SOUTHERN COMPANY

UFSAR Table of Contents

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Color	Description
Black	Original Westinghouse AP1000 DCD Revision 19 content
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Blue	Standard FSAR content
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Chapter 3 Design of Structures, Components, Equipment and Systems

3.1 Conformance with Nuclear Regulatory Commission General Design Criteria

This section discusses the extent to which the AP1000 design criteria for safety-related structures, systems, and components comply with 10 CFR 50, Appendix A. As presented in this section, each criterion is first quoted and then discussed. For some criteria, the AP1000 advanced passive design features are deemed to be significantly different in certain specific areas from those design features considered when the General Design Criteria were formulated. In those instances, the means by which the AP1000 design complies with the intent of the General Design Criterion is indicated. Where additional information is required for a complete discussion, the appropriate Design Control Document (DCD) sections are referenced.

3.1.1 Overall Requirements

Criterion 1 – Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified, as necessary, to assure a quality product, in keeping with the required safety function.

A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

AP1000 Compliance

The Quality Assurance Program for the AP1000 provides confidence that safety-related items and services are designed, procured, fabricated, inspected, and tested to quality standards commensurate with the safety-related functions to be performed. This program also applies to design services subcontracted to external organizations. The quality assurance program for erection of structures, systems, and components will be identified before the construction phase of the AP1000 project. The AP1000 quality assurance program is described in Chapter 17, including its compliance with ASME NQA-1.

Design, procurement, fabrication, inspection, and testing are performed according to recognized codes, standards, and design criteria that comply with the requirements of 10 CFR 50.55a. As necessary, supplemental standards, design criteria, and requirements are developed by the AP1000 designers. A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(a)(3). The alternate classification is discussed in Subsection 5.2.1.3.

Appropriate records documenting that design, procurement, fabrication, inspection, and testing comply with the applicable codes, standards, and design criteria are maintained according to appropriate, applicable laws and regulations, either by or under the control of the Combined License applicant.

In the passive AP1000 design, systems necessary to provide the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures

comparable to the guideline exposures of 10 CFR 50.34 are classified as safety-related. Therefore, the AP1000 complies with the intent of Criterion 1.

The principal design criteria, design bases, codes, and standards applied to the facility are identified in Section 3.2. Additional details may be found in the pertinent sections dealing with safety-related structures, systems, and components.

Criterion 2 – Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of the capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

AP1000 Compliance

The safety-related structures, systems, and components are designed to withstand the effects of natural phenomena without loss of the capability to perform their safety-related functions, or are designed such that their response or failure will be in a safe condition. Those structures, systems, and components vital to the shutdown capability of the reactor are designed to withstand the maximum probable natural phenomena at the intended site.

Accident analyses consider conservative site conditions that envelope expected sites. Appropriate combinations of structural loadings from normal, accident, and natural phenomena are considered in the plant design. The design of the plant in relationship to those natural phenomena is addressed.

Seismic and quality group classifications and other pertinent standards and information are given in the sections discussing individual structures, systems, and components as well as in Chapter 3. The nature and magnitude of the natural phenomena considered in the design of this plant are discussed in Chapter 2.

Criterion 3 – Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

AP1000 Compliance

The safety-related structures, systems, and components are designed to minimize the probability and effect of fires and explosions. Noncombustible and fire-resistant materials are used in the containment and main control room. Additionally noncombustible and fire-resistant materials are used on components of safety-related systems, and elsewhere in the plant where fire is a potential risk to safety-related systems.

For example, electrical cables have a fire-retardant jacketing, and fire barriers are used at fire area boundaries. The AP1000 design approach includes designing the safety-related systems with redundant divisions, and locating these redundant divisions in separate safety-related areas.

Equipment and facilities for fire protection, including detecting, alarming, and extinguishing functions, are provided to help protect both plant equipment and personnel from fire, explosion, and the resultant release of toxic vapors. Fire protection is provided by deluge systems (water spray), sprinklers, and portable extinguishers. Fire fighting systems are designed so that their rupture or inadvertent operation will not prevent safety-related systems from performing their design functions.

The following codes, guides, and standards are used as guidelines in the design of the fire protection system and equipment. The system and equipment conform to the applicable portions of the following documents:

- National Fire Protection Association Codes and Standards
- BTP-CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," July 1981

Subsection 9.5.1 describes the AP1000 fire protection system and equipment, including conformance with the applicable portions of these codes and standards and reference to specific fire protection codes and standards.

Criterion 4 – Environmental and Missile Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

AP1000 Compliance

Safety-related structures, systems, and components are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents.

The AP1000 design has emphasized the minimization of missiles, pipe whip, and fluid discharge by a combination of separation of safe shutdown components and design to prevent the dynamic effects of postulated pipe ruptures based on the application of the leak-before-break approach. This analysis is discussed in Subsection 3.4.3.5 and Section 3.6.

The AP1000 structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. Details of the design, environmental testing, and construction of these structures, systems, and components are given in the sections that discuss individual structures, systems, and components, as well as in Sections 3.5 and 3.6.

Criterion 5 – Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit.

AP1000 Compliance

The AP1000 is a single-unit plant. If more than one unit were built on the same site, none of the safety-related systems would be shared.

3.1.2 Protection by Multiple Fission Product Barriers

Criterion 10 – Reactor Design

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

AP1000 Compliance

The reactor core and associated coolant, control, and protection systems are designed to the following criteria:

- No fuel damage occurs during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II). For normal operation, the plant is designed to accommodate a fuel defect level of up to 0.25 percent. Fuel damage, as used here, is defined as penetration of the fission product barrier, that is, the fuel rod cladding. The small number of clad defects that may occur are within the capability of the plant cleanup system and are consistent with the plant design bases. For additional information see Section 11.1.

- The reactor can be returned to a safe shutdown state following a Condition III event, with only a small fraction of the fuel rods damaged, although sufficient fuel damage might occur to preclude the immediate resumption of operation.

- The core remains intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

The reactor protection system is designed to actuate a reactor trip whenever necessary to prevent exceeding the fuel design limits. The core design, together with the process and decay heat removal systems, provide this capability under expected conditions of normal operation, with appropriate margins for uncertainties and anticipated transient situations. This includes the effects of the loss of reactor coolant flow, trip of the turbine generator, loss of normal feedwater, and loss of both normal and preferred power sources.

Chapter 4, Reactor, describes the mechanical components of the reactor and reactor core, including the fuel rods and fuel assemblies, the mechanical design, nuclear design, and the thermal hydraulic design. Chapter 7 provides details of the control and protection systems instrumentation design and logic. This information supports the accident analyses documented in Chapter 15. The acceptable fuel design limits are not exceeded for Condition I and II events. Acceptable core cooling is provided for Condition III and IV events.

Criterion 11 – Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power-operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

AP1000 Compliance

When the reactor is critical, the negative fuel temperature reactivity effects (Doppler feedback) provide prompt reactivity feedback to compensate for a rapid, uncontrolled reactivity excursion. The negative Doppler coefficient of reactivity is provided by the use of a low-enrichment fuel design. This Doppler feedback is the primary reactivity feedback mechanism to provide the inherent core reactivity protection during rapid core reactivity excursions.

For slower reactivity transients that result in moderator temperature increases, the nonpositive moderator temperature coefficient of reactivity provides compensatory reactivity feedback to help control these slower transients. The overall core design establishes a nonpositive moderator temperature coefficient of reactivity.

Chapter 4 provides information pertaining to the core design.

Criterion 12 – Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

AP1000 Compliance

Power oscillations of the fundamental mode are inherently eliminated by negative Doppler and nonpositive moderator temperature coefficients of reactivity.

Oscillations, due to xenon spatial effects, in the radial and azimuthal overtone modes are heavily damped because of the inherent design and due to the negative Doppler and nonpositive moderator temperature coefficients of reactivity.

Oscillations due to xenon spatial effects may occur in the axial first overtone mode. Reactor trip functions are provided, using the measured axial power imbalance as an input, so that the fuel design limits are not exceeded during axial xenon oscillations.

If it is necessary to maintain axial imbalance within the limits (that is, imbalances that are alarmed to the operator and are within the imbalance trip setpoints), the operator can suppress axial xenon oscillations by control rod motions or temporary power reductions or both.

Oscillations due to spatial xenon effects, in axial modes higher than the first overtone, are heavily damped because of the inherent design and the negative Doppler coefficient of reactivity.

The stability of the core against xenon-induced power oscillations and the functional requirements of instrumentation for monitoring and measuring core power distribution are discussed in Chapter 4. Details of the instrumentation design and logic are discussed in Chapter 7.

Criterion 13 – Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its

associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

AP1000 Compliance

Instrumentation and controls are provided to monitor and control neutron flux, control rod position, fluid temperatures, pressures, flows, and levels, as necessary, to maintain plant safety. Instrumentation is provided in the reactor coolant system, steam and power conversion system, containment, engineered safety systems, radioactive waste management systems, and other auxiliary systems.

See Section 7.5 for a discussion of indications that are required for operator use under normal operating and accident conditions. Criteria regarding layout of the controls and displays are provided in Chapter 18.

The quantity and types of process instrumentation used provide safe and orderly operation of systems over the design range of plant operations, including accident conditions.

Criterion 14 – Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

AP1000 Compliance

The reactor coolant pressure boundary is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, while maintaining stresses within applicable limits. Consideration is given to loadings under normal operating conditions and to abnormal loadings, such as seismic loadings. The piping is protected from overpressure by means of pressure-relieving devices, as required by ASME Code, Section III. See Subsection 5.2.2 for additional information.

Reactor coolant pressure boundary materials and fabrication techniques are such that there is a low probability of gross rupture or significant leakage. The AP1000 reactor coolant system design incorporates revised pipe-break criteria (leak-before-break) to reduce or eliminate the need to consider the dynamic effects of pipe breaks. The configuration and materials of the reactor coolant system have been selected such that the pipe stresses meet the leak-before-break criteria. See Subsection 3.6.3 for additional information.

The AP1000 reactor core and reactor internals are designed to limit neutron fluence on the reactor vessel. See Section 5.4 and Chapter 4 for additional information.

The reactor vessel is manufactured from low-alloy carbon steel clad with 308L stainless steel weld overlay on wetted surfaces. The vessel shell is constructed of ring-rolled forgings that eliminate vertical weld seams. Chemical composition of the forging material is controlled to improve radiation resistance of the vessel. (See Criterion 31 for further discussion of the reactor coolant pressure boundary.)

Coolant chemistry is controlled to protect the materials of construction of the reactor coolant pressure boundary from corrosion. See Subsection 5.2.3 for additional information.

The reactor coolant pressure boundary welds are accessible for in-service inspections to assess structural and leaktight integrity. For the reactor vessel, a material surveillance program is provided. Instrumentation is provided to detect significant leakage from the reactor coolant pressure boundary, with indication in the main control room. See Subsection 5.2.4 for additional information.

A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary is nonsafety-related. This portion of the system is capable of being automatically isolated by safety-related valves that are designed and qualified for the design requirements.

Criterion 15 – Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation, including anticipated operational occurrences.

AP1000 Compliance

Steady-state and transient analyses are performed to demonstrate that reactor coolant system design conditions are not exceeded during normal operation. Protection and control setpoints are based on these analyses. See Chapter 15 for additional information.

The reactor coolant system stress analysis and the leak-before-break analyses are described in Appendices 3B and 3C. See Section 5.3 for additional information.

Two safety valves are provided for the reactor coolant system. These valves and their setpoints meet the ASME Code, Section III criteria for overpressure protection. See Subsection 5.2.2 for additional information.

Criterion 16 – Containment Design

The reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

AP1000 Compliance

The containment is an integral part of the overall containment system, whose function is to contain the release of airborne radioactivity following postulated design basis accidents and to provide shielding for the reactor core and the reactor coolant system during normal operations. The containment consists of a steel containment vessel and is surrounded by a concrete shield building.

The containment vessel, which is a free-standing steel shell, is an integral part of the passive containment cooling system, whose function is to provide the safety-related ultimate heat sink for the removal of the reactor coolant system sensible heat, core decay heat, and stored energy. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents.

The containment is designed to house the reactor coolant system and other related systems. The containment vessel functions as an essentially leaktight barrier. It is protected against postulated missiles from external sources as well as missiles produced by internal equipment failures.

Containment penetrations are isolated according to the provisions of GDCs 54, 55, 56, and 57.

Criterion 17 – Electrical Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming that the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational

occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system shall have sufficient independence, redundancy, and testability to perform their safety functions, assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights-of-way) designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time, following a loss of all onsite alternating current power supplies and other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss of coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

AP1000 Compliance

The AP1000 plant design supports an exemption to the requirement of GDC 17 for two physically independent offsite circuits by providing safety-related passive systems for core cooling and containment integrity, and multiple nonsafety-related onsite and offsite electric power sources for other functions. See Section 6.3 for additional information on the systems for core cooling.

A reliable dc power source supplied by batteries provides power for the safety-related valves and instrumentation during transient and accident conditions.

The Class 1E dc and UPS system is the only safety-related power source required to monitor and actuate the safety-related passive systems. Otherwise, the plant is designed to maintain core cooling and containment integrity, independent of nonsafety-related ac power sources indefinitely. The only electric power source necessary to accomplish these safety-related functions is the Class 1E dc and UPS power system which includes the associated safety-related 120V ac distribution switchgear.

Although the AP1000 is designed with reliable nonsafety-related offsite and onsite ac power that are normally expected to be available for important plant functions, nonsafety-related ac power is not relied upon to maintain the core cooling or containment integrity.

The nonsafety-related ac power system is designed such that plant auxiliaries can be powered from the grid under all modes of operation. During loss of offsite power, the ac power is supplied by the onsite standby diesel-generators. Preassigned loads and equipment are automatically loaded on the diesel-generators in a predetermined sequence. Additional loads can be manually added as required. The onsite standby power system is not required for safe shutdown of the plant.

Criterion 18 – Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional

performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

AP1000 Compliance

The AP1000 is designed so that only the Class 1E dc and UPS system is required in order to initiate and actuate the systems necessary for maintaining core cooling and containment integrity. The safety-related dc power system design complies with GDC 18. Compliance with GDC 18 is achieved by designing testability and inspection capability into the system. The associated testing requirements are contained in Chapter 16.

Criterion 19 – Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

AP1000 Compliance

The AP1000 main control room provides the man-machine interfaces required to operate the plant safely and efficiently under normal conditions and to maintain it in a safe manner under accident conditions, including LOCAs. Simplified passive safety-related system designs are provided that do not rely upon operator action to maintain core cooling for design basis accidents. Operator action outside the main control room to mitigate the consequences of an accident is permitted.

The main control room is shielded by the containment and auxiliary building from direct gamma radiation and inhalation doses resulting from the postulated release of fission products inside containment. Refer to Chapter 15 for additional information on accident conditions. The main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS) allows access to and occupancy of the main control room under accident conditions as described in Subsection 9.4.1. Sufficient shielding and the main control room/control support area HVAC subsystem provide adequate protection so that personnel will not receive radiation exposure in excess of 5 rem whole-body or its equivalent to any part of the body for the duration of the accident.

If ac power is unavailable for more than 10 minutes or if "high-high" particulate, low pressurizer pressure is detected, or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The main control room emergency habitability system also allows access to and occupancy of the main control room under accident conditions. The emergency main control room habitability system is designed to satisfy seismic Category I requirements as described in Section 3.2; the system design is described in Section 6.4.

In the event that the operators are forced to abandon the main control room, a workstation is provided with remote shutdown capability. A main control room evacuation is not assumed to occur simultaneously with design basis events. The remote shutdown workstation is described in Section 7.4.

3.1.3 Protection and Reactivity Control Systems

Criterion 20 – Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

AP1000 Compliance

The protection system is a microprocessor-based system that trips the reactor and actuates engineered safety features when predetermined limits are exceeded or when manually initiated.

The reactor trip portion of the protection system includes four independent, redundant, physically separated, electrically-isolated divisions. The coincidence circuits guard against the loss of protection or the generation of false protection signals due to equipment failures through the use of a two-out-of-four logic and built-in operational bypasses.

Independent, redundant, physically separated, electrically-isolated engineered safety features trains are provided. Signal conditioning for the plant sensors is provided.

See Chapter 7 for additional information concerning the design of the protection system.

Criterion 21 – Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (l) no single failure results in the loss of the protection function and (2) removal from service of any component or channel does not result in the loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

AP1000 Compliance

The protection system is designed for functional reliability and in-service testability. The design employs redundant logic trains and measurement and equipment diversity.

The protection system equipment includes integral testing circuits. System equipment, from input to output, in the protection cabinets and the engineered safety features cabinets, is tested. Simulated inputs replace the field signals. Outputs are monitored for validity. Manual and automatic testing is used to test the final stages of the reactor trip circuits and the reactor trip switchgear. Testing of cabinets and communications links verifies the functional operation of the equipment and the hardware. See Chapter 7 for further information concerning the test capabilities of the protection system.

Criterion 22 – Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels

do not result in the loss of the protection function or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

AP1000 Compliance

Design of the protection systems includes consideration of natural phenomena, normal maintenance, testing, and accident conditions so that the protection functions are available.

Protection system components are designed, arranged, and qualified for operation in the environment accompanying any emergency situation in which the components are required to function.

Functional diversity has been designed into the system. The extent of this functional diversity is demonstrated for a variety of postulated accidents. Diverse protection functions automatically serve to mitigate the consequences of an event. Chapter 15 identifies the primary and diverse protective functions for each of the analyzed events.

Sufficient redundancy and independence are designed into the protection systems so that no single failure or removal from service of any component or channel of a system results in loss of the protection function. Functional diversity and location diversity are designed into the system.

Automatic reactor trip is initiated by neutron flux measurements, reactor coolant system overtemperature delta-T, reactor coolant system overpower delta-T, pressurizer pressure and level measurements, reactor coolant flow, reactor coolant pump speed, reactor coolant pump bearing water temperature, and steam generator water level measurements. Trips may also be initiated manually or by a safety injection signal.

For additional information pertaining to the reactor trip logic, see Section 7.2.

High-quality components, conservative design and quality control, inspection, calibration, and tests are used to guard against common-mode failure. Qualification testing and analysis are performed on the safety-related systems to demonstrate functional operation at normal and post-accident conditions of temperature, humidity, pressure, and radiation for specified periods, as required. Typical protection system equipment is subjected to type tests under simulated seismic conditions, using conservatively large accelerations and applicable frequencies.

See Section 7.1 for additional information concerning the equipment design of the protection and safety monitoring system.

See Sections 3.10 and 3.11 for information pertaining to environmental and seismic qualification of the protection system equipment.

The AP1000 includes a nonsafety-related diverse actuation system. The diverse actuation system provides specific automatic functions including control rod insertion, turbine trip, passive residual heat removal heat exchanger actuation, core makeup tank actuation, isolation of critical containment lines, and passive containment cooling system actuation. This system is diverse and independent from the reactor protection system from sensors up to the actuation devices.

See Section 7.7 for additional information concerning the diverse actuation system.

Criterion 23 – Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air) or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

AP1000 Compliance

The protection system is designed considering the most probable failure modes of the components under various perturbations of the environment and energy sources. Reactor trip channels are designed on the deenergize-to-trip principle so that a single event (that is, loss of power) that could affect many functions at the same time causes the channels to actuate to their tripped conditions.

Criterion 24 – Separation of Protection and Control Systems

The protection system shall be separated from the control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

AP1000 Compliance

The protection system is separate and distinct from the control systems. Control systems are, in some cases, dependent on the protection system for control signals that are derived from protection system measurements, where applicable. These signals are transferred to the control system by isolation devices classified as protection components.

The adequacy of the system isolation is verified by testing under conditions of postulated credible faults. The failure of a single control system component or channel, or the failure or removal from service of a single protection system component or channel common to the control and protection system, leaves intact a system that satisfies the requirements of the protection system. The removal of a protection division from service is allowed during testing of the division.

Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of the control rods.

AP1000 Compliance

The protection system is designed to limit reactivity transients so that the fuel design limits are not exceeded. Reactor shutdown by control rod insertion is independent of the normal control functions since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Thus, in the postulated accidental withdrawal of a control rod or control rod bank (assumed to be initiated by a control malfunction), neutron flux, temperature, pressure, level, and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

The AP1000 is designed to automatically terminate a boron dilution during manual or automatic operation at power, and also during startup and shutdown conditions. See Chapter 7 for a discussion of the signals used in the logic to terminate a boron dilution. Subsection 9.3.6.4.5 discusses the chemical and volume control system design features for addressing boron dilution. The Chapter 15 safety analyses demonstrate that fuel design limits are not exceeded.

Criterion 26 – Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that the acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

AP1000 Compliance

Two reactivity control systems are provided. These are rod cluster control assemblies and gray rod assemblies, and chemical shim (boric acid). The rod cluster control and gray rod assemblies are inserted into the core by the force of gravity.

During operation, the shutdown rod banks are fully withdrawn. The control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. See Section 4.3 for additional information.

The shutdown and control rod banks are designed to provide reactivity margin to shut down the reactor during normal operating conditions and during anticipated operational occurrences, without exceeding specified fuel design limits. The safety analyses assume the most restrictive time in the core operating cycle and that the most reactive control rod cluster assembly is in the fully withdrawn position. See Chapter 15 for summaries of the analyses, assumptions, and results.

The safety-related passive systems provide the required boration to establish and maintain safe shutdown condition for the reactor core. See Section 6.3 for additional information.

Criterion 27 – Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

AP1000 Compliance

The plant is provided with the means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. Combined use of the control rod and the chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth control rod assembly is assumed to be stuck in the fully withdrawn position for this determination.

Criterion 28 – Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (l) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

AP1000 Compliance

The maximum reactivity worth of the control rods and the maximum rates of reactivity increase employing control rods and boron removal are limited by design and operating procedures.

The appropriate reactivity addition rate for the withdrawal of control rods and the dilution rate of the boric acid in the reactor coolant system are specified in the precautions, limitations, and setpoint document and the control system setpoint study. Technical specifications explicitly specify control rod bank alignment and insertion limits in addition to shutdown margin reactivity requirements.

The control rod reactivity addition rate is determined by the allowable rod control system withdrawal speed, in conjunction with the control rod worth, which varies throughout the operating cycle. The capability to change boron concentration is determined by the various plant systems that provide makeup to the reactor coolant system. The reactivity insertion rates, rod withdrawal limits, and boron dilution limits are discussed in Chapter 4.

Core cooling capability following events such as rod ejection and steam line breaks is provided by keeping the reactor coolant pressure boundary stresses within faulted condition limits, as specified by applicable ASME codes. Structural deformations are also checked and limited to values that do not jeopardize the operation of needed safety-related features.

Criterion 29 – Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

AP1000 Compliance

The protection and reactivity control systems have an extremely high probability of performing their required safety-related functions in the event of anticipated operational occurrences. High-quality equipment, diversity, and redundancy, support this probability. Loss of power to the protection system results in a reactor trip. Defense in depth is designed into AP1000 to reduce challenges to the protection and reactivity control systems.

3.1.4 Fluid Systems

Criterion 30 – Quality of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

AP1000 Compliance

Reactor coolant pressure boundary components are designed, fabricated, inspected, and tested in conformance with the ASME Code, Section III. A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(a)(3). The alternate classification is discussed in Section 5.2.

Leakage detection monitoring is accomplished using instrumentation and other components of several systems. See Subsection 5.2.5 for additional information. Reactor coolant pressure boundary leakage is classified as either identified or unidentified leakage.

Auxiliary systems connected to the reactor coolant pressure boundary incorporate design and administrative provisions that limit leakage. Leakage is detected by increasing auxiliary system level,

temperature, flow, or pressure, by lifting of relief valves, or by increasing values of monitored radiation in the auxiliary system.

Leakage from the reactor coolant pressure boundary and other components not otherwise identified inside the containment will condense and flow by gravity via the floor drains and other drains to the containment sump. Leakage is indicated by an increase in the sump level.

Reactor coolant system inventory monitoring provides an indication of system leakage. The reactor coolant system inventory balance is a quantitative inventory or mass balance calculation.

Leakage from the reactor coolant pressure boundary will result in an increase in the radioactivity levels inside containment. The containment atmosphere is monitored for airborne gaseous radioactivity and F18 particulate. From the concentration of F18 particulate and the power level, reactor coolant pressure boundary leakage can be estimated.

Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (l) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (l) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.

AP1000 Compliance

Control is maintained over material selection and fabrication for the reactor coolant pressure boundary components so that the boundary behaves in a nonbrittle manner. The portion of the chemical and volume control system that uses an alternate classification is not required to meet the requirements to prevent brittle failure. The reactor coolant pressure boundary materials exposed to the coolant are corrosion-resistant stainless steel or nickel-chromium-iron alloy. The nil-ductility transition reference temperature of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR 50, Appendix G (Reference 1). See Section 5.3 for additional information.

The following requirements are imposed in addition to those specified by the ASME Code, Section III.

- A 100 percent volumetric ultrasonic shear wave test of reactor vessel plate and a post-hydrotest ultrasonic map of welds in the pressure vessel are required. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the ASME Code, Section III is also required in order to preclude interpretation problems during in-service inspection.

- In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and l/2T compact tension specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM, E-185 (Reference 2).

- Reactor vessel core region material chemistry (copper, phosphorous, and vanadium) is controlled to reduce sensitivity to embrittlement due to irradiation over the life of the plant.

The fabrication and quality control techniques used in the fabrication of the reactor coolant system are governed by ASME Code, Section III requirements.

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the ASME Code, Section III, Appendix G. The approach specifies that the allowable stress intensity factors for vessel-operating conditions do not exceed the reference stress intensity factor for the metal temperature. Operating specifications include conservative margins for predicted changes in the material reference temperatures due to irradiation.

Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (l) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel.

AP1000 Compliance

The design of the reactor coolant pressure boundary provides accessibility to the internal surfaces of the reactor vessel and most external zones of the vessel, including the nozzle-to-reactor coolant piping welds, the top and bottom heads, and external surfaces of the reactor coolant piping, except for the area of pipe within the primary shield concrete. The inspection capability complements the leakage detection systems in assessing the integrity of the pressure boundary components. The reactor coolant pressure boundary will be periodically inspected under the provisions of the ASME Code, Section XI. Section 5.1 provides the reactor coolant system primary loop drawings. The portion of the chemical and volume control system that uses an alternate classification is constructed to requirements that do not require in-service inspection.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates, forgings, weldments, and associated heat-treated zones is performed according to 10 CFR 50, Appendix H. Additionally, samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes conventional tensile and impact tests and fracture mechanics specimens. The observed shifts in the nil-ductility transition reference temperature of the core region materials with irradiation is used to confirm the allowable limits calculated for operational transients.

The design of the reactor coolant pressure boundary piping provides for accessibility of welds requiring in-service inspection under the provisions of the ASME Code, Section XI. Removable insulation is provided at welds requiring in-service inspection. See Section 5.3 and Subsection 5.2.4 for additional information.

Criterion 33 – Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

AP1000 Compliance

Changes in the reactor coolant volume will be accommodated by the pressurizer level program for normal power changes, including the transition from hot standby to full-power operation and returning to hot standby. In addition, the pressurizer has sufficient volume to accommodate minor reactor coolant system leakage.

Safety-related passive reactor coolant system makeup is provided to accommodate small leaks when the normal makeup system is unavailable and to accommodate larger leaks resulting from loss of coolant accidents. Safety-related reactor coolant makeup and safety injection are provided by two core makeup tanks, two accumulators, and an in-containment refueling water storage tank. Long-term cooling is provided by containment gravity recirculation of reactor coolant within containment. See Section 6.3 for additional information. The safety-related reactor coolant makeup relies on the Class 1E and UPS system. Neither onsite or offsite ac power is required.

In addition, the nonsafety-related chemical and volume control system automatically provides inventory control to accommodate minor leakage from the reactor coolant system, expansion during heatup from cold shutdown, and contraction during cooldown. This inventory control is provided by letdown and makeup connections to the chemical and volume control system purification loop. Redundant pumps with connections to redundant nonsafety-related onsite ac power are provided when offsite power is not available and these pumps can be supplied from offsite power when onsite power is not available. See Section 5.2 for additional information.

Criterion 34 – Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

"Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

AP1000 Compliance

The AP1000 design satisfies the intent of GDC 34 by reducing the risk associated with loss of the decay heat removal function through a combination of safety-related passive systems, together with nonsafety-related active systems. Specific decay heat removal systems include the following:

- A safety-related passive residual heat removal heat exchanger that uses natural circulation flow and that does not require electrical power for operation

- Automatic, safety-related feed and bleed using the core makeup tanks, accumulators, and the in-containment refueling water storage tank for injection and the automatic depressurization system valves for reactor coolant system venting

- The nonsafety-related main feedwater system with motor-driven pumps supplied by the main generator or by offsite power

- The nonsafety-related startup feedwater system with motor-driven pumps supplied by offsite or onsite power, including automatic sequencing on the nonsafety-related diesel generators

- The nonsafety-related normal residual heat removal system with motor-driven pumps supplied by offsite or onsite power, including nonsafety-related diesel generators, for use at low reactor coolant system pressures

A safety-related emergency feedwater system is not required for the AP1000 design. An active safety-related residual heat removal system is not required for the AP1000.

The AP1000 passive core cooling system, in conjunction with the passive containment cooling system, provides a reliable capability for removing decay heat from the reactor core and maintains sufficient water inventory to provide adequate core cooling for an extended period of time. The system does not depend upon pumped injection or recirculation, and actuates automatically, requiring no operator actions.

The containment arrangement addresses the Regulatory Guide 1.82 issues. Functional performance of the system addresses the guidelines of Regulatory Guide 1.139, except that cooldown rate is somewhat more limited when using the passive residual heat removal equipment. See Subsection 1.9.1 for additional information.

The passive core cooling system provides both gravity injection and gravity recirculation, automatically shifting injection modes when the proper containment flood-up conditions are achieved.

The AP1000 design provides a passive decay heat removal system that functions independent of nonsafety-related ac power supplies and can accommodate single active failures. (The Class 1E dc and UPS system supplies power to the safety-related monitoring and control instrumentation.) The passive core cooling system complies with General Design Criterion 34 by providing the capability to remove decay heat without relying on nonsafety-related ac power.

Criterion 35 – Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (l) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

AP1000 Compliance

The AP1000 design provides for safety-related passive reactor coolant makeup. Core makeup tanks accommodate small leaks when the normal makeup system is unavailable and provide safety injection for small-break loss of coolant accidents. Accumulators provide the high makeup flow required for a large loss of coolant accident and initiate injection when the reactor coolant system pressure is below the static accumulator pressure during a small-break loss of coolant accident.

The in-containment refueling water storage tank, and after containment flood-up, containment recirculation capability provide the long-term source of gravity injection to the core after the reactor coolant system is depressurized. The automatic depressurization system valves provide the vent path to transfer the core decay heat to the containment and then to the ultimate heat sink.

The AP1000 design provides a passive core cooling system that functions independent of ac power supplies, assuming single active failures. The passive core cooling system does not need the

nonsafety-related diesel-generators for electrical power to either actuate or operate the various system components. Therefore, the passive core cooling system complies with the intent of GDC 35 by providing the capability for core cooling without relying on nonsafety-related ac power sources.

Criterion 36 – Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

AP1000 Compliance

The AP1000 design includes a passive core cooling system that provides emergency core decay heat removal, emergency reactor coolant system makeup and boration, safety injection, and containment sump pH control. The system piping and components are designed to permit access for periodic inspection and testing of equipment, according to the ASME Code and technical specification requirements, to provide confidence in the integrity and capability of the system.

The core makeup tanks, accumulators, and passive residual heat removal heat exchanger have manways which permit access for inspection and required maintenance. The in-containment refueling water storage tank design provides access for both the tank itself and for the passive residual heat removal heat exchanger, spargers, and other components located inside the tank.

In addition, the system piping provides accessibility for inspection and maintenance to the extent practical. See Section 6.3 for additional information.

Criterion 37 – Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (l) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

AP1000 Compliance

The AP1000 passive core cooling system is designed to permit the periodic inspection and testing of the appropriate system components. The testing capabilities of the system including in-service testing and inspection to confirm the structural and leaktight integrity of various components, technical specification operability and performance of the active system components, and additional in-service testing to confirm the overall operability of the system.

The stage 1, 2, and 3 automatic depressurization system valves have provisions for shutdown in-service testing and at-power operability testing.

Planned shutdown testing includes operability testing of the component and system performance, including operation of applicable portions of the protection and safety monitoring system and the use of the appropriate power sources for the system.

The AP1000 design has significantly reduced the support systems required for system operation. In-service testing of the required support systems is also planned.

Criterion 38 – Containment Heat Removal System

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss of coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

AP1000 Compliance

The AP1000 design uses passive systems for post-loss of coolant accident core and containment heat removal and for the prevention of overpressurization failure of the containment building. Heat is transferred from the containment atmosphere to the steel containment shell by natural convection and condensation. Heat removal from the exterior of the containment shell is enhanced by a directed-flow natural convection design and a passive, external cooling water distribution system.

The AP1000 passive containment cooling system is designed with sufficient capacity to prevent the containment from exceeding its design pressure with no operator action or outside assistance for a minimum of 3 days. After 3 days, limited operator action is required.

The AP1000 passive containment cooling system consists of a steel containment shell and associated water supplies, piping, valves, and air baffle. The passive containment cooling system is a passive system that uses gravity and natural circulation as driving forces. The design of the AP1000 passive containment cooling system does not require the use of any pumps, and it functions independent of nonsafety-related ac power sources for 3 days. Therefore, the passive containment cooling system can function during loss of offsite or onsite power. GDC 38 is satisfied by using appropriate redundancy and by the design of the passive containment cooling system and its reliance on natural forces.

Criterion 39 – Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles and piping, to assure the integrity and capability of the system.

AP1000 Compliance

The AP1000 design uses safety-related passive means for containment heat removal. The design of the system allows for inspection of piping, valves, the containment shell and air baffle, and other components to provide confidence in the integrity and capability of the system.

The periodic inspections specified in the ASME Code and technical specifications provide confidence that the capability of these heat removal systems is retained through plant life.

Criterion 40 – Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (l) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable

portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

AP1000 Compliance

The AP1000 design includes a passive containment cooling system that provides containment heat removal to limit the peak containment pressure following design basis events. The system piping and components are designed to permit access for periodic inspection and testing of equipment, according to the ASME Code and technical specification requirements, to provide confidence in the integrity and capability of the system.

The passive containment cooling water storage tank design allows access for both the tank and for the various components located inside the tank.

In addition, the system piping provides accessibility for inspection and maintenance to the extent practical. See Section 6.2 for additional information.

Criterion 41 – Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided, as necessary, to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

AP1000 Compliance

Fission product control for the AP1000 plant is provided via natural removal processes within containment and by limiting containment leakage. The passive removal processes such as deposition and sedimentation are evaluated based on a physically-based source term with large scale core damage. See Section 6.5 for additional details. The containment and penetration design includes features specifically designed to minimize overall containment leakage. See Subsection 6.2.3 for additional details.

The generation of hydrogen in the containment under post-accident conditions has been evaluated, and the containment hydrogen control system has been designed such that the following criteria are satisfied:

- In compliance with Section 50.44 of 10 CFR 50, means are provided to measure and control post-loss of coolant accident hydrogen concentrations.

- The combustible concentrations of hydrogen do not accumulate in the areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

- Internal passive autocatalytic recombiners are provided for hydrogen control following a design basis loss of coolant accident.

- Hydrogen igniters are provided to limit local and global hydrogen concentrations to below 10 percent following a degraded core event with the reaction of 100 percent of the zircaloy cladding.

- The concentration of uniformly distributed hydrogen produced by the equivalent of a 75 percent active fuel-clad metal water reaction does not exceed 13 percent by volume during and following a degraded core event. (The AP1000 containment volume is large enough to provide passive protection for the hydrogen produced by 75 percent zircaloy cladding reaction following a severe accident.)

- The nonsafety-related ventilation system, normally used during refueling, is designed with the capability for a controlled purge of the containment atmosphere to assist in post-accident cleanup, but is not required for hydrogen control.

Criterion 42 – Inspection of Containment Atmosphere Cleanup System

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

AP1000 Compliance

The containment atmosphere cleanup systems are designed and located so that they can be inspected periodically, as appropriate.

Criterion 43 – Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

AP1000 Compliance

The appropriate portions of the containment atmosphere cleanup system are designed to permit periodic pressure and functionality testing.

As described in GDC 41, the containment atmosphere cleanup system has no safety-related post-accident cleanup functions. Dose mitigation is passively provided by the containment isolation and integrity, natural removal processes, and limited containment leakage. Periodic containment integrity is verified in accordance with 10 CFR 50 Appendix J testing as described in Subsection 6.2.3.

Criterion 44 – Cooling Water

A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

AP1000 Compliance

The passive containment cooling system is the ultimate heat sink for the AP1000 and does not rely upon offsite or onsite ac power sources. Heat transfer by convection from the containment shell to the atmosphere meets the intent of GDC 44. Additional information is provided in the responses for GDC 34 and GDC 38.

Criterion 45 – Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

AP1000 Compliance

Refer to the discussion provided for GDC 39.

Criterion 46 – Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss of coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

AP1000 Compliance

Refer to the discussion provided for GDC 40.

3.1.5 Reactor Containment

Criterion 50 – Containment Design Basis

The reactor containment structure, including access opening, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

AP1000 Compliance

The design of the containment structure is based on the containment design basis accidents, which include the rupture of a reactor coolant pipe or the rupture of a main steam or feedwater line. The maximum pressure and temperature reached, a description of the calculational model, and input parameters for a containment design basis accident are presented in Section 6.2. The containment design provides margin to the design basis limits.

Criterion 51 – Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in

a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

AP1000 Compliance

Principal load-carrying components of ferritic materials of the reactor containment boundary exposed to the external environment are selected so that they behave in a nonbrittle manner and so that the probability of fracture propagation is minimized. See Subsection 3.8.2 for additional information.

Criterion 52 – Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

AP1000 Compliance

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leakage rate tests according to the requirements of 10 CFR 50, Appendix J.

Criterion 53 – Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows.

AP1000 Compliance

Provisions exist for conducting individual leakage rate tests on containment penetrations. Penetrations are visually inspected and pressure-tested for leak tightness at periodic intervals. Other inspections are performed as required by 10 CFR 50, Appendix J.

Criterion 54 – Piping Systems Penetrating Containment

Piping systems penetrating the primary reactor containment shall be provided with leak detection, isolation and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

AP1000 Compliance

Piping systems penetrating the primary reactor containment are provided with containment isolation valves. Penetrations that close for containment isolation have redundant valving. Automatic isolation valves with air-, solenoid-, or motor-operators, which do not restrict normal plant operation, are periodically tested to verify operability.

The AP1000 containment isolation design satisfies the current NRC requirements including the post-TMI requirements, as discussed in Subsection 1.9.3. In general, this means that two barriers are provided, one inside containment and the other outside containment. Usually these barriers are valves, but in some cases they are closed piping systems not connected to the reactor coolant system or to the containment atmosphere.

The AP1000 design incorporates a reduction in the number of existing penetrations. Most penetrations are normally closed. Those few that are normally open and are required to close use remotely operated valves for isolation that close automatically. See Subsection 6.2.3 for additional information.

Nonessential systems that may be normally open, such as the mini-purge system, are provided with automatic containment isolation valves that close automatically on a containment isolation signal. The containment isolation signal is actuated by the protection and safety monitoring system. See Section 7.3 for additional information.

Piping and electrical containment penetrations are equipped with test connections and test vents or have other provisions to allow periodic leak rate testing so that leakage is within the acceptable limits established in technical specifications consistent with 10 CFR 50, Appendix J.

Criterion 55 – Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided, as necessary, to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, and use characteristics, and physical characteristics of the site environs.

AP1000 Compliance

Lines that penetrate containment that are connected to the reactor coolant pressure boundary are provided with containment isolation valves in accordance with one of the acceptable arrangements as described in GDC 55. Additional information is found in Subsection 6.2.3.

Criterion 56 – Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates the primary reactor containment shall be provided with containment isolation valves as follows, unless it can be

demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside the containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside the containment; or

3. One locked closed isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4. One automatic isolation valve inside and one automatic isolation valve outside the containment. A simple check valve may not be used as the automatic isolation valve outside the containment.

Isolation valves outside the containment shall be located as close to the containment as practical and, upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

AP1000 Compliance

Lines connecting directly with the containment atmosphere and penetrating the reactor containment are normally provided with two isolation valves in series, one inside and one outside the containment, in accordance with one of the acceptable arrangements as described in GDC 56. Isolation of instrument lines for containment pressure measurement is demonstrated on a different basis and does not require isolation valves. Additional information is found in Subsection 6.2.3.

Criterion 57 – Closed System Isolation Valves

Each line that penetrates the primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, locked closed, or capable of remote manual operation. This valve shall be outside the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

AP1000 Compliance

Lines that penetrate the containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere are considered closed systems within the containment and are equipped with at least one containment isolation valve of one of the following types:

- An automatic isolation valve (a simple check valve is not used as this automatic valve)

- A locked-closed valve

This valve is located outside the containment and as close to the containment wall as practical.

3.1.6 Fuel and Reactivity Control

Criterion 60 – Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during

normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for the retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

AP1000 Compliance

Means are provided to control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

The radioactive waste management systems are designed to minimize the potential for an inadvertent release of radioactivity from the facility and to provide confidence that the discharge of radioactive wastes is maintained below regulatory limits of 10 CFR 50, Appendix I, during normal operation. The gaseous radwaste and liquid radwaste processing systems include continuous radiation monitoring of their discharge paths. High radiation automatically closes a discharge isolation valve. The liquid radwaste system also has provisions to prevent inadvertent siphoning of its monitor tank contents which could cause an uncontrolled discharge. The radioactive waste management systems, the design bases, and the estimated amounts of radioactive effluent releases to the environment are described in Chapter 11.

Criterion 61 – Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

AP1000 Compliance

The spent fuel pool cooling system, and the fuel handling and refueling system are designed to provide cooling and shielding for the fuel assemblies stored in the spent fuel pit and to provide purification of the water in the pit. The system design provides adequate safety under normal and postulated accident conditions.

The spent fuel pool cooling system normal system operation is described in Subsection 9.1.3. Sampling of the spent fuel pool water for gross activity, tritium, and particulate matter is conducted periodically. The concentration of tritium in the spent fuel pool water is maintained at less than 0.5 microcuries per gram to provide confidence that the airborne concentration of tritium in the fuel handling area is within the limits specified in 10 CFR 20, Appendix B. See Subsection 12.2.2 for additional information.

The spent fuel pool is designed so that a water level is maintained above the spent fuel assemblies for at least 72 hours following a loss of the spent fuel pool cooling system, without ac power. See Subsection 9.1.2 for additional information.

The spent fuel pool cooling system maintains the water in the in-containment refueling water storage tank consistent with activity requirements of the water in the refueling cavity during a refueling. Two spent fuel pool cooling filters are provided, one downstream of each demineralizer in the purification branch line of each mechanical train. The filters are sized to collect particulates and suspended solids passed by the demineralizer.

The AP1000 spent fuel pool cooling system is not required to operate to mitigate design basis events. In the event the spent fuel pool cooling system is unavailable, the spent fuel pool cooling is provided by the heat capacity of the water in the pool and in the passive sources of makeup water.

Normal HVAC to the spent fuel pool area is provided by a subsystem of the radiologically controlled area ventilation system described in Subsection 9.4.3. No credit is taken for this system in evaluation of fuel handling accidents discussed in Subsection 15.7.4.

Connections to the spent fuel pool are provided at an elevation that prevents inadvertent draining of the water in the pool to an unacceptable level.

The design of spent fuel storage pool and the spent fuel pool cooling system satisfies GDC 61. See Subsection 9.1.3 for additional information.

Criterion 62 – Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

AP1000 Compliance

The restraints, interlocks, and physical arrangement provided for the safe handling and storage of new and spent fuel are discussed in Section 9.1. The spent fuel assemblies are stored in the spent fuel pit until fission product activity is low enough to permit shipment.

Criterion 63 – Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in the fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in the loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

AP1000 Compliance

Instrumentation is provided to monitor spent fuel storage pool temperature and water level. Indication and alarms are provided in the main control room. Area radiation monitoring is provided in the fuel storage area for personnel protection and general surveillance. The area monitor alarms locally and in the main control room.

If radiation levels in the ventilation effluent reach a predetermined point, an alarm is actuated in the main control room, and the ventilation discharge path is automatically transferred through filter absorber units that provide filtration before discharge from the plant vent.

Criterion 64 – Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss of coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

AP1000 Compliance

The containment atmosphere is monitored during normal and transient operations by the containment gaseous radiation monitors. Under accident conditions, samples of the containment atmosphere taken via the sampling system provide data on airborne radioactive concentrations within the containment.

No reactor coolant fluids are required to be recirculated outside of containment following an accident. Radioactivity levels contained in the facility effluent and discharge paths and in the plant environs are monitored during normal and accident conditions by the plant radiation monitoring systems. High radiation in a discharge path causes automatic closure of the discharge isolation valve.

Area radiation monitors (ARMs) are provided to supplement the personnel and area radiation survey provisions of the AP1000 health physics program described in Section 12.5 and to comply with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, and Regulatory Guides 1.97, 8.2, 8.8, and 8.12. In addition to the installed detectors, periodic plant environmental surveillance is established.

Measurement capability and reporting of effluents are based on the guidelines of Regulatory Guides 1.4 and 1.21, as discussed in Subsection 1.9.1. Additional information is contained in Chapters 11 and 12.

3.1.7 Combined License Information

This section contained no requirement for additional information.

3.1.8 References

1. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

2. American Society of Testing Materials E-185, Standard Recommended Practice for Surveillance Test for Nuclear Reactor Vessels, and the requirements for 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

3.2 Classification of Structures, Components, and Systems

Structures, systems, and components in the AP1000 are classified according to nuclear safety classification, quality groups, seismic category, and codes and standards. This section provides the methodology used for safety-related and seismic classification of AP1000 structures, systems, and components. The seismic classification is described in Subsection 3.2.1. Subsection 3.2.2 describes the classification including nuclear safety-related classification and the corresponding codes and standards. Additionally, Subsection 3.2.2 describes nonsafety-related equipment classifications.

3.2.1 Seismic Classification

General Design Criterion 2 requires that nuclear power plant "Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions." 10 CFR 100, Appendix A sets forth the criteria to which the plant design bases demonstrate the capability to function during and after vibratory ground motion associated with the safe shutdown earthquake conditions.

The seismic classification methodology used in AP1000 complies with the preceding criteria, as well as with recommendations stated within Regulatory Guide 1.29. Conformance with the recommendations of Regulatory Guide 1.29 is discussed in Subsection 1.9.1. The methodology classifies structures, systems, and components into three categories: seismic Category I (C-I), seismic Category II (C-II) and non-seismic (NS).

Seismic Category I applies to both functionality and integrity, and seismic Category II applies only to integrity. Non-seismic items located in the proximity of safety-related items, the failure of which during a safe shutdown earthquake could result in loss of function of safety-related items, are designated as seismic Category II.

There are no safety-related structures, systems, or components outside the scope of the DCD, except for engineered fill which is classified as a Seismic Category I, safety-related structure. See Table 3.2-2. Refer to Subsection 2.5.4 for a discussion of safety-related backfill.

The nonsafety-related structures, systems, and components outside the scope of the DCD are classified as non-seismic (NS).

3.2.1.1 Definitions

3.2.1.1.1 Seismic Category I (C-I)

Seismic Category I applies to, in general, safety-related structures, systems, and components, Seismic Category I also applies to those structures, systems, and components required to support or protect safety-related structures, systems, and components. The exceptions to this general rule are a limited number of structures, such as those required for tornado missile protection, which do not have a safety-related function to perform during or following a seismic event. (See Subsection 3.2.2.3.)

Safety-related items are those necessary to provide for the following:

- The integrity of the reactor coolant pressure boundary
- The capability to shut down the reactor and maintain it in a safe shutdown condition
- Capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34.

Seismic Category I structures, systems, and components are designed to withstand the appropriate seismic loads, as discussed in Section 3.7, and other applicable loads without loss of function. Seismic Category I structures are protected from interaction with adjacent non-seismic structures as described in Subsection 3.7.2.8.

Systems and components identified as safety-related systems and components in Table 3.2-3, and electrical and instrumentation components identified in Table 3.11-1, are the systems and components necessary for continued operation that must remain functional without undue risk of the health and safety of the public during and following an operating basis earthquake. Systems and components identified as Equipment Class A, B, and C in Table 3.2-3, and electrical and instrumentation components identified in Table 3.11-1, are the systems and components that per the criteria of 10 CFR Part 50, Appendix S, must be demonstrated, prior to resuming operations, to have no functional damage following a seismic ground motion exceeding the operating basis earthquake ground motion. See Section 3.7 for information on the operating basis earthquake.

Seismic Category I structures, systems, and components meet the quality assurance requirements of 10 CFR 50, Appendix B. The criteria used for the design of seismic Category I structures, systems, and components are discussed in Section 3.7.

3.2.1.1.2 Seismic Category II (C-II)

Seismic Category II applies to plant structures, systems, and components which perform no safety-related function, and the continued function of which is not required. Seismic Category II applies to structures, systems, and components designed to prevent their collapse under the safe shutdown earthquake. Structures, systems and components are classified as seismic Category II to preclude their structural failure during a safe shutdown earthquake or interaction with seismic Category I items which could degrade the functioning of a safety-related structure, system, or component to an unacceptable level, or could result in incapacitating injury to occupants of the main control room.

Seismic Category II structures, systems, and components are designed so that the safe shutdown earthquake does not cause unacceptable structural failure of or interaction with seismic Category I items. Seismic Category II fluid systems require an appropriate level of pressure boundary integrity if located near sensitive equipment.

The criteria used for the design of seismic Category II structures, systems, and components are discussed in Section 3.7.

Pertinent portions of 10 CFR 50, Appendix B apply to the analysis and design of seismic Category II structures, systems, and components. The quality assurance requirements for the analysis and design of seismic Category II structures, systems, and components are performed in accordance with the Westinghouse AP1000 quality plan as described in Section 17.3 and are sufficient to provide that these components will meet the requirement to not cause unacceptable structural failure of or interaction with seismic Category I items. See Section 17.7 for the Combined License applicant quality assurance program requirement.

3.2.1.1.3 Non-Seismic

Non-seismic (NS) structures, systems, and components are those that are not classified seismic Category I or Category II.

The criteria used for the design of non-seismic structures, components and systems are discussed in Section 3.7.

The non-seismic lines and associated equipment are routed, to the extent practicable, outside of safety-related buildings and rooms to avoid adverse system interactions. In cases where these lines are routed in safety-related areas, the non-seismic item is evaluated for the safe shutdown earthquake and is upgraded to seismic Category II if a credible failure could cause an unacceptable interaction.

Although the seismic category for an item located in the proximity of safety-related structures, systems, and components may be upgraded to seismic Category II, its pre-assigned equipment class remains unchanged.

3.2.1.2 Classifications

Table 3.2-1 illustrates the general relationship between safety-related equipment classes and seismic categories. In most cases, except as noted in Subsection 3.2.2.5, safety-related items are also seismic Category I items. When portions of systems are identified as seismic Category I, the boundaries of seismic Category I portions of the system are shown on the piping and instrumentation diagram (P&ID) of that system. See Subsection 1.7.2 for a list of the piping and instrumentation diagrams.

3.2.1.3 Classification of Building Structures

Building structures are assigned a seismic category as indicated in Table 3.2-2. Codes and standards used in the design and construction of building structures are given in Section 3.8. The building structures are not assigned a safety classification in Subsection 3.2.2 with the exception of the containment vessel.

3.2.2 AP1000 Classification System

The assignment of safety-related classification and use of codes and standards conforms to the requirements of 10 CFR 50.55a for the development of a Quality Group classification and the use of codes and standards. The description of the equipment classification which follows identifies the classifications requiring the full 10 CFR 50, Appendix B quality assurance program as described in Chapter 17 and the Quality Group associated with each classification.

The classification system provides a means of identifying the extent to which structures, systems, and components are related to safety-related and seismic requirements. The classification system provides an easily recognizable means of identifying the extent to which structures, systems, and components are related to ANS nuclear safety classification, NRC quality groups, ASME Code, Section III classification, seismic category and other applicable industry standards, as shown in Table 3.2-3.

There are no safety-related structures, systems, or components outside the scope of the DCD, except for engineered fill which is classified as a Seismic Category I, safety-related structure. See Table 3.2-2. Refer to Subsection 2.5.4 for a discussion of safety-related backfill.

3.2.2.1 Classification Definitions

The definitions used in the classification of structures, systems and components are provided in the following. Unless otherwise noted these definitions apply throughout the Design Control Document. These definitions are consistent with the ANS Definitions for Light Water Reactor Standards (ANS-58.14-1993).

Safety-related is a classification applied to items relied upon to remain functional during or following a design basis event to provide a safety-related function. Safety-related also applies to documentation and services affecting a safety-related item.

Safety-related function is a function that is relied upon during or following a design basis event to provide for the following:

- The integrity of the reactor coolant pressure boundary
- The capability to shut down the reactor and maintain it in a safe shutdown condition
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34.

Design basis event is an event that is a condition of normal operation (including anticipated operational occurrences), a design basis accident, an external event, or natural phenomena for which the plant must be designed so that the safety-related functions are achievable.

Design basis accidents and transients are those design basis events that are accidents and transients and are postulated in the safety analyses. The design basis accidents and transients are used in the design of the plant to establish acceptable performance requirements for structures, systems, and components.

3.2.2.2 Application of Classification

The AP1000 requires adaptation of safety classification documents and standards because of the way that the AP1000 accomplishes safety-related functions.

In addition to 10 CFR 50.55a, the AP1000 classification has been developed considering requirements and guidelines in the following:

- ANSI N18.2 (Reference 1) – safety classification
- ANS 51.1 (Reference 2) – safety classification
- Regulatory Guide 1.26 – Quality Groups
- Regulatory Guide 1.97 – instrumentation requirements
- 10 CFR 21.

Conformance with the guidelines of Regulatory Guides 1.26 and 1.97 is discussed in Subsection 1.9.1.

The general guidelines for safety classification in the ANSI and ANS standards are useful in the development of the AP1000 classification. The specific classifications for various structures, systems, and components included in Regulatory Guide 1.26 and ANSI 18.2 and ANS 51.1 are based on a nuclear power plant with active safety systems and are not necessarily appropriate for the passive safety systems of the AP1000.

For the purposes of equipment classification, structures, systems, and components are classified as Class A, B, C, D, E, F, G, L, P, R, or W. For mechanical equipment Classes A, B, and C are equivalent to ANS Safety Class 1, 2, and 3. For electrical equipment Class C is equivalent to Class 1E. Structures, systems, and components classified equipment class A, B, or C or seismic Category I are basic components as defined in 10 CFR Part 21.

Equipment Class D is a nonsafety-related class. Classes E, F, G, L, P, R, and W are nonsafety-related classes associated with different industry codes and standards.

Components are classified down to the replacement part level according to the definitions and criteria of the classification system. A single item or portion thereof, which provides two or more functions of different classes, is classified according to the most stringent function. Different portions of the same structure, system, or component may perform different functions and be assigned to different equipment classes if the structure, system, or component contains a suitable interface boundary.

The definitions and criteria for the AP1000 equipment classes follow.

3.2.2.3 Equipment Class A

Class A is a safety-related class equivalent to ANS Safety Class 1. It applies to the reactor coolant system pressure boundary, including the required isolation valves and mechanical supports. This class has the highest integrity, and the lowest probability of leakage.

10 CFR 21 applies to Class A structures, systems, and components. Class A structures, systems, and components are seismic Category I and use codes and standards consistent with the guidelines for NRC Quality Group A. 10 CFR 50, Appendix B applies. ASME Code, Section III, Class 1 applies to pressure retaining components.

3.2.2.4 Equipment Class B

Class B is a safety-related class equivalent to ANS Safety Class 2. It limits the leakage of radioactive material from the containment following a design basis accident. This class is designed to accomplish the following:

- It provides fission product barrier or primary containment radioactive material holdup or isolation.
- It provides the containment boundary including penetrations and isolation valves. This also includes piping that functions as the containment boundary. For example, the steam and feedwater system inside containment and the secondary shell of the steam generator are Class B by this criterion.
- It circulates a non-containment/non-reactor coolant fluid to provide a post-accident safety-related function into and out of the containment. These lines have a Class B pressure boundary inside the containment. The outside containment lines in this circulation loop can be Class C or a nonsafety-related class if suitable containment isolation valves are provided.
- It introduces emergency negative reactivity to make the reactor subcritical (for example, control rods).
- This class also applies to structures, systems, and components where leakage could cause a loss of adequate core cooling. In isolating leaks, credit can be taken for automatic safety-related isolation and for appropriate operator action. As a minimum, operator action needs redundant safety-related indication and alarm followed by 30 minutes for operator action.

10 CFR 21 applies to Class B structures, systems, and components. Class B structures, systems, and components are seismic Category I and use codes and standards consistent with the guidelines for NRC Quality Group B. 10 CFR 50, Appendix B applies. ASME Code, Section III, Class 2 or Class MC applies to pressure retaining components. ASME Code, Section III, Subsection NE applies to the containment vessel and guard pipes.

3.2.2.5 Equipment Class C

Class C is a safety-related class equivalent to ANS Safety Class 3. It applies to other safety-related functions required to mitigate design basis accidents and other design basis events. Minor leakage will not prevent Class C structures, systems, and components from meeting the safety-related function, either from the regard of radiation dose or system functioning.

This class also applies to equipment that, upon rupturing, would cause dose limits for unrestricted areas, as specified in 10 CFR 20, to be exceeded or would cause a loss of core cooling.

10 CFR 21 applies to Class C structures, systems, and components. Class C structures, systems, and components use codes and standards consistent with the guidelines for NRC Quality Group C. Class C structures, systems, and components are seismic Category I except those noted below which are not required to provide a safety-related function following a seismic event. 10 CFR 50, Appendix B applies. ASME Code, Section III, Class 3 applies to pressure retaining components. In addition to these requirements, for systems that provide emergency core cooling functions, full radiography in accordance with the requirements of ASME Code, Section III, ND-5222 of a random sample of welds will be conducted on the piping butt welds during construction. For Class C air and gas storage tanks fabricated without welding, ASME Code, Section VIII, Appendix 22 may be used in lieu of Section III, Class 3. 10 CFR 50, Appendix B requirements and 10 CFR 21 apply to the manufacture of safety-related air and gas storage tanks. For core support structures ASME Code, Section III, Subsection NG applies. For electrical systems, appropriate IEEE standards, including IEEE standard 323-74 (Reference 3) and IEEE standard 344-87 (Reference 4), apply.

Class C applies to structures, systems, and components not included in Class A or Class B that are designed and relied upon to accomplish one or more of the following safety-related functions:

- Provide safety injection or maintain sufficient reactor coolant inventory to allow for core cooling
- Provide core cooling
- Provide containment cooling
- Provide for removal of radiation from the containment atmosphere as necessary to meet the offsite dose limits
- Limit the buildup of radioactive material in the atmosphere of rooms and areas outside containment as necessary to meet the offsite dose limits
- Introduce negative reactivity control measures to achieve or maintain safe shutdown conditions (for example, boron addition)
- Maintain geometry of structures inside the reactor vessel so that the control rods can be inserted (when required) and the fuel remains in a coolable geometry
- Provide load-bearing structures and supports for Class A, B, and C structures, systems, and components. This applies to structures and supports that are not part of the pressure boundary.
- Provide structures and buildings to protect Class A, B, and C structures, systems, and components from events such as internal/external missiles, seismic, and flooding. Structures protecting equipment from nonseismic events are not required to be seismic Category I.

- Provide permanent radiation shielding to allow operator access to the main control room and to limit the exposure to Class A, B and C structures, systems, and components

- Provide safety support functions to Class A, B and C structures, systems, and components, such as, heat removal, room cooling, and electrical power

- Provide instrumentation and controls for automatic or manual actuation of Class A, B, and C structures, systems, and components necessary to perform the safety-related functions of the Class A, B, or C structure, system or component. This includes the processing of signals and interlock functions required for proper safety performance of these structures, systems, and components.

- Maintain spent fuel integrity, the failure of which could result in fuel damage such that significant quantities of radioactive material could be released from the fuel and results in offsite doses greater than normal limits (for example, spent fuel pool, fuel transfer tube isolation valve)

- Maintain spent fuel sub-critical

- Monitor radioactive effluent to confirm that release rates or total releases are within limits established for normal operations and transient operation

- Monitor variables to indicate status of Class A, B or C structures, systems, and components required for post-accident mitigation

- Provide for functions defined in Class B where structures, systems, and components, or portions thereof are not within the scope of the ASME Code, Section III, Class 2.

- Provide provisions for connecting temporary equipment to extend the use of safety related systems. See Subsection 1.9.5 for a discussion of actions required for an extended loss of onsite and offsite ac power sources.

The components and portions of systems that provide emergency core cooling functions and are required to have radiography of a random sample of welds during construction include the following:

- Accumulators

- Injection piping from the accumulators to the reactor coolant system isolation check valves in the direct vessel injection line

- Piping from the in-containment refueling water storage tank (IRWST) and recirculation screens to the reactor coolant system isolation check valves in the direct vessel injection line

- Piping from the Stage 1, 2, and 3 automatic depressurization system valves to the IRWST including the spargers.

The IRWST is formed from portions of structural modules that are elements of the containment internal structures. The inspection requirements for the welds in these structural modules are provided in Subsection 3.8.3.6.2.

3.2.2.6 Equipment Class D

Class D is nonsafety-related with some additional requirements on procurement, inspection or monitoring.

For Class D structures, systems, and components containing radioactivity, it is demonstrated by conservative analysis that the potential for failure due to a design basis event does not result in exceeding the normal offsite doses per 10 CFR 20. This criterion is in conformance with the definition of Class D in Regulatory Guide 1.26.

A structure, system or component is classified as Class D when it directly acts to prevent unnecessary actuation of the passive safety systems. Structures, systems and components which support those which directly act to prevent the actuation of passive safety systems are also Class D. The inclusion of these nonsafety-related structures, systems, and components in Class D recognizes that these systems provide an important first level of defense that helps to reduce the calculated probabilistic risk assessment core melt frequency. These structures, systems, and components are normally used to support plant cooldown and depressurization and to maintain shutdown conditions during maintenance and refueling outages.

For Class D structures, systems, and components considered to be risk significant as defined in the reliability assurance program (see Section 16.2). Provisions are made to check for operability, including appropriate testing and inspection, and to repair out-of-service structures, systems, and components. These provisions are documented and administered in the plant reliability assurance plan and operating and maintenance procedures.

A portion of chemical and volume control system is defined as the reactor coolant pressure boundary and is Class D. This portion of the chemical and volume control system is seismically analyzed. See Subsection 5.2.1.1 for the seismic analysis requirements.

Some Class D structures, systems, and components are assumed to function in a severe containment environment. The design requirements for these components include operation in such an environment. An evaluation is done to confirm that the structure, system, or component can be expected to function in such an environment.

Standard industrial quality assurance standards are applied to Class D structures, systems, and components to provide appropriate integrity and function although 10 CFR 50, Appendix B and 10 CFR 21 do not apply. 10 CFR 50, Appendix B and 10 CFR 21 do apply to Class D structures, systems, and components that are seismic Category I. Pertinent portions of 10 CFR 50, Appendix B are applied to seismic Category II applications as described in Subsection 3.2.1.1.2. These industrial quality assurance standards are consistent with the guidelines for NRC Quality Group D. The industry standards used for Class D structures, systems and components are widely used industry standards. Typical industrial standards used for Class D systems and components are provided as follows:

- Pressure vessels – ASME Code, Section VIII
- Piping – ANSI B 31.1. Power Piping, (Reference 5)
- Pumps – API 610 (Reference 6), or Hydraulic Institute Standards (Reference 7)
- Valves – ANSI B16.34 (Reference 8)
- Atmospheric storage tanks – API-650 (Reference 9), AWWA D 100 (Reference 10), or ANSI B96.1 (Reference 11)
- 0 - 15 psig Storage Tanks – API-620 (Reference 12)
- AC motor and generators – NEMA MG1 (Reference 13)

- Circuit breakers, switchgear, relays, substations and fuses – IEEE C37 (Reference 14).

The NS buildings (except for the NS portions of the turbine and annex buildings outlined in Table 3.2-2) containing Class D structures, systems, and components, as well as the anchorage of the structures, systems, and components to the building, are designed to the seismic requirements of the Uniform Building Code (Reference 15). The NS portions of the turbine building and annex building are designed to the requirements of the International Building Code, IBC-06 (Reference 19). The systems and components are not designed for seismic loads. However, when Class D structures, systems, and components are located near a Class A, B, or C structure, system, or component, the requirements for seismic Category II may apply.

For Class D structures, systems, and components required to be monitored for maintenance effectiveness by 10 CFR 50.65, the availability parameters and criteria are included in the maintenance monitoring plan for evaluating the effectiveness of the maintenance program.

As examples, Class D applies to structures, systems, and components not included in Class A, B or C that provide the following functions:

- Provide core or containment cooling which prevents challenges to the passive core cooling system and the passive containment cooling system
- Process, extract, encase, store or reuse radioactive fluid or waste
- Verify that plant operating conditions are within technical specification limits
- Provide permanent shielding for post accident access to Class A, B or C structures, systems, and components or of offsite personnel
- Handle spent fuel, the failure of which could result in fuel damage such that limited quantities of radioactive material could be released from the fuel (for example, fuel handling machine, spent fuel handling tool, new and spent fuel racks)
- Protect Class B or C structures, systems, and components necessary to attain or maintain safe shutdown following fire
- Indicate the status of protection system bypasses that are not automatically removed as a part of the protection system operation
- Aid in determining the cause or consequences of an event for post-accident investigation
- Prevent interaction that could result in preventing Class A, B or C structures, systems, and components from performing required safety-related functions
- Limit the buildup of hydrogen in the containment atmosphere to acceptable values

3.2.2.7 Other Equipment Classes

Equipment classes E, F, G, L, P, R, and W are nonsafety-related. They apply to structures, systems, and components not covered in the above classes. They have no safety-related function to perform. They do not contain sufficient radioactive material that a release could exceed applicable limits.

Structures, systems, and components that do not normally contain radioactive fluids, gases, or solids but have the potential to become radioactively contaminated are classified as one of these nonsafety-related classes if all of the following criteria are satisfied:

- The system is only potentially radioactive and does not normally contain radioactive material, and
- The system has shown in plant operations that the operation with the system containing radioactive material meets or can meet unrestricted area release limits, and
- An evaluation of the system confirms that the system contains features and components that keep the consequences of a system failure as low as reasonably achievable, and
- The system has no other regulatory guidance requiring its inclusion in Classes A, B, C or D.

This review of the system features and components includes the following as a minimum:

- Features and components that control and limit the radioactive contamination in the system
- Features that facilitate an expeditious cleanup should the system become contaminated
- Features and components that limit and control the radiological consequences of a potential system failure
- The means by which the system prevents propagation to an event of greater consequence.

There are no special quality assurance requirements for Class E, F, G, L, P, R, and W structures, systems, and components. Unless specifically specified, 10 CFR Part 21 and Part 50, Appendix B do not apply. The systems and components are normally not designed for seismic loading. However, there may be special cases where some seismic design is required. See Subsection 3.2.1 for more details.

Structures, systems, and components are designed in accordance with an industry standard at the discretion of the designer. The following provides examples of industry standards which may be used for these classes:

Class E – This class is used for nonsafety-related structures, systems, and components that do not have a specialized industry standard or classification, as noted in the following classes.

Class F and G – These classes are used for Fire Protection Systems. They comply with National Fire Protection Association Codes which invoke ANSI B31.1 (Reference 5), AWWA (American Water Works Association), API (American Petroleum Institute), Underwriters Laboratories (UL), and other codes, depending on service. See Subsection 9.5.1 for quality assurance requirements for fire protection structures, systems, and components. Portions of fire protection systems that protect safety-related SSCs are designated as AP1000 equipment **Class F**, which meets the requirements of ANSI B31-1 and requires seismic analysis.

Class L – This class is used in heating, ventilation and air-conditioning systems. It complies with SMACNA - 1985 (Reference 15). Components may also be procured to AMCA and ASHRAE standards.

Class P – This class is used for plumbing equipment. It complies with the National Plumbing Code (Reference 17).

Class R – This class is for air cleaning units and components that may be required to contain, clean, or exclude radioactively contaminated air. It complies with ASME 509 (Reference 18). When used with 10 CFR Part 50 Appendix B quality assurance, it is equivalent to Class C.

Class W – This class complies with American Water Works Association guidelines with no specific quality assurance requirements.

3.2.2.8 Instrumentation and Control Line Interface Criteria

Class C instrumentation, as defined in Subsection 3.2.2.5 have a safety-related equipment class pressure boundary including the sensing line, valves and instrument sensor. The pressure boundary is the same safety-related equipment class as the systems or components it is connected to. Sensing lines connected to the reactor coolant system pressure boundary are Class B if a suitable flow restrictor is provided.

The parts of the sensor, outside the pressure boundary, are designated Class C (1E) if they provide a safety-related function per Subsection 3.2.2.1. They are Class D if the instrument supports Class D functions per Subsection 3.2.2.6. Otherwise the parts are Class E.

3.2.2.9 Electrical Classifications

Safety-related electrical equipment is equipment Class C, as outlined in Subsection 3.2.2.5, and is constructed to IEEE standards for Class 1E. The nonsafety-related electrical equipment and instrumentation is constructed to standards including non-Class 1E IEEE standards and National Electrical Manufacturers Association (NEMA) standards. Safety-related electrical equipment and instrumentation is identified in Section 3.11.

3.2.3 Inspection Requirements

Safety-related structures, systems, and components built to the requirements of the ASME Code, Section III, are required by 10 CFR 50.55a to have in-service inspections. The requirements of the in-service inspection program for ASME Code, Section III structures, systems, and components are found in Section XI of the ASME Code.

The following ASME standards apply to safety-related structures, systems, and components:

- Pumps (Class A, B, C) – ASME OM Code, Subsection ISTB
- Valves (Class A, B, C) – ASME OM Code, Subsection ISTC
- Equipment supports (Class A, B, C) – ASME Code, Section XI, Subsection IWF
- Metal containments and vessels – ASME Code, Section XI, Subsection IWE
- Other Class A components such as pipes and tanks – ASME Code, Section XI, Subsection IWB
- Other Class B components such as pipes and tanks – ASME Code, Section XI, Subsection IWC
- Other Class C components such as pipes and tanks – ASME Code, Section XI, Subsection IWD.

The inspection requirements, if applicable, for Class D structures, systems, and components are established by the designer for each structure, system, and component. These inspection requirements are developed so that the reliability of the structures, systems, and components is not degraded. The inspection requirements are included in the administratively controlled inspection or maintenance plans.

3.2.4 Application of AP1000 Safety-Related Equipment and Seismic Classification System

The application of the AP1000 equipment and seismic classification system to AP1000 systems and components is shown in Table 3.2-3. Table 3.2-3 lists safety-related and seismic Category I mechanical and fluid system component and associated equipment class and seismic category as well as other related information. The table also provides information on the systems that contain Class D components. Additional information on the Class D functions of the various systems can be found in the description in the Design Certification Document (DCD) for the systems. Mechanical and fluid systems that contain no safety-related or Class D systems are included in the table and general information provided on the system. Supports for piping and components have the same classification as the component or piping supported. Supports for AP1000 equipment Class A, B, and C mechanical components and piping are constructed to ASME Code, Section III, Subsection NF requirements. The principal construction code for supports for nonsafety-related components and piping is the same as that for the supported component or piping.

Following the name of each system is the building location of the system components. Some of the systems supply all or most of the buildings. This is indicated by identifying the location as various. Where a system includes piping or ducts that only passed through a building without including any components that building is generally not included in the list.

The following list includes the systems in Table 3.2-3. The three letters in the beginning of each line is the acronym for the system. The systems included in Table 3.2-3 are listed alphabetically by three letter acronym. Those systems marked with an asterisk * are electrical or instrumentation systems and are not included in Table 3.2-3. The components in the incore instrumentation system that have a pressure boundary function are included in the table. See Section 3.11 for identification of safety-related electrical and instrumentation equipment.

NSSS/Steam Generator Controls and Auxiliaries

BDS	Steam Generator Blowdown System
CNS	Containment System
CVS	Chemical and Volume Control System
PCS	Passive Containment Cooling System
PXS	Passive Core Cooling System
RCS	Reactor Coolant System
RNS	Normal Residual Heat Removal System
RXS	Reactor System
SGS	Steam Generator System

Nuclear Control and Monitoring

*DAS	Diverse Actuation System
IIS	Incore Instrumentation System

*OCS	Operation and Control Centers
*PMS	Protection and Safety Monitoring System
PSS	Primary Sampling System
*RMS	Radiation Monitoring System
*SJS	Seismic Monitoring System
*SMS	Special Monitoring System

Main Power Cycle and Auxiliaries

CDS	Condensate System
CFS	Turbine Island Chemical Feed System
CPS	Condensate Polishing System
DTS	Demineralized Water Treatment System
DWS	Demineralized Water Transfer and Storage System
FWS	Main and Startup Feedwater System
GSS	Gland Seal System
HDS	Heater Drain System
MSS	Main Steam System
MTS	Main Turbine System
RWS	Raw Water System
TDS	Turbine Island Vents, Drains and Relief System

Class 1E and Emergency Power Systems

*IDS	Class 1E dc and UPS System

Cooling and Circulating Water

CCS	Component Cooling Water System
CES	Condenser Tube Cleaning System
CWS	Circulating Water System

SFS	Spent Fuel Pit Cooling System
SWS	Service Water System
TCS	Turbine Building Closed Cooling Water System

Auxiliary Steam

ASS	Auxiliary Steam Supply System

Generation and Transmission

*ZAS	Main Generation System
*ZBS	Transmission Switchyard and Offsite Power System
*ZVS	Excitation and Voltage Regulation System

Radwaste

WGS	Gaseous Radwaste System
WLS	Liquid Radwaste System
WRS	Radioactive Waste Drain System
WSS	Solid Radwaste System

HVAC

VAS	Radiologically Controlled Area Ventilation System
VBS	Nuclear Island Nonradioactive Ventilation System
VCS	Containment Recirculation Cooling System
VES	Main Control Room Emergency Habitability System
VFS	Containment Air Filtration System
VHS	Health Physics and Hot Machine Shop HVAC System
VLS	Containment Hydrogen Control System
VRS	Radwaste Building HVAC System
VTS	Turbine Building Ventilation System

VUS	Containment Leak Rate Test System
VWS	Central Chilled Water System
VXS	Annex/Auxiliary Nonradioactive Ventilation System
VYS	Hot Water Heating System
VZS	Diesel Generator Building Ventilation System

Turbine-Generator Controls and Auxiliary

CMS	Condenser Air Removal System
HCS	Generator Hydrogen and CO_2 Systems
HSS	Hydrogen Seal Oil System
LOS	Main Turbine and Generator Lube Oil System
*TOS	Main Turbine Control and Diagnostics System

Material Handling

FHS	Fuel Handling and Refueling System
MHS	Mechanical Handling System

Piping Services

CAS	Compressed and Instrument Air Systems
DOS	Standby Diesel Fuel Oil System
FPS	Fire Protection System
PGS	Plant Gas Systems
PWS	Potable Water System

Non-Class 1E Power Systems

*ECS	Main AC Power System
*EDS	Non-Class 1E dc and UPS System
ZOS	Onsite Standby Power System

*ZRS	Offsite Retail Power System

Miscellaneous Electrical Systems

*EFS	Communication Systems
*EGS	Grounding and Lightning Protection System
*EHS	Special Process Heat Tracing System
*ELS	Plant Lighting System
*EQS	Cathodic Protection System

Non-Nuclear Controls and Monitoring

*DDS	Data Display and Processing System
*MES	Meteorological and Environmental Monitoring System
*PLS	Plant Control System
*SES	Plant Security System
SSS	Secondary Sampling System
*TVS	Closed Circuit TV System

Non-Radioactive Drains

DRS	Storm Drain System
RDS	Gravity and Roof Drain Collection System
SDS	Sanitary Drainage System
WWS	Waste Water System

Those systems marked with an asterisk (*) are electrical or instrumentation systems and are not included in Table 3.2-3.

3.2.5 Combined License Information

This section contained no requirement for additional information.

3.2.6 References

1. ANSI N18.2a-75, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

2. ANS/ANSI 51.1-83, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.”

3. IEEE 323-74, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.”

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

5. ASME/ANSI B31.1-1989, “Power Piping, ASME Code for Pressure Piping.”

6. API 610-81, “Centrifugal Pumps for General Refinery Services.”

7. “Hydraulic Institute Standards,” 1975, Hydraulic Institute.

8. ASME/ANSI B16.34-81, “Valves - Flanged and Buttwelding End.”

9. API-650-80, “Welded Steel Tanks for Oil Storage,” Revision 1, February 1984.

10. AWWA D100-84, “Welded Steel Tanks for Water Storage.”

11. ANSI B96.1-81, “Welded Aluminum-Alloy Storage Tanks.”

12. API-620-82, “Recommended Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks,” Revision 1, April 1985.

13. NEMA MG-1-98, “Motors and Generators,” Revision 1, January 1998, National Electric Manufacturers Association.

14. IEEE C37, IEEE standards on circuit breakers, switch gear, relays, substations, fuses, etc.

15. “Uniform Building Code (1997),” International Conference of Building Officials.

16. SMACNA - 1995, HVAC Duct Construction Standards - Metal and Flexible, 1985 Edition, Sheet Metal and Air-Conditioning Contractors National Association.

17. The BOCA Basic/National Plumbing Code 1984: Model Plumbing Regulations for the Protection of Public Health, Safety and Welfare: Sixth Edition, Building Officials and Code Administrators International.

18. ASME/ANSI AG-1-1997, “Code on Nuclear Air and Gas Treatment.”

19. International Building Code, 2006.

Table 3.2-1
Comparison of Safety Classification Requirements

AP1000 Code Letter (1)	ANS Equipment Safety Class (2)	RG 1.29 Seismic Design Reqmnts (3)	ASME Code, Sec. III Class (4)	IEEE Requirements	RG 1.26 NRC Quality Group (5)	10 CFR 50 Appendix B (6)	Inspection & Testing Requirements	Required Test & Maint.
A	SC-1	I	1	NA	GROUP A	YES	YES(7)	(8)
B	SC-2	I	2	NA	GROUP B	YES	YES(7)	(8)
C	SC-3	I	3	1E	GROUP C	YES	YES(7)	(8)
D	NNS(2)	NA(9)	NA(10)	(10)	GROUP D	NO(10)	YES(11)	(11)
OTHER	NNS(2)	NA(13)	NA	NA	NA	NA(12)	NA	NA

NA - Not Applicable OTHER includes Classes E, F, L, P, R, and W.

Notes:

1. A single letter equipment classification identifies the safety class, quality group, and other classifications for AP1000. See the Subsection 3.2.2 for definition.
2. AP1000 safety classification is an adaptation of that defined in ANSI 51.1. The NNS defined in the ANSI 51.1 standard is divided into several AP1000 equipment classifications namely, Classes D E, F, L, P, R, and W.
3. See Subsection 3.2.1 for definition of seismic categories.
4. ASME Boiler and Pressure Vessel Code, Section III defines various classes of structures, systems, and components for nuclear power plants. It defines criteria and requirements based on the classification. It is not applicable for nonsafety-related components.
5. The quality group classification corresponds to those provided in Regulatory Guide 1.26.
6. "Yes" means quality assurance program is required according to 10 CFR 50 Appendix B.
   "No" means quality assurance program is not required according to 10 CFR 50 Appendix B.
7. Class A, B, and C, structures, systems, and components built to ASME Code, Section III are inspected to ASME Code, Section XI requirements. See the text for additional specification of requirements.
8. Class A, B, and C structures, systems, and components that are required to function to mitigate design base accidents have some testing requirements included in the plant technical specifications. In addition to the requirements in the technical specifications, testing and maintenance requirements are included in an administratively controlled reliability assurance plan.
9. See Subsection 3.2.1 for cases when seismic Category II requirements are applicable for Class D structures, systems, and components.
10. See the text for a discussion of the industry standards used in the construction of Class D structures, systems and components.
11. Class D structures, systems, and components have selected reliability assurance programs and procedures to provide availability when needed. These programs are administratively controlled programs and are not included in the technical specifications.
12. Normal industrial procedures are followed in procuring, designing, fabricating, and testing these nonsafety-related structures, systems, and components.
13. Some Class E, F, G, L, P, R, and W structures, systems, and components may be classified as seismic Category II. See Subsection 3.7.3.

Table 3.2-2
Seismic Classification of Building Structures

Structure	**Category(1)**
Nuclear Island Basemat Containment Interior Shield Building Auxiliary Building Containment Air Baffle	C-I
Containment Vessel	C-I
Plant Vent and Stair Structure	C-II
Turbine Building – First bay adjacent to Nuclear Island outlined by Columns I.1 to R, 11.05 to 11.2, and 11.02 to 11.2	C-II
Turbine Building – All portions of Turbine Building except first bay adjacent to Nuclear Island as outlined by Columns H.05 to R and 12.1 to 20	NS(2)
Turbine Building	NS(2)
Annex Building Area Outlined by Columns A - D and 8 - 13 Area Outlined by Columns A - G and 13 - 16	NS(2)
Annex Building Area Outlined by Columns E - I.1 and 2 - 13	C-II
Radwaste Building	NS(2)
Diesel-Generator Building	NS(3)
Circulating Water Pumphouse and Towers	NS
Safety-Related Backfill	C-I

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

Note:

1. Within the broad definition of seismic Category I and II structures, these buildings contain members and structural subsystems the failure of which would not impair the capability for safe shutdown. Examples of such systems would be elevators, stairwells not required for access in the event of a postulated earthquake, and nonstructural partitions in nonsafety-related areas. These substructures are classified as non-seismic.
2. The NS designation for the turbine building, the radwaste building, and a portion of the annex building indicates that the buildings are not seismic Category I or seismic Category II. The seismic requirements for these buildings are outlined in Subsection 3.7.2.8.
3. The NS designation for the diesel-generator building indicates that the building is not seismic Category I or seismic Category II. The seismic requirements for buildings containing Class D equipment, including the diesel generator building, are outlined in Subsection 3.2.2.6.

Table 3.2-3 (Sheet 1 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Auxiliary Steam Supply System (ASS)				Location: Turbine Building	
System components are Class E					
Steam Generator Blowdown System (BDS)				Location: Turbine Building	
System components are Class E					
Compressed and Instrument Air System (CAS)				Location: Various	
CAS-PL-V014	Instrument Air Supply Outside Containment Isolation	B	I	ASME III-2	
CAS-PL-V015	Instrument Air Supply Inside Containment Isolation	B	I	ASME III-2	
CAS-PL-V027	Containment Penetration Test Connection Isolation	B	I	ASME III-2	
CAS-PL-V204	Service Air Supply Outside Containment Isolation	B	I	ASME III-2	
CAS-PL-V205	Service Air Supply Inside Containment Isolation	B	I	ASME III-2	
CAS-PL-V219	Containment Penetration Test Connection Isolation	B	I	ASME III-2	
CAS-PY-C02	Containment Instrument Air Inlet Penetration	B	I	ASME III, MC	
CAS-PY-C03	Containment Service Air Inlet Penetration	B	I	ASME III, MC	
Balance of system components are Class E					
Component Cooling Water System (CCS)				Location: Auxiliary Building and Turbine Building	
n/a	Heat Exchangers, CCS and SWS Side	D	NS	ASME VIII	
n/a	Pumps	D	NS	Hydraulic Institute Stds.	
n/a	Tanks	D	NS	ASME VIII	
n/a	Valves Providing CCS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
CCS-PL-V200	CCS Containment Isolation Valve - Inlet Line ORC	B	I	ASME III-2	
CCS-PL-V201	CCS Containment Isolation Valve - Inlet Line IRC	B	I	ASME III-2	
CCS-PL-V207	CCS Containment Isolation Valve - Outlet Line IRC	B	I	ASME III-2	
CCS-PL-V208	CCS Containment Isolation Valve - Outlet Line ORC	B	I	ASME III-2	
CCS-PL-V209	Containment Isolation Valve Test Connection - Outlet Line	B	I	ASME III-2	
CCS-PL-V214	CCS Supply Containment Isolation - IRC	C	I	ASME III-3	
CCS-PL-V215	CCS Supply Containment Isolation Valve Test Connection - IRC	C	I	ASME III-3	

Table 3.2-3 (Sheet 2 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Component Cooling Water System (Continued)					
CCS-PL-V216	Containment Leak Test Outlet Line - IRC	C	I	ASME III-3	
CCS-PL-V217	Containment Isolation Valve V207 Body Test Connection Valve	C	I	ASME III-3	
CCS-PL-V270	CCS IRC Relief Valve	C	I	ASME III-3	
CCS-PL-V271	CCS IRC Relief Valve	C	I	ASME III-3	
CCS-PL-220	CCS Containment Isolation Relief Valve	B	I	ASME III-2	
CCS-PL-V257	Containment Isolation Valve Test Connection - Inlet Line	B	I	ASME III-2	
CCS-PY-C01	Containment Supply Header Penetration	B	I	ASME III, 2	
CCS-PY-C02	Containment Return Header Penetration	B	I	ASME III, 2	
Balance of system components are Class E					
Condensate System (CDS)				Location: Turbine Building	
System components are Class E					
Condenser Tube Cleaning System (CES)				Location: Turbine Building	
System components are Class E					
Turbine Island Chemical Feed System (CFS)				Location: Turbine Building and SWS Chemical Treatment Building/Area	
System components are Class E					
Condenser Air Removal System (CMS)				Location: Turbine Building	
n/a	Condenser Vacuum Breakers	E	NS	ANSI 16.34	
Balance of system components are Class D					
Containment System (CNS)				Location: Containment	
CNS-MV-01	Containment Vessel	B	I	ASME III, MC	
CNS-MY-Y01	Equipment Hatch	B	I	ASME III, MC	
CNS-MY-Y02	Maintenance Hatch	B	I	ASME III, MC	
CNS-MY-Y03	Personnel Hatch - 135′-3″	B	I	ASME III, MC	
CNS-MY-Y04	Personnel Hatch - 107′-2″	B	I	ASME III, MC	
n/a	Spare Containment Penetrations	B	I	ASME III, MC	
Condensate Polishing System (CPS)				Location: Turbine Building	
System components are Class E					
Chemical and Volume Control System (CVS)				Location: Containment, Auxiliary Building, and Annex Building	
n/a	Heat Exchangers, CVS and CCS Side	D	NS	ASME VIII/ TEMA	
n/a	Pumps	D	NS	Hydraulic Institute Stds.	
n/a	Tanks (Except CVS-MT-03 and CVS-MT-05)	D	NS	API 650	
CVS-MT-03	CVS Chemical Mixing Tank	E	NS	ASME VIII	

Table 3.2-3 (Sheet 3 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Chemical and Volume Control System (Continued)					
CVS-MT-05	CVS Air Intrusion Prevention Tank	D	NS	ASME VIII	
n/a	Demineralizers	D	NS	ASME VIII	
n/a	Filters	D	NS	ASME VIII	
n/a	Valves Providing CVS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
CVS-PL-V001	RCS Purification Stop	A	I	ASME III-1	
CVS-PL-V002	RCS Purification Stop	A	I	ASME III-1	
CVS-PL-V003	RCS Purification Stop	C	I	ASME III-3	
CVS-PL-V040	Resin Flush IRC Isolation	B	I	ASME III-2	
CVS-PL-V041	Resin Flush ORC Isolation	B	I	ASME III-2	
CVS-PL-V042	Flush Line Containment Isolation Relief	B	I	ASME III-2	
CVS-PL-V045	Letdown Containment Isolation IRC	B	I	ASME III-2	
CVS-PL-V046	Letdown Pressure Instrument Root	B	I	ASME III-2	
CVS-PL-V047	Letdown Containment Isolation ORC	B	I	ASME III-2	
CVS-PL-V058	Letdown Line Containment Isolation Relief	B	I	ASME III-2	
CVS-PL-V065	Zinc Addition – IRC Shutoff	C	I	ASME III-3	
CVS-PL-V067	Makeup Return Line Bypass Check Valve	A	I	ASME III-1	
CVS-PL-V080	RCS Purification Return Line Check Valve	C	I	ASME III-3	
CVS-PL-V081	RCS Purification Return Line Stop Valve	A	I	ASME III-1	
CVS-PL-V082	RCS Purification Return Line Check Valve	A	I	ASME III-1	
CVS-PL-V084	Auxiliary Pressurizer Spray Line Isolation	A	I	ASME III-1	
CVS-PL-V085	Auxiliary Pressurizer Spray Line	A	I	ASME III-1	
CVS-PL-V090	Makeup Line Containment Isolation	B	I	ASME III-2	
CVS-PL-V091	Makeup Line Containment Isolation	B	I	ASME III-2	
CVS-PL-V092	Zinc Injection Containment Isolation ORC	B	I	ASME III-2	
CVS-PL-V094	Zinc Injection Containment Isolation IRC	B	I	ASME III-2	
CVS-PL-V095	Zinc Add Containment Isolation Test Connection	C	I	ASME III-3	
CVS-PL-V096	Zinc Injection Containment Isolation Test Connection	B	I	ASME III-2	

Table 3.2-3 (Sheet 4 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Chemical and Volume Control System (Continued)					
CVS-PL-V098	Zinc Addition Line Containment Isolation Thermal Relief Valve	B	I	ASME III-2	
CVS-PL-V100	Makeup Line Containment Isolation Relief	B	I	ASME III-2	
CVS-PL-V136A	Demineralized Water System Isolation	C	I	ASME III-3	
CVS-PL-V136B	Demineralized Water System Isolation	C	I	ASME III-3	
CVS-PL-V215	Hydrogen Injection – IRC Shutoff	C	I	ASME III-3	
CVS-PL-V216	Hydrogen Injection Containment Isolation Test Connection	C	I	ASME III-3	
CVS-PL-V217	Hydrogen Injection Containment Isolation Check IRC	B	I	ASME III-2	
CVS-PL-V218	Hydrogen Injection Containment Isolation Test Connection	B	I	ASME III-2	
CVS-PL-V219	Hydrogen Injection Containment Isolation ORC	B	I	ASME III-2	
CVS-PY-C01	Demineralizer Resin Flush Line Containment Penetration	B	I	ASME III, MC	
CVS-PY-C02	Letdown Line Containment Penetration	B	I	ASME III, MC	
CVS-PY-C03	Makeup Line Containment Penetration	B	I	ASME III, MC	
CVS-PY-C04	Zinc Add Line Containment Penetration	B	I	ASME III, 2	
CVS-PY-C05	Hydrogen Add Line Containment Penetration	B	I	ASME III, 2	
Balance of system components are Class D or E					
Circulating Water System (CWS)				Location: Turbine Building and pump intake structure	
System components are Class E					
Standby Diesel Fuel Oil System (DOS)				Location: Diesel Generator Building and yard	
n/a	Fuel Oil Transfer Package	D	NS	Manufacturer Std.	
n/a	Fuel Oil Storage Tanks	D	NS	API 650	
n/a	Fuel Oil Day Tanks	D	NS	ASME VIII	
n/a	Valves Providing DOS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
n/a	Ancillary Diesel Generator Fuel Tank	D	II	ASME VIII	Located in Annex Building
Balance of system components are Class E					

Table 3.2-3 (Sheet 5 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Storm Drain System (DRS)				Location: Various	
System components are Class E					
Demineralized Water Treatment System (DTS)				Location: Turbine Building	
System components are Class E					
Demineralized Water Transfer and Storage System (DWS)				Location: Various	
n/a	Condensate Storage Tanks	D	NS	API 650	
n/a	Valves Providing DWS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
DWS-PL-V244	Demineralized Water Supply Containment Isolation - Outside	B	I	ASME III-2	
DWS-PL-V245	Demineralized Water Supply Containment Isolation - Inside	B	I	ASME III-2	
DWS-PL-V248	Containment Penetration Test Connection Isolation	B	I	ASME III-2	
DWS-PY-C01	Containment Demineralized Water Supply Penetration	B	I	ASME III, MC	
Balance of system components are Class E					
Electrical Distribution System (ECS)				Location: Annex Building	
n/a	Ancillary Diesel Generator Engines	D	NS	Manufacturer Standard	Anchorage is SCII
n/a	Ancillary Diesel Generator Radiators	D	NS	CAGI	
n/a	Ancillary Diesel Generator Silencers	D	NS	API661	
n/a	Valve providing fuel to ECS Ancillary Diesel Generators	D	NS	ANSI 16.34	
Balance of system components are Class E					
Fuel Handling and Refueling System (FHS)			Location: Containment and Auxiliary Building		
FHS-FH-01	Refueling Machine	D	II	AISC	
FHS-FH-02	Fuel Handling Machine	D	II	AISC	
FHS-FH-04	New Fuel Elevator	D	II	AISC	
FHS-FH-05	Fuel Transfer System	D	II	AISC	
FHS-FH-52	Spent Fuel Assembly Handling Tool	D	II	AISC	
FHS-FS-01	New Fuel Storage Rack	D	I	Manufacturer Std.	
FHS-FS-02	Spent Fuel Storage Rack	D	I	Manufacturer Std.	
FHS-FT-01	Fuel Transfer Tube	B	I	ASME III Class MC	Jurisdictional Boundary CV Only, Remaining Portion Optional
FHS-MT-01	Spent Fuel Pool	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only

Table 3.2-3 (Sheet 6 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Fuel Handling and Refueling System (Continued)					
FHS-MT-02	Fuel Transfer Canal	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MT-05	Spent Fuel Cask Loading Pit	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MT-06	Spent Fuel Cask Washdown Pit	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MY-Y01	Spent Fuel Transfer Gate	C	I	Manufacturer Std.	
FHS-MY-Y02	Spent Fuel Cask Loading Pit Gate	C	I	Manufacturer Std.	
FHS-PL-V001	Fuel transfer tube Isolation Valve	C	I	ASME-III-3	
FHS-PY-B01	Fuel Transfer Tube Blind Flange	B	I	ASME III Class MC	
Balance of system components are Class E					
Fire Protection System (FPS)				Location: Various	
FPS-PL-V050	Fire Water Containment Supply Isolation	B	I	ASME III-2	
FPS-PL-V051	Fire Water Containment Test Connection Isolation	B	I	ASME III-2	
FPS-PL-V052	Fire Water Containment Supply Isolation - Inside	B	I	ASME III-2	
FPS-PY-C01	Fire Protection Containment Penetration	B	I	ASME III, 2	
FPS-PL-V441	Auxiliary Connection to CCS Isolation	D	NS	ANSI B31.1	
Containment standpipe and suppression system components	Includes all FPS components Inside Rector Containment with the exception of those used for containment isolation and containment spray	F	NS	ANSI B31.1	Seismic Analysis Consistent with ASME Section III Class 3 Systems
Various	Auxiliary Building Standpipe and Non-1E Equipment Penetration Room Preaction Sprinkler System components	F	NS	ANSI B31.1	Seismic Analysis Consistent with ASME Section III Class 3 Systems
Balance of system components are Class E, F & G					
Main and Startup Feedwater System (FWS)				Location: Turbine Building	
n/a	Startup Feedwater Pumps	D	NS	Hydraulic Institute Standards	
n/a	Valves Providing SFW AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
Balance of system components are Class E					
Gland Seal System (GSS)				Location: Turbine Building	
System components are Class D					

Table 3.2-3 (Sheet 7 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Generator Hydrogen and CO_2 Systems (HCS)				Location: Turbine Building	
System components are Class E					
Heater Drain System (HDS)				Location: Turbine Building	
System components are Class E					
Hydrogen Seal Oil System (HSS)				Location: Turbine Building	
System components are Class E					
Incore Instrumentation System (IIS)				Location: Containment	
n/a	IIS Guide Tubes	A	I	ASME III-1	
n/a	Thimble assemblies	B	I	Manufacturer Std.	
Main Turbine and Generator Lube Oil System (LOS)				Location: Turbine Building	
System components are Class E					
Mechanical Handling System (MHS)				Location: Various	
MHS-MH-01	Containment Polar Crane	C	I	NUREG-0554 supplemented by ASME NOG-1	
MHS-MH-02	Cask Handling Crane	C	I	NUREG-0554 supplemented by ASME NOG-1	
MHS-MH-05	Equipment Hatch Hoist	C	I	NUREG-0554 supplemented by ASME NOG-1	
MHS-MH-06	Maintenance Hatch Hoist	C	I	NUREG-0554 supplemented by ASME NOG-1	
Balance of system components are Class E					
Main Steam System (MSS)				Location: Turbine Building	
System components are Class E					
Main Turbine System (MTS)				Location: Turbine Building	
System components are Class E					
Passive Containment Cooling System (PCS)			Location: Containment Shield Building and Auxiliary Building		
PCS-MT-01	Passive Containment Cooling Water Storage Tank	C	I	ACI 349	See subsection 6.2.2.2.3 for additional design requirements
PCS-MT-02	PCS Chemical Addition Tank	D	II	ASME VIII	
PCS-MT-03	Water Distribution Bucket	C	I	Manufacturer Std.	See subsection 6.2.2.2.3 for additional design requirements
PCS-MT-04	Water Collection Troughs	C	I	Manufacturer Std.	See subsection 6.2.2.2.3 for additional design requirements

Table 3.2-3 (Sheet 8 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-MT-05	Passive Containment Cooling Ancillary Water Storage Tank	D	II	API 650	
PCS-MT-06	PCCWST Leak Chase Collection Pot	D	II	ASME VIII	
PCS-PL-V001A	PCCWST Isolation	C	I	ASME III-3	
PCS-PL-V001B	PCCWST Isolation	C	I	ASME III-3	
PCS-PL-V001C	PCCWST Isolation	C	I	ASME III-3	
PCS-MP-01A	PCS Recirculation Pump	D	NS	Hydraulic Institute Standards	Equipment Anchorage is Seismic Category II
PCS-MP-01B	PCS Recirculation Pump	D	NS	Hydraulic Institute Standards	Equipment Anchorage is Seismic Category II
PCS-PL-V002A	PCCWST Series Isolation	C	I	ASME III-3	
PCS-PL-V002B	PCCWST Series Isolation	C	I	ASME III-3	
PCS-PL-V002C	PCCWST Series Isolation	C	I	ASME III-3	
PCS-PL-V004	Recirculation Bypass Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V005	PCCWST Supply to FPS Isolation	C	I	ASME III-3	
PCS-PL-V009	Spent Fuel Pool Emergency Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V010A	Flow Transmitter FT001 Root Valve	C	I	ASME III-3	
PCS-PL-V010B	Flow Transmitter FT001 Root Valve	C	I	ASME III-3	
PCS-PL-V011A	Flow Transmitter FT002 Root Valve	C	I	ASME III-3	
PCS-PL-V011B	Flow Transmitter FT002 Root Valve	C	I	ASME III-3	
PCS-PL-V012A	Flow Transmitter FT003 Root Valve	C	I	ASME III-3	
PCS-PL-V012B	Flow Transmitter FT003 Root Valve	C	I	ASME III-3	
PCS-PL-V013A	Flow Transmitter FT004 Root Valve	C	I	ASME III-3	
PCS-PL-V013B	Flow Transmitter FT004 Root Valve	C	I	ASME III-3	
PCS-PL-V014	Chemical Addition Tank Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V015	Water Bucket Makeup Line Drain Valve	C	I	ASME III	
PCS-PL-V016	PCCWST Drain Isolation Valve	C	I	ASME III-3	

Table 3.2-3 (Sheet 9 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-PL-V017	Chemical Addition Tank Vent Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V018	Recirculation Pump Throttle Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V019	Chemical Addition Tank Fill Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V020	Water Bucket Makeup Line Isolation Valve	C	I	ASME III-3	
PCS-PL-V021	PCCWST TO Recirculation Pump Suction Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V022	Chemical Addition Tank Drain Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V023	PCS Recirculation Return Isolation	C	I	ASME III-3	
PCS-PL-V025	Pressure Transmitter PT 031 Root Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V029	PCCWST Isolation Valve Leakage Detection Drain	C	I	ASME III-3	
PCS-PL-V030	PCCWST Isolation Valve Leakage Detection Crossconnect Valve	C	I	ASME III-3	
PCS-PL-V031A	Level Transmitter LT 016 & 010 Root Isolation Valve	C	I	ASME III-3	
PCS-PL-V031B	Level Transmitter LT 015 & 011 Root Isolation Valve	C	I	ASME III-3	
PCS-PL-V033	Recirculation Pump Suction from Long Term Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V035A	Recirculation Pump Suction Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V035B	Recirculation Pump Suction Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V036A/B	Recirculation Pump Discharge Check Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V037	PCCAWST Discharge Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V038	PCCAWST Drain Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V039	PCCWST Long-Term Makeup Check Valve	C	I	ASME III-3	
PCS-PL-V040	Recirculation Pump Suction from PCCAWST Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V041	PCCAWST Recirculation Return Line Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V042	PCCWST Long-Term Makeup Isolation Drain Valve	C	I	ASME III-3	PCS-PL-V043

Table 3.2-3 (Sheet 10 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-PL-V043	PCCAWST Recirculation Return Line Drain Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V044	PCCWST Long-Term Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V045	Emergency Makeup to the Spent Fuel Pool Isolation Valve	C	I	ASME III-3	
PCS-PL-V046	PCCWST Recirculation Return Isolation Valve	C	I	ASME III-3	
PCS-PL-V047A/B	PCS Recirculation Pump Discharge Isolation Valve	D	NS	ANSI B16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V048	Recirculation Pump Fire Suction Isolation Valve	D	NS	ANSI 16.34	Seismically Analyzed for Operability
PCS-PL-V049	Emergency Makeup to the Spent Fuel Pool Drain Isolation Valve	C	I	ASME III-3	
PCS-PL-V050	Spent Fuel Pool Long Term Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V051	Spent Fuel Pool Emergency Makeup Lower Isolation	C	I	ASME III-3	
PCS-PL-V052	Spent Fuel Pool Emergency Makeup Isolation Valve	C	I	ASME III-3	
PCS-PL-V053	PCS Recirculation Heater Pressure Relief Valve	D	NS	ASME VIII	
PCS-PL-V060A	Shutoff Valve for Leakage Sensor	C	I	ASME III-3	
PCS-PL-V060B	Shutoff Valve for Leakage Sensor	C	I	ASME III-3	
PCS-PL-V100	Temporary Containment Washdown Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V301	PCCWST to Recirculation Pump Suction Drain Isolation Valve	D	NS	ANSI 16.34	Equipment Anchorage is Seismic Category II
PCS-PL-V303	Recirculation Header Discharge to SFS Pool Vent Isolation Valve	C	I	ASME III-3	
PCS-PL-V304	Recirculation Header Discharge to SFS Pool Drain Isolation Valve	C	I	ASME III-3	
PCS-PL-V305	PCCWST Recirculation Return Drain Isolation Valve	C	I	ASME III-3	
PCS-PY-B01	Spent Fuel Pool Emergency Makeup Isolation	C	I	ASME III-3	
PCS-PY-C01	Containment Pressure Instrument Line Penetration	B	I	ASME, MC	
PCS-PY-C02	Containment Pressure Instrument Line Penetration	B	I	ASME, MC	

Table 3.2-3 (Sheet 11 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Containment Cooling System (Continued)					
PCS-PY-C03	Containment Pressure Instrument Line Penetration	B	I	ASME, MC	
PCS-PY-C04	Containment Pressure Instrument Line Penetration	B	I	ASME, MC	
Balance of system components are Class E or F					
Plant Gas Systems (PGS)				Location: Various	
System components are Class E					
Primary Sampling System (PSS)			Location: Containment and Auxiliary Building		
n/a	Grab Sample Unit	D	NS	Manufacturer Std.	
n/a	Sample Cooler, PSS and CCS Side	D	NS	ASME VIII/ TEMA	
n/a	Valves Providing PSS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
PSS-PL-V001A	Hot Leg Sample Isolation	B	I	ASME III-2	
PSS-PL-V001B	Hot Leg Sample Isolation	B	I	ASME III-2	
PSS-PL-V003	Pressurizer Liquid Isolation	B	I	ASME III-2	
PSS-PL-V004A	PXS Accumulator Sample Isolation	C	I	ASME III-3	
PSS-PL-V004B	PXS Accumulator Sample Isolation	C	I	ASME III-3	
PSS-PL-V005A	PXS CMT A Sample Isolation	B	I	ASME III-2	
PSS-PL-V005B	PXS CMT B Sample Isolation	B	I	ASME III-2	
PSS-PL-V005C	PXS CMT A Sample Isolation	B	I	ASME III-2	
PSS-PL-V005D	PXS CMT B Sample Isolation	B	I	ASME III-2	
PSS-PL-V008	Containment Air Sample Containment Isolation IRC	B	I	ASME III-2	
PSS-PL-V010A	Liquid Sample Line Containment Isolation IRC	B	I	ASME III-2	
PSS-PL-V010B	Liquid Sample Line Containment Isolation IRC	B	I	ASME III-2	
PSS-PL-V011A	Liquid Sample Line Containment Isolation ORC	B	I	ASME III-2	
PSS-PL-V011B	Liquid Sample Line Containment Isolation ORC	B	I	ASME III-2	
PSS-PL-V012A	Liquid Sample Isolation Valve	C	I	ASME III-3	
PSS-PL-V012B	Liquid Sample Check Valve	C	I	ASME III-3	
PSS-PL-V013	RCS Pressurizer Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V014A	RCS Hot Leg 1 Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V014B	RCS Hot Leg 2 Sample Isolation Valve	B	I	ASME III-2	

Table 3.2-3 (Sheet 12 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Primary Sampling System (Continued)					
PSS-PL-V015A	PXS Accumulator Sample Isolation Valve	C	I	ASME III-3	
PSS-PL-V015B	PXS Accumulator Sample Isolation Valve	C	I	ASME III-3	
PSS-PL-V016A	PXS CMT A Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V016B	PXS CMT B Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V016C	PXS CMT A Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V016D	PXS CMT B Sample Isolation Valve	B	I	ASME III-2	
PSS-PL-V023	Sample Return Line Containment Isolation ORC	B	I	ASME III-2	
PSS-PL-V024	Sample Return Containment Isolation IRC	B	I	ASME III-2	
PSS-PL-V046	Air Sample Line Containment Isolation ORC	B	I	ASME III-2	
PSS-PL-V076A	Containment Testing Boundary Isolation Valve	C	I	ASME III-3	
PSS-PL-V076B	Containment Testing Boundary Isolation Valve	C	I	ASME III-3	
PSS-PL-V082	Containment Isolation Test Connection Isolation Valve	C	I	ASME III-3	
PSS-PL-V083	Containment Isolation Test Connection Isolation Valve	C	I	ASME III-3	
PSS-PL-V085	Containment Isolation Test Connection Isolation Valve	B	I	ASME III-2	
PSS-PL-V086	Containment Isolation Test Connection Isolation Valve	C	I	ASME III-3	
PSS-PY-C01	Common Primary Sample Line Penetration	B	I	ASME III, MC	
PSS-PY-C02	Containment Atmosphere Sample Line Penetration	B	I	ASME III, MC	
PSS-PY-C03	Containment Atmosphere Sample Line Penetration	B	I	ASME III, 2	
PSS-PY-C04	RCS Hot Leg Sample Line Penetration	B	I	ASME III, MC	
PSS-PY-Y01	Delay Coil 1 for RCS Hot Leg 1	C	I	ASME III-3	
PSS-PY-Y02	Delay Coil 2 for RCS Hot Leg 2	C	I	ASME III-3	
PSS-MY-Y05	Delay Coil Assembly	C	I	ASME III-3	
Balance of system components are Class E					

Table 3.2-3 (Sheet 13 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Potable Water System (PWS)				Location: Various	
PWS-PL-V418	PWS MCR Isolation Valve	C	I	ASME III-3	
PWS-PL-V420	PWS MCR Isolation Valve	C	I	ASME III-3	
PWS-PL-V498	PWS MCR Vacuum Relief	C	I	ASME III-3	
Balance of system components are Class E, P, and W					
Passive Core Cooling System (PXS)				Location: Containment	
PXS-ME-01	Passive Residual Heat Removal Heat Exchanger	A	I	ASME III-1	
PXS-MT-01A	Accumulator Tank A	C	I	ASME III-3	
PXS-MT-01B	Accumulator Tank B	C	I	ASME III-3	
PXS-MT-02A	Core Makeup Tank A	A	I	ASME III-1	
PXS-MT-02B	Core Makeup Tank B	A	I	ASME III-1	
PXS-MT-03	In-Containment Refueling Water Storage Tank	C	I	ACI 349/AISC N690	ACI 349 is used for Evaluation of Structural Boundary
PXS-MT-04	IRWST Gutter	C	I	Manufacturer Std.	
PXS-MW-01A	Reactor Coolant Depressurization Sparger A	C	I	ASME III-3	
PXS-MW-01B	Reactor Coolant Depressurization Sparger B	C	I	ASME III-3	
PXS-MY-Y01A	IRWST Screen A	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y01B	IRWST Screen B	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y01C	IRWST Screen C	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y02A	Containment Recirculation Screen A	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.
PXS-MY-Y02B	Containment Recirculation Screen B	C	I	Manufacturer Std.	Structural frame and attachment use ASME III, Subsection NF criteria. Screen modules use manufacturer std.

Table 3.2-3 (Sheet 14 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-MY-Y03A	pH Adjustment Basket 3A	C	I	Manufacturer Std.	
PXS-MY-Y03B	pH Adjustment Basket 3B	C	I	Manufacturer Std.	
PXS-MY-Y04A	pH Adjustment Basket 4A	C	I	Manufacturer Std.	
PXS-MY-Y04B	pH Adjustment Basket 4B	C	I	Manufacturer Std.	
PXS-PL-V002A	CMT A CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V002B	CMT B CL Inlet Isolation	A	I	ASME III-1	
PXS-PL-V010A	CMT A Upper Sample	B	I	ASME III-2	
PXS-PL-V010B	CMT B Upper Sample	B	I	ASME III-2	
PXS-PL-V011A	CMT A Lower Sample	B	I	ASME III-2	
PXS-PL-V011B	CMT B Lower Sample	B	I	ASME III-2	
PXS-PL-V012A	CMT A Drain	A	I	ASME III-1	
PXS-PL-V012B	CMT B Drain	A	I	ASME III-1	
PXS-PL-V013A	CMT A Discharge Manual Isolation	A	I	ASME III-1	
PXS-PL-V013B	CMT B Discharge Manual Isolation	A	I	ASME III-1	
PXS-PL-V014A	CMT A Discharge Isolation	A	I	ASME III-1	
PXS-PL-V014B	CMT B Discharge Isolation	A	I	ASME III-1	
PXS-PL-V015A	CMT A Discharge Isolation	A	I	ASME III-1	
PXS-PL-V015B	CMT B Discharge Isolation	A	I	ASME III-1	
PXS-PL-V016A	CMT A Discharge Check	A	I	ASME III-1	
PXS-PL-V016B	CMT B Discharge Check	A	I	ASME III-1	
PXS-PL-V017A	CMT A Discharge Check	A	I	ASME III-1	
PXS-PL-V017B	CMT B Discharge Check	A	I	ASME III-1	
PXS-PL-V019A	RNS to CMT Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V019B	RNS to CMT Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V020A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V020B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V021A	Accumulator A Nitrogen Vent	C	I	ASME III-3	
PXS-PL-V021B	Accumulator B Nitrogen Vent	C	I	ASME III-3	
PXS-PL-V022A	Accumulator A Pressure Relief	C	I	ASME III-3	
PXS-PL-V022B	Accumulator B Pressure Relief	C	I	ASME III-3	
PXS-PL-V023A	Accumulator A Pressure Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V023B	Accumulator B Pressure Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V024A	Accumulator A Pressure Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V024B	Accumulator B Pressure Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V025A	Accumulator A Sample	C	I	ASME III-3	
PXS-PL-V025B	Accumulator B Sample	C	I	ASME III-3	
PXS-PL-V026A	Accumulator A Drain	C	I	ASME III-3	

Table 3.2-3 (Sheet 15 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V026B	Accumulator B Drain	C	I	ASME III-3	
PXS-PL-V027A	Accumulator A Discharge Isolation	C	I	ASME III-3	
PXS-PL-V027B	Accumulator B Discharge Isolation	C	I	ASME III-3	
PXS-PL-V028A	Accumulator A Discharge Check	A	I	ASME III-1	
PXS-PL-V028B	Accumulator B Discharge Check	A	I	ASME III-1	
PXS-PL-V029A	Accumulator A Discharge Check	A	I	ASME III-1	
PXS-PL-V029B	Accumulator B Discharge Check	A	I	ASME III-1	
PXS-PL-V030A	CMT A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V030B	CMT B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V031A	CMT A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V031B	CMT B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V033A	Accumulator A Check Valve Drain	B	I	ASME III-2	
PXS-PL-V033B	Accumulator B Check Valve Drain	B	I	ASME III-2	
PXS-PL-V042	Nitrogen Supply Containment Isolation ORC	B	I	ASME III-2	
PXS-PL-V043	Nitrogen Supply Containment Isolation IRC	B	I	ASME III-2	
PXS-PL-V052	Accumulator Nitrogen Containment Penetration TC	B	I	ASME III-2	
PXS-PL-V080A	CMT A WR Level Isolation	B	I	ASME III-2	
PXS-PL-V080B	CMT B WR Level Isolation	B	I	ASME III-2	
PXS-PL-V081A	CMT A WR Level Isolation	B	I	ASME III-2	
PXS-PL-V081B	CMT B WR Level Isolation	B	I	ASME III-2	
PXS-PL-V082A	CMT A Upper Level A Isolation 1	B	I	ASME III-2	
PXS-PL-V082B	CMT B Upper Level A Isolation 1	B	I	ASME III-2	
PXS-PL-V083A	CMT A Upper Level A Isolation 2	B	I	ASME III-2	
PXS-PL-V083B	CMT B Upper Level A Isolation 2	B	I	ASME III-2	
PXS-PL-V084A	CMT A Upper Level A Vent	B	I	ASME III-2	
PXS-PL-V084B	CMT B Upper Level A Vent	B	I	ASME III-2	
PXS-PL-V085A	CMT A Upper Level A Drain	B	I	ASME III-2	
PXS-PL-V085B	CMT B Upper Level A Drain	B	I	ASME III-2	
PXS-PL-V086A	CMT A Upper Level B Isolation 1	B	I	ASME III-2	
PXS-PL-V086B	CMT B Upper Level B Isolation 1	B	I	ASME III-2	

Table 3.2-3 (Sheet 16 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V087A	CMT A Upper Level B Isolation 2	B	I	ASME III-2	
PXS-PL-V087B	CMT B Upper Level B Isolation 2	B	I	ASME III-2	
PXS-PL-V088A	CMT A Upper Level B Vent	B	I	ASME III-2	
PXS-PL-V088B	CMT B Upper Level B Vent	B	I	ASME III-2	
PXS-PL-V089A	CMT A Upper Level B Drain	B	I	ASME III-2	
PXS-PL-V089B	CMT B Upper Level B Drain	B	I	ASME III-2	
PXS-PL-V092A	CMT A Lower Level A Isolation 1	B	I	ASME III-2	
PXS-PL-V092B	CMT B Lower Level A Isolation 1	B	I	ASME III-2	
PXS-PL-V093A	CMT A Lower Level A Isolation 2	B	I	ASME III-2	
PXS-PL-V093B	CMT B Lower Level A Isolation 2	B	I	ASME III-2	
PXS-PL-V094A	CMT A Lower Level A Vent	B	I	ASME III-2	
PXS-PL-V094B	CMT B Lower Level A Vent	B	I	ASME III-2	
PXS-PL-V095A	CMT A Lower Level A Drain	B	I	ASME III-2	
PXS-PL-V095B	CMT B Lower Level A Drain	B	I	ASME III-2	
PXS-PL-V096A	CMT A Lower Level B Isolation 1	B	I	ASME III-2	
PXS-PL-V096B	CMT B Lower Level B Isolation 1	B	I	ASME III-2	
PXS-PL-V097A	CMT A Lower Level B Isolation 2	B	I	ASME III-2	
PXS-PL-V097B	CMT B Lower Level B Isolation 2	B	I	ASME III-2	
PXS-PL-V098A	CMT A Lower Level B Vent	B	I	ASME III-2	
PXS-PL-V098B	CMT B Lower Level B Vent	B	I	ASME III-2	
PXS-PL-V099A	CMT A Lower Level B Drain	B	I	ASME III-2	
PXS-PL-V099B	CMT B Lower Level B Drain	B	I	ASME III-2	
PXS-PL-V101	PRHR HX Inlet Isolation	A	I	ASME III-1	
PXS-PL-V102A	PRHR HX Inlet Head Vent	B	I	ASME III-2	
PXS-PL-V102B	PRHR HX Inlet Head Drain	B	I	ASME III-2	
PXS-PL-V103A	PRHR HX Outlet Head Vent	B	I	ASME III-2	
PXS-PL-V103B	PRHR HX Outlet Head Drain	B	I	ASME III-2	
PXS-PL-V104A	PRHR HX Flow Transmitter A Isolation	B	I	ASME III-2	
PXS-PL-V104B	PRHR HX Flow Transmitter B Isolation	B	I	ASME III-2	
PXS-PL-V105A	PRHR HX Flow Transmitter A Isolation	B	I	ASME III-2	
PXS-PL-V105B	PRHR HX Flow Transmitter B Isolation	B	I	ASME III-2	

Table 3.2-3 (Sheet 17 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V106	Containment Recirculation A Highpoint Vent	C	I	ASME III-3	
PXS-PL-V107	Containment Recirculation A Highpoint Vent	C	I	ASME III-3	
PXS-PL-V108A	PRHR HX Control	A	I	ASME III-1	
PXS-PL-V108B	PRHR HX Control	A	I	ASME III-1	
PXS-PL-V109	PRHR HX/RCS Return Isolation	A	I	ASME III-1	
PXS-PL-V111A	PRHR HX Highpoint Vent	B	I	ASME III-2	
PXS-PL-V111B	PRHR HX Highpoint Vent	B	I	ASME III-2	
PXS-PL-V113	PRHR HX Pressure Transmitter Isolation	B	I	ASME III-2	
PXS-PL-V115A	Containment Recirculation A Drain	C	I	ASME III-3	
PXS-PL-V115B	Containment Recirculation B Drain	C	I	ASME III-3	
PXS-PL-V116A	Containment Recirculation A Drain	C	I	ASME III-3	
PXS-PL-V116B	Containment Recirculation B Drain	C	I	ASME III-3	
PXS-PL-V117A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V117B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V118A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V118B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V119A	Containment Recirculation A Check	C	I	ASME III-3	
PXS-PL-V119B	Containment Recirculation B Check	C	I	ASME III-3	
PXS-PL-V120A	Containment Recirculation A Isolation	C	I	ASME III-3	
PXS-PL-V120B	Containment Recirculation B Isolation	C	I	ASME III-3	
PXS-PL-V121A	IRWST Line A Isolation	C	I	ASME III-3	
PXS-PL-V121B	IRWST Line B Isolation	C	I	ASME III-3	
PXS-PL-V122A	IRWST Injection A Check	A	I	ASME III-1	
PXS-PL-V122B	IRWST Injection B Check	A	I	ASME III-1	
PXS-PL-V123A	IRWST Injection A Isolation	A	I	ASME III-1	
PXS-PL-V123B	IRWST Injection B Isolation	A	I	ASME III-1	
PXS-PL-V124A	IRWST Injection A Check	A	I	ASME III-1	
PXS-PL-V124B	IRWST Injection B Check	A	I	ASME III-1	
PXS-PL-V125A	IRWST Injection A Isolation	A	I	ASME III-1	
PXS-PL-V125B	IRWST Injection B Isolation	A	I	ASME III-1	

Table 3.2-3 (Sheet 18 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V126A	IRWST Injection Check Test	C	I	ASME III-3	
PXS-PL-V126B	IRWST Injection Check Test	C	I	ASME III-3	
PXS-PL-V127	IRWST Injection Line A Drain	C	I	ASME III-3	
PXS-PL-V128A	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V128B	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V129A	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V129B	IRWST Injection Check Test	A	I	ASME III-1	
PXS-PL-V130A	IRWST Gutter Bypass A Isolation	C	I	ASME III-3	
PXS-PL-V130B	IRWST Gutter Bypass B Isolation	C	I	ASME III-3	
PXS-PL-V131A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V131B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V132A	IRWST Injection Line A Drain	B	I	ASME III-2	
PXS-PL-V132B	IRWST Injection Line B Drain	B	I	ASME III-2	
PXS-PL-V133A	IRWST Injection Line A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V133B	IRWST Injection Line B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V134A	IRWST Injection Line A Highpoint Vent	B	I	ASME III-2	
PXS-PL-V134B	IRWST Injection Line B Highpoint Vent	B	I	ASME III-2	
PXS-PL-V135A	IRWST Injection Line A Highpoint Vent Isolation	B	I	ASME III-2	
PXS-PL-V135B	IRWST Injection Line B Highpoint Vent Isolation	B	I	ASME III-2	
PXS-PL-V149	RNS Suction Pump Line Drain	C	I	ASME III-3	
PXS-PL-V150A	IRWST Level Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V150B	IRWST Level Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V150C	IRWST Level Transmitter C Isolation	C	I	ASME III-3	
PXS-PL-V150D	IRWST Level Transmitter D Isolation	C	I	ASME III-3	
PXS-PL-V151A	IRWST Level Transmitter A Isolation	C	I	ASME III-3	
PXS-PL-V151B	IRWST Level Transmitter B Isolation	C	I	ASME III-3	
PXS-PL-V151C	IRWST Level Transmitter C Isolation	C	I	ASME III-3	
PXS-PL-V151D	IRWST Level Transmitter D Isolation	C	I	ASME III-3	
PXS-PL-V201A	Accumulator A Leak Test	B	I	ASME III-2	
PXS-PL-V201B	Accumulator B Leak Test	B	I	ASME III-2	

Table 3.2-3 (Sheet 19 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Passive Core Cooling System (Continued)					
PXS-PL-V202A	Accumulator A Leak Test	C	I	ASME III-3	
PXS-PL-V202B	Accumulator B Leak Test	C	I	ASME III-3	
PXS-PL-V205A	RNS Discharge Leak Test	B	I	ASME III-2	
PXS-PL-V205B	RNS Discharge Leak Test	B	I	ASME III-2	
PXS-PL-V206	RNS Discharge Leak Test	C	I	ASME III-3	
PXS-PL-V207A	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V207B	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V208A	RNS Suction Leak Test	B	I	ASME III-2	
PXS-PL-V217	PXS Leak Test Line Isolation	D	NS	ANSI B31.1	
PXS-PL-V221	Test Header to IRWST	D	NS	ANSI B31.1	
PXS-PL-V230A	CMT A Fill Isolation	B	I	ASME III-2	
PXS-PL-V230B	CMT B Fill Isolation	B	I	ASME III-2	
PXS-PL-V231A	CMT A Fill Check	B	I	ASME III-2	
PXS-PL-V231B	CMT B Fill Check	B	I	ASME III-2	
PXS-PL-V232A	Accumulator A Fill/Drain Isolation	C	I	ASME III-3	
PXS-PL-V232B	Accumulator B Fill/Drain Isolation	C	I	ASME III-3	
PXS-PL-V250A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V250B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V251A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V251B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V252A	CMT A Check Valve Test Valve	A	I	ASME III-1	
PXS-PL-V252B	CMT B Check Valve Test Valve	A	I	ASME III-1	
PXS-PY-C01	Nitrogen Makeup Containment Penetration	B	I	ASME III, 2	
Balance of system components are Class E					
Reactor Coolant System (RCS)				Location: Containment	
RCS-MB-01	Steam Generator 1	A	I	ASME III-1	
RCS-MB-02	Steam Generator 2	A	I	ASME III-1	
RCS-MP-01A/B	SG 1A(B) Reactor Coolant Pump	A	I	ASME III-1	Pump Motor – Class D
n/a	Rotor Shaft	C	I	Manufacturer Std	
n/a	Impeller	C	I	Manufacturer Std	
n/a	Flywheel	C	I	Manufacturer Std	
n/a	RCP Heat Exchanger (Tube Side)	A	I	ASME III-1	Shellside – Class D, ASME VIII, Div. 1
n/a	Pump Motor Cooling Water to HX Inlet Connector	A	I	ASME III-1	

Table 3.2-3 (Sheet 20 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
n/a	Pump Motor Cooling Water from HX Outlet Connector	A	I	ASME III-1	
RCS-MP-02A/B	SG 2A(B) Reactor Coolant Pump	A	I	ASME III-1	Pump Motor – Class D
n/a	Rotor Shaft	C	I	Manufacturer Std	
n/a	Impeller	C	I	Manufacturer Std	
n/a	Flywheel	C	I	Manufacturer Std	
n/a	RCP Heat Exchanger (Tube Side)	A	I	ASME III-1	Shellside – Class D, ASME VIII, Div. 1
n/a	Pump Motor Cooling Water to HX Inlet Connector	A	I	ASME III-1	
n/a	Pump Motor Cooling Water from HX Outlet Connector	A	I	ASME III-1	
RCS-MV-01	Reactor Vessel	A	I	ASME III-1	
RCS-MV-02	Pressurizer	A	I	ASME III-1	
RCS-MY-Y11	SG 1 Shell	B	I	ASME III-1	
RCS-MY-Y12	SG 1 Channel Head Divider Plate	B	I	ASME III-1	
RCS-MY-Y13	SG 1 Tube Bundle Support Assembly	C	I	ASME III, NG	
RCS-MY-Y14	SG 1 Steam Flow Limiting Venturi	B	I	ASME III, NG	
RCS-MY-Y15	SG 1 Feedwater Distribution Ring Supports	B	I	ASME III, NG	
RCS-MY-Y21	SG 2 Shell	B	I	ASME III-1	
RCS-MY-Y22	SG 2 Channel Head Divider Plate	B	I	ASME III-1	
RCS-MY-Y23	SG 2 Tube Bundle Support Assembly	C	I	ASME III, NG	
RCS-MY-Y24	SG 2 Steam Flow Limiting Venturi	B	I	ASME III, NG	
RCS-MY-Y25	SG 2 Feedwater Distribution Ring Supports	B	I	ASME III, NG	
RCS-PL-V001A	First Stage ADS	A	I	ASME III-1	
RCS-PL-V001B	First Stage ADS	A	I	ASME III-1	
RCS-PL-V002A	Second Stage ADS	A	I	ASME III-1	
RCS-PL-V002B	Second Stage ADS	A	I	ASME III-1	
RCS-PL-V003A	Third Stage ADS	A	I	ASME III-1	
RCS-PL-V003B	Third Stage ADS	A	I	ASME III-1	
RCS-PL-V004A	Fourth Stage ADS	A	I	ASME III-1	
RCS-PL-V004B	Fourth Stage ADS	A	I	ASME III-1	
RCS-PL-V004C	Fourth Stage ADS	A	I	ASME III-1	
RCS-PL-V004D	Fourth Stage ADS	A	I	ASME III-1	
RCS-PL-V005A	Pressurizer Safety Valve	A	I	ASME III-1	
RCS-PL-V005B	Pressurizer Safety Valve	A	I	ASME III-1	
RCS-PL-V007A	ADS Test Valve	B	I	ASME III-2	

Table 3.2-3 (Sheet 21 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
RCS-PL-V007B	ADS Test Valve	B	I	ASME III-2	
RCS-PL-V007C	ADS Test Valve	B	I	ASME III-2	
RCS-PL-V008	ADS Valve Leakage Check Valve	C	I	ASME III-3	
RCS-PL-V010A	ADS Discharge Header A Vacuum Relief	C	I	ASME III-3	
RCS-PL-V010B	ADS Discharge Header B Vacuum Relief	C	I	ASME III-3	
RCS-PL-V011A	First Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V011B	First Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V012A	Second Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V012B	Second Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V013A	Third Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V013B	Third Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V014A	Fourth Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V014B	Fourth Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V014C	Fourth Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V014D	Fourth Stage ADS Isolation	A	I	ASME III-1	
RCS-PL-V095	Hot Leg 2 Level Instrument Root	B	I	ASME III-2	
RCS-PL-V096	Hot Leg 2 Level Instrument Root	B	I	ASME III-2	ADS Test Valve
RCS-PL-V097	Hot Leg 1 Level Instrument Root	B	I	ASME III-2	
RCS-PL-V098	Hot Leg 1 Level Instrument Root	B	I	ASME III-2	
RCS-PL-V101A	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101B	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101C	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101D	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101E	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V101F	Hot Leg 1 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102A	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102B	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102C	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102D	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	

Table 3.2-3 (Sheet 22 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
RCS-PL-V102E	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V102F	Hot Leg 2 Flow Instrument Root	B	I	ASME III-2	
RCS-PL-V103	PRHR HX Outlet Line Drain	B	I	ASME III-2	
RCS-PL-V171A	Cold Leg 1A Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V171B	Cold Leg 1A Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V172A	Cold Leg 1B Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V172B	Cold Leg 1B Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V173A	Cold Leg 2A Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V173B	Cold Leg 2A Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V174A	Cold Leg 2B Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V174B	Cold Leg 2B Bend Instrument Root	B	I	ASME III-2	
RCS-PL-V108A	Hot Leg 1 Sample Isolation	B	I	ASME III-2	
RCS-PL-V108B	Hot Leg 2 Sample Isolation	B	I	ASME III-2	
RCS-PL-V110A	Pressurizer Spray Valve	A	I	ASME III-1	
RCS-PL-V110B	Pressurizer Spray Valve	A	I	ASME III-1	
RCS-PL-V111A	Pressurizer Spray Block Valve	A	I	ASME III-1	
RCS-PL-V111B	Pressurizer Spray Block Valve	A	I	ASME III-1	
RCS-PL-V120	Reactor Vessel Flange Leakoff	D	NS	ANSI B31.1	
RCS-PL-V121	Reactor Vessel Flange Leakoff	D	NS	ANSI B31.1	
RCS-PL-V122A	Reactor Vessel Flange Leakoff	D	NS	ANSI B31.1	
RCS-PL-V122B	Reactor Vessel Flange Leakoff	D	NS	ANSI B31.1	
RCS-PL-V150A	Reactor Vessel Head Vent	A	I	ASME III-1	
RCS-PL-V150B	Reactor Vessel Head Vent	A	I	ASME III-1	
RCS-PL-V150C	Reactor Vessel Head Vent	A	I	ASME III-1	
RCS-PL-V150D	Reactor Vessel Head Vent	A	I	ASME III-1	
RCS-PL-V204	Pressurizer Manual Vent	A	I	ASME III-1	
RCS-PL-V205	Pressurizer Manual Vent	A	I	ASME III-1	
RCS-PL-V210A	Pressurizer Spray Bypass	B	I	ASME III-2	
RCS-PL-V210B	Pressurizer Spray Bypass	B	I	ASME III-2	
RCS-PL-V015	Pressurizer Vent to RCDT Test Valve	D	NS	ANSI 16.34	

Table 3.2-3 (Sheet 23 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor Coolant System (Continued)					
RCS-PL-V225A	Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V225B	Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V225C	Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V225D	Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V226A	Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V226B	Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V226C	Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V226D	Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V228	Wide Range Pressurizer Level Steam Space Instrument Root	B	I	ASME III-2	
RCS-PL-V229	Wide Range Pressurizer Level Liquid Space Instrument Root	B	I	ASME III-2	
RCS-PL-V232	Manual Head Vent	C	I	ASME III-3	
RCS-PL-V233	Head Vent Isolation	C	I	ASME III-3	
RCS-PL-V241	ADS Valve Discharge Header Drain Isolation	C	I	ASME III-3	
RCS-PL-V242	ADS Valve Discharge Header Drain Check	D	NS	ANSI 16.34	
RCS-PL-V261A	RCP 1A Drain	A	I	ASME III-1	
RCS-PL-V261B	RCP 1B Drain	A	I	ASME III-1	
RCS-PL-V261C	RCP 2A Drain	A	I	ASME III-1	
RCS-PL-V261D	RCP 2B Drain	A	I	ASME III-1	
RCS-PL-V260A	RCP 1A Vent	A	I	ASME III-1	
RCS-PL-V260B	RCP 1B Vent	A	I	ASME III-1	
RCS-PL-V260C	RCP 2A Vent	A	I	ASME III-1	
RCS-PL-V260D	RCP 2B Vent	A	I	ASME III-1	
RCS-PY-K03	Safety Valve Discharge Chamber Rupture Disk	C	I	ASME III-3	
RCS-PY-K04	Safety Valve Discharge Chamber Rupture Disk	C	I	ASME III-3	
Gravity and Roof Drain Collection System (RDS)				Location: Various	
System components are Class E					

Table 3.2-3 (Sheet 24 of 58) AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Normal Residual Heat Removal System (RNS)			Location: Containment and Auxiliary Building		
RNS-ME-01A	Normal Residual Heat Removal Heat Exchanger A (Tube Side)	C	I	ASME III-3	Shellside – Class D ASME VIII, Div. 1
RNS-ME-01B	Normal Residual Heat Removal Heat Exchanger B (Tube Side)	C	I	ASME III-3	Shellside – Class D ASME VIII, Div. 1
RNS-MP-01A	Residual Heat Removal Pump A	C	I	ASME III-3	Pump Motor - Class D
RNS-MP-01B	Residual Heat Removal Pump B	C	I	ASME III-3	Pump Motor - Class D
RNS-PL-V001A	RNS HL Suction Isolation - Inner	A	I	ASME III-1	
RNS-PL-V001B	RNS HL Suction Isolation - Inner	A	I	ASME III-1	
RNS-PL-V002A	RNS HL Suction and Containment Isolation - Outer	A	I	ASME III-1	
RNS-PL-V002B	RNS HL Suction and Containment Isolation - Outer	A	I	ASME III-1	
RNS-PL-V003A	RCS Pressure Boundary Valve Thermal Relief	B	I	ASME III-2	
RNS-PL-V003B	RCS Pressure Boundary Valve Thermal Relief	B	I	ASME III-2	
RNS-PL-V004A	RCS Pressure Boundary Valve Thermal Relief Isolation	B	I	ASME III-2	
RNS-PL-V004B	RCS Pressure Boundary Valve Thermal Relief Isolation	B	I	ASME III-2	
RNS-PL-V005A	RNS Pump A Suction Isolation	C	I	ASME III-3	
RNS-PL-V005B	RNS Pump B Suction Isolation	C	I	ASME III-3	
RNS-PL-V006A	RNS HX A Outlet Flow Control	C	I	ASME III-3	
RNS-PL-V006B	RNS HX B Outlet Flow Control	C	I	ASME III-3	
RNS-PL-V007A	RNS Pump A Discharge Isolation	C	I	ASME III-3	
RNS-PL-V007B	RNS Pump B Discharge Isolation	C	I	ASME III-3	
RNS-PL-V008A	RNS HX A Bypass Flow Control	C	I	ASME III-3	
RNS-PL-V008B	RNS HX B Bypass Flow Control	C	I	ASME III-3	
RNS-PL-V010	RNS Discharge Containment Isolation Valve Test	C	I	ASME III-3	
RNS-PL-V011	RNS Discharge Containment Isolation Valve - ORC	B	I	ASME III-2	
RNS-PL-V012	RNS Discharge Containment Isolation Valve Test Connection ORC	B	I	ASME III-2	
RNS-PL-V013	RNS Discharge Containment Isolation - IRC	B	I	ASME III-2	

Table 3.2-3 (Sheet 25 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Normal Residual Heat Removal System (Continued)					
RNS-PL-V014	RNS Discharge Containment Isolation Valve Test Connection	C	I	ASME III-3	
RNS-PL-V015A	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V015B	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V016	RNS Discharge Containment Penetration Isolation Valves Test	C	I	ASME III-3	
RNS-PL-V017A	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V017B	RNS Discharge RCS Pressure Boundary	A	I	ASME III-1	
RNS-PL-V021	RNS HL Suction Pressure Relief	B	I	ASME III-2	
RNS-PL-V022	RNS Suction Header Containment Isolation - ORC	B	I	ASME III-2	
RNS-PL-V023	RNS Suction from IRWST - Containment Isolation	B	I	ASME III-2	
RNS-PL-V024	RNS Discharge to IRWST Isolation	C	I	ASME III-3	
RNS-PL-V029	RNS Discharge to CVS	C	I	ASME III-3	
RNS-PL-V030A	RNS HX A Shell Drain	D	NS	ANSI B31.1	
RNS-PL-V030B	RNS HX B Shell Drain	D	NS	ANSI B31.1	
RNS-PL-V031A	RNS Train A Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V031B	RNS Train B Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V032A	RNS Train A Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V032B	RNS Train B Discharge Flow Instrument Isolation	C	I	ASME III-3	
RNS-PL-V033A	RNS Pump A Suction Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V033B	RNS Pump B Suction Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V034A	RNS Pump A Discharge Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V034B	RNS Pump B Discharge Pressure Instrument Isolation	C	I	ASME III-3	
RNS-PL-V035A	RNS HX A Shell Vent	D	NS	ANSI 16.34	
RNS-PL-V035B	RNS HX B Shell Vent	D	NS	ANSI 16.34	
RNS-PL-V036A	RNS Pump A Suction Piping Drain. Isolation	C	I	ASME III-3	
RNS-PL-V036B	RNS Pump B Suction Piping Drain. Isolation	C	I	ASME III-3	

Table 3.2-3 (Sheet 26 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Normal Residual Heat Removal System (Continued)					
RNS-PL-V045	RNS Pump Discharge Relief	C	I	ASME III-3	
RNS-PL-V048A	RNS Pump Seal Cooler A Vent Isolation	C	I	ASME III-3	
RNS-PL-V048B	RNS Pump Seal Cooler B Vent Isolation	C	I	ASME III-3	
RNS-PL-V049A	RNS Pump Seal Cooler A Drain Isolation	C	I	ASME III-3	
RNS-PL-V049B	RNS Pump Seal Cooler B Drain Isolation	C	I	ASME III-3	
RNS-PL-V050	RNS Pump A Casing Drain. Isolation	C	I	ASME III-3	
RNS-PL-V051	RNS Pump B Casing Drain. Isolation	C	I	ASME III-3	
RNS-PL-V052	RNS Pump Suction From Spent Fuel Pool Isolation	C	I	ASME III-3	
RNS-PL-V053	RNS Pump Discharge to Spent Fuel Pool Isolation	C	I	ASME III-3	
RNS-PL-V055	RNS Pump Suction to Cask Loading Pit Isolation	C	I	ASME III-3	
RNS-PL-V056	RNS Pump Suction to Cask Loading Pit Isolation	C	I	ASME III-3	
RNS-PL-V057A	RNS Pump A Miniflow Isolation	C	I	ASME III-3	
RNS-PL-V057B	RNS Pump B Miniflow Isolation	C	I	ASME III-3	
RNS-PL-V059	RNS Pump Suction Containment Isolation Test Connection	C	I	ASME III-3	
RNS-PL-V061	RNS Return from CVS - Containment Isolation	B	I	ASME III-2	
RNS-PL-V065	RNS Discharge Drain Valve	C	I	ASME III-3	
RNS-PL-V066A	RNS Discharge to DVI Line A Drain	C	I	ASME III-3	
RNS-PL-V066B	RNS Discharge to DVI Line B Drain	C	I	ASME III-3	
RNS-PL-V067A	RNS Discharge to DVI Line A Drain	B	I	ASME III-2	
RNS-PL-V067B	RNS Discharge to DVI Line B Drain	B	I	ASME III-2	
RNS-PL-V068	RNS Discharge to IRWST Drain	C	I	ASME III-3	
RNS-PL-V069A	RNS Pump A Miniflow Vent	C	I	ASME III-3	
RNS-PL-V069B	RNS Pump B Miniflow Vent	C	I	ASME III-3	
RNS-PL-V071A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V071B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	

Table 3.2-3 (Sheet 27 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Normal Residual Heat Removal System (Continued)					
RNS-PL-V072A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V072B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V073A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V073B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V074A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V074B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V075A	RNS HX A Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V075B	RNS HX B Channel Head Drain Isolation	C	I	ASME III-3	
RNS-PL-V080	IRWST Suction Line to RNS Pump Vent	B	I	ASME III-2	
RNS-PL-V081	RNS Cask Loading Pit Suction Line Vent	C	I	ASME III-3	
RNS-PL-V082	RNS Discharge Drain	C	I	ASME III-3	
RNS-PY-C01	Normal Residual Heat Removal Suction Line Penetration	B	I	ASME III, MC	
RNS-PY-C02	Normal Residual Heat Removal Discharge Line Penetration	B	I	ASME III, MC	
Balance of system components are Class E					
Raw Water System (RWS)				Location: Yard, Turbine Building	
System components are Class E					
Reactor System (RXS)				Location: Containment	
n/a	Fuel Assemblies	C	I	Manufacturer Std.	
RXS-FR-B06	Control Rod Cluster B6	B	I	Manufacturer Std.	
RXS-FR-B10	Control Rod Cluster B10	B	I	Manufacturer Std.	
RXS-FR-C05	Control Rod Cluster C5	B	I	Manufacturer Std.	
RXS-FR-C07	Control Rod Cluster C7	B	I	Manufacturer Std.	
RXS-FR-C09	Control Rod Cluster C9	B	I	Manufacturer Std.	
RXS-FR-C11	Control Rod Cluster C11	B	I	Manufacturer Std.	
RXS-FR-D06	Control Rod Cluster D6	B	I	Manufacturer Std.	
RXS-FR-D08	Control Rod Cluster D8	B	I	Manufacturer Std.	
RXS-FR-D10	Control Rod Cluster D10	B	I	Manufacturer Std.	
RXS-FR-E03	Control Rod Cluster E3	B	I	Manufacturer Std.	
RXS-FR-E05	Control Rod Cluster E5	B	I	Manufacturer Std.	
RXS-FR-E07	Control Rod Cluster E7	B	I	Manufacturer Std.	
RXS-FR-E09	Control Rod Cluster E9	B	I	Manufacturer Std.	
RXS-FR-E11	Control Rod Cluster E11	B	I	Manufacturer Std.	

