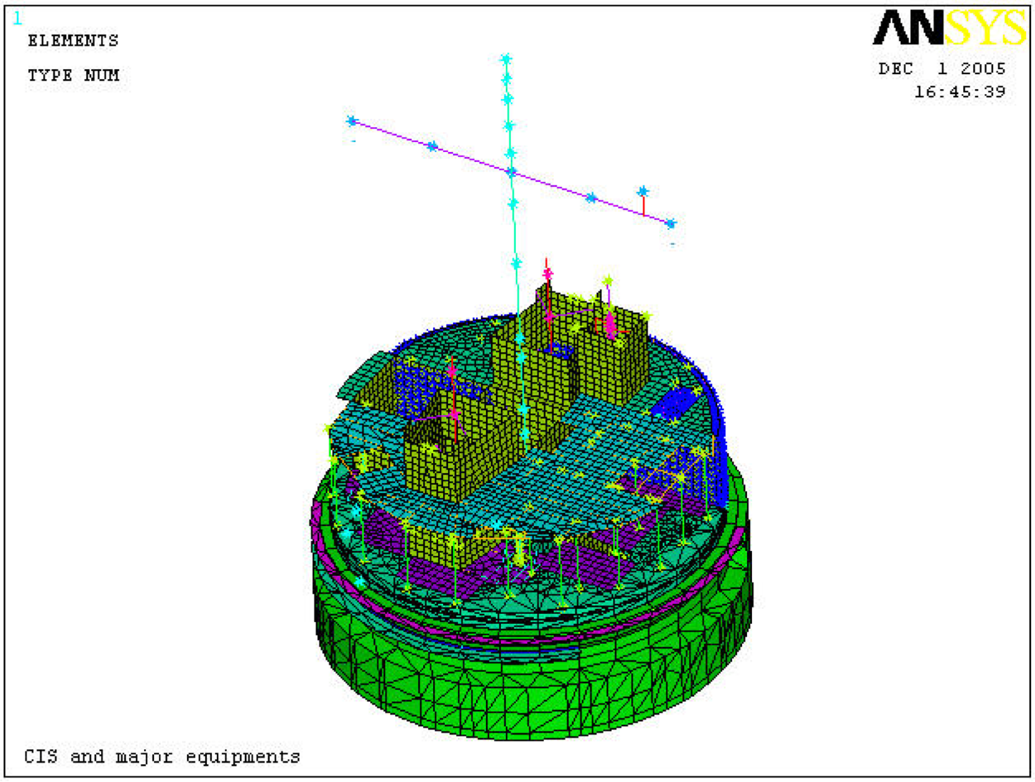
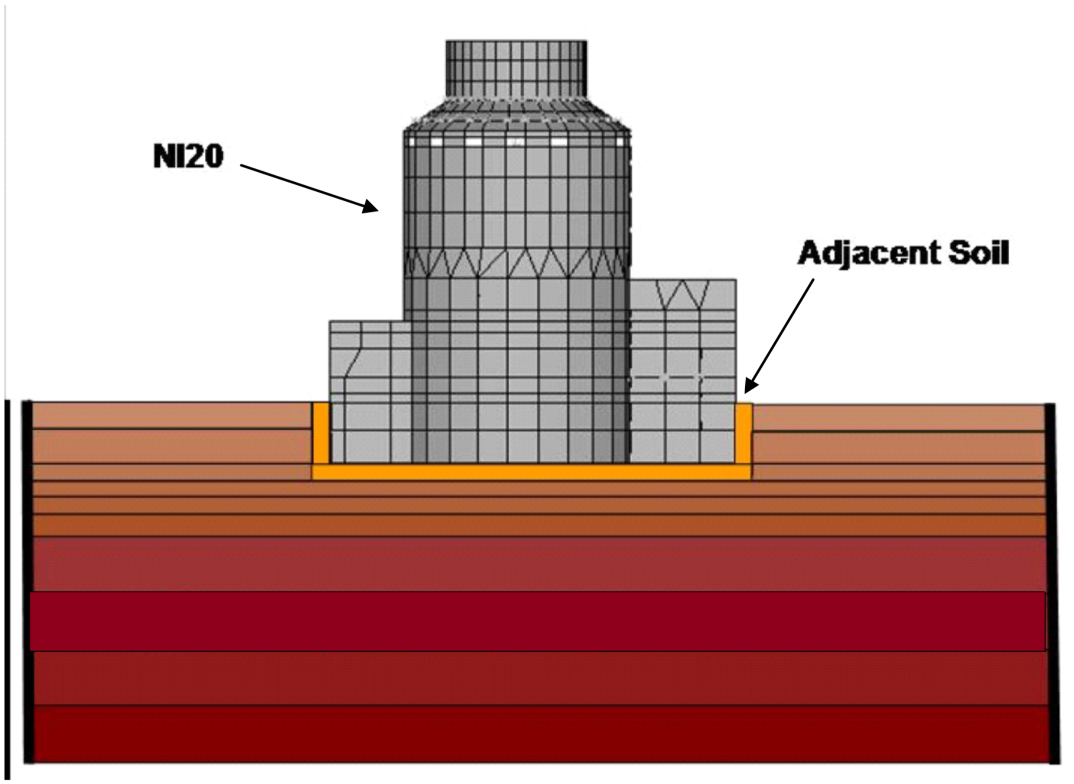
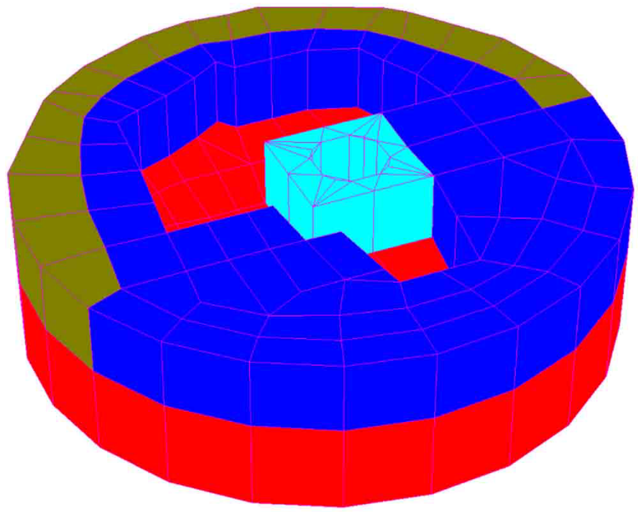


Figure 3G.2-10
Containment Internal Structure with the SCV, PC, Reactor Coolant Loop, and Pressurizer



Note: The adjacent soil elements are part of the structural portion of SASSI and have the same material properties as the soil. These elements are used to obtain soil lateral and bearing soil pressures.

Figure 3G.2-11
Soil Structure Interaction Model – NI20 Looking East



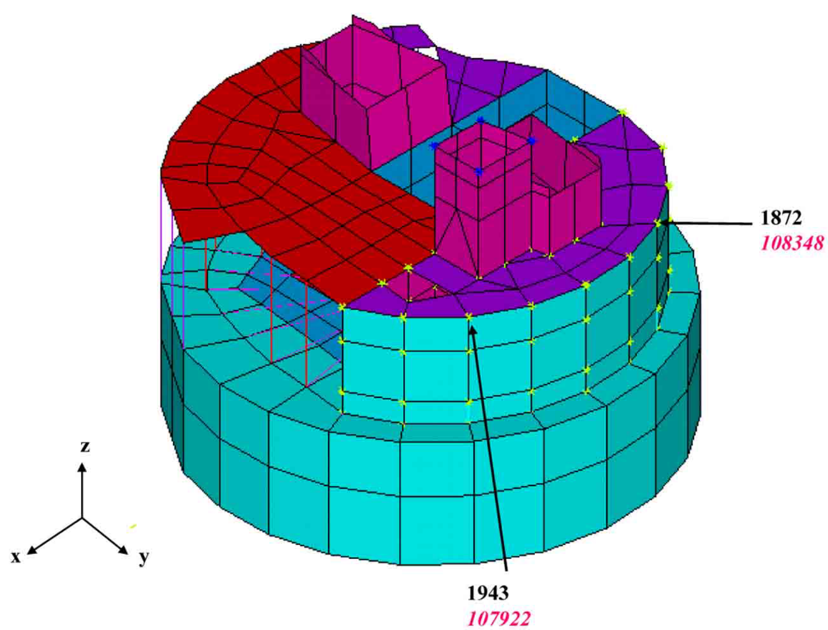


Figure 3G.2-12
Coarse Model of Containment Internal Structures

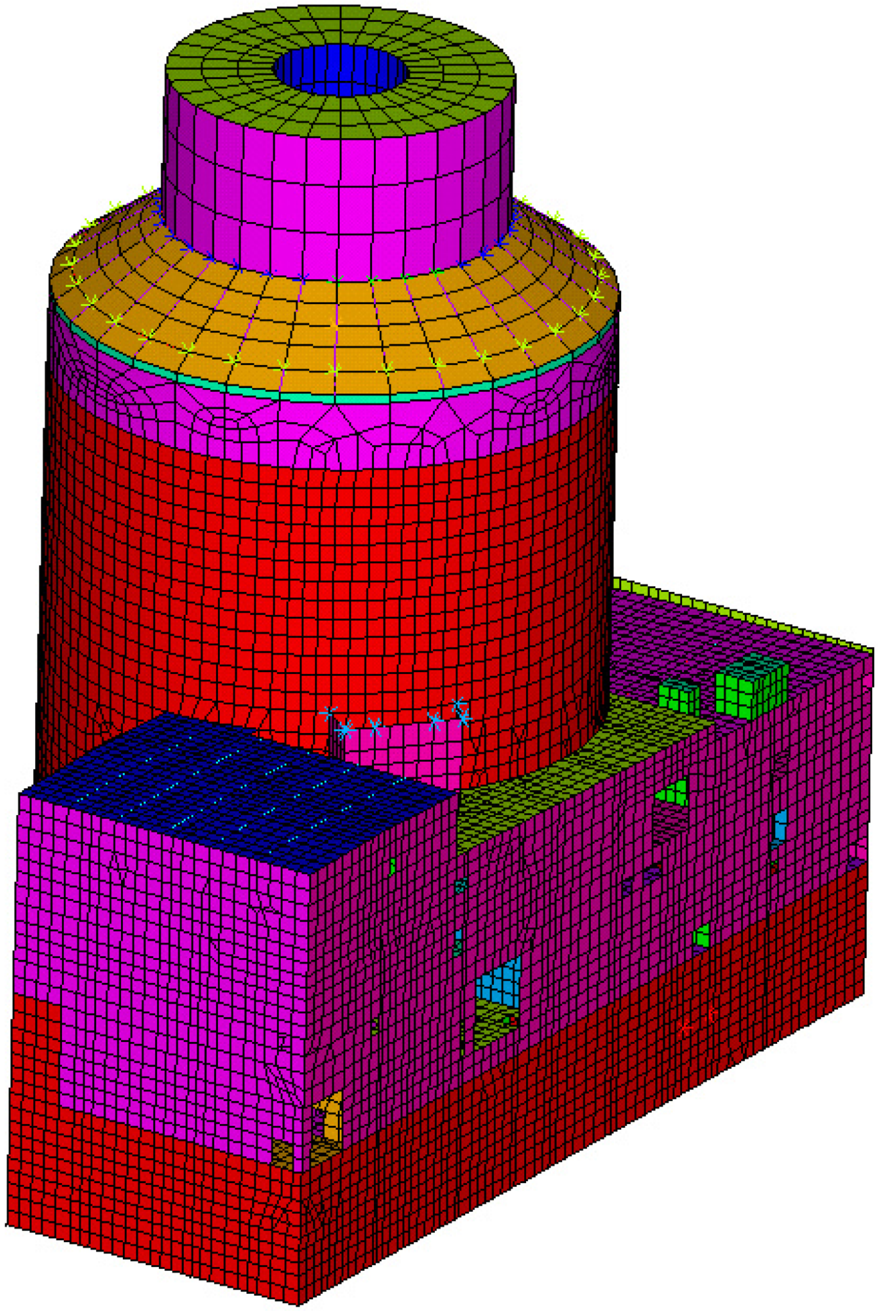


Figure 3G.2-13
Fine Mesh (NI05) Model of Auxiliary and Shield Building

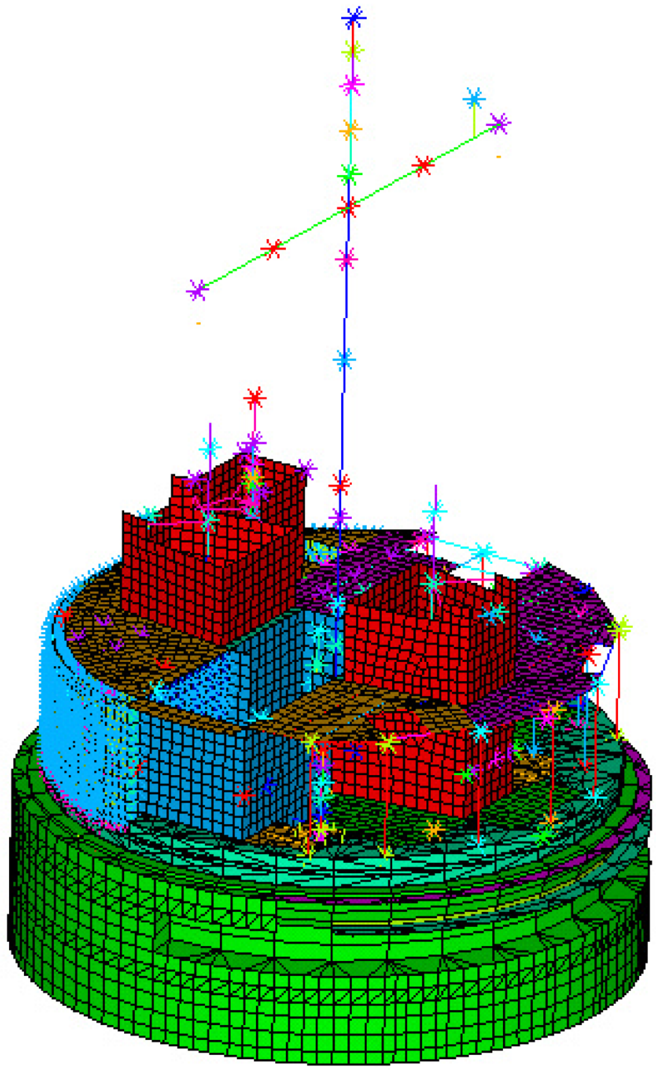


Figure 3G.2-14
NI05 Model of Containment Internal Structures

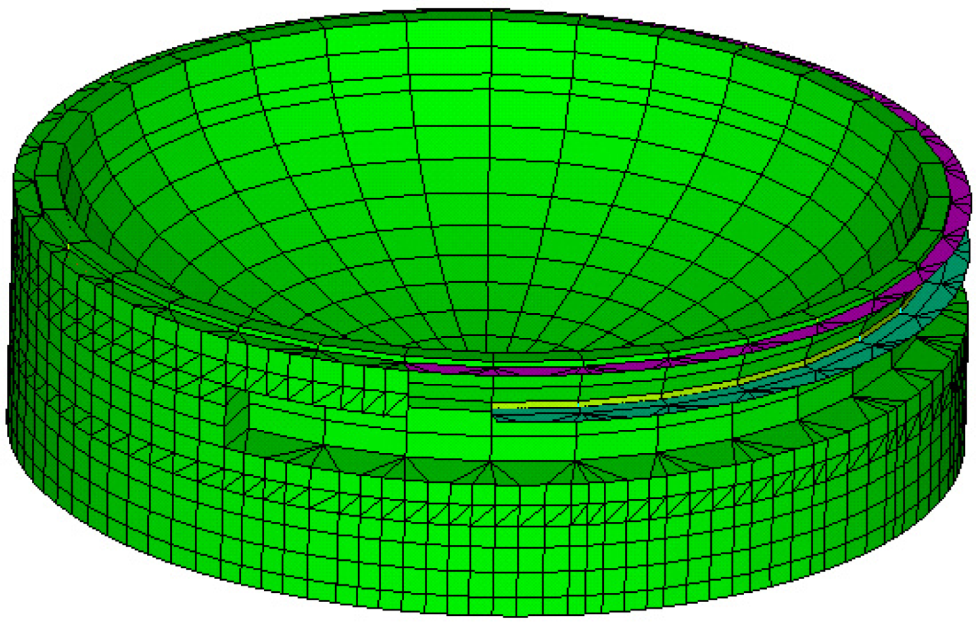


Figure 3G.2-15
3D NI05 Refined Mesh Model of Outer Containment Basemat (Dish)

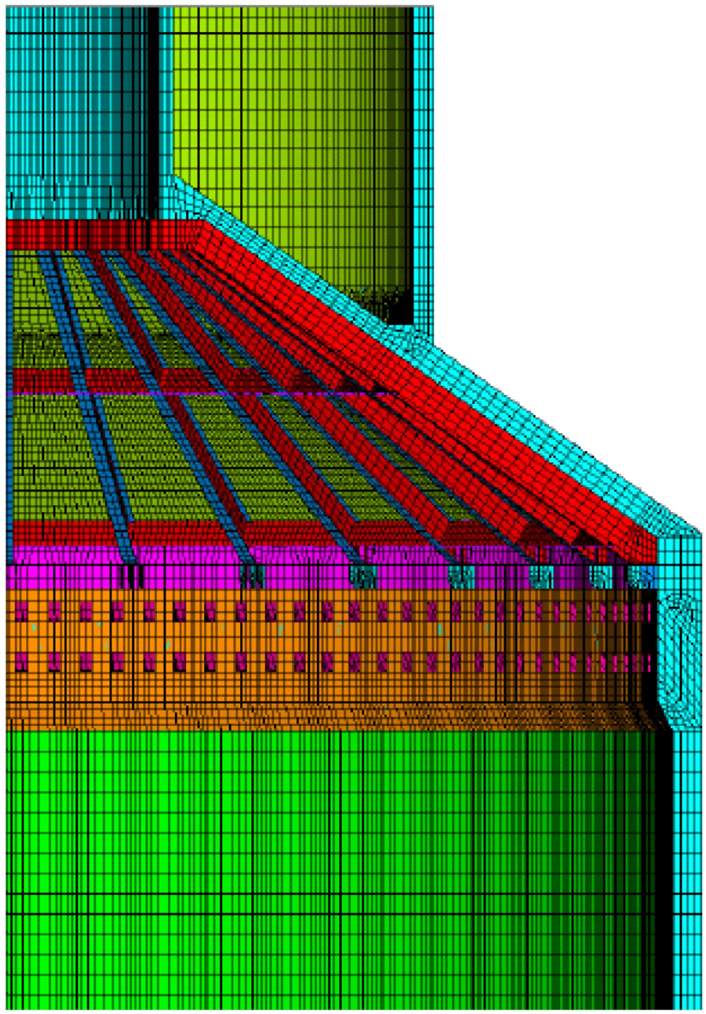
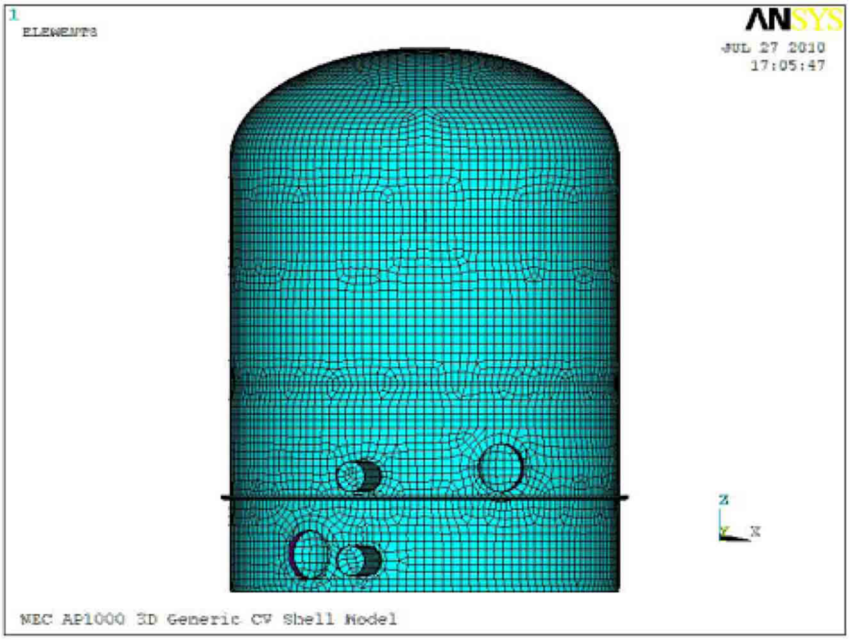
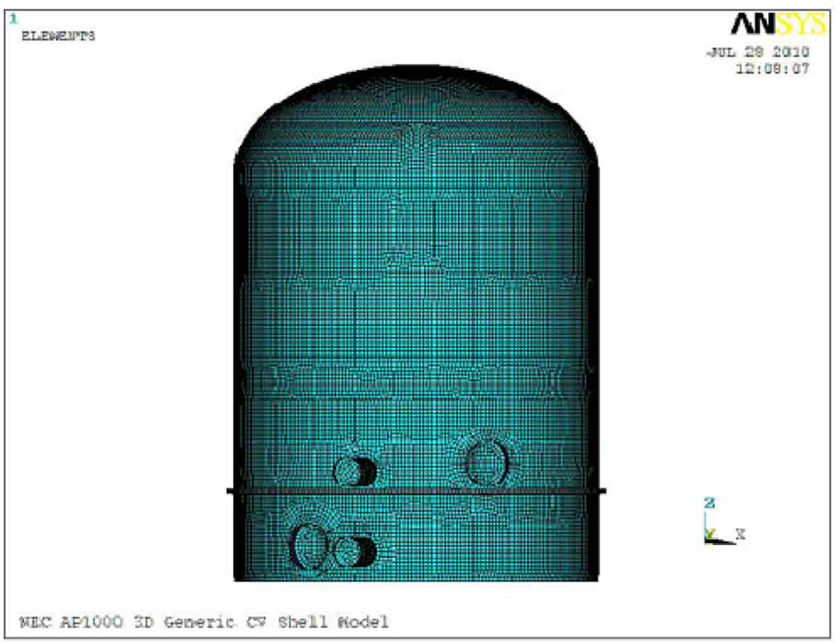


Figure 3G.2-16
Quadrant Model of Shield Building Roof



Mesh Size 37″ x 37″



Mesh Size 18″ x 18″

Figure 3G.2-17
Detailed 3D Finite Element Model of Containment Vessel Including Large Penetrations

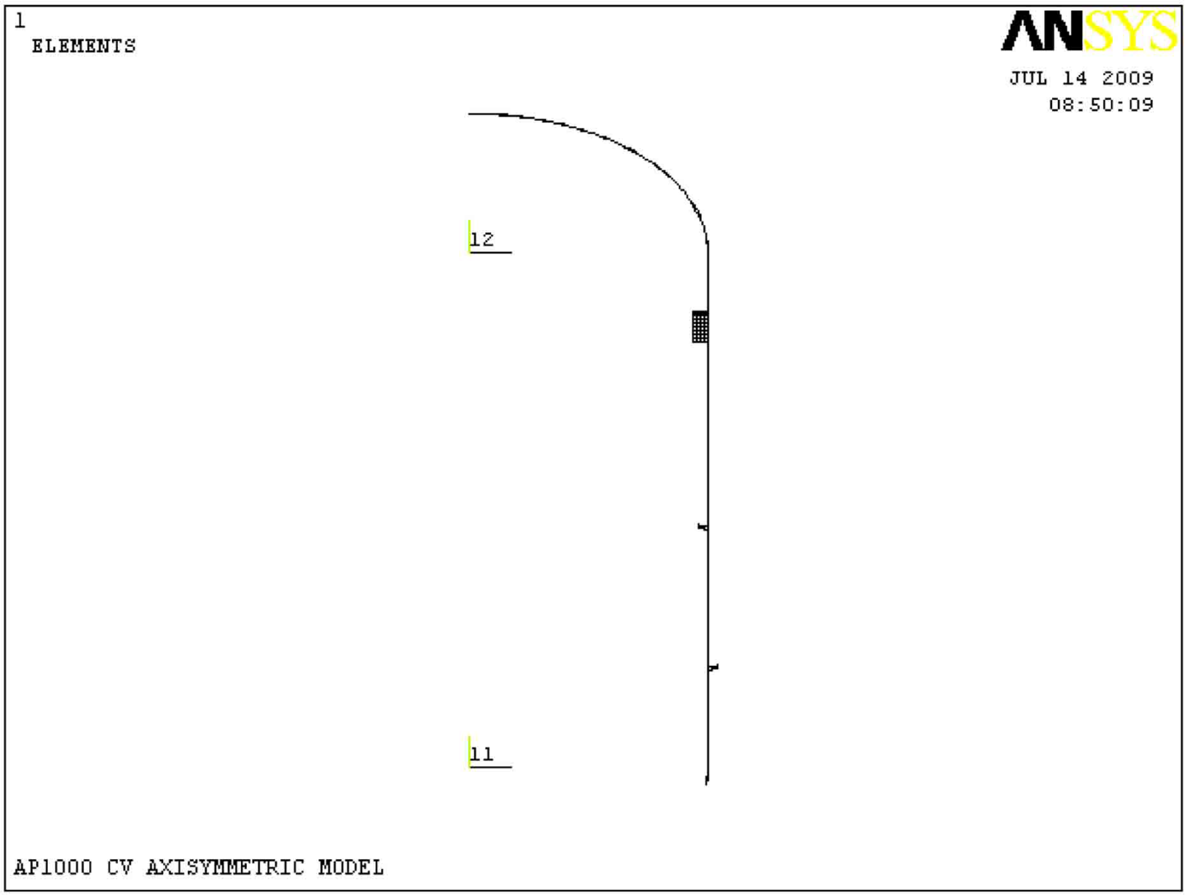


Figure 3G.2-18
Axisymmetric Model of Containment Vessel

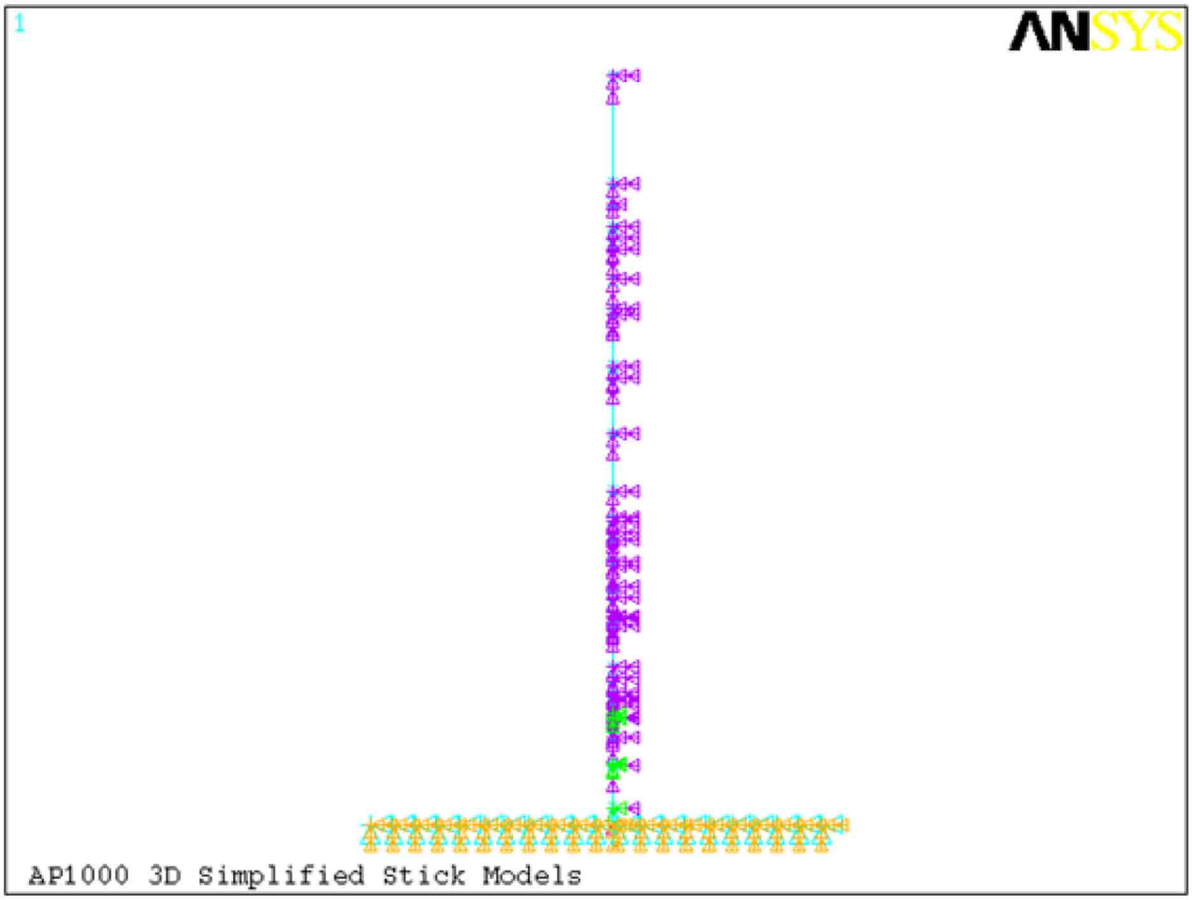
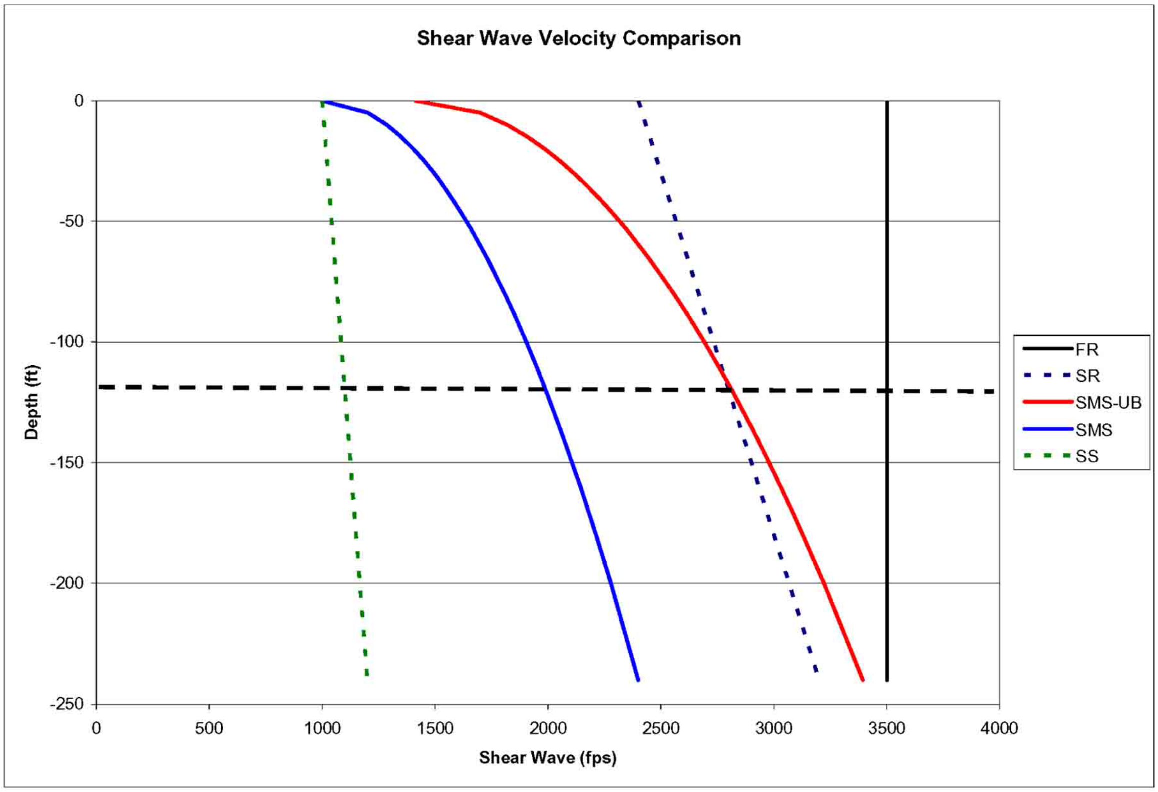


Figure 3G.2-19
Schematic of Non-linear 2D East-West Nuclear Island Stick Model Used for Stability Evaluation that Addresses Sliding and Overturning



Note: Fixed base analyses were performed for hard rock sites. These analyses are applicable for shear wave velocity greater than 8000 feet per second.

Figure 3G.3-1
Generic Soil Profiles

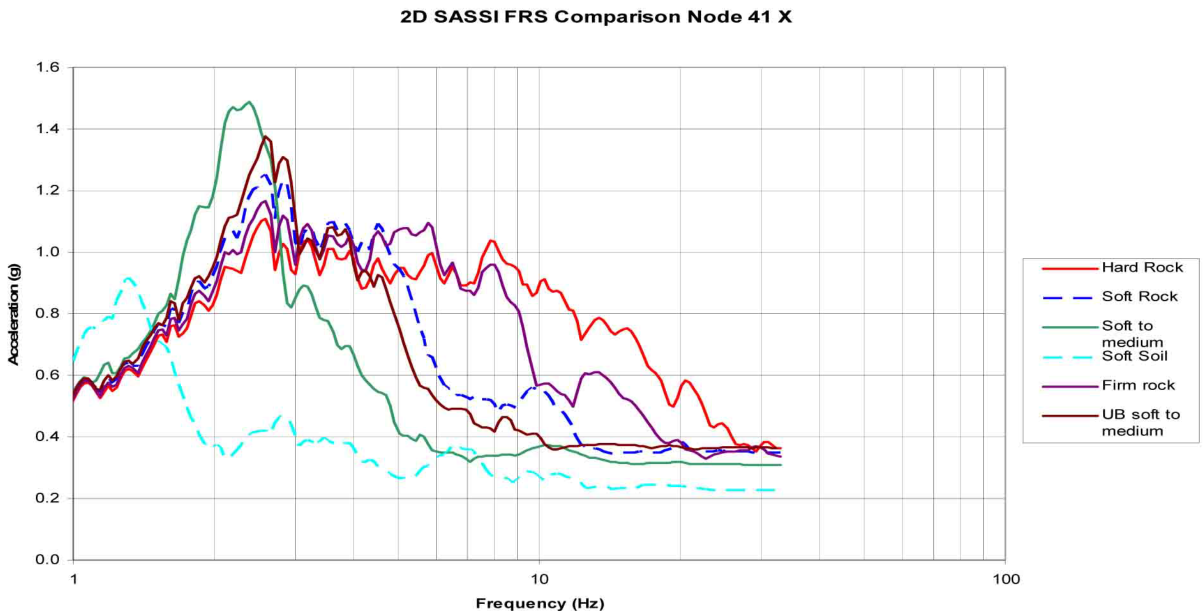


Figure 3G.3-2
2D SASSI FRS – Node 41 X (ASB El. 99′)

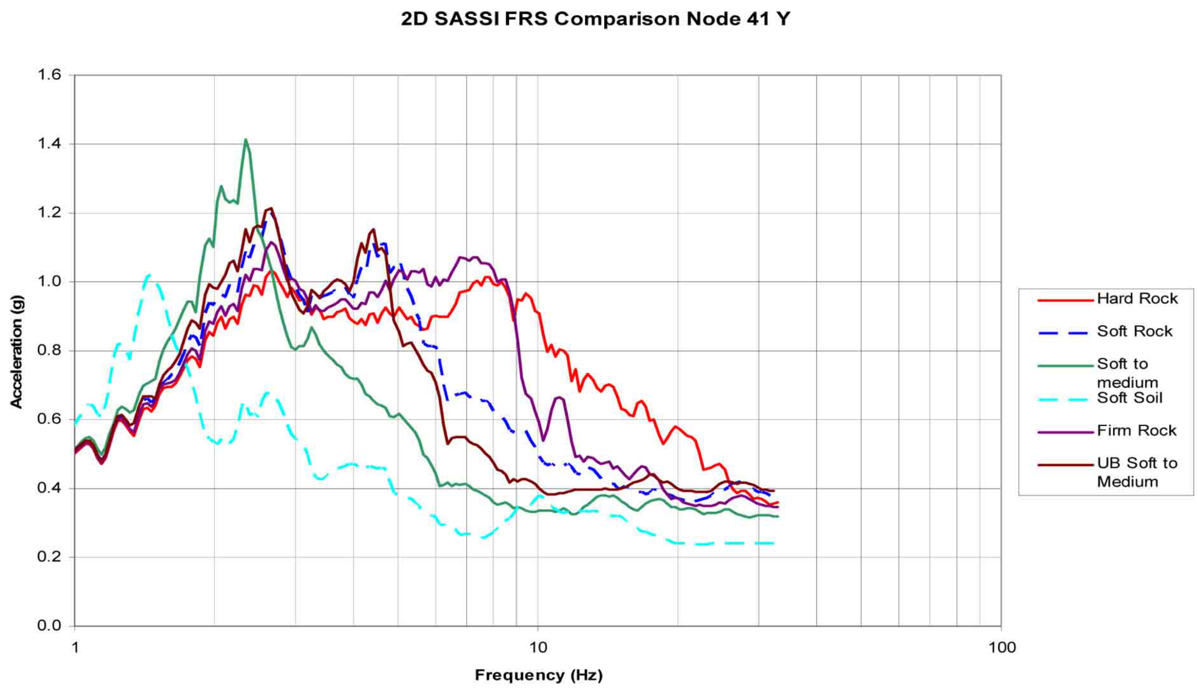


Figure 3G.3-3
2D SASSI FRS – Node 41 Y (ASB El. 99′)

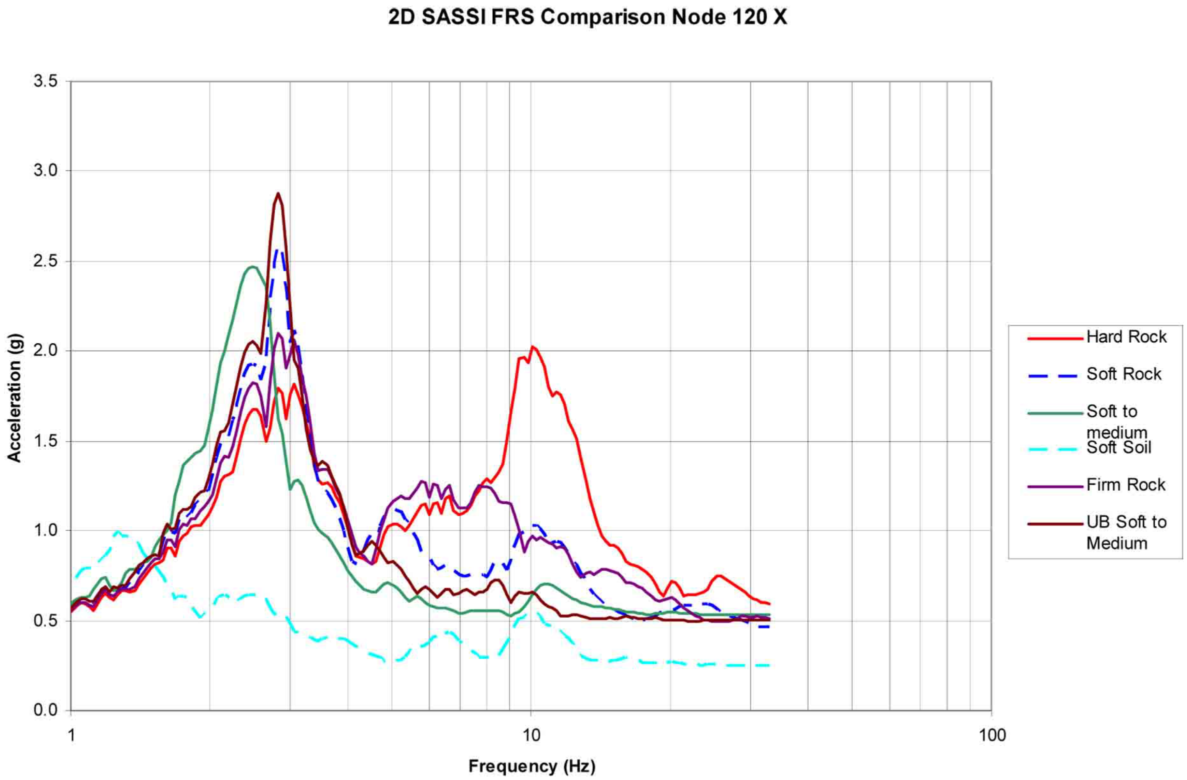


Figure 3G.3-4
2D SASSI FRS – Node 120 X (ASB El. 179.6′)

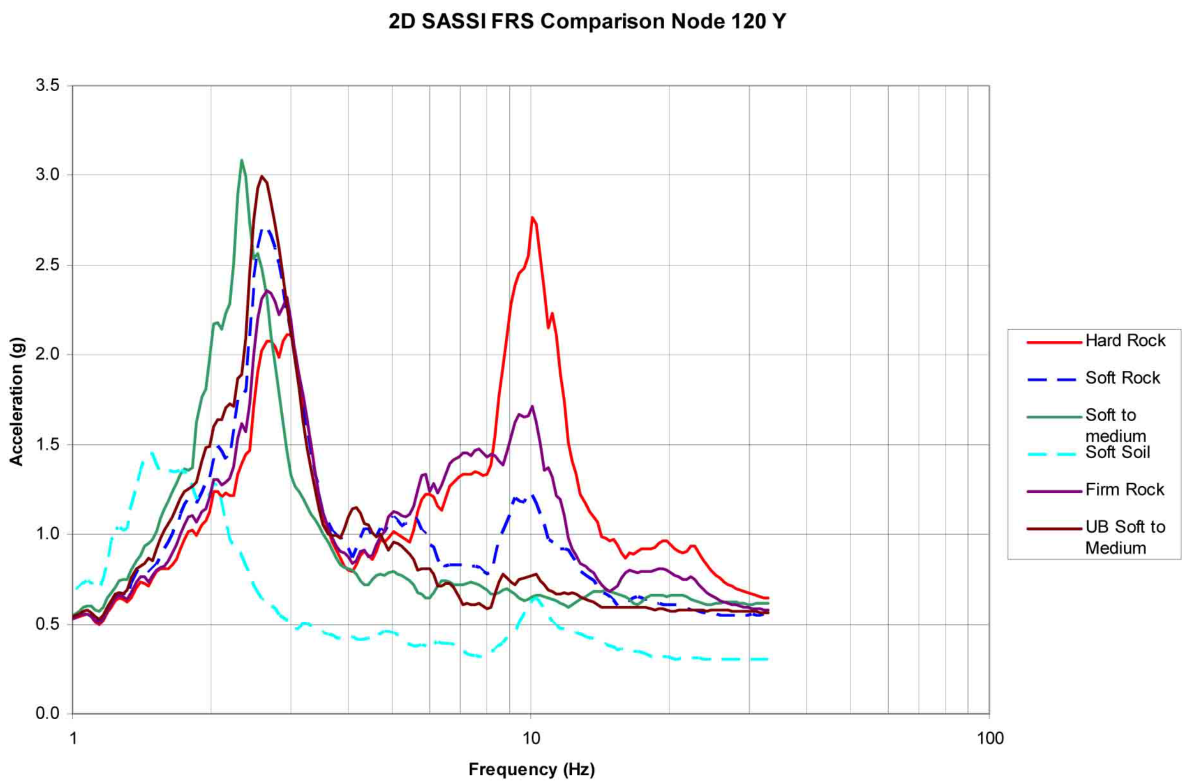


Figure 3G.3-5
2D SASSI FRS – Node 120 Y (ASB El. 179.6′)

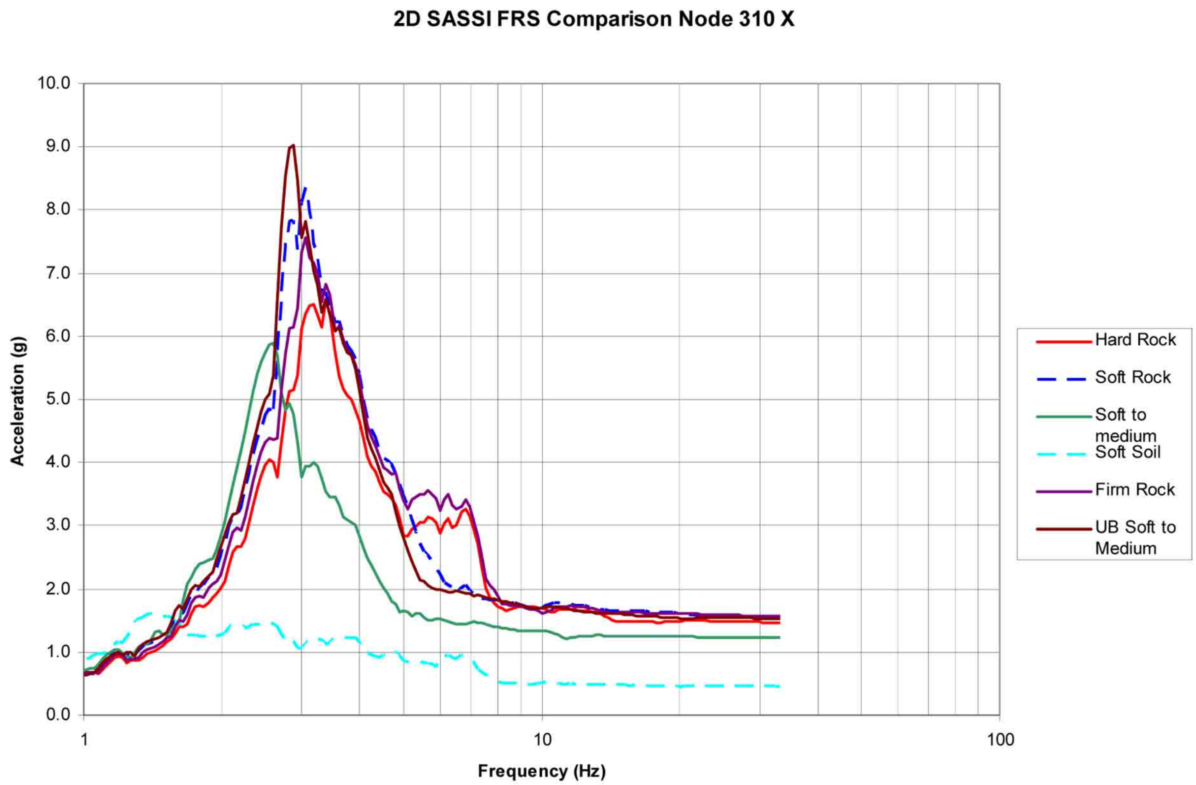


Figure 3G.3-6
2D SASSI FRS – Node 310 X (ASB El. 333.2′)

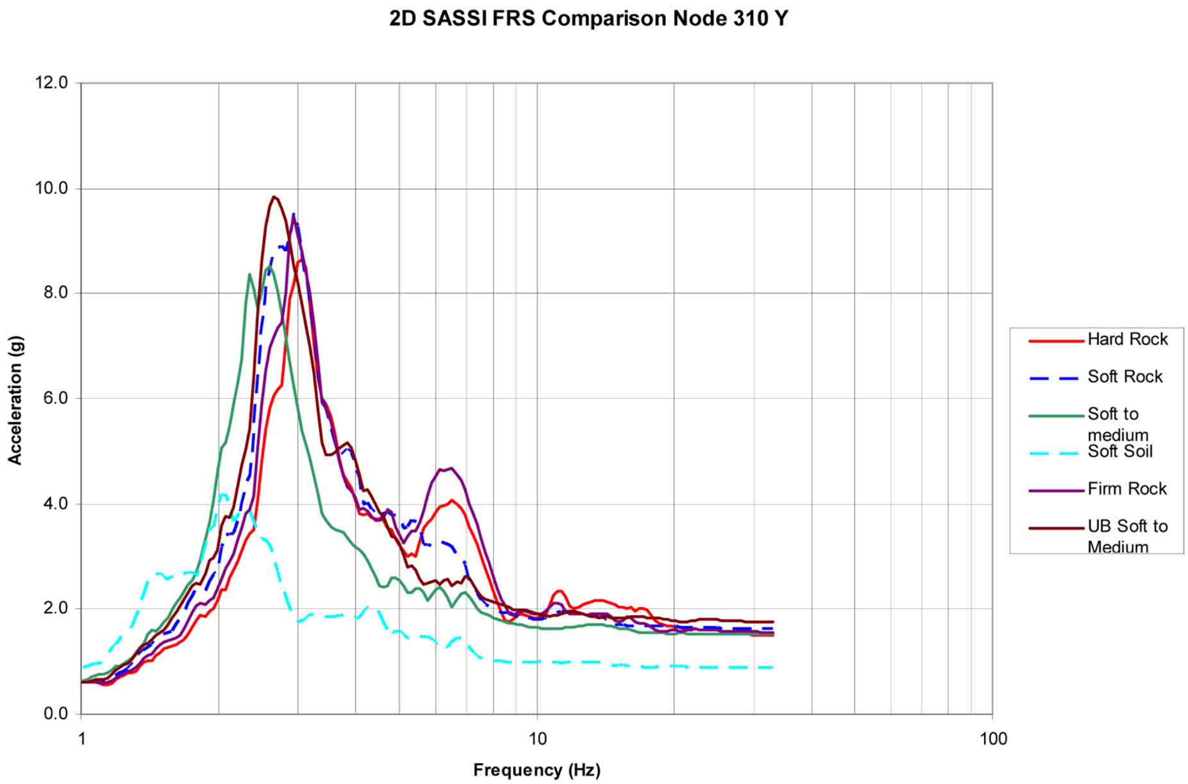


Figure 3G.3-7
2D SASSI FRS – Node 310 Y (ASB El. 333.2′)

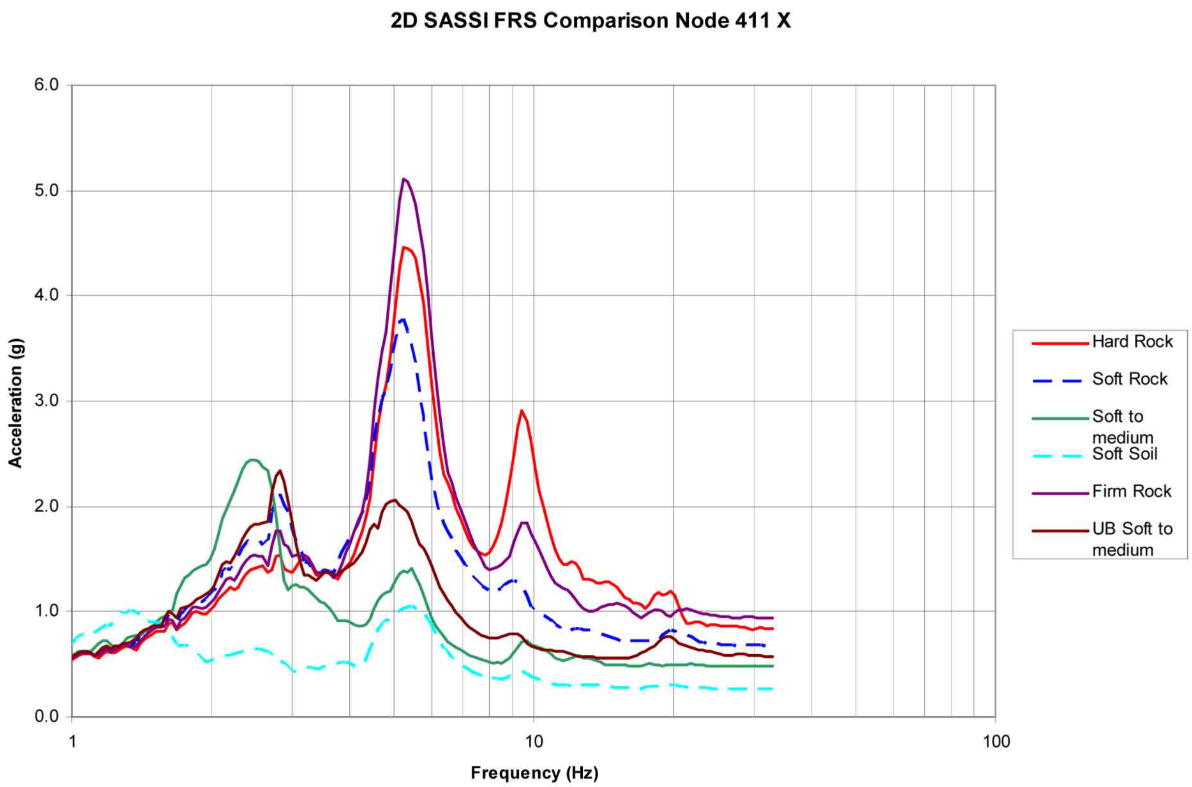


Figure 3G.3-8
2D SASSI FRS – Node 411 X (SCV El. 200.0′)

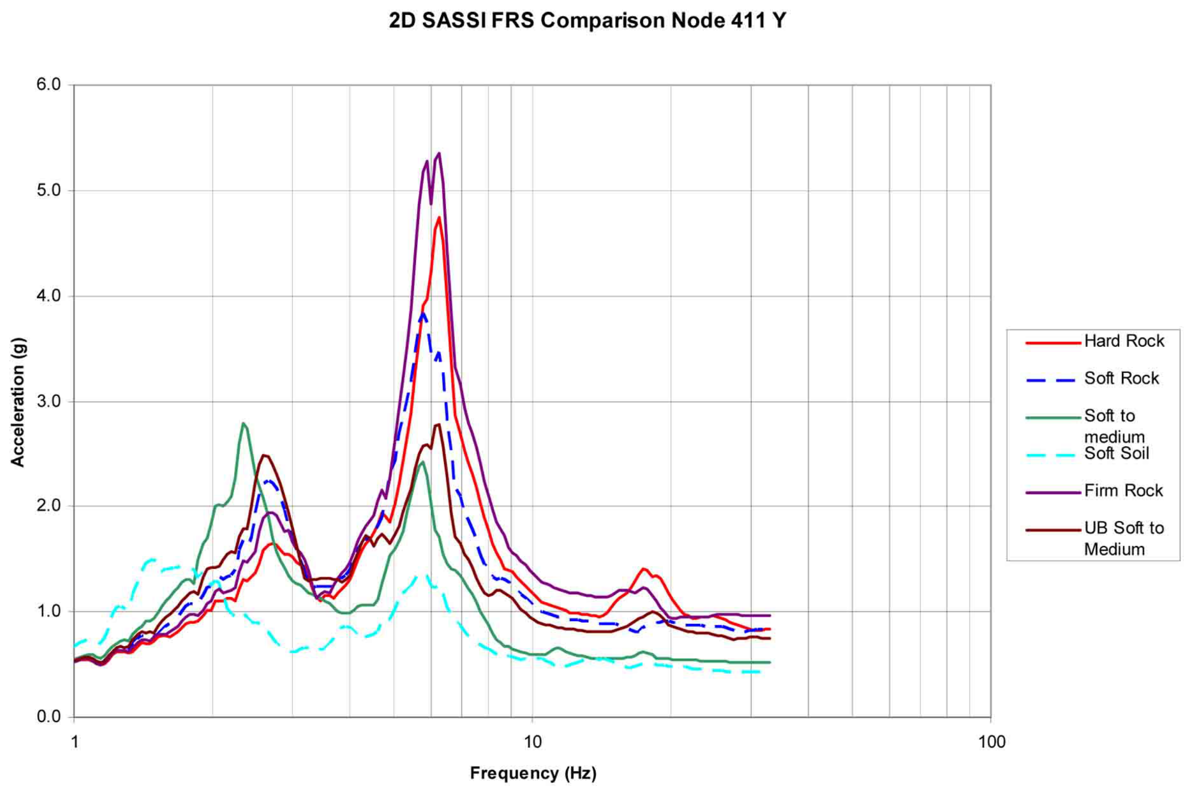


Figure 3G.3-9
2D SASSI FRS – Node 411 Y (SCV El. 200.0′)

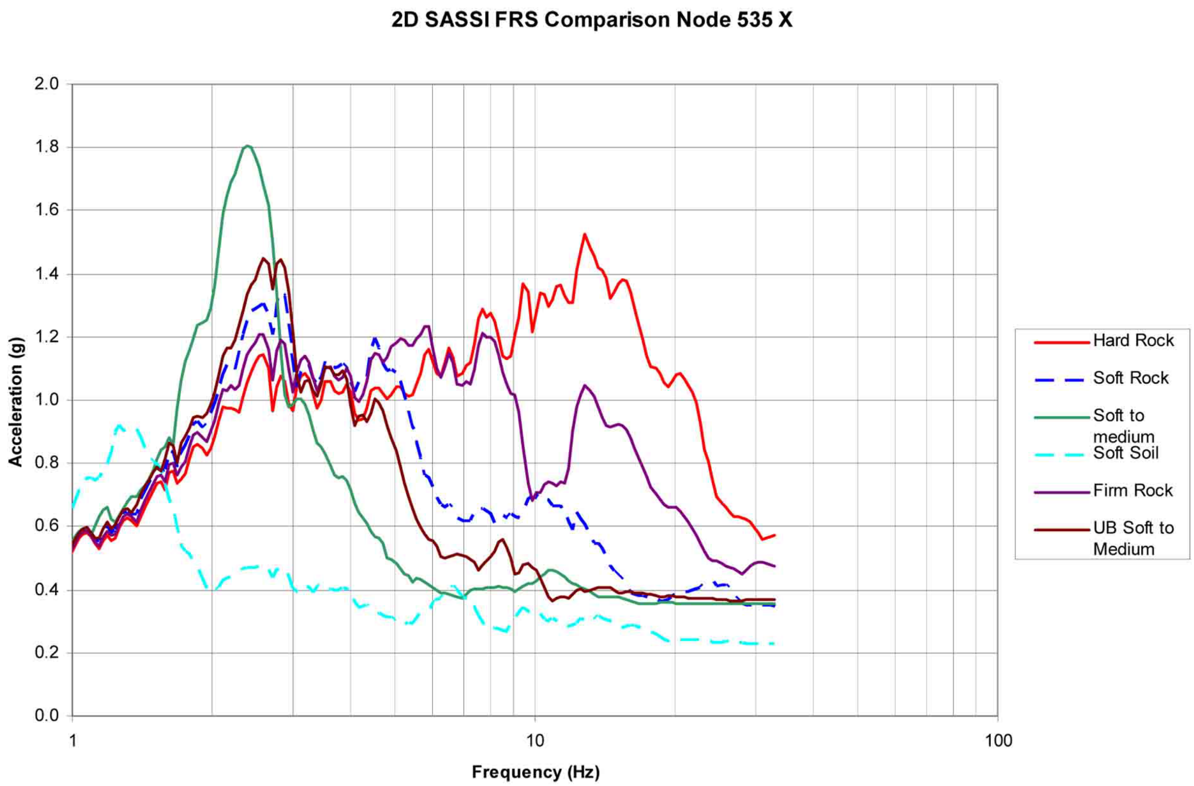


Figure 3G.3-10
2D SASSI FRS – Node 535 X (CIS El. 134.3′)

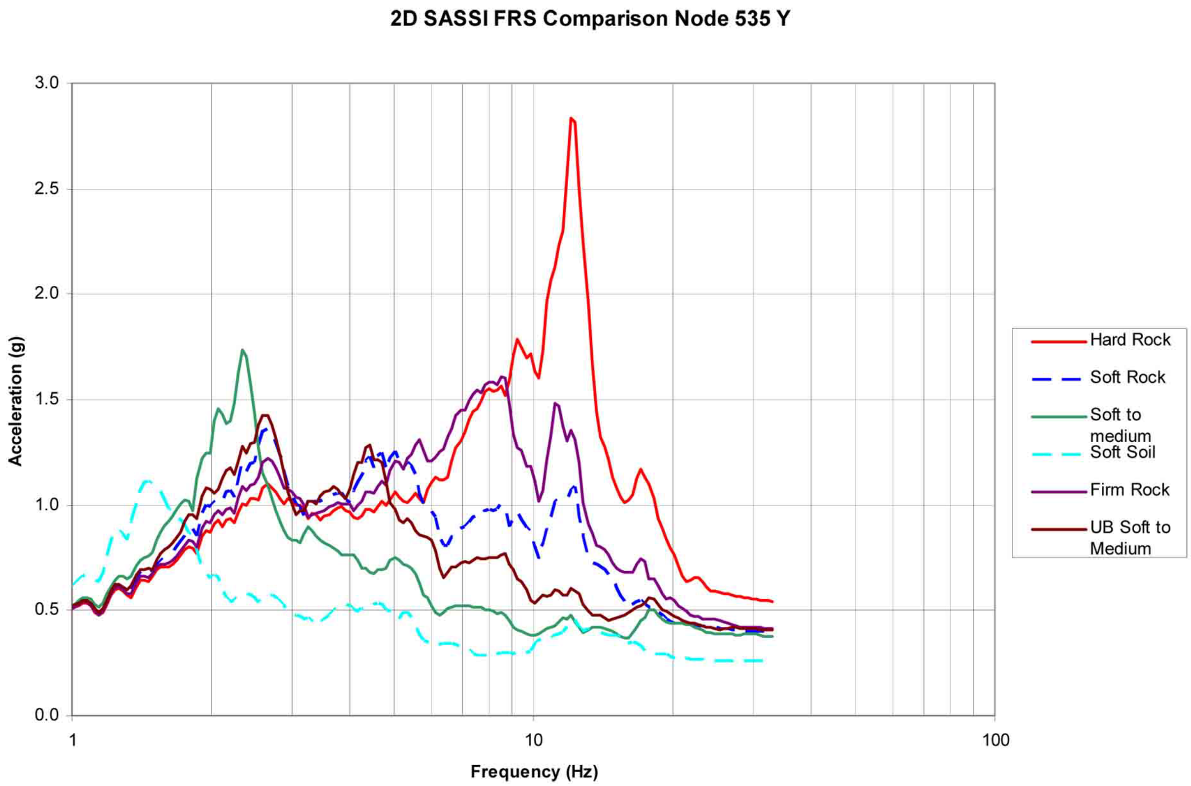


Figure 3G.3-11
2D SASSI FRS – Node 535 Y (CIS El. 134.3′)

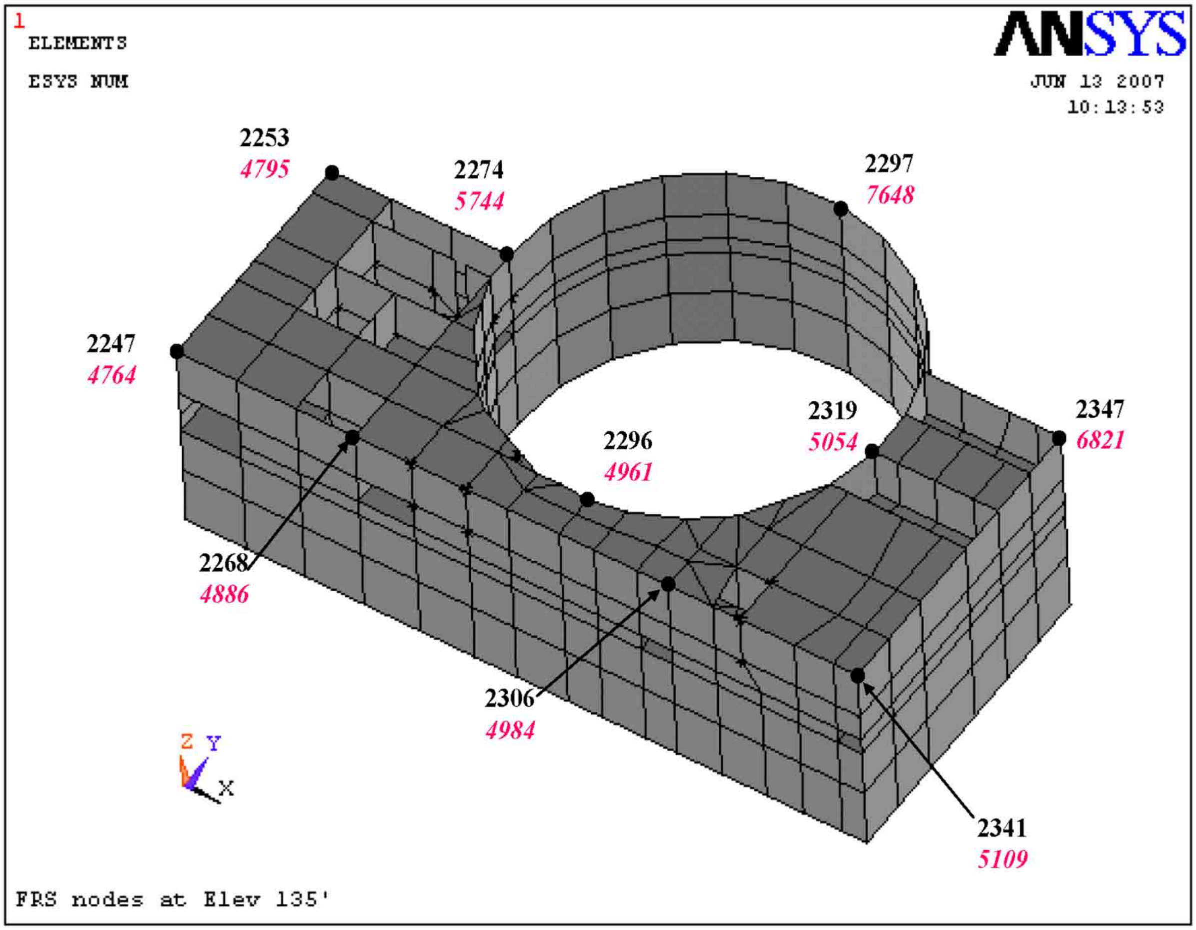


Figure 3G.4-1
Auxiliary Shield Building "Rigid" Nodes at El. 135′

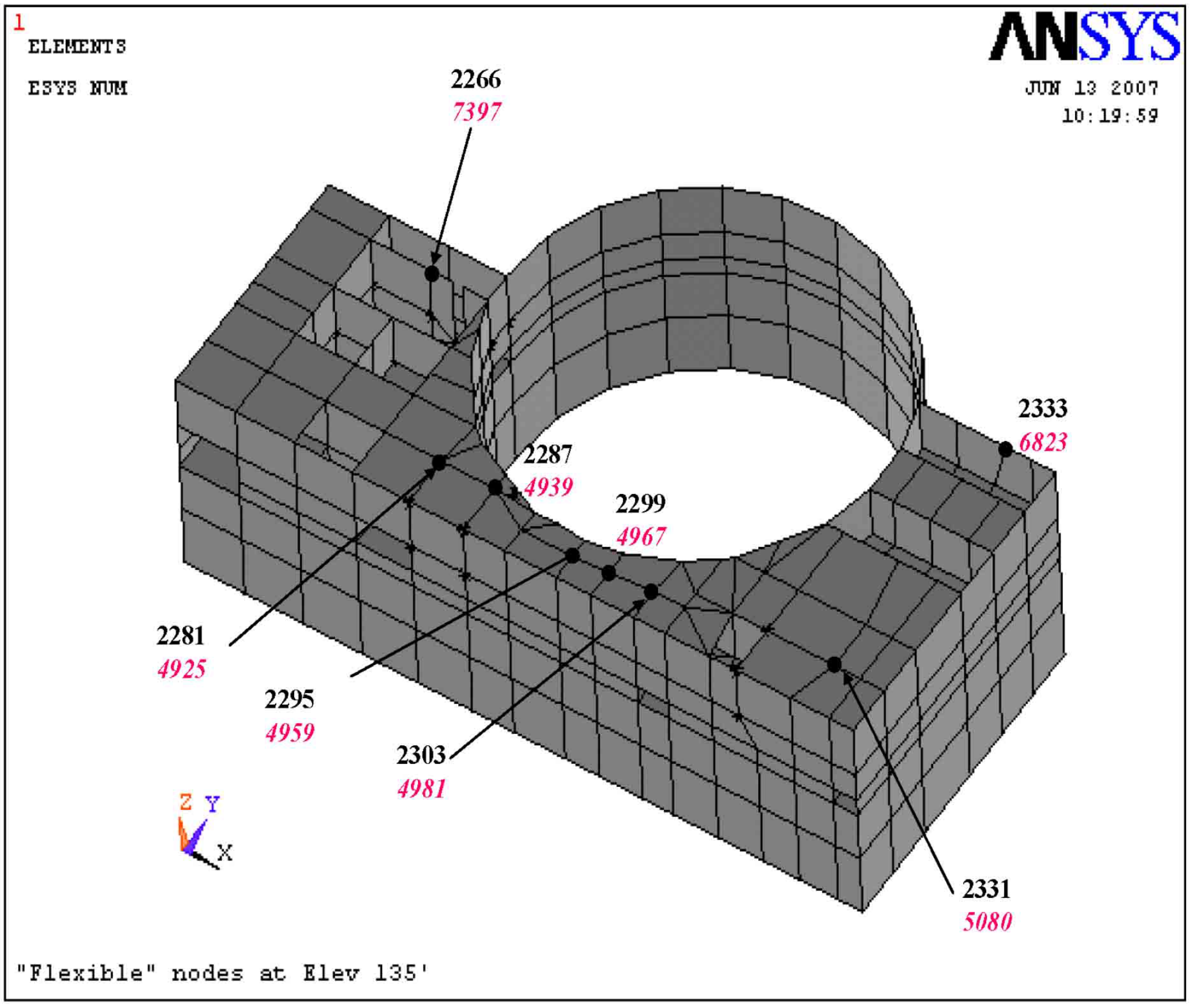


Figure 3G.4-2
Auxiliary Shield Building "Flexible" Nodes at El. 135′

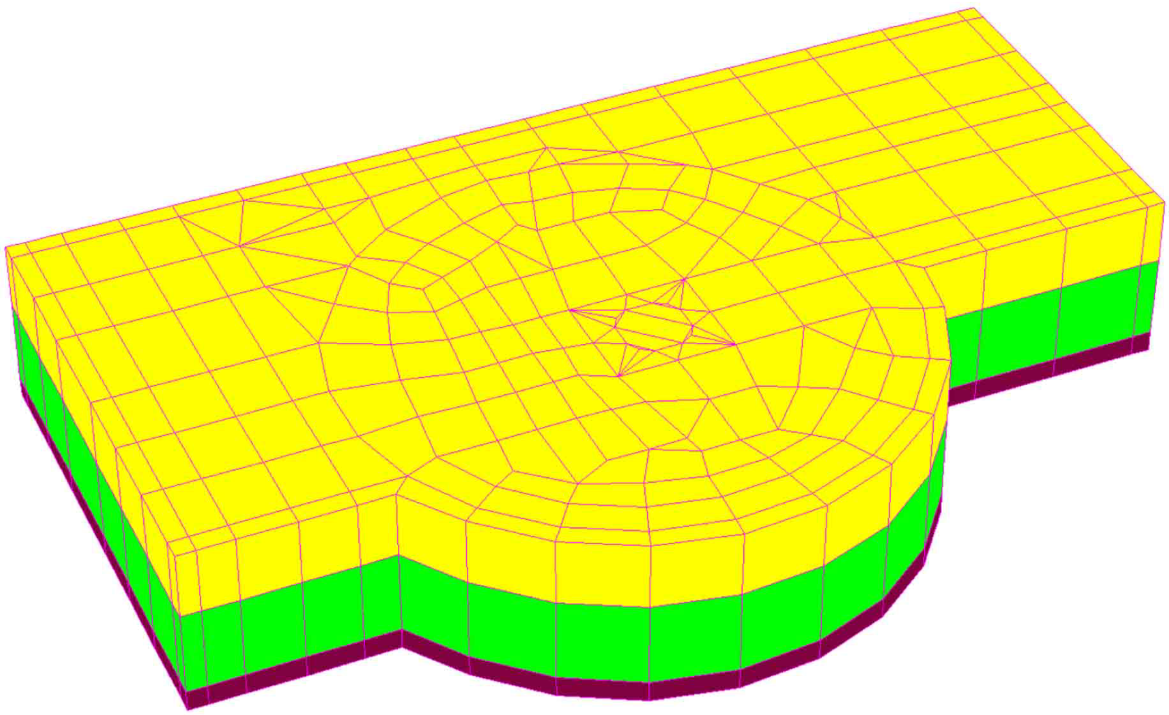


Figure 3G.4-3
Excavated Soil

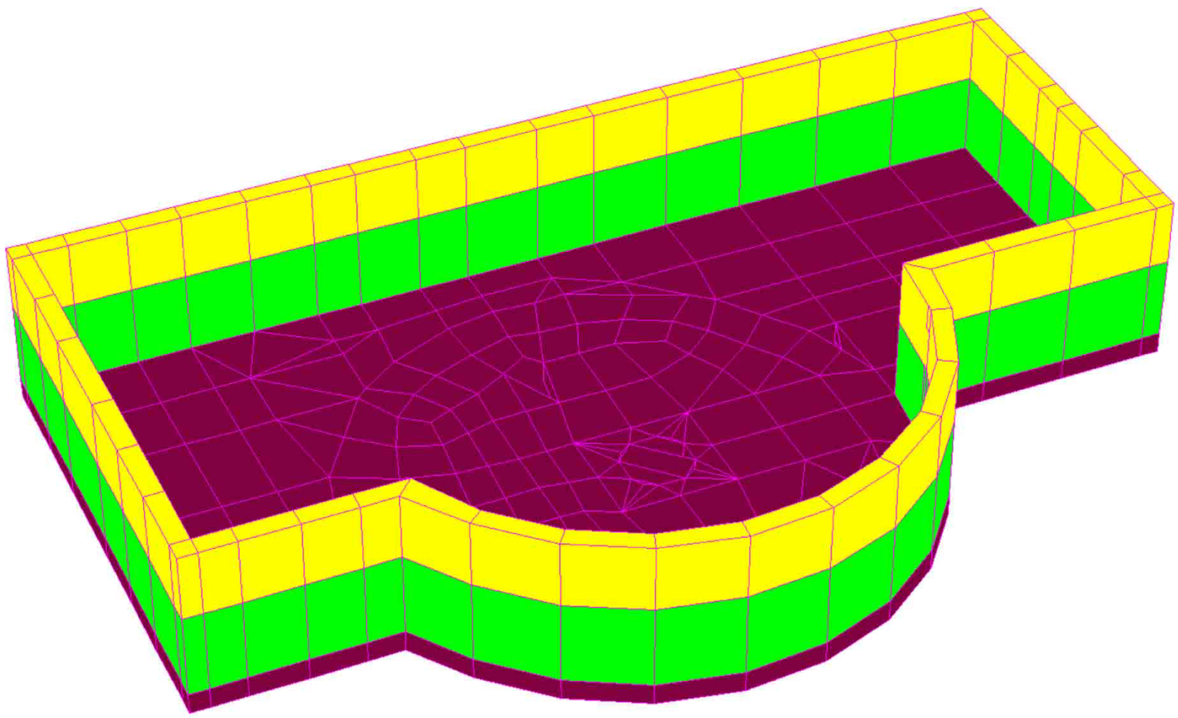
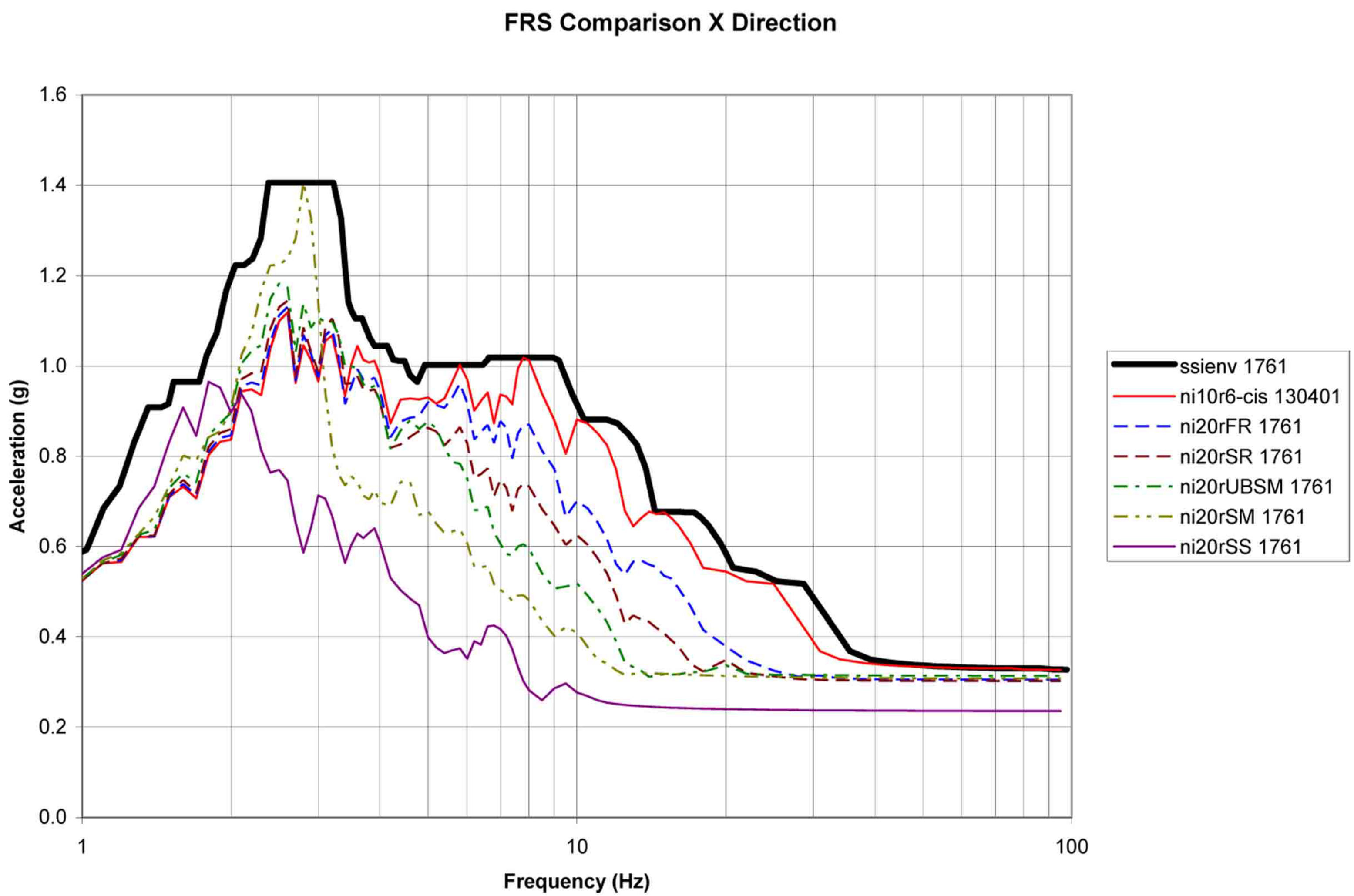


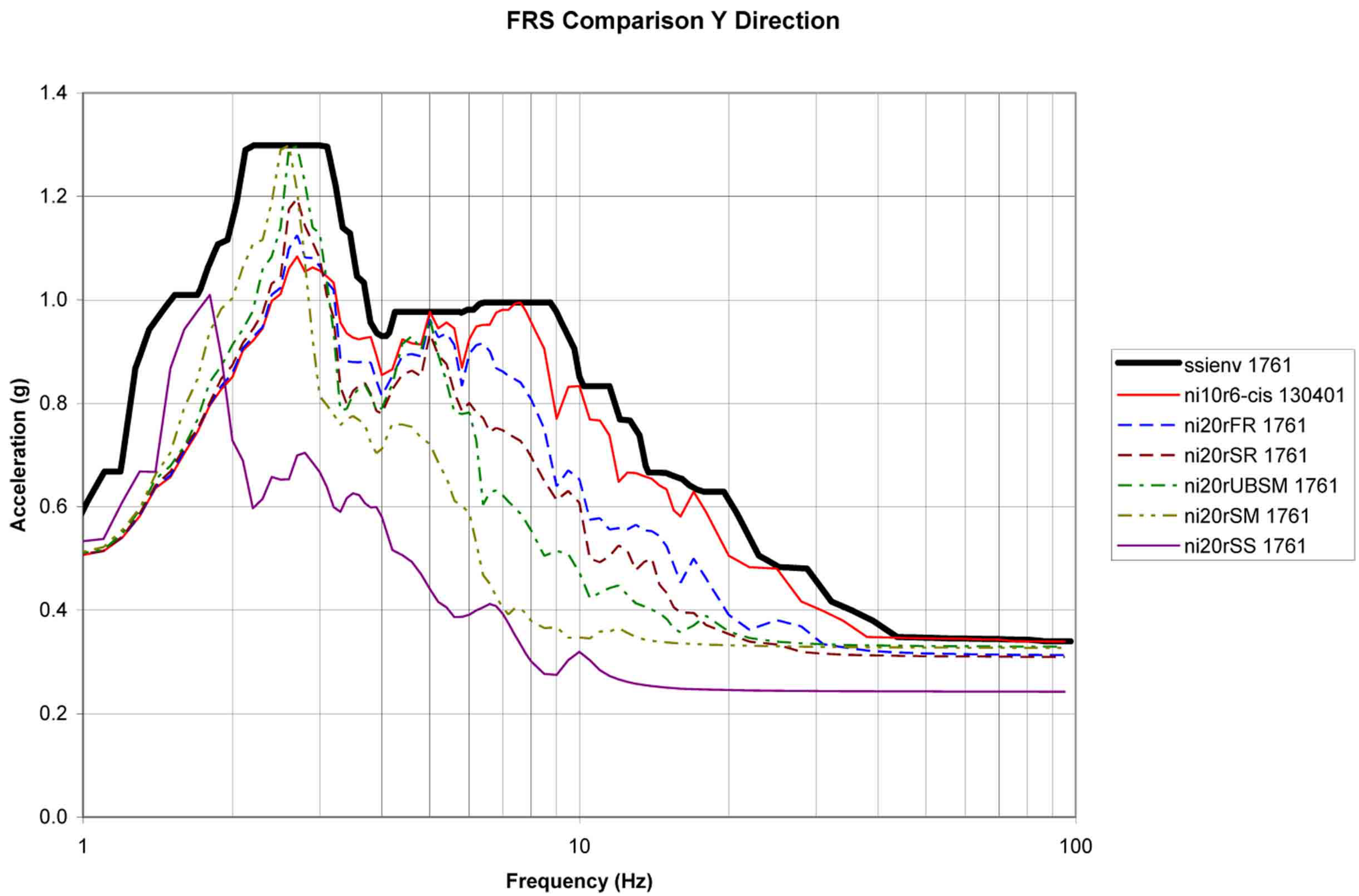
Figure 3G.4-4
Additional Elements for Soil Pressure Calculations



[*Figure 3G.4-5X*

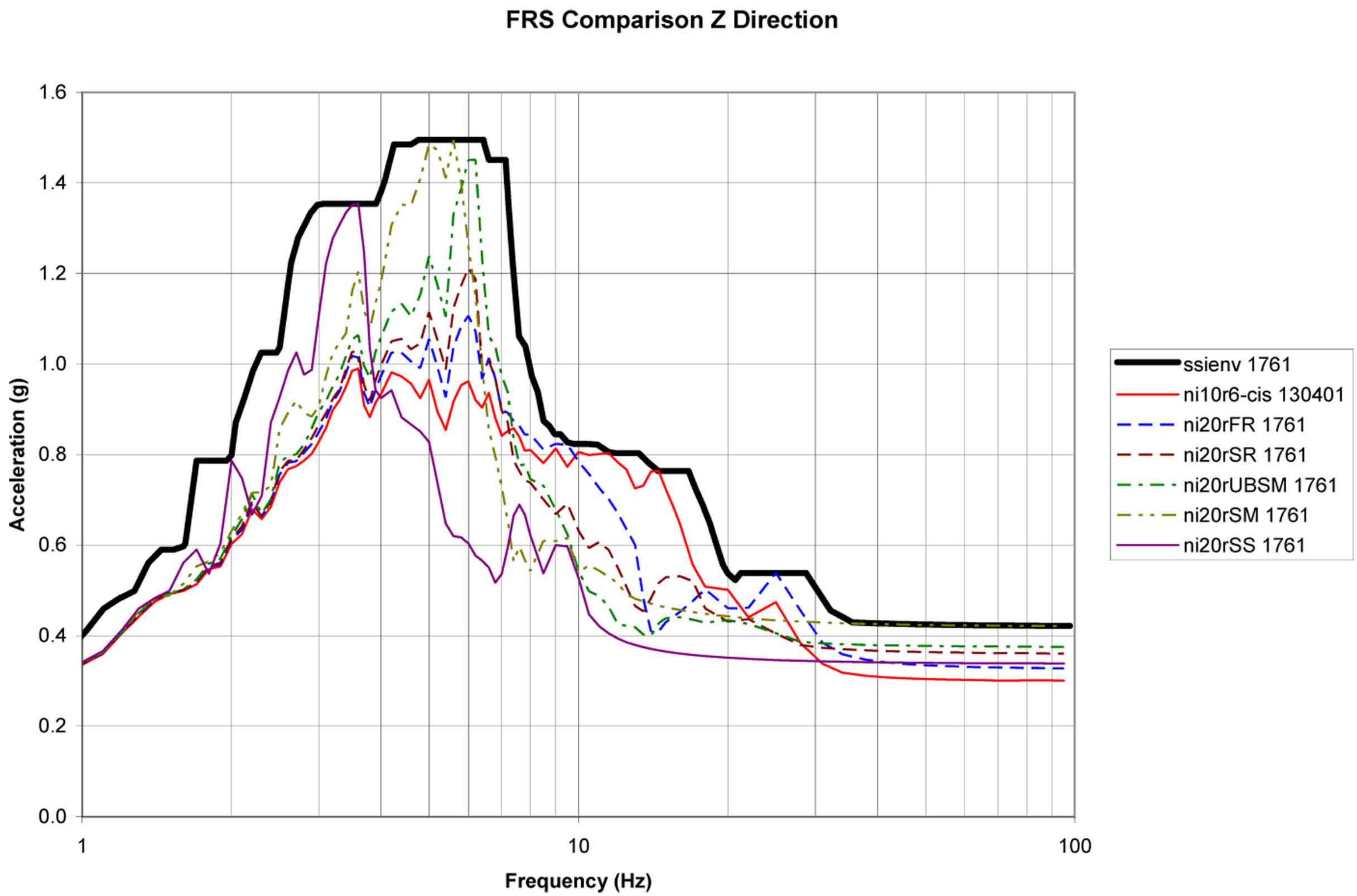
X Direction FRS for Node 130401 (NI10) or 1761 (NI20) CIS at Reactor Vessel Support Elevation of 100']*

*NRC Staff approval is required prior to implementing a change in this information.



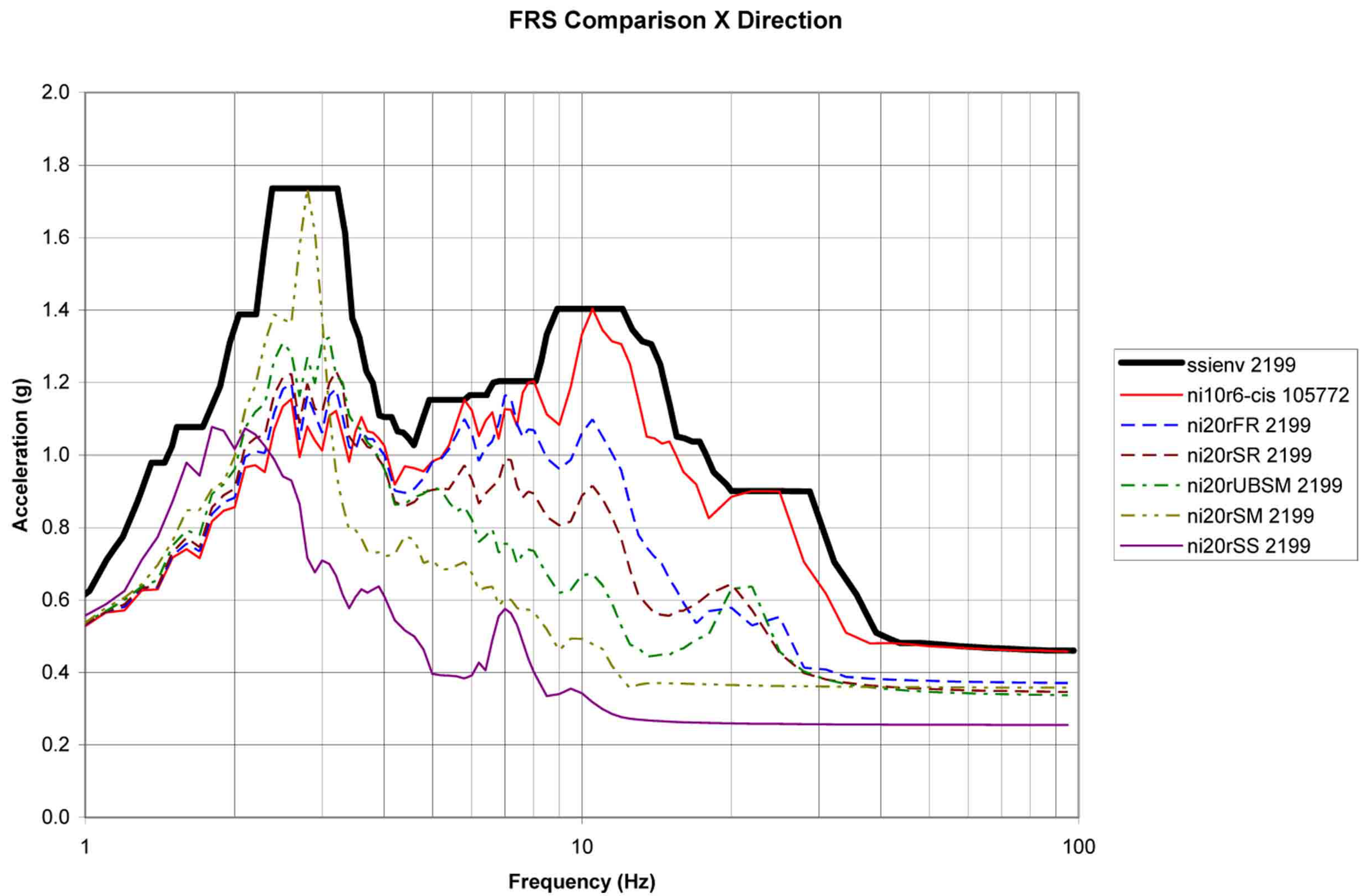
[*Figure 3G.4-5Y*
Y Direction FRS for Node 130401 (NI10) or 1761 (NI20) CIS at Reactor Vessel Support Elevation of 100']*

*NRC Staff approval is required prior to implementing a change in this information.



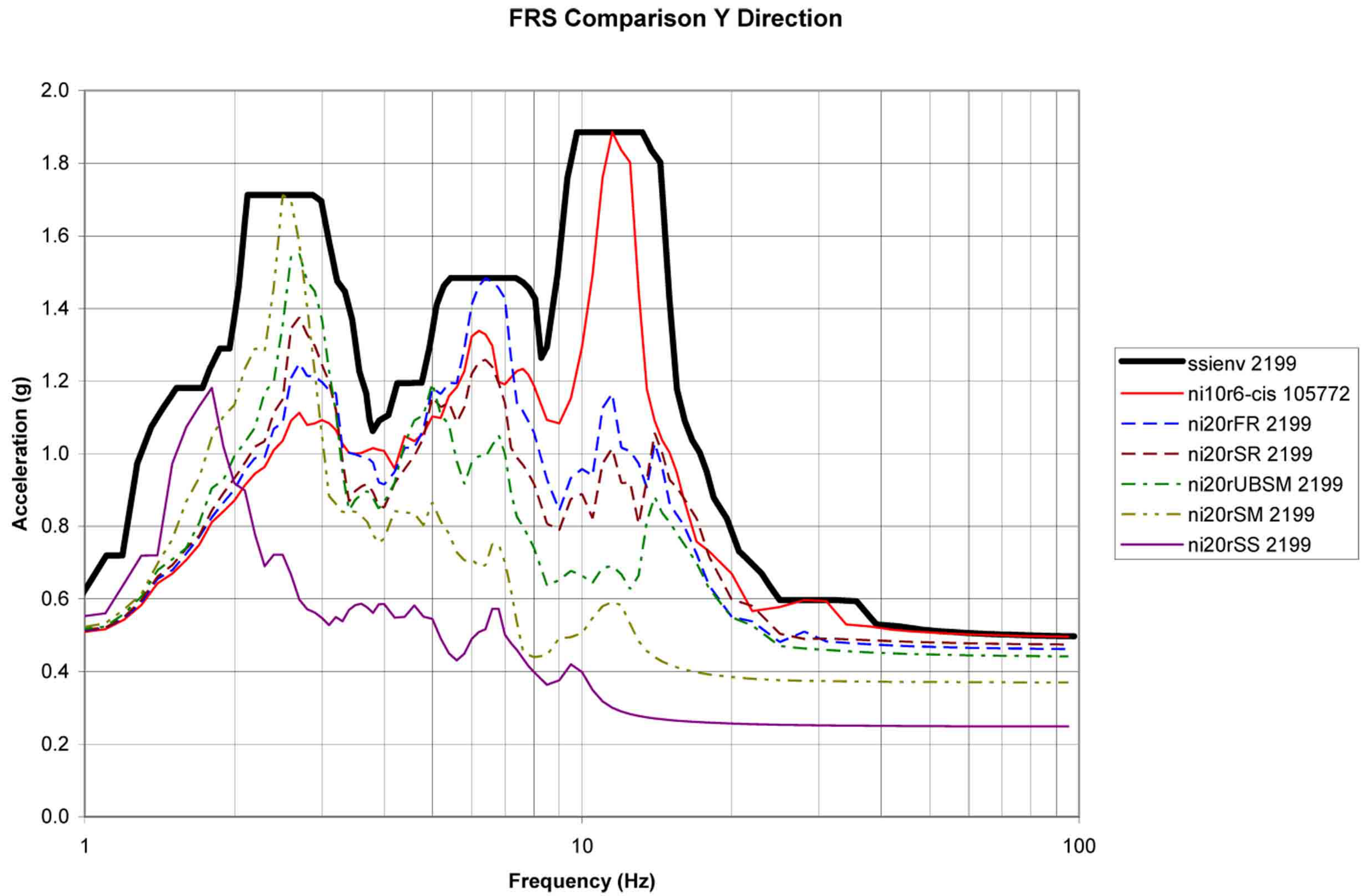
[*Figure 3G.4-5Z*
Z Direction FRS for Node 130401 (NI10) or 1761 (NI20) CIS at Reactor Vessel Support Elevation of 100']*

*NRC Staff approval is required prior to implementing a change in this information.



[*Figure 3G.4-6X*
***X Direction FRS for Node 105772 (NI10) or 2199 (NI20) CIS at Operating Deck Elevation 134.25'*]*

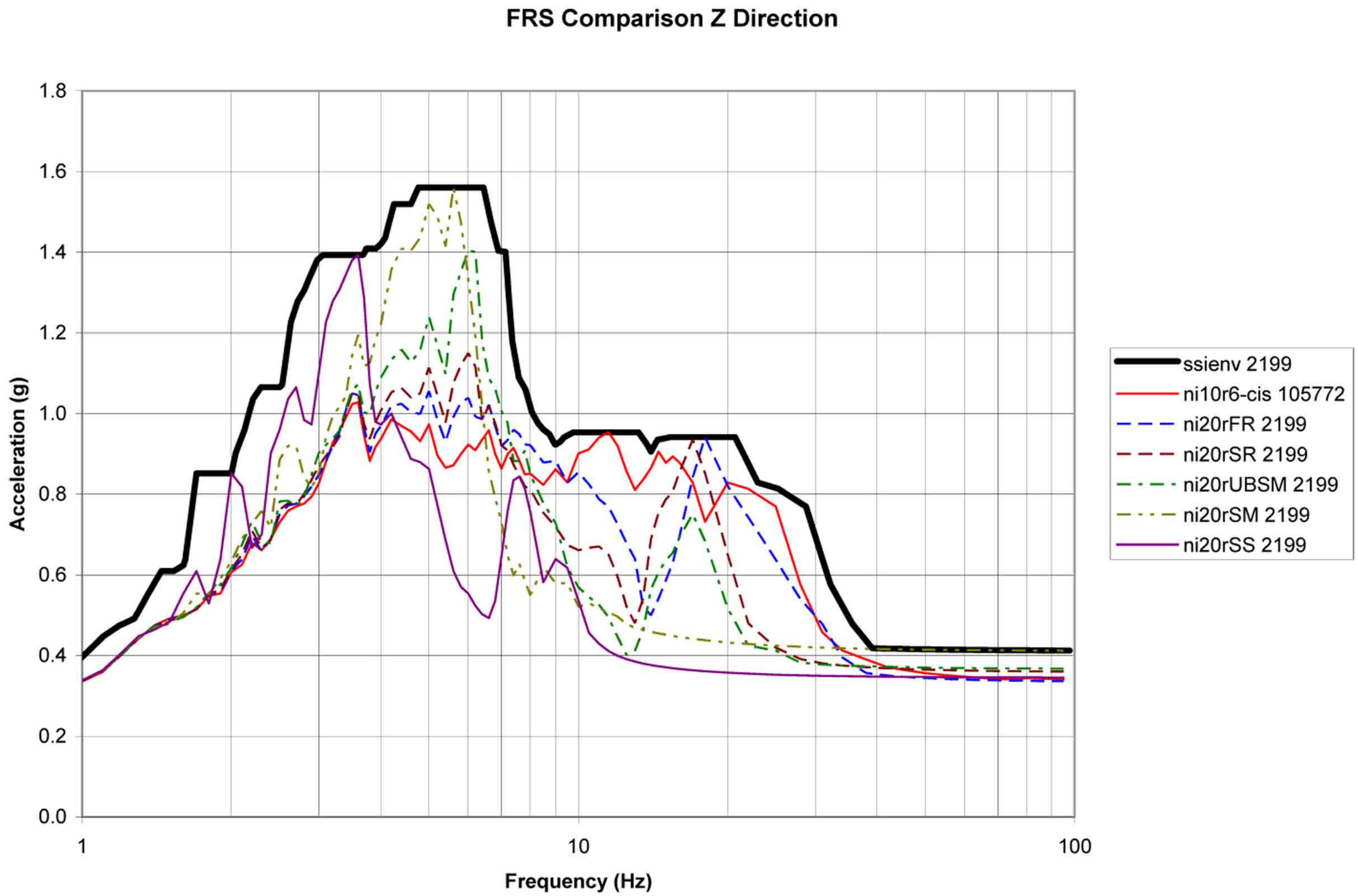
*NRC Staff approval is required prior to implementing a change in this information.



[Figure 3G.4-6Y

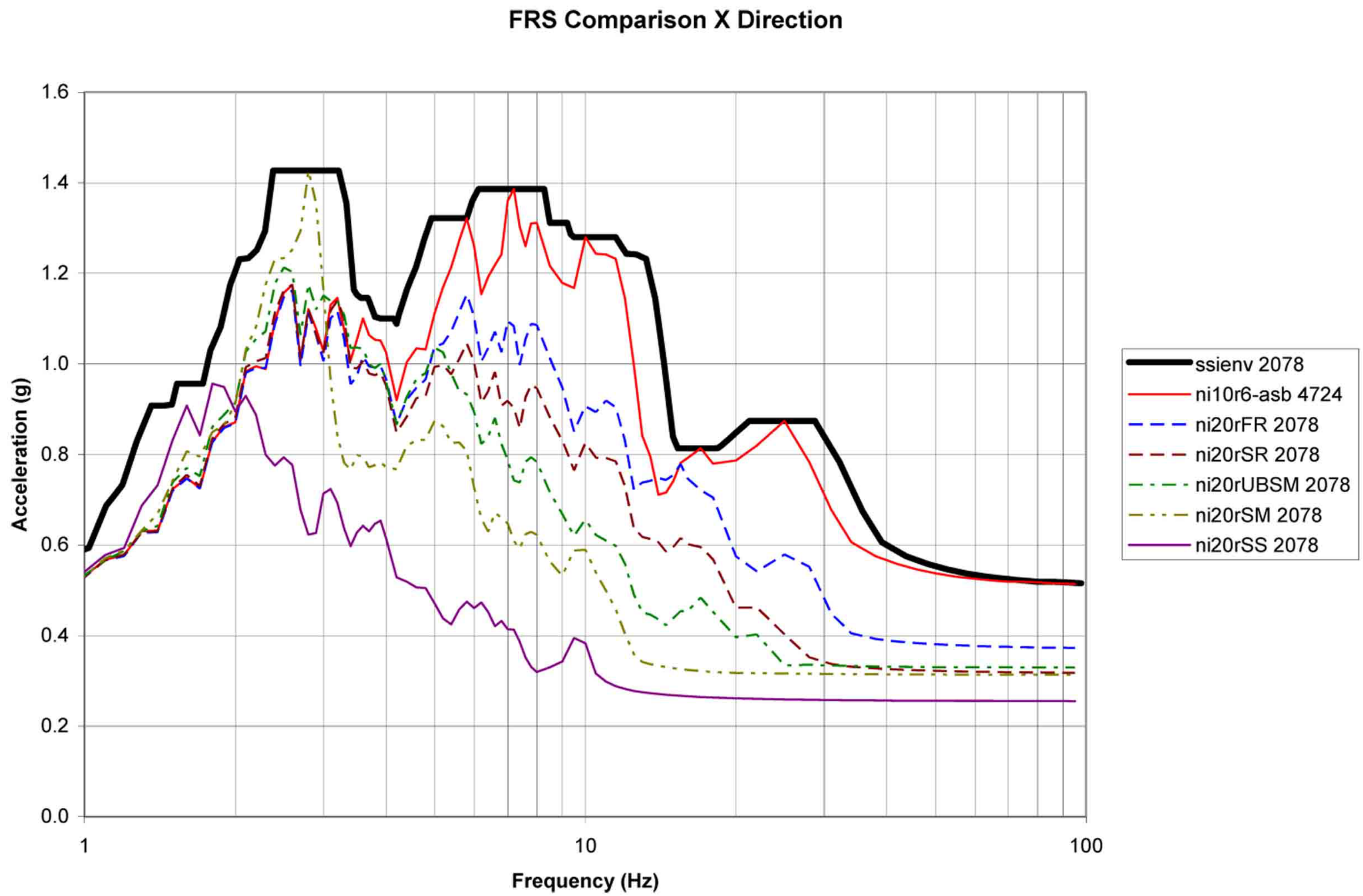
Y Direction FRS for Node 105772 (NI10) or 2199 (NI20) CIS at Operating Deck Elevation 134.25']*

*NRC Staff approval is required prior to implementing a change in this information.



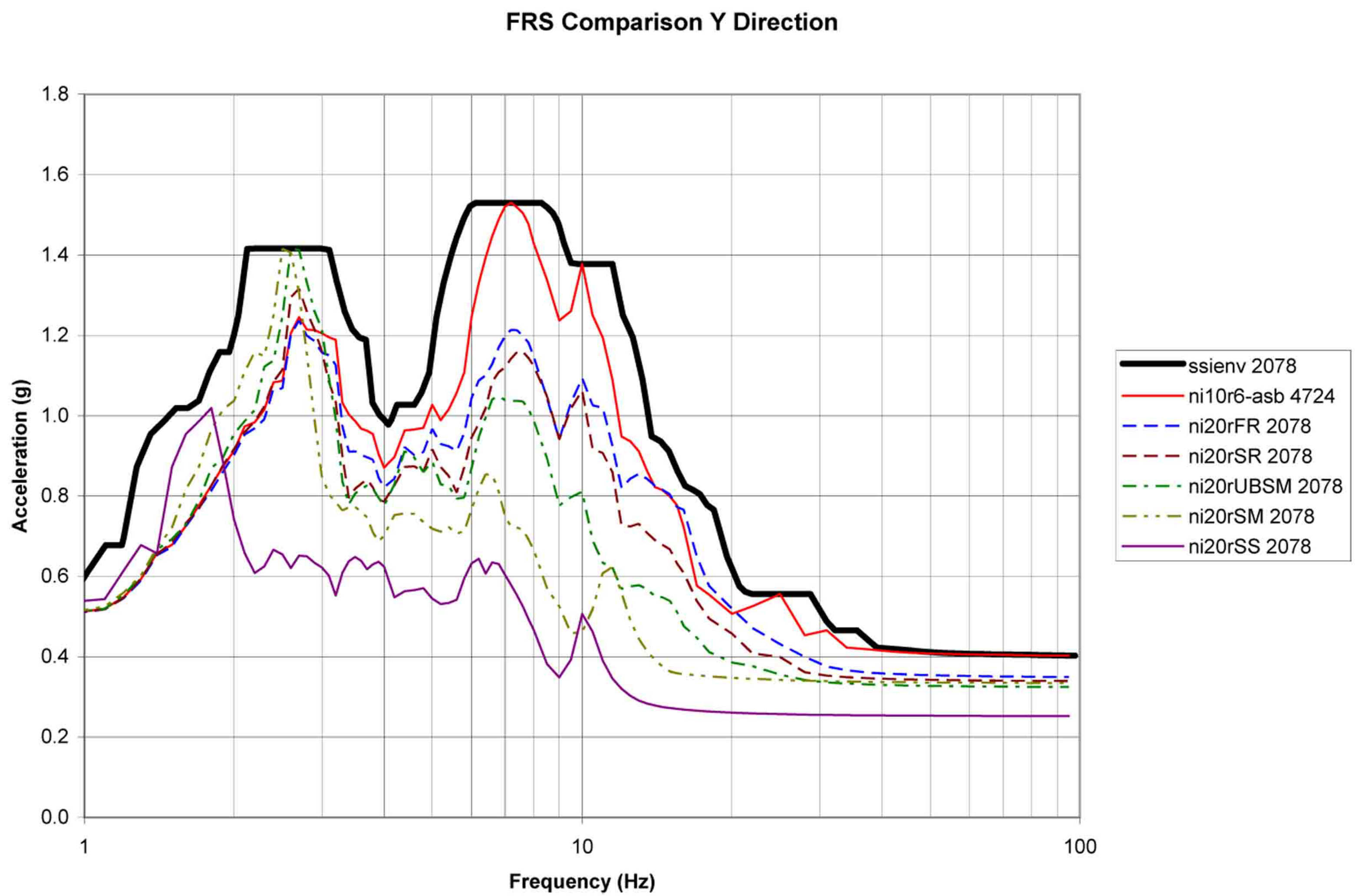
[*Figure 3G.4-6Z*
Z Direction FRS for Node 105772 (NI10) or 2199 (NI20) CIS at Operating Deck Elevation 134.25']*

*NRC Staff approval is required prior to implementing a change in this information.



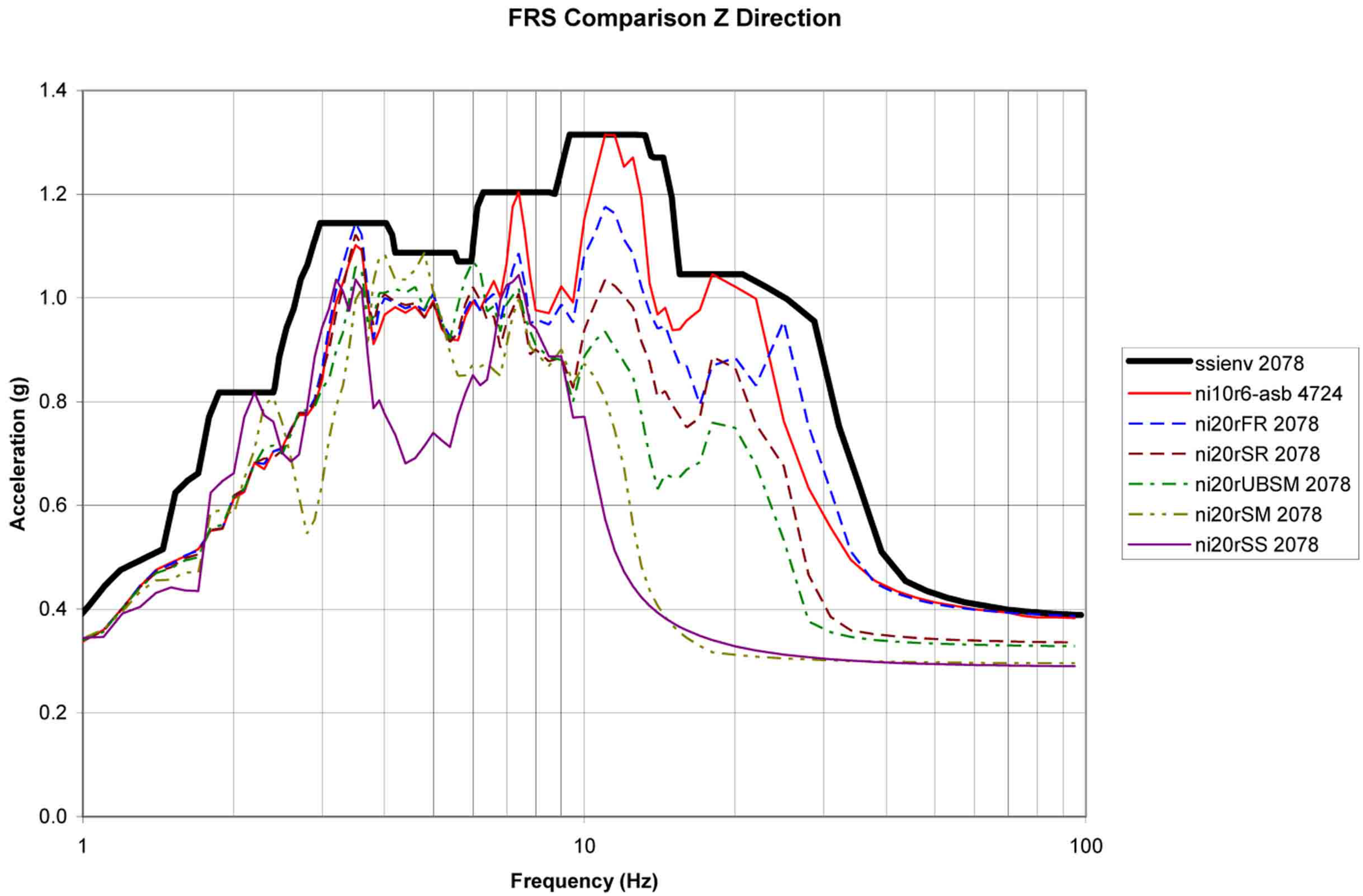
[Figure 3G.4-7X
X Direction FRS for Node 4724 (NI10) or 2078 (NI20) ASB Control Room Side Elevation 116.50'*]

*NRC Staff approval is required prior to implementing a change in this information.



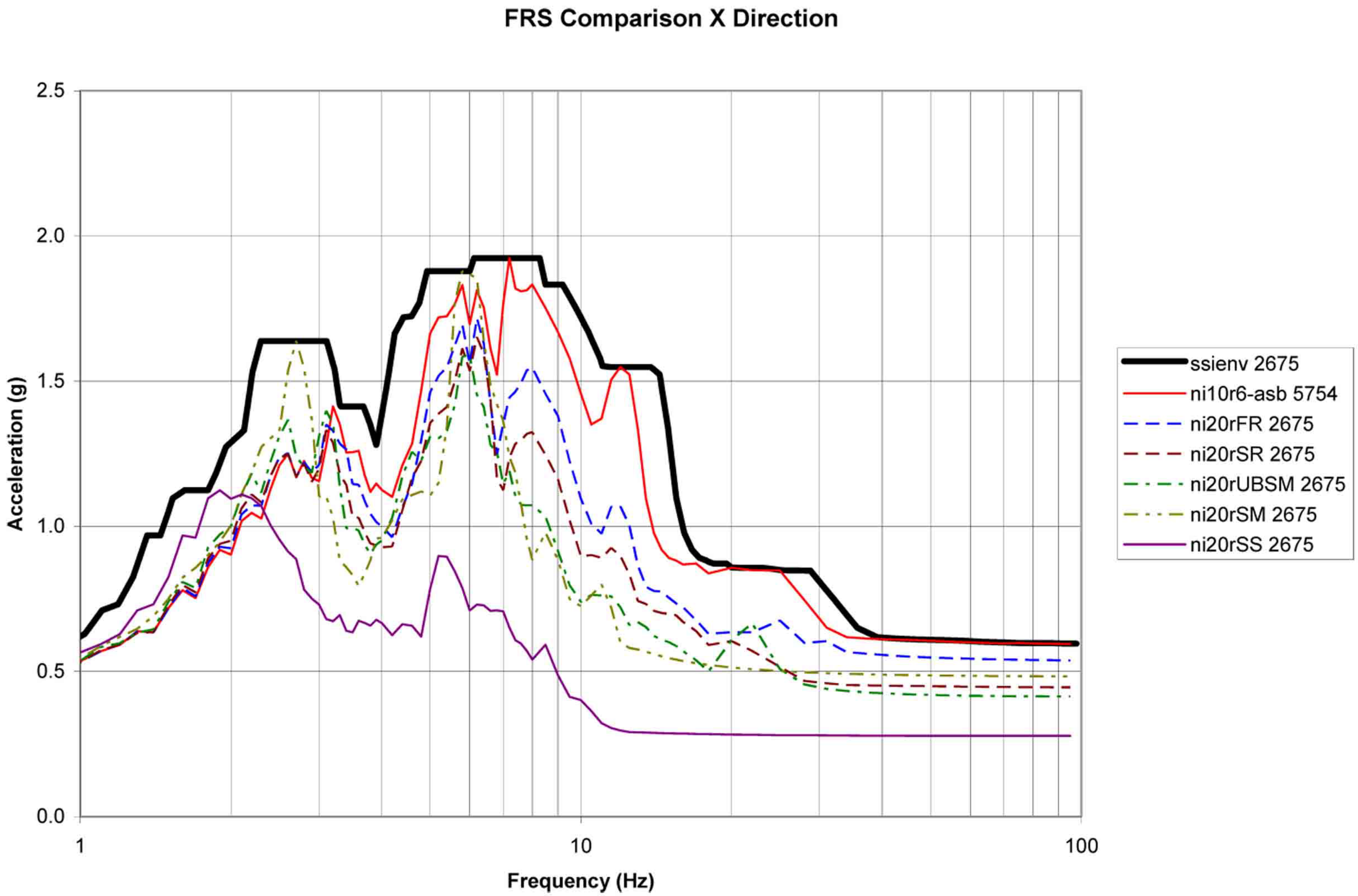
[Figure 3G.4-7Y
Y Direction FRS for Node 4724 (NI10) or 2078 (NI20) ASB Control Room Side Elevation 116.50']*

*NRC Staff approval is required prior to implementing a change in this information.



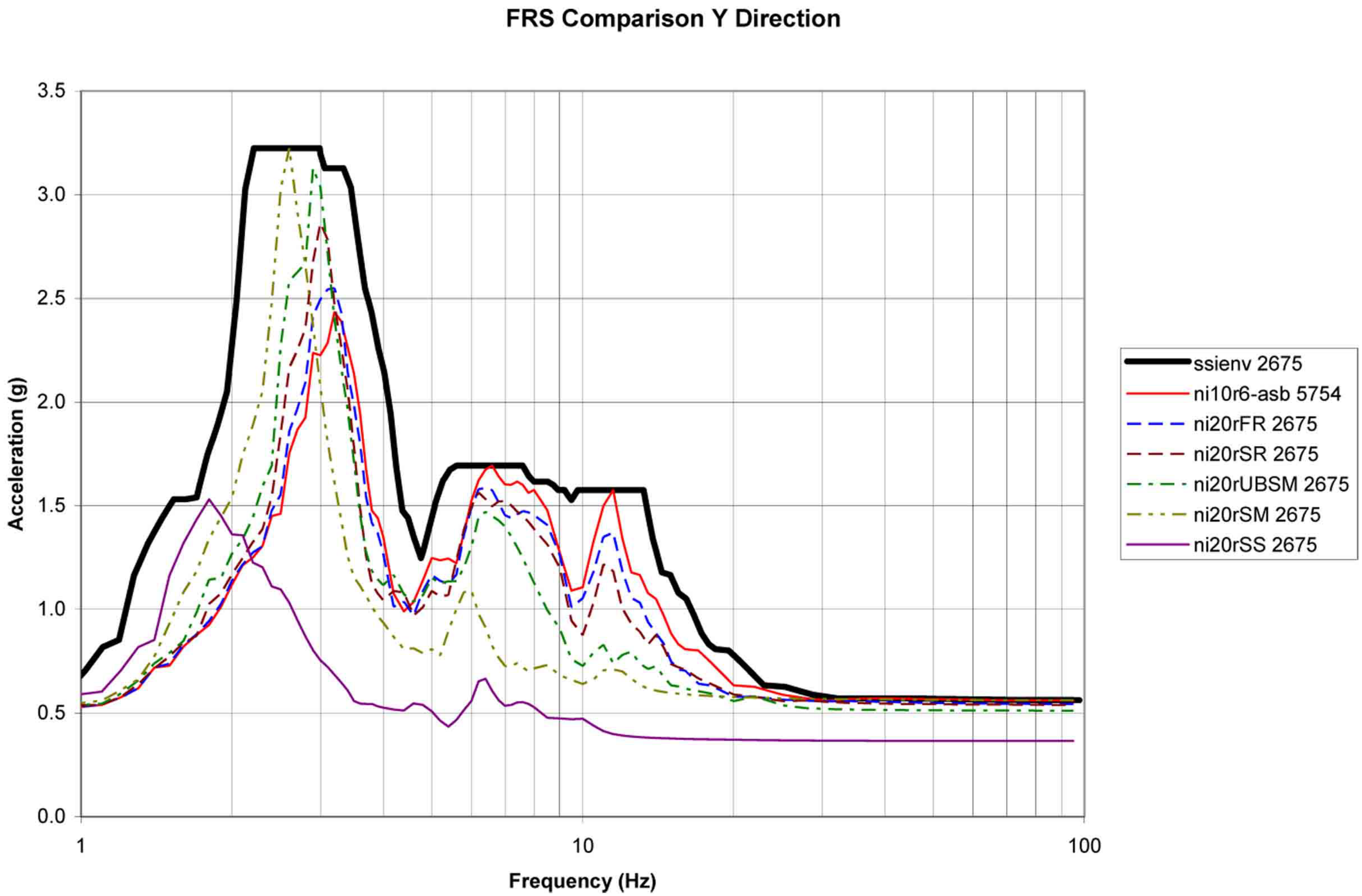
[*Figure 3G.4-7Z*
Z Direction FRS for Node 4724 (NI10) or 2078 (NI20) ASB Control Room Side Elevation 116.50'*]

*NRC Staff approval is required prior to implementing a change in this information.



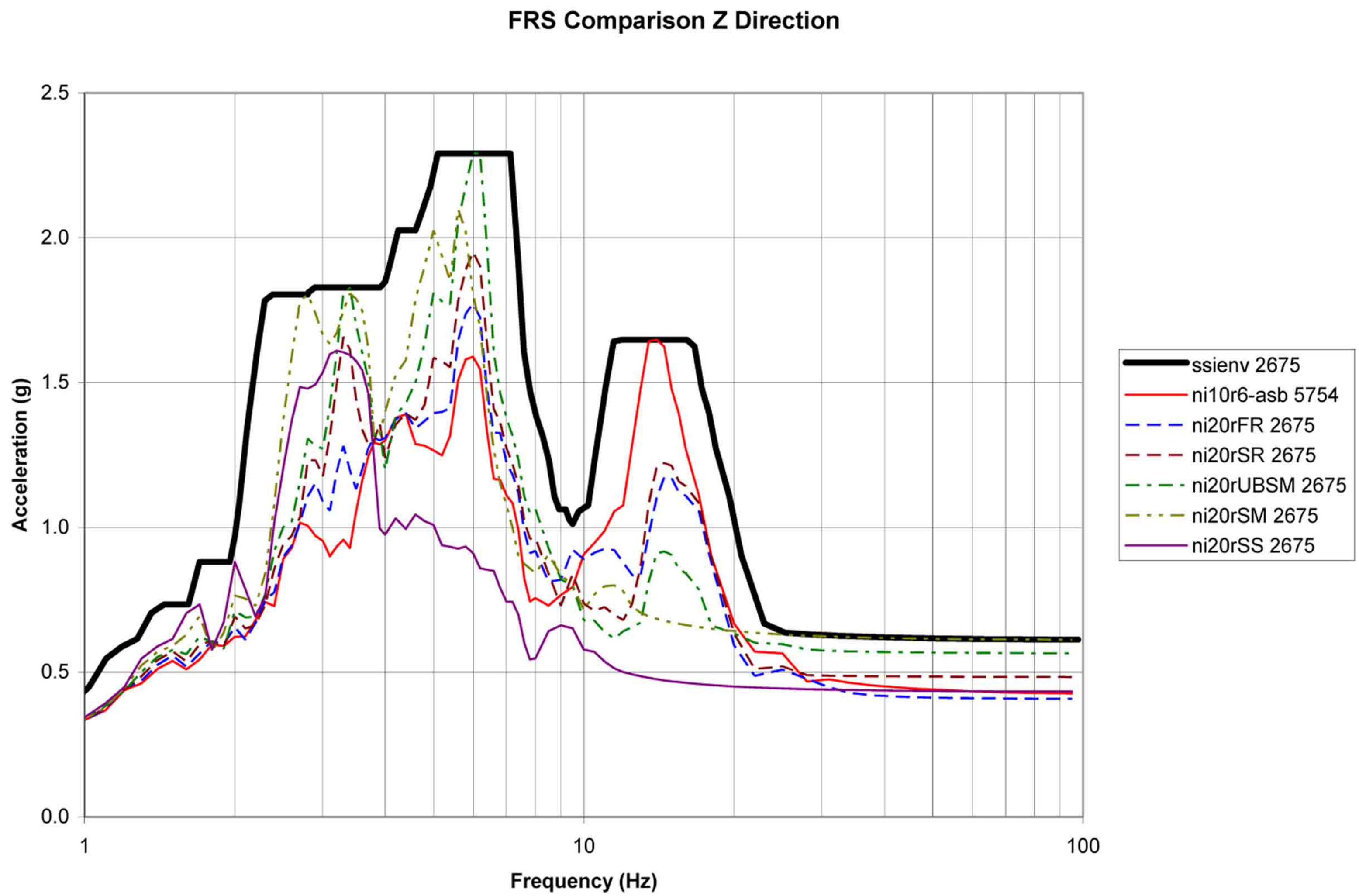
[*Figure 3G.4-8X*
X Direction FRS for Node 5754 (NI10) or 2675 (NI20) ASB Fuel Building Roof Elevation 179.19'*]

*NRC Staff approval is required prior to implementing a change in this information.