Table 3.2-3 (Sheet 28 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-FR-E13	Control Rod Cluster E13	B	I	Manufacturer Std.	
RXS-FR-F02	Control Rod Cluster F2	B	I	Manufacturer Std.	
RXS-FR-F04	Control Rod Cluster F4	B	I	Manufacturer Std.	
RXS-FR-F12	Control Rod Cluster F12	B	I	Manufacturer Std.	
RXS-FR-F14	Control Rod Cluster F14	B	I	Manufacturer Std	
RXS-FR-G03	Control Rod Cluster G3	B	I	Manufacturer Std.	
RXS-FR-G05	Control Rod Cluster G5	B	I	Manufacturer Std.	
RXS-FR-G07	Control Rod Cluster G7	B	I	Manufacturer Std.	
RXS-FR-G09	Control Rod Cluster G9	B	I	Manufacturer Std.	
RXS-FR-G11	Control Rod Cluster G11	B	I	Manufacturer Std.	
RXS-FR-G13	Control Rod Cluster G13	B	I	Manufacturer Std.	
RXS-FR-H04	Control Rod Cluster H4	B	I	Manufacturer Std.	
RXS-FR-H08	Control Rod Cluster H8	B	I	Manufacturer Std.	
RXS-FR-H12	Control Rod Cluster H12	B	I	Manufacturer Std.	
RXS-FR-J03	Control Rod Cluster J3	B	I	Manufacturer Std.	
RXS-FR-J05	Control Rod Cluster J5	B	I	Manufacturer Std.	
RXS-FR-J07	Control Rod Cluster J7	B	I	Manufacturer Std.	
RXS-FR-J09	Control Rod Cluster J9	B	I	Manufacturer Std.	
RXS-FR-J11	Control Rod Cluster J11	B	I	Manufacturer Std.	
RXS-FR-J13	Control Rod Cluster J13	B	I	Manufacturer Std.	
RXS-FR-K02	Control Rod Cluster K2	B	I	Manufacturer Std.	
RXS-FR-K04	Control Rod Cluster K4	B	I	Manufacturer Std.	
RXS-FR-K12	Control Rod Cluster K12	B	I	Manufacturer Std.	
RXS-FR-K14	Control Rod Cluster K14	B	I	Manufacturer Std.	
RXS-FR-L03	Control Rod Cluster L3	B	I	Manufacturer Std.	
RXS-FR-L05	Control Rod Cluster L5	B	I	Manufacturer Std.	
RXS-FR-L07	Control Rod Cluster L7	B	I	Manufacturer Std.	
RXS-FR-L09	Control Rod Cluster L9	B	I	Manufacturer Std.	
RXS-FR-L11	Control Rod Cluster L11	B	I	Manufacturer Std.	
RXS-FR-L13	Control Rod Cluster L13	B	I	Manufacturer Std.	
RXS-FR-M06	Control Rod Cluster M6	B	I	Manufacturer Std.	
RXS-FR-M08	Control Rod Cluster M8	B	I	Manufacturer Std.	
RXS-FR-M10	Control Rod Cluster M10	B	I	Manufacturer Std.	
RXS-FR-N5	Control Rod Cluster N5	B	I	Manufacturer Std.	
RXS-FR-N7	Control Rod Cluster N7	B	I	Manufacturer Std.	
RXS-FR-N9	Control Rod Cluster N9	B	I	Manufacturer Std.	
RXS-FR-N11	Control Rod Cluster N11	B	I	Manufacturer Std.	
RXS-FR-P6	Control Rod Cluster P6	B	I	Manufacturer Std.	
RXS-FR-P10	Control Rod Cluster P10	B	I	Manufacturer Std.	
RXS-MI-01	Reactor Upper Internals	C	I	ASME III, CS	
RXS-MI-02	Reactor Lower Internals	C	I	ASME III, CS	
RXS-MI-10	Non-Threaded Fasteners	D	II	Manufacturer Std.	
RXS-MI-11	Threaded Structural Fasteners	C	I	ASME III, CS	
RXS-MI-20	Lower Core Support Plate	C	I	ASME III, CS	

Table 3.2-3 (Sheet 29 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MI-22	Vortex Suppression Plate	D	II	Manufacturer Std.	
RXS-MI-23	Core Shroud Assembly	D	II	Manufacturer Std.	
RXS-MI-24	Radial Supports [4]	C	I	ASME III, CS	
RXS-MI-25	Core Barrel	C	I	ASME III, CS	
RXS-MI-26	Core Barrel Nozzle	C	I	ASME III, CS	
RXS-MI-27	Head and Vessel Pins	D	II	Manufacturer Std.	
RXS-MI-28	Lower Support Plate Fuel Alignment Pins	C	I	ASME III, CS	
RXS-MI-29	Core Barrel Hold Down Spring	C	I	Manufacturer Std.	
RXS-MI-50	Upper Support	C	I	ASME III, CS	
RXS-MI-51	Upper Core Plate	C	I	ASME III, CS	
RXS-MI-52	Support Columns [42]	C	I	ASME III, CS	
RXS-MI-53	Guide Tube Assemblies [69]	C	I	Manufacturer Std.	
RXS-MI-54	Upper Core Plate Fuel Alignment Pins	C	I	ASME III, CS	
RXS-MI-55	Upper Core Plate Inserts	C	I	ASME III, CS	
RXS-MI-56	Safety Injection Deflector	D	II	Manufacturer Std.	
RXS-MI-57	Irradiation Specimen Guide Tubes	D	II	Manufacturer Std.	
RXS-MI-58	Head Cooling Nozzles	D	II	Manufacturer Std.	
n/a	Neutron Pad	D	II	Manufacturer Std.	
n/a	Instrument Grid Assembly	D	II	Manufacturer Std.	
n/a	DVI Flow Diverter	D	II	Manufacturer Std.	
RXS-MI-80	Reactor Vessel Flow Skirt	D	II	Manufacturer Std.	
RXS-MN-01	Reactor Vessel Cavity Reflective Insulation	D	II	Manufacturer Std.	
RXS-MV-10	Reactor Integrated Head Package	C	I	AISC-690	
RXS-MV-10A	Integrated Head Package Shroud	C	I	ASME-NF	
RXS-MV-10B	Integrated Head Package Seismic Support System	C	I	ASME-NF	
RXS-MV-11B06	CRDM Latch Assembly B6	D	NS	Manufacturer Std.	
RXS-MV-11B06	CRDM Drive Rod Assembly B6	D	NS	Manufacturer Std.	
RXS-MV-11B06	CRDM Coil Stack Assembly B6	D	NS	Manufacturer Std.	
RXS-MV-11B06L	CRDM Latch Housing B6	A	I	ASME III-1	
RXS-MV-11B06R	CRDM Rod Travel Housing B6	A	I	ASME III-1	
RXS-MV-11B08	CRDM Latch Assembly B8	D	NS	Manufacturer Std.	
RXS-MV-11B08	CRDM Drive Rod Assembly B8	D	NS	Manufacturer Std.	
RXS-MV-11B08	CRDM Coil Stack Assembly B8	D	NS	Manufacturer Std.	
RXS-MV-11B08L	CRDM Latch Housing B8	A	I	ASME III-1	
RXS-MV-11B08R	CRDM Rod Travel Housing B8	A	I	ASME III-1	

Table 3.2-3 (Sheet 30 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11B10	CRDM Latch Assembly B10	D	NS	Manufacturer Std.	
RXS-MV-11B10	CRDM Drive Rod Assembly B10	D	NS	Manufacturer Std.	
RXS-MV-11B10	CRDM Coil Stack Assembly B10	D	NS	Manufacturer Std.	
RXS-MV-11B10L	CRDM Latch Housing B10	A	I	ASME III-1	
RXS-MV-11B10R	CRDM Rod Travel Housing B10	A	I	ASME III-1	
RXS-MV-11C05	CRDM Latch Assembly C5	D	NS	Manufacturer Std.	
RXS-MV-11C05	CRDM Drive Rod Assembly C5	D	NS	Manufacturer Std.	
RXS-MV-11C05	CRDM Coil Stack Assembly C5	D	NS	Manufacturer Std.	
RXS-MV-11C05L	CRDM Latch Housing C5	A	I	ASME III-1	
RXS-MV-11C05R	CRDM Rod Travel Housing C5	A	I	ASME III-1	
RXS-MV-11C07	CRDM Latch Assembly C7	D	NS	Manufacturer Std.	
RXS-MV-11C07	CRDM Drive Rod Assembly C7	D	NS	Manufacturer Std.	
RXS-MV-11C07	CRDM Coil Stack Assembly C7	D	NS	Manufacturer Std.	
RXS-MV-11C07L	CRDM Latch Housing C7	A	I	ASME III-1	
RXS-MV-11C07R	CRDM Rod Travel Housing C7	A	I	ASME III-1	
RXS-MV-11C09	CRDM Latch Assembly C9	D	NS	Manufacturer Std.	
RXS-MV-11C09	CRDM Drive Rod Assembly C9	D	NS	Manufacturer Std.	
RXS-MV-11C09	CRDM Coil Stack Assembly C9	D	NS	Manufacturer Std.	
RXS-MV-11C09L	CRDM Latch Housing C9	A	I	ASME III-1	
RXS-MV-11C09R	CRDM Rod Travel Housing C9	A	I	ASME III-1	
RXS-MV-11C11	CRDM Latch Assembly C11	D	NS	Manufacturer Std.	
RXS-MV-11C11	CRDM Drive Rod Assembly C11	D	NS	Manufacturer Std.	
RXS-MV-11C11	CRDM Coil Stack Assembly C11	D	NS	Manufacturer Std.	
RXS-MV-11C11L	CRDM Latch Housing C11	A	I	ASME III-1	
RXS-MV-11C11R	CRDM Rod Travel Housing C11	A	I	ASME III-1	
RXS-MV-11D04	CRDM Latch Assembly D4	D	NS	Manufacturer Std.	
RXS-MV-11D04	CRDM Drive Rod Assembly D4	D	NS	Manufacturer Std.	
RXS-MV-11D04	CRDM Coil Stack Assembly D4	D	NS	Manufacturer Std.	
RXS-MV-11D04L	CRDM Latch Housing D4	A	I	ASME III-1	

Table 3.2-3 (Sheet 31 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11D04R	CRDM Rod Travel Housing D4	A	I	ASME III-1	
RXS-MV-11D06	CRDM Latch Assembly D6	D	NS	Manufacturer Std.	
RXS-MV-11D06	CRDM Drive Rod Assembly D6	D	NS	Manufacturer Std.	
RXS-MV-11D06	CRDM Coil Stack Assembly D6	D	NS	Manufacturer Std.	
RXS-MV-11D06L	CRDM Latch Housing D6	A	I	ASME III-1	
RXS-MV-11D06R	CRDM Rod Travel Housing D6	A	I	ASME III-1	
RXS-MV-11D08	CRDM Latch Assembly D8	D	NS	Manufacturer Std.	
RXS-MV-11D08	CRDM Drive Rod Assembly D8	D	NS	Manufacturer Std.	
RXS-MV-11D08	CRDM Coil Stack Assembly D8	D	NS	Manufacturer Std.	
RXS-MV-11D08L	CRDM Latch Housing D8	A	I	ASME III-1	
RXS-MV-11D08R	CRDM Rod Travel Housing D8	A	I	ASME III-1	
RXS-MV-11D10	CRDM Latch Assembly D10	D	NS	Manufacturer Std.	
RXS-MV-11D10	CRDM Drive Rod Assembly D10	D	NS	Manufacturer Std.	
RXS-MV-11D10	CRDM Coil Stack Assembly D10	D	NS	Manufacturer Std.	
RXS-MV-11D10L	CRDM Latch Housing D10	A	I	ASME III-1	
RXS-MV-11D10R	CRDM Rod Travel Housing D10	A	I	ASME III-1	
RXS-MV-11D12	CRDM Latch Assembly D12	D	NS	Manufacturer Std.	
RXS-MV-11D12	CRDM Drive Rod Assembly D12	D	NS	Manufacturer Std.	
RXS-MV-11D12	CRDM Coil Stack Assembly D12	D	NS	Manufacturer Std.	
RXS-MV-11D12L	CRDM Latch Housing D12	A	I	ASME III-1	
RXS-MV-11D12R	CRDM Rod Travel Housing D12	A	I	ASME III-1	
RXS-MV-11E03	CRDM Latch Assembly E3	D	NS	Manufacturer Std.	
RXS-MV-11E03	CRDM Drive Rod Assembly E3	D	NS	Manufacturer Std.	
RXS-MV-11E03	CRDM Coil Stack Assembly E3	D	NS	Manufacturer Std.	
RXS-MV-11E03L	CRDM Latch Housing E3	A	I	ASME III-1	
RXS-MV-11E03R	CRDM Rod Travel Housing E3	A	I	ASME III-1	
RXS-MV-11E05	CRDM Latch Assembly E5	D	NS	Manufacturer Std.	
RXS-MV-11E05	CRDM Drive Rod Assembly E5	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 32 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11E05	CRDM Coil Stack Assembly E5	D	NS	Manufacturer Std.	
RXS-MV-11E05L	CRDM Latch Housing E5	A	I	ASME III-1	
RXS-MV-11E05R	CRDM Rod Travel Housing E5	A	I	ASME III-1	
RXS-MV-11E07	CRDM Latch Assembly E7	D	NS	Manufacturer Std.	
RXS-MV-11E07	CRDM Drive Rod Assembly E7	D	NS	Manufacturer Std.	
RXS-MV-11E07	CRDM Coil Stack Assembly E7	D	NS	Manufacturer Std.	
RXS-MV-11E07L	CRDM Latch Housing E7	A	I	ASME III-1	
RXS-MV-11E07R	CRDM Rod Travel Housing E7	A	I	ASME III-1	
RXS-MV-11E09	CRDM Latch Assembly E9	D	NS	Manufacturer Std.	
RXS-MV-11E09	CRDM Drive Rod Assembly E9	D	NS	Manufacturer Std.	
RXS-MV-11E09	CRDM Coil Stack Assembly E9	D	NS	Manufacturer Std.	
RXS-MV-11E09L	CRDM Latch Housing E9	A	I	ASME III-1	
RXS-MV-11E09R	CRDM Rod Travel Housing E9	A	I	ASME III-1	
RXS-MV-11E11	CRDM Latch Assembly E11	D	NS	Manufacturer Std.	
RXS-MV-11E11	CRDM Drive Rod Assembly E11	D	NS	Manufacturer Std.	
RXS-MV-11E11	CRDM Coil Stack Assembly E11	D	NS	Manufacturer Std.	
RXS-MV-11E11L	CRDM Latch Housing E11	A	I	ASME III-1	
RXS-MV-11E11R	CRDM Rod Travel Housing E11	A	I	ASME III-1	
RXS-MV-11E13	CRDM Latch Assembly E13	D	NS	Manufacturer Std.	
RXS-MV-11E13	CRDM Drive Rod Assembly E13	D	NS	Manufacturer Std.	
RXS-MV-11E13	CRDM Coil Stack Assembly E13	D	NS	Manufacturer Std.	
RXS-MV-11E13L	CRDM Latch Housing E13	A	I	ASME III-1	
RXS MV-11E13R	CRDM Rod Travel Housing E13	A	I	ASME III-1	
RXS-MV-11F02	CRDM Latch Assembly F2	D	NS	Manufacturer Std.	
RXS-MV-11F02	CRDM Drive Rod Assembly F2	D	NS	Manufacturer Std.	
RXS-MV-11F02	CRDM Coil Stack Assembly F2	D	NS	Manufacturer Std.	
RXS-MV-11F02L	CRDM Latch Housing F2	A	I	ASME III-1	
RXS-MV-11F02R	CRDM Rod Travel Housing F2	A	I	ASME III-1	

Table 3.2-3 (Sheet 33 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11F04	CRDM Latch Assembly F4	D	NS	Manufacturer Std.	
RXS-MV-11F04	CRDM Drive Rod Assembly F4	D	NS	Manufacturer Std.	
RXS-MV-11F04	CRDM Coil Stack Assembly F4	D	NS	Manufacturer Std.	
RXS-MV-11F04L	CRDM Latch Housing F4	A	I	ASME III-1	
RXS-MV-11F04R	CRDM Rod Travel Housing F4	A	I	ASME III-1	
RXS-MV-11F06	CRDM Latch Assembly F6	D	NS	Manufacturer Std.	
RXS-MV-11F06	CRDM Drive Rod Assembly F6	D	NS	Manufacturer Std.	
RXS-MV-11F06	CRDM Coil Stack Assembly F6	D	NS	Manufacturer Std.	
RXS-MV-11F06L	CRDM Latch Housing F6	A	I	ASME III-1	
RXS-MV-11F06R	CRDM Rod Travel Housing F6	A	I	ASME III-1	
RXS-MV-11F08	CRDM Latch Assembly F8	D	NS	Manufacturer Std.	
RXS-MV-11F08	CRDM Drive Rod Assembly F8	D	NS	Manufacturer Std.	
RXS-MV-11F08	CRDM Coil Stack Assembly F8	D	NS	Manufacturer Std.	
RXS-MV-11F08L	CRDM Latch Housing F8	A	I	ASME III-1	
RXS-MV-11F08R	CRDM Rod Travel Housing F8	A	I	ASME III-1	
RXS-MV-11F10	CRDM Latch Assembly F10	D	NS	Manufacturer Std.	
RXS-MV-11F10	CRDM Drive Rod Assembly F10	D	NS	Manufacturer Std.	
RXS-MV-11F10	CRDM Coil Stack Assembly F10	D	NS	Manufacturer Std.	
RXS-MV-11F10L	CRDM Latch Housing F10	A	I	ASME III-1	
RXS-MV-11F10R	CRDM Rod Travel Housing F10	A	I	ASME III-1	
RXS-MV-11F12	CRDM Latch Assembly F12	D	NS	Manufacturer Std.	
RXS-MV-11F12	CRDM Drive Rod Assembly F12	D	NS	Manufacturer Std.	
RXS-MV-11F12	CRDM Coil Stack Assembly F12	D	NS	Manufacturer Std.	
RXS-MV-11F12L	CRDM Latch Housing F12	A	I	ASME III-1	
RXS-MV-11F12R	CRDM Rod Travel Housing F12	A	I	ASME III-1	
RXS-MV-11F14	CRDM Latch Assembly F14	D	NS	Manufacturer Std.	
RXS-MV-11F14	CRDM Drive Rod Assembly F14	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 34 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11F14	CRDM Coil Stack Assembly F14	D	NS	Manufacturer Std.	
RXS-MV-11F14L	CRDM Latch Housing F14	A	I	ASME III-1	
RXS-MV-11F14R	CRDM Rod Travel Housing F14	A	I	ASME III-1	
RXS-MV-11G03	CRDM Latch Assembly G3	D	NS	Manufacturer Std.	
RXS-MV-11G03	CRDM Drive Rod Assembly G3	D	NS	Manufacturer Std.	
RXS-MV-11G03	CRDM Coil Stack Assembly G3	D	NS	Manufacturer Std.	
RXS-MV-11G03L	CRDM Latch Housing G3	A	I	ASME III-1	
RXS-MV-11G03R	CRDM Rod Travel Housing G3	A	I	ASME III-1	
RXS-MV-11G05	CRDM Latch Assembly G5	D	NS	Manufacturer Std.	
RXS-MV-11G05	CRDM Drive Rod Assembly G5	D	NS	Manufacturer Std.	
RXS-MV-11G05	CRDM Coil Stack Assembly G5	D	NS	Manufacturer Std.	
RXS-MV-11G05L	CRDM Latch Housing G5	A	I	ASME III-1	
RXS-MV-11G05R	CRDM Rod Travel Housing G5	A	I	ASME III-1	
RXS-MV-11G07	CRDM Latch Assembly G7	D	NS	Manufacturer Std.	
RXS-MV-11G07	CRDM Drive Rod Assembly G7	D	NS	Manufacturer Std.	
RXS-MV-11G07	CRDM Coil Stack Assembly G7	D	NS	Manufacturer Std.	
RXS-MV-11G07L	CRDM Latch Housing G7	A	I	ASME III-1	
RXS-MV-11G07R	CRDM Rod Travel Housing G7	A	I	ASME III-1	
RXS-MV-11G09	CRDM Latch Assembly G9	D	NS	Manufacturer Std.	
RXS-MV-11G09	CRDM Drive Rod Assembly G9	D	NS	Manufacturer Std.	
RXS-MV-11G09	CRDM Coil Stack Assembly G9	D	NS	Manufacturer Std.	
RXS-MV-11G09L	CRDM Latch Housing G9	A	I	ASME III-1	
RXS-MV-11G09R	CRDM Rod Travel Housing G9	A	I	ASME III-1	
RXS-MV-11G11	CRDM Latch Assembly G11	D	NS	Manufacturer Std.	
RXS-MV-11G11	CRDM Drive Rod Assembly G11	D	NS	Manufacturer Std.	
RXS-MV-11G11	CRDM Coil Stack Assembly G11	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 35 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11G11L	CRDM Latch Housing G11	A	I	ASME III-1	
RXS-MV-11G11R	CRDM Rod Travel Housing G11	A	I	ASME III-1	
RXS-MV-11G13	CRDM Latch Assembly G13	D	NS	Manufacturer Std.	
RXS-MV-11G13	CRDM Drive Rod Assembly G13	D	NS	Manufacturer Std.	
RXS-MV-11G13	CRDM Coil Stack Assembly G13	D	NS	Manufacturer Std.	
RXS-MV-11G13L	CRDM Latch Housing G13	A	I	ASME III-1	
RXS-MV-11G13R	CRDM Rod Travel Housing G13	A	I	ASME III-1	
RXS-MV-11H02	CRDM Latch Assembly H2	D	NS	Manufacturer Std.	
RXS-MV-11H02	CRDM Drive Rod Assembly H2	D	NS	Manufacturer Std.	
RXS-MV-11H02	CRDM Coil Stack Assembly H2	D	NS	Manufacturer Std.	
RXS-MV-11H02L	CRDM Latch Housing H2	A	I	ASME III-1	
RXS-MV-11H02R	CRDM Rod Travel Housing H2	A	I	ASME III-1	
RXS-MV-11H04	CRDM Latch Assembly H4	D	NS	Manufacturer Std.	
RXS-MV-11H04	CRDM Drive Rod Assembly H4	D	NS	Manufacturer Std.	
RXS-MV-11H04	CRDM Coil Stack Assembly H4	D	NS	Manufacturer Std.	
RXS-MV-11H04L	CRDM Latch Housing H4	A	I	ASME III-1	
RXS-MV-11H04R	CRDM Rod Travel Housing H4	A	I	ASME III-1	
RXS-MV-11H06	CRDM Latch Assembly H6	D	NS	Manufacturer Std.	
RXS-MV-11H06	CRDM Drive Rod Assembly H6	D	NS	Manufacturer Std.	
RXS-MV-11H06	CRDM Coil Stack Assembly H6	D	NS	Manufacturer Std.	
RXS-MV-11H06L	CRDM Latch Housing H6	A	I	ASME III-1	
RXS-MV-11H06R	CRDM Rod Travel Housing H6	A	I	ASME III-1	
RXS-MV-11H08	CRDM Latch Assembly H8	D	NS	Manufacturer Std.	
RXS-MV-11H08	CRDM Drive Rod Assembly H8	D	NS	Manufacturer Std.	
RXS-MV-11H08	CRDM Coil Stack Assembly H8	D	NS	Manufacturer Std.	
RXS-MV-11H08L	CRDM Latch Housing H8	A	I	ASME III-1	
RXS-MV-11H08R	CRDM Rod Travel Housing H8	A	I	ASME III-1	
RXS-MV-11H10	CRDM Latch Assembly H10	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 36 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11H10	CRDM Drive Rod Assembly H10	D	NS	Manufacturer Std.	
RXS-MV-11H10	CRDM Coil Stack Assembly H10	D	NS	Manufacturer Std.	
RXS-MV-11H10L	CRDM Latch Housing H10	A	I	ASME III-1	
RXS-MV-11H10R	CRDM Rod Travel Housing H10	A	I	ASME III-1	
RXS-MV-11H12	CRDM Latch Assembly H12	D	NS	Manufacturer Std.	
RXS-MV-11H12	CRDM Drive Rod Assembly H12	D	NS	Manufacturer Std.	
RXS-MV-11H12	CRDM Coil Stack Assembly H12	D	NS	Manufacturer Std.	
RXS-MV-11H12L	CRDM Latch Housing H12	A	I	ASME III-1	
RXS-MV-11H12R	CRDM Rod Travel Housing H12	A	I	ASME III-1	
RXS-MV-11H14	CRDM Latch Assembly H14	D	NS	Manufacturer Std.	
RXS-MV-11H14	CRDM Drive Rod Assembly H14	D	NS	Manufacturer Std.	
RXS-MV-11H14	CRDM Coil Stack Assembly H14	D	NS	Manufacturer Std.	
RXS-MV-11H14L	CRDM Latch Housing H14	A	I	ASME III-1	
RXS-MV-11H14R	CRDM Rod Travel Housing H14	A	I	ASME III-1	
RXS-MV-11J03	CRDM Latch Assembly J3	D	NS	Manufacturer Std.	
RXS-MV-11J03	CRDM Drive Rod Assembly J3	D	NS	Manufacturer Std.	
RXS-MV-11J03	CRDM Coil Stack Assembly J3	D	NS	Manufacturer Std.	
RXS-MV-11J03L	CRDM Latch Housing J3	A	I	ASME III-1	
RXS-MV-11J03R	CRDM Rod Travel Housing J3	A	I	ASME III-1	
RXS-MV-11J05	CRDM Latch Assembly J5	D	NS	Manufacturer Std.	
RXS-MV-11J05	CRDM Drive Rod Assembly J5	D	NS	Manufacturer Std.	
RXS-MV-11J05	CRDM Coil Stack Assembly J5	D	NS	Manufacturer Std.	
RXS-MV-11J05L	CRDM Latch Housing J5	A	I	ASME III-1	
RXS-MV-11J05R	CRDM Rod Travel Housing J5	A	I	ASME III-1	
RXS-MV-11J07	CRDM Latch Assembly J7	D	NS	Manufacturer Std.	
RXS-MV-11J07	CRDM Drive Rod Assembly J7	D	NS	Manufacturer Std.	
RXS-MV-11J07	CRDM Coil Stack Assembly J7	D	NS	Manufacturer Std.	
RXS-MV-11J07L	CRDM Latch Housing J7	A	I	ASME III-1	
RXS-MV-11J07R	CRDM Rod Travel Housing J7	A	I	ASME III-1	
RXS-MV-11J09	CRDM Latch Assembly J9	D	NS	Manufacturer Std.	
RXS-MV-11J09	CRDM Drive Rod Assembly J9	D	NS	Manufacturer Std.	
RXS-MV-11J09	CRDM Coil Stack Assembly J9	D	NS	Manufacturer Std.	
RXS-MV-11J09L	CRDM Latch Housing J9	A	I	ASME III-1	
RXS-MV-11J09R	CRDM Rod Travel Housing J9	A	I	ASME III-1	
RXS-MV-11J11	CRDM Latch Assembly J11	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 37 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11J11	CRDM Drive Rod Assembly J11	D	NS	Manufacturer Std.	
RXS-MV-11J11	CRDM Coil Stack Assembly J11	D	NS	Manufacturer Std.	
RXS-MV-11J11L	CRDM Latch Housing J11	A	I	ASME III-1	
RXS-MV-11J11R	CRDM Rod Travel Housing J11	A	I	ASME III-1	
RXS-MV-11J13	CRDM Latch Assembly J13	D	NS	Manufacturer Std.	
RXS-MV-11J13	CRDM Drive Rod Assembly J13	D	NS	Manufacturer Std.	
RXS-MV-11J13	CRDM Coil Stack Assembly J13	D	NS	Manufacturer Std.	
RXS-MV-11J13L	CRDM Latch Housing J13	A	I	ASME III-1	
RXS-MV-11J13R	CRDM Rod Travel Housing J13	A	I	ASME III-1	
RXS-MV-11K02	CRDM Latch Assembly K2	D	NS	Manufacturer Std.	
RXS-MV-11K02	CRDM Drive Rod Assembly K2	D	NS	Manufacturer Std.	
RXS-MV-11K02	CRDM Coil Stack Assembly K2	D	NS	Manufacturer Std.	
RXS-MV-11K02L	CRDM Latch Housing K2	A	I	ASME III-1	
RXS-MV-11K02R	CRDM Rod Travel Housing K2	A	I	ASME III-1	
RXS-MV-11K04	CRDM Latch Assembly K4	D	NS	Manufacturer Std.	
RXS-MV-11K04	CRDM Drive Rod Assembly K4	D	NS	Manufacturer Std.	
RXS-MV-11K04	CRDM Coil Stack Assembly K4	D	NS	Manufacturer Std.	
RXS-MV-11K04L	CRDM Latch Housing K4	A	I	ASME III-1	
RXS-MV-11K04R	CRDM Rod Travel Housing K4	A	I	ASME III-1	
RXS-MV-11K06	CRDM Latch Assembly K6	D	NS	Manufacturer Std.	
RXS-MV-11K06	CRDM Drive Rod Assembly K6	D	NS	Manufacturer Std.	
RXS-MV-11K06	CRDM Coil Stack Assembly K6	D	NS	Manufacturer Std.	
RXS-MV-11K06L	CRDM Latch Housing K6	A	I	ASME III-1	
RXS-MV-11K06R	CRDM Rod Travel Housing K6	A	I	ASME III-1	
RXS-MV-11K08	CRDM Latch Assembly K8	D	NS	Manufacturer Std.	
RXS-MV-11K08	CRDM Drive Rod Assembly K8	D	NS	Manufacturer Std.	
RXS-MV-11K08	CRDM Coil Stack Assembly K8	D	NS	Manufacturer Std.	
RXS-MV-11K08L	CRDM Latch Housing K8	A	I	ASME III-1	
RXS-MV-11K08R	CRDM Rod Travel Housing K8	A	I	ASME III-1	
RXS-MV-11K10	CRDM Latch Assembly K10	D	NS	Manufacturer Std.	
RXS-MV-11K10	CRDM Drive Rod Assembly K10	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 38 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11K10	CRDM Coil Stack Assembly K10	D	NS	Manufacturer Std.	
RXS-MV-11K10L	CRDM Latch Housing K10	A	I	ASME III-1	
RXS-MV-11K10R	CRDM Rod Travel Housing K10	A	I	ASME III-1	
RXS-MV-11K12	CRDM Latch Assembly K12	D	NS	Manufacturer Std.	
RXS-MV-11K12	CRDM Drive Rod Assembly K12	D	NS	Manufacturer Std.	
RXS-MV-11K12	CRDM Coil Stack Assembly K12	D	NS	Manufacturer Std.	
RXS-MV-11K12L	CRDM Latch Housing K12	A	I	ASME III-1	
RXS-MV-11K12R	CRDM Rod Travel Housing K12	A	I	ASME III-1	
RXS-MV-11K14	CRDM Latch Assembly K14	D	NS	Manufacturer Std.	
RXS-MV-11K14	CRDM Drive Rod Assembly K14	D	NS	Manufacturer Std.	
RXS-MV-11K14	CRDM Coil Stack Assembly K14	D	NS	Manufacturer Std.	
RXS-MV-11K14L	CRDM Latch Housing K14	A	I	ASME III-1	
RXS-MV-11K14R	CRDM Rod Travel Housing K14	A	I	ASME III-1	
RXS-MV-11L03	CRDM Latch Assembly L3	D	NS	Manufacturer Std.	
RXS-MV-11L03	CRDM Drive Rod Assembly L3	D	NS	Manufacturer Std.	
RXS-MV-11L03	CRDM Coil Stack Assembly L3	D	NS	Manufacturer Std.	
RXS-MV-11L03L	CRDM Latch Housing L3	A	I	ASME III-1	
RXS-MV-11L03R	CRDM Rod Travel Housing L3	A	I	ASME III-1	
RXS-MV-11L05	CRDM Latch Assembly L5	D	NS	Manufacturer Std.	
RXS-MV-11L05	CRDM Drive Rod Assembly L5	D	NS	Manufacturer Std.	
RXS-MV-11L05	CRDM Coil Stack Assembly L5	D	NS	Manufacturer Std.	
RXS-MV-11L05L	CRDM Latch Housing L5	A	I	ASME III-1	
RXS-MV-11L05R	CRDM Rod Travel Housing L5	A	I	ASME III-1	
RXS-MV-11L07	CRDM Latch Assembly L7	D	NS	Manufacturer Std.	
RXS-MV-11L07	CRDM Drive Rod Assembly L7	D	NS	Manufacturer Std.	
RXS-MV-11L07	CRDM Coil Stack Assembly L7	D	NS	Manufacturer Std.	
RXS-MV-11L07L	CRDM Latch Housing L7	A	I	ASME III-1	
RXS-MV-11L07R	CRDM Rod Travel Housing L7	A	I	ASME III-1	
RXS-MV-11L09	CRDM Latch Assembly L9	D	NS	Manufacturer Std.	
RXS-MV-11L09	CRDM Drive Rod Assembly L9	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 39 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11L09	CRDM Coil Stack Assembly L9	D	NS	Manufacturer Std.	
RXS-MV-11L09L	CRDM Latch Housing L9	A	I	ASME III-1	
RXS-MV-11L09R	CRDM Rod Travel Housing L9	A	I	ASME III-1	
RXS-MV-11L11	CRDM Latch Assembly L11	D	NS	Manufacturer Std.	
RXS-MV-11L11	CRDM Drive Rod Assembly L11	D	NS	Manufacturer Std.	
RXS-MV-11L11	CRDM Coil Stack Assembly L11	D	NS	Manufacturer Std.	
RXS-MV-11L11L	CRDM Latch Housing L11	A	I	ASME III-1	
RXS-MV-11L11R	CRDM Rod Travel Housing L11	A	I	ASME III-1	
RXS-MV-11L13	CRDM Latch Assembly L13	D	NS	Manufacturer Std.	
RXS-MV-11L13	CRDM Drive Rod Assembly L13	D	NS	Manufacturer Std.	
RXS-MV-11L13	CRDM Coil Stack Assembly L13	D	NS	Manufacturer Std.	
RXS-MV-11L13L	CRDM Latch Housing L13	A	I	ASME III-1	
RXS-MV-11L13R	CRDM Rod Travel Housing L13	A	I	ASME III-1	
RXS-MV-11M04	CRDM Latch Assembly M4	D	NS	Manufacturer Std.	
RXS-MV-11M04	CRDM Drive Rod Assembly M4	D	NS	Manufacturer Std.	
RXS-MV-11M04	CRDM Coil Stack Assembly M4	D	NS	Manufacturer Std.	
RXS-MV-11M04L	CRDM Latch Housing M4	A	I	ASME III-1	
RXS-MV-11M04R	CRDM Rod Travel Housing M4	A	I	ASME III-1	
RXS-MV-11M06	CRDM Latch Assembly M6	D	NS	Manufacturer Std.	
RXS-MV-11M06	CRDM Drive Rod Assembly M6	D	NS	Manufacturer Std.	
RXS-MV-11M06	CRDM Coil Stack Assembly M6	D	NS	Manufacturer Std.	
RXS-MV-11M06L	CRDM Latch Housing M6	A	I	ASME III-1	
RXS-MV-11M06R	CRDM Rod Travel Housing M6	A	I	ASME III-1	
RXS-MV-11M08	CRDM Latch Assembly M8	D	NS	Manufacturer Std.	
RXS-MV-11M08	CRDM Drive Rod Assembly M8	D	NS	Manufacturer Std.	
RXS-MV-11M08	CRDM Coil Stack Assembly M8	D	NS	Manufacturer Std.	
RXS-MV-11M08L	CRDM Latch Housing M8	A	I	ASME III-1	
RXS-MV-11M08R	CRDM Rod Travel Housing M8	A	I	ASME III-1	
RXS-MV-11M10	CRDM Latch Assembly M10	D	NS	Manufacturer Std.	
RXS-MV-11M10	CRDM Drive Rod Assembly M10	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 40 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11M10	CRDM Coil Stack Assembly M10	D	NS	Manufacturer Std.	
RXS-MV-11M10L	CRDM Latch Housing M10	A	I	ASME III-1	
RXS-MV-11M10R	CRDM Rod Travel Housing M10	A	I	ASME III-1	
RXS-MV-11M12	CRDM Latch Assembly M12	D	NS	Manufacturer Std.	
RXS-MV-11M12	CRDM Drive Rod Assembly M12	D	NS	Manufacturer Std.	
RXS-MV-11M12	CRDM Coil Stack Assembly M12	D	NS	Manufacturer Std.	
RXS-MV-11M12L	CRDM Latch Housing M12	A	I	ASME III-1	
RXS-MV-11M12R	CRDM Rod Travel Housing M12	A	I	ASME III-1	
RRXS-MV-11N05	CRDM Latch Assembly N5	D	NS	Manufacturer Std.	
RXS-MV-11N05	CRDM Drive Rod Assembly N5	D	NS	Manufacturer Std.	
RXS-MV-11N05	CRDM Coil Stack Assembly N5	D	NS	Manufacturer Std.	
RXS-MV-11N05L	CRDM Latch Housing N5	A	I	ASME III-1	
RXS-MV-11N05R	CRDM Rod Travel Housing N5	A	I	ASME III-1	
RXS-MV-11N07	CRDM Latch Assembly N7	D	NS	Manufacturer Std.	
RXS-MV-11N07	CRDM Drive Rod Assembly N7	D	NS	Manufacturer Std.	
RXS-MV-11N07	CRDM Coil Stack Assembly N7	D	NS	Manufacturer Std.	
RXS-MV-11N07L	CRDM Latch Housing N7	A	I	ASME III-1	
RXS-MV-11N07R	CRDM Rod Travel Housing N7	A	I	ASME III-1	
RXS-MV-11N09	CRDM Latch Assembly N9	D	NS	Manufacturer Std.	
RXS-MV-11N09	CRDM Drive Rod Assembly N9	D	NS	Manufacturer Std.	
RXS-MV-11N09	CRDM Coil Stack Assembly N9	D	NS	Manufacturer Std.	
RXS-MV-11N09L	CRDM Latch Housing N9	A	I	ASME III-1	
RXS-MV-11N09R	CRDM Rod Travel Housing N9	A	I	ASME III-1	
RXS-MV-11N11	CRDM Latch Assembly N11	D	NS	Manufacturer Std.	
RXS-MV-11N11	CRDM Drive Rod Assembly N11	D	NS	Manufacturer Std.	
RXS-MV-11N11	CRDM Coil Stack Assembly N11	D	NS	Manufacturer Std.	
RXS-MV-11N11L	CRDM Latch Housing N11	A	I	ASME III-1	
RXS-MV-11N11R	CRDM Rod Travel Housing N11	A	I	ASME III-1	
RXS-MV-11P06	CRDM Latch Assembly P6	D	NS	Manufacturer Std.	
RXS-MV-11P06	CRDM Drive Rod Assembly P6	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 41 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Reactor System (Continued)					
RXS-MV-11P06	CRDM Coil Stack Assembly P6	D	NS	Manufacturer Std.	
RXS-MV-11P06L	CRDM Latch Housing P6	A	I	ASME III-1	
RXS-MV-11P06R	CRDM Rod Travel Housing P6	A	I	ASME III-1	
RXS-MV-11P08	CRDM Latch Assembly P8	D	NS	Manufacturer Std.	
RXS-MV-11P08	CRDM Drive Rod Assembly P8	D	NS	Manufacturer Std.	
RXS-MV-11P08	CRDM Coil Stack Assembly P8	D	NS	Manufacturer Std.	
RXS-MV-11P08L	CRDM Latch Housing P8	A	I	ASME III-1	
RXS-MV-11P08R	CRDM Rod Travel Housing P8	A	I	ASME III-1	
RXS-MV-11P10	CRDM Latch Assembly P10	D	NS	Manufacturer Std.	
RXS-MV-11P10	CRDM Drive Rod Assembly P10	D	NS	Manufacturer Std.	
RXS-MV-11P10	CRDM Coil Stack Assembly P10	D	NS	Manufacturer Std.	
RXS-MV-11P10L	CRDM Latch Housing P10	A	I	ASME III-1	
RXS-MV-11P10R	CRDM Rod Travel Housing P10	A	I	ASME III-1	
RXS-MY-Y01	IHP Lower Shroud Assembly	C	I	ASME-NF	
Balance of system components are Class E					
Sanitary Drainage System (SDS)				Location: Various	
SDS-PL-V001	SDS MCR Isolation Valve	C	I	ASME III-3	
SDS-PL-V002	SDS MCR Isolation Valve	C	I	ASME III-3	
Balance of system components are Class D, P, and W					
Spent Fuel Pool Cooling System (SFS)			Location: Auxiliary Building, Containment		
n/a	Heat Exchangers, SFS and CCS Side	D	NS	ASME VIII	
n/a	Pumps	D	NS	Hydraulic Institute Std.	
n/a	Demineralizers	D	NS	ASME VIII	
n/a	Filters	D	NS	ASME VIII	
n/a	Valves Providing SFS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
SFS-PL-V024A	Spent Fuel Pool Level Instrument Isolation	C	I	ASME III-3	
SFS-PL-V024B	Spent Fuel Pool Level Instrument Isolation	C	I	ASME III-3	
SFS-PL-V024C	Spent Fuel Pool Level Instrument Isolation	C	I	ASME III-3	
SFS-PL-V028	Cask Washdown Pit Level Instrument Isolation	C	I	ASME III-3	
SFS-PL-V031	SFS Refueling Cavity Drain to SGS Compartment Isolation	C	I	ASME III-3	

Table 3.2-3 (Sheet 42 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Spent Fuel Pool Cooling System (Continued)					
SFS-PL-V032	SFS Refueling Cavity Suction Isolation	C	I	ASME III-3	
SFS-PL-V033	SFS Refueling Cavity Drain to Containment Sump Isolation	C	I	ASME III-3	
SFS-PL-V034	SFS Suction Line Containment Isolation	B	I	ASME III-2	
SFS-PL-V035	SFS Suction Line Containment Isolation	B	I	ASME III-2	
SFS-PL-V037	SFS Discharge Line Containment Isolation	B	I	ASME III-2	
SFS-PL-V038	SFS Discharge Line Containment Isolation	B	I	ASME III-2	
SFS-PL-V039	SFS Suction Line from IRWST Isolation	C	I	ASME III-3	
SFS-PL-V040	SFS Fuel Transfer Canal Drain Isolation	C	I	ASME III-3	
SFS-PL-V041	SFS Cask Loading Pit Drain Isolation	C	I	ASME III-3	
SFS-PL-V042	Cask Loading Pit to Pump Suction Isolation	C	I	ASME III-3	
SFS-PL-V043	SFS CVS Makeup Reverse Flow Prevention	C	I	ASME III-3	
SFS-PL-V045	SFS Discharge to Cask Loading Pit Isolation	C	I	ASME III-3	
SFS-PL-V047	SFS Demineralized Water Makeup to SFP Reverse Flow Prevent	D	NS	ANSI 16.34	
SFS-PL-V048	SFS Containment Penetration Test Connection	B	I	ASME III-2	
SFS-PL-V049	SFS Cask Loading Pit Drain to WLS Isolation	C	I	ASME III-3	
SFS-PL-V056	SFS Containment Penetration Test Connection Isolation	B	I	ASME III-2	
SFS-PL-V058	SFS Containment Isolation Valve V034 Test	C	I	ASME III-3	
SFS-PL-V066	Spent Fuel Pool to Cask Washdown Pit Isolation	C	I	ASME III-3	
SFS-PL-V067	SFS Containment Isolation Relief Valve	B	I	ASME III-2	
SFS-PL-V068	Cask Washdown Pit Drain Isolation	C	I	ASME III-3	
SFS-PL-V071	Refueling Cavity Overflow to SG Compartment	C	I	ASME III-3	
SFS-PL-V072	Refueling Cavity Overflow to SG Compartment	C	I	ASME III-3	
SFS-PL-V075	SFS Containment Floodup Isolation Valve	C	I	ASME III-3	

**Table 3.2-3 (Sheet 43 of 58)
AP1000 Classification of Mechanical and
Fluid Systems, Components, and Equipment**

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Spent Fuel Pool Cooling System (Continued)					
SFS-PY-C01	Spent Fuel Cooling Pump Discharge to IRWST	B	I	ASME III, MC	
SFS-PY-C02	Spent Fuel Cooling Pump Suction from IRWST	B	I	ASME III, MC	
Balance of system components are Class D					
Steam Generator System (SGS)			Location: Containment and Auxiliary Building		
SGS-MY-Y01A	Steam Generator A PORV Silencer	D	NS	Manufacturer Std.	
SGS-MY-Y01B	Steam Generator B PORV Silencer	D	NS	Manufacturer Std.	
SGS-PL-V001A	LT001 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V001B	LT005 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V002A	LT001 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V002B	LT005 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V003A	LT002 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V003B	LT006 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V004A	LT002 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V004B	LT006 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V005A	LT003 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V005B	LT007 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V006A	LT003 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V006B	LT007 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V007A	LT004 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V007B	LT008 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V008A	LT004 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V008B	LT008 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V010A	LT011 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V010B	LT013 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V011A	LT011 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V011B	LT013 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V012A	LT012 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V012B	LT014 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V013A	LT012 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V013B	LT014 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V014A	PORV Discharge Condensate Drain Isolation	D	NS	ANSI B31.1	
SGS-PL-V014B	PORV Discharge Condensate Drain Isolation	D	NS	ANSI B31.1	
SGS-PL-V015A	FT021 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V015B	FT023 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V016A	FT020 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V016B	FT022 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V017A	FT021 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V017B	FT023 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V018A	FT020 Root Isolation Valve	B	I	ASME III-2	

Table 3.2-3 (Sheet 44 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V018B	FT022 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V019A	Main Steam Line Vent Isolation	B	I	ASME III-2	
SGS-PL-V019B	Main Steam Line Vent Isolation	B	I	ASME III-2	
SGS-PL-V022A	PT030 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V022B	PT034 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V023A	PT031 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V023B	PT035 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V024A	PT032 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V024B	PT036 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V025A	PT033 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V025B	PT037 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V027A	PORV Block Valve SG 01	B	I	ASME III-2	
SGS-PL-V027B	PORV Block Valve SG 02	B	I	ASME III-2	
SGS-PL-V030A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V030B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V031A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V031B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V032A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V032B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V033A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V033B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V034A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V034B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V035A	Main Steam Safety Valve SG 01	B	I	ASME III-2	
SGS-PL-V035B	Main Steam Safety Valve SG 02	B	I	ASME III-2	
SGS-PL-V036A	Steam Line Condensate Drain Isolation	B	I	ASME III-2	
SGS-PL-V036B	Steam Line Condensate Drain Isolation	B	I	ASME III-2	
SGS-PL-V038A	Steam Line #1 Nitrogen Supply Isolation	B	I	ASME III-2	
SGS-PL-V038B	Steam Line #2 Nitrogen Supply Isolation	B	I	ASME III-2	

Table 3.2-3 (Sheet 45 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V040A	Main Steam Line Isolation	B	I	ASME III-2	
SGS-PL-V040B	Main Steam Line Isolation	B	I	ASME III-2	
SGS-PL-V042A	MSIV Bypass Control Isolation	B	I	ASME III-2	
SGS-PL-V042B	MSIV Bypass Control Isolation	B	I	ASME III-2	
SGS-PL-V043A	MSIV Bypass Control Isolation	C	I	ASME III-3	
SGS-PL-V043B	MSIV Bypass Control Isolation	C	I	ASME III-3	
SGS-PL-V045A	SG 1 Condensate Pipe Drain Valve	B	I	ASME III-2	
SGS-PL-V045B	SG 2 Condensate Pipe Drain Valve	B	I	ASME III-2	
SGS-PL-V046A	LT015 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V046B	LT017 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V047A	LT015 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V047B	LT017 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V048A	LT016 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V048B	LT018 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V049A	LT016 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V049B	LT018 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V050A	LT044 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V050B	LT046 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V051A	LT044 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V051B	LT046 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V052A	LT045 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V052B	LT047 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V053A	LT045 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V053B	LT047 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V056A	PT062 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V056B	PT063 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V057A	Main Feedwater Isolation	B	I	ASME III-2	
SGS-PL-V057B	Main Feedwater Isolation	B	I	ASME III-2	
SGS-PL-V058A	Main Feedwater Check	B	I	ASME III-2	
SGS-PL-V058B	Main Feedwater Check	B	I	ASME III-2	
SGS-PL-V062A	FT055A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V062B	FT056A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V063A	FT055A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V063B	FT056A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V064A	FT055A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V064B	FT056A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V065A	FT055A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V065B	FT056A Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V067A	Startup Feedwater Isolation	B	I	ASME III-2	
SGS-PL-V067B	Startup Feedwater Isolation	B	I	ASME III-2	
SGS-PL-V074A	SG Blowdown Isolation	B	I	ASME III-2	
SGS-PL-V074B	SG Blowdown Isolation	B	I	ASME III-2	

Table 3.2-3 (Sheet 46 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PL-V075A	SG Series Blowdown Isolation	C	I	ASME III-3	
SGS-PL-V075B	SG Series Blowdown Isolation	C	I	ASME III-3	
SGS-PL-V084A	SG 1 Nitrogen Sparging Isolation	B	I	ASME III-2	
SGS-PL-V084B	SG 2 Nitrogen Sparging Isolation	B	I	ASME III-2	
SGS-PL-V086A	Steam Line Condensate Drain Control	C	I	ASME III-3	
SGS-PL-V086B	Steam Line Condensate Drain Control	C	I	ASME III-3	
SGS-PL-V093A	Orifice Isolation Valve	C	I	ASME III-3	
SGS-PL-V093B	Orifice Isolation Valve	C	I	ASME III-3	
SGS-PL-V094A	Orifice Cleanout Line Isolation Valve	C	I	ASME III-3	
SGS-PL-V094B	Orifice Cleanout Line Isolation Valve	C	I	ASME III-3	
SGS-PL-V095A	Orifice Isolation Valve	C	I	ASME III-3	
SGS-PL-V095B	Orifice Isolation Valve	C	I	ASME III-3	
SGS-PL-V096A	Steam Line Condensate Drain Level Isolation Valve	B	I	ASME III-2	
SGS-PL-V096B	Steam Line Condensate Drain Level Isolation Valve	B	I	ASME III-2	
SGS-PL-V097A	Steam Line Condensate Drain Level Isolation Valve	B	I	ASME III-2	
SGS-PL-V097B	Steam Line Condensate Drain Level Isolation Valve	B	I	ASME III-2	
SGS-PL-V233A	Power Operated Relief Valve	C	I	ASME III-3	
SGS-PL-V233B	Power Operated Relief Valve	C	I	ASME III-3	
SGS-PL-V240A	MSIV Bypass Isolation	B	I	ASME III-2	
SGS-PL-V240B	MSIV Bypass Isolation	B	I	ASME III-2	
SGS-PL-V250A	Main Feedwater Control	C	I	ASME III-3	
SGS-PL-V250B	Main Feedwater Control	C	I	ASME III-3	
SGS-PL-V255A	Startup Feedwater Control	C	I	ASME III-3	
SGS-PL-V255B	Startup Feedwater Control	C	I	ASME III-3	
SGS-PL-V256A	Startup Feedwater Check Valve	C	I	ASME III-3	
SGS-PL-V256B	Startup Feedwater Check Valve	C	I	ASME III-3	
SGS-PY-C01A	Main Steam Line A Penetration	B	I	ASME III, MC	
SGS-PY-C01B	Main Steam Line B Penetration	B	I	ASME III, MC	
SGS-PY-C02A	Main Feedwater Line A Penetration	B	I	ASME III, MC	

Table 3.2-3 (Sheet 47 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Steam Generator System (Continued)					
SGS-PY-C02B	Main Feedwater Line B Penetration	B	I	ASME III, MC	
SGS-PY-C03A	Steam Generator A Blowdown Line Penetration	B	I	ASME III, MC	
SGS-PY-C03B	Steam Generator B Blowdown Line Penetration				
SGS-PY-C05A	Startup Feedwater Line A Penetration	B	I	ASME III, MC	
SGS-PY-C05B	Startup Feedwater Line B Penetration	B	I	ASME III, MC	
Secondary Sampling System (SSS)				Location: Turbine Building	
System components are Class E					
Service Water System (SWS)				Location: Turbine Building and Yard	
n/a	Service Water Cooling Tower Fans	D	NS	Manufacturer Std.	
n/a	Service Water Cooling Tower	D	NS	Manufacturer Std.	
n/a	Service Water Pumps	D	NS	Hydraulic Institute Std.	
n/a	Valves Providing SWS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
Turbine Building Closed Cooling Water System (TCS)				Location: Turbine Building	
System components are Class E					
Turbine Island Vents, Drains and Relief System (TDS)				Location: Turbine Building	
n/a	Piping and components that provide the path from the GSS and CMS to atmosphere and rad monitor	D	NS	ANSI B31.1	
Balance of system components are Class E					
Main Turbine Control and Diagnostic System (TOS)				Location: Turbine Building	
System components are Class E					
Radiologically Controlled Area Ventilation System (VAS)			Location: Auxiliary Building and Annex Building		
n/a	CVS and RNS Pump Room Coolers	Note 2	NS	Manufacturer Std.	
n/a	Valves Providing VAS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500	
n/a	Fire Dampers	Note 3	NS	UL-555	
n/a	Air Handling Units	L	NS	Manufacturer Std.	
n/a	Filters	L	NS	UL 900	
n/a	Fans, Ductwork	L	NS	SMACNA	
n/a	Ductwork in Auxiliary Building except ductwork attached to mechanical modules	L	II	SMACNA	
Balance of system components are Class L					

Table 3.2-3 (Sheet 48 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Nuclear Island Nonradioactive Ventilation System (VBS)			Location: Auxiliary Building and Annex Building		
n/a	Battery Rooms Exhaust Fans	Note 2	NS	AMCA	
n/a	PCS Room Heaters	Note 2	NS	Manufacturer Std.	
n/a	Fire Dampers	Note 3	NS	UL-555S	
n/a	Dampers Providing AP1000 Equipment Class D Function	Note 2	NS	ANSI/AMCA-500	
n/a	Dampers in lines isolating radioactive contamination	R	NS	ASME-509	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500	
VBS-MP-01A	Sample Pump A	C	I	Manufacturer Std.	
Nuclear Island Nonradioactive Ventilation System (VBS) (Continued)					
VBS-MP-01B	Sample Pump B	C	I	Manufacturer Std.	
n/a	MCR/CSA Supplemental Air Filtration Units	Note 2	NS	ASME AG-1, Note 4	
VBS-PL-V186	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V187	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V188	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V189	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V190	MCR Isolation Valve	C	I	ASME III-3	
VBS-PL-V191	MCR Isolation Valve	C	I	ASME III-3	
n/a	Valves Providing VBS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
n/a	Other Air Handling Units	Note 2	NS	Manufacturer Std.	
n/a	Filters	Note 2	NS	UL 900	
n/a	Fans, Ductwork	Note 2, L or R	NS	SMACNA	
VBS-MA-10A	Ancillary Fan	D	NS	ANSI/AMCA 210, 211, 300	Equipment Anchorage is Seismic Category II
VBS-MA-10B	Ancillary Fan	D	NS	ANSI/AMCA 210, 211, 300	Equipment Anchorage is Seismic Category II
VBS-MA-11	Ancillary Fan	D	NS	ANSI/AMCA 210, 211, 300	Equipment Anchorage is Seismic Category II
VBS-MA-12	Ancillary Fan	D	NS	ANSI/AMCA 210, 211, 300	Equipment Anchorage is Seismic Category II
n/a	Ductwork in Auxiliary Building except ductwork attached to mechanical modules	L	II	SMACNA	
Balance of system components are Class L					
Containment Recirculation Cooling System (VCS)				Location: Containment	
n/a	Dampers	L	NS	ANSI/AMCA-500	
n/a	Fan Coil Units	L	NS	Manufacturer Std.	
n/a	Fans, Ductwork	L	NS	SMACNA	
Balance of system components are Class L					
Main Control Room Emergency Habitability System (VES)				Location: Auxiliary Building	
VES-MD-D001A	Relief Damper	Note 1	I	ASME 509/510	
VES-MD-D001B	Relief Damper	Note 1	I	ASME 509/510	

Table 3.2-3 (Sheet 49 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Main Control Room Emergency Habitability System (Continued)					
VES-MT-01	Emergency Air Storage Tank 01	C	I	ASME VIII, Appendix 22	
VES-MT-02	Emergency Air Storage Tank 02	C	I	ASME VIII, Appendix 22	
VES-MT-03	Emergency Air Storage Tank 03	C	I	ASME VIII, Appendix 22	
VES-MT-04	Emergency Air Storage Tank 04	C	I	ASME VIII, Appendix 22	
VES-MT-05	Emergency Air Storage Tank 05	C	I	ASME VIII, Appendix 22	
VES-MT-06	Emergency Air Storage Tank 06	C	I	ASME VIII, Appendix 22	
VES-MT-07	Emergency Air Storage Tank 07	C	I	ASME VIII, Appendix 22	
VES-MT-08	Emergency Air Storage Tank 08	C	I	ASME VIII, Appendix 22	
VES-MT-09	Emergency Air Storage Tank 09	C	I	ASME VIII, Appendix 22	
VES-MT-10	Emergency Air Storage Tank 10	C	I	ASME VIII, Appendix 22	
VES-MT-11	Emergency Air Storage Tank 11	C	I	ASME VIII, Appendix 22	
VES-MT-12	Emergency Air Storage Tank 12	C	I	ASME VIII, Appendix 22	
VES-MT-13	Emergency Air Storage Tank 13	C	I	ASME VIII, Appendix 22	
VES-MT-14	Emergency Air Storage Tank 14	C	I	ASME VIII, Appendix 22	
VES-MT-15	Emergency Air Storage Tank 15	C	I	ASME VIII, Appendix 22	
VES-MT-16	Emergency Air Storage Tank 16	C	I	ASME VIII, Appendix 22	
VES-MT-17	Emergency Air Storage Tank 17	C	I	ASME VIII, Appendix 22	
VES-MT-18	Emergency Air Storage Tank 18	C	I	ASME VIII, Appendix 22	
VES-MT-19	Emergency Air Storage Tank 19	C	I	ASME VIII, Appendix 22	
VES-MT-20	Emergency Air Storage Tank 20	C	I	ASME VIII, Appendix 22	
VES-MT-21	Emergency Air Storage Tank 21	C	I	ASME VIII, Appendix 22	
VES-MT-22	Emergency Air Storage Tank 22	C	I	ASME VIII, Appendix 22	
VES-MT-23	Emergency Air Storage Tank 23	C	I	ASME VIII, Appendix 22	
VES-MT-24	Emergency Air Storage Tank 24	C	I	ASME VIII, Appendix 22	

Table 3.2-3 (Sheet 50 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Main Control Room Emergency Habitability System (Continued)					
VES-MT-25	Emergency Air Storage Tank 25	C	I	ASME VIII, Appendix 22	
VES-MT-26	Emergency Air Storage Tank 26	C	I	ASME VIII, Appendix 22	
VES-MT-27	Emergency Air Storage Tank 27	C	I	ASME VIII, Appendix 22	
VES-MT-28	Emergency Air Storage Tank 28	C	I	ASME VIII, Appendix 22	
VES-MT-29	Emergency Air Storage Tank 29	C	I	ASME VIII, Appendix 22	
VES-MT-30	Emergency Air Storage Tank 30	C	I	ASME VIII, Appendix 22	
VES-MT-31	Emergency Air Storage Tank 31	C	I	ASME VIII, Appendix 22	
VES-MT-32	Emergency Air Storage Tank 32	C	I	ASME VIII, Appendix 22	
VES-PL-V001	Air Delivery Alternate Isolation Valve	C	I	ASME III-3	
VES-PL-V002A	Pressure Regulating Valve A	C	I	ASME III-3	
VES-PL-V002B	Pressure Regulating Valve B	C	I	ASME III-3	
VES-PL-V005A	Air Delivery Main Isolation Valve A	C	I	ASME III-3	
VES-PL-V005B	Air Delivery Main Isolation Valve B	C	I	ASME III-3	
VES-PL-V006A	Air Delivery Line Pressure Instrument Isolation Valve A	C	I	ASME III-3	
VES-PL-V006B	Air Delivery Line Pressure Instrument Isolation Valve B	C	I	ASME III-3	
VES-PL-V010A	Air Delivery Line Maintenance Isolation Valve A	C	I	ASME III-3	
VES-PL-V010B	Air Delivery Line Maintenance Isolation Valve B	C	I	ASME III-3	
VES-PL-V011A	Air Delivery Line Maintenance Isolation Valve A	C	I	ASME III-3	
VES-PL-V011B	Air Delivery Line Maintenance Isolation Valve B	C	I	ASME III-3	
VES-PL-V016	Temporary Instrument Isolation Valve A	C	I	ASME III-3	
VES-PL-V018	Temporary Instrument Isolation Valve A	C	I	ASME III-3	
VES-PL-V019	Temporary Instrument Isolation Valve B	C	I	ASME III-3	
VES-PL-V020	Temporary Instrument Isolation Valve B	C	I	ASME III-3	
VES-PL-V022A	Pressure Relief Isolation Valve A	C	I	ASME III-3	

Table 3.2-3 (Sheet 51 of 58)
AP1000 Classification of Mechanical and
Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Main Control Room Emergency Habitability System (Continued)					
VES-PL-V022B	Pressure Relief Isolation Valve B	C	I	ASME III-3	
VES-PL-V024A	Air Bank 1 Isolation Valve A	C	I	ASME III-3	
VES-PL-V024B	Air Bank 2 Isolation Valve B	C	I	ASME III-3	
VES-PL-V024C	Air Bank 3 Isolation Valve C	C	I	ASME III-3	
VES-PL-V024D	Air Bank 4 Isolation Valve D	C	I	ASME III-3	
VES-PL-V025A	Air Bank 1 Isolation Valve A	C	I	ASME III-3	
VES-PL-V025B	Air Bank 2 Isolation Valve B	C	I	ASME III-3	
VES-PL-V025C	Air Bank 3 Isolation Valve C	C	I	ASME III-3	
VES-PL-V025D	Air Bank 4 Isolation Valve D	C	I	ASME III-3	
VES-PL-V026A	Air Bank 1 Fill/Vent Isolation Valve A	C	I	ASME III-3	
VES-PL-V026B	Air Bank 2 Fill/Vent Isolation Valve B	C	I	ASME III-3	
VES-PL-V026C	Air Bank 3 Fill/Vent Isolation Valve C	C	I	ASME III-3	
VES-PL-V026D	Air Bank 4 Fill/Vent Isolation Valve D	C	I	ASME III-3	
VES-PL-V040A	Air Tank Safety Relief Valve A	C	I	ASME III-3	
VES-PL-V040B	Air Tank Safety Relief Valve B	C	I	ASME III-3	
VES-PL-V040C	Air Tank Safety Relief Valve C	C	I	ASME III-3	
VES-PL-V040D	Air Tank Safety Relief Valve D	C	I	ASME III-3	
VES-PL-V043A	Differential Pressure Instrument Line Isolation Valve A	C	I	ASME III-3	
VES-PL-V043B	Differential Pressure Instrument Line Isolation Valve B	C	I	ASME III-3	
VES-PL-V044	Main Air Flowpath Isolation Valve	C	I	ASME III-3	
VES-PL-V045	Eductor Flow Path Isolation Valve	C	I	ASME III-3	
VES-PL-V046	Eductor Bypass Isolation Valve	C	I	ASME III-3	
VES-PY-N01	MCR Air Filtration Line Eductor	C	I	ASME III-3	
VES-MY-F01	MCR Air Filtration Line Charcoal Filter	Note 1	I	ASME AG-1 Section FD	
VES-MY-F02	MCR Air Filtration Line HEPA Filter	Note 1	I	ASME AG-1 Section FC	
VES-MY-F03	MCR Air Filtration Line Postfilter	Note 1	1	ASME AG-1	
VES-MD-D001A	MCR Gravity Relief Dampers	Note 1	I	ASME AG-1	
VES-MD-D001B	MCR Gravity Relief Dampers	Note 1	I	ASME AG-1	
VES-MD-D002	MCR Air Filtration Line Supply Damper	Note 1	I	ASME AG-1 Section DA	

Table 3.2-3 (Sheet 52 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Main Control Room Emergency Habitability System (Continued)					
VES-MD-D003	MCR Air Filtration Line Supply Damper	Note 1	I	ASME AG-1 Section DA	
VES-MY-Y01	MCR Air Filtration Line Silencer	Note 1	I	ASME AG-1 Section SA	
VES-MY-Y02	MCR Air Filtration Line Silencer	Note 1	I	ASME AG-1 Section SA	
Containment Air Filtration System (VFS)			Location: Auxiliary Building and Annex Building		
VFS-PY-C01	Containment Supply Duct Penetration	B	I	ASME III, 2	
VFS-PY-C02	Containment Exhaust Duct Penetration	B	I	ASME III, 2	
VFS-MY-Y01	Containment Air Supply Debris Screen	C	I	ASME Sec. III Class 3	
VFS-MY-Y02	Containment Air Exhaust Debris Screen	C	I	ASME Sec. III Class 3	
VFS-PL-V003	Containment Purge Supply Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V004	Containment Purge Supply Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V008	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V009	Containment Purge Discharge Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V010	Containment Purge Discharge Containment Isolation Valve	B	I	ASME III-2	
VFS-PL-V012	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V015	Containment Isolation Test Connection	B	I	ASME III-2	
VFS-PL-V800A	Vacuum Relief Containment Isolation A – ORC	B	I	ASME III-2	
VFS-PL-V800B	Vacuum Relief Containment Isolation B – ORC	B	I	ASME III-2	
VFS-PL-V803A	Vacuum Relief Containment Isolation Check Valve A – IRC	B	I	ASME III-2	
VFS-PL-V803B	Vacuum Relief Containment Isolation Check Valve B – IRC	B	I	ASME III-2	
n/a	Valves Providing VFS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
n/a	Dampers in lines isolating radioactive contamination	R	NS	ASME-509	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500	
n/a	Fire Dampers	Note 3	NS	UL-555	
n/a	Supply Air Handling Units	L	NS	Manufacturer Std.	
n/a	Air Exhaust Filtration Units	R	NS	ASME AG-1, Note 4	

Table 3.2-3 (Sheet 53 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Containment Air Filtration System (Continued)					
n/a	Fans, Ductwork	L or R	NS	SMACNA or ASME AG-1, Note 4	
Balance of system components are Class L and Class R					
Health Physics and Hot Machine Shop HVAC System (VHS)				Location: Annex Building	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500	
n/a	Fire Dampers	Note 3	NS	UL-555	
n/a	Air Handling Units w/ Filters	L	NS	Manufacturer Std.	
n/a	Fans, Ductwork	L	NS	SMACNA	
Balance of system components are Class E or Class L					
Containment Hydrogen Control System (VLS)				Location: Containment	
n/a	Hydrogen Igniters	D	NS	Manufacturer Std.	Provides Hydrogen Control Following Severe Accidents
VLS-MY-E01A	Catalytic Hydrogen Recombiner A	D	NS	Manufacturer Std.	
VLS-MY-E01B	Catalytic Hydrogen Recombiner B	D	NS	Manufacturer Std.	
Balance of system components are Class E or Class L					
Radwaste Building Ventilation System (VRS)				Location: Radwaste Building	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500	
n/a	Fire Damper	Note 3	NS	UL-555	
n/a	Air Handling Units	L	NS	Manufacturer Std.	
n/a	Filters	L	NS	UL 900	
n/a	Fans, Ductwork	L	NS	SMACNA	
Balance of system components are Class E or Class L					
Turbine Building Ventilation System (VTS)				Location: Turbine Building	
n/a	Shutoff, Isolation, and Balancing Dampers	L	NS	ANSI/AMCA-500	
n/a	Fire Dampers	Note 3	NS	UL-555	
n/a	Air Handling Units w/ Filters	L	NS	Manufacturer Std., UL-900	
n/a	Fans, Ductwork	L	NS	SMACNA	
Balance of system components are Class L					
Containment Leak Rate Test System (VUS)				Location: Auxiliary Building	
VUS-PL-V015	Main Equipment Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V016	Maintenance Equipment Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V017	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V018	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V019	Personnel Hatch Test Connection	B	I	ASME III-2	

Table 3.2-3 (Sheet 54 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Containment Leak Rate Test System (Continued)					
VUS-PL-V020	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V021	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V022	Personnel Hatch Test Connection	B	I	ASME III-2	
VUS-PL-V023	Fuel Transfer Tube Test Connection	B	I	ASME III-2	
VUS-PL-V101	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V102	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V103	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V104	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V109	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V110	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V111	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V112	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V113	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V114	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V115	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V116	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V117	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V118	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V119	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V120	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V121	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V122	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V123	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V124	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	

Table 3.2-3 (Sheet 55 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Containment Leak Rate Test System (Continued)					
VUS-PL-V125	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V126	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V127	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V128	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V129	Electrical Penetration Test Isolation Valve	B	I	ASME III-2	
VUS-PL-V140	Spare Penetration Test Connection	B	I	ASME III-2	
VUS-PL-V141	Spare Penetration Test Connection	B	I	ASME III-2	
VUS-PL-V142	Spare Penetration Test Connection	B	I	ASME III-2	
Balance of system components are Class E					
Central Chilled Water System (VWS)				Location: Various	
n/a	Air Cooled Chiller	D	NS	Manufacturer Std.	
n/a	Pumps	D	NS	Manufacturer Std.	
n/a	Tanks	D	NS	ASME VIII	
n/a	Valves Providing VWS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
VWS-PY-C01	Containment Chilled Water Supply Penetration	B	I	ASME III, 2	
VWS-PY-C02	Containment Chilled Water Return Penetration	B	I	ASME III, 2	
VWS-PL-V058	Fan Coolers Supply Containment Isolation	B	I	ASME III-2	
VWS-PL-V062	Fan Coolers Supply Containment Isolation	B	I	ASME III-2	
VWS-PL-V080	VWS Containment Isolation Relief Valve	B	I	ASME III-2	
VWS-PL-V082	Fan Coolers Return Containment Isolation	B	I	ASME III-2	
VWS-PL-V086	Fan Coolers Return Containment Isolation	B	I	ASME III-2	
VWS-PL-V424	Containment Penetration Test Connection	B	I	ASME III-2	
VWS-PL-V425	Containment Penetration Test Connection	B	I	ASME III-2	
Balance of system components are Class E					

Table 3.2-3 (Sheet 56 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Annex/Auxiliary Nonradioactive Ventilation System (VXS)			Location: Auxiliary Building and Annex Building		
n/a	Air Handling Unit Fans Providing AP1000 Equipment Class D Function	Note 2	NS	AMCA	
n/a	Dampers Providing VXS AP1000 Equipment Class D Function	Note 2	NS	ANSI/AMCA-500	
n/a	Exhaust Fan Providing Ancillary Diesel Room Ventilation	Note 2	NS	AMCA	
n/a	Fire Dampers	Note 3	NS	UL-555 or UL-555S	
n/a	Air Handling Units	L	NS	Manufacturer Std.	
n/a	Filters	L	NS	UL 900	
n/a	Fans, Ductwork	L	NS	SMACNA	
n/a	Ductwork in Auxiliary Building except ductwork attached to mechanical modules	L	II	SMACNA	
Balance of system components are Class E or Class L					
Hot Water Heating System (VYS)				Location: Various	
System components are Class E					
Diesel Generator Building Ventilation System (VZS)				Location: Diesel Generator Building	
n/a	Unit Heaters Providing AP1000 Equipment Class D Function	Note 2	NS	UL-1025; NFPA 70	
n/a	Fans Providing AP1000 Equipment Class D Function	Note 2	NS	AMCA	
n/a	Dampers Providing VZS AP1000 Equipment Class D Function	Note 2	NS	AMCA	
n/a	Fire Dampers	Note 3	NS	UL-555	
n/a	Air Handling Units	L	NS	Manufacturer Std.	
n/a	Filters	L	NS	UL 900	
n/a	Fans, Ductwork	L	NS	SMACNA	
Balance of system components are Class E					
Gaseous Radwaste System (WGS)				Location: Auxiliary Building	
n/a	Gas Cooler	D	NS	Manufacturer Std.	
n/a	Sample Pumps	D	NS	Manufacturer Std.	
n/a	Guard and Delay Beds	D	NS	ASME VIII	Design for 1/2 SSE
n/a	Moisture Separator	D	NS	ASME VIII	
n/a	Valves Providing WGS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
Liquid Radwaste System (WLS)			Location: Containment, Auxiliary, and Radwaste Buildings		
n/a	Heat Exchangers, WLS and CCS Side	D	NS	ASME VIII/ TEMA	
n/a	Pumps	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 57 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Liquid Radwaste System (WLS) (Continued)					
n/a	Tanks (except WLS-MT-13A, WLS-MT-13B, WLS-MT-17, WLS-MT-23A, WLS-MT-23B	D	NS	ASME III without Code Stamp	
WLS-MT-13A	Waste Holdup Tank A Chemical Addition Pot	D	NS	ASME VIII-1	
WLS-MT-13B	Waste Holdup Tank B Chemical Addition Pot	D	NS	ASME-VIII-1	
WLS-MT-17	Chemical Waste Tank Chemical Addition Pot	D	NS	ASME VIII-1	
WLS-MT-23A	WLS Leak Chase Collection Pot A	D	NS	ASME VIII-1	
WLS-MT-23B	WLS Leak Chase Collection Pot B	D	NS	ASME VIII-1	
n/a	Degasifier	D	NS	ASME VIII	
n/a	Ion Exchangers	D	NS	ASME VIII	
n/a	Filters	D	NS	ASME VIII	
n/a	Valves Providing WLS AP1000 Equipment Class D Function (local drain valves in Radwaste Building)	D	NS	ANSI 16.34	
n/a	Floor Drain Hubs	D	NS	Manufacturer Std.	
WLS-MT-02	Containment Sump	D	NS	ACI 349	ACI 349 Evaluation of Structural Boundary Only
WLS-PL-V055	Sump Discharge Containment Isolation IRC	B	I	ASME III-2	
WLS-PL-V057	Sump Discharge Containment Isolation ORC	B	I	ASME III-2	
WLS-PL-V058	WLS Containment Isolation Relief Valve	B	I	ASME III-2	
WLS-PL-V067	RCDT Gas Outlet Containment Isolation IRC	B	I	ASME III-2	
WLS-PL-V068	RCDT Gas Outlet Containment Isolation ORC	B	I	ASME III-2	
WLS-PL-V071A	CVS Compartment to Sump	C	I	ASME III-3	
WLS-PL-V071B	PXS A Compartment to Sump	C	I	ASME III-3	
WLS-PL-V071C	PXS B Compartment to Sump	C	I	ASME III-3	
WLS-PL-V072A	CVS Compartment to Sump	C	I	ASME III-3	
WLS-PL-V072B	PXS A Compartment to Sump	C	I	ASME III-3	
WLS-PL-V072C	PXS B Compartment to Sump	C	I	ASME III-3	
WLS-PY-C02	Reactor Coolant Drain Tank WLS Connection Penetration	B	I	ASME III, 2	
WLS-PY-C03	Containment Sump Pumps Combined Discharge Penetration	B	I	ASME III, MC	
WLS-MY-Y34	Containment Sump Level Instrument Stilling Well	D	I	Manufacturer Std.	

Table 3.2-3 (Sheet 58 of 58)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Liquid Radwaste System (WLS) (Continued)					
WLS-MY-Y35	Containment Sump Level Instrument Stilling Well	D	I	Manufacturer Std.	
WLS-MY-Y36	Containment Sump Level Instrument Stilling Well	D	I	Manufacturer Std.	
Balance of system components are Class E					
Radioactive Waste Drain System (WRS)				Location: Auxiliary Building	
n/a	Pumps	D	NS	Manufacturer Std.	
n/a	Valves Providing WRS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
n/a	Floor Drain Hubs	D	NS	Manufacturer Std.	
WRS-MT-02A	WRS Leak Chase Collection Pot A	D	NS	ASME VIII-1	
WRS-MT-02B	WRS Leak Chase Collection Pot B	D	NS	ASME VIII-1	
Solid Radwaste System (WSS)				Location: Auxiliary Building	
n/a	Pumps	D	NS	Manufacturer Std.	
n/a	Tanks	D	NS	ASME VIII	
n/a	Filters	D	NS	ASME VIII	
n/a	Valves Providing WSS AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
Balance of system components are Class E					
Waste Water System (WWS)				Location: Various	
WWS-PL-V506	MCR WWS Isolation Valve	C	I	ASME III-3	
Balance of system components are Class E					
Onsite Standby Power System (ZOS)				Location: Diesel Generator Building	
n/a	Diesel Generator Engines	D	NS	Manufacturer Std.	
n/a	Diesel Generator Starting Units	D	NS	Manufacturer Std.	
n/a	Diesel Generator Radiators	D	NS	CAGI	
n/a	Diesel Generator Silencers	D	NS	API 661	
n/a	Valves Providing ZOS Diesel Generator Engines AP1000 Equipment Class D Function	D	NS	ANSI 16.34	
Balance of system components are Class E					

Notes:
1. Component performs a safety-related function equivalent to AP1000 equipment Class C. The component is constructed using the standards for Class R and a quality assurance program in conformance with 10 CFR Part 50 Appendix B.
2. Component performs an AP1000 equipment Class D function and is constructed using the standards for Class L or Class R.
3. Fire dampers are constructed to the requirements of UL-555 or UL-555S if they are fire and smoke dampers and are located in Class D, Class L, and Class R ducts.
4. Construction is non-seismic and meets applicable portions of ASME AG-1 consistent with RG 1.140.

Table 3.2-201
Not Used

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

The wind loadings for seismic Category I structures are in accordance with American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7-98 (Reference 1).

3.3.1.1 Design Wind Velocity

The design wind is specified as a basic wind speed of 145 mph with an annual probability of occurrence of 0.02 based on the most severe location identified in Reference 1 This wind speed is the 3 second gust speed at 33 feet above the ground in open terrain (Reference 1, exposure C). The basic wind speed of 145 mph is the 3 second gust speed that has become the basis of wind design codes since 1995. It corresponds to the 110 mph fastest mile wind used as the basis for the AP600 design in accordance with the 1988 edition of Reference 1.

Higher winds with a probability of occurrence of 0.01 are used in the design of seismic Category I structures by using an importance factor of 1.15. This is obtained by classifying the AP1000 seismic Category I structures as essential facilities and using the design provisions for Category IV of Reference 1.

Velocity pressure exposure coefficients and gust response factors are calculated according to Reference 1 for exposure C, which is applicable to shorelines in hurricane prone areas in the 1998 edition of Reference 1. The topographic factor is taken as unity.

The design wind loads calculated as described above exceed those required at other locations in the United States, where the more severe Exposure Category D is specified in Reference 1. Exposure Category D is applicable for sites near the open inland waterways, the Great Lakes, and the coastal areas of California, Oregon, Washington, and Alaska. For such locations, the basic wind speed is less than 130 mph.

The wind velocity characteristics for the Vogtle Electric Generating Plant, Units 3 and 4 (VEGP), are given in Subsection 2.3.1.3.1. These values are bounded by the design wind velocity values given in Subsection 3.3.1.1 for the AP1000 plant.

3.3.1.2 Determination of Applied Forces

The procedures used in transforming the wind velocity into an effective pressure to be applied to structures and parts and portions of structures follow the guidelines of Reference 1.

Effective pressures applied to interior and exterior surfaces of the buildings and corresponding shape coefficients are calculated according to Reference 1 for exposure C. Shape coefficients, defining the variation around the circumference of the shield building, are calculated using ASCE Paper No. 3269 (Reference 2). These shape coefficients are consistent with those observed in the model tests described in Reference 6.

3.3.2 Tornado Loadings

Seismic Category I structures are designed to resist tornado wind loads without exceeding the allowable stresses defined in Subsection 3.8.4. These tornado loads exceed the loads for hurricanes with a probability of occurrence comparable to that of the tornado. In addition, seismic Category I structures are designed to remain functional when subjected to tornado-generated missiles as discussed in Subsection 3.5.1.4. Seismic Category I structures are permitted to sustain local missile damage such as partial penetration and local cracking or permanent deformation or both, provided

that structural integrity is maintained and seismic Category I systems, components, and equipment required to function during or after passage of a tornado are not subject to damage by secondary missiles, such as from concrete spalling. See Subsection 3.5.2.

3.3.2.1 Applicable Design Parameters

The design parameters applicable to the design basis tornado are as follows:

- Maximum wind speed – 300 mph
- Maximum rotational speed – 240 mph
- Maximum translational speed – 60 mph
- Radius of maximum rotational wind from center of tornado – 150 ft
- Atmospheric pressure drop – 2.0 psi
- Rate of pressure change – 1.2 psi/sec

It is estimated that the probability of wind speeds greater than the design basis tornado is between 10^{-6} and 10^{-7} per year for an AP1000 at a "worst location" anywhere within the contiguous United States.

The tornado characteristics for the VEGP are given in Subsection 2.3.1.3.2. These values are bounded by the tornado design parameters given in Subsection 3.3.2.1 for the AP1000 plant.

3.3.2.2 Determination of Forces on Structures

The procedures described in Subsection 3.3.1.2 are used to transform the tornado wind loading and differential pressure loading into effective loads on structures, with a wind velocity of 300 mph (translational plus rotational velocities). The dynamic wind pressure is applied to the structures in the same manner as the wind loads described in Subsection 3.3.1.2, except that the importance factor, gust factor, and the variation of wind speed with height do not apply. Loading combinations and load factors used are as follows:

$$W_t = W_w$$

$$W_t = W_p$$

$$W_t = W_m$$

$$W_t = W_w + 0.5\,W_p$$

$$W_t = W_w + W_m$$

$$W_t = W_w + 0.5\,W_p + W_m$$

where:

W_t = total tornado load

W_w = total wind load

W_p = total differential pressure load

W_m = total missile load

The maximum pressure drop of 2.0 psi, applicable to a nonvented structure, is used for W_p for all structures except the upper portion of the shield building. The portion of the shield building surrounding the upper annulus is designed as fully vented (zero differential pressure) due to the large area of the air inlets and discharge stack. Figure 3.3-1 shows the velocity pressure variation with the radius from the center of the tornado. When the tornado loading includes the missile load, the structure locally may go into the plastic range because of missile impact. Subsection 3.5.3 discusses the procedure for analyzing local missile effects.

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

The failure of structures not designed for tornado loadings does not affect the capability of seismic Category I structures or safety-related systems performance. This is accomplished by one of the following:

- Designing the adjacent structure to seismic Category I structure tornado loading
- Investigating the effect of adjacent structure failure on seismic Category I structures to determine that no impairment of function results
- Designing a structural barrier to protect seismic Category I structures from adjacent structural failure.

The structures adjacent to the nuclear island are the annex building, the radwaste building, and the turbine building.

The portion of the annex building adjacent to the nuclear island is classified as seismic Category II and is designed to seismic Category I structure tornado loading. The acceptance criteria are based on ACI 349 for concrete structures and on AISC N690 for steel structures. The structure is constructed to the same requirements as nonseismic structures, ACI 318 for concrete structures, and AISC-S355 for steel structures. Siding is permitted to blow off during the tornado.

The radwaste building is a small steel-frame building. If it were to collapse in the tornado, it would not impair the integrity of the reinforced concrete nuclear island.

The main area of the turbine building is classified as nonseismic and is designed to seismic Category I structure tornado loading. The acceptance criteria for tornado loading are based on ACI 318 for concrete structures using a load factor of 1.0 and on 1.7 times the AISC 360 allowables for steel structures. Siding is permitted to blow off during the tornado.

Consideration of the effects of wind and tornado due to failures in an adjacent AP1000 plant and VEGP Units 1 and 2 are bounded by the evaluation of the buildings and structures in a single unit.

3.3.2.4 Tornado Loads on the Passive Containment Cooling System Air Baffle

The containment air baffle is located within the annulus between the containment vessel and the shield building. It interfaces with the passive containment cooling system and separates downward flowing air entering at the air intake openings at the top of the cylindrical portion of the shield building from upward flowing air that cools the containment vessel and flows out of the discharge diffuser.

Loads due to the atmospheric pressure drop (W_p) are calculated assuming the tornado is centered over the containment. Differential pressure between the air intakes and the discharge is calculated based on the radius of the shield building and the parameters of the tornado defined in

Subsection 3.3.2.1. The differential pressure is used with the pressure loss coefficients in the air flow path to determine pressures throughout the flow path.

The development of loads on the air baffle due to the design wind and tornado (W_w) are described in the test reports (References 3, 4, and 5). Models of the AP600 were tested in a wind tunnel and subjected to representative wind profiles. Pressures were measured on each side of the baffle, and the differential pressures were normalized to the input wind velocity. The pressure coefficients are applied to the effective dynamic pressure for the design wind and the tornado to obtain the wind loads across the baffle. The tornado wind is specified to be constant with height. The tornado loads calculated for the AP600 are applicable to the AP1000. The AP1000 configuration is similar to the AP600. The height of the shield building roof increases by 20′ 6"; the exterior diameter of the passive containment cooling storage tank increases from 80′ 0" to 89′ 0". The pressure coefficients measured in the AP600 tests are not significantly affected by these changes in geometry.

Wind conditions result in a pressure reduction in the annulus between the shield building and the containment vessel as well as above the containment dome. This reduced pressure is equivalent to an increase in containment internal pressure and is within the normal operating range for containment pressure (-0.2 to 1.0 psig).

Wind conditions result in a small wind load across the containment vessel. This is maximum opposite the air intakes where positive pressures occur on the windward side and negative pressures occur on the leeward side. Lateral loads on the containment vessel are developed in Reference 5.

3.3.3 Combined License Information

The site interface criteria for wind and tornado are addressed in APP-GW-GLR-020 (Reference 7).

The VEGP site satisfies the site interface criteria for wind and tornado (see Subsections 3.3.1.1, 3.3.2.1 and 3.3.2.3) and will not have a tornado-initiated failure of structures and components within the applicant's scope that compromises the safety of AP1000 safety-related structures and components (see also Subsection 3.5.4).

Subsection 1.2.2 discusses differences between the plant specific site plan (see Figure 1.1-202) and the AP1000 typical site plan shown in Figure 1.2-2

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

Missiles caused by external events separate from the tornado are addressed in Subsections 3.5.1.3, 3.5.1.5, and 3.5.1.6.

3.3.4 References

1. American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7-98.

2. ASCE Paper No. 3269, "Wind Forces on Structures," Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961).

3. WCAP-13323-P and WCAP-13324-NP, "Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor," August 1992.

4. WCAP-14068-P, "Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor," May, 1994.

5. WCAP-14169-P, "Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report," September, 1994.

6. WCAP-13294-P and WCAP-13295-NP, "Phase I Wind Tunnel Testing for the Westinghouse AP600 Reactor," April 1992.

7. APP-GW-GLR-020, "Wind and Tornado Site Interface Criteria," Westinghouse Electric Company LLC.

8. American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7-05.

9. AISC 360, "Specification for Structural Steel Buildings," March 9, 2005.

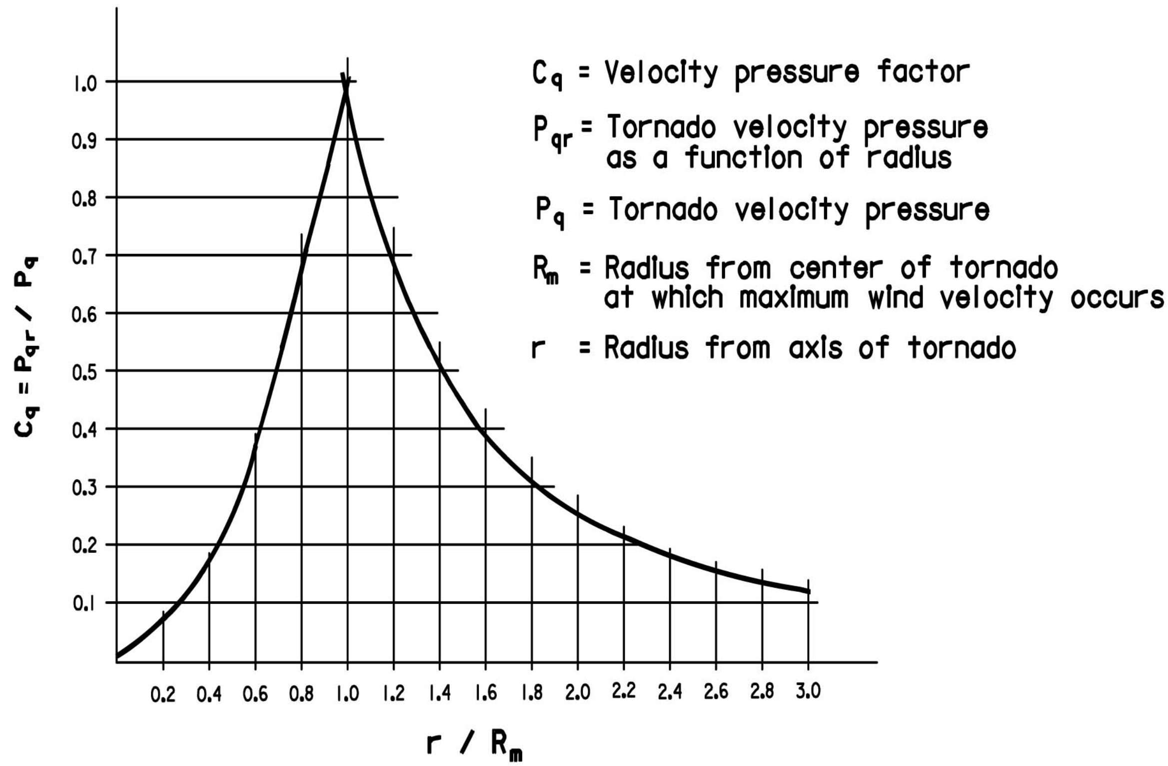


Figure 3.3-1
Velocity Pressure Variation with Radius from Center of Tornado

3.4 Water Level (Flood) Design

External flooding of a nuclear power plant from natural causes can be attributed to probable maximum flood, site and adjacent area probable maximum precipitation runoff, seiche, and ground water. Criteria for the design basis flood are in accordance with the provisions of Regulatory Guide 1.59, Design Basis Floods for Nuclear Power Plants, and Regulatory Guide 1.102, Flood Protection for Nuclear Power Plants. Conformance with the Regulatory Guides is described in Section 1.9. External events are described in Section 2.4. Chapter 2 provides interface data for AP1000 which has an interface flood level at plant grade.

Internal plant flooding can be attributed to piping ruptures, tank failures, or the actuation of fire suppression systems.

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Seismic Category I Structures, Systems, and Components

The seismic Category I structures, systems, and components identified in Section 3.2 are designed to withstand the effects of flooding due to natural phenomena or postulated component failures. A description of the structures is provided in Subsections 3.8.2, 3.8.3, and 3.8.4. None of the nonsafety-related structures, systems and components were found to be important based on flooding considerations. As a result, nonsafety-related structures, systems and components are not important in mitigation of flood events and are not required to be protected from either internal or external flooding.

3.4.1.1.1 Protection from External Flooding

The probable maximum flood for the AP1000 has been established at less than plant elevation 100′ as discussed previously in Section 2.4. The probable maximum flood results from site specific events, such as river flooding, upstream dam failure, or other natural causes.

Flooding does not occur from the probable maximum precipitation. The roofs do not have internal roof drains. The annex, radwaste, and diesel/generator buildings have parapets with large openings to drain to scuppers/drains to preclude accumulation of water on the roofs. The roofs are sloped such that rainfall is directed towards gutters located along the edges of the roofs. Therefore, ponding of water on the roofs is precluded. Water from roof drains and/or scuppers, as well as runoff from the plant site and adjacent areas, is conveyed to catch basins, underground pipes, or directly to open ditches by sloping the tributary surface area. The site is graded to offer protection to the seismic Category I structures.

The high ground water table interface is at two feet below the grade elevation, as discussed in Section 2.4.

The components that may be potential sources for external flooding are nonsafety-related, nonseismic tanks as shown in Figure 1.2-2:

- Fire water tanks as described in Subsection 9.5.1. These two tanks have volumes of approximately 325,000 and 490,000 gallons, and are located at the north end of the turbine building. Water will drain from the tanks away from the nuclear island and adjacent buildings due to the required site grading.

- Condensate storage tank as described in Subsection 9.2.4. This tank has a volume of 657,000 gallons, and is located at the west side of the turbine building. Water will drain from the tank away from the turbine and auxiliary buildings due to site grading.

- Demineralized water tank as described in Subsection 9.2.4. This tank has a volume of approximately 100,000 gallons and is located adjacent to the annex building at elevation 107′-2″. Water will drain from the tank away from the annex building to elevation 100′-0″. Nearby doors lead to areas in the annex building which do not contain safety-related components or systems.

- Boric acid storage tank as described in Subsection 9.3.6. This tank has a volume of approximately 70,000 gallons and is located adjacent to the demineralized water storage tank.

- Diesel fuel oil tanks as described in Subsection 9.5.4. These two tanks have volumes of approximately 100,000 gallons each. They are located remote from safety-related structures and are provided with dikes to retain leaks and spills.

- Passive containment cooling ancillary water storage tank as described in Subsection 6.2.2.3. This tank has a volume of 780,000 gallons and is located at the west side of the auxiliary building. Water will drain from the tank away from the auxiliary building due to site grading.

In addition, failure of the cooling tower or the service water or circulating water piping under the yard could result in a potential flood source. However, these potential sources are located far from safety-related structures and the consequences of a failure in the yard would be enveloped by the analysis described in Subsection 10.4.5.

For the AP1000, the 100′-0″ building floor elevations are slightly above the grade elevation. In addition, the slope of the yard grade directs water away from the buildings. Because the probable maximum flood for AP1000 is less than grade elevation, the exterior doors are not required to be watertight for protection from external flooding.

Process piping penetrations through the exterior walls of the nuclear island below grade are embedded in the wall or are welded to a steel sleeve embedded in the wall. There are no access openings or tunnels penetrating the exterior walls of the nuclear island below grade.

The reinforced concrete seismic Category I structures, incorporating the waterproofing and sealing features described above and in Subsection 3.4.1.1.1.1, provide hardened protection for safety-related structures, systems, and components as defined in Regulatory Guide 1.59.

3.4.1.1.1.1 Waterproofing

A waterproof membrane or waterproofing system for the seismic Category I structures below grade will be installed as an architectural aide to limit the infiltration of subsurface water. The COL applicant will use a waterproofing system for foundation mat (mudmat) and the below grade exterior walls exposed to flood and groundwater that will demonstrate a friction coefficient ≥ 0.55 with all horizontal concrete surfaces. This friction coefficient is maintained for the life expectancy of the plant and will not introduce a horizontal slip plane increasing the potential for movement during an earthquake (see Subsection 3.8.5.5.3). Typical waterproofing approaches are described as follows:

- HDPE Double-Sided Textured Waterproof Membrane

  Figures 3.4-1and 3.4-2 show the typical application of this waterproofing approach for a mechanically stabilized earth (MSE) wall and for a step-back configuration.

- HDPE Single-Sided Self-Adhering Sheet Waterproofing Membrane

  The HDPE single-sided adhesive sheet membrane is interchangeable with the HDPE double-sided textured membrane for use in the mudmat, and it may be used in certain circumstances to waterproof the walls.

- Self-adhesive, Rubberized Asphalt/Polyethylene Waterproofing Membrane

  The self-adhesive rubberized membrane is for application to waterproof the walls only.

- Sprayed-on Waterproofing Membrane

  This method may be used either for soil sites, in conjunction with an MSE wall, or for rock sites, where an open excavation may be used. The membrane consists of 100-percent solids materials based on polymer-modified asphalt or polyurea. This system may include a polyester reinforcement fabric having properties necessary to meet the performance requirements for the system. Figure 3.4-4 shows the typical installation using MSE walls with the sprayed-on, liquid-applied waterproofing membrane placed on the MSE wall panels and between the two layers of the mudmat. Where the vertical face of excavation is used as a form for the exterior walls, the waterproof membrane is installed on the vertical face of the excavation prior to placement of concrete in the exterior walls. An alternate waterproofing system for the seismic Category I structures below grade is as presented in Subsection 3.8.5.1.1.

- HDPE Waterproof Membrane System

  The HDPE waterproof membrane system uses an embedded HDPE Liner, with additional rubberized waterproof sealant materials forming a transition to the waterproof membrane on the MSE walls. It may be used for vertical waterproofing applications on the NI wall.

The waterproof function of the membrane is not safety-related; however, the membrane between the mudmats must provide adequate shear strength to transfer horizontal shear forces due to seismic (SSE) loading. This function is seismic Category I. The waterproof membrane will have physical properties, including surface and texture, to achieve the required coefficient of friction. Primer or geotextile may be added as required.

3.4.1.1.2 Protection from Internal Flooding

The nuclear island general arrangement drawings provided in Section 1.2 are a useful reference for the internal flooding discussion.

The AP1000 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components. As a result, component failures resulting from internal flooding do not prevent safe shutdown of the plant or prevent mitigation of the flooding event. Protection mechanisms are described in Section 3.6. The protection mechanisms related to minimizing the consequences of internal flooding include the following:

- Structural enclosures
- Structural barriers
- Curbs and elevated thresholds
- Leak detection systems
- Drain systems

The AP1000 minimizes the number of penetrations through enclosure or barrier walls below the flood level. Those few penetrations through flood protection walls that are below the maximum flood level

are watertight. Any process piping penetrating below the maximum flood level either is embedded in the wall or floor or is welded to a steel sleeve embedded in the wall or floor. There are no watertight doors in the AP1000 used for internal flood protection because, as described in Subsection 3.4.1.2.2, they are not needed to protect safe shutdown components from the effects of internal flooding. The walls, floors, and penetrations are designed to withstand the maximum anticipated hydrodynamic loads associated with a pipe failure as described in Section 3.6. The two watertight doors on the waste holdup tank compartments limit the consequence of a failure on spent fuel pool water level.

3.4.1.2 Evaluation of Flooding Events

3.4.1.2.1 External Flooding

Base mat and exterior walls of seismic Category I structures are designed to resist upward and lateral pressures caused by the probable maximum flood and high ground water level. The vertical hydrostatic pressure acting uniformly at the bottom of the base mat is the product of the height to the high water level and the unit weight of water assumed as 62.4 lb/ft^3. The horizontal hydrostatic pressure acting on the exterior walls varies with height, with the maximum at the bottom of the wall and zero at the maximum water level. Minimum factors of safety for overturning, sliding, and flotation are described in Subsection 3.8.5. There are no dynamic water forces associated with the probable maximum flood or high ground water level because they are below the finished grade. Dynamic forces associated with the probable maximum precipitation are not factors in the analysis or design since the finished grade is adequately sloped.

There are no safety-related hydraulic structures for AP1000.

3.4.1.2.2 Internal Flooding

This section describes the consequences of compartment flooding for various postulated component failures. The equipment required to achieve and maintain safe shutdown depends on the initiating event. The safety-related systems and components available for safe shutdown are described in Section 7.4. This equipment is located in the auxiliary building and inside containment. Except for floor drains, no credit is taken in this evaluation for the availability of nonsafety-related systems or components.

Each area of the plant containing safety-related systems or equipment is reviewed to determine the postulated fluid system failures which would result in the most adverse internal flooding conditions. For the internal flooding analysis, the failure of safety-related systems, structures or components is acceptable provided they have no safe shutdown function or the safe shutdown function is otherwise accomplished. The internal flooding analysis shows that systems, structures, and components are not prevented from performing their required safe shutdown functions due to the effects of the postulated failure. In addition, the analysis identifies the protection features that mitigate the consequences of flooding in an area that contains safety-related equipment.

The flooding sources considered in the analysis consist of the following:

- High-energy piping (breaks and cracks)
- Through-wall cracks in seismically-supported moderate energy piping
- Breaks and through-wall cracks in non-seismically-supported moderate energy piping
- Pump mechanical seal failures
- Storage tank ruptures
- Actuation of fire suppression systems
- Flow from upper elevations and adjacent areas

The analysis is performed based on the criteria and assumptions provided in Section 3.6 and ANS-56.11 (Reference 1). Section 3.6 provides the criteria used to define break and crack locations and configurations for high and moderate-energy piping failures. Additional design criteria pertaining to the internal flooding analysis are provided in this section.

The analysis consists of the following steps:

- Identification of the flood sources
- Identification of essential equipment in area
- Determination of flowrates and flood levels
- Evaluation of effects on essential equipment

As stated in Section 3.6, high-energy ASME Code Class 1, 2, and 3 piping of 6 inch nominal diameter or larger inside the containment is evaluated for mechanistic pipe break (leak-before-break) for AP1000. Those high-energy piping systems that do not satisfy the mechanistic pipe break requirements inside containment and high-energy lines outside containment are evaluated for non-mechanistic breaks and cracks, as above.

Fluid flowrates from high- and moderate-energy piping ruptures are determined based on the criteria provided in Section 3.6 and ANSI 56.11 (Reference 1). Fluid flowrates through stairwells, floor openings, and floor sleeves are determined in accordance with the formulas given in Reference 1.

No breaks are assumed for piping with nominal diameters of 1 inch or less. For each storage tank rupture, it is assumed that the entire tank inventory is drained.

The analysis of potential flooding events is performed on a floor-by-floor and room-by-room basis depending upon the relative location of safety-related equipment. No credit is taken for operation of sump pumps to mitigate the consequences of flooding.

3.4.1.2.2.1 Containment Flooding Events

General

The safe shutdown systems and components located inside the containment are associated with the passive core cooling system (PXS), the automatic depressurization system (ADS), and containment isolation.

The evaluation of containment flooding events addresses the impact of flooding on the safe shutdown systems and components. The AP1000 passive core cooling system, the internal containment compartments, and the equipment locations are designed for internal flooding to maintain post accident long-term cooling flow to the reactor core from the flooded volumes.

In the unlikely event of a loss-of-coolant accident (LOCA), the combined water inventory from available sources within the containment is sufficient to flood the reactor and steam generator compartments to a level above the reactor coolant system piping to provide water flow back into the reactor coolant system via the break location or via the passive core cooling system containment recirculation subsystem (see Section 6.3) flow path.

The potential for flooding safe shutdown components inside containment that would be required to perform safe shutdown functions is limited to two equipment compartments. These compartments are located in the southeast and northeast quadrants of the containment below the floor at elevation 107′-2″. For flood evaluation, these compartments extend up to the top of the curbs through the openings in the floor. These two compartments contain passive core cooling system components that provide two redundant means for delivering borated water to the reactor coolant system when required for safe shutdown.

The two passive core cooling system compartments primarily contain passive core cooling system components. The southeast compartment is referred to as the PXS-A compartment and the northeast compartment as the PXS-B compartment. The principal passive core cooling system component in each passive core cooling system compartment is an accumulator. A passive core cooling system core makeup tank is located above each passive core cooling system compartment. Each passive core cooling system compartment also contains isolation valves for the accumulator, the core makeup tank, the in-containment refueling water storage tank, and the passive core cooling system containment recirculation subsystem line.

There are seven automatically actuated containment isolation valves inside containment subject to flooding. These normally closed containment isolation valves are not required to operate during a safe shutdown operation and they would not fail open as a result of the compartment flooding. Also, there is a redundant, normally closed, containment isolation valve located outside containment in series with each of these valves.

The PXS-A compartment contains one normally closed spent fuel pit cooling system containment isolation valve. The PXS-B compartment contains four normally closed normal residual heat removal system containment isolation valves. The maintenance floor contains two normally closed liquid radwaste system containment isolation valves located partially below the maximum flood level.

Except for the valves mentioned above, the rest of the automatically actuated containment isolation valves are located above the maximum flood level; therefore, these components would not be adversely affected by postulated flooding.

Flooding can be postulated from a failure of several systems located inside the containment. The worst case flooding scenario is a LOCA. The maximum flood level for a LOCA is based on the combined inventory of the reactor coolant system, the two accumulators, the two core makeup tanks, and the in-containment refueling water storage tank flooding the containment. The maximum inventory also considers makeup from the cask loading pit and boric acid tank.

Curbs are provided around openings through the maintenance floor at elevation 107′-2″ to control flooding. Overflow into the refueling canal occurs through a pipe centered at elevation 110′-0″. Curbs around openings into the chemical and volume control system compartment extend up to elevation 110′-0″. Curbs around openings into the PXS-A compartment extend up to elevation 110′-2″. Curbs around openings into the PXS-B compartment extend up to elevation 110′-1″. With these curb elevations, water flooding the maintenance floor is directed first into the refueling canal, then into the CVS compartment, then into the PXS-B compartment, and finally into the PXS-A compartment.

The evaluation of containment flooding from postulated component failures includes the compartments that are located below the maximum flood level. There are seven subcompartments that contain components below the floor at elevation 107′-2″. The active safe shutdown components inside containment which are located below the maximum flood level are located in only two of the seven floodable compartments.

The seven compartments partially or completely below the maximum flood level include the reactor vessel cavity, the two steam generator compartments, the vertical access tunnel, the two passive core cooling system compartments, and the chemical and volume control system compartment. The safe shutdown components are located in the two passive core cooling system compartments.

The reactor vessel cavity and the two steam generator compartments are interconnected by a large vertical access tunnel. These four compartments are treated, in this discussion, as one large floodable volume and they are referred to as the reactor coolant system compartment. Flooding of

this compartment above elevation 107′-2″ also includes the maintenance floor outside the curbs around the other three compartments.

The PXS-A compartment (Room 11206), PXS-B compartment (Room 11207), and the chemical and volume control system compartment (Room 11209) are physically separated and isolated from each other by structural walls and curbs such that flooding in any one of these compartments or in the reactor coolant system compartment cannot cause flooding in any of the other compartments. The access hatch to the PXS-B compartment is located near the containment wall and is normally closed to address severe accident considerations. The access hatch to the PXS-B compartment is accessible from Room 11300 on elevation 107′-2″.

The fire protection system and the demineralized water transfer and storage system are open-cycle systems that enter the containment. During plant operation, the containment piping for these systems is isolated by containment isolation valves and is not a potential flooding source. These systems are not open systems as defined in Bulletin 80-24 (one that has an essentially unlimited source).

Reactor Coolant System Compartment

The reactor coolant system compartment, represented by the reactor vessel cavity, the two steam generator compartments, and the large vertical access tunnel, is the largest of the separate floodable compartments. With the exception of the pressurizer which is at a higher elevation, the principal components of the reactor coolant system are contained in this compartment.

The reactor vessel cavity and the adjoining equipment room are at the lowest level in the containment. The floor level of these rooms is at elevation 71′-6″. The floor level of the two steam generator compartments is at elevation 83′-0″. A portion of each compartment has low point areas at elevation 80′-0″.

The containment sump pumps are located in the equipment room at elevation 71′-6″. The arrangement for the floor drains from the two passive core cooling system compartments and the chemical and volume control system compartment provide a drain path for each compartment to the lowest level of containment (elevation 71′-6″) where the containment sump is located. Therefore, the source of the flooding in the reactor coolant system compartment is not limited to the components or systems contained within this compartment.

Any leakage that occurs within the containment drains by gravity to the elevation 71′-6″ equipment room. Reverse flow into the two passive core cooling system compartments and the chemical and volume control system compartment is prevented by redundant backflow preventers in each of the three compartment drain lines.

Flooding in any compartment of the containment is detected by the containment sump level monitoring system and the containment flood-up level instrumentation.

The containment sump level monitoring system consists of two seismically qualified level sensors in the containment sump. These sensors transmit the sump level indication to the main control room and the plant instrumentation system.

The plant instrumentation system monitors the rate of the sump level rise, calculates the leakage collection rate, and initiates the appropriate alarms in the main control room. A description of this leak detection system is provided in Subsection 5.2.5.3.1.

Another indication of flooding in this compartment is provided by the containment flood-up level instrumentation consisting of three redundant Class 1E level sensor racks. Multiple discrete level signals are provided from the bottom of the reactor vessel cavity to the top of the vertical access

tunnel. These level sensors transmit the containment sump water level indication to the main control room.

In the event that the source of the containment flooding can not be terminated, the water level in the reactor vessel cavity and the steam generator compartment continues to increase until the water source has been depleted or the leak has been isolated. The maximum level that could occur in the compartment from all of the water which is available in containment is elevation 108′-10″.

Since the reactor coolant system compartment contains no active safe shutdown components below the maximum flood-up level, the flooding of this compartment has no impact on safe shutdown capability.

Passive Core Cooling System Compartments

The PXS-A and PXS-B compartments, located in the southeast and northeast quadrants of the containment, primarily contain components associated with the passive core cooling system. The safe shutdown related components of the passive core cooling system located in these two compartments are redundant and essentially identical. One set of the redundant equipment is located in each of the two separate compartments.

The redundant passive core cooling system components located in these two compartments provide coolant to the reactor vessel from the two core makeup tanks, the two accumulators, and the in-containment refueling water storage tank via two independent and redundant direct vessel injection lines.

Each passive core cooling system compartment contains a parallel set of normally closed, air operated, core makeup tank isolation valves that receive actuation signals to open during a safe shutdown operation. These valves are approximately 10 feet above the floor level of the passive core cooling system compartments and 26 feet above the floor of the reactor vessel cavity.

Each passive core cooling system compartment also contains one normally open accumulator isolation valve and one normally open in-containment refueling water storage tank isolation valve. These valves do not have to be repositioned during a safe shutdown operation and a coincident flooding event.

In addition, each passive core cooling system compartment contains four passive core cooling system containment recirculation subsystem isolation valves. A normally closed, squib valve is located in each of two parallel flow paths. One of the lines includes a check valve in series with the squib valve. The other line includes a normally open, motor-operated valve in series with the squib valve. The squib valve and motor-operated valves are opened on a low in-containment refueling water storage tank level signal to provide a redundant flow path from the flooded reactor/steam generator compartments to the reactor vessel. One set of these redundant containment recirculation subsystem isolation valves is required to open to provide a redundant recirculation flow path to the reactor vessel. In the unlikely event that one of the two passive core cooling system compartments were to be flooded, the set of recirculation valves in the other, unflooded, compartment could be opened. Note that these squib valves are qualified to operate after being flooded.

The design bases for this system are described in Section 6.3. The passive core cooling system is designed to perform its safety functions in the unlikely event of the most limiting single failure occurring coincident with any design basis event. For example, a direct vessel injection line could break in one of the two passive core cooling system compartments, thus preventing the core makeup tank and the accumulator located in the compartment from delivering borated water to the reactor vessel. A coincident single failure in the other passive core cooling system compartment would prevent only one of the two parallel injection paths from opening. This series of events would not prevent the passive core cooling system from performing its safety function.

The maximum flooding rate to either of these passive core cooling system compartments would occur on a postulated LOCA of one of the eight inch direct vessel injection lines at a location inside one of the two compartments. This postulated rupture would result in direct blowdown from the reactor coolant system to the compartment as well as blowdown of the associated core makeup tank and accumulator. The resulting flooding in one of two passive core cooling system compartments would not prevent the passive core cooling system from performing its safe shutdown function.

Another postulated LOCA, that would cause rapid flooding in the PXS-B compartment, is a rupture of the 12 inch normal residual heat removal system line. This line is routed from one of two reactor coolant system hot legs to a containment penetration in the PXS-B compartment.

The evaluation of containment flooding events is also concerned with non-LOCA flooding events. The maximum flooding rate to either of the passive core cooling system compartments, for a non-LOCA event, would be based on a postulated rupture of one of the two in-containment refueling water storage tank lines or a postulated rupture of one of the two accumulator injection lines.

A 10-inch line is routed from the in-containment refueling water storage tank to the PXS-A compartment and a 10-inch line is routed to the PXS-B compartment. The driving head from a full in-containment refueling water storage tank to either of these two compartments is approximately 35 feet. A rupture in one of these lines would result in flooding of the associated passive core cooling system compartment and the reactor coolant system compartment via the normal drain path or by overflowing the passive core cooling system compartment.

The 8-inch accumulator injection lines are routed from the accumulators to the 8-inch direct vessel injection lines. A rupture of either of these two injection lines at a point upstream of the two series reactor coolant system pressure boundary check valves would result in the blowdown of the accumulator to the associated compartment. The water level attained in this case would be limited to the water volume of the accumulator. The water level would not reach the level of the core makeup tank isolation valves.

The total flood-up of either the PXS-A or PXS-B compartments from any source of water is acceptable and does not prevent the passive core cooling system from performing its required safe shutdown function.

The PXS-A and the PXS-B compartments are physically separated and isolated from each other by a structural wall so that flooding in one compartment can not cause flooding in the other compartment. They are located below the maintenance floor level which is at elevation 107′-2″. A curb is provided around openings that penetrate through the maintenance floor into these compartments from the elevation 107′-2″ floor level.

There are several HVAC ducts, cable trays, and pipes that penetrate the maintenance floor into the passive core cooling system compartments. These penetrations are properly protected to prevent leakage into the passive core cooling system compartments.

The floor drains for these two compartments are located at elevation 84′-6″. Reverse flow through the floor drains is blocked by redundant, safety-related backflow preventers in the drain lines.

When the flooding rate exceeds the ability of the floor drain lines to drain the water from the compartment, or in the event that the floor drain line is blocked, the water level in that compartment increases to the entrance curb elevation.

Should the flooding continue, the water overflows from that compartment to the maintenance floor at elevation 107′-2″. The water overflowing to this level would immediately drain to the reactor coolant system compartment via the vertical access tunnel. There is no curb at the entrance to the vertical

access tunnel; therefore, water on the maintenance floor (elevation 107′-2″) flows freely into the reactor coolant system compartment. For LOCA events, flooding via this path continues to a level above the reactor coolant system cold legs.

If the leakage rate into PXS-A or PXS-B were not excessive, the compartment drain lines would prevent significant flood-up in that compartment. Consequently, the flooding of the components could be prevented for postulated flooding events of limited duration and flowrates less than the drain line capacity.

The flowrate from the compartments is a function of the water height in the PXS compartments and the water height in the reactor coolant system compartment. The differential head between the two water levels establishes the flowrate from the compartment.

The draining of these compartments initiates flooding of the reactor vessel cavity and the adjoining cavity equipment room. If the operator does not terminate the leak, action is taken to shut down the reactor.

If the flooding rate is not greater than the compartment drain line capacity, the large volume of the reactor vessel cavity and the adjoining equipment room provides additional time for the operator to identify the source of leakage before any significant flooding occurs in the compartments containing the passive core cooling system equipment.

Should the flooding continue, the water level eventually reaches the steam generator compartment floor at elevation 83′-0″. The large floor area of the two steam generator compartments and the vertical access tunnel provides additional volume for flood-up and reduces the rate of level increase.

The containment isolation valves in these two passive core cooling system compartments are located above elevation 95′-0″, but below the maximum flood-up level. The PXS-A compartment contains one normally closed, motor operated, spent fuel pool cooling system containment isolation valve. The PXS-B compartment contains four normally closed, motor operated, normal residual heat removal system containment isolation valves. These containment isolation valves are not required to operate for safe shutdown and they do not fail open as a result of compartment flooding. Also, there are redundant outside containment isolation valves for each line that penetrates the containment boundary.

Chemical and Volume Control System Compartment

The majority of the components associated with the chemical and volume control system are located inside the containment in a separate compartment in the north quadrant of the containment below elevation 107′-2″.

There are several HVAC ducts, cable trays, and pipes that penetrate the maintenance floor into the chemical and volume control system compartment. These penetrations are properly protected to prevent leakage around the ducts into the chemical and volume control system compartment. The entrance curb elevation for the chemical and volume control system compartment is lower than the PXS-A and B compartment curbs to preferentially flood the chemical and volume control system compartment.

A single floor drain line is routed from this compartment to the containment sump at elevation 71′-6″. Reverse flow from the containment sump to this compartment is prevented by redundant, safety-related backflow preventers in the drain lines.

In the event that the single drain line were to be blocked, the water level in the chemical and volume control system compartment would flood to the level of the entrance curb elevation and would over flow to the maintenance floor at elevation 107′-2″. The water overflowing to this level would drain to

the reactor coolant system compartment via the vertical access tunnel. There is no adverse effect on safe shutdown of the plant from flooding of the chemical and volume control system compartment.

The fire protection system and the demineralized water transfer and storage system are open-cycle systems that enter the containment. During plant operation, the containment piping for these systems is isolated by containment isolation valves and is not a potential flooding source. These systems are not open systems as defined in Bulletin 80-24 (one that has an essentially unlimited source).

3.4.1.2.2.2 Auxiliary Building Flooding Events

General

The AP1000 auxiliary building contains radiologically controlled areas and nonradiologically controlled areas which are physically separated by 2 and 3 foot structural walls and floor slabs. These structural barriers are designed to prevent flooding across the boundary between these areas by locating penetrations for piping and HVAC duct above maximum flood levels, or by sealing these penetrations. Process piping penetrations between the radiologically controlled areas and nonradiologically controlled areas are embedded in the wall or are welded to a steel sleeve in the wall. Electrical penetrations between the radiologically controlled areas and nonradiologically controlled areas are located above the maximum flood level. Electrical penetrations subject to the effects of the local build up of water on floors above the maximum flood level are also sealed.

For example, flooding in the auxiliary building at elevation 66′-6″ of the radiologically controlled area would not cause flooding in the nonradiologically controlled areas since the two areas are completely separated by a three foot thick structural wall. In the non-radiologically controlled area (non-RCA) of the auxiliary building, the four Class 1E electrical divisions are separated by 3-hour fire barriers. Portions of these fire barriers also serve as flood barriers.

- **Nonradiologically Controlled Areas**

  The safe shutdown systems and components that are located in the nonradiologically controlled area are associated with the protection and safety monitoring and Class 1E dc system, and containment isolation. The safe shutdown components associated with the protection and safety monitoring system are the instrumentation and control (I&C) cabinets located in the nonradioactive controlled area on level 3 (elevation 100′-0″). The safe shutdown components associated with the Class 1E dc system are the Class 1E batteries on level 1 (elevation 66′-6″) and level 2 (elevation 82′-6″) and dc electrical equipment also on level 2.

  The nonradiologically controlled areas of the auxiliary building are designed to provide maximum separation between the mechanical and electrical equipment areas. This separation prevents the propagation of leaks from the piping areas and the mechanical equipment areas to the Class 1E electrical and Class 1E I&C equipment rooms.

  The major piping compartments in the nonradiologically controlled area are the main steam isolation valve compartments on levels 4 and 5 (elevations 117′-6″ and 135′-3″, respectively) and the valve/piping penetration compartment on level 3 (elevation 100′-0″). The mechanical equipment rooms in the nonradiologically controlled area are the HVAC compartments on levels 4 and 5.

  Drain lines are provided in each of the piping and mechanical equipment compartments which drain to the turbine building drain tank. Leakage from postulated pipe ruptures in these compartments will drain to the turbine building.

- **Radiologically Controlled Areas**

The safe shutdown components located in radiologically controlled areas (RCA) are primarily containment isolation valves which are located near the containment vessel and above elevation 82′-6″. These containment isolation valves are located above the maximum flood level for this area. They are required to either close or remain closed during a safe shutdown operation.

The evaluation of potential flooding within the radiologically and nonradiologically controlled areas of the auxiliary building is performed on a floor-by-floor basis as described below.

Auxiliary Building Level 1 (Elevation 66′-6″)

- **Nonradiologically Controlled Area**

  Level 1 of the nonradiologically controlled area has five individual rooms that contain Class 1E batteries: four divisional (A, B, C, and D) Class 1E battery rooms and one Class 1E spare battery room. The doors are not water tight.

  The primary line of defense for level 1 is to exclude fluid systems and their associated piping from this area. The only fluid systems in level 1 are the potable water and fire protection systems. Potable water is used for battery washdown and the emergency eye wash/shower facilities. The maximum nominal diameter of potable water piping in this area is 1 inch; therefore, it is excluded from consideration as a source of flooding.

  The potential for flooding on level 1 is limited to fire fighting activities. The seismically qualified fire protection system piping routed through levels 1, 2, 3, and 4 is the only piping in this area that is greater than 1 inch in diameter.

  Fire fighting activities in levels 1, 2, 3, or 4 would contribute to flooding in level 1. The drain lines, stairwells, and the elevator shaft direct the water from fire fighting activities down to the auxiliary building nonradiologically controlled area sump located on level 1.

  Fire fighting in these five battery rooms is accomplished by manual means from two fire hose stations located adjacent to the two stairwells. The maximum flowrate to this area from the two hose stations is assumed to be 250 gpm.

  A limited supply of water is initially provided to the fire protection system standpipe fire hose stations (See Subsection 9.5.1) from the passive containment cooling system storage tank. A nominal volume of 18,000 gallons is provided for the fire protection system. A volume of 42,000 gallons is conservatively assumed; this is the volume in the tank between the elevations of the fire protection system inlet and the tank overflow. In the event that both fire hose stations are used to fight a fire in one of the five battery rooms, the maximum water depth would be less than 12 inches, assuming that the water could propagate into all rooms on this level. This maximum water depth is substantially below the terminal height on the first row of batteries which is located approximately 30 inches above the floor.

  Since a limited supply of fire water is provided, inadvertent initiation of the fire protection system can not exceed the flooding levels described above. Operator action to stop inadvertent water flow from the fire protection system is expected to limit flooding to only a small fraction of this water supply.

  Structural walls, drain line routing, and raised platforms prevent leakage that may occur in piping or mechanical areas on levels 4 and 5 from propagating to the electrical areas on levels 1, 2, 3, or 4.

Dual sump pumps and water level sensors are also provided in the sump on level 1. The level sensors transmit water level indication to the main control room and the plant control system. Level alarms alert the operator to take corrective action.

The sump pumps are sized to remove approximately 250 gpm (with two pumps operating) based on a maximum flow from two fire hose stations of 250 gpm. The discharge of these pumps is directed to the turbine building drain tank of the waste water system (WWS) located on elevation 89′-0″ of the turbine building as described in Subsection 9.2.9. The discharge line into the tank is provided with a standpipe to prevent siphoning back to the auxiliary building nonradiologically controlled area sump. These sump pumps and level sensors are not required to maintain safe shutdown capability.

- **Radiologically Controlled Area**

There are no safe shutdown components located on level 1 of the radiologically controlled area. The radiologically controlled area of the auxiliary building is subject to flooding from a variety of potential sources including the component cooling water, central chilled water, hot water, spent fuel pool cooling, normal residual heat removal system, and chemical and volume control system, as well as various tanks. Most of the piping associated with these systems is above level 1; however, the flow from any postulated rupture in the radiologically controlled area will eventually flood level 1. The principal flow paths to level 1 are the vertical pipe chase and the floor gratings provided in the elevator lobbies on levels 2 and 3. Other flow paths include the floor drain system, the stairwell, and the elevator shaft.

The auxiliary building radiologically controlled area sump is located on level 1 with dual sump pumps and water level sensor provided in the sump. The level sensor transmits water level indication to the main control room and the plant control system. High level alarms alert the operator to take corrective action.

The sump pumps are sized to remove approximately 250 gpm (with two pumps operating) based on a maximum flow from two fire hose stations of 250 gpm. The discharge of these pumps is directed to the waste holdup tanks of the liquid radioactive waste system as described in Section 11.2. These sump pumps and level sensor are not required to maintain safe shutdown capability.

For the component cooling water and central chilled water systems, the maximum flooding volume is bounded by the system volume plus a reasonable period of makeup. This includes any discharges from the pressure relief valves on the cooling water lines to the RNS heat exchangers. Flow from these two valves is also directed to the auxiliary building sump through the WWS. For the spent fuel pool cooling system, the maximum flooding volume is limited to the volume of water above the spent fuel pool strainer plus a reasonable period of makeup. This flooding volume is approximately equal to that of the component cooling water and chilled water systems above.

The normal residual heat removal system is operated only when the plant is shutdown. Since it is not normally operating, it is evaluated as a moderate-energy system. Flooding is determined based on the maximum flowrate from a through-wall crack in a 8 inch normal residual heat removal system discharge line. Assuming that the leakage is detected and isolated within 30 minutes after initiation, the maximum flooding volume is approximately equal to those above.

Flooding due to a break in the high-energy chemical and volume control system makeup pump discharge line is bounded by the normal residual heat removal system through-wall crack.

Flow from the postulated break spreads throughout the level 1 rooms and corridor via flow under doors and interconnecting floor drains if the auxiliary building radiologically controlled area sump pumps are inoperable. The maximum flood level in the area, for any of the cases above, is less than 12 inches. This flooding has no impact on safe shutdown since there are no components on level 1 required for safe shutdown.

Normal residual heat removal systems components with systems important missions are expected to remain functional following the flooding event since the pump motors and valve operators are above the maximum flood level if the flood source is not a break in the normal residual heat removal system piping itself.

Flow from a tank rupture in one of the tank rooms will initially flood the tank room, and begin to flow to the auxiliary building radiologically controlled area sump via floor drains. If the sump pumps are inoperable, the tank volume floods the balance of level 1 via the interconnecting floor drains. The maximum flood level for this event is less than for the piping failures discussed above.

Auxiliary Building Level 2 (Elevation 82′-6″)

- **Nonradiologically Controlled Area**

  Level 2 of the nonradiologically controlled area has two Class 1E battery rooms, four divisional Class 1E dc electrical equipment rooms, and one Class 1E reactor coolant pump trip switchgear room. The doors to these rooms are not water tight.

  Level 2 contains an arrangement of fire protection and potable water piping similar to level 1.

  The potential for flooding on this level is limited to fire fighting activities. Fire fighting in these rooms is accomplished by manual means from two fire hose stations located adjacent to the two stairwells. The maximum flowrate to this area from the two hose stations is assumed to be 250 gpm.

  The drains, elevator shafts, and stairwells drain water spilled on this level to level 1. Therefore, no significant accumulation of water occurs on level 2.

- **Radiologically Controlled Area**

  The radiologically controlled area on level 2 contains a few containment isolation valves and an effluent monitor tank at elevation 92’-6". The horizontal pipe chase at elevation 92′-6″ contains two normally closed normal residual heat removal system isolation valves. One spent fuel pool cooling system containment isolation valve is located, above 92′-6″, in the adjacent vertical pipe chase. The area on the north side of the lower annulus contains two chemical and volume control system and two liquid radwaste automatically operated containment isolation valves above elevation 82′-6″. These valves are required to close or remain closed during a safe shutdown operation.

  Two chemical and volume control system valves used to isolate the chemical and volume control system makeup pump suction from the demineralized water storage tank are located in the makeup pump compartment at 82′-6″. These safety-related valves close or remain closed to prevent boron dilution events. They are not required for safe shutdown.

  Potential sources of flooding for this area include the chemical and volume control system, a liquid radioactive waste system effluent monitor tank, and the fire protection system, including an automatic suppression system in the CVS makeup pump room. Flow from a component rupture or from fire fighting activities on level 2 drains to level 1 as described below.

To protect the above valves from flooding, the makeup pump compartment at elevation 82′-6″ drains, via the floor grating located in the corridor adjacent to the stairwell, directly to elevation 66′-6″. Flooding in the lower annulus drains directly to elevation 66′-6″ via the floor grating and various openings to the tank rooms. The stairwell and elevator shaft on the east wall are additional flow paths to level 1. The horizontal pipe chase at elevation 92′-6″ drains under the door directly to elevation 66′-6″ via the vertical pipe chase. As a result of these drain paths, there is no significant accumulation of water in the makeup pump compartment, lower annulus or the horizontal pipe chase from any postulated pipe ruptures. The containment isolation valves are above the maximum flood level in these areas. The chemical and volume control system makeup pumps and the normal residual heat removal valves in the valve compartment are nonsafety-related defense-in-depth equipment and are expected to remain functional following the flooding event since the pump motors and valve operators are above the calculated flood level of 6 inches.

Auxiliary Building Level 3 (Elevation 100′-0″)

- **Nonradiologically Controlled Area**

Level 3 of the nonradiologically controlled area includes the remote shutdown room, one reactor coolant pump trip switchgear room, four divisional Class 1E I&C rooms, one equipment room, and the valve/piping penetration room. The division A, B, C and D I&C rooms and the electrical room also include containment electrical penetrations. The doors are not water tight.

The level 3 Class 1E and non-Class 1E electrical areas contain only fire protection system piping. Fire hose stations are provided near each of the two stairwells and normally dry fire protection piping, supplied from the passive containment system tank, serves the preaction sprinkler system in the non-1E equipment/penetration room. The potential for flooding in the electrical areas on this level is limited to fire fighting activities. The maximum flowrate to this area from either automatic or manual fire fighting activities is assumed to be 250 gpm. The floor drains, stairwells, and elevator shaft drain water spilled on this level down to level 1. Therefore, no significant accumulation of water occurs in this area.

The valve/piping penetration room on level 3 is physically separated from the electrical rooms. The valve/piping penetration room contains automatically actuated containment isolation valves for the steam generator blowdown system and the hydrogen line in the chemical and volume control system. Access to this room is from the turbine building. The access door and drain lines provided in this room drain from the auxiliary building to the turbine building. Maximum postulated flood level for this room is less than 36 inches. The containment isolation valves in the area are located above this maximum flood level.

- **Radiologically Controlled Area**

There are no safe shutdown components located on level 3.

Potential sources of flooding for this area include the normal residual heat removal system, the component cooling water system, an effluent monitor tank and the fire protection system, including an automatic suppression system in the rail car bay. Flow from a component rupture or from fire fighting activities on level 3 drains directly to level 1.

Auxiliary Building Level 4 (Elevation 117′-6″)

- **Nonradiologically Controlled Area**

Level 4 of the nonradiologically controlled area includes the main control room, one divisional Class 1E penetration room, one non-Class 1E electrical penetration room, two main steam isolation valve compartments, and one mechanical equipment room.

The doors to these rooms are not water tight. There are no doors from the main steam isolation valve compartments to the Class 1E electrical areas. The main steam isolation valve compartments are only accessible from the turbine building at elevation 135′-3″. The mechanical equipment room is only accessible from the turbine building at elevation 117′-6″.

The potential for flooding Class 1E electrical areas on this level is limited to fire fighting activities. The Class 1E electrical penetration room and main control room are accessible from a hose station near the east stairwell. While the main control room kitchen and restroom are provided with potable water, the lines are 1 inch and smaller, and are not evaluated for pipe ruptures.

Fire fighting in the control room is done manually using portable extinguishers or a fire hose from a hose station in the east corridor. In the event that a hose is brought into the main control room through the east corridor access doors, water accumulation is limited by flow through the access doors which are open. The threshold of the east corridor access door is at the elevation of the floor slab. Once in the corridor this flow drains, via floor drains, the stairwell and elevator shaft, to level 1. An emergency egress door and stairwell is located on the west end of the main control room, which leads down to the remote shutdown room. The threshold of the emergency egress door is flush with the raised portion of flooring in the main control room, which is approximately 14 inches above the east corridor entrance. Water being discharged in this area will flow through the porous raised flooring and flow back out the east access doors. The main control room has a normally closed floor drain which can be manually opened to drain water to the auxiliary building non-RCA sump at level 1. The drain paths prevent significant flooding of the adjacent rooms.

In the event of fire fighting activity in the non-Class 1E electrical penetration rooms, the accumulation of water is prevented by floor drains and flows through the stairwell and elevator shaft to level 1.

The mechanical equipment room contains containment isolation valves for the chilled water, compressed air, component cooling water, and passive core cooling (nitrogen) systems. Flooding in the mechanical equipment room due to fire fighting or piping ruptures is directed to the turbine building through the access door at elevation 117′-6″ or through floor drains to the turbine building. The maximum flood level for this room is 4 inches. The containment isolation valves in this area are located above this maximum flood level.

The main steam isolation valve compartments contain the main steam and main feedwater piping and their isolation valves. In the event of a pipe break or leak in the area, floor drains to the turbine building are provided. Structural walls and floors are designed to prevent flow of water to levels 1, 2, or 3. For larger flows, wall openings and pressure relief panels, located at floor elevation, open to drain the rooms to the turbine building. The maximum flood level for these rooms is less than 36 inches. The isolation valves in this area are located above this maximum flood level.

- **Radiologically Controlled Area**

In the radiologically controlled area, there are six containment isolation valves on level 4. Five of these are located in the vertical pipe chase. These are for the primary sampling system, spent fuel pool cooling system and containment air filtration system. The primary source of flooding in the vertical pipe chase is the spent fuel cooling line. Flow from this break will be directed through grating down to level 3 where water will flow under the door to the staging area and through floor drains to the auxiliary building RCA sump which limits the flood level to less than 7 inches. The

containment isolation valves are located above the spent fuel cooling line and there are no other sources of flooding located above them. The other containment isolation valve for the containment air filtration system is located in a separate compartment adjacent to column line 5. The principal source of flooding for this area is fire fighting from a hose station located at elevation 107′-2″. Flow from this source will be directed under the door and through floor drains to the auxiliary building radiologically controlled area sump which limits the flood level to less than 3 inches. No other safe shutdown equipment is located in this area.

Auxiliary Building Level 5 (Elevation 135′-3″)

- **Nonradiologically Controlled Area**

Level 5 of the nonradiologically controlled area contains two mechanical HVAC equipment rooms and the upper portion of the two main steam isolation valve compartments. There is no safety-related equipment on level 5.

The evaluation of the main steam isolation valve compartments is addressed in the discussion of level 4.

Water from fire fighting, postulated pipe, or potable water storage tank ruptures in the main mechanical HVAC equipment rooms drains to the turbine building via floor drains or to the annex building via flow under the doors. Therefore, no significant accumulation of water occurs in this room. Floor penetrations are sealed and a 6 inch platform is provided at the elevator and stairwell such that flooding in these rooms does not propagate to levels below.

The mechanical room between the main steam isolation valve compartments at level 5 is accessed from the turbine building on the same level. This room is drained to the turbine building. In the event of fire fighting or postulated pipe ruptures, the accumulation of water is prevented by directing the flow to the drains or under the doors into the turbine building. Floor penetrations are sealed such that flooding in this area does not propagate to other areas of the auxiliary building.

- **Radiologically Controlled Area**

Level 5 of the radiologically controlled area contains the fuel handling area operating deck, HVAC equipment and access rooms, the main equipment hatch staging area, and the component cooling water system valve room. The only safety-related equipment on level 5 are the compressed air tanks for the main control room emergency habitability system located in the main equipment hatch staging area.

Over-filling of the spent fuel pool would flood the fuel handling area operating deck. The flooding flowrate is limited by the makeup capacity from the demineralized water or chemical and volume control systems. Accumulation of water in this area is prevented by floor drains and by flow to the stairwells and elevator shaft which drain to level 1. Spent fuel pool cooling is not adversely affected by this event. There is no safe shutdown equipment in this area. The component cooling water system valves with the regulatory treatment of nonsafety-related system important missions located in the component cooling water system valve room which support the normal residual heat removal system, are located well above the maximum flood level for this area and are expected to remain functional in a flooding event.

The shield building stairwell serves as a pipe chase for passive containment cooling system supply and return lines, drains for the passive containment cooling system valve room and passive containment cooling system air outlet shield plug, and a fire water line. Leakage from a crack in one of these lines flows down the stairwell to level 5, under the stairwell door to the HVAC equipment room, and then to the auxiliary building radiologically controlled area sump via floor drains or to the annex building. There is no significant accumulation in the stairwell or the

equipment and access rooms. There is no safe shutdown equipment in this area. The passive containment cooling system supply and return line connections to the passive containment cooling system storage tank are above the minimum water level, thus a leak in these lines would not adversely affect the safe shutdown capability of the passive containment cooling system.

Water from fire fighting in the main equipment hatch staging area drains to the auxiliary building radiologically controlled area sump via floor drains, or to the annex building via flow under the roll-up door. Therefore, no significant accumulation of water occurs in this area.

Auxiliary Building Upper Annulus (Elevation 132′-3″)

This area serves as the air flow path for the passive containment cooling system. It is bounded by the seismic Category I shield building on the outside and the seismic Category I containment vessel on the inside. The floor has a curb on the outside with a flexible seal connected to the shield building. The curb and seal block communication with the middle annulus area, below. The outside wall of the annulus is provided with redundant safety-related drains to the yard drainage system.

The worst case flooding scenario is postulated as blockage of the nonsafety-related floor drains concurrent with inadvertent opening of a passive containment cooling system cooling water isolation valve. The maximum flood level is determined by the flow gradient to the operating drains. Maximum level will be approximately 2 feet. This level does not affect the capability of passive containment cooling system air cooling. No other safe shutdown equipment is affected. Passive containment cooling system operation or leakage is detected by sensors on the passive containment cooling system discharge line. During non-accident conditions the annulus is accessible to manually clear any drain blockage.

PCS Valve Room (Elevation 284′-10″)

This room contains three redundant safety-related valve trains for the passive containment cooling system water cooling subsystem. One train must open to provide the required containment cooling. The only source of flooding for this room is a through-wall crack in the passive containment cooling system piping. The worst crack location is in the 6 inch line between the valves and the flow control orifices. This leak is not isolable from the 756,700-gallon passive containment cooling system water storage tank above the valve room. Flow is by gravity.

Leakage will flow down to the landing at elevation 277′-2″ where the water will flow through floor drains or under doors to the upper annulus which is then discharged through redundant drains to the storm drain. There will be negligible water accumulation in the valve room. The passive containment cooling system isolation valves are located above the maximum flood level in the valve room, so they remain operable.

Sensors in the valve room drain sump are provided for leak detection. An alarm is provided in the main control room to alert the operator to take corrective action if the level sensors detect an abnormal water level in the valve room. The leakage does not adversely affect containment or any other essential system.

3.4.1.2.2.3 Adjacent Structures Flooding Events

Turbine Building

The turbine building is subject to flooding from a variety of potential sources including the circulating water, service water, condensate/feedwater, component cooling water, turbine building cooling water, demineralized water and fire protection systems as well as the deaerator storage tank. Flow from any postulated ruptures above elevation 100′-0″ flows down to elevation 100′-0″ via floor grating and stairwells. Thus, there will be a negligible contribution from these sources to flooding of the auxiliary

building compartments at elevations 135′-3″ and 117′-6″ via flow under doors. Auxiliary building flooding is bounded by the effects of postulated breaks in the compartments.

The bounding flooding source for the turbine building is a break in the circulating water piping which would result in flooding of the elevation 100′-0″ floor. Flow from this break runs out of the building to the yard through a relief panel in the turbine building west wall and limits the maximum flood level to less than 6 inches. The only area of the auxiliary building which interfaces with the turbine building at elevation 100′-0″ is the valve/piping penetration room. This room could be flooded via flow under the door or backflow through the drains, however the flood level would be less than postulated for a break in the valve/piping penetration room itself.

The waste water system (WWS) sump pumps located in the nonradiologically controlled area of the auxiliary building discharge to the turbine building drain tank. Backflow from the drain tank is prevented as described in Subsection 3.4.1.2.2.2

There is no safety-related equipment in the turbine building. The component cooling water and service water components on elevation 100′-0″, which provide the regulatory treatment of nonsafety-related systems important to support the normal residual heat removal system, are expected to remain functional following a flooding event in the turbine building since the pump motors and valve operators are above the expected flood level.

Annex Building

- **Nonradiologically Controlled Areas**

  The primary sources of flooding in the nonradiologically controlled areas of the annex building are the component cooling water, chilled water and fire protection systems. Water from postulated breaks above elevation 100′-0″ flows primarily through floor drains to the annex building sump that discharges to the turbine building drain tank. Alternate paths include flows to the turbine building via flow under access doors at elevations 135′-3″ and 117′-6″ and flows down to elevation 100′-0″ via stairwells and elevator shaft. Water accumulation at elevation 100′-0″ is minimized by floor drains to the annex building sump and by flow under the access doors leading directly to the yard area. The floors of the annex building are sloped away from the access doors to the nuclear island in the vicinity of the access doors to prevent migration of flood water to the nonradiologically controlled areas of the nuclear island.

  There is no safety-related equipment in the nonradiologically controlled area portion of the annex building. The main ac power system components with regulatory treatment of nonsafety-related systems important missions are located on elevation 117′-6″ in the electrical switchgear rooms, which are separated from potential flood sources. Water from manual fire fighting operations is collected by floor drains discharging to the annex building sump or down a hatch or stairwell to elevation 100′-0″. The non-Class 1E dc and UPS system (EDS) equipment with regulatory treatment of nonsafety-related systems important missions is located on elevation 100′-0″ in separate battery rooms. Water in one of these rooms due to manual fire fighting in the room is collected by floor drains to the annex building sump and by flow under the access doors leading directly to the yard area. This is not expected to affect functionality of equipment in the adjacent rooms.

- **Radiologically Controlled Areas**

  There is no safety-related equipment in the radiologically controlled area portion of the annex building. The primary sources of flooding in the radiologically controlled areas of the annex building are the component cooling water, chilled water and fire protection systems, including an automatic suppression system that protects the containment access corridor. Water from postulated breaks above elevation 100′-0″ drains through floor drains to the radioactive waste

drain system sump in the radiologically controlled area of the auxiliary building or drains to elevation 100′-0″ via stairwells and equipment handling hatches or under access doors to the radiologically controlled area portion of the auxiliary building. Accumulated water at elevation 100′-0″ is minimized by floor drains discharging to the radioactive waste drain system sump or chemical waste tank in the auxiliary building. The contribution of water to the flooding of the radiologically controlled area portion of the auxiliary building is bounded by flooding events which could occur in the auxiliary building.

Radwaste Building

The potential sources of flooding in the radwaste building are the chilled water, hot water, and fire protection systems or from failure of one of the three waste monitor tanks. Flow from postulated breaks is directed to floor drains via a curb/sloped floor around the perimeter to drain to the radioactive waste drain system sump in the radiologically controlled area of auxiliary building. The contribution of water to flooding of the auxiliary building is bounded by flooding events which could occur in the auxiliary building. There are no safety-related systems or components or equipment with regulatory treatment of nonsafety-related systems important missions in the radwaste building.

Diesel Generator Building

The potential source of flooding in the diesel generator building is the fire protection system. There is no safety-related equipment in the diesel generator building. The diesel generator system which has regulatory treatment of nonsafety-related systems important mission has each diesel and associated auxiliaries in a separate compartment. Flooding due to a break in a fire water header is directed to the respective diesel generator building sump and subsequently pumped to the yard oil separator or is drained by gravity to the yard area under the access doors. The equipment in the adjacent diesel generator compartment should remain functional following the event.

3.4.1.3 Permanent Dewatering System

The need for a permanent dewatering system is site specific and is defined as discussed in Subsection 3.4.3.

No permanent dewatering system is required because site groundwater levels are two feet or more below site grade level as described in Subsection 2.4.12.

3.4.2 Analytical and Test Procedures

The AP1000 is designed so that the maximum water levels considered due to natural phenomena or internal flooding do not jeopardize the safety of the plant or the ability to achieve and maintain safe shutdown conditions. The analytical approach in the consideration of external and internal flooding events is described in Subsection 3.4.1.2.

3.4.3 Combined License Information

The site-specific water levels given in Subsection 3.4.1.3 and Section 2.4 satisfy the interface requirements identified in Section 2.4.

3.4.4 References

1. ANSI/ANS-56.11-1988, "Design Criteria for Protection against the Effects of Compartment Flooding in Light Water Reactor Plants."