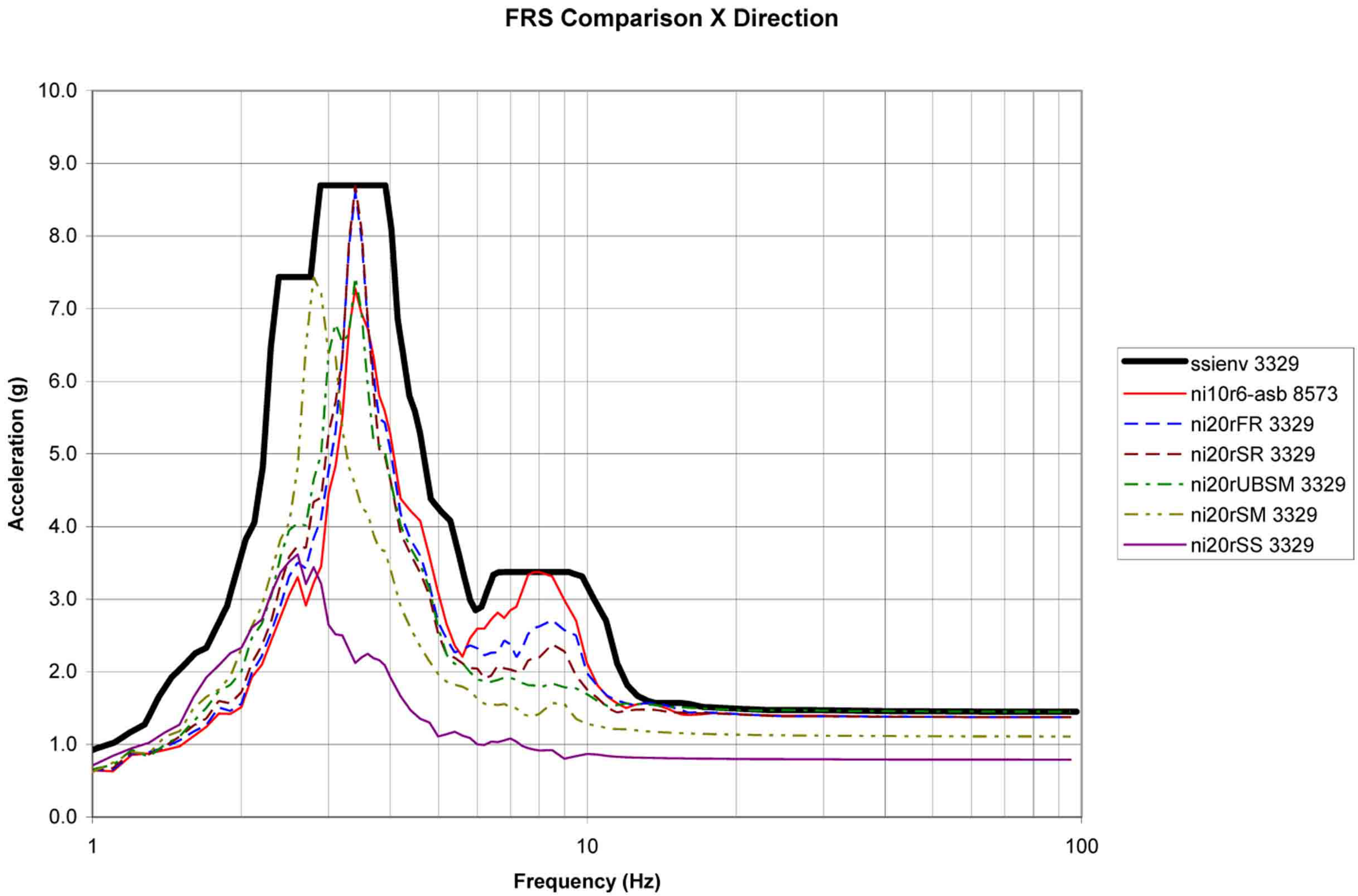
[*Figure 3G.4-8Y*
***Y Direction FRS for Node 5754 (NI10) or 2675 (NI20) ASB Fuel Building Roof Elevation 179.19'*]*

*NRC Staff approval is required prior to implementing a change in this information.



[*Figure 3G.4-8Z*
Z Direction FRS for Node 5754 (NI10) or 2675 (NI20) ASB Fuel Building Roof Elevation 179.19']*

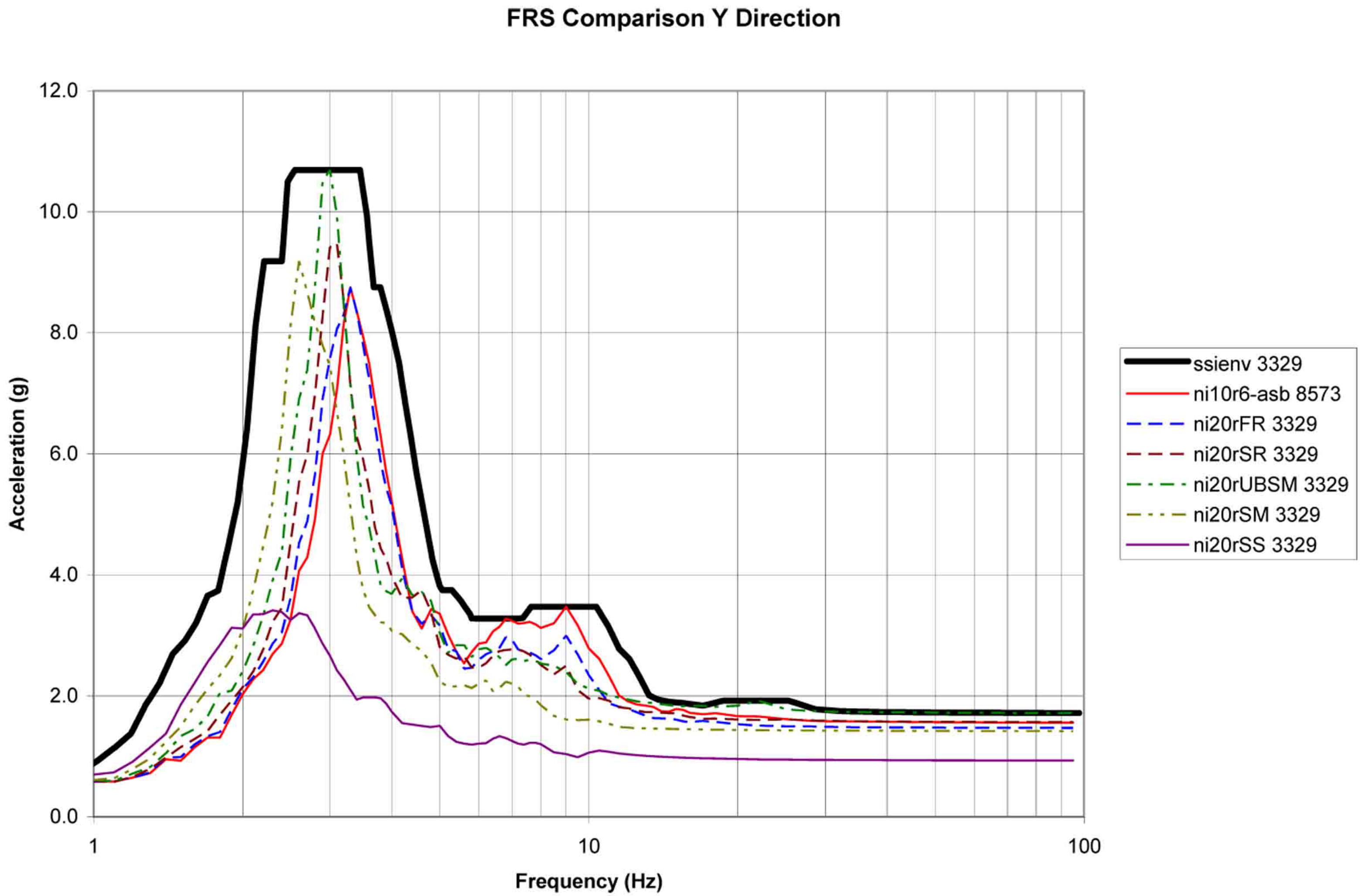
*NRC Staff approval is required prior to implementing a change in this information.



[Figure 3G.4-9X

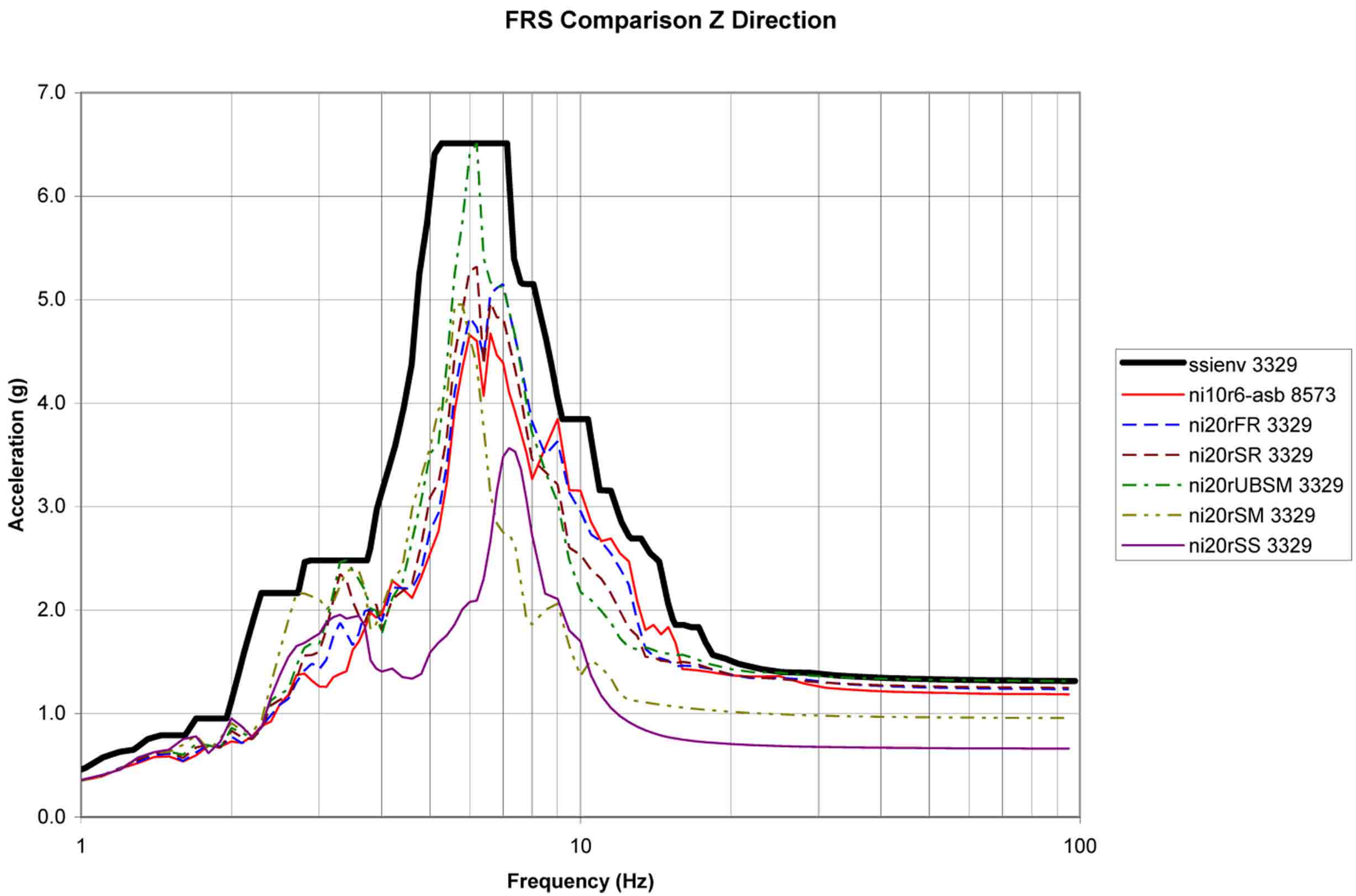
X Direction FRS for Node 8573 (NI10) or 3329 (NI20) ASB Shield Building Roof Elevation 327.41']*

*NRC Staff approval is required prior to implementing a change in this information.



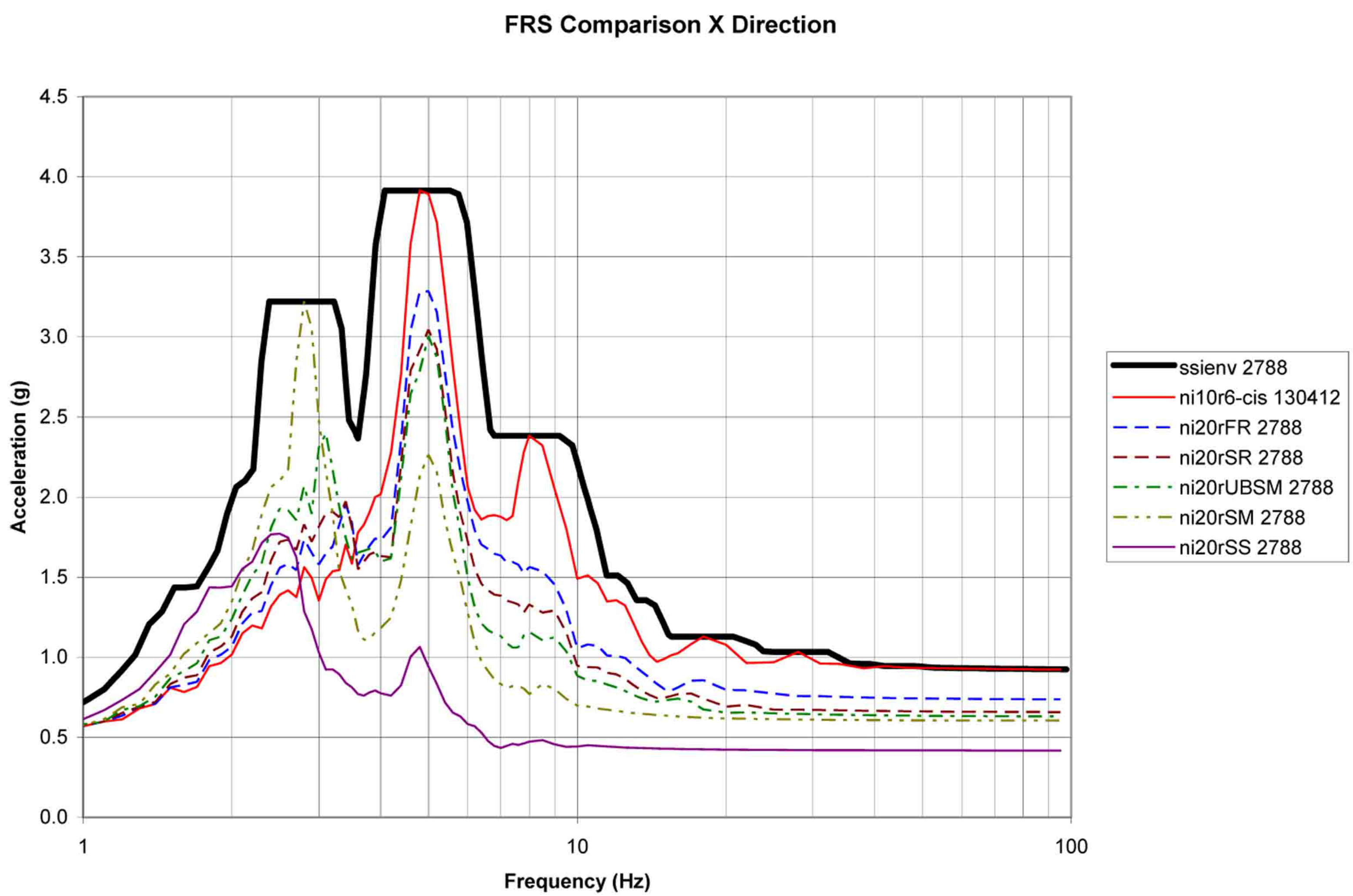
[*Figure 3G.4-9Y*
Y Direction FRS for Node 8573 (NI10) or 3329 (NI20) ASB Shield Building Roof Elevation 327.41']*

*NRC Staff approval is required prior to implementing a change in this information.



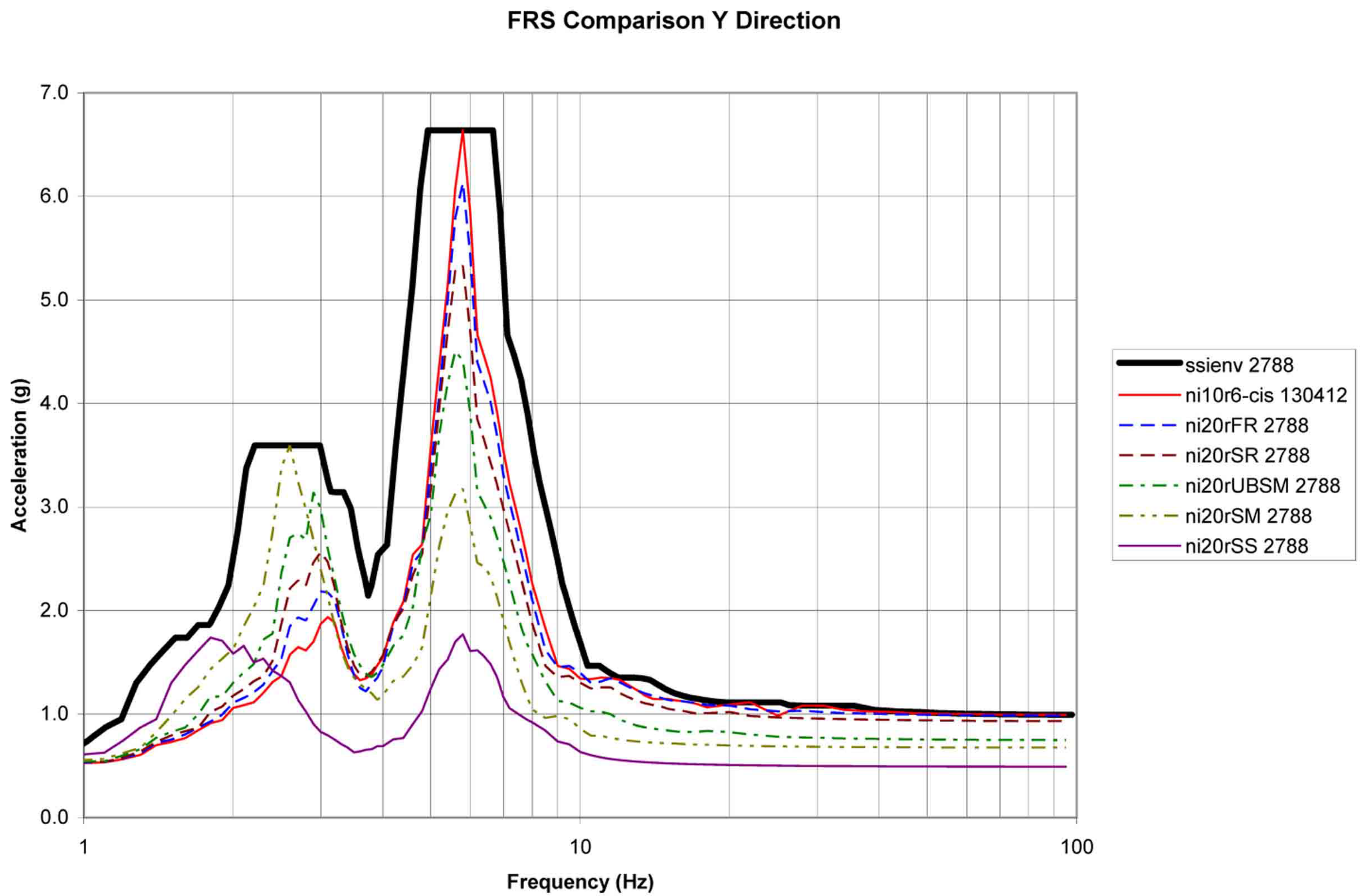
[*Figure 3G.4-9Z*
Z Direction FRS for Node 8573 (NI10) or 3329 (NI20) ASB Shield Building Roof Elevation 327.41']*

*NRC Staff approval is required prior to implementing a change in this information.



[*Figure 3G.4-10X*
X Direction FRS for Node 130412 (NI10) or 2788 (NI20) SCV Near Polar Crane Elevation 224.00']*

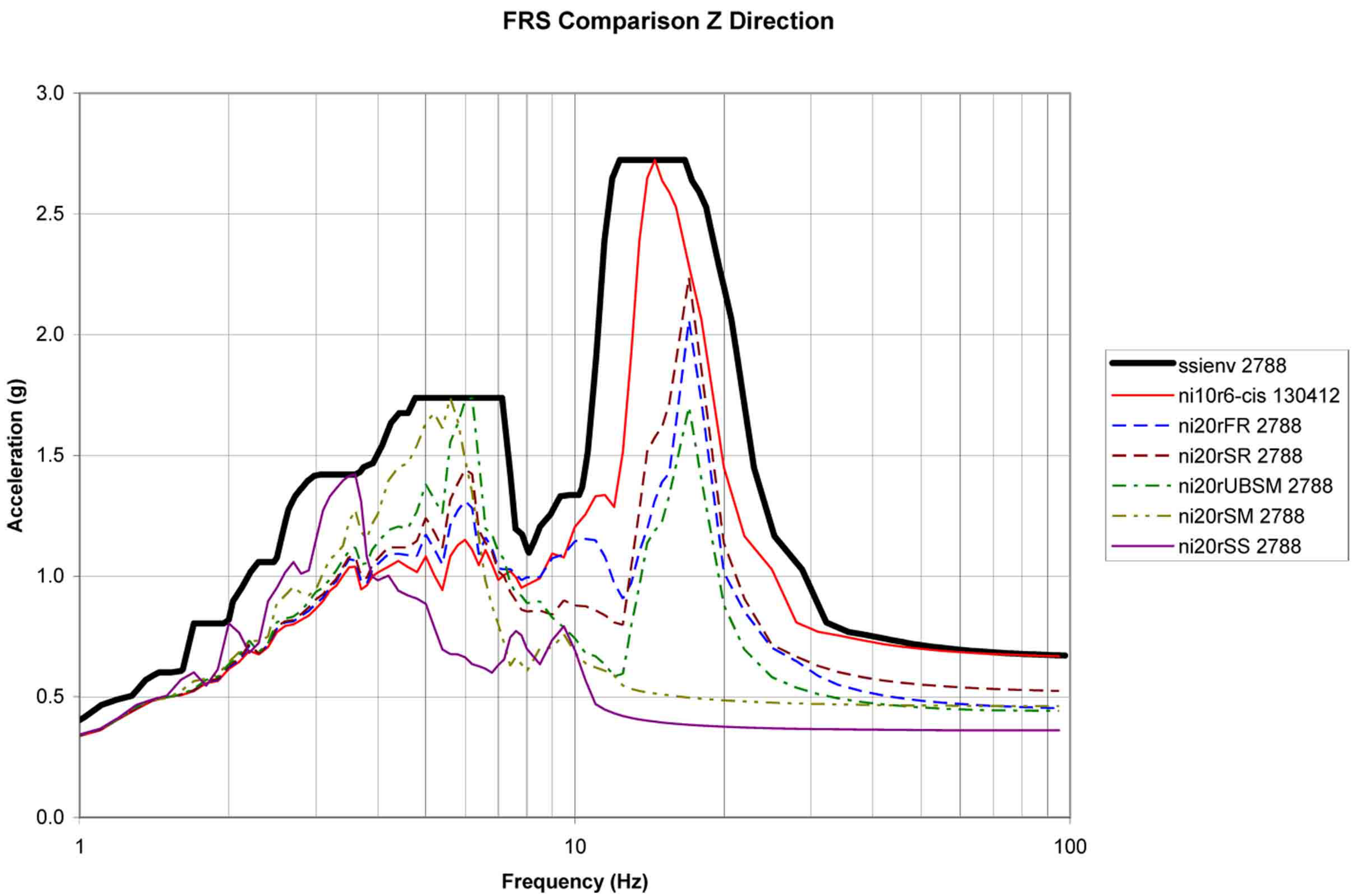
*NRC Staff approval is required prior to implementing a change in this information.



[Figure 3G.4-10Y

Y Direction FRS for Node 130412 (NI10) or 2788 (NI20) SCV Near Polar Crane Elevation 224.00']*

*NRC Staff approval is required prior to implementing a change in this information.



[Figure 3G.4-10Z
Z Direction FRS for Node 130412 (NI10) or 2788 (NI20) SCV Near Polar Crane Elevation 224.00']*

*NRC Staff approval is required prior to implementing a change in this information.

Appendix 3H Auxiliary and Shield Building Critical Sections

3H.1 Introduction

[*This appendix summarizes the structural design and analysis of structures identified as "Critical Sections" in the auxiliary and shield buildings. The design summaries include the following information:*

- *Description of buildings*
- *Governing codes and regulations*
- *Structural loads and load combinations*
- *Global analyses*
- *Structural design of critical structural elements*

Subsections 3H.2 through 3H.5 include a general description of the auxiliary building and shield building, a summary of the design criteria and the global analyses. Examples of the structural design are shown for 14 critical sections which are identified in subsection 3H.5 and shown in Figures 3H.5-1 (3 sheets). The exact locations of the critical sections related to the shield building cylinder shown in Figure 3H.5-16. Representative design details are provided for these structures in subsection 3H.5.]*

3H.2 Description of Auxiliary and Shield Buildings

3H.2.1 Description of Auxiliary Building

[*The auxiliary building is a reinforced concrete structure. The auxiliary building is one of the three buildings that make up the nuclear island and shares a common basemat with the containment building and the shield building. The auxiliary building general layout is shown in Figure 3H.2-1. It is a C-shaped section of the nuclear island that wraps around approximately half of the circumference of the shield building. The building dimensions are shown on key structural dimension drawings, Figure 3.7.2-12.*

The auxiliary building is divided into six areas, which are identified in Figure 3H.2-1. It is a 5-story building; three stories are located above grade and two are located below grade. Areas 1 and 2 (Figure 3H.2-1) have five floors, including two floors below grade level. The lowest floor at elevation 66'-6" is used exclusively for housing battery racks. The next higher floor, at elevation 82'-6", also has battery racks and some electrical equipment. The floor at the grade level, elevation 100'-0", has electrical penetration areas, a remote shutdown workstation room, and some Division A and Division C equipment. The main control room is situated on the floor at elevation 117'-6", which also has rooms for the main steam and feedwater lines. The floor at elevation 135'-3" carries air filtration and air handling units, chiller pumps, and other mechanical and electrical equipment. The roof for areas 1 and 2 is at elevation 153'-0".

Areas 3 and 4 of the auxiliary building are the areas east of the containment shield building. Valve and piping areas, and some mechanical equipment, are located in the basement floor at elevation 66'-6". The floor at elevation 82'-6" has a piping penetration area, a radiation chemistry laboratory, makeup pumps, and other mechanical equipment. The floor at grade level elevation 100'-0" has an electrical penetration room, a staging area for the equipment hatch, and the access opening to the annex building. The electrical penetration area, trip switchgears, and motor control centers occupy most of the floor at elevation 117'-6". The floor at elevation 135'-3" is used for

*NRC Staff approval is required prior to implementing a change in this information.

the storage of main control room air cylinders and provides access to the annex building. The roof for these areas is at elevation 160'-6".

Areas 5 and 6 include facilities for storage and handling of new and spent fuel. The spent fuel pool, fuel transfer canal, and cask loading and cask washdown pits have concrete walls and floors. They are lined on the inside surface with stainless steel plate for leak prevention. The walls and major floors are constructed using concrete filled steel plate modules. The new fuel storage area is a separate reinforced concrete pit providing temporary dry storage for the new fuel assemblies. A 150-ton cask handling crane travels in the east-west direction. The location and travel of this crane prevents the crane from carrying loads over the spent fuel pool to preclude them from falling into the spent fuel pool. Mechanical equipment is also located in this area for spent fuel cooling, residual heat removal, and liquid waste processing. This equipment is generally nonsafety-related.]*

3H.2.2 Description of Shield Building

The shield building is the structure and annulus area that surrounds the containment building. It shares a common basemat with the containment building and the auxiliary building. The shield building uses concrete-filled steel plate construction (SC) as well as reinforced concrete (RC) structure. The figures in Section 1.2 show the layout of the shield building and its interface with the other buildings of the nuclear island.

Figure 3.8.4-5 shows the following significant features and the principal systems and components of the shield building:

- Shield building cylindrical structure
- Shield building roof structure
- RC/SC connections
- Air inlets and tension ring
- Knuckle region (connection to exterior wall of PCS tank)
- Compression ring (connection to interior wall of PCS tank)
- Passive containment cooling system (PCS) water storage tank (PCCWST)

The overall configuration of the shield building is established from functional requirements related to radiation shielding, missile barrier, passive containment cooling, tornado, and seismic event protection. These functional requirements led to establishing the design based on two primary design codes used for nuclear plant structures: 1) ACI 349 for reinforced concrete design, and 2) ANSI/AISC N690 for structural steel design.

The shield building SC walls are anchored to the RC basemat and shield building RC wall by mechanical connections. These RC-to-SC connections are also used in the other regions of the shield building, including:

- Auxiliary building RC roof connection to the shield building SC wall
- Auxiliary building RC wall connection to shield building SC wall
- Tension ring connection to the shield building RC roof

The connections provide for the direct transfer of forces from the RC reinforcing steel to the SC liner plates.

The cylindrical shield wall has an outside radius of 72.5 feet and a thickness of 36 inches. The cylindrical wall section that is a few feet below the auxiliary building roof line is a reinforced concrete (RC) structure. The section that is not protected by the auxiliary building is a steel concrete (SC) composite structure (see Figure 3H.5-16). The overall thickness of 36 inches is the same as the RC wall below. The concrete for the SC portion is standard concrete with compressive strength of

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6000 psi. The SC portion is constructed with steel surface plates, which act as concrete reinforcement. The 0.75-inch tie bars are welded to the steel faceplates to develop composite behavior of the steel faceplates and concrete. The shear studs are welded to the inside surface of the steel plate. The tie bar spacing is reduced in the higher stress regions. A typical SC wall panel is shown in Figure 3H.5-13.

The tension ring is located at the interface of the shield building steel concrete composite air inlet structures and the shield building reinforced concrete roof. The top of the tension ring interfaces with the RC roof slab. The tension ring supports the roof girders that are located under the RC roof slab. The bottom of the tension ring is attached to the air inlets structure. The bottom of the air inlets structure is attached to the top of the cylindrical SC wall of the shield building. The connection of the tension ring to the roof is of RC design and is described above.

The primary function of the tension ring is to resist the thrust from the shield building roof. The air inlets structure is located directly below the tension ring and includes the air openings that provide for natural circulation of cooling air. Though its steel plates are connected to the concrete infill by studs and tie bars, the tension ring is conservatively designed as a hollow steel box girder. The concrete infill is credited only for out-of-plane shear transfer and for stability of the steel plates. The tension ring is designed to have high stiffness and to remain elastic under required load combinations.

The air inlets structure is a 4.5-foot-thick SC structure with through-wall openings for air flow. The air inlet openings consist of circular pipes at a downward inclination of 38 degrees from the vertical. Steel plates on each face, aligned with the inner and outer flanges of the tension ring, serve as primary reinforcement. The concrete infill is connected to the steel plates with tie bars and studs. The top of the air inlets structure is welded to the underside of the tension ring. The bottom of the air inlets structure is welded to the SC wall.

The shield building conical roof steel structure consists of 32 radial beams. Between each pair of radial beams there are circumferential beams. A steel plate is welded to the top flanges of each beam and forms a surface on which the concrete is placed. The steel structure forms a conical shell that spans the area from the compression ring to the tension ring.

The outside diameter of the PCS tank (passive containment cooling water storage tank) intersects with the shield building roof at the knuckle region. Outside of the PCS tank, the concrete roof slab thickness is 3 feet and at the bottom of the PCS tank, the concrete thickness is 2 feet. The wall from the PCS tank applies a load to the roof slab, and also provides stiffness and increases the strength of the roof in that region.

The inside diameter of the PCS tank intersects with the roof slab at the compression ring. The compression ring provides the compression support for the conical roof dome. It consists of a composite structure having a curved steel beam section, which supports the concrete roof directly above it. The inside wall of the PCS tank is located above the concrete roof. Studs are placed on the top flange of the steel girder to allow the steel and concrete sections to act as a composite unit. The curved girder is designed to provide support for the steel structure during construction and during the initial placement of the concrete roof before the concrete has hardened sufficiently.

The PCS tank sits on top of the shield building roof. It is supported by and acts integrally with the conical roof. The inside surface has a liner that functions to provide leak protection, but is not required to provide structural strength to the structure. Leak chase channels are provided over the liner welds. The top elevation of the water inside the tank for the PCS has sufficient freeboard to preclude impact on the roof during the SSE.

3H.3 Design Criteria

[*The auxiliary and shield building structures are reinforced concrete structures, structural modules, and horizontal concrete slabs supported by composite structural steel framing.*

- *Seismic forces are obtained from the response spectrum analysis of the three-dimensional finite element analysis models as described in subsection 3H.4. The shear wall and floor slab design also considers out-of-plane bending and shear forces due to loading, such as live load, dead load, seismic, lateral earth pressure, hydrostatic, hydrodynamic, and wind pressure.*

- *The shield building roof and the passive containment cooling water storage tank are analyzed using three-dimensional finite element models with the ANSYS computer code*]* as described in subsection 3.8.4.4.1. [*Loads and load combinations include construction, dead, live, thermal, wind, and seismic. The response spectrum analysis of the nuclear island is supplemented by equivalent static acceleration analysis of a more detailed model of a quadrant of the shield building roof. The results from the more detailed analysis are used in the evaluation of the tension ring, air inlets, and radial beams. The seismic response of the water in the tank is analyzed in a separate analysis with seismic input defined by the floor response spectrum.*

- *The structural steel framing is used primarily to support the concrete slabs and roofs. Metal decking, supported by the steel framing, is used as form work for the concrete slabs and roofs.*

- *The finned floors for the main control room and the instrumentation and control room ceilings are designed as reinforced concrete slabs in accordance with American Concrete Institute standard ACI 349. The steel panels are designed and constructed in accordance with American Institute of Steel Construction Standard AISC N690. For positive bending, the steel plate is in tension and the steel plate with fin stiffeners serves as the bottom reinforcement. For negative bending, compression is resisted by the stiffened plate and tension by top reinforcement in the concrete.*]*

3H.3.1 Governing Codes and Standards

[*The primary codes and standards used in the design of the auxiliary and shield buildings are listed below:*

- *ACI 349-01, "Code Requirement for Nuclear Safety-Related Structure Steel" (refer to subsection 3.8.4.5 for supplementary requirements)*

- *ANSI/AISC N690-1994, "Specification for the Design, Fabrication and Erection of Safety-Related Steel Structures for Nuclear Facilities" (refer to subsection 3.8.4.5 for supplemental requirements).*]*

3H.3.2 Seismic Input

The SSE design response spectra are given in Figures 3.7.1-1 and 3.7.1-2. [*They are based on the Regulatory Guide 1.60 response spectra anchored to 0.30g, but are amplified at 25 Hertz to reflect larger high-frequency seismic energy content observed for eastern United States sites.*]* The nuclear island seismic analyses are summarized in Subsection 3.7.2.

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3H.3.3 Loads

[*The auxiliary and shield buildings are seismic Category I structures. The loads listed in the following subsections are used for the design of the building structures. All the listed loads are not necessarily applicable to all structures and their elements. Loads for which each structural element is designed are based on the conditions to which that particular structural element is potentially subjected.*]*

Dead Load (D):

[*The weight of all permanent construction and installations, including fixed equipment, is included as the dead load during its normal operating condition.*

The weight of minor equipment (not specifically included in the dead load), piping, cables and cable trays, ducts, and their supports was included as equivalent dead load (EDL). A minimum of 50 pounds per square foot (psf) was used as EDL. For floors with a significant number of small pieces of equipment, the total weight of miscellaneous small pieces of equipment, divided by the floor area of the room plus an additional 50 psf was used as the equivalent dead load.]*

Earth Pressure (H):

[*The static earth pressure acting on the structures during normal operation is considered in the design of exterior walls. The dynamic soil pressure, induced during a safe shutdown earthquake (SSE), is included as a seismic load.*]*

Live Loads (L):

[*The load imposed by the use and occupancy of the building is included as the live load. Live loads include floor area loads, laydown loads, fuel transfer casks, equipment handling loads, trucks, railroad vehicles, and similar items. The floor area live load is not applied on areas occupied by equipment whose weight is specifically included in the dead load. Live load is applicable on areas under equipment where access is provided, for instance, the floor under an elevated tank supported on legs.*

Floor loading diagrams are prepared for areas for component laydown. The diagrams show the location of major pieces of equipment and their foot-print loads or equivalent uniformly distributed loads.

The following live load items are considered in design:

A. Building floor loads

The following minimum values for live loads are used.

– *Structural platforms and gratings*	*100 psf*
– *Ground floors*	*250 psf*
– *All other elevated floors*	*200 psf* *(This load is reduced if the equivalent dead load for the floor is more than 50 psf. The sum of the live load and the equivalent dead load is 250 psf.)*

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B. Roof loads

The roof is designed for a uniform snow load of 63 psf calculated in accordance with ASCE 7-98. This corresponds to ground snow load of 75 psf, exposure factor of 1.0, thermal factor of 1.0, and an importance factor of 1.2.

C. Concentrated loads for the design of local members

– *Concentrated load on beams and girders (in load combinations that do not include seismic load)*	*5,000 pounds so applied as to maximize moment or shear. This load is not carried to columns or walls. It is not applied in areas where no heavy equipment will be located or transported, such as the access control areas.*
– *Concentrated load on slabs (considered with dead load only)*	*5,000 pounds so applied as to maximize moment or shear. This load is not carried to columns or walls. It is not applied in access control areas.*

In design reconciliation analysis, if actual loads are established to be lower than the above loads, the actual loads are used for reconciliation.

D. Temporary exterior wall surcharge

When applicable, a minimum surcharge outside and adjacent to subsurface wall of 250 psf is applied.

E. Construction loads

The additional construction loads produced by cranes, trucks, and the like, with their pickup loads, are considered. For steel beams supporting concrete floors, the weight of the wet concrete plus 100 psf uniform load and 5,000 pounds concentrated load, distributed near points of maximum shear and moment, is applied. A one-third increase in allowable stress is permitted.

Metal decking and precast concrete panels, used as formwork for concrete floors are designed for the wet weight of the concrete plus a construction live load of 20 psf uniform or 150 pounds concentrated. The deflection during normal operation is limited to span in inches divided by 180, or 0.75 inch, whichever is less.

F. Crane loads

The impact allowance for traveling crane supports and runway horizontal forces is in accordance with AISC N690.

G. Elevator loads

The impact allowance used for the elevator supports is 100 percent, applied to design capacity and weight of car plus appurtenances, unless otherwise specified by the equipment supplier.

H. Equipment laydown and major maintenance

Floors are designed for planned refueling and maintenance activities as defined on equipment laydown drawings.]*

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Wind Load

[*The wind loads are as follows:*

- *Design wind (W)*

  For the design of the exterior walls, wind loads are applied in accordance with ASCE 7-98 with a basic wind speed of 145 mph. The importance factor is 1.15, and the exposure category is C. Wind loads are not combined with seismic loads.

- *Tornado load (W_t)*

  The exterior walls of the auxiliary and shield buildings are designed for tornado. A maximum wind speed of 300 mph (maximum rotational speed: 240 mph, maximum translational speed: 60 mph) is used to design the structures.]*

Seismic Loads (E_s)

[*The SSE (E_s) is used for evaluation of the structures of the auxiliary and shield buildings. E_s is defined as the loads generated by the SSE specified for the plant, including the associated hydrodynamic loads and dynamic incremental soil pressure.*]*

Operating Thermal Loads (T_o)

[*Normal thermal loads for the exterior walls and roofs are addressed in the design. These correspond to positive and negative linear temperature gradients with the inside surface at an average 70°F and the outside air temperature at -40°F and +115°F, respectively. These loads are considered for the seismic Category I structures in combination with the SSE also. All exterior walls of the nuclear island above grade not protected by adjacent buildings are designed for these thermal loads. The thermal gradient is also applied to the portion of the shield building between the upper annulus and the auxiliary building.*

Normal thermal loads for the passive containment cooling system (PCS) tank design are calculated based on the outside air temperature extremes specified for the safety-related design. The PCS tank is assumed to be at 40°F when the outside air temperature is -40°F. The water in the PCS tank is assumed to be at 70°F when the outside air temperature is postulated to be at 115°F.

Normal thermal loads due to a thermal gradient in the structures below the grade level (exterior walls and basemat) are small and are not considered in the design.]*

Effects of Pipe Rupture (Y)

[*The evaluations consider the following loads:*

- *Accident design pressure load, P_a, within or across a compartment and/or building generated by the postulated pipe rupture, including the dynamic effects due to the pressure time history.*

  Main steam isolation valve (MSIV) and steam generator blowdown valve compartments are designed for a pressurization load of 6 pounds per square inch (psi).

- *Accident thermal loads, T_a, due to thermal conditions generated by the postulated pipe break and including T_o.*

Temperature gradients are based on an exterior air temperature of -40°F.

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The structural integrity of the west wall of the main control room is also evaluated for the jet impingement (Y_j)]*

3H.3.4 Load Combinations and Acceptance Criteria

[*Concrete structures are designed in accordance with ACI 349 for the load combinations and load factors given in Table 3.8.4-2. Steel structures are designed in accordance with AISC N690 for the load combinations and stress limit coefficients given in Table 3.8.4-1. The following supplemental requirements are applied for the use of AISC N690:*

- *In Section Q1.0.2, the definition of secondary stress applies to stresses developed by temperature loading only.*
- *In Section Q1.3, where the structural effects of differential settlement are present, they are included with the dead load, D.*
- *In Table Q1.5.7.1, the stress limit coefficients for compression are as follows:*
  - *1.3 instead of 1.5 in load combinations 2, 5, and 6*
  - *1.4 instead of 1.6 in load combinations 7, 8, and 9*
  - *1.6 instead of 1.7 in load combination 11*
- *In Section Q1.5.8, for constrained members (rotation and/or displacement constraint such that a thermal load causes significant stresses) supporting safety-related structures, systems, or components, the stresses under load combinations 9, 10, and 11 are limited to those allowed in Table Q1.5.7.1 as modified above.*]*

3H.4 Seismic Analyses

[*A global seismic analysis of the AP1000 nuclear island structure is performed to obtain building seismic response for the seismic design of nuclear safety-related structures. The seismic loads for the design of the shear walls and the slabs in the auxiliary building are based on a response spectrum analysis of the auxiliary building and the shield building 3D finite element models.*]* This analysis is described in Subsection 3.7.2. [*For determining the out-of-plane seismic loads on flexible slabs and wall segments, spectral accelerations are obtained from time history analyses or from the relevant response spectra, using the 7 percent damping curve. Hand calculations are performed to estimate the out-of-plane seismic forces and the corresponding bending moment in each shear wall and floor slab element to supplement the loads obtained from the global seismic analysis.*]*

3H.4.1 Live Load for Seismic Design

[*Floor live loads, based on requirements during plant construction and maintenance activities, are specified varying from 50 to 250 pounds per square foot.*

For the local design of members, such as the floors and beams, seismic loads include the response due to masses equal to 25 percent of the specified floor live loads or 75 percent of the roof snow load, whichever is applicable. These seismic loads are combined with 100 percent of the specified live loads, or 75 percent of the roof snow load, whichever is applicable. These live and snow loads are included as mass in calculating the vertical seismic forces on the floors and roof. The mass of equipment and distributed systems is included in both the dead and seismic loads.]*

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3H.5 Structural Design of Critical Sections

[*This subsection summarizes the structural design of representative seismic Category I structural elements in the auxiliary building and shield building. These structures are listed below and the corresponding location numbers are shown on Figure 3H.5-1. The basis for their selection to this list is also provided for each structure.*

(1) South wall of auxiliary building (column line 1), elevation 66'-6" to elevation 180'-0". (This exterior wall illustrates typical loads such as soil pressure, surcharge, temperature gradients, seismic, and tornado.) – see subsection 3H.5.1.1 and Figures 3H.5-2 and 3H.5-3

(2) Interior wall of auxiliary building (column line 7.3), elevation 66'-6" to elevation 160'-6" (This is one of the most highly stressed shear walls.) – see subsection 3H.5.1.2 and Figure 3H.5-4

(3) West wall of main control room in auxiliary building (column line L), elevation 117'-6" to elevation 153'-0". (This illustrates design of a wall for subcompartment pressurization.) – see subsection 3H.5.1.3 and Figure 3H.5-12

(4) North wall of MSIV east compartment (column line 11 between column lines L and M), elevation 117'-6" to elevation 153'-0". (The main steam line is anchored to this wall segment.) – see subsection 3H.5.1.4 and Figure 3H.5-5

(5) Roof slab at elevation 180'-0" adjacent to shield building cylinder. (This is the connection between the two buildings at the highest elevation.) – see subsection 3H.5.2.1 and Figure 3H.5-7

(6) Floor slab on metal decking at elevation 135'-3". (This is a typical slab on metal decking and structural steel framing.) – see subsection 3H.5.2.2 and Figure 3H.5-6

(7) 2'-0" slab in auxiliary building (operations work area (tagging room) ceiling) at elevation 135'-3". (This illustrates the design of a typical 2'-0" thick concrete slab.) – see subsection 3H.5.3.1 and Figure 3H.5-8. (Note: The 'Tagging Room' has been renamed as "Operations Work Area." However, to avoid changing the associated design and analysis documents, this room is referred to as the 'Tagging Room.')

(8) Finned floor in the main control room at elevation 135'-3". (This illustrates the design of the finned floors.) – see subsection 3H.5.4 and Figure 3H.5-9

(9) Shield building roof/exterior wall of PCS water storage tank. (This is a unique area of the roof and water tank.) – see subsection 3H.5.6.3

(10) Shield building roof/interior wall of PCS water storage tank. (This is a unique area of the roof and water tank.) – see subsection 3H.5.6.2

(11) Shield building roof, tension ring, and air inlet. (This is the junction between the shield building roof and the cylindrical wall of the shield building.) – see subsections 3H.5.6 and 3H.5.6.1

(12) Divider wall between the spent fuel pool and the fuel transfer canal. (This wall is subjected to thermal and seismic sloshing loads.) – see subsection 3H.5.5.1 and Figure 3H.5-10

(13) Shield building SC cylinder is the exposed portions of the shield building that are not protected by the Auxiliary Building and is a steel concrete composite structure – see

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subsection 3H.5.7.1, Figure 3H.5-16, and Figures 5 and 6 of APP-GW-GLR-602 (Reference 1)

(14) *Shield building SC to RC connection is the region of the shield building that anchors the SC cylindrical wall modules to the RC basemat and wall of the shield building – see subsection 3H.5.7.2, Figure 3H.5-16, and Figures 1, 2, and 3 of APP-GW-GLR-602 (Reference 1)*

The design implemented in fabrication and construction drawings and instructions will have the design shown, an equal design, or a better design for the key structural elements.]*

3H.5.1 Shear Walls

Structural Description

[*Shear walls in the auxiliary building vary in size, configuration, aspect ratio, and amount of reinforcement. The stress levels in shear walls depend on these parameters and the seismic acceleration level. The range of these parameters and the stress levels in various regions of the most severely stressed shear wall are described in the following paragraphs.*

The height of the major structural shear walls in the auxiliary building ranges between 30 to 120 feet. The length ranges between 40 and 260 feet. The aspect ratio of these walls (full height/full length) is generally less than 1.0 and often less than 0.25. The walls are typically 2 to 5 feet thick, and are monolithically cast with the concrete floor slabs, which are 9 inches to 2 feet thick. Exterior shear walls are several stories high and do not have many large openings. Interior shear walls, however, are discontinuous in both vertical and horizontal directions. The in-plane behavior of these shear walls, including the large openings, is adequately represented in the analytical models for the global seismic response. Where the refinement of these finite element models is insufficient for design of the reinforcement, for example in walls with a large number of openings, detailed finite element models are used.

The shear walls are used as the primary system for resisting the lateral loads, such as earthquakes. The auxiliary building shear walls are also evaluated for flexure and shear due to the out-of-plane loads.]*

Design Approach

[*The auxiliary building shear walls are designed to withstand the loads specified in subsection 3H.3.3. Beside dead, live, and other normal operating condition loads, the following loads are considered in the shear wall design:*

- *Seismic loads*
  - *The SSE loads for the wall are obtained from the seismic analyses of auxiliary/shield buildings that are described in subsection 3H.4.*
  - *Calculations are performed by considering shear wall segments bounded by the floors below and above the segment and the adjacent walls perpendicular to, on both sides of, the segment under consideration. Appropriate boundary conditions are assumed for the four edges of the segment. Natural frequencies of wall segments are determined using finite element models or text book formulas for the frequency of plate structures. Corresponding spectral acceleration is determined from the applicable response spectrum.*

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- *Exterior walls, below grade level, are also evaluated for dynamic earth pressure exerted during an SSE for two cases:*

  - *Dynamic earth pressure calculated in accordance with ASCE 4-98*

  - *Passive earth pressure*

- *Accident pressure load*

  - *Shear walls of the main steam isolation valves (MSIV) rooms are designed for 6 pounds per square inch (psi) differential pressure acting in conjunction with the seismic loads. Member forces due to accident pressure and SSE are combined by absolute sum.*

  - *The main control room wall of the east MSIV compartment is evaluated for the pressure and the jet load due to a postulated main steamline break.*

- *Tornado load*

  For exterior walls above grade level, tornado loads are considered.

The design temperatures for thermal gradient are included in Table 3H.5-1.

The shear walls are designed for the load combinations, as applicable, contained in Table 3.8.4-2. The wall sections are designed in accordance with the requirements of ACI 349-01.]*

3H.5.1.1 Exterior Wall at Column Line 1

[*The wall at column line 1 is the exterior wall at the south end of the nuclear island. The reinforced concrete wall extends from the top of the basemat at elevation 66'-6" to the roof at elevation 180'-0". It is 3'-0" thick below the grade and 2'-3" thick above the grade.*

The wall is designed for the applicable loads including dead load, live load, hydrostatic load, static and dynamic lateral soil pressure loads, seismic loads, and thermal loads. For various segments of this wall, Table 3H.5-2 provides the listing and magnitude of the various design loads and Table 3H.5-3 presents the details of the wall reinforcement. The sections where the required reinforcement is calculated are shown in Figure 3H.5-2 (Sheet 1). Typical wall reinforcement is shown on Figure 3H.5-3.]*

3H.5.1.2 Wall at Column Line 7.3

[*The wall at column line 7.3 is a shear wall that connects the shield building and the nuclear island exterior wall at column line I. It extends from the top of the basemat at elevation 66'-6" to the top of the roof. The wall is 3 feet thick below the grade at elevation 100'-0" and 2 feet thick above the grade. Out-of-plane lateral support is provided to the wall by the floor slabs on either side of it and the roof at the top.*

The auxiliary building design loads are described in Section 3H.3.3, and the wall is designed for the applicable loads.

For various segments of this wall, the corresponding governing load combination and associated design loads are shown in Table 3H.5-4. Table 3H.5-5 presents the details of the wall reinforcement. The sections where the required reinforcement is calculated are shown in Figure 3H.5-2 (Sheet 2). Typical wall reinforcement is shown on Figure 3H.5-4]*

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3H.5.1.3 Wall at Column Line L

[*The wall at column line L is a shear wall on the west side of the Main Control Room. It extends from the top of the basemat at elevation 66'-6" to the top of the roof. The wall is 2 feet thick. Out-of-plane lateral support is provided to the wall by the floor slabs on either side of it and the roof at the top. The segment of the wall that is a part of the main control room boundary is from elevation 117'-6" to elevation 135'-3".*

The auxiliary building design loads are described in subsection 3H.3.3, and the wall is designed for the applicable loads. In addition to the dead, live and seismic loads, the wall is designed to withstand a 6 pounds per square inch pressure load due to a pipe break in the MSIV room even though it is a break exclusion area. This wall segment is also designed to withstand a jet load due to the pipe break.

The governing load combination and associated design loads are those due to the postulated pipe rupture and are shown in Table 3H.5-6. Table 3H.5-7 and *Figure 3H.5-12 present the details of the wall reinforcement. The sections where the required reinforcement is calculated are shown in Figure 3H.5-2 (Sheet 3).*]*

3H.5.1.4 Wall at Column Line 11

[*The north wall of the MSIV east compartment, at column line 11 between elevation 117'-6" and elevation 153'-0", has been identified as a critical section.*

The segment of the wall between elevation 117'-6" and elevation 135'-3" is 4 feet thick, and several pipes such as the main steam line, main feed water line, and the start-up feed water line are anchored to this wall at the interface with the turbine building.

The wall segment from elevation 135'-3" to elevation 153'-0" does not provide support to any high energy lines, and is 2 feet thick. This portion does not have to withstand reactions from high energy line breaks.

The wall is designed to withstand loads such as the dead load, live load, seismic load and the thermal load. The MSIV room is a break exclusion area, but the design also considered the loads associated with one square foot pipe rupture in the MSIV room, such as compartment pressurization, jet load, and the reactions at the pipe anchors. The loads on the pipe anchor include pipe rupture loads for breaks in the turbine building.

The wall structure is analyzed using three dimensional finite element analyses supplemented by hand calculations. Analyses are performed for individual loads, and design loads are determined for applicable load combinations from Table 3.8.4-2.

Typical wall reinforcement is shown in Figure 3H.5-5.]*

3H.5.2 Composite Structures (Floors and Roof)

[*The floors consist of a concrete slab on metal deck, which rests on structural steel floor beams. Several floors in the auxiliary building are designed as one-way reinforced concrete slabs supported continuously on steel beams. Typically, the beams span between two reinforced concrete walls. The beams are designed as composite with formed metal deck spanning perpendicular to the members. Unshored construction is used. For the floors, beams are typically spaced at about 6-feet intervals and spans are between 16 feet and 25 feet.*]*

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Structural Description

[*A typical layout of these floors is shown in Figure 3H.5-6. The metal deck rests on the top flange of the structural steel floor beam, with the longitudinal axes of the metal deck ribs and floor beams placed perpendicular to each other. The depth of the ribs for 9-inch concrete floor slabs and 15-inch deep concrete roof slabs are 3 inches and 4.5 inches respectively. The concrete slab is tied to the structural steel floor beam by shear connectors, which are welded to the top flange of the floor beam. The concrete slab and the floor beams form a composite floor system. For the design loads after hardening of concrete, the transformed section is used to check the stresses.*

The construction sequence is as follows:

- *The structural steel floor (floor beam, metal deck, and shear connectors) is fabricated in the shop, brought to the floor location, and placed in position. In some cases, the beams and deck are preassembled and placed as a module.*
- *The metal deck is used as the formwork, and concrete is poured on the metal deck. Until concrete hardens, the load is carried by the metal deck and the steel floor beam.*
- *During concreting, no shoring is provided.*]*

Design Approach

[*The floor design considers the dead, live, construction, extreme environmental, and other applicable loads identified in Section 3H.3.3. The design floor loading includes the equipment attached to the floor. The end condition for the steel beams is simply supported, or continuous. The seismic load is obtained using the applicable floor acceleration response spectrum (7 percent damping for the SSE loads).*

The load combinations applicable to the design of these floors are shown in Tables 3.8.4-1 and 3.8.4-2. The design of the floor system is performed in two parts:

- *Design of structural steel beams*
  - *The structural steel floor beams are evaluated to withstand the weight of wet concrete during the placement of concrete. The composite section is designed for the design loads during normal and extreme environment conditions. Shear connectors are also designed.*
- *Design of concrete slab*
  - *The concrete slab and the steel reinforcement of the composite section are evaluated for normal and extreme environmental conditions. The slab concrete and the reinforcement is designed to meet the requirements of American Concrete Institute standard ACI 349-01 "Code Requirements for Nuclear Safety-Related Structures."*
  - *The slab design considers the in-plane and out-of-plane seismic forces. The global in-plane and out-of-plane forces are obtained from the response spectrum analysis of the 3D finite element model of the auxiliary and shield buildings. The out-of plane seismic forces due to floor self-excitation are determined by hand calculations using the applicable vertical seismic response spectrum and slab frequency.*]*

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3H.5.2.1 Roof at Elevation 180′-0″, Area 6 (Critical Section is between Col. Lines N & K-2 and 3 & 4)

[*The layout of this segment of the roof is shown in Figure 3H.5-7 as Region "B." The concrete slab is 15 inches thick, plus 4.5-inch deep metal deck ribs. It is composite with 5 feet deep plate girders, spaced 14'-2" center to center, by using shear connectors. The girder flanges are 20" x 2" and the web is 56" x 7/16". The girders span approximately 64 feet in the north-south direction and are designed as simply supported. The concrete slab between the girders behaves as a one-way slab and is designed to span between the girders.*

The roof girders are designed for dead and live loads, including construction loads (with wet concrete) with simple support end conditions. A one-third increase in allowable stress is permitted for the construction load combination.

The girders are also evaluated as part of the composite beam after drying of concrete. The composite roof structure is designed to withstand dead and live load / snow load, as well as the wind, tornado and seismic loads.

A typical connection of the roof slab to the shield building is shown in Figure 3H.5-7. The figure shows the arrangement of reinforcement at the connection in the fuel building roof, the shield building cylindrical wall, and the walls of the auxiliary building just below the roof. The design summary is shown in Table 3H.5-10.]*

3H.5.2.2 Floor at Elevation 135′-3″, Area 1 (Between Column Lines M and P)

[*The design of a typical composite floor is shown in Figure 3H.5-6. The design summary is shown in Table 3H.5-11. The concrete slab is 9 inches thick, plus 3-inch deep metal deck ribs. The floor beams are typically W14x26.*

- *The floor beams are designed for construction load (with wet concrete) with simple support end conditions. The design loads include the dead load and a construction live load of 100 pounds per square foot (psf) distributed load plus 5000 pounds concentrated load near the point of maximum shear and moment. A one-third increase in allowable stress is permitted.*

- *The floor beams are also designed as part of the composite beam after drying of the concrete. Because of continuity of rebars into the wall and the connection of the bottom flange to the support embedment, the end support condition is considered as fixed.*]*

3H.5.3 Reinforced Concrete Slabs

[*Reinforced concrete floors in auxiliary building are 24 inch or 36 inch thick. These floors are constructed with 16" or 28" of reinforced concrete placed on the top of 8 inch thick precast concrete panels. The 8" thick precast concrete panels are installed at the bottom to serve as the formwork and withstand the load of wet concrete slab. The main reinforcement is provided in the precast panels which are connected to the concrete placed above it by shear reinforcement. The precast panels and the cast-in-place concrete act together as a composite reinforced concrete slab. Examples of such floors are the Operations Work Area (Tagging Room) ceiling slab at elevation 135 ft 3 inches in Area 2, and the Area 5/6 elevation 100'-0" slab between column lines 1 & 2.*]*

*NRC Staff approval is required prior to implementing a change in this information.

3H.5.3.1 Operations Work Area (Tagging Room) Ceiling

The tagging room (room number 12401) location is shown on Figure 1.2-8. [*Figure 3H.5-8 shows the typical cross section and reinforcement. The design summary is shown in Table 3H.5-12. Design dimensions of the Operations Work Area (Tagging Room) Ceiling are as follows:*

Room Size:	*16'-0" x 11'-10"*
Boundary Conditions:	*Fixed at Walls J and K*
Clear Span:	*16'-0"*
Slab Thickness:	*Total = 24 inches*
	Precast Panel = 8 inches
	Cast-in-Place = 16 inches

The two precast concrete panels, each 5'-11" wide and spanning over 16'-0" clear span, are installed to serve as the formwork.]*

3H.5.4 Concrete Finned Floors

[*The ceilings of the main control room and the instrumentation and control rooms in the auxiliary building are designed as finned-floor modules. A typical floor design is shown in Figure 3H.5-9. A finned floor consists of a 24-inch-thick concrete slab poured over a stiffened steel plate ceiling. The fins, welded to stiffen the steel plate, are half inch by 9 inch rectangular sections perpendicular to the plate. Shear studs are welded on the other side of the steel plate, and the steel and concrete act as a composite section. The fins are exposed to the environment of the room and enhance the heat-absorbing capacity of the ceiling. Several shop-fabricated steel panels, cut to room width and placed side by side perpendicular to the room length, are used to construct the stiffened plate ceiling in a modularized fashion. The stiffened plate with fins is designed to withstand construction loads prior to concrete hardening.*

The main control room ceiling fin floor is designed for the dead, live, and the seismic loads. The design summary is shown in Table 3H.5-13.