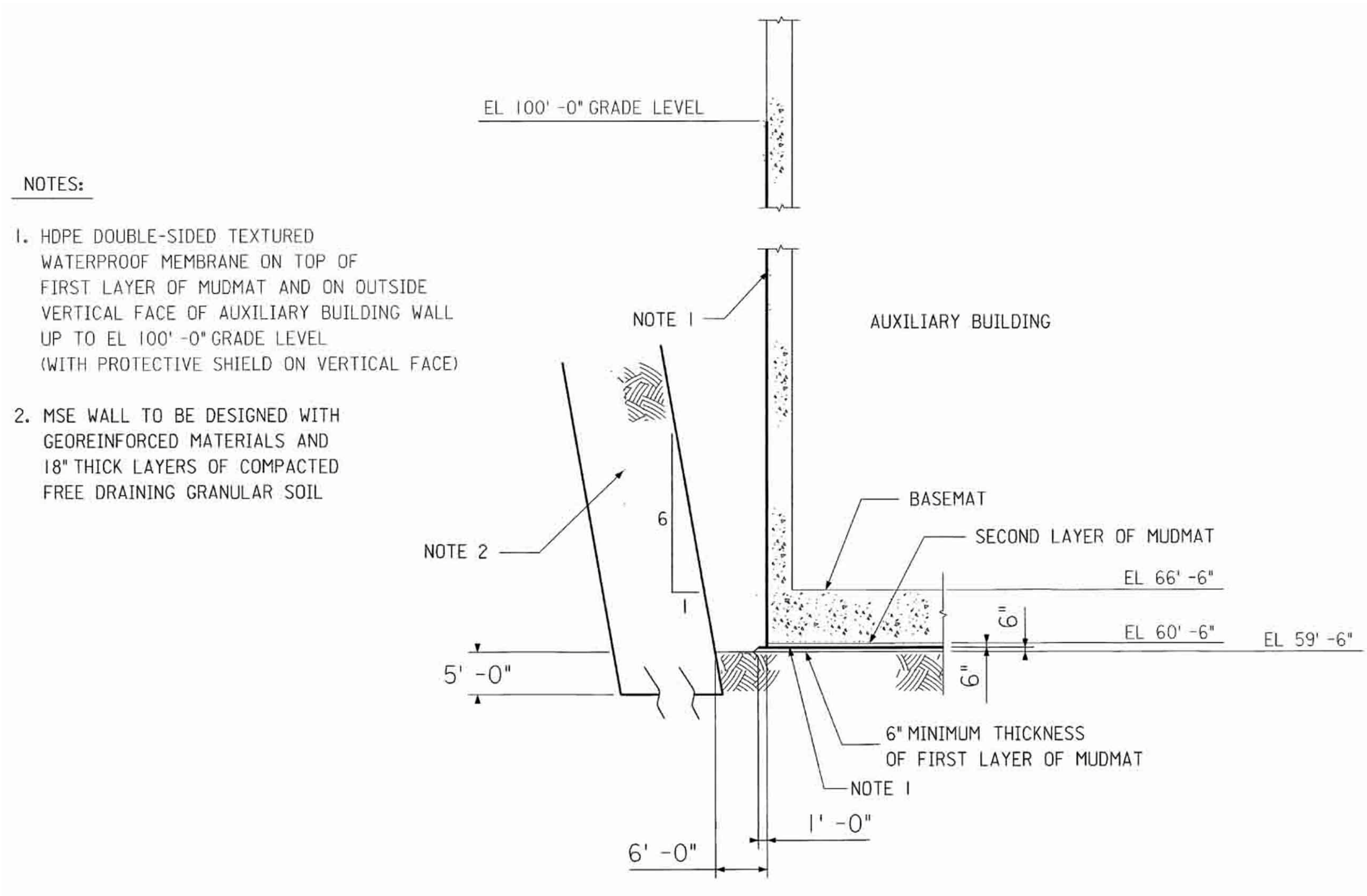


Figure 3.4-1
Typical Details of
Nuclear Island Waterproofing Below Grade

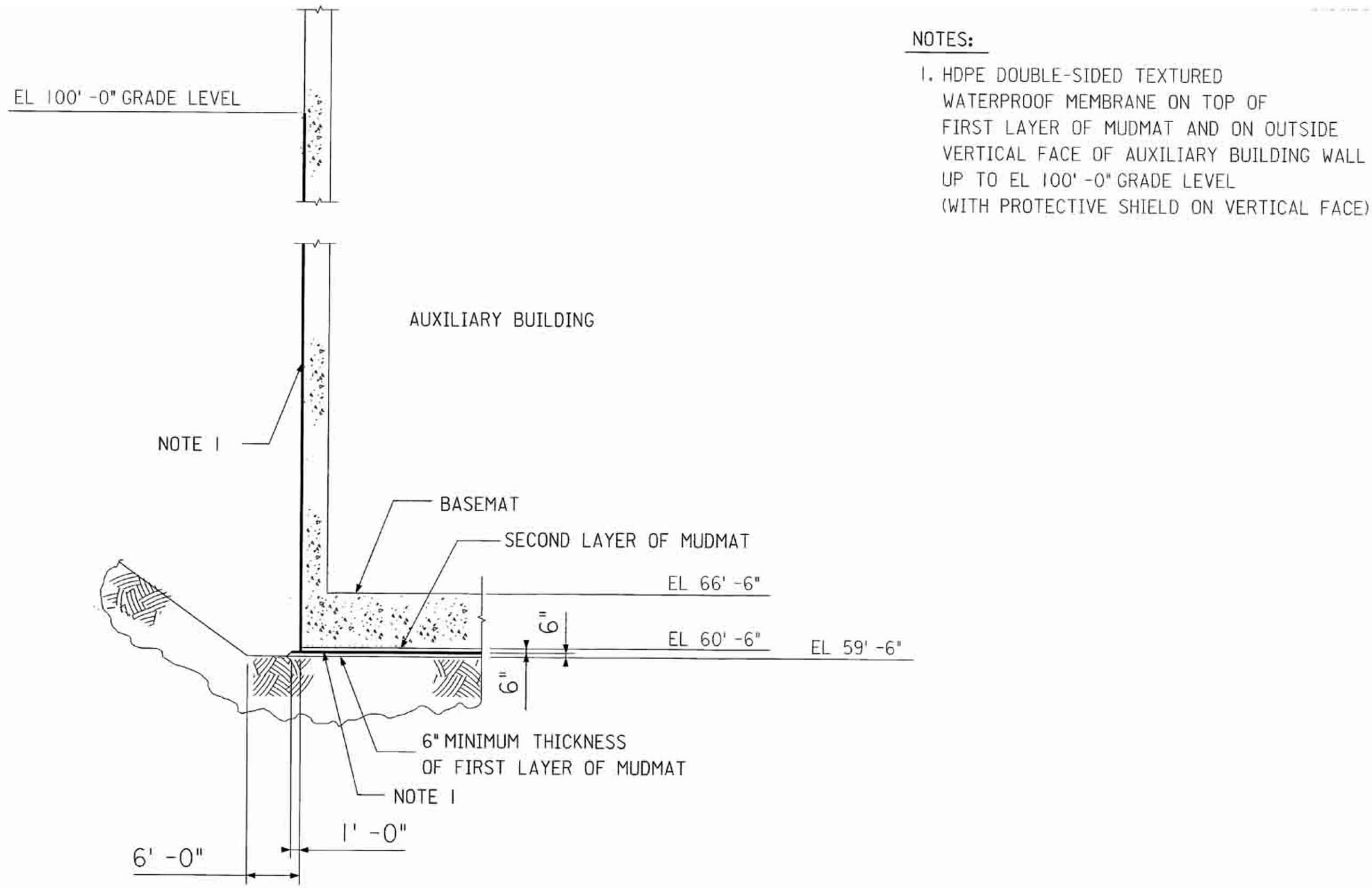


Figure 3.4-2
Typical Details of Nuclear Island Waterproofing Below Grade with Step Back

Figure 3.4-3 Not Used

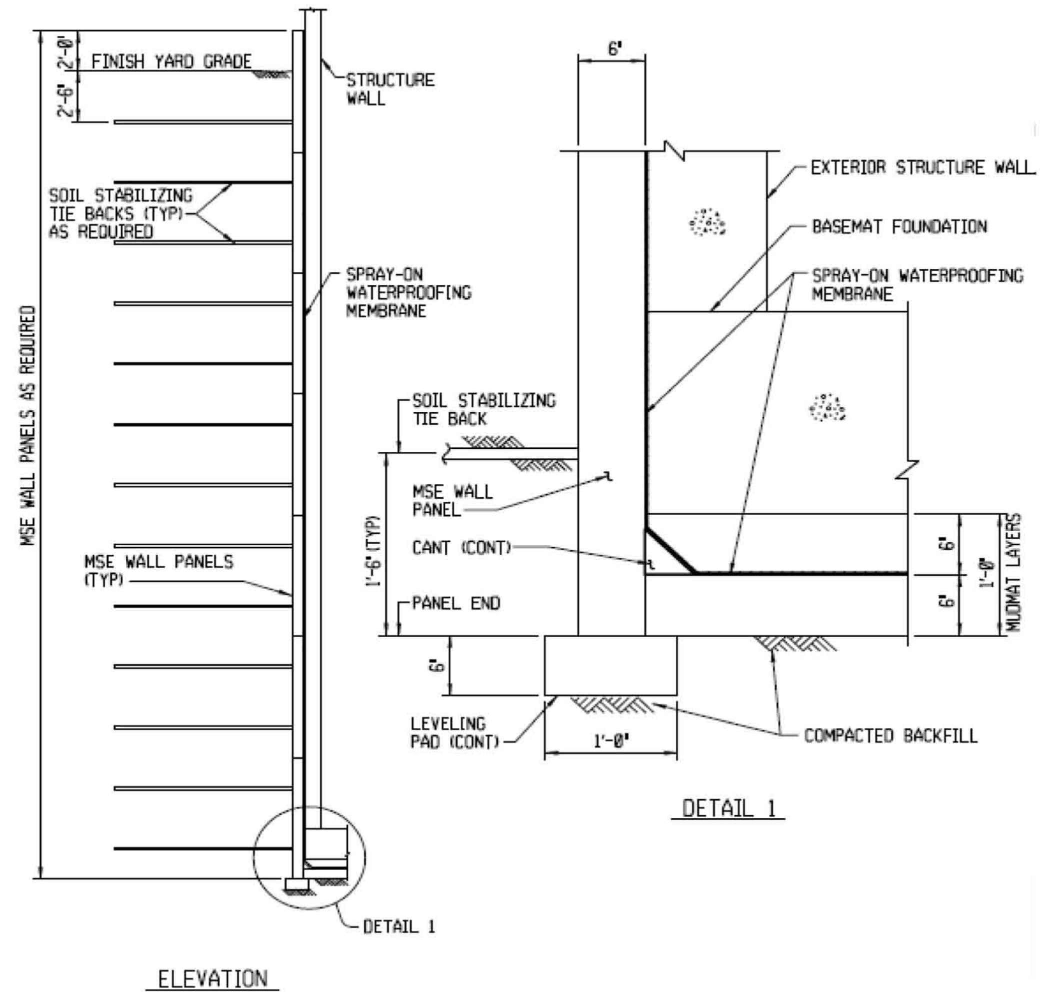


Figure 3.4-4
Typical Details of Membrane Corner Detail at Basemat and Exterior Wall

3.5 Missile Protection

General Design Criterion 4 of Appendix A to 10 CFR 50 requires that structures systems and components important to safety be protected from the effects of missiles. The AP1000 criteria for protection from postulated missiles provide the capability to safely shut down the reactor and maintain it in a safe shutdown condition. The AP1000 criteria also protect the integrity of the reactor coolant system pressure boundary and maintain offsite radiological dose/concentration levels within the limits defined in 10 CFR 50.34.

Missiles may be generated by pressurized components, rotating machinery, and explosions within the plant and by tornadoes or transportation accidents external to the plant. Potential missile hazards are eliminated to the extent practical by minimizing the potential sources of missiles through proper selection of equipment, and by arrangement of structures and equipment in a manner to minimize the potential for damage from missiles. Potential missiles due to failures of nonseismic items are addressed in Subsection 3.7.3.13. Heavy load-drop evaluations are described in Subsection 9.1.5.

The following are definitions for missile protection terminology:

Internally Generated Missile – A mass that may be accelerated by energy sources continuously present on site.

Single Active Failure – Malfunction or loss of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic, or electrical malfunction, but not the loss of component structural integrity.

High-Energy System – Fluid systems that, during normal plant conditions, are operated or maintained pressurized with a maximum operating temperature greater than 200°F and/or a maximum operating pressure greater than 275 psig, as discussed in Subsection 3.6.1.

The following criteria are applied in the identification of missiles and the protection requirements that must be satisfied:

- A missile must not damage structures, systems, or components to the extent that could prevent achieving or maintaining safe shutdown of the plant or result in a significant release of radioactivity.

- A single active component failure is assumed in systems used to mitigate the consequences of the postulated missile and achieve a safe shutdown condition. The single active component failure is assumed to occur in addition to the postulated missile and any direct consequences of the missile. When the postulated missile is generated in one of two or more redundant trains of a dual-purpose safety-related fluid system, which is designed to seismic Category I standards and is capable of being powered from both onsite and offsite sources, a single active component failure need not be assumed in the remaining train(s), or associated supporting trains.

- Walls, partitions, and other items that enclose safety-related systems, or separate redundant trains of safety related equipment, must be constructed so that a postulated missile cannot damage components required to achieve safe shutdown nor damage components required to prevent a release of radioactivity producing offsite doses in excess of 10 CFR 50.34 limits.

- A postulated missile from the reactor coolant system must not cause loss of integrity of the primary containment, main steam, feedwater, or other loop of the reactor coolant system.

- A postulated missile from any system other than the reactor coolant system must not cause loss of integrity of the containment or the reactor coolant system pressure boundary.

- Other plant accidents or severe natural phenomena are not assumed to occur in conjunction with a postulated missile (except for tornado).

- Offsite power is assumed to be unavailable if a trip of the turbine-generator or reactor protection system is a direct consequence of the postulated missile.

- Safe shutdown is accomplished using only safety-related systems with a coincident single active failure, although nonsafety-related systems not affected by the missile are available to support safe shutdown.

- Missiles are postulated to occur where the single failure of a retention mechanism can result in a missile, unless the missile is not considered credible as discussed later. Missiles created by the independent failures of two retention mechanisms are not postulated.

- The energy of postulated missiles produced by rotating components is based on a 120 percent overspeed condition, unless such an overspeed condition is not possible (such as a synchronous motor).

- Equipment required for safe shutdown is located in plant areas separate from potential missile sources wherever practical.

- Spatial separation may be used to demonstrate protection from missile hazards when it is shown that the range and trajectory of the generated missile is less than the distance to or is directed away from the potential target.

The AP1000 passive design minimizes the number of safety-related structures, systems, and components required for safe shutdown. Systems required for safe shutdown are identified in Chapter 7. Safety class structures, systems and components, their location, seismic category, and quality group classifications are given in Section 3.2. General arrangement drawings showing locations of the structures, systems, and components are given in Section 1.2 The areas required for safe shutdown, and the major systems and components housed therein that are required to be protected from internally and externally generated missiles for safe shutdown, are summarized below:

- The containment vessel, including the reactor coolant loop, and passive core cooling system inside containment

- The shield building, including the passive containment cooling system

- Containment penetration areas, including containment isolation valves and Class IE cables

- The control complex including the main control room, reactor protection system, batteries, and dc switchgear

- The spent fuel pit

The AP1000 relies on safety-related systems and equipment to establish and maintain safe shutdown conditions. There are no nonsafety-related systems or components that require protection from missiles.

Evaluations are performed to demonstrate that the criteria are satisfied in the event a credible missile is produced coincident with a single active component failure. These evaluations include the following:

- For those potential missiles considered to be credible, a realistic assessment is made of the postulated missile size and energy, and its potential trajectories.
- Potentially impacted components associated with systems required to achieve and maintain safe shutdown are identified.
- Loss of these potentially impacted components coincident with an assumed single active component failure is evaluated to determine if sufficient redundancy remains to achieve and maintain a safe shutdown condition. If these criteria are satisfied, no further protection is required for the identified missile. If these conditions are not satisfied, additional protective features are incorporated (for example, plant layout is modified, or barriers are added).

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

3.5.1.1.1 Criteria for Missile Prevention

Equipment for the AP1000 is selected to minimize the potential for missiles to be generated. Missiles are postulated as described in Subsection 3.5.1.1.2. The following items are the major equipment selection considerations with regards to missile prevention:

- Safety-related rotating equipment is designed so that the surrounding housings would contain fragments in the event of failure of the rotating parts.
- Valves that have only a threaded connection between the body and the bonnet are not used in high-energy systems. ASME Code, Section III valves with removable bonnets should be of the pressure-seal type or have bolted bonnets.
- Valve stems of valves located in high-energy systems have at least two retention features. In addition to the stem threads, acceptable features include back seats on the stem or a power actuator, such as an air or motor operator.
- Thermowells and other instrument wells, vents, drains, test connections, and other fittings located in high-energy systems are attached to the piping or pressurized equipment by welding. The completed joint should have a greater design strength than the parent metal. Threaded connections in high-energy systems are avoided.
- High-pressure gas cylinders permanently installed in safety-related areas are constructed to the criteria of ASME Code, Section III or Section VIII. Portable and temporary cylinders and cylinders periodically replaced in safety-related areas are constructed and handled in accordance with applicable Department of Transportation requirements for seamless steel cylinders.

3.5.1.1.2 Missile Selection

3.5.1.1.2.1 Missiles not Considered Credible

This subsection describes internally generated missiles (outside of containment) not considered credible. Missiles not considered credible include the following:

- Catastrophic failure of safety-related rotating equipment (such as pumps, fans, and compressors) leading to the generation of missiles is not considered credible. These components are designed to preclude having sufficient energy to move the masses of their rotating parts through the housings in which they are contained. In addition, material characteristics, inspections, quality control during fabrication and erection, and prudent operation as applied to the particular component reduce the likelihood of missile generation.

- Catastrophic failure of nonsafety-related rotating equipment is not considered credible in situations where measures similar to those just described for safety-related rotating equipment are applied to them. Protection from nonsafety-related equipment will normally be provided by separation. In special situations, equipment features may be used to prevent missile formation.

- Provisions to preclude generation of missiles due to failure of the turbine generator are discussed in Subsection 3.5.1.3.

- Missiles originating in non-high-energy fluid systems are not considered credible because these systems have insufficient stored energy.

- The valve bonnets of pressure-seal, bonnet-type valves, constructed in accordance with ASME Code, Section III, are not considered credible missiles. The valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke capturing the bonnet or reducing bonnet energy. Because of the conservative design of the retaining ring of these valves, bonnet ejection is unlikely.

- The valves of the bolted bonnet design, constructed in accordance with ASME Code, Section III, are not considered credible missiles. These bolted bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material according to ASME Code, Section III requirements, and by designing flanges in accordance with applicable code requirements. Even if bolt failure would occur, the likelihood of all bolts experiencing simultaneous complete severance failure is not credible. The widespread use of valves with bolted bonnets, and the low historical incidence of complete severance failure of the bonnet, confirm that bolted valve bonnets are not credible missiles. Safety-relief valves in high energy systems use the bolted bonnet design.

- Valve stems are not considered as credible missiles if at least one feature (in addition to the stem threads) is included in their design to prevent ejection. Valve stems with back seats are prevented from becoming missiles by this feature. In addition, the valve stems of valves with power actuators, such as air- or motor-operated valves, are effectively restrained by the valve actuator. Valve stems of rotary motion valves, such as plug valves, ball valves (except single-seat ball valves) and butterfly valves, as well as diaphragm and bellows type valves are not considered as credible missiles. Because these valves do not have a large reservoir of pressurized fluid acting on the valve stem, there is little stored energy available to produce a missile.

- Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are not considered as credible missiles.

- Thermowells and similar fittings attached to piping or pressurized equipment by welding are not considered as credible missiles where the completed joint has a greater design strength than the parent metal. Such a design makes missile formation not credible. Threaded connections are not used to connect instrumentation to high-energy systems or components.

- Instrumentation such as pressure, level, and flow transmitters and associated piping and tubing are not considered as credible missiles. The quantity of high energy fluid in these instruments is limited and will not result in the generation of missiles. The connecting piping and tubing is made up using welded joints or compression fittings for the tubing. Tubing is small diameter and has only a small amount of stored energy.

- ASME Code, Section III vessel ruptures and ruptures of gas storage vessels constructed without welding using ASME Code, Section VIII criteria are not considered credible due to the conservative design, material characteristics, inspections, quality control during fabrication and erection, and prudent operation.

- Rotating components that operate less than 2 percent of the time are not considered credible sources of missiles. Components that are excluded by this criterion include motors on valve operators and pumps in systems that operate infrequently, such as the chemical and volume control makeup pumps. This exclusion is similar to the exclusion mentioned in Subsection 3.6.1.1, that is, of lines from the high-energy category of lines that have limited operating time in high energy conditions.

- Valves, rotating equipment, vessels, and small fittings not otherwise considered to be credible missiles due to design features or other considerations are not considered to be a potential source of missiles when struck by a falling object.

3.5.1.1.2.2 Explosions

Missiles can potentially be generated by a hydrogen explosion. Missiles that could prevent achieving or maintaining a safe shutdown or result in significant release of radioactivity are precluded by design of the plant systems that use or generate hydrogen.

- The battery compartments are ventilated by a system that is designed to preclude the possibility of hydrogen accumulation. Therefore, a hydrogen explosion in a battery compartment is not postulated.

- Gaseous hydrogen is supplied to the nuclear island from bottles (high-pressure tanks) adjacent to the turbine building and near the nuclear island. The hydrogen is not located in a compartment that contains safety-related systems or components. The hydrogen is supplied to the chemical and volume control system inside the containment. The source of hydrogen is normally isolated from the hydrogen supply line, which is routed into the containment by a normally closed/fail-closed solenoid valve. This quantity would not lead to an explosion even if the entire volume is assumed to remain in the compartment in which it is released.

- Mixing within a compartment is achieved by normal convection caused by thermal forces from hot surfaces and air movement due to operation of HVAC systems. The hydrogen supply line is not routed through compartments that do not have air movement due to HVAC systems.

- The bulk gas plant storage area for the plant gas system (PGS) stores liquid hydrogen for use in generator cooling. This storage area is located sufficiently far from the nuclear island that an explosion would not result in missiles more energetic than the tornado missiles for which the nuclear island is designed. The liquid hydrogen is converted to gas in the storage area and then piped to the generator in the turbine building. The turbine building includes sufficient ventilation to prevent an explosive concentration of hydrogen in the event of a leak.

- A detonation of a flammable vapor cloud (delayed ignition) due to the accidental release of hydrogen from the PGS bulk gas storage area or from the high-pressure hydrogen bottles area would not result in missiles more energetic than the tornado missiles for which the nuclear island is designed.

3.5.1.1.2.3 Missiles to be Considered

The following missiles are considered:

- Nonsafety-related rotating equipment, not excluded above,
- Pressurized components, not excluded above, located in high-energy systems
- High pressure gas storage cylinders that may experience a failure of the outlet pipe or valve if accidentally impacted.

3.5.1.1.2.4 Credible Sources of Internally Generated Missiles (Outside Containment)

The consideration of missile sources outside containment that can adversely affect safety-related structures, systems or components is limited to a few rotating components inside the auxiliary building and a few pressurized components in the chemical volume and control system. The safety-related systems and components needed as described in Section 7.4 to bring the plant to a safe shutdown are located inside the containment shield building and auxiliary building, both of which have thick structural concrete exterior walls that provide protection from missiles generated in other portions of the plant. Safety-related systems and components located in the auxiliary building, including the main control room, are protected from missiles generated in other portions of the auxiliary building by the structural concrete interior walls and floors. Protection against potential missiles from the turbine-generator is discussed in Subsection 3.5.1.3.

Rotating components located inside the auxiliary building that are either safety-related or are constructed as canned motor pumps would contain fragments from a postulated fracture of the rotating elements. These are excluded from evaluation as missile sources. Rotating components used less than 2 percent of the time are also excluded from evaluation as missile sources. This exclusion of equipment that is used for a limited time is similar to the approach used for the definition of high-energy systems. Nonsafety-related rotating equipment in compartments surrounded by structural concrete walls with no safety-related systems or components inside the compartment is not considered a missile source. Rotating equipment with a housing or an enclosure that contains the fragments of a postulated impeller failure is not considered a credible source of missiles. For one or more of these reasons the nonsafety-related rotating equipment inside the auxiliary building is not considered to be a credible missile source. Nonsafety-related rotating equipment in compartments with safety-related systems or components that do not provide other separation features have design requirements for a housing or an enclosure to retain fragments from postulated failures of rotating elements.

The high-energy system inside the auxiliary building that includes pressurized components in the high-energy portions that are constructed to standards other than the ASME Code criteria outlined in Subsection 3.5.1.1.1 is the chemical and volume control system. The high-energy portion of this system inside the auxiliary building that is not constructed to ASME Code criteria outlined in Subsection 3.5.1.1.1 is from the makeup pumps to the containment and system isolation valves. The nonsafety-related, high-energy portion of this system is not required to be protected from missiles. The nonsafety-related, high-energy portion of the chemical and volume control system is not to be considered a missile source. It includes the design features that are outlined above to exclude components from consideration as missile sources. These considerations include features such as a pump housing or enclosure that contains fragments of a postulated impeller fracture, valve design

requirements, vessel design requirements, or enclosure requirements. See Table 3.6-1 for a list of the high-energy systems.

Falling objects (i.e. gravitational missiles) heavy enough to generate a secondary missile are postulated as a result of movement of a heavy load or from a nonseismically designed structure, system, or component during a seismic event. Movements of heavy loads are controlled to protect safety-related structures, systems, and components, see Subsection 9.1.5. Safety-related structures, systems, or components are protected from nonseismically designed structures, systems, or components or the interaction is evaluated. See Subsection 3.7.3.13 for additional discussion on the interaction of other systems with Seismic Category I systems. Valves, rotating equipment, vessels, and small fittings not otherwise considered to be credible missiles due to design features or other considerations are not considered to be a potential source of missiles when struck by a falling object. The outlet pipes and valves for the air storage bottles for the main control room are constructed to the ASME Code, Section III, requirements and are designed for seismic loads. The attached pipes and valves are not credible missile sources due to an accidental impact. The air storage bottles are located within a structural steel frame and are in an area with no activity directly above. For the reasons noted above, secondary missiles are not considered credible missiles.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Selection of equipment for the AP1000 considers provisions to minimize the potential for missiles to be generated. The considerations previously discussed in Subsection 3.5.1.1 are also applicable to equipment inside the containment.

3.5.1.2.1 Missile Selection

3.5.1.2.1.1 Missiles not Considered Credible

Potential missiles are not considered credible when sufficient energy is not available to produce the missile, or by design the probability of creating a missile is negligible. The following are not considered credible sources of internally generated missiles:

- Reactor coolant pump design requirements are established so that any failure of the rotating parts would be retained within the casing at specified overspeed conditions. This is discussed in Subsection 5.4.1.3.6.

- Catastrophic failure of rotating equipment such as pumps, fans, and compressors leading to the generation of missiles is not considered credible as described previously in Subsection 3.5.1.1.2.

- Failure of the reactor vessel, steam generators, pressurizer, core makeup tanks, accumulators, reactor coolant pump castings, passive residual heat exchangers, and piping leading to the generation of missiles is not considered credible. This is due to the material characteristics, preservice and inservice inspections, quality control during fabrication, erection and operation, conservative design, and prudent operation as applied to the particular component.

- Gross failure of a control rod drive mechanism housing, sufficient to create a missile from a piece of the housing or to allow a control rod to be ejected rapidly from the core, is not considered credible. This is because of the same reasons listed above for the reactor vessel and other components and is based on the following:

  – The control rod drive mechanisms are shop hydrotested to 125 percent of system design pressure.

- The housings are hydrotested to 125 percent of system design pressure after they are installed on the reactor vessel to the head adapters. They are checked again during the hydrotest of the completed reactor coolant system.
- The housings are made of Type 304 or 316 stainless steel, which exhibits excellent notch toughness.
- Stress levels in the mechanism are not affected by system thermal transients at power or by thermal movement of the coolant loops.
- The welds in the pressure boundary of the control rod drive mechanism meet the same design, procedure, examination, and inspection requirements as the welds on other ASME Code, Section III, Class 1 components.
- A nonmechanistic control rod ejection is considered in the safety analyses in Chapter 15 and the design transients in Subsection 3.9.1.1. The integrated head package and control rod drive mechanisms are not designed for the dynamic effects of a missile generated by a rupture of the control rod housing.

- Valves, valve stems, nuts and bolts, and thermowells in high-energy fluid systems and missiles originating in non-high-energy fluid systems are not considered credible missiles as discussed previously in Subsection 3.5.1.1.1.

3.5.1.2.1.2 Explosions

Missiles can potentially be generated by a hydrogen explosion. Missiles that could prevent achieving or maintaining a safe shutdown or result in significant release of radioactivity are precluded by design of the plant systems that use or generate hydrogen.

- Gaseous hydrogen is supplied to the nuclear island from bottles (high-pressure tanks) adjacent to the turbine building and near the nuclear island. The hydrogen is supplied to the chemical and volume control system inside the containment. The source of hydrogen is normally isolated from the hydrogen supply line, which is routed into the containment by a normally closed/fail-closed solenoid valve. The quantity of hydrogen that could be released inside the containment in the event of a failure of the hydrogen supply line is limited due to a flow restricting device and a normally closed/fail-closed solenoid valve isolating the system from the source of hydrogen. The quantity of hydrogen gas to be injected into the reactor coolant system is limited to the duration that the solenoid valve in the chemical and volume control system is open (which is opened for a calculated time via a manual plant control system entry) and the maximum gas flow allowed by the flow restriction. It is assumed that this quantity would be less than the contents of a single bottle. This quantity would not lead to an explosion even if the entire volume is assumed to remain in the compartment in which it is released. Mixing within a compartment is achieved by normal convection caused by thermal forces from hot surfaces and air movement due to operation of HVAC systems.

3.5.1.2.1.3 Missiles to be Considered

The following missiles are considered:

- Nonsafety related rotating equipment, not excluded above,
- Pressurized components, not excluded above, located in high-energy systems

3.5.1.2.1.4 Evaluation of Internally Generated Missiles (Inside Containment)

The consideration of credible missile sources inside containment that can adversely affect safety-related structures, systems, or components is limited to a few rotating components. The safety-related systems and components needed to bring the plant to a safe shutdown are inside the containment shield building and auxiliary building both of which have thick structural concrete exterior walls that provide protection from missiles generated in other portions of the plant.

Rotating components inside containment that are either safety-related or are constructed as sealless pumps would contain fragments from a postulated fracture of the rotating elements and are excluded from evaluation as missile sources. Rotating components in use less than 2 percent of the time are also excluded from evaluation as missile sources. This exclusion of equipment that is used for a limited time is similar to the approach used for the definition of high-energy systems. This includes the reactor coolant drain pumps, the containment sump pumps and motors for valve operators, and mechanical handling equipment. Non-safety-related rotating equipment in compartments surrounded by structural concrete walls with no safety-related systems or components inside the compartment is not considered a missile source. Rotating equipment with a housing or an enclosure that contains the fragments of a postulated impeller failure is not considered a credible source of missiles. For one or more of these reasons the nonsafety-related rotating equipment inside containment is considered not to be a credible missile source. Non-safety-related rotating equipment in compartments with safety-related systems or components that do not provide other separation features has design requirements for a housing or an enclosure to retain fragments from postulated failures of rotating elements.

The high-energy portions of high-energy systems inside the containment shield building except for a portion of the chemical and volume control system are constructed to the requirements of the ASME Code, Section III. The nonsafety-related, high-energy portion of the chemical and volume control system between the inside containment isolation valves and the outermost reactor coolant system isolation valves is not required to be protected from missiles and is not to be considered a missile source. It includes design features outlined above to exclude components from consideration as missile sources. In addition most of the nonsafety-related portion of the chemical and volume control system is contained in a compartment located away from safety-related equipment. See Table 3.6-1 for a list of the high-energy systems.

Falling objects heavy enough to generate a secondary missile are postulated as a result of movement of a heavy load or from a nonseismically designed structure, system, or component during a seismic event. Movements of heavy loads are controlled to protect safety-related structures, systems, and components (see Subsection 9.1.5). Design and operational procedures of the polar crane inside containment precludes dropping a heavy load. Additionally, movements of heavy loads inside containment occur during shutdown periods when most of the high-energy systems are depressurized. Valves, rotating equipment, vessels, and small fittings not otherwise considered to be credible missiles due to design features or other considerations are not considered to be a potential source of missiles when struck by a falling object. Secondary missiles are not considered credible. Striking a component with a falling object will not generate a secondary missile if design of the component precludes generation of missiles due to pressurization of the component. Safety-related structures, systems, or components are protected from nonseismically designed structures, systems, or components or the interaction is evaluated. Nonsafety-related equipment that could fall and damage safety-related equipment during an earthquake is classified as seismic Category II and is designed and supported to preclude such failure. See Subsection 3.7.3.13 for additional discussion on the interaction of other systems with Seismic Category I systems. There are no high-pressure gas storage cylinders inside the containment shield building. For the reasons noted above, secondary missiles are not considered credible missiles.

3.5.1.3 Turbine Missiles

The turbine generator is located north of the nuclear island with its shaft oriented north-south. In this orientation, the potential for damage from turbine missiles is negligible. Safety-related structures, systems and components are located outside the high-velocity, low-trajectory missile strike zone, as defined by Regulatory Guide 1.115. Thus, postulated low-trajectory missiles cannot directly strike safety-related areas.

The turbine and rotor design is described in Section 10.2. Protection is provided by the orientation of the turbine-generator and by the use of robust turbine rotors as described in Section 10.2. The rotor design, manufacturing, and material specification and the inspections recommended for the AP1000 provide an acceptably very low probability (see Subsection 10.2.2) of missile generation. Turbine rotor integrity is discussed in Subsection 10.2.3. This discussion includes fatigue and fracture analysis, material selection, and the maintenance program requirements.

The potential for a high-trajectory missile to impact safety-related areas of the AP1000 is less than 10^{-7}. Based on this very low probability, the potential damage from a high-trajectory missile is not evaluated. The probability of an impact in the safety-related areas is the product of the probability of missile generation from the turbine; the probability, assuming a turbine failure, that a high-trajectory missile would land within a few hundred feet from the turbine (10^{-7} per square foot); and the area of the safety-related area. In the AP1000, the safety-related area is contained within the containment shield building and the auxiliary building.

The potential for a turbine missile from another AP1000 plant in close proximity has been considered. As noted in Subsection 10.2.2, the probability of generation of a turbine missile (or P1 as identified in SRP 3.5.1.3) is less than 1 x 10^{-5} per year. This missile generation probability (P1) combined with an unfavorable orientation P2 x P3 conservative product value of 10^{-2} (from SRP 3.5.1.3) results in a probability of unacceptable damage from turbine missiles (or P4 value) of less than 10^{-7} per year per plant which meets the SRP 3.5.1.3 acceptance criterion and the guidance of Regulatory Guide 1.115. Thus, neither the orientation of the side-by-side AP1000 turbines nor the separation distance is pertinent to meeting the turbine missile generation acceptance criterion. In addition, the shield building and auxiliary building walls, roofs, and floors, provide further conservative, inherent protection of the safety-related SSCs from a turbine missile.

The orientation of the Units 1 and 2 turbines has been evaluated and Vogtle Units 3 and 4 are located outside of the low trajectory strike zones as described in Regulatory Guide 1.115. Therefore, there is no potential for a turbine missile from Units 1 and 2 to impact Units 3 and 4.

The turbine system maintenance and inspection program is discussed in Subsection 10.2.3.6.

3.5.1.4 Missiles Generated by Natural Phenomena

Tornado missiles are defined in accordance with Standard Review Plan, Subsection 3.5.1.4. The velocities are adjusted to the maximum wind velocity defined in Section 3.3. The following missiles are postulated:

- A massive high-kinetic-energy missile, which deforms on impact. It is assumed to be a 4000-pound automobile impacting the structure at normal incidence with a horizontal velocity of 105 mph or a vertical velocity of 74 mph. This missile is considered at all plant elevations up to 30 feet above grade. In addition, to consider automobiles parked within half a mile of the plant at higher elevations than the plant grade elevation, the evaluation of the automobile missile is considered at all plant elevations up to the junction of the outer wall of the passive containment cooling water storage tank with the roof of the shield building. This elevation is

approximately 193 feet above grade. This evaluation bounds sites with automobiles parked within half a mile of the shield building and auxiliary building at elevations up to the equivalent of 163 feet above grade.

- A rigid missile of a size sufficient to test penetration resistance. It is assumed to be a 275 pound, eight inch armor-piercing artillery shell impacting the structure at normal incidence with a horizontal velocity of 105 mph or a vertical velocity of 74 mph.
- A small rigid missile of a size sufficient to just pass through any openings in protective barriers. It is assumed to be a one inch diameter solid steel sphere assumed to impinge upon barrier openings in the most damaging direction at a velocity of 105 mph.

In addition to the missile spectrum specified above, the impact of tornado-driven sheet metal siding on the shield building is evaluated. The evaluation considers siding representative of the siding used on the turbine building, radwaste building, diesel generator building, and portions of the annex building. The evaluation considers a flat steel sheet, which bounds the corrugated siding design used on the buildings adjacent to the nuclear island.

3.5.1.5 Missiles Generated by Events Near the Site

As described previously in Section 2.2, the site interface is established to address site specific missiles as discussed in Subsection 3.5.4. The AP1000 missile interface criteria are based on the tornado missiles described in Subsection 3.5.1.4. Additional analyses are required to evaluate other site specific missiles.

The primary access point, administrative building, communications support center, warehouse and shops, engineering and administrative building, maintenance support building and miscellaneous structures are common structures that are located at a nuclear power plant. They are of similar design and construction to those that are typical at nuclear power plants. Therefore, any missiles resulting from a tornado-initiated failure are not more energetic than tornado missiles postulated for design of the AP1000. Additionally, there are no other structures adjacent to the nuclear island other than the turbine building, annex building, radwaste building and passive containment cooling ancillary water storage tank.

In accordance with Subsection 2.2.3, the effects of explosions have been evaluated and it has been determined that the overpressure criteria of Regulatory Guide 1.91 is not exceeded. Consistent with Regulatory Guide 1.91, the effects of blast-generated missiles will be less than those associated with the blast overpressure levels considered; therefore, no further evaluation of blast-generated missiles is required.

3.5.1.6 Aircraft Hazards

As described previously in Section 2.2, the site interface is established to address aircraft hazards as discussed in Subsection 3.5.4. The AP1000 missile interface criteria are based on the tornado missiles described in Subsection 3.5.1.4. Additional analyses are required to evaluate other site specific missiles. Aircraft crash probability, and the effects of this hazard on the plant, is determined as described in Section 2.2.

Airports and airways in the VEGP site vicinity are discussed in Subsection 2.2.2.6. Aircraft hazards related to these airports and airways (shown in Figure 3.5-201) have been evaluated in accordance with Regulatory Standard 002, *Processing Applications for Early Site Permits*, May 2004 (RS-002), and NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, Draft Revision 3, 1996 (NUREG-0800), Subsection 3.5.1.6.

3.5.1.6.1 Airports

RS-002 acceptance criteria provide a distance threshold for evaluating aircraft hazards due to nearby airports.

All airports in the VEGP site vicinity are greater than 10 mi from the site. The hazard probability for these airports is considered acceptable if the projected annual number of operations is less than 1,000 D^2, where D is the site-to-airport distance.

Bush Field is the closest (17 mi) and largest commercial airport in the VEGP site vicinity. The Federal Aviation Administration (FAA) (Reference 201) has projected the number of aircraft that will be in operation at Bush Field for every year up to 2025 for each of the following four types of aircraft: general aviation, air taxi and commuter, commercial air carrier, and military. The projected flight data (which include landings and takeoffs) are provided in Table 3.5-201. As noted in the table, the total number of projected aircraft operations is substantially less than 1,000 D^2 (289,000).

The other airports in the vicinity are much smaller than Bush Field. Since they are all at least 10 mi from the VEGP site, their aircraft hazard threshold is greater than 100,000 operations, which significantly exceeds their annual traffic.

As discussed in Subsection 2.2.2.6.1, a small unimproved grass airstrip is located immediately north of the VEGP site (north of Hancock Landing Road and west of the Savannah River). This privately owned and operated airstrip has a 1,650-foot turf runway oriented 80° East – 260° West. The airstrip is for personal use and the associated traffic consists only of small single-engine aircraft. In addition, there is a small helicopter landing pad on the VEGP site. This facility exists for corporate use and for use in case of emergency. The traffic associated with either of these facilities may be characterized as sporadic. Due to the small amount and the nature of the traffic, these facilities do not present a safety hazard to the VEGP site.

3.5.1.6.2 Airway V185

The VEGP site is approximately 1.5 mi east of the centerline of Federal Airway V185, which runs between Augusta and Savannah. A more detailed review of aircraft hazards was performed because the VEGP site is within the 2 statute mile limit. This review is summarized below.

Airways are typically used by commercial flights and by general aviation for inclement weather and nighttime operations. In general, military aircraft do not use the federal airways. To be allowed to fly in a federal airway, an aircraft needs to have the proper communication equipment and the pilot needs to have specific qualifications. In addition, most general aviation flights do not use a federal airway in favorable weather conditions. When these factors are considered, along with the fact that there are no regularly scheduled direct commercial flights between Augusta and Savannah, it is expected that the total number of aircraft using Airway V185 is relatively small.

Although the FAA does not maintain records of air traffic in Airway V185, informal communications with air traffic control personnel at the Augusta airport revealed that the southeast quadrant of the air space around the airport (of which Airway V185 is a part) has the least air traffic compared to the other quadrants and that the total traffic in Airway V185 is a fraction of the total operations into and out of the Augusta airport.

Because of the unavailability of traffic data for Airway V185, the following evaluation calculates the maximum number of airway flights per year above which the acceptance guideline probability of 10^{-7} per year contained in RS-002 and NUREG-0800 is exceeded. Regulation 14 CFR 71 provides the criteria for determining the width of the airway. It is 4 nautical miles on either side of the centerline, for a total width of 8 nautical miles (9.2 mi).

$P_{FA} = C \times N \times A / W$

where:

P_{FA} = probability per year of an aircraft crashing into a VEGP Units 3 and 4 safety-related structure, 1×10^{-7}

C = in-flight crash rate per mile for aircraft using airway = 4×10^{-10} (RS-002)

N = number of flights per year along the airway

A = effective area of plant or site area in square miles, see below

W = airway width, 9.2 mi

By rearranging this equation, the maximum number of flights corresponding to the acceptance guideline probability of 10^{-7} may be calculated.

NUREG-0800 and RS-002 also provide alternate guidance on the acceptable method for calculating area *A*. RS-002 specifies the use of the site area because, for ESP Applications where the type of power plant has not been selected, the plant cross-sectional area cannot be defined. However, because the Westinghouse AP1000 design has been selected, the effective area of the plant was used in this analysis.

The effective plant area (*A*) depends on the length, width, and height of the facility, as well as the aircraft's wingspan, skid distance, and impact angle (Reference 203).

The safety-related structures of the AP1000 design include only the containment and the auxiliary building; the remainder of the structures is not safety related. The AP1000 containment height is about 234 ft above grade, and the diameter is about 146 ft (Reference 204).

For traffic in Airway V185, the fractions of the types of aircraft using the airway were assumed to be the same as the fractions of the types of aircraft using Bush Field. Representative values for wingspan, skid distance, and impact angle for each aircraft type follow those suggested in (Reference 203). For military aviation, large aircraft are conservatively used in the estimates. The effective areas for general aviation, air taxi and commuter, commercial air carrier, and military aircraft are 0.025, 0.061, 0.073, and 0.086 sq mi, respectively. Using these effective areas and the fractions of aircraft types (52.9, 29.3, 12.8, and 5 percent for general aviation, air taxi and commuter, commercial air carrier, and military aircraft, respectively), the average of the weighted effective plant area, 0.045 mi^2, is determined for the calculation.

Among the representative wingspans, the large military aircraft has the longest wingspan of 223 ft (Reference 203). The physical separation of the new reactor buildings is about 650 ft. Since this distance is longer than the largest representative wingspan (223 ft), the estimate of the effective area involves only one unit. In addition, Subsection 3.5.1.6 of NUREG-0800 also suggests the use of an effective area of one unit of the plant.

To reach the permissible crash probability of 1×10^{-7}, the total number of flights traveling along Airway V185 would need to be about 51,100 per year. This value is higher than the total of all projected itinerant flights for 2025 at Bush Field (see Table 3.5-201).

Although the flight data associated with Airway V185 are not available from the FAA, the number of flights in this airway is expected to be only a fraction of the total Bush Field flights. Therefore, the presence of Airway V185 is not a safety concern for the VEGP site.

3.5.2 Protection from Externally Generated Missiles

Systems required for safe shutdown are protected from the effects of missiles. These systems are identified in Section 7.4. Protection from external missiles, including those generated by natural phenomena, is provided by the external walls and roof of the Seismic Category I nuclear island structures. The external walls and roofs are reinforced concrete. The structural design requirements for the shield building and auxiliary building are outlined in Subsection 3.8.4. Openings through these walls are evaluated on a case-by-case basis to provide confidence that a missile passing through the opening would not prevent safe shutdown and would not result in an offsite release exceeding the limits defined in 10 CFR 50.34. The evaluation of site-specific hazards for external events that may produce missiles more energetic than tornado missiles is discussed in Subsection 2.2.1.

Evaluation of turbine missiles is provided in Subsection 3.5.1.3. Evaluation of tornado missiles is provided in Subsection 3.5.1.4. Conformance with regulatory guide recommendations is provided in Appendix 1A.

3.5.3 Barrier Design Procedures

Missile barriers and protective structures are designed to withstand and absorb missile impact loads to prevent damage to safety-related components.

Formulae used for missile penetration calculations into steel or concrete barriers are the Modified National Defense Research Committee (NDRC) formula for concrete and either the Ballistic Research Laboratory (BRL) or Stanford formulae for steel.

Concrete (Modified NDRC Formula)

$$x = \left[4\,KNWd\left(\frac{V}{1000\,d}\right)^{1.8}\right]^{0.5} \qquad \text{for } \frac{x}{d} \le 2.0$$

$$x = KNW\left(\frac{V}{1000\,d}\right)^{1.8} + d \qquad \text{for } \frac{x}{d} > 2.0$$

where

x	=	penetration depth, inches
W	=	missile weight, lbs
d	=	missile diameter, inches
N	=	missile shape factor = 1.0
V	=	impact velocity, feet/sec
K	=	experimentally obtained material coefficient for penetration = $\frac{180}{\sqrt{f_c'}}$
fc′	=	concrete compressive strength

Scabbing thickness, t_s, and perforation thickness, t_p is given by:

$$\frac{t_s}{d} = 2.12 + 1.36\frac{x}{d} \qquad \text{for } 0.65 \le \frac{x}{d} \le 11.75$$

$$\frac{t_s}{d} = 7.91\left(\frac{x}{d}\right) - 5.06\left(\frac{x}{d}\right)^2 \qquad \text{for } \frac{x}{d} \le 0.65$$

$$\frac{t_p}{d} = 1.32 + 1.24\frac{x}{d} \qquad \text{for } 1.35 \le \frac{x}{d} \le 13.5$$

$$\frac{t_p}{d} = 3.19\,(\frac{x}{d}) - 0.718\,(\frac{x}{d})^2 \qquad \text{for } \frac{x}{d} \le 13.5$$

Steel (Stanford Formula)

$$\frac{E}{D} = \frac{S}{46{,}500}\left(16{,}000\,T^2 + 1{,}500\,\frac{W}{W_s}\,T\right)$$

Where:

E = critical kinetic energy required for perforation, foot pounds

D = effective missile diameter, inches

S = ultimate tensile strength of the target (steel plate), pounds per square inch

T = target plate thickness, inches

W = length of a square side between rigid supports, inches

Ws = length of a standard window, 4 inches

The ultimate tensile strength is directly reduced by the amount of bilateral tension stress already in the target. The equation is good within the following ranges:

$0.1 < T/D < 0.8$,

$0.002 < T/L < 0.05$,

$10 < L/D < 50$,

$5 < W/D < 8$,

$8 < W/T < 100$,

$70 < V < 400$

Where:

L = missile length, inches

V = impact velocity, feet/second

Steel (BRL Formula)

$$t_p = \frac{(E_k)^{2.3}}{672D}$$

Where:

t_p	=	steel plate thickness for threshold of perforation, inches
D	=	equivalent missile diameter, inches
E_k	=	missile kinetic energy, foot pounds
	=	$M V^2/2$
M	=	mass of the missile, lb-sec^2/ft.

In using the Modified NDRC, BRL and Stanford formulae for missile penetration, it is assumed that the missile impacts normal to the plane of the wall on a minimum impact area and, in the case of reinforced concrete, does not strike the reinforcing. Due to the conservative nature of these assumptions, the minimum thickness required for missile shields is taken as the thickness just perforated.

Structural members designed to resist missile impact are designed for flexural, shear, and buckling effects using the equivalent static load obtained from the evaluation of structural response. Stress and strain limits for the equivalent static load comply with applicable codes and Regulatory Guide 1.142, and the limits on ductility of steel structures as given in Subsection 3.5.3.1. The consequences of scabbing are evaluated if the thickness is less than the minimum thickness to preclude scabbing.

The thicknesses of the exterior walls above grade and of the roof of the nuclear island are 24 inches and 15 inches, respectively. The roof is constructed using left-in-place metal deck. These thicknesses exceed the minimum thicknesses for Region II tornado missiles specified in Standard Review Plan 3.5.3.

3.5.3.1 Ductility Factors for Steel Structures

Ductility factors for the design of steel structures are as follows:

- For tension due to flexure, $\mu \leq 10.0$
- For columns with slenderness ratio (L/r) equal to or less than 20, $\mu \leq 1.3$
- For columns with slenderness ratio greater than 20, $\mu \leq 1.0$
  Where: L = effective length of the member
  r = the least radius of gyration
- For members subjected to tension, $\mu \leq .5*(e_u/e_y)$
  Where: e_u = ultimate strain
  e_y = yield strain

3.5.4 Combined License Information

The evaluation for those external events that produce missiles that are more energetic than the tornado missiles postulated for design of the AP1000 is addressed in APP-GW-GLR-020 (Reference 1).

In addition, the VEGP site satisfies the site interface criteria for wind and tornado (see Subsections 3.3.1.1, 3.3.2.1 and 3.3.2.3) and will not have a tornado-initiated failure of structures and components within the applicant's scope that compromises the safety of AP1000 safety-related structures and components (see also Subsection 3.3.3).

Subsection 1.2.2 discusses differences between the plant specific site plan (see Figure 1.1-202) and the AP1000 typical site plan shown in Figure 1.2-2.

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

Missiles caused by external events separate from the tornado are addressed in Subsections 3.5.1.3, 3.5.1.5, and 3.5.1.6.

3.5.5 References

1. APP-GW-GLR-020, "Wind and Tornado Site Interface Criteria," Westinghouse Electric Company LLC.

201. **(APO 2006)** *APO Terminal Area Forecast Summary Report*, Federal Aviation Administration, http://www.apo.data.faa.gov/wtaf/, issued February 2006, accessed 5/2/2006.

202. **(Atlanta 2005)** *Atlanta Sectional Aeronautical Chart*, 74th Edition, U.S. Department of Transportation, Federal Aviation Administration, March 17, 2005.

203. **(DOE 1996)** *Accident Analysis for Aircraft Crash into Hazardous Facilities*, DOE Standard, DOE-STD-3014-96, US Department of Transportation, October 1996.

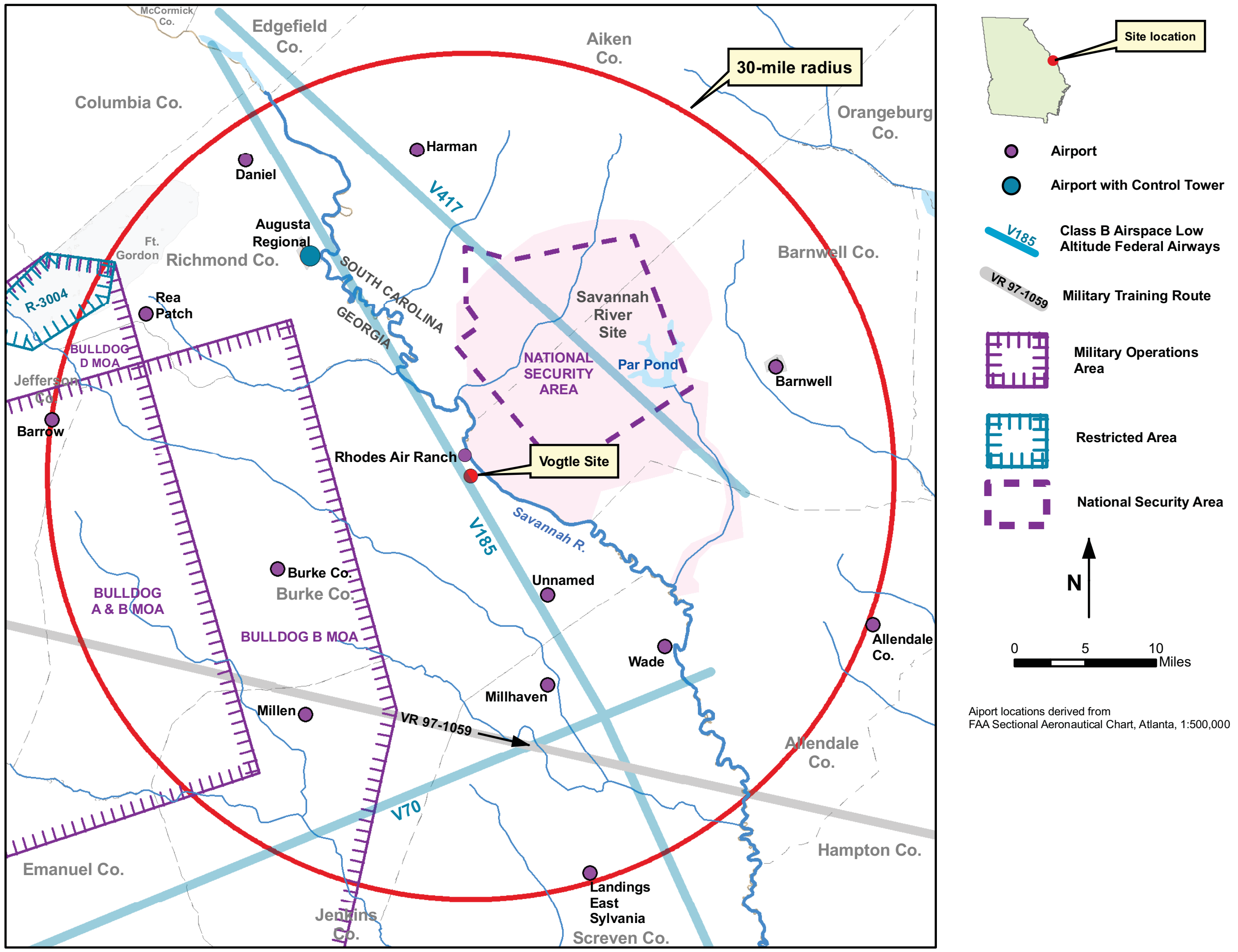
204. **(Westinghouse 2001)** *Nuclear Island General Arrangement, AP1000 Advanced Passive Light Water Reactor*, Rev. 0, Section B-B, DCD Number APP 1000 P2 902, Westinghouse Electric Company, 08/06/2001.

Table 3.5-201
Augusta APO Terminal Area Forecast Summary Report – Itinerant Operations

Year	General Aviation	Air Taxi & Commuter	Commercial Air Carrier	Military	Total
1990	22,023	14,941	6,495	4,522	47,981
1991	19,175	9,462	6,576	3,242	38,455
1992	17,872	9,393	7,196	3,221	37,682
1993	16,902	8,821	6,455	4,068	36,246
1994	16,896	5,961	6,473	3,727	33,057
1995	16,597	8,876	5,024	3,511	34,008
1996	17,016	9,325	4,225	2,780	33,346
1997	18,995	8,304	4,599	2,561	34,459
1998	19,611	7,518	5,028	2,271	34,428
1999	22,653	6,954	5,183	2,841	37,631
2000	21,975	6,663	4,969	3,354	36,961
2001	19,961	7,378	4,929	2,954	35,222
2002	20,085	7,164	4,286	3,082	34,617
2003	17,622	9,058	4,393	2,843	33,916
2004	18,658	9,441	4,934	2,528	35,561
2005	13,307	8,226	4,585	1,799	27,917
2006	13,618	8,328	4,585	1,799	28,330
2007	13,937	8,432	4,585	1,799	28,753
2008	14,263	8,537	4,585	1,799	29,184
2009	14,597	8,644	4,585	1,799	29,625
2010	14,939	8,751	4,585	1,799	30,074
2011	15,288	8,860	4,585	1,799	30,532
2012	15,646	8,971	4,585	1,799	31,001
2013	16,012	9,083	4,585	1,799	31,479
2014	16,387	9,196	4,585	1,799	31,967
2015	16,611	9,310	4,585	1,799	32,305
2016	16,837	9,426	4,585	1,799	32,647
2017	17,067	9,544	4,585	1,799	32,995
2018	17,300	9,663	4,585	1,799	33,347
2019	17,536	9,783	4,585	1,799	33,703
2020	17,776	9,905	4,585	1,799	34,065
2021	18,018	10,028	4,585	1,799	34,430
2022	18,264	10,153	4,585	1,799	34,801
2023	18,514	10,280	4,585	1,799	35,178
2024	18,766	10,408	4,585	1,799	35,558
2025	19,023	10,538	4,585	1,799	35,945

Source: Reference 201

Table 3.5-202
Deleted in Revision 2



Source: Reference 202

Figure 3.5-201
Airports Within 30 Miles of Vogtle Facility

3.6 Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping

The effects of a postulated pipe rupture in the AP1000 are of several types. This section considers the effects that are localized to the area of the break and are a result of the dynamic effects of the pipe rupture including jet impingement, pipe whip, subcompartment pressurization, and fluid system decompression. This section describes the evaluation of the potential for and effects of these dynamic effects. It describes measures taken to protect systems and equipment from dynamic effects of pipe rupture when necessary. This section also considers the effects of spray wetting and flooding from pipe ruptures and cracks.

Chapters 6 and 15 discuss the response of the system to changes in flow and pressure and loss of coolant and the response of the containment to the pressure and temperature changes. Pressure due to a break in a high energy line in the auxiliary building is vented into an adjacent building or to the atmosphere. The design transients listed in Subsection 3.9.1 are used in evaluating the components of the reactor coolant system for effects due to internal pressure and temperature changes from postulated accidents. Section 3.11 discusses the qualification of the equipment required to function in the adverse environmental conditions including temperature, humidity, pressure, and chemical consequences.

Pipe failure protection is provided according to the requirements of 10 CFR 50, Appendix A, General Design Criterion 4. In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided so that essential structures, systems, or components are not impacted by the adverse effects of postulated piping failure. Essential systems and components are those required to shut down the reactor and mitigate the consequences of the postulated piping failure. Nonsafety-related systems are not required to be protected from the dynamic and environmental effects associated with the postulated rupture of piping except as described in Subsection 3.6.1.1, item Q.

The criteria used to evaluate pipe failure protection are generally consistent with NRC guidelines including those in the Standard Review Plan Sections 3.6.1 and 3.6.2, NUREG-1061, Volume 3 (Reference 11) and applicable Branch Technical Positions.

Subsection 3.6.1 provides the design bases and criteria for the analysis required to demonstrate that essential systems are protected. The high- and moderate-energy systems representing the potential source of dynamic effects are listed. Additionally, the criteria for separation and the effects of adverse consequences are defined.

Subsection 3.6.2 defines the criteria for postulated break location and configuration. High-energy pipes are evaluated for the effects of circumferential and longitudinal pipe breaks and through-wall cracks. Moderate-energy pipes are evaluated for the effects of through-wall cracks. Analysis methods and criteria for evaluating pipe whip and evaluating the consequences of jet impingement, motions of the pipe, and system depressurization on integrity and operability are provided. The evaluation of containment penetrations, pipe whip restraints, guard pipes, and other protective devices is also described. The criteria for excluding breaks in high-energy piping adjacent to containment penetrations are also provided.

Evaluation of the dynamic effects of postulated breaks in the reactor coolant loop, main steam lines inside containment, and other primary piping inside containment equal to or greater than the 6-inch nominal pipe size (NPS) is eliminated for AP1000 based on mechanistic pipe break (leak-before-break) considerations. Those sections of high-energy piping that qualify for mechanistic pipe break are evaluated for only the effects of leakage cracks.

Subsection 3.6.3 describes the application of leak-before-break criteria to permit the elimination of pipe rupture dynamic effects considerations. Design guidelines aid in the design of piping systems

that satisfy the requirements for mechanistic pipe break. Dynamic effects of postulated breaks are evaluated for those analyzable sections of high-energy piping systems that do not use the mechanistic pipe break methods.

The safety analyses in Chapter 15 and the requirements for emergency core cooling discussed in Section 6.3 and the environmental qualification of equipment discussed in Section 3.11 of this report are not changed by the use of mechanistic pipe break considerations for pipe rupture dynamic effects evaluations. Chapter 6 describes the containment subcompartment pressurization analyses including mechanistic pipe break considerations.

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

A number of systems and components are necessary to shut the plant down in the event of a pipe rupture. These systems, termed essential systems, are protected from the postulated pipe ruptures. The essential systems for various pipe ruptures are the reactor coolant system, the steam generator system, the passive core cooling system, and the passive containment cooling system. In addition to these fluid systems, the protection and safety monitoring system and the Class 1E dc and UPS system are essential. The main control room and main control room habitability system are also protected as essential systems. In addition, containment penetrations and isolation valves (including those for nonessential systems) are essential.

Most of the equipment required for plant safety or safety-related shutdown is located inside containment. The piping inside containment also represents the most significant piping relative to plant safety and, therefore, is subject to the most stringent design and analysis requirements.

Essential equipment in the vicinity of piping that does not satisfy leak-before-break criteria is protected as required by the use of protective structures, pipe restraints, and separation. The need for protection of essential structures, systems and components is determined by evaluation of the dynamic effects. The design bases and criteria for the evaluation follow.

Evaluations are made based upon circumferential or longitudinal pipe breaks, through-wall cracks, or leakage cracks as determined by the appropriate criteria. At locations determined to be subject to a circumferential or longitudinal pipe break, dynamic effects such as jet impingement and pipe whip are evaluated.

At locations subject to through-wall cracks or leakage cracks, only effects such as spray wetting and flooding are evaluated. Through-wall cracks, which are postulated in high-energy piping and in moderate-energy lines, are larger and have a larger flowrate of water or steam than the leakage cracks postulated for high-energy piping, which satisfies the leak-before-break requirements.

The pressurization loads on structures and components are evaluated for postulated circumferential breaks and longitudinal breaks in piping that does not meet leak-before-break requirements and for postulated leakage cracks in piping that meets the leak-before-break requirements. See Subsection 3.8.3.4 and Subsection 3.8.4.3.1.4 for a discussion of pressurization loads on structures.

The in-containment refueling water storage tank is evaluated for pressurization as described in Subsection 3.6.1.2.1.

Pressurization loads for pipe failures in the main steam and feedwater break exclusion zones for high-energy lines in the vicinity of containment penetrations are evaluated for a 1.0 square foot break. Structures in the steam generator blowdown break exclusion zone are evaluated for subcompartment pressurization effects due to worst case circumferential pipe rupture in the 4-inch steam generator blowdown piping. Pipe whip and jet impingement are not evaluated for structures in the break exclusion zones per NRC Branch Technical Position MEB 3-1, section B.1.b, except that

the east wall and the floor at elevation 117′-6″ of the east main steam subcompartment is designed for pipe whip and jet impingement loads for worst case breaks in either the main steam line or the main feedwater line. See Subsection 3.6.2.1.1.4.

3.6.1.1 Design Basis

The following design bases relate to the evaluation of the effects of the pipe failures at locations determined in Subsection 3.6.2.

A. 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 two percent or less of the time during which the system is in operation or that experience high-energy pressures or temperatures for less than one percent of the plant operation time are considered moderate-energy.

B. The following assumptions are used to determine the thermodynamic state in the piping system for the calculation of fluid reaction forces:

   1. For those portions of piping systems normally pressurized during operation at power, the thermodynamic state in the pipe and associated reservoirs is that of normal full-power operation.

   2. For those portions of piping systems pressurized only during other normal plant conditions (for example, startup, hot standby, reactor cooldown), the thermodynamic state and associated operating condition are determined as the mode giving the most severe fluid reaction forces. Moderate-energy systems that are occasionally at higher temperature or pressure (see design basis A.) are not evaluated for pipe failures at the high-energy conditions.

   3. High-stress pipe rupture locations are based on calculated stresses due to Level A and Level B loading. Seismic loads are not included.

C. Circumferential and longitudinal breaks in high-energy pipes, except in pipes satisfying leak-before-break requirements, are evaluated for effects including subcompartment pressurization, pipe whip, jet impingement, jet reaction thrust, internal fluid decompression loads, spray wetting, and flooding.

D. High-energy and moderate-energy pipe through-wall cracks are evaluated for spray wetting and flooding effects. Dynamic effects are not evaluated for these cracks.

E. Through-wall cracks are not postulated in the break exclusion zones. The effects of flooding, spray wetting, and subcompartment pressurization are evaluated for a postulated 1.0 square foot break for the main steam and feedwater lines.

F. Where postulated, each longitudinal or circumferential break in high-energy fluid system piping, leakage crack in high-energy piping with mechanistic pipe break, or through-wall crack in high-energy or moderate-energy fluid system piping is considered separately as a single initial event occurring during normal plant conditions.

For systems not seismically analyzed for a safe shutdown earthquake, the safe shutdown earthquake is assumed to cause a pressure boundary failure, as described in Subsections 3.6.2.1.1.3 and 3.6.2.1.2.2.

G. Offsite power is not required for the actuation of the passive safety systems. The only electrical system required to function is the Class 1E dc and UPS system.

H. A single active component failure is assumed in systems used to mitigate the consequences of the postulated piping failure or to safely shut down the reactor. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power.

I. The function of the containment to act as the ultimate heat sink is maintained for any postulated pipe rupture.

J. Safety-related systems and components are used to mitigate the effects of postulated pipe ruptures. In addition, the turbine control and stop, moisture separator reheater 2nd stage steam isolation, and turbine bypass (steam dump) valves (which are not safety-related) are credited in single failure analyses to mitigate postulated steam line ruptures.

K. A whipping pipe is considered capable of rupturing impacted pipes of smaller nominal pipe diameter, irrespective of pipe-wall thickness. This is based on the assumption that only piping is determined to do the impacting. A whipping pipe is considered capable of developing a through-wall crack in a pipe of equal or larger nominal pipe size with equal or thinner wall thickness, assuming that only piping is determined to do the impacting. The preceding criterion is not used where the potential exists for valves or other components in the whipping pipe to impact the targets, since these are treated on a case-by-case basis.

L. Pipe whip is assumed to occur in the plane defined by the piping geometry and to cause movement in the direction of the jet reaction.

If unrestrained, a whipping pipe having a constant energy source sufficient to form a plastic hinge is considered to form a plastic hinge and rotate about the nearest rigid pipe whip restraint, anchor, or wall penetration capable of resisting the pipe whip loads or the calculated dynamic plastic hinge location.

If the direction of the initial pipe movement caused by the thrust force is such that the whipping pipe impacts a flat surface normal to its direction of travel, it is assumed that the pipe comes to rest against that surface, with no pipe whip in other directions.

Pipe whip restraints are provided wherever postulated pipe breaks could impair the capability of any essential system or component to perform its intended safety functions.

M. The calculation of thrust and jet impingement forces considers any line restrictions (that is, flow limiter) between the pressure source and break location and the absence of energy reservoirs, as applicable.

N. Breaks are not postulated to occur in pump and valve bodies since the wall thickness exceeds that of connecting pipe.

O. Components impacted by jets from breaks in piping containing high-pressure (870 to 2466 psia) steam or subcooled liquid (subcooled no more than 126°F) that would flash at the break, such as piping connected to the steam generators or reactor coolant loops, are evaluated as follows:

1. Impacted components within 10 piping inside diameters of the broken pipe are assumed to fail. Specific jet loads are calculated and evaluated only when failure of the component, when combined with a single active failure, could adversely affect safe shutdown or accident mitigation capability. These jet loads are calculated according to Subsection 3.6.2.3.1.

2. Components beyond 10 inside diameters of the broken pipe are considered to be undamaged by the jet and are not analyzed. The basis for these criteria is contained in NUREG/CR-2913 (Reference 1).

P. Pipe breaks are not postulated to occur in systems for which postulated leakage cracks have been shown to be stable for worst case loadings. (See Subsection 3.6.3.) Leak detection systems are provided that are capable of detecting the leakage from a postulated leakage crack.

For these systems, leakage cracks are postulated and evaluated for subcompartment pressure loads on structures and components. When the mechanistic pipe break approach is used, subcompartment pressure loads on structures and essential components are based on the small leakage crack determined from the mechanistic pipe break approach. Where the subcompartment includes lines not qualified for mechanistic pipe break, subcompartment pressurization is evaluated for a break in the line with the largest effect.

The leakage crack effects of jet impingement, pipe whip, and internal fluid system loads are considered negligible and are not evaluated. The leakage crack effects of flooding and environmental effects are less limiting than the corresponding effects for postulated high-energy through-wall cracks. These through-wall cracks are not eliminated by mechanistic pipe break.

Q. Nonessential systems, structures, and components are not required to meet the criteria outlined in this section. However, while none of the nonessential systems are needed during or following a pipe break event, pipe whip protection is evaluated in cases where a high-energy nonessential system failure could initiate a failure in an essential system or component or where a high-energy nonessential system failure could initiate a failure in another nonessential system whose failure could affect an essential system.

R. The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture will not preclude:

- Subsequent access to any areas, as required, to recover from the postulated pipe rupture

- Habitability of the control room

- Capability of essential instrumentation, electric power supplies, components, and controls to perform safety functions to the extent necessary to meet the criteria outlined in this section

3.6.1.2 Description

Essential systems are evaluated to demonstrate conformance with the design bases and to determine their susceptibility to the failure effects. Table 3.6-1 identifies systems which contain high and moderate-energy lines. The systems listed include all high- and moderate-energy systems inside containment plus the high- and moderate-energy systems in the auxiliary building near containment penetrations (including access hatches), the main control room, the Class 1E dc and UPS system or the portions of the passive containment cooling system located in the auxiliary building. The table does not list systems that operate at or close to atmospheric pressure including air handling and

gravity drains. High energy system piping in the turbine building adjacent to the auxiliary building is evaluated for potential effects on the main control room. These systems are included on Table 3.6-1.

The definition of high and moderate-energy systems is provided in paragraph A of Subsection 3.6.1.1.

The postulated break, through-wall crack, and leakage crack locations are determined according to Subsections 3.6.2 and 3.6.3.

Equipment is considered to be separated from the dynamic effects of pipe rupture when the equipment is located in a different subcompartment. For the case of pipe whip, equipment may be considered separated for dynamic effects based on the distance from the pipe and the length of pipe that is moving. For the case of jet impingement in a line with saturated or subcooled fluid, equipment more than ten inside pipe diameters from the break location and the tip of the pipe whip trajectory, including the resting location of the broken pipe, is considered separated for dynamic effects.

Equipment located in the same subcompartment as a break, through-wall crack, or leakage crack is subject to potential environmental and flooding effects. Equipment may also be subject to environmental and flooding effects of steam and water vented into a subcompartment from an adjoining subcompartment.

3.6.1.2.1 Pressurization Response

Pressurization response analyses are performed for subcompartments containing high-energy piping for which break locations are defined by Subsections 3.6.2.1.1.1, 3.6.2.1.1.2, and 3.6.2.1.1.3 or postulated leakage flaws are defined based on Subsection 3.6.3.3. Table 3.6-2 identifies those terminal end pipe breaks considered for the evaluation of the effects of pressurization loads on subcompartments. The terminal end pipe breaks inside containment that are postulated in piping that is not evaluated to the leak-before-break requirements of Subsection 3.6.3 are summarized in Table 3.6-2. The subcompartments are identified using the room numbers and room names given on Figures 1.2-4 through 1.2-10 as supplemented by Table 3.6-2. The subcompartments inside containment are designed to accommodate the pressurization loads from these breaks. In order to account for high stress break locations and the additional pressure boundary leakages from manways and flanges, pressurization loads on compartments inside containment enclosing high-energy piping are designed as described in Subsection 3.8.3.4.

There is no high-energy piping that can pressurize the annulus between the containment vessel and the shield building. Guard pipes are provided for the main steam, feedwater, and steam generator blowdown containment penetrations passing through the annulus as shown on Figure 3.8.2-4. The chemical and volume control system makeup piping is classified as high energy due to its design pressure, but does not cause pressurization because it is at ambient temperature.

The pressurization loads for the in-containment refueling water storage tank are based on the pressure and hydrodynamic loads due to the maximum discharge through the first, second, and third stages of the automatic depressurization system valves.

The pressurization loads for the reactor vessel annulus for the evaluation of asymmetric compartment pressurization are negligible based on a 5-gallon per minute leakage crack in the primary loop piping. The internal reactor pressure vessel asymmetric pressurization loads are based on a break in the largest pipe connected to the reactor coolant system that does not qualify for the application of mechanistic pipe break.

There are limited areas in the auxiliary building where the potential for pressurization loads from high-energy lines are considered. The pressurization loads for the steam tunnels are addressed in the

discussion of loads due to a break in the break exclusion zone of the main steam and feedwater lines. The pressurization loads for the Elevation 100′ containment penetration room containing the steam generator blowdown break exclusion zone are based on a circumferential rupture of the 4-inch steam generator blowdown piping. The areas through which the chemical and volume control system make-up line run, including the annulus between the containment and the containment shield building, are not subject to pressurization since the temperature of these lines is less than 212°F.

For a discussion of the criteria and analysis methods for subcompartment pressurization analysis, see Subsection 6.2.1.2. The analytical methods for transient mass distribution, used for pressure response analysis, are the same as described in WCAP-8077 (Reference 2).

3.6.1.2.2 Main Control Room Habitability

The high-energy lines in closest proximity to the main control room are the main steam line and main feedwater line. The portions of these lines near the main control room are in the main steam line isolation valve compartment and are part of the break exclusion areas.

The main control room is separated from the isolation valve compartment by two structural walls. The areas between the two walls is used for nonessential office and administrative space associated with the control room. The walls separating the main control room from the main steam isolation valve compartment are thick, reinforced-concrete walls.

Consistent with the criteria for evaluation of leaks in the break exclusion area, the subcompartment, including the walls, is evaluated for the effects of flooding, spray wetting and subcompartment pressurization from a 1-square-foot break from either main steam or feedwater line within the respective break exclusion areas. The wall between the main steam line isolation valve compartment and the main control room, and the floor slab between the main steam line isolation valve compartment and the safety related electrical equipment room are also evaluated for pipe whip and jet impingement loads for worse case breaks in either the main steam line or the main feedwater line. The subcompartment pressure loads from the 1-square-foot break are not combined with the pipe whip and jet impingement loads for the worse case breaks.

The effects upon the habitability of the main control room resulting from postulated pipe breaks and cracks in the auxiliary building are evaluated. In addition to pipe ruptures and cracks in lines in the auxiliary building, the main control room is evaluated for the dynamic effects and environmental effects of a postulated circumferential or longitudinal break of either the main steam line or main feedwater line in the turbine building.

Further description of the control room habitability systems, including options for remote shutdown, is provided in Section 6.4. The remote shutdown workstation is not subject to adverse effects of high-energy pipe rupture.

3.6.1.3 Safety Evaluation

3.6.1.3.1 General

An analysis of postulated pipe failures is performed to determine the impact of such failures on those safety-related systems or components that provide protective actions and are required to mitigate the consequences of the failure. Through such protective measures, as separation, barriers, and pipe whip restraints, the effects of breaks, through-wall cracks, and leakage cracks are prevented from damaging essential items to an extent that would impair their essential function or necessary component operability.

Typical measures used for protecting the essential systems, components, and equipment are outlined in the next subsection and are discussed in Subsection 3.6.2. The capability of specific safety-related systems to withstand a single active failure concurrent with the postulated event is discussed, as applicable. When the results of the pipe failure effects analysis show that the effects of a postulated pipe failure are isolated, physically remote, or restrained by protective measures from essential systems or components, no further dynamic analysis is performed.

3.6.1.3.2 Protection Mechanisms

The plant arrangement is based on maximizing the physical separation of redundant or diverse safety-related components and systems from each other and from nonsafety-related items. Therefore, in the event a pipe failure occurs, there is a minimal effect on other essential systems or components required for safe shutdown of the plant or to mitigate the consequences of the failure.

The effects associated with a particular pipe failure are mechanistically consistent with the failure. Thus, pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the specific measures for protection against the consequences of postulated failures.

Protection against the dynamic effects of pipe failures is provided by physical separation of systems and components, barriers, equipment shields, and pipe whip restraints. The precise method chosen depends largely upon considerations such as accessibility and maintenance. The preferred method of providing protection is by separation. When separation is not practical pipe whip restraints are used. Barriers or shields are used when neither separation nor pipe whip restraints are practical. This protection is not required when piping satisfies leak-before-break criteria.

Separation

The plant arrangement provides separation, to the extent practicable, between redundant safety systems (including their appurtenances) to prevent loss of safety function as a result of events for which the system is required to be functional. Separation between redundant safety systems, with their related appurtenances, therefore, is the basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

In general, separation is achieved by:

- Safety-related systems located remotely from high-energy piping, where practicable
- Redundant safety systems located in separate compartments, where practicable
- Specific components enclosed to retain the redundancy required for those systems that must function to mitigate specific piping failures
- Drainage systems provided for flooding control

Where physical separation is not possible, the pipe rupture hazard analysis includes an evaluation to determine the systems and components that require a structure for separation from the effects of a break in a high energy line. For these structures specifically included to separate breaks from essential systems or components, the evaluation considers that the break may be at the closest point in the line to the separating structure; not only at the break locations identified in Subsection 3.6.2.1.1. High energy lines qualified as leak-before-break lines and the lines in containment penetration break exclusion areas are not included as possible break locations in this evaluation. For a discussion of the information included in the pipe rupture hazard analysis see Subsection 3.6.2.5.

Barriers and Shields

Protection requirements are met through the protection afforded by walls, floors, columns, abutments, and foundations. Where adequate protection does not already exist as a result of separation, a separating structure such as additional barriers, deflectors, or shields is provided to meet the functional protection requirements.

Inside the containment, the secondary shield wall serves as a barrier between the reactor coolant loops and the containment. In addition, the refueling cavity walls, operating floor, and secondary shield walls minimize the possibility of an accident that may occur in any one reactor coolant loop affecting the other loop or the containment. Those portions of the steam and feedwater lines located within the containment are routed in such a manner that possible interaction between these lines and the reactor coolant piping is minimized. The direct vessel injection valves for train A and train B are separated by the secondary shield wall.

Barriers and shields that are identified as required by the pipe rupture hazard analysis are designed for loads from a break in the line at the closest location to the structure. This criterion is in conformance with the guidance of Branch Technical Position MEB 3-1. Rev. 2. Subsection 3.6.2.4 further discusses barriers and shields.

Piping Restraint Protection

Measures for protection against pipe whip are provided where the unrestrained pipe movement of either end of the ruptured pipe could cause damage at an unacceptable level to any structure, system, or components required to meet the criteria outlined in this subsection.

Subsection 3.6.2.3 gives the design criteria for and description of pipe whip restraints.

3.6.1.3.3 Specific Protection Considerations

The analysis of the consequences of pipe breaks, through-wall cracks, and leakage cracks uses the following criteria.

- High-energy containment penetrations are subject to special protection mechanisms. Restraints are provided to maintain the operability of the isolation valves and the integrity of the penetration due to a break in the safety-related and nonsafety piping beyond the restraint if required. These restraints are located as close as practicable to the containment isolation valves associated with these penetrations.

- Instrumentation required to function following a pipe rupture is protected.

- High-energy fluid system pipe whip restraints and protective measures are designed so that a postulated break in one pipe cannot lead to a rupture of other nearby essential pipes or components, if the secondary rupture results in consequences that are unacceptable for the initial postulated break.

For those cases in which the rupture of the main steam or feedwater piping inside containment is the postulated initiating event, the turbine control, turbine stop, moisture separator reheater 2nd stage steam isolation, and turbine bypass valves, and to a limited extent, the control systems for the turbine stop and feedwater control valves (which are nonsafety-related equipment), are credited in single failure analysis to mitigate the event. This equipment is not protected from pipe ruptures in the turbine building because the postulated pipe rupture for which it provides protection is inside containment. The assumed single active failure for this analysis is the function of the safety-related valve that would normally isolate the piping. This isolation function is addressed in more detail in Chapter 10.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This subsection describes the design bases for locating postulated breaks and cracks in high- and moderate-energy piping systems inside and outside the containment; the procedures used to define the jet thrust reaction at the break location; the procedures used to define the jet impingement loading on adjacent essential structures, systems, or components; pipe whip restraint design; and the protective assembly design. Pipe breaks in several high-energy systems, including the reactor coolant loop and surge line, are replaced by small leakage cracks when the leak-before-break criteria are applied. (See Subsection 3.6.3.) Jet impingement and pipe whip effects are not evaluated for these small leakage cracks.

3.6.2.1 Criteria Used to Define High- and Moderate-Energy Break and Crack Locations and Configurations

The NRC Branch Technical Position MEB 3-1 is used as the basis of the criteria for the postulation of high-energy pipe breaks and through-wall cracks, except for piping that satisfies the requirements for mechanistic pipe break, as described in Subsection 3.6.3.

A postulated high-energy pipe break is defined as a sudden, gross failure of the pressure boundary of a pipe either in the form of a complete circumferential severance (that is, a guillotine break) or as a sudden longitudinal, uncontrolled crack. For high-energy and moderate-energy fluid systems, pipe failures are also defined by postulation of controlled through-wall cracks in piping. For those piping lines that satisfy leak-before-break requirements, the guillotine breaks and sudden longitudinal cracks are replaced by postulated controlled leakage cracks.

Subsection 3.6.1 describes the evaluation and criteria for the effects of these breaks and cracks on the safety-related equipment.

3.6.2.1.1 High-Energy Break Locations

The locations for postulated breaks in high-energy piping are dependent on the classification, quality group, and design standards used for the piping system. The break locations for high-energy piping are described in the following subsections. These locations are based on the design configuration and include changes due to the as-built piping configuration. As a result of piping reanalysis due to differences between the design configuration and the as-built configuration, the high stress and usage factor location may be shifted. The intermediate break (if any) locations need not be changed unless one of the following conditions exists:

A. The dynamic effects from new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

B. There is a significant change in pipe design parameters such as pipe size, wall thickness or pressure rating.

Breaks are not postulated in piping in the vicinity of containment penetrations. The portion of the piping that does not have postulated breaks is the break exclusion area. Subsection 3.6.2.1.1.4 identifies the requirements for the piping in the containment penetration break exclusion area.

Breaks are not postulated for those sections of pipe, including the reactor coolant loop and pressurizer surge line, that meet the requirements for leak-before-break as described in Subsection 3.6.3.

The leak-before-break methodology is applied to the candidate high-energy lines in the nuclear island identified in Appendix 3E. This appendix also identifies other high-energy lines in the nuclear island with diameters larger than 1 inch and the break exclusion areas outside containment. The evaluation criteria for lines that do not satisfy the leak-before-break criteria are described in Subsection 3.6.2.

3.6.2.1.1.1 ASME Code, Section III, Division 1 – Class 1 Piping

[*Pipe breaks are postulated to occur at the following locations in piping designed and constructed to the requirements for Class 1 piping in the ASME Code, Section III, Division 1.*

- *At terminal ends of the piping, including:*
  - *The extremity of piping connected to structures, components, or anchors that act as essentially rigid restraints to piping translation and rotational motion due to static or dynamic loading.*
  - *Branch intersection points are considered a terminal end for the branch line unless the following are met: The branch and the main piping systems are modeled in the same static, dynamic and thermal analyses, and the branch and main run are of comparable size and fixity (that is, the nominal size of the branch is at least one-half of that of the main run).*
  - *In piping runs that are maintained pressurized during normal plant conditions for only a portion of the run, the terminal end, for purposes of defining break locations, is the piping connection to the first normally closed valve.*
- *At intermediate locations where the following conditions are satisfied:*
  - *Intermediate locations where the maximum stress range as calculated by Equation (10) of Paragraph NB-3653 of the ASME Code, Section III exceeds 2.4 Sm (where Sm is the design stress intensity), and either Equation (12) or Equation (13) of Paragraph NB-3653.6, exceed 2.4 Sm.*
  - *Intermediate locations where the cumulative usage factor as determined by the ASME Code exceeds 0.1.*
  - *Efforts will be made to avoid intermediate break locations through appropriate piping layout and pipe support design.*

The loading conditions considered for the stress range and usage factors calculated to determine break locations are those defined for Level A and B Service conditions for the piping system with the exception that seismic loads do not need to be considered for the postulation of intermediate break locations.

For those sections of pipe that satisfy the requirements for leak-before-break, leakage cracks are postulated for evaluation of subcompartment pressurization.]*

3.6.2.1.1.2 ASME Code, Section III – Class 2 and Class 3 Piping Systems

[*For those piping system lines designed and analyzed to the requirements of the ASME Code, Section III, Class 2 and 3, except for those sections that satisfy the mechanistic pipe break criteria (subsection 3.6.3), the following criteria apply.*

*NRC Staff approval is required prior to implementing a change in this information.

- *Pipe breaks are postulated to occur at terminal ends, using the same definition for terminal ends as for Class 1 pipe.*

- *Pipe breaks are postulated at intermediate locations between terminal ends where the maximum stress value, as calculated by the sum of Equations (9) and (10) in Subarticle NC-3600 (Class 2) and ND-3600 (Class 3) of the ASME Code, Section III, considering Level A and B Service conditions. That is, breaks are postulated at locations for sustained loads, occasional loads, and thermal expansion exceeding 0.8 (1.8 Sh + SA) or 0.8 (1.5 Sy + SA), where Sh, SA, and Sy are the allowable stress at maximum hot temperature stress, allowable stress range for thermal expansion, and yield strength, respectively, for Class 2 and 3 piping, as defined in Subarticle NC-3600 and Subarticle ND-3600 of the ASME Code, Section III. Efforts will be made to avoid intermediate break locations through appropriate piping layout and pipe support design.*

For those ASME Code, Section III, Class 2 and 3 systems that satisfy the leak-before-break criteria, postulated leakage crack locations are defined in the same way as for the Class 1 systems.]*

3.6.2.1.1.3 Piping Not Designed to ASME Code

[*Breaks in piping systems designed to requirements other than the ASME Code, such as ASME-B31.1 (Reference 3), are postulated at the following locations:*

- *If the piping is analyzed and supported to withstand safe shutdown earthquake loadings, pipe ruptures are postulated to occur at the following locations:*

  - *At terminal ends, using the same definition for terminal ends as for Class 1 pipe*

  - *At intermediate locations where the stresses, as calculated by the sum of Equations (9) and (10) in Subarticle NC3600 of the ASME Code, Section III, considering normal and upset plant conditions, exceeds 0.8 (1.8 Sh + SA) or 0.8 (1.5 Sy + SA)*

  - *Efforts will be made to avoid intermediate break locations through appropriate piping layout and pipe support design.*]*

- In the absence of stress analysis, breaks in non-nuclear piping are postulated at the following locations in each run or branch run:

  - Terminal ends

  - Intermediate fittings; (short- and long-radius elbows, crosses, flanges, nonstandard fittings, tees, reducers, welded attachments, and valves)

3.6.2.1.1.4 High-Energy Piping in Containment Penetration Areas

The AP1000 does not have any ASME Code, Section III Class 1 pipe in containment penetration areas. Breaks are not postulated in the portions of ASME Code, Section III, Class 2 or Class 3 piping, defined below as break exclusion piping, provided subject piping meets the following provisions:

- Stresses do not exceed those specified in Subsection 3.6.2.1.1.2.

- The maximum stress in this piping as calculated by Equation (9), of paragraph NC-3652 of ASME Code Section III, when subjected to the combined loadings of internal pressure, deadweight, and postulated pipe rupture outside the break exclusion zone, does not exceed 2.25 Sh or 1.8 Sy.

*NRC Staff approval is required prior to implementing a change in this information.

- The number of circumferential piping welds is minimized by using pipe bends in place of welding elbows when practicable. There are no longitudinal piping welds in the break exclusion zone. Where guard pipes are used, there are no circumferential or longitudinal welds in the piping enclosed within the guard pipe. Details of the arrangement are shown in Figure 3.8.2-4.

- When required for isolation valve operability, structural integrity, or containment integrity, anchors or five-way restraints capable of resisting torsional and bending moments produced by a postulated pipe break, either upstream or downstream of the piping and valves which form the containment isolation boundary, are located reasonably close to the isolation valves or penetration.

  The anchors or five-way restraints do not prevent the access required to conduct in-service inspection examinations specified in Section XI of the ASME Code. In-service examinations completed during each inspection interval provide 100-percent volumetric examination (according to IWA-2400, ASME Code, Section XI) of circumferential pipe welds within the boundary of these portions of piping during each inspection interval. This volumetric inspection applies to piping that is equal to or greater than a 3-inch nominal diameter.

- Welded attachments to these portions of piping for pipe supports or other purposes are avoided. Where welded attachments are necessary, detailed stress analyses are performed to demonstrate compliance with the limits of Subsection 3.6.2.1.1 and applicable requirements of Section XI of the ASME Code.

- The requirements of ASME Code, Section III, Subarticle NE-1120, are satisfied for the containment penetration.

- Class 3 pipe satisfies the fabrication and inspection requirements for Section III, Class 2 pipe.

- For evaluation of spray wetting, flooding, and subcompartment pressurization effects, longitudinal cracks (with crack flow areas of 1 square foot) are postulated in the main steam and main feedwater piping. The dynamic effects of pipe whip and jet impingement are not evaluated for these cracks. Locations having the greatest effect on essential equipment are chosen.

- Guard pipe assemblies for high-energy piping in the containment annulus region between the containment shell and shield building that are part of the containment boundary are designed according to the rules of Class MC, subsection NE, of the ASME Code. The following requirements also apply. The design pressure and temperature are equal to or greater than the maximum operating pressure and temperature of the enclosed process pipe under normal plant conditions. Level C service limits of the ASME Code, Section III, Paragraph NE-3221, are not exceeded by the loadings associated with containment design pressure and temperature in combination with a safe shutdown earthquake. The guard pipe assemblies are subjected to a pressure test performed at the maximum operating pressure of the enclosed process pipe.

Areas of system piping where no breaks, except as noted in Subsections 3.6.1.3 and 3.6.1.2.2, are postulated are as follows:

- The main steam piping from the containment penetration flued head inboard weld to the auxiliary building anchor downstream of the main steam isolation valves, including the main steam safety valves and the connecting branch piping

- The main feedwater piping from the auxiliary building side of the containment penetration flued head to the auxiliary building anchor upstream of the isolation valve
- The startup feedwater piping from the auxiliary building side of the containment penetration flued head to the auxiliary building anchor upstream of the isolation valve
- The steam generator blowdown piping from the auxiliary building side of the containment penetration flued head to the auxiliary building anchor downstream of the isolation valve
- The chemical and volume control system makeup piping from the containment penetration flued head to the outboard isolation valve
- The chemical and volume control system makeup piping from the containment penetration flued head to the inboard isolation valve

Those portions of the containment penetration flued heads identified above that have the same nominal dimensions as the connected pipe are also considered as part of the break exclusion zone piping. The auxiliary building anchors also have flued head designs and the same requirement applies to these.

The main steam and main feedwater containment penetration flued heads are attached to expansion bellows, which are attached to the containment vessel via insert plates (Subsection 3.8.2.1.5, Figure 3.8.2-4, Sheet 1). The function of the expansion bellows is to minimize any piping loads applied to the containment vessel. The containment is not a terminal end for these piping analyses; the terminal ends are the main steam and main feedwater piping anchors in the auxiliary building exterior wall and their respective steam generator nozzles inside containment. The portion of the main steam piping that is inside containment is evaluated to meet the leak-before-break mechanistic pipe break criteria in accordance with Subsection 3.6.3; the portion of the main feedwater piping that is inside containment is analyzed to meet the high-energy pipe break criteria in accordance with Subsection 3.6.2.

All other fluid system containment penetrations are for moderate-energy systems or for pipe of 1-inch nominal diameter or smaller. See Subsection 6.2.3 for a discussion of containment penetrations.

3.6.2.1.2 Types of Breaks/Cracks Postulated

3.6.2.1.2.1 Break in Piping – High-Energy

The following types of breaks are postulated to occur in ASME Code Class 1, 2, and 3 and non-ASME Code, Section III high-energy piping at the locations determined according to Subsection 3.6.2.1.1, except when the leak-before-break criteria are satisfied.

- In piping with a nominal diameter of greater than or equal to 4 inches, both circumferential and longitudinal breaks are postulated at each selected break location unless eliminated by comparison of longitudinal and axial stresses with the maximum stress as follows:
  - If the maximum stress range exceeds the limits specified in Subsections 3.6.2.1.1.1, 3.6.2.1.1.2, and 3.6.2.1.1.3, but the circumferential stress range is at least 1.5 times the axial stress range, only a longitudinal break is postulated.
  - If the maximum stress range exceeds the limits specified in Subsections 3.6.2.1.1.1, 3.6.2.1.1.2, and 3.6.2.1.1.3, but the axial stress is at least 1.5 times the circumferential stress range, only a circumferential break is postulated.

  - Longitudinal breaks, however, are not postulated at terminal ends.

- In piping with a nominal diameter of greater than 1 inch but less than 4 inches, only circumferential breaks are postulated at each selected break location.

- No breaks are postulated for piping with a nominal diameter of 1 inch or less.

3.6.2.1.2.2 Through-Wall Cracks in High- or Moderate-Energy Piping

Through-wall cracks are postulated in high-energy or moderate-energy piping, including branch runs larger than 1-inch nominal diameter as defined in the following paragraphs:

A. Through-wall cracks are not postulated in the break exclusion areas of high-energy pipe defined in Subsection 3.6.2.1.1.4 and in those portions of moderate-energy piping between containment isolation valves, provided the containment penetration meets the requirements of ASME Code, Section III, Sub-article NE-1120, and the piping is designed so that the maximum stress range based on the sum of equations (9) and (10) in Subarticle NC3600 of the ASME Code, Section III, does not exceed 0.4 (1.2 Sh + SA).

B. Through-wall cracks are not postulated in high- or moderate-energy fluid system piping located in an area where a break in the high-energy fluid system is postulated, provided that such cracks do not result in environmental conditions more limiting than the high-energy pipe break.

C. Subject to Paragraphs A and D, through-wall cracks are postulated in:

   - ASME Code, Section III, Division 1 – Class 1 piping where the maximum stress range as calculated by Equation (10) of Paragraph NB-3653 of the ASME Code, Section III exceeds 1.2 Sm. Cracks are also postulated at locations where the cumulative usage factor exceeds 0.1.

   - ASME Code, Section III, Division 1 – Class 2 or 3 piping at locations where the maximum stress range, as calculated by the sum of Equations (9) and (10) in Subarticle NC-3600 (Class 2) and ND-3600 (Class 3) of the ASME Code, Section III, considering Level A and B Service conditions, in the piping is greater than 0.4 (1.8 Sh + SA) or 0.4 (1.5 Sy + SA).

   - Seismically analyzed ASME-B31.1 piping at locations defined in the same way as ASME Code, Section III, Class 3 piping.

   - Nonseismically analyzed ASME-B31.1 piping at the following locations:

     - Terminal ends

     - Intermediate fittings; (short- and long-radius elbows, crosses, flanges, nonstandard fittings, tees, reducers, welded attachments, and valves)

D. Individual through-wall cracks are not postulated at specific locations determined by stress analyses when a review of the piping layout and plant arrangement drawings shows that the effects of through-wall leakage cracks at any location in the piping designed to seismic or nonseismic standards are isolated or are physically remote from structures, systems, and components required for safe shutdown.

E. Through-wall cracks are postulated to be in those circumferential locations that result in the most severe environmental consequences.

3.6.2.1.2.3 Leakage Cracks in High-Energy Piping with Leak-before-Break

In those sections of piping that satisfy the requirements for leak-before-break, leakage cracks are postulated for evaluation of subcompartment pressurization. The size of the crack is such that the expected leakage is 10 times the minimum leak detection capability for that location. See Subsection 3.6.3 for a discussion of crack size and leakage detection.

3.6.2.1.3 Break and Crack Configuration

3.6.2.1.3.1 High-Energy Break Configuration

Following a circumferential break, the two ends of the broken pipe are assumed to move clear of each other unless physically limited by piping restraints, structural members, or piping stiffness. The effective cross-sectional (inside diameter) flow area of the pipe is used in the jet discharge evaluation. Movement is assumed to be in the direction of the jet reaction initially with the total path controlled by the piping geometry.

The orientation of a longitudinal break, except when otherwise justified by a detailed stress analysis, is assumed to be at opposing points on a line perpendicular to the plane of a fitting for a non-axisymmetric fitting. The flow area of such a break is equal to the cross-sectional flow area of the pipe. The geometry of the longitudinal break may be assumed elliptical (2D along pipe axis and D/2 along pipe transverse) or circular. Both circumferential and longitudinal breaks are postulated to occur, but not concurrently, in high-energy piping systems at the locations specified in Subsection 3.6.2.1.2.1, except as follows:

- Where the postulated break location is at a tee or elbow, the locations and types of breaks are determined as follows:

  - Without the benefit of a detailed stress analysis, such as a finite element analysis, circumferential breaks are postulated to occur individually at each pipe-to-fitting weld. Longitudinal breaks are postulated to occur individually (except in piping with a nominal diameter less than 4-inches) on each side of the fitting at its center and oriented perpendicular to the plane of the fitting, or

  - Alternatively, if a detailed stress analysis or test is performed, the results may be used to predict the most probable rupture location(s) and type of break.

- Where the postulated break location is at a branch/run connection, a circumferential break is postulated at the branch pipe-to-branch fitting weld unless otherwise justified by detailed analysis.

- Where the postulated break location is at a welded attachment (lugs, stanchions), a circumferentially oriented break is postulated at the centerline of the welded attachment unless otherwise justified by a detailed analysis. The break area is equal to the pipe surface area that is bounded by the welded attachment.

- Where the postulated break location is at a reducer, circumferential breaks are postulated at each pipe-to-fitting weld. Longitudinal breaks are oriented to produce out-of-plane bending of the piping configuration on both sides of the reducer at each pipe-to-fitting weld.

3.6.2.1.3.2 High-Energy and Moderate-Energy Through-Wall Crack Configuration

High-and moderate-energy through-wall crack openings are assumed to be a circular orifice with cross-sectional flow area equal to that of a rectangle one-half the pipe inside diameter in length and

one-half pipe wall thickness in width. The flow from a through-wall crack is assumed to result in an environment that wets unprotected components within the compartment with consequent flooding in the compartment and communicating compartments, unless analysis shows otherwise. Flooding effects are determined on the basis of a conservatively estimated time period required to take corrective actions.

3.6.2.2 Analytical Methods to Define Jet Thrust Forcing Functions and Response Models

To determine the forcing function, the fluid conditions at the upstream source and at the break exit dictate the analytical approach and approximations that are used.

Analytical methods for calculation of jet thrust for the preceding situations are discussed in ANS-58.2-1988 (Reference 4) and Moody, F. J. (Reference 5). The discussion of the jet thrust forcing functions on the reactor coolant loop follows.

Since a rupture of the large-diameter reactor coolant loop piping does not have to be considered, based on satisfying mechanistic pipe break criteria, the jet thrust and reactive loads considered in the analysis are those associated with breaks in branch line sections that do not satisfy the mechanistic pipe break criteria.

To determine the thrust and reactive force loads to be applied to the reactor coolant loop during the postulated pipe rupture, it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the reactor coolant loops as a result of a postulated loss of coolant accident. These forces result from the transient flow and pressure histories in the reactor coolant system (RCS).

The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flowrates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates, and calculates the time-history of forces at appropriate locations in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the entire reactor coolant system. Key parameters calculated by the hydraulic model are pressure, mass flowrate, and density. These are supplied to the thrust calculation, together with plant layout information, to determine the time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated loss of coolant accident, the pressure and momentum flux terms are dominant. The inertia and gravitational terms are taken into account in the evaluation of the local fluid conditions in the hydraulic model.

The blowdown hydraulic analysis provides the basic information concerning the dynamic behavior of the reactor core environment for the loop forces. This requires the ability to predict the flow, quality, and pressure of the fluid throughout the reactor system. [*MULTIFLEX (Reference 6) or an equivalent computer code is used to provide this information.*]*

MULTIFLEX calculates the hydraulic transients within the entire primary coolant system. This hydraulic program considers a coupled, fluid-structure interaction by accounting for the deflection of the core support barrel. The depressurization of the system is calculated using the method of characteristics applicable to transient flow of a homogenous fluid in thermal equilibrium.

The ability to treat multiple flow branches and a large number of mesh points gives MULTIFLEX the flexibility to represent the various flow passages within the primary reactor coolant system. The system geometry is represented by a network of one-dimensional flow passages.

*NRC Staff approval is required prior to implementing a change in this information.

[*The THRUST computer program or equivalent is used to compute the transient (blowdown) hydraulic loads resulting from a loss of coolant accident.*]*

The blowdown hydraulic loads on primary loop components are computed from the equation:

$$F = 144\,A\left[(P - 14.7) + \left(\frac{\dot{m}^2}{144\,\rho\,g\,(A_m)^2}\right)\right]$$

where:

F	=	Force (lbf)
A	=	Aperture area (ft^2)
P	=	System pressure (psia)
$\dot{m}$	=	Mass flowrate (lbm/s)
ρ	=	Density (lbm/ft^3)
g	=	Gravitational constant = 32.174 ft-lbm/lbf - s2
A_m	=	Mass flow area (ft^2)

In the model to compute forcing functions, the reactor coolant loop system is represented by a model similar to that employed in the blowdown analysis. The entire loop layout is represented in a global coordinate system. Each node is described by blowdown hydraulic information and the orientation of the streamline of the force nodes in the system, which includes flow areas and projection coefficients along the three axes of the global coordinate system.

Each node is modeled as a separate control volume with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area.

Each force is divided into its x, y and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, a total y force, and a total z force. These thrust forces serve as input to the piping/ restraint dynamic analysis.

[*The THRUST code calculates forces the same way as the STHRUST code described in WCAP-8252 (Reference 7).*]*

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

This subsection describes the pipe rupture design criteria for auxiliary piping systems. Subsection 3.6.2.2 describes the analysis methods for thrust loadings. To mitigate each postulated pipe rupture, auxiliary piping systems required to maintain pressure boundary integrity or to provide for fluid flow are identified. The loadings on these systems may consist of jet impingement loads, transient motions at terminal end connections, or internal system depressurization loadings.

The application of leak-before-break analysis eliminates evaluation of postulated pipe ruptures in the primary coolant loop piping and selected piping systems of 6-inch nominal size or larger. The piping

*NRC Staff approval is required prior to implementing a change in this information.

system mechanical components and supports are designed for the effects of the remaining postulated pipe ruptures and leaks.

To confirm the continued integrity of the essential components and the engineered safety systems, consideration is given to the consequential effects of the pipe break to the extent that:

- The minimum performance capabilities of the engineered safety systems are not reduced below that required to protect against the postulated break.
- The containment leaktightness is not decreased below the design value if the break leads to a loss of coolant accident.
- Propagation of damage is limited in type or degree or both to the extent that:
  - A pipe break that is not a loss of coolant accident, steam line break, or main feedwater break will not cause a loss of coolant accident or steam line or feedwater line break.
  - A break in the nonsafety portion of the chemical and volume control system purification loop will not cause a break in the safety-related portion of the system. In addition, the ability to isolate the reactor coolant system flow will not be adversely affected.
  - A reactor coolant system pipe break will not cause a steam or feedwater system pipe break, and vice versa.

3.6.2.3.1 Jet Impingement

Analytical methods for the calculation of jet impingement forces are based on Moody, F. J. (Reference 5), NUREG/CR-2913 (Reference 1), and Section 7.3 of ANS-58.2-1988 (Reference 4). For piping systems this loading is a suddenly applied load that can have significant energy content. These loads are generally treated as statically applied constant loads.

Two separate structural evaluations are performed. For the short-term response, snubber supports are considered to be active and a dynamic load factor of 2 is used. For the longer-term response, snubber supports are considered inactive, and no dynamic load factor is used.

If simplified static analysis is performed instead of a dynamic analysis, the preceding jet load (FT) is multiplied by a dynamic load factor. For an equivalent static analysis of the target structure, the jet impingement force is multiplied by a dynamic load factor of 1.2 to 2.0, depending upon the time variance of the jet load and the elastic/plastic behavior of the target. This factor assumes that the target can be represented as essentially a one-degree-of-freedom system.

3.6.2.3.2 Transient Motions at Terminal Ends

This loading is displacement limited and has a short duration of about 0.5 seconds. An example is the motions of the primary loop piping at the terminal end connection of the Class 1 pressurizer surge line piping due to a postulated pipe rupture in a Class 2 pipe connected to the steam generator.

When there are active in-line components in the piping system that must function to mitigate the postulated pipe rupture, dynamic structural analyses are performed for the terminal end motions. The calculated accelerations are evaluated to confirm the operability of the active in-line components. For piping systems with no active in-line components, static structural analyses with no dynamic amplification are performed for the terminal end motions.

These analyses may consider nonlinear geometric and material characteristics of the piping system.

3.6.2.3.3 Internal System Depressurization

This loading has a short duration of approximately 0.5 seconds and arises from rapidly traveling pressure waves in piping systems connected to the broken piping system. Two types of configurations are possible: systems without check valves and systems with check valves. In systems with check valves, the valve closure can increase the duration and magnitude of these loads.

An example of the former is the pressure waves in the Class 1 letdown line of the chemical and volume control system piping due to a postulated pipe rupture in a Class 1 pipe connected to the primary loop piping. An example of the latter is the closure of the feedwater check valve due to a postulated pipe rupture upstream of the valve.

For piping systems without closing check valves, there is little energy in the high-frequency depressurization loadings. These loadings are therefore not considered in the piping and support analysis.

For piping system with closing check valves, the magnitude of the loadings depends on the valve closure time, with shorter closing times generally causing higher loadings. For this loading the potential system failure mechanisms evaluated are: 1) excessive pipe and valve hoop stress; 2) tensile loads on the valve pressure boundary bolting; and 3) excessive distortion of the valve disc or seat.

The maximum internal pressure and the kinetic energy of the valve disc at the time of closure are used to verify the pressure boundary integrity of the piping and valve based on the preceding failure mechanisms. MULIFLEXSG is used to calculate the pressure and kinetic energy. The supports on these systems are designed in such a way that support failure will occur prior to local pipe pressure boundary failure at the support connection.

3.6.2.3.4 Pipe Whip Restraints

To satisfy varying requirements of available space, permissible pipe deflection, and equipment operability, the restraints are generally designed as a combination of an energy-absorbing element and a restraint structure suitable for the geometry required to pass the restraint load from the whipping pipe to the main building structure. The restraint structure is typically a structural steel frame or truss, and the energy-absorbing element is usually either stainless steel U-bars or energy-absorbing material.

3.6.2.3.4.1 Location of Pipe Whip Restraints

For purposes of determining pipe hinge length and thus locating the pipe whip restraints, the plastic moment of the pipe is determined in the following manner:

$$Mp = 1.1\ zpSy$$

where:

zp = Plastic section modulus of pipe

Sy = Yield stress at pipe operating temperature

1.1 = 10-percent factor to account for strain hardening.

Pipe whip restraints are located as close to the axis of the reaction thrust force as practicable. Pipe whip restraints are generally located so that a plastic hinge does not form in the pipe. If, because of physical limitations, pipe whip restraints are located so that a plastic hinge can form, the consequences of the whipping pipe and the jet impingement effect are further investigated. Lateral guides are provided where necessary to predict and control pipe motion.

Generally, pipe whip restraints are designed and located with sufficient clearances between the pipe and the restraint in such a way that they do not interact and cause additional piping stresses. A design hot position gap is provided that allows maximum predicted thermal, seismic, and seismic anchor movement displacements to occur without interaction.

Exception to this general criterion may occur when a pipe support and restraint are incorporated into the same structural steel frame, or when a zero design gap is required. In these cases, the pipe whip restraint is included in the piping analysis and designed to the requirements of pipe support structures for all loads except pipe break and designed to the requirements of pipe whip restraints when pipe break loads are included.

In general, the pipe whip restraints do not prevent the access required to conduct in-service inspection examination of piping welds. When the location of the restraint makes the piping welds inaccessible for in-service inspection, a portion of the restraint is designed to be removable to provide accessibility.

3.6.2.3.4.2 Analysis and Design of Pipe Whip Restraints

The criteria for analysis and design of pipe whip restraints for postulated pipe break effects are provided in the following. These criteria are consistent with the guidelines in ANS-58.2-1988 (Reference 4).

- Pipe whip restraints are designed based on energy absorption principles by considering the elastic-plastic, strain-hardening behavior of the materials used.

- Non-energy absorbing portions of the pipe whip restraints are designed to the requirements of AISC N690 Code supplemented by the requirements given in Subsection 3.8.4.5.

- A rebound factor of 1.1 is applied to the jet thrust force.

- Except in cases where calculations are performed to verify that a plastic hinge is formed, the energy absorbed by the ruptured pipe is conservatively assumed to be zero. That is, the thrust force developed goes directly into moving the broken pipe and is not reduced by the force required to bend the pipe.

- Other structural members of the pipe whip restraints are designed for elastic response. A dynamic increase factor is used for those members that are designed to remain elastic.

- The criteria for allowable strain in a pipe whip restraint are dependent on the type of restraint. The following discussions address the types of restraints used and the allowable strain for each. Note -ε = allowable strain used in design, and δ = allowable crushable length used in design.

Stainless Steel U-Bar – This type of restraint consists of one or more U-shaped, upset-threaded rods of stainless steel looped around the pipe but not in contact with the pipe. This allows unimpeded pipe motion during seismic and thermal movement of the pipe. At rupture, the pipe moves against the U-bars, which absorb the kinetic energy of pipe motion by yielding plastically. Figure 3.6-1 shows a typical example of a U-bar restraint.

$$\varepsilon = 0.5\varepsilon u$$

where:

εu = ultimate uniform strain of stainless steel (strain at ultimate stress)

Energy-Absorbing Material – This type of restraint consists of a crushable, stainless steel, internally honeycomb-shaped element designed to yield plastically under impact of the whipping pipe. A design hot position gap is provided between the pipe and the energy-absorbing material to allow unimpeded pipe motion during seismic and thermal pipe movements. Figure 3.6-2 shows a typical example of an energy-absorbing material restraint. The allowable capacity of crushable material shall be limited to 80 percent of its rated energy dissipating capacity as determined by dynamic testing, at loading rates within ± 50 percent of the specified design loading rate. The rated energy dissipating capacity shall be taken as not greater than the area under the load-deflection curve as illustrated in Figure 3.6.2-1 of NUREG-0800, Standard Review Plan, Section 3.6.2, Revision 2.

3.6.2.4 Protective Assembly Design Criteria

In addition to pipe whip restraints, other protective devices are designed to protect against the effects of postulated pipe ruptures. Barriers and shields are designed to protect against jet impingement. Guard pipes in the break exclusion zones provide additional confidence that pipes will not leak into the annulus between the containment vessel and the shield building.

3.6.2.4.1 Jet Impingement Barriers and Shields

Barriers and shields, constructed of either steel or concrete, are provided to protect essential equipment, including instrumentation, from the effects of jet impingement resulting from postulated pipe breaks. Barriers differ from shields in that they may also accept the impact of whipping pipes. Barriers and shields include walls, floors, and structures specifically designed to provide protection from postulated pipe breaks. Barrier and shield design is based on elastic methods and the elastic-plastic methods for dynamic analysis included in Biggs, J. M. (Reference 9). Design criteria and loading combinations are according to Subsections 3.8.3 and 3.8.4.

3.6.2.4.2 Auxiliary Guardpipes

The use of guard pipes has been minimized by plant arrangement and routing of high-energy piping. Guard pipes in the containment annulus areas of the break exclusion zones are designed as described in Subsection 3.6.2.1.1.4. Other guard pipes are designed and constructed to the same ASME rules as the enclosed process pipe.

3.6.2.5 Evaluation of Dynamic Effects of Pipe Ruptures

The preceding information provides the criteria and methods for the evaluation of the dynamic effects of pipe ruptures. The pipe rupture hazard analysis report (also referred to as the pipe break evaluation report) includes the following:

- Prepare a stress summary
- Identify pipe break locations in high energy piping
- Identify through-wall crack locations in high and moderate energy piping
- Identify and locate essential structures, systems, and components

- Evaluate consequences of pipe whip and jet impingement

  For rooms with both high energy breaks and essential items, confirm that there is no adverse interaction between the essential items and the whipping pipe or jet.

  The plant layout is modified as required to provide separation to protect essential systems.

- Evaluate consequences of flooding, environment, and compartment pressurization

  Evaluate compartment pressurization in the break exclusion zones in the vicinity of containment penetrations due to 1.0 square foot breaks in the main steam and feedwater lines.

- Design and locate protective hardware

- Prepare isometric piping sketches that identify the break locations, the basis for these locations and the protective hardware which mitigates the consequences of these breaks.

- Reconciliation of as-built condition

Pipe breaks that are larger than 1-inch nominal diameter are evaluated for pipe whip and jet impingement. Lines that are located in a break exclusion zone or are qualified to leak-before-break are not evaluated for pipe whip and jet impingement effects on systems and components, except for the portions of the lines in the MSIV compartment adjacent to the main control room as noted in Subsection 3.6.1.2.2.

Where these systems are qualified for mechanistic pipe break and pipe rupture loads prior to fabrication, the qualification is based on design information, not on as-built information. As-built information and the final configuration of valves and other equipment is used to verify the design analysis.

High Energy Break Locations

High energy break locations evaluated are on the nuclear island and in the turbine building for evaluation of the wall loadings in the south end of the turbine building adjacent to the main control room.

For ASME Class 1 piping terminal end locations are determined from the piping isometric drawings. Intermediate break locations depend on the ASME Code stress report fatigue analysis results. These results are not available at design certification. For the design of the AP1000, breaks are postulated at locations typically associated with a high cumulative fatigue usage factor. These locations are at valves, tees, and branch connections which have significant structural discontinuities. These locations are part of the as-built reconciliation as discussed in Subsection 3.6.4.1. The following ASME Class 1 lines are evaluated to terminal end and intermediate high energy break locations if applicable.

Line	**Diameter (inches)**
Pressurizer Spray	4
Automatic Depressurization Stage 1	4
Chemical and Volume Control Letdown	3
Chemical and Volume Control Makeup	3
Pressurizer Auxiliary Spray	2

For ASME Class 2 and 3 piping, terminal end break locations are determined from the piping isometric drawings. The intermediate break locations depend on the stress level. The AP1000 ASME Class 2 and 3 lines do not have intermediate breaks based on the low stress. The following ASME Class 2 and 3 lines have terminal end high energy break locations.

Line	**Diameter (inches)**
Main Feedwater	16, 20
Startup Feedwater	6
Steam Generator Blowdown	4

For B31.1 piping, terminal end break locations are determined from the piping isometric drawings. The intermediate break locations in seismically analyzed pipe depend on the stress level. The AP1000 ASME seismically analyzed B31.1 piping does not have intermediate breaks based on the low stress. For nonseismically analyzed high-energy ASME B31.1, intermediate breaks locations are postulated at each fitting.

Rooms subject to pressurization due to high energy pipe break are listed in Table 3.6-2 with the terminal end location.

Essential Systems and Components

In rooms that contain high energy pipe breaks, the systems and components that are needed to mitigate the postulated break and achieve a safe plant shutdown are identified. Rooms that contain both high energy pipe break locations and essential systems or components that must be protected are listed in Table 3.6-3. No high energy pipe break protection is required in other areas of the plant.

Essential Target Evaluation

To complete the essential target evaluation jet parameters, volumetric area of affected compartments, plant layout, and separating structures are considered. Parameters that determine the shape of the jet and the magnitude of the jet and thrust loads include pressure, temperature, and friction losses between the break and the reservoir. The volumetric area affected is determined by considering jet shape and loads at the postulated location of the breaks. Where an initial evaluation of essential targets indicated adverse effects, layout may be changed to relocate the target or postulated break. If necessary, the location of whip restraints and jet shields is established to protect essential systems and components. Essential equipment protected by pipe whip restraints or jet shields is listed in Table 3.6-3. The criteria for the break location postulated for evaluation of separating structures is outlined in Subsection 3.6.1.3.2.

Verification of the Pipe Break Hazard Analysis

A pipe rupture hazard analysis is prepared based on the as-designed piping stress analyses and pipe whip restraint design information. The as-designed piping analysis is based on piping routings, layouts, and isometrics. Intermediate break locations are identified using the as-designed piping stress analysis, including the fatigue analysis required for ASME Code Class 1 piping. As-designed piping stress analysis information is used to confirm the location and configuration of pipe whip restraints and jet impingement shields. The information included in Tables 3.6-2 and 3.6-3 is updated and validated as part of the as-designed pipe rupture hazard analysis. Large leakage cracks in moderate energy pipes are evaluated for adverse effects as part of the pipe break hazard evaluation.

The ASME Code, Section III, requires that each plant have a Design Report for the piping system that includes as-built information. Included in the Design Reports are the loads and loading combinations used in the analysis. Where mechanistic pipe break requirements are used to eliminate

the evaluation of dynamic effects of pipe rupture in ASME Code, Section III, Class 1, 2, and 3 piping system, the basis for the exclusion is documented in the Design Report.

The final piping stress analyses, pipe whip restraint design, and as-built reconciliation of the pipe break hazard analysis is discussed in Subsection 3.6.4.1. The final piping stress analysis includes design properties and characteristics of procured components selected to be included in the piping system that are not available for the as-designed evaluation. The as-built reconciliation is required prior to fuel loading and includes evaluation of the ASME Code fatigue analysis, pipe break dynamic loads, reconciliation to the certified design floor response spectra, confirmation of the reactor coolant loop time history seismic analyses, changes in support locations, preoperational testing, and construction deviations.

3.6.2.6 Evaluation of Flooding Effects from Pipe Failures

The effect of flooding due to high and moderate energy pipe failures on essential systems and components is described in Section 3.4.

3.6.2.7 Evaluation of Spray Effects from High- and Moderate-Energy Through-Wall Cracks

Essential systems and components are evaluated for the potential effects of spray from high- and moderate-energy through-wall cracks. Spray effects are assumed to be limited to the compartment where the pipe failure occurs. The spray is assumed to wet unprotected components in the compartment. It is further assumed the spray does not damage non-electrical passive components, including piping, ducts, valve bodies, or mechanical components of valve operators. Spray may cause failure of electrical components not designed to withstand wetting. Components protected by NEMA 4 or NEMA 12 enclosures are not affected by spray effects.

The safe shutdown components inside containment are subject to wetting from design basis events inside containment. These conditions bound the effects of spray from moderate energy cracks. Sensitive components are qualified for this environment as described in Section 3.11.

The doors to the auxiliary Class 1E battery rooms are normally closed, so spray cannot affect the batteries if fire fighting activities or a pipe crack were to occur in the corridor. If fire fighting activities were to occur in a particular room, all of the equipment is assumed inoperable due to the fire, therefore, no further spray effects need be considered. The containment isolation valves subject to spray and the safe shutdown components in the main steam tunnels are provided with spray protection. The sensitive components of the main control room emergency habitability system are protected from spray effects.

3.6.3 Leak-before-Break Evaluation Procedures

This subsection describes the design basis for mechanistic pipe break (leak-before-break) evaluation of high-energy piping systems.

Mechanistic pipe break evaluations demonstrate that for piping lines meeting the criteria, sudden catastrophic failure of the pipe is not credible. It is demonstrated that piping that satisfies the criteria leaks at a detectable rate from postulated flaws prior to growth of the flaw to a size that would fail because applied loads resulting from normal conditions, anticipated transients, and a postulated safe shutdown earthquake.

The use of mechanistic pipe break criteria represents a higher level of confidence of the integrity of piping systems based on additional criteria compared to the existing high level of integrity provided

by the requirements of the ASME Code. Evaluations of the mechanistic pipe break criteria are commonly called leak-before-break evaluations.

The use of mechanistic pipe break criteria permits the elimination of the evaluation of dynamic effects of sudden circumferential and longitudinal pipe breaks in the design basis analysis of structures, systems, and components. General Design Criterion 4 of Appendix A, 10 CFR Part 50 allows the use of analyses to eliminate from the design basis the dynamic effects of pipe ruptures.

Without the application of mechanistic pipe break criteria, the dynamic effects are evaluated for pipe ruptures postulated at locations defined in Subsection 3.6.2. Dynamic effects include jet impingement, pipe whip, jet reaction forces on other portions of the piping and components, subcompartment pressurization including reactor cavity asymmetric pressurization transients, pump overspeed and traveling pressure waves from the depressurization of the system.

Incorporating leak-before-break criteria and guidelines into the design process maximizes the benefits of applying mechanistic pipe break. Eliminating the dynamic effects permits minimizing the size and number of protective structures and eliminates the use of pipe whip restraints. This permits design optimization and avoids obstruction of pipe welds for in-service inspection by protective structures and restraints.

High-energy ASME Code Section III piping that is evaluated to the leak-before-break criteria is identified in Appendix 3E. This applies to the main steam piping as follows. The main steam piping from the steam generator outlet nozzle to the anchor downstream of the isolation valve is analyzed for applicable loadings including the safe shutdown earthquake. This anchor is at the exterior wall of the auxiliary building. The portion of this piping from the containment penetration flued head inboard weld to the above anchor satisfies the break exclusion zone requirements described in Subsection 3.6.2. The portion of this piping from the steam generator outlet nozzle to flued head inboard weld, including the welds, is evaluated to the leak-before-break criteria. The portion of the auxiliary building flued head (anchor in the wall) that has the same nominal dimensions as the main steam pipe is also classified as a break exclusion zone. High-energy piping that does not satisfy the leak-before-break criteria is designed to the requirements discussed in Subsections 3.6.1 and 3.6.2.

The piping to which mechanistic pipe break is applied is analyzed to demonstrate that the piping has leak-before-break characteristics. The leak-before-break analysis is either a fracture-mechanics based stability analysis or a plastic-instability limit load analysis as appropriate. The analysis combines normal and abnormal (including seismic) loads to determine a critical crack size for a postulated through-wall crack. The critical crack size is compared to the size of a leakage crack for which, with appropriate margin, detection is certain. When the critical crack size is sufficiently larger than the leakage crack size the leak-before-break requirements are satisfied.

Mechanistic pipe break is not used for purposes of specifying non-structural design criteria for emergency core cooling, containment systems, or other non-structural engineered safety features, or for the evaluation of environmental effects including spray wetting, humidity, and adverse reactions with chemicals in the coolant. This includes piping for which leak-before-break is demonstrated.

A bounding analysis is performed for each piping system. The bounding analysis is applied as discussed in Subsection 3.6.4.2 to verify that the as-built piping satisfies the requirements for leak-before-break.

3.6.3.1 Application of Mechanistic Pipe Break Criteria

Piping systems to which mechanistic pipe break are applied are high integrity systems with well understood loading combinations and conditions. The piping systems to which it is applied satisfy the

requirements of the ASME Code, Section III. ASME Code requirements also apply to the pre-service and in-service inspection which confirm continued integrity.

The mechanistic pipe break approach is applicable to high-energy piping provided plant design, operating experience, tests, or analyses have indicated low probability of failure from effects of intergranular stress corrosion cracking, water hammer, steam hammer, fatigue (thermal or mechanical), or erosion.

The plant design and operating features permit the application of the mechanistic pipe break approach. The piping to which the leak-before-break criteria is applied is evaluated for fatigue due to cyclic loads as required by the appropriate requirements of the ASME Code.

The piping in the AP1000 does not operate at temperatures for which creep or creep fatigue must be considered.

The reactor coolant loop piping, branch lines, and other lines in contact with reactor coolant are fabricated of austenitic stainless steel, which is very resistant to erosion and corrosion in typical reactor coolant chemistries and flowrates. Intergranular stress corrosion cracking has not been associated with reactor coolant piping in pressurized water reactors.

The design of the reactor coolant loop is not conducive to the generation of water hammer loads. The reactor coolant loop does not have any valves that could result in a water hammer due to rapid valve closure. The steam bubble in the pressurizer is not subject to the introduction of a large volume of cold water sufficient to result in a bubble collapse water hammer.

The design and component selection of reactor coolant branch lines and other lines evaluated for mechanistic pipe break follow design guidelines intended to minimize the potential for water hammer. Comparison of the AP1000 piping to the screening criteria in Subsection 5.29 of NUREG/CR-6519 (Reference 13) demonstrates that there is not a significant potential for water hammer in the leak-before-break piping.

Thermal stratification of water in stagnant or slowly flowing lines can result in thermal fatigue in a pipe. The piping and system design requirements for AP1000 address the potential for thermal stratification. For additional information of thermal stratification, see Subsections 3.9.3, 5.4.3, and 5.4.5.

The composition of the main steam lines has been selected to minimize the potential for erosion and corrosion. The main steam lines are fabricated from SA335 Grade P11 Alloy steel, which is composed of sufficient levels of chromium to preclude erosion and corrosion mechanisms. The main steam lines are not subject to water hammer or thermal stratification by the nature of the fluid transported.

The steam line is protected from being filled with water due to steam generator overfill by implementation of operating instructions or isolation requirements included in the protection system logic or both. See Section 7.3 for information on the protection system design to prevent overfill.

In addition to requirements on the design, fabrication, and inspection of the piping systems, the application of mechanistic pipe break requires a qualified leak detection capability. Leak detection systems inside containment meet the guidelines of Regulatory Guide 1.45. See Subsection 5.2.5 for a discussion of the leak detection system for the reactor coolant system and connected piping.

3.6.3.2 Design Criteria for Leak-before-Break

The methods and criteria to evaluate leak-before-break in the AP1000 are consistent with the guidance in NUREG-1061 (Reference 11) and Draft Standard Review Plan 3.6.3 (Reference 12) The application of the mechanistic pipe break in AP1000 requires that the following design requirements are met.

- Pre-service inspection of welds is required.

- For ASME Code Class 1, Class 2, and Class 3 systems for which leak-before break is demonstrated, the ASME Code, Section III and Section XI preservice and inservice inspection requirements will provide for the integrity of each system. The weld and welder qualification, and weld inspection requirements for ASME Code, Section III, Class 3 leak-before-break lines are equivalent to the requirements for Class 2. The inservice inspection requirement for each Class 3 leak-before-break line includes a volumetric inspection equivalent to the requirements for Class 2 for the weld at or closest to the high stress location.

- Inservice inspection and testing of snubbers (if used) are performed to provide for a low snubber failure rate.

- For the maximum stress due to steady-state vibration refer to Subsection 3.9.2.

- The leak-before-break bounding analysis curves are developed for each applicable piping system. The bounding analysis methods are described in Appendix 3B. These curves give the design guidance to satisfy the stress limits and leak-before-break acceptance criteria. The highest stressed point (critical location) determined from the piping stress analysis is compared to the bounding analysis curve and has to fall on or under the curve. The points on or under the bounding analysis curve satisfy the requirements for leak-before-break.

  The analyzed normal stress and maximum stress are not required to construct the bounding analysis curve. The analyzed stresses are calculated by the equation;

$$\sigma = \frac{F_x}{A} + \frac{M}{Z}$$

where:

σ is the stress

F_x is the axial force

M is the applied moment

A is the piping cross-sectional area

Z is the piping section modulus.

The normal stress is calculated by the algebraic summation of load combination method and the maximum stress is calculated by the absolute summation of load combination method.

- The corrosion-resistant piping materials, including base metal and welds, have an appropriate toughness. The piping materials containing primary coolant are wrought stainless steel. The welds in stainless steel pipe are made using the gas tungsten arc (GTAW) process. These materials are very resistant to crack extension. The tensile properties for the leak-before-break evaluation are those found in the Section II Appendices of the ASME Code. During the design stage, the material properties used are based on the ASME Code minimum values. During the as-built reconciliation stage, certified material test report values are reviewed to verify that ASME Code requirements are satisfied.

- For those lines fabricated using non-stainless ferritic materials, the materials used and the associated welds have adequate toughness to demonstrate that leak-before-break criteria are satisfied. The welds are made using the gas tungsten arc (GTAW) process. The tensile properties for the leak-before-break evaluation are obtained from actual material tests. During the design stage, the material properties are based on test results. During the as-built reconciliation stage, certified material test report values are reviewed to verify that the toughness and strength requirements of the ASME Code, Section III are satisfied.

- Potential degradation by erosion, erosion/corrosion and erosion cavitation is examined to provide low probability of pipe failure.

- Wall thicknesses in elbows and other fittings are evaluated to confirm that ASME Code, Section III piping requirements are met as a minimum.

- The as-built condition of the piping and support system is evaluated based on the guidelines in EPRI NP-5630 (Reference 10) and reconciled to the analysis of the leak-before-break criteria based on the design information. The locations and characteristics of the supports, including any gaps between the supports and piping, or other configurations that result in a nonlinear response are included in the as-built evaluation.

- Adjacent structures and components are designed for the safe shutdown earthquake event to provide low probability of indirect pipe failure.

- The piping supports are anchored to reinforced concrete structures, to concrete-filled steel plate structures, or to steel structures anchored to these types of structures. Piping is not supported by masonry block walls.

3.6.3.3 Analysis Methods and Criteria

The methods used to develop the bounding analysis curves are described in Appendix 3B. Development of the bounding analysis curves provides an evaluation method that is consistent with NRC requirements and guidance. The calculation method and computer codes used for AP1000 are benchmarked to test data and have been previously accepted by the NRC for leak-before-break evaluations in operating nuclear power plants.

Analyzable sections run from one terminal end or anchor to another terminal end or anchor. A terminal end is typically a connection to a larger pipe or a component. For the structural analysis, a normally closed valve between pressurized and unpressurized portions of a line is not considered a terminal end. Figure 3.6-3 is a schematic of a portion of a piping system that illustrates the meaning of analyzable segments. In the figure the analyzable portion of the pipe runs from point A to point D.

The leak-before-break evaluation is based on a fracture mechanics stability analysis comparing the selected leakage crack to the critical crack size. The following discussion outlines the analysis method.

The development of leak-before-break bounding analysis curves assume that circumferentially oriented postulated cracks are limiting. Stability is established by analyzing through-wall flaws.

Leakage Flaw

Through-wall flaws in candidate leak-before break piping systems are postulated. [*The size of the postulated flaws are large enough so that the leakage is detectable with adequate margin, using 10 times the minimum installed leak detection capability when the pipes are subjected to normal operational loads combining by algebraic sum method.*]* That is, the size of the leakage flaw postulated would be expected to have a leak rate 10 times the size of the rated leak rate detection capability.

As noted in Subsection 5.2.5, the rated capability of the leak detection systems for the primary coolant inside containment is 0.5 gpm. The methods used to detect leakage are described in Subsection 5.2.5.3. The methods used for primary coolant are the containment sump level, inventory balance, and containment atmosphere radiation. The method used to detect leakage from the main steam line inside containment is the containment sump level. Containment air cooler condensate flow, and containment atmosphere pressure, temperature, and humidity also provide an indication of possible leakage.

Stability and Critical Flaw Sizes

The local and global failure mechanisms are evaluated, as appropriate, to provide margin on flaw size and load. The local mode of failure addresses crack tip behavior: blunting, initiation, extension, and instability. The local failure mechanism is evaluated for ferritic steel piping systems using the J-integral method. The global mode of failure addresses the behavior of the net section: initial yielding, strain hardening, and plastic hinge formation. The global failure mechanism (limit load method) is evaluated for stainless steel piping with no cast material and GTAW welding. From these evaluations a critical crack size is determined. That is, a crack larger than the critical crack size would have unstable growth characteristics.

Acceptance Standards

[*The results of the preceding evaluations are compared to show that the critical flaw size, which is shown to be stable when the maximum loads are combined based on individual absolute values, is at least twice the size (to satisfy margin of 2 on flaw size) of the leakage flaw size. To satisfy a margin on load of 1.0, the maximum loads are combined using absolute summation of individual values.*]* The maximum loads are described in Appendix 3B Subsection 3B.3.3.

Bounding Analyses

Evaluations are provided for each different combination of material type, pipe size, pressure, and temperature. These evaluations are used to develop a set of curves of maximum faulted stress versus the corresponding normal stress that satisfy the criteria for leak-before-break. These curves are used in the design of the piping systems and will be used to verify that the as-built piping satisfies the requirements for leak-before-break as discussed in Subsection 3.6.4.2.

3.6.3.4 Documentation of Leak-before-Break Evaluations

The leak-before-break evaluation is used to support the elimination of dynamic effects of pipe breaks from the loading conditions for the piping analysis. An evaluation of leak-before-break using the as-built configuration of the piping system and supports is required as part of the Design Report (also referred to as LBB evaluation report where applicable) of the as-built configuration required to meet ASME Code requirements and LBB criteria. Appendix 3B contains a discussion of the bounding analysis methods for the leak-before-break evaluation.

*NRC Staff approval is required prior to implementing a change in this information.

The analysis methods, criteria, and loads used for evaluation of stress in piping systems are outlined in Subsections 3.7.3 and 3.9.3.

3.6.4 Combined License Information

3.6.4.1 Pipe Break Hazard Analysis

The as-built reconciliation of the pipe break hazards analysis and as-built pipe rupture hazard analysis are addressed in APP-GW-GLR-021 (Reference 14).

The as-designed pipe rupture hazards evaluation is made available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets protected by associated mitigation features (Reference is Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate.

A pipe rupture hazard analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The report will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design.

The pipe whip restraint and jet shield design includes the properties and characteristics of procured components connected to the piping, components, and walls at identified break and target locations. The design will be completed prior to installation of the piping and connected components.

The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in Subsections 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load (in accordance with DCD Tier 1 Table 3.3-6, item 8).

This COL item is also addressed in Subsection 14.3.3.

3.6.4.2 Leak-before-Break Evaluation of As-Designed Piping

The leak-before-break evaluation of the as-designed piping is addressed in APP-GW-GLR-022 (Reference 15).

3.6.4.3 Leak-before-Break Evaluation of As-Built Piping

Not used.

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

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

3.6.5 References

1. NUREG/CR-2913, “Two-Phase Jet Loads,” January 1983.

2. WCAP-8077 (Proprietary) and WCAP-8078 (Nonproprietary), “Ice Condenser Containment Pressure Transient Analysis Methods,” March 1973.

3. ASME/ANSI-B31.1-1989 Edition, “Power Piping,” including 1989 Addendum.

4. ANSI/ANS-58.2-1988, “Design Bases for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture.”

5. Moody, F. J., Fluid Reaction and Impingement Loads, paper presented at the ASCE Specialty Conference, Chicago, December 1973.

6. “MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics,” WCAP-8708 (Proprietary) and WCAP-8709 (Nonproprietary), February 1976.

7. WCAP-8252, “Documentation of Selected Westinghouse Structural Analysis Computer Codes,” Revision 1, May 1977.

8. Not used.

9. Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill Book Company, New York, 1964.

10. EPRI NP-5630, “Guidelines for Piping System Reconciliation” (NCIG-05, Revision 1), May 1988.

11. NUREG-1061, Volume 3, Report of the U. S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks, November 1984.

12. 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.

13. NUREG/CR-6519, Screening Reactor Steam/Water Systems for Water Hammer, November 1996.

14. APP-GW-GLR-021, “AP1000 As-Built COL Information Items,” Westinghouse Electric Company LLC.

15. APP-GW-GLR-022, “AP1000 Leak-Before-Break Evaluation of As-Designed Piping,” Westinghouse Electric Company LLC.

Table 3.6-1
High-Energy and Moderate-Energy Fluid Systems Considered for Protection of Essential Systems[(a)]

System	High-Energy	Moderate-Energy
Reactor coolant (RCS)	•	
Steam generator (SGS)[(b)]	•	
Passive core cooling (PXS)	•	
Passive containment cooling (PCS)[(c)]		•
Main control room habitability (VES)	•	
Chemical and volume control (CVS)	•	
Primary sampling (PSS)	•	
Compressed and instrument air (CAS)		•
Normal residual heat removal (RNS)[(a)]		•
Component cooling water (CCS)		•
Spent fuel pit cooling (SFS)		•
Demineralized water (DWS)		•
Liquid radwaste (WLS)		•
Radioactive drain (WRS)		•
Central chilled water (VWS)[(a)]		•
Fire protection (FPS)		•
Steam generator blowdown (BDS)[(d)]	•	
Main and startup feedwater (FWS)[(d)]	•	
Main steam (MSS)[(d)]	•	
Hot water heating (VYS)		•

Notes:

a. Systems included on this list are high-energy or moderate-energy fluid systems located in the containment or the auxiliary building. Systems that operate at or close to atmospheric pressure such as ventilation and gravity drains are not included. The normal residual heat removal system lines are classified as moderate-energy based on the 1 percent rule. These lines experience high-energy conditions for less than 1 percent of the plant operating time. The portions of the normal residual heat removal system from the connections to the reactor coolant system and passive core cooling system to the first closed valve in each line are high energy. The spent fuel pit cooling system is classified as moderate energy based on the 2 percent rule. These systems experience high-energy conditions for less than 2 percent of the system operating time. See Subsection 3.6.1.1 Item A and Subsection 3.6.1.2 for additional information.

b. Main and startup feedwater, main steam, and steam generator blowdown lines located in the containment and auxiliary building are part of the steam generator system.

c. The essential portion of the system is at atmospheric pressure.

d. The portion of these systems in the turbine building adjacent to the auxiliary building are evaluated for the effect of a circumferential or longitudinal break on the main control room.

Table 3.6-2 (Sheet 1 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	**Room Number**	**Description**	**Terminal End Break Location Excluded by LBB**	**Description**	**Terminal End Break Location**
Steam Generator Compart-ment 1	11201	22 in. Cold Leg (RCS)	RC Pump Nozzles (2)	4 in. Pressurizer Spray (RCS)	Cold Leg Nozzles (2)
		18 in. Fourth Stage ADS (RCS)	Hot Leg Nozzle		
	11301	31 in. Hot Leg (RCS)	SG Nozzle	3 in. Purification (CVS)	3 in. SG Channel Head Nozzle
		18 in. Surge Line (RCS)	Hot Leg Nozzle		
		18 in. & 14 in. Fourth Stage ADS (RCS)	Valves: V004A/C		
		14 in. PRHR Return (RCS)	SG Channel Head Nozzle		
	11401	None		4 in. SG Blowdown (SGS)	4 in. SG Nozzle
	11501	None		None	
	11601			16 in and 20 in. Feedwater (SGS) 6 in. Startup Feedwater (SGS)	SG Nozzle SG Nozzle
	11701	38 in. Main Steam (SGS)	SG Nozzle	None	

Table 3.6-2 (Sheet 2 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
Steam Generator Compart-ment 2	11202	22 in. Cold Leg (RCS)	RC Pump Nozzles (2)	None	
		18 in. Fourth Stage ADS (RCS)	Hot Leg Nozzle		
		20 in. Normal RHR (RCS)	Hot Leg Nozzle		
		12 in. Normal RHR (RCS)	20 in. x 12 in. Reducer (This is not a terminal end)		
	11302	31 in. Hot Leg (RCS)	SG Nozzle	None	
		18 in. & 14 in. Fourth Stage ADS (RCS)	Valves: V004B/D		
		8 in. Cold Leg to CMT (RCS)	Cold Leg Nozzles (2)		
	11402	None		4 in. SG Blowdown (SGS)	4 in. SG Nozzle
	11502	None		None	
	11602			16 in. and 20 in. Feedwater (SGS) 6 in. Startup Feedwater (SGS)	SG Nozzle SG Nozzle

Table 3.6-2 (Sheet 3 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	**Room Number**	**Description**	**Terminal End Break Location Excluded by LBB**	**Description**	**Terminal End Break Location**
	11702	38 in. Main Steam (SGS)	SG Nozzle	None	
Reactor Vessel Nozzle Area	11205	31 in. Hot Leg (RCS)	Reactor Vessel Nozzles (2)	None	
		22 in. Cold Leg (RCS)	Reactor Vessel Nozzles (4)		
		8 in. Direct Vessel Injection (RCS)	Reactor Vessel Nozzles (2)		
PXS Valve and Accumulator Room A	11206	8 in. Accumulator Injection (PXS)	Accumulator Nozzle	None	
		8 in. CMT Injection (PXS)	CMT Nozzle		
		6 in. Line from Normal RHR (RNS)	Valve: V017A		
		8 in. Line from IRWST (PXS)	Valves: V125A & V123A		
PXS Valve Room B	11207 PXS	6 in. Line from Normal RHR (RNS)	Valve: V017B	None	
		8 in. Line from IRWST (PXS)	Valves: V125B & V123B		
Accumulator Room B	11207 ACCUM	8 in. Accumulator Injection (PXS)	Accumulator Nozzle	None	
		8 in. CMT Injection (PXS)	CMT Nozzle		
RNS Valve Room	11208	10 in. Normal RHR (RNS)	Valves: V001A/B	None	

Table 3.6-2 (Sheet 4 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	**Room Number**	**Description**	**Terminal End Break Location Excluded by LBB**	**Description**	**Terminal End Break Location**
Vertical Access	11204	None		3 in. Line from Regen HX to SG 01 (CVS)	Anchor to Wall
				3 in. Purification from Cold Leg to Regen HX (CVS)	Anchor to Wall
RNS Valve Room	11208	10 in. Normal RHR (RNS)	Valves: V001A/B	None	
Lower Pressurizer Compartment	11303	18 in. Surge Line (RCS)	Pressurizer Nozzle	None	
Upper Pressurizer Compartment	11503	14 in. ADS (RCS)	Pressurizer Nozzle (2)	4 in. Pressurizer Spray (RCS)	Pressurizer Nozzle
Lower ADS Valve Area	11603	14 in. & 8 in. ADS (RCS) 6 in. Pressurizer Safety (RCS)	Valves: V012B & V013B Valves: V005A and V005B 14 in. x 6 in. Tees (2)	4 in. ADS (RCS)	Valve V0011B & 14 in. x 4 in. Branch
Upper ADS Valve Area	11703	14 in. & 8 in., ADS (RCS)	Valves: V012A & V013A	4 in. ADS (RCS)	Valve V0011A & 14 in. x 4 in. Branch
Maintenance Floor/ Mezzanine	11400	38 in. Main Steam (SGS)	Non-terminal End Location (2) at Boundary of Break Exclusion Zone	6 in. Startup Feedwater (SGS)	Anchors (2) at Containment Penetration
		14 in. Passive RHR (PXS)	PRHR HX Inlet Nozzle		
		8 in. CMT Balance Line Piping	CMT Nozzles (2)		

Table 3.6-2 (Sheet 5 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	Room Number	Description	Terminal End Break Location Excluded by LBB	Description	Terminal End Break Location
SG01 Access Room	11304	None		None	
Pressurizer Spray Valve Room	11403	None		None	
Maintenance Floor	11300	14 in. Passive RHR (PXS)	PRHR HX Outlet Nozzle	None	
Operating Deck	11500	None		None	
CVS Room	11209	None		3 in. Purification from Pressurizer Spray to Regen HX (CVS)	Regen HX Nozzle
				3 in. Return, Auxiliary Spray (CVS)	Regen HX Nozzle
				3 in. Return to RNS from Regen HX (CVS)	Valve: V079
				3 in. Supply from RNS to Letdown HX (CVS)	Valve: V072
				3 in. Supply from Regen HX to Letdown HX (CVS)	Nozzles: Regen HX, Letdown HX
CVS Room	11209 Pipe Chase	None		3 in. Purification from Anchor to Regen HX	Anchor
				3 in. Return from Regen HX to Anchor (CVS)	Anchor
				4 in. SG Blowdown (SGS)	Anchors (2) at Containment Penetration

Table 3.6-2 (Sheet 6 of 7)
Subcompartments and Postulated Pipe Ruptures

Compartment		Lines Evaluated to LBB		Lines Not Evaluated to LBB	
Name	**Room Number**	**Description**	**Terminal End Break Location Excluded by LBB**	**Description**	**Terminal End Break Location**
Reactor Coolant Drain Tank Room	11104	None		None	
Reactor Vessel Cavity	11105	None		None	
MSIV Compartment B	12504/ 12404	None		Main Steam Main Feedwater Startup Feedwater Lines[(a)]	Longitudinal Cracks with Crack Flow Areas of 1 Square Foot are Postulated
MSIV Compartment A	12506/ 12406	None		Main Steam Main Feedwater Startup Feedwater Lines[(a)]	Longitudinal Cracks with Crack Flow Areas of 1 Square Foot are Postulated
Valve/Piping Penetration Room	12306	None		4 in. Steam Generator Blowdown[(a)]	Anchors (2) at Containment Penetrations Anchors (2) at Wall to Turbine Building

Note:

a. The piping in these areas is included in break exclusion zones. For additional information on the evaluation of these lines, see Subsection 3.6.1.2.1 for the steam generator blowdown line; Subsection 3.6.1.2.2 for information on the evaluation of lines in MSIV compartment B because of the proximity to the main control room; and Subsection 3.6.2.1.1.4 for general break exclusion zone requirements.

Table 3.6-2 (Sheet 7 of 7)
Subcompartments and Postulated Pipe Ruptures

Room #	Description	Bottom Elevation	Top Elevation
11104	RCDT Room	66′-6″	81′-0″
11105	Reactor Vessel Cavity	66′-6″	98′
11205	Reactor Vessel Nozzle Area	98′	107′-2″
11201	SG Compartment 1	83′	104′-7″
11202	SG Compartment 2	83′	104′-7″
11204	Vertical Access	83′	107′-2″
11206	PXS Valve Room A	87′-6″	105′-2″
11300	Maintenance Floor	107′-2″	118′-6″
11301	SG Compartment 1	104′-7″	116′-6″
11302	SG Compartment 2	104′-7″	116′-6″
11400	Maintenance Floor/Mezzanine	118′-6″	133′-3″
11401	SG Compartment 1	116′-6″	135′-3″
11402	SG Compartment 2	116′-6″	135′-3″
11501	SG Compartment 1	135′-3″	153′-0″
11502	SG Compartment 2	135′-3″	153′-0″
11601	SG Compartment 1	153′-0″	166′-4″
11602	SG Compartment 2	153′-0″	166′-6″
11701	SG Compartment 1	166′-4″	----
11702	SG Compartment 2	166′-4″	----
11500	Operating Deck	135′-3″	281′-8 3/8″
11303	Pressurizer Lower Compartment	107′-2″	135′-3″
11304	SG01 Access Room	107′-2″	118′-6″
11403	Pressurizer Spray Valve Room	118′-6″	133′-3″
11503	Pressurizer Upper Compartment	135′-3″	166′-1.5″
11603	Lower ADS Valve Area	166′-1.5″	176′-10.5″
11703	Upper ADS Valve Area	176′-10.5″	----
11207 ACCUM	Accumulator Room B	87′-6″	105′-2″
11207 PXS	PXS Valve Room B	87′-6″	105′-2″
11208	RNS Valve Room	94′	105′-2″
11209	CVS Room	80′-6″	105′-2″
11209 PIPE	CVS Room Pipe Tunnel	100′-0″	105′-2″
12306	Valve/Piping Penetration Room	100′-0″	117′-6″
12504/12404	MSIV Compartment B (Upper/Lower)	117′-6″	153′-0″
12506/12406	MSIV Compartment A (Upper/Lower)	117′-6″	153′-0″

Table 3.6-3 (Sheet 1 of 7)
NI Rooms With Pipe Whip Restraints and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Pipe Whip Restraint	Hazard Source/Room	Essential Target Description/Room
11201	Steam Generator Compartment-01, Below the Lower Manway	PWR-RCS002	Reactor Coolant System (RCS)-Pressurizer Spray Line, 4" L110A: Terminal End Break at RCS Cold Leg L002A.	Raceways and cables. Passive Core Cooling System (PXS) containment level instrumentation. Steam Generator System (SGS) level instrumentation. RCS pressurizer instrumentation. Reactor coolant loop (RCL) (steam generator, pumps, hot leg, and cold legs) and branch line piping/valves.
		PWR-RCS003	RCS-Pressurizer Spray Line, 4" L106: Terminal End Break at RCS Cold Leg L002B.	Raceways and cables. PXS containment level instrumentation. SGS level instrumentation. RCS pressurizer instrumentation. RCL branch line piping/valves.
11209 Chase	Pipe Chase to CVS Equipment Room	PWR-SGS004	SGS-Blowdown Line, 4" L009A: Terminal End Break at Containment Penetration P27.	SGS blowdown piping (L009B). CVS makeup piping (L056). CVS letdown piping (L049). CVS hydrogen supply piping (L215). Liquid Radwaste System (WLS) containment sump piping (L072).