The finned floor structure is evaluated for the load combinations listed in Tables 3.8.4-1 and 3.8.4-2.]*

Design Methodology

[*The finned floors are designed as reinforced concrete slabs in accordance with ACI Standard 349. For positive bending, the steel plate is in tension. The steel plate with fin stiffeners serves the function of bottom rebars. For negative bending, the potential for buckling due to compression in this element is checked by using the criteria of American National Standards Institute/American Institute of Steel Construction standards ANSI/AISC N690-94. Twisting, and therefore lateral buckling of the stiffener, is restrained by the concrete.*

The finned floors resist vertical and in-plane forces for both normal and extreme loading conditions. For positive bending, the concrete above the neutral axis carries compressive stresses and the stiffened steel plate resists tension. Negative bending compression is resisted by the stiffened plate and tension by top rebars in the concrete. The neutral axis for negative bending is located in the stiffened plate section, and the concrete in tension is assumed inactive. Horizontal in-plane forces are resisted by the stiffened plate and longitudinal rebars.

Minimum top reinforcement is provided in the slab in each direction for shrinkage and temperature crack control. In addition, top reinforcement located parallel to the stiffeners is used as tension

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reinforcement in negative bending. The stiffened plate provides crack control capability for the bottom of the slab in the transverse direction.

Composite section properties, based on an all steel-transformed section, as detailed in Section Q1.11 of ANSI/AISC N690-94, are used to design the following:

- *Weld strength between stiffener and the steel plate*
- *Spacing of the shear studs for the composite action*

The stiffened plate alone is designed to resist all construction loads prior to the concrete hardening. The plate is designed against the criteria for bending and shear, specified in ANSI/AISC N690-94, Sections Q1.5.1.4 and Q1.5.1.2. In addition, the weld between the stiffener and the steel plate is designed to satisfy the code requirements.]*

3H.5.5 Structural Modules

[*Structural modules are used for some of the structural elements on the south side of the auxiliary building. These structural modules are structural elements built up with welded steel structural shapes and plates. The modules consist of steel faceplates connected by steel trusses as shown in Figure 3.8.3-2. The primary purpose of the trusses is to stiffen and hold together the faceplates during handling, erection, and concrete placement. The thickness of the steel faceplates is 0.5 inch except in a few local areas. The nominal spacing of the trusses is 30 inches. Shear studs are welded to the inside faces of the steel faceplates. Faceplates are welded to adjacent faceplates with full penetration welds so that the weld is at least as strong as the plate. The structural wall modules are anchored to the concrete base by reinforcing steel dowels or other types of connections embedded in the reinforced concrete below. After erection, concrete is placed between the faceplates.*

These modules include the spent fuel pool, fuel transfer canal, and cask loading and cask washdown pits. The structural modules are similar to the structural modules for the containment internal structures (see description in subsection 3.8.3 and Figures 3.8.3-8, 3.8.3-14, 3.8.3-15 and 3.8.3-17). Figure 3.8.4-5 shows the location of the structural modules in the auxiliary building. The structural modules extend from elevation 66'-6" to elevation 135'-3".

The loads and load combinations applicable to the structural modules in the auxiliary building are the same as for the containment internal structures]* (Subsection 3.8.3.5.3) [*except that there are no ADS nor pressure loads due to pipe breaks.*

The design methodology of these modules in the auxiliary building is similar to the design of the structural modules in the containment internal structures]* described in Subsection 3.8.3.5.3.

3H.5.5.1 West Wall of Spent Fuel Pool

[*Figure 3H.5-10 shows an elevation of the west wall of the spent fuel pool (column line L-2), and element numbers in the finite element model. The wall is a 4 feet thick concrete filled structural wall module.*

A finite element analysis is performed for seismic, thermal, and hydrostatic loads with the following assumptions:

- *The seismic in-plane and out-of-plane forces are obtained from the response spectrum analysis of the 3D finite element model of the auxiliary and shield buildings.*

- *The thermal loads are applied as linearly varying temperatures between the inner and outer faces of the walls and floors.*

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- *The hydrostatic loads are applied to the spent fuel pool walls and floors, which is considered full with water. This provides the loads for the design of the divider wall.*

- *The seismic sloshing is modeled in the spent fuel pool.*

The concrete filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349. The face plates are treated as reinforcing steel.

Methods of analysis are based on accepted principles of structural mechanics and are consistent with the geometry and boundary conditions of the structures. Both computer codes and hand calculations are used.

Table 3H.5-8 shows the required plate thickness for certain critical locations. The steel plates are half inch thick.]*

3H.5.6 Shield Building Roof and Connections

[*The shield building roof is a reinforced concrete shell (supporting the passive containment cooling system tank and air diffuser), which is supported on a structural steel module. The structural configuration is shown on sheets 7, 8, and 9 of Figure 3.7.2-12. Air intakes are located at the top of the cylindrical portion of the shield building. The conical roof supports the passive containment cooling system tank. The conical roof is constructed as a structural steel module and lifted into place during construction. Steel beams provide permanent structural support for steel liner and concrete. The concrete is cast in place. Connection between concrete and steel liner are made using shear studs.*

The design of the shield building is shown in Figure 3H.5-11 (Sheets 1-6). These figures show the typical details of the "Tension Ring," the "Air Inlet Structure," and the "Exterior Wall of the Passive Containment Cooling System Tank." Figure 3H.5-16, Sheets 1 and 2, also shows the typical dimensions of the surface plates and the SC to RC connections on the shield building cylindrical segment.

A detailed ANSYS model was used to represent these components of the enhanced design. Analyses were performed to determine the response of the structures for the dead weight, hydrostatic load due to PCS water, snow load, wind load, tornado load, seismic load (including seismic-induced pressure on PCS wall), and thermal loads. The design was evaluated to comply with the requirements of ANSI/AISC N690-94 and of ACI 349-01.

The design summaries of the components are included in Table 3H.5-9.

The steel frame for the shield building roof and the concrete placed directly thereon is designed to AISC N690.

- *In the radial direction, the steel beams, the steel surface plate, and the concrete are evaluated as a composite section using the axial and bending member forces in the steel and concrete section from the finite element analyses. The steel stresses and the end connection are calculated assuming the steel alone resists all loads applied before the concrete has reached 75 percent of its required strength and the effective composite section resists all loads applied after that time.*

- *The concrete is evaluated using all member forces in the concrete and surface steel plate from the finite element analyses (in-plane and out-of-plane forces and moments). The circumferential channels are provided for construction only and are not modeled in the finite*

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element analysis or credited for resisting permanent loads. The concrete section is evaluated by the strength method of ACI-349. The steel plate is not considered as reinforcement in the circumferential direction.

Additional information is provided in Table 3H.5-15.]*

3H.5.6.1 Air Inlets and Tension Ring

[*The configuration and plate size of the air inlets enhance their structural performance. The air inlets structure (as shown on Figure 3H.5-14) is located at the top of the cylindrical wall portion of the shield building, beginning at approximately elevation 251' and rising to approximately elevation 266'. The air inlets serve as the intake for air as part of the PCS.*

Above the air inlets, at approximately elevation 266', is the connection designated as the tension ring that connects and supports the conical roof. The tension ring also contains 32 radial beam seat connections where the W36 x 393 radial beams for the conical roof are connected.

The air inlets region is 4.5-feet thick with steel plates on each face as the primary reinforcement, which are connected using tie bars. Near the bottom of the air inlet structure, the thickness transitions to 3 feet thick to connect with the shield building cylinder. The air inlet openings are formed using pipe at a downward inclination of 38 degrees from the vertical. The pipe spacing is approximately 2.81 degrees circumferentially with shear studs welded to the outside surface of the pipes. The tie bars are located with three bars between adjacent air inlets at each elevation at maximum design spacing of 8.5 inches vertically. At approximately the same elevations as the tie bars, two 3/4-inch by 6-inch (minimum) shear studs are located between the tie bars except at elevations where there is interference with the air inlet pipes. Tie bars and studs may be omitted in local areas due to design features and other obstructions.

The tension ring is designed as a structural steel box structure with concrete infill and shear studs. Also the connection of the RC conical roof to the tension ring is designed to be a mechanical connection. The air inlets and tension ring design methodology is supported by linear analysis and benchmarked nonlinear analysis. The tension ring is designed to ANSI/AISC N690 and is a concrete-filled box girder, with two continuous 1.5-inch-thick steel plates top and bottom, which connect the inner liner plate to the outer liner plate, as shown in Figure 3H.5-15.]*

3H.5.6.2 Compression Ring and Interior Wall of Passive Containment Cooling Water Storage Tank

[*The other areas of the shield building are designed to existing industry code requirements, and include the conical roof, the passive containment cooling water storage tank, the compression ring, the knuckle region, and their related attachments. These areas are designed as RC structures in accordance with ACI 349. The steel frame for the roof is designed for the applicable building code ANSI/AISC N690. The concrete roof is designed to ACI 349 requirements without credit for the steel plate on the bottom of the concrete. The configuration and reinforcement of the compression ring and the connection to the interior wall of the passive containment cooling water storage tank is shown in Figure 3H.5-11.*

Additional information is provided in Table 3H.5-15.]*

3H.5.6.3 Knuckle Region and Exterior Wall of Passive Containment Cooling System Tank

[*The exterior wall of the passive containment cooling system tank is two feet thick. The wall starts at the tank floor elevation of 293' 9". There is a stainless steel liner on the inside surface of the tank. The*

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wall liner consists of a plate with stiffeners and welded studs on the concrete side of the plate. Leak chase channels are provided over the liner welds. The reinforcement in the concrete wall is designed without taking credit for the strength provided by the liner. The governing loads for design of the exterior wall are the hydrostatic pressure of the water, the in-plane and out-of-plane seismic response, and the temperature gradient across the wall. The reinforcement is shown in Figure 3H.5-11. The reinforcement required and the reinforcement provided is summarized in Table 3H.5-9.

Additional information is provided in Table 3H.5-15.]*

3H.5.7 Shield Building Cylinder (SC)

3H.5.7.1 Shield Building Cylindrical Wall

[*The shield building surrounds the containment vessel and shares a common basemat with the containment vessel and the auxiliary building. The cylindrical shield wall has an outside radius of 72.5 feet and a thickness of 36 inches. The cylindrical wall section that is below the auxiliary building roof line is a reinforced concrete structure. The section that is not protected by the auxiliary building is a steel concrete composite structure, where two 0.75-inch plates act compositely with 34.5 inches of concrete via tie bars and shear studs. The steel plate modules are connected to the reinforced concrete basemat and walls by mechanical connectors as described below.*

A typical configuration of the SC wall is shown in Figure 3H.5-13. The overall thickness of 36 inches is the same as the RC wall below. The concrete for the SC portion is standard concrete with a compressive strength of 6000 psi. The SC portion is constructed with steel surface plates, which act as concrete reinforcement. The nominal thickness of the steel faceplates is 0.75 inches. In each module, tie bars are welded to the steel faceplates to develop composite behavior of the steel faceplates and concrete. The shear studs are welded to the inside surface of the steel plate to provide composite action. The tie bars are at closer spacing in the higher stress regions. The reinforcement detailing incorporates ACI 349 requirements.

The panels of the SC wall are welded together with a complete joint penetration weld.

The wall is designed for the applicable loads described in subsection 3H.3.3. A finite element analysis is performed to determine the design forces.

Table 3H.5-14 shows the design summary for the enhanced shield SC cylindrical wall. The three sheets represent locations in the shield building cylinder that have some of the largest demands due to mechanical loads. The element on the west side at grade near the RC/SC connection has large tension forces due to overturning of the cylinder under seismic demand. This area is one of the most stressed elements in tension. The element near the fuel handling building roof at elevation 180' is an element with high out-of-plane shear due to the interaction between the fuel handling building and the cylinder during an earthquake. This element is located close to the fuel building roof. The element above wall 7.3 at elevation 175' has the largest demand for out-of-plane shear in the general part of the cylindrical wall away from the SC/RC connection and the interface with the auxiliary building roof.

Additional discussion and information are provided in Section 4 and Figures 5 and 6 of APP-GW-GLR-602 (Reference 1).]*

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3H.5.7.2 Reinforced Concrete (RC)/Steel Concrete Composite (SC) Horizontal and Vertical Connections

[*The steel plate modules are anchored to the RC basemat and walls of the shield building by mechanical rebar connections. The connectors provide for the direct transfer of forces from the RC reinforcing steel to the SC liner plates.*

At the horizontal connection at the interface with the RC structure that occurs on the bottom of the lowest SC wall module, each vertical reinforcing bar in the RC basemat wall is connected to a mechanical coupler. A similar vertical connection occurs on the vertical edges of SC wall modules that interface with the RC portion of the shield building wall. In the vertical connection, each hoop reinforcing bar in the RC wall is connected to a mechanical coupler and forces are transferred directly from the hoop bars to the SC liner plate. The mechanical connections are designed to the stress limits of ANSI/AISC N690 for loads in the reinforcing bars equivalent to 125 percent of the specified yield strength of the weaker of the steel plate or reinforcing bar and are proven components used in existing structures. This design basis exceeds the maximum demand that occurs on the west side of the shield building at grade and is summarized in Sheet 3 of Table 3H.5-14. This connection improves the overall ductility of the RC/SC connection.

Additional discussion and information are provided in Section 4 and Figures 1, 2, 3, and 4 of APP-GW-GLR-602 (*Reference 1*).]*

3H.5.8 References

1. [*APP-GW-GLR-602, Revision 1 (Proprietary) and APP-GW-GLR-603, Revision 1 (Non-Proprietary), "AP1000 Shield Building Design Details for Select Wall and RC/SC Connections," Westinghouse Electric Company LLC.*]*

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Table 3H.5-1
Nuclear Island: Design Temperatures for Thermal Gradient

Structure (See detail in Subsection 3H.3.3.)	Load	Temperature (°F)		Remark
PCS Tank Walls	Normal Thermal, T_o	[*(Outside)* *-40* *+115*	*(Inside)* *+40* *+70*]*	–
Roofs and Exterior Walls Above Grade Air Temperatures	Normal Thermal, T_o Accident Thermal, T_a	[*(Outside)* *-40* *+115* *-40* *-40*	*(Inside)* *+70* *+70* *+132* *+212*]*	– MSIV room Fuel handling area
Roofs and Exterior Walls Above Grade Concrete Temperatures	Normal Thermal, T_o Accident Thermal, T_a	[*(Outside)* *-21.6* *-22.8* *-25.4* *+3.2* *+109.1* *+108.0* *+107.5* *+98.6* *-40* *-40* *+63*	*(Inside)* *+47* *+48.4* *+51.5* *+46.6* *+79.2* *+80.7* *+81.3* *+81.3* *+132* *+212* *+212*]*	24″ thickness 27″ thickness 36″ thickness 15″ insulated roof 24″ thickness 27″ thickness 36″ thickness 15″ insulated roof MSIV room Fuel handling area Insulated roof
Interior Walls/Slabs Concrete Temperatures	Normal Thermal, T_o Accident Thermal, T_a	[*(Side 1)* *N/R* *+70* *+70*	*(Side 2)* *N/R* *+132* *+212*]*	– MSIV room Fuel handling area
Exterior Walls Below Grade	Normal Thermal, T_o Accident Thermal, T_a	N/R N/R	N/R N/R	– –
Basemat	Normal Thermal, T_o Accident Thermal, T_a	N/R N/R	N/R N/R	– –
Shield Building (Between Upper Annulus and Auxiliary Building)	Normal Thermal, T_o Accident Thermal, T_a	[*(Outside)* *-40* *+115* *-40* *N/R*	*(Inside)* *+70* *+70* *+132* *N/R*]*	– MSIV room wall Rest of wall

Notes:
1. N/R means loads due to a thermal gradient are not required to be considered.
2. Based on ACI 349-01 (Appendix A), the base temperature for the construction is assumed to be 70°F.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-2
Exterior Wall at Column Line 1 Forces and Moments in Critical Locations
(Units: kips, ft)

Load Combination	M_X	M_Y	M_{XY}	T_X	T_Y	T_{XY}
Elevation 180′-0″ to 135′-3″						
[*D + L + H + Ta*		177.8	3.1		115.5	8.8
1.05 D + 1.3 L + 1.3 H + 1.2 To]*	106.4		5.6	117.0		23.9
Elevation 135′-3″ to 100′-0″						
[*D + L + H + Ta*		50.8	0.3		89.8	104.8
D + L + H + Ta	82.9		7.6	172.9		24.8
D + L + H + Ta]*	60.0		3.6	165.7		106.0
Elevation 100′-0″ to 82′-6″						
[*1.05 D + 1.3 L + 1.3 H + 1.2 To*		48.1	8.4		106.1	17.3
D + L+ Es]*	1.8		5.4	15.6		58.6
Elevation 82′-6″ to 66′-6″						
[*D + L – Es*		93.8	26.5		170.7	31.5
0.9 D + Es		32.7	27.2		182.1	42.4
0.9 D + Es]*	15.5		27.2	18.6		42.4

Note:
X is along the horizontal direction, and Y is in the vertical direction.

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Table 3H.5-3
Exterior Wall on Column Line 1 Details of Wall Reinforcement (in^2/ft)
(See Figure 3H.5-2 for Locations of Wall Sections.)

Wall Segment (See detail in Subsection 3H.5.1.1.)	Location	Required(2)			*[Provided (Minimum)]**		
		Vertical	Horizontal	Shear	*Vertical*	*Horizontal*	*Shear*
Wall Section 1, 6							
Elevation 180′-0″ to 135′-3″				NR			None
	Outside Face	3.48	2.65		[*3.91*	*3.12*	
	Inside Face	1.94	1.52		*3.12*	*3.12*]*	
Wall Section 2, 3, 7							
Elevation 135′-3″ to 100′-0″				NR			None
	Outside Face	1.88	3.04		[*3.12*	*3.12*	
	Inside Face	1.77	2.23		*3.12*	*3.12*]*	
Wall Section 4, 8							
Elevation 100′-0″ to 82′-6″				0.003			[*0.44*]*
	Outside Face	1.42	0.70		[*3.12*	*1.56*	
	Inside Face	1.01	0.70		*3.12*	*1.27*]*	
Wall Section 5, 9							
Elevation 82′-6″ to 66′-6″				0.27			[*1.00*]*
	Outside Face	2.29	0.87		[*4.39*	*1.27*	
	Inside Face	1.87	0.87		*3.12*	*1.27*]*	

Note:

1. NR = not required.
2. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-4
Interior Wall at Column Line 7.3 Forces and Moments in Critical Locations
(Units: kips, ft)

Load Combination	M_X	M_Y	M_{XY}	T_X	T_Y	T_{XY}
From Roof to Elevation 155′-6″						
[*1.05 D + 1.3 L + 1.2 To*		135.3	10.9		117.3	210.2
1.05 D + 1.3 L + 1.2 To]*	75.5		4.1	229.8		94.3
Elevation 155′-6″ to 135′-3″						
[*0.9 D – Es*		14.1	1.3		160.8	228.7
D + L – Es]*	28.0		1.0	29.8		231.7
Elevation 135′-3″ to 117′-6″						
[*0.9 D – Es*		3.3	1.3		142.2	140.9
D + L – Es]*	10.0		1.0	41.7		175.0
Elevation 117′-6″ to 100′-0″						
[*0.9 D – Es*		4.7	2.8		143.9	184.9
D + L + Es]*	6.4		1.5	172.8		107.9
Elevation 100′-0″ to 82′-6″						
[*0.9 D – Es*		15.4	2.6		90.4	169.8
D + L – Es]*	8.7		2.6	46.6		175.6
Elevation 82′-6″ to 66′-6″						
[*0.9 D – Es*		23.5	1.3		80.9	49.3
D + L – Es]*	0.8		1.3	1.7		74.1

Note:
X is along the horizontal direction, and Y is in the vertical direction.

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Table 3H.5-5
Interior Wall on Column Line 7.3 Details of Wall Reinforcement
(See Figure 3H.5-2 for Locations of Wall Sections.)

Wall Segment (See detail in Subsection 3H.5.1.2.)	**Location**	**Wall Section**	**Reinforcement on Each Face (in²/ft)**	
			Required(1)	**[*Provided (Min.)*]***
From Roof to Elevation 155′-6″	Horizontal	1	3.96	[*4.12*
	Vertical	7	3.60	*3.72*
Elevation 155′-6″ to 135′-3″	Horizontal	2	2.80	*3.12*
	Vertical	8	3.59	*3.72*
Elevation 135′-3″ to 117′-6″	Horizontal	3	2.03	*2.54*
	Vertical	9	2.63	*3.12*
Elevation 117′-6″ to 100′-0″	Horizontal	4	2.29	*2.54*
	Vertical	10	2.98	*3.12*
Elevation 100′-0″ to 82′-6″	Horizontal	5	1.69	*2.54*
	Vertical	11	2.08	*3.12*
Elevation 82′-6″ to 66′-6″	Horizontal	6	0.85	*1.27*
	Vertical	12	0.98	*1.56*
Shear Reinforcement (in²/ft²)				
From Roof to Elevation 155′-6″	Standard hook or T headed bar	7	0.38	*0.44*]*

Note:

1. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-6
Interior Wall at Column Line L Forces and Moments in Critical Locations
(Units: kips, ft)

Load Combination	M_X	M_Y	M_{XY}	T_X	T_Y	T_{XY}
Elevation 154′-2″ to 135′-3″						
[*0.9 D + Es+ Pa + Yj*		6.0	3.5		115.4	170.2
0.9 D + Es+ Pa + Yj]*	14.3		3.5	46.0		170.2
Elevation 135′-3″ to 117′-6″						
[*0.9 D + Es+ Pa + Yj*		145.3	12.2		26.0	38.2
0.9 D + Es+ Pa + Yj]*	24.5		7.1	15.5		114.9

Note:
X is along the horizontal direction, and Y is in the vertical direction.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-7
Interior Wall on Column Line L Details of Wall Reinforcement
(See Figure 3H.5-2, Sheet 3, for Locations of Wall Sections.)

Wall Segment (See detail in Subsection 3H.5.1.3.)	Location	Wall Section	Reinforcement on Each Face (in^2/ft^2) Required(1)	[*Provided (Min.)*]*
Elevation 154′-2″ to 135′-3″	Horizontal	1	2.08	[*2.27*
	Vertical	3	2.59	*3.12*
Elevation 135′-3″ to 117′-6″	Horizontal	2	1.36	*4.39*
	Vertical	4	2.02	*5.66*]*
Shear Reinforcement (in^2/ft^2)				
Elevation 154′-2″ to 135′-3″	Standard hook or T headed bar	5	0.01	[*0.11*
Elevation 135′-3″ to 117′-6″	Standard hook or T headed bar	6	0.33	*2.00*]*

Note:

1. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-8 (Sheet 1 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 20477

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-16.15	-22.92	-28.34	-1.34	-1.06	-0.32	-0.32	
Live (L)	1.46	0.32	-1.57	-0.06	-0.21	0.04	0.03	
Hydro (F)	37.52	12.36	-4.32	-100.50	-14.49	62.14	-9.95	
Seismic (Es)	46.21	56.51	183.20	81.72	28.70	103.00	14.79	
Thermal (To)	-561.80	-267.70	-51.15	-426.90	-145.50	90.32	-23.66	
Thermal (Ta)	-955.80	-444.60	-139.70	-1401.0	-450.00	227.50	-83.16	
LC(1a)	32.40	-14.25	-48.39	-142.68	-22.12	86.61	-14.33	[*1.4D+1.7L+1.4F*
LC(3a)	84.05	51.21	147.24	-60.38	7.15	189.71	0.56	*D+L+F+Es*
LC(3b)	84.05	51.21	-219.16	-223.82	-50.25	-16.29	-29.02	*D+L+F+E's*
LC(3e)	-267.08	-116.11	115.28	-327.19	-83.79	246.16	-14.22	*D+L+F+Es+To*
LC(3f)	-267.08	-116.11	-251.12	-490.63	-141.19	40.16	-43.80	*D+L+F+E's+To*
LC(3m)	84.20	53.18	151.64	-60.18	7.46	189.71	0.57	*0.9D+F+Es*
LC(3n)	84.20	53.18	-214.76	-223.62	-49.94	-16.29	-29.01	*0.9D+F+E's*
LC(3o)	-266.92	-114.13	119.68	-326.99	-83.47	246.16	-14.22	*0.9D+F+Es+To*
LC(3p)	-266.92	-114.13	-246.72	-490.43	-140.87	40.16	-43.80	*0.9D+F+E's+To*
LC(5a)	-574.55	-288.12	-121.54	-977.52	-297.00	204.04	-62.22	*D+L+F+Ta*
LC(5b)	-825.30	-421.18	-153.29	-53.19	-5.28	63.89	-15.73	*D+L+F+Ta*
LC(7a)	-397.01	-211.45	-74.69	-427.19	-125.72	132.70	-28.49	*1.05D+1.3L+1.05F+1.2To*]*

Notes:

x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.42 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for load combination 5 including thermal:	46.33 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combination 5 including thermal:	46.3 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-8 (Sheet 2 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 10529

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-24.40	-96.30	-20.71	-1.16	-2.27	-0.28	-0.34	
Live (L)	-0.44	-2.48	-0.55	-0.01	-0.24	0.01	0.08	
Hydro (F)	9.86	-5.49	6.22	8.37	-73.49	16.94	16.02	
Seismic (Es)	110.80	335.20	95.73	19.03	93.81	22.15	29.34	
Thermal (To)	-215.70	-479.30	-150.10	-99.69	-357.90	16.39	19.34	
Thermal (Ta)	-389.40	-883.60	-273.20	-364.10	-982.20	40.42	17.26	
LC(1a)	-21.10	-146.72	-21.23	10.09	-106.48	23.34	22.09	[*1.4D+1.7L+1.4F*
LC(3a)	99.77	228.74	83.17	29.58	-11.59	45.60	51.51	*D+L+F+Es*
LC(3b)	99.77	228.74	-108.29	-8.48	-199.21	1.30	-7.17	*D+L+F+E's*
LC(3e)	-35.05	-70.83	-10.64	-32.72	-235.28	55.84	63.60	*D+L+F+Es+To*
LC(3f)	-35.05	-70.83	-202.10	-70.78	-422.90	11.54	4.92	*D+L+F+E's+To*
LC(3m)	102.64	240.85	85.80	29.71	-11.12	45.61	51.47	*0.9D+F+Es*
LC(3n)	102.64	240.85	-105.66	-8.35	-198.74	1.31	-7.21	*0.9D+F+E's*
LC(3o)	-32.17	-58.72	-8.02	-32.60	-234.81	55.86	63.55	*0.9D+F+Es+To*
LC(3p)	-32.17	-58.72	-199.48	-70.66	-422.43	11.56	4.87	*0.9D+F+E's+To*
LC(5a)	-258.35	-656.52	-185.79	-220.36	-689.88	41.93	26.55	*D+L+F+Ta*
LC(5b)	-362.67	-963.64	-260.17	7.94	-144.07	12.21	12.80	*D+L+F+Ta*
LC(7a)	-177.61	-469.58	-128.51	-67.20	-348.29	29.80	31.07	*1.05D+1.3L+1.05F+1.2To*]*

Notes:

x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.47 inches (maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for load combination 5 including thermal:	40.3 ksi
[*Yield stress:*	*65.0 ksi (Minimum*]*
Maximum stress intensity range for load combination 5 including thermal:	50.8 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-8 (Sheet 3 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 10544

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-20.03	-75.69	-42.72	3.53	-2.18	-0.01	-1.93	
Live (L)	-0.64	-1.98	-1.22	0.36	-0.06	0.02	-0.07	
Hydro (F)	-4.13	-2.97	-4.10	39.78	3.54	0.99	-4.80	
Seismic (Es)	67.42	185.70	113.20	48.28	7.62	5.78	5.32	
Thermal (To)	-121.60	-387.30	-239.80	75.83	-107.40	39.64	49.91	
Thermal (Ta)	-215.20	-670.10	-416.60	184.20	-269.30	115.50	136.20	
LC(1a)	-34.91	-113.49	-67.62	61.25	1.81	1.40	-9.54	[*1.4D+1.7L+1.4F*
LC(3a)	40.97	103.87	63.52	107.86	10.34	7.18	-3.41	*D+L+F+Es*
LC(3b)	40.97	103.87	-162.88	11.30	-4.90	-4.39	-14.04	*D+L+F+E's*
LC(3e)	-35.03	-138.19	-86.36	155.26	-56.79	31.95	27.79	*D+L+F+Es+To*
LC(3f)	-35.03	-138.19	-312.76	58.70	-72.02	20.39	17.15	*D+L+F+E's+To*
LC(3m)	43.61	113.42	69.01	107.15	10.61	7.16	-3.14	*0.9D+F+Es*
LC(3n)	43.61	113.42	-157.39	10.59	-4.62	-4.41	-13.78	*0.9D+F+E's*
LC(3o)	-32.39	-128.64	-80.87	154.54	-56.51	31.93	28.05	*0.9D+F+Es+To*
LC(3p)	-32.39	-128.64	-307.27	57.98	-71.75	20.37	17.41	*0.9D+F+E's+To*
LC(5a)	-159.30	-499.45	-308.41	158.79	-167.01	73.19	78.32	*D+L+F+Ta*
LC(5b)	-267.05	-805.64	-503.54	51.38	-38.58	1.37	-9.65	*D+L+F+Ta*
LC(7a)	-117.40	-375.64	-230.60	102.82	-79.20	30.78	30.27	*1.05D+1.3L+1.05F+1.2To*]*

Notes:

x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.31 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for load combination 5 including thermal:	46.95 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combination 5 including thermal:	84.9 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-8 (Sheet 4 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 10524

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-35.61	-104.80	0.68	-4.70	7.72	-0.55	-2.22	
Live (L)	-0.45	-2.21	-0.72	-0.25	-0.49	0.00	0.10	
Hydro (F)	11.85	-1.35	4.92	28.52	16.50	3.71	3.79	
Seismic (Es)	76.80	225.60	79.29	53.31	177.00	6.83	55.70	
Thermal (To)	-369.10	-433.40	179.90	-215.40	-109.40	-7.32	-59.63	
Thermal (Ta)	-696.60	-730.00	329.40	-555.10	-487.60	-13.58	-95.78	
LC(1a)	-34.04	-152.37	6.62	32.92	33.09	4.43	2.37	[*1.4D+1.7L+1.4F*
LC(3a)	57.33	116.69	86.14	88.29	207.34	11.48	58.89	*D+L+F+Es*
LC(3b)	57.33	116.69	-72.44	-18.33	-146.66	-2.18	-52.51	*D+L+F+E's*
LC(3e)	-173.36	-154.18	198.57	-46.34	138.96	6.90	21.62	*D+L+F+Es+To*
LC(3f)	-173.36	-154.18	39.99	-152.96	-215.04	-6.76	-89.78	*D+L+F+E's+To*
LC(3m)	61.34	129.38	86.78	89.00	207.05	11.53	59.02	*0.9D+F+Es*
LC(3n)	61.34	129.38	-71.80	-17.62	-146.95	-2.13	-52.38	*0.9D+F+E's*
LC(3o)	-169.35	-141.49	199.22	-45.62	138.68	6.96	21.75	*0.9D+F+Es+To*
LC(3p)	-169.35	-141.49	40.64	-152.24	-215.32	-6.71	-89.65	*0.9D+F+E's+To*
LC(5a)	-459.59	-564.62	210.75	-323.37	-281.01	-5.32	-58.19	*D+L+F+Ta*
LC(5b)	-741.71	-755.24	398.88	19.86	124.99	-105.77	-114.64	*D+L+F+Ta*
LC(7a)	-302.36	-439.4	139.9	136.9	57.2	-2.2	-42.9	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.32 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for load combination 5 including thermal:	42.1 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combination 5 including thermal:	72.5 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-8 (Sheet 5 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 20462

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-7.31	-29.13	-1.51	-1.45	-3.75	-0.06	0.35	
Live (L)	-0.11	-0.55	0.21	-0.14	-0.60	0.00	0.05	
Hydro (F)	5.04	-0.04	-1.61	-16.58	64.59	-1.48	-20.87	
Seismic (Es)	25.64	33.82	32.90	10.45	114.90	2.48	12.55	
Thermal (To)	-286.10	-78.70	66.37	-208.70	-130.00	0.86	-1.51	
Thermal (Ta)	-616.80	-121.80	116.60	-650.20	-502.40	6.16	3.93	
LC(1a)	-3.36	-41.77	-4.01	-25.47	84.16	-2.15	-28.64	[*1.4D+1.7L+1.4F*
LC(3a)	25.28	4.09	29.35	-14.35	200.98	0.35	-16.27	*D+L+F+Es*
LC(3b)	25.28	4.09	-36.45	-35.25	-28.82	-4.61	-41.37	*D+L+F+E's*
LC(3e)	-153.54	-45.10	70.83	-144.78	119.73	0.89	-17.21	*D+L+F+Es+To*
LC(3f)	-153.54	-45.10	5.03	-165.68	-110.07	-4.07	-42.31	*D+L+F+E's+To*
LC(3m)	26.11	7.55	29.29	-14.06	201.95	0.35	-16.35	*0.9D+F+Es*
LC(3n)	26.11	7.55	-36.51	-34.96	-27.85	-4.61	-41.45	*0.9D+F+E's*
LC(3o)	-152.70	-41.63	70.77	-144.50	120.70	0.89	-17.29	*0.9D+F+Es+To*
LC(3p)	-152.70	-41.63	4.97	-165.40	-109.10	-4.07	-42.39	*0.9D+F+E's+To*
LC(5a)	-387.88	-105.84	69.97	-424.54	-253.76	2.31	-18.01	*D+L+F+Ta*
LC(5b)	-646.13	-113.41	80.41	35.38	175.18	-4.36	-31.38	*D+L+F+Ta*
LC(7a)	-217.10	-90.37	46.78	-175.63	-34.40	-0.96	-22.61	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.20 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for load combination 5 including thermal:	20.6 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combination 5 including thermal:	20.6 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-8 (Sheet 6 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 21402

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	-1.82	-17.93	4.00	0.92	0.93	-0.32	0.22	
Live (L)	-0.21	-0.98	0.41	0.19	-0.04	-0.02	-0.03	
Hydro (F)	7.14	0.29	-2.18	104.60	15.51	-16.65	3.08	
Seismic (Es)	36.81	21.41	17.68	139.90	28.75	12.42	12.08	
Thermal (To)	-228.50	-181.90	85.52	-291.30	-212.00	11.34	6.92	
Thermal (Ta)	-379.10	-378.40	159.80	-783.80	-661.10	41.72	28.29	
LC(1a)	7.08	-26.36	3.24	148.06	22.95	-23.80	4.56	[*1.4D+1.7L+1.4F*
LC(3a)	44.77	2.90	19.03	287.45	51.36	-11.24	16.58	*D+L+F+Es*
LC(3b)	44.77	2.90	-16.33	7.65	-6.14	-36.08	-7.58	*D+L+F+E's*
LC(3e)	-98.05	-110.78	72.48	105.39	-81.14	-4.15	20.90	*D+L+F+Es+To*
LC(3f)	-98.05	-110.78	37.12	-174.41	-138.64	-28.99	-3.26	*D+L+F+E's+To*
LC(3m)	45.16	5.68	18.23	287.17	51.31	-11.18	16.59	*0.9D+F+Es*
LC(3n)	45.16	5.68	-17.13	7.37	-6.19	-36.02	-7.57	*0.9D+F+E's*
LC(3o)	-97.65	-108.01	71.68	105.11	-81.19	-4.09	20.91	*0.9D+F+Es+To*
LC(3p)	-97.65	-108.01	36.32	-174.69	-138.69	-28.93	-3.25	*0.9D+F+E's+To*
LC(5a)	-231.84	-255.12	102.10	-384.16	-396.79	9.08	20.95	*D+L+F+Ta*
LC(5b)	-268.90	-468.00	168.35	-17.41	14.23	-18.83	13.88	*D+L+F+Ta*
LC(7a)	-166.1	-156.2	66.6	-107.4	-141.8	-9.3	8.6	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

Plate thickness required for load combinations excluding thermal:	0.28 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for load combination 5 including thermal:	25.1 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combination 5 including thermal:	31.3 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-8 (Sheet 7 of 7)
Design Summary of Spent Fuel Pool Wall Design Loads, Load Combinations, and Comparisons to Acceptance Criteria – Element No. 21414

Load/ Combination	S_{xx} kip/ft	S_{yy} kip/ft	S_{xy} kip/ft	M_{xx} k-ft/ft	M_{yy} k-ft/ft	N_x kip/ft	N_y kip/ft	Comments
Dead (D)	0.69	-10.62	-2.57	-0.52	-0.22	-0.03	0.12	
Live (L)	0.18	0.12	-0.45	0.00	-0.11	-0.01	0.02	
Hydro (F)	4.25	0.56	-2.73	-27.01	-31.06	-1.46	1.82	
Seismic (Es)	26.90	13.88	36.68	26.35	21.70	2.17	4.34	
Thermal (To)	-79.35	-40.69	49.04	-129.00	-119.30	10.01	6.90	
Thermal (Ta)	-129.60	-66.37	57.50	-374.60	-374.70	26.38	24.34	
LC(1a)	7.24	-13.89	-8.19	-38.54	-43.97	-2.09	2.75	[*1.4D+1.7L+1.4F*
LC(3a)	33.73	4.16	29.84	-11.98	-22.11	0.10	7.03	*D+L+F+Es*
LC(3b)	33.73	4.16	-43.52	-64.68	-65.51	-4.24	-1.66	*D+L+F+E's*
LC(3e)	-15.86	-21.27	60.49	-92.61	-96.67	6.36	11.34	*D+L+F+Es+To*
LC(3f)	-15.86	-21.27	-12.87	-145.31	-140.07	2.01	2.66	*D+L+F+E's+To*
LC(3m)	33.48	5.10	30.55	-11.93	-21.98	0.11	7.00	*0.9D+F+Es*
LC(3n)	33.48	5.10	-42.81	-64.63	-65.38	-4.23	-1.69	*0.9D+F+E's*
LC(3o)	-16.12	-20.33	61.20	-92.56	-96.54	6.37	11.31	*0.9D+F+Es+To*
LC(3p)	-16.12	-20.33	-12.16	-145.26	-139.94	2.02	2.62	*0.9D+F+E's+To*
LC(5a)	-75.87	-51.43	30.19	-261.65	-265.57	15.00	17.17	*D+L+F+Ta*
LC(5b)	-114.31	-96.07	55.47	-35.06	-36.08	2.55	-1.61	*D+L+F+Ta*
LC(7a)	-54.08	-40.93	30.63	-125.65	-122.46	5.94	7.24	*1.05D+1.3L+1.05F+1.2To*]*

Notes:
x – direction is horizontal; y – direction is vertical.
See Figure 3H.5-10 for element location.

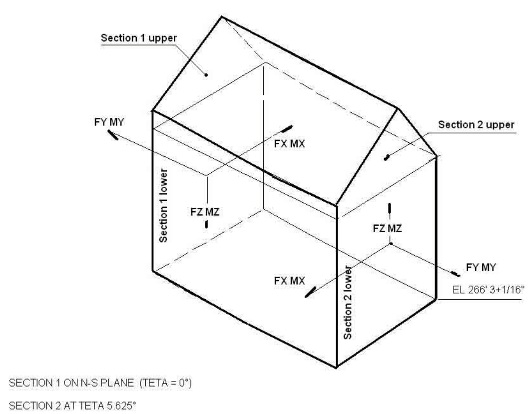
Plate thickness required for load combinations excluding thermal:	0.14 inches (Maximum)
[*Plate thickness provided:*	*0.50 -0.01 +0.10 inches*]*
Maximum principal stress for load combination 5 including thermal:	22.1 ksi
[*Yield stress:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combination 5 including thermal:	22.1 ksi
Allowable stress intensity:	130.0 ksi (Minimum)

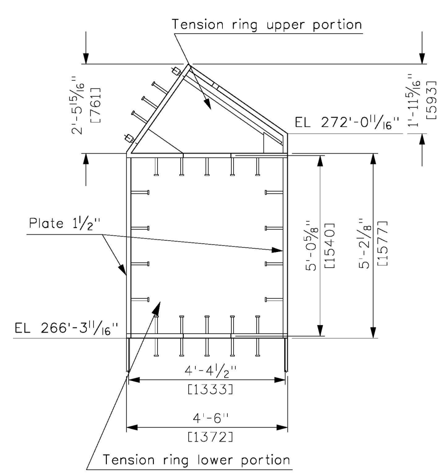
*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-9 (Sheet 1 of 3)
Shield Building Roof Reinforcement Summary
(Tension Ring)

Tension Ring – Axial Force and Bending Verification								
Location		**Seismic Maximum Stresses**		**Maximum Stresses ksi**	**F_y ksi**	**Maximum Steel Area Required(2) (in^2/ft)**	[***Steel Area Provided***]*	[***Design Limit(1) for Ratio Max Required/ Provided***]*
Section	**Angles**	**Seismic L/C**	**f_a ksi**					
2 lower	5.625°	9	14.31	14.31	50	7.74	[*Liner 1 1/2" = 18 (in^2/ft) (Min)*]*	[*0.43 + 2%*]*
	84.375°	17	12.52					
1 lower	0°	9	12.97					
	90°	17	11.39					

Tension Ring – Shear Force and Torsion Verification								
Location		**Seismic Maximum Stresses**		**Maximum Stresses ksi**	**F_y ksi**	**Maximum Steel Area Required(2) (in^2/ft)**	[***Steel Area Provided***]*	[***Design Limit for Ratio Max Required/ Provided***]*
Section	**Angles**	**Seismic L/C**	**f_v ksi**					
2 lower	5.625°	17	4.83	5.52	50	5.04	[*Liner 1 1/2" = 18 (in^2/ft) (Min.)*]*	[*0.28 + 2%*]*
	84.375°	9	5.52					
1 lower	0°	18	3.20					
	90°	11	4.00					





Notes:
1. [*Two percent of the value may be added to the design limit as an allowance for minor variances in analysis results.*]*
2. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-9 (Sheet 2a of 3)
Shield Building Roof Reinforcement Summary
(Air Inlet)

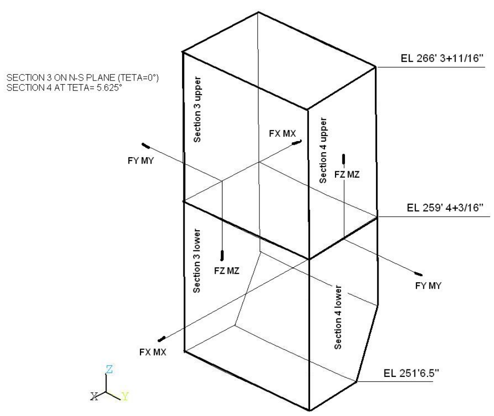
AIS Reinforcement Summary – Horizontal Sections						
Locations (Figure 3H.5-11)		Steel Area (Vertical Direction – Z Local Dir.)				
		Required - Seismic Load Combinations (in^2/ft)		Maximum Required[2] (in^2/ft)	[*Provided*]*	[*Design Limit*[1] *for Ratio Max Required/ Provided*]*
Sections	**Angles**	**Seismic L/C**	**Values**			
5+6	0°-5.625°	16	1.65	2.10	[*Liner 1" = 12 (in*^{2}*/ft) (Min.)*]*	[*0.175 + 2%*]*
	84.375°-90°	8	1.41			
8	0°-5.625°	16	2.10			
	84.375°-90°	8	1.69			
9	0°-5.625°	16	2.10			
	84.375°-90°	8	1.68			
11	0°-5.625°	16	1.61	1.61	[*Liner 3/4" = 9 (in*^{2}*/ft) (Min.)*]*	[*0.18 + 2%*]*
	84.375°-90°	24	1.21			

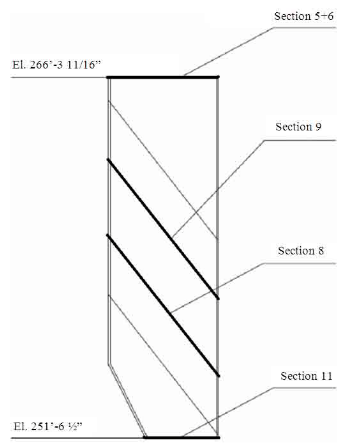
Notes:
1. [*Two percent of the value may be added to the design limit as an allowance for minor variances in analysis results*.]*
2. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-9 (Sheet 2b of 3)
Shield Building Roof Reinforcement Summary
(Air Inlet)

AIS Reinforcement Summary – Vertical Sections						
Locations (Figure 3H.5-11)		**Steel Area (Hoop Direction – Y Local Dir.)**				
		Required - Seismic Load Combinations (in^2/ft)		**Maximum Required(2) (in^2/ft)**		[***Design Limit(1) for Ratio Max Required/ Provided***]*
Sections	**Angles**	**Seismic L/C**	**Values**		[***Provided***]*	
3 Upper	0°	9	9.56	10.04	[*Liner 1" = 12 (in^2/ft) (Min.)*]*	[*0.84 + 2%*]*
	90°	17	8.32			
3 Lower	0°	9	8.14			
	90°	18	7.03			
4 Upper	5.625°	9	10.04			
	84.375°	17	8.69			
4 Lower	5.625°	9	7.98			
	84.375°	19	6.82			





Notes:
1. [*Two percent of the value may be added to the design limit as an allowance for minor variances in analysis results.*]*
2. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-9 (Sheet 2c of 3)
Shield Building Roof Reinforcement Summary
(Air Inlet)

Out of Plane Shear Reinforcement Summary – AIS							
Locations (Figure 3H.5-11)		**Required – Seismic Load Combinations (in^2/ft)**			**Maximum Required(2) (in^2/ft)**	[***Steel Area Provided***]*	[***Design Limit(1) for Ratio Max Required/ Provided***]*
Angles	**Sections**	**Seismic L/C**	**Values**	**Sum**			
0° - 5.625°	Max of Vertical Sections 3 upper - 4 upper	1	0.10	0.10	0.34	[*3 #6 TIE BAR @2.8125° (41.36") (8 1/2" in vertical direction) = 0.54 (in^2/ft) (Min.)*]*	[*0.63 + 2%*]*
	Horizontal Section 5+6		0.00				
84.375° - 90°	Max of Vertical sections 3 upper - 4 upper	1	0.10	0.10			
	Horizontal Section 5+6		0.00				
0° - 5.625°	Max of Vertical Sections 3 upper – 4 upper	9	0.10	0.34			
	Horizontal Section 8		0.24				
84.375° - 90°	Max of Vertical Sections 3 upper – 4 upper	1	0.10	0.30			
	Horizontal Section 8		0.20				
0° - 5.625°	Max of Vertical Sections 3 lower - 4 lower	0	0.093	0.22			
	Horizontal Section 9		0.127				
84.375° - 90°	Max of Vertical Sections 3 lower - 4 lower	0	0.183	0.18			
	Horizontal Section 9		0.000				
0° - 5.625°	Max of Vertical Sections 3 lower - 4 lower	1	0.167	0.17			
	Horizontal Section 11		0.000				
84.375° - 90°	Max of Vertical Sections 3 lower - 4 lower	0	0.02	0.02			
	Horizontal Section 11		0.00				

Notes:
1. [*Two percent of the value may be added to the design limit as an allowance for minor variances in analysis results.*]*
2. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-9 (Sheet 3 of 3)
Shield Building Roof Reinforcement Summary
(Exterior Wall Of Passive Containment Cooling System Tank)

Wall Segment	Location (Figure 3H.5-11 Sheet 5 of 6)	Reinforcement on Each Face, in^2/ft Maximum Required	Provided (Minimum)		Ratio Required/ Provided
Bottom	Vertical	1.37	1#11@1.2°	[*1.72*	0.80
	Hoop	0.67	1#9@6"	*2*	0.33
	Shear	0.07	1#6@1.2°x12"	*0.48*	0.15
Mid-height	Vertical	0.64	1#11@1.2°	*1.72*	0.37
	Hoop	1.85	1#9@6"	*2*	0.92
Top	Vertical	0.52	1#11@1.2°	*1.72*	0.30
	Hoop	0.79	1#9@6"	*2*]*	0.39

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-10
Design Summary Of Roof At Elevation 180′-0″, Area 6
(Near Shield Building Interface)

Governing Load Combination (Roof Girder)	
Combination Number	3 – Extreme Environmental Condition Downward Seismic Acceleration
Bending Moment	= 7125 kips-ft
Corresponding Stress	= 24.1 ksi
Allowable Stress	= 38.0 ksi
Shear Force	= 447 kips
Corresponding Stress	= 17.0 ksi
Allowable Stress	= 20.1 ksi
Governing Load Combination (Concrete Slab)	
Parallel to Girders	
Combination Numbers	3 – Extreme Environmental Condition
Reinforcement (Each Face)	
Required(1)	= 1.74 in^2/ft
[*Provided*	*= 2.54* in^2*/ft (Minimum)*]*
Perpendicular to Girders	
Combination Numbers	3 – Extreme Environmental Condition
Reinforcement (Each Face)	
Required(1)	= 1.68 in^2/ft
[*Provided*	*= 3.12* in^2*/ft (Minimum)*]*

Note:

1. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-11
Design Summary Of Floor At Elevation 135′-3″ Area 1 (Between Column Lines M And P)

Governing Load Combination (Steel Beam)	
Load Combination	3 – Extreme Environmental Condition Downward Seismic
Bending Moment	=(-) 63.9 kips-ft
Corresponding Stress	= 17.0 ksi
Allowable Stress	=33.26 ksi
Shear Force	= 30.7 kips
Corresponding Stress	= 8.7 ksi
Allowable Stress	= 20.1 ksi
Governing Load Combination (Concrete Slab)	
Parallel to the Beams	
Load Combination	3 – Extreme Environmental Condition Downward Seismic
Bending Moment	=(-) 16.0 kips-ft/ft
In-plane Shear	=20.0 kips (per foot width of the slab)
Reinforcement (Each Face)	
Required(1)	= 0.41 in^2/ft
[*Provided*	*= 0.44 in^2/ft (Min.)*]*
Perpendicular to the Beams	
Combination Number	Normal Condition
Bending Moment	=(+) 6.66 kips-ft (per foot width of the slab)
Reinforcement (Each Face)	
Required(1)	= 0.28 in^2/ft
[*Provided*	*= 0.60 in^2/ft (Min.)*]*

Note:
1. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-12
Design Summary Of Floor At Elevation 135′-3″
(Operations Work Area (Previously Known As 'Tagging Room') Ceiling)

Design of Precast Concrete Panels	
Governing Load Combination	Construction
Design Bending Moment (Midspan)	= 14.53 kip-ft/ft
Bottom Reinforcement (E/W Direction)	
Required(1)	= 0.58 in^2/ft
[*Provided*	*= 0.79 in^2/ft (Min.)*]*
Top Reinforcement (E/W Direction)	
Required(1)	= (Minimum required by Code)
[*Provided*	*= 0.20 in^2/ft (Min.)*]*
Top and Bottom Reinforcement (N/S Direction)	
Required(1)	= (Minimum required by Code)
[*Provided*	*= 0.20 in^2/ft (Min.)*]*
Design of 24-inch-Thick Slab	
Governing Load Combination	Extreme Environmental Condition (SSE)
Design Bending Moment (E/W Direction) Midspan	= 14.40 kips ft/ft
Design In-plane Shear	= 31.9 kips ft
Design In-plane Tension	= 21.9 kips ft
Bottom Reinforcement (E/W Direction)	
Required(1)	= 0.53 in^2/ft
[*Provided*	*= 0.79 in^2/ft (Min.)*]*
Design Bending Moment (E/W Direction) at Support	= 28.81 kips-ft/ft
Design In-plane Shear	= 31.9 kips/ft
Design In-plane Tension	= 21.9 kips/ft
Top Reinforcement (E/W Direction)	
Required(1)	= 0.93 in^2/ft
[*Provided*	*= 1.00 in^2/ft (Min.)*]*
Design Bending Moment (N/S Direction)	= 8.47 kips ft/ft
Design In-plane Shear	= 31.9 kips/ft
Design In-plane Tension	= 27.2 kip/ ft
Top and Bottom Reinforcement (N/S Direction)	
Required(1)	= 0.59 in^2/ft
[*Provided*	*= 0.79 in^2/ft (Min.)*]*

Note:
1. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

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Table 3H.5-13
Design Summary Of Floor At Elevation 135′-3″ Area 1 (Main Control Room Ceiling)

The design of the bottom plate with fins is governed by the construction load.
For the composite floor, the design forces used for the evaluation of a typical 9-inch-wide strip of the slab are as follows: Maximum bending moment=+35.0 (-24.4) kips-ft Maximum shear force=22.3 kips
The design evaluation results are summarized below: (1) • [*The actual area of the tension steel is 9.0 in^2 (Min.),*]* which provides a design strength of 518.5 kips-ft bending moment capacity. • [*The design shear strength is 23.22 kips.* • *The shear studs are spaced a maximum of 9 inches c/c, in both directions.*]* The calculated required spacing is 9.06 inches.

Note:

1. Thermal loads have been considered in the design of critical sections. The required reinforcement values shown do not include the load case where seismic and normal thermal loads are numerically combined as the normal thermal loads were assessed to be insignificant. When the seismic and normal thermal loads are numerically combined, the value of required reinforcement may increase; however, in all cases the required reinforcement is less than the provided reinforcement and thus the design of the critical section reinforcement is acceptable.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-14 (Sheet 1 of 3)
Design Summary of Enhanced Shield Building Cylindrical Wall Load Combinations, and Comparison to Acceptance Criteria Elevation 180 Feet Near Fuel Handling Building Roof

Load/Combination	TX	TY	TXY	MX	MY	MXY	NX	NY	Comments
	kip/ft	kip/ft	kip/ft	k-ft/ft	k-ft/ft	k-ft/ft	kip/ft	kip/ft	
Dead	-6	-118	15	-25	-17	4	-6	-5	
Live	1	-1	0	0	0	0	0	0	
Seismic	155	385	163	299	209	35	71	33	
1	-7	-167	22	-35	-24	5	-8	-7	1.4 D + 1.7 L
2	150	266	179	274	192	38	65	28	D + L + Es
3	150	266	-147	-324	-226	-31	-76	-38	D + L + E's
4	-160	-504	-147	-324	-226	-31	-76	-38	D + L - Es
5	-160	-504	179	274	192	38	65	28	D + L - E's
6	150	278	177	277	193	38	66	28	0.9 D + Es
7	150	278	-149	-322	-224	-31	-76	-37	0.9 D + E's
35(2)	211	369	229	453	294	64	105	33	0.9 D + E's + αTo(W1)
37(2)	226	357	234	463	302	64	108	33	0.9 D + E's + αTo(W2)

x-direction is horizontal; y-direction is vertical.
Element number: 12164
[*Plate thickness required for load combinations excluding thermal:* *0.43 inches + 2%(1)*]*
[*Plate thickness required for load combinations including thermal:* *0.57 inches + 2%(1)*]*
[*Plate thickness provided:* *0.75 inches*]*
[*Shear reinforcement required for load combinations excluding thermal:* *0.64* in^2/ft^2 *+ 2%(1)*]*
[*Shear reinforcement required for load combinations including thermal:* *0.93* in^2/ft^2 *+ 2%(1)*]*
Shear reinforcement provided: See [*APP-GW-GLR-602, Section 4.*]*

Notes:
1. [*The Tier 2* designation for "Plate thickness required" requires NRC approval if this value is exceeded as a result of design changes or detail design adjustments identified during preparation of fabrication or construction drawings or instructions.*]*
2. Load cases 35 and 37 are the two governing load combinations for element 12164 that include thermal and seismic loads combined numerically. W1 designates the winter conditions with the spent fuel pool at the normal operating temperature limit. W2 designates the winter conditions with the spent fuel pool and fuel transfer canal at the normal operating temperature limit. Es is SRSS (member forces are positive) of the SSE loads. E's is Es with all member forces except axial forces (TX, TY) reversed to negative.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-14 (Sheet 2 of 3)
Design Summary of Enhanced Shield Building Load Combinations, and Comparison to Acceptance Criteria Elevation 175 Feet Near Intersection With Wall 7.3

Load/Combination	TX kip/ft	TY kip/ft	TXY kip/ft	MX k-ft/ft	MY k-ft/ft	MXY k-ft/ft	NX kip/ft	NY kip/ft	Comments
Dead	-6	-105	12	-6	5	1	0	2	
Live	0	-1	0	0	0	0	0	0	
Seismic	34	325	176	38	25	13	2	8	
1	-9	-149	17	-9	7	1	0	3	1.4 D + 1.7 L
2	28	219	188	31	30	14	2	10	D + L + Es
3	28	219	-164	-44	-20	-12	-3	-6	D + L + E's
4	-40	-431	-164	-44	-20	-12	-3	-6	D + L - Es
5	-40	-431	188	31	30	14	2	10	D + L - E's
6	28	230	187	32	29	14	2	10	0.9 D + Es
7	28	230	-166	-44	-20	-12	-3	-7	0.9 D + E's
19(2)	77	227	186	36	-58	7	3	11	D + L + E's + αTo(W1)
37(2)	77	238	186	36	-58	7	3	11	0.9 D + E's + αTo(W2)

x-direction is horizontal; y-direction is vertical.

Element number:	11514
[*Plate thickness required for load combinations excluding thermal:*	*0.40 inches + 2%(1)*]*
[*Plate thickness required for load combinations including thermal:*	*0.40 inches + 2%(1)*]*
[*Plate thickness provided:*	*0.75 inches*]*
[*Shear reinforcement required for load combinations excluding thermal:*	*0.07 in^2/ft^2 + 2%(1)*]*
[*Shear reinforcement required for load combinations including thermal:*	*0.08 in^2/ft^2 + 2%(1)*]*
Shear reinforcement provided:	See [*APP-GW-GLR-602, Section 4.*]*

Notes:

1. [*The Tier 2* designation for "Plate thickness required" requires NRC approval if this value is exceeded as a result of design changes or detail design adjustments identified during preparation of fabrication or construction drawings or instructions.*]*
2. Load cases 19 and 37 are the two governing load combinations for element 11514 that include thermal and seismic loads combined numerically. W1 designates the winter conditions with the spent fuel pool at the normal operating temperature limit. W2 designates the winter conditions with the spent fuel pool and fuel transfer canal at the normal operating temperature limit. Es is SRSS (member forces are positive) of the SSE loads. E's is Es with all member forces except axial forces (TX, TY) reversed to negative.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-14 (Sheet 3 of 3)
Design Summary of Enhanced Shield Building Load Combinations, and Comparison to Acceptance Criteria Elevation Grade on West Side

Load/Combination	TX kip/ft	TY kip/ft	TXY kip/ft	MX k-ft/ft	MY k-ft/ft	MXY k-ft/ft	NX kip/ft	NY kip/ft	Comments
Dead	2	-127	0	2	19	0	0	-2	
Live	0	1	0	0	0	0	0	0	
Seismic	58	477	231	2	16	17	4	7	
1	2	-176	-1	3	26	0	0	-3	1.4 D + 1.7 L
2	60	352	231	4	35	18	4	5	D + L + Es
3	60	352	-232	1	2	-17	-4	-9	D + L + E's
4	-57	-603	-232	1	2	-17	-4	-9	D + L - Es
5	-57	-603	231	4	35	18	4	5	D + L - E's
6	60	364	231	4	33	18	4	5	0.9 D + Es
7	60	364	-232	0	1	-17	-4	-9	0.9 D + E's
23(2)	182	364	-238	113	155	-17	-4	-31	D + L + E's + αTo(W1)
41(2)	182	380	-238	113	153	-17	-4	-31	0.9 D + E's + αTo(W2)

x-direction is horizontal; y-direction is vertical.

Element number:	23752
[*Plate thickness required for load combinations excluding thermal:*	*0.56 inches + 2%(1)*]*
[*Plate thickness required for load combinations including thermal:*	*0.58 inches + 2%(1)*]*
[*Plate thickness provided:*	*0.75 inches*]*
[*Shear reinforcement required for load combinations excluding thermal:*	*0.06 in^2/ft^2 + 2%(1)*]*
[*Shear reinforcement required for load combinations including thermal:*	*0.21 in^2/ft^2 + 2%(1)*]*
Shear reinforcement provided:	See [*APP-GW-GLR-602, Section 4.*]*

Notes:

1. [*The Tier 2* designation for “Plate thickness required” requires NRC approval if this value is exceeded as a result of design changes or detail design adjustments identified during preparation of fabrication or construction drawings or instructions.*]*
2. Load cases 23 and 41 are the two governing load combinations for element 23752 that include thermal and seismic loads combined numerically. W1 designates the winter conditions with the spent fuel pool at the normal operating temperature limit. W2 designates the winter conditions with the spent fuel pool and fuel transfer canal at the normal operating temperature limit. Es is SRSS (member forces are positive) of the SSE loads. E's is Es with all member forces except axial forces (TX, TY) reversed to negative.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3H.5-15
Shield Building Roof Reinforcement Ratio of Code Required Versus Provided

Critical Sections	Stress Component	Required in^2/ft	Provided (Minimum) in^2/ft	Reinforcement Ratio
[*Conical Roof Steel Beams*]* (1)	Axial + Bending	-	[*Radial Beams W36 X 393*]*	1.33
	Shear	-		8.33
[*Conical Roof Near Tension Ring*]*	Radial	1.80	[*1.96*]*	1.09
	Hoop	4.31	[*4.68*]*	1.09
[*Knuckle Region*]*	Vertical	1.37	[*1.72*]*	1.25
	Radial	1.52	[*2.23*]*	1.47
	Hoop	1.37	[*3.12*]*	2.28
[*Compression Ring*]*	Vertical	1.04	[*1.48*]*	1.42
	Radial	3.09	[*4.42*]*	1.43
	Hoop	2.14	[*3.12*]*	1.45

Note:

1. Steel beams are not considered as reinforcement for the reinforced concrete roof. Ratio for conical roof steel beams is based on demand and allowable stresses in psi.

*NRC Staff approval is required prior to implementing a change in this information.

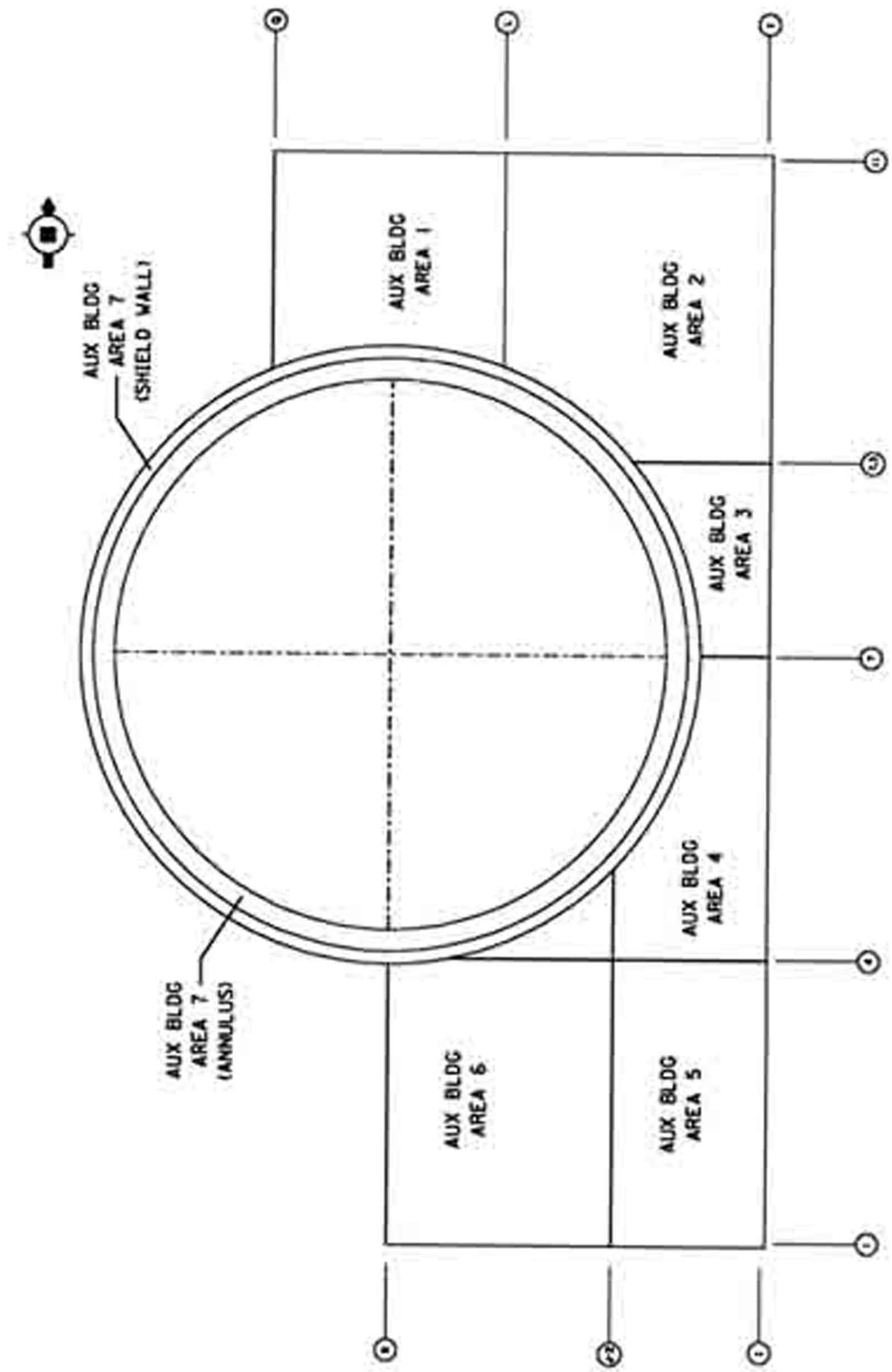


Figure 3H.2-1
[*General Layout of Auxiliary Building*]*

*NRC Staff approval is required prior to implementing a change in this information.

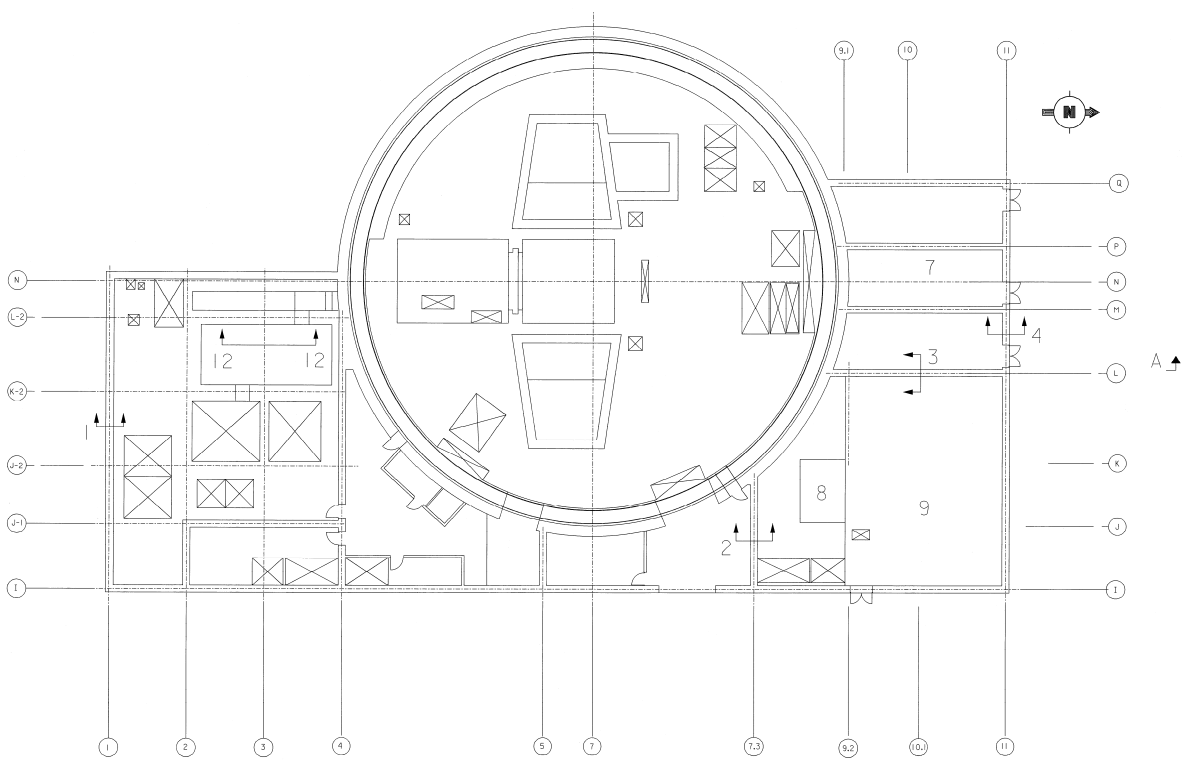


Figure 3H.5-1 (Sheet 1 of 3)
[*Nuclear Island Critical Sections Plan at El. 135'-3"*]*

*NRC Staff approval is required prior to implementing a change in this information.

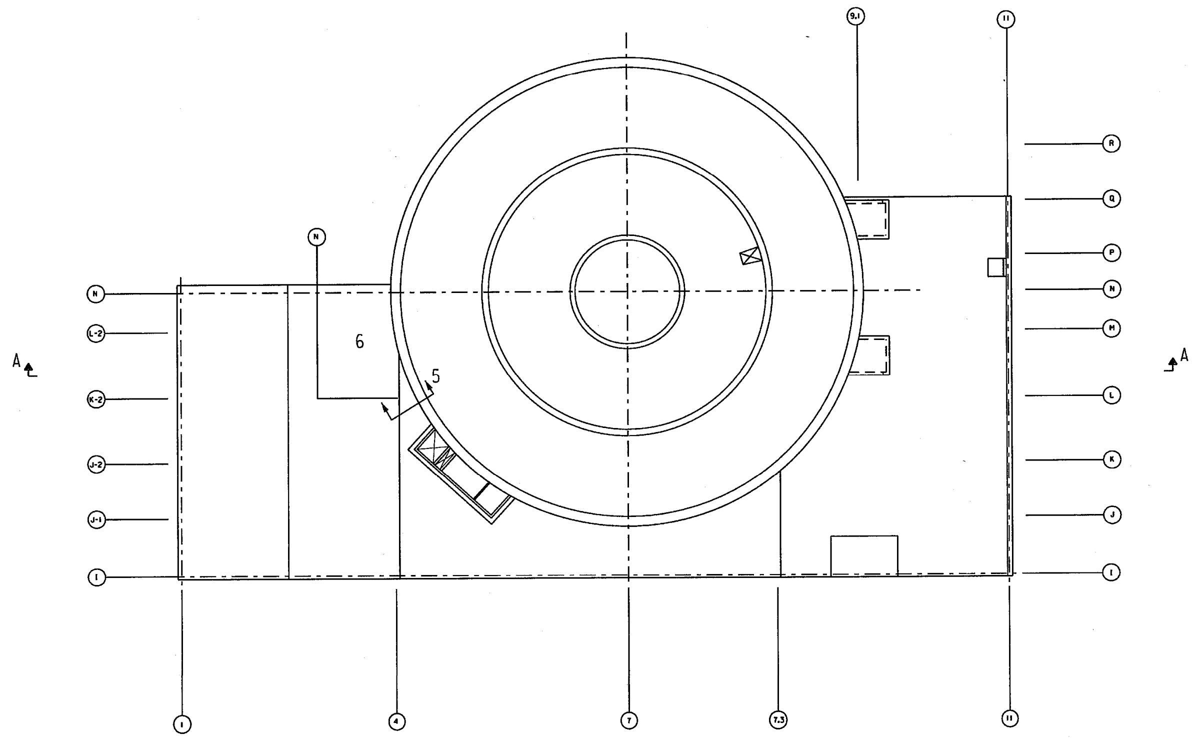


Figure 3H.5-1 (Sheet 2 of 3)
[*Nuclear Island Critical Sections Plan at El. 180'-0"*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withhold Under 10 CFR 2.390d

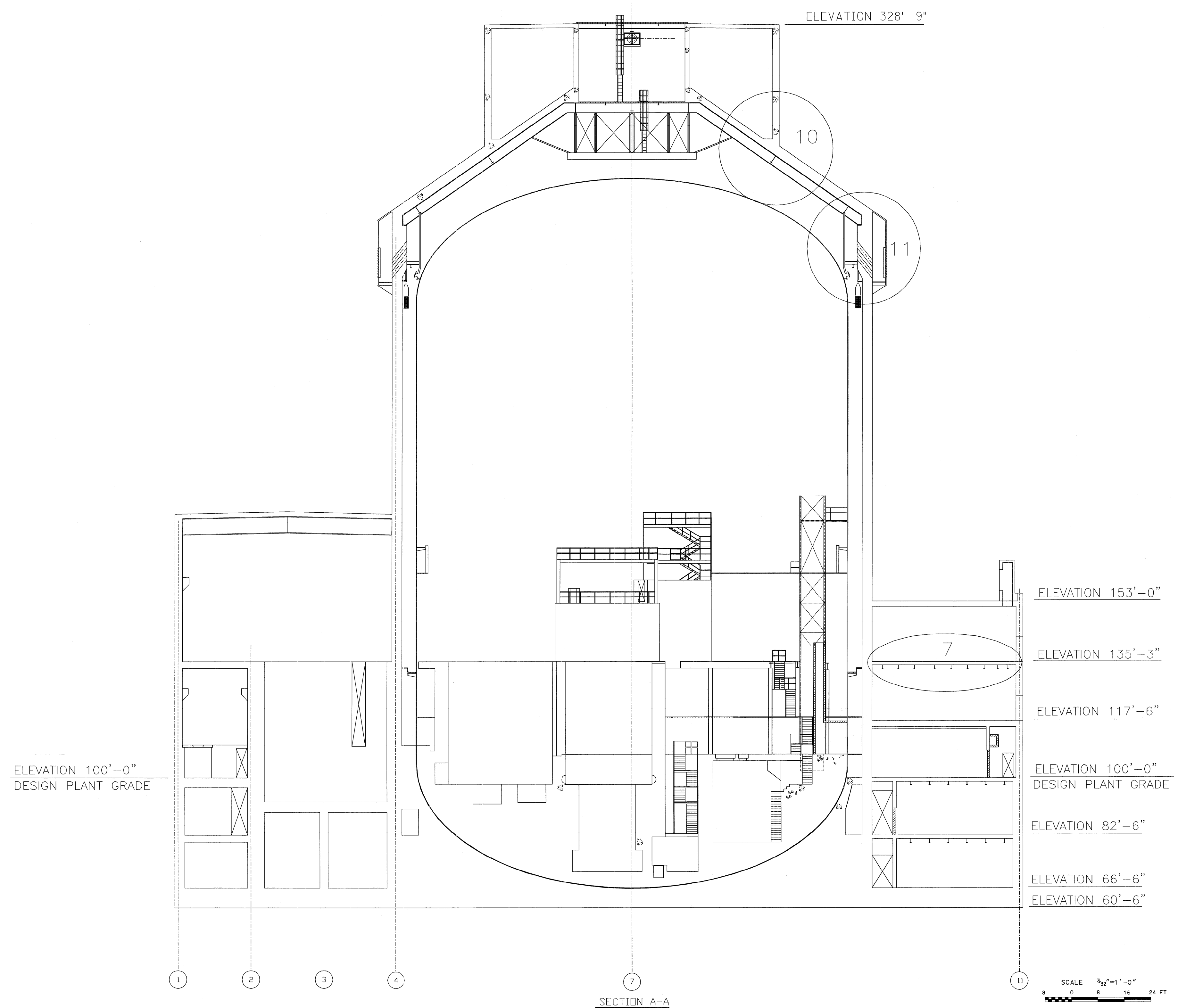


Figure 3H.5-1 (Sheet 3 of 3)
[*Nuclear Island Critical Sections Section A-A*]*

*NRC Staff approval is required prior to implementing a change in this information.

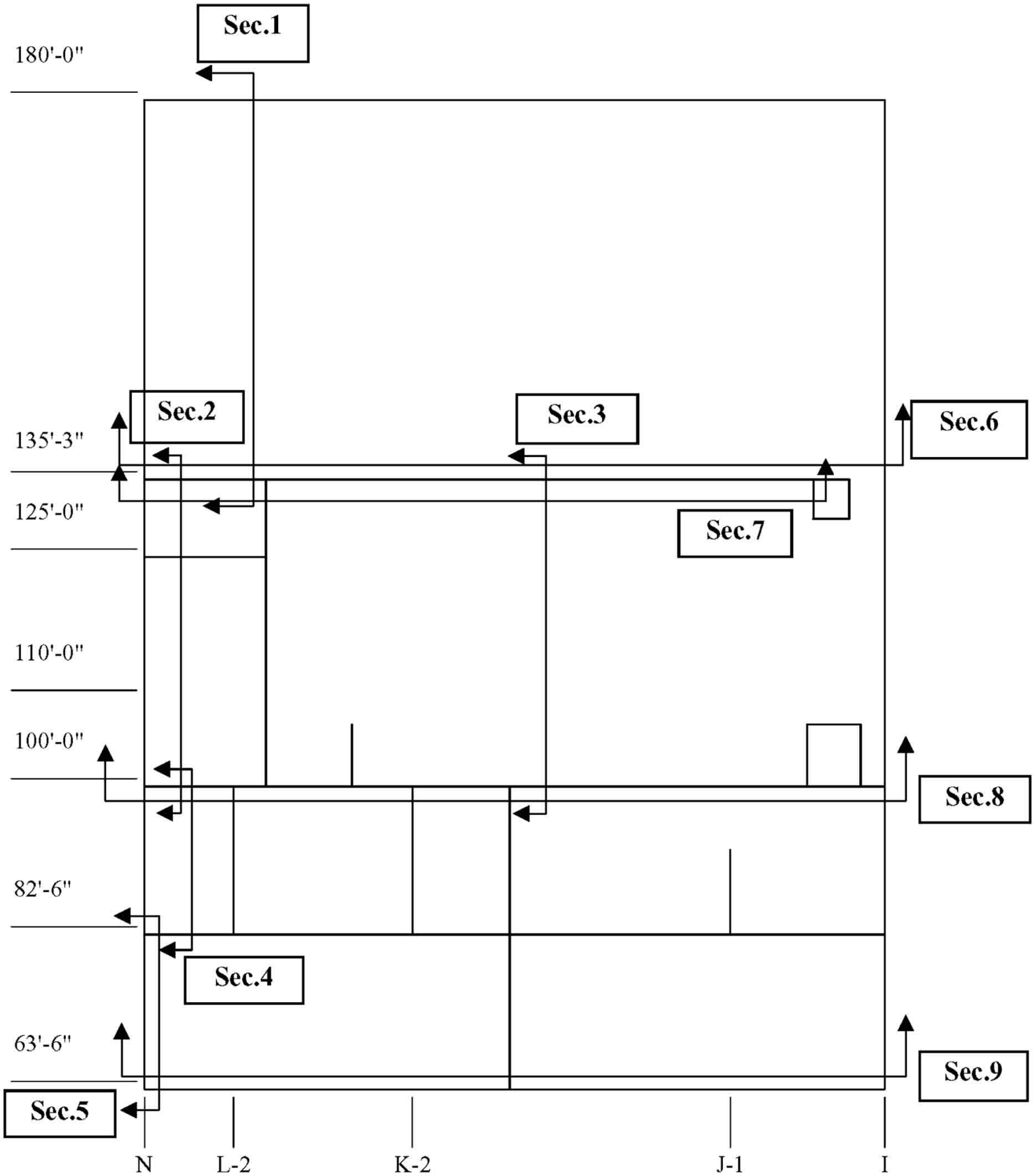


Figure 3H.5-2 (Sheet 1 of 3)
[*Wall on Column Line 1*]*

*NRC Staff approval is required prior to implementing a change in this information.

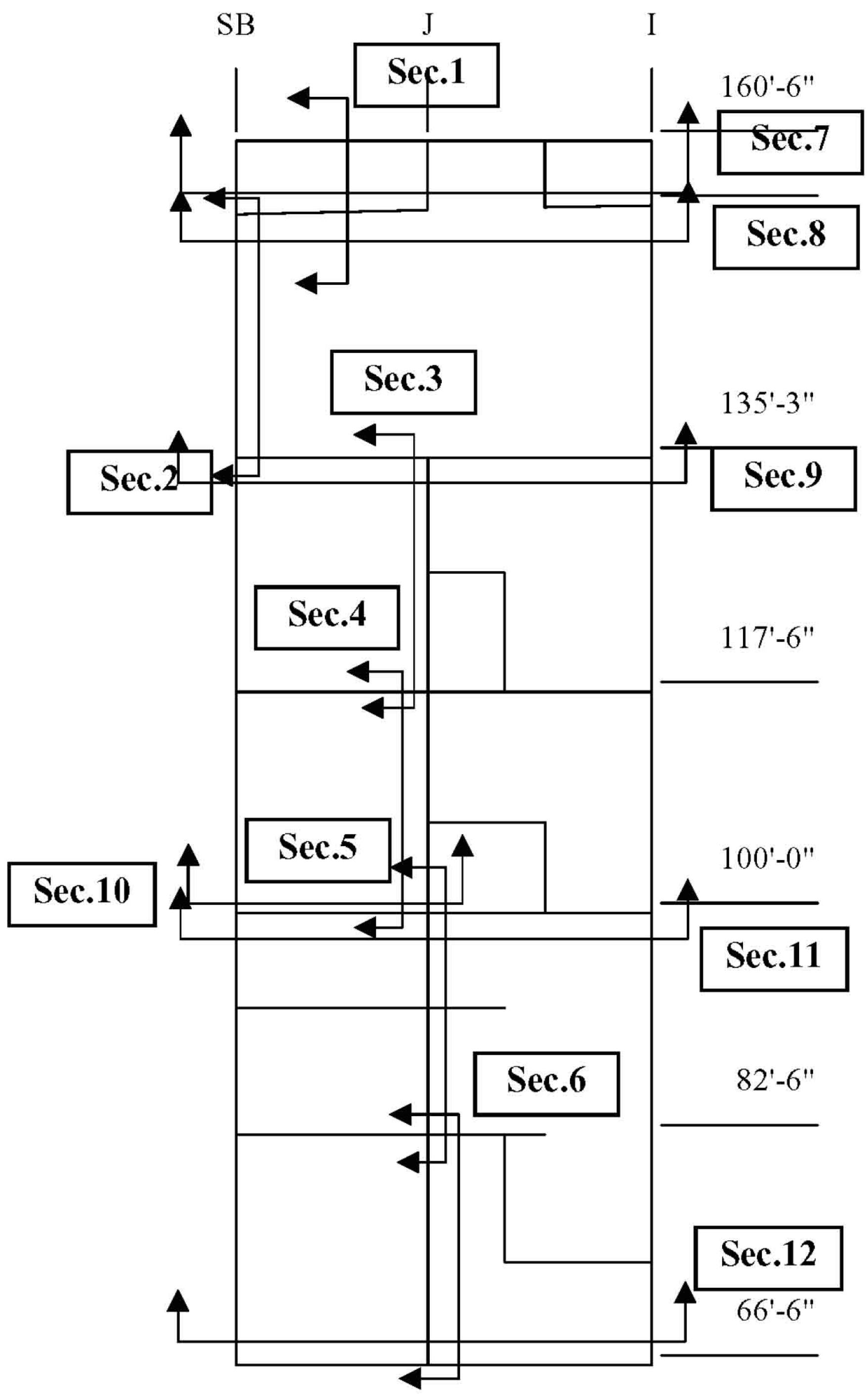


Figure 3H.5-2 (Sheet 2 of 3)
[*Wall on Column Line 7.3*]*

*NRC Staff approval is required prior to implementing a change in this information.

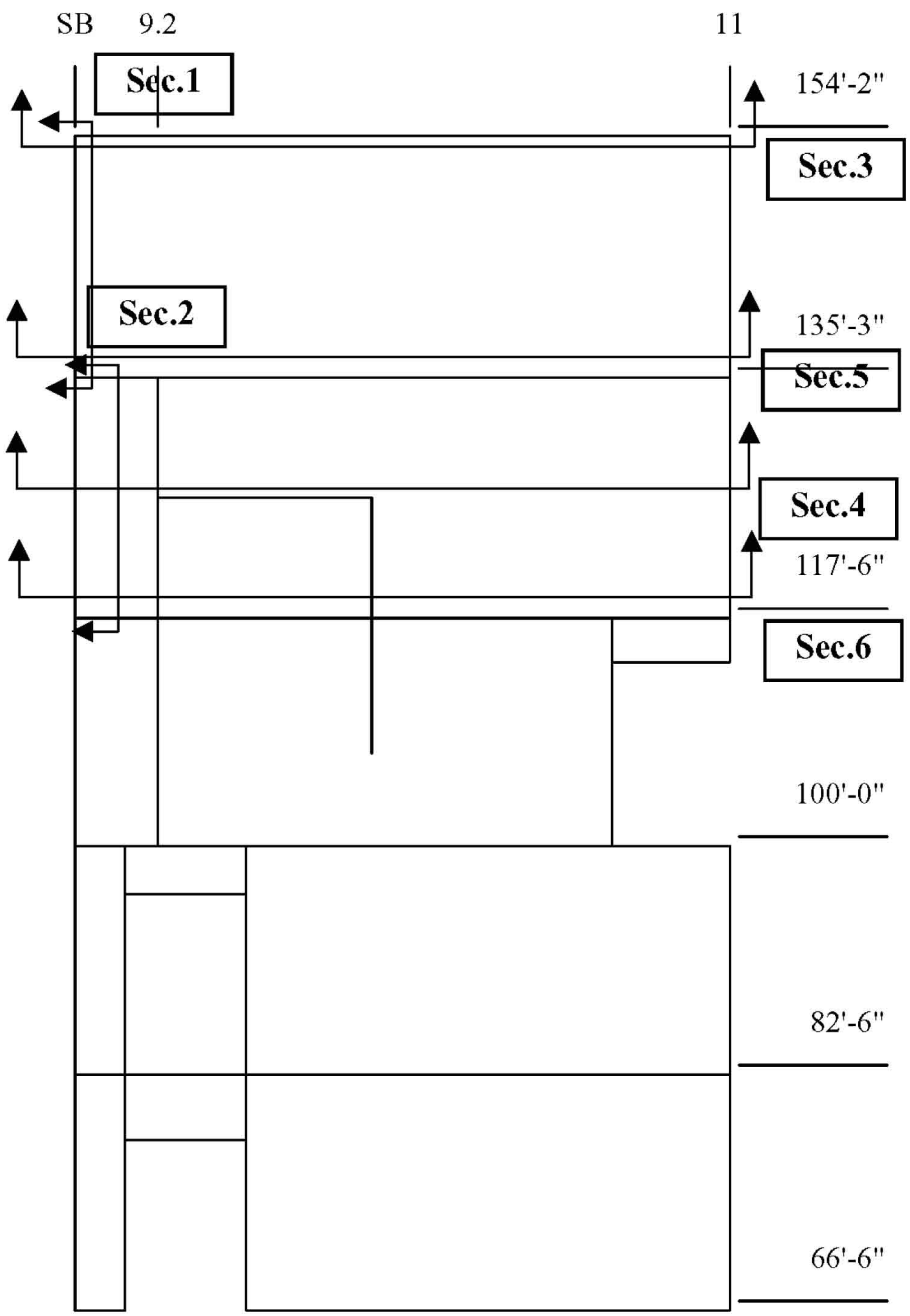


Figure 3H.5-2 (Sheet 3 of 3)
[*Wall on Column Line L*]*

*NRC Staff approval is required prior to implementing a change in this information.

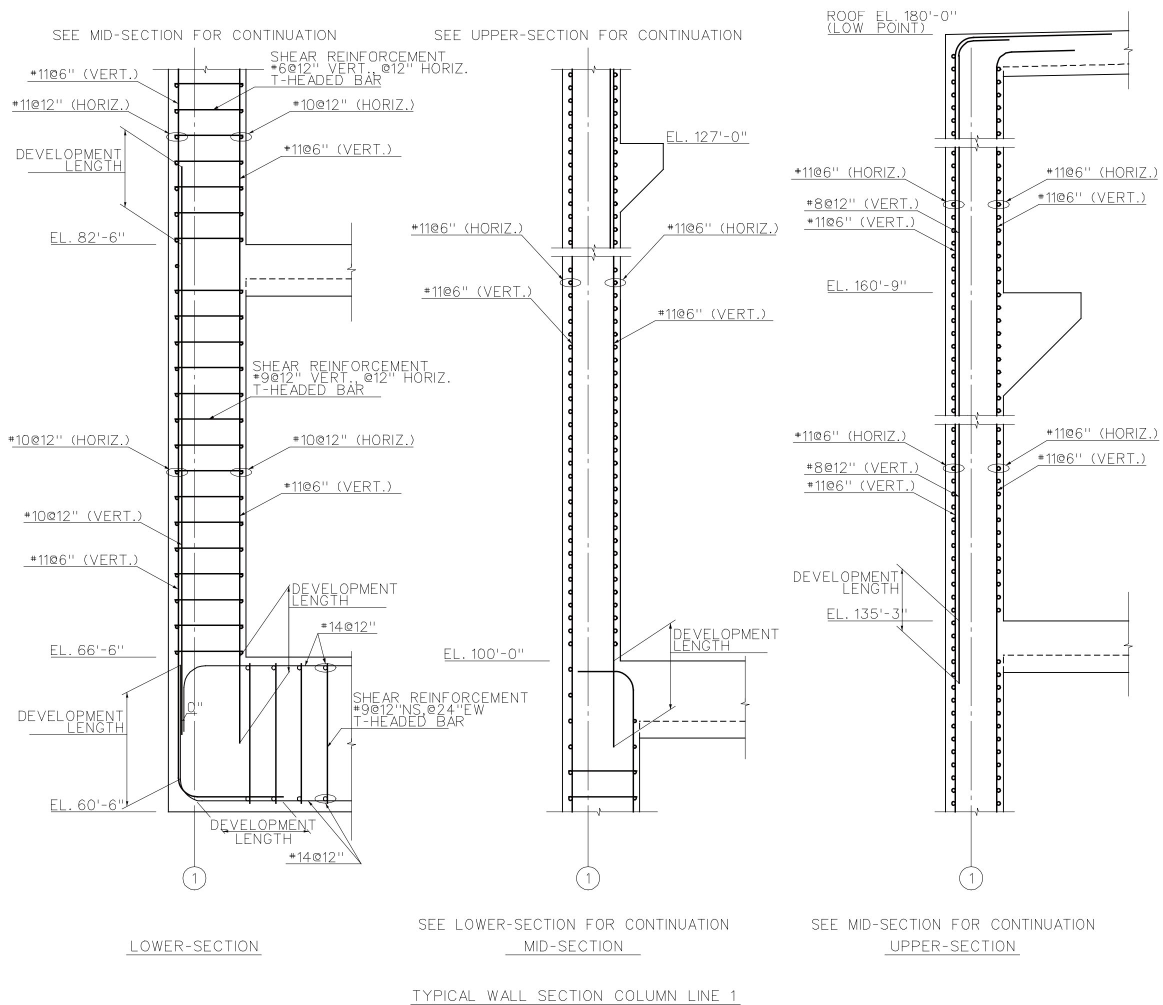


Figure 3H.5-3
[*Typical Reinforcement in Wall on Column Line 1*]*

*NRC Staff approval is required prior to implementing a change in this information.

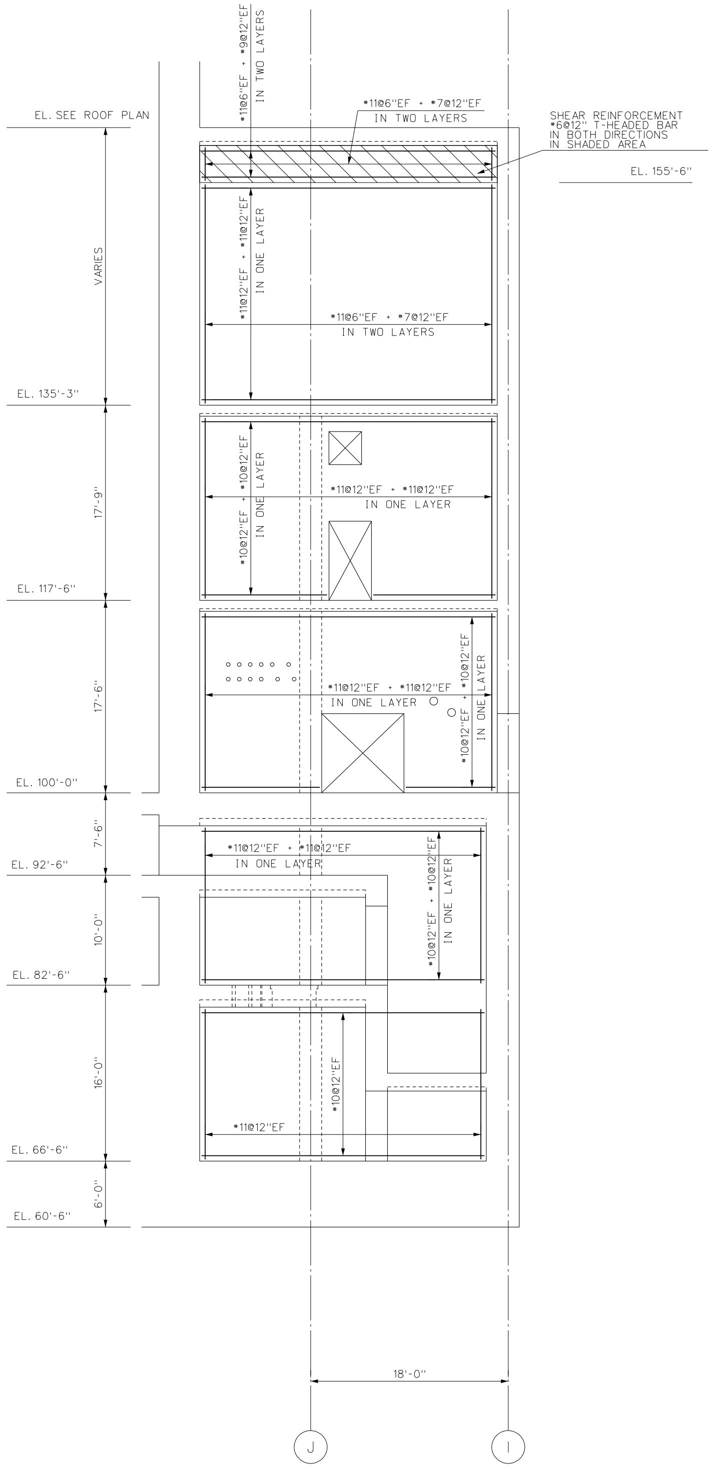


Figure 3H.5-4
[*Typical Reinforcement in Wall 7.3*]*

*NRC Staff approval is required prior to implementing a change in this information.

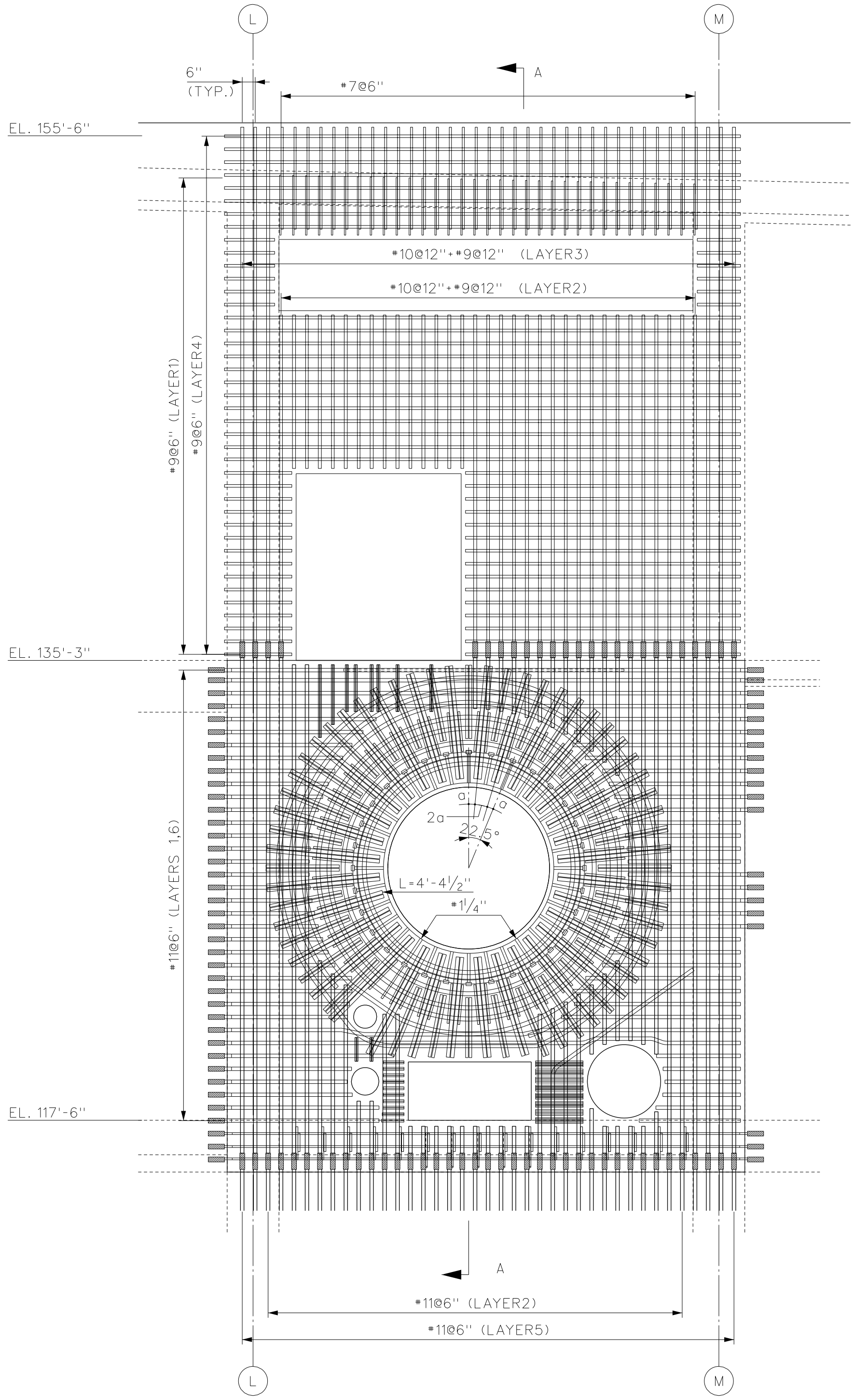


Figure 3H.5-5 (Sheet 1 of 3)
[*Concrete Reinforcement in Wall 11*]*

*NRC Staff approval is required prior to implementing a change in this information.

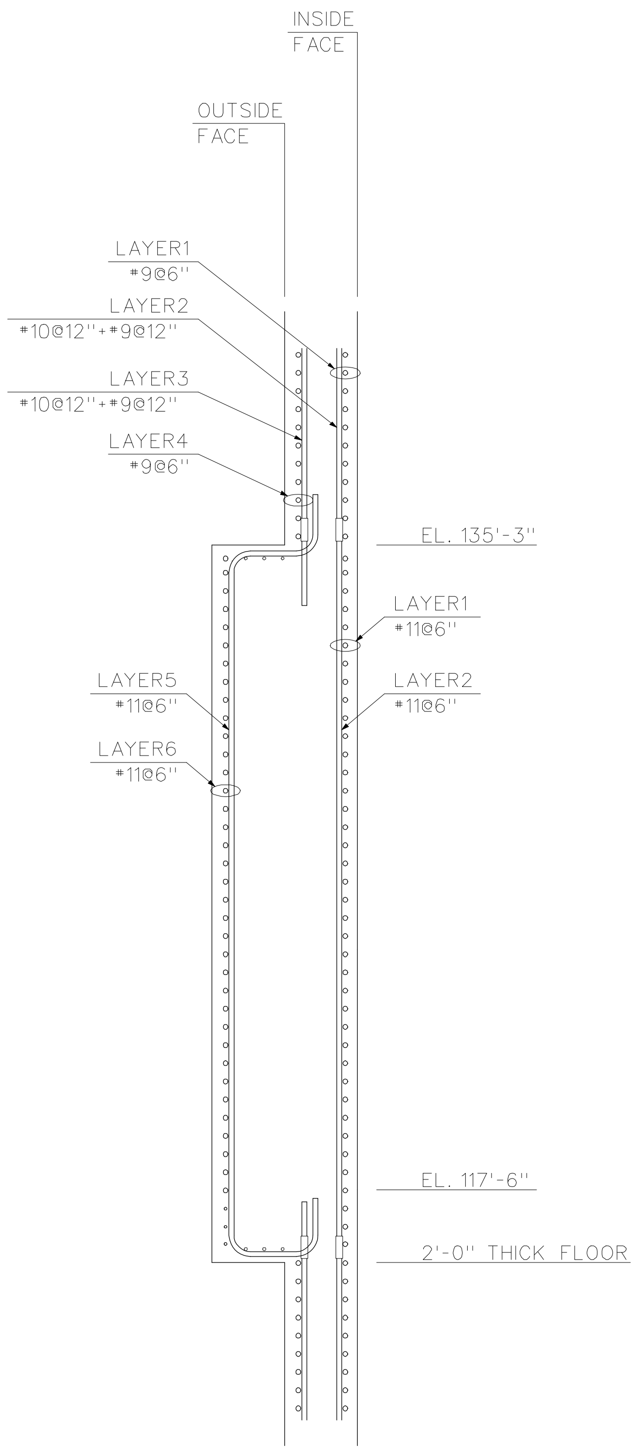


Figure 3H.5-5 (Sheet 2 of 3)
[*Concrete Reinforcement Layers in Wall 11 (Looking East)*]*

*NRC Staff approval is required prior to implementing a change in this information.

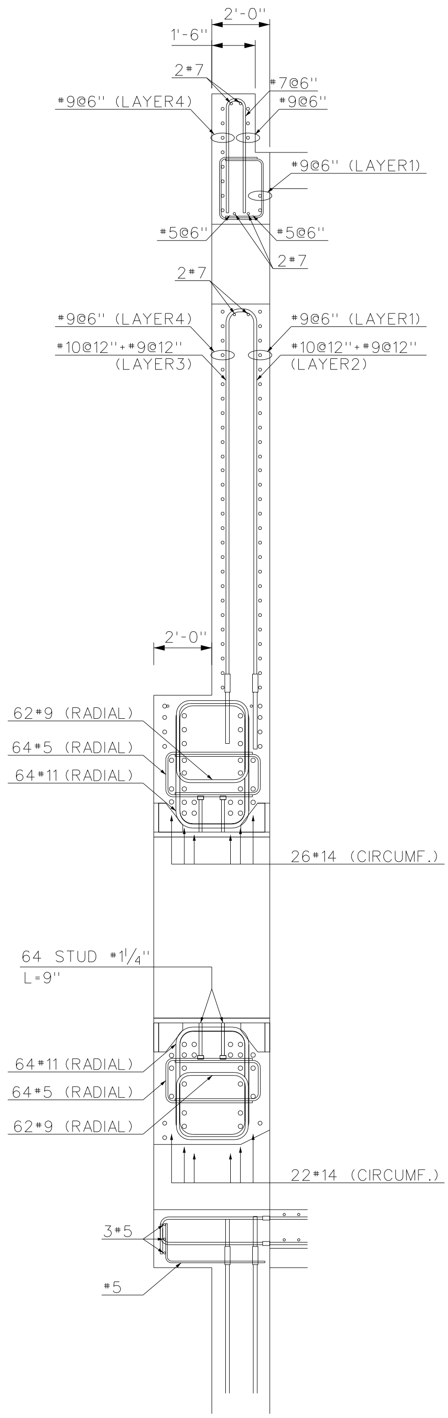
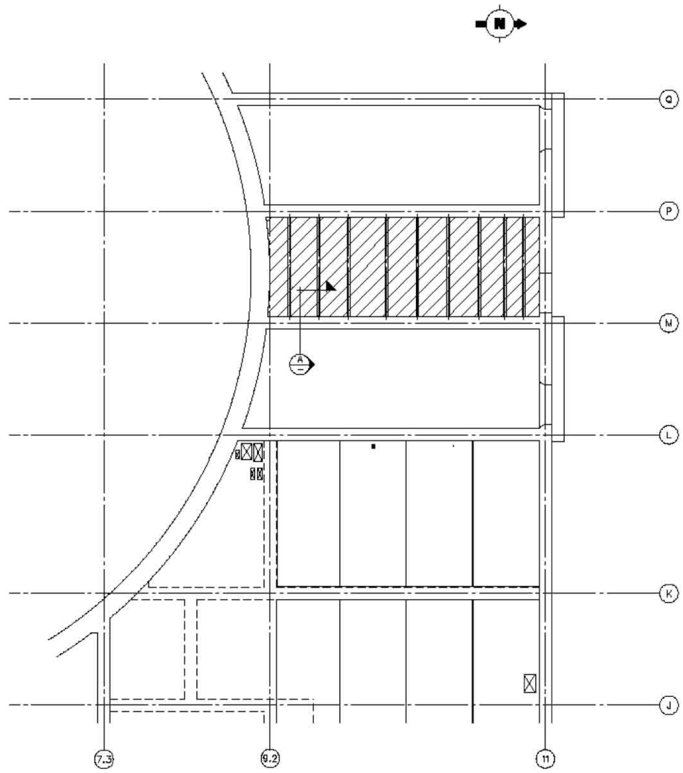
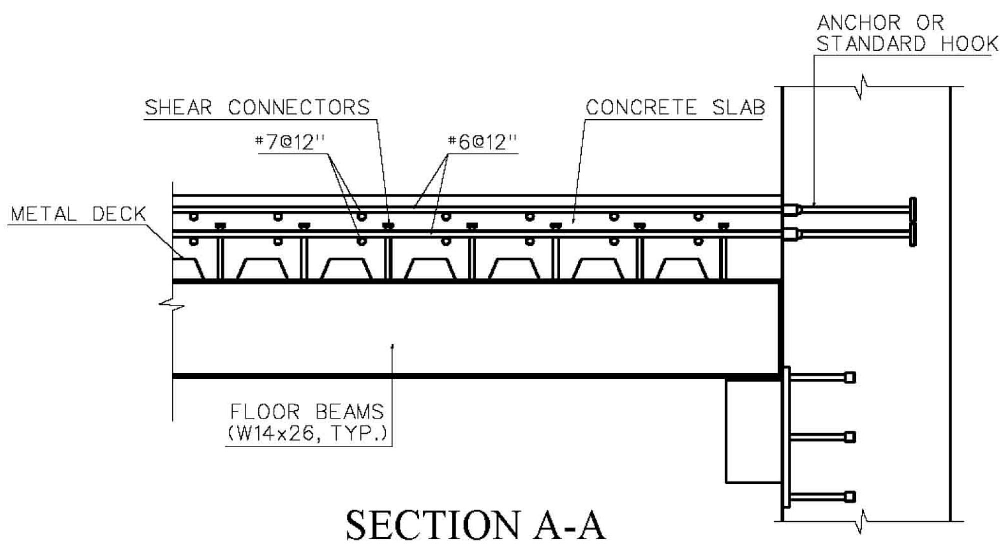


Figure 3H.5-5 (Sheet 3 of 3)
[*Wall 11 at Main Steamline Anchor Section A-A*]*

*NRC Staff approval is required prior to implementing a change in this information.



PLAN VIEW



SECTION A-A

Figure 3H.5-6
[*Auxiliary Building Typical Composite Floor*]*

*NRC Staff approval is required prior to implementing a change in this information.

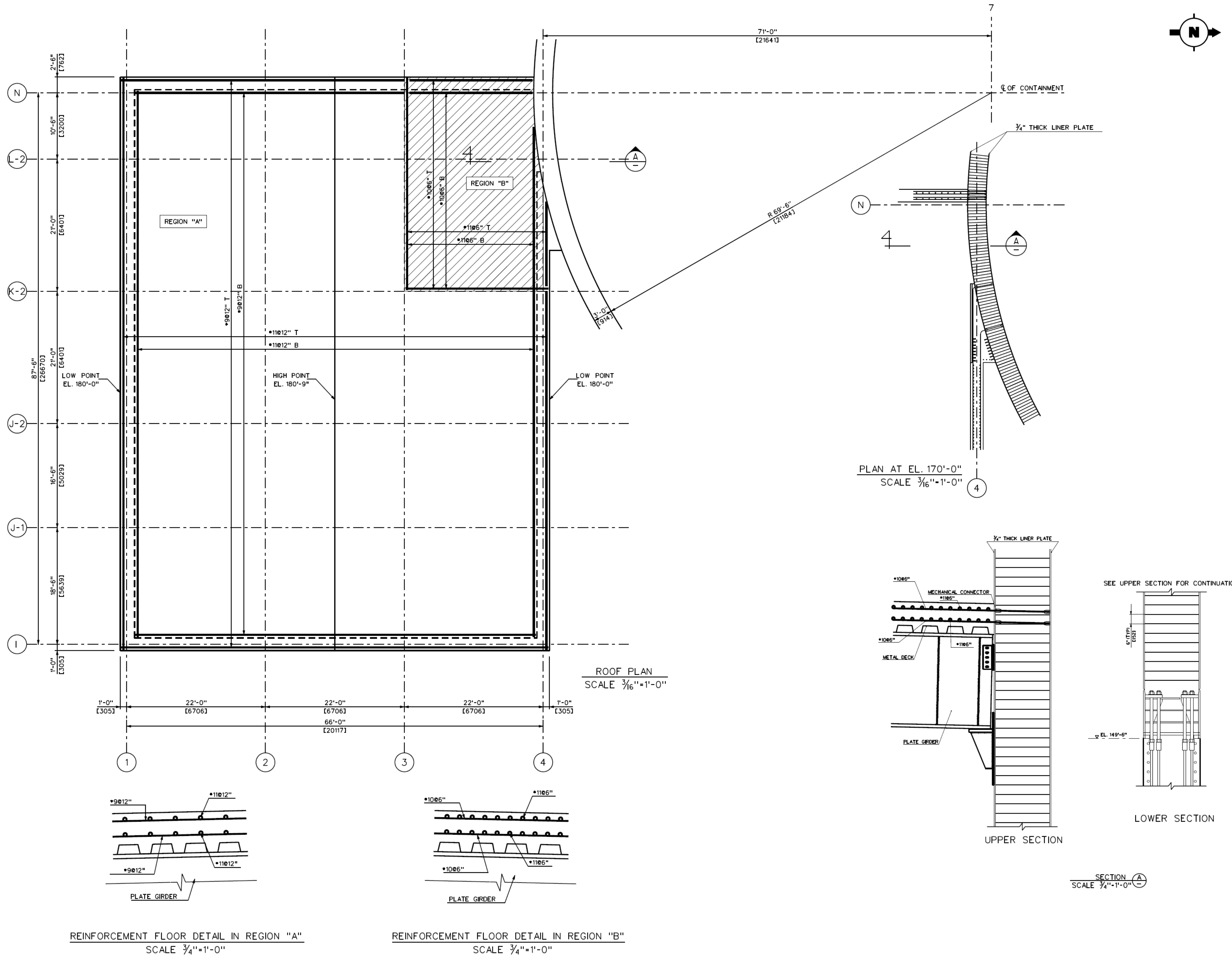


Figure 3H.5-7
[*Typical Reinforcement and Connection to Shield Building*]*

*NRC Staff approval is required prior to implementing a change in this information.

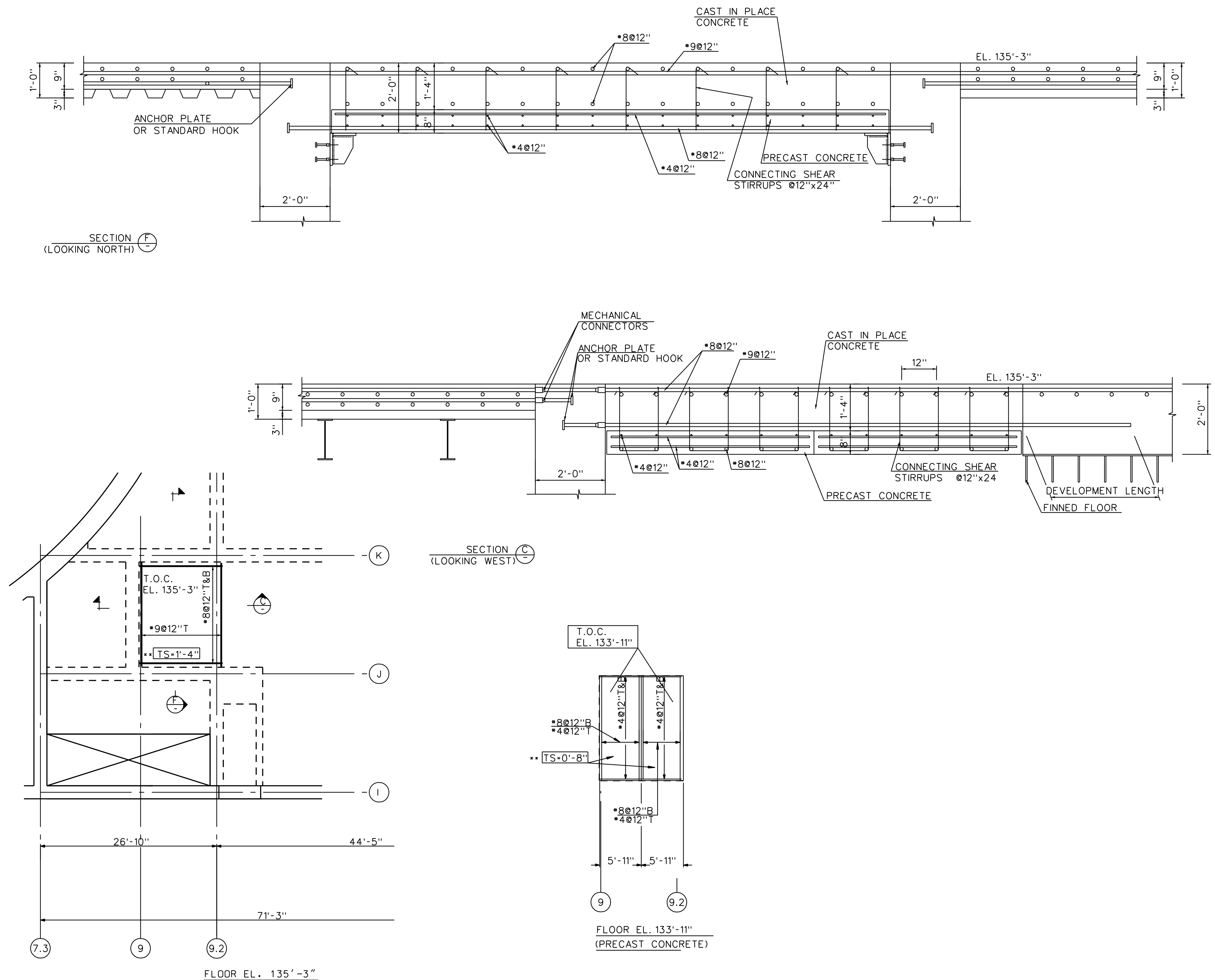


Figure 3H.5-8
[Auxiliary Building Operations Work Area (Tagging Room) Ceiling]*

*NRC Staff approval is required prior to implementing a change in this information.

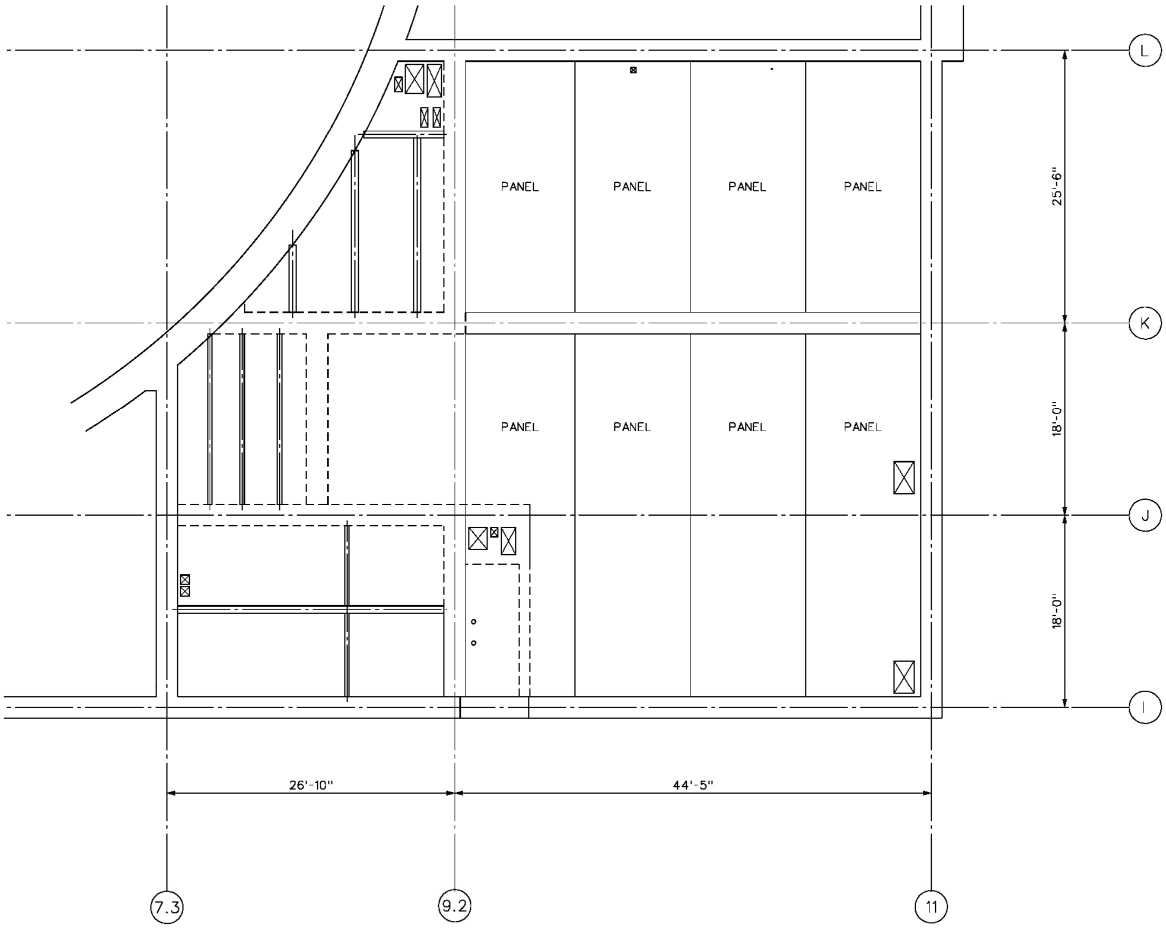


Figure 3H.5-9 (Sheet 1 of 3)
[*Auxiliary Building Finned Floor*]*

*NRC Staff approval is required prior to implementing a change in this information.

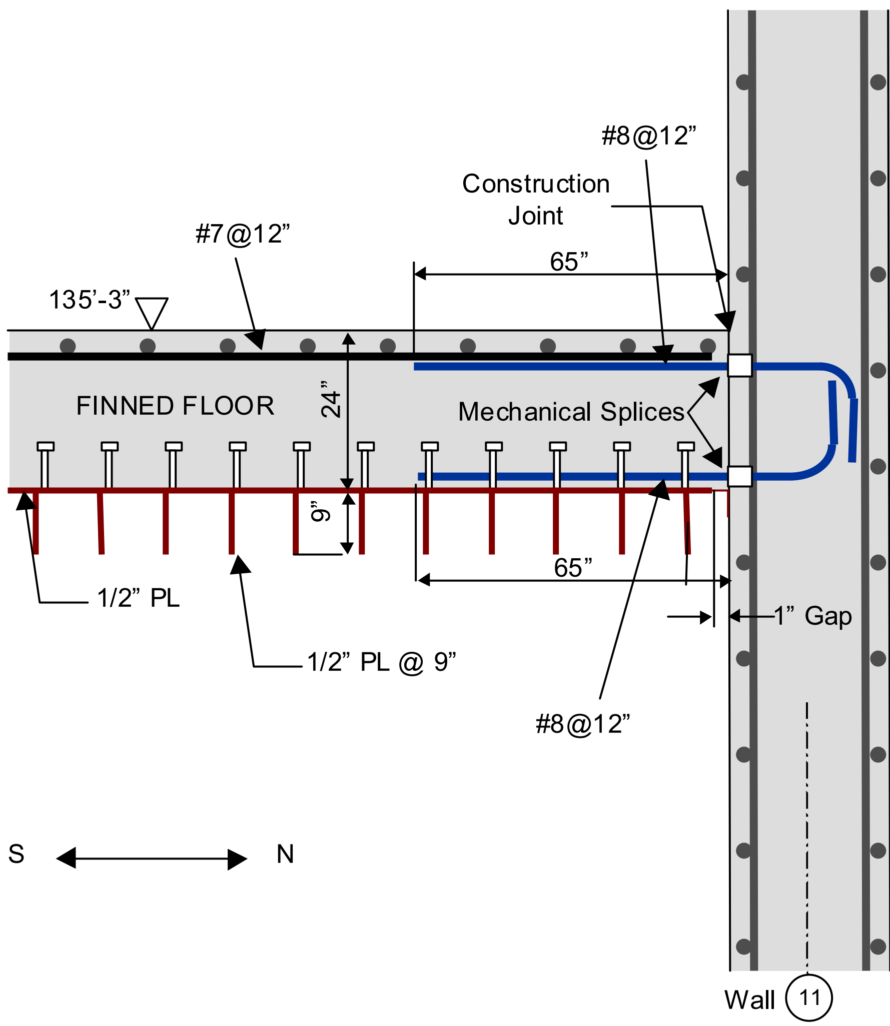


Figure 3H.5-9 (Sheet 2 of 3)
[*Auxiliary Building Finned Floor*]*

*NRC Staff approval is required prior to implementing a change in this information.