Table 3.6-3 (Sheet 2 of 7)
NI Rooms With Pipe Whip Restraints and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Pipe Whip Restraint	Hazard Source/Room	Essential Target Description/Room
		PWR-SGS008	SGS-Blowdown Line, 4" L009B: Terminal End Break at Containment Penetration P28.	CVS makeup piping (L056). CVS letdown piping (L049). CVS hydrogen supply piping (L215).
		PWR-CVS056	CVS-Makeup Line, 3" L056: Terminal End Break at In-Line Anchor.	SGS blowdown piping (L009B). CVS makeup valve (CVS-V091), (Room 11300).
11300	Maintenance Floor	PWR CVS047 A/B	CVS-Letdown Line, 2" L049: Terminal End Break at inlet to Valve V059.	Raceways and cables (Rooms 11300 and 11400). SGS MB01 level instrumentation piping (Room 11400). CVS makeup valves (CVS-V091 and V100). CVS hydrogen supply valves (CVS-V215, V216, V217, and V218). WLS containment sump valve (WLS-V055). RCS pressurizer instrumentation.
11301	Steam Generator Compartment-01, Lower Manway Area	PWR-RCS001 A/B	CVS-Makeup Line, 3" RCS L112, Terminal End Break at Steam Generator MB01.	Steam generator MB01 support. RCS Passive Residual Heat Removal (PRHR) Heat Exchanger (HX) return piping (L113).
		PWR-SGS003	SGS-SG Blowdown 4" SGS-L009A (Room 11401)	Raceways and cables (Room 11201). PXS containment level instrumentation (Room 11201). SGS level instrumentation (Room 11201). RCS pressurizer instrumentation (Room 11201). RCL (steam generator, pumps, hot leg, and cold legs) and branch line piping/valves (Room 11201). Steam generator MB01 support. RCS PRHR return piping (L113).

Table 3.6-3 (Sheet 3 of 7)
NI Rooms With Pipe Whip Restraints and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Pipe Whip Restraint	Hazard Source/Room	Essential Target Description/Room
11302	Steam Generator Compartment-02, Lower Manway Area	PWR-SGS007	SGS- SG Blowdown 4" SGS-L009B (Room 11402).	SGS level instrumentation (Rooms 11202, 11302). RCL and drain line piping/valves (Rooms 11202, 11302). Steam generator MB02 support (Rooms 11202, 11302). Raceways and cables (Rooms 11202, 11302). PXS containment recirculation screen (Room 11202).
11400	Maintenance Floor Mezzanine	PWR-SGS002 A/B/C	SGS-Startup Feedwater Line, 6" L005A: Terminal End Break at Containment Penetration P44.	PXS PRHR HX ME01 upper head. PXS PRHR HX ME01 upper head vent and drain piping and valves (PXS-V102A/B). SGS MB01 level instrumentation piping. Raceways and cables.
11402	Steam Generator Compartment-02, Tube Sheet Area	PWR-SGS006A/B	SGS-Startup Feedwater Line, 6" L005B: Terminal End Break at Containment Penetration P45 (Room 11400).	Steam generator MB02 and supports. Primary Sampling System (PSS) core makeup tank (CMT) (MT02B) sample piping and valves (PSS V005B/C), (Room 11400). PXS CMT MT02B vent piping and valve (PXS-V030B), (Room 11400). PXS CMT MT02B balance line piping and valve (PXS-V002B), (Room 11400). PXS CMT MT02B sample line piping and valve (PXS-V010B), (Room 11400). PXS CMT MT02B makeup line piping (PXS-L012B), (Room 11400). PXS CMT MT02B sample line piping (PXS-L011B), (Room 11400). Raceways and cables, (Rooms 11400 and 11402).

Table 3.6-3 (Sheet 4 of 7)
NI Rooms With Pipe Whip Restraints and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Pipe Whip Restraint	Hazard Source/Room	Essential Target Description/Room
11503	Upper Pressurizer Compartment	PWR-RCS006	RCS-Pressurizer Spray Line, 4" L215: Terminal End Break at Pressurizer Nozzle.	ADS Stage 1, 2, and 3 valves (RCS-V001B, RCS-V002B, RCSV003B, RCS-V011B, RCS-V012B, and RCS-V013B), (Room 11603). ADS Stage 1, 2, and 3 valves (RCS-V001A, RCS-V002A, RCSV003A, RCS-V011A, RCS-V012A, and RCS-V013A), (Room 11703). ADS Stage 1, 2, and 3 support steel (Rooms 11503, 11603, and 11703). RCS pressurizer support steel (Room 11503).
11601	Steam Generator-01, Feedwater Nozzle Area	PWR-SGS001	SGS-Startup Feedwater Line, 6" L005A: Terminal End Break at Steam Generator MB01 Nozzle.	RCS head vent piping/valves. RCS ADS stage 1, 2, 3 discharge header (RCS-L064B). SGS level instrumentation piping (Rooms 11601, 11501, 11401, 11301, and 11201). RCS ADS stage 4 valves (RCS-V004A, RCS-V004C, RCS-V014A, and RCS-V014C), (Room 11401). Raceways and cables (Rooms 11601, 11501, 11401, 11301, 11201, 11603, and 11703). Steam generator MB01 supports (Rooms 11601, 11401, 11301, and 11201). RCL (steam generator, pumps, hot leg, and cold legs) and branch line piping/valves (Rooms 11601, 11501, 11401, 11301, and 11201).

Table 3.6-3 (Sheet 5 of 7)
NI Rooms With Pipe Whip Restraints and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Pipe Whip Restraint	Hazard Source/Room	Essential Target Description/Room
				RCS ADS Stage 1, 2, and 3 piping, valves and support steel (Rooms 11503, 11603, and 11703). RCS pressurizer support steel (Room 11503). RCS pressurizer spray line (Room 11503). RCS pressurizer level instrumentation (Room 11503).
		PWR-SGS021	SGS-Main Feedwater Line, 16" L003A: Terminal End Break at Steam Generator MB01 Nozzle.	RCS ADS stage 1, 2, 3 discharge header (RCS-L064B). SGS level instrumentation piping (Rooms 11601, 11501, 11401, 11301, and 11201). RCS ADS stage 4 valves (RCS-V004A, RCS-V004C, RCS-V014A, and RCS-V014C), (Room 11401). Raceways and cables (Rooms 11601, 11501, 11401, 11301, 11201, 11603, and 11703). Steam generator MB01 supports (Rooms 11601, 11401, 11301, and 11201). RCL and branch line piping/valves (Rooms 11601, 11501, 11401, 11301, and 11201). RCS ADS Stage 1, 2, and 3 piping, valves and support steel (Rooms 11503, 11603, and 11703). RCS pressurizer support steel (Room 11503). RCS pressurizer spray line (Room 11503). RCS pressurizer level instrumentation (Room 11503).

Table 3.6-3 (Sheet 6 of 7)
NI Rooms With Pipe Whip Restraints and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Pipe Whip Restraint	Hazard Source/Room	Essential Target Description/Room
11602	Steam Generator-02, Feedwater Nozzle Area	PWR-SGS005	SGS-Startup Feedwater Line, 6" L005B: Terminal End Break at Steam Generator MB02 Nozzle.	SGS level instrumentation piping (Rooms 11602, 11502, 11402, 11302, and 11202). RCS ADS stage 4 valves (RCS-V004B, RCS-V004D, RCS-V014B, and RCS-V014D), (Room 11402). Raceways and cables (Rooms 11602, 11502, 11402, 11302, 11202). Steam generator MB02 supports (Rooms 11602, 11402, 11302, and 11202). RCL and branch line piping/valves (Rooms 11602, 11502, 11402, 11302, and 11202). PXS containment recirculation screen (Room 11202).
		PWR-SGS022	SGS-Main Feedwater line, 16" L003B: Terminal End Break at Steam Generator MB02 Nozzle.	SGS level instrumentation piping (Rooms 11602, 11502, 11402, 11302, and 11202). RCS ADS stage 4 valves (RCS-V004B, RCS-V004D, RCS-V014B, and RCS-V014D), (Room 11402). Raceways and cables (Rooms 11602, 11502, and 11402). Steam generator MB02 supports (Rooms 11602, 11402, 11302, 11202). RCL and branch line piping/valves (Rooms 11602, 11502, 11402, 11302, and 11202). PXS containment recirculation screen (Room 11202).

Table 3.6-3 (Sheet 7 of 7)
NI Rooms With Pipe Whip Restraints and Corresponding Hazard Sources and Essential Targets

Room Number	Room Description	Pipe Whip Restraint	Hazard Source/Room	Essential Target Description/Room
11603	Lower ADS Valve Area	PWR-RCS108 A/B	RCS-Automatic Depressurization System Stage 1 Line, 4" L010B: Terminal End Break at Inlet to Valve RCS V011B.	ADS Stage 2 and 3 valves (RCS-V002B, RCS-V003B, RCS-V012B, and RCS-V013B). ADS Stage 1, 2, and 3 support steel (Rooms 11503, 11603, and 11703). RCS pressurizer support steel (Room 11503). Raceways and cables (Rooms 11603 and 11703).
		PWR-RCS107	RCS-Automatic Depressurization System Stage 1 Line, 4" L010B: Terminal End Break at Outlet of 14x4” Tee.	RCS ADS Stage 3 valve (RCS-V013B). ADS Stage 1, 2, and 3 support steel (Rooms 11503, 11603, and 11703). RCS pressurizer support steel (Room 11503). Raceways and cables (Rooms 11603 and 11703).
11703	Upper ADS Valve Area	PWR-RCS106 A/B	RCS-Automatic Depressurization System Stage 1 Line, 4" L010A: Terminal End Break at Inlet to Valve RCS V011A.	RCS ADS Stage 2 and 3 valves (RCS-V002A, RCS-V003A, RCS-V012A, and RCS-V013A). ADS Stage 1, 2, and 3 support steel (Rooms 11503, 11603, and 11703). RCS pressurizer support steel (Room 11503). Raceways and cables (Rooms 11603, 11703).
		PWR-RCS105	RCS-Automatic Depressurization System Stage 1 Line, 4" L010A: Terminal End Break at Outlet of 14x4” Tee.	ADS Stage 1, 2, and 3 support steel (Rooms 11503, 11603, 11703). RCS ADS Stage 3 valve (RCS-V013A). RCS pressurizer support steel (Room 11503). Raceways and cables (Rooms 11603, 11703).

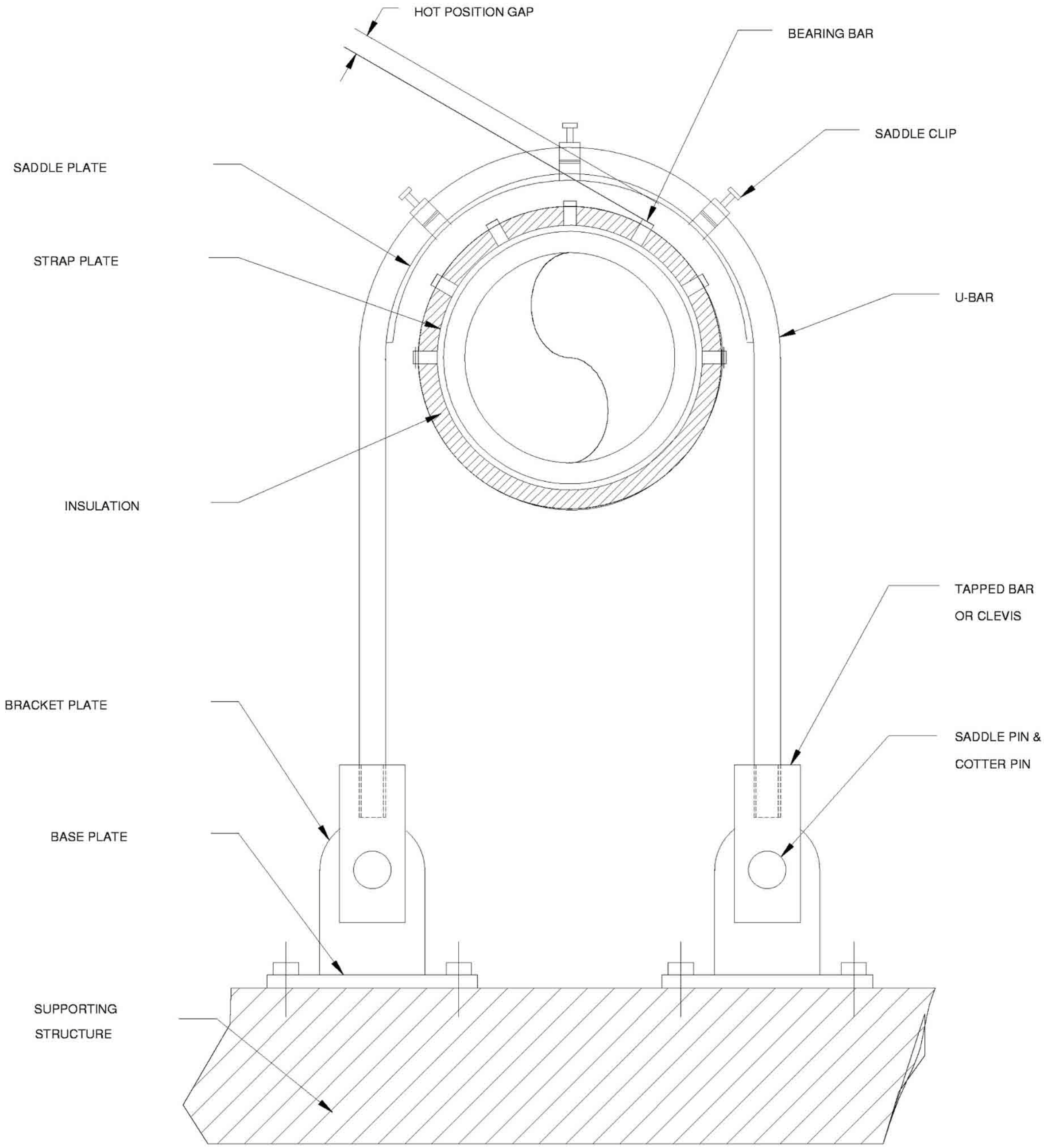


Figure 3.6-1
Typical U-Bar Restraint

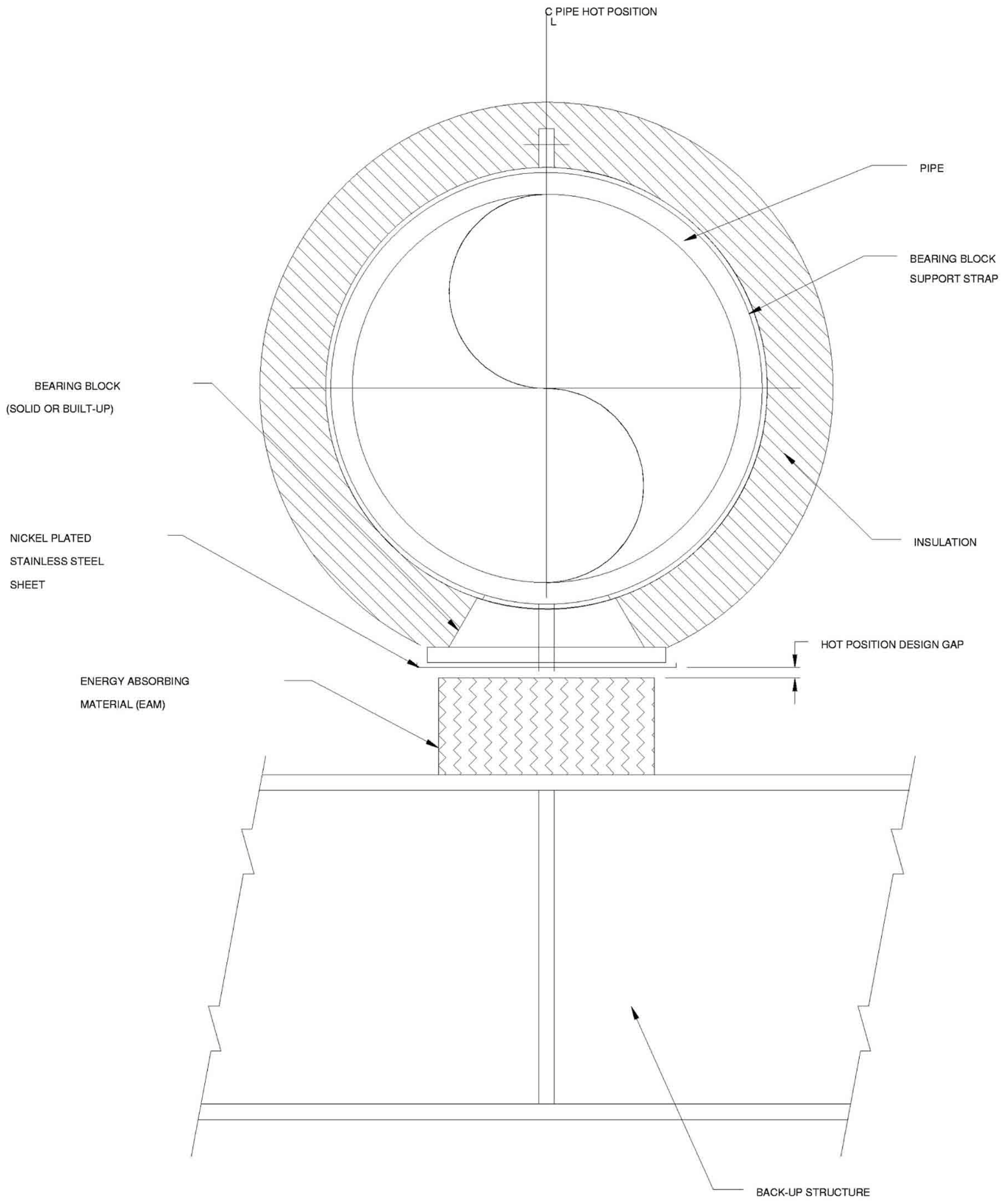
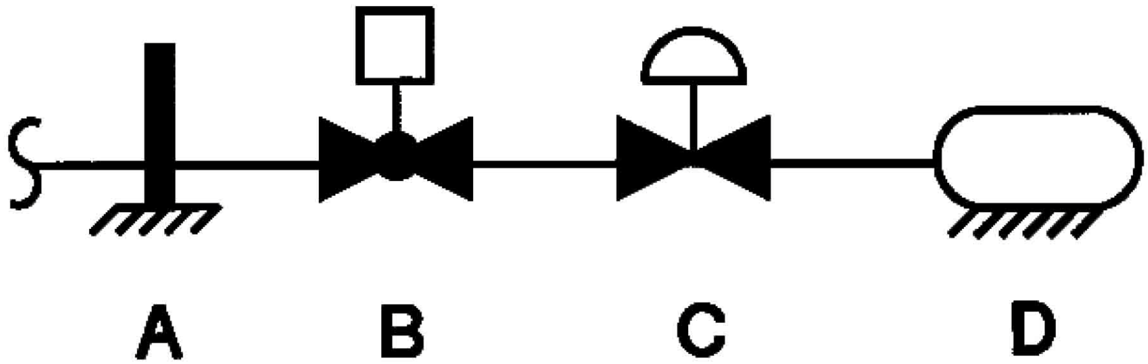


Figure 3.6-2
Typical Energy Absorbing Material Restraint



A – Anchor
B – Closed Valve
C – Closed Valve
D – Terminal End

A to B – High Energy
B to D – Moderate Energy

Figure 3.6-3
Terminal Ends Definitions

3.7 Seismic Design

Plant structures, systems, and components important to safety are required by General Design Criterion (GDC) 2 of Appendix A of 10 CFR 50 to be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

Each plant structure, system, equipment, and component is classified in an applicable seismic category depending on its function. A three-level seismic classification system is used for the AP1000: seismic Category I, seismic Category II, and nonseismic. The definitions of the seismic classifications and a seismic classifications listing of structures, systems, equipment, and components are presented in Section 3.2.

Seismic design of the AP1000 seismic Categories I and II structures, systems, equipment, and components is based on the safe shutdown earthquake (SSE). The safe shutdown earthquake is defined as the maximum potential vibratory ground motion at the generic plant site as identified in Section 2.5.

The operating basis earthquake (OBE) has been eliminated as a design requirement for the AP1000. Low-level seismic effects are included in the design of certain equipment potentially sensitive to a number of such events based on a percentage of the responses calculated for the safe shutdown earthquake. Criteria for evaluating the need to shut down the plant following an earthquake are established using the cumulative absolute velocity approach according to EPRI Report NP-5930 (Reference 1) and EPRI Report TR-100082 (Reference 17). For the purposes of the shutdown criteria in Reference 1 the operating basis earthquake for shutdown is considered to be one-third of the safe shutdown earthquake.

Seismic Category I structures, systems, and components are designed to withstand the effects of the safe shutdown earthquake event and to maintain the specified design functions. Seismic Category II and nonseismic structures are designed or physically arranged (or both) so that the safe shutdown earthquake could not cause unacceptable structural interaction with or failure of seismic Category I structures, systems, and components.

3.7.1 Seismic Input

The geologic and seismologic considerations of the plant site are discussed in Section 2.5.

The peak ground acceleration of the safe shutdown earthquake, now referred to as the Certified Seismic Design Response Spectra (CSDRS), has been established as 0.30g for the AP1000 design. The vertical peak ground acceleration is conservatively assumed to equal the horizontal value of 0.30g as discussed in Section 2.5.

3.7.1.1 Design Response Spectra

The AP1000 design response spectra of the safe shutdown earthquake, now referred to as the Certified Seismic Design Response Spectra (CSDRS), are provided in Figures 3.7.1-1 and 3.7.1-2 for the horizontal and the vertical components, respectively.

The horizontal design response spectra for the AP1000 plant are developed, using the Regulatory Guide 1.60 spectra as the base and several evaluations to investigate the high frequency amplification effects. These evaluations included:

- Comparison of Regulatory Guide 1.60 spectra with the spectra predicted by recent eastern U.S. spectral velocity attenuation relations (References 23, 24, 25, and 26) using a suite of magnitudes and distances giving a 0.3 g peak acceleration

- Comparison of Regulatory Guide 1.60 spectra with the 10^{-4} annual probability uniform hazard spectra developed for eastern U.S. nuclear power plants by both Lawrence Livermore National Laboratory (Reference 27) and Electric Power Research Institute (Reference 28)

- Comparison of Regulatory Guide 1.60 spectra with the spectra of 79 additional old and newer components of strong earthquake time histories not considered in the original derivation of Regulatory Guide 1.60

Based on the above described evaluations, it is concluded that the eastern U.S. seismic data exceed Regulatory Guide 1.60 spectra by a modest amount in the 15 to 33 hertz frequency range when derived either from published attenuation relations or from the 10^{-4} annual probability of exceedance uniform hazard spectra at eastern U.S. sites. This conclusion is consistent with findings of other investigators that eastern North American earthquakes have more energy at high frequencies than western earthquakes. Exceedance of Regulatory Guide 1.60 spectra at the high frequency range, therefore, would be expected since Regulatory Guide 1.60 spectra are based primarily on western U.S. earthquakes. The evaluation shows that, at 25 hertz (approximately in the middle of the range of high frequencies being considered, and a frequency for which spectral amplitudes are explicitly evaluated) the mean-plus-one-standard-deviation spectral amplitudes for 5 percent damping range from about 2.1 to 4 cm/sec and average 2.7 cm/sec. Whereas, the Regulatory Guide 1.60 spectral amplitude at the same frequency and damping value equal just over 2 cm/sec.

It is concluded, therefore, that an appropriate augmented 5 percent damping horizontal design velocity response spectrum for the AP1000 project is one with spectral amplitudes equal to the Regulatory Guide 1.60 spectrum at control frequencies 0.25, 2.5, 9 and 33 hertz augmented by an additional control frequency at 25 hertz with an amplitude equal to 3 cm/sec. This spectral amplitude equals 1.3 times the Regulatory Guide 1.60 amplitude at the same frequency. The additional control point's spectral amplitude of other damping values were determined by increasing the Regulatory Guide 1.60 spectral amplitude by 30 percent.

The AP1000 design vertical response spectrum is, similarly, based on the Regulatory Guide 1.60 vertical spectra at lower frequencies but is augmented at the higher frequencies equal to the horizontal response spectrum.

The AP1000 design response spectra's relative values of spectrum amplification factors for control points are presented in Table 3.7.1-3.

The design response spectra are applied at the foundation level in the free field at hard rock sites and at the finished grade in the free field at firm rock and soil sites. The resulting peak horizontal ground acceleration values are above 0.1g. This satisfies 10 CFR Part 50, Appendix S, which requires that the horizontal component of the SSE ground motion in the free-field at the foundation elevation (that is, bottom of foundation) has a peak ground acceleration of at least 0.1g together with an appropriate response spectrum. The definitions (characteristics) of hard rock, firm rock, and soil sites are provided in Subsection 3.7.1.4.

3.7.1.1.1 Design Ground Motion Response Spectra

The Vogtle site-specific safe shutdown earthquake (SSE) design response spectra (DRS) are the site-specific ground motion response spectra (GMRS) determined in Subsection 2.5.2.6. These response spectra are determined in the free-field on the ground surface.

The Vogtle foundation input response spectra (FIRS) are at an outcrop located at the 40' depth. The development of these FIRS is discussed in Subsection 2.5.2.7. These Vogtle response spectra are compared to the AP1000 SSE design response spectra that are also referred to as the AP1000 certified seismic design response spectra (CSDRS). The CSDRS also represents the AP1000 FIRS.

This is because: (1) the CSDRS at a hard rock site is essentially the same at grade and at foundation; and (2) the CSDRS envelopes the in-column motions of the other generic soil conditions. The AP1000 CSDRS are applied at the foundation level in the free field at hard rock sites, and at the finished grade for the other soil generic conditions. The comparisons are shown in Figures 3.7-201 and 3.7-202. As seen from those comparisons, there are exceedances above the CSDRS; therefore, plant specific seismic evaluations are performed that demonstrate that the AP1000 plant designed for the CSDRS is acceptable for the Vogtle site. The results from a Vogtle site specific two-dimensional seismic evaluation that demonstrates the acceptability of the Vogtle site are given in Appendix 2.5E. Additionally, a Vogtle site specific three-dimensional seismic evaluation that demonstrates the acceptability of the Vogtle site is given in Appendix 3GG. Based on these Vogtle site specific seismic evaluations it can also be concluded that the standard AP1000 plant certified design is fully acceptable to a SSE design response spectra level of the CSDRS at Vogtle's plant grade.

As discussed in Subsection 2.5.4.13, the heavy lift derrick (HLD) counterweight and ring foundation were abandoned in place after construction. The HLD counterweight is outside the defined excavation of Unit 3 and Unit 4 and therefore does not need to be evaluated. Portions of the HLD ring foundation extend over the Unit 3 and Unit 4 excavation slopes within the engineered granular backfill (EGB); but outside the Category 1 and 2 backfill. The presence of the HLD ring foundation has no effect on the VEGP site-specific 3D SASSI SSI analyses of the Nuclear Island (NI) presented in Appendix 3GG based on the following information.

The VEGP site-specific 3D SASSI SSI of the NI is consistent with the accepted DCD 3D SASSI NI modeling approach of not including structure-to-structure interaction of the adjacent structures such as the Annex Building and the Turbine Building; and therefore the more distant abandoned HLD ring foundation has even less structure-to-structure effects on the NI seismic response. Additionally only a portion of the abandoned HLD ring foundation is within a limited area of the non-safety EGB over the slopes of the excavation. It has been demonstrated in the ESP as amended that a large variation of the EGB properties does not significantly affect the site-specific seismic analyses; therefore, it is concluded the abandoned portion of the HLD ring foundation in the EGB has no significant effect on the site-specific seismic analyses.

The operating basis earthquake ground motion (OBE) spectral values are used as one measure of potential damage to those structures, systems, and components designed to the SSE design ground motion to determine the severity of the seismic event and make a determination of whether the plant must be shut down. For the AP1000 certified design, OBE is not an explicit design load; as such it is therefore defined as one-third the CSDRS. Since it has been demonstrated that the Vogtle site characteristics do not limit the AP1000 design to the CSDRS, the Vogtle OBE for the AP1000 is defined as one-third the AP1000 CSDRS.

The FIRS and the CSDRS in the horizontal direction in the free-field at the foundation of the AP1000 Nuclear Island exceed the minimum spectrum requirements of 10 CFR50 Appendix S.

3.7.1.2 Design Time History

A "single" set of three mutually orthogonal, statistically independent, synthetic acceleration time histories is used as the input in the dynamic analysis of seismic Category I structures. The synthetic time histories were generated by modifying a set of actual recorded "TAFT" earthquake time histories. The design time histories include a total time duration equal to 20 seconds and a corresponding stationary phase, strong motion duration greater than 6 seconds. The acceleration, velocity, and displacement time-history plots for the three orthogonal earthquake components, "H1," "H2," and "V," are presented in Figures 3.7.1-3, 3.7.1-4, and 3.7.1-5. Design horizontal time history, H1, is applied in the north-south (Global X or 1) direction; design horizontal time history, H2, is applied in the east-west (global Y or 2) direction; and design vertical time history is applied in the

vertical (global Z or 3) direction. The cross-correlation coefficients between the three components of the design time histories are as follows:

$$\rho_{12} = 0.05,\ \rho_{23} = 0.043,\ \text{and}\ \rho_{31} = 0.140$$

where 1, 2, 3 are the three global directions.

Since the three coefficients are less than 0.16 as recommended in Reference 30, which was referenced by NRC Regulatory Guide 1.92, Revision 1, it is concluded that these three components are statistically independent. The design time histories are applied at the foundation level in the free field.

The ground motion time histories (H1, H2, and V) are generated with time step size of 0.010 second for applications in soil structure interaction analyses. For applications in the fixed-base mode superposition time-history analyses, the time step size is reduced to 0.005 second by linear interpolation. The maximum frequency of interest in the horizontal and vertical seismic analysis of the nuclear island is 33 hertz. Modes with higher frequencies are included in the analysis so that the mass in these higher modes is included in the member forces. The cutoff frequencies used in the soil structure interaction analyses are 33 hertz. The maximum "cut-off" frequency for the soil structure interaction analyses and the fixed-base analyses is well within the Nyquist frequency limit.

The comparison plots of the acceleration response spectra of the time histories versus the design response spectra for 2, 3, 4, 5, and 7 percent critical damping are shown in Figures 3.7.1-6, 3.7.1-7, and 3.7.1-8. The SRP 3.7.1, Table 3.7.1-1, provision of frequency intervals is used in the computation of these response spectra.

In SRP 3.7.1 the NRC introduced the requirement of minimum power spectral density to prevent the design ground acceleration time histories from having a deficiency of power over any frequency range. SRP 3.7.1, Revision 2, specifies that the use of a single time history is justified by satisfying a target power spectral density (PSD) requirement in addition to the design response spectra enveloping requirements. Furthermore, it specifies that when spectra other than Regulatory Guide 1.60 spectra are used, a compatible power spectral density shall be developed using procedures outlined in NUREG/CR-5347 (Reference 29).

The NUREG/CR-5347 procedures involve ad hoc hybridization of two earlier power spectral density envelopes. Since the modification to the RG 1.60 design spectra adopted for AP1000 (see Subsection 3.7.1.1) is relatively small (compared to the uncertainty in the fit to RG 1.60 of power spectral density-compatible time histories referenced in NUREG/CR-5347) and occurs only in the frequency range between 9 to 33 hertz, a project-specific power spectral density is developed using a slightly different hybridization for the higher frequencies.

Since the original RG 1.60 spectrum and the project-specific modified RG 1.60 spectrum are identical for frequencies less than 9 hertz, no modification to the power spectral density is done in this frequency range. At frequencies above 9 hertz, the third and the fourth legs of the power spectral density are slightly modified as follows:

- The frequency at which the design response spectrum inflected towards a 1.0 amplification factor at 33 hertz takes place at 25 hertz in the AP1000 spectrum rather than at 9 hertz as in the RG 1.60 spectrum. The third leg of the power spectral density, therefore, is extended to about 25 hertz rather than 16 hertz.

- The lead coefficient to the fourth leg of the power spectral density is changed to connect with the extended third leg.

The AP1000 augmented power spectral density, anchored to 0.3 g, is as follows:

$S_0(f) = 58.5\ (f/2.5)^{0.2}\ in^2/sec^3$, $f \leq 2.5$ hertz

$S_0(f) = 58.5\ (2.5/f)^{1.8}\ in^2/sec^3$, 2.5 hertz $\leq f \leq 9$ hertz

$S_0(f) = 5.832\ (9/f)^3\ in^2/sec^3$, 9 hertz $\leq f \leq 25$ hertz

$S_0(f) = 0.27\ (25/f)^8\ in^2/sec^3$, 25 hertz $\leq f$

The AP1000 Minimum Power Spectral Density is presented in Figure 3.7.1-9. This AP1000 target power spectral density is compatible with the AP1000 horizontal design response spectra and envelops a target power spectral density compatible with the AP1000 vertical design response spectra. This AP1000 target power spectral density, therefore, is conservatively applied to the vertical response spectra.

The comparison plots of the power spectral density curve of the AP1000 acceleration time histories versus the target power spectral density curve are presented in Figures 3.7.1-10, 3.7.1-11, and 3.7.1-12. The power spectral density functions of the design time histories are calculated at uniform frequency steps of 0.0489 hertz. The power spectral densities presented in Figures 3.7.1-10 through 3.7.1-12 are the averaged power spectral density obtained over a moving frequency band of ±20 percent centered at each frequency. The power spectral density amplitude at frequency (f) has the averaged power spectral density amplitude between the frequency range of 0.8 f and 1.2 f as stated in appendix A of Revision 2 of SRP 3.7.1.

3.7.1.3 Critical Damping Values

Energy dissipation within a structural system is represented by equivalent viscous dampers in the mathematical model. The damping coefficients used are based on the material, load conditions, and type of construction used in the structural system. The safe shutdown earthquake damping values used in the dynamic analysis of various structures, supports, and equipment are presented in Table 3.7.1-1. The damping values are based on Regulatory Guide 1.61, Revision 0, ASCE Standard 4-98 (Reference 3), except for the damping value of the primary coolant loop piping, which is based on Reference 22, and conduits, cable trays and their related supports.

The damping values for conduits, cable trays and their related supports are shown in Table 3.7.1-1. The damping value of conduit, empty cable trays, and their related supports is similar to that of a bolted structure, namely 7 percent of critical. The damping value of filled cable trays and supports increases with increased cable fill and level of seismic excitation. Full cable trays use a 10-percent damping value consistent with RG 1.61, Revision 1. The limiting condition for design of the AP1000 standard cable tray supports is for full cable tray weight.

For structures or components composed of different material types, the composite modal damping is calculated using the stiffness-weighted method based on Reference 3. The modal damping values equal:

$$\beta_n = \sum_{i=1}^{nc} \frac{\{\phi_n\}^T\ \beta_i\ [K_t]_i\ \{\phi_n\}}{\{\phi_n\}^T\ [K_t]\ \{\phi_n\}}$$

where:

β_n	=	ratio of critical damping for mode n
nc	=	number of elements
$\{\phi_n\}$	=	mode n (eigenvector)
$[K_t]_i$	=	stiffness matrix of element i
β_i	=	ratio of critical damping associated with element i
$[K_t]$	=	total system stiffness matrix

The linear structural damping values were defined in the modeling codes as a parameter of material property defined for each element. This form of structural damping is used for seismic time history analyses. The structural models analyzed follow the damping criteria stated in Table 3.7.1-1 using 5-percent SSE damping for steel composite (SC) structures, including the shield building wall and modules, and 7-percent SSE damping for the remaining reinforced concrete (RC) structures throughout the nuclear island. A time history non-linear analysis confirms only minor cracking in the nuclear island structure.

3.7.1.4 Supporting Media for Seismic Category I Structures

The supporting media will be described consistent with the information items in Subsection 2.5.4. Seismic analyses for both rock and soil sites are described in Subsection 3.7.2 and Appendix 3G.

The AP1000 nuclear island consists of three seismic Category I structures founded on a common basemat. The three structures that make up the nuclear island are the coupled auxiliary and shield buildings, the steel containment vessel, and the containment internal structures. [*The nuclear island is shown in Figure 3.7.1-14.*]* The foundation embedment depth, foundation size, and total height of the seismic Category I structures are presented in Table 3.7.1-2.

For the design of seismic Category I structures, a set of six design soil profiles (that include hard rock) of various shear wave velocities is established from parametric studies as described in Appendix 3G.3. The soil cases selected for the AP1000 use parameters from the AP600 design, and the AP600 conclusions are applicable to the AP1000 due to the identical footprint to the AP600 and the similarity in overall mass.

For the AP1000 2D and 3D soil-structure interaction analyses, although some of the parabolic soil profiles are defined using a depth of 240 feet, the actual soil profile defined in SASSI (System for Analysis of Soil-Structure Interaction) (base rock) goes to only elevation 120′.

Soil-structure interaction analyses on soil sites for the AP1000 used the latest soil degradation curves recommended by EPRI TR-102293, although these represent more recent soils data and differ slightly for those used for the AP600.

These six profiles are sufficient to envelope sites where the shear wave velocity of the supporting medium at the foundation level exceeds 1000 feet per second (see Subsection 2.5.2). The design soil profiles include a hard rock site, a soft rock site, a firm rock site, an upper bound soft-to-medium soil site, a soft-to-medium soil site, and a soft soil site. The shear wave velocity profiles and related governing parameters of the six sites considered 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.

*NRC Staff approval is required prior to implementing a change in this information.

- 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 strain-dependent shear modulus curves for the foundation materials, together with the corresponding damping curves are taken from References 37 and 38 and are shown in Figures 3.7.1-15 and 3.7.1-16 for rock material and soil material respectively. The different curves for soil in Figure 3.7.1-16 apply to the range of depth within a soil column below grade. The strain-dependent soil material damping is limited to 15 percent of critical damping. The strain-dependent properties used in the SSI analyses for the safe shutdown earthquake are shown in Table 3.7.1-4 and Figure 3.7.1-17 for the firm rock, soft rock, upper bound soft-to-medium soil, soft-to-medium soil, and soft soil properties.

Some variation of soil modeling (water table, soil layering, soil degradation model, and the like) and combinations of these have been demonstrated to have no significant effect on the seismic response of the nuclear island structures. The governing parameters obtained for the AP600 soil studies are also applicable to the AP1000. Each of the parameters deemed not significant has been analyzed.

For instance, the combination of effects of the different strain dependent soil parameters that affect the strain-iterated shear wave velocity profiles was evaluated and shown not to result in exceedances of the envelope of the generic seismic design in-structure response spectra (ISRS).

3.7.2 Seismic System Analysis

Seismic Category I structures, systems, and components are classified according to Regulatory Guide 1.29. Seismic Category I building structures of AP1000 consist of the containment building (the steel containment vessel and the containment internal structures), 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, such as thickness of the basemat, floor slabs, roofs and walls, of the seismic Category I building structures are shown in Figure 3.7.2-12.*]*

Seismic systems are defined, according to SRP 3.7.2, 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. The following subsections describe the seismic analyses performed for the nuclear island. Other seismic Category I structures, systems, equipment, and components not designated as seismic systems (that is, heating, ventilation, and air-conditioning systems; electrical

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cable trays; piping systems) are designated as seismic subsystems. The analysis of seismic subsystems is presented in Subsection 3.7.3.

Seismic Category I building structures are on the nuclear island. Other building structures are classified nonseismic or seismic Category II. Nonseismic structures are analyzed and designed for seismic loads according to the Uniform Building Code (Reference 2) requirements for Zone 2A. The main area of the turbine building structure is analyzed and designed for seismic loads in accordance with International Building Code requirements for an earthquake magnitude equivalent to the Uniform Building Code, Zone 3. Seismic Category II building structures are designed for the safe shutdown earthquake using the same methods and design allowables as are used for seismic Category I structures. The acceptance criteria are based on ACI 349 for concrete structures and on AISC N690 for steel structures including the supplemental requirements described in Subsections 3.8.4.4.1 and 3.8.4.5.

Separate seismic analyses are performed for the nuclear island for each of the six design soil profiles defined in Subsection 3.7.1.4. The analyses generate one set of in-structure responses for each of the design soil profiles. The six sets of in-structure responses are enveloped to obtain the seismic design envelope (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.

Appendix 3G summarizes 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. The seismic analyses of the nuclear island are summarized in a seismic analysis summary report. This report describes the development of the finite element models, the soil structure interaction and fixed base analyses, and the results thereof. Seismic response spectra are given in Appendix 3G for the six key locations:

- Containment internal structures at reactor vessel support elevation 100.00′.
- Containment internal structures at operating deck elevation 134.25′.
- Auxiliary shield building north east corner at control room floor elevation 116.50′.
- Auxiliary shield building corner of fuel building roof at shield building elevation 179.19′.
- Auxiliary shield building roof area elevation 327.41′.
- Steel containment vessel near polar crane elevation 224.000′.

3.7.2.1 Seismic Analysis Methods

Seismic analyses of the nuclear island are performed in conformance with the criteria within SRP 3.7.2.

Seismic analyses – using response spectra analysis, the equivalent static acceleration method, the mode superposition time-history method, and the complex frequency response analysis method – are performed for the safe shutdown earthquake to determine the seismic force distribution for use in the design of the nuclear island structures, and to develop in-structure seismic responses (accelerations, displacements, and floor response spectra) for use in the analysis and design of seismic subsystems.

3.7.2.1.1 Equivalent Static Acceleration Analysis

Equivalent static analyses, using computer program ANSYS (Reference 36), are performed to obtain the seismic forces and moments required for the structural design of the steel containment vessel and the nuclear island basemat (see Subsection 3.8.2.4.1.1). Equivalent static loads are applied to the finite element models using the maximum acceleration results from the time history analyses for

the six design soil profiles. Accidental torsional moments are applied as described in Subsection 3.7.2.11.

Equivalent static analyses are also performed for design of the shield building roof and radial roof beams, PCS tank, tension ring, and air inlet structure (see Subsection 3.8.4.4.1). The equivalent static loads are based on the maximum acceleration results from time history dynamic analysis of the nuclear island in Subsection 3.7.2.1.2.

3.7.2.1.2 Time-History Analysis and Complex Frequency Response Analysis

Mode superposition time-history analyses using computer program ANSYS and complex frequency response analysis using computer program SASSI are performed to obtain the in-structure seismic response needed in the analysis and design of seismic subsystems. Three-dimensional finite element shell models of the nuclear island structures are used in conjunction with the design soil profiles presented in Subsection 3.7.1.4 to obtain the in-structure responses. Stick models are coupled to the shell models of the concrete structures for the containment vessel, polar crane, reactor coolant loop, pressurizer, and core makeup tanks. Three models are used. The fine (NI10) model, as described in Subsection 3G.2.2.1, is used to define the seismic response for the hard rock site. The coarse (NI20) model, as described in Subsection 3G.2.2.2, is used for the soil structure interaction (SSI) analyses and is set up in both ANSYS and SASSI. The NI05 model, as described in Appendix 3G.2.2.4, 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 models and analyses are described in Appendix 3G.

For the hard rock site, the soil-structure interaction effect is negligible. Therefore, for the hard rock site, the nuclear island is analyzed as a fixed-base structure, using computer program ANSYS without the foundation media. The three components of earthquake (two horizontal and one vertical time histories) are applied simultaneously in the analysis. Since the NI10 finite element model of the auxiliary and shield building uses shell elements to represent the 6-foot-thick basemat, the nodes of the basemat element are at the center of the basemat (elevation 63′-6″). The finite element model of the containment internal structures uses solid elements, which extend down to elevation 60′-6″. When the finite element models are combined and used in the time history analyses, the auxiliary building finite element model is fixed at the shell element basemat nodes (elevation 63′-6″) and the base of the containment internal structures is fixed at the bottom of the solid element base nodes (elevation 60′-6″). This difference in elevation of the base fixity is not significant since the concrete between elevations 60′-6″ and 63′-6″, below the auxiliary building, is nearly rigid. There is no lateral support due to soil or hard rock below grade. This case results in higher response than a case analyzed with full lateral support below grade.

For additional information on the method used to calculate displacement, see Appendix 3G.4.1 and Appendix 3G.4.2.

3.7.2.1.3 Response Spectrum Analysis

Response spectral analysis is used for the evaluation of the nuclear island structures. Response spectrum analyses are used to perform an analysis of a particular structure or portion of structure using the procedures described in Appendix 3G.4.3.1 and Subsections 3.7.2.6, 3.7.2.7, and 3.7.3.

Seismic response spectrum analysis of the auxiliary building, shield building, and containment internal structure is performed to develop the seismic design loads for these buildings, and the loads generated include the amplified load due to flexibility and the distribution of this load to the surrounding structures.

3.7.2.2 Natural Frequencies and Response Loads

Modal analyses are performed for the shell and lumped-mass stick models of the seismic Category I structures on the nuclear island, as described in Appendix 3G. Seismic response spectra at the six key locations (Subsection 3.7.2) are given in Appendix 3G.

3.7.2.3 Procedure Used for Modeling

Based on the general plant arrangement, three-dimensional, finite element models are developed for the nuclear island structures: a finite element model of the coupled shield and auxiliary buildings, a finite element model of the containment internal structures, a finite element model of the shield building roof, and an axisymmetric shell model of the steel containment vessel. These three-dimensional, finite element models provide the basis for the development of the dynamic model of the nuclear island structures.

The finite element models of the coupled shield and auxiliary buildings, and the containment internal structures are based on the gross concrete section with the modulus based on the specified compressive strength of concrete reduced by a factor of 0.8 to consider the effect of cracking as recommended in Table 6-5 of FEMA 356 (Reference 5). This 80-percent value is supported by non-linear ABAQUS analyses performed on the nuclear island finite element model. The comparison between linear and non-linear models shows that the 80-percent stiffness model response spectra enveloped the non-linear model, providing a conservative approach in terms of response spectra and maximum stresses obtained in the shield building wall.

Seismic subsystems coupled to the overall dynamic model of the nuclear island include the coupling of the reactor coolant loop model to the model of the containment internal structures, and the coupling of the polar crane model to the model of the steel containment vessel. The criteria used for decoupling seismic subsystems from the nuclear island model are according to Section II.3.b of SRP 3.7.2, Revision 2. The total mass of other major subsystems and equipment is less than one percent of the respective supporting nuclear island structures; therefore, the mass of other major subsystems and equipment is included as concentrated lumped-mass only.

Several minor (basic building configuration not modified) design changes and model improvements include the following:

- Provision for heavier fuel racks in the spent fuel pool area. Fuel and rack masses are updated, and pool water volumes are modeled as lumped masses.

- Changes in the annulus configuration are incorporated into the dish model, and lower shield building and upper containment internal structures basemat nodes and elements are modified for compatibility.

- The core makeup tanks were added as stick models.

- The polar crane model has a reduced weight and updated steel containment vessel local stiffness, and now includes polar crane truck stiffness.

The seismic analysis of the water inside the PCCWST was performed for the AP600. It was concluded that the low-frequency sloshing mode is not significant to the response of the nuclear island away from the shield building roof and that this conclusion could be extended to the AP1000 design. Further analysis indicated that the sloshing mass ratio remained essentially unchanged between the AP600 and AP1000.

3.7.2.3.1 Coupled Shield and Auxiliary Buildings and Containment Internal Structures

The finite element models of the coupled shield and auxiliary buildings and the reinforced concrete portions of the containment internal structures are based on the gross concrete section with the modulus based on the specified compressive strength of concrete of contributing structural walls and slabs. The properties of the concrete-filled structural modules are computed using the combined gross concrete section and the transformed steel face plates of the structural modules. The modulus is reduced by a factor of 0.8 to consider the effect of cracking. Furthermore, the weight density of concrete plus the uniformly distributed miscellaneous dead weights are considered by adding surface mass or by adjusting the material mass density of the structural elements. An equivalent tributary slab area load of 50 pounds per square foot is considered to represent miscellaneous deadweight such as minor equipment, piping and raceways. 25 percent of the floor live load or 75 percent of the roof snow load, whichever is applicable, is considered as mass in the global seismic models.

Major equipment weights are distributed over the floor area or are included as concentrated lumped masses at the equipment locations. The major equipment supported by the containment internal structures is represented by stick models connected to the containment internal structures, and includes the reactor coolant loop, the pressurizer, and the core makeup tank. 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). The finite element models of the coupled shield and auxiliary buildings and the containment internal structures are described in Appendix 3G. The auxiliary and shield building is modeled with shell elements and the base of the finite element model is at the middle of the basemat at elevation 63′-6″. The bottom of the containment and internal structures are modeled with solid elements and the base of the finite element model is at the underside of the basemat at elevation 60′-6″. The interface between the models is at a radius of 71′-0″ at the mid-surface of the shield building.

3.7.2.3.2 Steel Containment Vessel

The steel containment vessel is a freestanding, cylindrical, steel shell structure with ellipsoidal upper and lower steel domes. The three-dimensional, lumped-mass stick model of the steel containment vessel is developed based on the axisymmetric shell model. Figure 3G.2-4 presents the steel containment vessel 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.

This 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.

Additional details of the steel containment vessel stick model are included in Appendix 3G.2.1.3.

The containment air baffle, presented in Subsection 3.8.4.1.3, is supported from the steel containment vessel at regular intervals so that a gap is maintained for airflow. It is constructed with individual panels which do not contribute to the stiffness of the containment vessel. The fundamental frequency of the baffle panels and supports is about twice the fundamental frequency of the containment vessel. The mass of the air baffle is small, equal to approximately 10 percent of the vessel plates to which it is attached. The air baffle, therefore, is assumed to have negligible interaction with the steel containment vessel. Only the mass of the air baffle is considered and added at the appropriate elevations of the steel containment vessel stick model.

The interaction between the polar crane and the containment vessel is significant and is included in the model. This polar crane model reflects the polar crane wheel assemblies. The polar crane is supported on a ring girder which is an integral part of the steel containment vessel 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 as shown in Figure 3G.2-5 with five masses at the mid-height of the bridge at elevation 233′-6″ and one mass for the trolley. 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 3.7 hertz transverse to the bridge, 6.4 hertz vertically, and 8.5 hertz along the bridge.

[*During plant operating conditions, the polar crane is parked in the plant north-south direction with the trolley located at one end near the containment shell.*]* In the seismic model, the crane bridge spans in the north-south direction and the mass eccentricity of the trolley is considered by locating the mass of the trolley at the northern limit of travel of the main hook. Furthermore, the mass eccentricity of the two equipment hatches and the two personnel airlocks are considered by placing their mass at their respective center of mass as shown in Figure 3G.2-4. Any modeling change due to the as-procured crane data is resolved with the COL holder item in Subsection 3.7.5.4, "Reconciliation of Seismic Analyses of Nuclear Island Structures."

3.7.2.3.3 Nuclear Island Seismic Model

The nuclear island seismic models are described in Appendix 3G. The various building models are interconnected to form the overall dynamic model of the nuclear island. The mass properties of the models include all tributary mass expected to be present during plant operating conditions. This includes the dead weight of walls and slabs, weight of major equipment, and equivalent tributary slab area loads representing miscellaneous equipment, piping and raceways.

The hydrodynamic mass effect of the water within the passive containment cooling system water tank on the shield building roof, the in-containment refueling water storage tank within the containment internal structures, and the spent fuel pool in the auxiliary building is evaluated. Since the water in the PCCS tank responds at a very low frequency (sloshing) and does not affect building response, the PCCS tank water horizontal mass is reduced to exclude the low frequency water sloshing mass. The total mass of the water in the in-containment refueling water storage tank within the containment

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internal structures, and the spent fuel pool in the auxiliary building is included in the nuclear island seismic model.

Seismic response spectra are developed at the locations of the nodes. These response spectra are grouped and enveloped to define the seismic design response spectra. The nodes associated with a specific elevation and building structure (i.e., auxiliary and shield building and containment internal structures) are grouped. For the auxiliary and shield building where the floor at the elevation of interest is rigid (i.e. frequency > 33 hertz), it is only necessary to envelop the response spectra at edge points and interior nodes at the shield wall to obtain the largest seismic response spectra because of rigid motion. The edge nodes reflect the largest rocking and translational response of the auxiliary building, and the response spectra associated with the nodes on the shield wall will reflect the shield wall dynamic response. It is not necessary to include any nodes between the shield wall and auxiliary building edge since the floor is rigid, and the response cannot be worse than those enveloped.

A refined finite element shell model of the nuclear island concrete structures is reviewed for flexible regions, which may produce amplified response spectra. This model, called the NI05 model, has a tetrahedral mesh size of approximately 5 feet by 5 feet. Each of the principal walls and floors in the auxiliary and shield building as well as the containment internal structures are reviewed. A modal analysis of the NI05 model for both auxiliary and shield building and containment internal structures is reviewed for each of these regions for the existence of out of plane modes, which are considered flexible (less than 33 hertz) with significant participating mass. 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. [*These regions, which have flexible areas, are evaluated in one of two ways:*

- *Flexible areas, which have been previously identified, have amplified response spectra developed directly from the time history analyses for the envelope of soil sites.*

- *Flexible regions, which require a detailed analysis to obtain the amplified response spectra, use input directly from time history analysis. The NI05 finite element model is used to capture out-of-plane flexibilities that, because of mesh refinement, a more course model could not capture.*

If equipment or a structure is supported at more than one elevation, then the seismic input as an envelope of multiple groups based on the support locations will be defined. Therefore, if the equipment or structure is supported on rigid and flexible floor areas the response spectra (horizontal and vertical directions) used by the analysts will be the envelope of the rigid and flexible areas that include inside and outside nodes.

If an equipment or structure is supported exclusively by a floor or wall, only that spectra will be used for design.]*

3.7.2.4 Soil-Structure Interaction

Soil-structure interaction is not significant for the nuclear island founded on rock with a shear wave velocity greater than 8000 feet per second. The soil-structure interaction analyses for the firm rock and soil sites are described in Appendix 3G.

The computer program SASSI is used to perform the soil-structure interaction analysis. The SASSI model of the nuclear island is based on the NI20 Coarse Finite Element model. Soil-structure interaction analyses are performed based on the nuclear island 3D SASSI model for the three soil conditions established from the AP1000 2D SASSI analyses, in addition to soft rock and soft soil.

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SASSI uses key frequencies to perform its transfer function calculations. For a large model, resting on a very stiff soil (hard rock), SASSI gives conservative results at high frequencies. The significant responses for AP1000 soil cases occur at less than 10 hertz so the SASSI model is adequate for use.

Analyses are performed with large solid-shell finite element models at two levels. 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 analyses. 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.

3.7.2.5 Development of Floor Response Spectra

The design floor response spectra are generated according to Regulatory Guide 1.122.

Seismic floor response spectra are computed using time-history responses determined from the nuclear island seismic analyses. The time-history responses for the hard rock condition are determined from a mode superposition time history analysis using computer program ANSYS.

The time-history responses for the firm rock and soil conditions are determined from a complex frequency response analysis using computer program SASSI. Floor response spectra for damping values equal to 2, 3, 4, 5, 7, 10, and 20 percent of critical damping are computed at the required locations.

The floor response spectra for the design of subsystems and components are generated by broadening the enveloped nodal response spectra determined for the hard rock site and soil sites.

The spectral peaks are broadened by ±15 percent to account for the variation in the structural frequencies, due to the uncertainties in parameters such as material and mass properties of the structure and soil, damping values, seismic analysis technique, and the seismic modeling technique. Figure 3.7.2-14 shows the broadening procedure used to generate the design floor response spectra. Spectral peaks at frequencies associated with fundamental soil structure interaction frequencies are reviewed. If there is a “valley” between peaks due to different soil profiles and not the building modal response, then this valley is filled by extending the broadening of the lower peak horizontally until it meets the broadened upper peak.

Floor response spectra for the auxiliary building are obtained from the three-dimensional model as described in Appendix 3G. These spectra are developed for the specific location in the auxiliary building. Where spectra at a number of nodes have similar characteristics, a single set of spectra may be developed by enveloping the broadened spectra at each of the nodes.

The safe shutdown earthquake floor response spectra for 5 percent damping, at representative locations of the coupled auxiliary and shield buildings, the steel containment vessel, and the containment internal structures are presented in Appendix 3G.

3.7.2.6 Three Components of Earthquake Motion

Seismic system analyses are performed considering the simultaneous occurrences of the two horizontal and the vertical components of earthquake.

In mode superposition time-history analyses using computer program ANSYS, the three components of earthquake are applied either simultaneously or separately. In the ANSYS analyses with the three earthquake components applied simultaneously, the effect of the three components of earthquake motion is included within the analytical procedure so that further combination is not necessary.

In analyses with the earthquake components applied separately and in the response spectrum and equivalent static analyses, the effect of the three components of earthquake motion are combined using one of the following methods:

- For seismic analyses with the statistically independent earthquake components applied separately, the time-history responses from the three earthquake components are combined algebraically at each time step to obtain the combined response time-history. This method is used in the SASSI analyses.

- The peak responses due to the three earthquake components from the response spectrum and equivalent static analyses are combined using the square root of the sum of squares (SRSS) method.

- The peak responses due to the three earthquake components from the equivalent static analyses are combined directly, using the assumption that when the peak response from one component occurs, the responses from the other two components are 40 percent of the peak (100 percent-40 percent-40 percent method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus), are considered. This method is used in the nuclear island basemat analyses, the containment vessel analyses, and the shield building roof analyses.

The containment vessel is analyzed using axisymmetric finite element models. These axisymmetric building structures are analyzed for one horizontal seismic input from any horizontal direction and one vertical earthquake component. Responses are combined by either the square root of the sum of squares method or by the 100 percent-40 percent-40 percent method in which one component is taken at 100 percent of its maximum value and the other components are taken at 40 percent of their maximum value.

For the seismic responses presented in Appendix 3G, the effect of three components of earthquake are considered as follows:

- Mode Superposition Time History Analysis (program ANSYS) and the Complex Frequency Response Analysis (program SASSI) – the time history responses from the three components of earthquake motion are combined algebraically at each time step.

A summary of the dynamic analyses performed and the combination techniques used are presented in Appendix 3G.

3.7.2.7 Combination of Modal Responses

The modal responses of the response spectrum system structural analysis are combined using the procedures described in Appendix 3G.4.3. In the fixed base mode superposition time history analysis of the hard rock site, the total seismic response is obtained by superposing the modal responses within the analytical procedure so that further combination is not necessary.

A summary of the dynamic analyses performed and the combination techniques used are presented in Appendix 3G.

3.7.2.8 Interaction of Seismic Category II and Nonseismic Structures with Seismic Category I Structures, Systems, or Components

Nonseismic structures are evaluated to determine that their seismic response does not preclude the safety functions of seismic Category I structures, systems or components. This is accomplished by satisfying one of the following:

- The collapse of the nonseismic structure will not cause the nonseismic structure to strike a seismic Category I structure, system or component.
- The collapse of the nonseismic structure will not impair the integrity of seismic Category I structures, systems or components.
- The structure is classified as seismic Category II and is analyzed and designed to prevent its collapse under the safe shutdown earthquake.

The structures adjacent to the nuclear island are the annex building, the radwaste building, and the turbine building.

3.7.2.8.1 Annex Building

The portion of the annex building adjacent to the nuclear island is classified as seismic Category II. The structural configuration is shown in Figure 3.7.2-19. The annex building is analyzed for the safe shutdown earthquake for the six soil profiles described in Subsection 3.7.1.4. For the hard rock site, a range of soil properties is assumed for the layer above rock at the level of the nuclear island foundation. Seismic input is defined by response spectra applied at the base of a dynamic model of the annex building. The seismic response spectra input at the base of the annex building are the envelopes of the range of soil sites and also envelope the AP1000 design free field ground spectra shown in Figures 3.7.1-1 and 3.7.1-2. The envelope of the maximum building response acceleration values is applied as equivalent static loads to a more detailed static model. See Subsection 3.7.2.8.4 for more discussion of modeling and seismic analysis.

The minimum space required between the annex building and the nuclear island to avoid contact is obtained by absolute summation of the deflections of each structure obtained from either a time history or a response spectrum analysis for each structure. The maximum displacement of the roof of the annex building is 1.6 inches in the east-west direction. The minimum clearance between the structural elements of the annex building above grade and the nuclear island is 4 inches.

3.7.2.8.2 Radwaste Building

The radwaste building is classified as nonseismic and is designed to the seismic requirements of the Uniform Building Code, Zone 2A with an Importance Factor of 1.25. As shown in the radwaste building general arrangement in Figure 1.2-22, it is a small steel framed building. If it were to impact the nuclear island or collapse in the safe shutdown earthquake, it would not impair the integrity of the reinforced concrete nuclear island. The minimum clearance between the structural elements of the radwaste building above grade and the nuclear island is 4 inches.

Three methods are used to demonstrate that a potential radwaste building impact on the nuclear island during a seismic event will not impair its structural integrity:

- The maximum kinetic energy of the impact during a seismic event considers the maximum radwaste building and nuclear island velocities. The total kinetic energy is considered to be absorbed by the nuclear island and converted to strain energy. The deflection of the nuclear island is less than 0.2″. The shear forces in the nuclear island walls are less than the ultimate shear strength based on a minus one standard deviation of test data.
- Stress wave evaluation shows that the stress wave resulting from the impact of the radwaste building on the nuclear island has a maximum compressive stress less than the concrete compressive strength.

- An energy comparison shows that the kinetic energy of the radwaste building is less than the kinetic energy of tornado missiles for which the exterior walls of the nuclear island are designed.

3.7.2.8.3 Turbine Building

The south end of the turbine building is separated from the rest of the turbine building by a 2'-0" thick reinforced concrete wall that provides a robust structure around the first bay. This wall isolates the first bay of the turbine building from the general area of the turbine building and from the adjacent yard area. The main segment of this wall is located on column line 11.2. This wall extends from El.100'-0" basemat to El.169'-0". The first bay of the turbine building is classified as seismic Category II. The other bays are classified as non-seismic. The structure configuration is shown in Figure 3.7.2-20.

The first bay of the turbine building is analyzed for the safe shutdown earthquake for the six soil profiles described in Subsection 3.7.1.4. For the hard rock site, a range of soil properties is assumed for the layer above rock at the level of the nuclear island foundation. Seismic input is defined by response spectra applied at the base of a dynamic model of the first bay of the turbine building. The seismic response spectra input at the base of the first bay of the turbine building are the envelopes of the range of soil sites and also envelope the AP1000 design free field ground spectra shown in Figures 3.7.1-1 and 3.7.1-2. See Subsection 3.7.2.8.4 for more discussion of modeling and seismic analysis.

The first bay is designed in accordance with ACI-349 for concrete features and AISC-N690 for steel features.

For the non-seismic portion of the Turbine Building, seismic design is upgraded in order to provide margin against collapse during the safe shutdown earthquake. The turbine building is a braced steel frame structure designed to meet the following criteria:

- The turbine building is designed in accordance with the 2006 International Building Code (Reference 40). This references ACI-318 for concrete structures and AISC for steel structures. Seismic loads are defined in accordance with the International Building Code with the maximum considered earthquake spectral parameters S_{DS} = 0.9, S_{D1} = 0.54 for Site Class D. This is consistent with the 1997 Uniform Building Code provisions for Zone 3 with an Importance Factor of 1.0. For a braced structure that has a mix of eccentric and special concentric bracing, the response modification factor is 6 (ASCE 7-05, Reference 42) using strength design.

- The design complies with the seismic requirements for eccentrically braced frames and special concentrically braced frames given in the 2005 AISC Seismic Provisions for Structural Steel Buildings (Reference 41). Quality assurance is in accordance with ASCE 7-05 (Reference 42).

3.7.2.8.4 Seismic Modeling and Analysis of Seismic Category II Building Structures

Seismic Category II structures, systems, and components are designed so that the safe shutdown earthquake does not cause unacceptable structural failure or interaction with seismic Category I items. Therefore, the seismic response of seismic Category II buildings must be obtained so that they can be designed to meet the seismic Category II requirements as given in Subsection 3.2.1.1.2. Seismic Category II structures are analyzed and evaluated in the same manner as seismic Category I structures. The foundation of the non-seismic portion is modeled with the associated mass distributed on it so that the soil structure interaction during a seismic event is reflected in the analysis.

The seismic analyses performed for the adjacent seismic Category II structures are simulated 3D analyses. The seismic analyses are performed primarily using 2D SASSI models. To properly account for the 3D effect, the response from 2D and 3D SASSI analyses of the seismic Category II buildings on rigid foundations are compared and a 3D effects factor is developed from this comparison. Three soil cases (upper bound soft to medium UBSM, soft to medium SM, and soft soil SS) are used to determine the 3D factor. Shown in Figures 3.7.2-20 and 3.7.2-21 are the 2D SASSI models with adjacent building structures. The seismic Category II buildings are modeled as stick models. The 3D model with adjacent structures is shown in Figure 3.7.2-22.

Seismic Category II buildings are designed using envelope foundation input response spectra (FIRS). The development of these FIRS shall be based on a number of analyses results from the SASSI analyses. The seismic Category II FIRS shall be the envelope of the SASSI seismic Category II foundation response spectra resulting from the following seismic inputs/soil profiles:

- AP1000 CSDRS – Hard rock at El. 60.5′.
- AP1000 CSDRS – Firm rock, soft rock, upper bound soft to medium, soft to medium, and soft soil soil profiles with AP1000 CSDRS spectra input at plant grade; and
- AP1000 hard rock high frequency (HRHF) – For rock sites, HRHF at plant grade shall be developed using AP1000 HRHF spectra at El. 60.5′ and a range of backfill soil profiles. The backfill soil under the annex and turbine buildings has a parabolic soil profile as a function of depth (El. 100′ to El. 60.5′) and uses EPRI (1993) strain dependent curves. The HRHF at plant grade spectrum shall be generated using soil profiles corresponding to a shear wave velocity of 500 fps, 750 fps, and 1000 fps at El. 100′. The HRHF at plant grade shall be used as input to SASSI analyses to determine the FIRS at the base of the seismic Category II structures.

For each soil case, 2D SASSI analyses shall be performed and the results at three locations at the base of the seismic Category II structures are enveloped. The maximum bearing demand and maximum relative displacement shall be established from the 2D SASSI analyses. The 3D effect factor is applied to the envelope foundation spectra and used for the design of the annex building and turbine building first bay.

Response spectrum analyses (using detailed finite element building models) shall be used to obtain seismic design loads for the seismic Category II building design. The seismic input to the response spectrum analyses is the envelope foundation response spectra obtained from the SASSI analyses. The COL applicant will perform the following screening criteria to determine if it has to perform further analysis for its site. If the requirements given below are not met, then the site applicant can perform site-specific analyses to demonstrate that its site-specific seismic Category II foundation seismic response spectra are less than the AP1000 annex building and turbine building first bay generic design envelope foundation spectra.

- The site meets Subsection 2.5.4.2 DCD soil uniformity requirements.
- For soil sites, the site GMRS is enveloped by the AP1000 CSDRS with soil profiles SS, SM, UBSM, SR, FR, and HR.
- For HRHF sites, the site GMRS is enveloped by the AP1000 HRHF response spectra with a minimum backfill surface shear wave velocity of 500 fps, and a minimum lateral extent of the backfill corresponding to a line extending down from the surface at a one horizontal to one vertical (1H:1V) slope from the outside footprint limit of the seismic Category II structure.

- The bearing capacity with appropriate factor of safety is greater than or equal to the bearing demand.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Seismic model uncertainties due to, among other things, uncertainties in material properties, mass properties, damping values, the effect of concrete cracking, and the modeling techniques are accounted for in the widening of floor response spectra, as described in Subsection 3.7.2.5. The effect of cracking of the concrete-filled structural modules inside containment due to thermal loads is discussed in Subsection 3.8.3.4.2.

3.7.2.10 Use of Constant Vertical Static Factors

The vertical component of the safe shutdown earthquake is considered to occur simultaneously with the two horizontal components in the seismic analyses. Therefore, constant vertical static factors are not used for the design of seismic Category I structures.

3.7.2.11 Method Used to Account for Torsional Effects

The seismic analysis models of the nuclear island incorporate the mass and stiffness eccentricities of the seismic Category I structures and the torsional degrees of freedom.

For the response spectrum analysis of the nuclear island, the seismic loads are combined by means of the square root of the sum of the squares (SRSS). The equation for SRSS is shown below.

$$\sqrt{(\alpha \cdot A_{NS})^2 + (\alpha \cdot A_{EW})^2 + (A_{VT})^2}$$

where,

A_{NS} maximum element forces due to SSE response analysis in X (NS)

A_{EW} maximum element forces due to SSE response analysis in Y (EW)

A_{VT} maximum element forces due to SSE response analysis in Z (VT)

α factor to account for accidental torsion effect in NS or EW (1.05)

Alternatively, for equivalent static analysis, the 100-40-40 rule is applicable in order to cover both negative and positive member forces. The equation for the 100-40-40 rule is shown below.

$$k_1 \cdot \mathrm{sign}(A_{NS}) \cdot (\alpha \cdot A_{NS}) + k_2 \cdot \mathrm{sign}(A_{EW}) \cdot (\alpha \cdot A_{EW}) + k_3 \cdot A_{VT}$$

where,

k_i combination factors (±1.0, ±0.4, ±0.4)

sign(X) sign of variable X: $X < 0$ results -1; $X \geq 0$ results +1

α factor to account for accidental torsion effect in NS or EW (1.05)

3.7.2.12 Methods for Seismic Analysis of Dams

Seismic analysis of dams is site specific design.

The evaluation of existing dams whose failure could affect the site interface flood level specified in Subsection 2.4.12, is included in Subsection 2.4.4. As discussed in Subsection 2.4.1.2.4, the U.S. Army Corps of Engineers has no current plans for the construction of additional reservoirs on the Savannah River.

3.7.2.13 Determination of Seismic Category I Structure Overturning Moments

Subsection 3.8.5.5.4 describes the effects of seismic overturning moments.

3.7.2.14 Analysis Procedure for Damping

Subsection 3.7.1.3 presents the damping values used in the seismic analyses. [*For structures comprised of different material types, the composite modal damping approach utilizing the strain energy method is used to determine the composite modal damping values.*]* Subsection 3.7.2.4 presents the damping values used in the soil-structure interaction analysis.

3.7.3 Seismic Subsystem Analysis

This subsection describes the seismic analysis methodology for subsystems, which are those structures and components that do not have an interface with the soil-structure interaction analyses. Structures and components considered as subsystems include the following:

- Structures, such as floor slabs, walls, miscellaneous steel platforms and framing
- Equipment modules consisting of components, piping, supports, and structural frames
- Equipment including vessels, tanks, heat exchangers, valves, and instrumentation
- Distributive systems including piping and supports, electrical cable trays and supports, HVAC ductwork and supports, instrumentation tubing and supports, and conduits and supports

Subsection 3.9.2 describes dynamic analysis methods for the reactor internals. Subsection 3.9.3 describes dynamic analysis methods for the primary coolant loop support system. Subsection 3.7.2 describes the analysis methods for seismic systems, which are those structures and components that are considered with the foundation and supporting media. Section 3.2 includes the seismic classification of building structures, systems, and components.

3.7.3.1 Seismic Analysis Methods

The methods used for seismic analysis of subsystems include, modal response spectrum analysis, time-history analysis, and equivalent static analysis. The methods described in this subsection are acceptable for any subsystem. The particular method used is selected by the designer based on its appropriateness for the specific item. Items analyzed by each method are identified in the descriptions of each method in the following paragraphs.

*NRC Staff approval is required prior to implementing a change in this information.

3.7.3.2 Determination of Number of Earthquake Cycles

Seismic Category I structures, systems, and components are evaluated for one occurrence of the safe shutdown earthquake (SSE). In addition, subsystems sensitive to fatigue are evaluated for cyclic motion due to earthquakes smaller than the safe shutdown earthquake. Using analysis methods, these effects are considered by inclusion of seismic events with an amplitude not less than one-third of the safe shutdown earthquake amplitude. The number of cycles is calculated based on IEEE-344-1987 (Reference 16) to provide the equivalent fatigue damage of two full safe shutdown earthquake events with 10 high-stress cycles per event. Typically, there are five seismic events with an amplitude equal to one-third of the safe shutdown earthquake response. Each of the one-third safe shutdown earthquake events has 63 high-stress cycles. [*For ASME Class 1 piping, the fatigue evaluation is performed based on five seismic events with an amplitude equal to one-third of the safe shutdown earthquake response. Each event has 63 high-stress cycles.*]*

When seismic qualification is based on dynamic testing for structures, systems, or components containing mechanisms that must change position in order to function, operability testing is performed for the safe shutdown earthquake preceded by one or more earthquakes. The number of preceding earthquakes is calculated based on IEEE-344-1987 (Reference 16) to provide the equivalent fatigue damage of one safe shutdown earthquake event. Typically, the preceding earthquake is one safe shutdown earthquake event or five one-half safe shutdown earthquake events.

3.7.3.3 Procedure Used for Modeling

The dynamic analysis of any complex system requires the discretization of its mass and elastic properties. This is accomplished by concentrating the mass of the system at distinct characteristic points or nodes, and interconnecting them by a network of elastic springs representing the stiffness properties of the systems. The stiffness properties are computed either by hand calculations for simple systems or by finite element methods for more complex systems.

Nodes are located at mass concentrations and at additional points within the system. They are selected in such a way as to provide an adequate representation of the mass distribution and high-stress concentration points of the system.

At each node, degrees of freedom corresponding to translations along three orthogonal axes, and rotations about these axes are assigned. The number of degrees of freedom is reduced by the number of constraints, where applicable. For equipment qualification, reduced degrees of freedom are acceptable provided that the analysis adequately and conservatively predicts the response of the equipment.

The size of the model is reviewed so that a sufficient number of masses or degrees of freedom are used to compute the response of the system. A model is considered adequate provided that additional degrees of freedom do not result in more than a 10 percent increase in response, or the number of degrees of freedom equals or exceeds twice the number of modes with frequencies less than 33 hertz.

Dynamic models of floor and roof slabs and miscellaneous steel platforms and framing include masses equal to 25 percent of the floor live load or 75 percent on the roof snow load, whichever is applicable.

Dynamic models are prepared for the following seismic Category I steel structures. Response spectrum or time history analyses are performed for structural design.

- Passive containment cooling valve room (room number 12701)

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- Steel framing around steam generators
- Containment air baffle

Seismic input for the subsystem and component design are the enveloped floor response spectra described in Subsection 3.7.2.5 or the response time histories as described in Subsection 3.7.2.1. Where amplified response spectra are required on the subsystem for design of components, such as for use in the decoupled analyses of piping or components described in Subsection 3.7.3.8.3, the amplified response spectra are generated and enveloped as described in Subsection 3.7.2.5.

3.7.3.4 Basis for Selection of Frequencies

The effect of the building amplification on equipment and components is addressed by the floor response spectra method or by a coupled analysis of the building and equipment. Certain components are designed for a natural frequency greater than 33 hertz. In those cases where it is practical to avoid resonance, the fundamental frequencies of components and equipment are selected to be less than one-half or more than twice the dominant frequencies of the support structure.

3.7.3.5 Equivalent Static Load Method of Analysis

[*The equivalent static load method involves equivalent horizontal and vertical static forces applied at the center of gravity of various masses. The equivalent force at a mass location is computed as the product of the mass and the seismic acceleration value applicable to that mass location. Loads, stresses, or deflections, obtained using the equivalent static load method, are adjusted to account for the relative motion between points of support when significant.*]*

3.7.3.5.1 Single Mode Dominant or Rigid Structures or Components

For rigid structures and components, or for cases where the response can be classified as single mode dominant, the following procedures are used. Examples of these systems, structures, and components are equipment, and piping lines, instrumentation tubing, cable trays, HVAC, and floor beams modeled on a span by span basis.

- For rigid systems, structures, and components (fundamental frequency ≥ 33 hertz), an equivalent seismic load is defined for the direction of excitation as the product of the component mass and the zero period acceleration value obtained from the applicable floor response spectra.

- A rigid component (fundamental frequency ≥ 33 hertz), whose support can be represented by a flexible spring, can be modelled as a single degree of freedom model in the direction of excitation (horizontal or vertical directions). The equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value at the natural frequency from the applicable floor response spectra. If the frequency is not determined, the peak acceleration from the applicable floor response spectrum is used.

[• *If the component has a distributed mass whose dynamic response will be single mode dominant, the equivalent static seismic load for the direction of excitation is defined as the product of the component mass and the seismic acceleration value at the component natural frequency from the applicable floor response spectra times a factor of 1.5. A factor of less than 1.5 may be used if justified. Static factors smaller than 1.5 are not used for piping systems.*]* A factor of 1.0 is used for structures or equipment that can be represented as uniformly loaded cantilever, simply supported, fixed-simply supported, or fixed-fixed beams

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(References 10 and 11) when the fundamental frequency is higher than the peak acceleration frequency associated with the applicable floor response spectrum. If the frequency is not determined, the peak acceleration from the applicable floor response spectrum is used.

3.7.3.5.2 Multiple Mode Dominant Response

This procedure applies to piping, instrumentation tubing, cable trays, and HVAC that are multiple span models. The equivalent static load method of analysis can be used for design of piping systems, instrumentation and supports that have significant responses at several vibrational frequencies. In this case, [*a static load factor of 1.5 is applied to the peak accelerations of the applicable floor response spectra. For runs with axial supports which are rigid in the axial direction (fundamental frequency greater than or equal to 33 hertz), the acceleration value of the mass of piping in its axial direction may be limited to 1.0 times its calculated spectral acceleration value. The spectral acceleration value is based on the frequency of the piping system along the axial direction. The relative motion between support points is also considered.*]*

3.7.3.6 Three Components of Earthquake Motion

[*Two horizontal components and one vertical component of seismic response spectra are employed as input to a modal response spectrum analysis.*]* The spectra are associated with the safe shutdown earthquake. In the response spectrum and equivalent static analyses, the effects of the three components of earthquake motion are combined using one of the following methods:

[● *The peak responses due to the three earthquake components from the response spectrum analyses are combined using the square root of the sum of squares (SRSS) method.*

● *The peak responses due to the three earthquake components are combined directly, using the assumption that when the peak response from one component occurs, the responses from the other two components are 40 percent of the peak (100 percent-40 percent-40 percent method). Combinations of seismic responses from the three earthquake components, together with variations in sign (plus or minus), are considered. This method is not used for piping systems.*

One set of three mutually orthogonal artificial time histories is used when time-history analyses are performed. The components of earthquake motion specified in the three directions are statistically independent and applied simultaneously. When this method is used, the responses from each of the three components of motion are combined algebraically at each time step.]*

In addition, an optional method for combining the response of the three components of earthquake motion is presented in the following paragraphs.

[*The time-history safe shutdown earthquake analysis of a subsystem can be performed by simultaneously applying the displacements and rotations at the interface point(s) between the subsystem and the system. These displacements and rotations are the results obtained from a model of a larger subsystem or a system that includes a simplified representation of the subsystem. The time-history safe shutdown earthquake analysis of the system is performed by applying three mutually orthogonal and statistically independent, artificial time histories.*]* Possible examples of the use of this method of seismic analysis include the following:

● The subsystem analysis is a flexible floor or miscellaneous structural steel frame. The corresponding system analysis is the soil-structure interaction analysis of the nuclear island structures.

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- The subsystem analysis is the primary loop piping system and interior concrete building structure. The interface point is the top of the basemat. The corresponding system analysis is the soil-structure interaction analysis of the nuclear island structures.

- The subsystem analysis is the reactor coolant pump and internal components. The interface points are the welds on the pump suction and discharge nozzles. The corresponding system analysis is the primary loop piping system and interior concrete building structure.

3.7.3.7 Combination of Modal Responses

[*For the seismic response spectra analyses, the zero period acceleration cut-off frequency is 33 hertz. High frequency or rigid modes are considered using the left-out-force method or the missing mass method*]* described in Subsection 3.7.3.7.1. The method to combine the low frequency modes is described in Subsection 3.7.3.7.2. [*The rigid mode results in the three perpendicular directions of the seismic input are combined by the SRSS method. The resultant response of the rigid modes is combined by SRSS with the flexible mode results.*]* The combination of modal responses in time history analyses of piping systems is described in Subsection 3.7.3.17 Modal responses in time history analyses of other subsystems are combined as described in Subsection 3.7.2.6.

3.7.3.7.1 Combination of High-Frequency Modes

This subsection describes alternative methods of accounting for high-frequency modes (generally greater than 33 hertz) in seismic response spectrum analysis. Higher-frequency modes can be excluded from the response calculation if the change in response is less than or equal to 10 percent.

3.7.3.7.1.1 Left-Out-Force Method or Missing Mass Correction for High Frequency Modes

The left-out-force method is based on the Left-Out-Force Theorem. This theorem states that for every time history load there is a frequency, f_r, called the "rigid mode cutoff frequency" above which the response in modes with natural frequencies above f_r will very closely resemble the applied load at each instant of time. These modes are called "rigid modes." [*The left-out-force method is used in program PIPESTRESS.*]*

The left-out-force vector, $\{Fr\}$, is calculated based on lower modes:

$$\{Fr\} = \left[1 - \Sigma M e_j e_j^T\right] f(t)$$

where:

f (t)	=	the applied load vector
M	=	the mass matrix
e_j	=	the eigenvector

Note that Σ is only for all the flexible modes, not including the rigid modes.

In the response spectra analysis, the total inertia force contribution of higher modes can be interpreted as:

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$$\{Fr\} = Am[M]\left[\{r\} - \Sigma P_j\, e_j\right]$$

where:

Am = the maximum spectral acceleration beyond the flexible modes

[M] = the mass matrix

$\{r\}$ = the influence vector or displacement vector due to unit displacement

P_j = participation factor

Since,

$$P_j = e_j^{\,T}[M]\{r\}, \{Fr\} = Am[M]\{r\}\left[1 - \Sigma M\, e_j\, e_j^{\,T}\right]$$

[*In PIPESTRESS, the low frequency modes are combined by one of the Regulatory Guide 1.92 methods in the response spectrum analysis.*]* For each support level, there is a pseudo-load vector or left-out-force vector in the X, Y and Z directions. These left-out-force vectors are used to generate left-out-force solutions which are multiplied by a scalar amplitude equal to a magnification factor specified by the user. This factor is usually the ZPA (zero period acceleration) of the response spectrum for the corresponding direction. The resultant low frequency responses are combined by square root of the sum of the squares with the high frequency responses (rigid modes results).

[*In GAPPIPE, the results from the high frequency responses are also combined by the square root of the sum of the squares with those from the resultant loads contributed by lower modes.*]* The missing mass correction for an independent support motion or multiple response spectra analysis is exactly the same as that for the single enveloped response spectrum analysis except that Am used is the envelope of all the zero period accelerations of all the independent support inputs.

3.7.3.7.1.2 SRP 3.7.2 Method

[*The method described in SRP Section 3.7.2 may also be used for combination of high-frequency modes.*]*

The following is the procedure for incorporating responses associated with high-frequency modes.

Step 1 Determine the modal responses only for those modes having natural frequencies less than that at which the spectral acceleration approximately returns to the zero period acceleration (33 hertz for the Regulatory Guide 1.60 response spectra). Combine such modes according to the methods discussed in Subsection 3.7.3.7.2.

Step 2 For each degree of freedom included in the dynamic analysis, determine the fraction of degree of freedom mass included in the summation of all modes included in Step 1. This fraction d_i for each degree of freedom is given by:

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$$d_i = \sum_{n=1}^{N} C_n \times \phi_{n,i}$$

where:

n = order of mode under consideration

N = number of modes included in Step 1

$\phi_{n,i}$ = nth natural mode of the system

C_n is the participation factor given by:

$$C_n = \frac{(\phi_n)^T [m] (1)}{(\phi_n)^T [m] (\phi_n)}$$

Next, determine the fraction of degree of freedom mass not included in the summation of these modes:

$$e_i = d_i - \delta_{ij}$$

where δ_{ij} is the Kronecker delta, which is 1 if degree of freedom i is in the direction of the earthquake motion and 0 if degree of freedom i is a rotation or not in the direction of the earthquake input motion.

If, for any degree of freedom i, the absolute value of this fraction e_i exceeds 0.1, the response from higher modes is included with those included in Step 1.

Step 3 Higher modes can be assumed to respond in phase with the zero period acceleration and, thus, with each other. Hence, these modes are combined algebraically, which is equivalent to pseudostatic response to the inertial forces from these higher modes excited at the zero period acceleration. The pseudostatic inertial forces associated with the summation of all higher modes for each degree of freedom i are given by:

$$P_i = ZPA \times M_i \times e_i$$

where:

P_i = force or moment to be applied by degree of freedom i

M_i = mass or mass moment of inertia associated with degree of freedom i.

The subsystem is then statically analyzed for this set of pseudo static inertial forces applied to all degrees of freedom to determine the maximum responses associated with high-frequency modes not included in Step 1.

Step 4 The total combined response to high-frequency modes (Step 3) is combined by the square root of sum of the squares method with the total combined response from lower-frequency modes (Step 1) to determine the overall structural peak responses.

3.7.3.7.2 Combination of Low-Frequency Modes

This subsection describes the method for combining modal responses in the seismic response spectra analysis. [*The total unidirectional seismic response for subsystems is obtained by combining the individual modal responses using the square root sum of the squares method. For subsystems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. For piping systems, the methods in Regulatory Guide 1.92 are used for modal combinations.*]* For other subsystems, the methods in Regulatory Guide 1.92 or the following alternative methods may be used. [*The groups of closely spaced modes are chosen so that the differences between the frequencies of the first mode and the last mode in the group do not exceed 10 percent of the lower frequency.*

Combined total response for systems having such closely spaced modal frequencies is obtained by adding to the square root sum of squares of all modes the product of the responses of the modes in each group of closely spaced modes and coupling factor.]* This can be represented mathematically as:

$$R_T^{\,2} = \sum_{i=1}^{N} R_i^{\,2} + 2\sum_{j=1}^{S} \sum_{k=Mj}^{Nj-1} \sum_{\ell=k+1}^{Nj} R_k \, R_\ell \, \varepsilon_{k\ell}$$

where:

R_T	=	total unidirectional response
R_i	=	absolute value of response of mode i
N	=	total number of modes considered
S	=	number of groups of closely spaced modes
	=	lowest modal number associated with group j of closely spaced modes
N_j	=	highest modal number associated with group j of closely spaced modes
$\varepsilon_{k\ell}$	=	coupling factor, defined as follows:

$$\varepsilon_{k\ell} = \left(1 + \frac{\left(w_k^{'} - w_\ell^{'}\right)^2}{\left(\beta_k^{'}\, w_k + \beta_\ell^{'}\, w_\ell\right)^2} \right)^{-1}$$

and,

$$w_k^{'} = w_k \left[1 - \left(\beta_k\right)^2 \right]^{1/2}$$

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$$\beta_k' = \beta_k + \frac{2}{w_k t_d}$$

where:

w_k = frequency of closely spaced mode k

β_k = fraction of critical damping in closely spaced mode k

t_d = duration of the earthquake (= 30 seconds)

[*Alternatively, a more conservative grouping method can be used in the seismic response spectra analyses. The groups of closely spaced modes are chosen so that the difference between two frequencies is no greater than 10 percent.*]* Therefore,

$$R_T{}^2 = \sum_{i=1}^{N} R_i{}^2 + 2\Sigma\, \varepsilon_{k\ell}\, R_k\, R_\ell$$

where:

$$\frac{|\, w_k - w_\ell \,|}{w_\ell} \leq 0.1$$

All other terms for the modal combination remain the same. The 10 percent grouping method is more conservative than the grouping method because the same mode can appear in more than one group.

In addition to the above methods, any of the other methods in Regulatory Guide 1.92 may be used for modal combination.

[*3.7.3.8 Analytical Procedure for Piping*

This subsection describes the modeling methods and analytical procedures for piping systems.

The piping system is modeled as beam elements with lump masses connected by a network of elastic springs representing the stiffness properties of the piping system. Concentrated weights such as valves or flanges are also modeled as lump masses. The effects of torsion (including eccentric masses), bending, shear, and axial deformations, and effects due to the changes in stiffness values of curved members are accounted for in the piping dynamic model.

The lump masses are selected so that the maximum spacing is not greater than the length that would produce a natural frequency equal to the zero period acceleration (ZPA) frequency of the seismic input when calculated based on a simply supported beam. As a minimum, the number of degrees of freedom is equal to twice the number of modes with frequencies less than the zero period acceleration frequency.

The piping system analysis model includes the effect of piping support mass when the contributory mass of the support is greater than 10 percent of the total mass of the effected piping spans. The contributory mass of the support is the portion of the support mass that is attached to the piping; such as clamps, bolts, trunnions, struts, and snubbers. Supports that are not directly attached to the

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piping, such as box frames, need not be considered for mass effects. The mass of the applicable support will not affect the response of the system in the supported direction, herefore only the unsupported direction needs to be considered. Based on this reasoning, the mass of full anchors can be neglected. The total mass of each effected piping span includes the mass of the piping, fluid contents, insulation, and any concentrated masses (for example, valves or flanges) between the adjacent supports in each unrestrained direction on both sides of the applicable support. For example; the contributory mass of an X direction support must be compared to the mass of the piping spans in the unrestrained Y and Z directions. A contributory support mass that is less than 10 percent of the masses of the effected spans will have insignificant effect on the response of the piping system and can be neglected.

The stiffness matrix of the piping system is calculated based on the stiffness values of the pipe elements and support elements. Default rigid or calculated support stiffness values are used (see subsections 3.9.3.1.5 and 3.9.3.4). When the support deflections are limited to 1/8 inches for the dynamic combined faulted loads, default rigid support stiffness values are used. If the dynamic combined faulted load deflection for any support exceeds 1/8 inches, calculated support stiffness values are used for the affected support.

Valves, equipment and piping modules are considered as rigid if the natural frequencies are greater than 33 hertz. Valves with lower frequencies are included in the piping system model. See subsection 3.7.3.8.2.1 for flexible equipment and subsection 3.7.3.8.3 for flexible modules.

See subsection 3.9.3.1.4 for the primary loop piping and support system.]*

3.7.3.8.1 Supporting Systems

This subsection deals with the analysis of piping systems that provide support to other piping systems. The methods used for the analysis of the primary loop piping are described in Appendix 3C. [*The supported piping system may be excluded from the analysis of the supporting piping system when the ratio of the supported pipe to supporting pipe moment of inertia is less than or equal to 0.04.*

If the ratio of the run piping outside diameter to the branch piping outside diameter (nominal pipe size) exceeds or equals 3.0, the branch piping can be excluded from the analysis of the run piping. The mass and stiffness effects of the branch piping are considered as described below.

Stiffness Effect

The stiffness effect of the decoupled branch pipe is considered significant when the distance from the run pipe outside diameter to the first rigid or seismic support on the decoupled branch pipe is less than or equal to one half the deadweight span of the branch pipe (given in ASME III Code Subsection NF).

Mass Effect

Considering one direction at a time, the mass effect is significant when the weight of half the span (from the decoupling point) of the branch pipe in one direction is more than 20 percent the weight of the main run pipe span in the same direction. Concentrated weights in the branch pipe are considered. A branch pipe span in x direction is the span between the decoupled branch point and the first seismic or rigid support in the x direction. A main run pipe span in the x direction is the piping bounded by the first seismic or rigid support in the x direction on both sides of the decoupled branch point. Similarly, the same definition applies to the spans in other directions (y and z).

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If the calculated branch pipe weight is less than 20 percent but more than 10 percent of the main run pipe weight, this weight is lumped at the decoupling point of the run pipe for the run pipe analysis. This weight can be neglected if it is less than 10 percent of the main run pipe weight.

Required Coupled Branch Piping

If the stiffness and/or mass effects are considered significant, the branch piping is included in the piping analysis for the run pipe analysis. The portion of branch piping considered in the analysis adequately represents the behavior of the run pipe and branch pipe. The branch line model ends in one of the following ways:

- *First six-way anchor*
- *Four rigid/seismic supports in each of the three perpendicular directions*
- *Rigidly supported zone as described* in *subsection 3.7.3.13.4.2*]*

3.7.3.8.2 Supported Systems

This subsection deals with the analysis of piping systems that are supported by other piping systems or by equipment.

3.7.3.8.2.1 Large Diameter Auxiliary Piping

[*This subsection deals with ASME Class 1 piping larger than 1-inch nominal pipe size and ASME Class 2 and 3 piping with nominal pipe size larger than 2 inches. The response spectra methodology is used.*

When the supporting system is a piping system, the supported pipe (branch) can be decoupled from the supporting pipe (run) when the ratio of the run piping nominal pipe size to branch pipe nominal pipe size is greater than or equal to three to one. Decoupling can also be done when the moment of inertia of the branch pipe is less than or equal to 4 percent of the moment of inertia of the run pipe.

During the analysis of the branch piping, resulting values of tee anchor reactions are checked against the capabilities of the tee.

The seismic inertia effects of equipment and piping that provide support to supported (branch) piping systems are considered when significant. When the frequency of the supporting equipment is less than 33 hertz, then either a coupled dynamic model of the piping and equipment is used, or the amplified response spectra at the equipment connection point is used with a decoupled model of the supported piping. When supported piping is supported by larger piping, one of the following methods is used:

- *A coupled dynamic model of the supported piping and the supporting piping*

- *Amplified response spectra at the connection point to the supporting piping with a decoupled model of the supported piping*]*

3.7.3.8.2.2 Small-Diameter Auxiliary Piping

[*This subsection deals with ASME Code Class 1 piping equal to or less than 1-inch nominal pipe size and ASME Class 2 and 3 piping with nominal pipe sizes less than or equal to 2 inches. This includes instrumentation tubing. These piping systems may be supported by equipment or primary loop piping or other auxiliary piping or both. The response spectra or equivalent static load methodology is used. One of the following methods may be used for these systems:*

- *Same method as described in subsection 3.7.3.8.2.1*

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- *Equivalent static analysis based on appropriate load factors applied to the response spectra acceleration values*]*

Subsections 3.9.3 and 3.9.8.2 discuss the final design and as built reconciliation of small bore piping.

3.7.3.8.3 Piping Systems on Modules

Many portions of the systems for the AP1000 are assembled as modules offsite and shipped to the plant as completed units. This method of construction does not result in any unique requirements for the analysis of these structures, systems, or components. Existing industry standards and regulatory requirements and guidelines are appropriate for the evaluation of structures, systems, and components included in modules.

The modules are constructed using a structural steel framework to support the equipment, pipe, and pipe supports in the module. The structural steel framework is designed as part of the building structure according to the criteria given in Subsection 3.8.4.

One exception is the pressurizer and safety relief valve module, which is attached to the top of the pressurizer. For this module the structures and piping arrangements support valves off the pressurizer and not the building structure. The structural steel frame is designed as a component support according to ASME Code, Section III, Subsection NF. [*Piping in modules is routed and analyzed in the same manner as in a plant not employing modules. Piping is analyzed from anchor point to anchor point, which are not necessarily at the boundaries of the module.*]* This is consistent with the manner in which room walls are treated in a nonmodule plant.

[*The supported piping or component may be decoupled from the seismic analysis of the structural frame based on the following criteria. The mass ratio, Rm, and the frequency ratio, Rf, are defined as follows:*

- *Rm = mass of supported component or piping/mass of supporting structural frame*
- *Rf = frequency of the component or piping/frequency of the structural frame*

Decoupling may be done when:

- *Rm < 0.01, for any Rf, or*
- *Rm $\geq$ 0.01 and $\leq$ 0.10, if Rf $\leq$ 0.8 or if Rf is $\geq$ 1.25.*

In addition, supported piping may be decoupled if analysis shows that the effect on the structural frame is small, that is, when the change in response is less than 10 percent. When piping or components are decoupled from the analysis of the frame, the contributory mass of the piping and components is included as a rigid mass in the model of the structural frame.]*

When piping or components are decoupled from the analysis of the frame using the preceding criteria, the effect of the frame is accounted for in the analysis of the decoupled components or piping. Either an amplified response spectra or a coupled model is used. The amplified response spectra are obtained from the time history safe shutdown earthquake analysis of the frame. The coupled model consists of a simplified mass and stiffness model of the frame connected to the seismic model of the components or piping.

Alternative criteria may be applied to simple frames that behave as pipe support miscellaneous steel. Decoupling may be done when the deflection of the frame due to dynamic combined faulted loading is less than or equal to 1/8 inch. These deflections are defined with respect to the structure to which the structural frame is attached. The stiffness of the intervening elements between the frame and the supported piping or component is considered as follows: Default rigid stiffness values are used for

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supports except that vendor stiffness values are used for snubbers and rigid gapped supports. The mass of the structural frame is evaluated as a self-weight excitation loading on the frame and the structures supporting the frame. The same approach is used for pipe support miscellaneous steel, as described in Subsection 3.9.3.4.

When the supported components or piping cannot be decoupled, they are included in the analysis model of the structural frame. The interaction between the piping and the frame is incorporated by including the appropriate stiffness and mass properties of the components, piping, and frame in the coupled model.

[*3.7.3.8.4 Piping Systems with Gapped Supports*

This subsection describes the analysis methods for piping systems with rigid gapped supports. These supports may be used to minimize the number of pipe support snubbers and the corresponding inservice testing and maintenance activities.

The analysis consists of an iterative response spectra analysis of the piping and support system. Iterations are performed to establish calculated piping displacements that are compatible with the stiffness and gap of the rigid gapped supports. The results of the computer program GAPPIPE, which uses this methodology, are supported with test data (Reference 13).

The method implemented in GAPPIPE to analyze piping systems supported by rigid gapped supports is based on the equivalent linearization technique. GAPPIPE analysis is performed whenever snubber supports are replaced by rigid gapped supports.

The basis of the concept is to find an equivalent linear spring with a response-dependent stiffness for each nonlinear rigid gapped support, or limit stop, in the mathematical model of the piping system. The equivalent linearized stiffness minimizes the mean difference in force in the support between the equivalent spring and the corresponding original gapped support. The mean difference is estimated by an averaging process in the time domain, that is, across the response duration, using the concept of random vibration. Details of the design and analysis methods and modeling assumptions are described in Reference 12.]*

3.7.3.9 Combination of Support Responses

This subsection describes alternative methods for combining the responses from the individual support or attachment points that connect the supported system or subsystem to the supporting system or subsystem. There are two aspects to the responses from the support or attachment points: seismic anchor motions and envelope or multiple-input response spectra methodology.

Seismic Anchor Motions – The response due to differential seismic anchor motions is calculated using static analysis (without including a dynamic load factor). In this analysis, the static model is identical to the static portion of the dynamic model used to compute the seismic response due to inertial loading. In particular, the structural system supports in the static model are identical to those in the dynamic model.

[*The effect of relative seismic anchor displacements is obtained either by using the worst combination of the peak displacements or by proper representation of the relative phasing characteristics associated with different support inputs. For components supported by a single concrete building (coupled shield and auxiliary buildings, or containment internal structures), the seismic motions at all elevations above the basemat are taken to be in phase. When the component supports are in the same structure, the relative seismic anchor motions are small and the effects are neglected. This is applicable to building structures and to those supplemental steel frames that are rigid in comparison to the components. Supplemental steel frames that are flexible can have*

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significant seismic anchor motions which are considered. When the components supports are in different structures, the relative seismic anchor motion between the structures is taken to be out-of-phase and the effects are considered. The results of the modal spectra analysis (multiple input or envelope) are combined with the results from seismic anchor motion by the absolute sum method or the SRSS method, as described in Tables 3.9-5 and 3.9-6.]*

Response Spectra Methods – The envelope broadened uniform-input response spectra can lead to excessive conservatism and unnecessary pipe supports. The peak shifting method and independent support motion spectra method are used to avoid unnecessary conservatism.

Seismic Response Spectra Peak Shifting

The peak shifting method may be used in place of the broadened spectra method, as described below.

Determine the natural frequencies $(f_e)_n$ of the system to be qualified in the broadened range of the maximum spectrum acceleration peak.

If no equipment or piping system natural frequencies exist in the ±15 percent interval associated with the maximum spectrum acceleration peak, then the interval associated with the next highest spectrum acceleration peak is selected and used in the following procedure.

Consider all N natural frequencies in the interval

$f_j - 0.15f_j \leq (f_e)_n \leq f_j + 0.15f_j$

where:

f_j = the frequency of maximum acceleration in the envelope spectra

n = 1 to N

The system is then evaluated by performing N + 3 separate analyses using the envelope unbroadened floor design response spectrum and the envelope unbroadened spectrum modified by shifting the frequencies associated with each of the spectral values by a factor of +0.15; -0.15; and

$$\frac{(f_e)_n - f_j}{f_j}$$

where:

n = 1 to N

The results of these separate seismic analyses are then enveloped to obtain the final result desired (e.g., stress, support loads, acceleration, etc.) at any given point in the system. If three different floor response spectrum curves are used to define the response in the two horizontal and the vertical directions, then the shifting of the spectral values as defined above may be applied independently to these three response spectrum curves.

Independent Support Response Spectrum Methods

The use of multiple-input response spectra accounts for the phasing and interdependence characteristics of the various support points. The following alternative methods are used for the

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AP1000 plant. These are based on the guidelines provided by the "Pressure Vessel Research Committee Technical Committee on Piping Systems" (Reference 14).

[*Envelope Uniform Response Spectra - Method A - The seismic response spectrum that envelopes the supports is used in place of the spectra at each support in the envelope uniform response spectra. Also, the contribution from the seismic anchor motion of the support points is assumed to be in phase and is added algebraically as follows:*

$$q_i = d_i \sum_{j=1}^{N} P_{ij}$$

where:

q_i	=	*combined displacement response in the normal coordinate for mode i*
d_i	=	*maximum value of* d_{ij}
d_{ij}	=	*displacement spectral value for mode i associated with support "j"*
P_{ij}	=	*participation factor for mode i associated with support j*
N	=	*number of support points*

Enveloped response spectra are developed as the seismic input in three perpendicular directions of the piping coordinate system to include the spectra at the floor elevations of the attachment points and the piping module or equipment if applicable. The mode shapes and frequencies below the cut-off frequency are calculated in the response spectrum analysis. The modal participation factors in each direction of the earthquake motion and the spectral accelerations for each significant mode are calculated. Based on the calculated mode shapes, participation factors, and spectral accelerations of individual modes, the modal inertia response forces, moments, displacements, and accelerations are calculated. For a given direction, these modal inertia responses are combined based on consideration of closely spaced modes and high frequency modes to obtain the resultant forces, moments, displacements, accelerations, and support loads. The total seismic responses are combined by square-root-sum-of-the-squares method for all three earthquake directions.

Independent Support Motion - Method B - When there are more than one supporting structure, the independent support motion (ISM) method for seismic response spectra may be used.

Each support group is considered to be in a random-phase relationship to the other support groups. The responses caused by each support group are combined by the square-root-sum-of-the-square method. The displacement response in the modal coordinate becomes:

$$q_i = \left[\sum_{j=1}^{N} \left(P_{ij}\, d_{ij} \right)^2 \right]^{1/2}$$

A support group is defined by supports that have the same time-history input. This usually means all supports located on the same floor (or portions of a floor) of a structure.]*

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3.7.3.10 Vertical Static Factors

Constant static factors can be used in some cases for the design of seismic Category I subsystems and equipment. The criteria for using this method are presented in Subsection 3.7.3.5.

3.7.3.11 Torsional Effects of Eccentric Masses

[*The methods used to account for the torsional effects of valves and other eccentric masses (for example, valve operators) in the seismic subsystem analyses are as follows:*

- *When valves and other eccentric masses are considered rigid, the mass of the operator and valve body or other eccentric mass are located at their respective center of gravity. The eccentric components (that is, yoke, valve body) are modeled as rigid members.*

- *When valves and other eccentric masses are not considered rigid, the dynamic models are simulated by the lumped masses in discrete locations (that is, center of gravity of valve body and valve operator), coupled by elastic members with properties of the eccentric components.*]*

3.7.3.12 Seismic Category I Buried Piping Systems and Tunnels

[*There are no seismic Category I buried piping systems and tunnels in the AP1000 design.*]*

3.7.3.13 Interaction of Other Systems with Seismic Category I Systems

The safety functions of seismic Category I structures, systems, and components are protected from interaction with nonseismic structures, systems, and components; or their interaction is evaluated. The safety-related systems and components required for safe shutdown are described in Section 7.4. This equipment is located in selected areas of the auxiliary building and inside containment. The primary means of protecting safety-related structures, systems, and components from adverse seismic interactions are discussed in the following paragraphs in the order of preference.

- Separation – separation with the use of physical barriers

- Segregation – routing away from location of seismic Category I systems, structures, and components

- Impact Evaluation – contact with seismic Category I systems, structures, and components may occur, and there is insufficient energy in the impact to cause loss of safety function

- Support as seismic Category II

[*Interaction of connected systems with seismic Category I piping is considered by including the other piping in the analysis of the seismic Category I system.*]* Interaction of piping systems that are adjacent to Category I structures, systems, and components is also considered. This is discussed in Subsection 3.7.3.13.4.

The containment and each room outside containment containing safety-related systems or equipment, as identified in Table 3.7.3-1, are reviewed for potential adverse seismic interactions to demonstrate that systems, structures, and components are not prevented from performing their required safe shutdown functions. In addition, the review identifies the protection features required to mitigate the consequences of seismic interaction in an area that contains safety-related equipment.

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The evaluation steps to address seismic interaction taken for each room or building area containing seismic Category I systems, structures, and components are:

1. Define targets susceptible to damage (sensitive targets);
   Sensitive targets are those seismic Category I components for which adverse spatial interaction can result in loss of safety function.

2. Define sources which can potentially interact in an adverse manner with the target.

3. If possible, assure adequate free space to eliminate the possibility of seismically-induced damaging impacts for the sensitive targets.

4. Assess impact effects (interaction) when adequate free space is not present.

5. Correct adverse seismic interaction conditions.

The three-dimensional computer model and composites developed for the nuclear island are used during the design process of the systems and components in the nuclear island, to aid in evaluating and documenting the review for seismic interactions. This review is performed using the design criteria and guidelines described in Subsections 3.7.3.13.1 through 3.7.3.13.4.

The seismic interaction review is discussed in Subsection 3.7.5.3. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition.

3.7.3.13.1 Separation and Segregation

Separation – The general plant arrangement provides physical separation between the seismic Category I and nonseismic structures, systems, and components to the maximum extent practicable in the nuclear island. The objective is to assist in the preclusion of a potential adverse interaction if the nonseismic structures, systems and components were to fail during a seismic event. Whenever possible, nonseismic pipe, electrical raceway, or ductwork is not routed above or adjacent to safety-related equipment, pipe, electrical raceway, or ductwork, thereby eliminating the possibility of seismic interaction.

Workstations and other equipment in the Main Control Room are separated from piping. Further, as stated in Subsection 3.2.1.1.2, structures, systems, and components that are located overhead in the Main Control Room are supported as seismic Category II.

Segregation – Where separation by physical means cannot be accomplished and it becomes necessary to locate or route nonseismic structures, systems, and components in or through safety-related areas, the nonseismic structures, systems and components are segregated from the seismic Category I items to the extent practicable.

Nonseismic cabinets are separated or segregated from seismic Category I cabinets. Also, if a cabinet is a source or a target, the cabinet doors must be secured by latches or fasteners to assure they do not open during a seismic event.

3.7.3.13.2 Impact Analysis

Adverse spatial interaction (i.e., loss of structural integrity or function effecting safety) can potentially occur when two items are in close proximity. Adverse spatial interaction can result from contact or impact from overturning. Seismic Category I systems, structures, and components that are sensitive to seismic interaction are identified as potential targets. Sources are structures or components that

can have adverse spatial interaction with the seismic Category I systems, structures, and components. Identification and evaluation of spatial interactions includes the following considerations:

- Proximity of the source to the target. That is, the location of the source within the impact evaluation zone (shown in Figure 3.7.3-1)

  If a source is outside the impact evaluation zone, and does not enter this zone if overturning occurs, no adverse spatial interaction can occur with the identified target. If the source is within the impact evaluation zone and the supports of the source fail, the source could free fall, potentially impacting the target.

- Robustness of target

  If a target has significant structural integrity, and its function is not an issue, adverse spatial interaction could not occur with the identified source.

- Energy of impact

  The energy of the source impacting the target may be so low as not to cause adverse spatial interaction with the target.

A specific nonseismic structure, system, or component identified as a source to a specific safety-related component can be acceptable without being supported as seismic Category II, if an analysis demonstrates that the weight and configuration of the source, relative to the target, and the trajectory of the source are such that the interaction would not cause unacceptable damage to the target. For example, a nonseismic instrument tube routed above a seismic Category I electrical cable tray would not pose a hazard and would be acceptable.

Nonseismic equipment can overturn as a result of a safe shutdown earthquake. The trajectory of its fall is evaluated to determine if it poses a potential impact hazard to a safety-related structure, system, or component. If it poses a hazard, the equipment is relocated, or it is supported as described in Subsection 3.7.3.13.3.

Nonseismic walls, platforms, stairs, ladders, grating, handrail installations, or other structures next to safety-related structures, systems, and components are evaluated to determine if their failure is credible.

Should a nonseismic structure, system, or component be capable of being dislodged from its supports, the trajectory of its fall is evaluated for potential adverse impacts. If these present a hazard, the structure, system or component is relocated or supported as described in Subsections 3.7.3.13.3 and 3.7.3.13.4. Impact is assumed for sources within an impact evaluation zone around the safety-related equipment. The impact evaluation zone is defined as the envelope around the target for which a source, if located outside of the envelope, would not impact the target during a safe shutdown earthquake in the event the supports of the source were to fail and allow the source to fall. The impact evaluation zone is defined by the volume extending 6 feet horizontally from the perimeter of the seismic Category I object up to a height of 35 feet. The impact evaluation zone above 35 feet is defined by a 10-degree cone radiating vertically from the foot of the object, projected from its perimeter. This definition of the impact evaluation zone is illustrated in Figure 3.7.3-1. The impact evaluation zone need not extend beyond seismic Category I structures such as walls or floor slabs.

Using seismic experience data, the following seismic Category I equipment (potential targets) are not sensitive to piping, HVAC ducts, and cable tray interaction because they are robust to these types of impact:

- Tanks, "heavy" equipment (for example, heat exchangers)
- Mechanical or electrical penetrations
- Heating, ventilation, and air conditioning (HVAC)
- Adjacent piping
- Conduits
- Cable trays
- Structures

3.7.3.13.3 Seismic Category II Supports

Where the preceding approaches of separation, segregation, or impact analysis cannot prevent unacceptable interaction, the source is classified and supported as seismic Category II. The seismic Category II designation provides confidence that these nonseismic structures, systems, and components can withstand the forces of a safe shutdown earthquake in addition to the loading imparted on the seismic Category II supports due to failure of the remaining nonseismically supported portions. This includes nozzle loads from the nonseismic piping. Design methods and stress criteria for systems, structures, and components classified as seismic Category II are the same as for seismic Category I systems, structures, and components, except for piping which is described in Subsection 3.7.3.13.4.2. However, the functionality of these seismic Category II sources does not have to be maintained following a safe shutdown earthquake.

HVAC duct and/or cable trays within the impact evaluation zone are seismically supported using the criteria given in Appendices 3F and 3A for seismic Category I assuring that the HVAC and cable tray segments identified as a source will not fall or adversely impact the sensitive target. Adequate free space between the source and target is assured using the load combination that includes the safe shutdown earthquake. The seismic displacement of the HVAC duct and/or cable tray is 6 inches or the calculated displacement.

Nonseismic equipment identified as a source within the impact evaluation zone is supported as seismic Category II. Support seismic loads include seismic inertia loads of the equipment determined as described in Subsection 3.7.3.5 and nozzle loads from attached piping determined as described in Subsection 3.7.3.13.4.2. Adequate free space is assessed considering a 6-inch deflection envelope for equipment identified as a source, or calculated deflections obtained using the safe shutdown earthquake load combination and elastic analysis.

[*3.7.3.13.4 Interaction of Piping with Seismic Category I Piping Systems, Structures, and Components*

This subsection describes the design methods for piping to prevent adverse spatial interactions.

3.7.3.13.4.1 Seismic Category I Piping

The safe shutdown earthquake piping displacements obtained for the seismic Category I piping are used for the evaluation of seismic interaction with sensitive equipment. Adequate free space between a source and a target is checked adding absolutely the piping safe shutdown earthquake deflection and the safe shutdown earthquake target deflection along with the other loads (e.g., dead weight, thermal) that are in the appropriate design criteria load combinations. Sensitive equipment for piping as the source is seismic Category I equipment shown in Table 3.7.3-2 along with the portion that must be protected ("zone of protection"). Supports may be added to limit seismic movement to eliminate potential adverse interaction.

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3.7.3.13.4.2 Seismic Category II Piping

This subsection describes the methods and criteria for piping that is connected to seismic Category I piping. Interaction of seismic Category I piping and nonseismic Category I piping connected to it is achieved by incorporating into the analysis of the seismic Category I system a length of pipe that represents the actual dynamic behavior of the complete run of the nonseismic Category I system. The length considered is classified as seismic Category II and extends to the interface anchor or rigid support as described below.

The seismic Category II portion of the line, up to the interface anchor or interface rigid support (last seismic support), is analyzed according to Equation 9 of ASME Code, Section III, Class 3, with a stress limit equal to the smaller of 4.5 S_h and 3.0 S_y. In either case, the nonseismic piping is isolated from the seismic Category I piping by anchors or seismic supports. The anchor or seismic Category II supports are designed for loads from the nonseismic piping. This includes three plastic moment components (M_{p1}, M_{p2}, or M_{p3}) in each of three local coordinate directions. The responses to the three moments are evaluated independently. The seismic Category II portion of the line is analyzed by the response spectrum or equivalent static load method for safe shutdown earthquake.

Single Interface Anchor

The seismic Category II piping may be terminated at a single interface anchor (six-way). This anchor and the supports on the seismic Category II piping are evaluated for safe shutdown earthquake loadings using the rules of ASME III Subsection NF. If the anchor is an equipment nozzle, then the equipment load path through the equipment supports are evaluated to the same acceptance criteria as seismic Category I equipment.

Anchor Followed by a Series of Seismic Supports

The seismic Category II piping may be terminated at the last seismic support which follows a six-way anchor on the seismic Category II piping. This last seismic support and the supports on the seismic Category II piping are evaluated for safe shutdown earthquake loadings using the rules of ASME III Subsection NF. From the anchor to the last seismic support, the response to the plastic moments (M_{p1}, M_{p2}, or M_{p3}) is combined with the responses to seismic anchor motions and equivalent static seismic inertia of the piping system by the absolute sum method. The responses to these moments are evaluated independently. The support and anchor loads due to the plastic moments (M_{p1}, M_{p2}, or M_{p3}) of the seismically analyzed and supported section can be reduced if the elbow/bend resultant moments have exceeded the plastic limit moments of the elbow/bend. The value of the reduction factor RF is as follows:

RF	=	*Multiplier used to reduce the interface anchor and support loads*
RF	=	*< I, (if RF > I, no reduction is applicable)*
RF	=	M_L/M_a
M_a	=	*Resultant moment at elbow/bend. Use maximum value if several elbows/bends are within seismically supported region.*
M_L	=	$0.8h^{0.6} D^2 t\ Sy$ *for* $h < 1.45$
M_L	=	$D^2 t\ Sy$ *for* $h > 1.45$
h	=	*Flexibility characteristic of elbow/bend*
D	=	*Outside diameter of elbow/bend*

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t = *Thickness of elbow/bend*

R = *Bend radius of elbow/bend*

Rigid Region

The seismic Category II piping may be terminated at the last seismic support of a rigidly supported region of the piping system. The rigid region is typically defined as either four bi-lateral supports around an elbow or six bilateral supports around a tee. The structural behavior of the rigid region is similar to that of a six-way anchor. The frequency of the piping system in the rigid region is greater than or equal to 33 hertz. This last seismic support in the rigid region and the supports on the seismic Category II piping are evaluated for safe shutdown earthquake loadings using the rules of ASME III Subsection NF.

3.7.3.13.4.3 Nonseismic Piping

Nonseismic piping within the impact evaluation zone is seismically supported, thereby ensuring that the pipe segment identified as a source will not fall or adversely impact the sensitive target (Table 3.7-2). This situation is shown in Figure 3.7.3-2, and the seismic supported piping criteria described below:

Supports within the impact evaluation zone, plus one transverse support in each transverse direction beyond the impact evaluation zone, are classified as seismic Category II and are evaluated for the safe shutdown earthquake loading using the rules of ASME III, Subsection NF.

- *Piping within the impact evaluation zone plus one transverse support in each transverse direction are evaluated to Equation 9 of ASME Code, Section III, Class 3, with a stress limit equal to the smaller of 4.5 S_h and 3.0 S_y. Outside the impact evaluation zone, the nonseismic piping meets ASME/ANSI B31.1 requirements.*

- *The nonseismic piping and seismic Category II supports are designed for loads from the nonseismic piping beyond the impact evaluation zone. This includes three plastic moment components (M_{p1}, M_{p2}, or M_{p3}) in each of three local coordinate directions applied at the first and last seismic Category II support. The responses to the three moments are evaluated independently. The response from the moments applied at the first seismic*

- *Category II support is combined with the response from the moments applied at the last seismic Category II support and with the responses to seismic anchor motions and equivalent static seismic inertia of the piping system by the absolute sum method. The support and anchor loads due to the plastic moments (M_{p1}, M_{p2}, or M_{p3}) of the seismically analyzed and supported section can be reduced if the elbow/bend resultant moments have exceeded the plastic limit moments of the elbow/bend. The value of the reduction factor RF is the same as the value for connected seismic Category II piping described above.*

- *The piping segment identified as the source has at least one effective axial support.*

- *Adequate free space between a source and a target is checked adding absolutely the piping safe shutdown earthquake deflections (defined following seismic Category II piping analysis methodology) and the safe shutdown earthquake target deflection. Also included are the displacements associated with the appropriate load cases.*

- *When the anchor is an equipment nozzle, the equipment is supported as seismic Category II as described in subsection 3.7.3.13.3.*]*

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3.7.3.14 Seismic Analyses for Reactor Internals

See Subsection 3.9.2 for the dynamic analyses of reactor internals.

3.7.3.15 Analysis Procedure for Damping

Damping values used in the seismic analyses of subsystems are presented in Subsection 3.7.1.3. Safe shutdown earthquake damping values used for different types of analysis are provided in Table 3.7.1-1. For subsystems that are composed of different material types, the composite modal damping approach with the weighted stiffness method is used to determine the composite modal damping value. Alternately, the minimum damping value may be used for these systems. [*Composite modal damping for coupled building and piping systems is used for piping systems that are coupled to the primary coolant loop system and the interior concrete building. Composite modal damping is used for piping systems that are coupled to flexible equipment or flexible valves. Piping systems analyzed by the uniform envelope response spectra method with rigid valves can be evaluated with 5 percent damping. Five percent damping is not used in piping systems that are susceptible to stress corrosion cracking.*]*

For the time history dynamic analysis and independent support motion response spectra analysis of piping systems, 4 percent, 3 percent, and 2 percent damping values are used as described in Table 3.7.1-1.

When piping systems and nonsimple module steel frames (simple frames are described in Subsection 3.7.3.8.3) are in a single coupled model, composite damping, as described in Subsection 3.7.1.3 is used.

3.7.3.16 Analysis of Seismic Category I Tanks

This subsection describes the seismic analyses for the large, atmospheric seismic Category I pools and tanks. These are reinforced concrete structures with stainless steel liners or with structural modules, as discussed in Subsections 3.8.3 and 3.8.4. They include the spent fuel pit in the auxiliary building, the in-containment refueling water storage tank, and the passive containment cooling water tank incorporated into the shield building roof. There are no other seismic Category I tanks.

The seismic analyses of the tank consider the impulsive and convective forces of the water as well as the flexibility of the walls. For the spent fuel pit, cask loading pit, cask washdown pit and fuel transfer canal, the impulsive loads are calculated by considering a portion of the water mass responding with the concrete walls. The impulsive forces are calculated by conventional methods for rigid tanks. The passive containment cooling water tank is analyzed using methods described in Reference 15 for toroidal tanks. It is also analyzed by finite element methods. The in-containment refueling water storage tank is irregular in plan and is analyzed by finite element methods.

3.7.3.17 Time History Analysis of Piping Systems

[*The time history dynamic analysis is an alternate seismic analysis method for response spectrum analysis when time history seismic input is used. This method is also used for dynamic analyses of piping systems subjected to time history hydraulic transient loadings or forcing functions induced by postulated pipe breaks. The modal superposition method is used to solve the equations of motion. The computer programs used are GAPPIPE, PIPESTRESS, ANSYS, and WECAN.*

The modal superposition method is based on the equations of motion which can be decoupled as long as the piping system is within its elastic limit. The modal responses are obtained from integrating the decoupled equations. The total responses are obtained by the algebraic sum of the individual responses of the individual modes at each time step. The cutoff frequency is selected based on the

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frequency content of the input forcing function and the highest significant frequency of the piping system. The integration time step is no larger than 10 percent of the period of the cutoff frequency.

For dynamic analysis, including seismic analysis at a hard rock site, three separate analyses are performed for each loading case to account for uncertainties. The three analyses correspond to three different time scales: normal time, time expanded by 15 percent, and time compressed by 15 percent. For time history analysis of piping system models that include a dynamic model of the supporting concrete building either the building stiffness is varied by + and - 30 percent, or the time scale is shifted by + and - 15 percent. Alternately, when uniform enveloping time history analysis is performed, modeling uncertainties are accounted for by the spreading that is included in the broadened response spectra.

For time history analysis using the PIPESTRESS program, the response from the high frequency modes above the cutoff frequency is calculated based on the static response to the left-out-forces. This response is combined with the response from the low frequency modes by algebraic sum at each time step. Composite modal damping is used with PIPESTRESS program. The damping of the individual components is as listed in Table 3.7.1-1.

Alternately, for time history analysis using the PIPESTRESS, GAPPIPE, ANSYS, or WECAN programs, the number of modes used in the modal analysis is chosen so that the results of the dynamic analysis based on the chosen number of modes are within 10 percent of the results of the dynamic analysis based on the next higher number of modes used. The number of modes analyzed is selected to account for the principal vibration modes of the piping system. The modes are combined by algebraic sum. Composite modal damping is used with the ANSYS or WECAN programs. The damping of the individual components is as listed in Table 3.7.1-1.]*

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with Regulatory Guide 1.12

Compliance with Regulatory Guide 1.12 is discussed in this section and in Subsection 1.9.1.

Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operation and shutdown in accordance with Regulatory Guide 1.12.

3.7.4.1.1 Safety Design Basis

The seismic instrumentation serves no safety-related function and therefore has no nuclear safety design basis.

3.7.4.1.2 Power Generation Design Basis

The seismic instrumentation is designed to provide the following:

- Collection of seismic data in digital format
- Analysis of seismic data after a seismic event
- Operator notification that a seismic event exceeding a preset value has occurred
- Operator notification (after analysis of data) that a predetermined cumulative absolute velocity value has been exceeded

*NRC Staff approval is required prior to implementing a change in this information.

3.7.4.2 Location and Description of Instrumentation

The following instrumentation and associated equipment are used to measure plant response to earthquake motion.Four triaxial acceleration sensor units, located as stated in Subsection 3.7.4.2.1, are connected to a time-history analyzer. The time-history analyzer recording and playback system is located in a panel in the nuclear island in a room near the main control room. Seismic event data from these sensors are recorded on a solid-state digital recording system at 200 samples per second per data channel.

This solid-state recording and analysis system has internal batteries and a charger to prevent the loss of data during a power outage, and to allow data collection and analysis in a seismic event during which the power fails. Normally 120 volt alternating current power is supplied from the non-Class 1E dc and uninterruptible power supply system. The system uses triaxial acceleration sensor input signals to initiate the time-history analyzer recording and main control room alarms. The system initiation value is adjustable from 0.002g to 0.02g.

The time-history analyzer starts recording triaxial acceleration data from each of the triaxial acceleration sensors after the initiation value has been exceeded. Pre-event recording time is adjustable from 1.2 to 15.0 seconds, and will be set to record at least 3 seconds of pre-event signal. Post-event run time is adjustable from 10 to 90 seconds. A minimum of 25 minutes of continuous recording is provided. Each recording channel has an associated timing mark record with 2 marks per second, with an accuracy of about 0.02 percent.

The instrumentation components are qualified to IEEE 344-1987 (Reference 16).

The sensor installation anchors are rigid so that the vibratory transmissibility over the design spectra frequency range is essentially unity.

3.7.4.2.1 Triaxial Acceleration Sensors

Each sensor unit contains three accelerometers mounted in a mutually orthogonal array mounted with one horizontal axis parallel to the major axis assumed in the seismic analysis. The triaxial acceleration sensors have a dynamic range of 1000 to 1 (0.001 to 1.0g) and a frequency range of 0.2 to 50 hertz.

One sensor unit will be located in the free field, as discussed in below. The AP1000 seismic monitoring system will provide for signal input from the free field sensor.

A second sensor unit is located on the nuclear island basemat in the spare battery charger room at elevation 66′-6″ near column lines 9 and L.

A third sensor unit is located on the shield building structure at elevation 266′ near column lines 4-1 and K.

The fourth sensor unit is located on the containment internal structure on the east wall of the east steam generator compartment just above the operating floor at elevation 138′ close to column lines 6 and K.

Seismic instrumentation is not located on equipment, piping, or supports since experience has shown that data obtained at these locations are obscured by vibratory motion associated with normal plant operation.

A free-field sensor will be located and installed to record the ground surface motion representative of the site. To be representative of this site in regards to seismic response of structures, systems, and

components, the free-field sensor is located on the ground surface of the engineered backfill. The backfill directly supports the Nuclear Island and the adjacent structures and extends out from these structures a significant distance. The free-field sensor is located where the backfill vertically extends from the top of the Blue Bluff Marl to the ground surface, but horizontally at a distance where possible effects on recorded ground motion associated with surface features, buildings, and components would be minimized. The trigger value is initially set at 0.01g.

3.7.4.2.2 Time-History Analyzer

The time-history analyzer receives input from the triaxial acceleration sensors and, when activated as described in Subsection 3.7.4.3, begins recording the triaxial data from each triaxial acceleration sensor and initiates audio and visual alarms in the main control room.

This recorded data will be used to evaluate the seismic acceleration of the structure on which the triaxial acceleration sensors are mounted.

The time-history analyzer is a multichannel, digital recording system with the capability to automatically download the recorded acceleration data to a dedicated computer for data storage, playback, and analysis after a seismic event.

The time-history analyzer can compute cumulative absolute velocity (CAV) and the 5 percent of critical damping response spectrum for frequencies between 1 and 10 Hz. The operator may select the analysis of either CAV or the response spectrum. Analysis results are printed out on a dedicated graphics printer that is part of the system and is located in the same panel as the time-history analyzer.

3.7.4.3 Control Room Operator Notification

The time-history analyzer provides for initiation of audible and visual alarms in the main control room when predetermined seismic acceleration values sensed by any of the triaxial acceleration sensors are exceeded and when the system is activated to record a seismic event. In addition to alarming when the system is activated, the analyzer portion of the system will provide a second alarm if the predetermined cumulative absolute velocity value has been exceeded by any of the sensors. Alarms are annunciated in the main control room.

3.7.4.4 Comparison of Measured and Predicted Responses

The recorded seismic data is used by the combined license holder operations and engineering departments to evaluate the effects of the earthquake on the plant structures and equipment.

The criterion for initiating a plant shutdown following a seismic event will be exceedance of a specified response spectrum limit or a cumulative absolute velocity limit. The seismic instrumentation system is capable of computing the cumulative absolute velocity as described in EPRI Report NP-5930 (Reference 1) and EPRI Report TR-100082 (Reference 17).

Post-earthquake operating procedures utilize the guidance of EPRI Reports NP-5930, TR-100082, and NP-6695, as modified and endorsed by the NRC in Regulatory Guides 1.166 and 1.167. A response spectrum check up to 10Hz and the cumulative absolute velocity will be calculated based on the recorded motions at the free field instrument. If the operating basis earthquake ground motion is exceeded or significant plant damage occurs, the plant must be shutdown in an orderly manner.

In addition, the procedures address measurement of the post-seismic event gaps between the new fuel rack and walls of the new fuel storage pit, between the individual spent fuel racks, and from the

spent fuel racks to the spent fuel pool walls, and provide for appropriate corrective actions to be taken if needed (such as repositioning the racks or analysis of the as-found condition).

3.7.4.5 Tests and Inspections

Periodic testing of the seismic instrumentation system is accomplished by the functional test feature included in the software of the time-history recording accelerograph. The system is modular and is capable of single-channel testing or single channel maintenance without disabling the remainder of the system.

Installation and acceptance testing of the triaxial acceleration sensors described in Subsection 3.7.4.2.1 is completed prior to initial startup. Installation and acceptance testing of the time-history analyzer described in Subsection 3.7.4.2.2 is completed prior to initial startup.

3.7.5 Combined License Information

3.7.5.1 Seismic Analysis of Dams

Dams whose failure could affect the site interface flood level are addressed in Subsection 3.7.2.12 and Subsection 2.4.4.

3.7.5.2 Post-Earthquake Procedures

Site-specific procedures for activities following an earthquake are addressed in Subsection 3.7.4.4.

3.7.5.3 Seismic Interaction Review

The seismic interaction review will be updated for as-built information. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition. The as-built seismic interaction review is completed prior to fuel load.

3.7.5.4 Reconciliation of Seismic Analyses of Nuclear Island Structures

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

3.7.5.5 Free Field Acceleration Sensor

The location for the free-field acceleration sensor is addressed in Subsection 3.7.4.2.1.

3.7.6 References

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Table 3.7.1-1
Safe Shutdown Earthquake Damping Values

	Percent
Welded and friction-bolted steel structures and equipment	4
Bearing bolted structures and equipment	7
Prestressed concrete structures	5
Reinforced concrete structures	7
Concrete filled steel plate structures	5
[*Piping (for uniform envelope response spectra analysis)*	*5*
Piping (alternative for time history analysis and independent support motion response spectra analysis)	
Less than or equal to 12-inch diameter	*2*
Greater than 12-inch diameter	*3*
Primary coolant loop	*4*]*
Fuel assemblies	20
Control rod drive mechanisms	5
Full cable trays and related supports	10
Empty cable trays and related supports	7
Conduits and related supports	7
HVAC ductwork	7
HVAC welded ductwork	4
Cabinets and panels for electrical equipment	5
Equipment such as welded instrument racks and tanks	3

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.7.1-2
Embedment Depth and Related
Dimensions of Category I Structures

Structure	Foundation Embedment Depth (ft)	Least Foundation Width (ft)	Structure Height (ft)
Shield Building	See Note	See Note	268.25
Steel Containment Vessel	See Note	See Note	215.33
Auxiliary Building	See Note	See Note	119.50

Note:

1. The seismic Category I structures are founded on a common basemat embedded 39.5 feet, [*with dimensions shown in Figure 3.7.1-14*.]*

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.7.1-3
AP1000 Design Response Spectra
Amplification Factors for Control Points

HORIZONTAL					
Percent of Critical Damping	**Acceleration[1]**				**Displacement[1]**
	A (33 cps)	**B' (25 cps)[2]**	**B (9 cps)**	**C (2.5 cps)**	**D (0.25 cps)**
2.0	1.0	1.70	3.54	4.25	2.50
3.0	1.0	1.66	3.13	3.76	2.34
4.0	1.0	1.63	2.84	3.41	2.19
5.0	1.0	1.60	2.61	3.13	2.05
7.0	1.0	1.55	2.27	2.72	1.88
VERTICAL					
Percent of Critical Damping	**Acceleration[1]**				**Displacement[1]**
	A (33 cps)	**B' (25 cps)[2]**	**B (9 cps)**	**C (3.5 cps)**	**D (0.25 cps)**
2.0	1.0	1.70	3.54	4.05	1.67
3.0	1.0	1.66	3.13	3.58	1.56
4.0	1.0	1.63	2.84	3.25	1.46
5.0	1.0	1.60	2.61	2.98	1.37
7.0	1.0	1.55	2.27	2.59	1.25

Notes:
1. Maximum ground displacement is taken proportional to maximum ground acceleration, and is 36 inches for ground acceleration of 1.0 gravity.
2. The 5 percent damping amplification factor for control point B' is derived per discussion in Subsection 3.7.1.1. This 5 percent damping amplification factor equals 1.3 times the RG 1.60 response spectra at 25 hertz. The amplification factors at control point B' for other damping values are determined by increasing the RG 1.60 response spectra at 25 hertz by 30 percent.