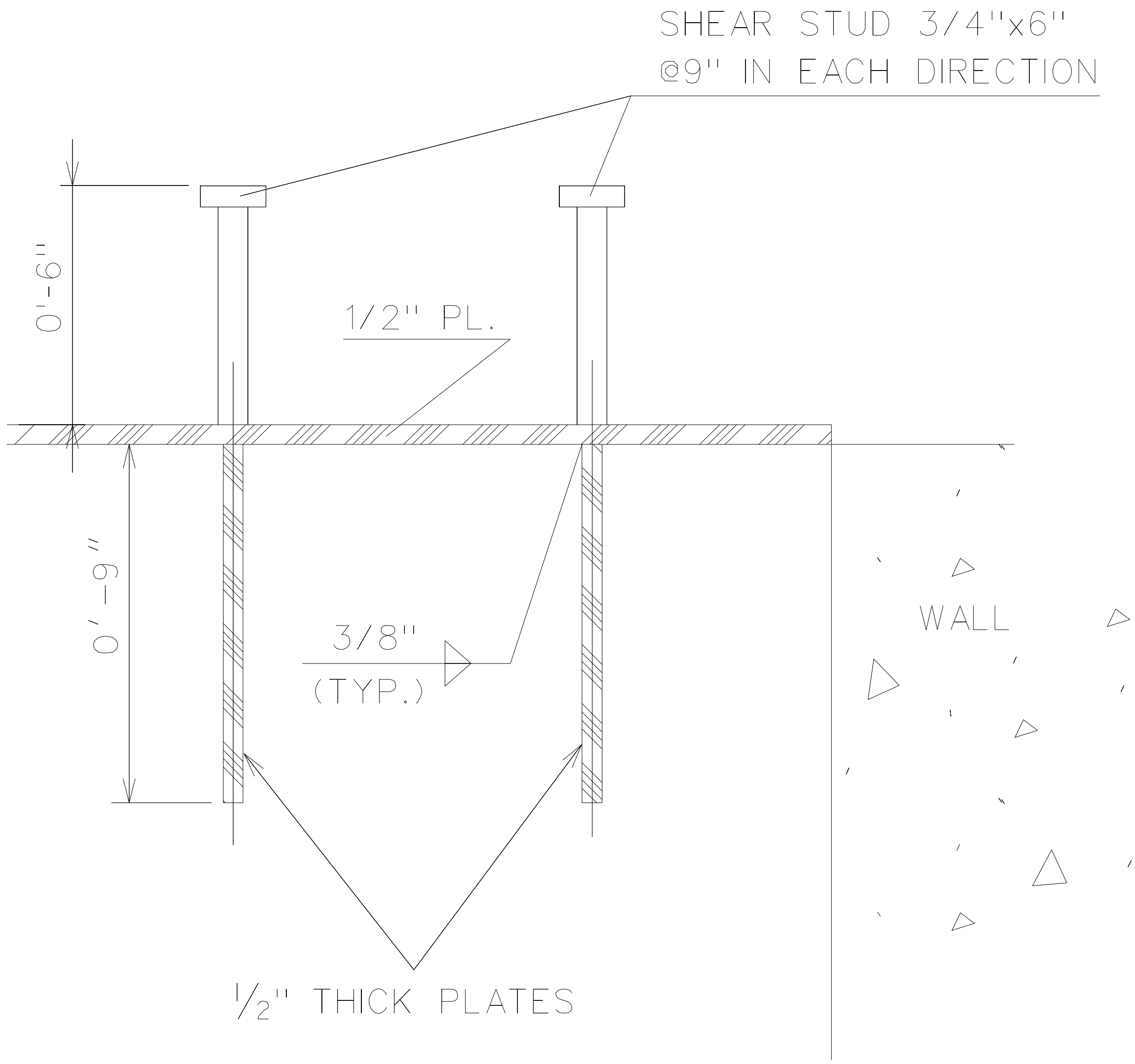


Figure 3H.5-9 (Sheet 3 of 3)
[*Auxiliary Building Finned Floor*]*

*NRC Staff approval is required prior to implementing a change in this information.

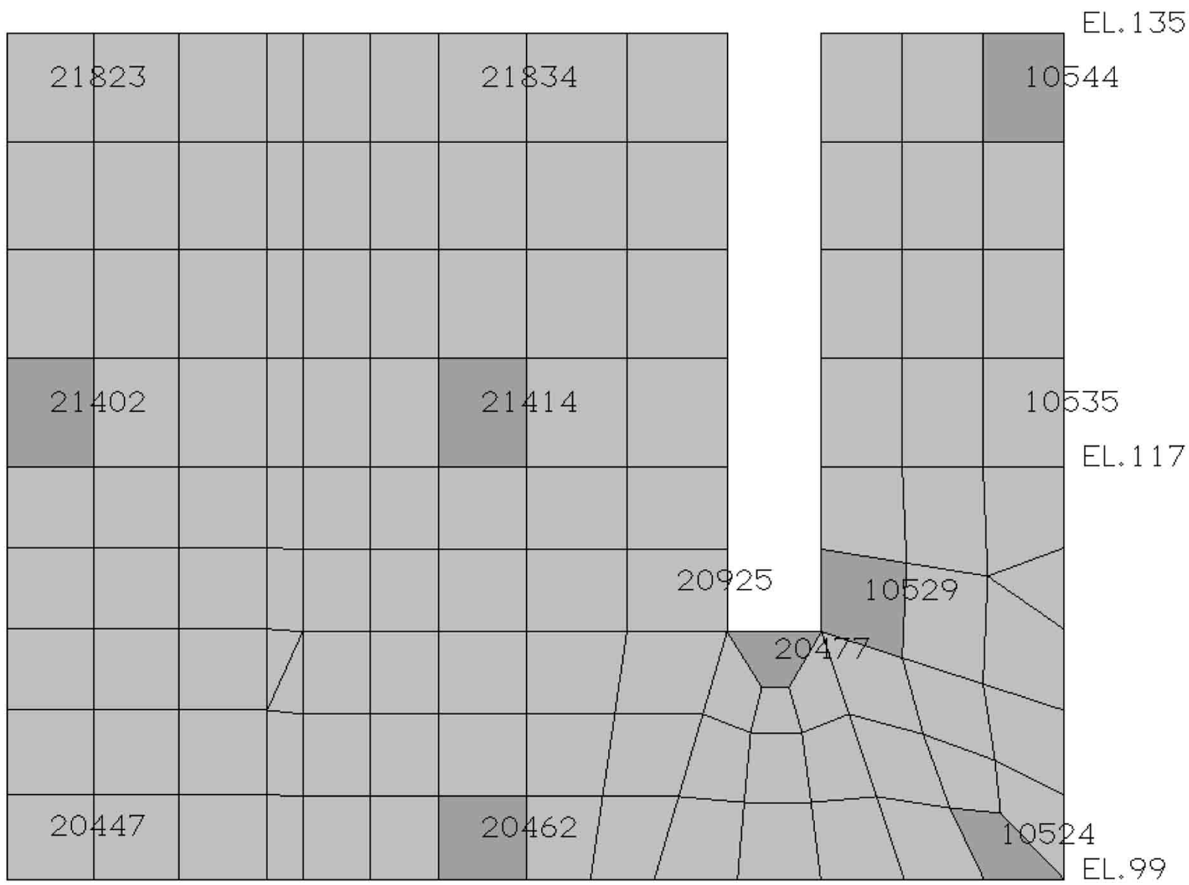


Figure 3H.5-10
[*Spent Fuel Pool Wall Divider Wall Element Locations*]*

*NRC Staff approval is required prior to implementing a change in this information.

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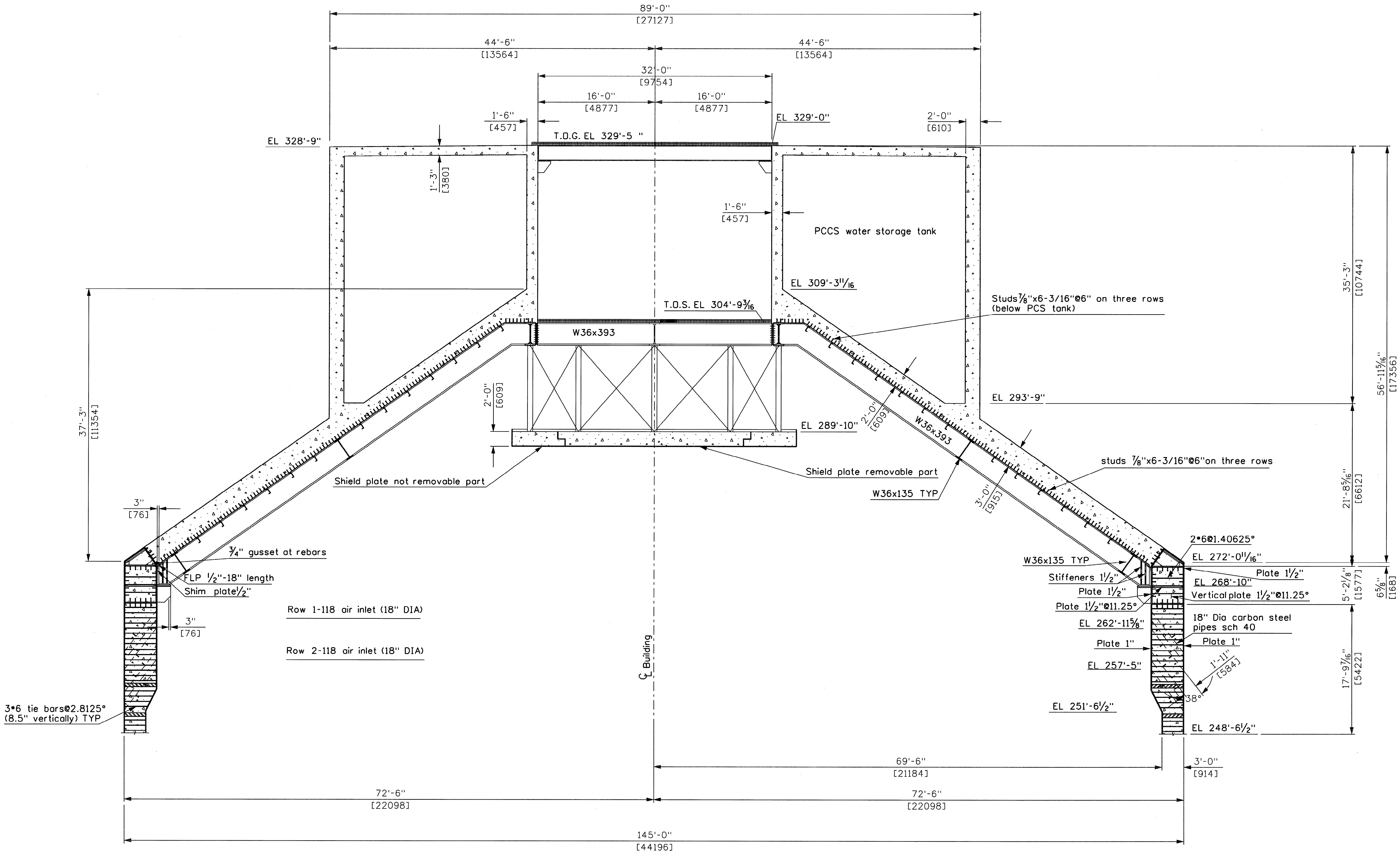

Figure 3H.5-11 (Sheet 1 of 6)
[*Design of Shield Building: Roof and Air Inlets*]*

*NRC Staff approval is required prior to implementing a change in this information.

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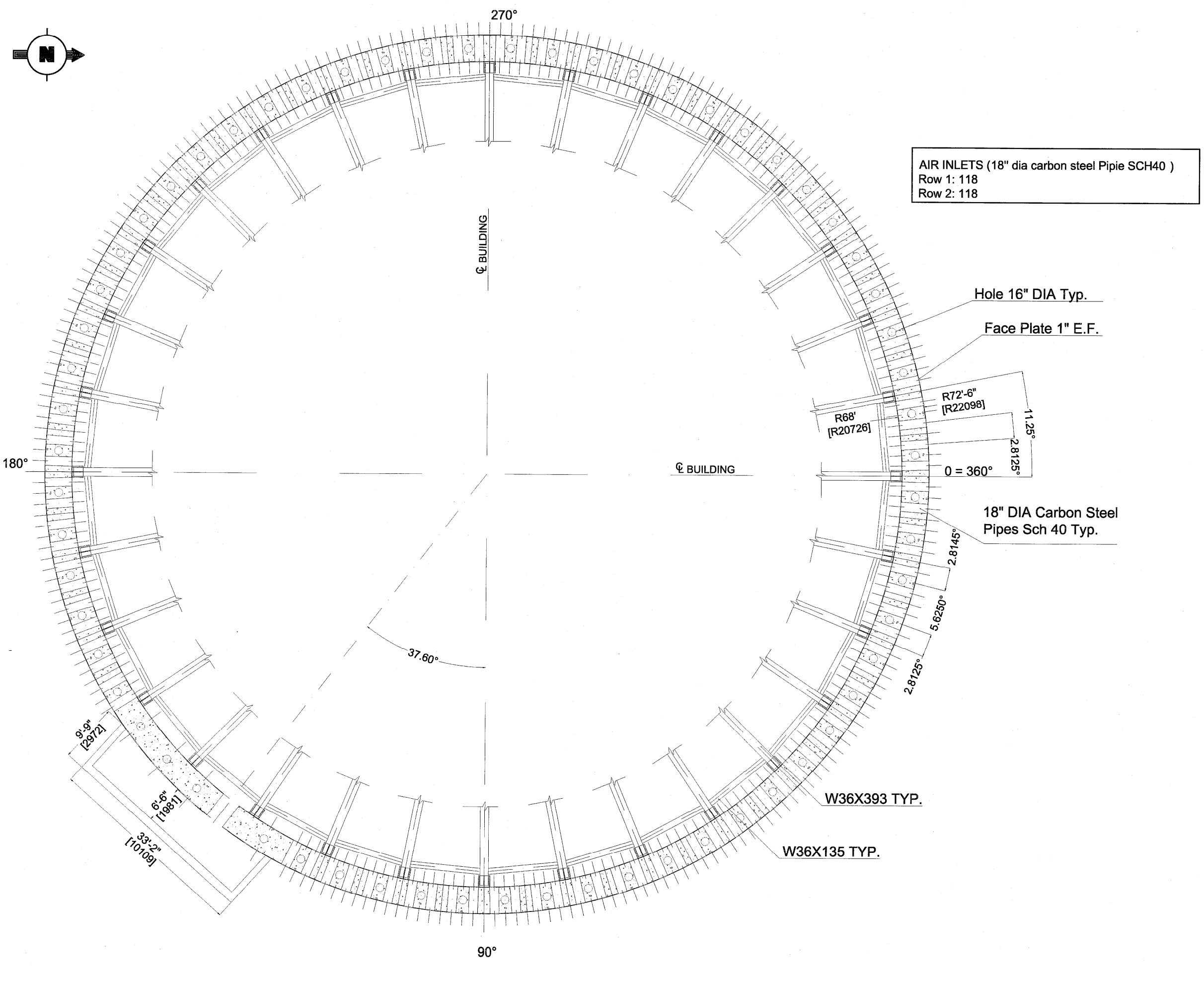

Figure 3H.5-11 (Sheet 2 of 6)
[*Design of Shield Building: Concrete Detail at Tension Ring*]*

*NRC Staff approval is required prior to implementing a change in this information.

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Critical Sections of Air Inlet and Tension Ring
at the Connection with Conical Roof

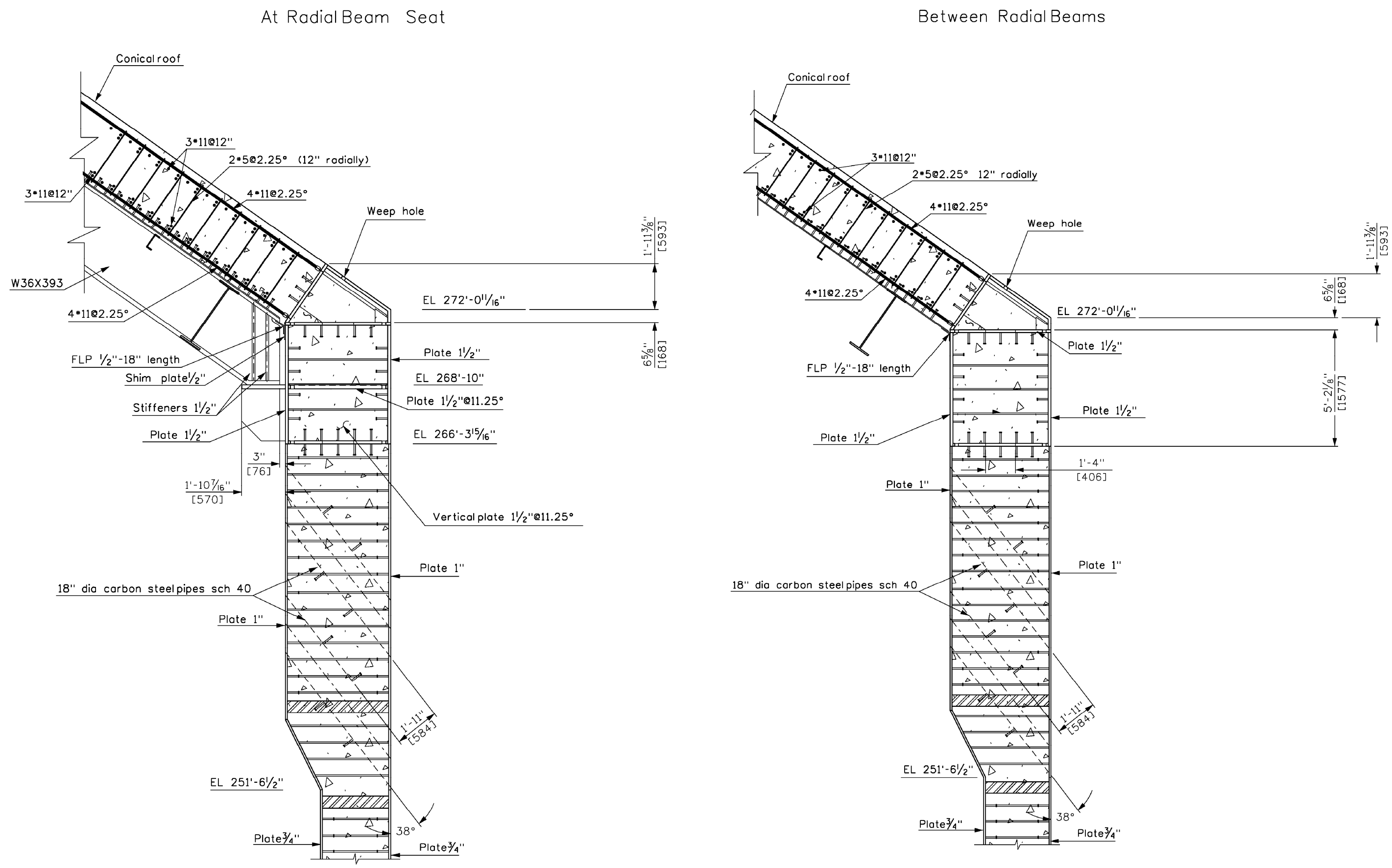

Figure 3H.5-11 (Sheet 3 of 6)
[*Design of Shield Building: Roof/Air Inlet Interface*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withhold Under 10 CFR 2.390d

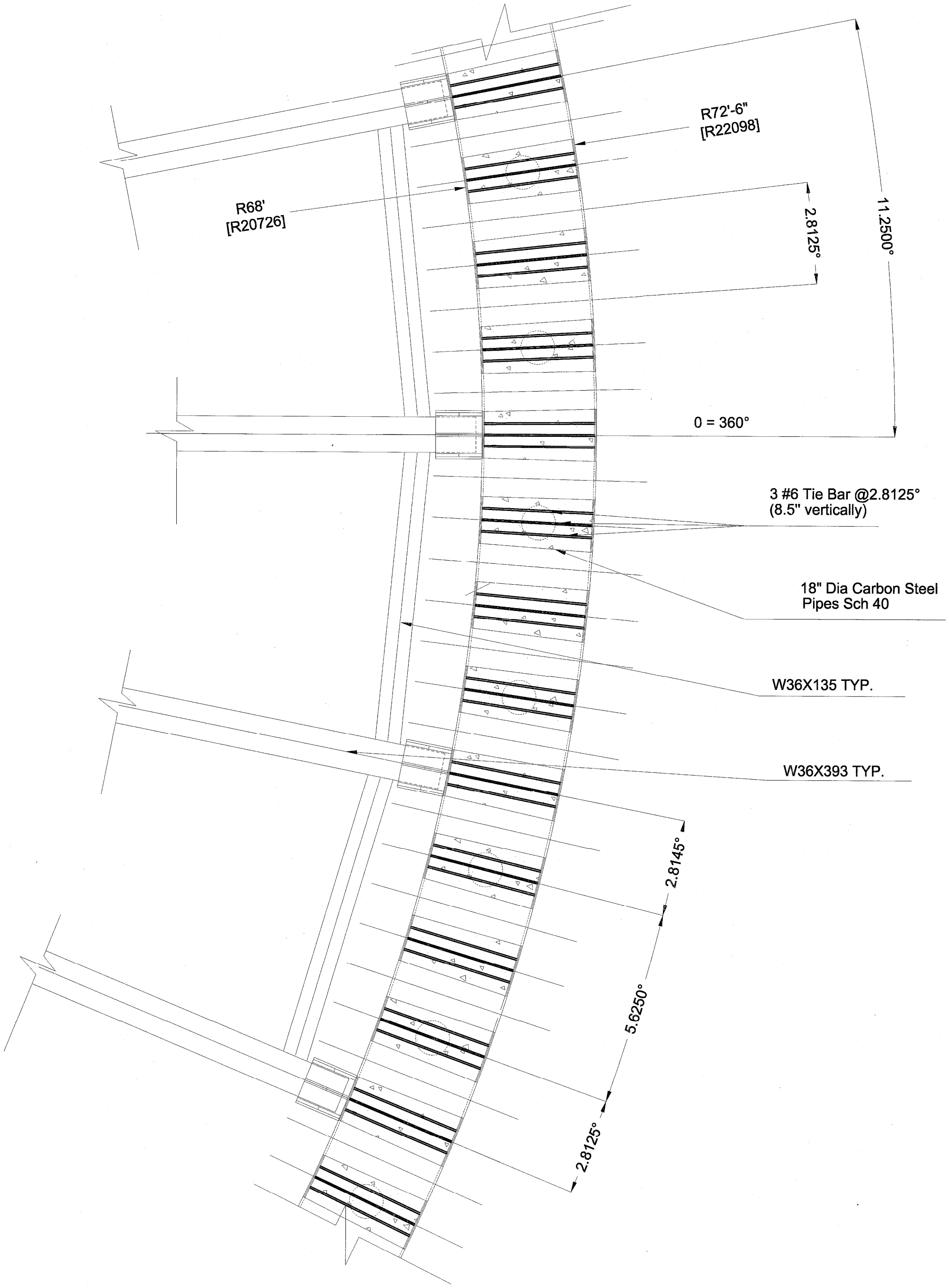

Figure 3H.5-11 (Sheet 4 of 6)
[*Design of Shield Building at Air Inlets*]*

*NRC Staff approval is required prior to implementing a change in this information.

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CRITICAL SECTION AT THE CONNECTION
BETWEEN CONICAL ROOF AND PCCS TANK

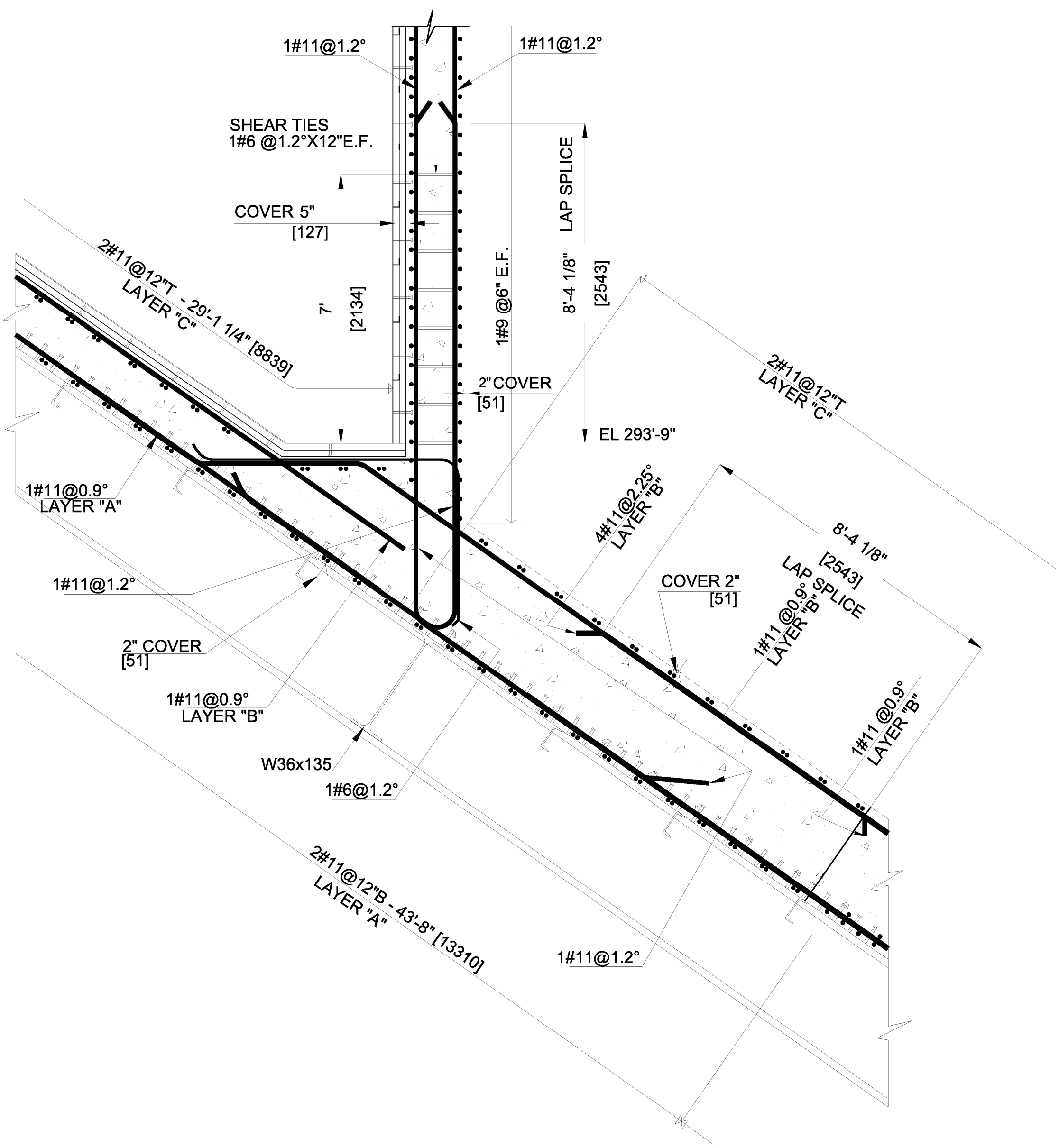

Figure 3H.5-11 (Sheet 5 of 6)
[*Design of Shield Building: Tank/Roof Interface Reinforcement*]*

*NRC Staff approval is required prior to implementing a change in this information.

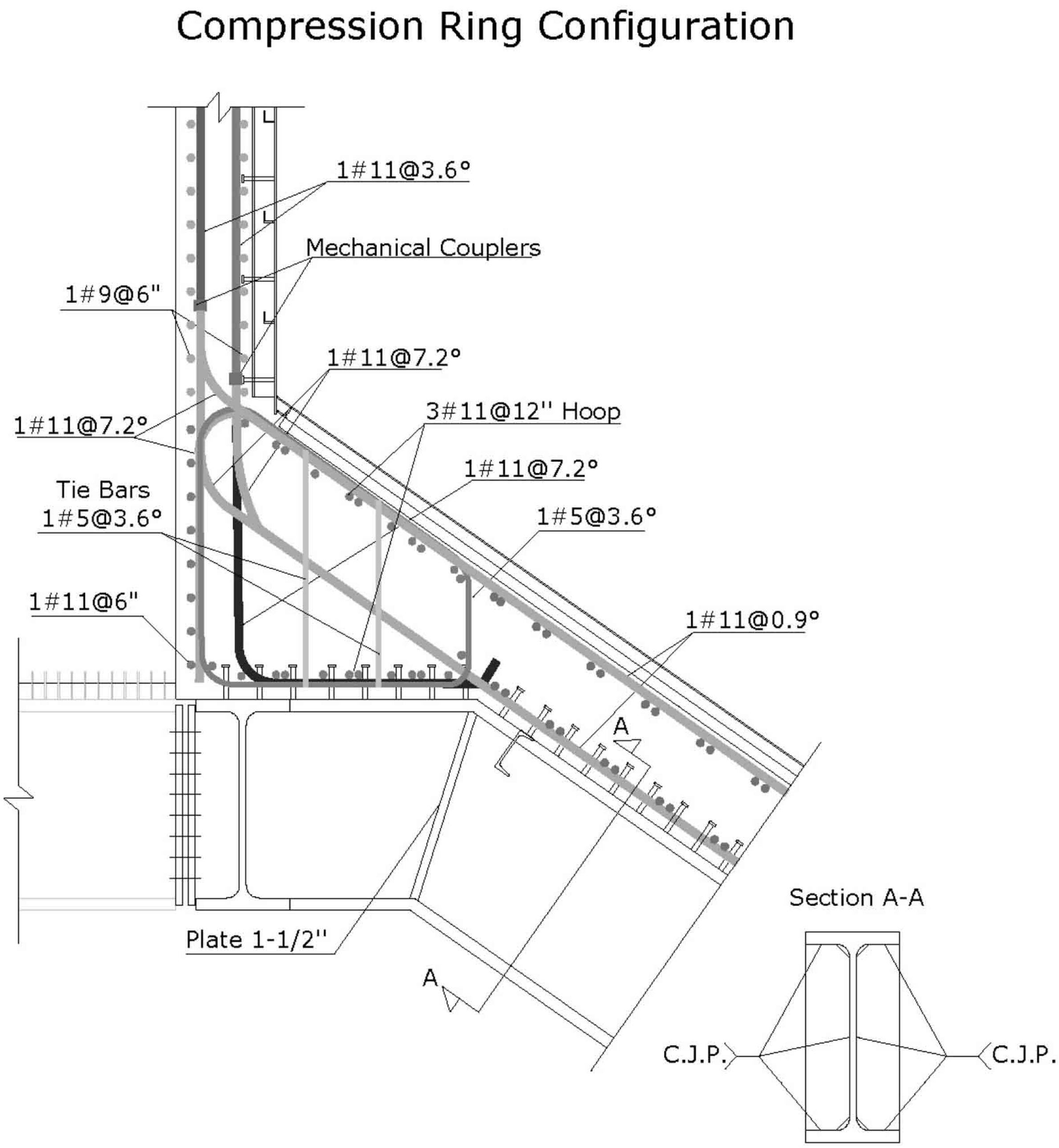

Figure 3H.5-11 (Sheet 6 of 6)
Design of Shield Building: Tank/Compression Ring Roof Interface Reinforcement

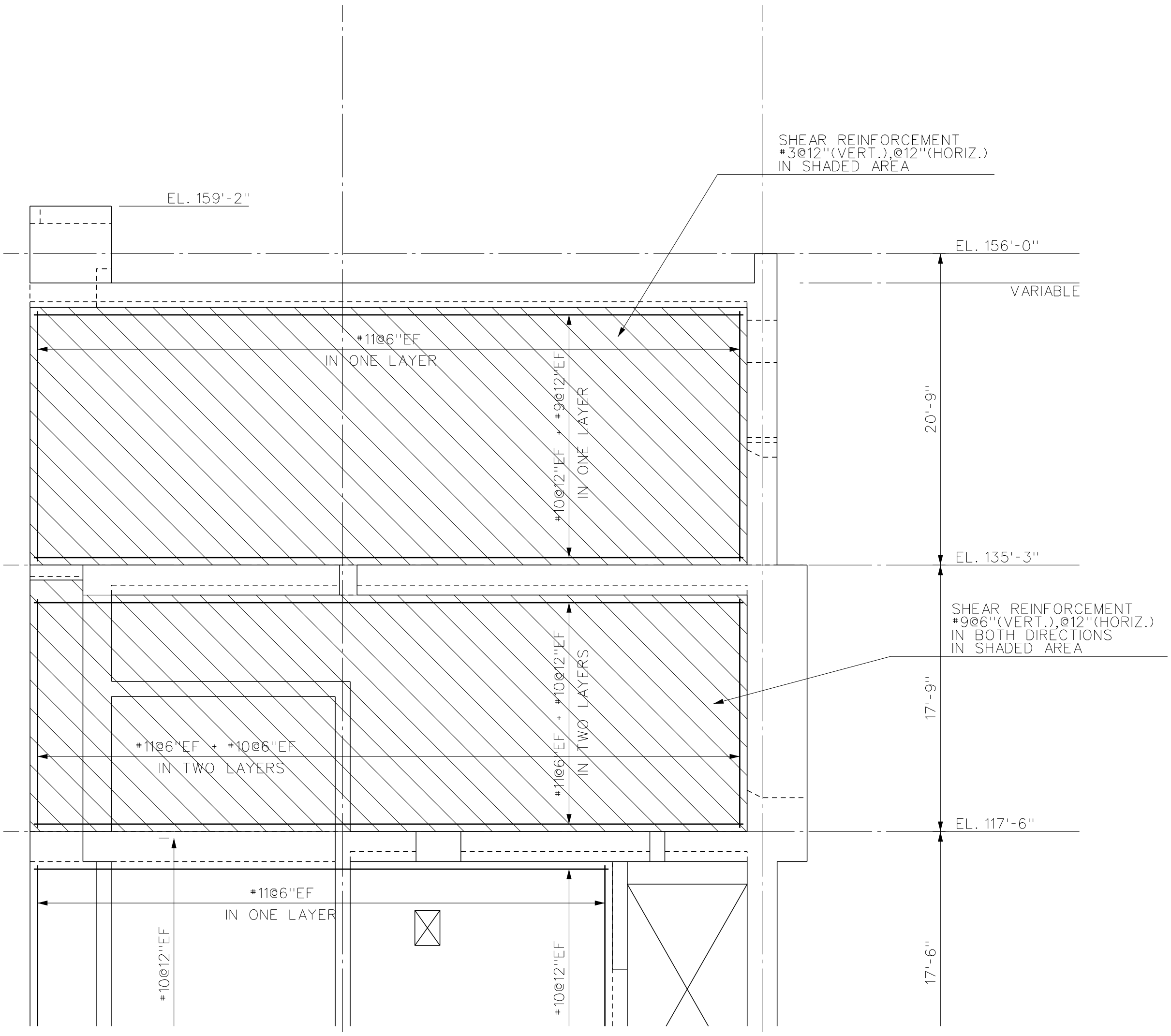

Figure 3H.5-12
[*Typical Reinforcement in Wall L*]*

*NRC Staff approval is required prior to implementing a change in this information.

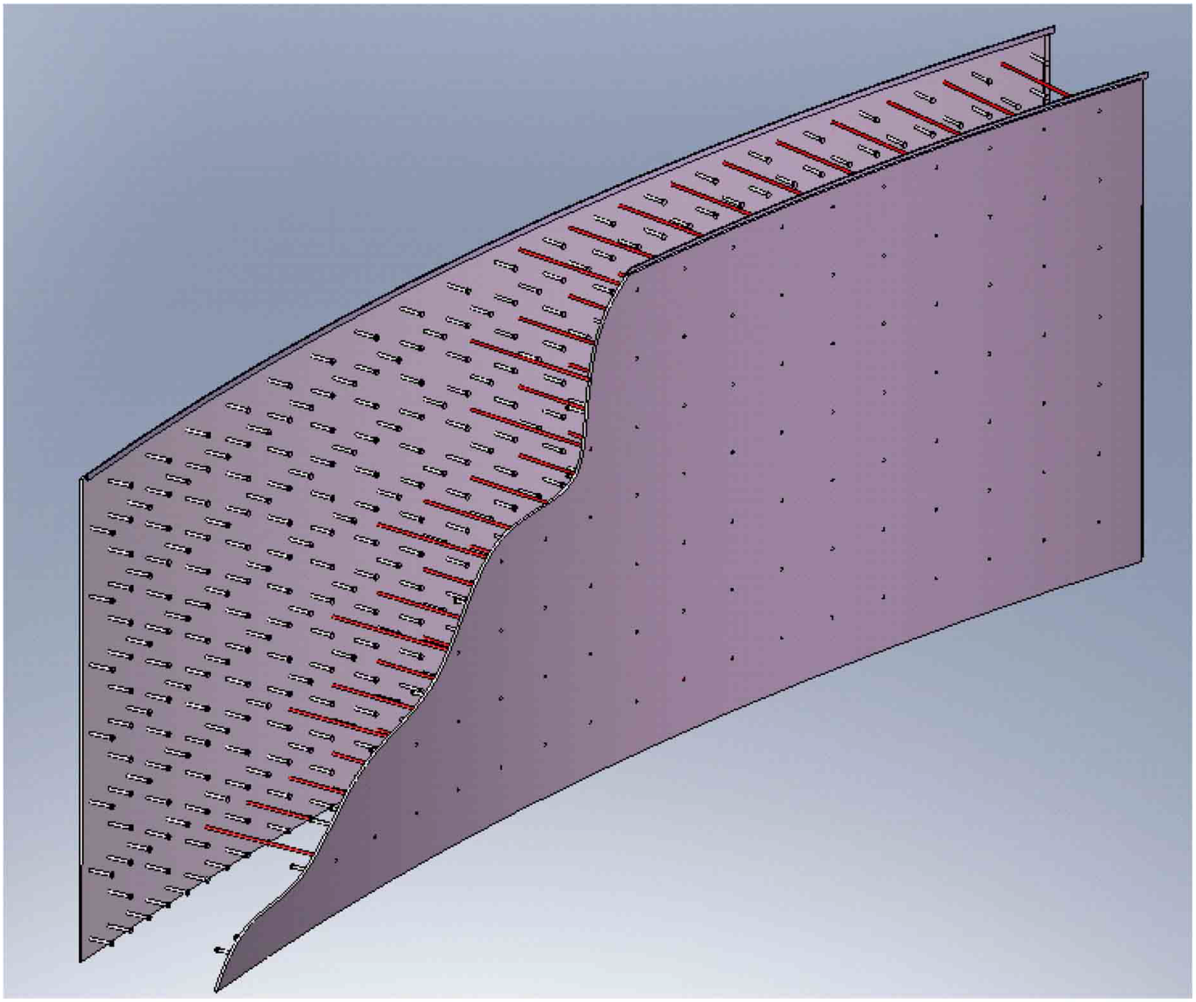

Figure 3H.5-13
Enhanced Shield Building Wall Panel Layout

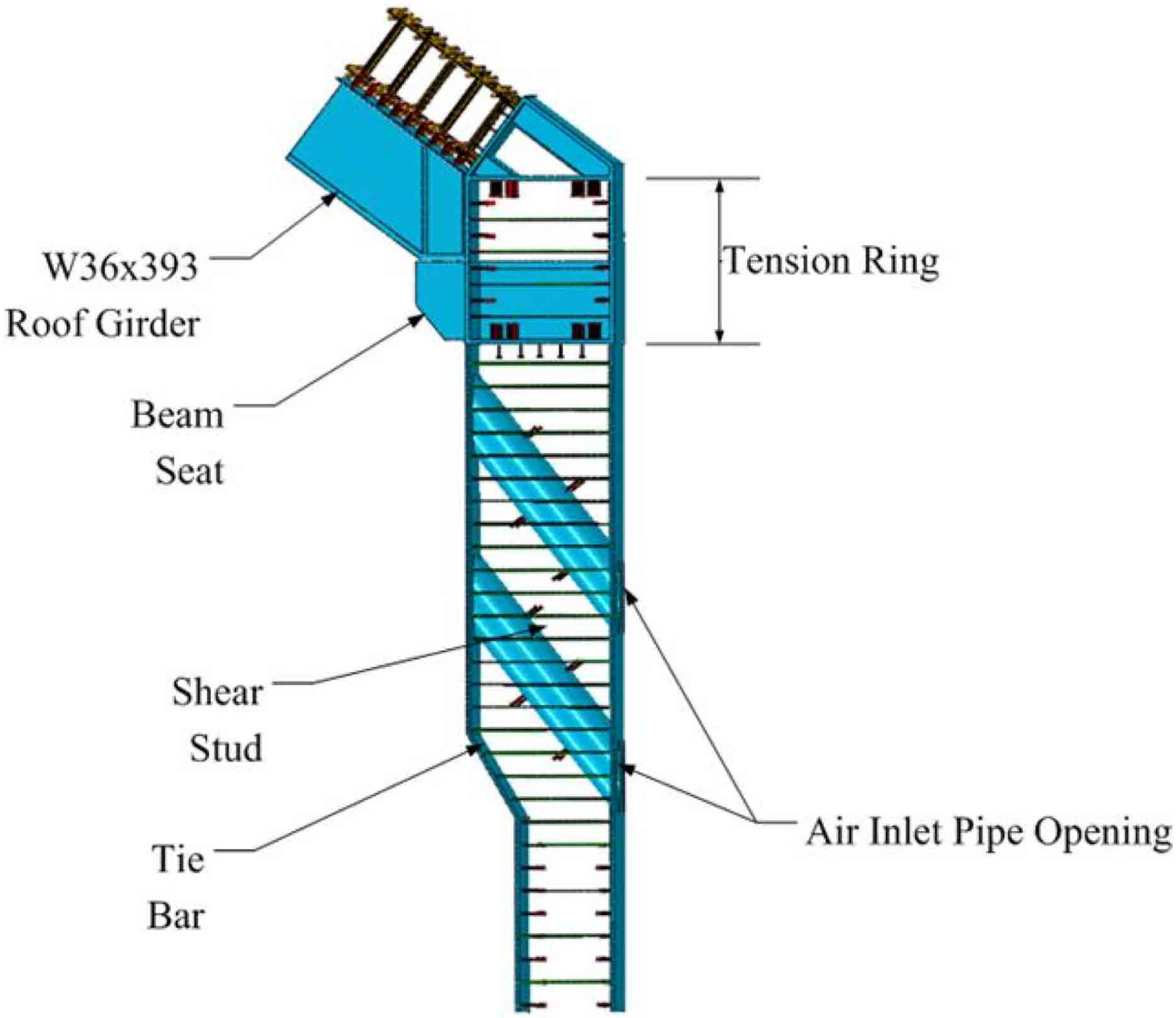

Figure 3H.5-14
Elevation View of Tension Ring and Air Inlets

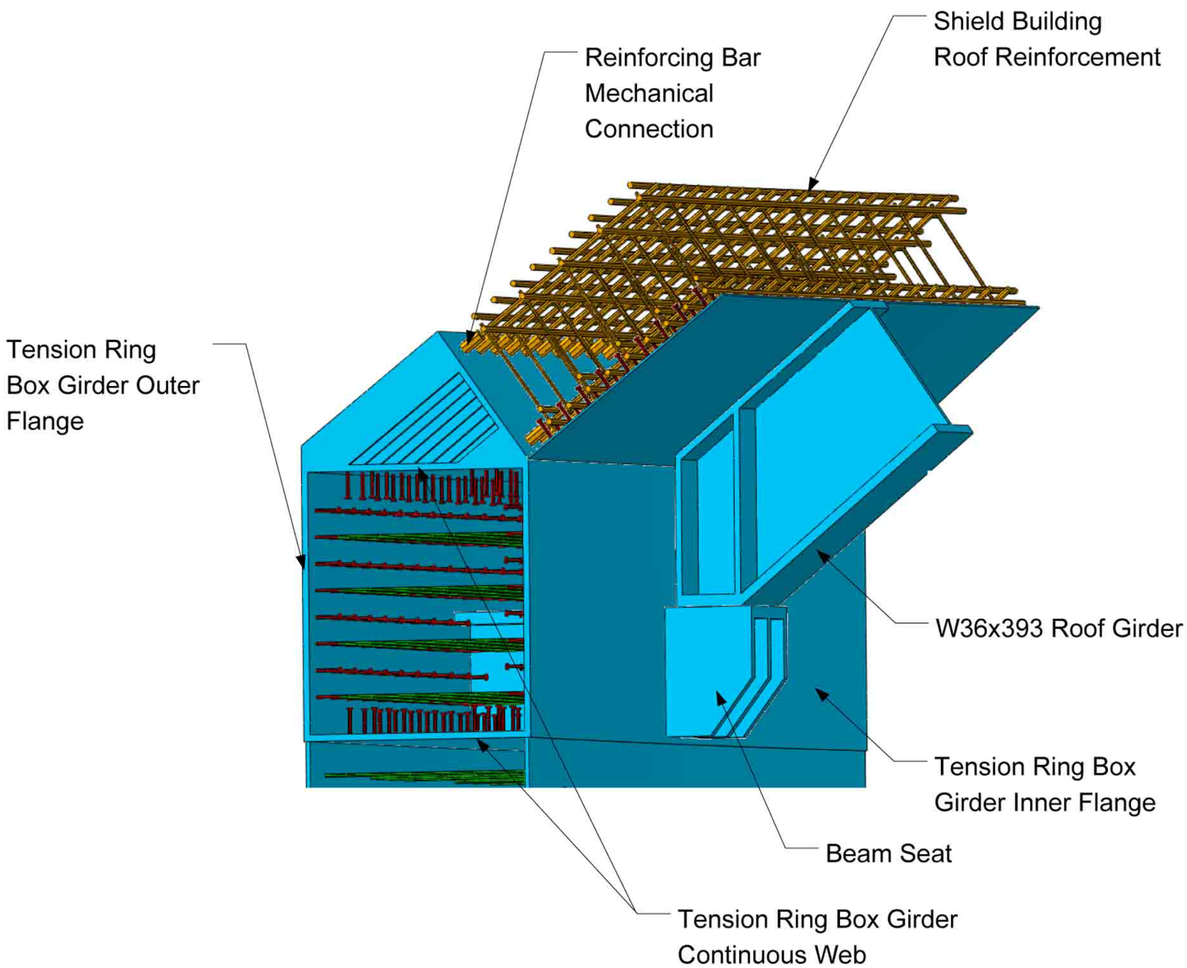

Figure 3H.5-15
Shield Building Tension Ring

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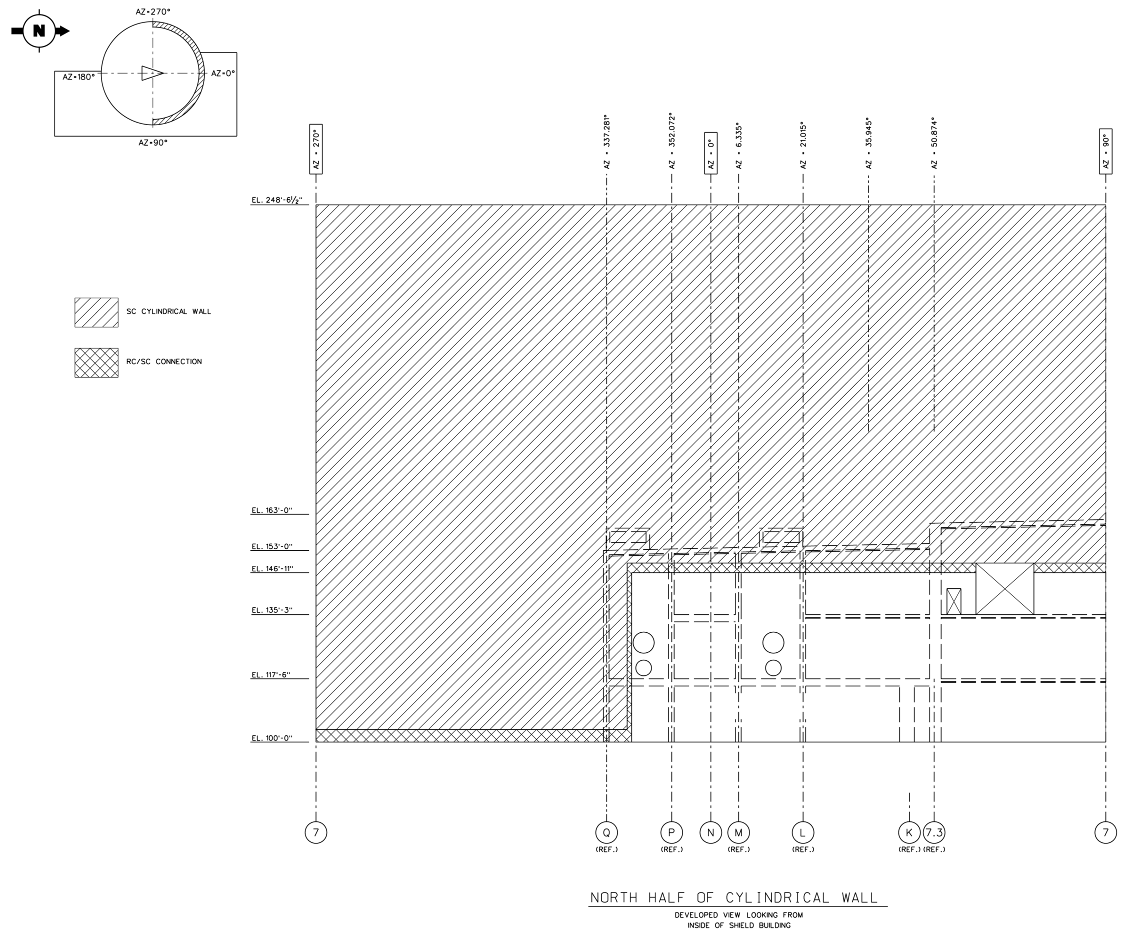

For additional information, see Figure 6 of APP-GW-GLR-602 (Reference 1).

Figure 3H.5-16 (Sheet 1 of 2)
[*Design of Shield Building: Surface Plates on Cylindrical Section – Developed View 90-270 Degrees*]*

*NRC Staff approval is required prior to implementing a change in this information.

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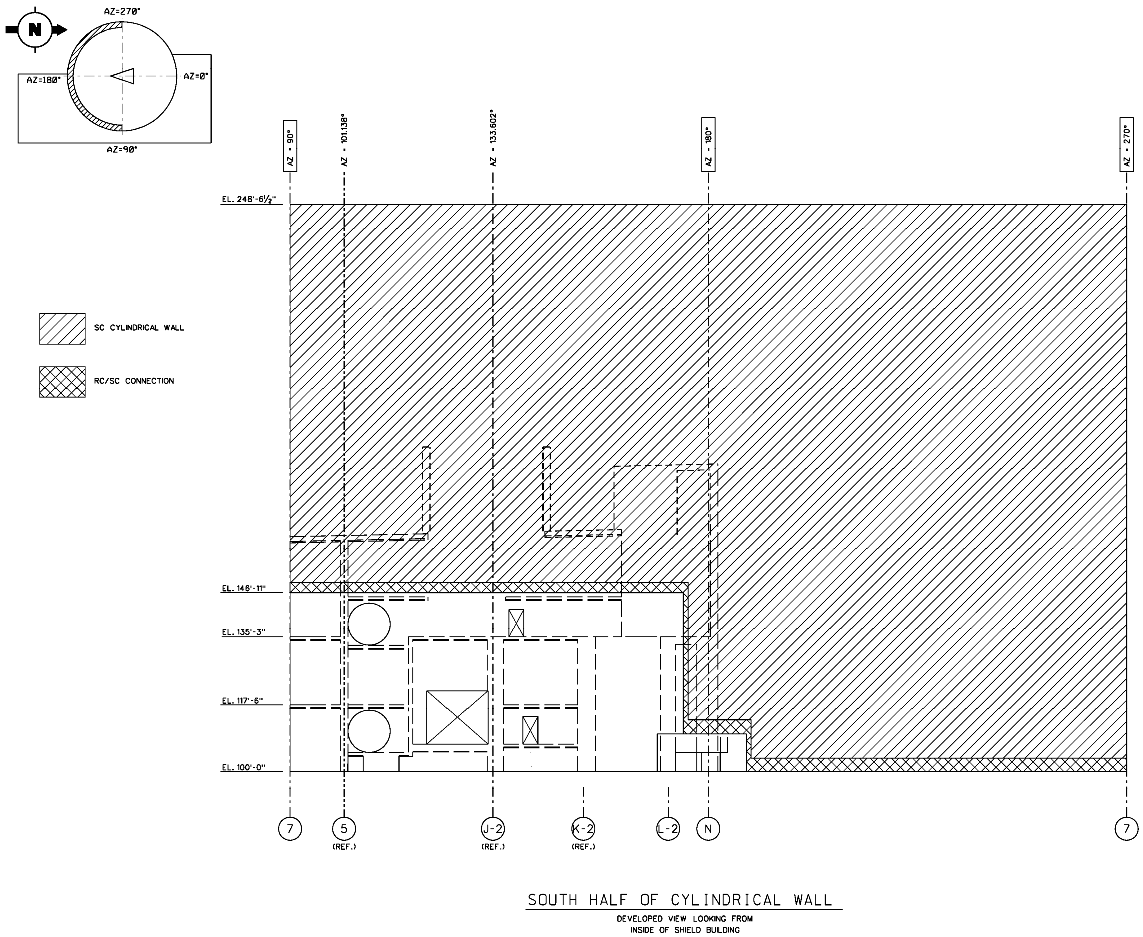

For additional information, see Figure 6 of APP-GW-GLR-602 (Reference 1).

Figure 3H.5-16 (Sheet 2 of 2)
[*Design of Shield Building: Surface Plates on Cylindrical Section – Developed View 270-90 Degrees*]*

*NRC Staff approval is required prior to implementing a change in this information.

Appendix 3I Evaluation for High Frequency Seismic Input

3I.1 Introduction

The seismic analysis and design of the AP1000 plant is based on the Certified Seismic Design Response Spectra (CSDRS) shown in Subsection 3.7.1.1. These spectra are based on Regulatory Guide 1.60 with an increase in the 25 hertz region. Ground Motion Response Spectra (GMRS) for some Central and Eastern United States rock sites show higher amplitude at high frequency than the CSDRS. Evaluations are described in this appendix for an envelope response spectra with high frequency for the seismic input. The resulting spectra of this site are shown in Figure 3I.1-1 and Figure 3I.1-2 and compare this hard rock high frequency (HRHF) envelope response spectra at the foundation level against the AP1000 CSDRS for both the horizontal and vertical directions for 5% damping. The HRHF envelope response spectra exceed the CSDRS for frequencies above about 15 Hz.

High frequency seismic input is generally considered to be non-damaging as described in Reference 1. The evaluation of the AP1000 nuclear island for the high frequency input is based on the analysis of a limited sample of structures, components, supports, and piping to demonstrate that the high frequency seismic response is non-damaging. The evaluation includes building structures, reactor pressure vessel and internals, primary component supports, primary loop nozzles, piping, and equipment.

This appendix describes the methodology and criteria used in the evaluation to confirm that the high frequency input is not damaging to equipment and structures qualified by analysis for the AP1000 CSDRS. It provides supplemental criteria for selection and testing of equipment whose function might be sensitive to high frequency. The results of the high frequency evaluation demonstrating that the AP1000 plant is qualified for this type of input are documented in a technical report (Reference 2). This report will provide a summary of the analysis and test results.

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

Because the NI FIRS are not enveloped by the AP1000 HRHF spectra, a site-specific analysis is performed to evaluate and justify exceedances. Technical report WLG-GW-GLR-815 (Reference 201) provides a summary of those evaluations and results. This report presents in-structure response spectra throughout the Nuclear Island resulting from the site-specific input. These in-structure response spectra were investigated and all exceedances of the CSDRS or HRHF spectra were identified. Three instances of largest exceedances were noted, and these three instances were investigated as bounding conditions and justified by further evaluations.

3I.2 High Frequency Seismic Input

Presented in Figures 3I.1-1 and 3I.1-2 is a comparison of the horizontal and vertical HRHF envelope response spectra and the AP1000 CSDRS. The HRHF envelope response spectra presented are calculated at foundation level (39.5' below grade), at the upper most competent material and treated as an outcrop for calculation purposes.

For each direction, the HRHF envelope response spectra exceed the design spectra in higher frequencies (greater than 15 Hz horizontal and 20 Hz vertical). The spectra are used for the HRHF

envelope response spectra. If necessary, the HRHF envelope response spectra are enhanced at low frequencies so that HRHF envelope response spectra fully envelope all of the hard rock sites.

These HRHF envelope response spectra are further limited in that the shear wave velocity limitation is defined at the bottom of the basemat equal to or higher than 7,500 fps, while maintaining a shear wave velocity equal to or above 8,000 fps at the lower depths.

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

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

3I.3 NI Models Used To Develop High Frequency Response

The NI20 nuclear island model described in Appendix 3G is analyzed in ACS SASSI using the HRHF time histories applied at foundation level to obtain the motion at the base.

A modal analysis of the NI05 model for both the auxiliary and shield buildings and containment internal structure (CIS) has been performed for each of these regions. Specific areas within each wall or floor where out-of-plane modes, which may respond to either CSDRS or HRHF input (including structures with modes less than 33 Hz and between 33 Hz to 50 Hz), have been identified. The survey reveals that some regions, typically in the middle of a floor or wall, exhibit amplified behavior compared to the critical nodes at the corner and edge building locations. The amplified FRS for these regions is generated in addition to the typical set of critical nodes for building analysis by a single time history analysis of the NI05 building model subject to the HRHF time history input. Seismic response spectra for each of the “flexible” nodes are considered when selecting the pre-existing “group” spectra, which is the envelope of the entire floor in that area.

Evaluation of incoherent HRHF spectra has been performed. The CSDRS and HRHF seismic responses were compared with coherent and incoherent considerations at a number of locations in the nuclear island. There are some exceedances, mostly above the 15 hertz region, and these are typical of the plant comparative responses. The steel containment vessel (SCV) was excluded from the evaluation because the HRHF spectra at the base of the SCV are enveloped by the AP1000 CSDRS spectra at the base of the SCV.

Structures designed to the CSDRS input are adequately designed for the HRHF input because the HRHF coherent results are enveloped by the CSDRS results.

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

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

Evaluation of incoherent NI FIRS has been performed. In-structure response spectra for the AP1000 CSDRS, incoherent HRHF spectra and the incoherent NI FIRS were compared at a number of

locations/elevations in the Nuclear Island. Several minor exceedances were noted that are addressed as part of the sampling evaluation outlined in Subsection 3I.6.

3I.4 Evaluation Methodology

The demonstration that the AP1000 nuclear power plant is qualified for the high frequency seismic response does not require the analysis of the total plant. The evaluations made are of representative systems, structures, and components, selected by screening, as potentially sensitive to high frequency input in locations where there were exceedances in the high frequency region. Acceptability of this sample is considered sufficient to demonstrate that the AP1000 is qualified.

The high frequency seismic analyses that are performed use time history or broadened response spectra. The analysis is not performed using the combination spectra of the CSDRS and the HRHF envelope response spectra. Separate analyses with each spectra are used.

The high frequency seismic analyses used the soil-structure interaction code ACS SASSI. The results presented in this report are based on the stochastic (multiple, statistical analyses) seismic incoherent soil-structure interaction analysis approach referred herein as the simulation approach.

The evaluations performed assess the ability of the system, structure, or component to maintain its safety function.

Supplementary analyses are performed as needed to show that high frequency floor response spectra exceedances are not damaging. These analyses can include: gap nonlinearities; material inelastic behavior; multi point response spectra analyses where the high frequency response excites a local part of the system. Tests on equipment are specified as needed where function cannot be demonstrated by analysis, or analysis is not appropriate.

3I.5 General Selection Screening Criteria

The following general screening criteria are used to identify representative AP1000 systems, structures, and components (SSCs) for the samples to be evaluated to demonstrate acceptability of the AP1000 nuclear power plant for the high frequency motion.

- Select systems, structures, and components based on their importance to safety. This includes the review of component safety function for the SSE event and its potential failure modes due to an SSE. Those components whose failure modes would result in safe shutdown are excluded.

- Select systems, structures, and components that are located in areas of the plant that experience large high frequency seismic response.

- Select systems, structures, and components that have significant modal response within the region of high frequency amplification. Significance is defined by such items as modal mass; participation factor, stress and/or deflection.

- Select systems, structures, and components that have significant stress as compared to allowable when considering load combinations that include seismic.

3I.6 Evaluation

In this section the portions of structures, the components, and the systems that are evaluated for the high frequency seismic response are identified. The sample to be evaluated, based on the screening criteria applicable to the SSCs consists of the following:

- Building Structures
  - Auxiliary Building – 3 locations
  - Shield Building – 8 locations
  - CIS – 2 locations
- Primary Coolant Loop
  - Reactor Vessel and Internals
  - Primary Component Supports
  - Reactor Coolant Loop Primary Equipment Nozzles
- Piping Systems – ASME Class 1, 2, and 3 piping systems will be evaluated for the HRHF GMRS. This evaluation is within the scope of the piping DAC (see COL Information Item 3.9-7).
- Electro-Mechanical Equipment – Equipment that is potentially sensitive to high frequency input (see Table 3I.6-1)

These structures, systems, and equipment are discussed in more detail in the sections that follow.

As described in Lee Nuclear Station site-specific Technical Report WLG-GW-GLR-815 (Reference 201), all exceedances of the in-structure response spectra resulting from the Lee Nuclear Station NI FIRS input were identified. Three instances of largest in-structure response spectra exceedances were investigated as bounding conditions and justified by further evaluation. Therefore, the sample of structures, systems and components selected for evaluation remains unchanged.

3I.6.1 Building Structures

Maintaining the NI buildings structural integrity is important to the safety of the plant. Representative portions of the buildings that are evaluated for the effect of high frequency input are selected based on those areas that can experience high seismic shear and moment loads due to the seismic event. Areas chosen are at the base of the shield building, in the vicinity of auxiliary building floors that have fundamental frequencies in the high frequency region, and the corners of the auxiliary building. Three locations are selected on the auxiliary building that reflect the bottom of a wall where the shear and moment would be large, a wall in the vicinity of a floor that is influenced by high frequency response, and a corner intersection of walls. Eight locations are evaluated on the shield building. Four at elevation 107' and four at elevation 211'. These locations are located on the east, west, north and south sides. The south-west wall of the refueling canal is evaluated since it is a representative wall on the refueling canal. The CA02 wall in the CIS building is evaluated since it is a representative wall associated with the IRWST.

The evaluation consists of a comparison of the loads from the high frequency input to those obtained from the AP1000 design spectra, shown in Figures 3I.1-1 and 3I.1-2, for these representative building structures. The NI building structures are considered qualified for the high frequency input if the seismic loads from the Regulatory Guide 1.60 (modified) envelope those from the high frequency input. If there is any exceedance, this is evaluated further to confirm that the existing design is adequate.

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

3I.6.2 Primary Coolant Loop

A failure within the reactor coolant loop could challenge the integrity of the reactor coolant pressure boundary. Therefore, it is chosen for evaluation. The components evaluated are as follows:

- Reactor vessel and internals
- Reactor vessel supports
- Steam generator supports
- Reactor coolant loop primary equipment nozzles

The reactor vessel and internals are selected since they are important to safety and their analysis is representative of major primary components. The building structure below the reactor vessel supports is fairly stiff and there may be significant vertical amplification at the supports of the reactor pressure vessel. Further, reactor vessel internals have relatively complex structural systems including gap nonlinearities and sliding elements. Also, they may be sensitive to high frequency input as summarized below:

- Vertical and horizontal modes of the upper internals and the reactor vessel modes are in the relatively high frequency range.

- Additional high frequencies are associated with nonlinear impact

The evaluation consists of a comparison of the loads from the high frequency input to those obtained from the Regulatory Guide 1.60 (modified) input. Qualification is shown for the high frequency input if the seismic loads from the Regulatory Guide 1.60 (modified) envelope those from the high frequency input. If there is exceedance, then comparison is made for the combination of the seismic with the design basis pipe break loads and steady state loads. Qualification is then shown if the high frequency loads are relatively insignificant compared to the other loads, or there are no required design changes.

Maintaining the integrity of the reactor vessel and steam generator supports is important to preserving the primary component safety function. They are representative of supports on components, and see high loads.

The reactor coolant loop nozzles at the cold and hot leg interfaces of the reactor pressure vessel, reactor coolant pumps, and steam generators are important to include in the evaluation since these are critical areas of components.

The evaluation of the primary component supports and reactor coolant loop nozzles consists of a comparison of the loads from the high frequency input to those obtained from the Regulatory Guide 1.60 (modified) input. These items are considered qualified for the high frequency input if the seismic loads from the Regulatory Guide 1.60 (modified) envelope those from the high frequency input. If there is any exceedance, then an evaluation is made combining the high frequency loads with the other load components (e.g., thermal, pressure, dead) and a comparison made to the design loads. If the design loads envelope the load combinations that include the high frequency seismic input, then the nozzles and supports are considered qualified for the high frequency input.

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

3I.6.3 Piping Systems

ASME Class 1, 2, and 3 piping systems will be evaluated for the HRHF GMRS. This evaluation is within the scope of the piping DAC (see COL Information Item 3.9-7).

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

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

The stress results of the sample piping analysis packages show that the AP1000 HRHF stresses were greater than the NI FIRS stresses for all nodes in the ADS 4th Stage and SFS Aux. Building 4 to 6 piping packages and only slightly less in the Pressurizer Surge Line piping package. Stress comparison results show that AP1000 CSDRS stresses are greater than the NI FIRS stresses at all nodes in all three piping packages except for one node in the SFS Aux. Building 4 to 6 piping package where there was a slight NI FIRS exceedance. At this one point, the stresses resulting from the NI FIRS were less than those from the HRHF spectra. Therefore, the design practices for standard plant AP1000 piping systems have considered cases that envelope the Lee site-specific requirements.

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

3I.6.4 Electrical and Electro-Mechanical Equipment

The groups of safety-related equipment considered for evaluation are those that may be sensitive to the high frequency input. This includes those cabinet-mounted equipment, field sensors, and appurtenants that may be sensitive to high frequency seismic inputs identified in Table 3I.6-1.

Sample safety-related cabinets have been identified that are typically sensitive to seismic input. Evaluations were performed to verify these cabinets do not have excessive seismic excitation on their mounted equipment, the cabinet designs do not require changes due to the high frequency input, and the cabinets will maintain their structural integrity during the high frequency input. Time history analyses of these cabinets were performed for both the Regulatory Guide 1.60 (modified) and the high frequency inputs so that comparisons can be made to their seismic response from both seismic inputs. This analytical study reported in APP-GW-GLR-115 (Reference 2) concluded that safety-related equipment may be screened.

The AP1000 HRHF screening program for determination and evaluation of potential high frequency sensitive equipment is in compliance with the NRC requirements in Section 4.0, “Identification and Evaluation of HF Sensitive Mechanical and Electrical Equipment/Components,” of COL/DC-ISG-1 (Reference 3). The AP1000 HRHF screening program is also consistent with the guidelines developed as part of an industry review document in the EPRI White Paper, “Seismic Screening of Components Sensitive to High Frequency Vibratory Motions” (Reference 4), transmitted to the NRC on June 28, 2007, for determining the safety-related equipment and components that may be HRHF sensitive, and screening procedures to ensure that any safety-related equipment and components

sensitive to HRHF seismic excitation are screened out. This industry review of HF exceedance and further evaluations of SSCs performed by Westinghouse concluded that HRHF envelope response spectra are less harmful than the CSDRS except for the functionality of potential HRHF-sensitive components.

The AP1000 HRHF screening program is based on an HF evaluation study reported in APP-GW-GLR-115 (Reference 2). The HF evaluation study concluded that AP1000 In-Structure Response Spectra (ISRS) developed from the AP1000 CSDRS would, in the majority of cases, produce equipment stress results of the same magnitude or higher than the stress results produced from HRHF seismic excitation. The exception to this condition is when the dominant natural frequency of the equipment is in the HRHF exceedance range and there can be significantly more response because the frequency coincides with the input driving force. Under this condition, forces/stresses generated in the equipment could be due to the acceleration exceedance; therefore, the equipment will be subjected to HRHF seismic evaluation/testing to screen out equipment by verifying its performance and acceptability under HRHF excitation. Review of seismic test data for electrical and microprocessor based cabinets performed to generic and high frequency excitation concluded that seismic testing that peaks in the lower frequency range will produce larger displacements and velocities, and will result in higher stresses in the equipment.

The goal of the AP1000 HRHF screening program is to identify the potential safety-related equipment and components that have the potential to be HRHF-sensitive and show them to be acceptable for their specific application (screened-out). The AP1000 HRHF screening program is a two step process. The first step is an HRHF susceptibility review to identify potential high frequency sensitive safety-related equipment. The second step is the screened-out equipment process to demonstrate its acceptability for the HRHF seismic excitation. Evaluation of screened-in equipment as defined in COL/DC-ISG-1 (Reference 3) is not performed because all safety-related equipment that is screened-in will be eliminated or shown to be acceptable through a design change process.

For the AP1000 HRHF screening program, the following conditions must exist:

1. Plant-specific HRHF GMRS exceeds the AP1000 CSDRS in the high frequency range at 5% critical damping.

2. Safety-related equipment has potential failure modes involving change of state, chatter, signal change/drift, and connection problems.

Table 3I.6-2 is a list of potential HRHF-sensitive AP1000 safety-related equipment developed based on Table 3.11-1 of Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment.” The equipment in Table 3.2-3 of Section 3.2, “AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment,” and Table 3I.6-3 is not HRHF-sensitive. The structural integrity and operability of equipment in Table 3.2-3 and Table 3I.6-3 will not be impacted by the high frequency excitation.

The HRHF susceptibility review of AP1000 safety-related equipment is not performed for potential failure modes associated with mounting, connections, fasteners, joints, and structural interface. These potential failure modes are addressed through the seismic qualification of the safety-related equipment to the AP1000 ISRS testing performed in compliance with IEEE Standard 344-1987. The AP1000 ISRS qualification testing generates higher displacements and velocities than those resulting from HRHF seismic excitation since the AP1000 ISRS is controlled by the lower frequency range. The higher displacement, velocities, and accelerations will detect these equipment structural failure modes if they exist.

At locations where HRHF response spectra show exceedance of the CSDRS and there is a likelihood of equipment damage, further evaluations would be performed to verify that the existing qualification

is adequate for equipment not high frequency sensitive, as listed in Table 3I.6-3, under the following conditions:

- Safety-related equipment must have modes or natural frequencies in the range of interest.
- Evaluation will apply the same acceptance criteria and methodologies used in CSDRS qualification.

To demonstrate acceptability for both CSDRS and HRHF testing, the test response spectra must envelop the CSDRS and HRHF spectra, respectively, with margin over the frequency range of interest in compliance with IEEE Standard 344-1987. In the event that the CSDRS and/or HRHF response spectra would be revised after the qualification program has been completed, a reconciliation effort would be performed to verify that the CSDRS and HRHF testing is still valid. The reconciliation effort may result in requalification activities and qualification documentation revisions.

High Frequency Screening Process – Step 1

The potential failure modes of high frequency sensitive component types and assemblies are important considerations in the high frequency program. The following are potential failure modes of high frequency sensitive components/equipment:

- Inadvertent change of state
- Chatter
- Change in accuracy and drift in output signal or set-point
- Electrical connection failure or intermediacy (e.g., poor quality solder joints)
- Mechanical connection failure
- Mechanical misalignment/binding (e.g., latches, plungers)
- Fatigue failure (e.g., solder joints, ceramics, self-taping screws, spot welds)
- Improperly and unrestrained mounted components
- Inadequately secured/locked mechanical fasteners and connections

Components and equipment determined to have potential failure modes involve change of state, chatter, signal change/drift, and connection problems will be demonstrated to be acceptable through the performance of a supplemental high frequency screening test. Those high frequency sensitive components having failure modes associated with mounting, connections and fasteners, joints, and interface are considered to be acceptable as a result of the AP1000 ISRS qualification testing per IEEE Standard 344-1987 and/or require quality assurance inspection and process/design controls.

High Frequency Screening Process – Step 2

The HRHF susceptibility review is to verify that the subject equipment is capable of performing its safety-related function under HRHF seismic excitation. All AP1000 safety-related equipment will be qualified to the AP1000 ISRS, and the dominant natural frequencies of the equipment will be determined. The EPRI White Paper (Reference 4) identifies the following three evaluation methods to demonstrate that potential HRHF-sensitive safety-related equipment is not HRHF vulnerable:

1. Existing seismic qualification test data for potential high frequency sensitive equipment should be reviewed for applicability and adequacy of the test method to demonstrate sufficient high frequency content.

2. Systems/circuits containing potentially sensitive items should be reviewed for inappropriate/unacceptable system actions due to assumed change of state, contact chatter/intermittency, set point drifts, or loss of calibration.

3. HRHF vibration screening test is conducted to identify any HRHF sensitivities/abnormalities of the components. Several conventional test methods are recommended.

The first and third evaluation methods are part of the AP1000 HRHF screening program and are further detailed below. The AP1000 HRHF seismic screening evaluation will employ the AP1000 HRHF SSE response spectra as input in verifying potential HF sensitive safety-related equipment is not vulnerable to HRHF seismic excitation. Additional seismic test margin will be introduced into the HRHF seismic screening evaluation as needed.

Method 1: Review of Seismic Test Data

Available seismic test data can be used for AP1000 HRHF plant applications when:

- Seismic qualification testing performed on potential HRHF-sensitive safety-related equipment meets as a minimum the AP1000 ISRS in compliance with IEEE Standard 344-1987.

- Safe shutdown earthquake (SSE) test had sufficient energy content in the HRHF region to verify that the safety-related equipment is not vulnerable to HRHF seismic excitation.

No additional seismic testing is required for safety-related equipment previously tested and whose qualification level envelops the HRHF required response spectra (RRS).

IEEE Standard 344-1987 provides guidance to ensure that the seismic test input is generated and in compliance with the frequency range of interest. To demonstrate acceptability for frequency content, it is necessary to show that the frequency content of the test waveform is at least as broad as the frequency content of the amplified region of the RRS except at the low frequencies where non-enveloping is permitted under certain conditions (refer to IEEE Standard 344-1987 subclauses 7.6.3.1(10) and 7.6.3.1(13)). An evaluation of the test input waveform should be conducted per IEEE Standard 344-1987 Annex B to verify the test data has sufficient content over the frequency range of interest throughout the input time history. If an evaluation of the test input is performed, and the data demonstrates sufficient frequency content in the high frequency range throughout the time history, then the data is acceptable.

Method 3: HRHF Screening Test

The HRHF screening test is a supplemental test to the required seismic qualification methods performed in accordance with IEEE Standard 344-1987 for those plants that have high frequency exceedance of the AP1000 CSDRS. The purpose of the HRHF screening test is to demonstrate that the potential HRHF-sensitive safety-related equipment will perform its safety-related function as required under HRHF seismic excitation. The HRHF screening test is performed in conjunction with the AP1000 ISRS seismic qualification testing, or it is performed as a supplemental test after completion of the AP1000 ISRS seismic qualification testing. The AP1000 ISRS and HRHF test input time histories have 30-second durations with frequency content up to the cutoff frequency developed in accordance with subclause 7.6.3 (Multiple-Frequency Tests) and Annex B (Frequency Content and Stationarity) of IEEE Standard 344-1987. During the AP1000 ISRS and HRHF testing, the equipment will be functional and monitored to verify the safety-related function was demonstrated. Screening testing will be performed using HRHF response spectra as defined in the EPRI White Paper (Reference 4) when AP1000 HRHF inputs are not available. The HRHF response spectra will be generated based on the 5g and 15g peak spectral acceleration at 5% critical damping in the 25 Hz to 50 Hz frequency range. If the HRHF screening test cannot demonstrate the equipment to be acceptable, then the safety-related equipment will be removed or modified and additional testing or justification will be required.

The AP1000 safety-related equipment will be seismic qualified to the AP1000 ISRS associated with the mounting location of the equipment as a minimum. Seismic qualification testing will consist of five

AP1000 ISRS operating basis earthquakes (OBEs) followed by one SSE as a minimum. The OBE level will be at least one-half the SSE level. The OBE testing is used to vibration age and address low-cycle fatigue of equipment prior to SSE testing. Cyclic fatiguing of equipment for HRHF exceedance area is adequately addressed by performing five OBE (one-half the SSE) and a minimum of one SSE seismic test runs in compliance with IEEE Standard 344-1987 prior to performing the supplemental HRHF screening test. Additional OBE testing in the high frequency exceedance range is adequately addressed by the demonstration that the peak stress cycles required for five one-half SSE events using the AP1000 HRHF ISRS are equivalent to or enveloped by the peak stress cycles resulting from five one-half SSE events and one full SSE event using the AP1000 CSD ISRS.

The test results of AP1000 seismic qualification programs with multiple operational states (for example, relays have three possible operational states: de-energized, energized, and change of state) will be used to determine the most sensitive equipment electrical operational state. The HRHF test run is performed on the equipment in its most sensitive electrical operational state to demonstrate its safety-related function under HRHF seismic excitation. If this is not possible, additional HRHF screening tests will be performed as needed to address the other most sensitive electrical operation states.

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

3I.7 References

1. EPRI Draft White Paper, “Considerations for NPP Equipment and Structures Subjected to Response Levels Caused by High Frequency Ground Motions,” Transmitted to NRC March 19, 2007.

2. APP-GW-GLR-115, “Effect of High Frequency Seismic Content on SSCs,” Westinghouse Electric Company LLC.

3. COL/DC-ISG-1, “Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion,” May 19, 2008.

4. EPRI White Paper, “Seismic Screening of Components Sensitive to High Frequency Vibratory Motions,” June 2007.

5. Letter, R. Sisk (Westinghouse) to NRC, “AP1000 Response to Request for Additional Information (SRP3.10),” DCP/NRC2280, October 17, 2008.

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

Table 3I.6-1
Potential High Frequency Sensitive Equipment List

• Equipment or components with moving parts and required to perform a switching function during the seismic event (e.g., circuit breakers, contactors, auxiliary switches, molded case circuit breakers, motor control center starters, and pneumatic control assemblies) • Components with moving parts that may bounce or chatter such as relays and actuation devices (e.g., shunt trips) • Unrestrained components • Potentiometers • Process switches and sensors (e.g., pressure/differential pressure, temperature, level, limit/position, and flow) • Components with accuracy requirements that may drift due to seismic loading • Interfaces such as secondary contacts – Connectors and connections (including circuit board connections for digital and analog equipment)

Table 3I.6-2 (Sheet 1 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Batteries	
IDSA 125V 60 Cell Battery 1A	IDSA-DB-1A
IDSA 125V 60 Cell Battery 1B	IDSA-DB-1B
IDSB 125V 60 Cell Battery 1A	IDSB-DB-1A
IDSB 125V 60 Cell Battery 1B	IDSB-DB-1B
IDSB 125V 60 Cell Battery 2A	IDSB-DB-2A
IDSB 125V 60 Cell Battery 2B	IDSB-DB-2B
IDSC 125V 60 Cell Battery 1A	IDSC-DB-1A
IDSC 125V 60 Cell Battery 1B	IDSC-DB-1B
IDSC 125V 60 Cell Battery 2A	IDSC-DB-2A
IDSC 125V 60 Cell Battery 2B	IDSC-DB-2B
IDSD 125V 60 Cell Battery 1A	IDSD-DB-1A
IDSD 125V 60 Cell Battery 1B	IDSD-DB-1B
Spare 125V 60 Cell Battery 1A	IDSS-DB-1A
Spare 125V 60 Cell Battery 1B	IDSS-DB-1B
Battery Chargers	
IDSA Battery Charger	IDSA-DC-1
IDSB Battery Charger	IDSB-DC-1
IDSB Battery Charger 2	IDSB-DC-2
IDSC Battery Charger 1	IDSC-DC-1
IDSC Battery Charger 2	IDSC-DC-2
IDSD Battery Charger	IDSD-DC-1
Spare Battery Charger	IDSS-DC-1

Table 3I.6-2 (Sheet 2 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Distribution Panels	
IDSA 250 Vdc Dist Panel	IDSA-DD-1
IDSB 250 Vdc Dist Panel	IDSB-DD-1
IDSC 250 Vdc Dist Panel	IDSC-DD-1
IDSD 250 Vdc Dist Panel	IDSD-DD-1
IDSA 120 Vac Dist Panel 1	IDSA-EA-1
IDSA 120 Vac Dist Panel 2	IDSA-EA-2
IDSB 120 Vac Dist Panel 1	IDSB-EA-1
IDSB 120 Vac Dist Panel 2	IDSB-EA-2
IDSB 120 Vac Dist Panel 3	IDSB-EA-3
IDSC 120 Vac Dist Panel 1	IDSC-EA-1
IDSC 120 Vac Dist Panel 2	IDSC-EA-2
IDSC 120 Vac Dist Panel 3	IDSC-EA-3
IDSD 120 Vac Dist Panel 1	IDSD-EA-1
IDSD 120 Vac Dist Panel 2	IDSD-EA-2
Fuse Panels	
IDSA Fuse Panel	IDSA-EA-4
IDSB Fuse Panel	IDSB-EA-4
IDSB Fuse Panel	IDSB-EA-5
IDSB Fuse Panel	IDSB-EA-6
IDSC Fuse Panel	IDSC-EA-4
IDSC Fuse Panel	IDSC-EA-5
IDSC Fuse Panel	IDSC-EA-6
IDSD Fuse Panel	IDSD-EA-4

Table 3I.6-2 (Sheet 3 of 28)
List of Potential High Frequency Sensitive AP1000 Safety-Related electrical and Electro-mechanical Equipment

Description	AP1000 Tag Number
Transfer Switches	
IDSA Fused Transfer Switch Box 1	IDSA-DF-1
IDSB Fused Transfer Switch Box 1	IDSB-DF-1
IDSB Fused Transfer Switch Box 2	IDSB-DF-2
IDSC Fused Transfer Switch Box 1	IDSC-DF-1
IDSC Fused Transfer Switch Box 2	IDSC-DF-2
IDSD Fused Transfer Switch Box 1	IDSD-DF-1
IDSS Fused Transfer Switch Box 1	IDSS-DF-1
Spare Battery 125/250 Vdc Disconnect Switch	IDSS-SW-1
IDSS Spare Termination Box	IDSS-DF-2
IDSS Spare Termination Box	IDSS–DF-3
IDSS Spare Termination Box	IDSS-DF-4
IDSS Spare Termination Box	IDSS-DF-5
IDSS Spare Termination Box	IDSS-DF-6
Motor Control Centers	
IDSA 250 Vdc MCC	IDSA-DK-1
IDSB 250 Vdc MCC	IDSB-DK-1
IDSC 250 Vdc MCC	IDSC-DK-1
IDSD 250 Vdc MCC	IDSD-DK-1
Switchboards	
IDSA 250 Vdc Switchboard 1	IDSA-DS-1
IDSB 250 Vdc Switchboard 1	IDSB-DS-1
IDSB 250 Vdc Switchboard 2	IDSB-DS-2
IDSC 250 Vdc Switchboard 1	IDSC-DS-1
IDSC 250 Vdc Switchboard 2	IDSC-DS-2
IDSD 250 Vdc Switchboard 1	IDSD-DS-1

Table 3I.6-2 (Sheet 4 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Transformers	
IDSA Regulating Transformer 1	IDSA-DT-1
IDSB Regulating Transformer 1	IDSB-DT-1
IDSC Regulating Transformer 1	IDSC-DT-1
IDSD Regulating Transformer 1	IDSD-DT-1
Inverters	
IDSA Inverter	IDSA-DU-1
IDSB Inverter 1	IDSB-DU-1
IDSB Inverter 2	IDSB-DU-2
IDSC Inverter 1	IDSC-DU-1
IDSC Inverter 2	IDSC-DU-2
IDSD Inverter	IDSD-DU-1
Switchgear	
RCP 1A 6900V Switchgear 31	ECS-ES-31
RCP 1A 6900V Switchgear 32	ECS-ES-32
RCP 2A 6900V Switchgear 51	ECS-ES-51
RCP 2A 6900V Switchgear 52	ECS-ES-52
RCP 1B 6900V Switchgear 41	ECS-ES-41
RCP 1B 6900V Switchgear 42	ECS-ES-42
RCP 2B 6900V Switchgear 61	ECS-ES-61
RCP 2B 6900V Switchgear 62	ECS-ES-62
Reactor Trip Switchgear	PMS-JD-RTSA01
Reactor Trip Switchgear	PMS-JD-RTSA02
Reactor Trip Switchgear	PMS-JD-RTSB01
Reactor Trip Switchgear	PMS-JD-RTSB02

Table 3I.6-2 (Sheet 5 of 28)
List of Potential High Frequency Sensitive AP1000 Safety-Related electrical and Electro-mechanical Equipment

Description	AP1000 Tag Number
Reactor Trip Switchgear	PMS-JD-RTSC01
Reactor Trip Switchgear	PMS-JD-RTSC02
Reactor Trip Switchgear	PMS-JD-RTSD01
Reactor Trip Switchgear	PMS-JD-RTSD02
Level Switches	
Core Makeup Tank A Narrow Range	PXS-JE-LS011A
Core Makeup Tank A Narrow Range	PXS-JE-LS011B
Core Makeup Tank A Narrow Range	PXS-JE-LS011C
Core Makeup Tank A Narrow Range	PXS-JE-LS011D
Core Makeup Tank B Narrow Range	PXS-JE-LS012A
Core Makeup Tank B Narrow Range	PXS-JE-LS012B
Core Makeup Tank B Narrow Range	PXS-JE-LS012C
Core Makeup Tank B Narrow Range	PXS-JE-LS012D
Core Makeup Tank A Narrow Range	PXS-JE-LS013A
Core Makeup Tank A Narrow Range	PXS-JE-LS013B
Core Makeup Tank A Narrow Range	PXS-JE-LS013C
Core Makeup Tank A Narrow Range	PXS-JE-LS013D
Core Makeup Tank B Narrow Range	PXS-JE-LS014A
Core Makeup Tank B Narrow Range	PXS-JE-LS014B
Core Makeup Tank B Narrow Range	PXS-JE-LS014C
Core Makeup Tank B Narrow Range	PXS-JE-LS014D
Containment Floodup Level	PXS-JE-LS050
Containment Floodup Level	PXS-JE-LS051
Containment Floodup Level	PXS-JE-LS052

Table 3I.6-2 (Sheet 6 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Neutron Detectors	
Source Range Neutron Detector	RXS-JE-NE001A
Source Range Neutron Detector	RXS-JE-NE001B
Source Range Neutron Detector	RXS-JE-NE001C
Source Range Neutron Detector	RXS-JE-NE001D
Intermediate Range Neutron Detector	RXS-JE-NE002A
Intermediate Range Neutron Detector	RXS-JE-NE002B
Intermediate Range Neutron Detector	RXS-JE-NE002C
Intermediate Range Neutron Detector	RXS-JE-NE002D
Power Range Neutron Detector (Lower)	RXS-JE-NE003A
Power Range Neutron Detector (Lower)	RXS-JE-NE003B
Power Range Neutron Detector (Lower)	RXS-JE-NE003C
Power Range Neutron Detector (Lower)	RXS-JE-NE003D
Power Range Neutron Detector (Upper)	RXS-JE-NE004A
Power Range Neutron Detector (Upper)	RXS–JE-NE004B
Power Range Neutron Detector (Upper)	RXS-JE-NE004C
Power Range Neutron Detector (Upper)	RXS-JE-NE004D
Radiation Monitors	
Containment High Range Area Monitor	PXS-JE-RE160
Containment High Range Area Monitor	PXS-JE-RE161
Containment High Range Area Monitor	PXS-JE-RE162
Containment High Range Area Monitor	PXS-JE-RE163
Control Room Supply Air Area Monitor	VBS-JE-RE001A
Control Room Supply Air Area Monitor	VBS-JE-RE001B

Table 3I.6-2 (Sheet 7 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Speed Sensors	
RCP 1A Pump Speed	RCS-JE-ST281
RCP 1B Pump Speed	RCS-JE-ST282
RCP 2A Pump Speed	RCS-JE-ST283
RCP 2B Pump Speed	RCS-JE-ST284
Transmitters	
PCS Water Delivery Flow	PCS-JE-FT001
PCS Water Delivery Flow	PCS-JE-FT002
PCS Water Delivery Flow	PCS-JE-FT003
PCS Water Delivery Flow	PCS-JE-FT004
PCS Storage Tank Water Level	PCS-JE-LT010
PCS Storage Tank Water Level	PCS-JE-LT011
PRHR HX Flow	PXS-JE-FT049A
PRHR HX Flow	PXS-JE-FT049B
RCS Hot Leg 1 Flow	RCS-JE-FT101A
RCS Hot Leg 1 Flow	RCS-JE-FT101B
RCS Hot Leg 1 Flow	RCS-JE-FT101C
RCS Hot Leg 1 Flow	RCS-JE-FT101D
RCS Hot Leg 2 Flow	RCS-JE-FT102A
RCS Hot Leg 2 Flow	RCS-JE-FT102B
RCS Hot Leg 2 Flow	RCS-JE-FT102C
RCS Hot Leg 2 Flow	RCS-JE-FT102D
SG1 Startup Feedwater Flow	SGS-JE-FT055A
SG1 Startup Feedwater Flow	SGS-JE-FT055B
SG2 Startup Feedwater Flow	SGS-JE-FT-056A

Table 3I.6-2 (Sheet 8 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
SG2 Startup Feedwater Flow	SGS-JE-FT056B
MCR Air Delivery Line Flow Rate – A	VES-JE-FT003A
MCR Air Delivery Line Flow Rate – B	VES-JE-FT003B
Plant Vent Flow	VFS-JE-FT101
IRWST Level	PXS-JE-LT045
IRWST Level	PXS-JE-LT046
IRWST Level	PXS-JE-LT047
IRWST Level	PXS-JE-LT048
RCS Hot Leg Water Level	RCS-JE-LT160A
RCS Hot Leg Water Level	RCS-JE-LT160B
PZR Level	RCS-JE-LT195A
PZR Level	RCS-JE-LT195B
PZR Level	RCS-JE-LT195C
PZR Level	RCS-JE-LT195D
SG1 Narrow Range Level	SGS-JE-LT001
SG1 Narrow Range Level	SGS-JE-LT002
SG1 Narrow Range Level	SGS-JE-LT003
SG1 Narrow Range Level	SGS-JE-LT004
SG2 Narrow Range Level	SGS-JE-LT005
SG2 Narrow Range Level	SGS-JE-LT006
SG2 Narrow Range Level	SGS-JE-LT007
SG2 Narrow Range Level	SGS-JE-LT008
SG1 Wide Range Level	SGS-JE-LT011
SG1 Wide Range Level	SGS-JE-LT012
SG1 Wide Range Level	SGS-JE-LT015

Table 3I.6-2 (Sheet 9 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
SG1 Wide Range Level	SGS-JE-LT016
SG2 Wide Range Level	SGS-JE-LT013
SG2 Wide Range Level	SGS-JE-LT014
SG2 Wide Range Level	SGS-JE-LT017
SG2 Wide Range Level	SGS-JE-LT018
Spent Fuel Pool Level	SFS-JE-LT019A
Spent Fuel Pool Level	SFS-JE-LT019B
Spent Fuel Pool Level	SFS-JE-LT019C
Air Storage Tank Pressure – A	VES-JE-PT001A
Air Storage Tank Pressure – B	VES-JE-PT001B
Containment Pressure Normal Range	PCS-JE-PT005
Containment Pressure Normal Range	PCS-JE-PT006
Containment Pressure Normal Range	PCS-JE-PT007
Containment Pressure Normal Range	PCS-JE-PT008
Containment Pressure Extended Range	PCS-JE-PT012
Containment Pressure Extended Range	PCS-JE-PT013
Containment Pressure Extended Range	PCS-JE-PT014
RCS Wide Range Pressure	RCS-JE-PT140A
RCS Wide Range Pressure	RCS-JE-PT140B
RCS Wide Range Pressure	RCS-JE-PT140C
RCS Wide Range Pressure	RCS-JE-PT140D
PZR Pressure	RCS-JE-PT191A
PZR Pressure	RCS-JE-PT191B
PZR Pressure	RCS-JE-PT191C
PZR Pressure	RCS-JE-PT191D

Table 3I.6-2 (Sheet 10 of 28)
List of Potential High Frequency Sensitive AP1000 Safety-Related electrical and Electro-mechanical Equipment

Description	AP1000 Tag Number
Main Steam Line SG1 Pressure	SGS-JE-PT030
Main Steam Line SG1 Pressure	SGS-JE-PT031
Main Steam Line SG1 Pressure	SGS-JE-PT032
Main Steam Line SG1 Pressure	SGS-JE-PT033
Main Steam Line SG2 Pressure	SGS-JE-PT034
Main Steam Line SG2 Pressure	SGS-JE-PT035
Main Steam Line SG2 Pressure	SGS-JE-PT036
Main Steam Line SG2 Pressure	SGS-JE-PT037
Main Control Room Differential Pressure	VES-JE-PDT004A
Main Control Room Differential Pressure	VES-JE-PDT004B
Protection and Safety Monitoring Systems	
Protection and Safety Monitoring System Cabinets	Multiple
MCR/RSW Transfer Switch Panel A	PMS-JW-004A
MCR/RSW Transfer Switch Panel B	PMS-JW-004B
MCR/RSW Transfer Switch Panel C	PMS-JW-004C
MCR/RSW Transfer Switch Panel D	PMS-JW-004D
Source Range Neutron Flux Preamplifier Panel A	PMS-JW-005A
Source Range Neutron Flux Preamplifier Panel B	PMS-JW-005B
Source Range Neutron Flux Preamplifier Panel C	PMS-JW-005C
Source Range Neutron Flux Preamplifier Panel D	PMS-JW-005D
Intermediate Range Neutron Flux Preamplifier Panel A	PMS-JW-006A
Intermediate Range Neutron Flux Preamplifier Panel B	PMS-JW-006B
Intermediate Range Neutron Flux Preamplifier Panel C	PMS-JW-006C

Table 3I.6-2 (Sheet 11 of 28)
List of Potential High Frequency Sensitive AP1000 Safety-Related electrical and Electro-mechanical Equipment

Description	AP1000 Tag Number
Intermediate Range Neutron Flux Preamplifier Panel D	PMS-JW-006D
Power Range Neutron Flux High Voltage Distribution Box A	PMS-JW-007A
Power Range Neutron Flux High Voltage Distribution Box B	PMS-JW-007B
Power Range Neutron Flux High Voltage Distribution Box C	PMS-JW-007C
Power Range Neutron Flux High Voltage Distribution Box D	PMS-JW-007D
Main Control Room	
Operator Workstation A	N/A
Operator Workstation B	N/A
Supervisor Workstation	N/A
Switch Station (Including Switches)	N/A
QDPS MCR Display Unit	PMS-JY-001B
QDPS MCR Display Unit	PMS-JY-001C
MCR Load Shed Panel 1	VES-EP-01
MCR Load Shed Panel 2	VES-EP-02
Active Valves	
Containment Isolation – Air Out Solenoid Valve Limit Switch	 CAS-PL-V014-S CAS-PL-V014-L
Containment Isolation – Inlet Limit Switch Motor Operator	 CCS-PL-V200-L CCS-PL-V200-M
Containment Isolation – Outlet Limit Switch Motor Operator	 CCS-PL-V207-L CCS-PL-V207-M
Containment Isolation – Outlet Limit Switch Motor Operator	 CCS-PL-V208-L CCS-PL-V208-M

Table 3I.6-2 (Sheet 12 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
RCS Purification Stop Valve	
Limit Switch	CVS-PL-V001-L
Motor Operator	CVS-PL-V001-M
RCS Purification Stop Valve	
Limit Switch	CVS-PL-V002-L
Motor Operator	CVS-PL-V002-M
RCS Letdown Stop Valve	
Limit Switch	CVS-PL-V003-L
Motor Operator	CVS-PL-V003-M
WLS Letdown IRC Isolation	
Limit Switch	CVS-PL-V045-L
Solenoid Valve	CVS-PL-V045-S1
Letdown Flow ORC Isolation	
Limit Switch	CVS-PL-V047-L
Solenoid Valve	CVS-PL-V047-S1
Auxiliary PZR Spray Isolation	
Limit Switch	CVS-PL-V084-L
Solenoid Valve	CVS-PL-V084-S
Makeup Line Containment Isolation	
Limit Switch	CVS-PL-V090-L
Motor Operator	CVS-PL-V090-M
Makeup Line Containment Isolation	
Limit Switch	CVS-PL-V091-L
Motor Operator	CVS-PL-V091-M

Table 3I.6-2 (Sheet 13 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Hydrogen Addition Containment Isolation	
Limit Switch	CVS-PL-V092-L
Solenoid Valve	CVS-PL-V092-S
Demineralizer Water System Isolation	
Limit Switch	CVS-PL-V136A-L
Solenoid Valve	CVS-PL-V136A-S
Demineralized Water System Isolation	
Limit Switch	CVS-PL-V136B-L
Solenoid Valve	CVS-PL-V136B-S
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V001A-L
Solenoid Valve	PCS-PL-V001A-S1
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V001B-L
Solenoid Valve	PCS-PL-V001B-S1
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V001C-L
Motor Operator	PCS-PL-V001C-M
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V002A-L
Motor Operator	PCS-PL-V002A-M
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V002B-L
Motor Operator	PCS-PL-V002B-M

**Table 3I.6-2 (Sheet 14 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment**

Description	AP1000 Tag Number
PCCWST Isolation Valve	
Limit Switch	PCS-PL-V002C-L
Motor Operator	PCS-PL-V002C-M
Containment Isolation – Air Sample Line	
Limit Switch	PSS-PL-V008-L
Solenoid Operator	PSS-PL-V008-S
Containment Isolation – Liquid Sample Line	
Limit Switch	PSS-PL-V010A-L
Solenoid Operator	PSS-PL-V010A-S
Containment Isolation – Liquid Sample Line	
Limit Switch	PSS-PL-V010B-L
Solenoid Operator	PSS-PL-V010B-S
Containment Isolation – Liquid Sample Line	
Limit Switch	PSS-PL-V011-L
Solenoid Valve	PSS-PL-V011-S
Containment Isolation - Sample Return Line	
Limit Switch	PSS-PL-V023-L
Solenoid Valve	PSS-PL-V023-S
Containment Isolation - Air Sample Line	
Limit Switch	PSS-PL-V046-L
Solenoid Valve	PSS-PL-V046-S
Core Makeup Tank A Discharge Isolation	
Limit Switch	PXS-PL-V014A-L
Solenoid Valve	PXS-PL-V014A-S1

Table 3I.6-2 (Sheet 15 of 28)
List of Potential High Frequency Sensitive AP1000 Safety-Related electrical and Electro-mechanical Equipment

Description	AP1000 Tag Number
Core Makeup Tank B Discharge Isolation	
Limit Switch	PXS-PL-V014B-L
Solenoid Valve	PXS-PL-V014B-S1
Core Makeup Tank A Discharge Isolation	
Limit Switch	PXS-PL-V015A-L
Solenoid Valve	PXS-PL-V015A-S1
Core Makeup Tank B Discharge Isolation	
Limit Switch	PXS-PL-V015B-L
Solenoid Valve	PXS-PL-V015B-S1
Nitrogen Supply Outside Containment Isolation	
Limit Switch	PXS-PL-V042-L
Solenoid Valve	PXS-PL-V042-S
PRHR HX Discharge Isolation	
Limit Switch	PXS-PL-V108A-L
Solenoid Valve	PXS-PL-V108A-S1
PRHR HX Discharge Isolation	
Limit Switch	PXS-PL-V108B-L
Solenoid Valve	PXS-PL-V108B-S1
Recirc Sump A Isolation	
Limit Switch	PXS-PL–V118A-L
Squib Operator	PXS-PL–V118A-T
Recirc Sump B Isolation	
Limit Switch	PXS-PL-V118B-L
Squib Operator	PXS-PL-V118B-T

Table 3I.6-2 (Sheet 16 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Recirc Sump A	
Limit Switch	PXS-PL-V120A-L
Squib Operator	PXS-PL-V120A-T
Recirc Sump B	
Limit Switch	PXS-PL-V120B-L
Squib Operator	PXS-PL-V120B-T
IRWST Injection A	
Limit Switch	PXS-PL-V123A-L
Squib Operator	PXS-PL-V123A-T
IRWST Injection B	
Limit Switch	PXS-PL-V123B-L
Squib Operator	PXS-PL-V123B-T
IRWST Injection A	
Limit Switch	PXS-PL-V125A-L
Squib Operator	PXS-PL-V125A-T
IRWST Injection B	
Limit Switch	PXS-PL-V125B-L
Squib Operator	PXS-PL-V125B-T
IRWST Gutter Drain Isolation A	
Limit Switch	PXS-PL-V130A-L
Solenoid Valve	PXS-PL-V130A-S1
IRWST Gutter Drain Isolation B	
Limit Switch	PXS-PL-V130B-L
Solenoid Valve	PXS-PL-V130B-S1

Table 3I.6-2 (Sheet 17 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
First Stage ADS	
Limit Switch	RCS-PL-V001A-L
Motor Operator	RCS-PL-V001A-M
First Stage ADS	
Limit Switch	RCS-PL-V001B-L
Motor Operator	RCS-PL-V001B-M
Second Stage ADS	
Limit Switch	RCS-PL-V002A-L
Motor Operator	RCS-PL-V002A-M
Second Stage ADS	
Limit Switch	RCS-PL-V002B-L
Motor Operator	RCS-PL-V002B-M
Third Stage ADS	
Limit Switch	RCS-PL-V003A-L
Motor Operator	RCS-PL-V003A-M
Third Stage ADS	
Limit Switch	RCS-PL-V003B-L
Motor Operator	RCS-PL-V003B-M
Fourth Stage ADS	
Limit Switch	RCS-PL-V004A-L
Squib Operator	RCS-PL-V004A-T
Fourth Stage ADS	
Limit Switch	RCS-PL-V004B-L
Squib Operator	RCS-PL-V004B-T

Table 3I.6-2 (Sheet 18 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Fourth Stage ADS	
Limit Switch	RCS-PL-V004C-L
Squib Operator	RCS-PL-V004C-T
Fourth Stage ADS	
Limit Switch	RCS-PL-V004D-L
Squib Operator	RCS-PL-V004D-T
First Stage ADS Isolation	
Limit Switch	RCS-PL-V011A-L
Motor Operator	RCS-PL-V011A-M
First Stage ADS Isolation	
Limit Switch	RCS-PL-V011B-L
Motor Operator	RCS-PL-V011B-M
Second Stage ADS Isolation	
Limit Switch	RCS-PL-V012A-L
Motor Operator	RCS-PL-V012A-M
Second Stage ADS Isolation	
Limit Switch	RCS-PL-V012B-L
Motor Operator	RCS-PL-V012B-M
Third Stage ADS Isolation	
Limit Switch	RCS-PL-V013A-L
Motor Operator	RCS-PL-V013A-M
Third Stage ADS Isolation	
Limit Switch	RCS-PL-V013B-L
Motor Operator	RCS-PL-V013B-M

Table 3I.6-2 (Sheet 19 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Reactor Vessel Head Vent	
Limit Switch	RCS-PL-V150A-L
Solenoid Operator	RCS-PV-V150A-S
Reactor Vessel Head Vent	
Limit Switch	RCS-PL-V150B-L
Solenoid Operator	RCS-PL-V150B-S
Reactor Vessel Head Vent	
Limit Switch	RCS-PL-V150C-L
Solenoid Operator	RCS-PL-V150C-S
Reactor Vessel Head Vent	
Limit Switch	RCS-PL-V150D-L
Solenoid Operator	RCS-PL-V150D-S
RCS Inner Suction Isolation	
Limit Switch	RNS-PL-V001A-L
Motor Operator	RNS-PL-V001A-M
RCS Inner Suction Isolation	
Limit Switch	RNS-PL-V001B-L
Motor Operator	RNS-PL-V001B-M
RCS Outer Suction Isolation	
Limit Switch	RNS-PL-V002A-L
Motor Operator	RNS-PL-V002A-M
RCS Outer Suction Isolation	
Limit Switch	RNS-PL-V002B-L
Motor Operator	RNS-PL-V002B-M

Table 3I.6-2 (Sheet 20 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
RHR Control/Isolation Valve	
Limit Switch	RNS-PL-V011-L
Motor Operator	RNS-PL-V011-M
RHR Pump Suction Header Isolation	
Limit Switch	RNS-PL-V022-L
Motor Operator	RNS-PL-V022-M
IRWST Suction Line Isolation	
Limit Switch	RNS-PL-V023-L
Motor Operator	RNS-PL-V023-M
RNS – CVS Containment Isolation	
Limit Switch	RNS-PL-V061-L
Air Operator	RNS-PL-V061-S
SDS – MCR Isolation	
Limit Switch	SDS-PL-V001-L
Motor Operator	SDS-PL-V001-M
SDS – MCR Isolation	
Limit Switch	SDS-PL-V002-L
Motor Operator	SDS-PL-V002-M
Containment Isolation	
Limit Switch	SFS-PL-V034-L
Motor Operator	SFS-PL–V034-M
Containment Isolation	
Limit Switch	SFS-PL-V035-L
Motor Operator	SFS- PL-V035-M
Containment Isolation	
Limit Switch	SFS-PL-V038-L
Motor Operator	SFS-PL-V038-M

Table 3I.6-2 (Sheet 21 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
PORV Block Valve	
Limit Switch	SGS-PL-V027A-L
Motor Operator	SGS-PL-V027A-M
PORV Block Valve	
Limit Switch	SGS-PL-V027B-L
Motor Operator	SGS-PL-V027B-M
Steam Line Condensate Drain Isolation	
Limit Switch	SGS-PL-V036A-L
Solenoid Valve	SGS-PL-V036A-S
Steam Line Condensate Isolation	
Limit Switch	SGS-PL-V036B-L
Solenoid Valve	SGS-PL-V036B-S
Main Steam Line Isolation	
Limit Switch	SGS-PL-V040A-L
Solenoid Valve	SGS-PL-V040A-S1
Solenoid Valve	SGS-PL-V040A-S2
Solenoid Valve	SGS-PL-V040A-S3
Solenoid Valve	SGS-PL-V040A-S4
Main Steam Line Isolation	
Limit Switch	SGS-PL-V040B-L
Solenoid Valve	SGS-PL-V040B-S1
Solenoid Valve	SGS-PL-V040B-S2
Solenoid Valve	SGS-PL-V040B-S3
Solenoid Valve	SGS-PL-V040B-S4

Table 3I.6-2 (Sheet 22 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Main Feedwater Isolation	
Limit Switch	SGS-PL-V057A-L
Solenoid Valve	SGS-PL-V057A-S1
Solenoid Valve	SGS-PL-V057A-S2
Solenoid Valve	SGS-PL-V057A-S3
Solenoid Valve	SGS-PL-V057A-S4
Main Feedwater Isolation	
Limit Switch	SGS-PL-V057B-L
Solenoid Valve	SGS-PL-V057B-S1
Solenoid Valve	SGS-PL-V057B-S2
Solenoid Valve	SGS-PL-V057B-S3
Solenoid Valve	SGS-PL-V057B-S4
Startup Feedwater Isolation	
Limit Switch	SGS-PL-V067A-L
Motor Operator	SGS-PL-V067A-M
Startup Feedwater Isolation	
Limit Switch	SGS-PL-V067B-L
Motor Operator	SGS-PL-V067B-M
SG Blowdown Isolation	
Limit Switch	SGS-PL-V074A-L
Solenoid Valve	SGS-PL-V074A-S
SG Blowdown Isolation	
Limit Switch	SGS-PL-V074B-L
Solenoid Valve	SGS-PL-V074B-S

Table 3I.6-2 (Sheet 23 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
SG Series Blowdown Isolation	
Limit Switch	SGS-PL-V075A-L
Solenoid Valve	SGS-PL-V075A-S
SG Series Blowdown Isolation	
Limit Switch	SGS-PL-V075B-L
Solenoid Valve	SGS-PL-V075B-S
Steam Line Condensate Drain Isolation Solenoid Valve	SGS-PL-V086A-S
Steam Line Condensate Drain Isolation Solenoid Valve	SGS-PL-V086B-S
Power Operated Relief Valve	
Limit Switch	SGS-PL-V233A-L
Solenoid Valve	SGS-PL-V233A-S
Power Operated Relief Valve	
Limit Switch	SGS-PL-V233B-L
Solenoid Valve	SGS-PL-V233B-S
MSIV Bypass Isolation Valve	
Limit Switch	SGS- PL-V240A-L
Solenoid Valve	SGS-PL-V240A-S1
Solenoid Valve	SGS-PL-V240A-S2
MSIV Bypass Isolation Valve	
Limit Switch	SGS-PL-V240B-L
Solenoid Valve	SGS-PL-V240B-S1
Solenoid Valve	SGS-PL-V240B-S2

Table 3I.6-2 (Sheet 24 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Main Feedwater Control Valve	
Limit Switch	SGS-PL-V250A-L
Solenoid Valve	SGS-PL-V250A-S
Main Feedwater Control Valve	
Limit Switch	SGS-PL-V250B-L
Solenoid Valve	SGS-PL-V250B-S
Startup Feedwater Control Valve	
Limit Switch	SGS-PL-V255A-L
Solenoid Valve	SGS-PL-V255A-S
Startup Feedwater Control Valve	
Limit Switch	SGS-PL-V255B-L
Solenoid Valve	SGS-PL-V255B-S
MCR Isolation Valve	
Limit Switch	VBS-PL-V186-L
Motor Operator	VBS-PL-V186-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V187-L
Motor Operator	VBS-PL-V187-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V188-L
Motor Operator	VBS-PL-V188-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V189-L
Motor Operator	VBS-PL-V189-M

Table 3I.6-2 (Sheet 25 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
MCR Isolation Valve	
Limit Switch	VBS-PL-V190-L
Motor Operator	VBS-PL-V190-M
MCR Isolation Valve	
Limit Switch	VBS-PL-V191-L
Motor Operator	VBS-PL-V191-M
Actuation Valve A	
Limit Switch	VES-PL-V005A-L
Solenoid Operator	VES-PL-V005A-S
Actuation Valve B	
Limit Switch	VES-PL-V005B-L
Solenoid Operator	VES-PL-V005B-S
Relief Isolation Valve A	
Limit Switch	VES-PL-V022A-L
Solenoid Valve	VES-PL-V022A-S
Relief Isolation Valve B	
Limit Switch	VES-PL-V022B-L
Solenoid Valve	VES-PL-V022B-S
Containment Purge Inlet Isolation	
Limit Switch	VFS-PL-V003-L
Solenoid Valve	VFS-PL-V003-S1
Containment Purge Inlet Isolation	
Limit Switch	VFS-PL-V004-L
Solenoid Valve	VFS-PL-V004-S1

Table 3I.6-2 (Sheet 26 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Containment Purge Discharge Isolation	
Limit Switch	VFS-PL-V009-L
Solenoid Valve	VFS-PL-V009-S1
Containment Purge Discharge Isolation	
Limit Switch	VFS-PL-V010-L
Solenoid Valve	VFS-PL-V010-S1
Vacuum Relief Containment Isolation Valve A - ORC	
Limit Switch	VFS-PL-V800A-L
Motor Operator	VFS-PL-V800A-M
Vacuum Relief Containment Isolation Valve B - ORC	
Limit Switch	VFS-PL-V800B-L
Motor Operator	VFS-PL-V800B-M
Fan Cooler Supply Isolation	
Limit Switch	VWS-PL-V058-L
Solenoid Valve	VWS-PL-V058-S
Fan Cooler Return Isolation	
Limit Switch	VWS-PL-V082-L
Solenoid Valve	VWS-PL-V082-S
Fan Cooler Return Isolation	
Limit Switch	VWS-PL-V086-L
Solenoid Valve	VWS-PL-V086-S
Sump Containment Isolation IRC	
Limit Switch	WLS-PL-V055-L
Solenoid Valve	WLS-PL-V055-S1

Table 3I.6-2 (Sheet 27 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Sump Containment Isolation ORC	
Limit Switch	WLS-PL-V057-L
Solenoid Valve	WLS-PL-V057-S1
RCDT Gas Containment Isolation	
Limit Switch	WLS-PL-V067-L
Solenoid Valve	WLS-PL-V067-S
RCDT Gas Containment Isolation	
Limit Switch	WLS-PL-V068-L
Solenoid Valve	WLS-PL-V068-S
Hot Leg 1 Sample Isolation	
Limit Switch	PSS-PL-V001A-L
Hot Leg 2 Sample Isolation	
Limit Switch	PSS-PL-V001B-L
Core Makeup Tank A CL Inlet Isolation	
Limit Switch	PXS-PL-V002A-L
Motor Operator	PXS-PL-V002A-M
Core Makeup Tank B CL Inlet Isolation	
Limit Switch	PXS-PL-V002B-L
Motor Operator	PXS-PL-V002B-M
PRHR HX Inlet Isolation	
Limit Switch	PXS-PL-V101-L
Motor Operator	PXS-PL-V101-M
Recirc Sump A Isolation	
Limit Switch	PXS-PL-V117A-L
Motor Operator	PXS-PL-V117A-M

Table 3I.6-2 (Sheet 28 of 28)
List of Potential High Frequency Sensitive
AP1000 Safety-Related electrical and
Electro-mechanical Equipment

Description	AP1000 Tag Number
Recirc Sump B Isolation	
Limit Switch	PXS-PL-V117B-L
Motor Operator	PXS-PL-V117B-M
Fourth Stage ADS Isolation	
Limit Switch	RCS-PL-V014A-L
Motor Operator	RCS-PL-V014A-M
Fourth Stage ADS Isolation	
Limit Switch	RCS-PL-V014B-L
Motor Operator	RCS-PL-V014B-M
Fourth Stage ADS Isolation	
Limit Switch	RCS-PL-V014C-L
Motor Operator	RCS-PL-V014C-M
Fourth Stage ADS Isolation	
Limit Switch	RCS-PL-V014D-L
Motor Operator	RCS-PL-V014D-M

Table 3I.6-3 (Sheet 1 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Resistance Temperature Detectors		
PRHR HX Outlet Temperature	RCS-JE-TE161	1
RCS Cold Leg 1A Narrow Range Temperature	RCS-JE-TE121A	1
RCS Cold Leg 1A Narrow Range Temperature	RCS-JE-TE121D	1
RCS Cold Leg 1B Narrow Range Temperature	RCS-JE-TE121B	1
RCS Cold Leg 1B Narrow Range Temperature	RCS-JE-TE121C	1
RCS Cold Leg 2A Narrow Range Temperature	RCS-JE-TE122B	1
RCS Cold Leg 2A Narrow Range Temperature	RCS-JE-TE122C	1
RCS Cold Leg 2B Narrow Range Temperature	RCS-JE-TE122A	1
RCS Cold Leg 2B Narrow Range Temperature	RCS-JE-TE122D	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE131A	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE131C	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE132A	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE132C	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE133C	1
RCS Hot Leg 1 Narrow Range Temperature	RCS-JE-TE133A	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE131B	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE131D	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE132B	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE132D	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE133B	1
RCS Hot Leg 2 Narrow Range Temperature	RCS-JE-TE133D	1
RCS Cold Leg 1A Dual Range Temperature	RCS-JE-TE125A	1
RCS Cold Leg 1B Dual Range Temperature	RCS-JE-TE125C	1
RCS Cold Leg 2A Dual Range Temperature	RCS-JE-TE125B	1

Table 3I.6-3 (Sheet 2 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
RCS Cold Leg 2B Dual Range Temperature	RCS-JE-TE125D	1
RCS Hot Leg 1 Wide Range Temperature	RCS-JE-TE135A	1
RCS Hot Leg 2 Wide Range Temperature	RCS-JE-TE135B	1
PZR Reference Leg Level Temperature	RCS-JE-TE193A	1
PZR Reference Leg Level Temperature	RCS-JE-TE193B	1
PZR Reference Leg Level Temperature	RCS-JE-TE193C	1
PZR Reference Leg Level Temperature	RCS-JE-TE193D	1
Thermocouples		1
Incore Thermocouples	IIS-JE-TE001-TE042	1
RCP 1A Bearing Water Temperature	RCS-JE-TE211A	1
RCP 1A Bearing Water Temperature	RCS-JE-TE211B	1
RCP 1A Bearing Water Temperature	RCS-JE-TE211C	1
RCP 1A Bearing Water Temperature	RCS-JE-TE211D	1
RCP 1B Bearing Water Temperature	RCS-JE-TE212A	1
RCP 1B Bearing Water Temperature	RCS-JE-TE212B	1
RCP 1B Bearing Water Temperature	RCS-JE-TE212C	1
RCP 1B Bearing Water Temperature	RCS-JE-TE212D	1
RCP 2A Bearing Water Temperature	RCS-JE-TE213A	1
RCP 2A Bearing Water Temperature	RCS-JE-TE213B	1
RCP 2A Bearing Water Temperature	RCS-JE-TE213C	1
RCP 2A Bearing Water Temperature	RCS-JE-TE213D	1
RCP 2B Bearing Water Temperature	RCS-JE-TE214A	1
RCP 2B Bearing Water Temperature	RCS-JE-TE214B	1
RCP 2B Bearing Water Temperature	RCS-JE-TE214C	1
RCP 2B Bearing Water Temperature	RCS-JE-TE214D	1

Table 3I.6-3 (Sheet 3 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Penetrations		
Penetrations (Mechanical)		1
Penetrations (Electrical)		1
Active Valves		
Containment Isolation – Air Out	CAS-PL-V014	2
Containment Isolation – Air In	CAS-PL-V015	2
Containment Isolation – Inlet	CCS-PL-V200	2
Service Air Supply Inside Containment Isolation	CAS-PL-V205	2
Containment Isolation – Inlet	CCS-PL-V201	2
Containment Isolation – Outlet	CCS-PL-V207	2
Containment Isolation – Outlet	CCS-PL-V208	2
CCS Containment Isolation Relief	CCS-PL-V220	2
CCS IRC Relief Valve	CCS-PL-V270	2
CCS IRC Relief Valve	CCS-PL-V271	2
RCS Purification Stop Valve	CVS-PL-V001	2
RCS Purification Stop Valve	CVS-PL-V002	2
RCS Letdown Stop Valve	CVS-PL-V003	2
Demineralizer Flush Line Relief Valve	CVS-PL-V042	2
WLS Letdown IRC Isolation	CVS-PL-V045	2
Letdown Flow ORC Isolation	CVS-PL-V047	2
RCS Purification Check Valve	CVS-PL-V080	2
RCS Purification Stop Valve	CVS-PL-V081	2
RCS Purification Check Valve	CVS-PL-V082	2
Auxiliary PZR Spray Isolation	CVS-PL-V084	2
Auxiliary PZR Spray Isolation	CVS-PL-V085	2
Makeup Line Containment Isolation	CVS-PL-V090	2
Makeup Line Containment Isolation	CVS-PL-V091	2

Table 3I.6-3 (Sheet 4 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Hydrogen Addition Containment Isolation	CVS-PL-V092	2
Hydrogen Addition Containment Isolation	CVS-PL-V094	2
Hydrogen Addition Containment Isolation	CVS-PL-V092	2
Hydrogen Addition Containment Isolation	CVS-PL-V094	2
Makeup Containment Isolation	CVS-PL-V100	2
Demineralizer Water System Isolation	CVS-PL-V136A	2
Demineralized Water System Isolation	CVS-PL-V136B	2
Demin Water Supply Containment Isolation – Inside	DWS-PL-V245	2
Fuel Transfer Tube Gate Valve	FHS-PL-V001	2
Fire Water Containment Supply Isolation – Inside	FPS-PL-V052	2
PCCWST Isolation Valve	PCS-PL-V001A	2
PCCWST Isolation Valve	PCS-PL-V001B	2
PCCWST Isolation Valve	PCS-PL-V001C	2
PCCWST Isolation Valve	PCS-PL-V002A	2
PCCWST Isolation Valve	PCS-PL-V002B	2
PCCWST Isolation Valve	PCS-PL-V002C	2
PCCWST Fire Protection Isolation	PCS-PL-V005	2
PCCWST Emergency Spent Fuel Pool Makeup Isolation	PCS-PL-V009	2
Water Bucket Makeup Line Drain Valve	PCS-PL-V015	2
Water Bucket Makeup Line Isolation Valve	PCS-PL-V020	2
PCS Recirculation Isolation	PCS-PL-V023	2
PCCWST Long-Term Makeup Check Valve	PCS-PL-V039	2
PCCWST Long Term Makeup Isolation Drain Valve	PCS-PL-V042	2
PCCWST Long Term Makeup Isolation Valve	PCS-PL-V044	2
Emergency Makeup to the Spent Fuel Pool Isolation Valve	PCS-PL-V045	2