Table 3.7.1-4 (Sheet 1 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Firm Rock								
0.0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.15	57422	3500	57032	3499	0.015
10.0	5.0	2	0.15	57422	3500	56600	3486	0.016
15.0	5.0	3	0.15	57422	3500	55943	3465	0.017
20.0	5.0	4	0.15	57422	3500	55511	3452	0.018
25.0	5.0	5	0.15	56442	3500	55933	3465	0.016
30.0	5.0	6	0.15	56442	3500	55436	3450	0.017
33.5	3.5	7	0.15	57422	3500	56076	3470	0.015
39.5	6.0	8	0.15	57422	3500	55898	3464	0.015
45.0	5.5	9	0.15	57422	3500	55716	3458	0.016
50.0	5.0	10	0.15	57422	3500	55575	3454	0.016
60.0	10.0	11	0.15	56442	3500	55400	3449	0.017
80.0	20.0	12	0.15	56442	3500	54695	3427	0.019
100.0	20.0	13	0.15	56442	3500	53358	3384	0.021
120.0	20.0	14	0.15	56442	3500	52295	3351	0.023
Bedrock	–	–	0.15	300000	8000	298137	8000	0.02

Table 3.7.1-4 (Sheet 2 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Soft Rock								
0.0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.15	27660	2429	27425	2426	0.016
10.0	5.0	2	0.15	29180	2495	28318	2466	0.018
15.0	5.0	3	0.15	30262	2541	28819	2487	0.020
20.0	5.0	4	0.15	30620	2556	28589	2477	0.023
25.0	5.0	5	0.15	30920	2568	29290	2508	0.019
30.0	5.0	6	0.15	31384	2588	29481	2516	0.021
33.5	3.5	7	0.15	31932	2610	30768	2570	0.017
39.5	6.0	8	0.15	32464	2632	31144	2586	0.018
45.0	5.5	9	0.15	33042	2655	31314	2593	0.019
50.0	5.0	10	0.15	33668	2680	31598	2604	0.020
60.0	10.0	11	0.15	34341	2707	31826	2614	0.021
80.0	20.0	12	0.15	35021	2733	31738	2610	0.024
100.0	20.0	13	0.15	35708	2760	31585	2604	0.026
120.0	20.0	14	0.15	36401	2787	31585	2604	0.028
Bedrock	–	–	0.15	300000	8000	298137	8000	0.020

Table 3.7.1-4 (Sheet 3 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Upper Bound Soft to Medium Soil								
0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.11	6873	1414	6664	1397	0.018
10.0	5.0	2	0.11	9844	1692	9202	1641	0.023
15.0	5.0	3	0.11	13917	2012	12880	1942	0.024
20.0	5.0	4	0.11	14971	2087	13629	1997	0.027
25.0	5.0	5	0.11	15645	2133	14574	2065	0.022
30.0	5.0	6	0.11	16419	2186	15045	2099	0.024
33.5	3.5	7	0.11	17873	2280	16908	2225	0.019
39.5	6.0	8	0.11	19036	2353	17873	2287	0.020
45.0	5.5	9	0.11	20387	2435	18996	2358	0.021
50.0	5.0	10	0.11	21726	2514	20136	2428	0.021
60.0	10.0	11	0.11	23234	2600	21366	2501	0.022
80.0	20.0	12	0.11	24712	2681	22314	2556	0.024
100.0	20.0	13	0.11	26151	2758	23137	2602	0.026
120.0	20.0	14	0.11	27546	2831	24009	2651	0.027
Bedrock	–	–	0.15	300000	8000	298137	8000	0.020

Table 3.7.1-4 (Sheet 4 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Soft-to-Medium Soil								
0.0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.11	3438	1000	3222	971	0.023
10.0	5.0	2	0.11	4923	1197	4355	1129	0.031
15.0	5.0	3	0.11	6960	1423	5987	1324	0.035
20.0	5.0	4	0.11	7487	1476	6161	1343	0.040
25.0	5.0	5	0.11	7824	1509	6699	1400	0.031
30.0	5.0	6	0.11	8211	1546	6891	1420	0.033
33.5	3.5	7	0.11	8938	1613	7872	1518	0.026
39.5	6.0	8	0.11	9520	1664	8317	1560	0.027
45.0	5.5	9	0.11	10195	1722	8834	1608	0.028
50.0	5.0	10	0.11	10864	1778	9347	1654	0.029
60.0	10.0	11	0.11	11618	1838	9818	1695	0.031
80.0	20.0	12	0.11	12357	1896	10031	1714	0.036
100.0	20.0	13	0.11	13077	1950	10201	1728	0.040
120.0	20.0	14	0.11	13774	2002	10512	1754	0.043
Bedrock	–		0.15	300000	8000	298137	8000	0.020

Table 3.7.1-4 (Sheet 5 of 5)
Strain Compatible Soil Properties

Depth to Bottom of Layer (ft)	Thickness of Layer (ft)	Layer Number	Total Unit Weight (kcf)	Initial G (ksf)	Initial Vs (fps)	Final G (ksf)	Final Vs (fps)	Damping
Soft Soil								
0.0	–	–	–	–	–	–	–	–
5.0	5.0	1	0.11	3438	1000	3222	971	0.023
10.0	5.0	2	0.11	3633	1028	3042	944	0.038
15.0	5.0	3	0.11	3865	1060	2974	933	0.047
20.0	5.0	4	0.11	3921	1068	2752	898	0.059
25.0	5.0	5	0.11	3955	1073	2922	925	0.049
30.0	5.0	6	0.11	3994	1078	2762	899	0.056
33.5	3.5	7	0.11	4065	1088	3022	941	0.046
39.5	6.0	8	0.11	4121	1095	2958	931	0.049
45.0	5.5	9	0.11	4183	1103	2896	921	0.053
50.0	5.0	10	0.11	4244	1111	2851	914	0.056
60.0	10.0	11	0.11	4310	1120	2774	901	0.062
80.0	20.0	12	0.11	4374	1128	2668	884	0.068
100.0	20.0	13	0.11	4434	1136	2691	888	0.069
120.0	20.0	14	0.11	4492	1143	2718	892	0.069
Bedrock	–	–	0.15	300000	8000	298137	8000	0.020

Tables 3.7.2-1–3.7.2-16 Not Used

Table 3.7.3-1 (Sheet 1 of 3)
Seismic Category I Equipment Outside Containment by Room Number

Room No.	Room Name	Equipment Description
12101	Division A battery room	Batteries
12102	Division C battery room 1	Batteries
12103	Spare battery room	Spare batteries
12104	Division B battery room 1	Batteries
12105	Division D battery room	Batteries
12113	Spare battery charger room	
12162	RNS pump room A	RNS pressure boundary
12163	RNS pump room B	RNS pressure boundary
12201	Division A dc equipment room	dc equipment
12202	Division C battery room 2	Batteries
12203	Division C dc equipment room	dc equipment
12204	Division B battery room 2	Batteries
12205	Division D dc equipment room	dc equipment
12207	Division B dc equipment room	dc equipment
12211	Corridor	Divisional cables
12212	Division B RCP trip switchgear room	RCP trip switchgear
12244	Lower annulus valve area	CVS/WLS containment isolation valves
12251	Demineralizer/filter access area	CVS/DWS isolation valves
12254	SFS penetration room	SFS containment isolation valve
12256	Containment isolation valve room	RNS containment isolation valves
12259	Pipe chase	RNS piping
12262	Piping/Valve room	RNS pressure boundary, SFS piping
12265	Waste monitor tank room C	SFS piping
12269	Pipe chase	RNS pressure boundary
12300	Corridor	Divisional cable
12301	Division A I&C room	Divisional I&C
12302	Division C I&C room	Divisional I&C

Table 3.7.3-1 (Sheet 2 of 3)
Seismic Category I Equipment Outside Containment by Room Number

Room No.	Room Name	Equipment Description
12303	Remote shutdown room	Divisional cabling
12304	Division B I&C/penetration room	Divisional I&C/electrical penetrations
12305	Division D I&C/penetration room	Divisional I&C/electrical penetrations
12306	Valve/piping penetration room	CCS/CVS/DWS/FPS/SGS containment isolation valves
12311	Corridor	Divisional cabling
12312	Division C RCP trip switchgear room	RCP trip switchgear
12313	Division C I&C/penetration room	Divisional I&C/electrical penetrations
12321	Non-1E equipment/penetration room	Divisional cabling/electrical penetrations
12341	Middle annulus	Class 1E electrical penetrations Various mechanical piping penetrations
12343	Division C middle annulus penetration room	Class 1E electrical penetrations
12344	Division B middle annulus penetration room	Class 1E electrical penetrations
12345	Division D middle annulus penetration room	Class 1E electrical penetrations
12351	Maintenance floor staging area	Divisional cabling (ceiling)
12352	Personnel hatch	Personnel airlock (interlocks)
12354	Middle annulus access room	PSS/SFS containment isolation valves
12362	RNS HX room	RNS pressure boundary
12365	Waste monitor tank room B	SFS piping
12400	Control room vestibule	Control room access
12401	Main control room	Dedicated safety panel VBS HVAC dampers VES isolation valves Lighting circuits Mounting for lighting fixtures
12404	Lower MSIV compartment B	SGS containment isolation valves, instrumentation and controls
12405	Lower VBS B and D equipment room	VWS/PXS/CAS containment isolation valves
12406	Lower MSIV compartment A	SGS containment isolation valves, instrumentation and controls
12412	Electrical penetration room Division A	Divisional electrical penetrations

Table 3.7.3-1 (Sheet 3 of 3)
Seismic Category I Equipment Outside Containment by Room Number

Room No.	Room Name	Equipment Description
12421	Non 1E equipment/penetration room	Divisional cabling/electrical penetrations
12422	Reactor trip switchgear II	Reactor trip switchgear
12423	Reactor trip switchgear I	Reactor trip switchgear
12452	VFS penetration room	VFS containment isolation valves, divisional cabling
12454	VFS/SFS/PSS penetration room	SFS/PSS/VFS containment isolation valves, RNS pressure boundary
12462	Cask washdown pit	SFS piping
12504	Upper MSIV compartment B	SGS CIVs, instrumentation and controls
12506	Upper MSIV compartment A	SGS CIVs, instrumentation and controls
12541	Upper annulus	PCS piping and cabling PCS air baffle
12553	Personnel access area	Personnel airlock (interlocks)
12555	Operating deck staging area/VES air storage	VES high pressure air bottles
12651	VAS Equipment Room	VFS containment isolation valves
12562	Fuel handling area	Spent fuel storage racks
12701	PCS valve room	PCS isolation valves/instrumentation
12703	PCS water storage tank	PCS piping, level and temperature instrumentation

Table 3.7.3-2
Equipment Classified as Sensitive Targets for
Seismically Analyzed Piping, HVAC Ducting, Cable Trays

Component	Discussion	Zone of Protection
Seismic Category I Valve No Class 1E Electrical Equipment Not pressure sensitive	These are manual valves. The actuator must be protected from impact.	Valve body and actuator area
Seismic Category I Valve Class 1E Electrical Equipment Pressure sensitive	These valves have sensitive Class 1E equipment (e.g., Position indicators, limit switches, motor operator) or solenoid valves.	One support (acting in direction of impact) on each side of valve
Seismic Category I Dampers	The actuator must be protected along with any Class 1E equipment.	Within one support (acting in direction of impact) on each side of HVAC
Monitors	This includes: neutron detectors, radiation monitors, resistance temperature detectors, speed sensors, thermocouples, and transmitters.	Monitors and associated wiring
Sensitive Electrical Equipment Housed in Cabinets, Panels or Boards	This includes: relays, contractors, breakers, and switchgear.	Cabinets, panels, and boards housing sensitive devices
Class 1E exposed cables and wiring	Cables and wiring which are not housed in cable trays or conduits must be protected.	Exposed cables and wiring
Device or Instrument Tubing	Any device or tubing that could be damaged resulting in the loss of the pressure boundary of a safety class line.	Device or tubing
Penetrations	Rigid penetrations are considered robust. Floating penetrations with bellows are considered sensitive.	Floating penetration and associated bellows

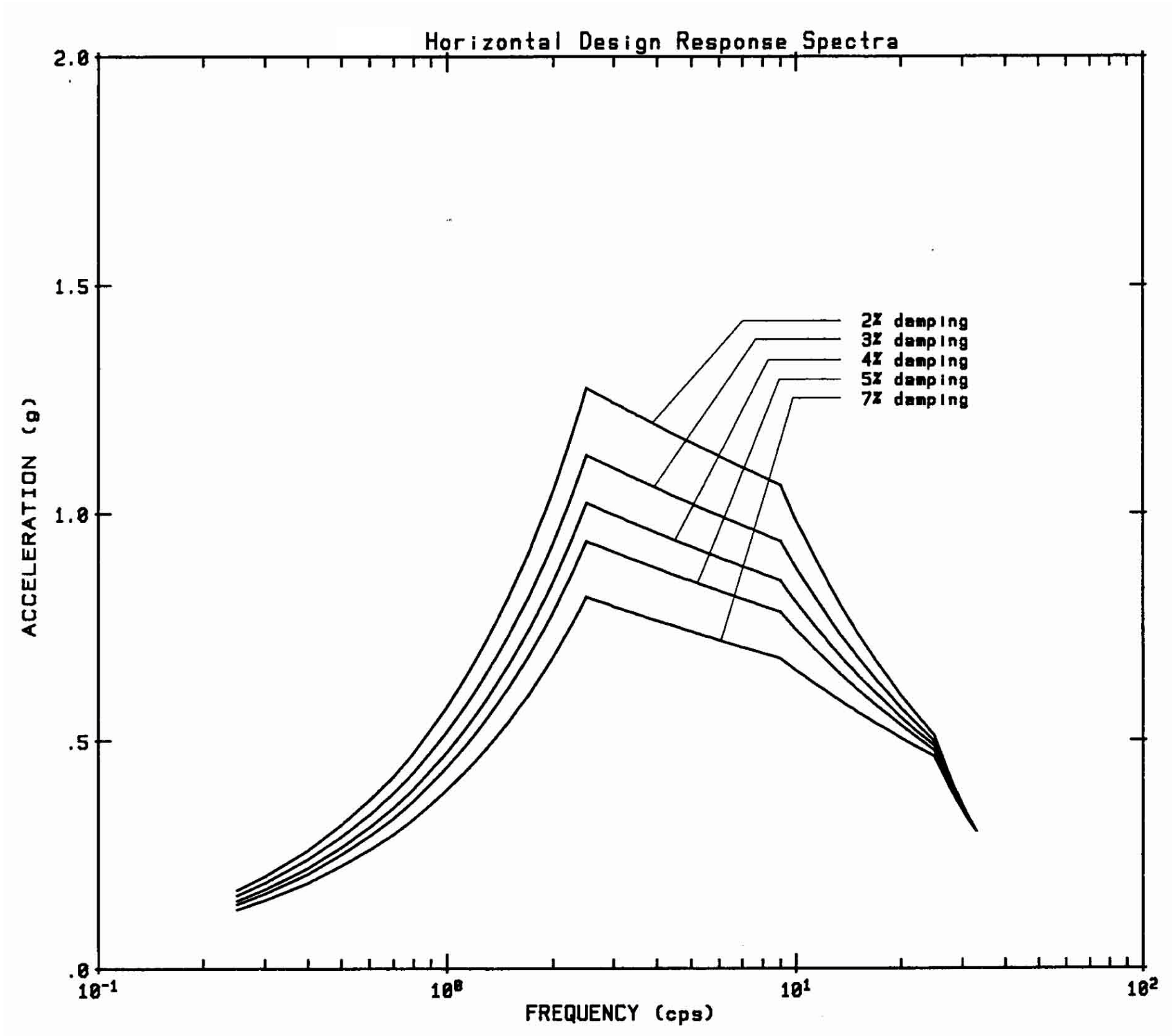


Figure 3.7.1-1
Horizontal Design Response Spectra
Safe Shutdown Earthquake

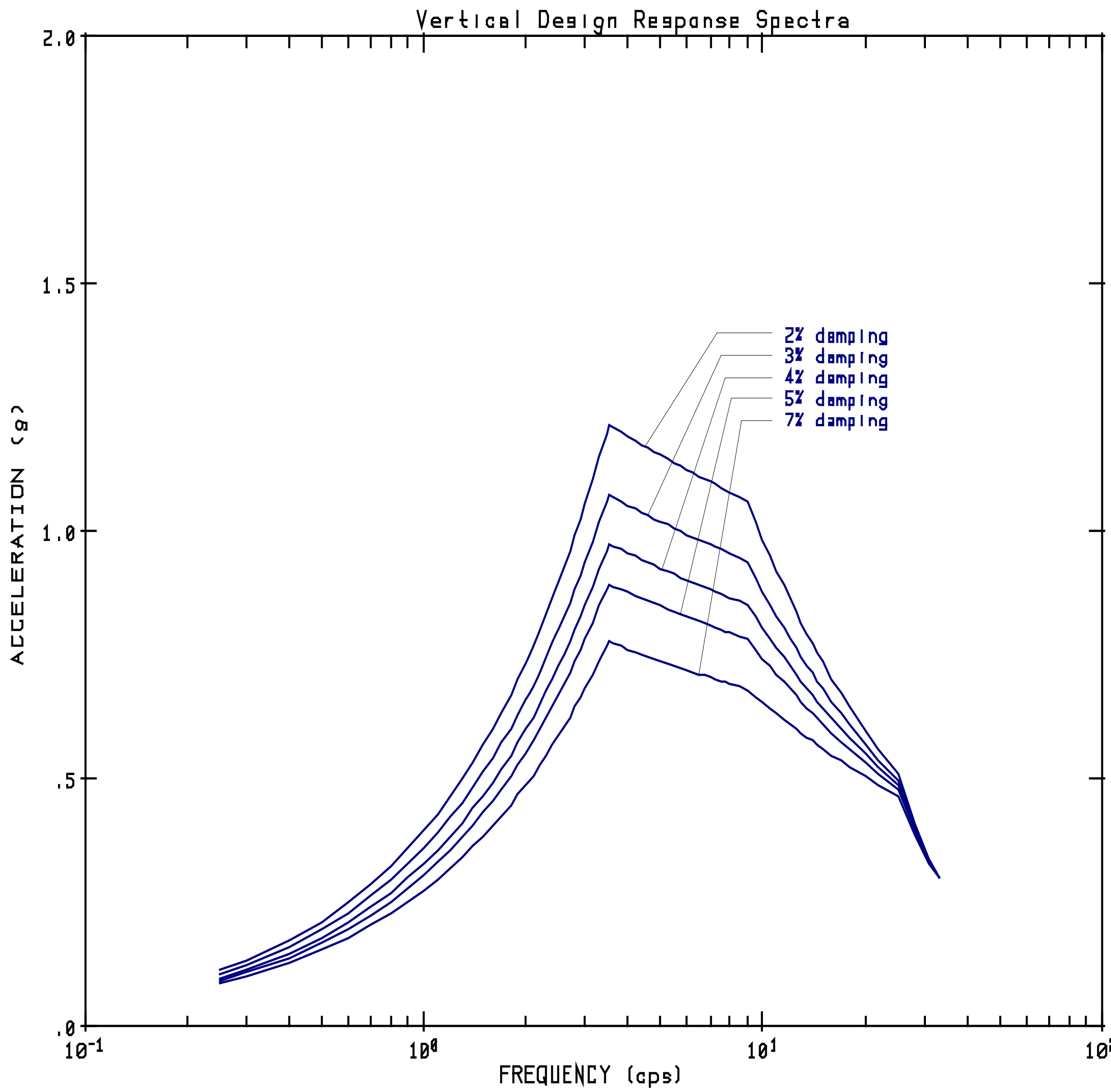


Figure 3.7.1-2
Vertical Design Response Spectra
Safe Shutdown Earthquake

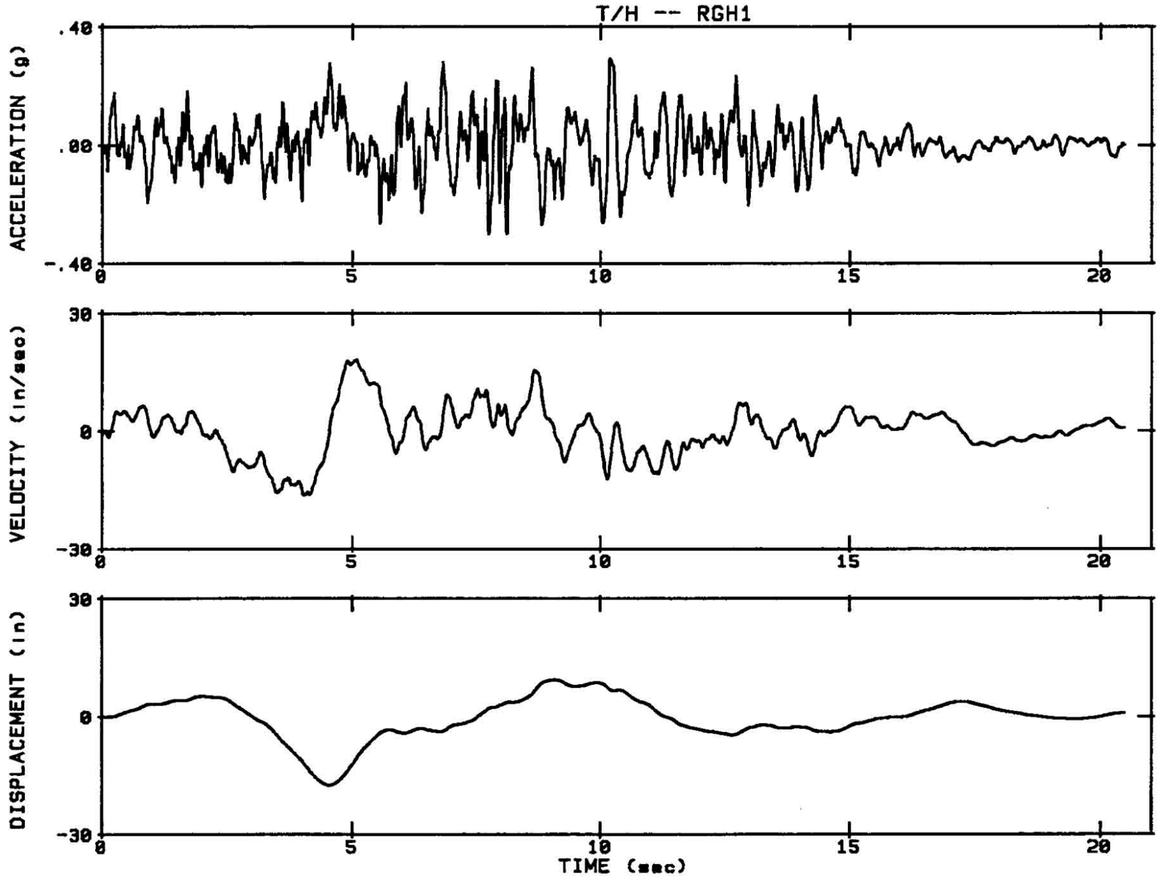


Figure 3.7.1-3
Design Horizontal Time History, "H1"
Acceleration, Velocity & Displacement Plots

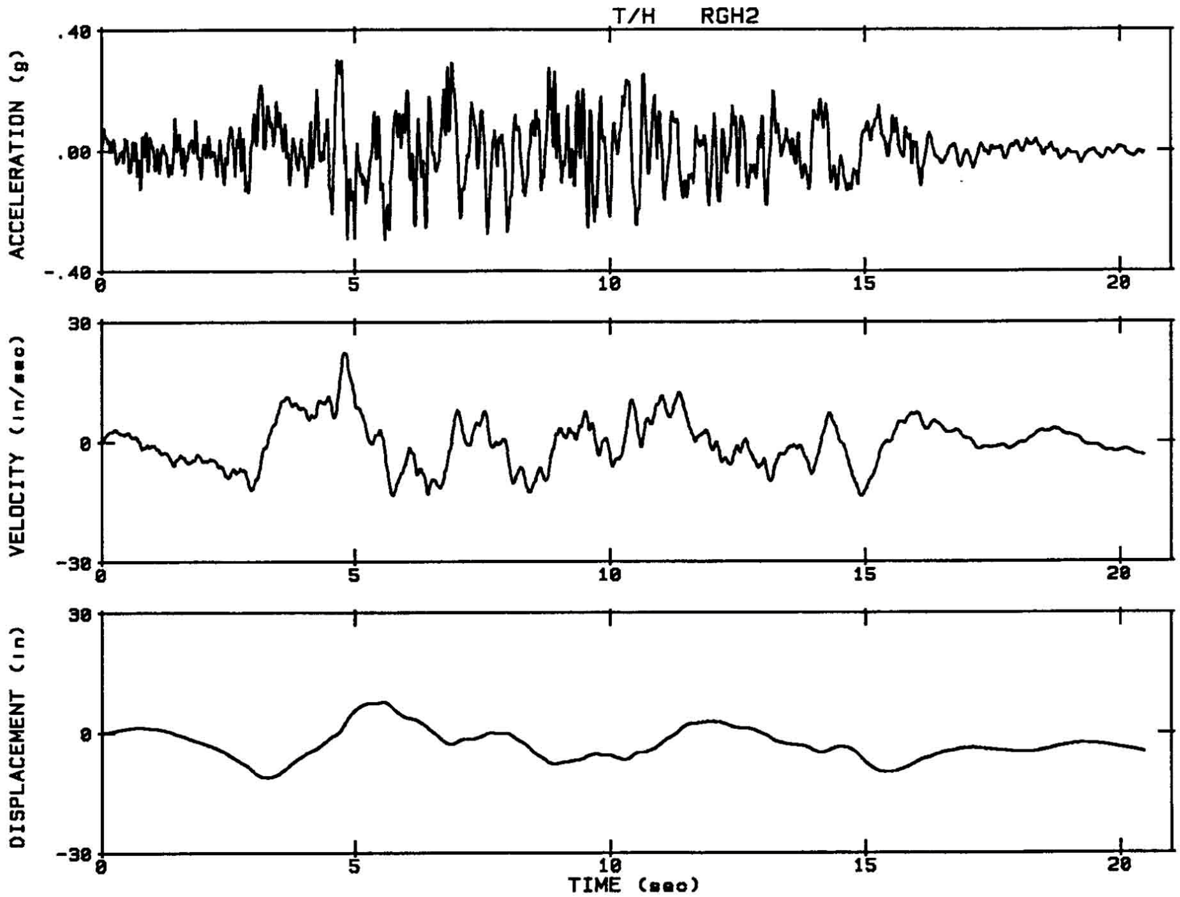


Figure 3.7.1-4
Design Horizontal Time History, "H2"
Acceleration, Velocity & Displacement Plots

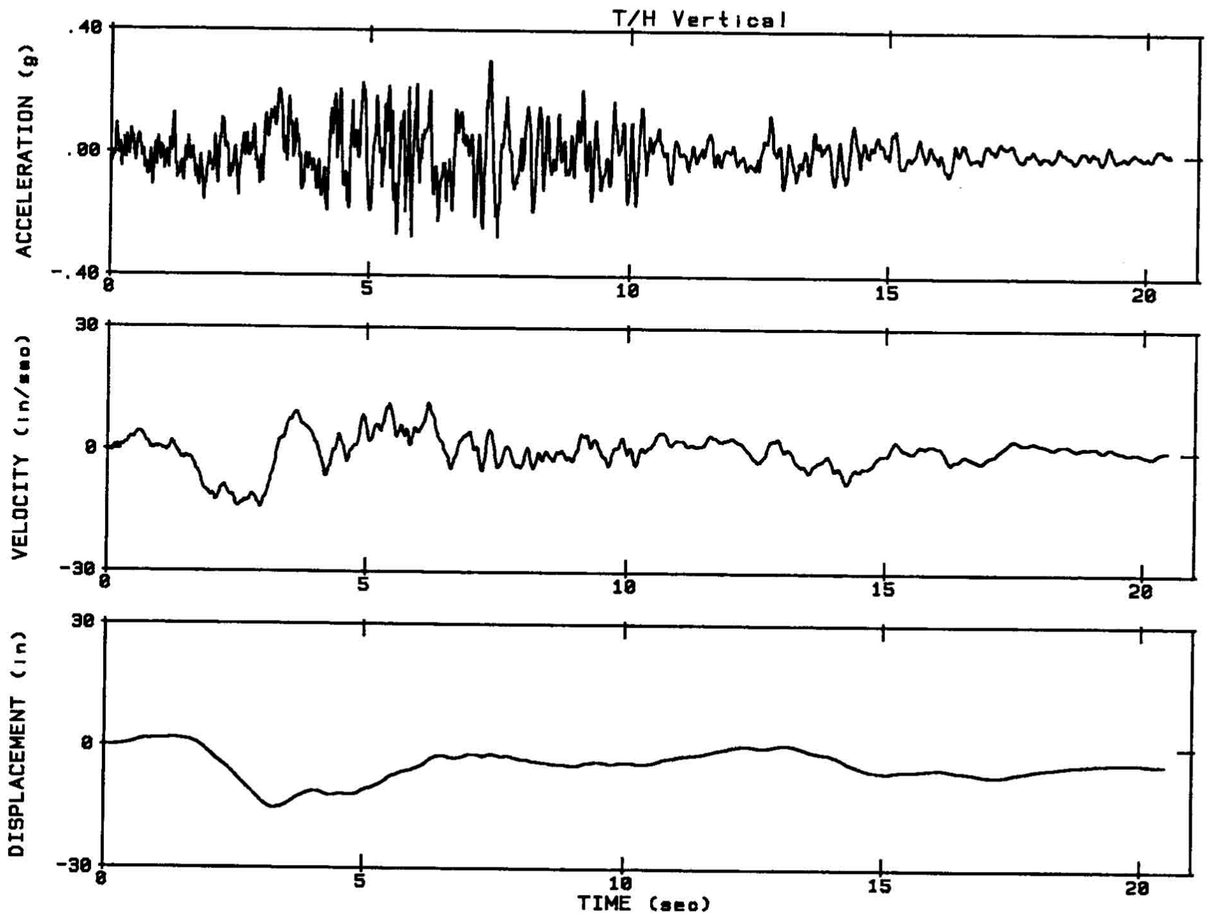


Figure 3.7.1-5
Design Vertical Time History
Acceleration, Velocity & Displacement Plots

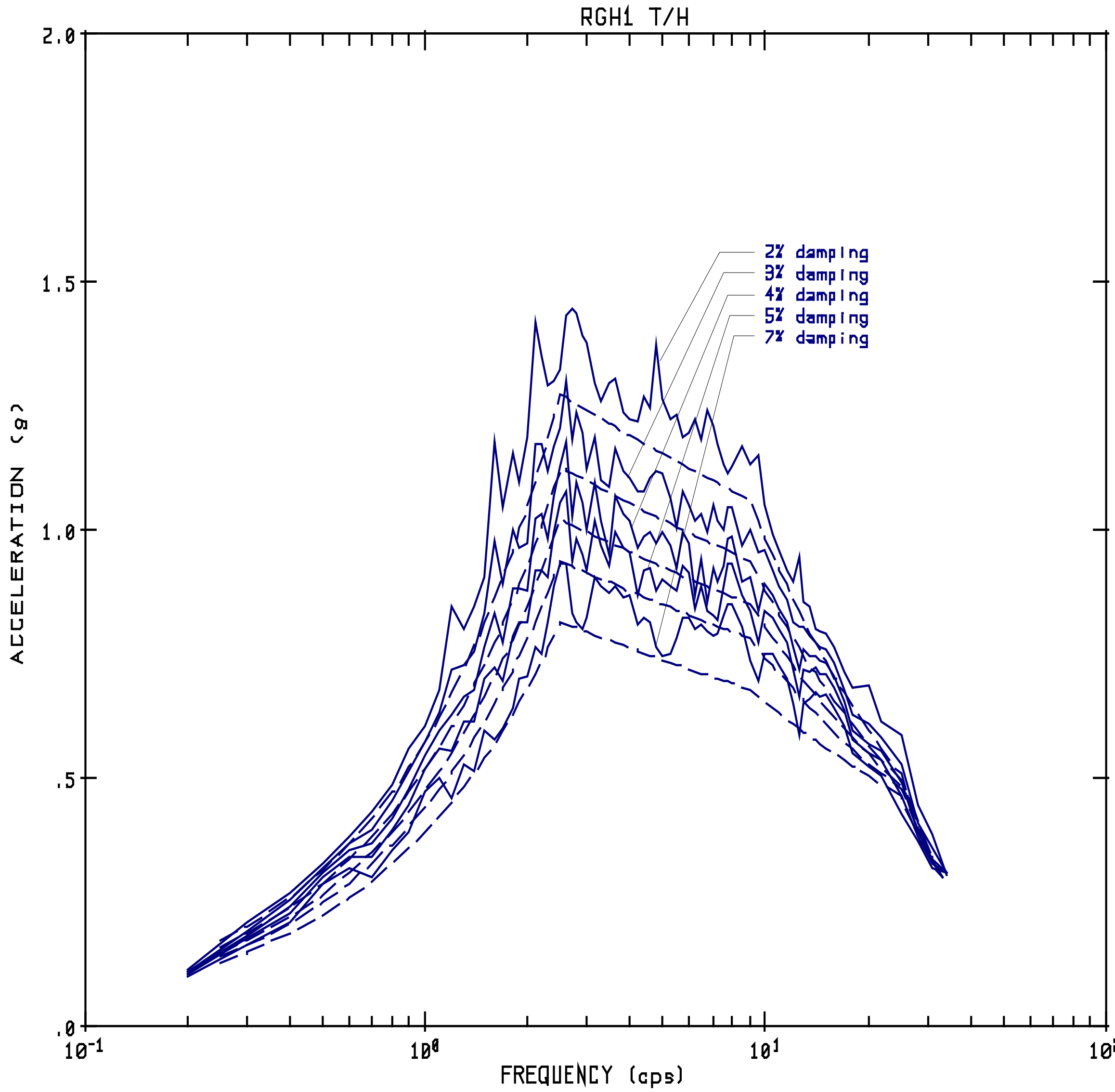


Figure 3.7.1-6
Acceleration Response Spectra of
Design Horizontal Time History, "H1"

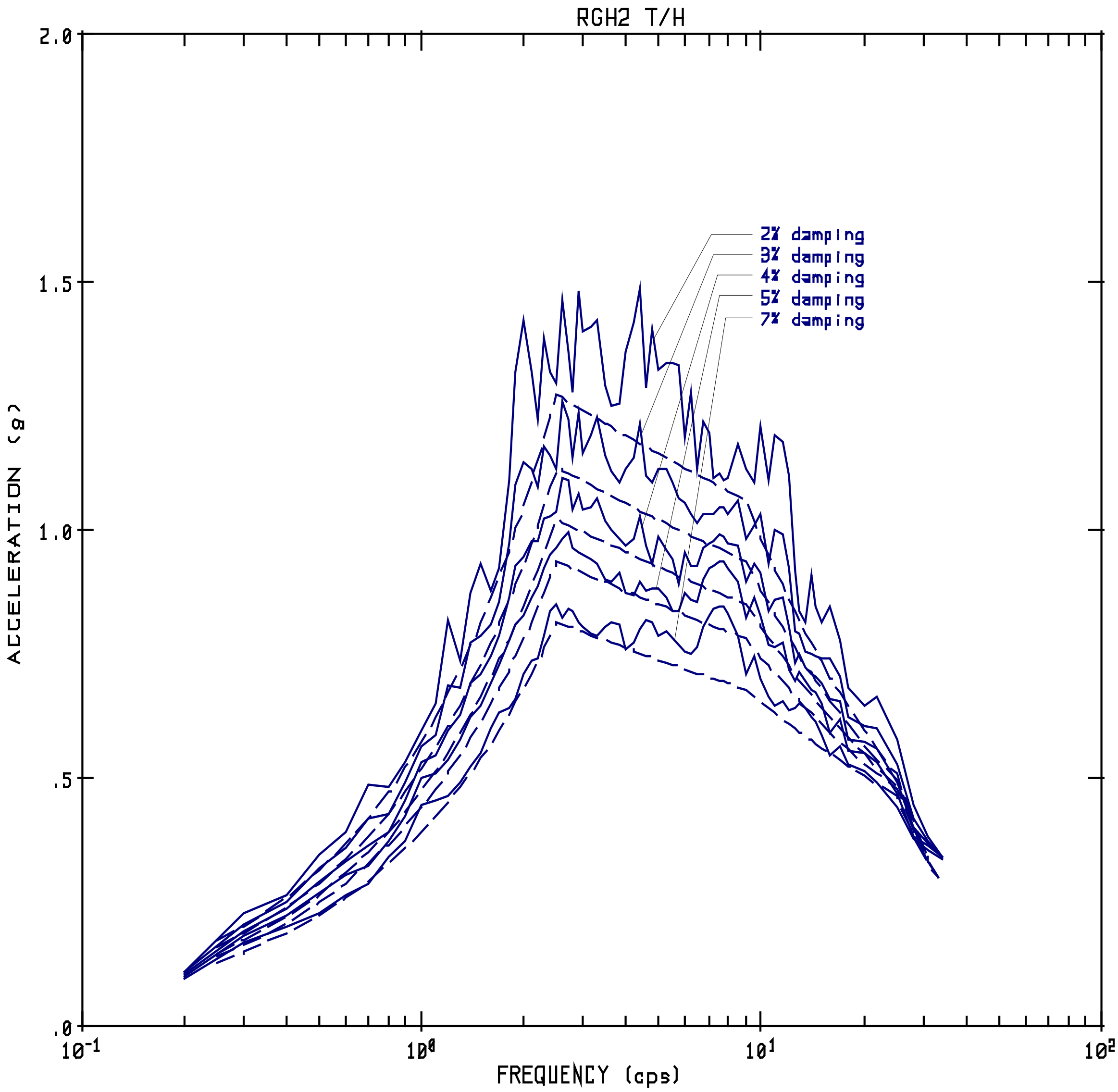


Figure 3.7.1-7
Acceleration Response Spectra of Design Horizontal Time History, "H2"

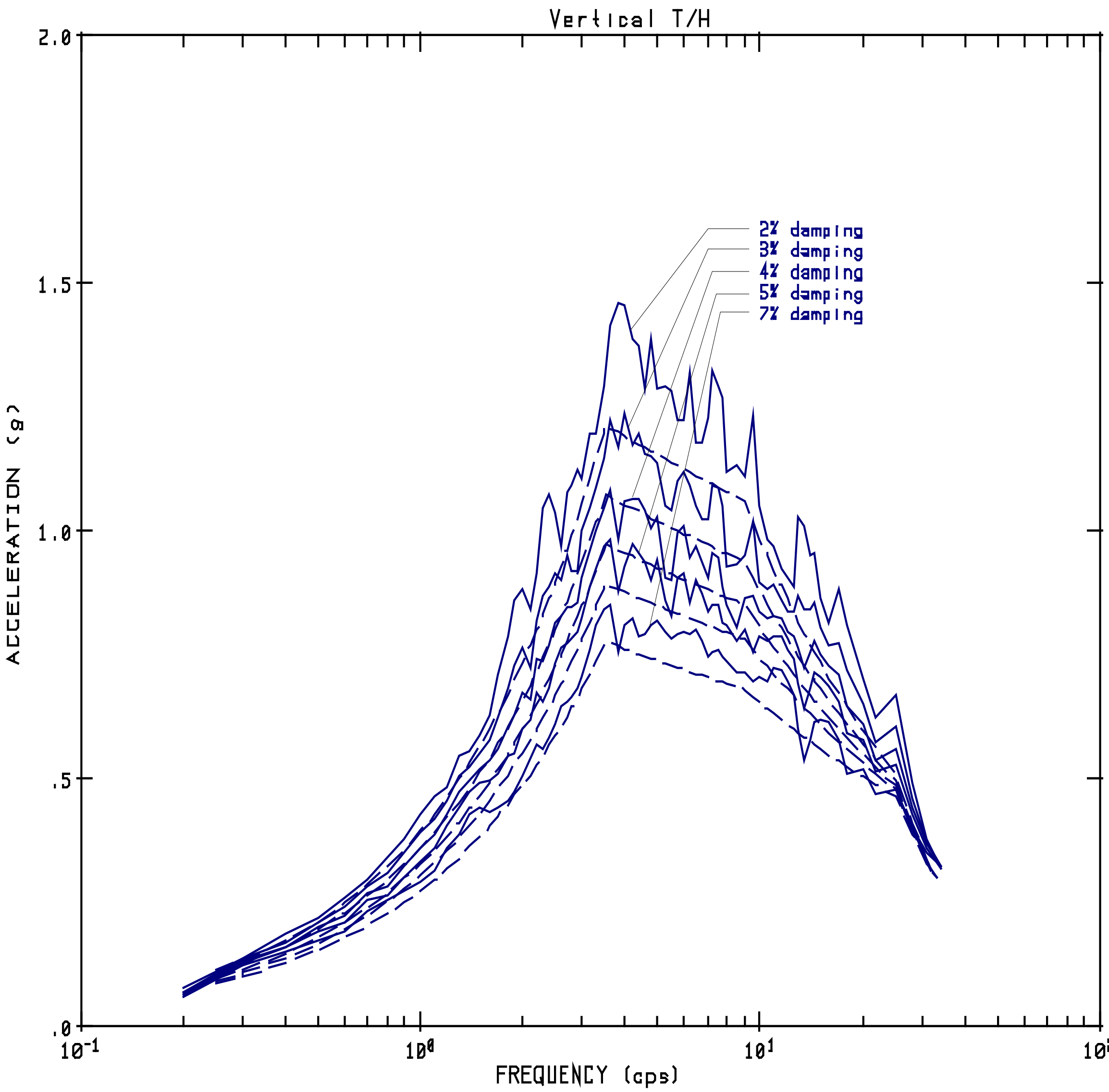


Figure 3.7.1-8
Acceleration Response Spectra of Design Vertical Time History

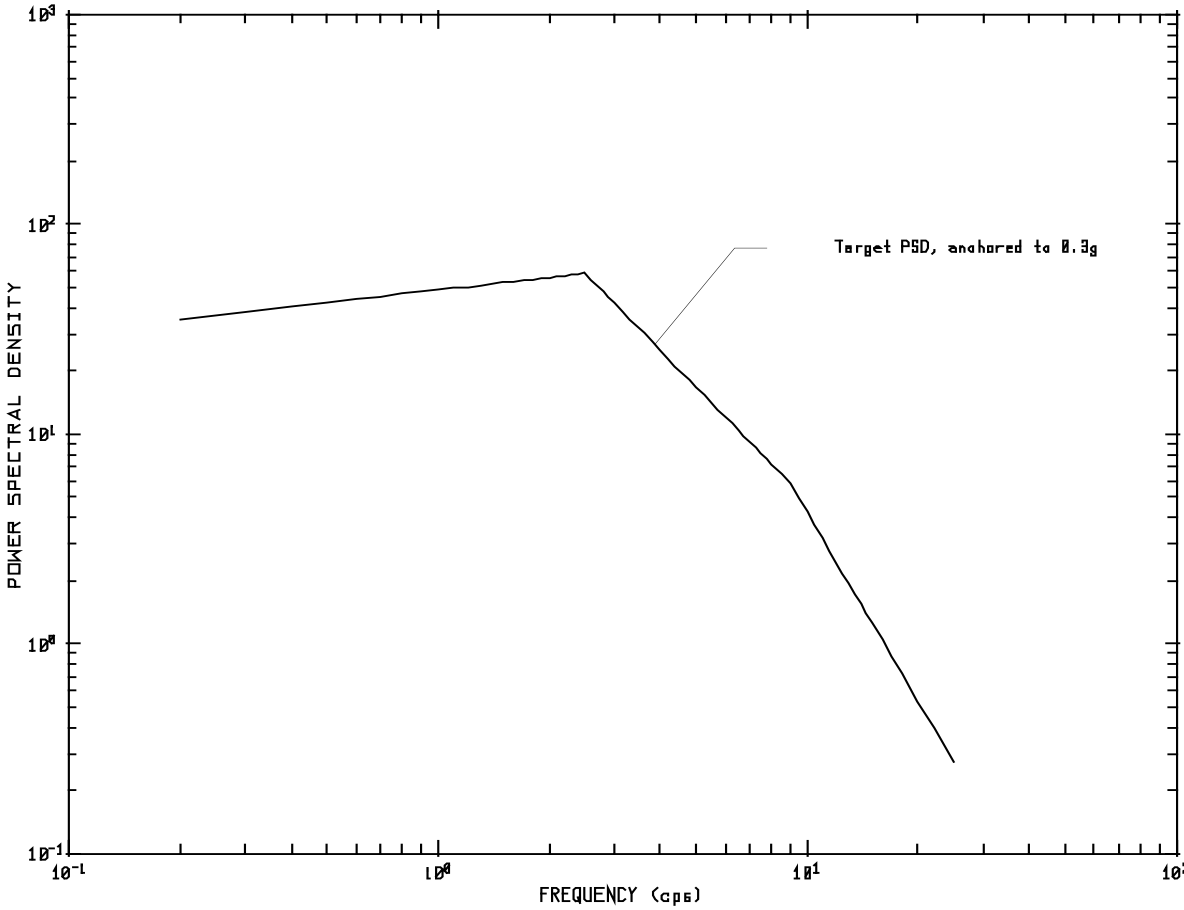


Figure 3.7.1-9
Minimum Power Spectral Density Curve
(Normalized to 0.3g)

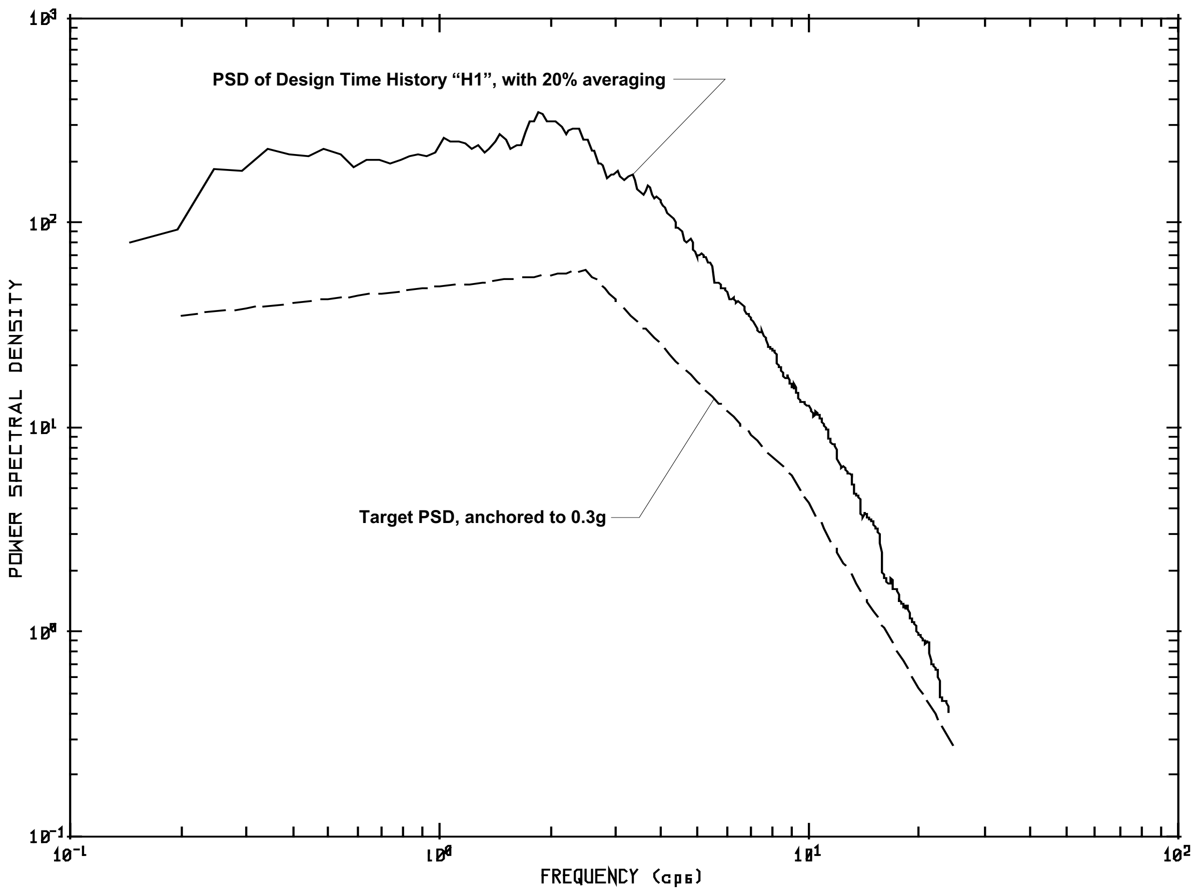


Figure 3.7.1-10
Power Spectral Density of
Design Horizontal Time History, "H1"

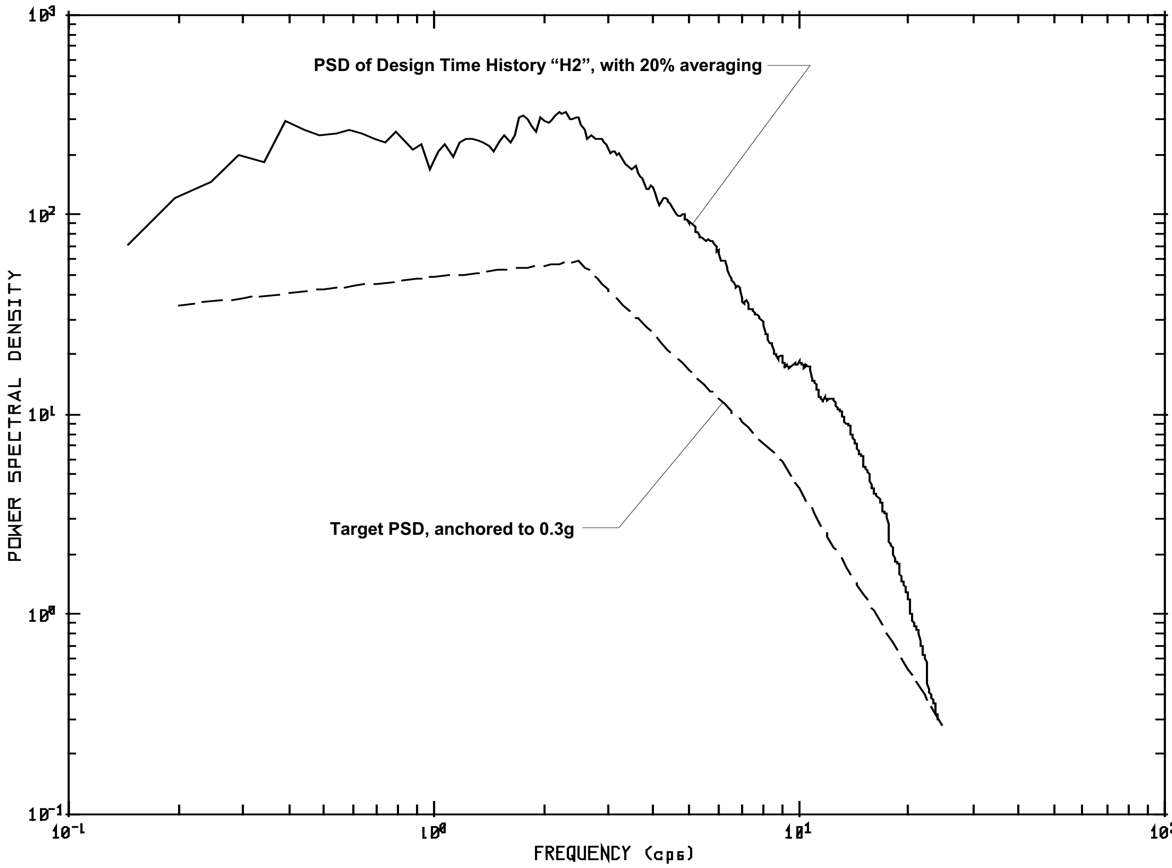


Figure 3.7.1-11
Power Spectral Density of
Design Horizontal Time History, "H2"

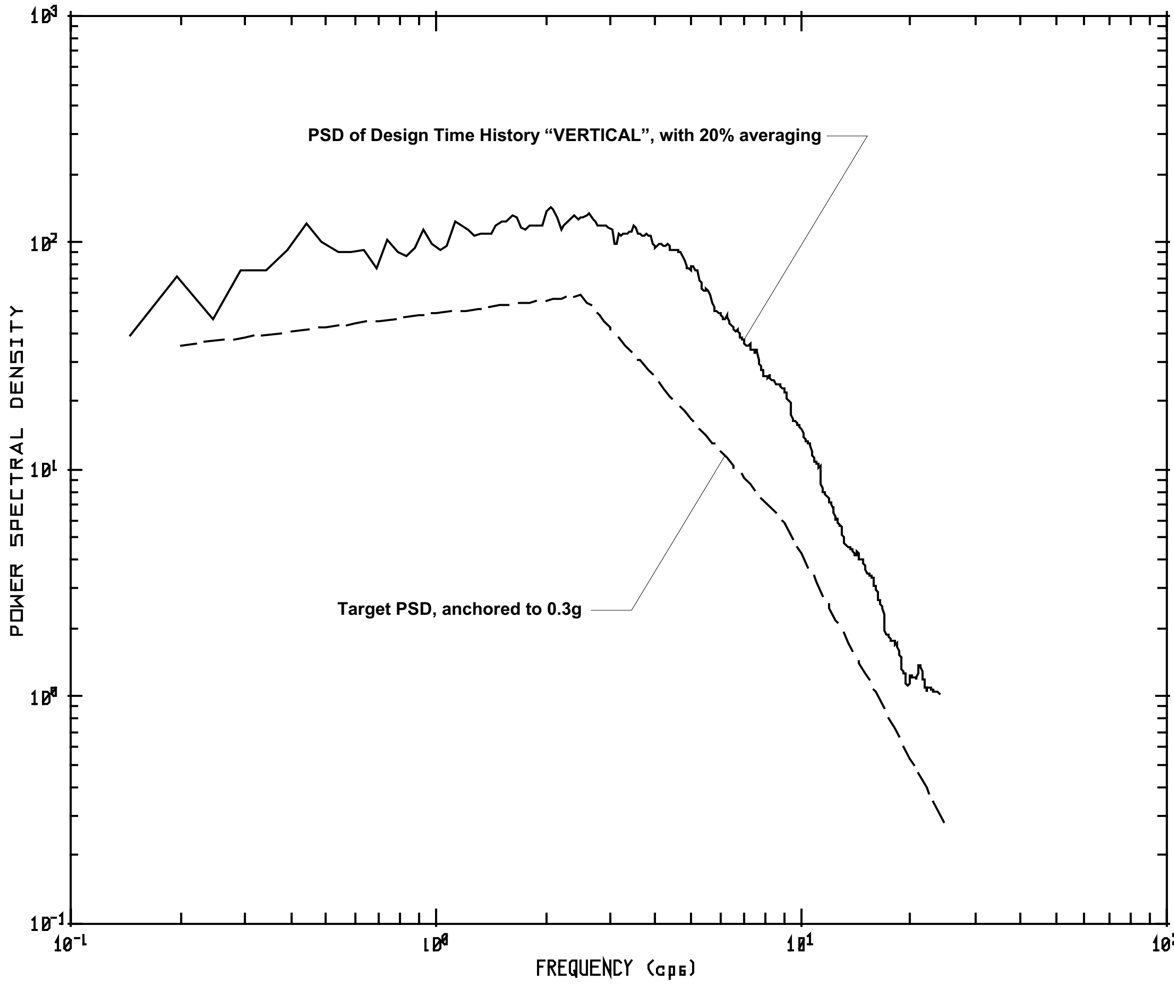


Figure 3.7.1-12
Power Spectral Density of Design Vertical Time History

Figure 3.7.1-13 Not Used

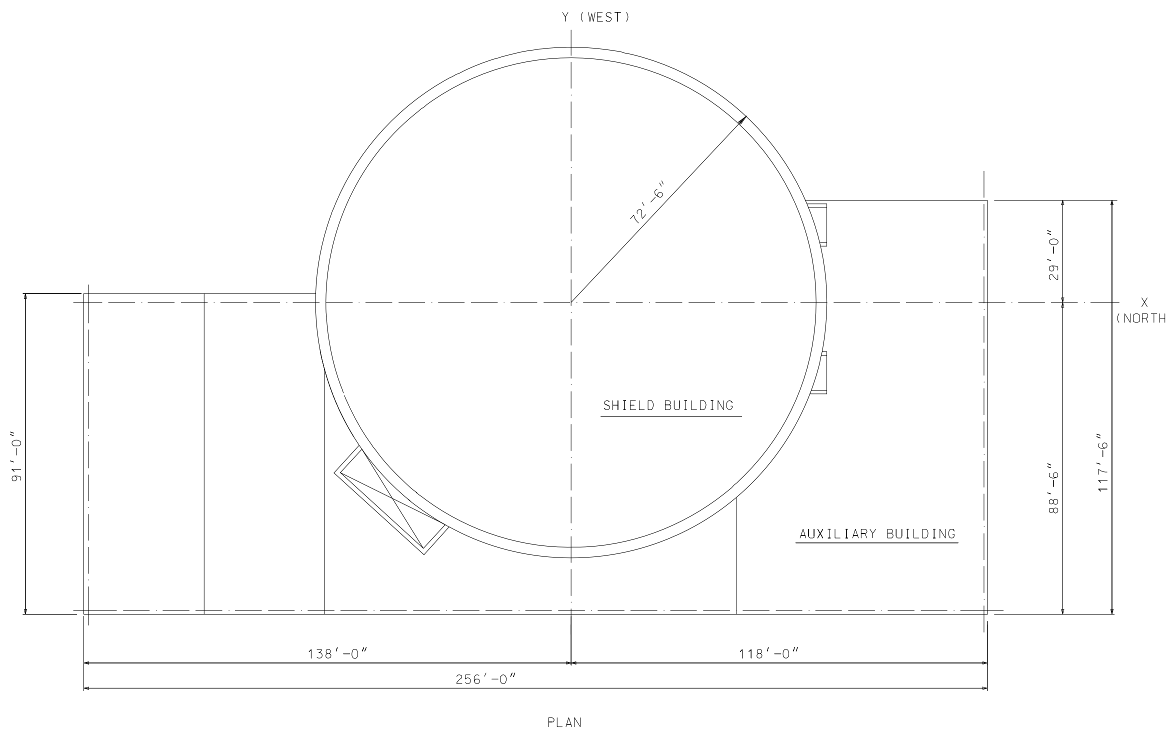
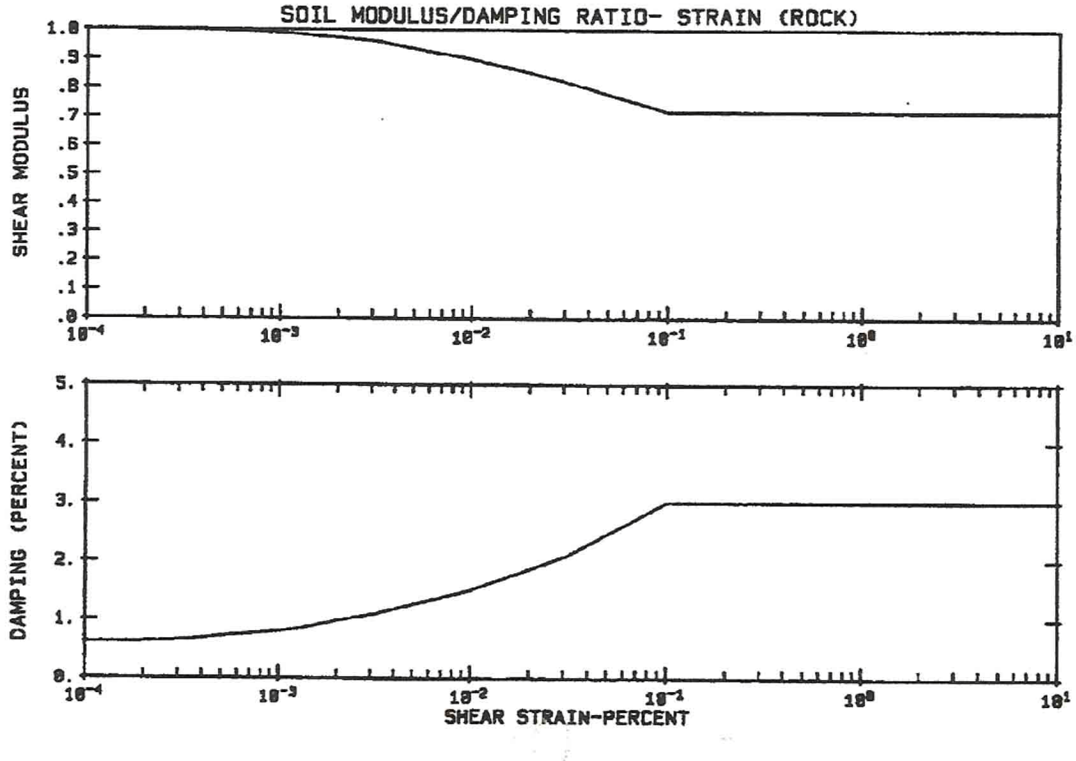


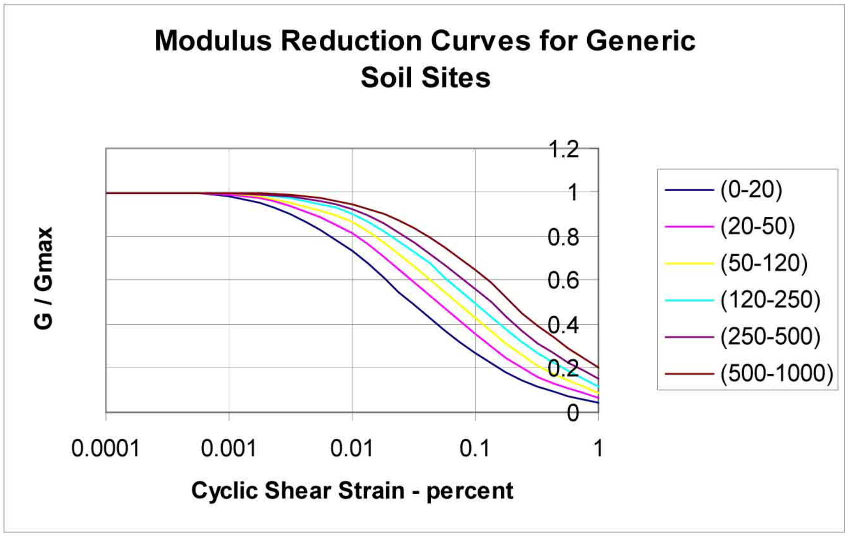
Figure 3.7.1-14
[*Nuclear Island Structures Dimensions*]*

*NRC Staff approval is required prior to implementing a change in this information.



Modulus Reduction Curves for Generic Soil Sites

Figure 3.7.1-15
Strain Dependent Properties of Rock Material (Ref. 37)



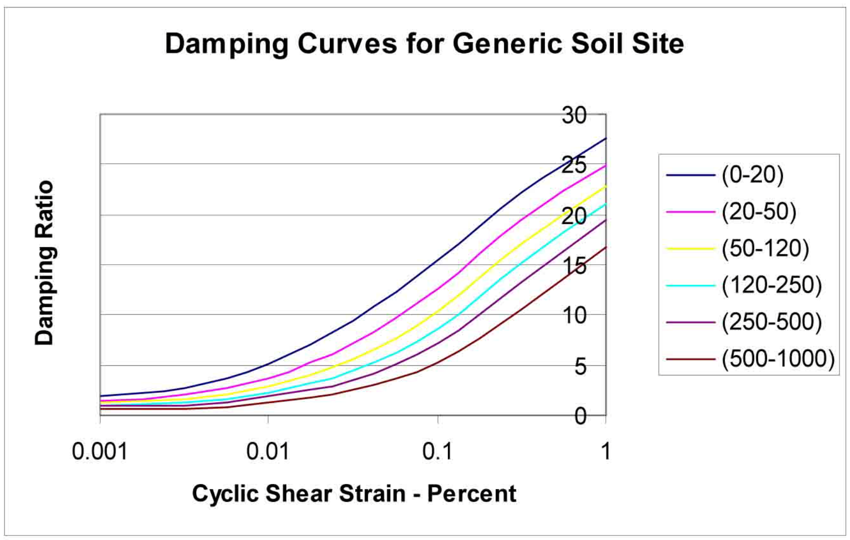
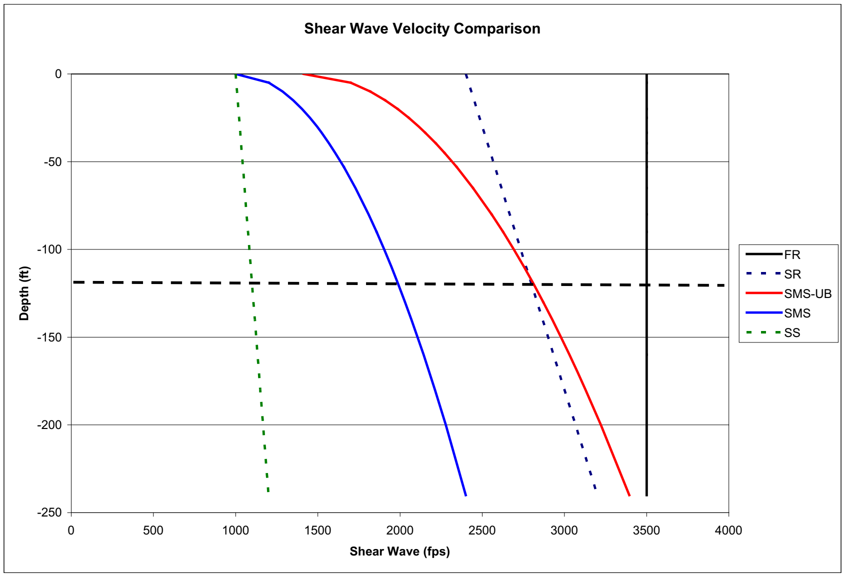
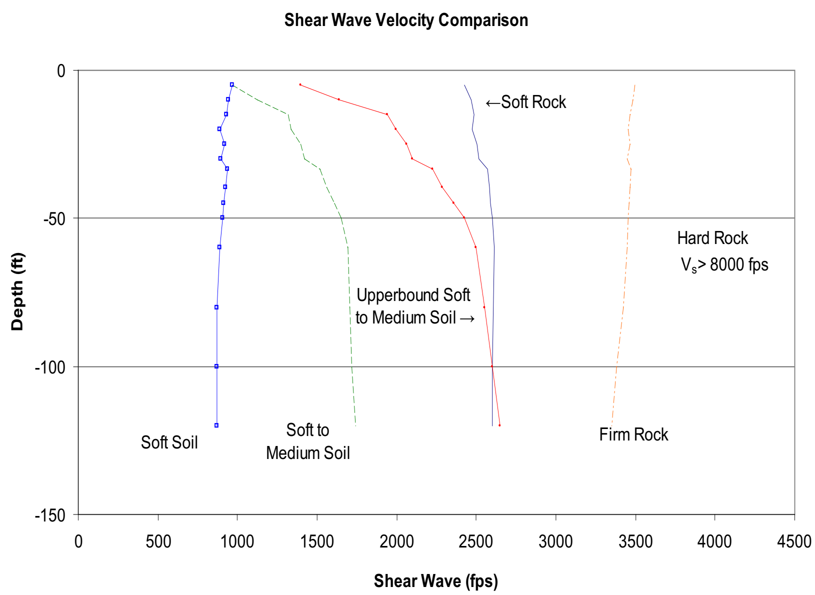


Figure 3.7.1-16
Strain Dependent Properties of Soil Material (Ref. 38)



Initial Properties



Strain-Iterated Shear Wave Velocity Profiles

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 3.7.1-17
Generic Soil Profiles

Figures 3.7.2-1–3.7.2-11   Not Used

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 1 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 66'-6"*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 2 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 82'-6"*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 3 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 100'-0" & 107'-2"*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 4 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 117'-6"*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 5 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 135'-3"*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 6 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 153'-0" & 160'-6"*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 7 of 12)
[*Nuclear Island Key Structural Dimensions Plan at El. 160'-6", 180'-0", & 329'-0"*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 8 of 12)
[*Nuclear Island Key Structural Dimensions Section A - A*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 9 of 12)
[*Nuclear Island Key Structural Dimensions Section B - B*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 10 of 12)
[*Nuclear Island Key Structural Dimensions Sections C - C and H - H*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 11 of 12)
[*Nuclear Island Key Structural Dimensions Section G - G*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-12 (Sheet 12 of 12)
[*Nuclear Island Key Structural Dimensions Section J - J*]*

*NRC Staff approval is required prior to implementing a change in this information.

Figure 3.7.2-13 Not Used

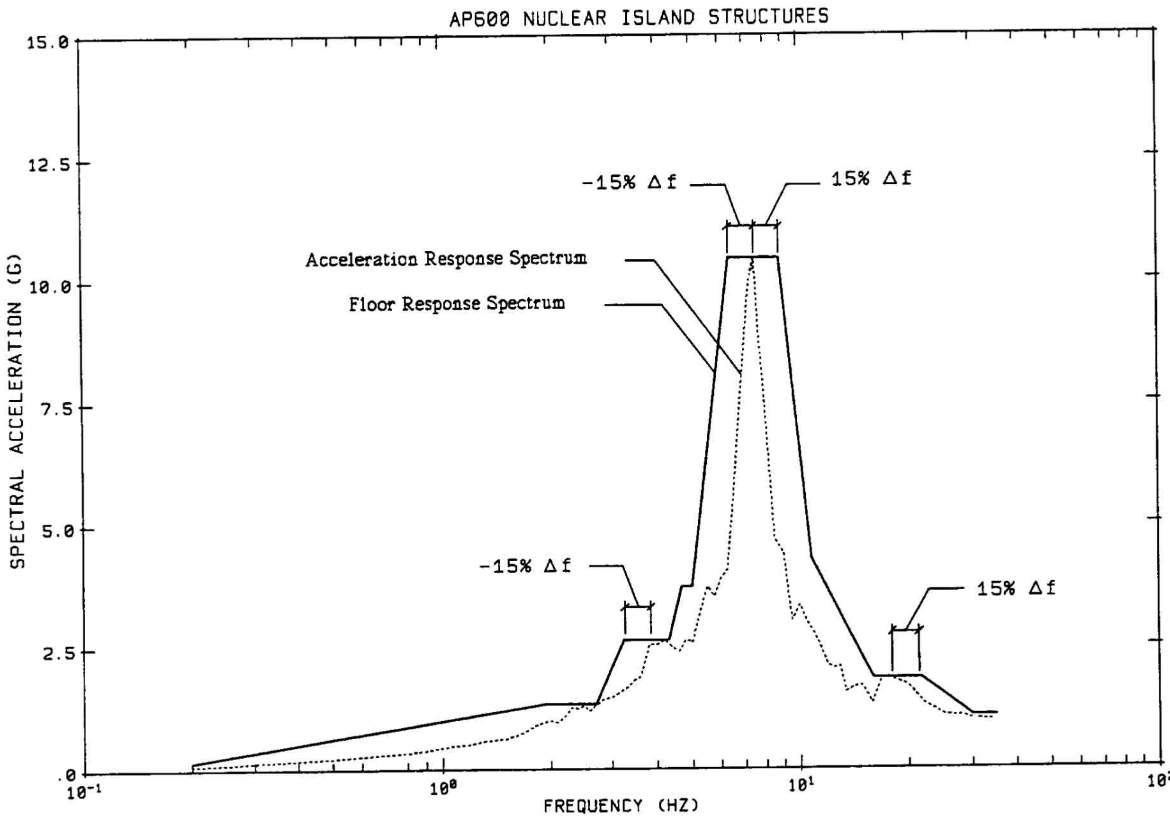


Figure 3.7.2-14
Typical Design Floor Response Spectrum

Figures 3.7.2-15–3.7.2-18 Not Used

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 1 of 10)
Annex Building Key Structural Dimensions
Plan at Elevation 100′-0″

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 2 of 10)
Annex Building Key Structural Dimensions
Plan at Elevation 107′-2″ and 117′-6″

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 3 of 10)
Annex Building Key Structural Dimensions
Plan at Elevation 135′-3″

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 4 of 10)
Annex Building Key Structural Dimensions
Plan at Elevation 158′-0″ and 150′-3″

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 5 of 10)
Annex Building Key Structural Dimensions
Roof Plan at Elevation 154′-0″ and 181′-11 3/4″

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 6 of 10)
Annex Building Key Structural Dimensions
Section A - A

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 7 of 10)
Annex Building Key Structural Dimensions
Section B - B

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 8 of 10)
Annex Building Key Structural Dimensions
Section C - C

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 9 of 10)
Annex Building Key Structural Dimensions
Sections D - D, E - E, & F - F

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.7.2-19 (Sheet 10 of 10)
Annex Building Key Structural Dimensions
Sections G - G, H - H, & J - J

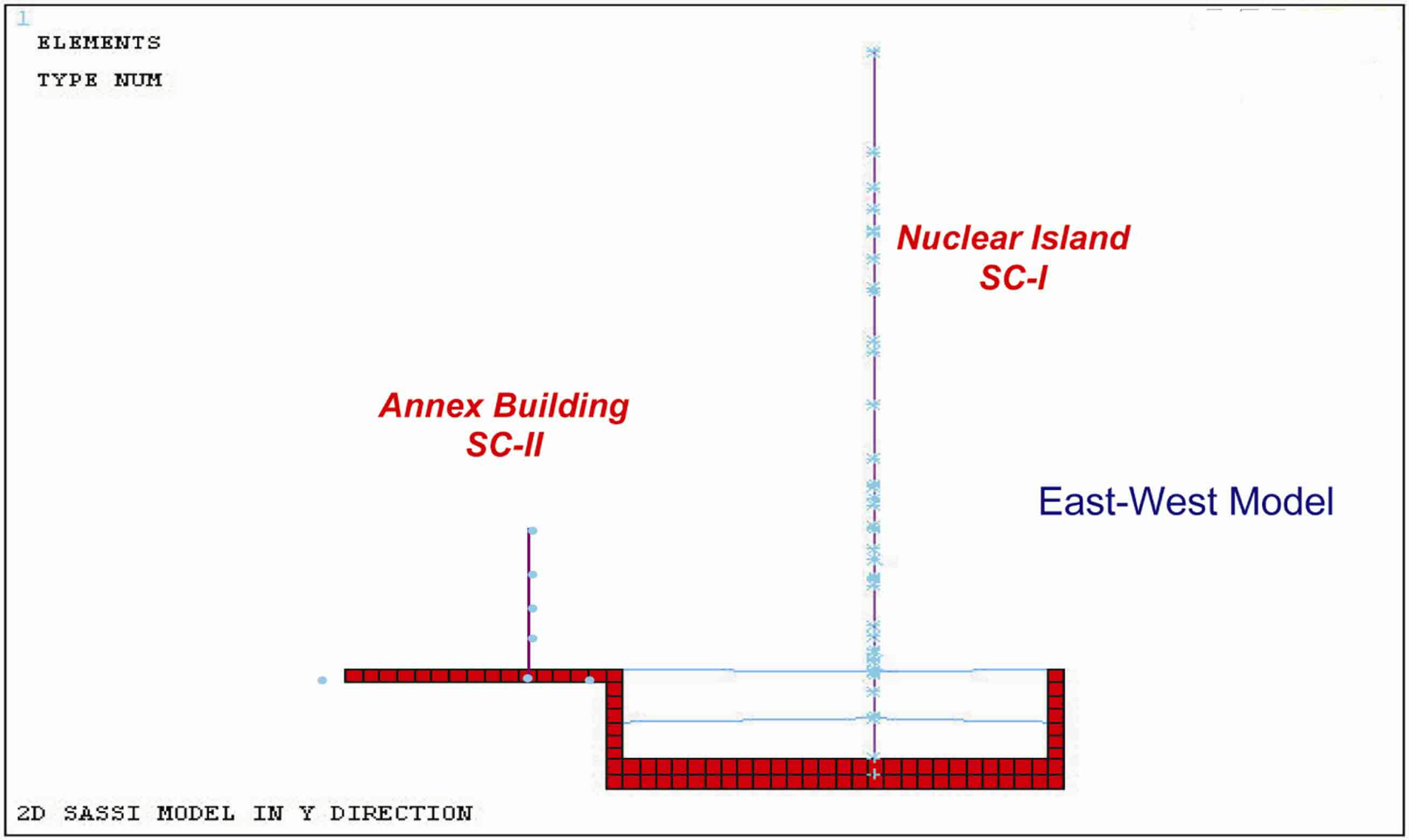


Figure 3.7.2-20
East-West 2D SASSI Model with Adjacent Buildings

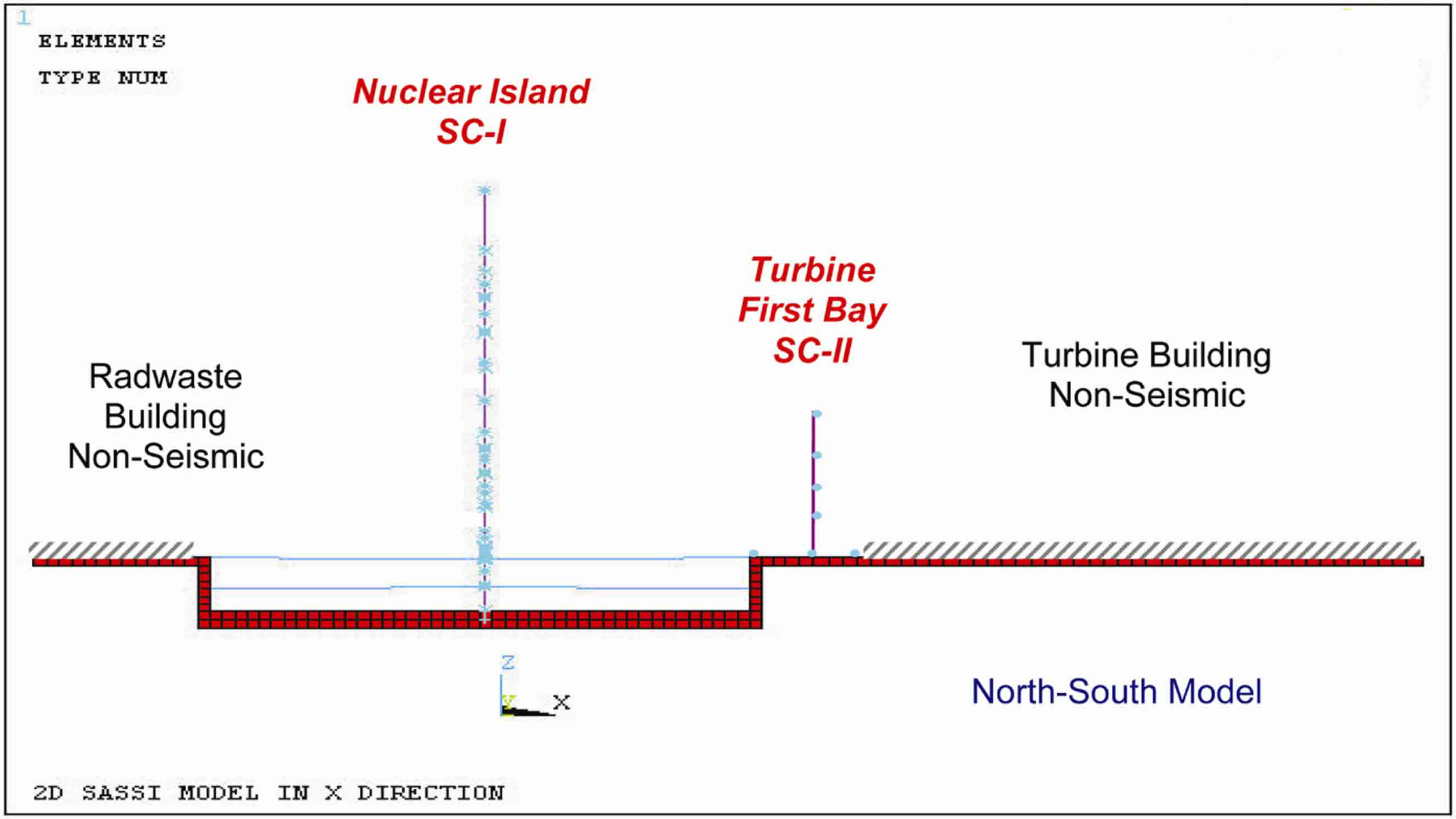


Figure 3.7.2-21
2D North-South SASSI Model with Adjacent Buildings

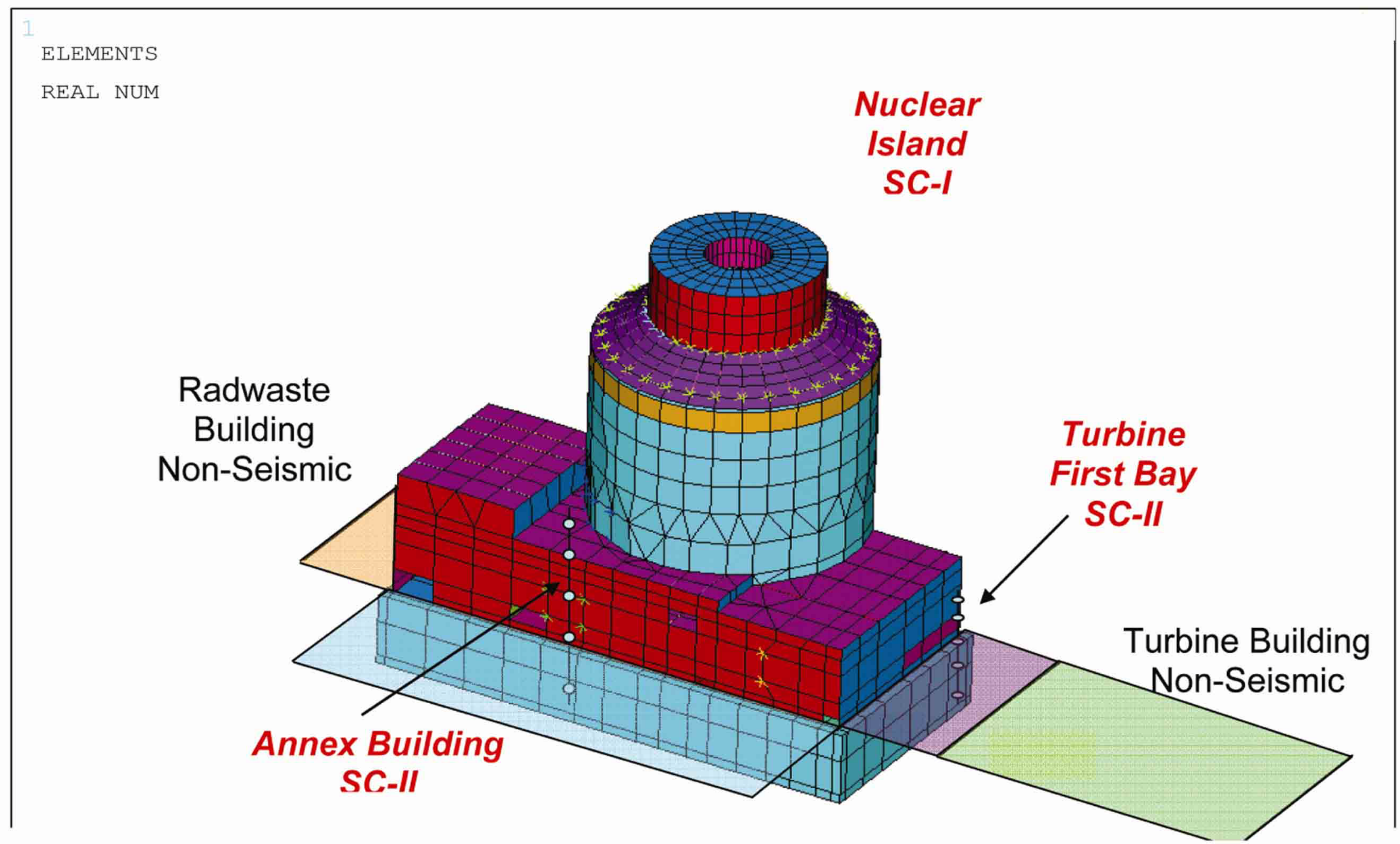


Figure 3.7.2-22
3D SASSI Model with Adjacent Buildings

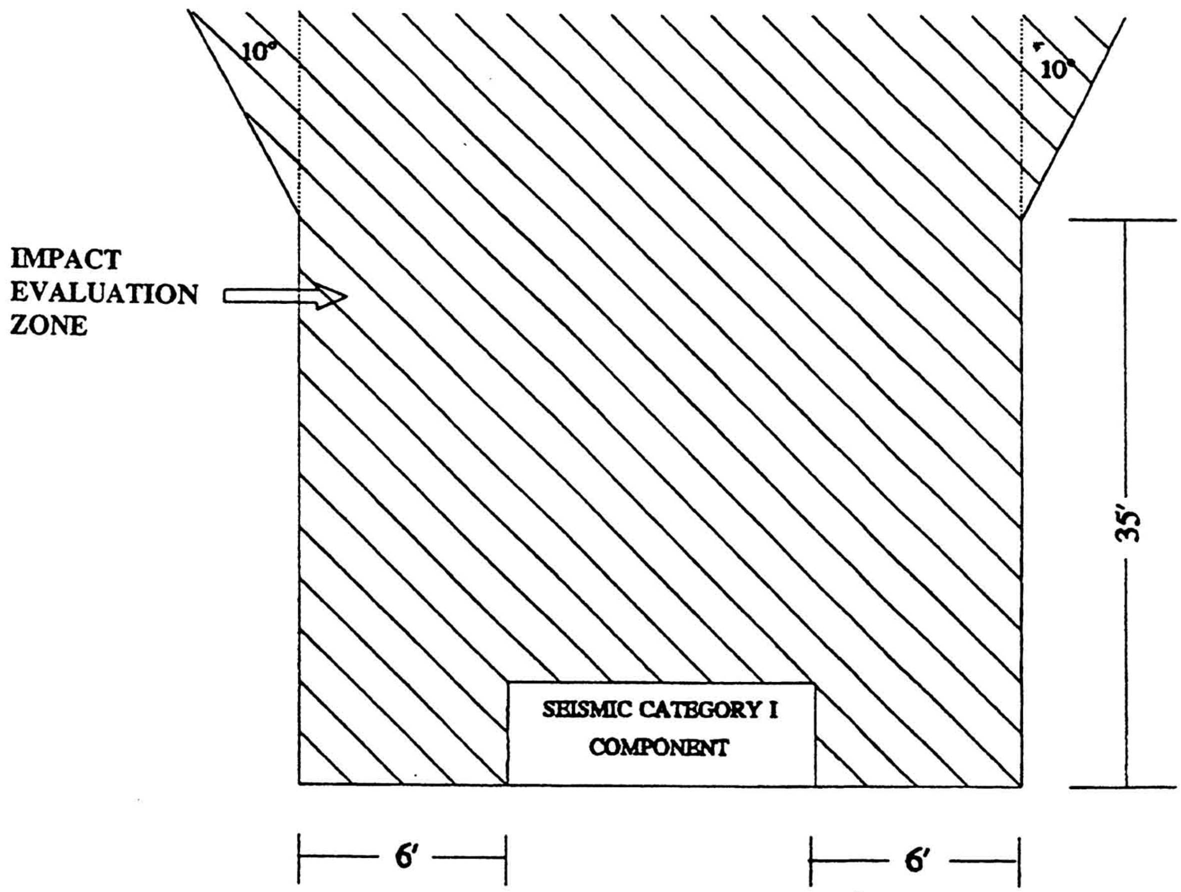


Figure 3.7.3-1
Impact Evaluation Zone

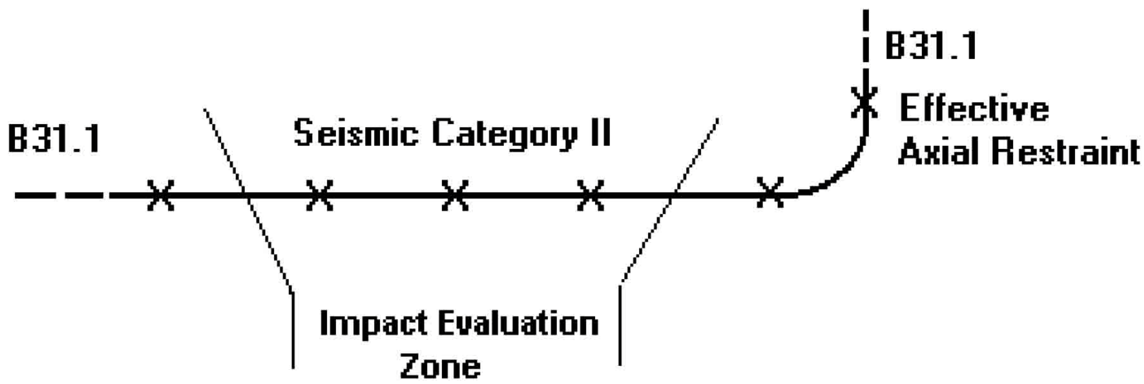


Figure 3.7.3-2
Impact Evaluation Zone and Seismic Supported Piping

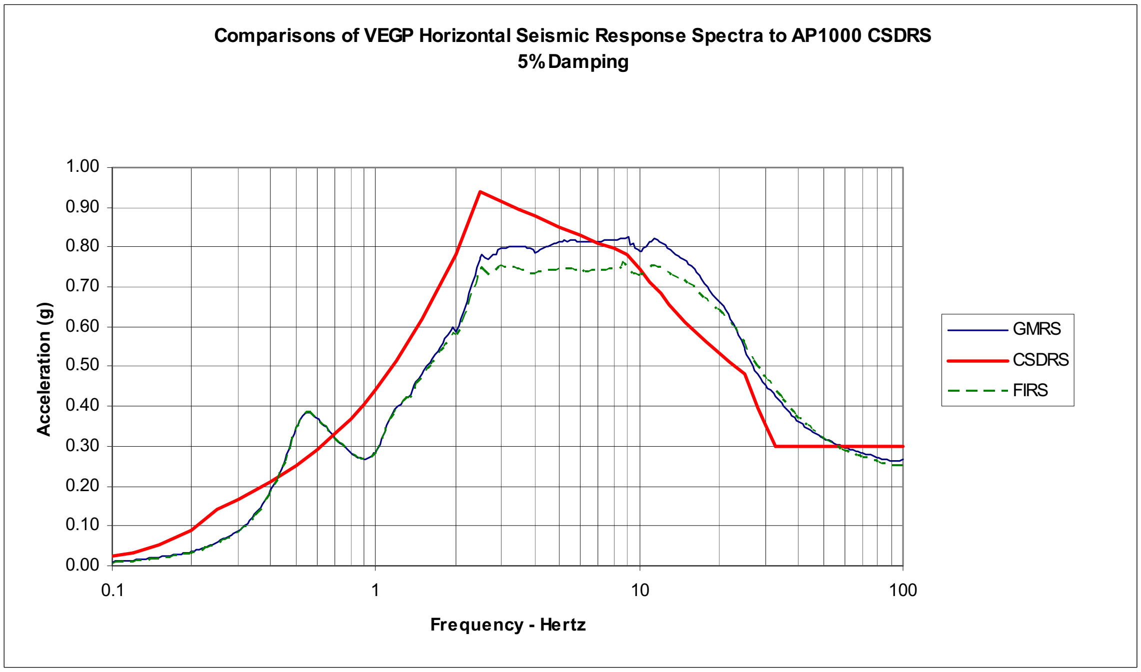


Figure 3.7-201
VEGP AP1000 Horizontal Spectra Comparison

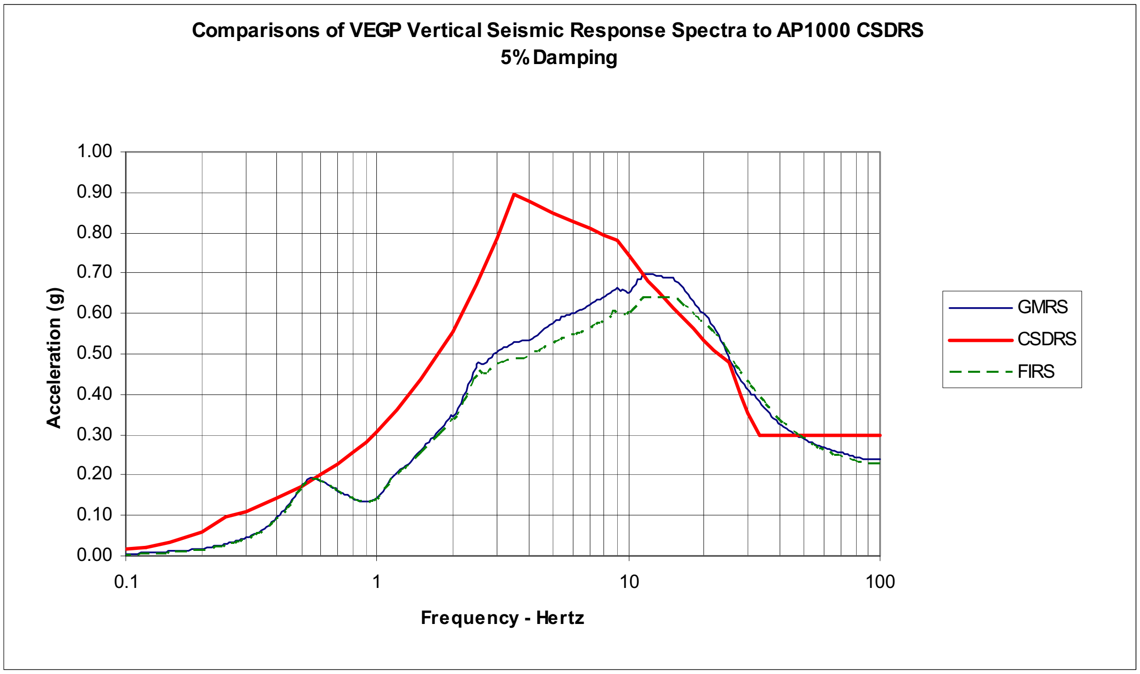


Figure 3.7-202
VEGP AP1000 Vertical Spectra Comparison

3.8 Design of Category I Structures

3.8.1 Concrete Containment

This subsection is not applicable to the AP1000.

3.8.2 Steel Containment

3.8.2.1 Description of the Containment

3.8.2.1.1 General

This subsection describes the structural design of the steel containment vessel and its parts and appurtenances. The steel containment vessel is an integral part of the containment system whose function is described in Section 6.2. It serves both to limit releases in the event of an accident and to provide the safety-related ultimate heat sink.

The containment vessel is an ASME metal containment. The information contained in this subsection is based on the design specification and preliminary design and analyses of the vessel. Final detailed analyses will be documented in the ASME Design Report.

The containment arrangement is indicated in the general arrangement figures in Section 1.2. The portion of the vessel above elevation 132′-3″ is surrounded by the shield building but is exposed to ambient conditions as part of the passive cooling flow path. A flexible watertight and airtight seal is provided at elevation 132′-3″ between the containment vessel and the shield building. The portion of the vessel below elevation 132′-3″ is fully enclosed within the shield building.

Figure 3.8.2-1 shows the containment vessel outline. It is a free-standing, cylindrical steel vessel with ellipsoidal upper and lower heads. [*The containment vessel has the following design characteristics:*

Diameter: 130 feet

Height: 215 feet 4 inches

Design Code: ASME III, Div. 1

Material: SA738, Grade B

Design Pressure: 59 psig

Design Temperature: 300°F

Design External Pressure: 1.7 psid

Lower Personnel Airlock: Elevation 110′-6″ and 107 degrees azimuth

Lower Equipment Hatch: Elevation 112′-6″ and 126 degrees azimuth

Upper Personnel Airlock Elevation 138′-7″ and 107 degrees azimuth

Upper Equipment Hatch Elevation 141′-6″ and 67 degrees azimuth

External Stiffener: Elevation 131′-9″

*NRC Staff approval is required prior to implementing a change in this information.

Internal Stiffener: Elevation 170′-0″

Bottom Head Tangent Line Elevation 104′-1 1/2″

Upper Head Tangent Line Elevation 244′-2 1/2″

The tangent line is the elevation at which the vessel transitions from the cylinder to the head.

The wall thickness in most of the cylinder is 1.75 inches. The wall thickness of the lowest course of the cylindrical shell is increased to 1.875 inches to provide margin in the event of corrosion in the embedment transition region. The thickness of the heads is 1.625 inches.]* The heads are [*ellipsoidal*]* with a major diameter of 130 feet and a height of 37 feet, 7.5 inches.

The containment vessel includes the shell, hoop stiffeners and crane girder, equipment hatches, personnel airlocks, penetration assemblies, and miscellaneous appurtenances and attachments. The design for external pressure is dependent on the spacing of the hoop stiffeners and crane girder, which are shown on Figure 3.8.2-1. [*The spacing between each pair of ring supports (the bottom flange of the crane girder, the hoop stiffeners, and the concrete floor at elevation 100′-0″) is less than 50 feet, 6 inches. The design of the stiffeners and polar crane girder provides equal or greater radial and rotational stiffness than the design evaluated for the design certification.*]*

The polar crane is designed for handling the reactor vessel head during normal refueling. The crane girder and wheel assemblies are designed to support a special trolley to be installed in the event of steam generator replacement.

The containment vessel supports most of the containment air baffle as described in Subsection 3.8.4. The air baffle is arranged to permit inspection of the exterior surface of the containment vessel. Steel plates are welded to the dome as part of the water distribution system, described in Subsection 6.2.2. The polar crane system is described in Subsection 9.1.5.

3.8.2.1.2 Containment Vessel Support

The bottom head is embedded in concrete, with concrete up to elevation 100′ on the outside and to the maintenance floor at elevation 107′-2″ on the inside. The containment vessel is assumed as an independent, free-standing structure above elevation 100′. The thickness of the lower head is the same as that of the upper head. There is no reduction in shell thickness even though credit could be taken for the concrete encasement of the lower head.

Vertical and lateral loads on the containment vessel and internal structures are transferred to the basemat below the vessel by shear studs, friction, and bearing. The shear studs are not required for design basis loads. They provide additional margin for earthquakes beyond the safe shutdown earthquake.

Seals are provided at the top of the concrete on the inside and outside of the vessel to prevent moisture between the vessel and concrete. A typical cross section design of the seal is presented in Figure 3.8.2-8, sheets 1 and 2.

3.8.2.1.3 Equipment Hatches

Two equipment hatches are provided. One is at the operating floor (elevation 135′-3″) with an inside diameter of 16 feet. The other is at floor elevation 107′-2″ to permit grade-level access into the containment, with an inside diameter of 16 feet. The hatches, shown in Figure 3.8.2-2, consist of a cylindrical sleeve with a pressure seated dished head bolted on the inside of the vessel. The containment internal pressure acts on the convex face of the dished head and the head is in

*NRC Staff approval is required prior to implementing a change in this information.

compression. The flanged joint has double O-ring or gum-drop seals with an annular space that may be pressurized for leak testing the seals. Each of the two equipment hatches is provided with an electrically powered hoist and with a set of hardware, tools, equipment and a self-contained power source for moving the hatch from its storage location and installing it in the opening. [*The information in Figure 3.8.2-2 that is considered to be Tier 2* information is the minimum thickness of the hatch cover, the inside diameter of the sleeve, the diameter of the insert plate, the minimum thickness of the insert plate, and the nominal spherical radius of the hatch cover.*]*

3.8.2.1.4 Personnel Airlocks

Two personnel airlocks are provided, one located adjacent to each of the equipment hatches. Figure 3.8.2-3 shows the typical arrangement. Each personnel airlock has about a 10-foot external diameter to accommodate a door opening of width 3 feet 6 inches and height 6 feet 8 inches. The airlocks are long enough to provide a clear distance of 8 feet, which is not impaired by the swing of the doors within the lock. The airlocks extend radially out from the containment vessel through the shield building. They are supported by the containment vessel. [*Area reinforcement for the personnel airlocks is provided by a minimum of 3-3/4-inch-thick insert plates. The surface area of the personnel airlock insert plate, not including the sleeve, is a minimum of 51.9 ft^2.*]*

Each airlock has two double-gasketed, pressure-seated doors in series. The doors are mechanically interlocked to prevent simultaneous opening of both doors and to allow one door to be completely closed before the second door can be opened. The interlock can be bypassed by using special tools and procedures.

3.8.2.1.5 Mechanical Penetrations

The mechanical penetrations consist of the fuel transfer penetration and mechanical piping penetrations and are listed in Table 6.2.3-1. Area is added to the shell by the addition of an insert plate that is thicker than the shell or by increasing the thickness of the nozzle neck or a combination of both. This piping penetration design is then evaluated for external loads on the penetration imposed by the piping system.

Design requirements for the mechanical penetrations are as follows:

- Design and construction of the process piping follow ASME, Section III, Subsection NC. Design and construction of the remaining portions follow ASME Code, Section III, Subsection NE. The boundary of jurisdiction is according to ASME Code, Section III, Subsection NE.

- Penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation.

- Guard pipes are designed for pipe ruptures as described in Subsection 3.6.2.1.1.4.

- Bellows are stainless steel or nickel alloy and are designed to accommodate axial and lateral displacements between the piping and the containment vessel. These displacements include thermal growth of the main steam and feedwater piping during plant operation, relative seismic movements, and containment accident and testing conditions. Cover plates are provided to protect the bellows from foreign objects during construction and operation. These cover plates are removable to permit in-service inspection.

*NRC Staff approval is required prior to implementing a change in this information.

Figure 3.8.2-4, sheet 1, shows typical details for the main steam penetration. This includes bellows to minimize piping loads applied to the containment vessel and a guardpipe to protect the bellows and to prevent overpressurization of the containment annulus in a postulated pipe rupture event. Similar details are used for the feedwater penetration. [*The main steam and feedwater penetrations are combined into a common 3-3/4-inch-thick insert plate. This thickness is a minimum value. The main steam penetration has an inside sleeve diameter of 57 inches. The feedwater penetration has an inside sleeve diameter of 38 inches.*]* The insert plates for the main steam and feedwater penetrations are shown in Figure 3.8.2-4, Sheet 7. The insert plate also includes the penetration for the 6-inch-diameter startup feedwater pipe. The insert plate is designed in accordance with NE-3330, “Openings and Reinforcement,” of the ASME Code.

Figure 3.8.2-4, sheet 2, shows typical details for the startup feedwater penetration. This includes a guardpipe to prevent overpressurization of the containment annulus in a postulated pipe rupture event. Similar details are used for the steam generator blowdown penetration.

Figure 3.8.2-4, sheet 3, shows typical details for the normal residual heat removal penetration. Similar details are used for other penetrations below elevation 107′-2″ where there is concrete inside the containment vessel. The flued head is integral with the process piping and is welded to the containment sleeve. The welds are accessible for in-service inspection. The containment sleeve is separated from the concrete by compressible material.

Figure 3.8.2-4, sheet 4 shows representative details for the other mechanical penetrations. These consist of:

- A sleeve welded to containment with flued head welded to the sleeve (detail A).
- A sleeve welded to containment with the process pipe welded directly to the sleeve and in direct contact with the process fluid (detail B).
- A sleeve welded to containment with process pipe passing through the sleeve (detail C).
- A sleeve welded to a thicker insert place welded to containment with sleeve welded to pipe and in contact with process fluid (detail D).

Flued heads are used for stainless piping greater than 2 inches in nominal diameter and for piping with high operating temperatures.

Note: The type of weld connecting the sleeve to pipe may vary.

The fuel transfer penetration, shown in Figure 3.8.2-4, sheet 5, is provided to transfer fuel between the containment and the fuel handling area of the auxiliary building. The fuel transfer tube is welded to the penetration sleeve. The containment boundary is a double-gasketed blind flange at the refueling canal end. The expansion bellows are not a part of the containment boundary. Rather, they are water seals during refueling operations and accommodate differential movement between the containment vessel, containment internal structures, and the auxiliary building.

3.8.2.1.6 Electrical Penetrations

Figure 3.8.2-4, sheet 6, shows a typical 18-inch-diameter electrical penetration. The penetration assemblies consist of conductor modules (or medium voltage cable modules in a similar 18-inch-diameter penetration) passing through a bulkhead attached to the containment nozzle. Electrical design of these penetrations is described in Subsection 8.3.1.1.6.

*NRC Staff approval is required prior to implementing a change in this information.

Electrical penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation. Double barriers permit testing of each assembly to verify that containment integrity is maintained. Design and testing is according to IEEE Standard 317-83 and IEEE Standard 323-74.

3.8.2.1.7 Instrument Line Penetrations

Instrument line penetrations are designed to maintain containment integrity under design basis accident conditions, including pressure, temperature, and radiation.

Figure 3.8.2-4, sheet 4, detail B, shows typical details for the containment pressure instrumentation penetrations. The penetrations consist of sleeves welded to the containment vessel. Pressure transmitters outside containment are connected to pressure sensors inside containment by sealed, fluid-filled tubing (capillary), which passes through the sleeves. The capillary tubing is welded directly to the sleeve at a tubing coupling, which has a thicker wall and larger diameter than the capillary tubing.

Design and construction of the penetrations are in accordance with ASME Section III. The penetration sleeves, including the welds to the tubing couplings, follow ASME Section III, Subsection NE. Because ASME Section III, Subsection NCA excludes the sealed-tubing instrument configuration from the scope of Section III, the capillary tubing is designed and fabricated in accordance with ASME B31.1.

3.8.2.2 Applicable Codes, Standards, and Specifications

[*The containment vessel is designed*]* and constructed [*according to the 2001 edition of the ASME Code, Section III, Subsection NE, Metal Containment, including the 2002 Addenda. Stability of the containment vessel and appurtenances is evaluated using ASME Code, Case N-284-1, Metal Containment Shell Buckling Design Methods, Class MC, Section III, Division 1, as published in the 2001 Code Cases, 2001 Edition, July 1, 2001.*]*

Structural steel nonpressure parts, such as ladders, walkways, and handrails are designed to the requirements for steel structures defined in Subsection 3.8.4.

Section 1.9 discusses compliance with the Regulatory Guides and the Standard Review Plans.

3.8.2.3 Loads and Load Combinations

Table 3.8.2-1 summarizes the design loads, load combinations and ASME Service Levels. They meet the requirements of the ASME Code, Section III, Subsection NE. The loads and load combinations used in the analysis are considered to be part of the method of evaluation. The containment vessel is designed for the following loads specified during construction, test, normal plant operation and shutdown, and during accident conditions:

D Dead loads or their related internal moments and forces, including any permanent piping and equipment loads

L Live loads or their related internal moments and forces, including crane loads

P_o Operating pressure loads during normal operating conditions resulting from pressure variations either inside or outside containment

T_o Thermal effects and loads during normal operating conditions, based on the most critical transient or steady-state condition

*NRC Staff approval is required prior to implementing a change in this information.

R_o Piping and equipment reactions during normal operating conditions, based on the most critical transient or steady-state condition

W Loads generated by the design wind on the portion of the containment vessel above elevation 132′, as described in Subsection 3.3.1.1

E_s Loads generated by the safe shutdown earthquake (SSE) as described in Section 3.7

W_t Loads generated by the design tornado on the portion of the containment vessel above elevation 132′, as described in Subsection 3.3.2

P_t Test pressure

P_d Containment vessel design pressure that exceeds the pressure load generated by the postulated pipebreak accidents and passive cooling function

P_e Containment vessel external pressure

T_a Thermal loads under thermal conditions generated by the postulated break or passive cooling function and including T_o. This includes variations around the shell due to the surrounding buildings and maldistribution of the passive containment cooling system water.

R_a Piping and equipment reactions under thermal conditions generated by the postulated break, as described in Section 3.6, and including R_o

Y_r Loads generated by the reaction on the broken high-energy pipe during the postulated break, as described in Section 3.6

Y_j Jet impingement load on a structure generated by the postulated break, as described in Section 3.6

Y_m Missile impact load on a structure generated by or during the postulated break, as from pipe whipping, as described in Section 3.6

Post-accident flooding load combination is not applicable in the design of the AP1000 containment vessel. The post-loss-of-coolant accident (LOCA) flooding event is enveloped by the other design cases.

The AP1000 addresses the production of large quantities of hydrogen from the oxidation of zirconium and other metals as a result of a postulated severe accident. The AP1000 includes hydrogen igniters inside containment to ensure that hydrogen generated in a severe accident is burned prior to reaching an explosive mixture. The discussion of the generation and burning of hydrogen as a result of a severe accident is included in Section 19.41.

The containment vessel is protected from the direct effects of wind/tornado loads (and associated potential missiles) by virtue of its location inside the shield building. The differential pressure effects of a tornado are also reduced because of the location and are bounded by other pressure loadings for which the containment vessel is designed.

The containment is evaluated for the deterministic severe accident pressure capacity. This evaluation is discussed in Subsection 3.8.2.4.2, “Evaluation of Ultimate Capacity.” According to 10 CFR 50.44, the hydrogen generated pressure loads from 100 percent fuel clad-coolant reaction plus the peak pressure from a hydrogen burn must be less than ASME Service Level C (not including buckling). The Service Level C maximum capacity is 117 psig at 300°F as presented in Subsection 3.8.2.4.2.8

The peak pressure from the 100 percent fuel clad-coolant reaction plus the hydrogen burn (Pg1 + Pg2) is 90.3 psig as reported in Section 41.11 and Table 41-4 of the AP1000 Probabilistic Risk Assessment report. The severe accident conditions are beyond design basis accidents, and the load combinations for these severe accident evaluations are not included in the load combinations and service limits for the containment vessel.

The AP1000 does not have a post-accident inerting system. Therefore, there is no load combination that includes inerting of the containment.

Note that loads associated with flooding of the containment below elevation 107′ are resisted by the concrete structures and not by the containment vessel.

3.8.2.4 Design and Analysis Procedures

The design and analysis procedures for the containment vessel are according to the requirements of the ASME Code, Section III, Subsection NE.

The analyses are summarized in Table 3.8.2-4. The detailed analyses will use a series of general-purpose finite element, axisymmetric shell and special purpose computer codes to conduct such analyses. Code development, verification, validation, configuration control, and error reporting and resolution are according to the Quality Assurance requirements of Chapter 17.

3.8.2.4.1 Analyses for Design Conditions

3.8.2.4.1.1 Axisymmetric Shell Analyses

The containment vessel is modelled as an axisymmetric shell and analyzed using the ANSYS computer program. A model used for static analyses is shown in Figure 3.8.2-6.

Dynamic analyses of the axisymmetric model, which is similar to that shown in Figure 3.8.2-6, are performed to obtain frequencies and mode shapes. These are used to confirm the adequacy of the containment vessel stick model as described in Subsection 3.7.2.3.2. Stress analyses are performed for each of the following loads:

- Dead load
- Internal pressure
- Seismic
- Polar crane wheel loads
- Wind loads
- Thermal loads

The seismic analysis performed envelopes all soil conditions. The global seismic loads are applied as equivalent static accelerations using the maximum accelerations shown in Table 3.8.2-5. These accelerations are the maximum accelerations from the nuclear island stick model on hard rock. The global member forces from the equivalent static case exceed those from the soil cases for soil conditions described in Appendix 3G. Based on these comparisons, the design acceleration values used for the global analyses are appropriate for both the hard rock and the soil sites. The seismic analysis of the nuclear island is discussed in Section 3.7 and Appendix 3G. The torsional moments, which include the effects of the eccentric masses, are increased to account for accidental torsion and are evaluated in a separate calculation.

The results of these load cases are factored and combined in accordance with the load combinations identified in Table 3.8.2-1. These results are used to evaluate the general shell away from local penetrations and attachments, that is, for areas of the shell represented by the axisymmetric

geometry. The results for the polar crane wheel loads are also used to establish local shell stiffnesses for inclusion in the containment vessel stick model described in Subsection 3.7.2.3. The results of the analyses and evaluations are included in the containment vessel design report.

Design of the containment shell is primarily controlled by the internal pressure of 59 psig. The meridional and circumferential stresses for the internal pressure case are shown in Figure 3.8.2-5. The most highly stressed regions for this load case are the portions of the shell away from the hoop stiffeners and the knuckle region of the top head. In these regions the stress intensity is close to the allowable for the design condition.

Table 3.8.2-1 includes a design load combination to address external pressure. For the design external pressure, a conservatively large magnitude of 1.7 psi differential pressure is used. Design external pressure is defined as a value greater than the external pressure at which the vacuum relief system will open and mitigate the external pressure. This is a part of the containment air filtration system (see Subsection 9.4.7). Upon actuation, the external pressure transient is immediately controlled and the external pressure is relieved. This design external pressure is combined with a coincident -40°F outside air temperature, which corresponds to a -18.5°F metal temperature for the portions of the containment vessel shell not insulated from ambient conditions. The portions of the containment vessel shell that are below the external stiffener are insulated from the cold outside air conditions and result in a metal temperature of 70°F.

A bounding case was analyzed to provide an indication of the margin to acceptance criteria associated with the minimum allowable service metal temperature for the AP1000 containment vessel. Various types of transients were considered to evaluate the minimum service metal temperature, including inadvertent fan cooler cases, inadvertent passive containment cooling system (PCS) actuation, and loss of ac power. The evaluation considered variations in initial conditions for parameters, including humidity, internal temperature, external temperature, and wind speed. These evaluations demonstrate that the -18.5°F service metal temperature is adequate.

Design external pressure is used in load combinations that include thermal loads and are used to evaluate Service Level A and D stress limits. These external pressure conditions are included in the loading combinations in Table 3.8.2-1.

Operating pressures range from -0.2 psig to 1.0 psig, which are then combined with an ambient temperature for the containment vessel. Design internal pressure is 59 psig combined with a containment vessel metal temperature of 300°F to be evaluated in the ASME service limits as well as the design conditions.

A load combination that combines design wind plus internal design pressure is not included in Table 3.8.2-1 because the wind loads are small (within the normal operating range for containment pressure) and because the combination of the design wind and accident pressure is a lower probability than either the design wind or the accident pressure acting alone.

Major loads that induce compressive stresses in the containment vessel are internal and external pressure and crane and seismic loads. Each of these loads and the evaluation of the compressive stresses are discussed below.

- Internal pressure causes compressive stresses in the knuckle region of the top head and in the equipment hatch covers. The evaluation methods are similar to those discussed in Subsection 3.8.2.4.2 for the ultimate capacity.

- Evaluation of external pressure loads is performed in accordance with ASME Code, Section III, Subsection NE, Paragraph NE-3133.

- Crane wheel loads due to crane dead load, live load, and seismic loads result in local compressive stresses in the vicinity of the crane girder. These are evaluated in accordance with ASME Code, Case N-284.

- Overall seismic loads result in axial compression and tangential shear stresses at the base of the cylindrical portion. These are evaluated in accordance with ASME Code, Case N-284.

The bottom head is embedded in the concrete base at elevation 100 feet. This leads to circumferential compressive stresses at the discontinuity under thermal loading associated with the design basis accident. The containment vessel design includes a Service Level A combination in which the vessel above elevation 107′-2″ is specified at the design temperature of 300°F and the portion of the embedded vessel (and concrete) below elevation 100 feet is specified at a temperature of 70°F. The temperature profile for the vessel is linear between these elevations. Containment shell buckling close to the base is evaluated against the criteria of ASME Code, Case N-284.

Revision 1 of Code Case N-284 is used for the evaluation of the containment shell and equipment hatches.

3.8.2.4.1.2 Local Analyses

The penetrations and penetration reinforcements are designed in accordance with the rules of ASME III, Subsection NE. The design of the large penetrations for the two equipment hatches and the two airlocks use the results of finite element analyses which consider the effect of the penetration and its dynamic response (Reference 53).

The personnel airlocks and equipment hatches are modeled in a 3-D shell finite element model of the containment. A 3D shell, finite element model of the containment vessel was developed in ANSYS to consider the effect of the penetrations and their quasi-static response due to a seismic event. The large masses and local stiffness of the personnel locks and equipment hatches are discretely modeled. The polar crane wheel loads are incorporated by appropriate loadings (dead load and seismic loadings). The bottom of the model is fixed at elevation 100′ where the containment vessel is embedded in concrete. This means that rotations and displacements are conservatively fixed at this location.

Static analyses are performed using the finite element model shown in Figure 3.8.2-7 for internal pressure, dead load (including the polar crane in the parked position), thermal loads and seismic loads. The global seismic loads are applied as equivalent static accelerations using the maximum accelerations shown in Table 3.8.2-5. The amplified local responses are included separately for each of the four penetrations. Local seismic axial and rotational accelerations about both horizontal and vertical axes are applied based on the maximum amplified response determined from a time history analysis on a less refined dynamic model with seismic time histories at elevation 100′.

Stresses are evaluated against the stress intensity criteria of ASME Section III, Subsection NE for the load combinations described in Table 3.8.2-1. Stability is evaluated against ASME Code Case N-284. Local stresses in the regions adjacent to the major penetrations are evaluated in accordance with paragraph 1700 of the code case. Stability is not evaluated in the reinforced penetration neck and insert plate which are substantially stiffer than the adjacent shell.

3.8.2.4.2 Evaluation of Ultimate Capacity

The capacity of the containment vessel has been calculated for internal pressure loads for use in the probabilistic risk assessment analyses and severe accident evaluations. These analyses include the evaluation of the peak pressure from the hydrogen-generated pressure loads from 100-percent fuel cladding metal-water reaction plus the hydrogen burn. Each element of the containment vessel

boundary was evaluated to estimate the maximum pressure at an ambient temperature of 100°F corresponding to the following stress and buckling criteria:

- Deterministic severe accident pressure capacity corresponding to ASME Service Level C limits on stress intensity, ASME paragraph NE-3222, and ASME Code Case N-284 for buckling of the equipment hatch covers, and 60 percent of critical buckling for the top head. The deterministic severe accident pressure capacity corresponds to the approach in SECY 93-087, to maintain a reliable leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges. This approach was approved by the Nuclear Regulatory Commission as outline in the Staff Requirements Memorandum on SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs, Dated July 21, 1993.

- Best estimate capacity corresponding to gross membrane yield at the ASME-specified minimum yield stress (SA738, Grade B, yield stress = 60 ksi, ultimate stress = 85 ksi), and critical buckling for the equipment hatch covers and top head.

The results are shown in Table 3.8.2-2. The analyses at a temperature of 100°F are described in the following paragraphs for each element. The critical regions identified in this table are then examined further for their response at higher temperatures. This results in the best-estimate capacity based on the ASME-specified minimum yield properties. The evaluation considered the containment boundary elements including:

- Cylindrical shell
- Top and bottom heads
- Equipment hatches and covers
- Personnel airlocks
- Mechanical and electrical penetrations

The evaluation identified the most likely failure mode to be that associated with gross yield of the cylindrical shell. Loss of containment function would be expected to occur because the large post-yield deflections would lead to local failures at penetrations, bellows, or other local discontinuities.

3.8.2.4.2.1 Tensile Stress Evaluation of Shell

Results of the axisymmetric analyses of the cylinder and top head described in Subsection 3.8.2.4.1 for dead load and internal pressure were evaluated to determine the pressure at which stresses reach yield at an ambient temperature of 100°F. The analyses assume the shell is fixed at elevation 100′, where the bottom head is embedded in concrete. The steel bottom head is identical to the top head and has a pressure capability greater than the top head due to the additional strength of the embedment concrete.

The allowable stress intensity under Service Level C loads is equal to yield. This corresponds to an internal pressure of 135 psig. The critical section is the cylinder, where the general primary membrane stress intensity is greatest.

The best-estimate yield analysis uses the von Mises criterion to establish yield rather than the more conservative ASME stress intensity approach. This increases the yield stress by about 15 percent for the cylinder, where the longitudinal stress is equal to one-half of the hoop stress resulting in first yield at an internal pressure of 155 psig. At this pressure, hoop stresses in the cylinder reach yield. The radial deflection is about 1.6 inches. As pressure increases further, large deflections occur. For a material such as SA738, where the yield plateau extends from a strain of 0.2 percent to 0.6 percent, deflections would increase to 4.8 inches at yield without a substantial increase in pressure. Strain

hardening would then permit a further increase in pressure with large radial deflections, as described in Subsection 3.8.2.4.2.6.

3.8.2.4.2.2 Buckling Evaluation of Top Head

The top head has a radius-to-height ratio of 1.728. This is not as shallow as most ellipsoidal or torispherical heads, which typically have a radius-to-height ratio of 2. The ratio was specifically selected to minimize the local stresses and buckling in the knuckle region due to internal pressure. As the ratio decreases, the magnitude of compressive stresses in the knuckle region decreases; for a radius-to-height ratio of 1.4 or smaller, there are no compressive stresses and therefore there is no potential for buckling.

Theoretical Buckling Capacity

The top head was analyzed using the BOSOR-5 computer code (Reference 1). This code permits consideration of both large displacements and nonlinear material properties. It calculates shell stresses and checks stability at each load step. The analysis included a portion of the cylinder with a thickness of 1.625 inches. In this analysis, yield of the cylinder started at a pressure of 144 psig using elastic – perfectly plastic material properties, a yield stress of 60 ksi, and the von Mises yield criterion. Yield of the top of the crown started at an internal pressure of 146 psig. Yield of the knuckle region started at 152 psig. A theoretical plastic buckling pressure of 174 psig was determined. At this pressure, the maximum effective prebuckling strain was 0.23 percent in the knuckle region where buckling occurred and 2.5 percent at the crown. The maximum deflection at the crown was 15.9 inches. A similar analysis was performed using nonlinear material properties considering the effects of residual stresses; buckling did not occur in this analysis, and failure would occur once strains at the crown reach ultimate. The failure mode was found to be an axisymmetric plastic collapse resulting from excessive vertical displacements at the crown. The maximum displacement was 43 inches at 195 psig.

Predicted Pressure Capacity

The actual buckling capacity may be lower than the theoretical buckling capacity because of effects not included in the analysis such as imperfections and residual stresses. This is considered by the use of capacity reduction factors that are based upon a correlation of theory and experiment. The capacity reduction factor for the top head was evaluated based on comparisons of BOSOR-5 analyses against test results of ellipsoidal and torispherical heads. This evaluation is described below and concludes that no reduction in capacity need be considered; that is, a capacity reduction factor of 1.0 is appropriate.

The knuckle region of ellipsoidal and torispherical heads is subjected to meridional tension and circumferential compression. The meridional tension tends to stabilize the knuckle region and reduces its sensitivity to imperfection. The radius-to-height ratio of 1.728 of the AP1000 head results in a larger ratio of meridional tension to circumferential compression than on shallower heads, further reducing the sensitivity to imperfection.

Welding Research Council Bulletin 267 (Reference 22) shows a comparison of BOSOR-5 predictions of buckling against the results of 20 tests of small head models. These results are summarized in Table 4 of the reference and show ratios (capacity reduction factors) of actual buckling to the BOSOR-5 prediction with an average of 1.2. Only one of the 20 cases shows a capacity reduction factor less than 1.0.

Table 3.8.2-3 shows the key parameters, test results, and BOSOR-5 predictions for two large, fabricated 2:1 torispherical heads tested and reported in NUREG/CR-4926 (Reference 23). The theoretical plastic buckling pressure predicted by BOSOR-5 represents initial buckling based on actual material properties. The initial buckling did not cause failure for either of the tests, and test

pressure continued to increase until rupture occurred in the spherical cap. The collapse pressures were three to four times the initial buckling pressures.

- **Test Head 1** – The test result of 58 psig is 79 percent of the predicted theoretical plastic buckling pressure of 74 psig. Many of the buckles occurred directly on the meridional weld seams of the knuckle. The knuckle welds were noticeably flatter than the corresponding welds of the Test 2 head. The as-built configuration extended inside the theoretical shape at some of the meridional weld seams and was most pronounced at the location of the first observed buckle. Model 1 exceeded the tolerances for formed heads specified for containment vessels in NE-4222.2 of ASME, Section III, Subsection NE.

- **Test Head 2** – The test result of 106 psi is 100 percent of the BOSOR-5 predicted theoretical plastic buckling pressure. For test head 2, the welds had no noticeable flat spots and there was a smooth transition between the sphere and knuckle sections. Test head 2 was well within the Code allowable deviations.

The low-capacity reduction factor of 0.79 for test head 1 is attributed to excessive imperfections associated with the fabrication of relatively thin plate (0.196 inch). These imperfections were visible and were outside the tolerances permitted by the ASME Code. The results of test head 1 are therefore not considered applicable to the AP1000. The results of test head 2 and of the small-scale models described in the Welding Research Council Bulletin support the application of a capacity reduction factor of 1.0.

The capacity of the AP1000 head was also investigated using an approach similar to that permitted in ASME Code, Case N284. This code case provides alternate rules for certain containment vessel geometries such as cylindrical shells. The theoretical elastic buckling pressure was calculated to be 536 psi using the linear elastic computer code, BOSOR-4 (Reference 24). A reduction factor (defined as the product of the capacity reduction factor and the plastic reduction factor) was established as 0.385 based on the lower bound curve of test results of 20 ellipsoidal and 28 torispherical test specimens, which also include the two large fabricated heads previously discussed. This resulted in a predicted buckling capacity of 206 psig.

The preceding paragraphs addressed incipient buckling. It is concluded that buckling would not occur prior to reaching the pressure of 174 psig predicted in the BOSOR-5 analyses. Tests indicate that pressure can be significantly increased prior to rupture after the formation of the initial buckles. Failure would occur when local strains reach ultimate either close to a local buckle in the knuckle or at the center of the crown. The best estimate capacity of the head is taken as the theoretical plastic buckling pressure of 174 psig predicted in the BOSOR-5 analyses.

The deterministic severe accident pressure capacity is taken as 60 percent of critical buckling. This is consistent with the safety factor for Service Level C in ASME Code, Case N-284 and results in a containment head capacity of 104 psig.

3.8.2.4.2.3 Equipment Hatches

SECY 93-087 permits evaluation of certain severe accident scenarios against ASME Service Level C limits. The equipment hatch covers were evaluated for buckling against ASME paragraph NE-3222 and according to ASME Code, Case N-284. Use of ASME Code, Case N-284 for this application was confirmed to be appropriate by ASME. The containment internal pressure acts on the convex face of the dished head and the hatch covers are in compression under containment internal pressure loads. The critical buckling capacity is based on classical buckling capacities reduced by capacity reduction factors to account for the effects of imperfections and plasticity. These capacity reduction factors are based on test data and are generally lower-bound values for the tolerances specified in the ASME Code.

The critical buckling pressure is 211 psig for the 16-foot-diameter hatch at an ambient temperature of 100°F. For the Service Level C limits in accordance with paragraph NE 3222, a safety factor of 2.50 is specified, resulting in capabilities of 84 psig (16-foot-diameter). For the Service Level C limits in accordance with Code Case N284, a safety factor of 1.67 is specified, resulting in capabilities of 126 psig (16-foot-diameter).

Typical gaskets have been tested for severe accident conditions as described in NUREG/CR-5096 (Reference 25). The gaskets for the AP1000 will be similar to those tested with material such as Presray EPDM E 603. For such gaskets the onset of leakage occurred at a temperature of about 600°F.

3.8.2.4.2.4 Personnel Airlocks

The capacity of the personnel airlocks was determined by comparing the airlock design to that tested and reported in NUREG/CR-5118 (Reference 3). Critical parameters are the same, so the results of the test apply directly. In the tests the inner door and end bulkhead of the airlock withstood a maximum pressure of 300 psig at 400°F. The capacity of the airlock is therefore at least 300 psig at ambient temperature. The maximum pressure corresponding to Service Level C is conservatively estimated by reducing this capacity in the ratio of the minimum specified material yield to ultimate.

3.8.2.4.2.5 Mechanical and Electrical Penetrations

Subsections 3.8.2.1.3 through 3.8.2.1.6 describe the containment penetrations. Penetration reinforcement is designed following the area replacement method of the ASME Code. The insert plates and sleeves permit development of the hoop tensile yield stresses predicted as the limiting capacity in Subsection 3.8.2.4.1. Capacities of the equipment hatch covers are discussed in Subsection 3.8.2.4.2.3 and of the personnel airlocks in Subsection 3.8.2.4.2.4.

Mechanical penetrations welded directly to the containment vessel are generally piping systems with design pressures greater than that of the containment vessel. Thicknesses of the flued head or end plate are established based on piping support loads or stiffness requirements. The capacities of these penetrations are greater than the capacity of the containment vessel cylinder.

Mechanical penetrations for the large-diameter high-energy lines, such as the main steam and feedwater piping, include expansion bellows. The piping and flued head have large pressure capability. The response of expansion bellows to severe pressure and deformations is described in NUREG/CR-5561 (Reference 4). The bellows can withstand large pressure loading but may tear once the containment vessel deflection becomes large. Testing reported in NUREG/CR-6154 (Reference 26) has shown that the bellows remain leaktight even when subjected to large deflections sufficient to fully compress the bellows. Such large deflections do not occur as long as the containment vessel remains elastic. As described in Subsection 3.8.2.4.2.6, the radial deflection of the shell increases substantially once the containment cylinder yields. The resulting deflections are assumed to cause loss of containment function. The containment penetration bellows are designed for a pressure of 90 psig at design temperature within Service Level C limits, concurrent with the relative displacements imposed on the bellows when the containment vessel is pressurized to these magnitudes.

Electrical penetrations have a pressure boundary consisting of the sleeve and an end plate containing a series of modules. The electrical pressure boundary is designed and built to the requirements of the ASME Code, Section III, Class MC, Subsection NE. The pressure capacity of these elements is large and is greater than the capacity of the containment vessel cylinder at temperatures up to the containment design temperature. Electrical penetration assemblies are also designed to satisfy ASME Service Level C stress limits under a pressure of 90 psig at design temperature. Tests at pressures and temperatures representative of severe accident conditions are

described in NUREG/CR-5334 (Reference 5), where typical nuclear industry penetrations were irradiated, aged, then tested. One design was tested to 135 psia at 700°F. Other electrical penetration assemblies were tested to 75 psia at 400°F and 155 psia at 361°F. These tests showed that the electrical penetration assemblies withstand severe accident conditions. The electrical penetration assemblies are qualified for the containment design basis event conditions as described in Appendix 3D. The assemblies are similar to one of those tested by Sandia as reported in NUREG/CR-5334 (Reference 5). The ultimate pressure capacity of the electrical penetration assemblies is primarily determined by the temperature. The maximum temperature of the containment vessel below the operating deck during a severe accident is below the temperature at which the assemblies from the three suppliers in the Sandia tests were tested.

3.8.2.4.2.6 Material Properties

The containment vessel is designed using SA738, Grade B material. This has a specified minimum yield of 60 ksi and ultimate of 85 ksi. Test data for materials having similar chemical properties were reviewed. In a sample of 122 tests for thicknesses equaling or exceeding 1.50 inches and less than 1.75 inches, the actual yield had a mean value of 69.1 ksi with a standard deviation of 3.3 ksi. Thus, the actual yield is expected to be about 15 percent higher than the minimum yield. Membrane yield of the cylinder is predicted to occur at an internal pressure of 178 psig.

A stress-strain curve for material with chemistry similar to SA738, Grade B, indicated constant yield stress of 81.3 ksi from a strain of 0.002 to 0.006 followed by strain-hardening up to a maximum stress of 94.5 ksi at a strain of 0.079. The first portion of the strain-hardening is nearly linear, with a stress of 90 ksi at a strain of 4 percent. This strain occurs at a stress 10 percent above yield. Thus, a pressure load 10 percent higher than that corresponding to yield of the shell would result in 4 percent strain and a 31-inch radial deflection of the containment cylinder. Such a deflection is expected to cause major distress for penetrations, the air flow path, and local areas where other structures are close to the containment vessel. Loss of function is therefore assumed for the containment once gross yield of the containment cylinder occurs.

3.8.2.4.2.7 Effect of Temperature

The evaluations described in the preceding subsections are based on an ambient temperature of 100°F. Nonmetallic items, such as gaskets, are qualified to function at the design temperature. The capacity of steel elements is reduced in proportion to the reduction due to temperature in yield stress, ultimate stress, or elastic modulus. The cylinder is governed by yield stress, and elastic buckling of the hatch covers is governed by the elastic modulus. The reduction in capacity is estimated using the tables given for material properties in the ASME Code. At 400°F, the yield stress is reduced by 17 percent and the pressure capacity corresponding to gross yield is reduced from 155 to 129 psig.

3.8.2.4.2.8 Summary of Containment Pressure Capacity

The ultimate pressure capacity for containment function is expected to be associated with leakage caused by excessive radial deflection of the containment cylindrical shell. This radial deflection causes distress to the mechanical penetrations, and leakage would be expected at the expansion bellows for the main steam and feedwater piping. There is high confidence that this failure would not occur before stresses in the shell reach the minimum specified material yield. This is calculated to occur at a pressure of 155 psig at ambient temperature and 129 psig at 400°F. Failure would be more likely to occur at a pressure about 15 percent higher based on expected actual material properties.

The deterministic severe accident pressure that can be accommodated according to the ASME Service Level C stress intensity limits and using a factor of safety of 1.67 for buckling of the top head is determined by the capacity of the 16-foot-diameter equipment hatch cover and the ellipsoidal head. The maximum capacity of the hatch cover, calculated according to ASME paragraph NE-3222,

Service Level C, is 84 psig at an ambient temperature of 100°F and 81 psig at 300°F. When calculated in accordance with ASME Code, Case N-284, Service Level C, the maximum capacity is 126 psig at an ambient temperature of 100°F and 121 psig at 300°F. The maximum capacity of the ellipsoidal head is 104 psig at 100°F and 91 psig at 300°F.

The maximum pressure that can be accommodated according to the ASME Service Level C stress intensity limits, excluding evaluation of instability, is determined by yield of the cylinder and is 135 psig at an ambient temperature of 100°F and 117 psig at 300°F. This limit is used in the evaluations required by 10 CFR 50.44.

3.8.2.5 Structural Criteria

The containment vessel is designed, fabricated, installed, and tested according to the ASME Code, Section III, Subsection NE, and will receive a code stamp.

Stress intensity limits are according to ASME Code, Section III, Paragraph NE-3221 and Table NE-3221-1. [*Critical buckling stresses are checked according to the provisions of ASME Code, Section III, Paragraph NE-3222, or ASME Code Case N-284.*]*

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

Materials for the containment vessel, including the equipment hatches, personnel locks, penetrations, attachments, and appurtenances meet the requirements of NE-2000 of the ASME Code. [*The basic containment material is SA-738, Grade B, plate. The procurement specification for the SA-738, Grade B, plate includes SA-20 supplemental requirements S1, Vacuum Treatment and S20, Maximum Carbon Equivalent for Weldability.*]* This material has been selected to satisfy the lowest service metal temperature requirement of -18.5°F. This temperature is established by analysis for the portion of the vessel exposed to the environment when the minimum ambient air temperature is -40°F. Impact test requirements are as specified in NE-2000.

[*The material of construction for the insert plates and fabricated nozzle necks of penetrations is SA-738 Grade B. The material of construction for forged nozzle neck forgings is SA-350, LF2, Class 1 for penetrations greater than 2 inches nominal diameter and less than 24 inches inside diameter, and for the containment air filtration system penetration nozzle necks. The maximum carbon equivalent for the SA-350, LF2, Class 1 used for penetrations is 0.52 percent. A vacuum refining process is required for SA-350, LF2, Class 1. The SA-350, LF2, Class 1 material is used in portions of the containment vessel that are below the external stiffener.*]* These portions of the containment vessel are insulated from the cold outside air temperature. Insulation is provided around the upper equipment hatch and personnel airlock, including the insert plates, in order to insulate the penetrations from the ambient air temperature in the upper annulus.

The containment vessel is coated with an inorganic zinc coating, except for those portions fully embedded in concrete. The inside of the vessel below the operating floor and up to 8 feet above the operating floor also has a phenolic top coat. Below elevation 100′ the vessel is fully embedded in concrete with the exception of the few penetrations at low elevations (see Figure 3.8.2-4, sheet 3 of 6, for typical details). Embedding the steel vessel in concrete protects the steel from corrosion.

The AP1000 configuration is shown in the general arrangement figures in Section 1.2 and in Figure 3.8.2-1. The exterior of the vessel is embedded at elevation 100′ and concrete is placed against the inside of the vessel up to the maintenance floor at elevation 107′-2″. Above this elevation the inside and outside of the containment vessel are accessible for inspection of the coating. The vessel is coated with an inorganic zinc primer to a level just below the concrete.

*NRC Staff approval is required prior to implementing a change in this information.

Seals are provided at the surface of the concrete inside and outside the vessel so that moisture is not trapped next to the steel vessel just below the top of the concrete. The seal on the inside accommodates radial growth of the vessel due to pressurization and heatup.

The plate thickness for the first course (elevation 104′-1.5″ to minimum 109′-0″) of the cylinder is 1.875 inches, which is 1/8-inch thicker than the rest of the vessel. This provides margin in the event there would be any corrosion in the transition region despite the coatings and seals described previously. Equivalent margin is available for the 1.625-inch-thick bottom head in the transition region (elevation 100′ to 104′-1.5″). The plate thickness for the head is a constant thickness and is established by the stresses in the knuckle. As a result, the pressure stresses in the transition zone are well below the allowable stress, providing margin in the event of corrosion in this region.

The quality control program involving welding procedures, erection tolerances, and nondestructive examination of shop- and field-fabricated welds conforms with Subsections NE-4000 and NE-5000 of the ASME Code. The containment vessel is designed to permit its construction using large subassemblies. These subassemblies consist of the two heads and three or more ring sections. These are assembled in an area near the final location, using plates fabricated in a shop facility.

3.8.2.7 Testing and In-Service Inspection Requirements

Testing of the containment vessel and the pipe assemblies forming the pressure boundary within the containment vessel will be according to the provisions of NE-6000 and NC-6000, respectively.

Subsection 6.2.5 describes leak-rate testing of the containment system including the containment vessel.

In-service inspection of the containment vessel will be performed. See Section 6.6 for information on inservice inspection for the containment vessel and penetrations.

3.8.3 Concrete and Steel Internal Structures of Steel Containment

3.8.3.1 Description of the Containment Internal Structures

The containment internal structures are those concrete and steel structures inside (not part of) the containment pressure boundary that support the reactor coolant system components and related piping systems and equipment. The concrete and steel structures also provide radiation shielding. The containment internal structures are shown on the general arrangement drawings in Section 1.2. The containment internal structures consist of the primary shield wall, reactor cavity, secondary shield walls, in-containment refueling water storage tank (IRWST), refueling cavity walls, operating floor, intermediate floors, and various platforms. The polar crane girders are considered part of the containment vessel. They are described in Subsection 3.8.2.

Component supports are those steel members designed to transmit loads from the reactor coolant system to the load-carrying building structures. The component configuration is described in this subsection including the local building structure backing up the component support. The design and construction of the component supports are described in Subsection 5.4.10.

The containment internal structures are designed using reinforced concrete and structural steel. At the lower elevations conventional concrete and reinforcing steel are used, except that permanent steel forms are used in some areas in lieu of removable forms based on constructibility considerations. These steel form modules (liners) consist of plate reinforced with angle stiffeners and tee sections, as shown in Figure 3.8.3-16. The angles and the tee sections are on the concrete side of the plate. Welded studs, or similar embedded steel elements, are attached on the concrete face of the permanent steel form where surface attachments transfer loads into the concrete. Where these

surface attachments are seismic Category I, the portion of the steel form module transferring the load into the concrete is classified as seismic Category I.

Walls and floors are concrete filled steel plate structural modules. The walls are supported on the mass concrete containment internal structures basemat with the steel surface plate extending down to the concrete floor on each side of the wall. The steel surface plates of the structural modules provide reinforcement in the concrete. The structural modules are anchored to the base concrete by mechanical connections welded to the steel plate as shown in Figure 3.8.3-8, Sheet 2. Figure 3.8.3-1 shows the location of the structural modules. Figures 3.8.3-2 and 3.8.3-15 show the typical structural configuration of the wall modules. Key structural elements of the module design are identified as Tier 2* information in the text and figures of this section. See DCD Introduction, Section 3.5 for a discussion of Tier 2* information. [*The information in Figure 3.8.3-2 that is Tier 2* is the minimum size of the angles and channels used to fabricate the modules. The information in Figure 3.8.3-15 that is Tier 2* is the design spacing between the face plates for the 4-foot-thick refueling canal wall in the containment internal structures and the design spacing between the trusses used to fabricate the modules in locations away from openings or penetrations in the wall. Local variation in the design of the trusses and spacing of the trusses and shear studs may be required to address internal obstructions and accessibility for fabrication and inspection. The obstructions include features such as leak chases; internal structures such as reinforcements, embedments, and backup structures; and internal conduit and piping.*]* See Subsection 3.8.3.5.3.6 for criteria for design of the shear stud size and spacing.

A representative floor module is shown in Figure 3.8.3-3 and also in Figure 3.8.3-16 combined with the liner module. These structural modules are structural elements built up with welded steel structural shapes and plates. The details shown in Figure 3.8.3-3 are a representative design for a specific portion of the operating deck floor in containment. The size, material, and configuration of the structural shapes and plates, the amount and arrangement of the reinforcement, and the type and size of the welds may vary in the final design of floor modules in this and other locations. Concrete is used where required for shielding, but reinforcing steel is not normally used.

Walls and floors exposed to water during normal operation or refueling are constructed using stainless steel plates.

3.8.3.1.1 Reactor Coolant Loop Supports

3.8.3.1.1.1 Reactor Vessel Support System

The reactor vessel is supported by four supports located under the cold legs, which are spaced 90 degrees apart in the primary shield wall. The supports are designed to provide for radial thermal growth of the reactor coolant system, including the reactor vessel, but they prevent the vessel from lateral and torsional movement. The ends of the supports are bolted to steel embedments in the concrete wall surrounding the reactor vessel. Each support connects to the embedments with eight 2-3/8-inch-diameter dowel studs. In addition, the center portion of the support is attached to the concrete with eight threaded anchor bolts embedded in the primary shield wall concrete. The reactor vessel supports loads are carried by the dowel studs to embedded steel structures and the anchor bolts to the concrete. Figure 3.8.3-4 shows the reactor vessel supports.

3.8.3.1.1.2 Steam Generator Support System

The steam generator vertical support consists of a single vertical column extending from the steam generator compartment floor to the bottom of the steam generator channel head. The column is constructed of heavy plate sections and is pinned at both ends to permit unrestricted radial displacement of the steam generator during plant heatup and cooldown. The location of this column is such that it will allow full access to the steam generator for routine maintenance activities. It is

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located a sufficient distance away from the reactor coolant pump motors to permit pump maintenance and in-service inspection.

The lower steam generator horizontal support is located at the top of the vertical column. It consists of a tension/compression strut oriented approximately perpendicular to the hot leg. The strut is pinned at both the wall bracket and the vertical column to permit movement of the generator during plant heatup and cooldown.

The upper steam generator horizontal support in the direction of the hot leg is located on the upper shell just above the transition cone. It consists of two large hydraulic snubbers oriented parallel with the hot leg centerline. One snubber is mounted on each side of the generator on top of the steam generator compartment wall. The hydraulic snubbers are valved to permit steam generator movement for thermal transition conditions, and to "lock-up" and act as rigid struts under dynamic loads.

The upper steam generator horizontal support in the direction normal to the hot leg is located on the lower shell just below the transition cone. It consists of two rigid tension/compression struts oriented perpendicular to the hot leg. The two rigid struts are mounted on the steam generator compartment wall at the elevation of the operating deck. The steam generator loads are transferred to the struts and snubbers through trunnions on the generator shell. Figure 3.8.3-5 shows the steam generator supports.

The steam generator supports are anchored using anchor bolts or steel weldments embedded in the concrete, designed in accordance with Appendix B of ACI 349. The lower portion of the column pedestal, embedded in the concrete, as shown on sheet 1 of Figure 3.8.3-5, transfers the vertical load into the reinforced concrete basemat. The lower and intermediate horizontal supports are located so that the loads are transferred into the plane of the adjacent floor. The upper supports are located so that the loads are transferred into the plane of the steam generator compartment walls.

3.8.3.1.1.3 Reactor Coolant Pump Support System

Because the reactor coolant pumps are integrated into the steam generator channel head, they do not have individual supports. They are supported by the steam generators.

3.8.3.1.1.4 Pressurizer Support System

The pressurizer is supported by four columns mounted from the pressurizer compartment floor. A lateral support is provided at the top of the columns. This lateral support consists of eight struts connecting it to the pressurizer compartment walls. A lateral support is also provided on the upper portion of the pressurizer. This lateral support consists of a ring girder around the pressurizer and eight struts connecting it to the pressurizer compartment walls. Figure 3.8.3-6 shows the pressurizer supports.

3.8.3.1.2 Containment Internal Structures Basemat

The containment internal structures basemat is the reinforced concrete structure filling the bottom head of the containment vessel. It extends from the bottom of the containment vessel head at elevation 66′-6″ up to the bottom of the structural modules that start between elevations 71′-6″ and 103′-0″. The basemat includes rooms as shown on Figure 1.2-5. The primary shield wall and reactor cavity extend from elevation 71′-6″ to elevation 107′-2″. They provide support for the reactor vessel and portions of the secondary shield walls and refueling cavity walls. The general arrangement drawings in Section 1.2 show the location and configuration of the primary shield wall and reactor cavity. The walls of the primary shield, the steam generator compartment and the CVS room are structural modules as shown in Figure 3.8.3-1. The rest of the basemat is reinforced concrete.

3.8.3.1.3 Structural Wall Modules

Structural wall modules are used for the primary shield wall around the reactor vessel, the wall between the vertical access and the CVS room, secondary shield walls around the steam generators and pressurizer, for the east side of the in-containment refueling water storage tank, and for the refueling cavity. The general arrangement drawings in Section 1.2 show the location and configuration. Locations of the structural modules are shown in Figure 3.8.3-1. Isometric views of the structural modules are shown in Figure 3.8.3-14. The figure shows the use of modules, overall configuration, and general design approach for incorporation of structural modules into the structural design. Alternatives to the design details shown in Figure 3.8.3-14, including the size, location, and type of structural shapes attached to the faceplates and embedded in the concrete, may be used. The embedded lower portions of the modules in the concrete, attachments, and penetrations may also be added or differ from that shown in the figure. The secondary shield walls are a series of walls that, together with the refueling cavity wall, enclose the steam generators. Each of the two secondary shield wall compartments provides support and houses a steam generator and reactor coolant loop piping. The in-containment refueling water storage tank is approximately 30 feet high. The floor elevation of this tank is 103′-0″. The tank extends up to about elevation 133′-3″, directly below the operating deck. On the west side, along the containment vessel wall, the tank wall consists of a stainless steel plate stiffened with structural steel sections in the vertical direction and angles in the horizontal direction. Structural steel modules, filled with concrete and forming, in part, the refueling cavity, steam generator compartment, and pressurizer compartment walls, compose the east wall. The refueling cavity has two floor elevations. The area around the reactor vessel flange is at elevation 107′-2″. The lower level is at elevation 98′-1″. The upper and lower reactor internals storage is at the lower elevation, as is the fuel transfer tube. The center line of the fuel transfer tube is at elevation 100′-5.75″.

Structural wall modules consist of steel faceplates connected by steel trusses. The primary purpose of the trusses is to stiffen and hold together the faceplates during handling, erection, and concrete placement. The nominal spacing of the trusses is 30 inches. The nominal thickness of the steel faceplates is 0.5 inch. [*In the CA01 and CA05 structural wall modules, the faceplate thickness is greater than the nominal thickness where required to support local demand. For areas subject to these high out-of-plane loads, the faceplate thickness may be increased up to a thickness of 1.5 inches. To support loading associated with the steam generator lateral supports, the faceplate thickness may be increased up to a thickness of 3.0 inches.*]* At corner locations, the trusses are replaced with diaphragms. These diaphragms have similar purpose as the trusses and are used to fabricate the modules in the corners. The detail design of the trusses and the design of the diaphragms shown in Figure 3.8.3-8, Sheet 1 are representative. Details such as the location, type, and size of welds and length and end configuration of channels used to assemble the internal trusses may vary in the design implemented from that shown in the figure. The size, shape, and location of welds, structural shapes, and other elements used to fabricate and stiffen the diaphragms and connect them to the faceplates and the location, size, and shape of the access opening may vary from that shown in the figure. Steel plates, structural shapes, reinforcement bars, or tie bars are installed between the structural wall module faceplates and embedded in the concrete as additional structural elements that support out-of-plane loads and structural integrity. This reinforcement is designed to the applicable requirements of ACI 349 and AISC N690. Each module has different characteristics that require the detailed design to be specific for that module and loading and interferences within that module.Shear studs are welded to the inside faces of the steel faceplates. Face plates are welded to adjacent plates with full penetration welds so that the weld is at least as strong as the plate. The full penetration welds are identified in Figure 3.8.3-8, Sheet 1 with the weld symbol that includes the notation CJP for complete joint penetration. Plates on each face of the wall module extend down to the elevation of the adjacent floor. Where the floors in the rooms on each side of the wall module are at different elevations, one of the plates at the bottom of the module extends further down than the other. This single-sided configuration is designated on Figure 3.8.3-1 as “CA Structure Module with Single Surface Plate” for lower elevations in containment. A typical

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example with the single-sided configuration at the bottom of the module is shown in Figure 3.8.3-8. Some modules or portions of modules do not have the single-sided configuration at the bottom of the module. The module functions as a wall above the upper floor level (elevation 103′-0″ in Figure 3.8.3-8). The single plate below this elevation is designed to transfer the reactions at the base of the wall into the basemat. This plate also acts as face reinforcement for the basemat. Basemat reinforcement dowels are provided at the bottom of the single plate as shown in Figure 3.8.3-8.

[*Shear studs, structural shapes, or horizontally oriented reinforcement bars mechanically connected or welded to the module faceplates may be used to anchor the CA01 and CA05 structural modules to the base concrete. Shear lugs designed in accordance with ACI 349 may also be used to transfer loads at the connection. Where horizontal bars are attached perpendicular to the faceplate, they function like shear studs to provide for faceplate development. Where additional reinforcement is required for detailing base concrete at module walls with offsets in faceplate elevation, the reinforcement is developed in accordance with ACI 349. Where horizontal bars are attached to the edge of a faceplate at an elevation offset, the bars are designed for a combination of applicable demand and developed in accordance with ACI 349.*]*

[*The information in Figure 3.8.3-8, Sheet 1 that is considered to be Tier 2* information is the design spacing of the faceplates, trusses, channels in the trusses and the minimum size and design spacing of the headed studs for the modular wall in the containment internal structure in locations away from openings or penetrations in the walls. Local variation in the design of the trusses and spacing of the trusses and shear studs may be required to address internal obstructions and accessibility for fabrication and inspection. As shown in Figure 3.8.3-1, Sheets 2 and 4, some CA01 and CA05 wall thicknesses are different than the wall thicknesses listed in Figure 3.8.3-8. Wall thickness increase due to local faceplate thickness increase is below the level of detail depicted in Figure 3.8.3-1. The module wall thicknesses are determined based on radiation protection considerations and are designed to the applicable requirements of AISC N690 and ACI 349. The use of full penetration welds to connect the faceplates of the modules is also considered to be Tier 2* information. The information in Figure 3.8.3-8, Sheet 2 that is considered to be Tier 2* information is the use of mechanical connectors and the development length requirement for the mechanical connectors.*]* The detail design of the mechanical connectors is governed by AISC N690 and ACI 349, and the representative design shown is not considered to be Tier 2* information. The material, size, and configuration of plates, size and type of welds, method of attaching the connectors to the modules, and size of deformed bars anchoring the embedments into the concrete may be different for each module and may vary from that shown in the figure provided that the embedment design satisfies the AISC N690 and ACI 349 codes. Sheet 3 of Figure 3.8.3-8 shows a wall of the IRWST that is a steel structure anchored in concrete and is not a concrete filled module. [*The information in Figure 3.8.3-8, Sheet 3 that is considered to be Tier 2* information is the plate thickness for the IRWST wall, the use, spacing, and size of angles and tees to stiffen the wall, the number, size, and use of studs provided to anchor the module, and the number, size, vertical spacing, and development length of the deformed bars provided to connect the module to the mass concrete. 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.*]*

The structural wall modules are anchored to the concrete base by reinforcing steel dowels or other types of connections mechanically connected or welded to the modules and embedded in the reinforced concrete below or adjacent to the module. [*The reinforcing steel used to anchor the modules to the concrete may be oriented horizontally or vertically and has a development that satisfies the requirements of ACI 349.*]* After erection, concrete is placed between the faceplates. Typical design details of the structural modules are shown in Figures 3.8.3-2 and 3.8.3-8. Shear-friction reinforcement may be included in construction joints at the transition between the base concrete and concrete in the modules and is not part of the mechanical connection between the module and base concrete described above. The shear-friction reinforcement at the construction

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joints is spaced and sized in accordance with ACI 349 requirements and is not connected to the module faceplates.

Representative design details of the connections with floor modules are shown in Figure 3.8.3-17.

3.8.3.1.4 Structural Floor Modules

Structural floor modules are used for the operating floor at elevation 135′-3″ over the in-containment refueling water storage tank, for the southeast quadrant of the operating floor and for the 107′-2″ floor over the rooms in the containment internal structures basemat. The floors are shown on the general arrangement drawings in Section 1.2. The 107′-2″ floors consist of steel tee and wide flange sections, welded to horizontal steel bottom plates stiffened by transverse stiffeners. After erection, concrete is placed on top of the horizontal plate and around the structural steel section. The remaining region of the operating floor consists of a concrete slab, placed on Q decking supported by structural steel beams. The operating floor is supported by the in-containment refueling water storage tank walls, refueling cavity walls, the secondary shield walls, and steel columns originating at elevation 107′-2″. Structural details for a representative example of the containment operating deck floor structural module are shown in Figure 3.8.3-3.

3.8.3.1.5 Internal Steel Framing

The region of the operating floor away from the in-containment refueling water storage tank consists of a concrete slab, placed on Q decking supported by structural steel beams. The floor at elevation 118′-6″ consists of steel grating supported by structural steel framing. In addition, a number of steel platforms are located above and below the operating floor. These platforms support either grating floors or equipment, such as piping and valves.

3.8.3.2 Applicable Codes, Standards, and Specifications

The following documents are applicable to the design, materials, fabrication, construction, inspection, or testing of the containment internal structures:

[• *American Concrete Institute (ACI), Code Requirements for Nuclear Safety Related Structures, ACI-349-01*]* (refer to Subsection 3.8.4.5 for supplemental requirements)

- American Concrete Institute (ACI), ACI Detailing Manual

- American Concrete Institute (ACI), Standard Specifications for Tolerances for Concrete Construction and Materials, ACI-117

- American Concrete Institute (ACI), Guide to Formwork for Concrete, ACI-347

[• *American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, AISC-N690-1994*]* (refer to Subsection 3.8.4.5 for supplemental requirements)

Nationally recognized industry standards, such as American Society for Testing and Materials, American Concrete Institute, and American Iron and Steel Institute, are used to specify material properties, testing procedures, fabrication, and construction methods. Section 1.9 describes conformance with the Regulatory Guides and the Standard Review Plans.

Welding and inspection activities for seismic Category I structural steel, including building structures, structural modules, cable tray supports, and heating, ventilating and air-conditioning (HVAC) duct supports are accomplished in accordance with written procedures and meet the requirements of the

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American Institute of Steel Construction (AISC N-690). The welded seams of the plates forming part of the leaktight boundary of the in-containment refueling water storage tank are examined by liquid penetrant and vacuum box after fabrication to confirm that the boundary does not leak.

3.8.3.3 Loads and Load Combinations

The loads and load combinations used in the analysis are considered to be part of the method of evaluation. The loads and load combinations for the containment internal structures are the same as for other Category I structures described in Subsection 3.8.4.3 and the associated tables, except for the following modifications:

> Wind loads (W), tornado loads (W_t), and precipitation loads (N) are not applicable to the design of the containment internal structures because of the protection provided by the steel containment. Therefore, these loading terms have been excluded in the load combinations for the containment internal structures.

3.8.3.3.1 Passive Core Cooling System Loads

Structures are evaluated for pressure and thermal transients associated with operation of the passive core cooling system. The effects of temperatures higher than 100°F on the modulus of elasticity and yield strength of steel are considered.

The passive core cooling system and the automatic depressurization system (ADS) are described in Section 6.3. The automatic depressurization system is in part composed of two spargers that are submerged in the in-containment refueling water storage tank. The spargers provide a controlled distribution of steam flow to prevent imposing excessive dynamic loads on the tank structures. Capped vent pipes are installed in the roof of the tank on the side near the containment wall.

These caps prevent debris from entering the tank from the containment operating deck, but they open under slight pressurization of the in-containment refueling water storage tank. This provides a path to vent steam released by the spargers. An overflow is provided from the in-containment refueling water storage tank to the refueling cavity to accommodate volume and mass increases during automatic depressurization system operation. Two sets of loads representing bounding operational or inadvertent transients are considered in the design of the in-containment refueling water storage tank.

- ADS_1 – This automatic depressurization system load is associated with blowdown of the primary system through the spargers when the water in the in-containment refueling water storage tank is cold and the tank is at ambient pressure. Dynamic loads on the in-containment refueling water storage tank due to automatic depressurization system operation are determined using the results from the automatic depressurization system hydraulic test as described in Subsection 3.8.3.4.2. The hydrodynamic analyses described in Subsection 3.8.3.4.2 show that member forces in the walls of the in-containment refueling water storage tank are bounded by a case with a uniform pressure of 5 psi applied to the walls. The in-containment refueling water storage tank is designed for a uniform pressure of 5 psi applied to the walls. This pressure is taken as both positive and negative due to the oscillatory nature of the hydrodynamic loads. This automatic depressurization system transient is of short duration such that the concrete walls do not heat up significantly. It is combined with ambient thermal conditions. Long-term heating of the tank is bounded by the design for the ADS_2 load.

- ADS_2 – This automatic depressurization system transient considers heatup of the water in the in-containment refueling water storage tank. This may be due to prolonged operation of the passive residual heat removal heat exchanger or due to an automatic depressurization

system discharge. For structural design, an extreme transient is defined starting at 50°F since this maximizes the temperature gradient across the concrete-filled structural module walls. Prolonged operation of the passive residual heat removal heat exchanger raises the water temperature from an ambient temperature of 50°F to saturation in about 4 hours, increasing to about 260°F within about 10 hours. Steaming to the containment atmosphere initiates once the water reaches its saturation temperature. The temperature transient is shown in Figure 3.8.3-7. Blowdown of the primary system through the spargers may occur during this transient and occurs prior to 24 hours after the initiation of the event. Since the flow through the sparger cannot fully condense in the saturated conditions, the pressure increases in the in-containment refueling water storage tank and steam is vented through the in-containment refueling water storage tank roof. The in-containment refueling water storage tank is designed for an equivalent static internal pressure of 5 psi in addition to the hydrostatic pressure occurring at any time up to 24 hours after the initiation of the event.

The ADS_1 and ADS_2 loads are considered as live loads. The dynamic ADS_1 load is combined with the safe shutdown earthquake by the square root sum of the squares (SRSS). ADS_2 is an equivalent static pressure which is included algebraically with other normal loads and then combined with plus/minus SSE loads.

3.8.3.3.2 Concrete Placement Loads

The steel faceplates of the structural wall modules, designed for the hydrostatic pressure of the concrete, act as concrete forms. The concrete placement loads are 1050 pounds per square foot determined in accordance with ACI-347. The bending stress in the faceplate due to this hydrostatic pressure of the concrete is approximately 13 ksi, based on the assumption of a continuous faceplate, or 20 ksi based on the assumption of simple spans. The minimum yield strength of material for the faceplates is 36 ksi. The stress is well below the allowable, since the plate is designed to limit the out-of-plane deflection. After the concrete has gained strength, these stresses remain in the steel; however, since the average residual stress is zero and since the concrete no longer requires hydrostatic support, the ultimate strength of the composite section is not affected, and the full steel plate is available to carry other loads as described below.

The steel plates and the concrete act as a composite section after the concrete has reached sufficient strength. The composite section resists bending moment by one face resisting tension and the other face resisting compression. The steel plate resists the tension and behaves as reinforcing steel in reinforced concrete. The composite section is underreinforced so that the steel would yield before the concrete reaches its strain limit of 0.003 in/in. As the steel faceplates are strained beyond yield to allow the composite section to attain its ultimate capacity, the modest residual bending stress from concrete placement is relieved, since the stress across the entire faceplate in tension is at yield. The small residual strain induced by the concrete placement loads is secondary and has negligible effect on the ultimate bending capacity of the composite section. The stresses in the faceplates resulting from concrete placement are therefore not combined with the stresses in the post-construction load combinations.

3.8.3.4 Analysis Procedures

This subsection describes the modelling and overall analyses of the containment internal structures, including the concrete-filled structural modules. Concrete and steel composite structures are used extensively in conventional construction. Applications include concrete slabs on steel beams and concrete-filled steel columns. Testing of concrete-filled structural modules is described in References 27 through 29 for in-plane loading and in References 30 through 33 for out-of-plane loading. The tests indicate that these composite structures behave in a manner similar to reinforced concrete structures. The initial load deflection behavior is well predicted using the gross properties of the steel and concrete. This is similar to the behavior of reinforced concrete elements where the

initial stiffness is predicted by the gross properties. As the load is increased on reinforced concrete members, cracking of the concrete occurs and the stiffness decreases. The behavior of concrete and steel composite structures is similar in its trends to reinforced concrete but has a superior performance. The results of the test program by Akiyama et al. (Reference 27)

indicate that concrete and steel composites similar to the structural modules have significant advantages over reinforced concrete elements of equivalent thickness and reinforcement ratios:

- Over 50 percent higher ultimate load carrying capacity
- Three times higher ductility
- Less stiffness degradation under peak cyclic loads, 30 percent for concrete and steel composites versus 65 percent for reinforced concrete

Methods of analysis for the structural modules are similar to the methods used for reinforced concrete. Table 3.8.3-2 summarizes the finite element analyses of the containment internal structures and identifies the purpose of each analysis and the stiffness assumptions for the concrete filled steel modules. For static loads the analyses use the monolithic (uncracked) stiffness of each concrete element. The elastic modulus is taken as 0.80 times the value calculated based on the ACI Code. This reduced elastic modulus considers a small degree of cracking as described in the seismic analyses in Subsection 3.7.2.3. For thermal and dynamic loads the analyses consider the extent of concrete cracking as described in later subsections.

Stiffnesses are established based on analyses of the behavior and review of the test data related to concrete-filled structural modules. The stiffnesses directly affect the member forces resulting from restraint of thermal growth. The in-plane shear stiffness of the module influences the fundamental horizontal natural frequencies of the containment internal structures in the nuclear island seismic analyses described in Subsection 3.7.2. The out-of-plane flexural stiffness of the module influences the local wall frequencies in the seismic and hydrodynamic analyses of the in-containment refueling water storage tank. Member forces are evaluated against the strength of the section calculated as a reinforced concrete section with zero strength assigned to the concrete in tension.

ACI 349, Section 9.5.2.3 specifies an effective moment of inertia for calculating the deflection of reinforced concrete beams. For loads less than the cracking moment, the moment of inertia is the gross (uncracked) inertia of the section. The cracking moment is specified as the moment corresponding to a maximum flexural tensile stress of $7.5\sqrt{f_c'}$. For large loads, the moment of inertia is that of the cracked section transformed to concrete. The effective moment of inertia provides a transition between these two dependent on the ratio of the cracking moment to the maximum moment in the beam at the stage the deflection is to be computed.

Table 3.8.3-1 summarizes in-plane shear and out-of-plane flexural stiffness properties of the 48-inch and 30-inch walls based on a series of different assumptions. The stiffnesses are expressed for unit length and height of each wall. The ratio of the stiffness to the stiffness of the monolithic case is also shown.

- Case 1 assumes monolithic behavior of the steel plate and uncracked concrete. This stiffness is supported by the test data described in References 27 through 33 for loading that does not cause significant cracking. This stiffness is the basis for the stiffness of the concrete-filled steel module walls in the nuclear island seismic analyses and in the uncracked case for the hydrodynamic analyses.

- Case 2 considers the full thickness of the wall as uncracked concrete. This stiffness value is shown for comparison purposes. It is applicable for loads that do not result in significant cracking of the concrete and is the basis for the stiffness of the reinforced concrete walls in the nuclear island seismic analyses (prior to the reduction in concrete stiffness by a factor of 0.8). This stiffness was used in the harmonic analyses of the internal structures described in Subsection 3.8.3.4.2.2.

- Case 3 assumes that the concrete in tension has no stiffness. For the flexural stiffness this is the conventional stiffness value used in working stress design of reinforced concrete sections. For in-plane shear stiffness, a 45-degree diagonal concrete compression strut is assumed with tensile loads carried only by the steel plate. The in-plane stiffnesses calculated by these assumptions are lower than the stiffnesses measured in the tests described in References 27 through 29 for loading that causes cracking.

3.8.3.4.1 Seismic Analyses

3.8.3.4.1.1 Finite Element Model

The structural modules are simulated within the finite element model using 3D shell elements. Equivalent shell element thickness and modulus of elasticity of the structural modules are computed as shown below. The shell element properties are computed using the combined gross concrete section and the transformed steel faceplates of the structural modules. This representation models the composite behavior of the steel and concrete. The equivalent modulus of concrete, E_m, is reduced by a factor of 0.8 to consider the effect of cracking as recommended in Table 6-5 of FEMA 356 (Reference 5 in Subsection 3.7.6). See Section 3.7 and Appendix 3G for further discussion of the containment internal structures finite element model.

- Axial and Shear Stiffnesses of module:

$$\sum EA = E_c(Lt + 2(n-1)Lt_s)$$

- Bending Stiffness of module:

$$\sum EI = E_c\left[\frac{L}{12}t^3 + 2\frac{L}{12}(n-1)\ t_s^3 + 2(n-1)Lt_s\left(\frac{t}{2}\right)^2\right]$$

where:

E_c	=	concrete modulus of elasticity
n	=	modular ratio of steel to concrete
L	=	length of wall module
t	=	thickness of wall module
t_s	=	thickness of plate on each face of wall module

These equations lead to an equivalent thickness, t_m, and modulus of elasticity of the plate elements, E_m, as shown below:

$$t_m = \left[\frac{1+3\alpha(n-1)}{1+\alpha(n-1)} \right]^{1/2} t$$

$$E_m = [1+\alpha(n-1)] \left[\frac{1+3\alpha(n-1)}{1+\alpha(n-1)} \right]^{-1/2} E_c$$

where $\alpha = 2t_s / t$ and terms of order α^3 are neglected (for a typical 30-inch thick wall with 1/2-inch steel plates, $\alpha = 0.033$).

3.8.3.4.1.2 Stiffness Assumptions for Local Seismic Analyses of In-Containment Refueling Water Storage Tank

The seismic analyses of the in-containment refueling water storage tank address the local response of the walls and water and are performed to verify the structural design of the tank. The local analyses performed uses the cracked section stiffness values based on composite behavior with zero stiffness for the concrete in tension (Case 3 of Table 3.8.3-1). The local analyses use the finite element model described in Subsection 3.8.3.4.2.2. Response spectrum analyses are performed using the floor response spectra at the base of the tank.

3.8.3.4.1.3 Damping of Structural Modules

Damping of the structural modules is reported in Reference 27 based on the cyclic load tests of a containment internal structure model. The equivalent viscous damping at the design load level was 5 percent for the concrete-filled steel model. This was almost constant up to the load level at which the steel plate started yielding. Dynamic analyses are performed using 7 percent damping for the reinforced concrete and 5 percent for the structural modules as shown in Subsection 3.7.1.

3.8.3.4.2 Hydrodynamic Analyses

This subsection describes the hydrodynamic analyses performed for the AP600 which demonstrated that design of the walls of the in-containment refueling water storage tank for 5 psi as described in Subsection 3.8.3.3.1 would bound the loads from the time history transient analysis. The analyses were performed using the AP600 test results. The peak values from these tests are also applicable to the AP1000 since they occur at the beginning of the transient, and the automatic depressurization system and the initial conditions are the same for the two plant designs (Reference 52). The structural configuration of the tank is identical. The minor differences in the height of the steam generator and pressurizer compartment walls and in the mass of the steam generators and pressurizer will have only a minor effect on the significant structural frequencies. Since the time histories applied in the AP600 analyses cover a broad range of frequencies the response of the AP1000 tank boundary will be similar to that of the AP600. The 5 psi pressure design basis for the tank boundary is therefore also applicable to the AP1000.

Hydrodynamic analyses were performed for the AP600 for automatic depressurization system discharge into the in-containment refueling water storage tank. This discharge is designated as ADS_1 in the load description of Subsection 3.8.3.3.1 and results in higher hydrodynamic loading than the ADS discharge into a hot tank in ADS_2. The first three stages of the automatic depressurization system valves discharge into the tank through spargers under water, producing hydrodynamic loads on the tank walls and equipment. Hydrodynamic loads, measured in hydraulic tests of the automatic depressurization system sparger in a test tank, are evaluated using the source load approach (Reference 34). Analyses of the tests define source pressure loads that are then used in analyses of

the in-containment refueling water storage tank to give the dynamic responses of the containment internal structures. The basic analysis approach consists of the following steps:

1. A pressure source, an impulsive forcing function at the sparger discharge, is selected from the tests using a coupled fluid structure finite element model of the test tank, taking into account fluid compressibility effects. This source development procedure is based on a comparison between analysis and test results, both near the sparger exit and at the boundaries of the test tank.

2. The pressure source is applied at each sparger location in a coupled fluid structure finite element model of the in-containment refueling water storage tank structure and of the contained water. The mesh characteristics of the model at the sparger locations and the applied forcing functions correspond to those of the test tank analysis.

3.8.3.4.2.1 Sparger Source Term Evaluation

A series of tests was conducted with discharge conditions representative of one sparger for the AP600 (References 35 and 36). Pressure traces measured during the test discharges were investigated, at both sparger exit and tank boundaries to (1) bound the expected discharge from the automatic depressurization system; (2) characterize the pressure wave transmission through the pool water; (3) determine the maximum pressure amplitudes and the frequency content; and (4) produce reference data for qualification of the analytical procedure. Pressure time histories and power spectrum densities were examined at reference sensors, both for the total duration of the discharge transient (about 50 seconds) and for critical time intervals.

Fluid-structure interaction analyses were performed with the ANSYS computer code (Reference 37). The mathematical model consists of a 3D sector finite element model, 15 degrees wide, as shown in Figure 3.8.3-9. It uses STIF30 fluid and STIF63 structural ANSYS finite elements, which take into account fluid compressibility and fluid-structure interaction. Rayleigh damping of 4 percent is used for the concrete structure, and fluid damping is neglected. Direct step-by-step time integration is used. The measured discharge pressures for single time intervals are imposed as uniform forcing functions on the idealized spherical surface of the steam/water interface, and pressures transmitted through the water to the tank boundary are calculated and compared with test measurements. The analyses of the test tank showed satisfactory agreement for the pressures at the tank boundary.

The examination of test results related to the structural design of the in-containment refueling water storage tank under automatic depressurization system hydrodynamic excitation and the comparison with the analytical procedure previously described, lead to the following conclusions regarding the sparger source term definition:

- The automatic depressurization system discharge into cold water produces the highest hydrodynamic pressures. The tests at higher water temperatures produce significantly lower pressures.

- Two pressure time histories, characterized by different shapes and frequency content, can be selected as representative of the sparger discharge pressures; they are assumed as acting on a spherical bubble centered on the sparger centerline and enveloping the ends of the sparger arms.

- The application of such time histories as forcing functions to an analytical model, simulating the fluid structure interaction effects in the test tank, has been found to predict the measured tank wall pressures, for the two selected reference time intervals.

- The two defined sparger source term pressure time histories can be used as forcing functions for global hydrodynamic analyses of the in-containment refueling water storage tank by developing a comprehensive fluid-structure finite element model and reproducing the test tank mesh pattern in the sparger region.

- The hydrodynamic loads on the vessel head support columns and ADS sparger piping located in the IRWST are developed from the forcing functions using the methodology documented in Reference 51.

3.8.3.4.2.2 In-Containment Refueling Water Storage Tank Analyses

The in-containment refueling water storage tank is constructed as an integral part of the containment internal structures as described in Subsection 3.8.3.1.3. It contains two depressurization spargers that are submerged approximately 9 feet below the normal water level. Transmission of the hydrodynamic pressures from the sparger discharge to the wetted in-containment refueling water storage tank is evaluated using the coupled fluid-structure interaction method similar to that described for the test tank analysis in the previous subsection.

The 3D ANSYS finite element model includes the in-containment refueling water storage tank boundary, the water within the in-containment refueling water storage tank, the adjacent structural walls of the containment internal structures, and the operating floor. The model of the in-containment refueling water storage tank, shown in Figures 3.8.3-10 (sheet 2), 3.8.3-11, and 3.8.3-12, represents the outer steel structures, the inner concrete walls, and the water. The model of the adjacent structural walls and floors is shown in Figure 3.8.3-10 (sheet 1). The flexible steel outer wall is represented using beam and shell elements; isotropic plate elements are used to represent the inner structural module walls. The water is modelled as a compressible fluid to provide an acoustic medium to transmit the source pressure. The model has two bubble boundaries representing the spargers. Pressure loads are applied to the solid element faces adjacent to the air bubbles. The forcing functions at the sparger locations are conservatively assumed to be in phase. Rayleigh damping of 5 percent is used for the concrete-filled structural modules and fluid damping is neglected. All degrees of freedom were retained in the step-by-step direct integration solution procedure for the in-containment refueling water storage tank boundary and the water. Degrees of freedom in the adjacent walls and floor were condensed by Guyon reduction.

Significant structural frequencies of the AP600 containment internal structures were analyzed using the harmonic response option with the ANSYS model of the in-containment refueling water storage tank and containment internal structures. A harmonic unit pressure is applied at the surface of the spherical bubble representing the automatic depressurization system spargers. Material properties for the concrete elements are based on the uncracked gross concrete section (Case 2 of Table 3.8.3-1). The results of these harmonic response analyses show the response deflection as a function of input frequency at nodes in the containment internal structures. The harmonic response analyses show that the largest responses are close to the wetted boundary of the in-containment refueling water storage tank and that the significant frequencies are from 18 to 50 hertz.

Two time histories are identified for the structural hydrodynamic analyses; one has significant frequencies below 40 hertz while the other has significant frequencies in the range of 40 to 60 hertz. Both time history inputs are used in the hydrodynamic analyses with the monolithic uncracked section properties for all walls. The lower frequency input is also applied in lower bound analyses using the cracked section stiffness values (Case 3 of Table 3.8.3-1) for the concrete walls that are boundaries of the in-containment refueling water storage tank. Monolithic properties are used for the other walls. Results from these cases are enveloped, thereby accounting for variabilities in the structural analyses.

The analyses of the AP600 in-containment refueling water storage tank give wall pressures, displacements, accelerations, hydrodynamic floor response spectra, and member forces due to the automatic depressurization system discharge pressure forcing functions. Consideration of pressure wave transmission and fluid-structure interaction shows a significant wall pressure attenuation with distance from the sparger region and with increasing wall flexibilities, relative to the measured sparger pressure forcing function. The member stresses are evaluated against the allowable stresses specified in Subsection 3.8.3.5 for seismic Category I structures, considering the hydrodynamic loads as live loads. The analyses show that the member forces in the walls of the in-containment refueling water storage tank are bounded by a case with a uniform pressure of 5 psi applied to the walls.

3.8.3.4.3 Thermal Analyses

The in-containment refueling water storage tank water and containment atmosphere are subject to temperature transients as described in Subsection 3.8.3.3.1. The temperature transients result in a nonlinear temperature distribution within the wall modules. Temperatures within the concrete wall are calculated in a unidimensional heat flow analysis. The average and equivalent linear gradients are applied to a finite element model of the containment internal structures at selected times during the transient. The effect of concrete cracking is considered in the stiffness properties for the concrete elements subjected to the thermal transient. The finite element model is that described in Subsection 3.8.3.4.2.2 except that the model of the water in the IRWST is not needed.

The structural modules are subject to a rapid temperature transient in the event of a loss-of-coolant accident (LOCA) or a main steam line break (MSLB). The structural modules were evaluated for these rapid temperature transients. The evaluation considered both carbon and stainless steel faceplates. The steel plate heats up most rapidly in the LOCA event with temperatures up to 270°F in the first few minutes for an ambient initial temperature of 50°F. The faceplate of the structural module will see differential temperatures of 220°F relative to the concrete. The concrete heats up more slowly and does not see a significant temperature increase during the early part of the transient. There is relative thermal growth of the faceplate, causing shear loads in the shear studs, and embedded angles of the structural steel trusses that are welded to the faceplate. The heatup of the surface plates during the initial portion of the LOCA transient results in cracking of the concrete walls except in regions where there is significant external restraint. The structural module maintains its integrity throughout the rapid thermal transient.

Thermal transients for the design basis accidents are described in Section 6.3. The analyses for these transients are similar to those described above.

3.8.3.5 Design Procedures and Acceptance Criteria

The containment internal structures that contain reinforcing steel including most of the areas below elevation 98′, are designed by the strength method, as specified in the ACI Code Requirements for Nuclear Safety Related Structures, ACI-349. This code includes ductility criteria for use in detailing, placing, anchoring, and splicing of the reinforcing steel.

The internal steel framing is designed according to the AISC Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, AISC-N690, supplemented by the requirements given in Subsection 3.8.4.5

[*The design and construction of anchors and embedments conform to the procedures and standards of Appendix B to ACI 349-01*]* and are in conformance with the regulatory positions of NRC Regulatory Guide 1.199, Revision 0. [*Alternative requirements to ACI 349, Appendix B apply to the anchoring of headed shear reinforcement for the basemat (see Subsection 3.8.4.4.1). Alternative*

*NRC Staff approval is required prior to implementing a change in this information.

requirements to ACI 349, Appendix B apply to the anchoring of headed reinforcement above the basemat (see Subsection 3.8.4.4.1).]*

The secondary shield walls, in-containment refueling water storage tank, refueling cavity, and operating floor above the in-containment refueling water storage tank are designed using structural modules. Concrete-filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349, as supplemented in the following paragraphs. Structural floor modules are designed as composite structures in accordance with AISC-N690. Methods of analysis used are based on accepted principles of structural mechanics and are consistent with the geometry and boundary conditions of the structures.

The methods described in Subsection 3.7.2 are employed to obtain the safe shutdown earthquake loads at various locations in the containment internal structures. The safe shutdown earthquake loads are derived from the equivalent static analysis of a three-dimensional, finite element model representing the entire containment internal structures.

The determination of pressure and temperature loads due to pipe breaks is described in Subsections 3.6.1 and 6.2.1.2. Subcompartments inside containment containing high energy piping are designed for a pressurization load of 5 psi. The pipe tunnel in the CVS room (room 11209, Figure 1.2-6) is designed for a pressurization load of 7.5 psi. These subcompartment design pressures bound the pressurization effects due to postulated breaks in high energy pipe. The design for the effects of postulated pipe breaks is performed as described in Subsection 3.6.2. Determination of pressure loads resulting from actuation of the automatic depressurization system is described in Subsection 3.8.3.3.1.

Determination of reactor coolant loop support loads is described in Subsection 3.9.3. Design of the reactor coolant loop supports within the jurisdiction of ASME Code, Section III, Division 1, Subsection NF is described in Subsections 3.9.3 and 5.4.10.

Computer codes used are general purpose codes. The code development, verification, validation, configuration control, and error reporting and resolution are according to the Quality Assurance requirements of Chapter 17.

3.8.3.5.1 Reactor Coolant Loop Supports

3.8.3.5.1.1 Reactor Vessel Support System

The reactor vessel supports are described in Subsection 3.8.3.1.1.1. Figure 3.8.3-4 shows the embedments and anchor bolts for the reactor vessel supports. The embedded portions of the reactor vessel supports, which are outside the ASME jurisdictional boundary, are designed by elastic methods of analysis. Note the dowel studs connecting the support to the steel embedment are not embedded in the concrete and are within the ASME jurisdictional boundary. The embedded portions, including the anchor bolts, are analyzed and designed to resist the applicable loads and load combinations given in Subsection 3.8.4.3. The design is according to AISC-N690 and ACI-349. The design of the steel to resist the loads satisfies the AISC-N690 requirements. The design and evaluation to assure that the embedment transfers the loads to the concrete and is retained in the concrete satisfies the requirements of ACI 349.

3.8.3.5.1.2 Steam Generator Support System

The embedded portions of the steam generator supports, which are outside the ASME jurisdictional boundary, are designed by elastic methods of analysis. They are analyzed and designed to resist the applicable loads and load combinations given in Subsection 3.8.4.3. The design is according to AISC-N690 and ACI-349. Figure 3.8.3-5 shows the jurisdictional boundaries.

*NRC Staff approval is required prior to implementing a change in this information.

3.8.3.5.1.3 Reactor Coolant Pump Support System

The reactor coolant pumps are integrated into the steam generator channel head and consequently do not have a separate support system.

3.8.3.5.1.4 Pressurizer Support System

The embedded portions of the pressurizer supports, which are outside the ASME jurisdictional boundary, are designed by elastic methods of analysis. They are analyzed and designed to resist the applicable loads and load combinations given in Subsection 3.8.4.3. The design is according to AISC-N690 and ACI-349. Figure 3.8.3-6 shows the jurisdictional boundaries.

3.8.3.5.2 Containment Internal Structures Basemat

The containment internal structures basemat including the primary shield wall and reactor cavity are designed for dead, live, thermal, pressure, and safe shutdown earthquake loads. The structural modules are designed as described in Subsection 3.8.3.5.3.

The reinforced concrete forming the base of the containment internal structures is designed according to ACI 349.

3.8.3.5.3 Structural Wall Modules

Structural modules without concrete fill, such as the west wall of the in-containment refueling water storage tank, are designed as steel structures, according to the requirements of AISC-N690. This code is applicable since the module is constructed entirely out of structural steel plates and shapes. In local areas stresses due to restraint of thermal growth may exceed yield and the allowable stress intensity is 3 S_{m1}. This allowable is based on the allowable stress intensity for Service Level A loads given in ASME Code, Section III, Subsection NE, Paragraph NE-3221.4.

The concrete-filled steel module walls are designed for dead, live, thermal, pressure, safe shutdown earthquake, and loads due to postulated pipe breaks. The in-containment refueling water storage tank walls are also designed for the hydrostatic head due to the water in the tank and the hydrodynamic pressure effects of the water due to the safe shutdown earthquake, and automatic depressurization system pressure loads due to sparger operation. The walls of the refueling cavity are also designed for the hydrostatic head due to the water in the refueling cavity and the hydrodynamic pressure effects of the water due to the safe shutdown earthquake.

Figure 3.8.3-8 shows the typical design details of the structural modules, typical configuration of the wall modules, typical anchorages of the wall modules to the reinforced base concrete, and connections between adjacent modules. Variations in the module design details and design elements used are described in the Figure 3.8.3-8 notes and in Subsections 3.8.3.1 and 3.8.3.1.3. Concrete-filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI-349, as supplemented in the following paragraphs. The faceplates are considered as the reinforcing steel, bonded to the concrete by headed studs. The application of ACI-349 and the supplemental requirements are supported by the behavior studies described in Subsection 3.8.3.4.1. The steel plate modules are anchored to the reinforced concrete basemat by mechanical connections welded to the steel plate. Loads are transferred directly from the faceplates to the base concrete using reinforcing bars, structural shapes, and shear studs connected directly to the modules with mechanical connectors or welds. The reinforcing steel used to anchor the modules to the concrete has a development that satisfies the requirements of ACI 349. The design of the surface plate, base plate, and vertical stiffeners is checked by finite element analysis. The design of critical sections is described in Subsection 3.8.3.5.8.

3.8.3.5.3.1 Design for Axial Loads and Bending

Design for axial load (tension and compression), in-plane bending, and out-of-plane bending is in accordance with the requirements of ACI-349, Chapters 10 and 14.

3.8.3.5.3.2 Design for In-Plane Shear

Design for in-plane shear is in accordance with the requirements of ACI-349, Chapters 11 and 14. The steel faceplates are treated as reinforcing steel, contributing as provided in Section 11.10 of ACI-349.

3.8.3.5.3.3 Design for Out-of-Plane Shear

Design for out-of-plane shear is in accordance with the requirements of ACI-349, Chapter 11.

3.8.3.5.3.4 Evaluation for Thermal Loads

The effect of thermal loads on the structural wall modules, with and without concrete fill, is evaluated by using the working stress design method for load combination 3 of Tables 3.8.4-1 and 3.8.4-2. This evaluation is in addition to the evaluation using the working stress design method of AISC N690 or the strength design method of ACI-349 for the load combinations without the thermal load. Acceptance for the load combination with normal thermal loads, which includes the thermal transients described in Subsection 3.8.3.3.1, is that the stress in general areas of the steel plate be less than yield. In local areas where the stress may exceed yield the total stress intensity range is less than twice yield. This evaluation of thermal loads is based on the ASME Code philosophy for Service Level A loads given in ASME Code, Section III, Subsection NE, Paragraphs NE-3213.13 and 3221.4.

3.8.3.5.3.5 Design of Trusses

The trusses provide a structural framework for the modules, maintain the separation between the faceplates, support the modules during transportation and erection, and act as "form ties" between the faceplates when concrete is being placed. After the concrete has cured, the trusses act with shear studs to provide interfacial shear transfer between the faceplates and concrete. The trusses also provide structural integrity and permit the development of the nominal out-of-plane shear strength of the concrete without shear reinforcement. In areas of a module wall where the hooked dowels from a floor slab do not overlap the hooked dowels of an opposing floor slab on the opposite face of the wall, the horizontal channel member of the trusses may be credited as a tension tie, by confinement of the faceplates, preventing separation of the concrete. The welded connection of the truss to the faceplate is sized to fully develop the specified minimum yield strength of the channel member. As described in Subsection 3.8.3.1.3, additional structural capacity is provided between the module faceplates in some areas to support localized loads. The trusses, connecting steel plates, structural shapes, reinforcement bars, and tie bars are designed according to the applicable requirements of AISC N690 and ACI 349.

3.8.3.5.3.6 Design of Shear Studs

The wall structural modules are designed as reinforced concrete elements, with the faceplates serving as reinforcing steel. Since the faceplates do not have deformation patterns typical of reinforcing steel, shear studs are provided to transfer the forces between the concrete and the steel faceplates. The shear studs make the concrete and steel faceplates behave compositely. In addition, the shear studs permit anchorage for piping and other items attached to the walls.

The design of the shear stud size and spacing includes consideration of shear transfer and faceplate buckling. Shear transfer is evaluated in accordance with AISC N690 Sections Q1.11.1 and Q1.11.4. Faceplate buckling is evaluated in accordance with AISC-N690 Section Q1.5.1.3. Conformance with these requirements precludes the possible failure modes for structural module construction for design basis loads.

3.8.3.5.4 Structural Floor Modules

Figure 3.8.3-3 shows the design details for a representative example of the floor modules. The details shown in Figure 3.8.3-3 are the representative design for a specific portion of the operating deck floor in containment. The size, material, and configuration of the structural shapes and plates, the amount and arrangement of the reinforcement, and the type and size of the welds may vary in the final design of floor modules in this and other locations. The operating floor is designed for dead, live, thermal, safe shutdown earthquake, and pressure due to automatic depressurization system operation or due to postulated pipe break loads. The operating floor region above the in-containment refueling water storage tank is a series of structural modules. The remaining floor is designed as a composite structure of concrete slab and steel beams in accordance with AISC-N690.

For vertical downward loads, the floor modules are designed as a composite section, according to the requirements of Section Q1.11 of AISC-N690. Composite action of the steel section and concrete fill is assumed based on meeting the intent of Section Q1.11.1 for beams totally encased in concrete. Although the bottom flange of the steel section is not encased within concrete, the design configuration of the floor module provides complete concrete confinement to prevent spalling. It also provides a better natural bonding than the code-required configuration.

For vertical upward loads, no credit is taken for composite action. The steel members are relied upon to provide load-carrying capacity. Concrete, together with the embedded angle stiffeners, is assumed to provide stability to the plates.

Floor modules are designed using the following basic assumptions and related requirements:

- Concrete provides restraint against buckling of steel plates. The buckling unbraced length of the steel plate, therefore, is assumed to equal the span length between the fully embedded steel plates and shapes.
- Although the floor modules forming the top (ceiling) of the in-containment refueling water storage tank are not in contact with water, stainless steel plates are used for the tank boundary.
- The floor modules are designed as simply supported beams.

3.8.3.5.4.1 Design for Vertical Downward Loads

The floor modules are designed as a one-way composite concrete slab and steel beam system in supporting the vertical downward loads. The effective width of the concrete slab is determined according to Section Q1.11.1 of AISC-N690. The effective concrete compression area is extended to the neutral axis of the composite section. The concrete compression area is treated as an equivalent steel area based on the modular ratio between steel and concrete material. Figure 3.8.3-13 shows the effective composite sections. The steel section is proportioned to support the dead load and construction loads existing prior to hardening of the concrete. The allowable stresses are provided in Table 3.8.4-1.

3.8.3.5.4.2 Design for Vertical Upward Loads

For vertical upward loads, the floor modules are designed as noncomposite steel structures. The effective width, b_e, of the faceplate in compression is based on post-buckling strength of steel plates and is determined from Equation (4.16) of Reference 44. The faceplates of the structural floor modules are stiffened and supported by embedded horizontal angles. Hence, the buckling unbraced length of the faceplates is equal to the span length between the horizontal angles. Since concrete provides restraint against buckling of the steel plates, a value of 0.65 is used for k when calculating the effective length of the steel plates and stiffeners whenever the plate or stiffener is continuous. The buckling stress, f_{cr}, of the faceplates is determined from Sections 9.2 and 9.3 of Reference 45. The effective width of the faceplates of the structural floor modules in compression is shown in Figure 3.8.3-13. The allowable stresses are provided in Table 3.8.4-1.

3.8.3.5.4.3 Design for In-Plane Loads

In-plane shear loads acting on the floor modules are assumed to be resisted only by the steel faceplate without reliance on the concrete for strength. The stresses in the faceplate due to the in-plane loads are combined with those due to vertical loads. The critical stress locations of the floor faceplate are evaluated for the combined normal and shear stress, based on the von Mises yield criterion:

For the particular case of a two-dimension stress condition the equation is:

$$(\sigma_1)^2 - \sigma_1\sigma_2 + (\sigma_2)^2 = (f_y)^2$$

where σ_1 and σ_2 are the principal stresses and f_y is the uniaxial yield stress.

For the faceplate where normal, σ, and shear, τ, stresses are calculated, the principal stresses can be expressed as follows:

$$\sigma_1 = \left(\frac{\sigma}{2}\right) + \sqrt{\frac{\sigma^2}{4} + \tau^2}$$

$$\sigma_2 = \left(\frac{\sigma}{2}\right) - \sqrt{\frac{\sigma^2}{4} + \tau^2}$$

Therefore, the condition at yield becomes:

$$\sigma^2 + 3\tau^2 = (f_y)^2$$

For the design of the structural floor module faceplate, the allowable stresses for the various loading conditions are as follows:

Normal condition:

$$\sigma^2 + 3\tau^2 \leq (0.6\ f_y)^2$$

Severe condition:

$$\sigma^2 + 3\tau^2 \leq (0.6\ f_y)^2$$

Extreme/abnormal condition:

$$\sigma^2 + 3\tau^2 \leq (0.96\ f_y)^2$$

Thermal stresses in the faceplates result from restraint of growth during the thermal transients described in Subsection 3.8.3.3.1. Evaluation for thermal stresses is the same as discussed in Subsection 3.8.3.5.3.4 for the wall modules.

3.8.3.5.5 Internal Steel Framing

Internal steel framing is analyzed and designed according to AISC-N690. Seismic analysis methods are described in Subsection 3.7.3.

3.8.3.5.6 Steel Form Modules

The steel form modules consist of plate reinforced with angle stiffeners and tee sections as shown in Figure 3.8.3-16. The steel form modules are designed for concrete placement loads defined in Subsection 3.8.3.3.2.

The steel form modules are designed as steel structures according to the requirements of AISC-N690. This code is applicable since the form modules are constructed entirely out of structural steel plates and shapes and the applied loads are resisted by the steel elements.

3.8.3.5.7 Design Summary Report

A design summary report is prepared for containment internal structures documenting that the structures meet the acceptance criteria specified in Subsection 3.8.3.5.

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 Section 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 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 judgement to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

3.8.3.5.8 Design Summary of Critical Sections

3.8.3.5.8.1 Structural Wall Modules

[*This subsection summarizes the design of the following critical sections:*

- *Southwest wall of the refueling cavity (4′0″ thick)*
- *South wall of west steam generator compartment (2′6″ thick)*
- *North east wall of in-containment refueling water storage tank (2′6″ thick)*

*NRC Staff approval is required prior to implementing a change in this information.

The thicknesses and locations of these walls which are part of the boundary of the in-containment refueling water storage tank are shown in Figure 3.8.3-18. The structural configuration and typical details are shown in Figures 3.8.3-1, 3.8.3-2, 3.8.3-8, 3.8.3-14, 3.8.3-15, and 3.8.3-17. The details shown in Figure 3.8.3-17 are representative of connections between floors in containment and walls constructed using steel plate concrete composite construction. Plate thickness, structural shape size, and reinforcement provided may be increased locally. The design implemented in fabrication and construction drawings and instructions may have alternative structural shapes or reinforcement arrangements if they provide equal or better load capacity. As described in Subsection 3.8.3.1.3, steel plates, structural shapes, reinforcement bars, or tie bars are used in other locations between the structural module faceplates to address local out-of-plane loads.]* The structural analyses are described in Subsection 3.8.3.4 and summarized in Table 3.8.3-2. The design procedures are described in Subsection 3.8.3.5.3.

[*The three walls extend from the floor of the in-containment refueling water storage tank at elevation 103′ 0″ to the operating floor at elevation 135′ 3″. The south west wall is also a boundary of the refueling cavity and has stainless steel plate on both faces. The other walls have stainless steel on one face and carbon steel on the other.*]* For each wall design information is summarized in Tables 3.8.3-3, 3.8.3-4, 3.8.3-5, and 3.8.3-6. [*Results are shown at the middle of the wall (mid span at mid height), at the base of the wall at its mid point (mid span at base) and at the base of the wall at the end experiencing greater demand (corner at base). The first part of each table shows the member forces due to individual loading. The lower part of the table shows governing load combinations. The steel plate thickness required to resist mechanical loads is shown at the bottom of the table as well as the thickness provided. The maximum principal stress for the load combination including thermal is also tabulated. If this value exceeds the yield stress at temperature a supplemental evaluation is performed*]* as described in Subsection 3.8.3.5.3.4; [*for these cases the maximum stress intensity range is shown together with the allowable stress intensity range which is twice the yield stress at temperature.*]*

3.8.3.5.8.2 In-Containment Refueling Water Storage Tank Steel Wall

[*The in-containment refueling water storage tank steel wall is the circular boundary of the in-containment refueling water storage tank. The structural configuration and typical details are shown in sheet 3 of Figure 3.8.3-8.*]* The structural analyses are described in Subsection 3.8.3.4 and summarized in Table 3.8.3-2. The design procedures are described in Subsection 3.8.3.5.3. [*The steel wall extends from the floor of the in-containment refueling water storage tank at elevation 103′ 0″ to the operating floor at elevation 135′ 3″. The wall is a 5/8″ thick stainless steel plate. It has internal vertical stainless steel T-section columns spaced 4′-8″ apart and external hoop carbon steel (L-section) angles spaced 18″ to 24″ apart. The wall is fixed to the adjacent modules and floor except for the top of columns which are free to slide radially and to rotate around the hoop direction.*

The wall is evaluated as vertical and horizontal beams. The vertical beams comprise the T-section columns plus the effective width of the plate. The horizontal beams comprise the L-section angles plus the effective width of the plate. Table 3.8.3-7 shows the ratio of the design stresses to the allowable stresses. When thermal effects result in stresses above yield, the evaluation is in accordance with the supplemental criteria]* as described in Subsection 3.8.3.5.3.4.

3.8.3.5.8.3 Column Supporting Operating Floor

[*This subsection summarizes the design of the most heavily loaded column in the containment internal structures. The column extends from elevation 107′-2″ to the underside of the operating floor at elevation 135′-3″. In addition to supporting the operating floor, it also supports a steel grating floor at elevation 118′-0″.*

*NRC Staff approval is required prior to implementing a change in this information.

The load combinations in Table 3.8.4-1 were used to assess the adequacy of the column. For mechanical load combinations, the maximum interaction factor due to biaxial bending and axial load is 0.59. For load combinations with thermal loads, the maximum interaction factor is 0.94. Since the interaction factors are less than 1, the column is adequate for all the applied loads.]*

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

Subsection 3.8.4.6 describes the materials and quality control program used in the construction of the containment internal structures.

[*The structural steel modules are constructed using carbon steel plates and shapes (ASTM A36, ASTM A992, or steel with equal or better material properties*). *Duplex 2101 (American Society for Testing and Materials A240, designation S32101) stainless steel plates or steel with equal or better material properties*]* are used on the surfaces of the modules in contact with water during normal operation or refueling. [*The material of construction for studs attached to the module plates used to transfer loads into the concrete is A-108 or steel with equal or better material properties*. *Bars used to anchor the modules in the concrete are deformed bars according to Reference 19, Grade 60, and Reference 20.*]*

The structural wall and floor modules are fabricated and erected in accordance with AISC-N690. Loads during fabrication and erection due to handling and shipping are considered as normal loads as described in Subsection 3.8.4.3.1.1. Packaging, shipping, receiving, storage and handling of structural modules are in accordance with NQA-1, Subpart 2.2 (formerly ANSI/ASME N45.2.2 as specified in AISC N690).

3.8.3.6.1 Fabrication, Erection, and Construction of Structural Modules

Modular construction techniques are used extensively in the containment internal structures (Figure 3.8.3-1). Subassemblies, sized for commercial rail shipment, are assembled offsite and transported to the site. Onsite fabrication consists of combining the subassemblies in structural modules, which are then installed in the plant. A typical modular construction technique is described in the following paragraphs for Module CA01, which is the main structural module in the containment internal structures.

The CA01 module is a multicompartmented structure which, in its final form, comprises the central walls of the containment internal structures. The vertical walls of the module house the refueling cavity, the reactor vessel compartment, and the two steam generator compartments. The module (Figure 3.8.3-14) is in the form of a "T" and is approximately 88 feet long, 95 feet wide and 86 feet high. The module is assembled from about 40 prefabricated wall sections called structural submodules (Figure 3.8.3-15). The submodules are designed for railroad transport from the fabricator's shop to the plant site with sizes up to 12 feet by 12 feet by 80 feet long, weighing up to 80 tons. A typical submodule weighs between 9 and 11 tons. The submodules are assembled outside the nuclear island with full penetration welds between the faceplates of adjacent subunits.

The completed CA01 module is lifted to its final location within the containment vessel by the heavy lift construction crane. Following placement of the CA01 module within the containment building, the hollow wall structures are filled with concrete, forming a portion of the structural walls of the containment internal structures.

Tolerances for fabrication, assembly and erection of the structural modules conform to the requirements of section 4 of ACI-117, applicable sections of AWS D1.1, and sections Q1.23 and Q1.25 of AISC-N690. Tolerances for shear stud spacing requirements are identified on Figure 3.8.3-8, Sheet 1 and conform to AWS D1.1, Paragraph 7.4.5.

*NRC Staff approval is required prior to implementing a change in this information.

3.8.3.6.2 Nondestructive Examination

Nondestructive examination of the submodules and module is performed according to AISC-N690 and AWS D 1.1. Welds are visually examined for 100 percent of their length. Full penetration welds are inspected by ultrasonic or radiographic examination for 10 percent of their length, or for 100 percent of the length of 1 weld in 10. Partial penetration welds are inspected by magnetic particle or liquid penetrant examination for 10 percent of their length, or for 100 percent of the length of 1 weld in 10.

3.8.3.6.3 Concrete Placement

After installation of the CA01 module in the containment, the hollow walls are filled with concrete. The concrete is placed through multiple delivery trunks located along the top of the wall or through windows in the module walls or pumping ports built into the module wall. It is placed in incremental layers with the placement rate based on the pressure of the wet concrete and its setting time. During concrete placement, workers and inspectors have access to the inside of the modules. The arrangement of the module internal trusses provides communication to aid in the free flow of concrete and movement of personnel.

3.8.3.7 In-Service Testing and Inspection Requirements

The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.3.8 Construction Inspection

Construction inspection is conducted to verify the concrete wall thickness and the surface plate thickness. The location for measurement of the structural wall modules is at the locations of the trusses used to provide the structural framework for the modules. Inspections will be measured at applicable sections excluding designed openings or penetrations. Inspections will confirm that each section provides the minimum required steel and concrete thicknesses as shown in Table 3.8.3-3. The minimum required steel and concrete thicknesses represent the minimum values to meet the design basis loads. Table 3.8.3-3 also indicates the steel plate thickness provided which may exceed the minimum required value for the following reasons:

- Structural margin
- Ease of construction
- Construction loads
- Use of standard thicknesses

[*As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.*]*

3.8.4 Other Category I Structures

The other seismic Category I structures are the shield building and the auxiliary building. New fuel and spent fuel racks are described in Section 9.1.

General criteria in this section describing the loads, load combinations, materials, and quality control are also applicable to the containment internal structures described in Subsection 3.8.3.

*NRC Staff approval is required prior to implementing a change in this information.

3.8.4.1 Description of the Structures

3.8.4.1.1 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.

A non-linear analyses performed on the nuclear island finite element model is used to validate the stiffness used in the evaluation of concrete-filled steel modules. This analysis used benchmarked layered shell elements for the reinforced concrete/steel concrete composite (RC/SC) connection, which transferred all stresses to the steel in the connections once a maximum yield stress in concrete was reached. These elements were included in the comparison made between linear and non-linear models. Results show the 80-percent stiffness model response spectra enveloped the non-linear model and provide a conservative approach in terms of response spectra and maximum stresses obtained in the shield building wall.

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 water storage (PCS) tank

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 shown in Figures 1, 2, 3, and 4 of APP-GW-GLR-602 (Reference 57).*]* 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. A typical auxiliary building RC roof connection to the shield building SC wall is shown in [*Figure 7 of APP-GW-GLR-602 (Reference 57).*]*

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 structure. The section that is not protected by the auxiliary building is a steel concrete 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 6000 psi. The SC

*NRC Staff approval is required prior to implementing a change in this information.

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 and [*Figure 5 of APP-GW-GLR-602 (Reference 57).*]*

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 tension ring and air inlets are shown in Figure 3H.5-14.

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.

*NRC Staff approval is required prior to implementing a change in this information.

Appendix 3H provides additional information regarding the shield building design, testing, and analysis that demonstrates the robust behavior of the design with respect to ductility, effects of creep, reserve strength of seismic margins, buckling, and shrinkage.

3.8.4.1.2 Auxiliary Building

The auxiliary building is a reinforced concrete and structural steel structure. Three floors are above grade and two are located below grade. It 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 is a C-shaped section of the nuclear island that wraps around approximately 50 percent of the circumference of the shield building. The floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

The figures in Section 1.2 show the layout of the auxiliary building and its interface with the other buildings of the nuclear island. The following are the significant features and the principal systems and components of the auxiliary building:

- Main control room
- Remote shutdown room
- Class 1E dc switchgear
- Class 1E batteries
- Reactor trip switchgear
- Reactor coolant pump trip switchgear
- Main steam and feedwater piping
- Main control room heating, ventilating, and air conditioning (HVAC)
- Class 1E switchgear rooms heating, ventilating, and air conditioning
- Spent fuel pool
- Fuel transfer canal
- Cask loading and washdown pits
- New fuel storage area
- Cask handling crane
- Fuel handling machine
- Chemical and volume control system (CVS) makeup pumps
- Normal residual heat removal system (RNS) pumps and heat exchangers
- Liquid radwaste tanks and components
- Spent fuel cooling system
- Gaseous radwaste processing system
- Mechanical and electrical containment penetrations

Structural modules are used for part of the south side of the auxiliary building. These structural modules are structural elements built up with welded steel structural shapes and plates. Concrete is used where required for shielding, but reinforcing steel is not normally used. These modules include the spent fuel pool, fuel transfer canal, and cask loading and cask washdown pits.

The configuration of the auxiliary building structural modules is similar to the structural modules described in Subsection 3.8.3.1 for the containment internal structures. As in the containment internal structures, the structural wall modules consist of steel faceplates connected by steel trusses with shear studs welded to the inside faces of the steel faceplates. Figure 3.8.4-4 shows the location of the structural modules in the auxiliary building. The structural modules extend from elevation 66′-6″ to elevation 135′-3″.

The auxiliary building modules are different in configuration and design details compared to the containment internal structures. The following examples are provided. The thickness of the structural

wall modules differs from the figures for the containment internal structures modules and ranges from 2′-6″ to 5′-0″ for the key structural walls. The auxiliary building modules also includes smaller structural walls that act as labyrinth walls or barrier access walls with thicknesses less than 2′-6″. Local variation in the design of the trusses and spacing of the trusses and shear studs may be required to address internal obstructions and accessibility for fabrication and inspection. The minimum design thickness of the faceplates is 0.5 inch. Portions of the faceplates are thicker than the nominal design thickness to provide strength for localized loads due to attachments and connections. The auxiliary building modules are placed on the nuclear island basemat and do not have an offset of elevation of the bottom of the module faceplates through the thickness of the wall that is shown in Figure 3.8.3-8 for a portion of a module inside containment. The detail design of the internal connections within the modules including those at the corners, and for design elements and structures within the modules such as the diaphragms associated with the corner design, is different than depicted for the containment internal structures. The design of the mechanical connection of the module to the basemat varies for details such as size of deformed bars, thickness and configuration of plates, design of welds, and use of bolted connections in lieu of welds.

The ceiling of the main control room (floor at elevation 135′-3″), and the instrumentation and control rooms (floor at elevation 117′-6″) are designed as finned floor modules (Figure 3H.5-9). A finned floor consists of a 24-inch-thick concrete slab poured over a stiffened steel plate ceiling. The fins are rectangular plates welded 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 (see Design Control Document (DCD) Subsection 6.4.2.2). Several shop-fabricated steel panels, placed side by side, are used to construct the stiffened plate ceiling in a modularized fashion. The stiffened plate is designed to withstand construction loads prior to concrete hardening.

The new fuel storage area is a separate reinforced concrete pit providing temporary dry storage for the new fuel assemblies.

A 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, thus precluding them from falling into the spent fuel pool.

3.8.4.1.3 Containment Air Baffle

The containment air baffle is located within the upper annulus of the shield building, providing an air flow path for the passive containment cooling system. The air baffle separates the downward air flow entering at the air inlets from the upward air flow that cools the containment vessel and flows out of the discharge stack. The upper portion is supported from the shield building roof and the remainder is supported from the containment vessel. The air baffle is a seismic Category I structure designed to withstand the wind and tornado loads defined in Section 3.3. The air baffle structural configuration is depicted in Figures 1.2-14 and 3.8.4-1. The baffle includes the following sections:

- A wall supported off the shield building roof (see Figure 1.2-14)
- A series of panels attached to the containment vessel cylindrical wall and the knuckle region of the dome
- A flexible seal closing the gap between the wall and the panels fixed to the containment vessel, designed to accommodate the differential movements between the containment vessel and shield building
- Flow guides attached at the bottom of the air baffle to minimize pressure drop

The air baffle is designed to meet the following functional requirements:

- The baffle and its supports are configured to minimize pressure losses as air flows through the system
- The baffle and its supports have a design objective of 60 years
- The baffle and its supports are configured to permit visual inspection and maintenance of the air baffle as well as the containment vessel. Periodic visual inspections are primarily to inspect the condition of the coatings
- The baffle is designed to maintain its function during postulated design basis accidents
- The baffle is designed to maintain its function under specified external events including earthquakes, hurricanes and tornadoes

The general arrangement of the containment air baffle is shown in Figure 3.8.4-1. The lowest elevation of the air baffle is approximately 142′-0". The upper air baffle panels are on the head. The air baffle is supported by U-shaped support attachments welded to the containment vessel. The attachments can be installed in the subassembly area and, therefore, should not interfere with the containment vessel erection welds. The only penetrations through the containment vessel above the operating deck at elevation 135′-3″ are the main equipment hatch and personnel airlock. Five panels are deleted at the equipment hatch and two flow guides at the personnel airlock.

Panels are attached to the containment vessel above the cylindrical portion. The panels are angled to follow the curvature of the knuckle region of the containment vessel head, forming a conical baffle that provides a transitional flow region into the upper shield building. A flexible seal is provided between this upper row of panels and the air baffle that is attached directly to the shield building roof as shown in sheet 4 of Figure 3.8.4-1. This flexible seal is attached to the top of the upper row of panels. The flexible seal is set at ambient conditions to permit relative movements from minus 2 inches to plus 3 inches radially and minus 1 inch to plus 4 inches vertically. This accommodates the differential movement between the containment vessel and the shield building, based on the absolute sum of the containment pressure and temperature deflections and of the seismic deflections, such that the integrity of the air baffle is maintained.

The panels accommodate displacements between each panel due to containment pressure and thermal growth. Radial and circumferential growth of the containment vessel are accommodated by slip at the bolts between the horizontal beams and the U shaped attachment resulting in small gaps between adjacent panels. Vertical growth is accommodated by slip between the panel and the horizontal beam supporting the top of the panel. Seal plates between the panels limit leakage during and after occurrence of these differential displacements.

3.8.4.1.4 Seismic Category I Cable Tray Supports

Electric cables are routed in horizontal and vertical steel trays supported by channel type struts made out of cold rolled channel type sections. Spacing of the supports is determined by allowable loads in the trays and stresses in the supports. The supports are attached to the walls, floors, and ceiling of the structures as required by the arrangement of the cable trays. Longitudinal and transverse bracing is provided where required.

3.8.4.1.5 Seismic Category I Heating, Ventilating, and Air Conditioning Duct Supports

Heating, ventilating, and air conditioning duct supports consist of structural steel members or cold rolled channel type sections attached to the walls, floors, and ceiling of the structures as required by

the arrangement of the duct. Spacing of the supports is determined by allowable stresses in the duct work and supports. Longitudinal and transverse bracing is provided where required.

3.8.4.2 Applicable Codes, Standards, and Specifications

The following standards are applicable to the design, materials, fabrication, construction, inspection, or testing:

- [*American Concrete Institute (ACI), Code Requirements for Nuclear Safety Related Structures, ACI-349-01*]* (refer to Subsection 3.8.4.5 for supplemental requirements)
- [*American Concrete Institute (ACI), Building Code Requirements for Structural Concrete, ACI 318-11, Section 12.6*]* (refer to subsection 3.8.4.4.1 for application of ACI 318-11 Section 12.6)
- American Concrete Institute (ACI), ACI Detailing Manual
- American Concrete Institute (ACI), Self-Consolidating Concrete, ACI-237R
- [*American Institute of Steel Construction (AISC), Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities, AISC-N690-1994*]* (refer to Subsection 3.8.4.5 for supplemental requirements)
- American Iron and Steel Institute (AISI), Specification for the Design of Cold Formed Steel Structural Members, Parts 1 and 2, 1996 Edition and 2000 Supplement

Section 1.9 describes conformance with the Regulatory Guides.

Welding and inspection activities for seismic Category I structural steel, including building structures, structural modules, cable tray supports and heating, ventilating, and air conditioning duct supports are accomplished in accordance with written procedures and meet the requirements of the American Institute of Steel Construction (AISC N-690). The welded seam of the plates forming part of the leaktight boundary of the spent fuel pool, fuel transfer canal, cask loading pit, and cask washdown pit is examined by liquid penetrant and vacuum box after fabrication to confirm that the boundary does not leak.

3.8.4.3 Loads and Load Combinations

The loads and load combinations used in the analysis are considered to be part of the method of evaluation.

3.8.4.3.1 Loads

The loads considered are normal loads, severe environmental loads, extreme environmental loads, and abnormal loads.

3.8.4.3.1.1 Normal Loads

Normal loads are those loads to be encountered, as specified, during initial construction stages, during test conditions, and later, during normal plant operation and shutdown. They include the following:

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D = Dead loads or their related internal moments and forces, including any permanent piping and equipment loads

F = Lateral and vertical pressure of liquids or their related internal moments and forces

L = Live loads or their related internal moments and forces, including any movable equipment loads and other loads that vary with intensity and occurrence

H = Static earth pressure or its related internal moments and forces

T_o = Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition

R_o = Piping and equipment reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

3.8.4.3.1.2 Severe Environmental Loads

The severe environmental load is the following:

W = Loads generated by the design wind specified for the plant in Subsection 3.3.1.1

3.8.4.3.1.3 Extreme Environmental Loads

Extreme environmental loads are the following:

E_s = Loads generated by the safe shutdown earthquake specified for the plant, including the associated hydrodynamic and dynamic incremental soil pressure. Loads generated by the safe shutdown earthquake are specified in Section 3.7.

Wt = Loads generated by the design tornado specified for the plant in Subsection 3.3.2, including loads due to tornado wind pressure, differential pressure, and tornado-generated missiles.

N = Loads generated by the probable maximum precipitation (provided previously in Table 2.0-201).

The application of the 48-hour PMWP and the 100-year return period ground-level snowpack in the roof design of safety-related structures is addressed in Subsection 2.3.1.3.4.

3.8.4.3.1.4 Abnormal Loads

Abnormal loads are those loads generated by a postulated high-energy pipe break accident for pipes not qualified for leak-before-break. Abnormal loads include the following:

P_a = Pressure load within or across a compartment generated by the postulated break. The main steam isolation valve (MSIV) and steam generator blowdown valve compartments are designed for a pressurization load of 6 psi. The subcompartment design pressure bounds the pressurization effects due to postulated breaks in high energy pipe. Determination of subcompartment pressure loads is discussed in Subsection 6.2.1.2.

T_a = Thermal loads under thermal conditions generated by the postulated break and including T_o. Determination of subcompartment temperatures is discussed in Subsection 6.2.1.2.

R_a = Piping and equipment reactions under thermal conditions generated by the postulated break and including R_o. Determination of pipe reactions generated by postulated breaks is discussed in Section 3.6.

Y_r = Load on the structure generated by the reaction on the broken high-energy pipe during the postulated break. Determination of the loads is discussed in Section 3.6.

Y_j = Jet impingement load on the structure generated by the postulated break. Determination of the loads is discussed in Section 3.6.

Y_m = Missile impact load on the structure generated by or during the postulated break, as from pipe whipping. Determination of the loads is discussed in Section 3.6.

3.8.4.3.1.5 Dynamic Effects of Abnormal Loads

The dynamic effects from the impulsive and impactive loads caused by P_a, R_a, Y_r, Y_j, Y_m, and tornado missiles are considered by one of the following methods:

- Applying an appropriate dynamic load factor to the peak value of the transient load
- Using impulse, momentum, and energy balance techniques
- Performing a time-history dynamic analysis

Elastoplastic behavior may be assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

Dynamic increase factors appropriate for the strain rates involved may be applied to static material strengths of steel and concrete for purposes of determining section strength.

3.8.4.3.2 Load Combinations

3.8.4.3.2.1 Steel Structures

The steel structures and components are designed according to the elastic working stress design methods of the AISC-N690 specification using the load combinations specified in Table 3.8.4-1.

3.8.4.3.2.2 Concrete Structures

The concrete structures and components are designed according to the strength design methods of ACI-349 Code, using the load combinations specified in Table 3.8.4-2.

3.8.4.3.2.3 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 (with the exception of the containment operating deck which is designed for 800 pounds per square foot specified for plant maintenance condition).

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 these specified

live loads, or 75 percent of the roof snow load, whichever is applicable, except in the case of the containment operating deck. For the seismic load combination, the containment operating deck is designed for a live load of 200 pounds per square foot which is appropriate for plant operating condition. The mass of equipment and distributed systems is included in both the dead and seismic loads.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Seismic Category I Structures

[*The design and analysis procedures for the seismic Category I structures (other than the containment vessel, containment internal structures, and other structures constructed using concrete-filled steel plate construction), including assumptions on boundary conditions and expected behavior under loads, are in accordance with ACI-349 with clarification as provided below for concrete structures, with AISC-N690 for steel structures, and AISI for cold formed steel structures.*]* The structural wall modules in the auxiliary building are designed using the same procedures as the structural modules in the containment internal structures described in Subsection 3.8.3.5.3. The shield building is designed using the procedures and requirements described in Subsection 3.8.4.5.5.

[*The criteria of ACI-349, Chapter 12, are applied in development and splicing of the reinforcing steel. The ductility criteria of ACI-349, Chapter 21, are applied in detailing and anchoring of the reinforcing steel. Provisions in ACI 318-11 Section 12.6*]* (Reference 58) [*for headed and mechanically anchored deformed bars apply to development of headed reinforcement as an alternative to the provisions in ACI 349 Appendix B.*]* The specific application of ACI 318-11 Section 12.6 is used for development of the headed reinforcement to provide ductile behavior. Alternatively, a rebar hook with requirements according to ACI 349 may also be used.

[*The application of Chapter 21 detailing is demonstrated in the reinforcement details of critical sections*]* in Subsection 3.8.5 and Appendix 3H.

[*Sections 21.2 through 21.5 of Chapter 21 of ACI 349 are applicable to frame members resisting earthquake effects. These requirements are considered in detailing structural elements subjected to significant flexure and out-of-plane shear. These elements include the following examples described in Appendix 3H*:]*

- Reinforcement details for the basemat are described in Subsection 3.8.5. [*Shear stirrups have T headed anchors at each end. Provisions in ACI 318-11 Section 12.6*]* (Reference 58) [*for headed and mechanically anchored deformed bars apply to headed shear reinforcement as an alternative to the provisions in ACI 349 Appendix B.*]* The specific application of ACI 318-11 Section 12.6 is used for development of the headed shear reinforcement to provide ductile behavior.

- Reinforcement details for the exterior walls below grade are described in Subsection 3H.5.1.1. [*Shear ties have T headed anchors at each end. Provisions in ACI 318-11 Section 12.6*]* (Reference 58) [*for headed and mechanically anchored deformed bars apply to headed reinforcement as an alternative to the provisions in ACI 349 Appendix B.*]* The specific application of ACI 318-11 Section 12.6 is used for development of the headed reinforcement to provide ductile behavior. Alternatively, a conventional tie with alternating 90 and 135 degree hooks around the longitudinal reinforcement may be used.

[*Sections 21.2 and 21.6 of Chapter 21 of ACI 349 are applicable to walls, diaphragms, and trusses serving as parts of the earthquake force-resisting systems as well as to diaphragms, struts, ties, chords and collector elements. These requirements are considered in the detailing of reinforcement in the walls and floors of the auxiliary building and in the shield building cylindrical wall and roof.*]*

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- Reinforcement for the shear walls and floors are shown in Subsections 3H.5.1 to 3H.5.4. [*Transverse reinforcement terminating at the edges of structural walls or at openings is detailed in accordance with 21.6.6.5 of ACI 349.*]*

The bases of design for the tornado, pipe breaks, and seismic effects are discussed in Sections 3.3, 3.6, and 3.7, respectively. The foundation design is described in Subsection 3.8.5.

The seismic Category I structures are reinforced concrete, concrete-filled steel plate, and structural module shear wall structures consisting of vertical shear/bearing walls and horizontal slabs supported by structural steel framing. Seismic forces are obtained from the response spectrum analysis of the three dimensional finite element models described in Table 3G.1-1. These results are modified to account for accidental torsion as described in Subsection 3.7.2.11. 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. Also evaluated and considered in the shear wall and floor slab design are out-of-plane bending and shear loads, such as live load, dead load, seismic, lateral earth pressure, hydrostatic, hydrodynamic, and wind pressure. These out-of-plane bending and shear loads are obtained from the response spectrum analyses supplemented by hand calculations.

The exterior walls of the seismic Category I structures below the grade are designed to resist the worst case lateral earth pressure loads (static and dynamic), soil surcharge loads, and loads due to external flooding as described in Section 3.4. The lateral earth pressure loads are evaluated for two cases:

- Lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with ASCE 4-98 (Reference 56), Subsection 3.5.3, Figure 3.5-1, "Variation of Normal Dynamic Soil Pressures for the Elastic Solution"

- Lateral earth pressure equal to the passive earth pressure

The shield building roof and the passive containment cooling water storage tank are analyzed using the three-dimensional finite element quadrant model described in Subsection 3G.2.3.1 with the ANSYS computer code. Loads and load combinations are given in Subsection 3.8.4.3 and include construction, dead, live, thermal, wind and seismic loads. The seismic response of the water in the tank is applied as static pressures corresponding to the impulsive and convective response. The results are used in the design of the tension ring, air inlet structure, PCS tank, shield building roof, and radial roof beams. 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 such that the member forces in these regions envelope those from a response spectrum analysis using the refined NI05 model, as described in Appendix 3G.2.2.4.

The liner for the passive containment cooling water storage system tank is analyzed by hand calculation. The design considers construction loads during concrete placement, loads due to handling and shipping, normal loads including thermal, and the safe shutdown earthquake. Buckling of the liner is prevented by anchoring the liner using the embedded stiffeners and welded studs. The liner is designed as a seismic Category I steel structure in accordance with AISC N690 with the supplemental requirements given in Subsection 3.8.4.

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 structural steel framing is designed for vertical loads. Appendix 3H shows typical structural steel framing in the auxiliary building.

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Computer codes used are general purpose computer codes. The code development, verification, validation, configuration control, and error reporting and resolution are according to the quality assurance requirements of Chapter 17.

[*The finned floors for the main control room and the instrumentation and control room ceilings are designed as reinforced concrete slabs in accordance with ACI-349. The steel panels are designed and constructed in accordance with 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.*]*

The concrete floors on steel plates, including the control room ceiling and the floors in the CA20 module, are designed as reinforced concrete slabs in accordance with ACI-349. The steel panels are designed and constructed in accordance with AISC-N690. For positive bending, the steel plate is in tension and the steel plate and stiffeners serve as the bottom reinforcement. For negative bending, compression is resisted by the concrete and stiffened plate and the tension by top reinforcement in the concrete. This methodology is described for the control room ceiling in Subsection 3H.5.4.

3.8.4.4.2 Seismic Category I Cable Tray Supports

The design and analysis procedures for seismic Category I cable trays and their supports are described in Appendix 3F.

3.8.4.4.3 Seismic Category I Heating, Ventilating, and Air Conditioning Duct Supports

The design and analysis procedures for seismic Category I heating, ventilating, and air conditioning ducts and their supports are described in Appendix 3A.

3.8.4.4.4 Below Grade Exterior Walls

The design and analysis procedures for seismic Category I exterior walls below grade are described below.

The nuclear island exterior walls below grade are subjected to various loads, including the lateral earth pressure loads. Lateral loads used in design of the nuclear island are based on conservative assumptions (soil profiles with highest lateral loads) for the properties of the soil adjacent to the exterior walls. Lateral loads are calculated for a range of possible soil properties and a conservative set of loads is specified for design.

The plant grade elevation is 100′-0″, the high groundwater level is 98′-0″, and the maximum flood level is at plant elevation 100′-0″.

Load Conditions

<u>Hydrostatic (groundwater) (Live, L)</u>

The design high groundwater level is at elevation 98′-0″, and the probable maximum flood level is at elevation 100′-0″. Both of these loads are treated as live loads. The hydrostatic unit water pressure (P_w) at a depth h (units: feet) below ground level is calculated as:

$$P_w = \gamma_w h \tag{1}$$

Where, γ_w = unit weight of water = 62.4 pcf

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At-Rest Earth Pressure (Earth, H)

Static earth pressure is based on "at -rest" conditions, and the coefficient of earth pressure for the at-rest condition (K_o) is determined from the following relationship:

$$P_o = K_o \gamma h \quad (2)$$

Where,

K_o = 1-SIN(φ)

φ = angle of internal friction

h = depth below grade (El. 100′-0″)

$\gamma = \gamma_s$ = Saturated unit weight of granular back fill above water table, or

$\gamma = \gamma_s - \gamma_w$ below water table

Static and Dynamic Surcharge Pressures

The static surcharge pressure (Dead, D) is considered a static pressure load, and therefore, the at-rest coefficient (K_o) is used. The static lateral surcharge pressure is defined for the same soil case as the at-rest pressure as follows:

$$P_{surch} = K_o q \quad (3)$$

Where,

q = static surcharge pressure based on footprint loads of adjacent structures

The dynamic lateral surcharge pressure (Seismic, Es_1) is based on 0.3 times the static surcharge pressure. The static and dynamic lateral surcharge pressures act uniformly along the height of the exterior walls.

Dynamic Earth Pressure (Seismic, Es_2)

The dynamic earth pressure is calculated in accordance with ASCE 4-98 (Reference 56), Subsection 3.5.3, Figure 3.5-1, "Variation of Normal Dynamic Soil Pressures for the Elastic Solution." The Poisson's ratio (v) for the soil varies between 0.35 and 0.4. The most conservative dynamic soil pressure distribution, obtained using 0.4 for v, is used. The seismic acceleration levels are the maximum accelerations associated with the seismic response of the nuclear island from elevation 60.5′ to elevation 100′. The north-south seismic excitation acceleration values are associated with Walls 1 and 11. The east-west seismic acceleration values apply to Walls N, Q, and I.

Passive Earth Pressure (Seismic, Es_3)

The nuclear island includes passive pressure on the exterior walls to resist sliding during an SSE. Therefore, the exterior walls below grade shall also be designed for passive earth pressure in the load combinations that include Es. The passive earth pressure is calculated from:

$$P_P = K_P \gamma h \quad (4)$$

Where,

$K_P = Tan^2(45^\circ + \varphi / 2)$

φ = angle of internal friction

h = depth below grade (El. 100′-0″)

$\gamma = \gamma_s$ = Saturated unit weight of granular back fill above water table or

$\gamma = \gamma_s - \gamma_w$ below water table

Load Combinations

Lateral loading conditions, identified above, are used for the following load combinations:

- Load Combination 3: (L) + (H) + (D) + (Es_1) + (Es_2)
- Load Combination 7: (L) + (D) + (Es_1) + (Es_3)
- Load Combinations 1 & 2: 1.4 x (D) + 1.7 x ((L) + (H))
- Load Combinations 4, 5 & 6: (L) + (H) + (D)
- Load Combinations 8 & 9: 1.05 x (D) + 1.3 x ((L) + (H))

The design of the exterior below grade walls is designed from the load combinations listed above, which incorporate the effects of full lateral passive earth pressure.

Based on the results for the load combinations, the critical lateral pressure distribution for the design loads along the exterior, below grade walls would result when combining the effects of Load Combinations (LC) 3 and 7 along the nuclear island's east-west axis. The pressure distribution for LC 3 would control from elevation 100′ (ground surface) to the intersection with the LC 7 pressure distribution. Graphically, this occurs at approximately elevation 91′. The LC 7 plot would control from this intersection to elevation 60.5′ (bottom of the basemat). LC 7 incorporates the effects of full lateral passive earth pressure.

Lateral Soil Pressure with Vertical Seismic Effects

Soil Pressure – Vertical seismic is the at-rest earth pressure for the generic soil model multiplied by 0.3 (seismic acceleration factor). This same factor was used to determine the dynamic lateral surcharge pressure (Seismic, Es_2) noted above. These tabular values where combined with the values from the dynamic soil pressure (along the east-west axis) by the SRSS Method (square root, sum of squares) and plotted against the dynamic soil pressure (along the east-west axis) alone. The contribution of vertical seismic effect on the at-rest earth pressure is negligible to the dynamic soil pressure. Therefore, vertical seismic effect is neglected and the dynamic soil pressure alone was used in determining the load combination.

3D SASSI Analyses

3D SASSI soil-structure interaction (SSI) analyses for soil conditions soft-to-medium (SM) and upper-bound-soft-to-medium (UBSM) soil were used to estimate the dynamic soil pressure on the exterior nuclear island walls. Five locations were used to calculate lateral pressure on the structure, including spring elements at elevation 100′ and 82.5′, and structure elements at elevation 91′, 71′, and 57′. Axial spring forces normal to and along each wall are tabulated and converted to a unit load and corresponding stress at each elevation. Similarly, element stresses normal to the plane of the wall are tabulated for each elevation, and the maximum normal stress was determined.

The 3D SASSI UBSM and SM dynamic soil pressure for the north-south and east-west vertical profiles are compared to the ASCE 4-98 lateral soil pressure distribution curve.

3D ANSYS Analyses

The analyses of the nuclear island for the lateral earth pressure loads are performed using 3D linear elastic ANSYS analysis. The lateral earth pressure loads are applied to the finite element nodes associated with embedded nuclear island structural walls. The analysis performed is linear elastic. Loads and moments at the base of the exterior walls are extracted from the analyses and are added in the design analyses of the basemat.

3.8.4.5 Structural Criteria

[*The analysis and design of concrete conform to ACI-349 as supplemented below and with clarifications provided in Subsection 3.8.4.4.1. The analysis and design of structural steel conform to AISC-N690. The analysis and design of cold-formed steel structures conform to AISI. The margins of structural safety are as specified by those codes.*]*

3.8.4.5.1 Supplemental Requirements for Concrete Structures

[*Supplemental requirements for ACI-349-01 are given in the position on Regulatory Guide 1.142 in Appendix 1A. The structural design meets the supplemental requirements identified in Regulatory Positions 2 through 8, 10 through 13, and 15.*]*

Paragraph 21.6.1 of ACI 349-01 should reference 21.6.6 instead of 21.6.5. Paragraph 21.6.5 in ACI 349-97 was renumbered to 21.6.6 in ACI 349-01, and the reference in 21.6.1 was not updated. The errata for ACI 349-01 are being updated to include this correction. This makes the paragraph consistent with ACI 349-97, which was endorsed by Regulatory Guide 1.142.

[*Design and construction of fastening to concrete is in accordance with ACI 349-01, Appendix B*]* and are in conformance with the regulatory positions of NRC Regulatory Guide 1.199, Revision 0. [*Alternative requirements to ACI 349, Appendix B apply to the anchoring of headed shear reinforcement for the basemat (see Subsection 3.8.4.4.1). Alternative requirements to ACI 349, Appendix B apply to the anchoring of headed reinforcement above the basement (see Subsection 3.8.4.4.1)*]*

3.8.4.5.2 Supplemental Requirements for Steel Structures

[*Supplemental requirements for use of AISC-N690 are as follows:*

- *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.*

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- *Sections Q1.24 and Q1.25.10 are supplemented as follows:*

  Shop painting is in accordance with Section M of the Manual of Steel Construction, Load and Resistance Factor Design, First Edition. Exposed areas after installation are field painted in accordance with the applicable portion of Chapter M of the Manual of Steel Construction, Load and Resistance Factor Design, First Edition.]* See Subsection 6.1.2.1 for additional description of the protective coatings.

3.8.4.5.3 Design Summary Report

A design summary report is prepared for seismic Category I structures documenting that the structures meet the acceptance criteria specified in Subsection 3.8.4.5.

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 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 judgement to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

3.8.4.5.4 Design Summary of Critical Sections

[*The design of representative critical elements of the following structures is described in Appendix 3H.*

- *South wall of auxiliary building (column line 1), elevation 66'-6" to elevation 180'-0" – see subsection 3H.5.1.1 and Figures 3H.5-2 and 3H.5-3*

- *Interior wall of auxiliary building (column line 7.3), elevation 66'-6" to elevation 160'-6" – see subsection 3H.5.1.2 and Figure 3H.5-4*

- *West wall of main control room in auxiliary building (column line L), elevation 117'-6" to elevation 153'-0" – see subsection 3H.5.1.3 and Figure 3H.5-12*

- *North wall of MSIV east compartment (column line 11 between lines L and M), elevation 117'-6" to elevation 153'-0" – see subsection 3H.5.1.4 and Figure 3H.5-5*

- *Roof slab at elevation 180'-0" adjacent to shield building cylinder – see subsection 3H.5.2.1 and Figure 3H.5-7*

- *Floor slab on metal decking at elevation 135'-3" – see subsection 3H.5.2.2 and Figure 3H.5-6*

- *2'-0" slab in auxiliary building (operations work area (tagging room) ceiling) at elevation 135'-3" – 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.')*

- *Finned floor in the main control room at elevation 135'-3" – see subsection 3H.5.4 and Figure 3H.5-9*

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- *Shield building roof, exterior wall of the PCS water storage tank – see subsection 3H.5.6.3 and Figure 3H.5-11*
- *Shield building roof, interior wall of the PCS water storage tank – see subsection 3H.5.6.2 and Figure 3H.5-11*
- *Shield building roof, tension ring, and air inlet structure – see subsections 3H.5.6 and 3H.5.6.1*
- *Divider wall between the spent fuel pool and the fuel transfer canal – see subsection 3H.5.5.1 and Figure 3H.5-10*
- *Shield building SC cylinder – see subsection 3H.5.7.1, Figure 3H.5-16, and Figures 5 and 6 of APP-GW-GLR-602 (Reference 57)*
- *Shield building SC to RC connection – see Subsection 3H.5.7.2, Figure 3H.5-16, and Figures 1, 2, 3, and 4 of APP-GW-GLR-602 (Reference 57)*]*

3.8.4.5.5 Shield Building Structural Wall Modules

[*The shield building concrete-filled steel module walls are designed for loads such as dead, live, thermal, wind, and safe shutdown earthquake loads as identified in Table 3.8.4-2. Concrete-filled structural wall modules are designed as reinforced concrete structures in accordance with the requirements of ACI 349, and supplemented with additional requirements discussed in subsection 3.8.3.5.3 and below. The faceplates are considered as the reinforcing steel, bonded to the concrete by headed studs. The steel plate modules are anchored to the reinforced concrete by mechanical connections welded to the steel plate.*]*

Figure 5 in Reference 57 shows the typical design details of the shield building concrete-filled steel module walls. The design of the shield building critical sections is described in Appendix 3H.

3.8.4.5.5.1 Design for Axial Loads and Bending

Design for axial load (tension and compression), in-plane bending, and out-of-plane bending is in accordance with the requirements of ACI 349, Chapters 10 and 14. The reinforcement for in-plane tension and compression combined with out-of-plane moments is calculated using a rectangular concrete stress distribution, in accordance with Sections 10.2 and 10.3 of the ACI 349 code.

3.8.4.5.5.2 Design for In-Plane Shear

Design for in-plane shear is in accordance with the requirements of ACI 349, Chapters 11 and 14. The steel faceplates are treated as reinforcing steel, contributing as provided in Section 11.10 of ACI 349.

3.8.4.5.5.3 Design for Out-of-Plane Shear

Design for out-of-plane shear is in accordance with the requirements of ACI 349, Chapter 11.

3.8.4.5.5.4 Evaluation for Thermal Loads

Thermal loads are combined with other loads according to the load combinations defined in Table 3.8.4-2. The effect of concrete cracking is considered in the stiffness properties for the concrete elements subjected to the thermal loads. Thermal forces and moments are calculated by multiplying linear elastic thermal stress analysis results by a stiffness reduction ratio α. The stiffness reduction

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ratio α is calculated as the ratio of the cracked stiffness to the elastic stiffness of the section to be evaluated.

3.8.4.5.5.5 Design of Shear Studs and Tie Bars

[*The shear stud connectors and tie bars are sized to carry the horizontal shear at the junction of the steel faceplates and the concrete infill. The shear stud and tie bar design is such that the shear stud and tie bar strength is sufficient to develop the full yield strength of the steel plate in approximately 3 x the thickness of the wall, or about 9 feet.*

The tie bars provide a structural framework for the modules, maintain the separation between the faceplates, support the modules during transportation and erection, and act as "form ties" between the faceplates when concrete is being placed. The tie bars provide additional shear capacity between the steel plates and concrete as well as additional strength similar to that provided by stirrups in reinforced concrete. The area and spacing of the tie bars satisfy the requirement for minimum shear reinforcement for beams given in the ACI codes. The connection between the tie bars and the steel faceplates is designed to develop the full tensile strength of the tie bar.]*

3.8.4.5.5.6 Design of Connections

The shield building steel plate modules are connected to standard reinforced concrete using a connection design satisfying the requirements outlined below. The locations of these connections include the following:

1. At the base of the shield building wall into the concrete below

2. Between SC and RC portions of the shield building wall

3. Where RC floors, roofs, and walls connect to the SC modules.

The steel plate modules are connected to the reinforced concrete by reinforcement or deformed bars directly connected to the modules. These connections are sized using a strength design approach, and the mechanical connection portion of the connection satisfies the requirements of AISC N690. In the area of the connections, the faceplates are thicker (up to 1.0-inch nominal thickness), as necessary, to address loads from the connections. The connection design satisfies the following requirements:

- The connections provide for the direct transfer of forces from the RC reinforcing steel to the SC faceplates. Lap splice type connections between dowels extending out of the RC into the SC adjacent to the headed shear studs are not used to connect the modules to the base concrete.

- Loads are transferred directly from the faceplates to the reinforced concrete using reinforcing bars, mechanical connection, and welds.

- The mechanical connections develop 125 percent of specified yield of the connected reinforcement bar.

- Loads that are out of plane of the wall, such as from roofs or walls, are carried through the full thickness of the shield building walls.

Figures 1, 2, 3, and 4 in APP-GW-GLR-602 (Reference 57) show the typical design details for the connection of the wall modules to the reinforced concrete. Figure 5 in Reference 57 shows the typical design details of the shield building concrete-filled steel module walls.

*NRC Staff approval is required prior to implementing a change in this information.

Figure 3H.5-7 and Figure 7 in Reference 57 show typical auxiliary building reinforced concrete roof connections to the shield building SC wall. The details of the connections between the auxiliary building roof and the shield building wall vary because of loads on the connection and the orientation of the wall to the roof reinforcement arrangement as outlined in the notes on Figure 3H.5-7 and in Section 4 of Reference 57. Figure 3H.5-7 also shows typical connections between the auxiliary building walls and the shield building SC wall modules. The connection elements used and the design details vary because of the orientation of the auxiliary building wall to the shield building wall. These connection design details satisfy the requirements of AISC N690 and the requirements identified above.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

This subsection contains information relating to the materials, quality control program, and special construction techniques used in the construction of the other seismic Category I structures, as well as the containment internal structures. An edition of the referenced specifications applicable after the start of construction or procurement activities will be used.

3.8.4.6.1 Materials

3.8.4.6.1.1 Concrete

[*The compressive strength of concrete used in the seismic Category I structures and containment internal structures is f_c' = 4000 psi except as noted in the following. For the nuclear island basemat (the nominal 6 ft. thick foundation described in Subsection 3.8.5.1) the compressive strength of concrete is f_c' = 5000 psi. For the SC composite portion of the shield building structure including the connection region below the SC/RC interface and the shield building roof, the compressive strength of concrete is f_c' = 6000 psi.*]* The test age of concrete containing pozzolan is up to 56 days. The test age of concrete without pozzolan is up to 28 days. Concrete is mixed, batched, and placed according to Reference 6, Reference 7, and ACI-349.

Portland cement conforms to Reference 8, Type II. Certified copies of mill test reports showing that the chemical composition and physical properties conform to the specification are obtained for each cement delivery.

Aggregates conform to Reference 9. The fineness modulus of fine aggregate (sand) is not less than 2.3, nor more than 3.1, per Reference 9. In at least four of five successive test samples, such modulus is not allowed to vary more than 0.20 from the moving average established by the last five tests. Coarse aggregates may be rejected if the loss from the Los Angeles abrasion test, Reference 10, using Grading A or Reference 11, exceeds 40 percent by weight at 500 revolutions. Acceptance of source and aggregates is based on the tests specified in Table 3.8.4-3 as applicable based on type of aggregate and ACI-349 requirements.

Water and ice used in mixing concrete do not contain more than 250 parts per million of chlorides (as Cl) as determined in accordance with Reference 12. They do not contain more than 2000 parts per million of total solids as determined in accordance with Reference 13. Water meets the criteria in Table 3.8.4-4 in regard to the effects of the proposed mixing water on hardened cement pastes and mortars compared with distilled water.

The concrete contains an air entraining admixture when required by ACI-349 and may contain pozzolans and a water-reducing admixture. A viscosity modifying admixture (VMA) may be used in self-consolidating concrete (SCC) for adjusting the viscosity and improving its stability as outlined in ACI 237R, "Self-Consolidating Concrete." Admixtures, except pozzolan, are stored in liquid solution.

*NRC Staff approval is required prior to implementing a change in this information.

Admixture types A, B, C, and F are used as noted herein to produce quality concrete. The admixtures shall be proven compatible with all constituents and other admixtures in mix prior to use in production of concrete. Furthermore, all admixtures shall be measured, introduced into the mix, and used in strict accordance to the manufacturer's recommendations.

Admixtures do not contain added chlorides except as contained in potable drinking water used for manufacture of the admixtures. The chloride content is stated in the manufacturer's material certification.

Pozzolan conforms to Reference 14, except that the ignition loss does not exceed 6 percent.

Pozzolan is sampled and tested in accordance with Reference 15 for source approval.

Air entraining admixture conforms to Reference 16.

Water-reducing admixtures conform to Reference 17 and are of types A or F. A type A admixture may be used to increase the slump for conventional mixes (may use type F, optional). A type F admixture shall be used for SCC admixtures.

Retarding admixtures conform to Reference 17 and are of type B. Type B admixtures may be used to retard set times for conventional and SCC mixes (i.e., for mass concrete).

Accelerating admixtures conform to Reference 17 and are type C. Type C admixtures may be used to accelerate the set time for conventional and SCC mixes (i.e., for thin members placed in cold weather).

Manufacturer's certification for the air entraining admixture is required demonstrating compliance with Reference 16, Section 4 requirements.

Manufacturer's certification for the water-reducing admixture is required demonstrating compliance with Reference 17, Section 5 requirements.

Manufacturer's test reports are required for each delivery of pozzolan showing the chemical composition and physical properties and certifying that the pozzolan complies with the specification.

Proportioning of the concrete mix is in accordance with Reference 18 and Option B of Reference 6.

A testing laboratory designs and tests the concrete mixes. Only mixes meeting the design requirements specified for concrete are used.

Forms for concrete are designed as recommended in ACI 347.

3.8.4.6.1.2 Reinforcing Steel

[*Reinforcing bars for concrete are deformed bars according to Reference 19, Grade 60, and Reference 20*.]* Certified material test reports are provided by the supplier for each heat of reinforcing steel delivered showing physical (both tensile and bend test results) and chemical analysis. In addition, a minimum of one tensile test is performed for each 50 tons of each bar size produced from each heat of steel.

In areas where reinforcing steel splices are necessary and lap splices are not practical, mechanical connections (e.g., threaded splices, swaged sleeves or cadwelds) are used.

*NRC Staff approval is required prior to implementing a change in this information.

[*Headed reinforcement meeting the requirements of ASTM A970 (Reference 49) is used where mechanical anchorage is required,*]* such as for shear reinforcement in the nuclear island basemat and in the exterior walls below grade.

As stated in Subsection 3.4.1.1.1, seismic Category I structures that are located below grade elevation are protected against flooding by a waterproofing system. This, in conjunction with the ACI 349 Code requirements for concrete cover for exposure to earth or weather for the reinforcing steel, provides sufficient protection for the reinforcing steel. Therefore, the use of coated reinforcing steel is not planned.

3.8.4.6.1.3 Structural Steel

Basic materials used in the structural and miscellaneous steel construction conform to the ASTM standards listed in Table 3.8.4-6. The table includes the major structural steel materials of construction. Materials used in limited applications and minor amounts may not be included. [*Additional information on structural and reinforcing steel for the shield building is provided in Table 1 of APP-GW-GLR-602 (Reference 57).*]*

3.8.4.6.1.4 Masonry Walls

There are no safety-related masonry walls used in the nuclear island.

3.8.4.6.2 Quality Control

The quality assurance program is described in Chapter 17. Conformance to Regulatory Guide 1.94 is as described in Section 1.9.

3.8.4.6.3 Special Construction Techniques

Construction techniques for the structural modules are the same as special construction techniques for the containment internal structures discussed previously in Subsection 3.8.3.6.1.

3.8.4.7 Testing and In-Service Inspection Requirements

Structures supporting the passive containment cooling water storage tank on the shield building roof will be examined before and after first filling of the tank.

- The boundaries of the passive containment cooling water storage tank and the shield building roof above the tension ring at the intersection of the shield building roof and the shield building cylinder will be inspected visually for excessive concrete cracking before and after first filling of the tank. Any significant concrete cracking will be documented and evaluated in accordance with ACI 349.3R-96 (Reference 50). The structure around the passive containment cooling water storage tank will be inspected for water leaking out of the tank through the concrete.

- The vertical elevation of the passive containment cooling water storage tank relative to the top of the shield building cylindrical wall at the tension ring will be measured before and after first filling. The change in relative elevation will be compared against the predicted deflection.

- A test will be performed to measure the leakage from the passive containment cooling water storage tank based on measuring the water flow out of the leak chase collection system.

- A report will be prepared summarizing the test and evaluating the results.

*NRC Staff approval is required prior to implementing a change in this information.

During the operation of the plant, the condition of these structures should be monitored by the Combined License holder to provide reasonable confidence that the structures are capable of fulfilling their intended functions. The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.4.8 Construction Inspection

Construction inspection is conducted to verify the concrete wall thickness and quantity of concrete reinforcement. The construction inspection includes concrete wall thickness and reinforcement expressed in units of in^2/ft (linear length) equivalent when compared to standard reinforcement bar sections. Inspections will be measured at applicable sections excluding designed openings or penetrations. Inspections will confirm that each applicable section provides the minimum required reinforcement, steel plate thickness, and concrete thickness. The minimum required reinforcement, steel plate thickness, and concrete thickness represent the minimum values to meet the design basis loads. Appendix 3H also indicates the reinforcement provided which may exceed the minimum required reinforcement for the following reasons:

- Structural margin
- Ease of construction
- Use of standardized reinforcement sizes and spacing

A shield building construction mockup program will be used to build full-scale replicas of areas of the shield building that present critical construction areas identified as challenging areas of construction. These mockups provide an opportunity to apply and evaluate alternate and innovative construction and inspection methods and procedures. The fabrication, assembly, and erection of a full-scale mockup provides the accurate physical representation needed to evaluate true working conditions, physical configuration, accessibility, and quality control issues that may be encountered in construction. Construction practices review and examination of the mockups will be used to confirm the adequacy of construction means, methods, and procedures. If defects are found, the procedures will be revised and the mockup repeated until the required result is obtained. The major tasks that will be performed on each mockup include the following: field performance testing of the quality of concrete mixes, methods of concrete placement, inspections and surveillance, and post-placement activities.

In addition to these important process control tasks performed on the mockups, an inspection program will be undertaken on the AP1000 construction site mockups that uses the enhanced shield building design. Both visual inspection and non-destructive examination (NDE) will be performed for assessing defects that may impact structural integrity, such as crack distress, deterioriation caused by honeycomb, voids, and delaminations. The NDE inspection of the mockups will be performed at the first construction site of an AP1000 plant with an enhanced shield building design, and mockups applicability will be evaluated for the subsequent AP1000 site. This is to demonstrate the construction quality process control for concrete placement, and develop and document insights and requirements for corrective action, if required, to be used in the construction inspection program for all AP1000 plants.

3.8.5 Foundations

3.8.5.1 Description of the Foundations

The nuclear island structures, consisting of the containment building, shield building, and auxiliary building are founded on a common 6-foot-thick, cast-in-place, reinforced concrete basemat foundation. The top of the foundation is at elevation 66′-6″.

The depth of overburden and depth of embedment are given in Subsection 2.5.4.5.

A description of the safety-related backfill, which supports Category I structures, is given in Subsection 2.5.4.5.

[*Adjoining buildings*[1]*, such as the turbine building and annex building, are structurally separated from the nuclear island structures by a 2-inch gap at and below the grade. A 4-inch minimum gap is provided above grade.*]* This provides space to prevent interaction between the nuclear island structures and the adjacent seismic Category II and non-seismic structures during a seismic event. The maximum relative seismic displacement between the roof of the nuclear island and any adjoining buildings is less than 3 inches. This results in a clearance (gap) between buildings greater than 1 inch during a seismic event. Therefore, there are no interactions between adjacent buildings during a seismic event. Figure 3.8.5-1 shows the foundations for the nuclear island structures and the adjoining structures.

Resistance to sliding of the concrete basemat foundation is provided by passive soil pressure and soil friction. This provides the required factor of safety against lateral movement under the most stringent loading conditions.

For ease of construction, the foundation is built on a mud mat. The mud mat is lean, nonstructural concrete and rests upon the load-bearing soil. Waterproofing requirements are described in Subsection 3.4.1.1.1.

The scope of the LWA foundation work includes: placing the mud mats, water proofing membrane, concrete forms, drains and other items necessary to prepare the Nuclear Island base slab for the first concrete pour.

After backfill beneath the NI (Nuclear Island) has been placed and compacted to roughly the required elevation for the first mud mat, the construction of the retaining wall will begin. The retaining wall will be a vertical mechanically-stabilized earth (MSE) wall with smooth-faced concrete panels. This wall will function as both a retaining wall as the backfill outside the NI volume is brought up to plant grade and as the exterior concrete form for the outer walls of the NI. Subsection 2.5.4.5, Excavation and Backfill, provides additional information on the backfill and MSE wall.

The construction of the MSE wall begins with installation of a concrete footer. The top surface of the MSE wall footer will be installed below the bottom elevation of the first mud mat. The size and reinforcement for the concrete footer will be as required by the designer of the MSE wall. The MSE wall footer is a relatively thin concrete structure that provides a stable, level surface for construction of the MSE wall. It provides no structural support for the mud mats or the NI itself.

The first course of the MSE wall will be placed on top of the footer at the surveyed locations required to outline the NI footprint. Inspections will be performed as required to assure that the outer dimensions of the NI are properly set.

Backfill around the outer sides of the MSE wall will commence as required by the designer of the MSE wall, with the standard large compaction equipment being used away from the wall, and smaller equipment providing the required compaction at the edges of the wall. During backfill placement and compaction, the backfill surface will be sloped away from the NI to drain surface water away from the NI excavation volume. Additional courses of the MSE wall will be added until final plant grade is reached.

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1. It should be noted that an evaluation of the radwaste building was made to consider its impact on the nuclear island or collapse in the safe shutdown earthquake; it was concluded that it would not impair the integrity of the nuclear island (see Subsection 3.7.2.8.2).

*NRC Staff approval is required prior to implementing a change in this information.

In parallel with the construction of the MSE wall, work within the NI footprint will continue. Temporary features to provide removal of surface water within the confined area of the NI will be installed as required. These features may include plastic sheeting, temporary sumps and pumps. In addition, the surface may be sloped to provide adequate drainage.

After the first few courses of the MSE wall have been placed, the backfill within the NI volume will be prepared for placement of the mud mat. Temporary drainage features will be removed, and material will be removed or added as required to establish the final elevation for the mud mat placement.

The first mud mat will consist of a 6-inch layer of non-reinforced concrete placed uniformly within the confines of the MSE wall. No additional formwork will be required. When this lower mud mat slab has reached the specified strength, a layer of waterproof membrane will be applied to the entire top of the slab, and extended vertically up the face of the MSE wall surface. The top portion of the mud mat slab (also a 6-inch layer of non-reinforced concrete) will then be placed, sandwiching the waterproof membrane. Additional detail concerning the waterproof membrane is provided in following Subsection 3.8.5.1.1. Figure 2.5-390 provides an illustration of the location of the waterproof membrane. Rebar and foundation embedments are not incorporated in either of these mud mats; therefore installation of such elements will not puncture the waterproofing membrane.

3.8.5.1.1 Waterproof Membrane

For Vogtle Electric Generating Plant (VEGP) Units 3 and 4 an alternate waterproofing system is presented as a departure from the DCD design. The alternate waterproofing system is an elastomeric “spray-on” waterproof membrane. The membrane is applied as a high-viscosity liquid that cures after exposure to air. This material may be applied by brush, roller or airless spray equipment.

Prior to procurement of the membrane material, a qualification program will be developed to demonstrate that the selected material will meet the waterproofing and friction requirements. This qualification program will address, as a minimum, the following:

- chemical properties of the membrane material,
- physical properties of the membrane material,
- surface finish requirements for the lower mudmat, and
- installation procedures necessary to achieve the required properties and coefficients of friction.

The qualification program will include testing to demonstrate that the ITAAC design commitment in Table 3.8-201 for friction coefficient has been met. Testing methods will simulate field conditions to demonstrate that a minimum 0.7 coefficient of friction is achieved by the mudmat waterproof membrane structural interface. A technical report will be provided for the ITAAC to document the basis for determining that the material will meet the required friction factor. Application procedures will be developed based on the results of qualification testing to assure that the conditions and assumptions of the qualification tests are maintained during product application.

Based upon the qualification program requirements, it is concluded that the installed waterproof membrane will provide a level of protection from external flooding and meet the coefficient of friction at the waterproof membrane-mudmat interface that is consistent with that of the existing DCD design.

The elastomeric waterproof membrane will be applied to the entire surface of the lower mudmat and the inner face of the MSE wall. Final thickness of the membrane will be specified based on the

physical properties of the selected material but is expected to be on the order of 80 to 120 mils. The membrane may be applied in multiple coats to achieve the required thickness.

The surface of both the mudmat and the MSE wall will be prepared in accordance with procedures that are consistent with the surface preparation requirements determined during the material qualification testing program. At the transition between the lower mudmat and the MSE wall, a small transition (chamfer or fillet) between the mudmat and wall will be provided to allow a smooth transition for the membrane.

The surface of the MSE wall will be prepared as necessary to assure that the waterproof coating application can bridge the small gaps and corners of the MSE blocks. This preparation process will likely include attaching a geo-textile material to the wall prior to application of the membrane material. It should be noted that the cured membrane has a degree of flexibility which allows it to accommodate thermal expansion and other minor movements between substrate members.

The application procedures will address all aspects of the coating application including batch qualification, surface preparation, application techniques, film thickness, cure time, and repair procedures.

The final mudmat will be placed on top of the waterproof membrane. Procedures will address inspection and testing as required to assure that the membrane surface will meet the required coefficient of friction.

The gap between the MSE wall and the NI wall is sealed to prevent water intrusion.

3.8.5.2 Applicable Codes, Standards, and Specifications

The applicable codes, standards, and specifications are described in Subsection 3.8.4.2.

3.8.5.3 Loads and Load Combinations

Loads and load combinations are described in Subsection 3.8.4.3. The loads and load combinations used in the analysis are considered to be part of the method of evaluation. As described in Subsection 3.8.2.1.2, the bottom head of the steel containment vessel is the same as the upper head and is capable of resisting the containment internal pressure without benefit of the nuclear island basemat. However, containment pressure loads affect the nuclear island basemat since the concrete is stiffer than the steel head. The containment design pressure is included in the design of the nuclear island basemat as an accident pressure in load combinations 5, 6, and 7 of Table 3.8.4-2. In addition to the load combinations described in Subsection 3.8.4.3, the nuclear island is checked for resistance against sliding and overturning due to the safe shutdown earthquake, winds and tornados, and against flotation due to floods and groundwater according to the load combinations presented in Table 3.8.5-1.

The effect of the groundwater level has been studied extensively. For the AP1000, Westinghouse has performed a time history analysis using a saturated and unsaturated soft-medium soil profile (Poisson’s ratio = 0.35) and compared the floor response spectra of the two analyses. Generic SSI analyses for the AP1000 assume the water table to be at grade level with saturated soil properties supporting the nuclear island. The unsaturated soil profile was produced where the water table was assumed to be well below the nuclear island.

The results of this analysis concluded that the depth of the water table used for SSI analyses has a negligible effect on the floor response spectra at the key nodes. Since the floor response spectra differences between the two models are negligible, no additional analyses are required to compare member forces or deformations.

3.8.5.4 Design and Analysis Procedures

The seismic Category I structures are concrete, shear-wall structures consisting of vertical shear/ bearing walls and horizontal floor slabs. The walls carry the vertical loads from the structure to the basemat. Lateral loads are transferred to the walls by the roof and floor slabs. The walls then transmit the loads to the basemat. The walls also provide stiffness to the basemat and distribute the foundation loads between them.

The design of the basemat consists primarily of applying the design loads to the structures, calculating shears and moments in the basemat, and determining the required reinforcement. For a site with hard rock below the underside of the basemat vertical loads are transmitted directly through the basemat into the rock. Horizontal loads due to seismic are distributed on the underside of the basemat resulting primarily in small membrane forces in the mat. The 6-foot-thick basemat is designed for the upward hydrostatic pressure due to groundwater reduced by the downward deadweight of the mat.

[*Seismic loads for the evaluation of the basemat of the Nuclear Island are developed from the results of the global seismic analyses on hard rock. They are specified as equivalent static accelerations.*

The equivalent static accelerations used in the non-linear design analyses of the nuclear island basemat are evaluated for the revised design with the enhanced shield building by comparing total base reactions and bearing pressures in a linear analysis using these equivalent static accelerations to those from a dynamic analysis of the nuclear island 3D finite element model. A time history fixed base analysis of the model is performed using time history inputs that envelope the basemat response given by the 3D SASSI analyses at the corners and centers of the basemat for all the specified generic soil cases.]*

The basemat reactions for the equivalent static analyses compare well against those of the “all soils” time history. [*Bearing pressures are calculated from the basemat reactions assuming a rigid basemat for dead live and seismic loads. Seismic loads are considered using the 1.0, 0.4, 0.4 combination method.*]* The bearing pressures resulting from the equivalent static accelerations are similar to those due to the “all soils” time history analysis demonstrating the adequacy of the equivalent static accelerations applied in the basemat analyses.

3.8.5.4.1 Analyses for Loads during Operation

The analyses of the basemat use the three-dimensional ANSYS finite element models of the auxiliary building and containment internal structures, which are described in Subsections 3G.2.1.1 and 3G.2.1.2 and shown in Figures 3G.2-1 and 3G.2-2. The model considers the interaction of the basemat with the overlying structures and with the soil. Provisions are made in the model for two possible uplifts. One is the uplift of the containment internal structures from the lower basemat. The other is the uplift of the basemat from the soil.

The three-dimensional finite element model of the basemat includes the structures above the basemat and their effect on the distribution of loads on the basemat. The finite element model of the basemat is shown on sheet 1 of Figure 3.8.5-2.

The subgrade is modeled with one vertical spring and two horizontal springs at each node of the basemat. The vertical springs act in compression only. The horizontal springs are active when the vertical spring is closed and inactive when the vertical spring lifts off. The analyses of the basemat accounted for the range of soil sites described in Section 2.5. Horizontal bearing reactions on the side walls below grade are conservatively neglected.

*NRC Staff approval is required prior to implementing a change in this information.

The nuclear island basemat below the containment vessel, and the containment internal structures basemat above the containment vessel, are simulated with solid tetrahedral elements. Nodes on the two basemats are connected with spring elements normal to the theoretical surface of the containment vessel.

Normal and extreme environmental loads and containment pressure loads are considered in the analysis. The normal loads include dead loads and live loads. Extreme environmental loads include the safe shutdown earthquake.

Dead loads are applied as inertia loads. Live loads and the safe shutdown earthquake loads are applied as concentrated loads on the nodes. The safe shutdown earthquake loads are applied as equivalent static loads using the assumption that while maximum response from one direction occurs, the responses from the other two directions are 40 percent of the maximum. Combinations of the three directions of the safe shutdown earthquake are considered.

Linear analyses are performed for all specified load combinations assuming that the soil springs can take tension. Critical load cases are then selected for non-linear analyses with basemat liftoff based on the results of the linear cases. The results from the analysis include the forces, shears, and moments in the basemat; the bearing pressures under the basemat; and the area of the basemat that is uplifted. Reinforcing steel areas are calculated from the member forces for each load combination case.

The required reinforcing steel for the portion of the basemat under the auxiliary building and under the shield building is determined by considering the reinforcement envelope for the full non-linear iteration of the most critical load combination cases. Additional reinforcement is provided in the design of the 6′ mat for soil sites such that the basemat can resist loads 20 percent greater than the demand calculated by the equivalent static acceleration analyses on uniform soil springs. This increase accommodates potential site specific lateral variability of the soil investigated separately in a series of parametric studies. Figure 3.8.5-3 shows the basemat reinforcement.

3.8.5.4.2 Analyses of Settlement During Construction

Construction loads are evaluated in the design of the nuclear island basemat. This evaluation is performed for soil sites meeting the site interface requirements of Subsection 2.5.4 at which settlement is predicted to be maximum. In the expected basemat construction sequence, concrete for the mat is placed in a single placement. Concrete may be placed below the sumps and elevator pits, if determined to enhance constructability, prior to the remaining single pour of the nuclear island basemat. The placement includes the first 6 feet of the thicker basemat below the containment vessel and shield building, but excludes the central zone directly below the bottom of the containment vessel. Construction continues with a portion of the shield building foundation and containment internal structure and the walls of the auxiliary building. The critical location for shear and moment in the basemat is around the perimeter of the shield building. Once the shield building and auxiliary building walls are completed to elevation 82′-6″, the load path changes and loads are resisted by the basemat stiffened by the shear walls.

The analyses account for the construction sequence, the associated time varying load and stiffness of the nuclear island structures, and the resulting settlement time history. To maximize the potential settlement, the analyses consider a 360 feet deep soft soil site with soil properties consistent with the soft soil case described in Subsection 2A.2. Two soil profiles are analyzed to represent limiting foundation conditions, and address both cohesive and cohesionless soils and combinations thereof:

- A soft soil site with alternating layers of sand and clay. The assumptions in this profile maximize the settlement in the early stages of construction and maximize the impact of dewatering.

- A soft soil site with clay. The assumptions maximize the settlement during the later stages of construction and during plant operation.

The analyses focus on the response of the basemat in the early stages of construction when it could be susceptible to differential loading and deformations. As subsequent construction incorporates concrete shear walls associated with the auxiliary building and the shield building, the structural system significantly strengthens, minimizing the impact of differential settlement. The displacements, and the moments and shear forces induced in the basemat are calculated at various stages in the construction sequence. These member forces are evaluated in accordance with ACI 349 using the load factors given in Table 3.8.4-2 Three construction sequences are examined to demonstrate construction flexibility within broad limits.

- A base construction sequence which assumes no unscheduled delays. The site is dewatered and excavated. Concrete for the basemat is placed. Concrete for the exterior walls below grade is placed after the basemat is in place. Exterior and interior walls of the auxiliary building are placed in 16 to 18-foot lifts.

- A delayed shield building case which assumes a delay in the placement of concrete in the shield building while construction continues in the auxiliary building. This bounding case maximizes tension stresses on the top of the basemat. The delayed shield building case assumes that no additional concrete is placed in the shield building after the pedestal for the containment vessel head is constructed. The analysis incorporates construction in the auxiliary building to elevation 117′-6″ and filling the CA20 module with concrete to elevation 135′-3″, and thereafter assumes that construction is suspended.

- A delayed auxiliary building case which assumes a delay in the construction of the auxiliary building while concrete placement for the shield building continues. This bounding case maximizes tension stresses in the bottom of the basemat. The delayed auxiliary building case assumes construction being suspended in the shield building at elevation 100′-0″. Resumption of construction of the shield building continues once the auxiliary building is completed to elevation 82′-6″, excluding wall N, with construction suspended at elevation 120′-0″, including setting of and concrete placement in CA01. Resumption in construction of the shield building above elevation 120′-0″ can proceed once the auxiliary building is completed to elevation 100′-0″ including wall N.

For the base construction sequence, the largest basemat moments and shears occur at the interface with the shield building before the connections between the auxiliary building and the shield building are credited. Once the shield building and auxiliary building walls are completed to elevation 82′-6″, the load path for successive loads changes and the loads are resisted by the basemat stiffened by the shear walls. Dewatering is discontinued once construction reaches grade, resulting in the rebound of the subsurface.

Of the three construction scenarios analyzed, the delayed auxiliary building case results in the largest demand for the bottom reinforcement in the basemat. The delayed shield building results in the largest demand for the top reinforcement in the basemat. The analyses of the three construction sequences demonstrate the following:

- The design of the basemat and superstructure accommodates the construction-induced stresses considering the construction sequence and the effects of the settlement time history.

- The design of the basemat can accommodate delays in the shield building so long as the auxiliary building construction is suspended at elevation 117′-0″. Resumption in construction of the auxiliary building can proceed once the shield building is advanced to elevation 100′-0″.

- The design of the basemat can accommodate delays in the auxiliary building so long as the shield building construction is suspended at elevation 100′-0″. Resumption in construction of the shield building can proceed once the auxiliary building is completed to elevation 82′-6″, excluding wall N, with construction suspended at elevation 120′-0″. Resumption in construction of the shield building above elevation 120′-0″ can proceed once the auxiliary building is completed to elevation 100′-0″ including wall N.

- After the structure is in place and cured to elevation 100′-0″, the basemat and structure act as an integral 40 foot deep structure and the loading due to construction above this elevation is not expected to cause significant additional flexural demand with respect to the basemat and the shield building concrete below the containment vessel. Accordingly, there is no need for placing constraints on the construction sequence above elevation 100′-0″.

The site conditions considered in the evaluation provide reasonable bounds on construction induced stresses in the basemat. Accordingly, the basemat design is adequate for practically all soil sites and it can tolerate major variations in the construction sequence without causing excessive deformations, moments and shears due to settlement over the plant life.

The analyses of alternate construction scenarios considered the softest material properties satisfying the shear wave velocity limit of 1000 feet per second. These analyses show that member forces in the basemat are acceptable subject to the limits shown below on the relative level of construction of the buildings. Construction of the AP1000 will satisfy the limits shown below, or a site-specific analysis of settlement and member forces will be completed. These limits do not apply to AP1000 units with a soil profile where the shear wave velocity exceeds 7500 feet per second.

Prior to completion of the shield building at elevation 82′-6″:

- Concrete may not be placed above elevation 117′-6″ in the auxiliary building, except in the CA20 structural module, where it may be placed to elevation 135′-3″.

Member forces in the basemat considering settlement during construction differ from those obtained from the analyses on uniform elastic soil springs described in Subsection 3.8.5.4.1.

Although the bearing pressures at the end of construction are similar in the two analyses, the resulting member forces differ due to the progressive changes in structural configuration during construction. The design using the results of the analyses of Subsection 3.8.5.4.1 provides sufficient structural strength to resist the specified loads including bearing reactions on the underside of the basemat. The member forces in these analyses are those due to primary externally applied loads and do not consider secondary stresses and strains locked in during early stages of construction. A confirmatory evaluation was performed to demonstrate that the member forces due to design basis loads, including locked-in forces due to construction settlement, remain within the capacity of the section. The evaluation was performed for critical locations which were selected as locations where the effect of locked in member forces were judged to be most significant.

The governing scenario is the case with a delay in the auxiliary building construction for the soft soil site with alternating layers of sand and clay. The delay is postulated to occur just prior to the stage where the auxiliary building walls are constructed. Member forces at the end of construction are calculated considering the effects of settlement during construction. The difference in these member forces from those calculated for dead load in the analyses on soil springs are added as additional dead loads in the critical safe shutdown earthquake load combination.

The member forces for the load combination of dead load plus safe shutdown earthquake, including the member forces locked-in during various stages of plant construction, are within the design

capacity for the five critical locations. The evaluation demonstrates that the member forces including locked-in forces calculated by elastic analyses remain within the capacity of the section.

3.8.5.4.3 Design Summary Report

A design summary report is prepared for the basemat documenting that the structures meet the acceptance criteria specified in Subsection 3.8.5.5.

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 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 judgement to performance of a revised analysis and design. The results of the evaluation will be documented in an as-built summary report.

3.8.5.4.4 Design Summary of Critical Sections

The basemat is designed to meet the acceptance criteria specified in Subsection 3.8.4.5. Two critical portions of the basemat are identified below together with a summary of their design. The boundaries are defined by the walls and column lines which are shown in Figure 3.7.2-12 (sheet 1 of 12). Table 3.8.5-3 shows the reinforcement required and the reinforcement provided for the critical sections.

[*<u>Basemat between column lines 9.1 and 11 and column lines K and L</u>*

This portion of the basemat is designed as a two way slab with the shorter directions spanning a distance of 23′6″ between the walls on column lines K and L. The slab is continuous with the adjacent slabs to the east and west. The critical loading is the bearing pressure on the underside of the slab due to dead and seismic loads. This establishes the demand for the top flexural reinforcement at mid span and for the bottom flexural and shear reinforcement at the walls. The basemat is designed for the member forces from the analyses]* described in Subsection 3.8.5.4.1. [*The top and bottom reinforcement in the east west direction of span are equal. The reinforcement provided is shown in sheets 1, 2 and 5 of Figure 3.8.5-3. Typical reinforcement details showing use of headed reinforcement for shear reinforcement are shown in Figure 3H.5-3.*

<u>Basemat between column lines 1 and 2 and column lines K-2 and N</u>

This portion of the basemat is designed as a two way slab with the shorter direction spanning a distance of 22′0″ between the walls on column lines 1 and 2. The slab is continuous with the adjacent slabs to the north and with the exterior wall to the south. The critical loading is the bearing pressure on the underside of the slab due to dead and seismic loads. This establishes the demand for the top flexural reinforcement at mid span and for the bottom flexural and shear reinforcement at wall 2. The basemat is designed for the member forces from the analyses on uniform soil springs]* described in Subsection 3.8.5.4.1. [*The reinforcement provided is shown in sheets 1, 2 and 5 of Figure 3.8.5-3. Typical reinforcement details showing use of headed reinforcement for shear reinforcement are shown in Figure 3H.5-3.*]*

Figure 3.8.5-3 shows minimum design reinforcement for the concrete in the areas defined. This location information does not define the exact length or position of the reinforcement bars. The

*NRC Staff approval is required prior to implementing a change in this information.

detailed reinforcement design, including requirements for detailing and concrete cover, is in conformance with ACI-349 requirements for foundations. The reinforcement arrangement implemented may have a greater amount of reinforcement than that shown in the figure. The size and spacing of the reinforcement bars may vary from that shown as long as the reinforcement area provided is equal to or greater than the area for the size and spacing identified in the figure. Incorporation of embedments, drains and other piping, support structures, and other obstructions may result in local variations from the reinforcement arrangement shown. 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 amplitude of the seismic floor response spectra do not exceed the design basis floor response spectra by more than 10 percent.

Depending on the extent of the deviations, the evaluation may range from documentation of an engineering judgement to performance of a revised analysis and design.

3.8.5.5 Structural Criteria

The analysis and design of the foundation for the nuclear island structures are according to ACI-349 with margins of structural safety as specified within it. The limiting conditions for the foundation medium, together with a comparison of actual capacity and estimated structure loads, are described in Section 2.5. The minimum required factors of safety against sliding, overturning, and flotation for the nuclear island structures are given in Table 3.8.5-1.

[*The design and construction of anchors and embedments conform to the procedures and standards of Appendix B to ACI 349-01*]* and are in conformance with the regulatory positions of NRC Regulatory Guide 1.199, Revision 0. [*Alternative requirements to ACI 349, Appendix B apply to the anchoring of headed shear reinforcement for the basemat (see Subsection 3.8.4.4.1).*]*

[*The basemat below the auxiliary building is designed for shear in accordance with the following supplemental provisions which are based on requirements for continuous deep flexural members in ACI 349-01. As permitted by paragraph 11.5.5.1 of ACI 349-01, shear reinforcement is not provided when the factored shear force, V_u, is less than one half of the shear strength provided by the concrete, φV_c.*

- *The design for shear is based on 11.1 through 11.5 of ACI 349-01 except that the critical section measured from the face of the support is taken at a distance of 0.15 l_n.*
- *Shear strength, V_n, is not taken greater than $8\sqrt{f_c'}\, b_w d$ when l_n/d is less than 2.*
  – When l_n/d is between 2 and 5, $V_n = 2/3\,(10 + l_n/d)\,\sqrt{f_c'}\, b_w d$
- *Minimum vertical shear reinforcement is provided in each bay. The area of vertical shear reinforcement, A_v, is not less than $0.0015\, b_w s$*
  – Spacing of shear reinforcement, s, based on provisions in Paragraph 11.5.4.1 does not exceed d/2, nor 24 in.
- *Shear reinforcement required at the critical section is used throughout the span.*

The terms φ, d, l_n, V_c, A_v, b_w, s, and f_c' are defined in ACI 349.]*

*NRC Staff approval is required prior to implementing a change in this information.

3.8.5.5.1 Nuclear Island Maximum Bearing Pressures

The foundation will be demonstrated to be capable of withstanding the bearing demand from the nuclear island as described in Subsection 2.5.4.10.1.

3.8.5.5.2 Flotation

The factor of safety against flotation of the nuclear island is shown in Table 3.8.5-2 and is calculated as follows:

$$F.S. = \frac{D}{(F \text{ or } B)}$$

where:

F.S.	=	factor of safety against flotation
D	=	total weight of structures and foundation
F	=	buoyant force due to the design basis flood
B	=	buoyant force due to high ground water table

3.8.5.5.3 Sliding

The factor of safety against sliding of the nuclear island during a tornado or a design wind is shown in Table 3.8.5-2 and is calculated as follows:

$$F.S. = \frac{F_S}{F_H}$$

where:

F.S.	=	factor of safety against sliding from tornado or design wind
F_S	=	shearing or sliding resistance at bottom of basemat
F_H	=	maximum lateral force due to active soil pressure, including surcharge, and tornado or design wind load

The factor of safety against sliding of the nuclear island during a safe shutdown earthquake is shown in Table 3.8.5-2 and is calculated as follows:

$$F.S. = \frac{F_S}{F_D}$$

where:

F.S.	=	factor of safety against sliding from a safe shutdown earthquake
F_S	=	shearing or sliding resistance at bottom of basemat

F_D = seismic force from safe shutdown earthquake

The sliding resistance is based on the friction force developed between the basemat and the foundation with a static coefficient of friction of 0.55. The governing friction value in the soil below the mudmat has an angle of internal friction of 35°. The effect of buoyancy due to the water table is included in calculating the sliding resistance. Passive soil pressure resistance is not included in the equations above because passive pressure is not considered for sliding stability. Since there is no passive pressure considered, active and overburden soil pressures are also not considered.

3.8.5.5.4 Overturning

The factor of safety against overturning of the nuclear island during a tornado or a design wind is shown in Table 3.8.5-2 and is calculated as follows:

$$F.S. = \frac{M_R}{M_O}$$

where:

F.S. = factor of safety against overturning from tornado or design wind

M_R = resisting moment

M_O = overturning moment of tornado or design wind

The factor of safety against overturning of the nuclear island during a safe shutdown earthquake is shown in Table 3.8.5-2 and is evaluated using the time history analysis assuming overturning about the edge of the nuclear island at the bottom of the basemat. The factor of safety is defined as follows:

$$F.S. = (M_R)/(M_O + M_{AO})$$

where:

F.S. = factor of safety against overturning from a safe shutdown earthquake

M_R = nuclear island's resisting moment against overturning

M_O = maximum safe shutdown earthquake induced overturning moment acting on the nuclear island, applied as a static moment

M_{AO} = Moment due to lateral forces caused by active and overburden pressures

The resisting moment is equal to the nuclear island dead weight, minus buoyant force from ground water table, multiplied by the distance from the edge of the nuclear island to its center of gravity. The overturning moment is the maximum moment about the same edge from the time history analyses of the nuclear island NI20 model described in Subsection 3.7.2 and Appendix 3G.2. Resistance moment due to passive pressure is not included in the equation above because passive pressure is not considered for overturning stability.

3.8.5.5.5 Seismic Stability Analysis

The factors of safety for sliding and overturning for the SSE are calculated for each soil case for the base reactions in terms of shear and bending moments about column lines 1, 11, I, and the west side of the shield building at each time step of the seismic time history. The 2D SASSI reactions (Fx, Fy, and Fz) are used to obtain seismic response factors between the hard rock (HR) case to the UBSM soil case, and the SM soil case. These factors are used to adjust the hard rock time history to reflect the seismic response for the other two potential governing soil cases UBSM and SM. The firm rock, soft rock, and soft soil cases have higher factors of safety against sliding and, therefore, are not considered.

A non-linear analysis with sliding friction elements using a 2D ANSYS model was performed. The 2D ANSYS model that was used to study the basemat uplift (see Subsection 3.8.5.5.6 and Appendix 3G.2.2.5). This 2D non-linear model is for the east-west direction. There is no need to consider the north-south direction since the nuclear island deflections calculated to maintain a factor of safety of 1.1 are largest in the east-west direction. This model was modified introducing friction elements along the bottom of the basemat and soil media interface. Direct time integration analysis was performed with vertical uplift and sliding allowed. The three cases that have the lowest factor of safety related to sliding were evaluated. These three cases are HR, UBSM, and SM. The seismic input was increased by 10 percent to maintain the factor of safety against sliding of 1.1. No passive soil resistance is considered. The resulting maximum displacement at the base of the nuclear island basemat (elevation 60.5′) using a coefficient of friction of 0.55 is 0.12 inch without buoyant force consideration, and 0.19 inch with buoyant force considered. This is negligible sliding during the seismic event, and no passive soil resistance is necessary from the backfill (side soil). Therefore, it can be concluded that the nuclear island is stable against sliding.

The minimum seismic stability factors of safety values are reported in Table 3.8.5-2.

3.8.5.5.6 Effect of Nuclear Island Basemat Uplift on Seismic Response

The effects of basemat uplift were evaluated using an east-west lumped-mass stick model of the nuclear island structures supported on a rigid basemat with nonlinear springs (see Appendix 3G.2.2.5). Floor response spectra from safe shutdown earthquake time history analyses, which included basemat uplift, were compared to those from analyses that did not include uplift. The comparisons showed that the effect of basemat uplift on the floor response spectra is not significant.

The design analyses of the nuclear island basemat include consideration of sliding and liftoff between the containment internal structures and the containment vessel and of sliding between the containment vessel and the nuclear island basemat. Analyses of stability demonstrate that there was no uplift or sliding at the interface of the containment internal structures and the containment vessel.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The materials and quality control program used in the construction of the nuclear island structures foundation are described in Subsection 3.8.4.6.

There are no special construction techniques used in the construction of the nuclear island structures foundation. Subsection 2.5.4.2 describes information related to the excavation, backfill, and mudmat.

3.8.5.7 In-Service Testing and Inspection Requirements

The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

The need for foundation settlement monitoring is site-specific as discussed in subsection Subsection 2.5.4.10.2.

3.8.5.8 Construction Inspection

Construction inspection is conducted to verify the concrete wall thickness and quantity of concrete reinforcement. The construction inspection includes concrete wall thickness and reinforcement expressed in units of in^2/ft (linear length) equivalent when compared to standard reinforcement bar sections. Inspections will be measured at applicable sections excluding designed openings or penetrations. Inspections will confirm that each section provides the minimum required reinforcement and concrete thickness as shown in Table 3.8.5-3. The minimum required reinforcement and concrete thickness represent the required minimum values to meet the design basis loads. Table 3.8.5-3 also indicates the reinforcement provided which may exceed the required minimum reinforcement for the following reasons:

- Structural margin
- Ease of construction
- Use of standardized reinforcement sizes and spacing

3.8.6 Combined License Information

3.8.6.1 Containment Vessel Design Adjacent to Large Penetrations

The final design of containment vessel elements (reinforcement) adjacent to concentrated masses (penetrations) is addressed in APP-GW-GLR-005 (Reference 53).

3.8.6.2 Passive Containment Cooling System Water Storage Tank Examination

Not used.

3.8.6.3 As-Built Summary Report

Not used.

3.8.6.4 In-Service Inspection of Containment Vessel

Not used.

3.8.6.5 Structures Inspection Program

The structures inspection program to address maintenance requirements for the seismic Category I and seismic Category II structures is addressed in Subsections 3.8.3.7, 3.8.4.7, 3.8.5.7, and 17.6.

3.8.6.6 Construction Procedures Program

Construction and inspection procedures for concrete filled steel plate modules address activities before and after concrete placement, use of construction mock-ups, and inspection of modules before and after concrete placement as discussed in Subsection 3.8.4.8. The procedures will be made available to NRC inspectors prior to use.

3.8.7 References

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*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.2-1
Load Combinations and Service Limits for Containment Vessel

Load Description		Load Combination and Service Limit										
		Con	**Test**	**Des.**	**Des.**	**A**	**A**	**A**	**C**	**D**	**C**	**D**
Dead	D	x	x	x	x	x	x	x	x	x	x	x
Live	L	x	x	x	x	x	x	x	x	x	x	x
Wind(5)	W	x						x				
Safe shutdown earthquake	E_s								x	x		x
Tornado	W_t										x	
Test pressure	P_t		x									
Test temperature	T_t		x									
Operating pressure	P_o							x			x	
Design pressure	P_d			x			x		x			x
Design external pressure	P_e				x	x				x		
Normal reaction	R_o				x	x		x		x	x	
Normal thermal(4)	T_o				x	x		(3)		x	(3)	
Accident thermal reactions	R_a			x			x		x			x
Accident thermal	T_a			x			x		x			x
Accident pipe reactions	Y_r											x
Jet impingement	Y_j											x
Pipe impact	Y_m											x

Notes:
1. Service limit levels are per ASME-NE.
2. Where any load reduces the effects of other loads, that load is to be taken as zero, unless it can be demonstrated that the load is always present or occurs simultaneously with the other loads.
3. Temperature of vessel is 70°F.
4. Temperature distribution for normal operation in cold weather.
5. Wind load for the construction load combination is based on a 70 mph wind. Wind load for the Service Level A load combination is analyzed as a reduction in external pressure.

Table 3.8.2-2
Containment Vessel Pressure Capabilities

Containment Element		Pressure Capability				
		Deterministic Severe Accident Capacity(1)			Maximum Pressure Capability(2)	
Temperature		100°F	300°F	400°F	100°F	400°F
Cylinder		135 psig	117 psig	112 psig	155 psig	129 psig
Ellipsoidal Head		104 psig	91 psig	87 psig	174 psig	144 psig
16-foot equipment hatch	F.S. = 1.67	126 psig	121 psig	118 psig	210 psig	198 psig
	F.S. = 2.50	84 psig	81 psig	79 psig		
Personnel airlocks(3)		>163 psig	>163 psig	>163 psig	>300 psig	>300 psig

Notes:
1. The buckling capacity of the ellipsoidal head is taken as 60 percent of the critical buckling pressure calculated by the BOSOR-5 nonlinear analyses; the buckling capacity at higher temperatures is calculated by reducing the capacity at 100°F by the ratio of yield at 100°F to yield at the higher temperature. Evaluations of the equipment hatch covers are shown both for ASME paragraph NE-3222 (F.S. = 2.50) and Code Case N-284 (F.S. = 1.67). Evaluations of the other elements are according to ASME Service Level C.
2. The estimated maximum pressure capability is based on minimum specified material properties.
3. The capacities of the personnel airlocks are estimated from test results.

Table 3.8.2-3
Analysis and Test Results of Fabricated Heads
(Reference 23)

	Test Model #1	Test Model #2
Cylinder radius	96.0 inches	96.0 inches
Knuckle radius	32.64 inches	32.64 inches
Spherical radius	172.8 inches	172.8 inches
Thickness	0.196 inches	0.27 inches
Head height/radius	0.5	0.5
Radius/thickness	490	356
Test initial buckling pressure	58 psig	106 psig
Test collapse pressure	229 psig	332 psig
Collapse pressure/initial buckling pressure	3.95	3.13
BOSOR-5 predicted buckling pressure	73.6 psig	106.6 psig

Table 3.8.2-4
Summary of Containment Vessel Models and Analysis Methods

Model	Analysis Method	Program	Purpose
Axisymmetric shell	Modal analysis	ANSYS	To calculate frequencies and mode shapes for comparison against stick model
Lumped mass stick model	Modal analysis	ANSYS	To create equivalent stick model for use in nuclear island seismic analyses
Axisymmetric shell	Static analyses using Fourier harmonic loads	ANSYS	To calculate containment vessel shell stresses
Axisymmetric shell	Nonlinear bifurcation	BOSOR5	To calculate buckling capacity close to base under thermal loads To calculate pressure capacity of top head
Finite element shell	Linear bifurcation	ANSYS	To study local effect of large penetrations and embedment on buckling capacity for axial and external pressure loads
Finite element shell	Modal analysis	ANSYS	To calculate frequencies and mode shapes for local effects of equipment hatches and personnel airlocks
Finite element shell	Static analyses	ANSYS	To calculate local shell stress in vicinity of the equipment hatches and personnel airlocks

Table 3.8.2-5
Maximum Absolute Nodal Acceleration (ZPA)
Steel Containment Vessel

Hard Rock Site Condition

Elevation (ft)	Maximum Absolute Nodal Acceleration, ZPA (g)					
	N-S Direction		E-W Direction		Vertical Direction	
	Mass Center	Edge	Mass Center	Edge	Mass Center	Edge
281.90	1.48		1.56		1.25	
273.83	1.43		1.50		1.02	
265.83	1.38		1.43		0.85	
255.02	1.31		1.34		0.73	
244.21	1.23	1.28	1.26	1.30	0.68	0.71
224.00	1.09	1.13	1.11	1.17	0.66	0.68
200.00	0.90	0.94	0.94	0.98	0.61	0.63
169.93	0.69	0.71	0.72	0.75	0.53	0.55
162.00	0.63	0.65	0.67	0.68	0.51	0.53
141.50	0.49	0.50	0.54	0.54	0.45	0.47
131.68	0.43	0.44	0.47	0.48	0.41	0.44
112.50	0.40	0.41	0.37	0.38	0.35	0.40
104.12	0.38	0.40	0.38	0.40	0.32	0.38
100.00	0.38	0.40	0.39	0.41	0.31	0.34

Notes:

1. Enveloped response results at the north, south, east, and west edge nodes of the structure are shown. This is the maximum value of the response at any of these edge nodes.
2. Results at elevation 233.50′ are mid-span of the polar crane bridge.

Table 3.8.3-1
Shear and Flexural Stiffnesses of Structural Module Walls

Case	Analysis Assumption	Shear Stiffness(1),(2)				Flexural Stiffness(1),(2)			
		48" Wall		30" Wall		48" Wall		30" Wall	
		GA $\times 10^6$ lbs	Ratio	GA $\times 10^6$ lbs	Ratio	EI $\times 10^9$ lbs. in^2	Ratio	EI $\times 10^9$ lbs. in^2	Ratio
1	Monolithic section considering steel plates and uncracked concrete. For shear stiffness this is ($A_c G_c + A_s G_s$).	83.5	1.0	55.8	1.0	47.5	1.0	13.6	1.0
2	Uncracked gross concrete section (full wall thickness considering steel plate as concrete)	73.9	0.89	46.2	0.83	33.2	0.70	8.1	0.60
3	Transformed cracked section considering steel plates and concrete (no concrete tension stiffness)	25.0	0.30	22.6	0.41	22.1	0.47	8.0	0.59

Notes:
1. The shear stiffness, GA, is calculated for the full thickness of wall. The flexural stiffness is calculated per unit length of the wall.
2. Stiffness calculations are based on the following material properties: E_c = 3,605,000 psi,n = 8, v_c = 0.17, v_s = 0.30

Table 3.8.3-2
Summary of Containment Internal Structures
Models and Analysis Methods

Computer Program and Model	Analysis Method	Purpose	Concrete Stiffness (1)
3D ANSYS finite element of containment internal structures	Static	To obtain the in-plane and out-of-plane mechanical forces for the design of floors and walls (dead, live, hydrostatic, pressure)	Monolithic Case 1 with E_c reduced by factor of 0.8
3D ANSYS finite element of containment internal structures	Response spectra analyses(2)	To obtain the in-plane and out-of-plane seismic forces for the design of floors and walls	Monolithic Case 1 with E_c reduced by factor of 0.8.
3D ANSYS finite element of containment internal structures	Static analyses	To obtain the in-plane and out-of-plane member forces for thermal loads	Cracked Case 3
The following AP600 analyses are used as background to develop the AP1000 design loads.			
3D ANSYS finite element of containment internal structures fixed at elevation 103′-0″	Harmonic analyses	To evaluate natural frequencies potentially excited by hydrodynamic loads	Uncracked Case 2
	Time history analyses	To obtain dynamic response of IRWST boundary for hydrodynamic loads	Monolithic and cracked Cases 1 & 3

Note:
1. See Table 3.8.3-1 for stiffness case description.
2. See Section 3.7 for discussion of the containment internal structures seismic analyses.

Table 3.8.3-3
Definition of Critical Locations and Thicknesses for Containment Internal Structures(1)(4)

Wall Description (see detail in subsection 3.8.3.5.8.1)	Applicable Column Lines	Applicable Elevation Range	[*Concrete Thickness*(2)]*	Required Thickness of Surface Plates (inches)(3)	[*Thickness of Surface Plates Provided (inches)*]*
Containment Structures					
Module Wall 1	Southwest wall of refueling cavity	Wall separating IRWST and refueling cavity from elevation 103′ to 135′-3″	[*4′-0" concrete-filled structural wall module with 0.5-in.-thick steel plate on inside and outside of wall*]*	0.12	[*0.5 -0.01 +0.1*]*
Module Wall 2	South wall of west steam generator compartment	Wall separating IRWST and west steam generator compartment from elevation 103′ to 135′-3″	[*2′-6" concrete-filled structural wall module with 0.5-in.-thick steel plate on inside and outside of wall*]*	0.43	[*0.5 -0.01 +0.1*]*
CA02 Module Wall	North east boundary wall of IRWST	Wall separating IRWST and maintenance floor from elevation 103′ to 135′-3″	[*2′-6" concrete-filled structural wall module with 0.5-in.-thick steel plate on inside and outside of wall*]*	0.36	[*0.5 -0.01 +0.1*]*

Notes:

1. The applicable column lines and elevation levels are identified and included in Figures 1.2-9, 3.7.2-12 (sheets 1 through 12), 3.7.2-19 (sheets 1 through 3) and on Table 1.2-1.

[2. *The concrete thickness includes the steel face plates. Thickness greater than 3'-0" have a construction tolerance of +1", -3/4". Thickness less than or equal to 3'-0" have a construction tolerance of +1/2", -3/8".*]*

3. These plate thicknesses represent the thickness required for operating and design basis loads except for designed openings or penetrations. These values apply for each face of the applicable wall unless specifically indicated on the table. For load combinations with thermal loads, the evaluation is performed as described in Subsection 3.8.3.5.3.4.

[4. *As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thicknesses.*]*

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-4 (Sheet 1 of 3)
Design Summary of Southwest Wall of Refueling Cavity Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Mid-Height

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	1	-20	1	1	1	0	0	0	–
Hydro (F)	2	0	1	24	30	-2	0	0	–
Live (L)	0	-7	0	3	2	0	0	1	During refueling
Live (L_o)	0	-2	0	1	1	0	0	0	During operation
ADS	2	7	6	19	19	-4	0	1	–
E_s	13	49	56	41	40	15	1	8	During operation
Thermal (T_0)	-74	-34	-47	141	113	0	4	-1	During operation
LC (1)	8	-18	14	70	76	-8	0	3	[*$1.4D+1.4F+1.7L_o+1.7ADS$*]*
LC (2)	5	-40	3	41	46	-2	0	2	[*$1.4D+1.4F+1.7L_r$*]*
LC (3)	8	-15	14	69	76	-8	0	3	[*$1.4D+1.4F+1.7ADS$*]*
LC (4)	18	35	65	87	90	17	1	10	[*$D+F+L_o+\|ADS\|+E_s$*]*
LC (5)	-12	-78	-60	-34	-28	-20	-1	-8	[*$D+F+L_o-\|ADS\|-E_s$*]*
LC (6)	-56	0	18	228	203	17	5	9	[*$D+F+L_o+\|ADS\|+T_0+E_s$*]*
LC (7)	-86	-112	-108	107	85	-19	3	-9	[*$D+F+L_o-\|ADS\|+T_0-E_s$*]*
LC (8)	18	39	65	86	90	17	1	10	[*$0.9D+1.0F+1.0\|ADS\|+1.0E_s$*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101870

Plate thickness required for load combinations excluding thermal: 0.12 inches (Maximum)

[*Plate thickness provided:* *0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal: 12.9 ksi

[*Yield stress at temperature:* *65.0 ksi (Minimum)*]*

Maximum stress intensity range for load combinations including thermal: 19.6 ksi

Allowable stress intensity range for load combinations including thermal: 130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-4 (Sheet 2 of 3)
Design Summary of Southwest Wall of Refueling Cavity Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Base

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-1	-31	1	-1	-4	0	0	1	–
Hydro (F)	3	1	2	-2	-41	-1	-1	17	–
Live (L)	-1	-6	0	0	-3	0	0	1	During refueling
Live (L_o)	0	-2	0	0	-1	0	0	0	During operation
ADS	2	6	7	-2	-32	-1	-1	10	–
E_s	11	55	65	12	87	3	3	18	During operation
Thermal (T_0)	-189	-31	-76	165	86	-2	-5	0	During operation
LC (1)	7	-35	16	-8	118	-3	-3	41	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	2	-53	5	-5	-68	-2	-1	25	[*1.4D+1.4F+1.7L_r*]*
LC (3)	7	-32	16	-8	117	-3	-3	41	[*1.4D+1.4F+1.7ADS*]*
LC (4)	15	29	75	11	73	3	4	45	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-11	-93	-68	-17	165	-6	-5	-10	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-174	-2	-1	175	160	1	-1	44	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-200	124	144	147	78	-8	-10	-11	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	16	34	74	11	74	3	4	45	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101788

Plate thickness required for load combinations excluding thermal:	0.09 inches (Maximum)
[*Plate thickness provided:*	*0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal:	15.5 ksi
[*Yield stress at temperature:*	*65.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal:	24.8 ksi
Allowable stress intensity range for load combinations including thermal:	130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-4 (Sheet 3 of 3)
Design Summary of Southwest Wall of Refueling Cavity Design Loads, Load Combinations, and Comparison to Acceptance Criteria North End Bottom Corner

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-2	-29	-7	0	-3	0	0	0	–
Hydro (F)	3	1	4	-10	-16	4	1	3	–
Live (L)	-1	-4	0	0	-1	0	0	0	During refueling
Live (L_o)	0	-1	0	0	0	0	0	0	During operation
ADS	3	-2	7	-7	-18	2	1	2	–
E_s	23	43	64	14	68	6	6	7	During operation
Thermal (T_0)	-173	-88	-42	138	94	-15	-30	13	During operation
LC (1)	6	-44	8	-27	-57	8	4	7	[*$1.4D+1.4F+1.7L_o+1.7ADS$*]*
LC (2)	0	-46	-4	-16	-28	5	3	4	[*$1.4D+1.4F+1.7L_r$*]*
LC (3)	6	-42	9	-27	-57	8	4	7	[*$1.4D+1.4F+1.7ADS$*]*
LC (4)	26	16	68	10	66	11	9	11	[*$D+F+L_o+\|ADS\|+E_s$*]*
LC (5)	-25	-74	-74	-32	-104	-4	-6	-5	[*$D+F+L_o-\|ADS\|-E_s$*]*
LC (6)	-147	-73	26	148	160	-4	-21	25	[*$D+F+L_o+\|ADS\|+T_0+E_s$*]*
LC (7)	-198	-162	-116	106	-11	-20	-36	8	[*$D+F+L_o-\|ADS\|+T_0-E_s$*]*
LC (8)	26	19	69	11	67	11	9	11	[*$0.9D+1.0F+1.0\|ADS\|+1.0E_s$*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101794

Plate thickness required for load combinations excluding thermal: 0.09 inches (Maximum)

[*Plate thickness provided:* *0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal: 11.1 ksi

[*Yield stress at temperature:* *65.0 ksi (Minimum)*]*

Maximum stress intensity range for load combinations including thermal: 21.3 ksi

Allowable stress intensity range for load combinations including thermal: 130.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-5 (Sheet 1 of 3)
Design Summary of South Wall of West Steam Generator Compartment Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Mid-Height

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-1	-23	0	1	1	0	0	0	–
Hydro (F_o)	-2	3	-5	20	22	0	0	-1	–
Live (L)	0	-7	-1	1	0	0	0	0	During refueling
Live (L_o)	0	-2	0	0	0	0	0	0	During operation
ADS	-1	9	-10	17	16	-1	0	1	–
E_s	7	65	58	30	26	5	1	4	During operation
Thermal (T_0)	-42	-53	19	69	59	2	-1	-1	During operation
LC (1)	-5	-17	-24	59	59	-2	1	0	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	-4	-39	-7	32	32	-1	1	-1	[*1.4D+1.4F+1.7L_r*]*
LC (3)	-5	-13	-24	59	59	-2	1	0	[*1.4D+1.4F+1.7ADS*]*
LC (4)	6	52	64	69	65	5	2	4	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-11	-96	-73	-26	-19	-6	-1	-6	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-36	-1	83	138	124	8	1	3	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-53	-149	-54	43	41	-4	-2	-7	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	6	56	64	68	64	5	2	4	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 104228

Plate thickness required for load combinations excluding thermal: 0.15 inches (Maximum)

[*Plate thickness provided:* *0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal: 16.5 ksi

[*Yield stress at temperature:* *36.0 ksi (Minimum)*]*

Maximum stress intensity range for load combinations including thermal: 16.5 ksi

Allowable stress intensity range for load combinations including thermal: 72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-5 (Sheet 2 of 3)
Design Summary of South Wall of West Steam Generator Compartment Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Base

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-3	-28	-1	0	2	0	0	0	–
Hydro (F)	3	4	-9	-4	-38	0	0	15	–
Live (L)	-1	-5	-1	0	-1	0	0	0	During refueling
Live (L_o)	0	-2	0	0	0	0	0	0	During operation
ADS	2	10	-10	-3	-30	0	0	9	–
E_s	14	74	52	9	62	1	1	14	During operation
Thermal (T_0)	-136	-13	43	79	65	5	4	-1	During operation
LC (1)	3	-20	-32	-10	-102	-1	-1	36	[*$1.4D+1.4F+1.7L_o+1.7ADS$*]*
LC (2)	-2	-43	-16	-5	-52	-1	0	21	[*$1.4D+1.4F+1.7L_r$*]*
LC (3)	4	-17	-32	-10	-102	-1	-1	36	[*$1.4D+1.4F+1.7ADS$*]*
LC (4)	16	58	52	9	55	1	1	38	[*$D+F+L_o+\|ADS\|+E_s$*]*
LC (5)	-17	-110	-73	-16	-128	-2	-2	-8	[*$D+F+L_o-\|ADS\|-E_s$*]*
LC (6)	-120	45	95	88	120	6	5	37	[*$D+F+L_o+\|ADS\|+T_0+E_s$*]*
LC (7)	-152	-122	-30	63	-63	3	3	-9	[*$D+F+L_o-\|ADS\|+T_0-E_s$*]*
LC (8)	16	62	52	9	55	1	1	38	[*$0.9D+1.0F+1.0\|ADS\|+1.0E_s$*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101943

Plate thickness required for load combinations excluding thermal:	0.14 inches (Maximum)
[*Plate thickness provided:*	*0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal:	17.6 ksi
[*Yield stress at temperature:*	*36.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal:	18.7 ksi
Allowable stress intensity range for load combinations including thermal:	72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-5 (Sheet 3 of 3)
Design Summary of South Wall of West Steam Generator Compartment Design Loads, Load Combinations, and Comparison to Acceptance Criteria West End Bottom Corner

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-6	-37	3	-1	6	1	-1	-3	–
Hydro (F)	5	14	-10	-6	-14	3	2	4	–
Live (L)	-2	-11	1	0	1	0	0	-1	During refueling
Live (L_o)	-1	-4	0	0	1	0	0	-1	During operation
ADS	9	36	-10	-3	-15	2	2	4	–
E_s	49	248	53	10	72	4	10	31	During operation
Thermal (T_0)	-125	-61	91	26	64	-4	-27	-23	During operation
LC (1)	14	21	-28	-15	-34	9	5	7	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	-4	-52	-8	-10	-8	6	0	-1	[*1.4D+1.4F+1.7L_r*]*
LC (3)	15	27	-28	-15	-35	9	5	8	[*1.4D+1.4F+1.7ADS*]*
LC (4)	57	256	57	6	80	10	13	35	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-59	-311	-71	-20	-93	-2	-12	-36	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	-67	195	147	32	144	6	-14	13	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	-184	-372	20	7	-29	-6	-39	-58	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	59	264	56	6	79	10	14	36	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 101933

Plate thickness required for load combinations excluding thermal:	0.43 inches (Maximum)
[*Plate thickness provided:*	*0.50 inches -0.01 +0.10*

As described in subsection 3.8.3.1.3, some CA01 and CA05 module wall faceplate thicknesses are greater than the 0.5-inch nominal faceplate thickness.]*

Maximum principal stress for load combinations including thermal:	31.9 ksi
[*Yield stress at temperature:*	*36.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal:	36.0 ksi
Allowable stress intensity range for load combinations including thermal:	72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-6 (Sheet 1 of 3)
Design Summary of North-East Wall of IRWST Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Mid-Height

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-1	-18	-1	0	1	-1	-2	-2	–
Hydro (F)	0	1	-1	16	17	3	2	0	–
Live (L)	0	-13	-1	5	5	-1	-2	-1	During refueling
Live (L_o)	0	-4	-1	4	5	-1	-2	-1	During operation
ADS	-2	3	0	16	14	5	2	1	–
E_s	16	42	47	35	39	12	11	16	During operation
Thermal (T_0)	-32	-13	48	49	51	-2	-5	-3	During operation
LC (1)	-5	-26	-3	58	58	8	0	-3	[*1.4D+1.4F+1.7*L_o*+1.7ADS*]*
LC (2)	-2	-46	-4	30	33	0	-3	-5	[*1.4D+1.4F+1.7*L_r]*
LC (3)	-5	-19	-1	51	48	10	3	-1	[*1.4D+1.4F+1.7ADS*]*
LC (4)	16	24	45	72	76	17	11	14	[*D+F+*L_o *+|ADS|+*E_s]*
LC (5)	-19	-67	-49	-31	-31	-16	-15	-21	[*D+F+*L_o *-|ADS|-* E_s]*
LC (6)	-16	11	93	121	127	15	6	11	[*D+F+*L_o *+|ADS|+*T_0*+*E_s]*
LC (7)	-51	-80	-1	18	20	-18	-20	-24	[*D+F+*L_o *-|ADS|+*T_0*-*E_s]*
LC (8)	16	29	46	68	71	18	13	16	[*0.9D+1.0F+1.0|ADS|+1.0*E_s]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 140027

Plate thickness required for load combinations excluding thermal:	0.13 inches (Maximum)
[*Plate thickness provided:*	*0.50 inches -0.01 +0.10*]*
Maximum principal stress for load combinations including thermal:	18.1 ksi
[*Yield stress at temperature:*	*36.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal:	18.1 ksi
Allowable stress intensity range for load combinations including thermal:	72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-6 (Sheet 2 of 3)
Design Summary of North-East Wall of IRWST Design Loads, Load Combinations, and Comparison to Acceptance Criteria Mid-Span at Bottom-Elevation 107'-2"

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	0	-19	1	0	0	0	0	0	–
Hydro (F)	1	2	-1	2	-8	2	0	11	–
Live (L)	-1	-7	0	0	-3	0	0	1	During refueling
Live (L_o)	-1	-1	0	0	-2	0	0	1	During operation
ADS	0	4	1	2	-7	3	0	8	–
E_s	23	54	47	13	48	8	6	17	During operation
Thermal (T_0)	-92	-58	71	64	63	-1	0	-1	During operation
LC (1)	0	-20	3	7	-27	8	-1	32	[*1.4D+1.4F+1.7L$_o$+1.7ADS*]*
LC (2)	1	-37	1	3	-16	3	-1	18	[*1.4D+1.4F+1.7L$_r$*]*
LC (3)	1	-18	3	6	-24	8	0	30	[*1.4D+1.4F+1.7ADS*]*
LC (4)	24	39	49	17	45	13	5	38	[*D+F+L$_o$ +|ADS|+E$_s$*]*
LC (5)	-23	-77	-47	-12	-65	-8	-7	-13	[*D+F+L$_o$ -|ADS|- E$_s$*]*
LC (6)	-68	-18	120	81	108	12	5	37	[*D+F+L$_o$ +|ADS|+T$_0$+E$_s$*]*
LC (7)	-114	-134	24	52	-2	-9	-7	-14	[*D+F+L$_o$ -|ADS|+T$_0$-E$_s$*]*
LC (8)	25	42	49	17	47	13	6	37	[*0.9D+1.0F+1.0|ADS|+1.0E$_s$*]*

Notes:
x-direction is horizontal; y-direction is vertical.
Element number 140005

Plate thickness required for load combinations excluding thermal:	0.12 inches (Maximum)
[*Plate thickness provided:*	*0.50 inches -0.01 +0.10*]*
Maximum principal stress for load combinations including thermal:	17.0 ksi
[*Yield stress at temperature:*	*36.0 ksi (Minimum)*]*
Maximum stress intensity range for load combinations including thermal:	21.9 ksi
Allowable stress intensity range for load combinations including thermal:	72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-6 (Sheet 3 of 3)
Design Summary of North-East Wall of IRWST Design Loads, Load Combinations, and Comparison to Acceptance Criteria
North End Bottom Corner-Elevation 107'-2"

Load/Comb.	TX k/ft	TY k/ft	TXY k/ft	MX kft/ft	MY kft/ft	MXY kft/ft	NX k/ft	NY k/ft	Comments
Dead (D)	-1	-19	1	0	-4	0	0	0	–
Hydro (F)	2	12	6	5	10	12	-7	-11	–
Live (L)	0	-10	1	-1	-4	1	1	1	During refueling
Live (L_o)	0	-3	0	0	-2	1	0	0	During operation
ADS	0	19	7	7	16	11	-5	-11	–
E_s	8	136	38	26	98	17	14	31	During operation
Thermal (T_0)	38	28	60	-39	24	-11	17	34	During operation
LC (1)	2	17	21	16	31	35	-17	-33	[*1.4D+1.4F+1.7L_o+1.7ADS*]*
LC (2)	2	-28	10	4	2	17	-8	-13	[*1.4D+1.4F+1.7L_r*]*
LC (3)	2	21	21	17	35	34	-18	-33	[*1.4D+1.4F+1.7ADS*]*
LC (4)	10	144	52	37	118	40	12	32	[*D+F+L_o +|ADS|+E_s*]*
LC (5)	-7	-165	-38	-29	-110	-16	-25	-52	[*D+F+L_o -|ADS|- E_s*]*
LC (6)	48	172	111	-2	142	29	29	66	[*D+F+L_o +|ADS|+T_0+E_s*]*
LC (7)	31	-137	22	-68	-86	-27	-8	-19	[*D+F+L_o -|ADS|+T_0-E_s*]*
LC (8)	10	149	51	37	120	39	12	32	[*0.9D+1.0F+1.0|ADS|+1.0E_s*]*

Notes:
x-direction is horizontal; y-direction is vertical.