Table 3I.6-3 (Sheet 5 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
PCCWST Recirculation Return Isolation Valve	PCS-PL-V046	2
Emergency Makeup to the Spent Fuel Pool Drain Isolation Valve	PCS-PL-V049	2
Spent Fuel Pool Long Term Makeup Isolation Valve	PCS-PL-V050	2
Spent Fuel Pool Emergency Makeup Lower Isolation Valve	PCS-PL-V051	2
Containment Isolation – Air Sample Line	PSS-PL-V008	2
Containment Isolation – Liquid Sample Line	PSS-PL-V010A	2
Containment Isolation – Liquid Sample Line	PSS-PL-V010B	2
Containment Isolation – Liquid Sample Line	PSS-PL-V011	2
Containment Isolation – Sample Return Line	PSS-PL-V023	2
Containment Isolation Sample Return	PSS-PL-V024	2
Containment Isolation – Air Sample Line	PSS-PL-V046	2
PWS MCR Isolation	PWS-PL-V418	2
PWS MCR Isolation	PWS-PL-V420	2
PWS MCR Vacuum Relief	PWS-PL-V498	2
Core Makeup Tank A Discharge Isolation	PXS-PL-V014A	2
Core Makeup Tank B Discharge Isolation	PXS-PL-V014B	2
Core Makeup Tank A Discharge Isolation	PXS-PL-V015A	2
Core Makeup Tank B Discharge Isolation	PXS-PL-V015B	2
Core Makeup Tank A Discharge	PXS-PL-V016A	2
Core Makeup Tank B Discharge	PXS-PL-V016B	2
Core Makeup Tank A Discharge	PXS-PL-V017A	2
Core Makeup Tank B Discharge	PXS-PL-V017B	2
Accumulator A Pressure Relief	PXS-PL-V022A	2
Accumulator B Pressure Relief	PXS-PL-V022B	2
Accumulator A Discharge	PXS-PL-V028A	2

Table 3I.6-3 (Sheet 6 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Accumulator B Discharge	PXS-PL-V028B	2
Accumulator A Discharge	PXS-PL-V029A	2
Accumulator B Discharge	PXS-PL-V029B	2
Nitrogen Supply Outside Containment Isolation	PXS-PL-V042	2
IRC Nitrogen Supply Inside Containment Isolation	PXS-PL-V043	2
PRHR HX Discharge Isolation	PXS-PL-V108A	2
PRHR HX Discharge Isolation	PXS-PL-V108B	2
Recirc Sump A Isolation	PXS-PL-V118A	2
Recirc Sump B Isolation	PXS-PL-V118B	2
Recirc Sump A Isolation	PXS-PL-V119A	2
Recirc Sump B Isolation	PXS-PL-V119B	2
Recirc Sump A Isolation	PXS-PL-V120A	2
Recirc Sump B Isolation	PXS- PL-V120B	2
IRWST Injection A	PXS-PL-V122A	2
IRWST Injection B	PXS-PL-V122B	2
IRWST Injection A	PXS-PL-V123A	2
IRWST Injection B	PXS-PL-V123B	2
IRWST Injection A	PXS-PL-V124A	2
IRWST Injection B	PXS-PL-V124B	2
IRWST Injection A	PXS-PL-V125A	2
IRWST Injection B	PXS-PL-V125B	2
IRWST Gutter Drain Isolation A	PXS-PL-V130A	2
IRWST Gutter Drain Isolation B	PXS-PL-V130B	2
First Stage ADS	RCS-PL-V001A	2
First Stage ADS	RCS-PL-V001B	2

Table 3I.6-3 (Sheet 7 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Second Stage ADS	RCS-PL-V002A	2
Second Stage ADS	RCS-PL-V002B	2
Third Stage ADS	RCS-PL-V003A	2
Third Stage ADS	RCS-PL-V003B	2
Fourth Stage ADS	RCS-PL-V004A	2
Fourth Stage ADS	RCS-PL-V004B	2
Fourth Stage ADS	RCS-PL-V004C	2
Fourth Stage ADS	RCS-PL-V004D	2
PZR Safety Valve	RCS-PL-V005A	2
PZR Safety Valve	RCS-PL-V005B	2
ADS Discharge Header A Relief	RCS-PL-V010A	2
ADS Discharge Header B Relief	RCS-PL-V010B	2
First Stage ADS Isolation	RCS-PL-V011A	2
First Stage ADS Isolation	RCS-PL-V011B	2
Second Stage ADS Isolation	RCS-PL-V012A	2
Second Stage ADS Isolation	RCS-PL-V012B	2
Third Stage ADS Isolation	RCS-PL-V013A	2
Third Stage ADS Isolation	RCS-PL-V013B	2
Reactor Vessel Head Vent	RCS-PL-V150A	2
Reactor Vessel Head Vent	RCS-PL-V150B	2
Reactor Vessel Head Vent	RCS-PL-V150C	2
Reactor Vessel Head Vent	RCS-PL-V150D	2
RCS Inner Suction Isolation	RNS-PL-V001A	2
RCS Inner Suction Isolation	RNS-PL-V001B	2
RCS Outer Suction Isolation	RNS-PL-V002A	2

Table 3I.6-3 (Sheet 8 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
RCS Outer Suction Isolation	RNS-PL-V002B	2
RCS Thermal Relief	RNS-PL-V003A	2
RCS Thermal Relief	RNS-PL-V003B	2
RHR Control/Isolation Valve	RNS-PL-V011	2
RNS Discharge Containment Isolation Valve Test Connection	RNS-PL-V012	2
RNS Discharge Containment Isolation	RNS-PL-V013	2
RNS Discharge RCP B Isolation	RNS-PL-V015A	2
RNS Discharge RCP B Isolation	RNS-PL-V015B	2
RNS Discharge RCP B Isolation	RNS-PL-V017A	2
RNS Discharge RCP B Isolation	RNS-PL-V017B	2
RNS Hot Leg Suction Relief	RNS-PL-V021	2
RHR Pump Suction Header Isolation	RNS-PL-V022	2
IRWST Suction Line Isolation	RNS-PL-V023	2
RNS Pump Discharge Relief	RNS-PL-V045	2
RNS – CVS Containment Isolation	RNS-PL-V061	2
Containment Isolation	SFS-PL-V034	2
Containment Isolation	SFS-PL-V035	2
SFS Discharge Containment Isolation	SFS-PL-V037	2
Containment Isolation	SFS-PL-V038	2
SFS Cask Loading Pit to SFS Pump	SFS-PL-V042	2
SFS Pump to Cask Loading Pit	SFS-PL-V045	2
Cask Loading Pit to WLS	SFS-PL-V049	2
Spent Fuel Pool to Cask Washdown Pit Isolation	SFS-PL-V066	2
SFS Containment Isolation Relief	SFS-PL-V067	2
Cask Washdown Pit Drain Isolation	SFS-PL-V068	2

Table 3I.6-3 (Sheet 9 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Refueling Cavity to SG Compartment	SFS-PL-V071	2
Refueling Cavity to SG Compartment	SFS-PL-V072	2
PORV Block Valve	SGS-PL-V027A	2
PORV Block Valve	SGS-PL-V027B	2
Steam Safety Valve SG01	SGS-PL-V030A	2
Steam Safety Valve SG02	SGS-PL-V030B	2
Steam Safety Valve SG01	SGS-PL-V031A	2
Steam Safety Valve SG02	SGS-PL-V031B	2
Steam Safety Valve SG01	SGS-PL-V032A	2
Steam Safety Valve SG02	SGS-PL-V032B	2
Steam Safety Valve SG01	SGS-PL-V033A	2
Steam Safety Valve SG02	SGS-PL-V033B	2
Steam Safety Valve SG01	SGS-PL-V034A	2
Steam Safety Valve SG02	SGS-PL-V034B	2
Steam Safety Valve SG01	SGS-PL-V035A	2
Steam Safety Valve SG02 Steam Line Condensate	SGS-PL-V035B	2
Drain Isolation	SGS-PL-V036A	2
Steam Line Condensate Isolation	SGS-PL-V036B	2
Main Steam Line Isolation	SGS-PL-V040A	2
Main Steam Line Isolation	SGS-PL-V040B	2
Main Feedwater Isolation	SGS-PL-V057A	2
Main Feedwater Isolation	SGS-PL-V057B	2
Startup Feedwater Isolation	SGS-PL-V067A	2
Startup Feedwater Isolation	SGS-PL-V067B	2
SG Blowdown Isolation	SGS-PL-V074A	2

Table 3I.6-3 (Sheet 10 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
SG Blowdown Isolation	SGS-PL-V074B	2
SG Series Blowdown Isolation	SGS-PL-V075A	2
SG Series Blowdown Isolation	SGS-PL-V075B	2
Steam Line Condensate Drain Isolation	SGS-PL-V086A	2
Steam Line Condensate Drain Isolation	SGS-PL-V086B	2
Power-Operated Relief Valve	SGS-PL-V233A	2
Power-Operated Relief Valve	SGS-PL-V233B	2
MSIV Bypass Isolation Valve	SGS-PL-V240A	2
MSIV Bypass Isolation Valve	SGSPL-V240B	2
Main Feedwater Control Valve	SGS-PL-V250A	2
Main Feedwater Control Valve	SGS-PL-V250B	2
Startup Feedwater Control Valve	SGS-PL-V255A	2
Startup Feedwater Control Valve	SGS-PL-V255B	2
MCR Isolation Valve	VBS-PL-V186	2
MCR Isolation Valve	VBS-PL-V187	2
MCR Isolation Valve	VBS-PL-V188	2
MCR Isolation Valve	VBS-PL-V189	2
MCR Isolation Valve	VBS-PL-V190	2
MCR Isolation Valve	VBS-PL-V191	2
Air Delivery Isolation Valve	VES-PL-V001	2
Pressure Regulator Valve A	VES-PL-V002A	2
Pressure Regulator Valve B	VES-PL-V002B	2
Actuation Valve A	VES-PL-V005A	2
Actuation Valve B	VES-PL-V005B	2
Temporary Instrument Isolation Valve A	VES-PL-V018	2
Temporary Instrument Isolation Valve B	VES-PL-V019	2
Relief Isolation Valve A	VES-PL-V022A	2
Relief Isolation Valve B	VES-PL-V022B	2

Table 3I.6-3 (Sheet 11 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Air Tank Relief A	VES-PL-V040A	2
Air Tank Relief B	VES-PL-V040B	2
Air Tank Relief A	VES-PL-V041A	2
Air Tank Relief B	VES-PL-V041B	2
Main Air Flow Path Isolation Valve	VES-PL-V044	2
Containment Purge Inlet Isolation	VFS-PL-V003	2
Containment Purge Inlet Isolation	VFS-PL-V004	2
Containment Purge Discharge Isolation	VFS-PL-V009	2
Containment Purge Discharge Isolation	VFS-PL-V010	2
Vacuum Relief Containment Isolation Valve A - ORC	VFS-PL-V800A	2
Vacuum Relief Containment Isolation Valve B - ORC	VFS-PL-V800B	2
Vacuum Relief Containment Isolation Check Valve A - IRC	VFS-PL-V803A	2
Vacuum Relief Containment Isolation Check Valve B - IRC	VFS-PL-V803B	2
Fan Cooler Supply Isolation	VWS-PL-V058	2
Fan Cooler Supply Isolation	VWS-PL-V062	2
VWS Containment Isolation Relief	VWS-PL-V080	2
Fan Cooler Return Isolation	VWS-PL-V082	2
Fan Cooler Return Isolation	VWS-PL-V086	2
Sump Containment Isolation IRC	WLS-PL-V055	2
Sump Containment Isolation ORC	WLS-PL-V057	2
WLS Containment Isolation Relief	WLS-PL-V058	2
RCDT Gas Containment Isolation	WLS-PL-V067	2
RCDT Gas Containment Isolation	WLS-PL-V068	2
CVS To Sump	WLS-PL-V071 A	2
PXS A To Sump	WLS-PL-V071 B	2

Table 3I.6-3 (Sheet 12 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
PXS B To Sump	WLS-PL-V071 C	2
CVS To Sump	WLS-PL-V072 A	2
PXS A To Sump	WLS-PL-V072 B	2
PXS B To Sump	WLS-PL-V072 C	2
Miscellaneous		
Nonactive Valves		
Containment Penetration Test Connection Isolation	CAS-PL-V027	2
Service Air Supply Outside Containment Isolation	CAS-PL-V204	2
Containment Penetration Test Connection Isolation	CAS-PL-V219	2
Containment Isolation Valve Test Connection – Outlet Line	CCS-PL-V209	2
CCS Supply Containment Isolation – IRC	CCS-PL-V214	2
CCS Supply Containment Isolation Valve Test Connection – IRC	CCS-PL-V215	2
Containment Leak Test Outlet Line – IRC	CCS-PL-V216	2
Containment Isolation Valve V207 Body Test Connection Valve	CCS-PL-V217	2
Containment Isolation Valve Test Connection – Inlet Line	CCS-PL-V257	2
Resin Flush IRC Isolation	CVS-PL-V040	2
Resin Flush ORC Isolation	CVS-PL-V041	2
Letdown PZR Instrument Root	CVS-PL-V046	2
H2 Mkup Containment Isolation Thermal Relief Valve	CVS-PL-V065	2
Hydrogen Add Cont Isolation Test Connection	CVS-PL-V095	2
Hydrogen Addition Containment Isolation Test Connection	CVS-PL-V096	2
Demin Water Supply Containment Isolation – Outside	DWS-PL-V244	2
Containment Penetration Test Connection Isolation	DWS-PL-V248	2
Fire Water Containment Test Connection Isolation	FPS-PL-V049	2
Fire Water Containment Supply Isolation	FPS-PL-V050	2

Table 3I.6-3 (Sheet 13 of 32)
List Of AP1000 Safety-Related Electrical
and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Fire Water Containment Test Connection Isolation	FPS-PL-V051	2
Flow Transmitter FT001 Root Valve	PCS-PL-V010A	2
Flow Transmitter FT001 Root Valve	PCS-PL-V010B	2
Flow Transmitter FT002 Root Valve	PCS-PL-V011A	2
Flow Transmitter FT001 Root Valve	PCS-PL-V011B	2
Flow Transmitter FT003 Root Valve	PCS-PL-V012A	2
Flow Transmitter FT003 Root Valve	PCS-PL-V012B	2
Flow Transmitter FT004 Root Valve	PCS-PL-V013A	2
Flow Transmitter FT004 Root Valve	PCS-PL-V013B	2
PCCWST Drain Isolation Valve	PCS-PL-V016	2
PCCWST Isolation Valve Leakage Detection Drain	PCS-PL-V029	2
PCCWST Isolation Valve Leakage Detection Crossconn	PCS-PL-V030	2
PCCWST Level Instrument Root Valve	PCS-PL-V031A	2
PCCWST Level Instrument Root Valve	PCS-PL-V031B	2
Recirculation Pump Suction from Long Term Makeup Isolation Valve	PCS-PL-V033	2
Spent Fuel Pool Emergency Makeup Isolation	PLS-PL-V052	2
Hot Leg 1 Sample Isolation	PSS-PL-V001A	2
Hot Leg 2 Sample Isolation	PSS-PL-V001B	2
Pressurizer Sample Isolation	PSS-PL-V003	2
PXS Accumulator Sample Isolation	PSS-PL-V004A	2
PXS Accumulator Sample Isolation	PSS-PL-V004B	2
PXS CMT A Sample Isolation	PSS-PL-V005A	2
PXS CMT B Sample Isolation	PSS-PL-V005B	2
PXS CMT A Sample Isolation	PSS-PL-V005C	2

Table 3I.6-3 (Sheet 14 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
PXS CMT B Sample Isolation	PSS-PL-V005D	2
Liquid Sample Check Valve	PSS-PL-V012A	2
Liquid Sample Check Valve	PSS-PL-V012B	2
Containment Testing Boundary Isolation Valve	PSS-PL-V076A	2
Containment Testing Boundary Isolation Valve	PSS-PL-V076B	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V082	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V083	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V085	2
Containment Isolation Test Connection Isolation Valve	PSS-PL-V086	2
Core Makeup Tank A CL Inlet Isolation	PXS-PL-V002A	2
Core Makeup Tank B CL Inlet Isolation	PXS-PL-V002B	2
Core Makeup Tank A Upper Sample	PXS-PL-V010A	2
Core Makeup Tank B Upper Sample	PXS-PL-V010B	2
Core Makeup Tank A Lower Sample	PXS-PL-V011A	2
Core Makeup Tank B Lower Sample	PXS-PL-V011B	2
Core Makeup Tank A Drain	PXS-PL-V012A	2
Core Makeup Tank B Drain	PXS-PL-V012B	2
Core Makeup Tank Discharge Manual Isolation	PXS-PL-V013A	2
Core Makeup Tank B Discharge Manual Isolation	PXS-PL-V013B	2
RNS to CMT Injection Line A Drain	PXS-PL-V019A	2
RNS to CMT Injection Line B Drain	PXS-PL-V019B	2
IRWST Injection Line A Drain	PXS-PL-V020A	2
IRWST Injection Line B Drain	PXS-PL-V020B	2
Accumulator A N_2 Vent	PXS-PL-V021A	2
Accumulator B N_2 Vent	PXS-PL-V021B	2

Table 3I.6-3 (Sheet 15 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Accumulator A PZR Transmitter Isolation	PXS-PL-V023A	2
Accumulator B PZR Transmitter Isolation	PXS-PL-V023B	2
Accumulator A PZR Transmitter Isolation	PXS-PL-V024A	2
Accumulator B PZR Transmitter Isolation	PXS-PL-V024B	2
Accumulator A Sample	PXS-PL-V025A	2
Accumulator B Sample	PXS-PL-V025B	2
Accumulator A Drain	PXS-PL-V026A	2
Accumulator B Drain	PXS-PL-V026B	2
Accumulator A Discharge Isolation	PXS-PL-V027A	2
Accumulator B Discharge Isolation	PXS-PL-V027B	2
Core Makeup Tank A Highpoint Vent	PXS-PL-V030A	2
Core Makeup Tank B Highpoint Vent	PXS-PL-V030B	2
Core Makeup Tank A Highpoint Vent	PXS-PL-V031A	2
Core Makeup Tank B Highpoint Vent	PXS-PL-V031B	2
Accumulator A Check Valve Drain	PXS-PL-V033A	2
Accumulator B Check Valve Drain	PXS-PL-V033B	2
Accumulator N_2 Containment Penetration Test Connection	PXS-PL-V052	2
CMT A Wide Level Upper Root	PXS-PL-V080A	2
CMT B Wide Level Upper Root	PXS-PL-V080B	2
CMT A Wide Level Lower Root	PXS-PL-V081A	2
CMT B Wide Level Lower Root	PXS-PL-V081B	2
CMT A Upper Level A Isolation 1	PXS-PL-V082A	2
CMT B Upper Level A Isolation 1	PXS-PL-V082B	2
CMT A Upper Level A Isolation 2	PXS-PL-V083A	2
CMT B Upper Level A Isolation 2	PXS-PL-V083B	2
CMT A Upper Level A Vent	PXS-PL-V084A	2

Table 3I.6-3 (Sheet 16 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
CMT B Upper Level A Vent	PXS-PL-V084B	2
CMT A Upper Level A Drain	PXS-PL-V085A	2
CMT B Upper Level A Drain	PXS-PL-V085B	2
CMT A Upper Level B Isolation 1	PXS-PL-V086A	2
CMT B Upper Level B Isolation 1	PXS-PL-V086B	2
CMT A Upper Level B Isolation 2	PXS-PL-V087A	2
CMT B Upper Level B Isolation 2	PXS-PL-V087B	2
CMT A Upper Level B Vent	PXS-PL-V088A	2
CMT B Upper Level B Vent	PXS-PL-V088B	2
CMT A Upper Level B Drain	PXS-PL-V089A	2
CMT B Upper Level B Drain	PXS-PL-V089B	2
CMT A Lower Level A Isolation 1	PXS-PL-V092A	2
CMT B Lower Level A Isolation 1	PXS-PL-V092B	2
CMT A Lower Level A Isolation 2	PXS-PL-V093A	2
CMT B Lower Level A Isolation 2	PXS-PL-V093B	2
CMT A Lower Level A Vent	PXS-PL-V094A	2
CMT B Lower Level A Vent	PXS-PL-V094B	2
CMT A Lower Level A Drain	PXS-PL-V095A	2
CMT B Lower Level A Drain	PXS-PL-V095B	2
CMT A Lower Level B Isolation 1	PXS-PL-V096A	2
CMT B Lower Level B Isolation 1	PXS-PL-V096B	2
CMT A Lower Level B Isolation 2	PXS-PL-V097A	2
CMT B Lower Level B Isolation 2	PXS-PL-V097B	2
CMT A Lower Level B Vent	PXS-PL-V098A	2
CMT B Lower Level B Vent	PXS-PL-V098B	2

Table 3I.6-3 (Sheet 17 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
CMT A Lower Level B Drain	PXS-PL-V099A	2
CMT B Lower Level B Drain	PXS-PL-V099B	2
PRHR HX Inlet Isolation	PXS-PL-V101	2
PRHR HX Inlet Head Vent	PXS-PL-V102A	2
PRHR HX Inlet Head Drain	PXS-PL-V102B	2
PRHR HX Outlet Head Vent	PXS-PL-V103A	2
PRHR HX Outlet Head Drain	PXS-PL-V103B	2
PRHR HX Flow Transmitter A Isolation	PXS-PL-V104A	2
PRHR HX Flow Transmitter B Isolation	PXS-PL-V104B	2
PRHR HX Flow Transmitter A Isolation	PXS-PL-V105A	2
PRHR HX Flow Transmitter B Isolation	PXS-PL-V105B	2
Containment Recirculation A Highpoint Vent	PXS-PL-V106	2
Containment Recirculation A Highpoint Vent	PXS-PL-V107	2
PRHR HX/RCS Return Isolation	PXS-PL-V109	2
PRHR HX Highpoint Vent	PXS-PL-V111A	2
PRHR HX Highpoint Vent	PXS-PL-V111B	2
PRHR HX PZR Transmitter Isolation	PXS-PL-V113	2
Containment Recirculation A Drain	PXS-PL-V115A	2
Containment Recirculation B Drain	PXS-PL-V115B	2
Containment Recirculation A Drain	PXS-PL-V116A	2
Containment Recirculation B Drain	PXS-PL-V116B	2
Recirc Sump A Isolation	PXS-PL-V117A	2
Recirc Sump B Isolation	PXS-PL-V117B	2
IRWST Line A Isolation	PXS-PL-V121A	2
IRWST Line B Isolation	PXS-PL-V121B	2
IRWST Injection Check Test	PXS-PL-V126A	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
IRWST Injection Check Test	PXS-PL-V126B	2
IRWST Injection Line A Drain	PXS-PL-V127	2
IRWST Injection Check Test	PXS-PL-V128A	2
IRWST Injection Check Test	PXS-PL-V128B	2
IRWST Injection Check Test	PXS-PL-V129A	2
IRWST Injection Check Test	PXS-PL-V129B	2
IRWST Injection Line A Drain	PXS-PL-V131A	2
IRWST Injection Line B Drain	PXS-PL-V131B	2
IRWST Injection Line A Drain	PXS-PL-V132A	2
IRWST Injection Line B Drain	PXS-PL-V132B	2
IRWST Injection Line A Highpoint Vent	PXS-PL-V133A	2
IRWST Injection Line B Highpoint Vent	PXS-PL-V133B	2
IRWST Injection Line A Highpoint Vent	PXS-PL-V134A	2
IRWST Injection Line B Highpoint Vent	PXS-PL-V134B	2
IRWST Injection Line A Highpoint Vent Isolation	PXS-PL-V135A	2
IRWST Injection Line B Highpoint Vent Isolation	PXS-PL-V135B	2
RNS Suction Pump Line Drain	PXS-PL-V149	2
IRWST Level Transmitter A Isolation	PXS-PL-V150A	2
IRWST Level Transmitter B Isolation	PXS-PL-V150B	2
IRWST Level Transmitter C Isolation	PXS-PL-V150C	2
IRWST Level Transmitter D Isolation	PXS-PL-V150D	2
IRWST Level Transmitter A Isolation	PXS-PL-V151A	2
IRWST Level Transmitter B Isolation	PXS-PL-V151B	2
IRWST Level Transmitter C Isolation	PXS-PL-V151C	2
IRWST Level Transmitter D Isolation	PXS-PL-V151D	2
Accumulator A Leak Test	PXS-PL-V201A	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Accumulator B Leak Test	PXS-PL-V201B	2
Accumulator A Leak Test	PXS-PL-V202A	2
Accumulator B Leak Test	PXS-PL-V202B	2
RNS Discharge Leak Test	PXS-PL-V205A	2
RNS Discharge Leak Test	PXS-PL-V205B	2
RNS Discharge Leak Test	PXS-PL-V206	2
RNS Suction Leak Test	PXS-PL-V207A	2
RNS Suction Leak Test	PXS-PL-V207B	2
RNS Suction Leak Test	PXS-PL-V208A	2
Core Makeup Tank A Fill Isolation	PXS-PL-V230A	2
Core Makeup Tank B Fill Isolation	PXS-PL-V230B	2
Core Makeup Tank A Fill Check	PXS-PL-V231A	2
Core Makeup Tank B Fill Check	PXS-PL-V231B	2
Accumulator A Fill/Drain Isolation	PXS-PL-V232A	2
Accumulator B Fill/Drain Isolation	PXS-PL-V232B	2
CMT A Check Valve Test Valve	PXS-PL-V250A	2
CMT B Check Valve Test Valve	PXS-PL-V250B	2
CMT A Check Valve Test Valve	PXS-PL-V251A	2
CMT B Check Valve Test Valve	PXS-PL-V251B	2
CMT A Check Valve Test Valve	PXS-PL-V252A	2
CMT B Check Valve Test Valve	PXS-PL-V252B	2
ADS Test Valve	RCS-PL-V007A	2
ADS Test Valve	RCS-PL-V007B	2
Fourth Stage ADS Isolation	RCS-PL-V014A	2
Fourth Stage ADS Isolation	RCS-PL-V014B	2
Fourth Stage ADS Isolation	RCS-PL-V014C	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Fourth Stage ADS Isolation	RCS-PL-V014D	2
Hot Leg 2 Level Instrument Root	RCS-PL-V095	2
Hot Leg 2 Level Instrument Root	RCS-PL-V096	2
Hot Leg 1 Level Instrument Root	RCS-PL-V097	2
Hot Leg 1 Level Instrument Root	RCS-PL-V098	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101A	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101B	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101C	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101D	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101E	2
Hot Leg 1 Flow Instrument Root	RCS-PL-V101F	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102A	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102B	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102C	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102D	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102E	2
Hot Leg 2 Flow Instrument Root	RCS-PL-V102F	2
PRHR HX Outlet Line Drain	RCS-PL-V103	2
Hot Leg 1 Sample Isolation	RCS-PL-V108A	2
Hot Leg 2 Sample Isolation	RCS-PL-V108B	2
PZR Spray Valve	RCS-PL-V110A	2
PZR Spray Valve	RCS-PL-V110B	2
PZR Spray Block Valve	RCS-PL-V111A	2
PZR Spray Block Valve	RCS-PL-V111B	2
Cold Leg 1A Bend Instrument Root	RCS-PL-V171A	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Cold Leg 1A Bend Instrument Root	RCS-PL-V171B	2
Cold Leg 1B Bend Instrument Root	RCS-PL-V172A	2
Cold Leg 1B Bend Instrument Root	RCS-PL-V172B	2
Cold Leg 2A Bend Instrument Root	RCS-PL-V173A	2
Cold Leg 2A Bend Instrument Root	RCS-PL-V173B	2
Cold Leg 2B Bend Instrument Root	RCS-PL-V174A	2
Cold Leg 2B Bend Instrument Root	RCS-PL-V174B	2
PZR Manual Vent	RCS-PL-V204	2
PZR Manual Vent	RCS-PL-V205	2
PZR Spray Bypass	RCS-PL-V210A	2
PZR Spray Bypass	RCS-PL-V210B	2
PZR Level Steam Space Instrument Root	RCS-PL-V225A	2
PZR Level Steam Space Instrument Root	RCS-PL-V225B	2
PZR Level Steam Space Instrument Root	RCS-PL-V225C	2
PZR Level Steam Space Instrument Root	RCS-PL-V225D	2
PZR Level Liquid Space Instrument Root	RCS-PL-V226A	2
PZR Level Liquid Space Instrument Root	RCS-PL-V226B	2
PZR Level Liquid Space Instrument Root	RCS-PL-V226C	2
PZR Level Liquid Space Instrument Root	RCS-PL-V226D	2
Wide Range PZR Level Steam Space Instrument Root	RCS-PL-V228	2
Wide Range PZR Level Liquid Space Instrument Root	RCS-PL-V229	2
Manual Head Vent	RCS-PL-V232	2
Head Vent Isolation	RCS-PL-V233	2
ADS Valve Discharge Header Drain Isolation	RCS-PL-V241	2
RCP 1A Flush	RCS-PL-V260A	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
RCP 1B Flush	RCS-PL-V260B	2
RCP 2A Flush	RCS-PL-V260C	2
RCP 2B Flush	RCS-PL-V260D	2
RCP 1A Drain	RCS-PL-V261A	2
RCP 1B Drain	RCS-PL-V261B	2
RCP 2A Drain	RCS-PL-V261C	2
RCP 2B Drain	RCS-PL-V261D	2
RCS Pressure Boundary Valve Thermal Relief Isolation	RNS-PL-V004A	2
RCS Pressure Boundary Valve Thermal Relief Isolation	RNS-PL-V004B	2
RNS Pump A Suction Isolation	RNS-PL-V005A	2
RNS Pump B Suction Isolation	RNS-PL-V005B	2
RNS HX A Outlet Flow Control	RNS-PL-V006A	2
RNS HX B Outlet Flow Control	RNS-PL-V006B	2
RNS Pump A Discharge Isolation	RNS-PL-V007A	2
RNS Pump B Discharge Isolation	RNS-PL-V007B	2
RNS HX A Bypass Flow Control	RNS-PL-V008A	2
RNS HX B Bypass Flow Control	RNS-PL-V008B	2
RNS Discharge Containment Isolation Valve Test	RNS-PL-V010	2
RNS Discharge Containment Isolation Valve Test Connection	RNS-PL-V014	2
RNS Discharge Containment Penetration Isolation Valves Test	RNS-PL-V016	2
RNS Discharge to IRWST Isolation	RNS-PL-V024	2
RNS Discharge to CVS	RNS-PL-V029	2
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V031A	2
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V031B	2

Table 3I.6-3 (Sheet 23 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
RNS Train A Discharge Flow Instrument Isolation	RNS-PL-V032A	2
RNS Train B Discharge Flow Instrument Isolation	RNS-PL-V032B	2
RNS Pump A Suction Pressure Instrument Isolation	RNS-PL-V033A	2
RNS Pump B Suction Pressure Instrument Isolation	RNS-PL-V033B	2
RNS Pump A Discharge Pressure Instrument Isolation	RNS-PL-V034A	2
RNS Pump B Discharge Pressure Instrument Isolation	RNS-PL-V034B	2
RNS Pump A Suction Piping Drain Isolation	RNS-PL-V036A	2
RNS Pump B Suction Piping Drain Isolation	RNS-PL-V036B	2
RNS HX A Channel Head Drain Isolation	RNS-PL-V046A	2
RNS HX B Channel Head Drain Isolation	RNS-PL-V046B	2
RNS Pump A Casing Drain Isolation	RNS-PL-V050	2
RNS Pump B Casing Drain Isolation	RNS-PL-V051	2
RNS Suction from SFP Isolation	RNS-PL-V052	2
RNS Discharge to SFP Isolation	RNS-PL-V053	2
RNS Suction from Cask Loading Pit Isolation Valve	RNS-PL-V055	2
RNS Pump Suction to Cask Loading Pit Isolation	RNS-PL-V056	2
RNS Train A Miniflow Isolation Valve	RNS-PL-V057A	2
RNS Train B Miniflow Isolation Valve	RNS-PL-V057B	2
RNS Pump Suction Containment Isolation Test Connection	RNS-PL-V059	2
RNS Discharge to DVI Line A Drain	RNS-PL-V066A	2
RNS Discharge to DVI Line B Drain	RNS-PL-V066B	2
RNS Discharge to DVI Line A Drain	RNS-PL-V067A	2
RNS Discharge to DVI Line B Drain	RNS-PL-V067B	2
RNS Discharge to IRWST Drain	RNS-PL-V068	2
LT019A Root Isolation Valve	SFS-PL-V024A	2
LT019B Root Isolation Valve	SFS-PL-V024B	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
LT019C Root Isolation Valve	SFS-PL-V024C	2
LT020 Root Isolation Valve	SFS-PL-V028	2
SFS Refueling Cavity Drain To SGS Compartment Isolation	SFS-PL-V031	2
SFS Refueling Cavity Suction Isolation	SFS-PL-V032	2
SFS Refueling Cavity Drain to Containment Sump Isolation	SFS-PL-V033	2
SFS Suction Line from IRWST Isolation	SFS-PL-V039	2
SFS Fuel Transfer Canal Suction Isolation	SFS-PL-V040	2
SFS Cask Loading Pit Suction Isolation	SFS-PL-V041	2
SFS CVS Makeup Reverse Flow Prevention	SFS-PL-V043	2
SFS Containment Penetration Test Connection	SFS-PL-V048	2
SFS Containment Penetration Test Connection Isolation	SFS-PL-V056	2
SFS Containment Isolation Valve V034 Test	SFS-PL-V058	2
SFS Containment Floodup Isolation Valve	SFS-PL-V075	2
LT001 Root Isolation Valve	SGS-PL-V001A	2
LT005 Root Isolation Valve	SGS-PL-V001B	2
LT001 Root Isolation Valve	SGS-PL-V002A	2
LT005 Root Isolation Valve	SGS-PL-V002B	2
LT002 Root Isolation Valve	SGS-PL-V003A	2
LT006 Root Isolation Valve	SGS-PL-V003B	2
LT002 Root Isolation Valve	SGS-PL-V004A	2
LT006 Root Isolation Valve	SGS-PL-V004B	2
LT003 Root Isolation Valve	SGS-PL-V005A	2
LT007 Root Isolation Valve	SGS-PL-V005B	2
LT003 Root Isolation Valve	SGS-PL-V006A	2
LT007 Root Isolation Valve	SGS-PL-V006B	2
LT004 Root Isolation Valve	SGS-PL-V007A	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
LT008 Root Isolation Valve	SGS-PL-V007B	2
LT004 Root Isolation Valve	SGS-PL-V008A	2
LT008 Root Isolation Valve	SGS-PL-V008B	2
LT011 Root Isolation Valve	SGS-PL-V010A	2
LT013 Root Isolation Valve	SGS-PL-V010B	2
LT011 Root Isolation Valve	SGS-PL-V011A	2
LT013 Root Isolation Valve	SGS-PL-V011B	2
LT012 Root Isolation Valve	SGS-PL-V012A	2
LT014 Root Isolation Valve	SGS-PL-V012B	2
LT012 Root Isolation Valve	SGS-PL-V013A	2
LT014 Root Isolation Valve	SGS-PL-V013B	2
FT021 Root Isolation Valve	SGS-PL-V015A	2
FT023 Root Isolation Valve	SGS-PL-V015B	2
FT020 Root Isolation Valve	SGS-PL-V016A	2
FT022 Root Isolation Valve	SGS-PL-V016B	2
FT021 Root Isolation Valve	SGS-PL-V017A	2
FT023 Root Isolation Valve	SGS-PL-V017B	2
FT020 Root Isolation Valve	SGS-PL-V018A	2
FT022 Root Isolation Valve	SGS-PL-V018B	2
Main Steam Line Vent Isolation	SGS-PL-V019A	2
Main Steam Line Vent Isolation	SGS-PL-V019B	2
PT030 Root Isolation Valve	SGS-PL-V022A	2
PT034 Root Isolation Valve	SGS-PL-V022B	2
PT031 Root Isolation Valve	SGS-PL-V023A	2
PT035 Root Isolation Valve	SGS-PL-V023B	2
PT032 Root Isolation Valve	SGS-PL-V024A	2

Table 3I.6-3 (Sheet 26 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
PT036 Root Isolation Valve	SGS-PL-V024B	2
PT033 Root Isolation Valve	SGS-PL-V025A	2
PT037 Root Isolation Valve	SGS-PL-V025B	2
Steam Line 1 Nitrogen Supply Isolation	SGS-PL-V038A	2
Steam Line 2 Nitrogen Supply Isolation	SGS-PL-V038B	2
MSIV Bypass Control Isolation	SGS-PL-V042A	2
MSIV Bypass Control Isolation	SGS-PL-V042B	2
MSIV Bypass Control Isolation	SGS-PL-V043A	2
MSIV Bypass Control Isolation	SGS-PL-V043B	2
SG1 Condensate Pipe Drain Valve	SGS-PL-V045A	2
SG2 Condensate Pipe Drain Valve	SGS-PL-V045B	2
LT015 Root Isolation Valve	SGS-PL-V046A	2
LT017 Root Isolation Valve	SGS-PL-V046B	2
LT015 Root Isolation Valve	SGS-PL-V047A	2
LT017 Root Isolation Valve	SGS-PL-V047B	2
LT016 Root Isolation Valve	SGS-PL-V048A	2
LT018 Root Isolation Valve	SGS-PL-V048B	2
LT016 Root Isolation Valve	SGS-PL-V049A	2
LT018 Root Isolation Valve	SGS-PL-V049B	2
LT044 Root Isolation Valve	SGS-PL-V050A	2
LT046 Root Isolation Valve	SGS-PL-V050B	2
LT044 Root Isolation Valve	SGS-PL-V051A	2
LT046 Root Isolation Valve	SGS-PL-V051B	2
LT045 Root Isolation Valve	SGS-PL-V052A	2
LT047 Root Isolation Valve	SGS-PL-V052B	2
LT045 Root Isolation Valve	SGS-PL-V053A	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
LT047 Root Isolation Valve	SGS-PL-V053B	2
PT062 Root Isolation Valve	SGS-PL-V056A	2
PT063 Root Isolation Valve	SGS-PL-V056B	2
Main Feedwater Check	SGS-PL-V058A	2
Main Feedwater Check	SGS-PL-V058B	2
FT055A Root Isolation Valve	SGS-PL-V062A	2
FT056A Root Isolation Valve	SGS-PL-V062B	2
FT055A Root Isolation Valve	SGS-PL-V063A	2
FT056A Root Isolation Valve	SGS-PL-V063B	2
FT055A Root Isolation Valve	SGS-PL-V064A	2
FT056A Root Isolation Valve	SGS-PL-V064B	2
FT055A Root Isolation Valve	SGS-PL-V065A	2
FT056A Root Isolation Valve	SGS-PL-V065B	2
SG1 Nitrogen Sparging Isolation	SGS-PL-V084A	2
SG2 Nitrogen Sparging Isolation	SGS-PL-V084B	2
Orifice Isolation Valve	SGS-PL-V093A	2
Orifice Isolation Valve	SGS-PL-V093B	2
Orifice Cleanout Line Isolation Valve	SGS-PL-V094A	2
Orifice Cleanout Line Isolation Valve	SGS-PL-V094B	2
Orifice Isolation Valve	SGS-PL-V095A	2
Orifice Isolation Valve	SGS-PL-V095B	2
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V096A	2
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V096B	2
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V097A	2
Steam Line Condensate Drain Level Isolation Valve	SGS-PL-V097B	2
Startup Feedwater Check Valve	SGS-PL-V256A	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Startup Feedwater Check Valve	SGS-PL-V256B	2
Air Delivery Line Pressure Instrument Isolation Valve A	VES-PL-V006A	2
Air Delivery Line Pressure Instrument Isolation Valve B	VES-PL-V006B	2
Temporary Instrument Isolation Valve A	VES-PL-V016	2
Deleted		
Deleted		
Temporary Instrument Isolation Valve B	VES-PL-V020	2
Air Tank Isolation Valve A	VES-PL-V024A	2
Air Tank Isolation Valve B	VES-PL-V024B	2
Air Tank Isolation Valve A	VES-PL-V025A	2
Air Tank Isolation Valve B	VES-PL-V025B	2
Refill Line Isolation Valve	VES-PL-V038	2
DP Instrument Line Isolation Valve A	VES-PL-V043A	2
DP Instrument Line Isolation Valve B	VES-PL-V043B	2
Containment Isolation Test Connection	VFS-PL-V008	2
Containment Isolation Test Connection	VFS-PL-V012	2
Containment Isolation Test Connection	VFS-PL-V015	2
Main Equipment Hatch Test Connection	VUS-PL-V015	2
Maintenance Equipment Hatch Test Connection	VUS-PL-V016	2
Personnel Hatch Test Connection	VUS-PL-V017	2
Personnel Hatch Test Connection	VUS-PL-V018	2
Personnel Hatch Test Connection	VUS-PL-V019	2
Personnel Hatch Test Connection	VUS-PL-V020	2
Personnel Hatch Test Connection	VUS-PL-V021	2
Personnel Hatch Test Connection	VUS-PL-V022	2
Fuel Transfer Tube Test Connection	VUS-PL-V023	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Electrical Penetration Test Isolation Valve	VUS-PL-V101	2
Electrical Penetration Test Isolation Valve	VUS-PL-V102	2
Electrical Penetration Test Isolation Valve	VUS-PL-V103	2
Electrical Penetration Test Isolation Valve	VUS-PL-V104	2
Electrical Penetration Test Isolation Valve	VUS-PL-V105	2
Electrical Penetration Test Isolation Valve	VUS-PL-V106	2
Electrical Penetration Test Isolation Valve	VUS-PL-V107	2
Electrical Penetration Test Isolation Valve	VUS-PL-V108	2
Electrical Penetration Test Isolation Valve	VUS-PL-V109	2
Electrical Penetration Test Isolation Valve	VUS-PL-V110	2
Electrical Penetration Test Isolation Valve	VUS-PL-V111	2
Electrical Penetration Test Isolation Valve	VUS-PL-V112	2
Electrical Penetration Test Isolation Valve	VUS-PL-V113	2
Electrical Penetration Test Isolation Valve	VUS-PL-V114	2
Electrical Penetration Test Isolation Valve	VUS-PL-V115	2
Electrical Penetration Test Isolation Valve	VUS-PL-V116	2
Electrical Penetration Test Isolation Valve	VUS-PL-V117	2
Electrical Penetration Test Isolation Valve	VUS-PL-V118	2
Electrical Penetration Test Isolation Valve	VUS-PL-V119	2
Electrical Penetration Test Isolation Valve	VUS-PL-V120	2
Electrical Penetration Test Isolation Valve	VUS-PL-V121	2
Electrical Penetration Test Isolation Valve	VUS-PL-V122	2
Electrical Penetration Test Isolation Valve	VUS-PL-V123	2
Electrical Penetration Test Isolation Valve	VUS-PL-V124	2
Electrical Penetration Test Isolation Valve	VUS-PL-V125	2

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Spare Penetration Test Connection	VUS-PL-V140	2
Spare Penetration Test Connection	VUS-PL-V141	2
Spare Penetration Test Connection	VUS-PL-V142	2
VWS Supply Containment Penetration IRC Test Connection/Vent	VWS-PL-V424	2
VWS Return Containment Penetration ORC Test Connection/Vent	VWS-PL-V425	2
Heat Exchangers		
Normal Residual Heat Removal Heat Exchanger A	RNS-ME-01A	3
Normal Residual Heat Removal Heat Exchanger B	RNS-ME-01B	3
Tanks		
Spent Fuel Pool	FHS-MT-01	3
Fuel Transfer Canal	FHS-MT-02	3
Spent Fuel Cask Loading Pit	FHS-MT-05	3
Passive Containment Cooling Water Storage Tank	PCS-MT-01	3
Water Distribution Bucket	PCS-MT-03	3
Water Collection Troughs	PCS-MT-04	3
Passive RHR Heat Exchanger	PXS-ME-01	3
Accumulator Tank A	PXS-MT-01A	3
Accumulator Tank B	PXS-MT-01B	3
Core Makeup Tank A	PXS-MT-02A	3
Core Makeup Tank B	PXS-MT-02B	3
In-Containment Refueling Water Storage Tank	PXS-MT-03	3
Emergency Air Storage Tank 01	VES-MT-01	3
Emergency Air Storage Tank 02	VES-MT-02	3
Emergency Air Storage Tank 03	VES-MT-03	3
Emergency Air Storage Tank 04	VES-MT-04	3

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List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Emergency Air Storage Tank 05	VES-MT-05	3
Emergency Air Storage Tank 06	VES-MT-06	3
Emergency Air Storage Tank 07	VES-MT-07	3
Emergency Air Storage Tank 08	VES-MT-08	3
Emergency Air Storage Tank 09	VES-MT-09	3
Emergency Air Storage Tank 10	VES-MT-10	3
Emergency Air Storage Tank 11	VES-MT-11	3
Emergency Air Storage Tank 12	VES-MT-12	3
Emergency Air Storage Tank 13	VES-MT-13	3
Emergency Air Storage Tank 14	VES-MT-14	3
Emergency Air Storage Tank 15	VES-MT-15	3
Emergency Air Storage Tank 16	VES-MT-16	3
Emergency Air Storage Tank 17	VES-MT-17	3
Emergency Air Storage Tank 18	VES-MT-18	3
Emergency Air Storage Tank 19	VES-MT-19	3
Emergency Air Storage Tank 20	VES-MT-20	3
Emergency Air Storage Tank 21	VES-MT-21	3
Emergency Air Storage Tank 22	VES-MT-22	3
Emergency Air Storage Tank 23	VES-MT-23	3
Emergency Air Storage Tank 24	VES-MT-24	3
Emergency Air Storage Tank 25	VES-MT-25	3
Emergency Air Storage Tank 26	VES-MT-26	3
Emergency Air Storage Tank 27	VES-MT-27	3
Emergency Air Storage Tank 28	VES-MT-28	3
Emergency Air Storage Tank 29	VES-MT-29	3
Emergency Air Storage Tank 30	VES-MT-30	3

Table 3I.6-3 (Sheet 32 of 32)
List Of AP1000 Safety-Related Electrical and Mechanical Equipment Not High Frequency Sensitive

Description	AP1000 Tag Number	Comment
Emergency Air Storage Tank 31	VES-MT-31	3
Emergency Air Storage Tank 32	VES-MT-32	3
Main Feed Pump A Status	ECS-ES-3-XXX	4
Main Feed Pump B Status	ECS-ES-4-XXX	4
Main Feed Pump C Status	ECS-ES-5-XXX	4

Notes:

1. Rugged AP1000 safety-related equipment with no moving parts required in demonstrating functional operability during a seismic event is considered to be not sensitive to HRHF seismic loadings. Seismic qualification is based on the seismic loads associated with the mounting location of the safety-related equipment as a minimum. AP1000 CSDRS seismic loads at the mounting location of the safety-related equipment produces comparable or higher equipment stresses and deflections than the HRHF seismic loadings based on the work reported in APP-GW-GLR-115, "Effect of High Frequency Seismic Content on SSCs." For rugged safety-related line-mounted equipment being qualified by test, seismic testing will be performed in compliance with IEEE Standard 382-1996 with a required input motion (RIM) curve extended to 64 Hz typically to a peak acceleration of 6g.
2. AP1000 safety-related valves are seismic qualified in accordance with ASME code for structural integrity to a maximum acceleration of 6g in all three principal orthogonal axes. AP1000 CSDRS seismic loads at the mounting location of the safety-related equipment produce comparable or higher equipment stresses and deflections than the HRHF seismic loadings based on the work reported in APP-GW-GLR-115, "Effect of High Frequency Seismic Content on SSCs." For rugged safety-related line-mounted equipment being qualified by test, seismic testing will be performed in compliance with IEEE Standard 382-1996 with a required input motion (RIM) curve extended to 64 Hz typically to a peak acceleration of 6g.
3. Seismic qualification is based on structural integrity alone to the seismic loadings associated with the mounting location of the safety-related equipment as a minimum. AP1000 CSDRS seismic loads at the mounting location of the safety-related equipment produce comparable or higher equipment stresses and deflections than the HRHF seismic loadings based on the work reported in APP-GW-GLR-115, "Effect of High Frequency Seismic Content on SSCs."
4. Seismic qualification is not required.

Table 3I-201
Not Used

Table 3I-202
Not Used

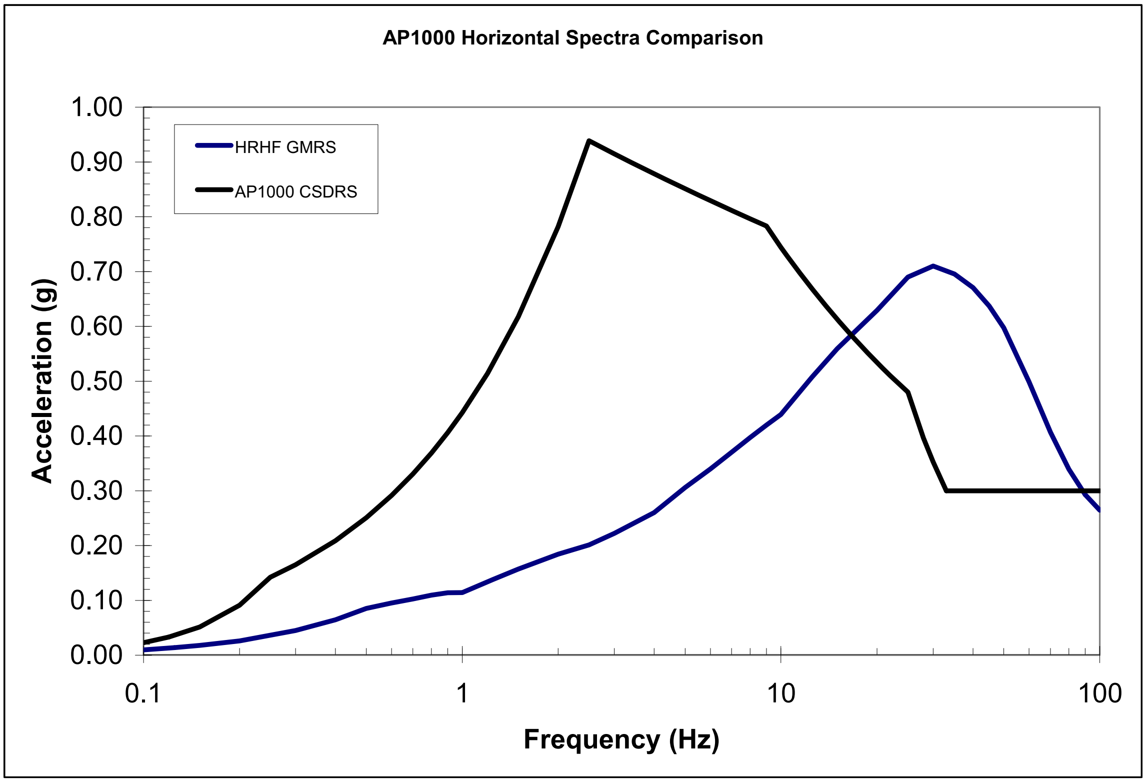

Figure 3I.1-1
Comparison of Horizontal AP1000 CSDRS and HRHF Envelope Response Spectra

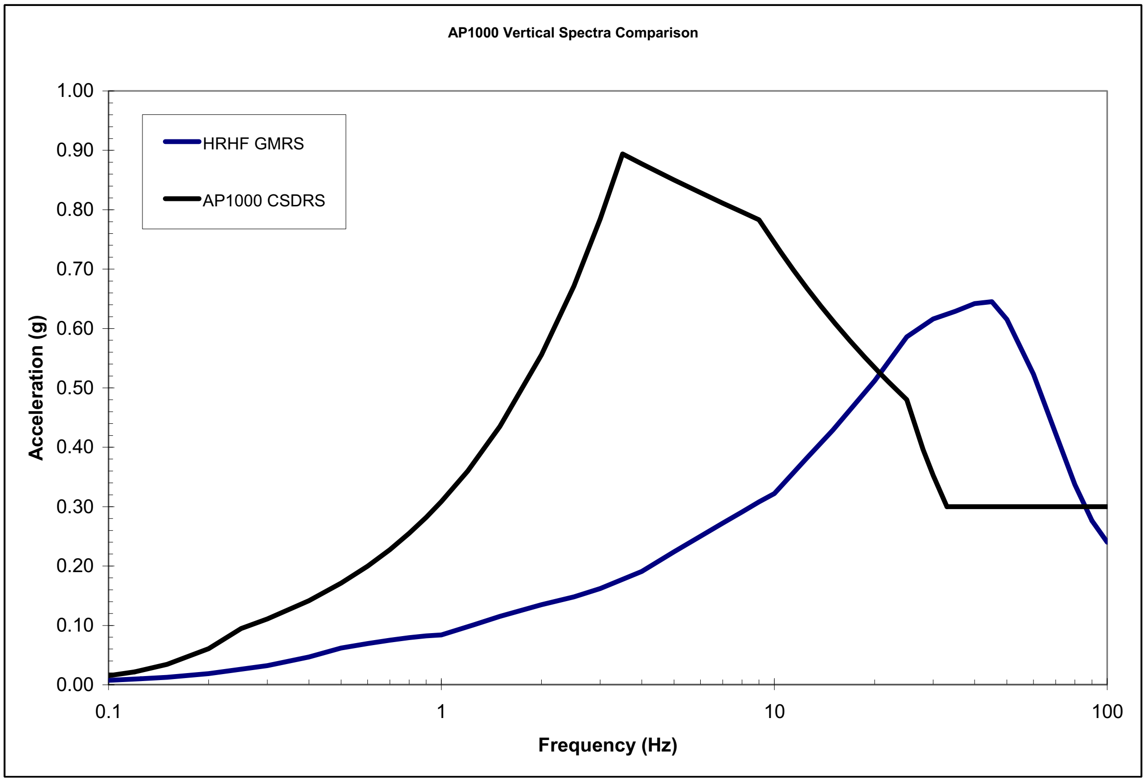

Figure 3I.1-2
Comparison of Vertical AP1000 CSDRS and HRHF Envelope Response Spectra

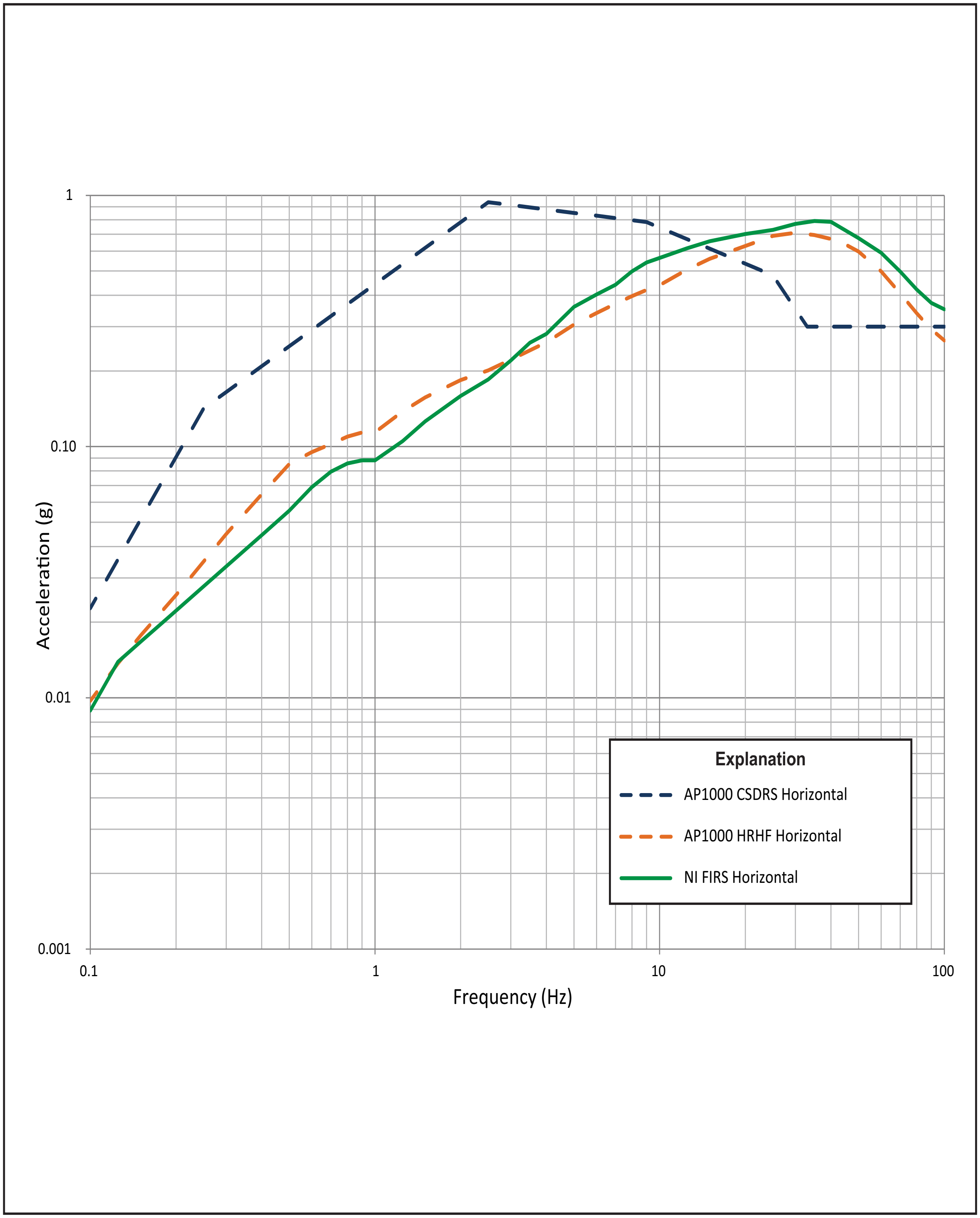

Figure 3I.1-201
Design Ground Motion Response Spectra - NI FIRS Horizontal

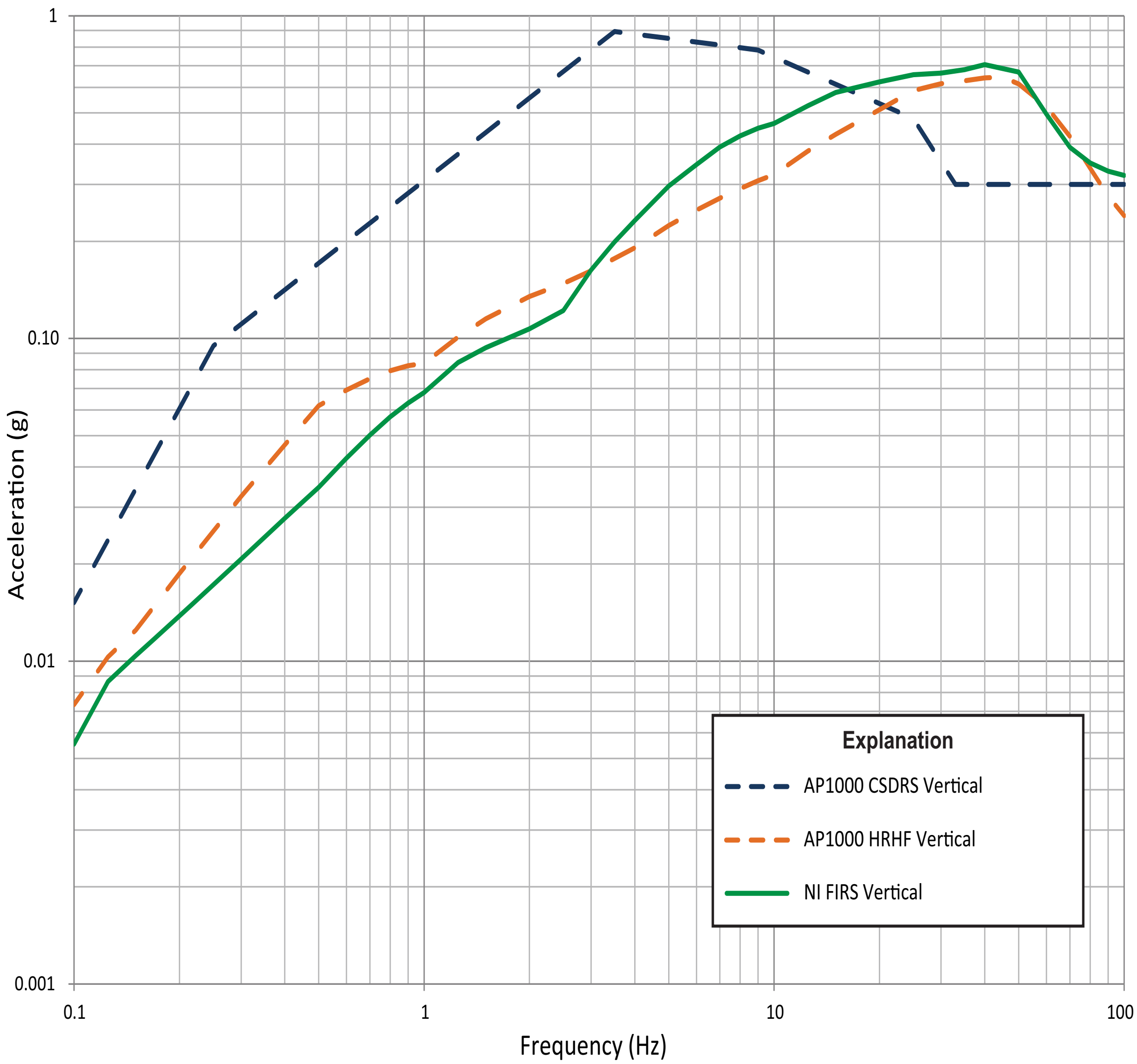

Figure 3I.1-202
Design Ground Motion Response Spectra - NI FIRS Vertical