Element number 140001

Plate thickness required for load combinations excluding thermal: 0.36 inches (Maximum)

[*Plate thickness provided:* *0.50 inches -0.01 +0.10*]*

Maximum principal stress for load combinations including thermal: 29.5 ksi

[*Yield stress at temperature:* *36.0 ksi (Minimum)*]*

Maximum stress intensity range for load combinations including thermal: 30.3 ksi

Allowable stress intensity range for load combinations including thermal: 72.0 ksi (Minimum)

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.3-7
Design Summary of Steel Wall of IRWST

Mechanical Loads Only AISC Interaction Ratio			
Section Location and Element Number	**T Section**	**L Section**	**Load Combination**
TYPICAL COLUMN AT MIDDLE OF WALL			
Top (139701)	0.13	0.64	*[D+F+L_o -|ADS|- E_s (LC#5)]**
Mid-height (139699)	0.33	0.33	*[D+F+L_o -|ADS|- E_s (LC#5)]**
	0.21	0.34	*[D+F+L_o +|ADS|+E_s (LC#4)]**
Bottom (139690)	0.69	0.09	*[D+F+L_o -|ADS|- E_s (LC#5)]**
ENVELOPE OF ALL LOCATIONS AND LOAD COMBINATIONS			
	0.94	0.94	*[D+F+ADS (LC#3)]**

Mechanical Plus Thermal Loads Ratio of Stress to AISC or ASME (2 * Sy = 110 ksi)				
Section Location and Element Number	**Flange of T Section**	**Flange of L Section**	**Plate**	**Load Combination**
TYPICAL COLUMN AT MIDDLE OF WALL				
Top (139701)	0.13 AISC	0.60 AISC	0.07 AISC	*[D+F+L_o -|ADS|+T_0-E_s (LC#7)]**
	0.07 AISC	0.061 AISC	0.47 AISC	*[D+F+L_o +|ADS|+T_0+E_s (LC#6)]**
Mid-height (139699)	0.32 AISC	0.87 AISC	0.11 AISC	*[D+F+L_o -|ADS|+T_0-E_s (LC#7)]**
Bottom (139690)	0.92 ASME	0.49 AISC	0.06 AISC	*[D+F+L_o +|ADS|+T_0+E_s (LC#6)]**
	0.28 ASME	0.67 AISC	0.23 AISC	*[D+F+L_o -|ADS|+T_0-E_s (LC#7)]**
ENVELOPE OF ALL LOCATIONS AND LOAD COMBINATIONS				
–	0.92 ASME	0.74 ASME	0.81 ASME	*[D+F+L_o +|ADS|+T_0+E_s (LC#6)]**

Note:
Results of the evaluation of mechanical and thermal loads are shown against the AISC allowables when the stresses are less than yield. Portions of the steel wall at the end of the wall exceed yield due to the restraint provided by the adjacent concrete. These areas are evaluated against the ASME allowables as described in Subsection 3.8.3.5.3.4.

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.4-1
[*LOAD COMBINATIONS AND LOAD FACTORS FOR SEISMIC CATEGORY I STEEL STRUCTURES*]*

Combination No.		***Load Combination and Factors***								
		1	***2***	***3***	***4***	***5***	***6***	***7***	***8***	***9***
Load Description										
Dead	D	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*
Liquid	F	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*
Live	L	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*
Earth pressure	H	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*
Normal reaction	R_o	*1.0*	*1.0*	*1.0*	*1.0*				*1.0*	*1.0*
Normal thermal	T_o			*1.0*	*1.0*				*1.0*	*1.0*
Wind	W		*1.0*							*1.0*
Safe shutdown earthquake	E_S			*1.0*				*1.0*		
Tornado	W_t				*1.0*					
Accident pressure	P_a					*1.0*	*1.0*	*1.0*		
Accident thermal	T_a					*1.0*	*1.0*	*1.0*		
Accident thermal reactions	R_a					*1.0*	*1.0*	*1.0*		
Accident pipe reactions	Y_r						*1.0*	*1.0*		
Jet impingement	Y_j						*1.0*	*1.0*		
Pipe impact	Y_m						*1.0*	*1.0*		
Stress Limit Coefficient [(1),(3)] *(except for compression)*		*1.0*	*1.0*	*1.6*	*1.6*	*1.6*	*1.6*	*1.7*	*1.5*	*1.5*
(for compression)		*1.0*	*1.0*	*1.4*	*1.4*	*1.4*	*1.4*	*1.6*	*1.3*	*1.3*

Notes:

1. *Allowable stress limits coefficients are applied to the basic stress allowables of AISI or AISC. The coefficients for AISC-N690 are supplemented by the requirements identified in subsection 3.8.4.5.*
2. *Where any load reduces the effects of other loads, the coefficient for that load is taken as zero unless it can be demonstrated that the load is always present or occurs simultaneously with the other loads.*
3. *In no instance does the allowable stress exceed $0.7F_u$ in axial tension nor $0.7F_u$ times the ratio of the plastic to elastic section modulus for tension plus bending.*
4. *Loads due to maximum precipitation are evaluated using load combination 4 with the maximum precipitation in place of the tornado load.*

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.4-2
[*LOAD COMBINATIONS AND LOAD FACTORS FOR SEISMIC CATEGORY I CONCRETE STRUCTURES*]*

Load Combination and Factors												
Combination No.		***1***	***2***	***3***	***4***	***5***	***6***	***7***	***8***	***9***	***10***	***11***
Dead	*D*	*1.4*	*1.4*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.05*	*1.05*	*1.4*	*1.0*
Liquid	*F*	*1.4*	*1.4*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.05*	*1.05*	*1.4*	*1.0*
Live	*L*	*1.7*	*1.7*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.3*	*1.3*	*1.7*	*1.0*
Earth	*H*	*1.7*	*1.7*	*1.0*	*1.0*	*1.0*	*1.0*	*1.0*	*1.3*	*1.3*	*1.7*	*1.0*
Design Pressure	P_d										*1.5*	*1.0*
Normal reaction	R_o	*1.7*	*1.7*	*1.0*	*1.0*				*1.3*	*1.3*	*1.7*	*1.0*
Normal thermal	T_o			*1.0*	*1.0*				*1.2*	*1.2*		
Wind	*W*		*1.7*							*1.3*		
Safe shutdown earthquake	E_s			*1.0*				*1.0*				*1.0*
Tornado	W_t				*1.0*							
Accident pressure	P_a					*1.4*	*1.25*	*1.0*				
Accident thermal	T_a					*1.0*	*1.0*	*1.0*				
Accident thermal reactions	R_a					*1.0*	*1.0*	*1.0*				
Accident pipe reactions	Y_r						*1.0*	*1.0*				
Jet impingement	Y_j						*1.0*	*1.0*				
Pipe impact	Y_m						*1.0*	*1.0*				

Notes:

1. *Design for mechanical loads is in accordance with ACI-349 Strength Design Method for all load combinations. Design for combinations including thermal loads is described in subsection 3.8.3.5.3.4. Shield building design for combinations including thermal loads is described in subsection 3.8.4.5.5.4.*
2. *Where any load reduces the effects of other loads, the corresponding coefficient for that load is taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise the coefficient for the load is taken as zero.*
3. *Loads due to maximum precipitation are evaluated using load combination 4 with the maximum precipitation in place of the tornado load.*
4. *Load combinations 10 and 11 are applicable to nuclear island basemat design only. P_d, the containment design pressure, is 59 psig.*

*NRC Staff approval is required prior to implementing a change in this information.

Table 3.8.4-3
Acceptance Tests for Concrete Aggregates

Method of Test	Designation
Organic impurities in sand	ASTM C 40
Effect of organic impurities on strength of mortar	ASTM C 87
Soundness of aggregates	ASTM C 88
Material finer than No. 200 sieve	ASTM C 117
Specific gravity and absorption - coarse aggregates	ASTM C 127
Specific gravity and absorption - fine aggregates	ASTM C 128
Los Angeles abrasion of small-size coarse aggregates	ASTM C 131
Sieve analysis	ASTM C 136
Friable particles	ASTM C 142
Potential reactivity of aggregates (chemical)	ASTM C 289
Petrographic examination of aggregates	ASTM C 295
Resistance to degradation of large-size coarse aggregates by abrasion and impact in the Los Angeles machine	ASTM C 535
Potential alkali reactivity of carbonate rocks for concrete aggregates	ASTM C 586
Resistance of concrete to rapid freezing and thawing	ASTM C 666
Flat and elongated particles	ASTM D4791

Table 3.8.4-4
Criteria for Water Used in Production of Concrete

Requirements and Test Method	Criteria
Compressive strength ASTM C 109	Reduction in strength not in excess of 10 percent
Soundness ASTM C 151	Increase in length limited to 0.10 percent
Time of setting ASTM C 191	± 10 min for initial set, ± 1 hour for final set

Table 3.8.4-5
Not Used

Table 3.8.4-6 (Sheet 1 of 2)
Materials Used in Structural and Miscellaneous Steel

Standard	Construction Material
ASTM A1	Carbon steel rails
ASTM A36/A36M	Rolled shapes, plates, and bars
ASTM A53	Welded and Seamless Steel Pipe, Grade B
ASTM A106	Seamless Carbon Steel Pipe For High Temperature Service
ASTM A108	Weld studs
ASTM A123	Zinc coatings (hot galvanized)
ASTM A167	Stainless and Heat-Resisting Chromium Nickel Steel Plate, Sheet and Strip
ASTM A193	Alloy Steel and Stainless Steel Bolting Materials for High-Temperature Service
ASTM A194	Carbon and Alloy Steel Nuts for Bolts for High-Pressure or High-Temperature Service, or Both
ASTM A240	Duplex 2101 stainless steel (designation S32101)
ASTM A242	High-strength low alloy structural steel
ASTM A276	Stainless and Heat-Resisting Steel Bars and Shapes
ASTM A307	Low carbon steel bolts
ASTM A312	Seamless and Welded Austenitic Stainless Steel Pipe
ASTM A325	High strength bolts
ASTM A354	Quenched and tempered alloy steel bolts (Grade BC)
ASTM A441	High-strength low alloy structural manganese vanadium steel
ASTM A490	Structural Bolts, Alloy Steel, Heat Treated, 150 ksi Minimum Tensile Strength
ASTM A496	High strength deformed wire
ASTM A500	Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes
ASTM A501	Hot-Formed Welded and Seamless Carbon Steel Structural Tubing
ASTM A505	Standard Specification for Steel, Sheet and Strip, Alloy, Hot-Rolled and Cold-Rolled
ASTM A514	High-Yield Strength Quenched and Tempered Alloy Steel Plate, Suitable for Welding
ASTM A517	Standard Specification for Pressure Vessel Plates, Alloy Steel, High-Strength, Quenched and Tempered
ASTM A540	Alloy Steel Bolting for Special Applications

Table 3.8.4-6 (Sheet 2 of 2)
Materials Used in Structural and Miscellaneous Steel

Standard	Construction Material
ASTM A564	Hot-Rolled and Cold-Finished, Age-Hardening Stainless and Heat-Resisting Steel Bars and Shapes
ASTM A570	Hot-Rolled Carbon Steel Sheets and Strip, Structural Quality, Grades C, D, and E
ASTM A572	High-strength low alloy structural steel
ASTM A588	High-strength low alloy structural steel
ASTM A607	Steel Sheet and Strip, Hot-Rolled and Cold-Rolled, High-Strength, Low Alloy, Columbium, and/or Vanadium
ASTM A615	Deformed and Plain Billet Steel Bars for Concrete Reinforcement
ASTM A618	Hot-Formed Welded and Seamless High-Strength Low Alloy Structural Tubing
ASTM A706	Low Alloy Steel Deformed Bars for Concrete Reinforcement
ASTM A970	Headed Steel Bars for Concrete Reinforcement
ASTM A992	Rolled shapes, plates, and bars
ASTM-F1554	Steel anchor bolts, 36, 55, and 105-ksi yield strength

Table 3.8.5-1
Minimum Required Factor of Safety
for Overturning and Sliding of Structures

Load Combination	Overturning	Sliding	Flotation
D + H + B + W	1.5	1.5	-
D + H + B + E_s	1.1	1.1	-
D + H + B + W_t	1.1	1.1	-
D + F	-	-	1.1
D + B	-	-	1.5

where:

D	=	dead load excluding the fluid loads
H	=	lateral earth pressure
W	=	wind load
E_s	=	safe shutdown earthquake load
W_t	=	tornado load
F	=	buoyant force due to the design basis flood
B	=	buoyant force on submerged structure due to high ground water table

Table 3.8.5-2
Factors of Safety for Flotation, Overturning and Sliding of Nuclear Island Structures

Environmental Effect	Factor of Safety[1]
Flotation	
High Ground Water Table	3.7
Design Basis Flood	3.5
Sliding	
Design Wind, North-South	14.0
Design Wind, East-West	10.1
Design Basis Tornado, North-South	7.7
Design Basis Tornado, East-West	5.9
Safe Shutdown Earthquake, North-South	1.1[2]
Safe Shutdown Earthquake, East-West	1.1[2]
Overturning	
Design Wind, North-South	51.5
Design Wind, East-West	27.9
Design Basis Tornado, North-South	17.7
Design Basis Tornado, East-West	9.6
Safe Shutdown Earthquake, North-South	1.77
Safe Shutdown Earthquake, East-West	1.17

Notes:
1. Factor of safety is calculated for the envelope of the soil and rock sites described in Subsection 3.7.1.4.
2. From non-linear sliding analysis using friction elements, the horizontal movement is negligible (0.12 inch without buoyant force consideration, and 0.19 inch with buoyant force considered).

Table 3.8.5-3
Definition of Critical Locations, Thicknesses and Reinforcement for Nuclear Island Basemat[1] (in²/ft)

Basemat Segment (see detail in subsection 3.8.5.4.4)	Location	Required[3][6]			*[Provided (Minimum)[4]]**		
		North-South	East-West	Shear	*North-South*	*East-West*	*Shear*
Auxiliary Building Basemat Elevation 66′ 6″							
Column line K to L and from Shield Building to Col. Line 11[2]							
	Top Face	Note 5	1.5		[*2.25*	*2.25*	
	Bottom Face	Note 5	1.6		*2.25*	*2.25*	
				0.23			*0.25*]*
Column line 1 to 2 and from Column Line K-2 to N wall[2]							
General area	Top Face	Note 5	Note 5		[*2.25*	*2.25*	
Central zone	Top Face	2.72	Note 5		*3.25*	*2.25*	
	Bottom Face	2.25	Note 5		*2.25*	*2.25*	
				0.47			*0.50*]*

Notes:

1. The applicable column lines and elevation levels are identified and included in Figures 1.2-9 and 3.7.2-12 (sheets 1 through 12).

[2. *The thickness of these sections is 6′0″ with a construction tolerance of +4 inches, -3/4 inch.*]*

3. These concrete reinforcement values represent the minimum reinforcement required for structural requirements except for designed openings, penetrations, sumps or elevator pits.

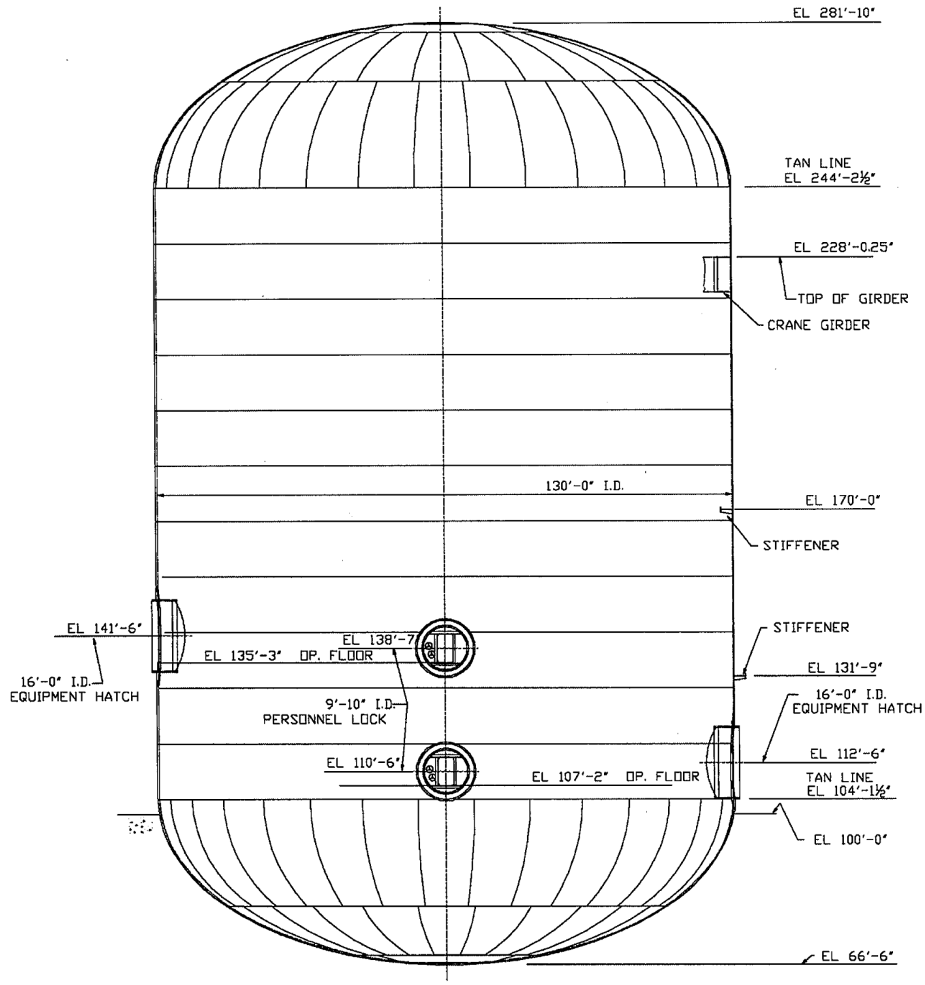
[4. *These concrete reinforcement values represent the provided reinforcement for structural requirements except for designed openings, penetrations, sumps or elevator pits.*]*

5. Reinforcement demand is low. Basemat reinforcement is fairly uniform across all of the basemat and the governing location is in other segments (see Figure 3.8.5-3, sheets 3 to 6).
6. 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 3.8-201
Waterproof Membrane Inspections, Tests, Analyses, and Acceptance Criteria

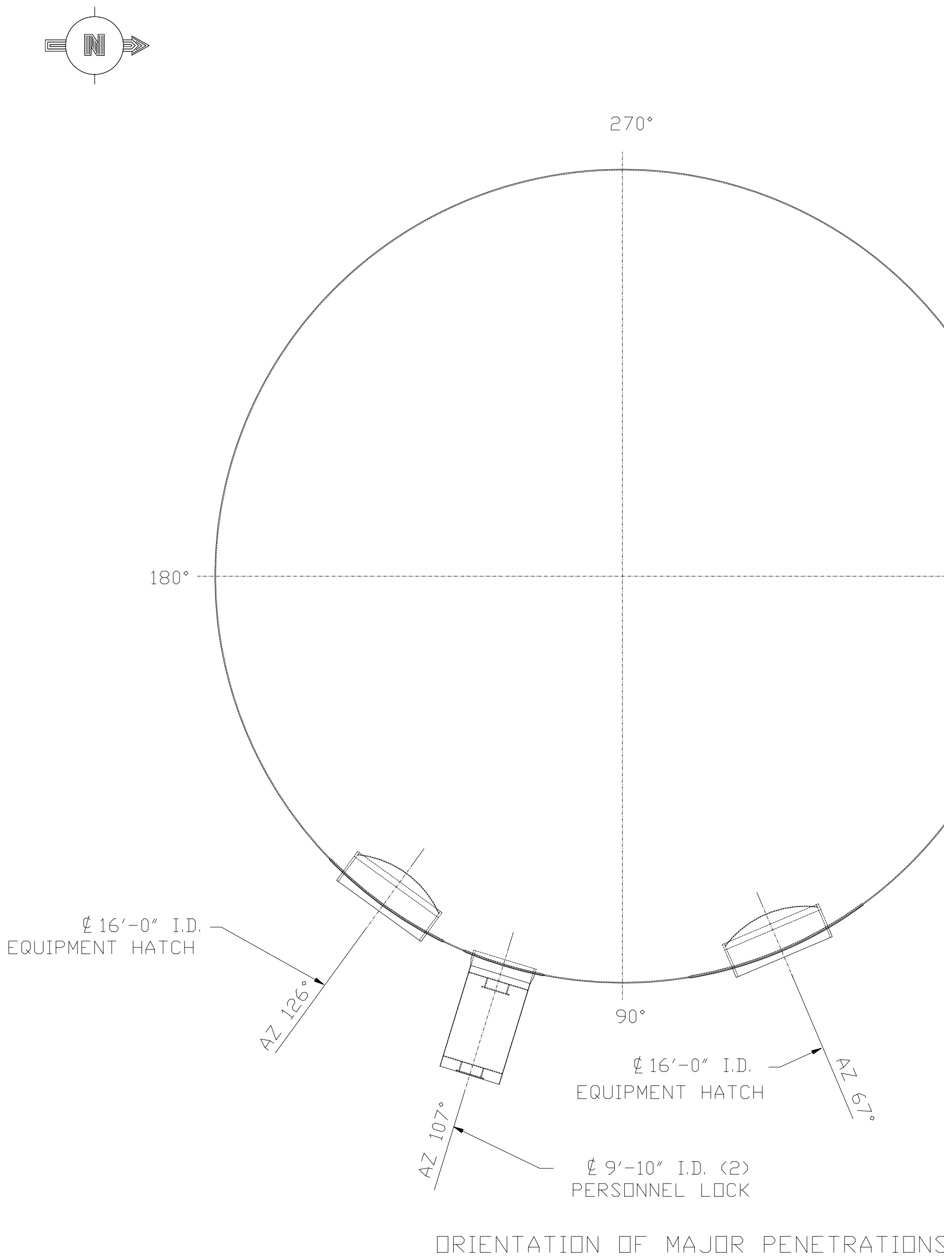
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1) The friction coefficient to resist sliding is 0.7 or higher.	Testing will be performed to confirm that the mudmat-waterproofing-mudmat interface beneath the Nuclear Island basemat has a minimum coefficient of friction to resist sliding of 0.7.	A report exists and documents that the as-built waterproof system (mudmat-waterproofing-mudmat interface) has a minimum coefficient of friction of 0.7 as demonstrated through material qualification testing.



Equipment hatches rotated out of position to show configuration: see Sheet 2 for azimuth orientation.

Note: Course layout, plate/insert plate geometry, and weld seams are shown for illustrative purposes only.

Figure 3.8.2-1 (Sheet 1 of 3)
Containment Vessel General Outline



Equipment hatch at azimuth of 126° is the lower hatch.

Equipment hatch at azimuth of 67° is the upper hatch.

Figure 3.8.2-1 (Sheet 2 of 3)
Containment Vessel General Outline

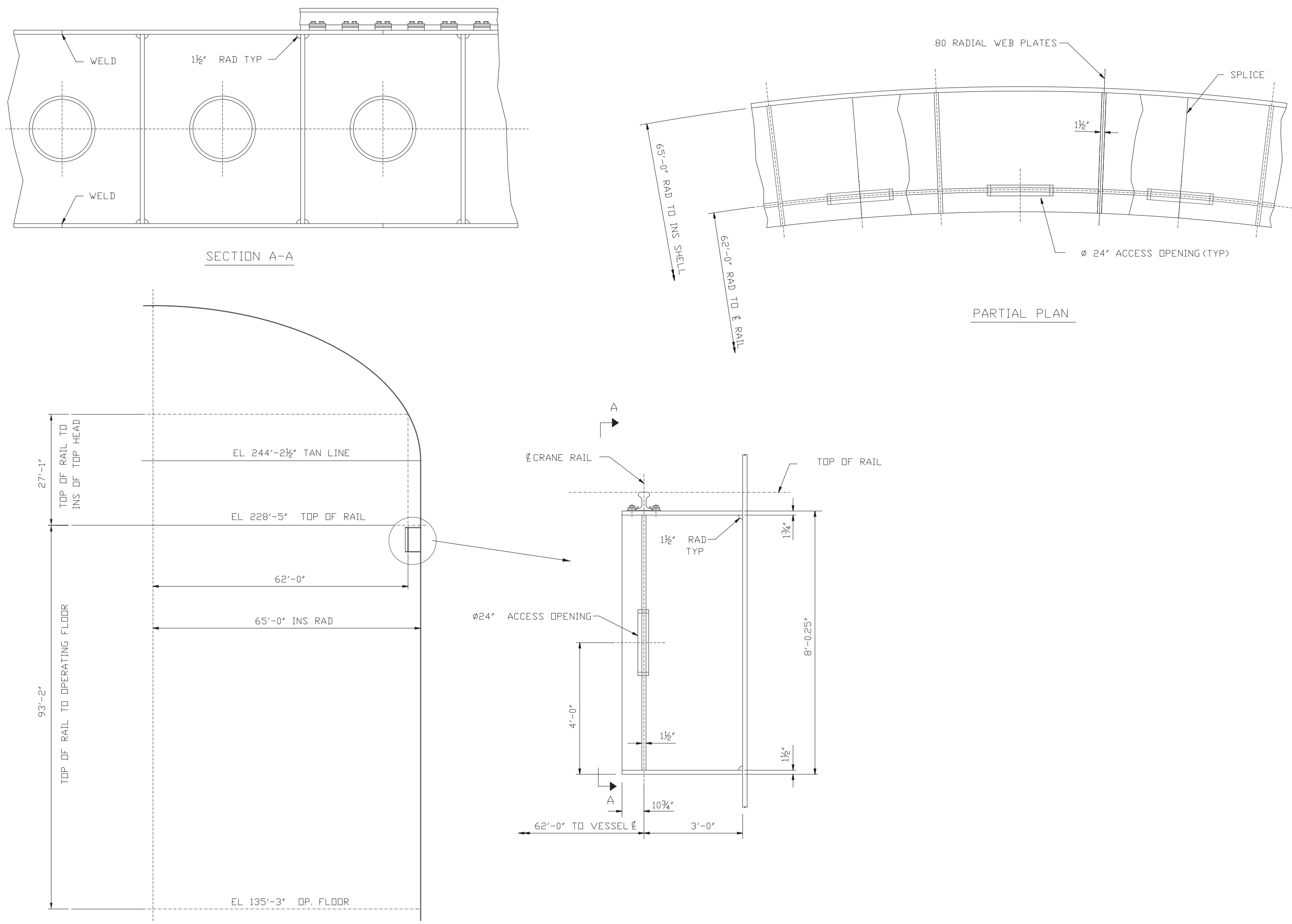
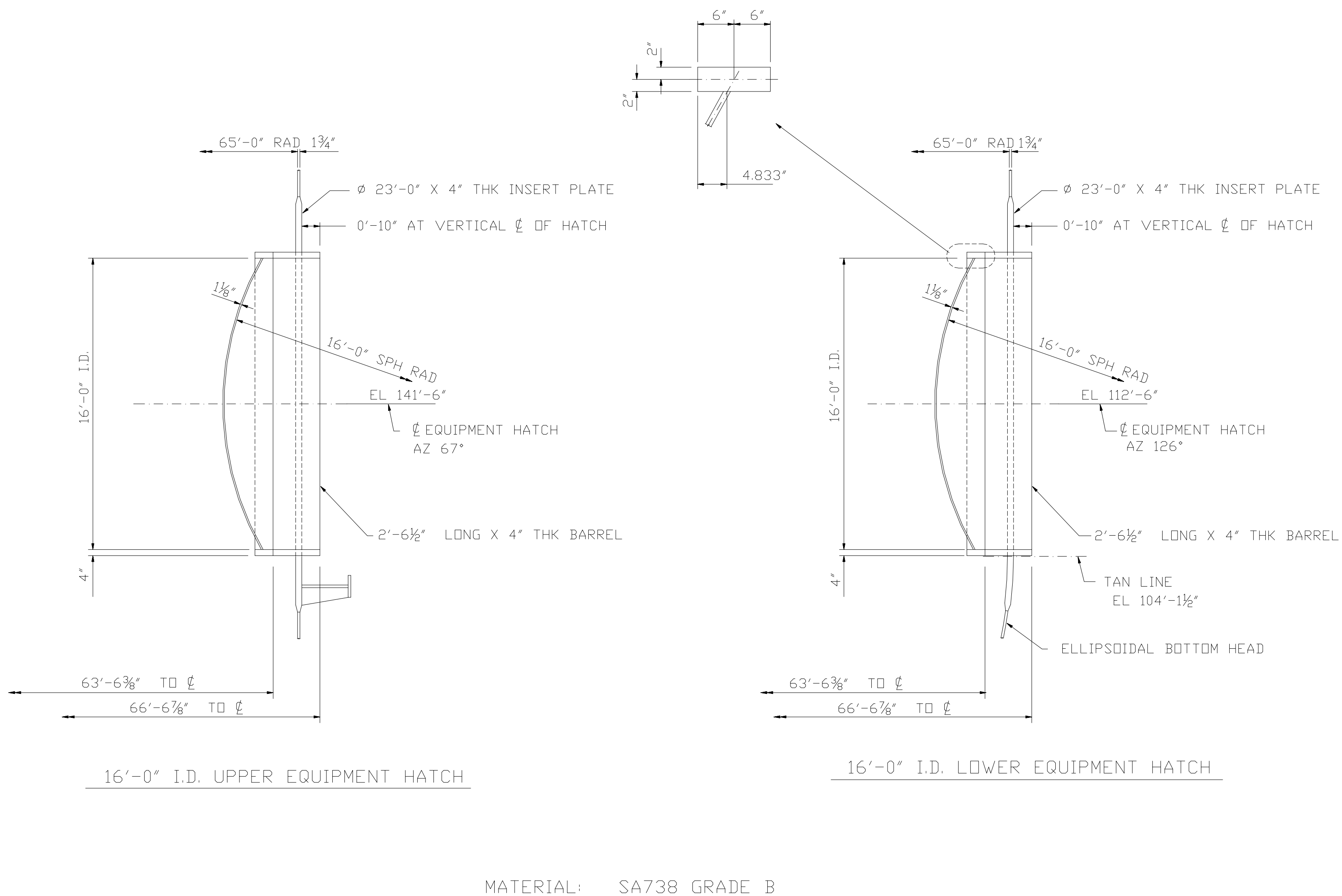


Figure 3.8.2-1 (Sheet 3 of 3)
Containment Vessel General Outline



See subsection 3.8.2.1.3 for information that is designated as Tier 2*.

Figure 3.8.2-2
Equipment Hatches

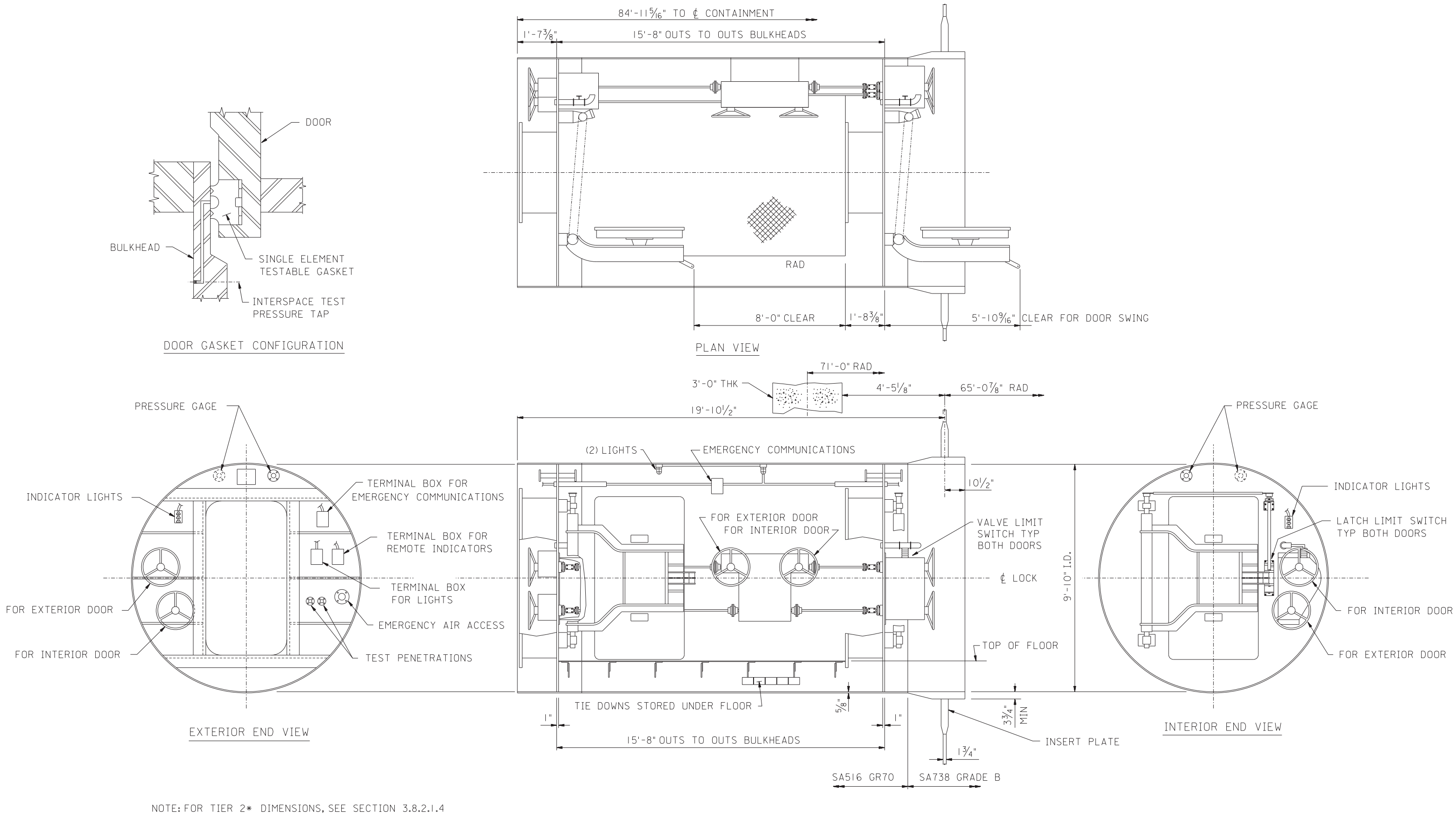


Figure 3.8.2-3
Personnel Airlock

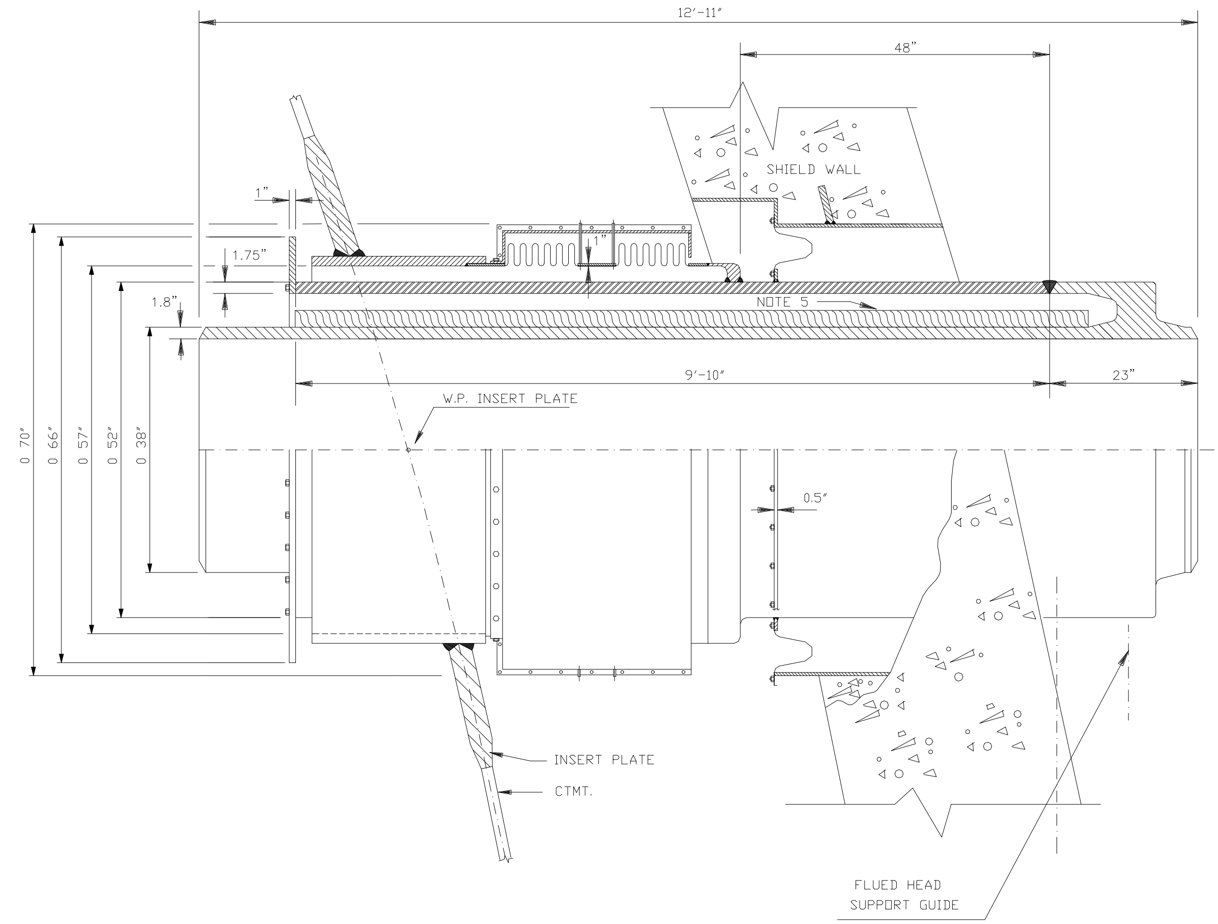


Figure 3.8.2-4 (Sheet 1 of 7)
Containment Penetrations Main Steam

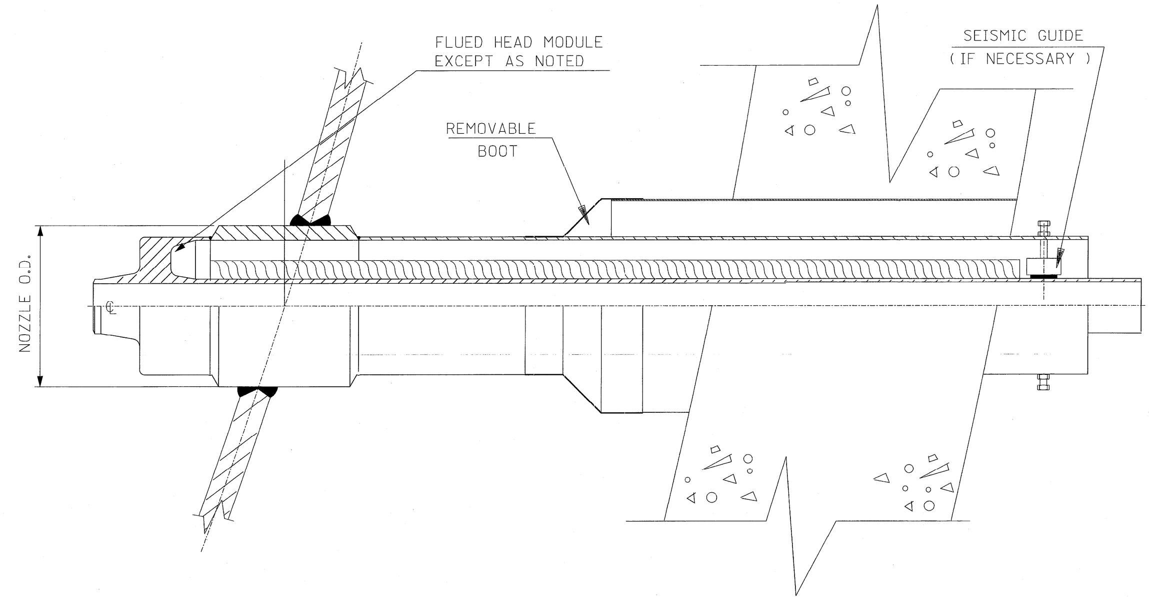


Figure 3.8.2-4 (Sheet 2 of 7)
Containment Penetrations Startup Feedwater

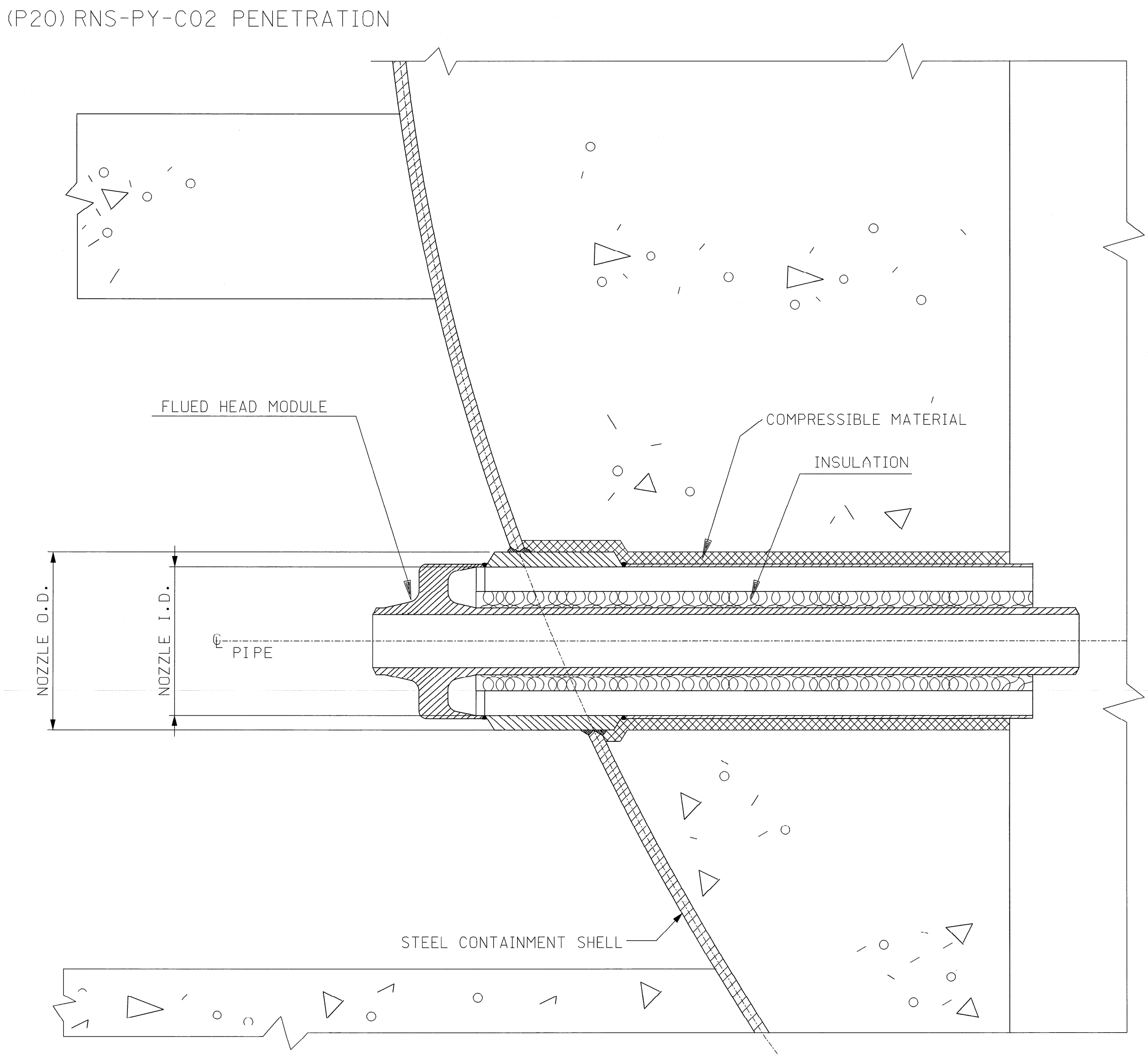


Figure 3.8.2-4 (Sheet 3 of 7)
Containment Penetrations Normal RHR Piping

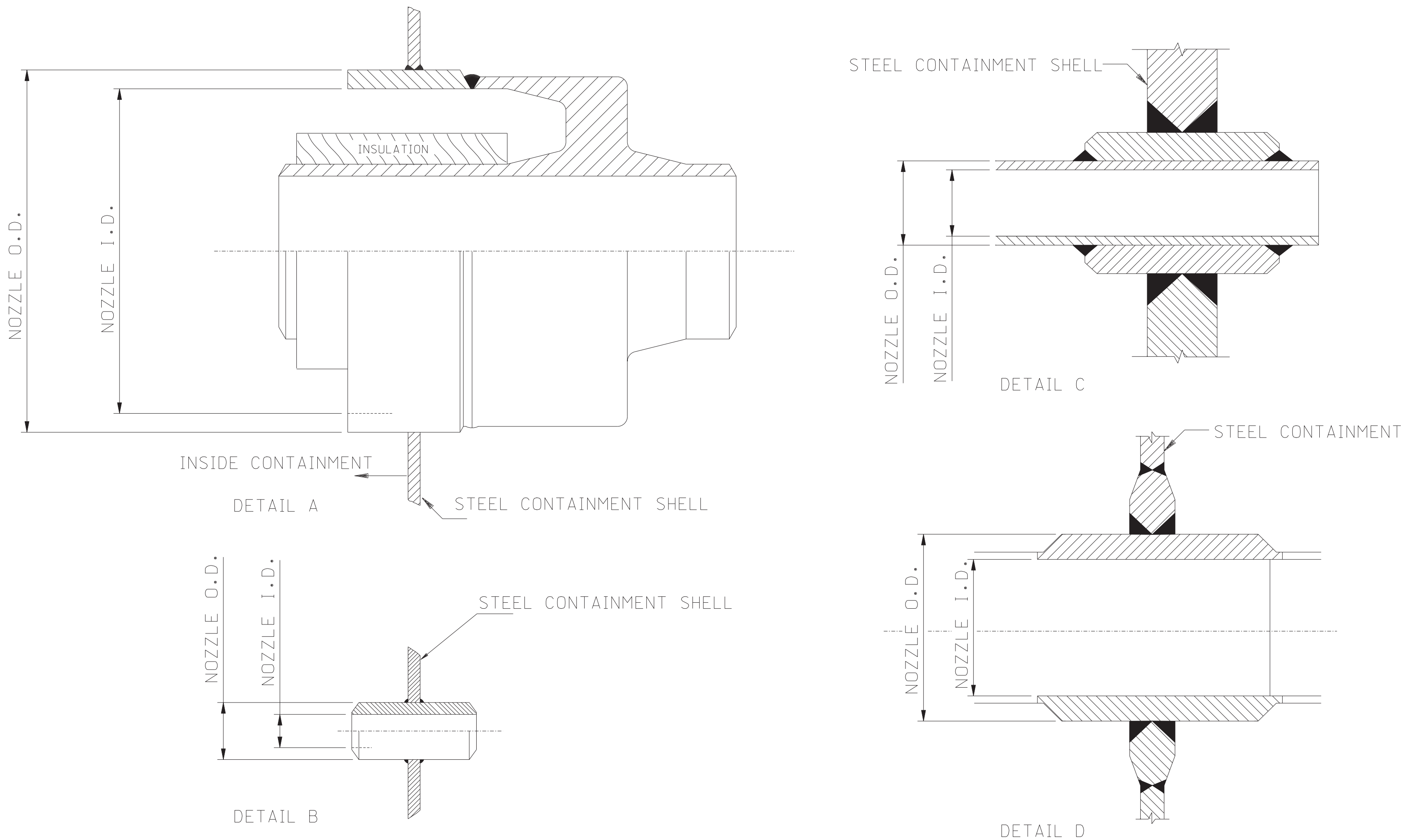


Figure 3.8.2-4 (Sheet 4 of 7)
Containment Penetrations

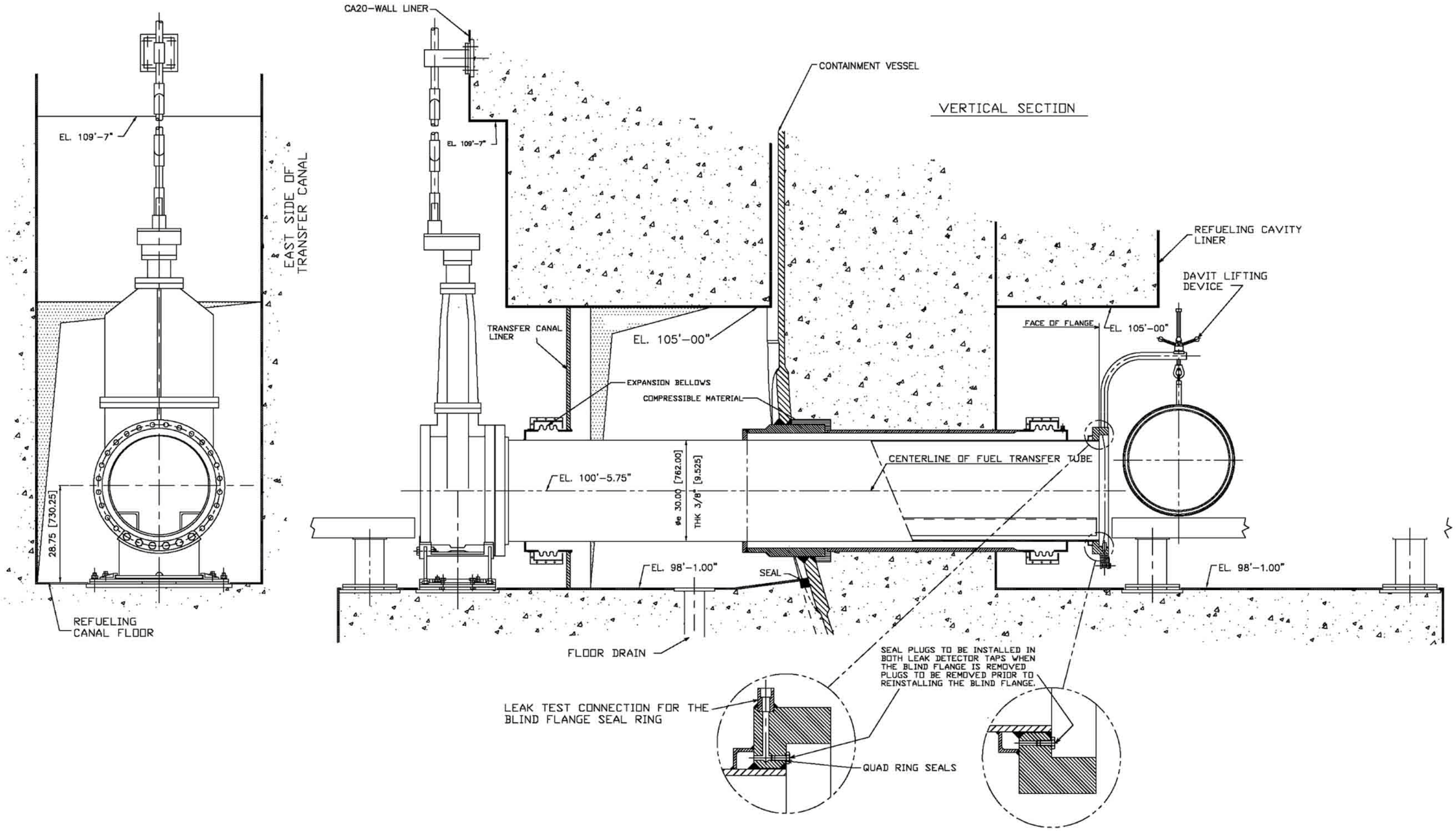
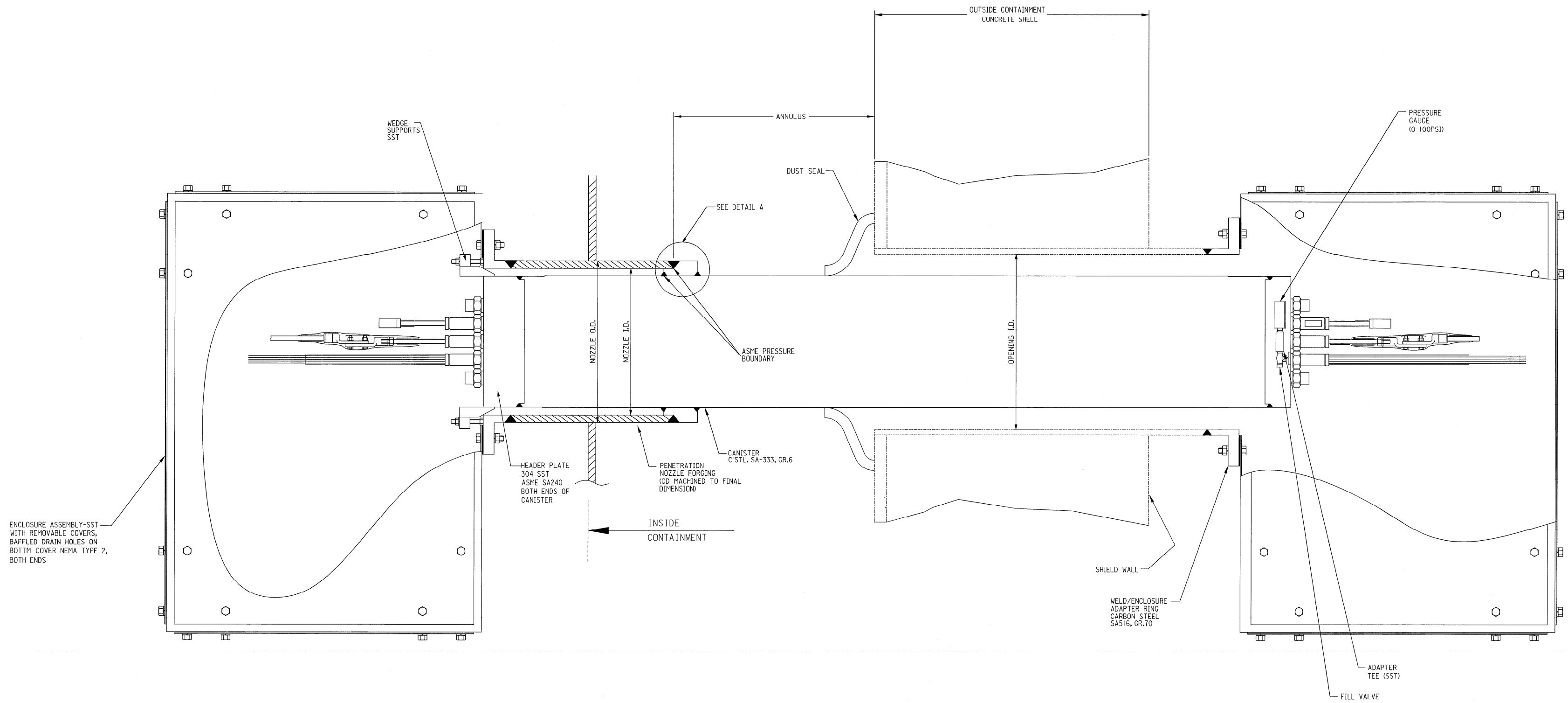


Figure 3.8.2-4 (Sheet 5 of 7)
Containment Penetrations
Fuel Transfer Penetration



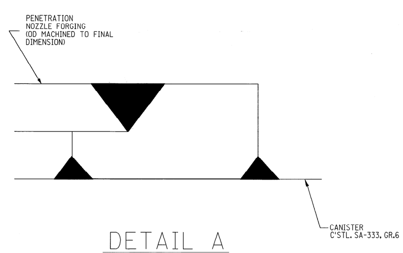
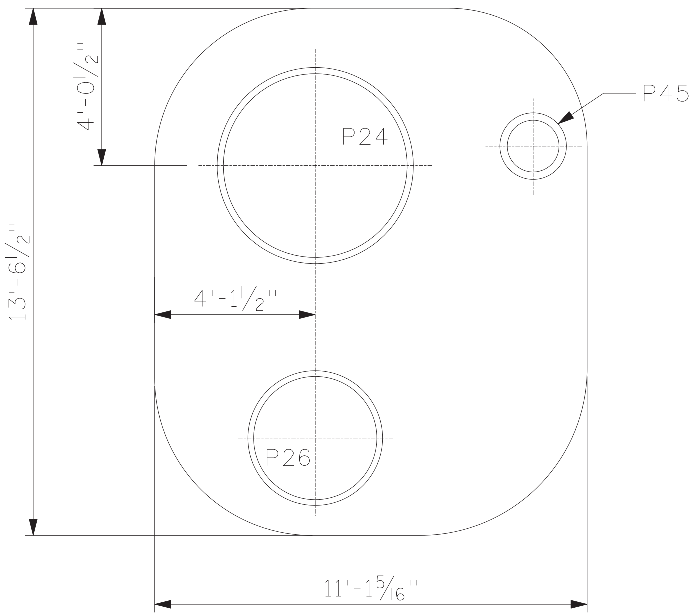


Figure 3.8.2-4 (Sheet 6 of 7)
Containment Penetrations
Typical Electrical Penetration



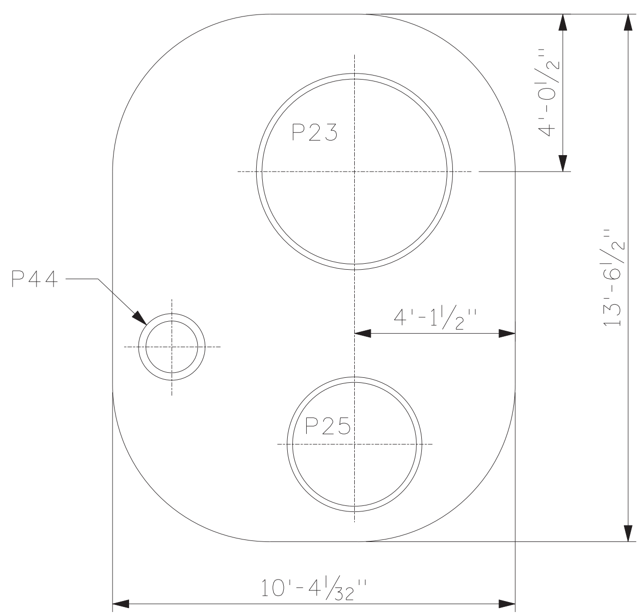


Figure 3.8.2-4 (Sheet 7 of 7)
Containment Penetrations
Steam Line and Feedwater Line Insert Plates

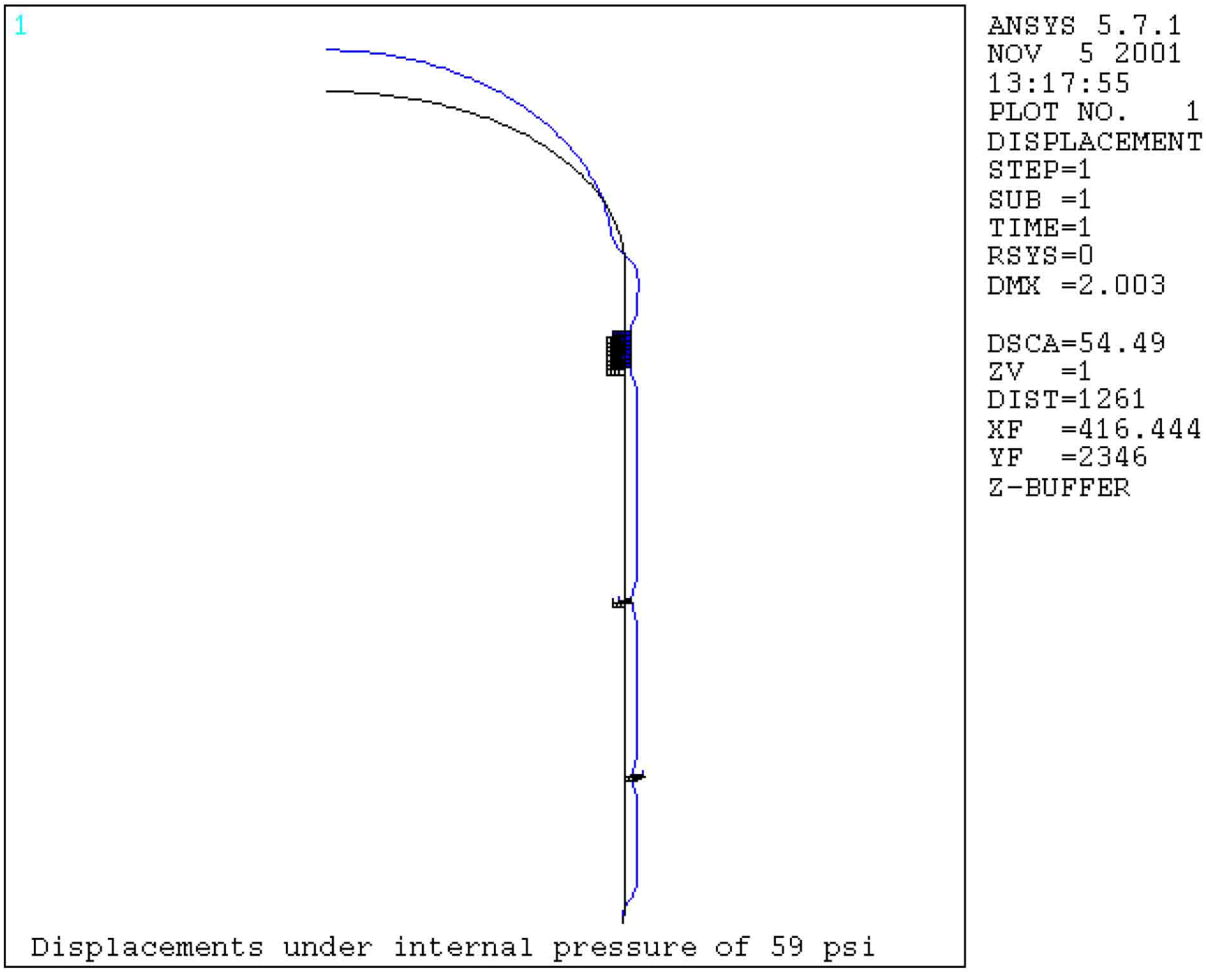
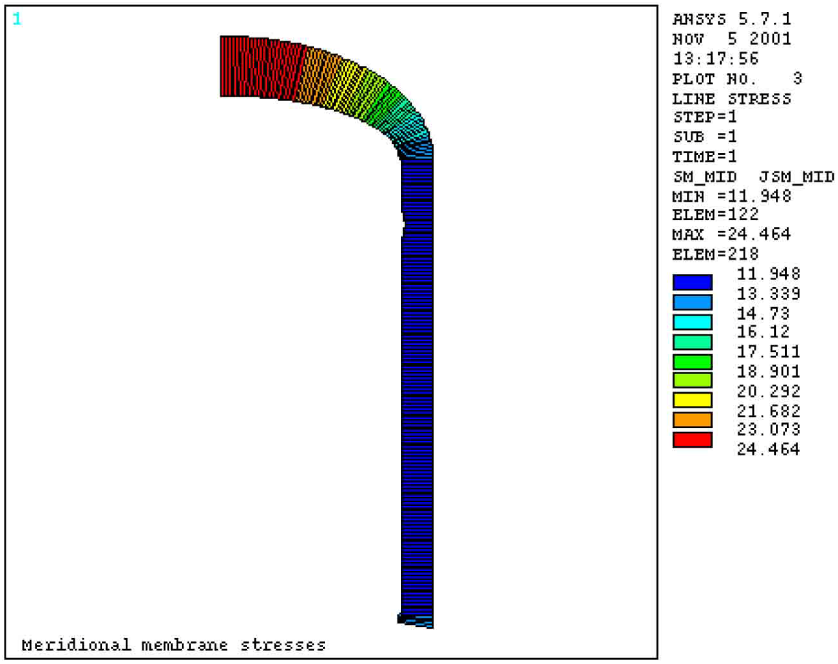
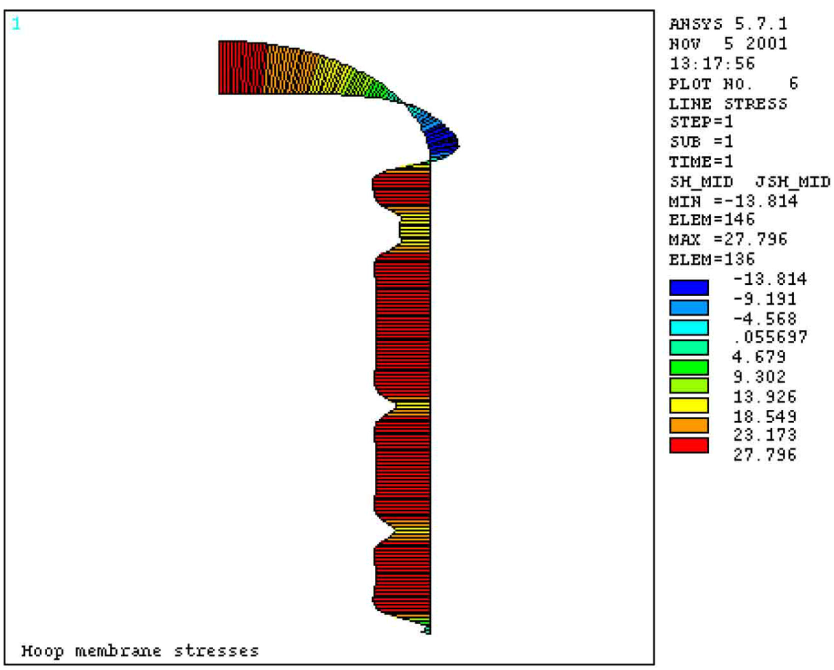


Figure 3.8.2-5 (Sheet 1 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Displaced Shape Plot



Meridional Membrane Stress (ksi)



Circumferential Membrane Stress (ksi)

Figure 3.8.2-5 (Sheet 2 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Membrane Stresses (ksi)

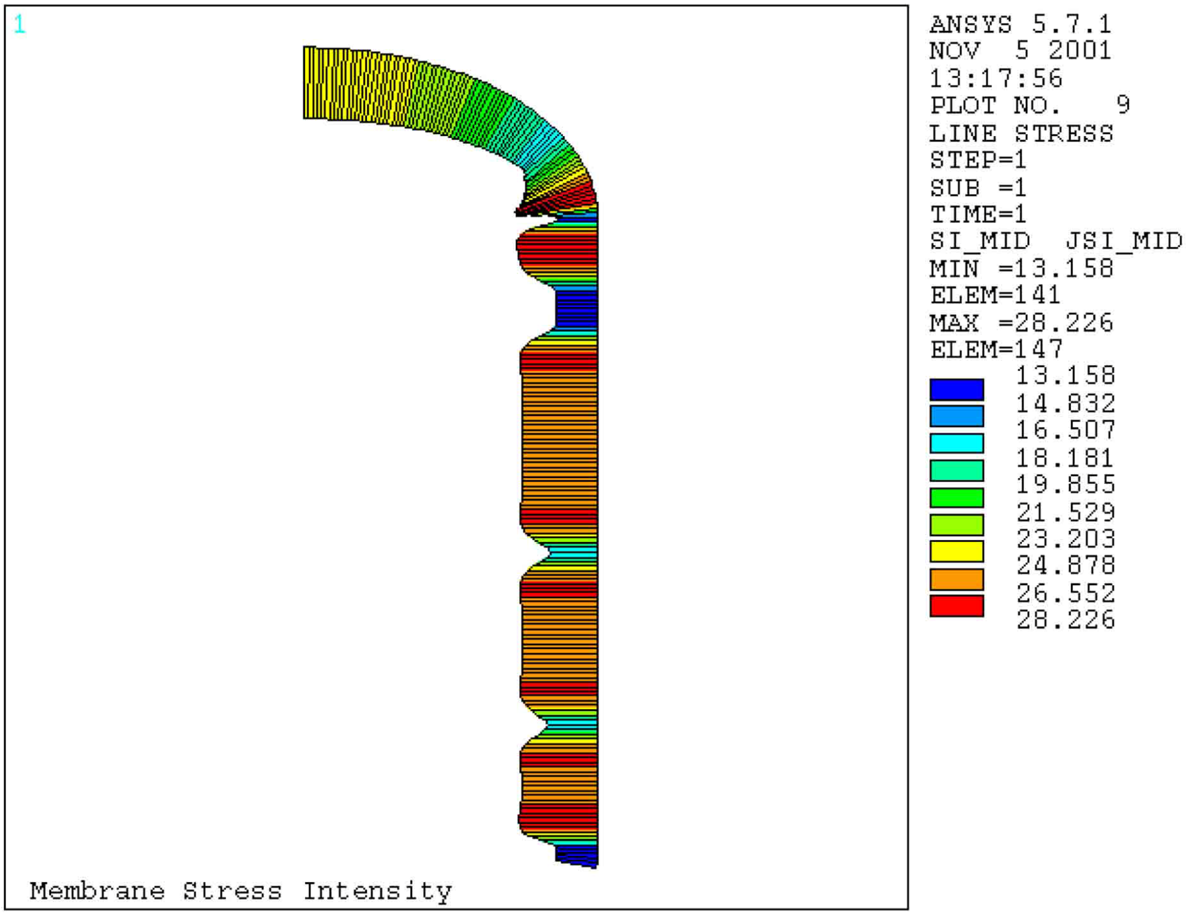
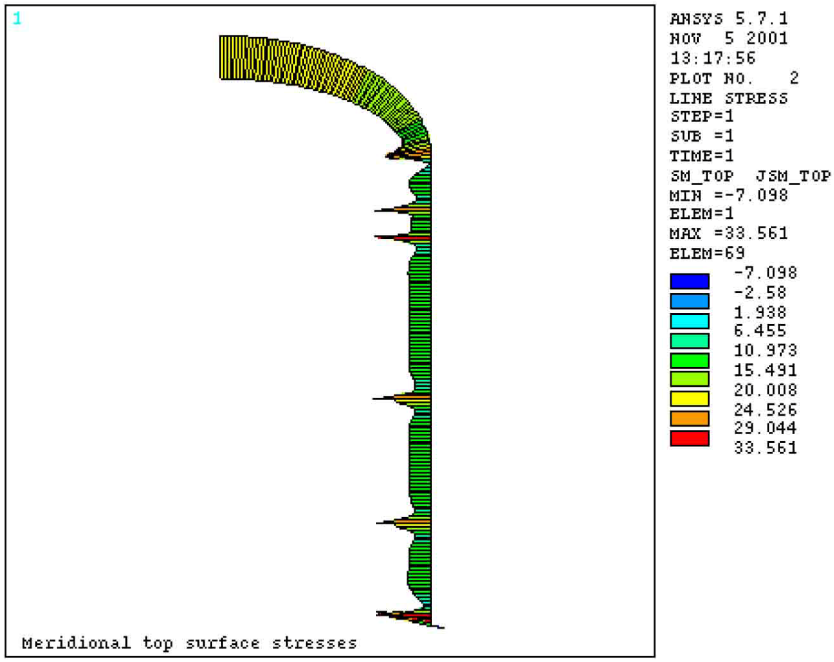
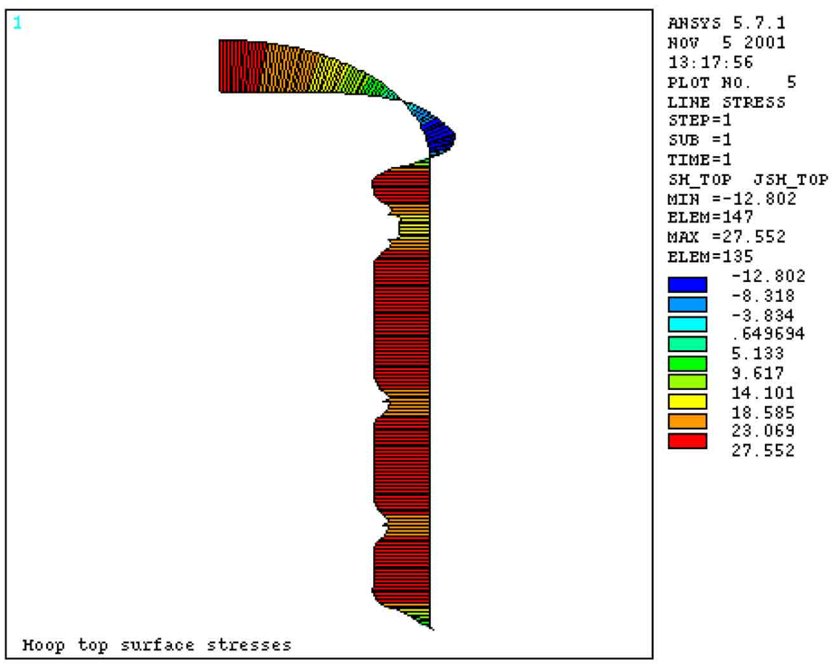


Figure 3.8.2-5 (Sheet 3 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Surface Meridional Stress (ksi)



Meridional Top Surface Stress (ksi)



Circumferential Top Surface Stress (ksi)

Figure 3.8.2-5 (Sheet 4 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Outside Surface Stresses (ksi)

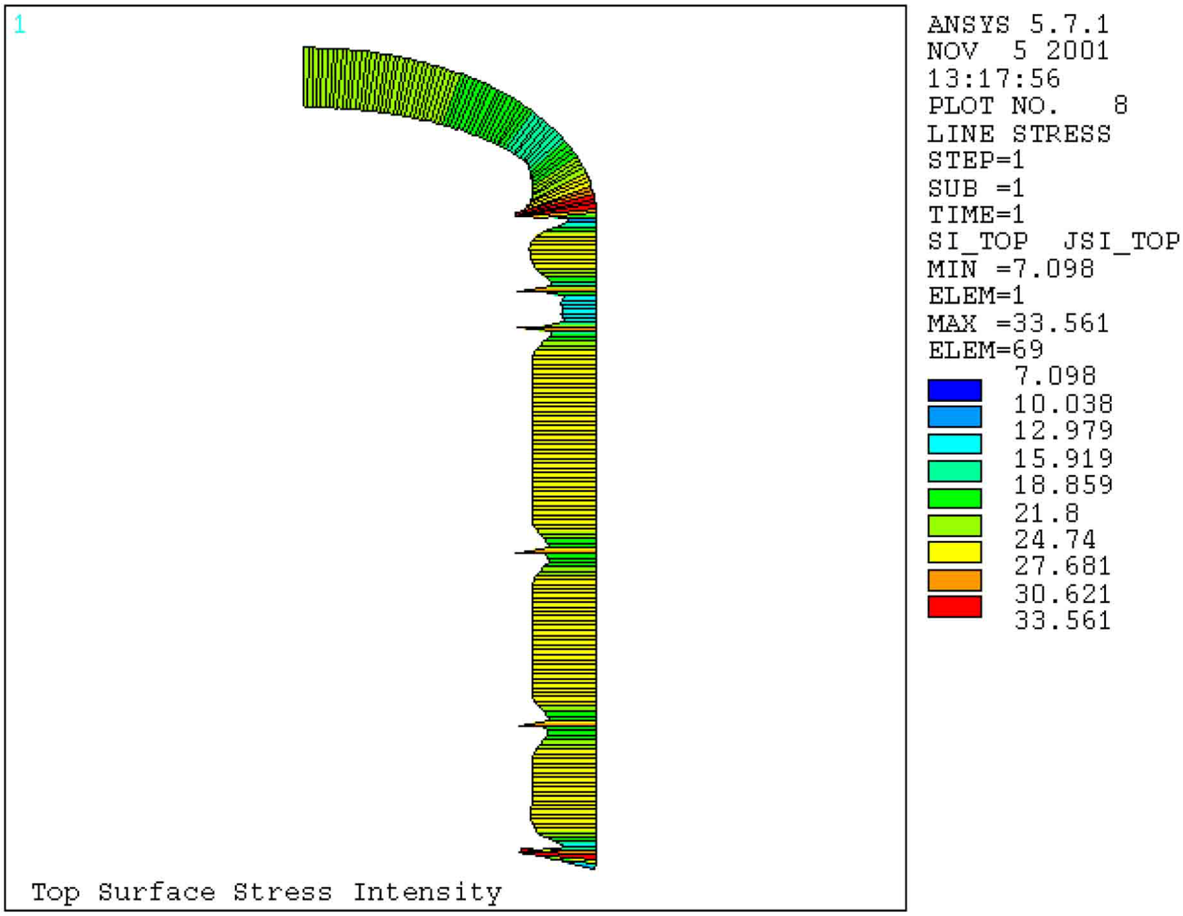


Figure 3.8.2-5 (Sheet 5 of 5)
Containment Vessel Response to Internal Pressure of 59 psig
Outer Stress Intensity (ksi)

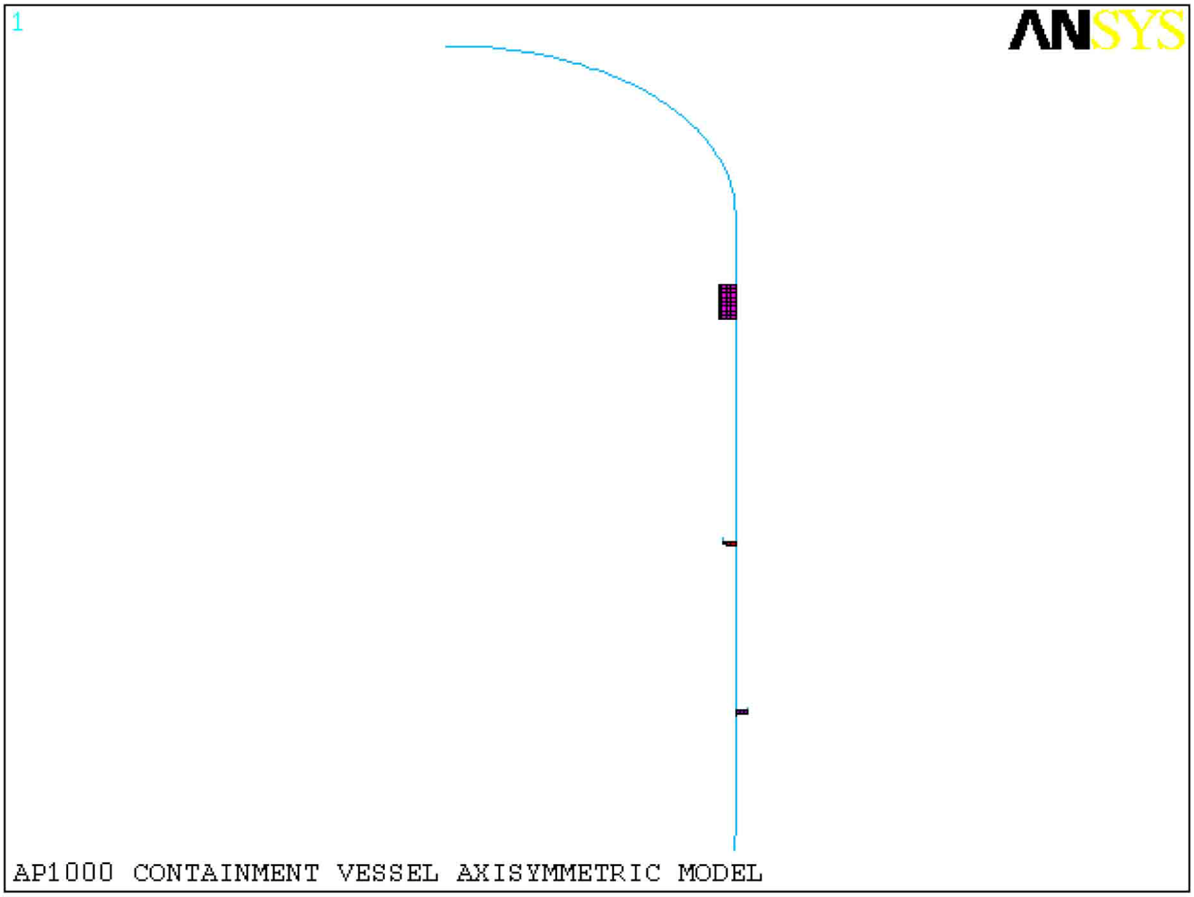
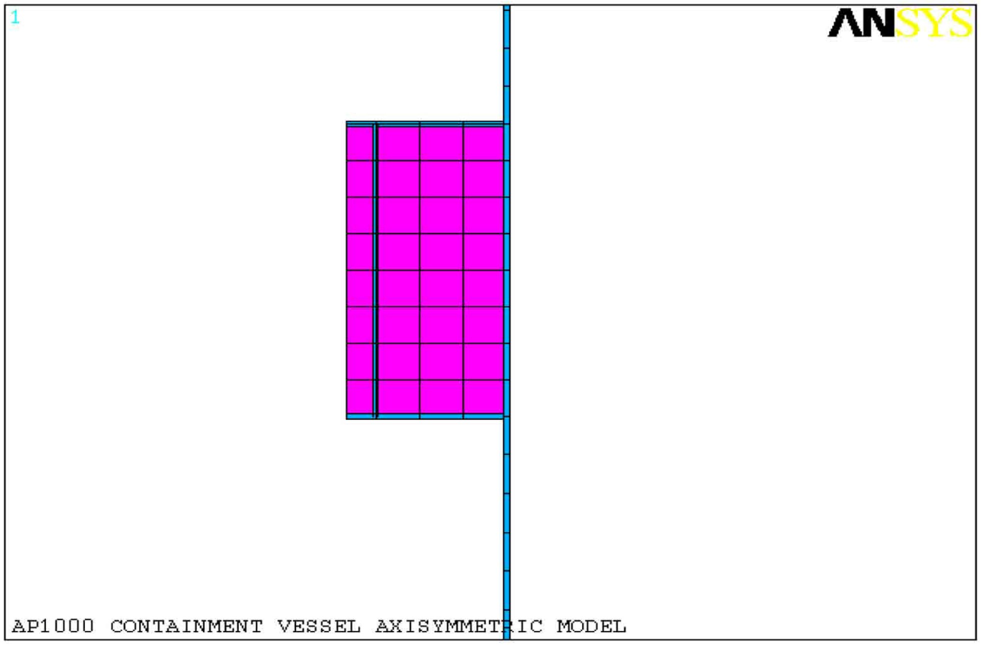
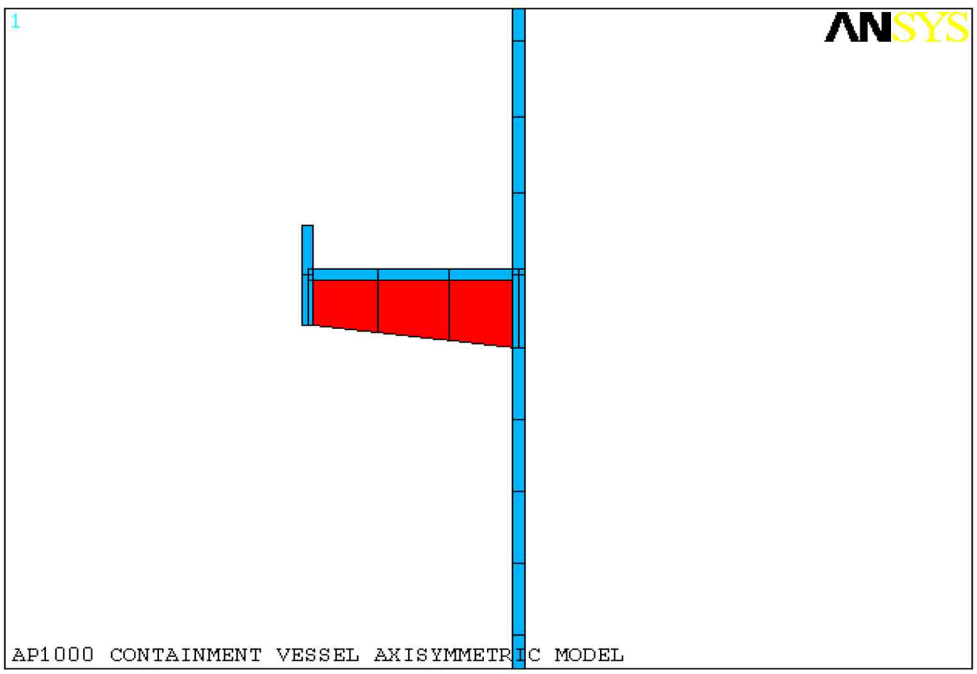


Figure 3.8.2-6 (Sheet 1 of 2)
Containment Vessel Axisymmetric Model

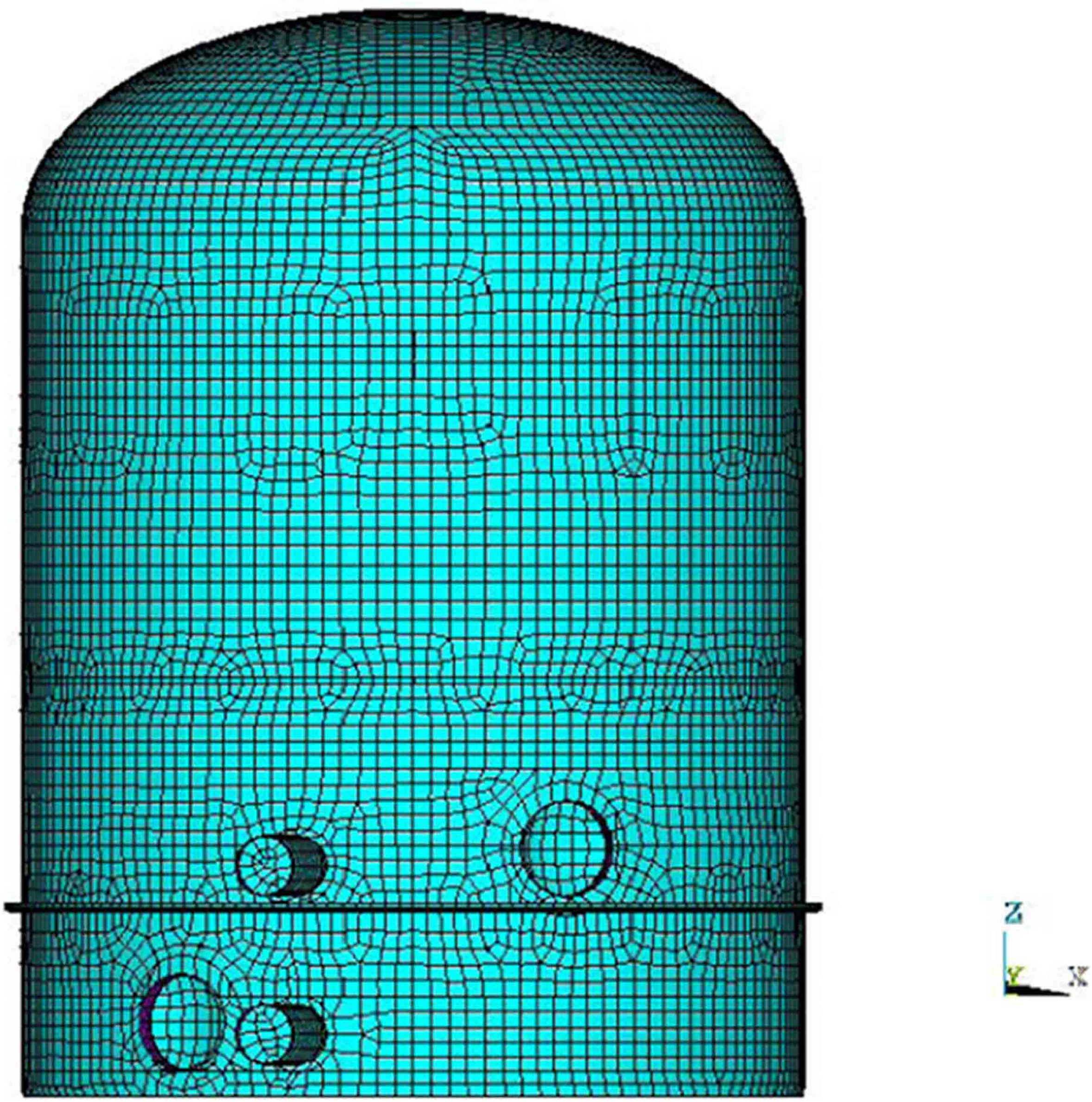


Crane Girder



Internal Stiffener at Elev. 170′-0″

Figure 3.8.2-6 (Sheet 2 of 2)
Containment Vessel Axisymmetric Model



AP1000 3D Generic CV Shell Model

Figure 3.8.2-7
Finite Element Model for Local Buckling Analyses

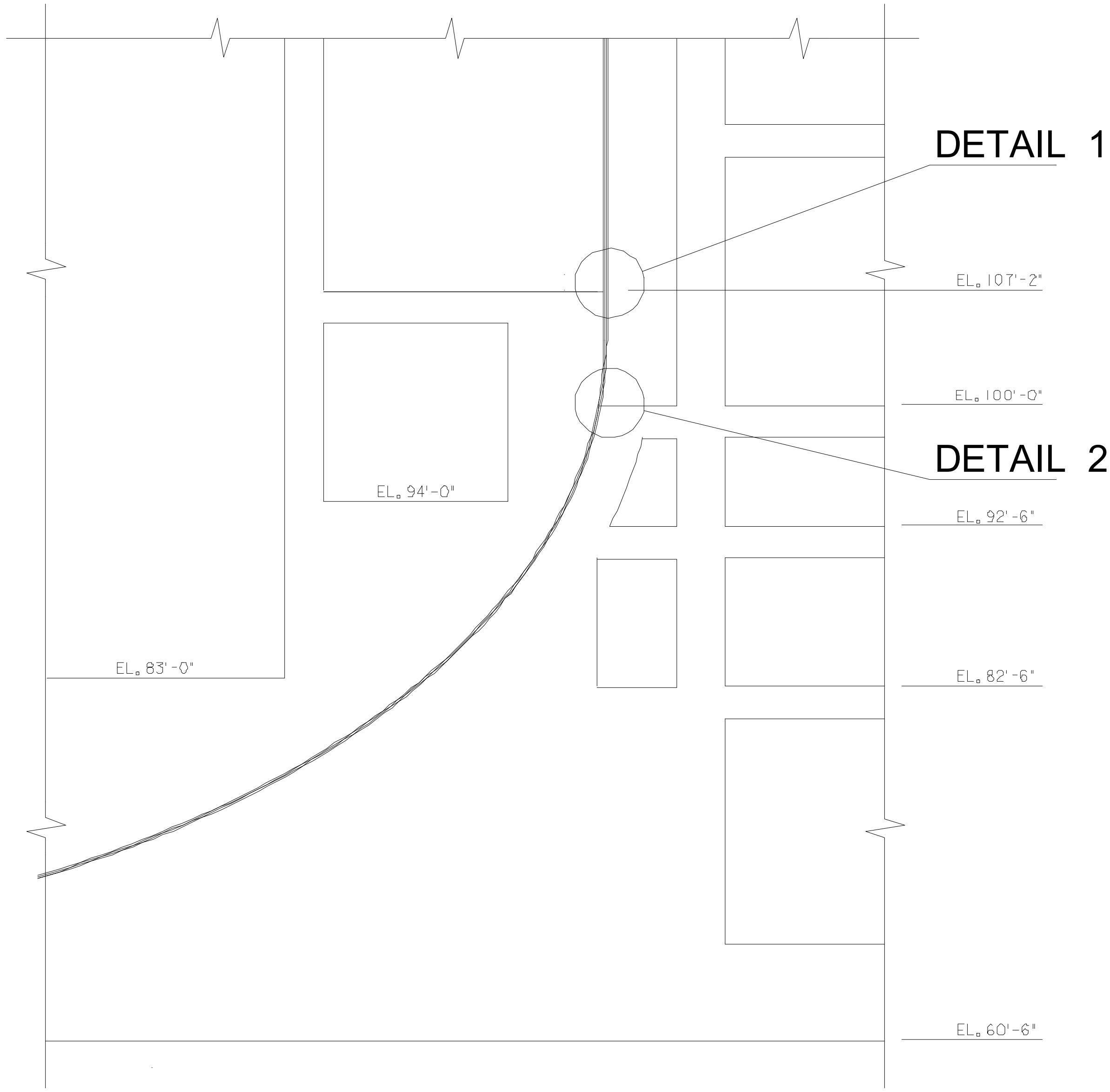


Figure 3.8.2-8 (Sheet 1 of 2)
Location of Containment Seal

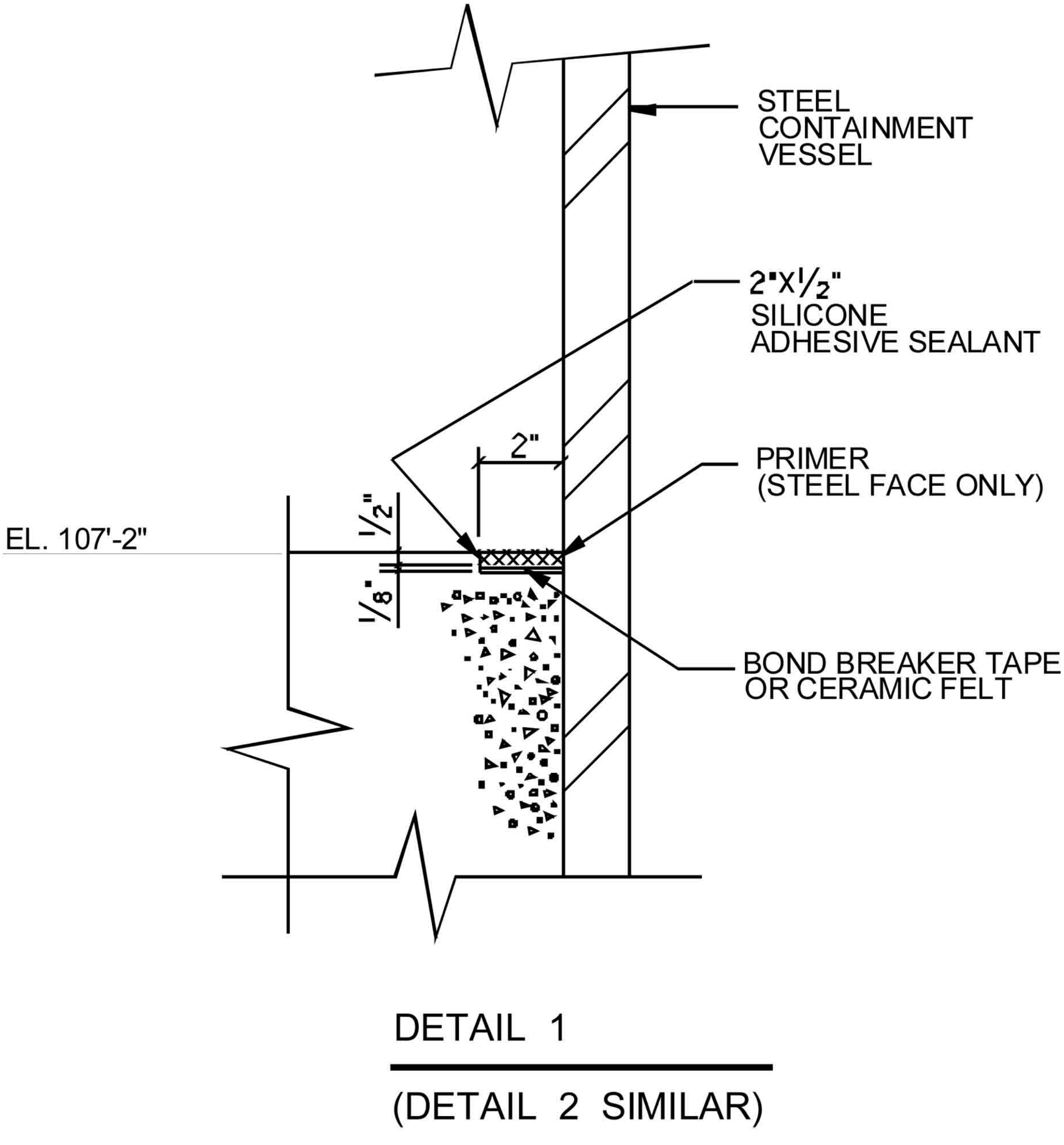


Figure 3.8.2-8 (Sheet 2 of 2)
Seal Sections and Details

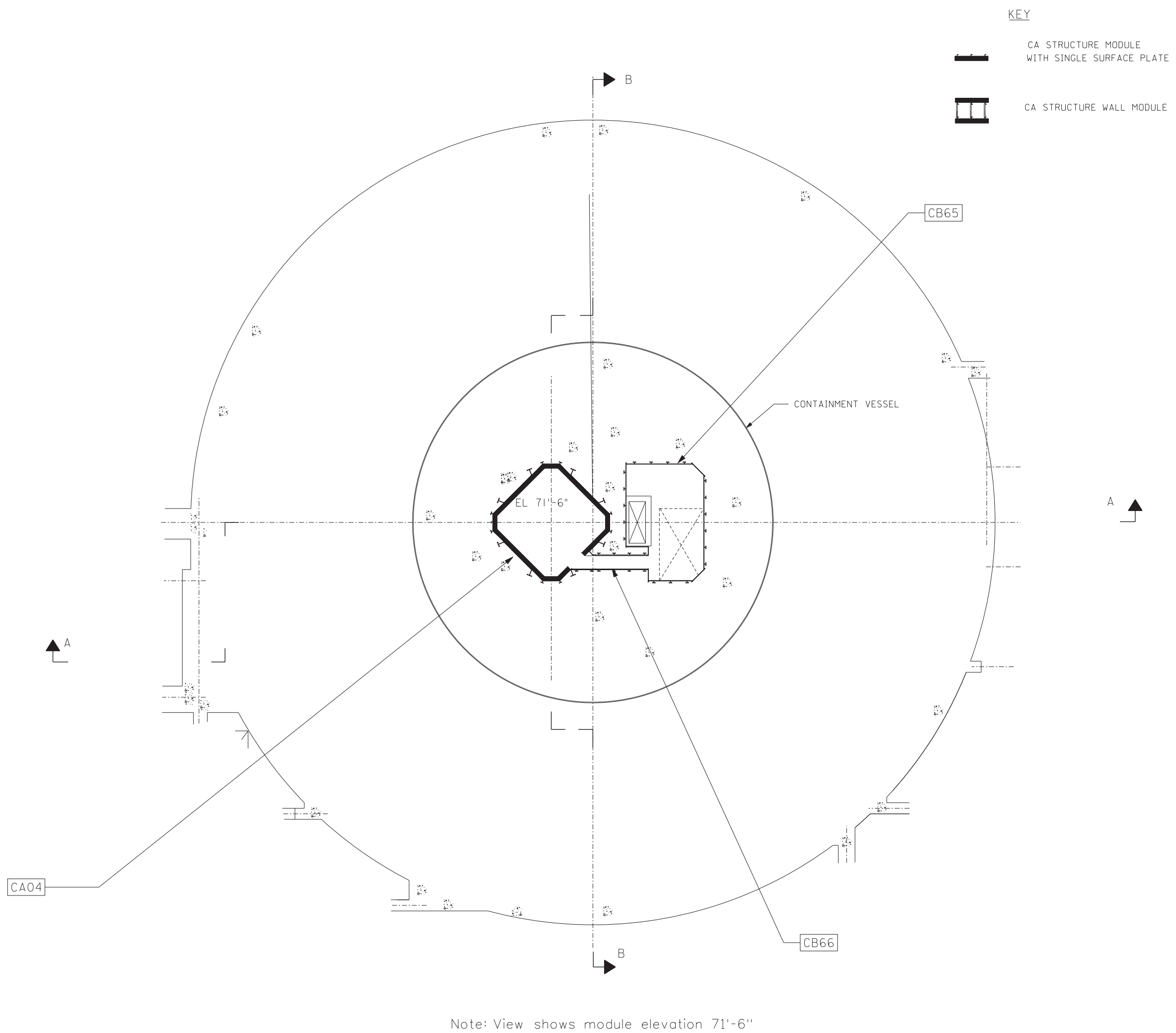


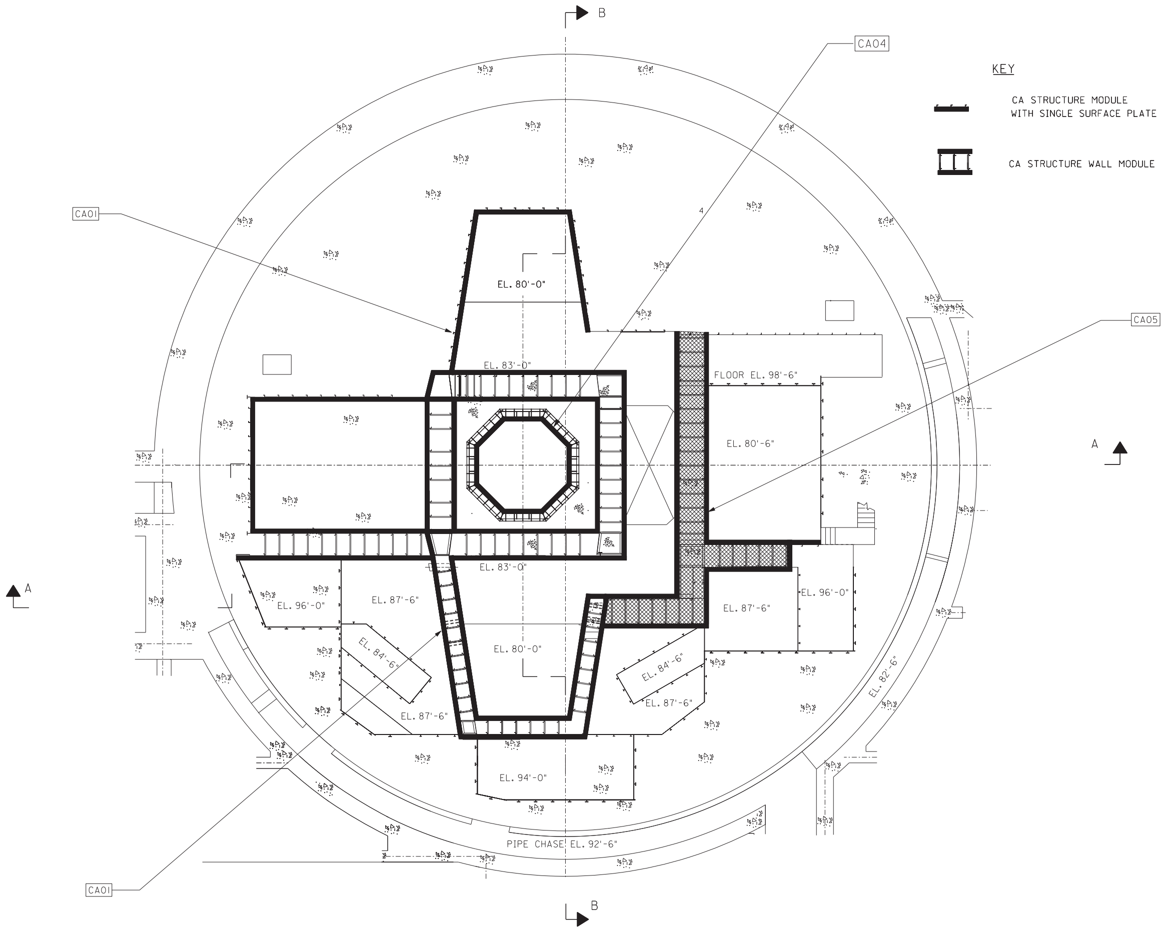
Figure 3.8.3-1 (Sheet 1 of 7)
[*Structural Modules in Containment Internal Structures*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.8.3-1 (Sheet 2 of 7)
[*Structural Modules in Containment Internal Structures*]*

*NRC Staff approval is required prior to implementing a change in this information.



Note: View shows module elevation 98'-6"

Figure 3.8.3-1 (Sheet 3 of 7)
[*Structural Modules in Containment Internal Structures*]*

*NRC Staff approval is required prior to implementing a change in this information.

Security-Related Information, Withheld Under 10 CFR 2.390d

Figure 3.8.3-1 (Sheet 4 of 7)
[*Structural Modules in Containment Internal Structures*]*

*NRC Staff approval is required prior to implementing a change in this information.

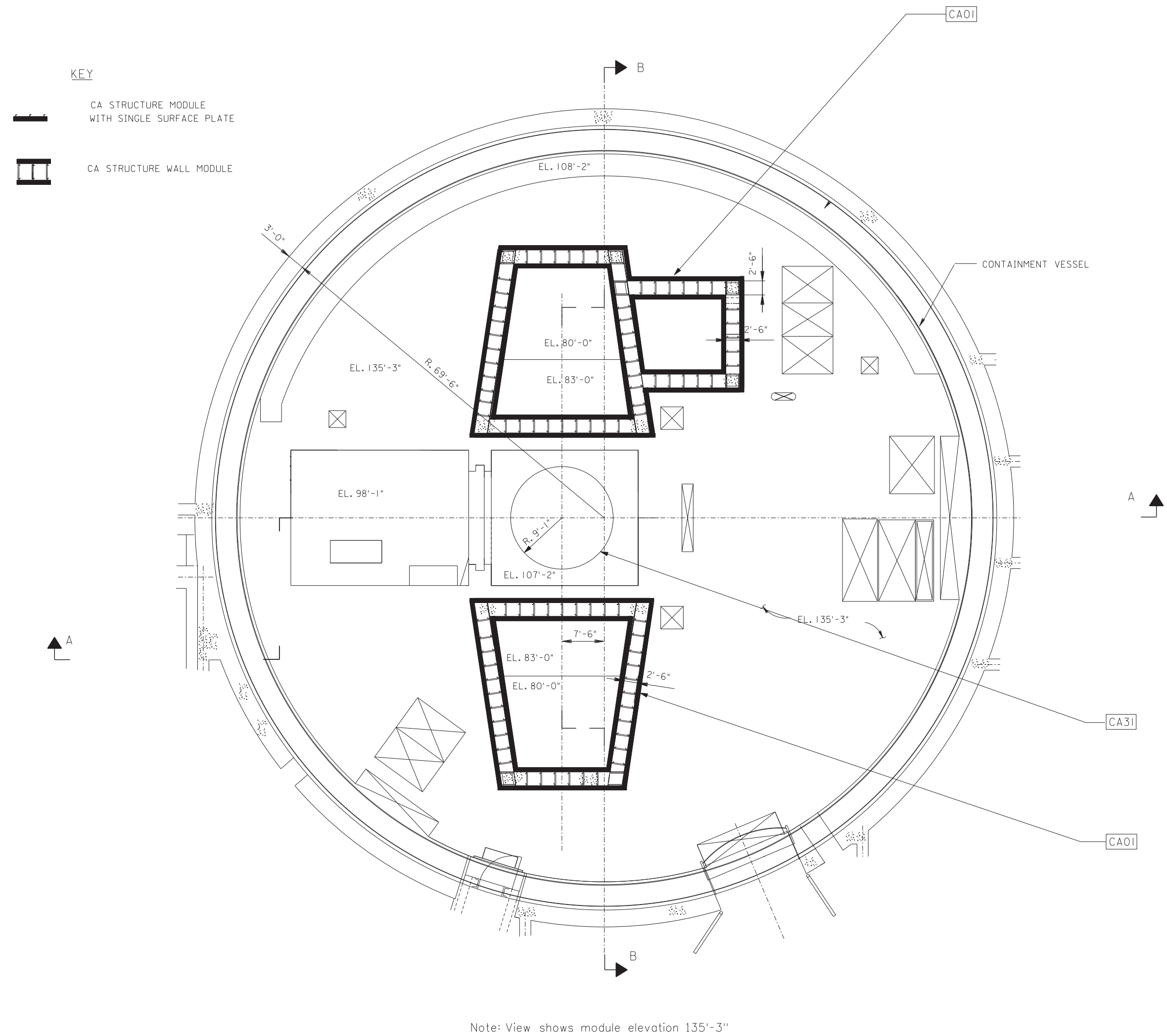


Figure 3.8.3-1 (Sheet 5 of 7)
[*Structural Modules in Containment Internal Structures*]*

*NRC Staff approval is required prior to implementing a change in this information.

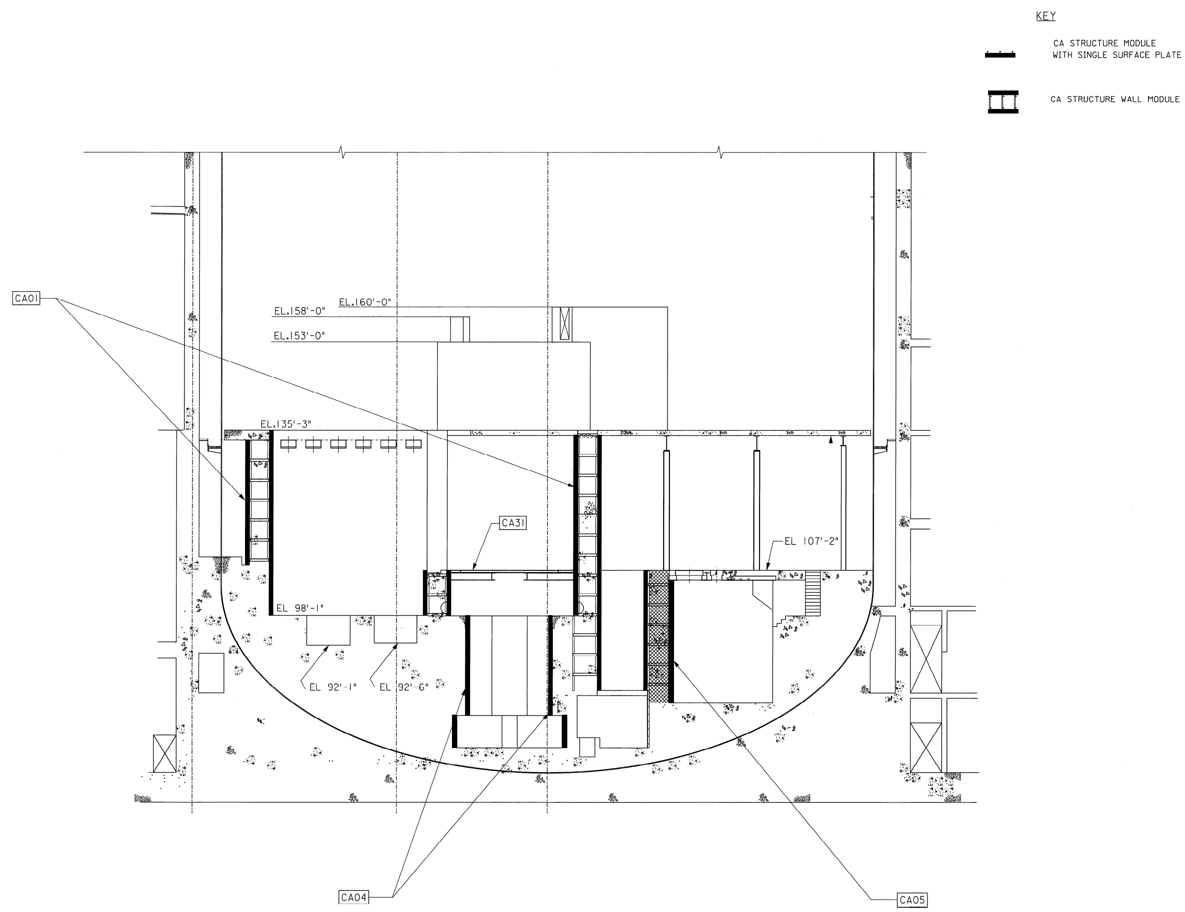


Figure 3.8.3-1 (Sheet 6 of 7)
[*Structural Modules in Containment Internal Structures*]*

*NRC Staff approval is required prior to implementing a change in this information.

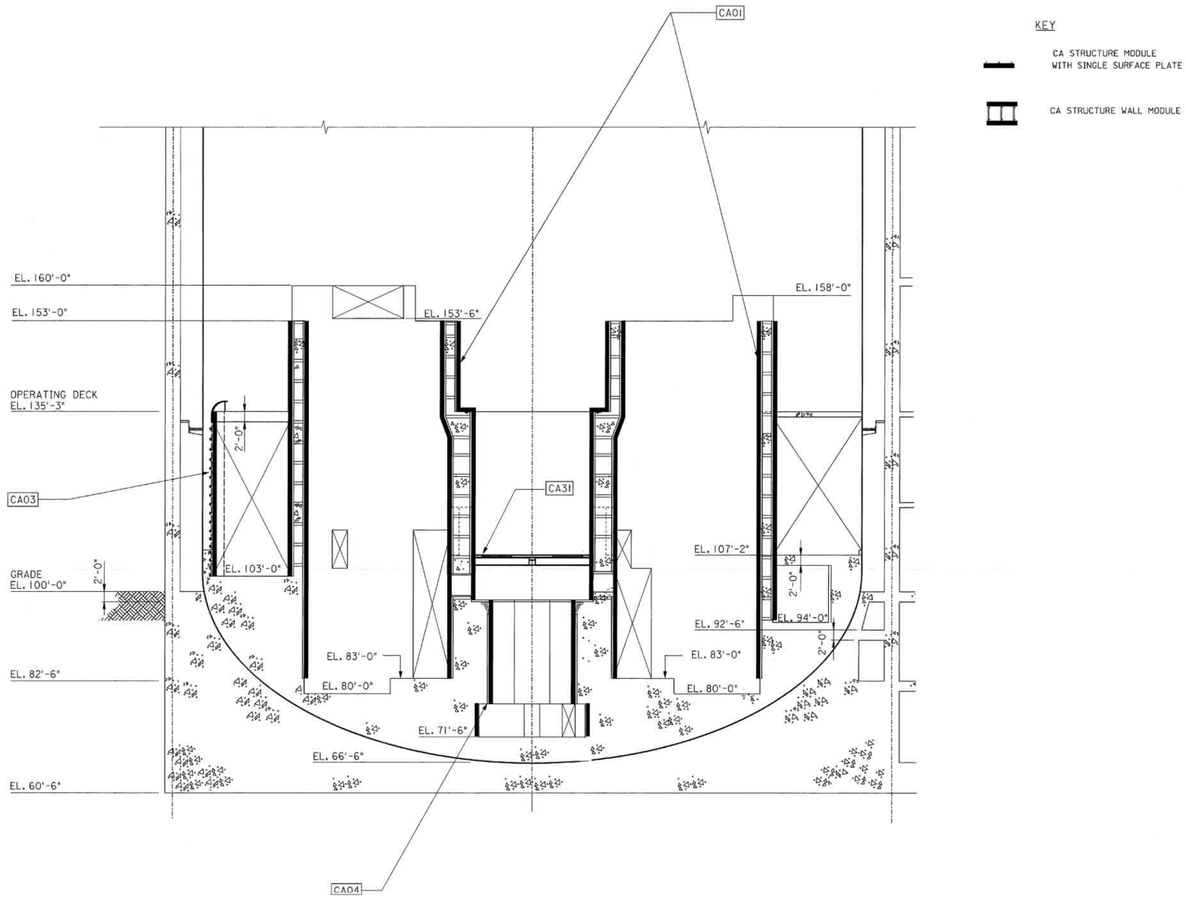
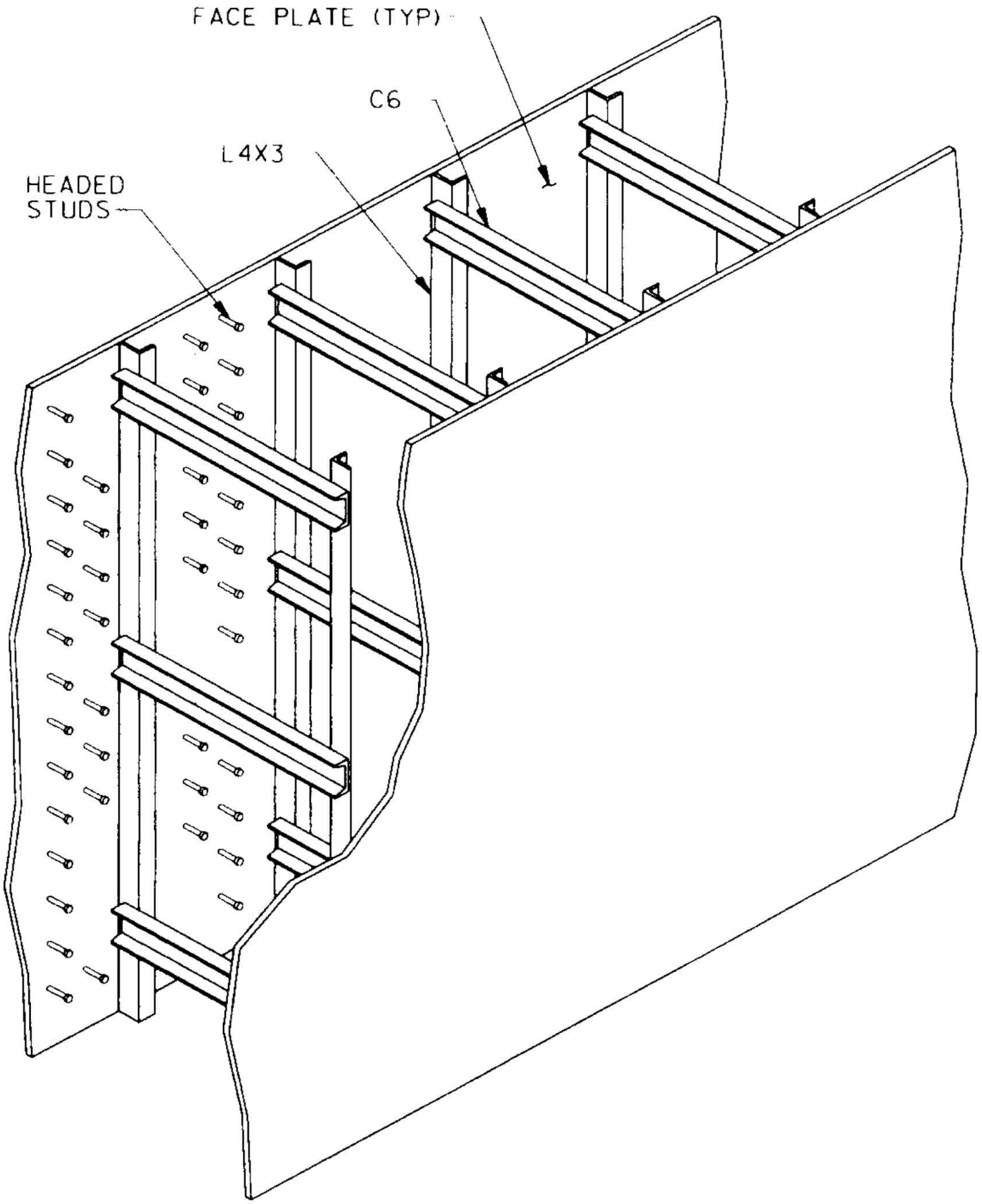


Figure 3.8.3-1 (Sheet 7 of 7)
[*Structural Modules in Containment Internal Structures*]*

*NRC Staff approval is required prior to implementing a change in this information.



Note: See Figure 3.8.3-8 for fabrication detail.

Figure 3.8.3-2
[*Typical Structural Wall Module*]*

*NRC Staff approval is required prior to implementing a change in this information.

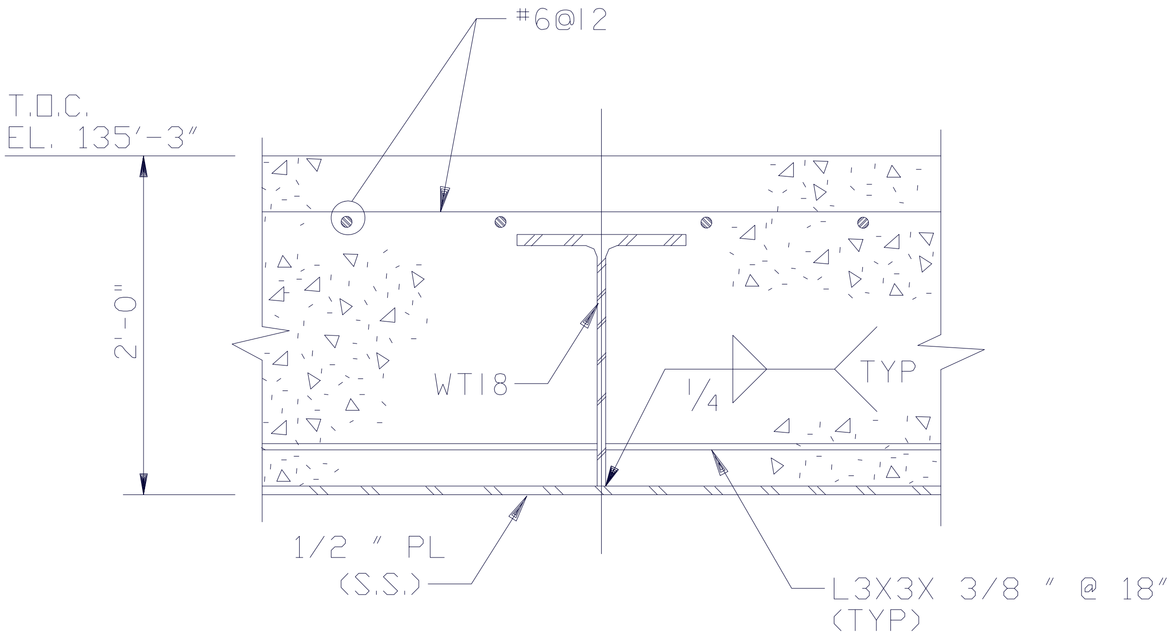


Figure 3.8.3-3
Structural Floor Module

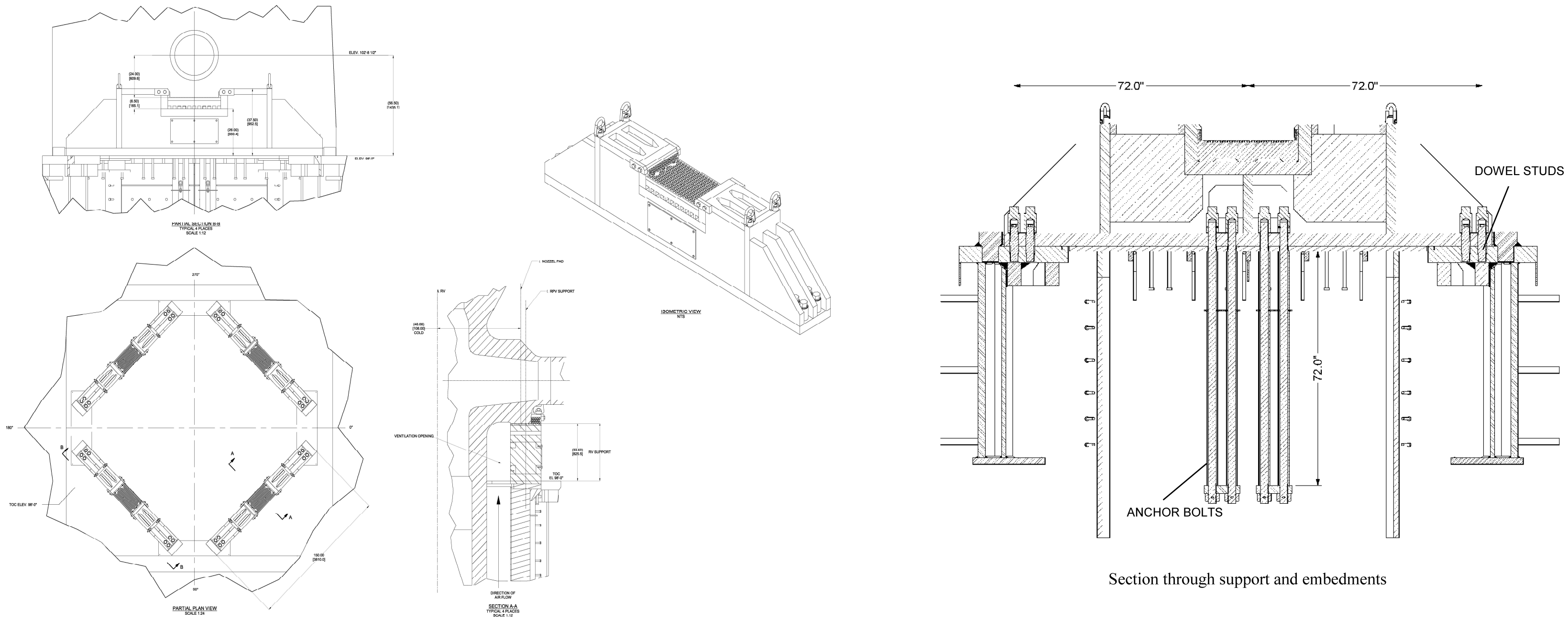


Figure 3.8.3-4
Reactor Vessel Supports

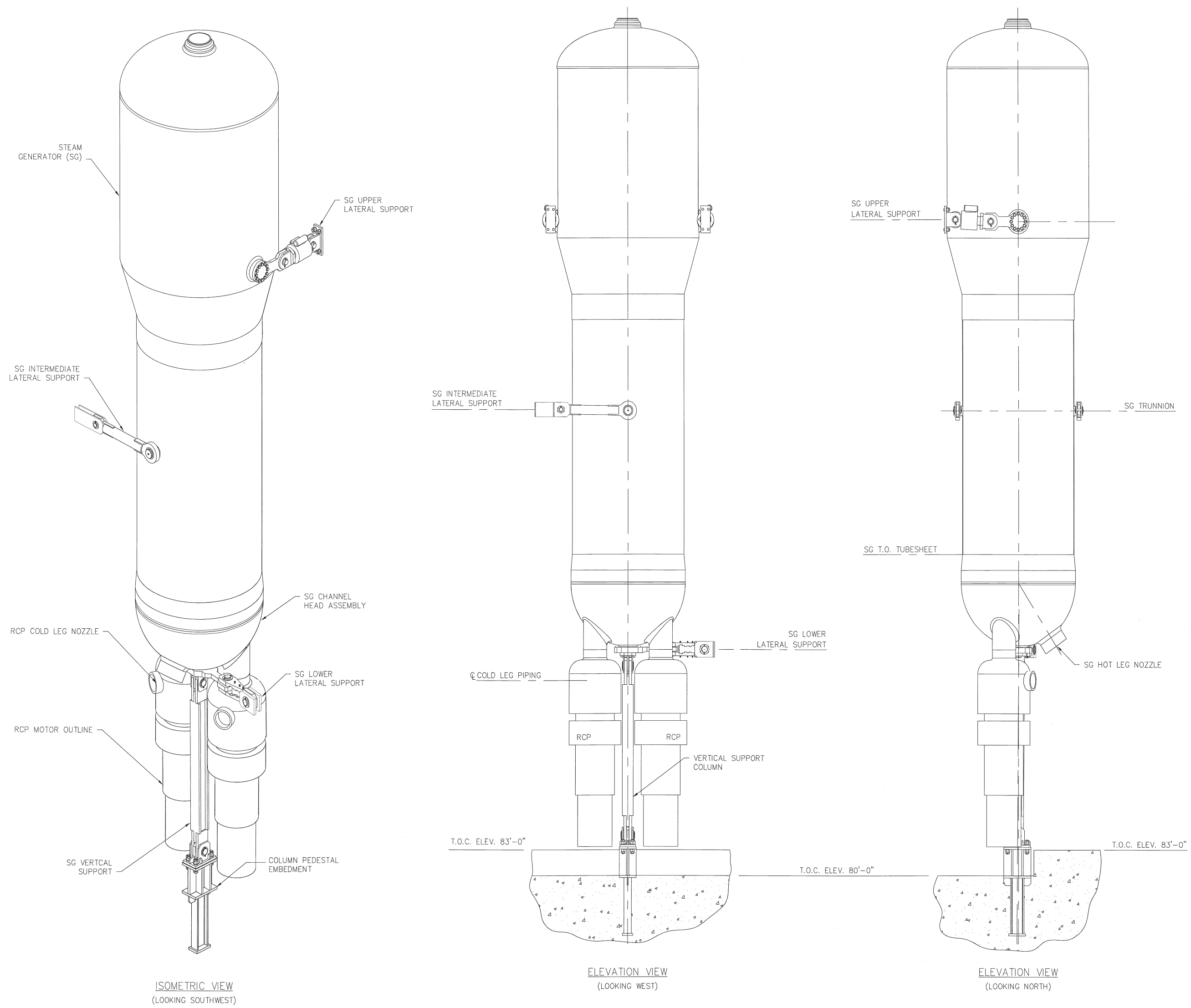


Figure 3.8.3-5 (Sheet 1 of 5)
Steam Generator Supports

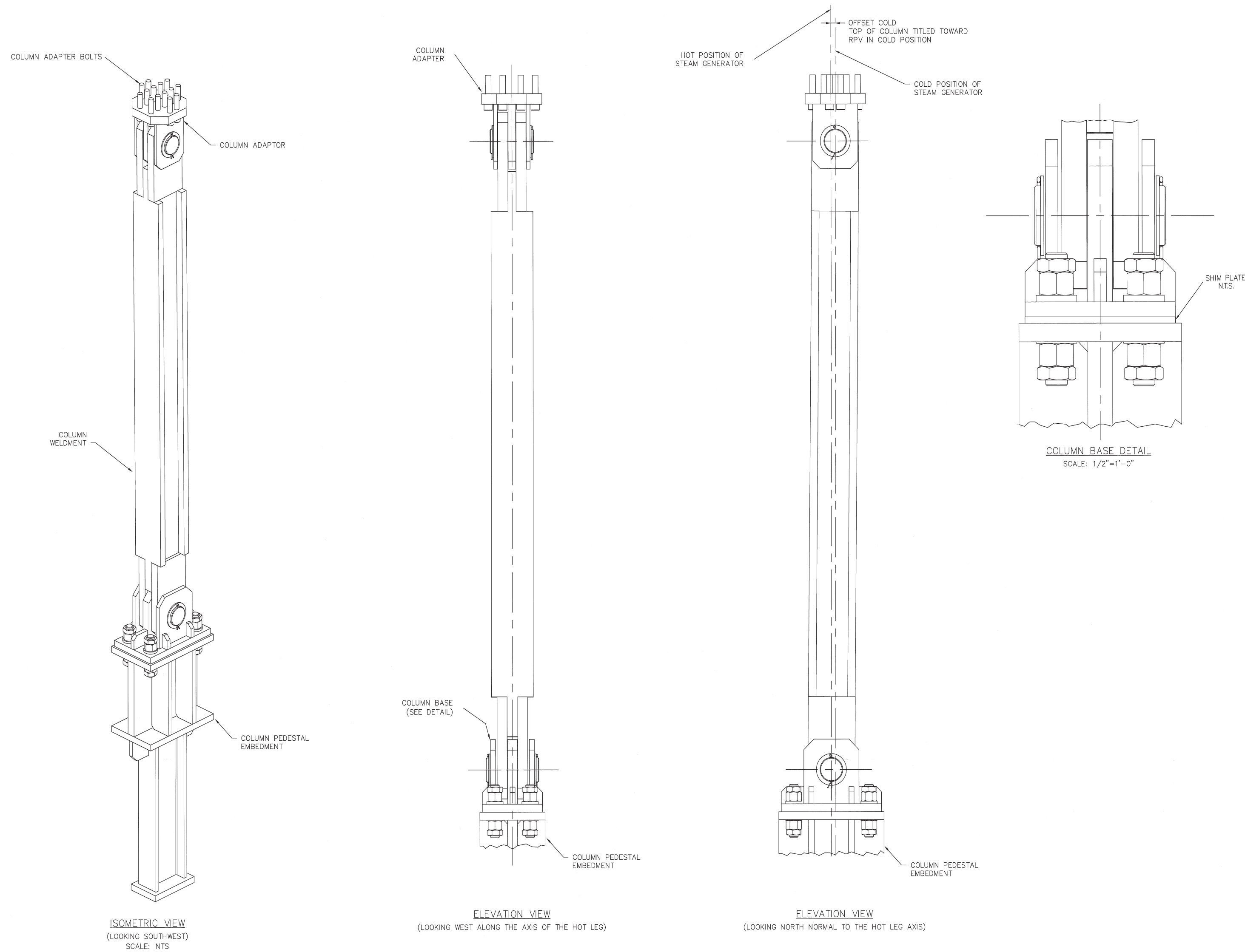
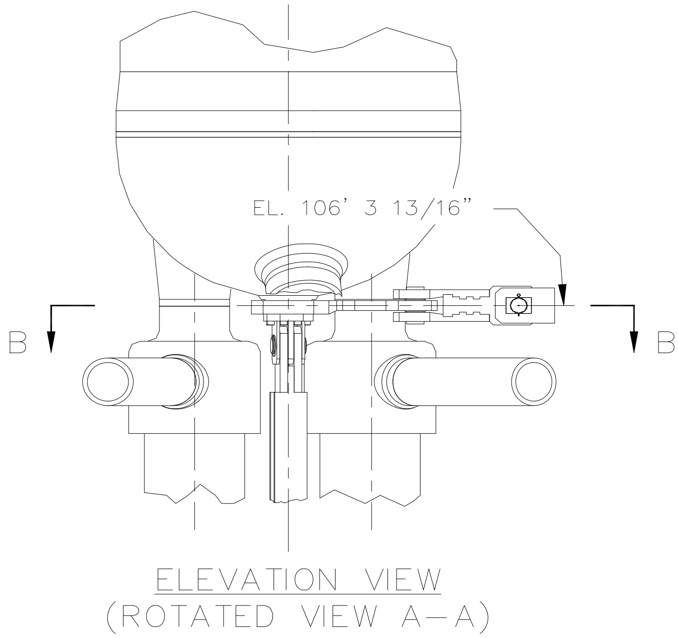
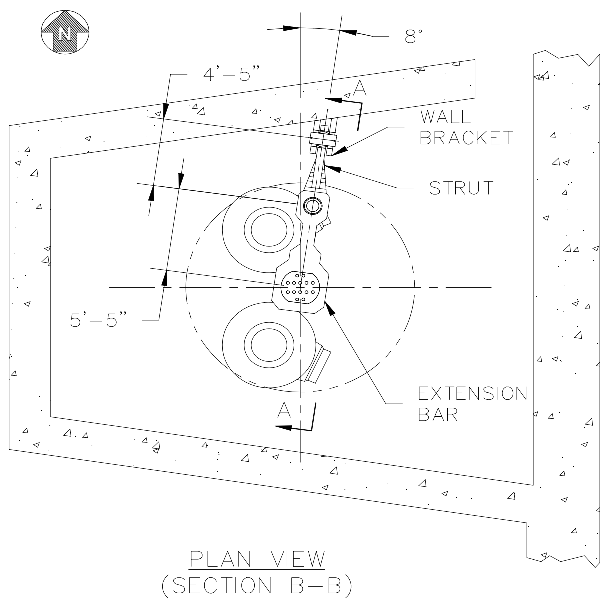


Figure 3.8.3-5 (Sheet 2 of 5)
Steam Generator Supports

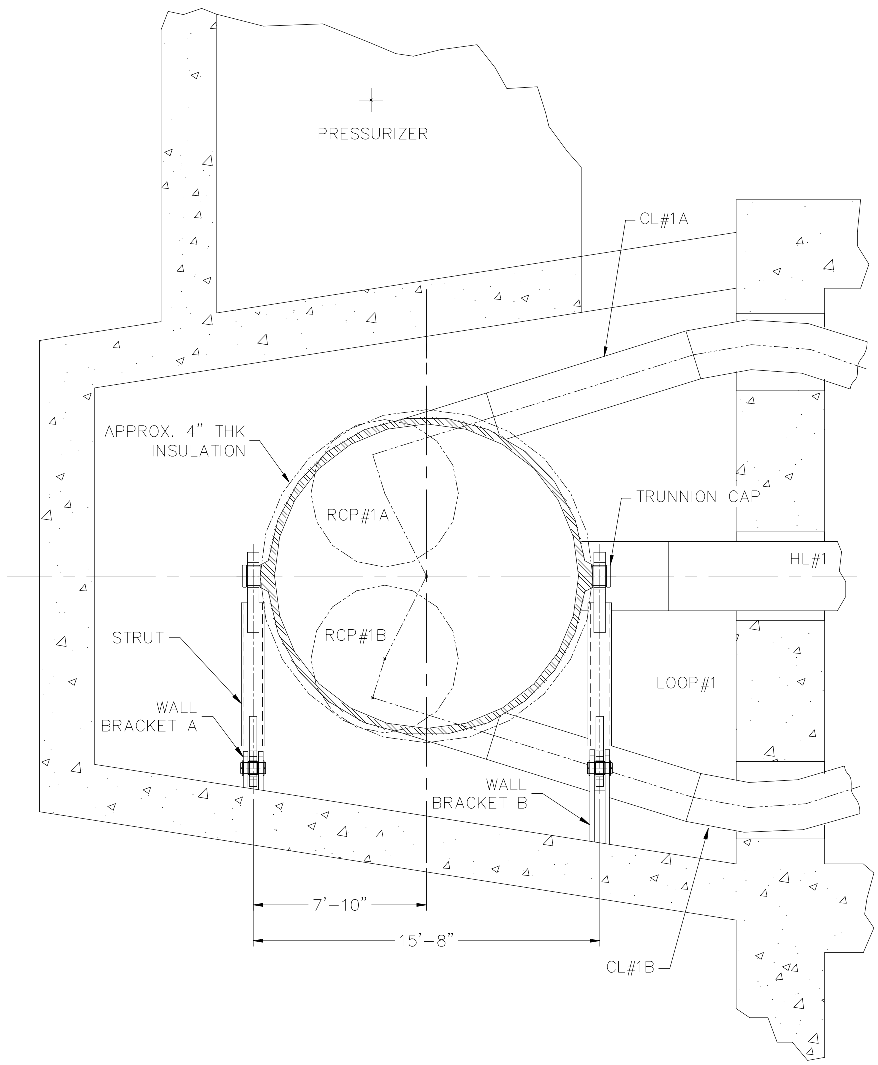


Lower Lateral Support Elevation View



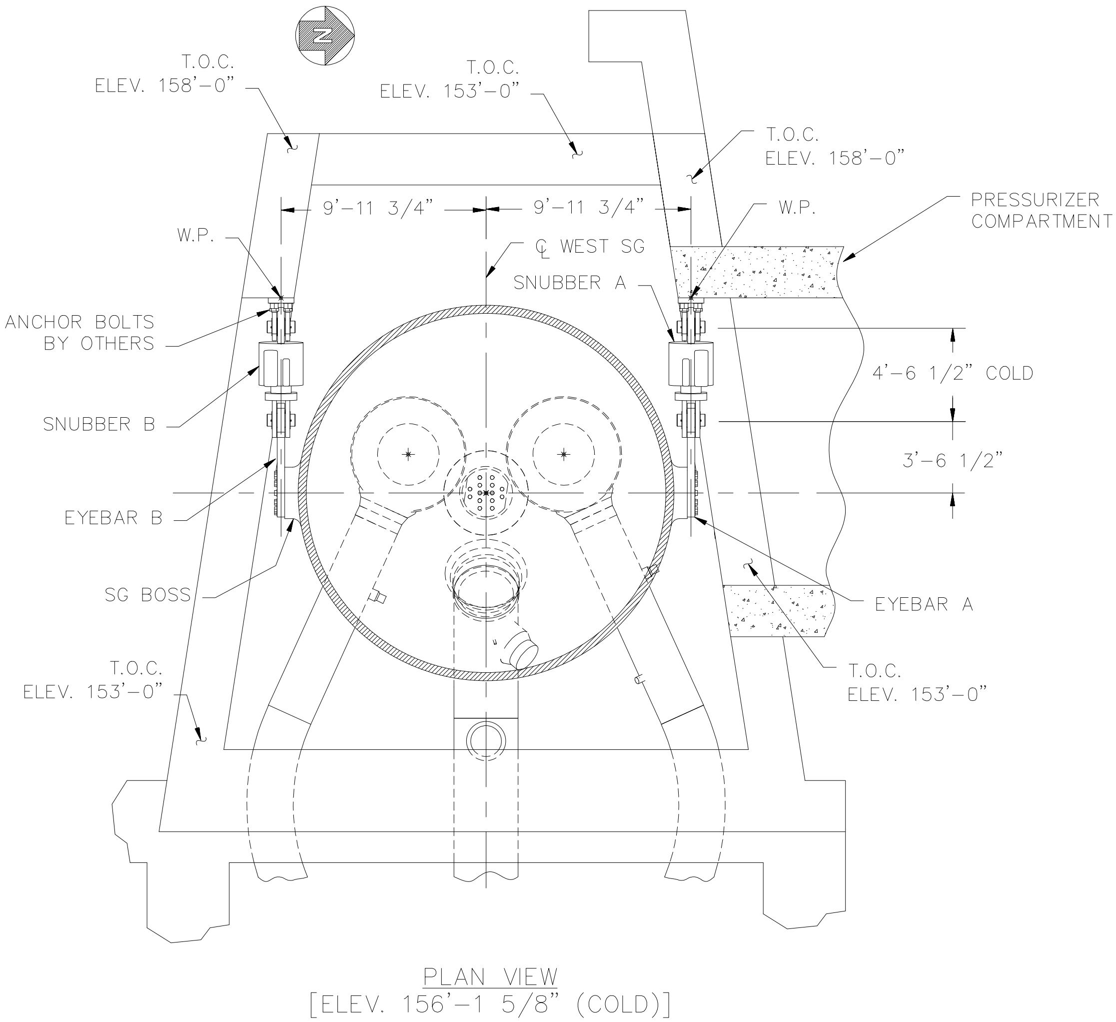
Lower Lateral Support Plan View

Figure 3.8.3-5 (Sheet 3 of 5)
Steam Generator Supports



Intermediate Lateral Support

Figure 3.8.3-5 (Sheet 4 of 5)
Steam Generator Supports



Upper Lateral Support

Figure 3.8.3-5 (Sheet 5 of 5)
Steam Generator Supports

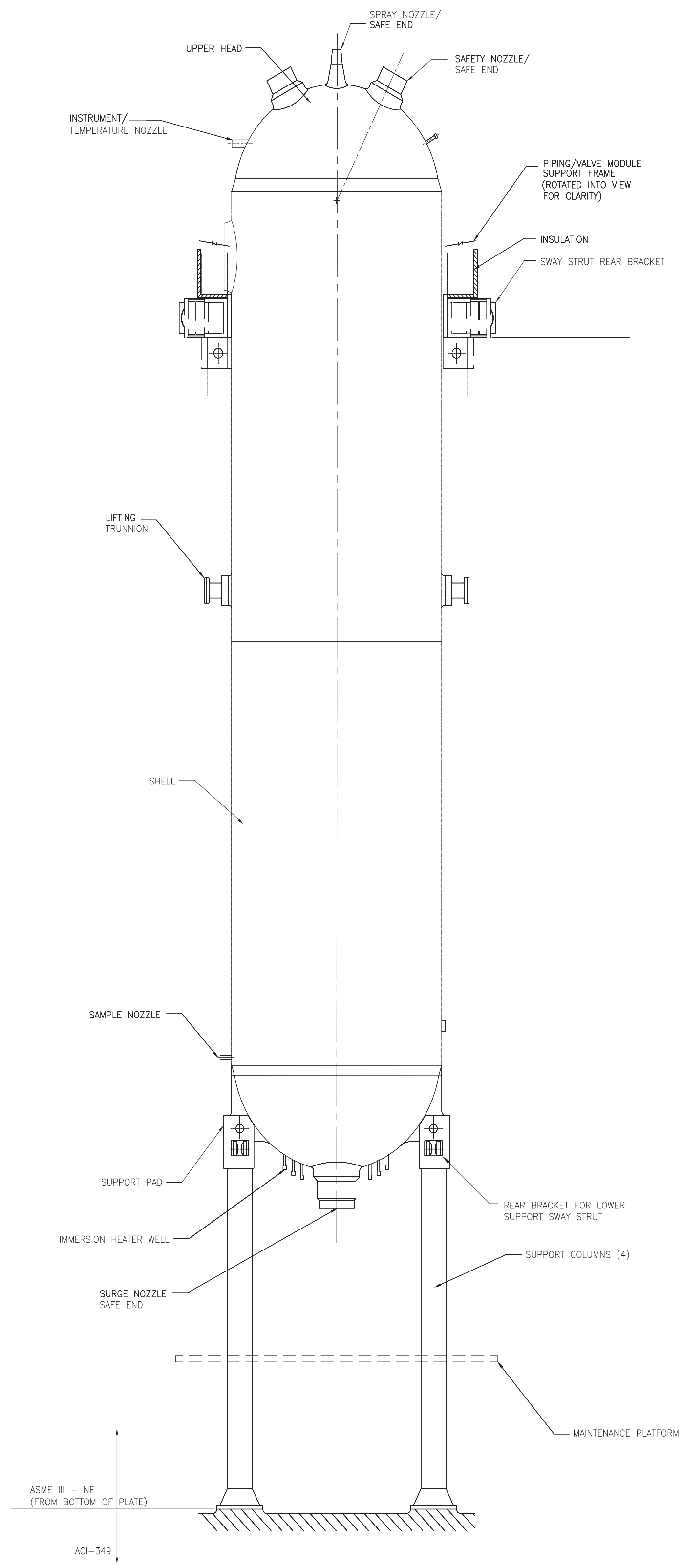


Figure 3.8.3-6 (Sheet 1 of 4)
Pressurizer Support Columns

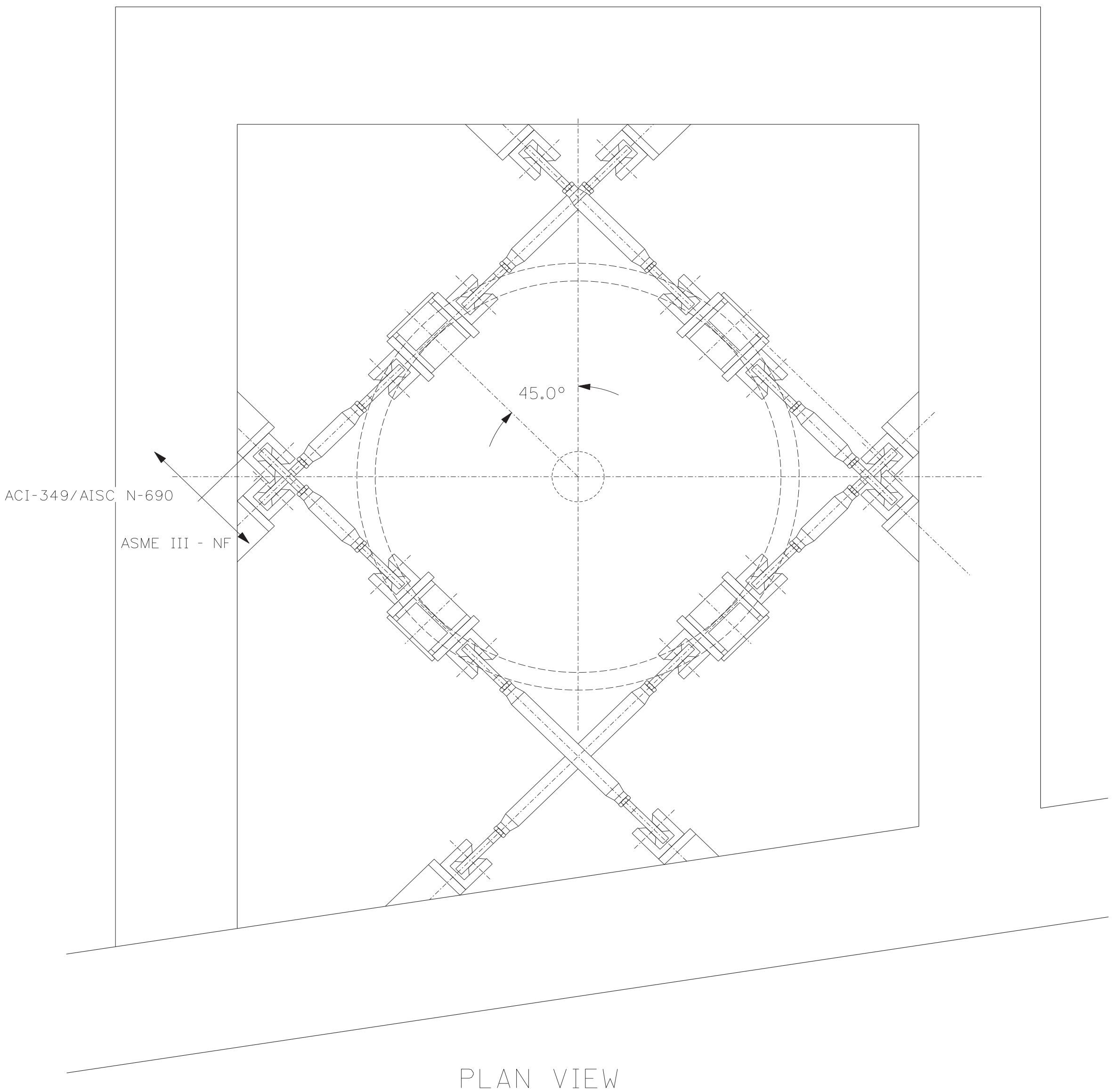
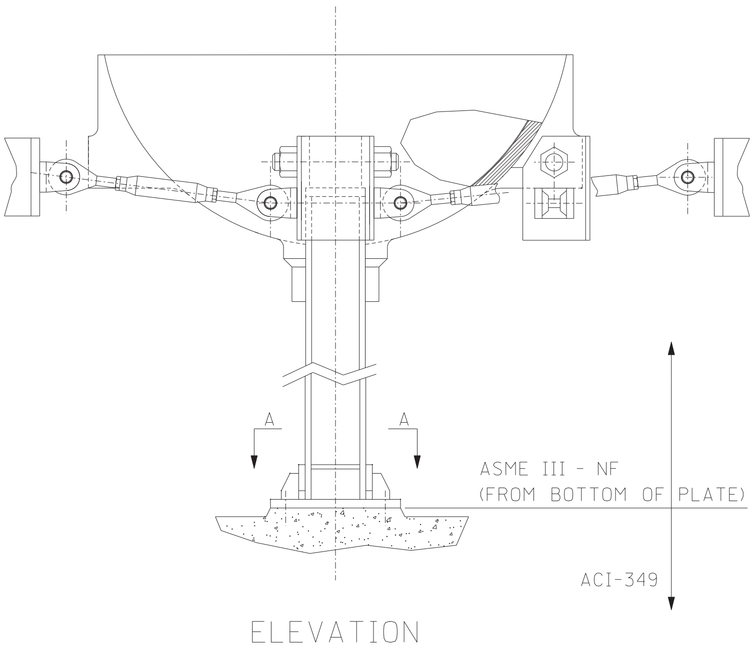


Figure 3.8.3-6 (Sheet 2 of 4)
Pressurizer Lower Lateral Supports



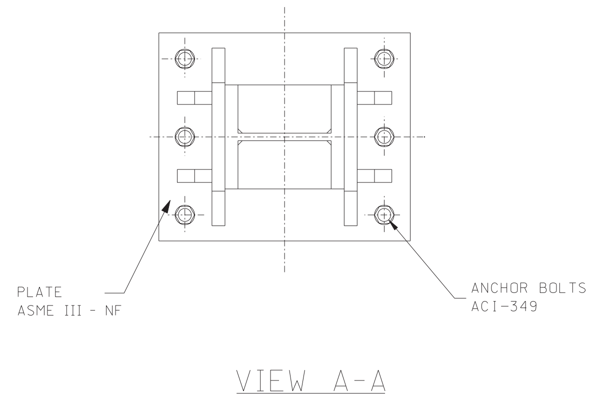


Figure 3.8.3-6 (Sheet 3 of 4)
Pressurizer Lower Supports

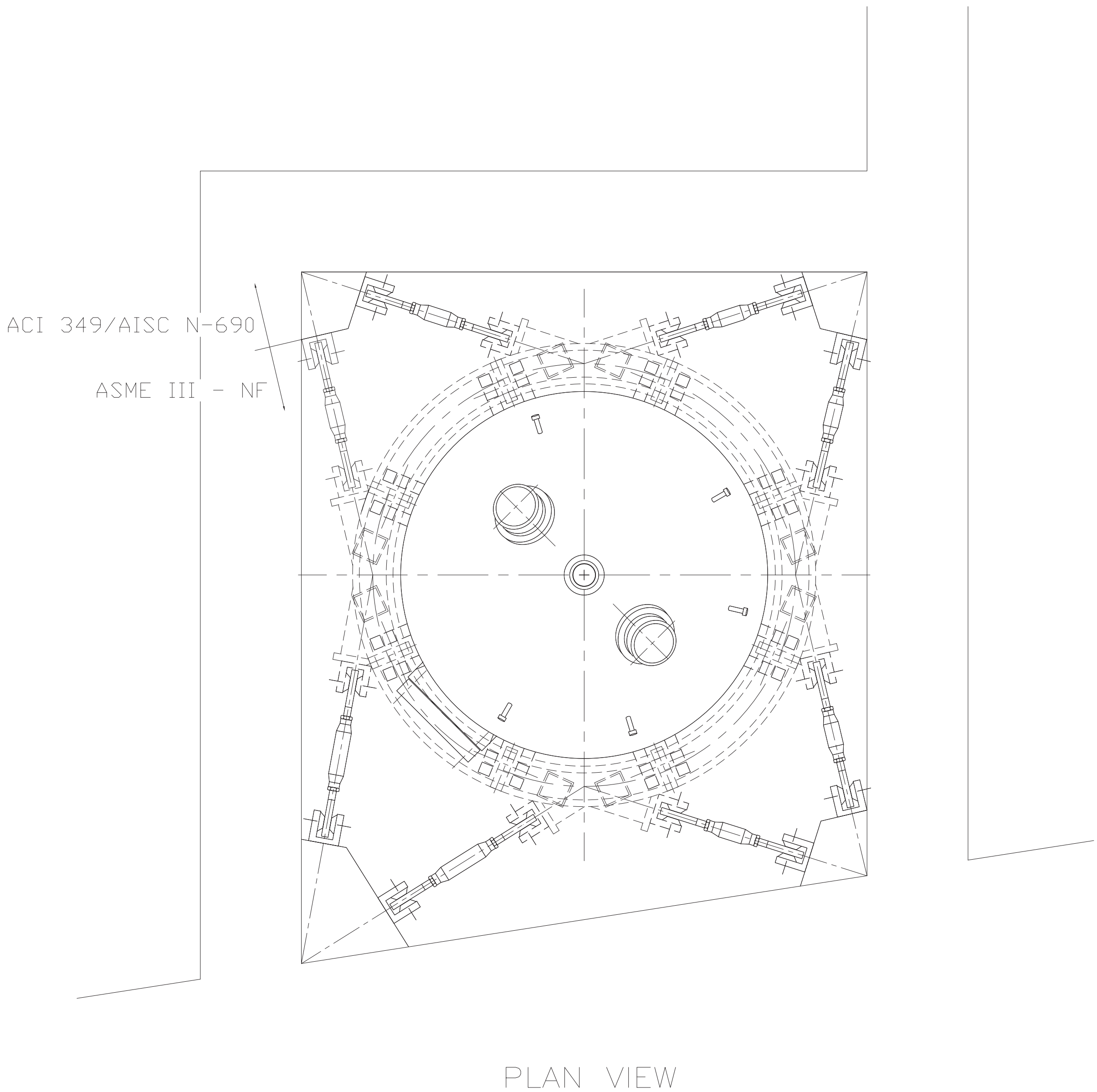


Figure 3.8.3-6 (Sheet 4 of 4)
Pressurizer Upper Supports

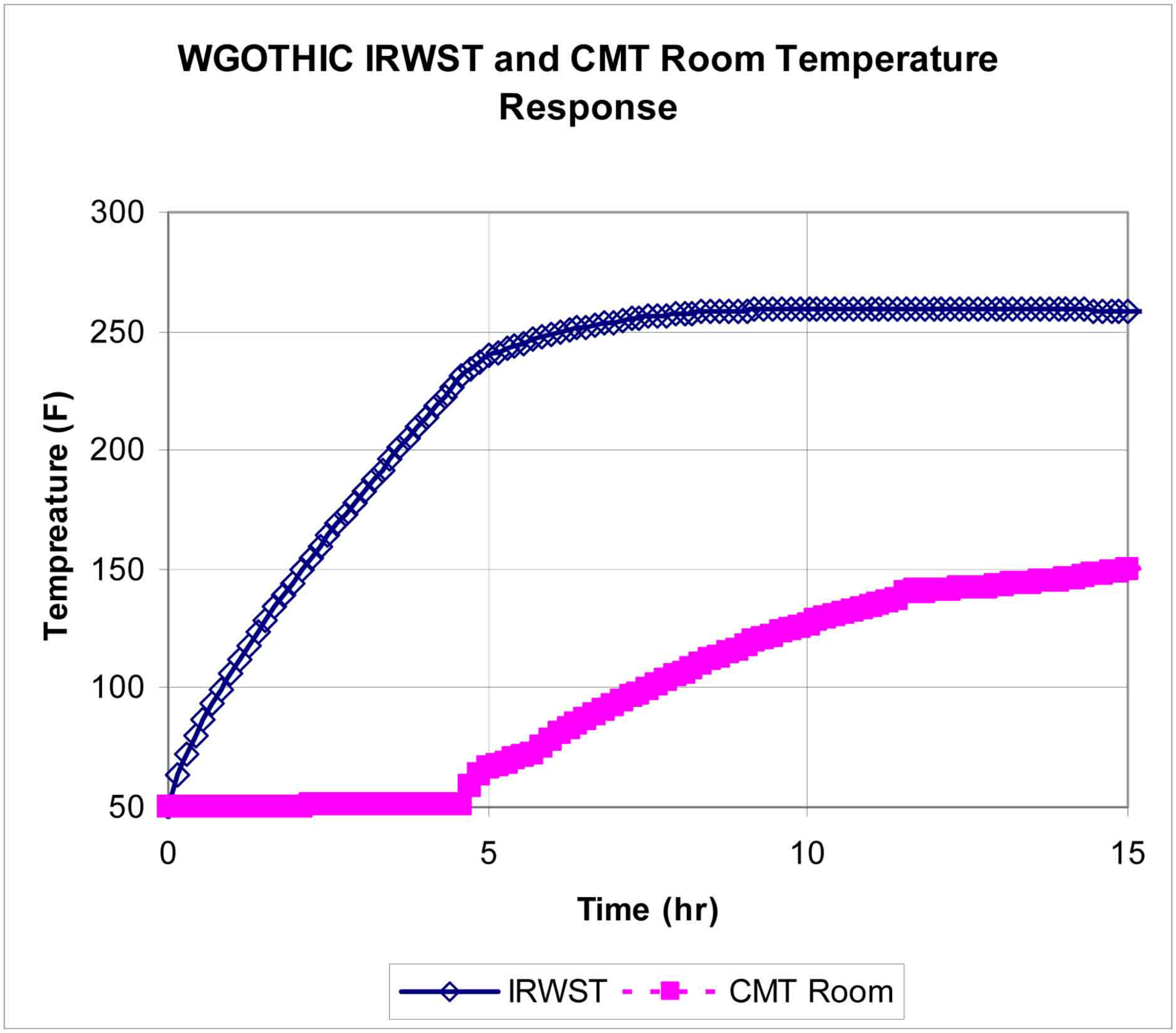
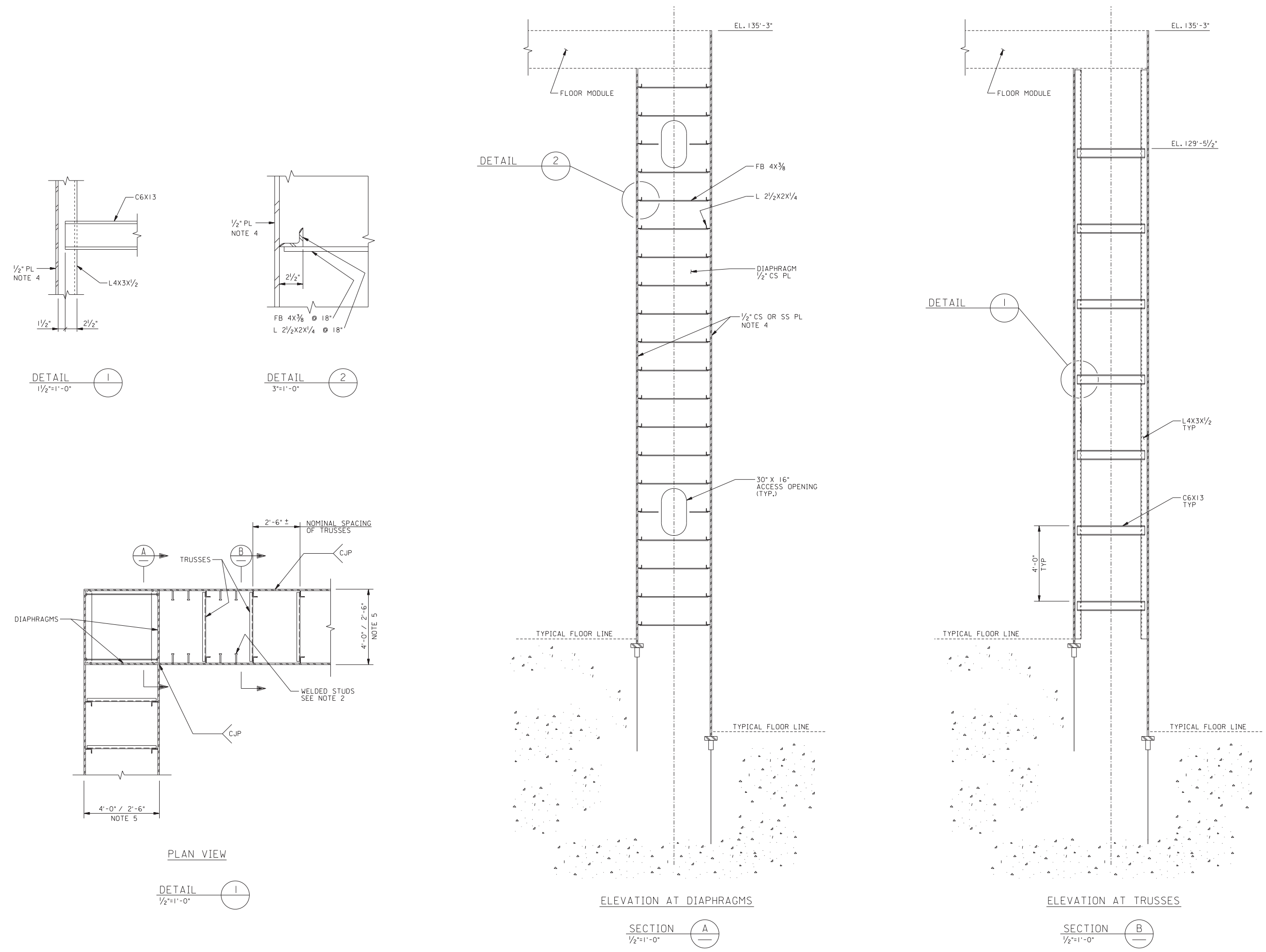


Figure 3.8.3-7
IRWST Temperature Transient



NOTES:

1. ALL WORK SHOWN ON THIS DRAWING IS SEISMIC CATEGORY 1.
2. DESIGN SPACING FOR WELDED STUDS IS AS FOLLOWS, UNLESS OTHERWISE REQUIRED BY THE SPECIFIC CODES AND STANDARDS INVOKED:
   3/4" ø X 6" @ 10" VERTICAL FOR A36 CS
   3/4" ø X 6" @ 10" HORIZONTAL FOR A36 CS
   5/8" ø X 6" @ 6" VERTICAL FOR SS
   5/8" ø X 6" @ 6" HORIZONTAL FOR SS
   THE TOLERANCE FOR THE SPACING IS ± 10% OF THE DESIGN SPACING
3. THE TYPICAL DETAILS SHOWN REPRESENT THE FUNDAMENTAL APPROACH FOR THE COMPOSITE WALL MODULES. THE FINAL DESIGN DETAILS MAY DIFFER FROM THOSE SHOWN FOR THE FOLLOWING REASONS.
   - ACCESSIBILITY FOR INSPECTION DURING FABRICATION AND CONSTRUCTION
   - LESSONS LEARNED FROM IMPLEMENTATION OF THE DESIGN
   - EASE OF FABRICATION AND CONSTRUCTION
   - RESOLUTION OF CONSTRUCTABILITY ISSUES AND SEQUENCES
   - VARIATION IN MODULE SIZE AND CONFIGURATION

   CHANGES MADE DURING DETAILED DESIGN ARE IN ACCORDANCE WITH THE SPECIFIC CODES AND STANDARDS INVOKED.
4. AS DESCRIBED IN SECTION 3.8.3.1.3. SOME CA01 AND CA05 MODULE WALL FACEPLATE THICKNESSES ARE GREATER THAN THE 0.5 INCH NOMINAL FACEPLATE THICKNESS.
5. SOME MODULE WALLS THAT SEPARATE EQUIPMENT SPACES FROM PERSONNEL ACCESS AREAS HAVE A WALL THICKNESS WHICH IS DIFFERENT THAN THE DIMENSIONS SHOWN. FIGURE 3.8.3-1 SHEETS 2 AND 4 IDENTIFY THE CA01 AND CA05 WALL THICKNESS DIFFERENCES.

TYPICAL STRUCTURAL WALL MODULE

See subsection 3.8.3.1.3 for information that is designated as Tier 2*.

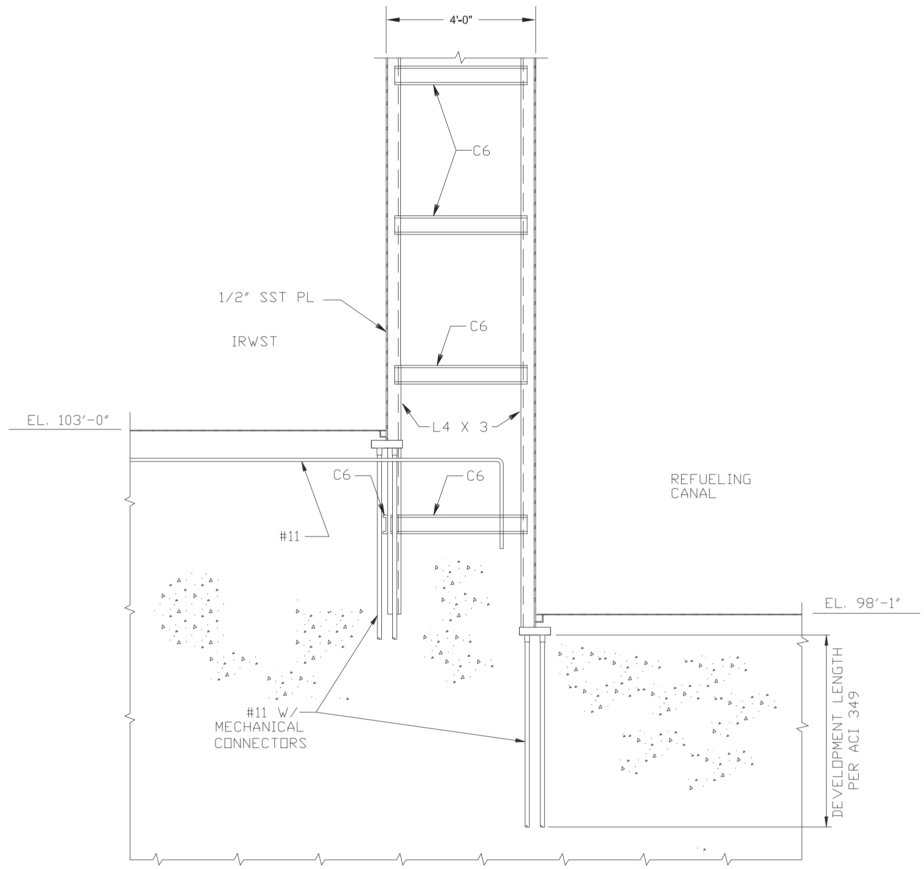
Figure 3.8.3-8 (Sheet 1 of 3)
[*Structural Modules – Typical Design Details*]*

*NRC Staff approval is required prior to implementing a change in this information.

NOTES:

THE TYPICAL DETAILS SHOWN REPRESENT THE FUNDAMENTAL APPROACH FOR THE COMPOSITE WALL MODULES. THE FINAL DESIGN DETAILS MAY DIFFER FROM THOSE SHOWN FOR THE FOLLOWING REASONS.

- ACCESSIBILITY FOR INSPECTION DURING FABRICATION AND CONSTRUCTION
- LESSONS LEARNED FROM IMPLEMENTATION OF THE DESIGN
- EASE OF FABRICATION AND CONSTRUCTION
- RESOLUTION OF CONSTRUCTABILTY ISSUES AND SEQUENCES
- VARIATION IN MODULE SIZE AND CONFIGURATION CHANGES MADE DURING DETAILED DESIGN ARE IN ACCORDANCE WITH THE SPECIFIC CODES AND STANDARDS INVOKED.
- AS DESCRIBED IN SECTION 3.8.3.1.3, SOME CA01 AND CA05 MODULE WALL FACEPLATE THICKNESSES ARE GREATER THAN THE 0.5 INCH NOMINAL FACEPLATE THICKNESS.
- SOME MODULE WALLS THAT SEPARATE EQUIPMENT SPACES FROM PERSONNEL ACCESS AREAS HAVE A WALL THICKNESS WHICH IS DIFFERENT THAN THE DIMENSIONS SHOWN. FIGURE 3.8.3-1 SHEETS 2 AND 4 IDENTIFY THE CA01 AND CA05 WALL THICKNESS DIFFERENCES.



CHANGES TO THE MECHANICAL CONNECTOR DETAIL SHOWN WILL MAINTAIN A DIRECT LOAD PATH TO TRANSFER LOADS FROM THE MODULE FACE PLATES TO THE REINFORCEMENT BARS IN THE BASE CONCRETE THROUGH THE USE OF INTERVENING PLATES, MECHANICAL CONNECTORS, AND WELDS.

See subsection 3.8.3.1.3 for information that is designated as Tier 2*.

Figure 3.8.3-8 (Sheet 2 of 3)
[*Structural Modules – Typical Design Details*]*

*NRC Staff approval is required prior to implementing a change in this information.

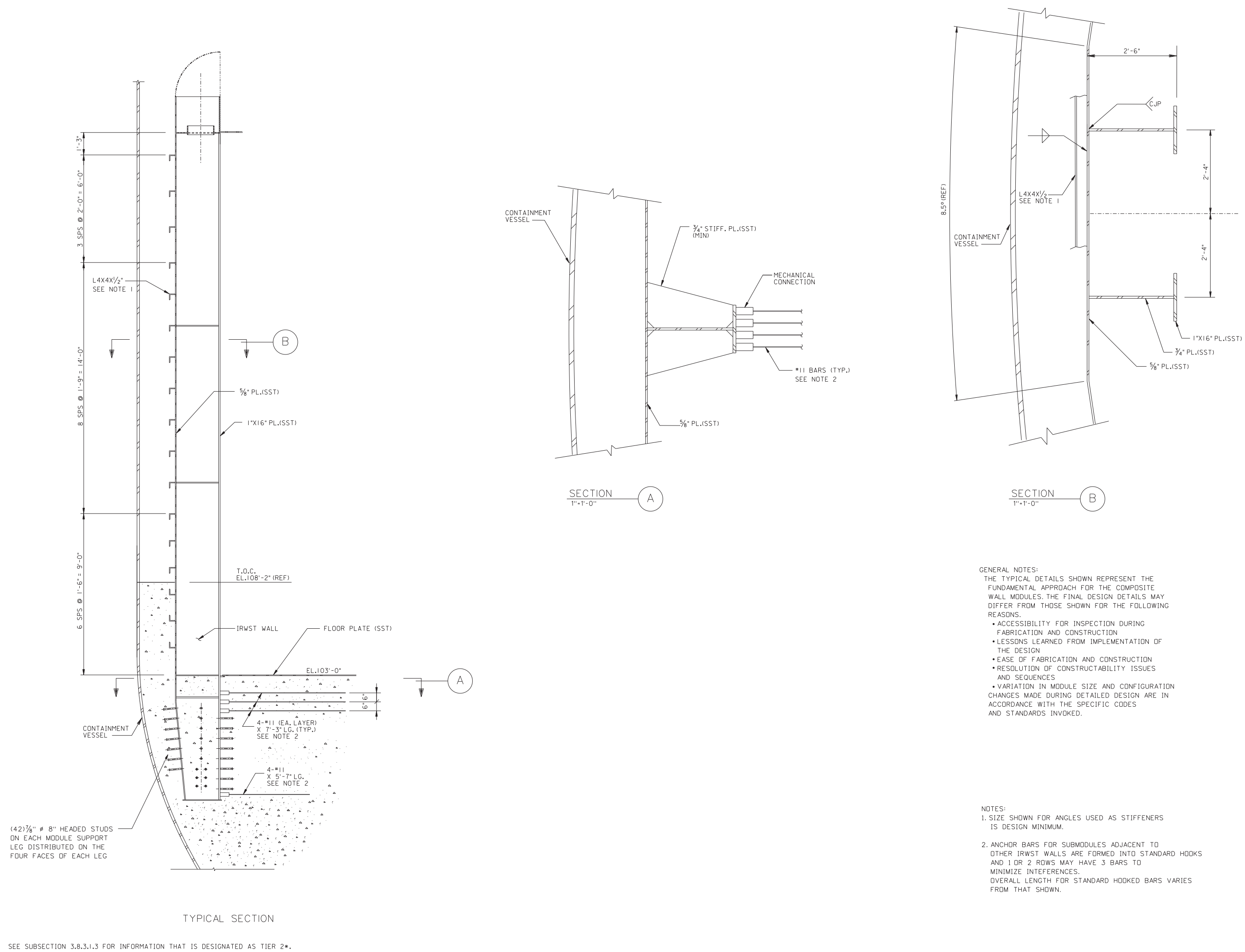


Figure 3.8.3-8 (Sheet 3 of 3)
[*Structural Modules – Typical Design Details*]*

*NRC Staff approval is required prior to implementing a change in this information.

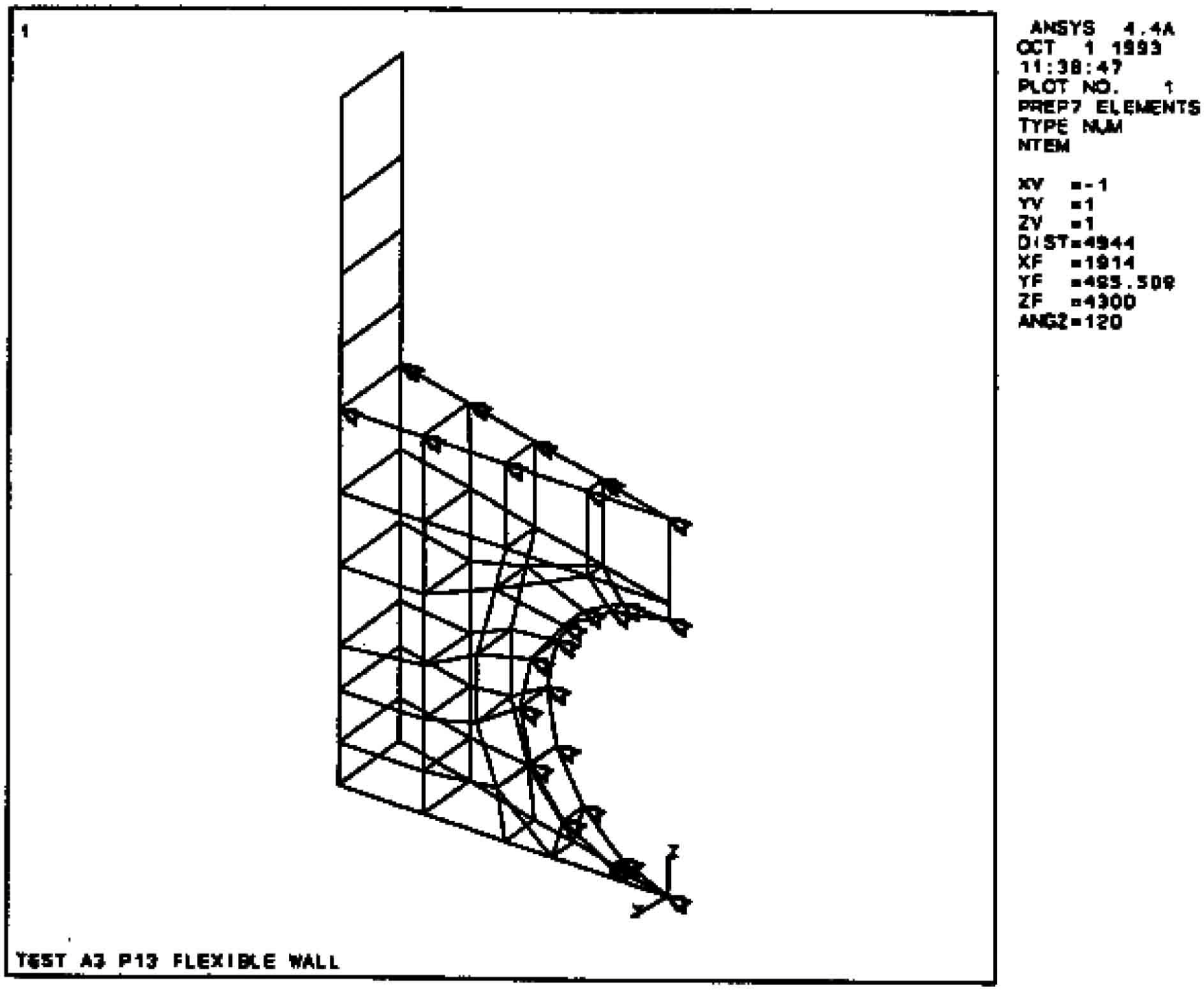


Figure 3.8.3-9
Test Tank Finite Element Model

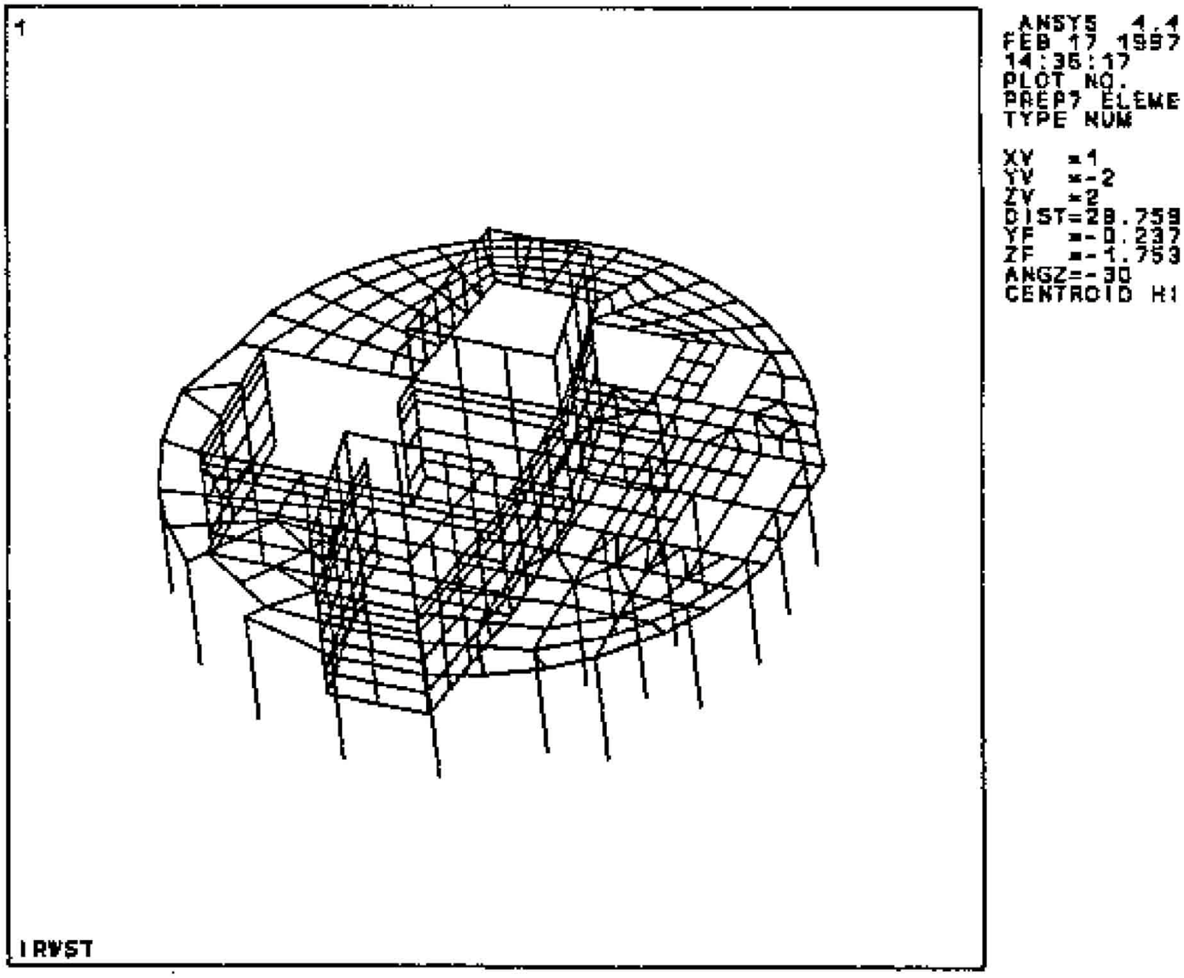


Figure 3.8.3-10 (Sheet 1 of 2)
IRWST Fluid Structure Finite Element Model
CIS Structural Model

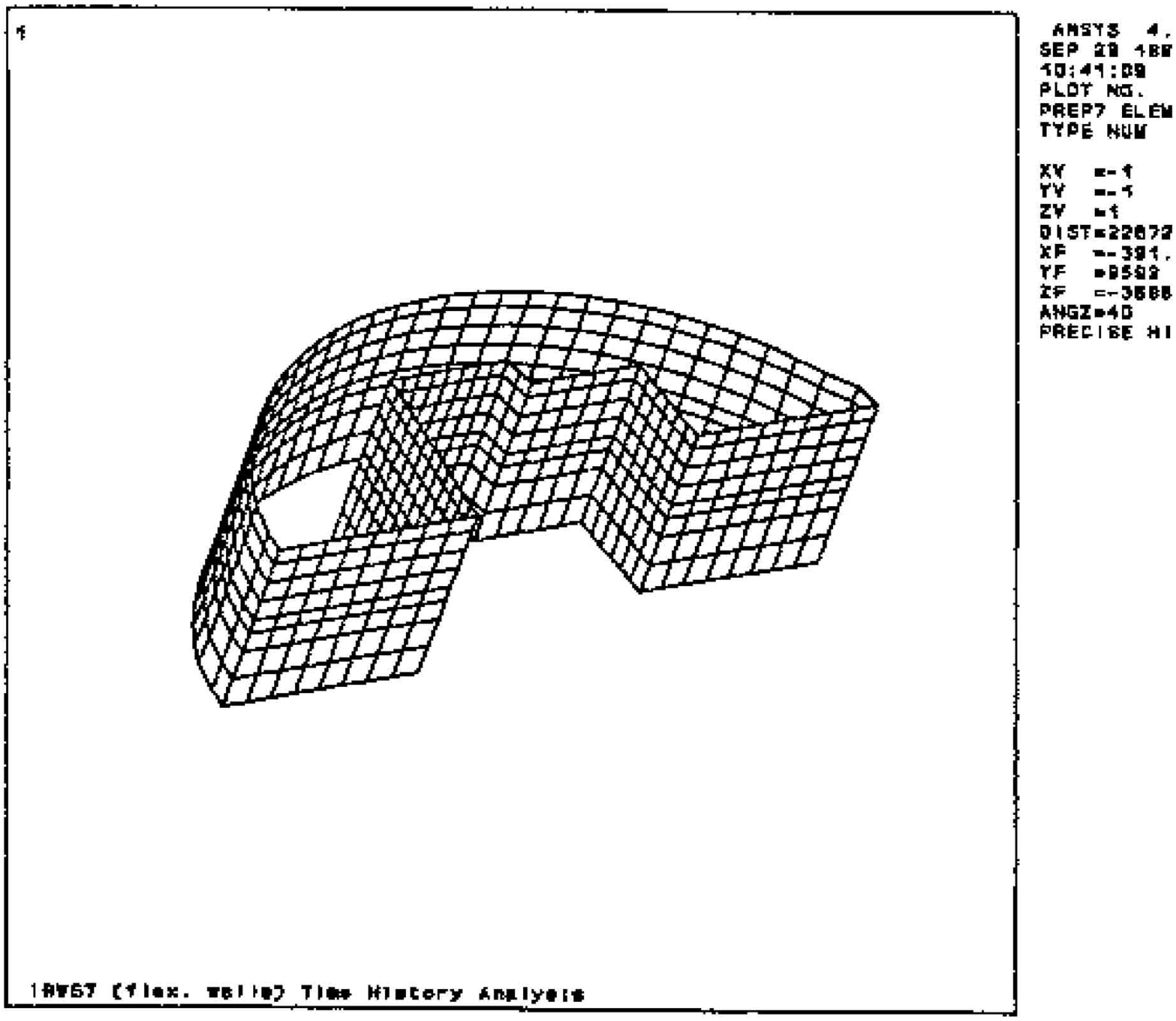


Figure 3.8.3-10 (Sheet 2 of 2)
IRWST Fluid Structure Finite Element Model
IRWST Structural Model

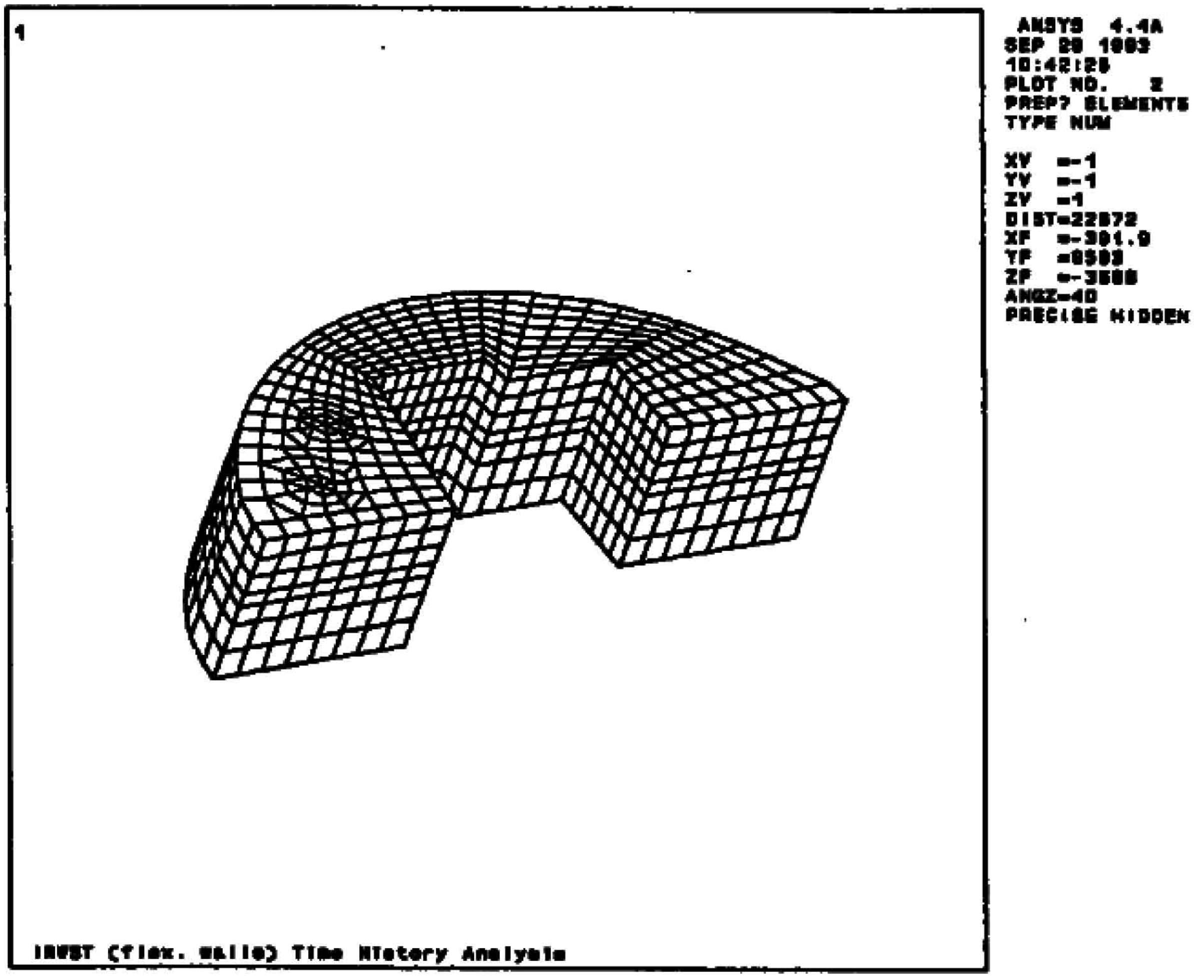


Figure 3.8.3-11
IRWST Fluid Structure Finite Element Model
Fluid Model

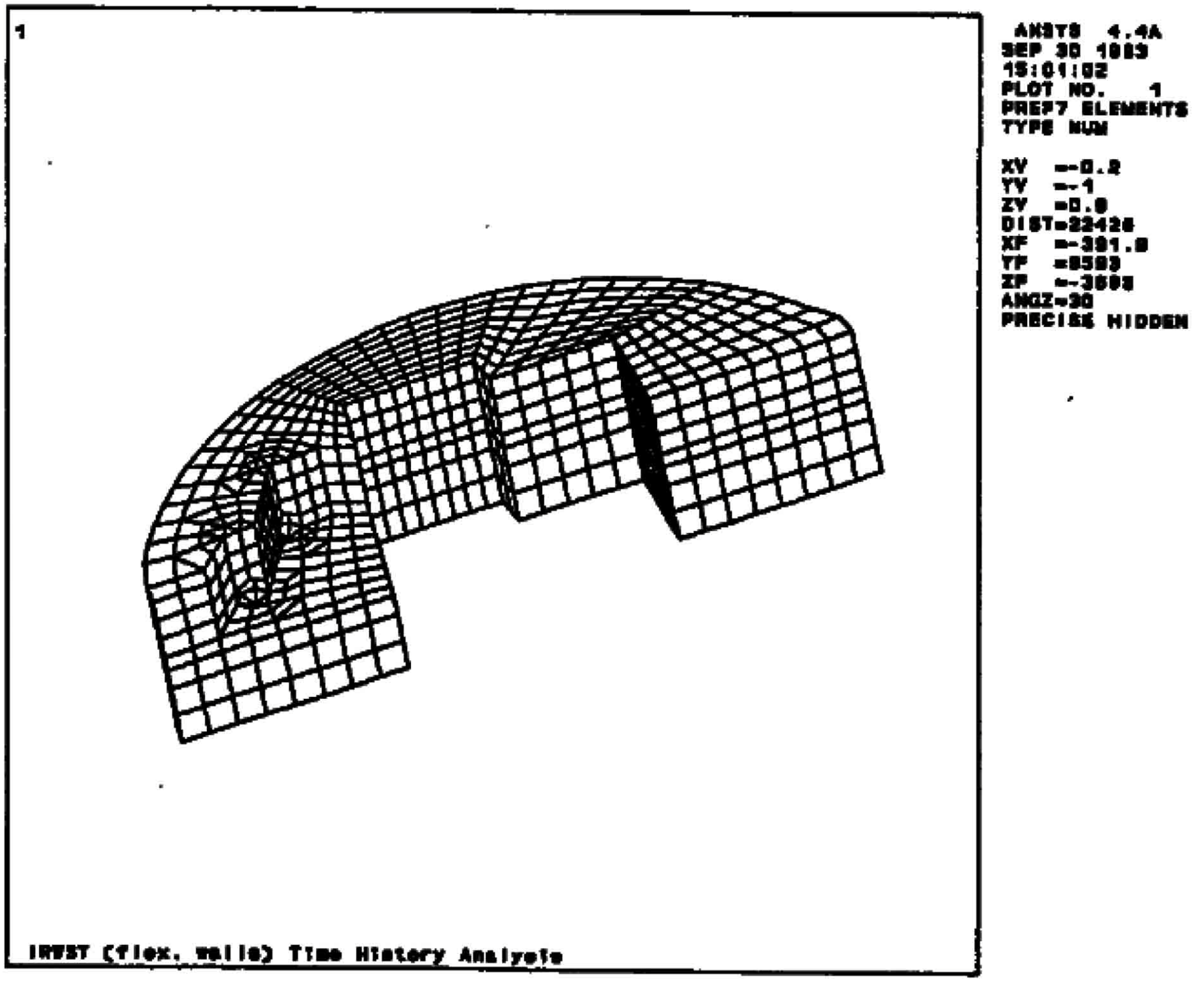
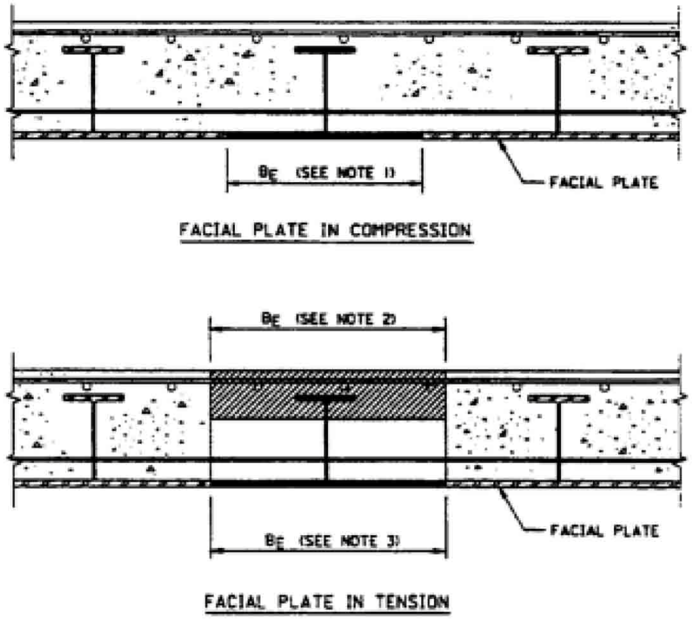


Figure 3.8.3-12
IRWST Fluid Structure Finite Element Model
Sparger Region Detail



EFFECTIVE SECTIONS FOR FLOOR MODULES

NOTES:

1. FOR FACIAL PLATE IN COMPRESSION, b_e, IS DETERMINED PER SECTION 3.8.3.5.4.2
2. EFFECTIVE WIDTH OF CONCRETE, b_e, IS DETERMINED PER SECTION Q 1.11.1 OF AISC N690.
3. FOR FACIAL PLATE IN TENSION, b_e, IS TAKEN TO BE ONE HALF OF THE DISTANCE TO THE ADJACENT BEAMS.

Figure 3.8.3-13
Effective Sections for Floor Modules

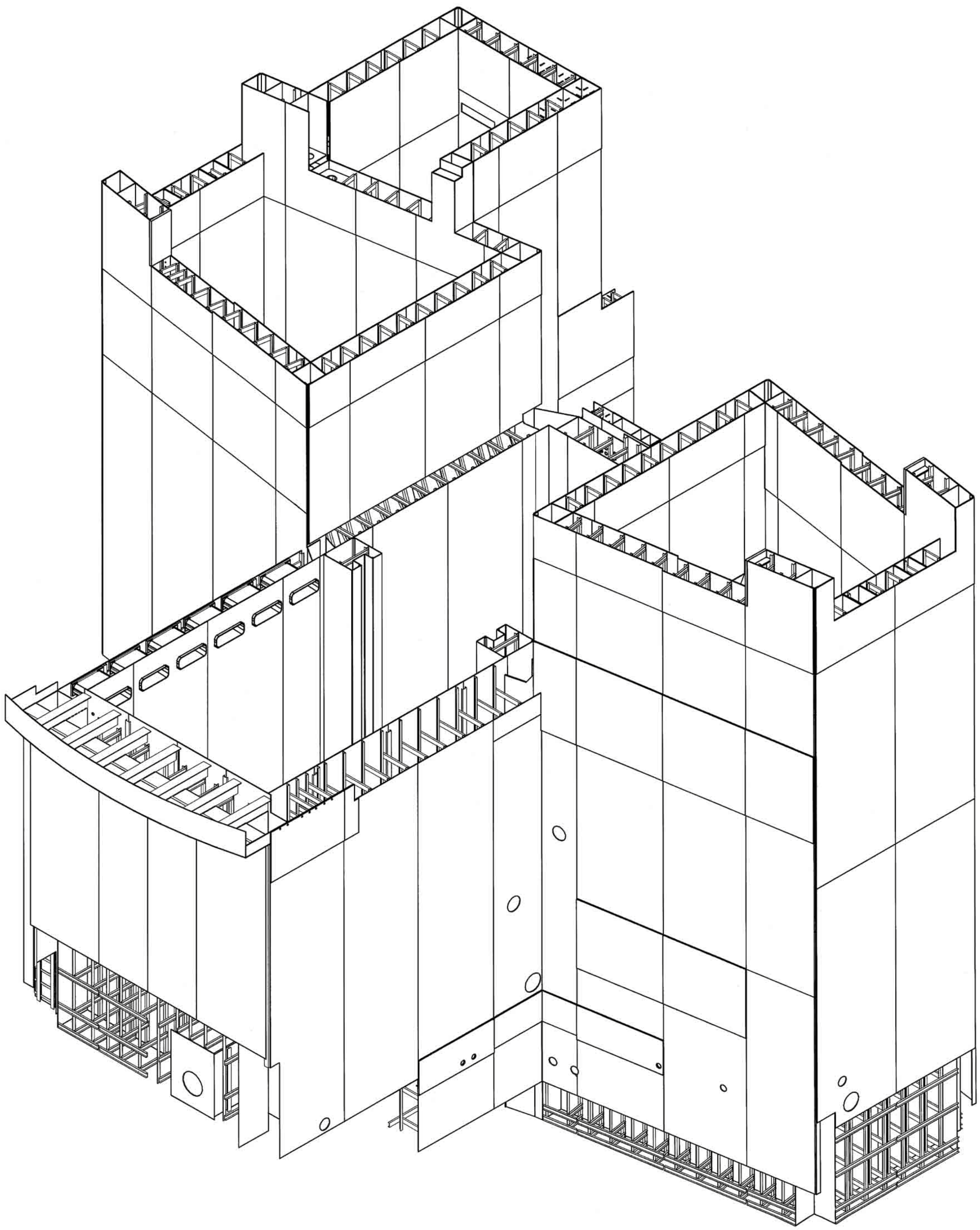


Figure 3.8.3-14 (Sheet 1 of 5)
[*CA-01 Module*]*

*NRC Staff approval is required prior to implementing a change in this information.

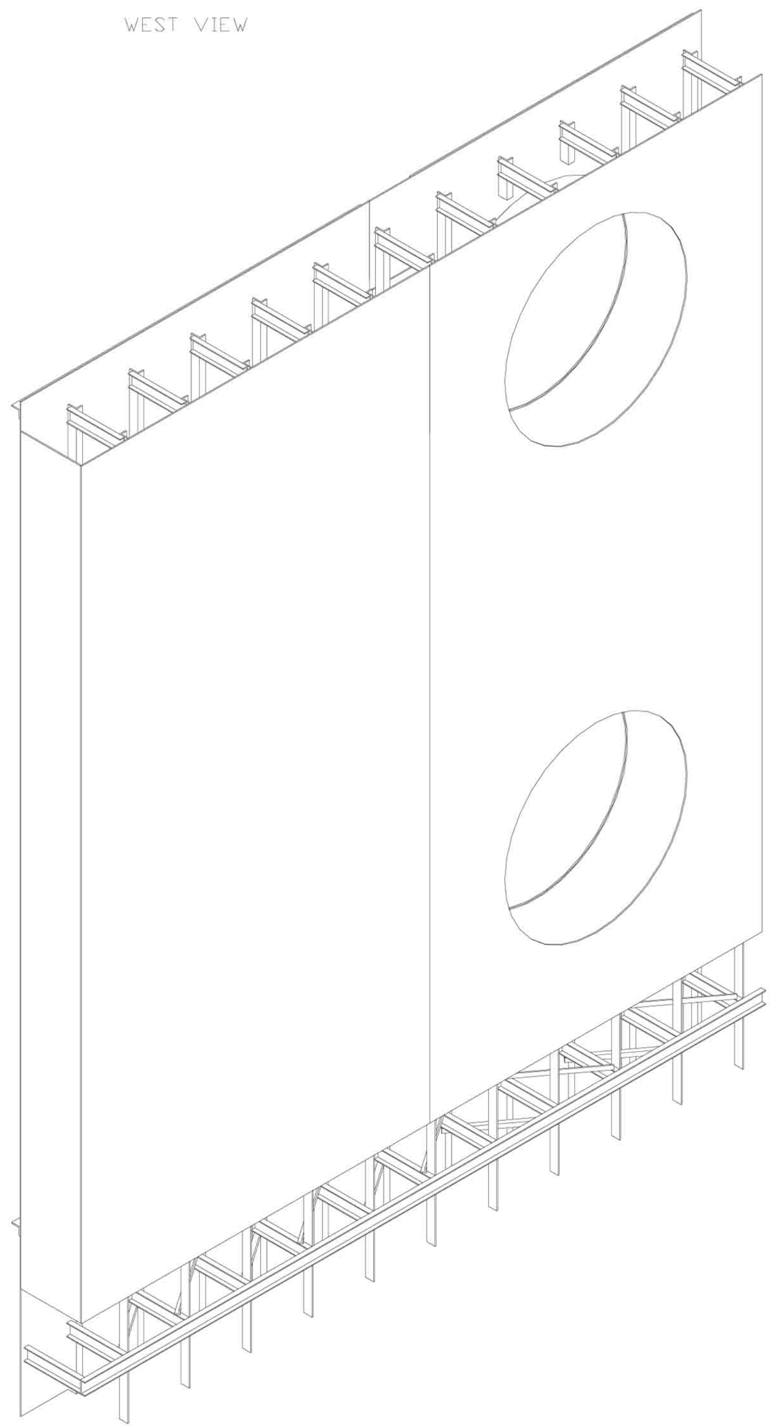


Figure 3.8.3-14 (Sheet 2 of 5)
[*CA-02 Module*]*

*NRC Staff approval is required prior to implementing a change in this information.

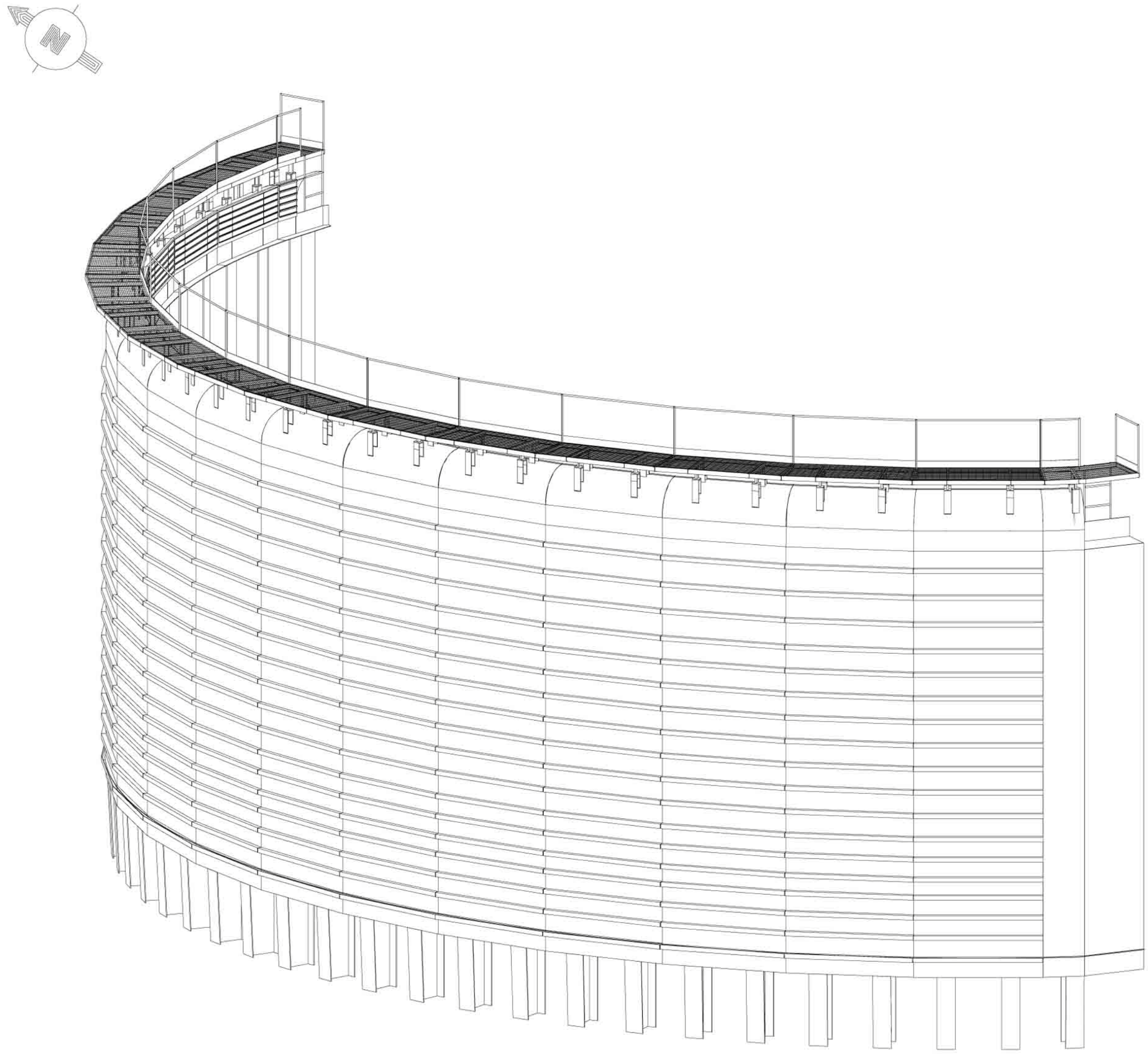


Figure 3.8.3-14 (Sheet 3 of 5)
[*CA-03 Module*]*

*NRC Staff approval is required prior to implementing a change in this information.

Figure 3.8.3-14 (Sheet 4 of 5)
[*CA-04 Structural Module*]*

*NRC Staff approval is required prior to implementing a change in this information.

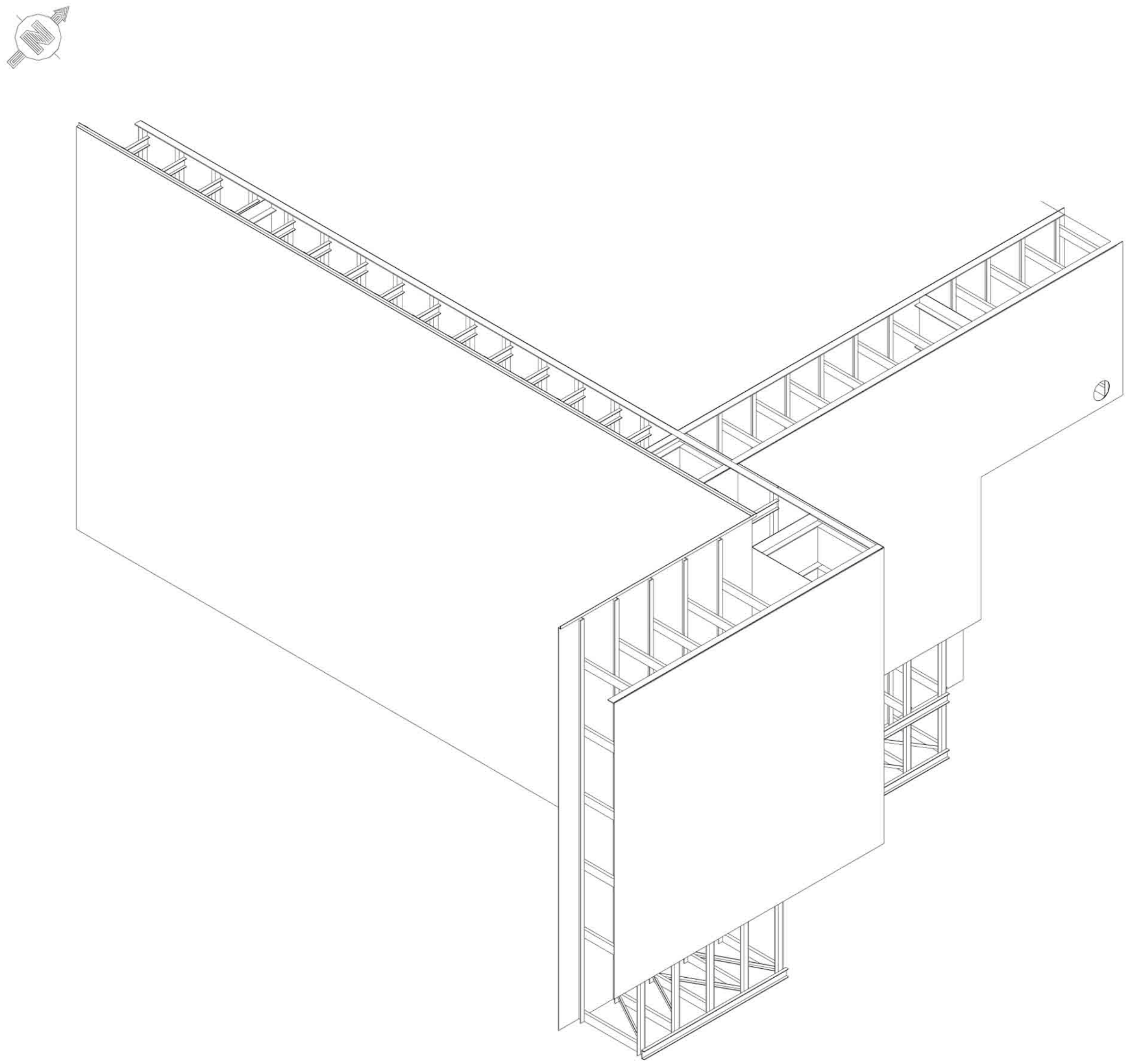
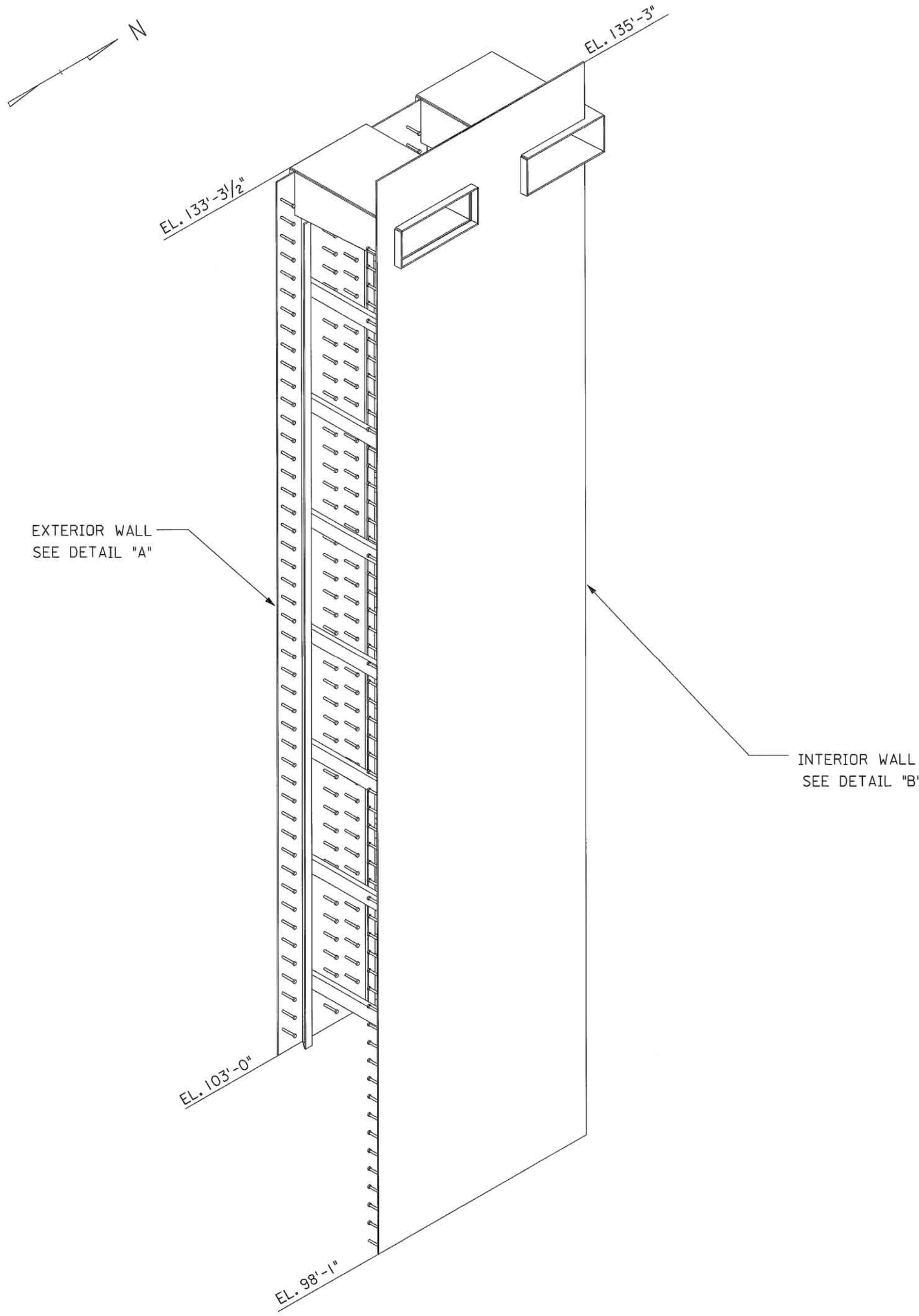


Figure 3.8.3-14 (Sheet 5 of 5)
[*CA-05 Module*]*

*NRC Staff approval is required prior to implementing a change in this information.



Note: See Figure 3.8.3-8 for fabrication detail.

Figure 3.8.3-15 (Sheet 1 of 2)
[*Typical Submodule*]*

*NRC Staff approval is required prior to implementing a change in this information.

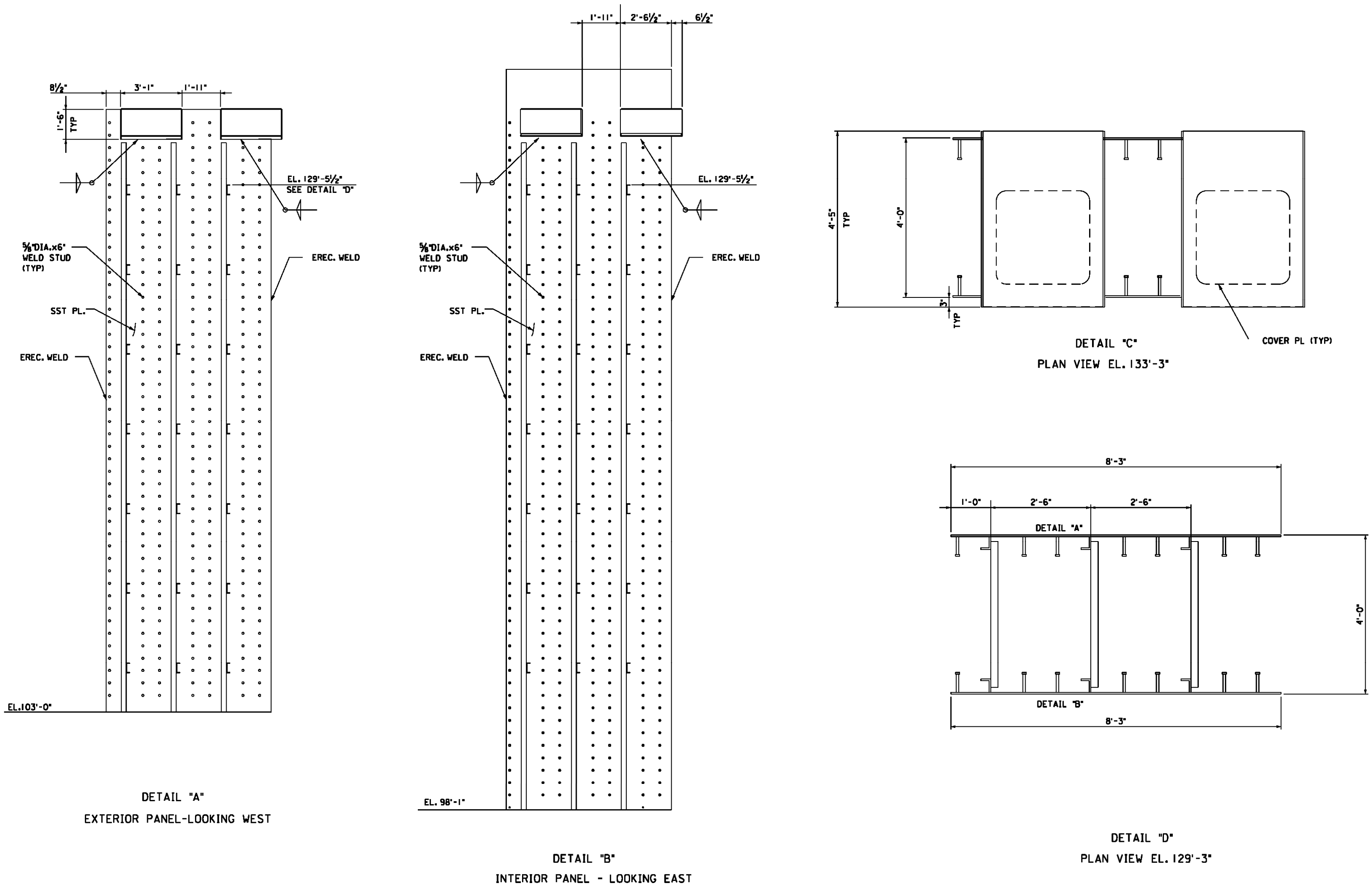


Figure 3.8.3-15 (Sheet 2 of 2)
[*Typical Submodule*]*

*NRC Staff approval is required prior to implementing a change in this information.

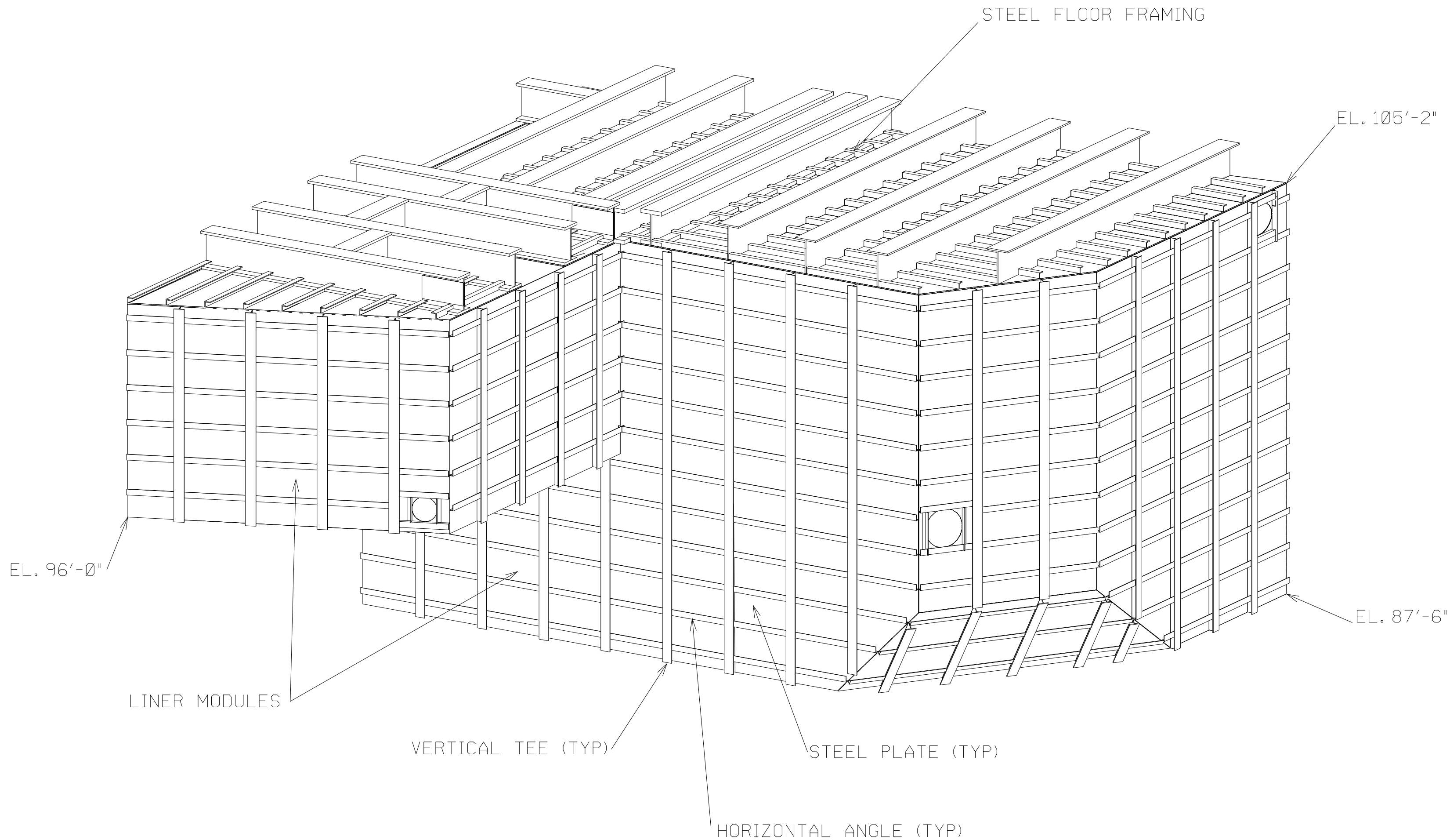
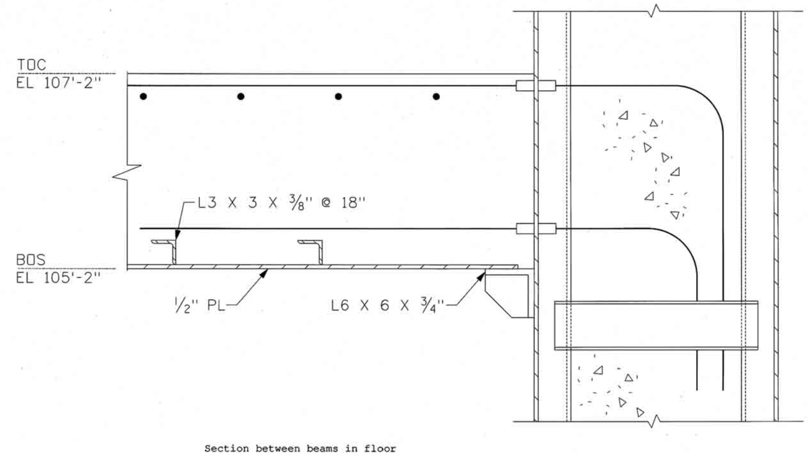
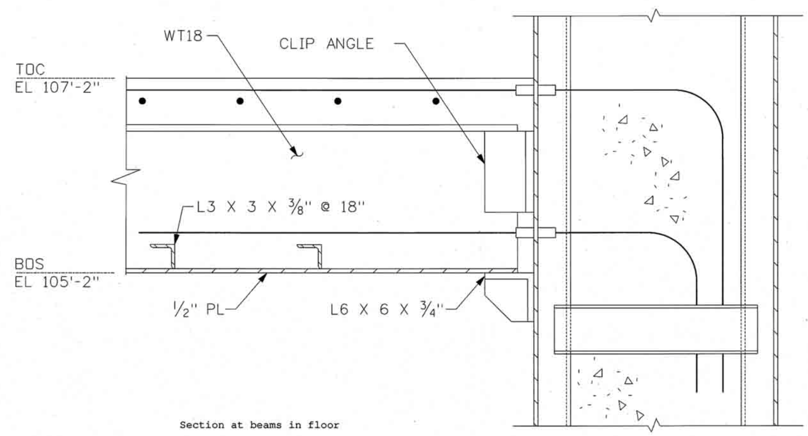


Figure 3.8.3-16
Liner Modules

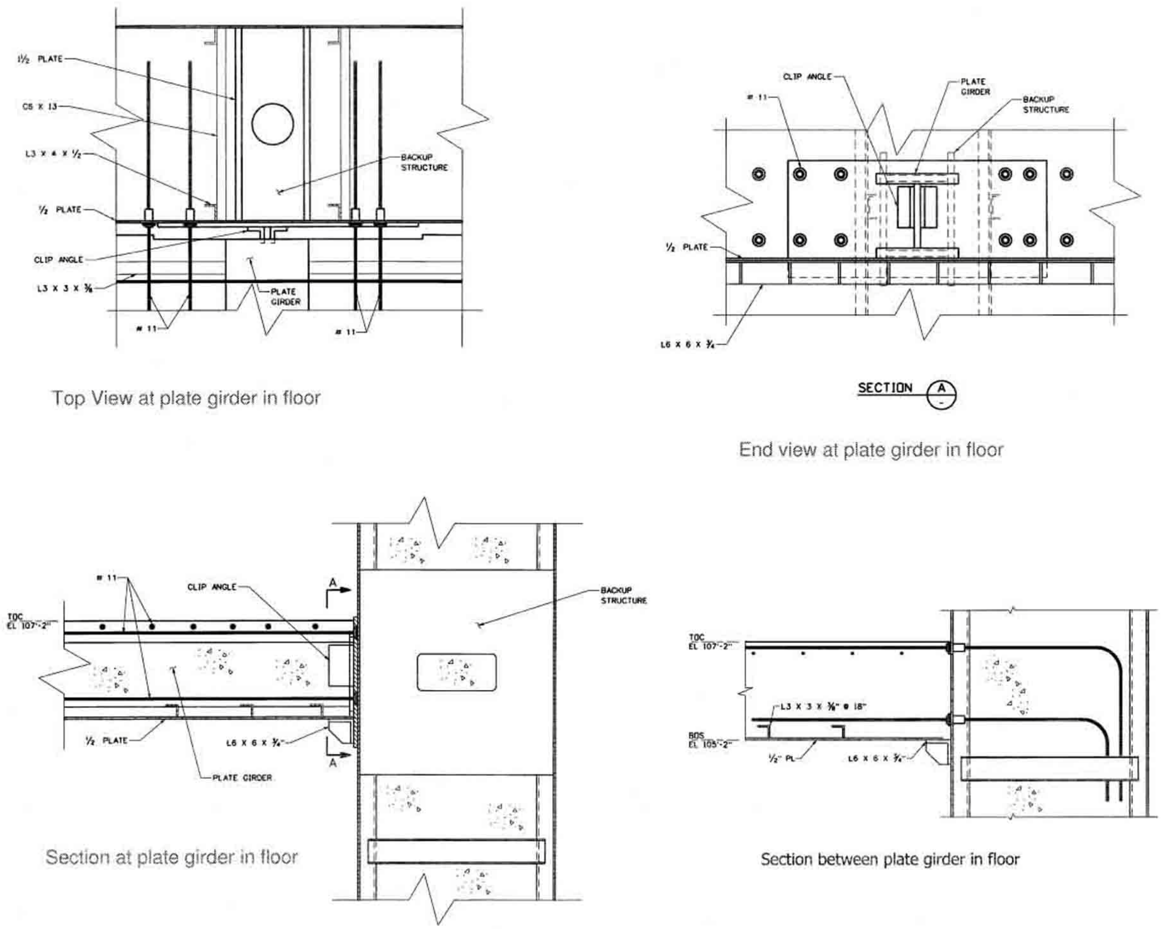




See subsection 3.8.3.5.8.1 for information on Tier 2* designation.

Figure 3.8.3-17 (Sheet 1 of 2)
[*Structural Modules – Design Details Standard Floor Connection*]*

*NRC Staff approval is required prior to implementing a change in this information.



See subsection 3.8.3.5.8.1 for information on Tier 2* designation.

Figure 3.8.3-17 (Sheet 2 of 2)
[*Structural Modules – Design Details Heavily Loaded Floor Connection*]*

*NRC Staff approval is required prior to implementing a change in this information.

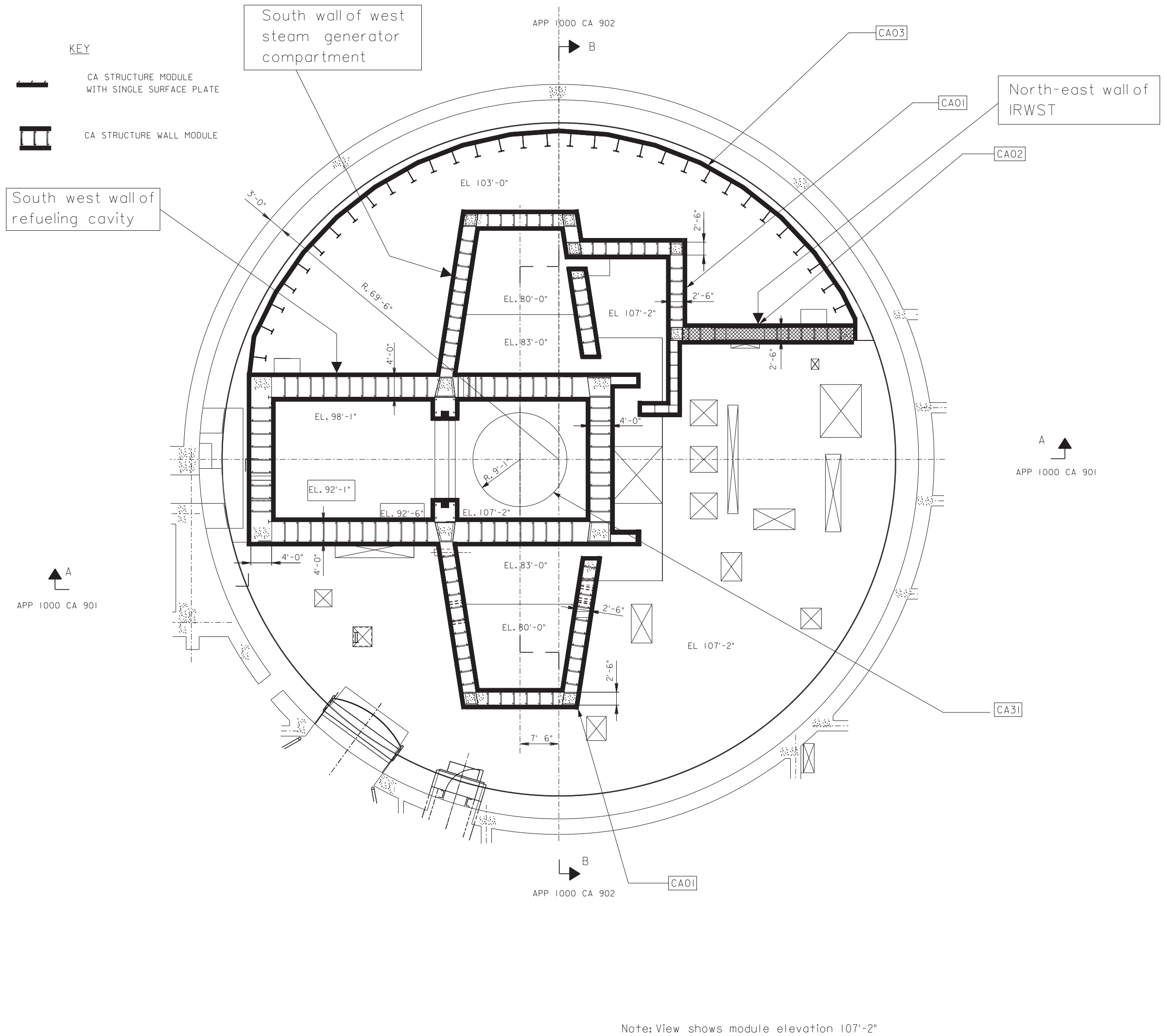
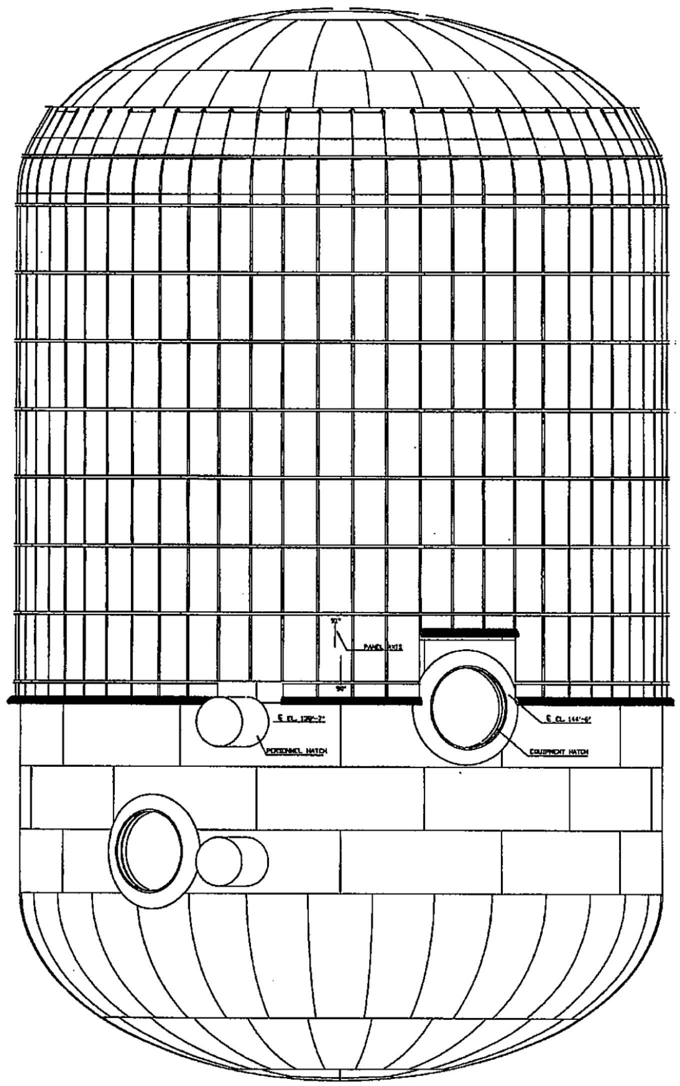


Figure 3.8.3-18
[*Location of Structural Wall Modules*]*

*NRC Staff approval is required prior to implementing a change in this information.



Note: Course layout, plate/insert plate geometry, and weld seams for the containment, and the number and configuration of the containment air baffle panels are shown for illustrative purposes only.

Figure 3.8.4-1 (Sheet 1 of 4)
Containment Air Baffle
General Arrangement

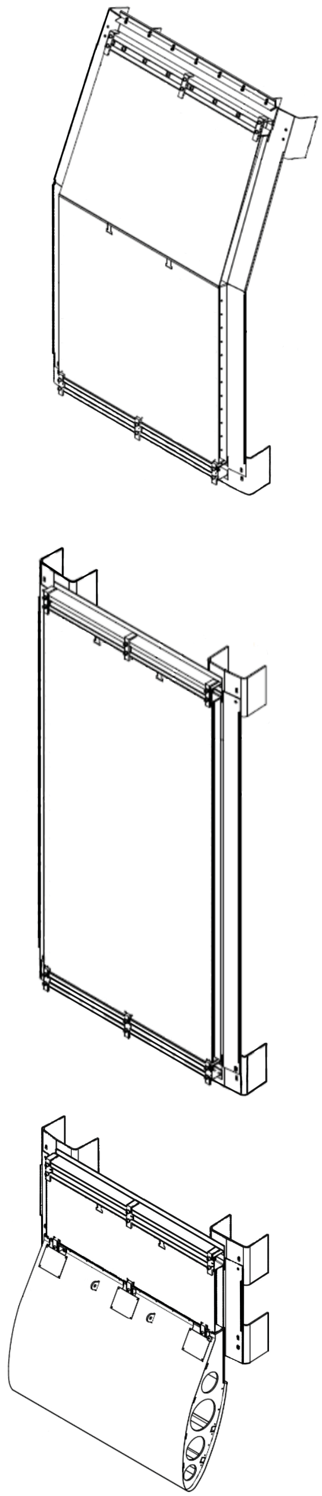


Figure 3.8.4-1 (Sheet 2 of 4)
Containment Air Baffle
Panel Types

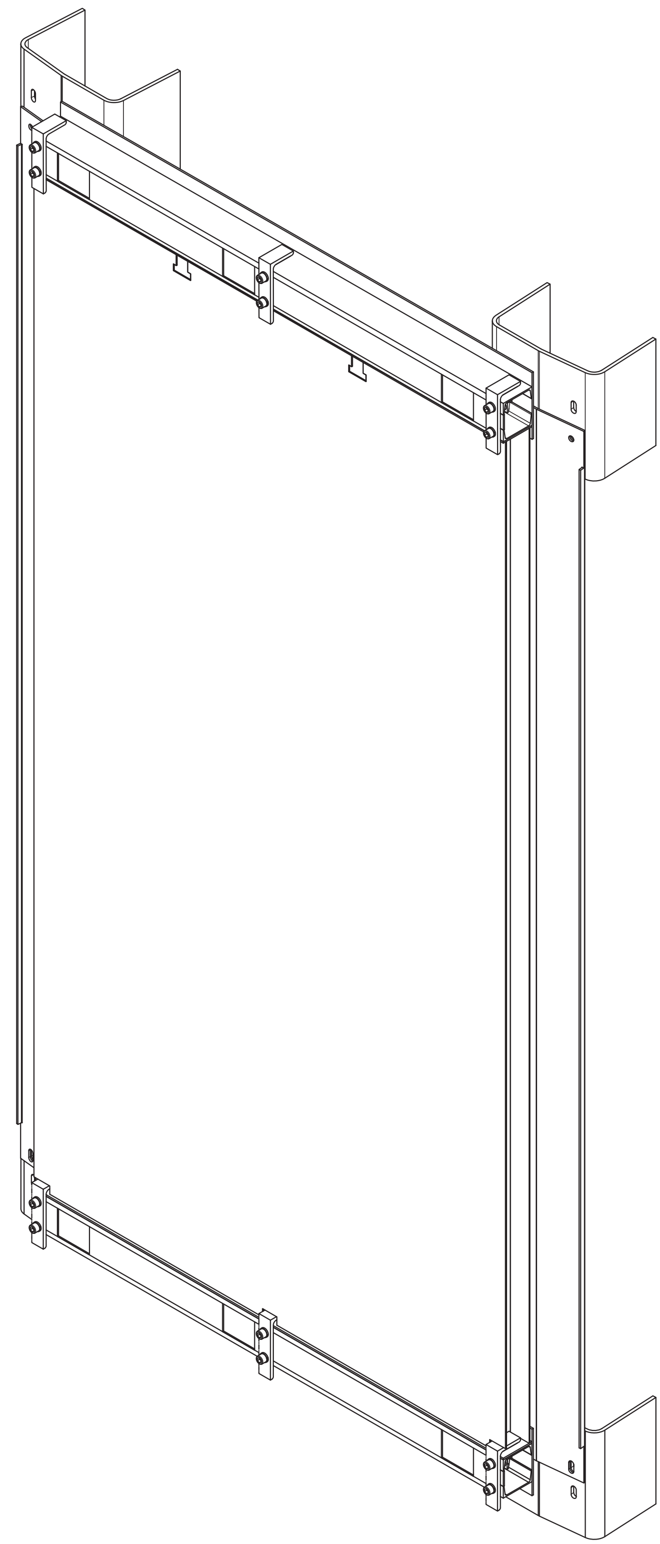


Figure 3.8.4-1 (Sheet 3 of 4)
Containment Air Baffle
Typical Panel on Cylinder

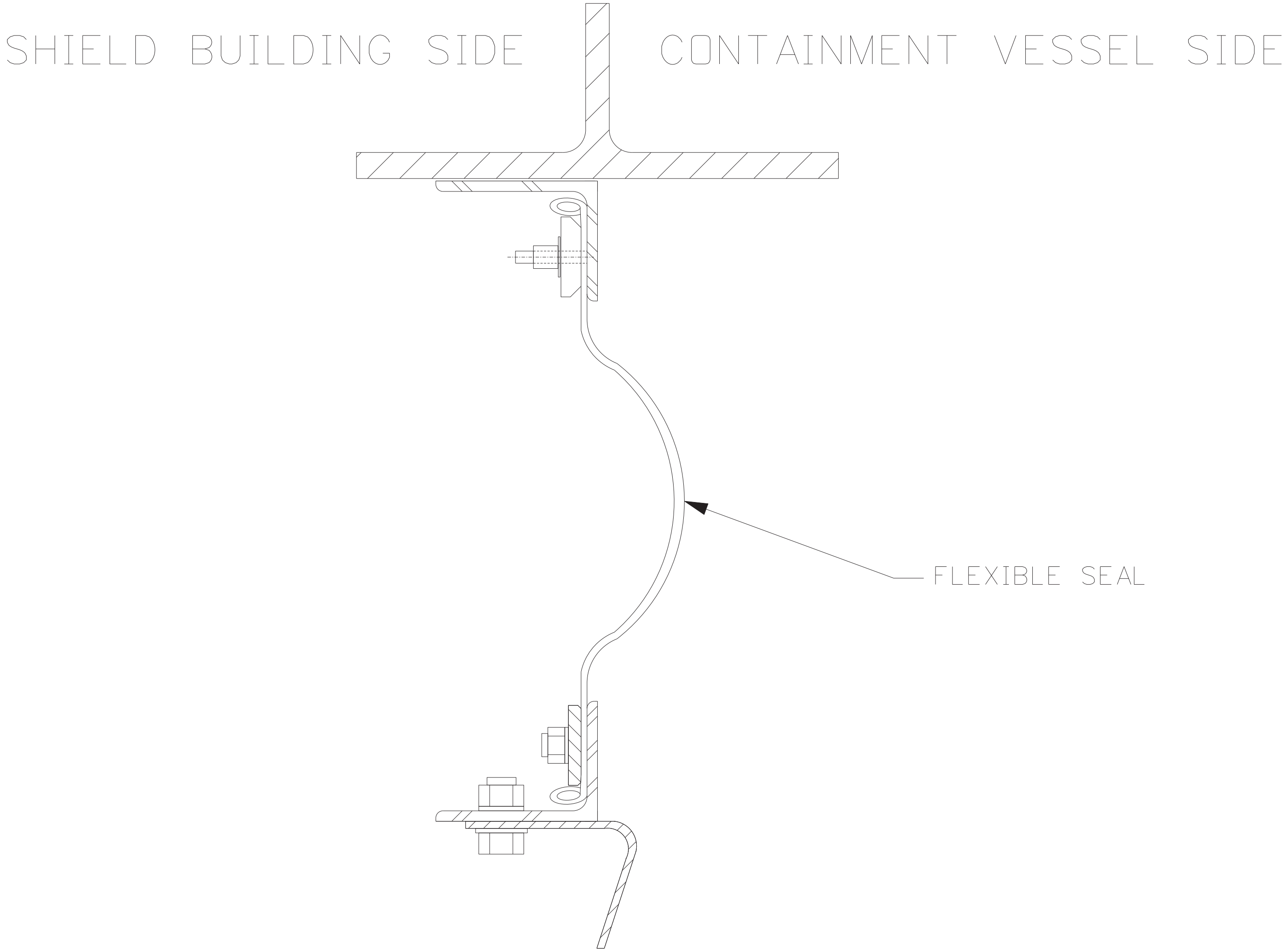


Figure 3.8.4-1 (Sheet 4 of 4)
Containment Air Baffle
Flexible Seal

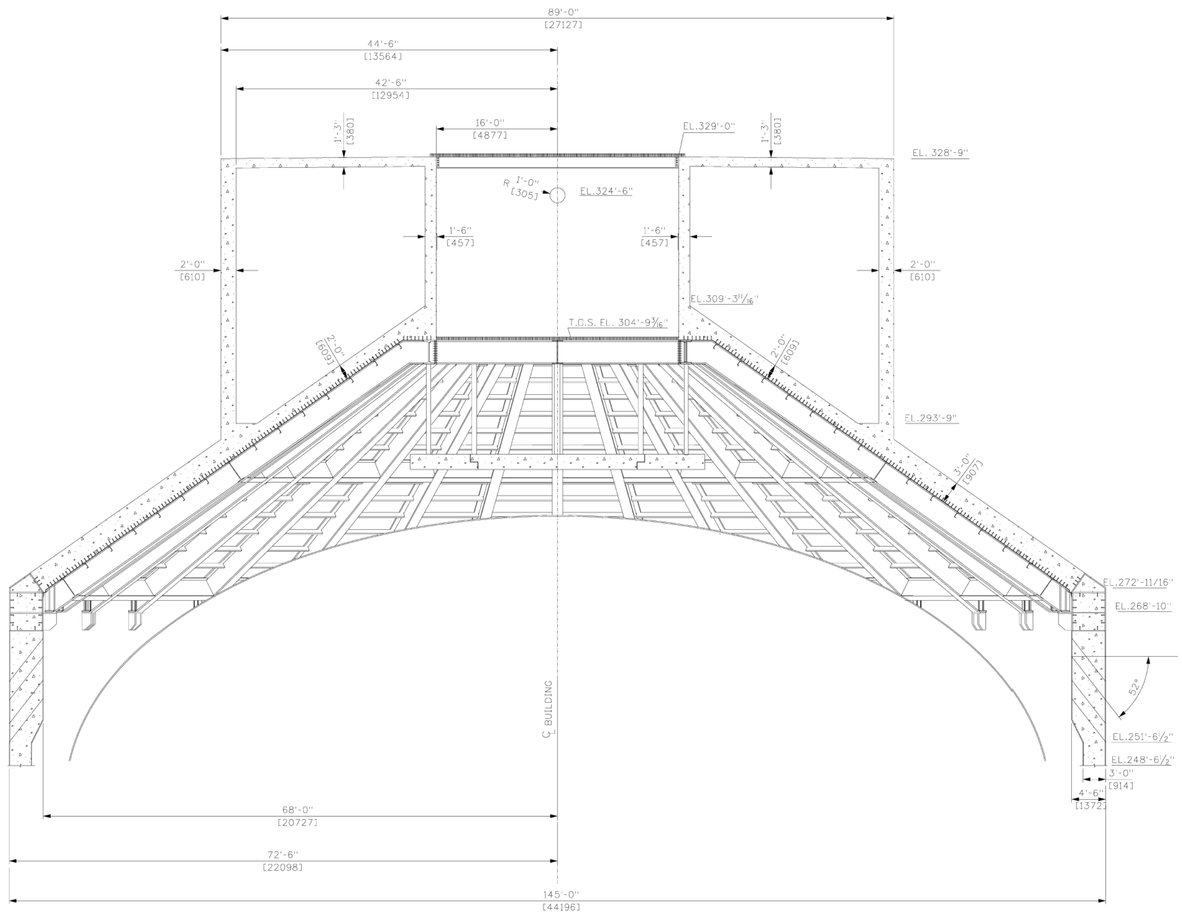


Figure 3.8.4-2
[*Passive Containment Cooling Tank*]*

*NRC Staff approval is required prior to implementing a change in this information.

Figure 3.8.4-3 Not Used

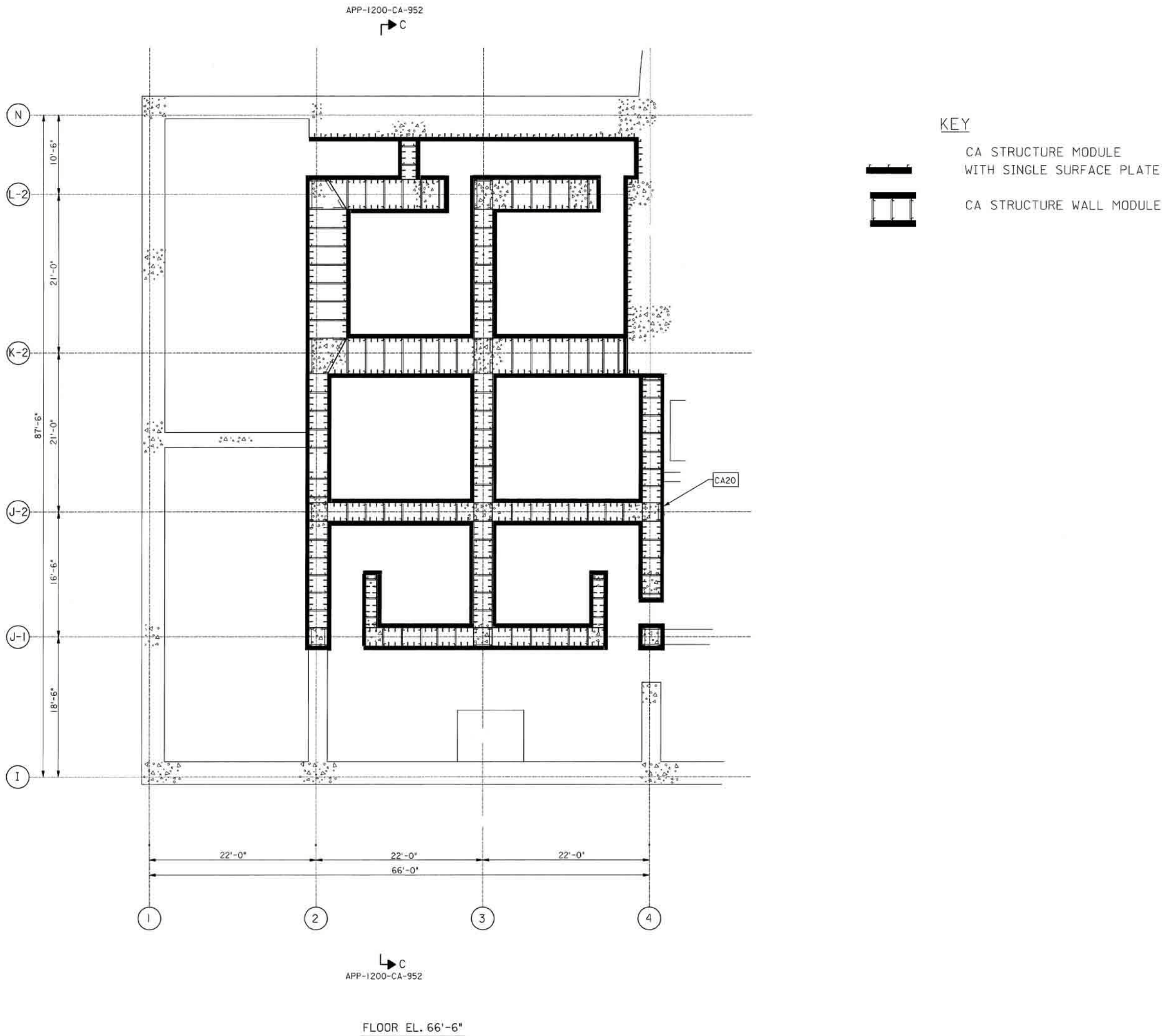


Figure 3.8.4-4 (Sheet 1 of 5)
[*Structural Modules in Auxiliary Building*]*

*NRC Staff approval is required prior to implementing a change in this information.

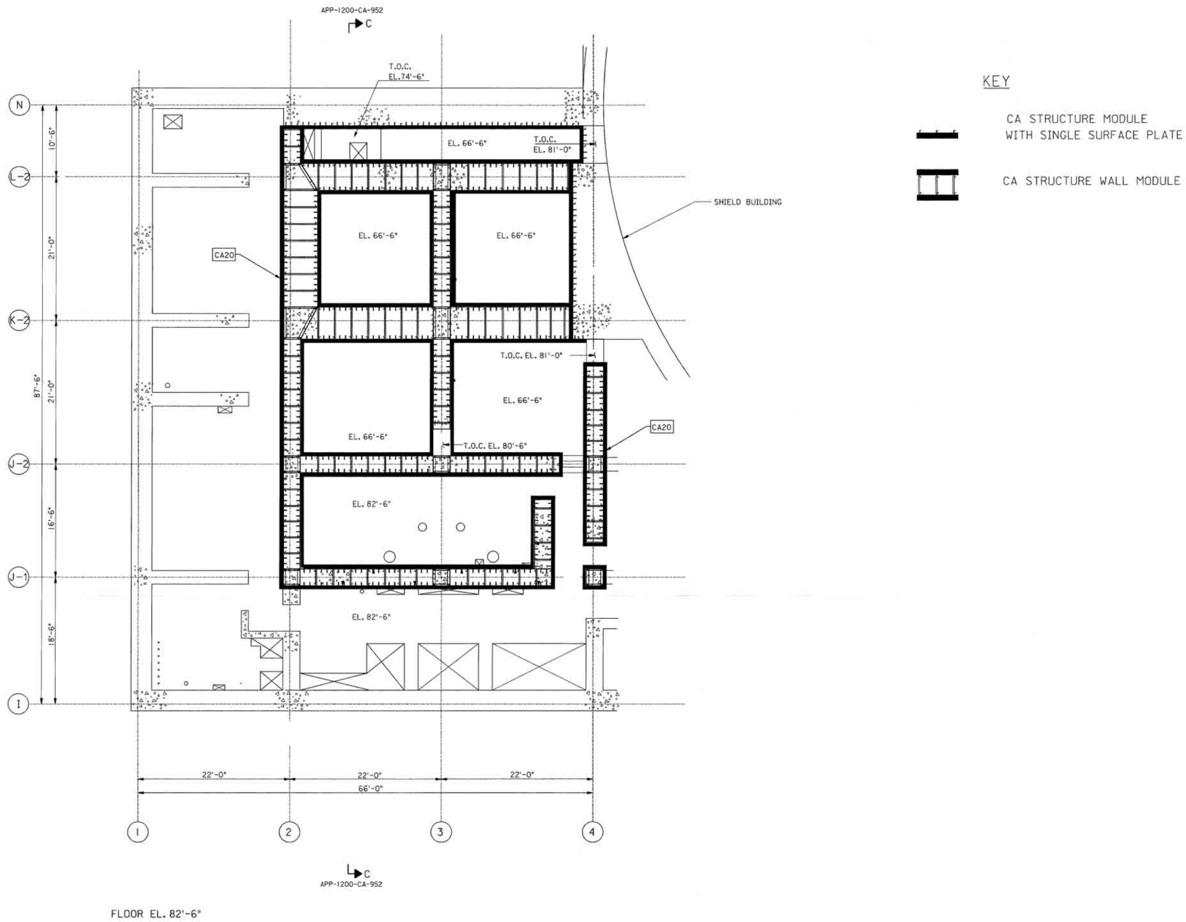


Figure 3.8.4-4 (Sheet 2 of 5)
[*Structural Modules in Auxiliary Building*]*

*NRC Staff approval is required prior to implementing a change in this information.

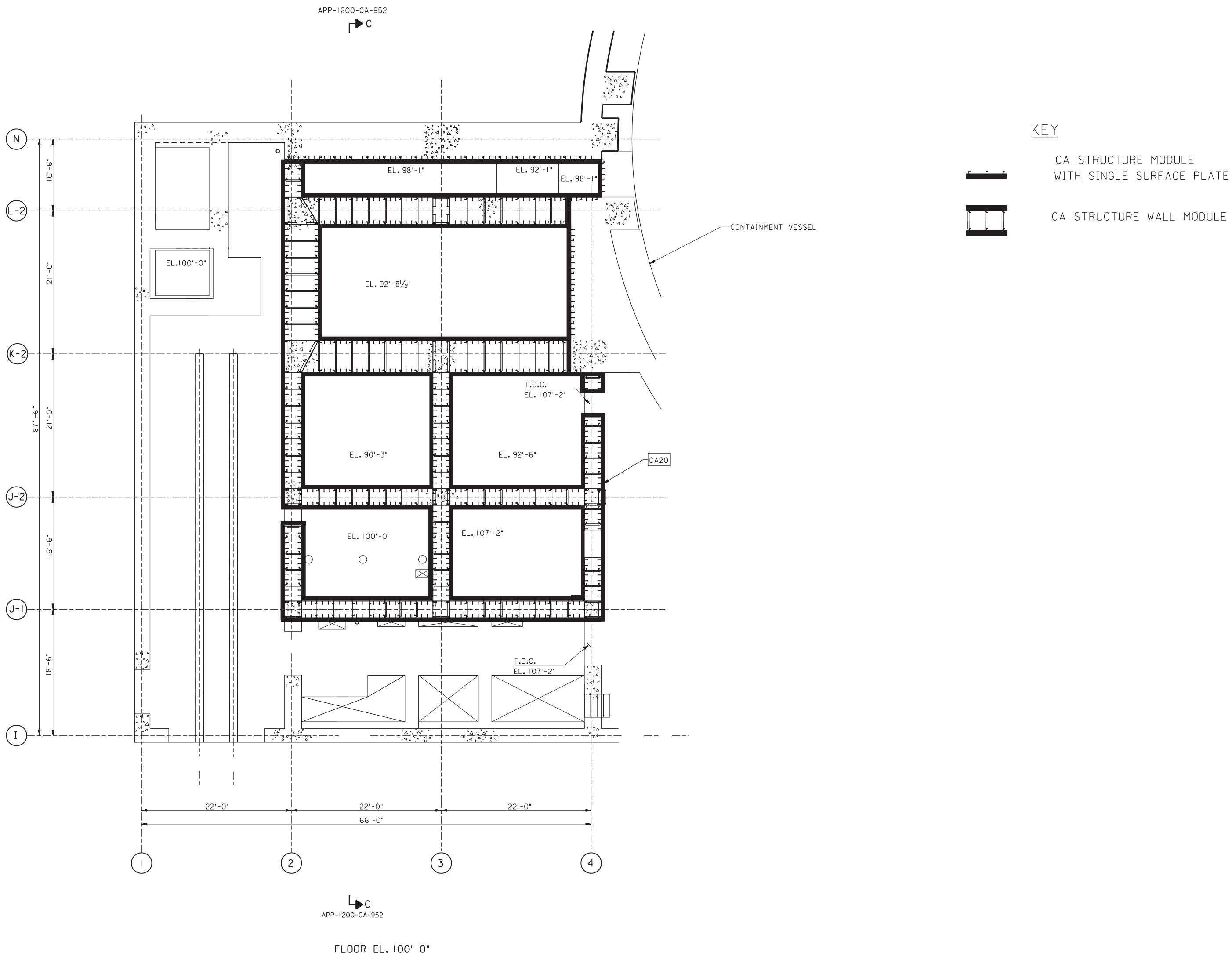


Figure 3.8.4-4 (Sheet 3 of 5)
[*Structural Modules in Auxiliary Building*]*

*NRC Staff approval is required prior to implementing a change in this information.

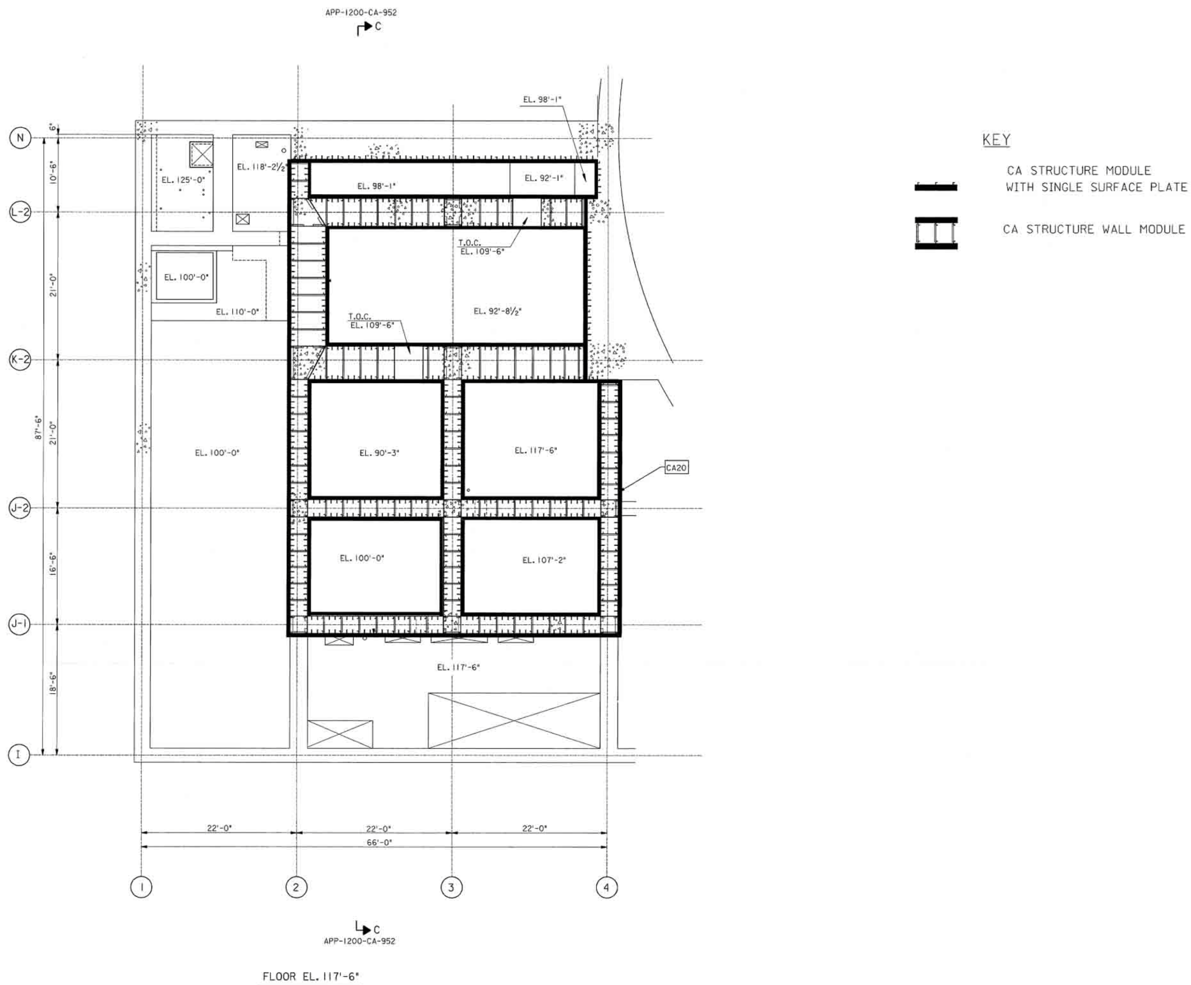


Figure 3.8.4-4 (Sheet 4 of 5)
[*Structural Modules in Auxiliary Building*]*

*NRC Staff approval is required prior to implementing a change in this information.

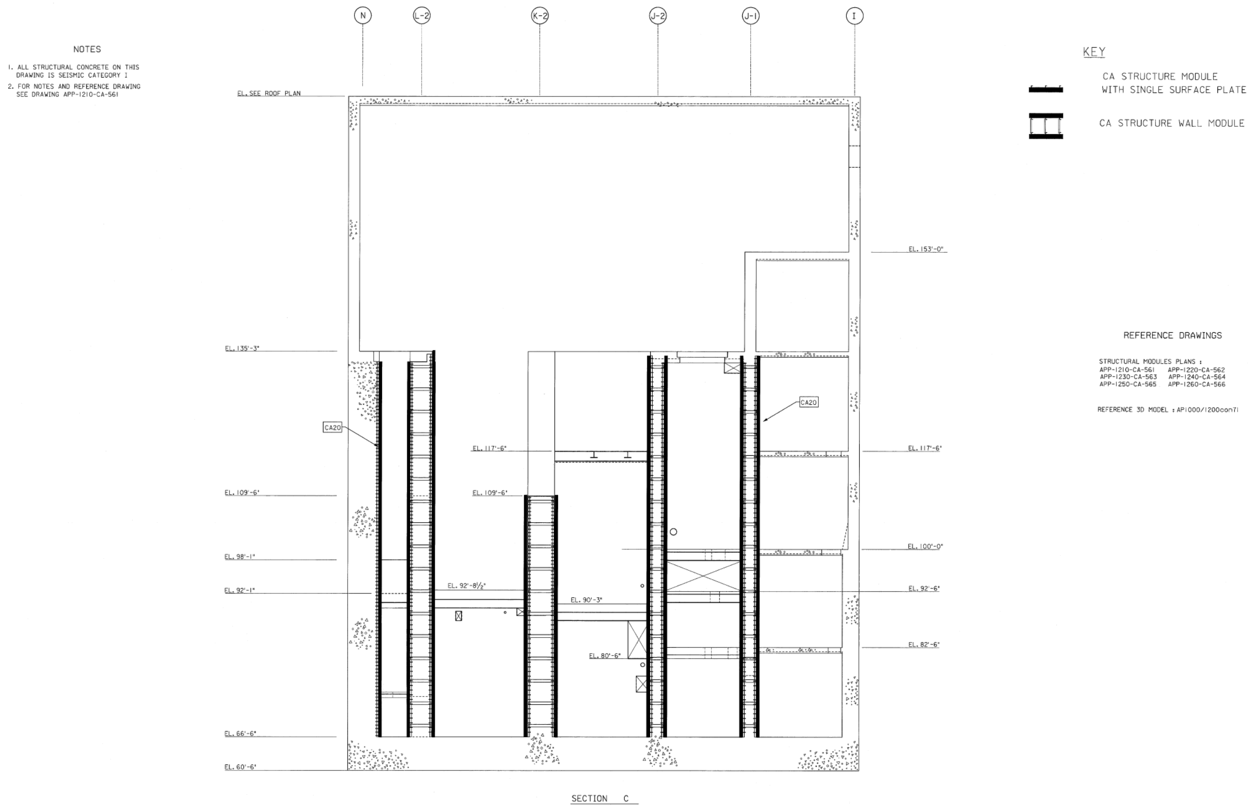


Figure 3.8.4-4 (Sheet 5 of 5)
[*Structural Modules in Auxiliary Building*]*

*NRC Staff approval is required prior to implementing a change in this information.

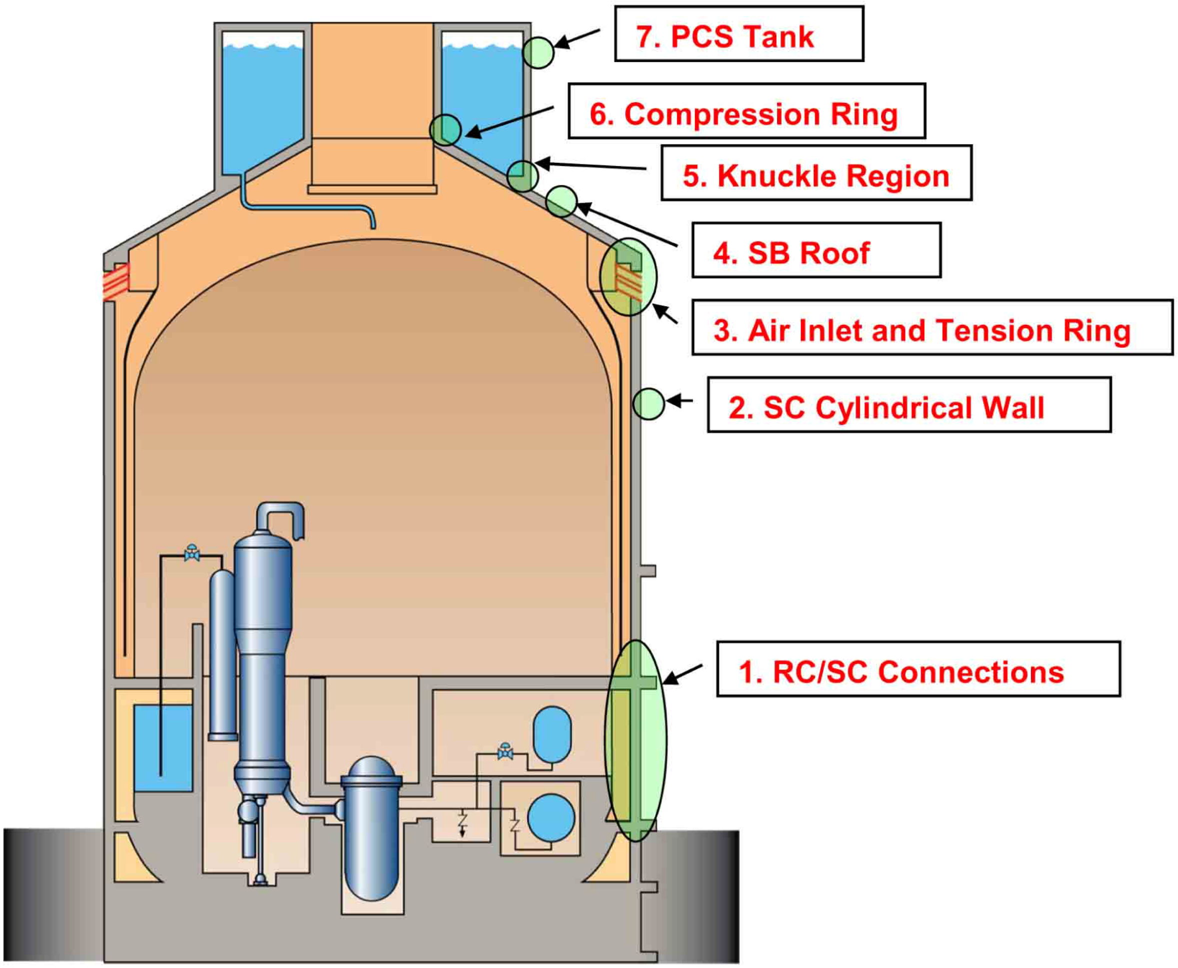


Figure 3.8.4-5
Shield Building Structure Key Areas

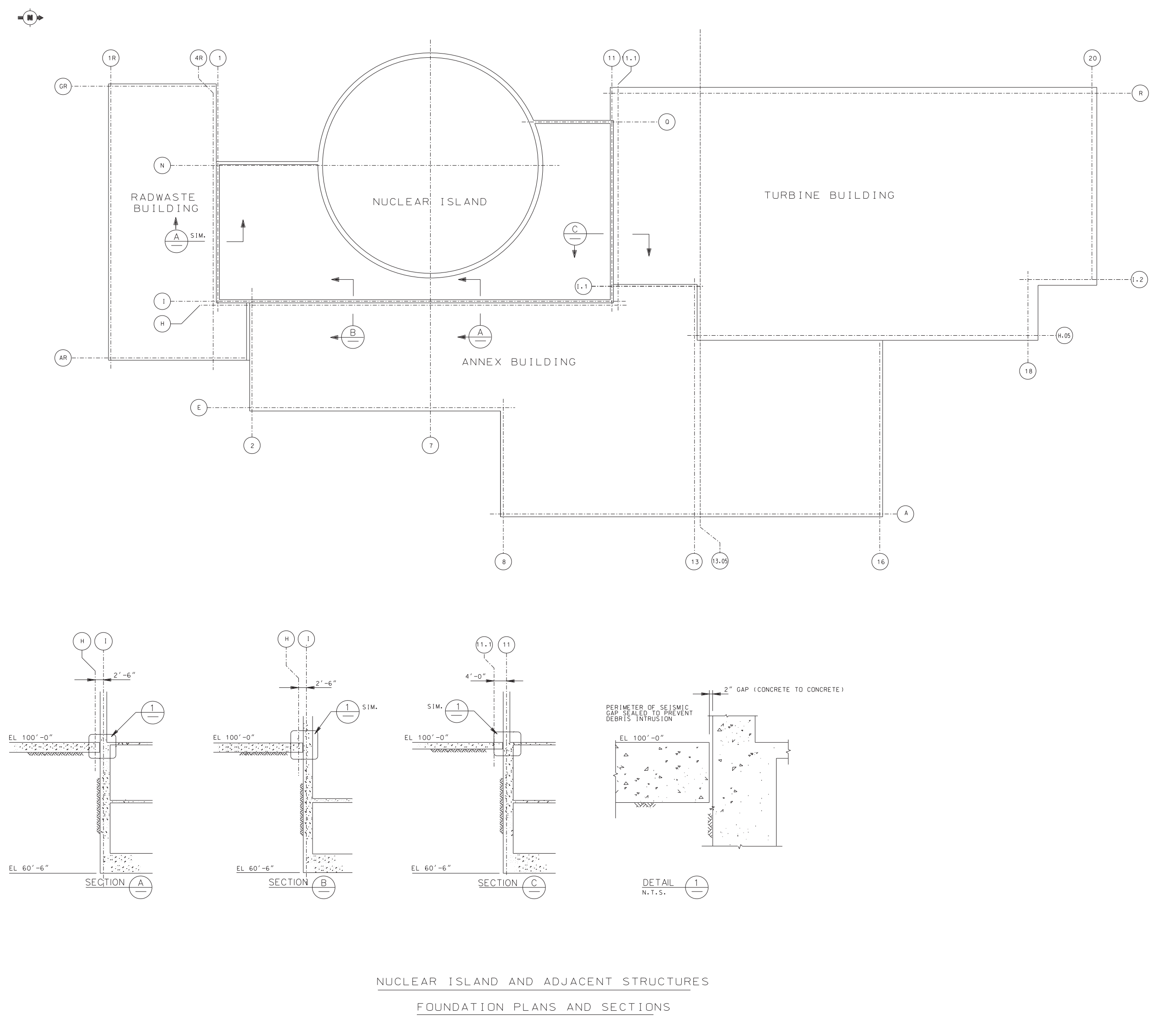


Figure 3.8.5-1
Foundation Plan

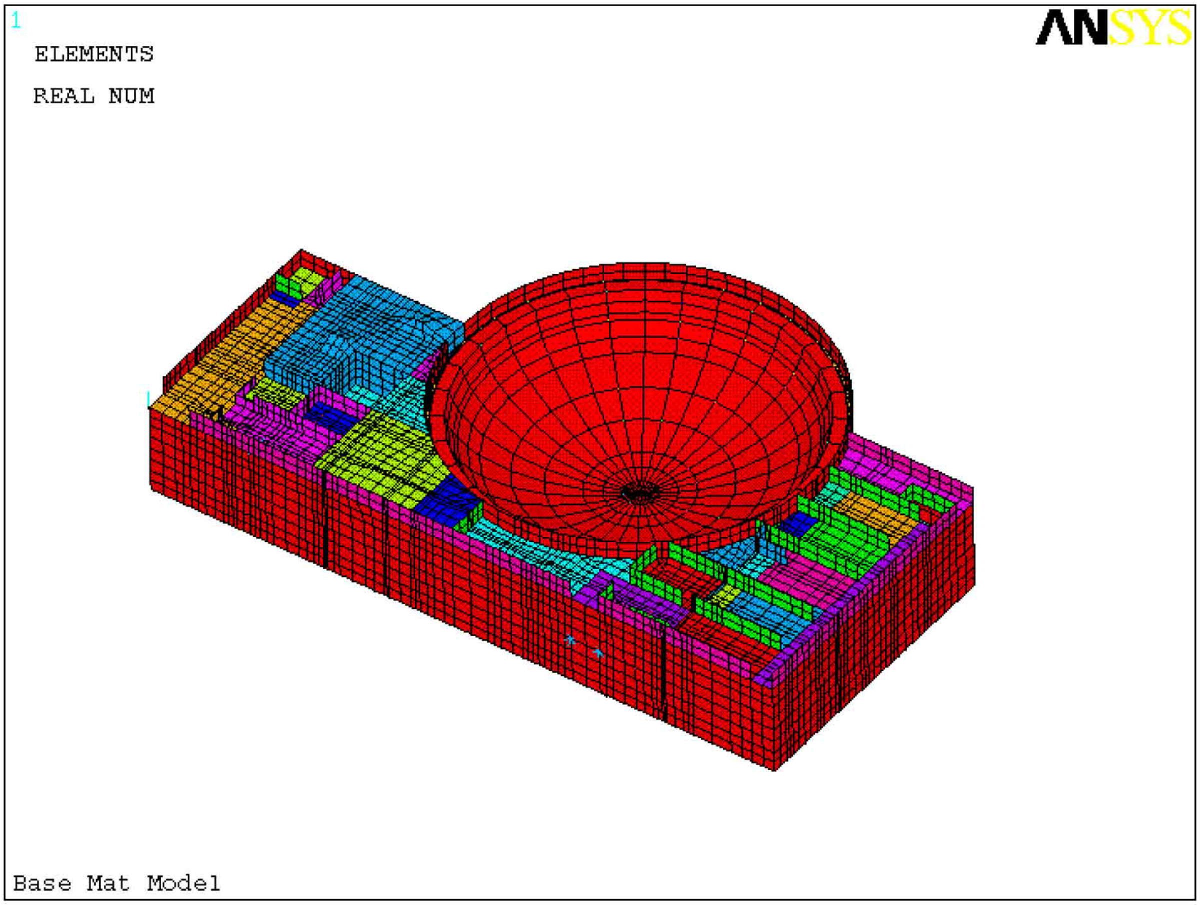


Figure 3.8.5-2
Isometric View of Finite Element Model

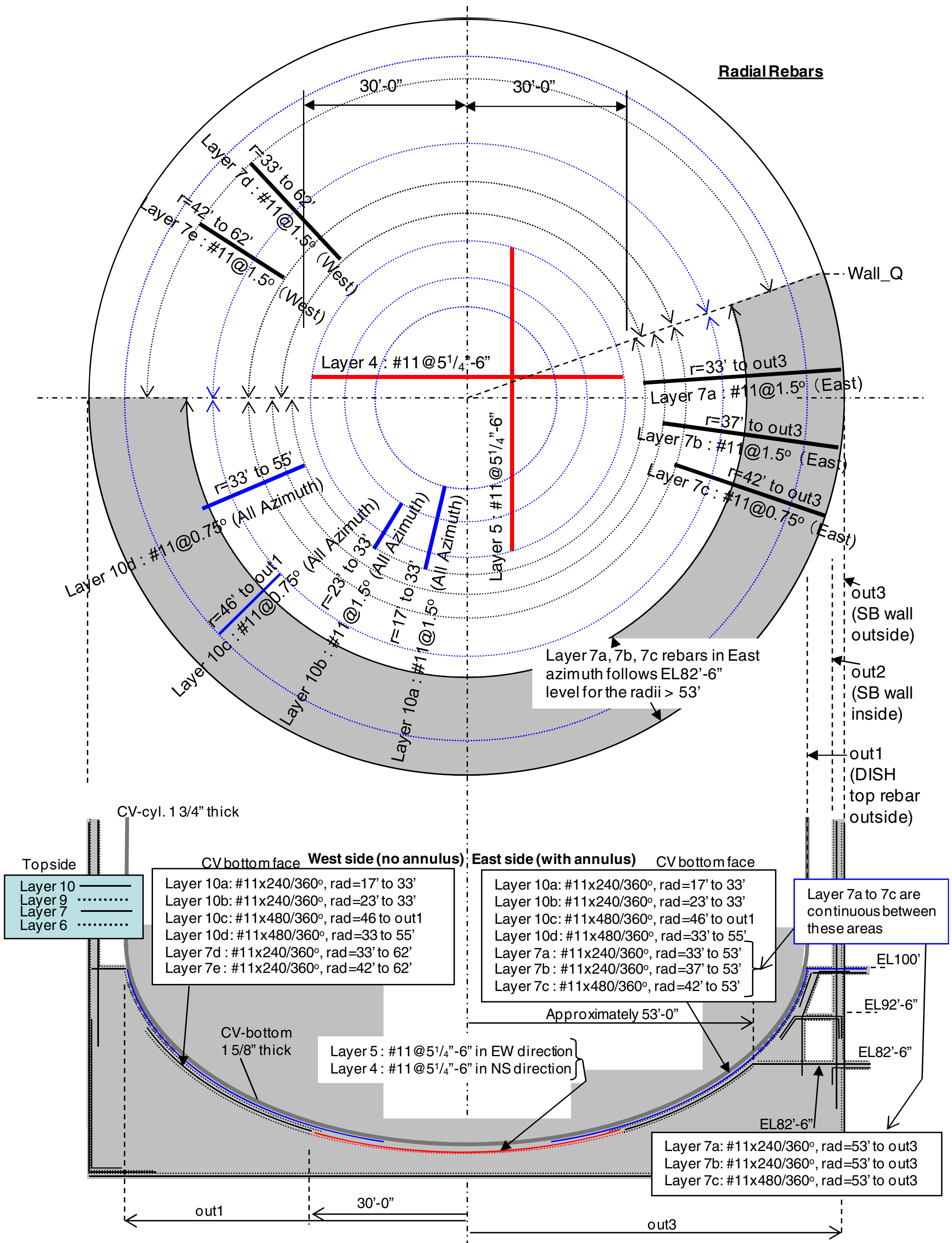


Figure 3.8.5-3 (Sheet 1 of 7)
Radial Reinforcement, Top Side of DISH

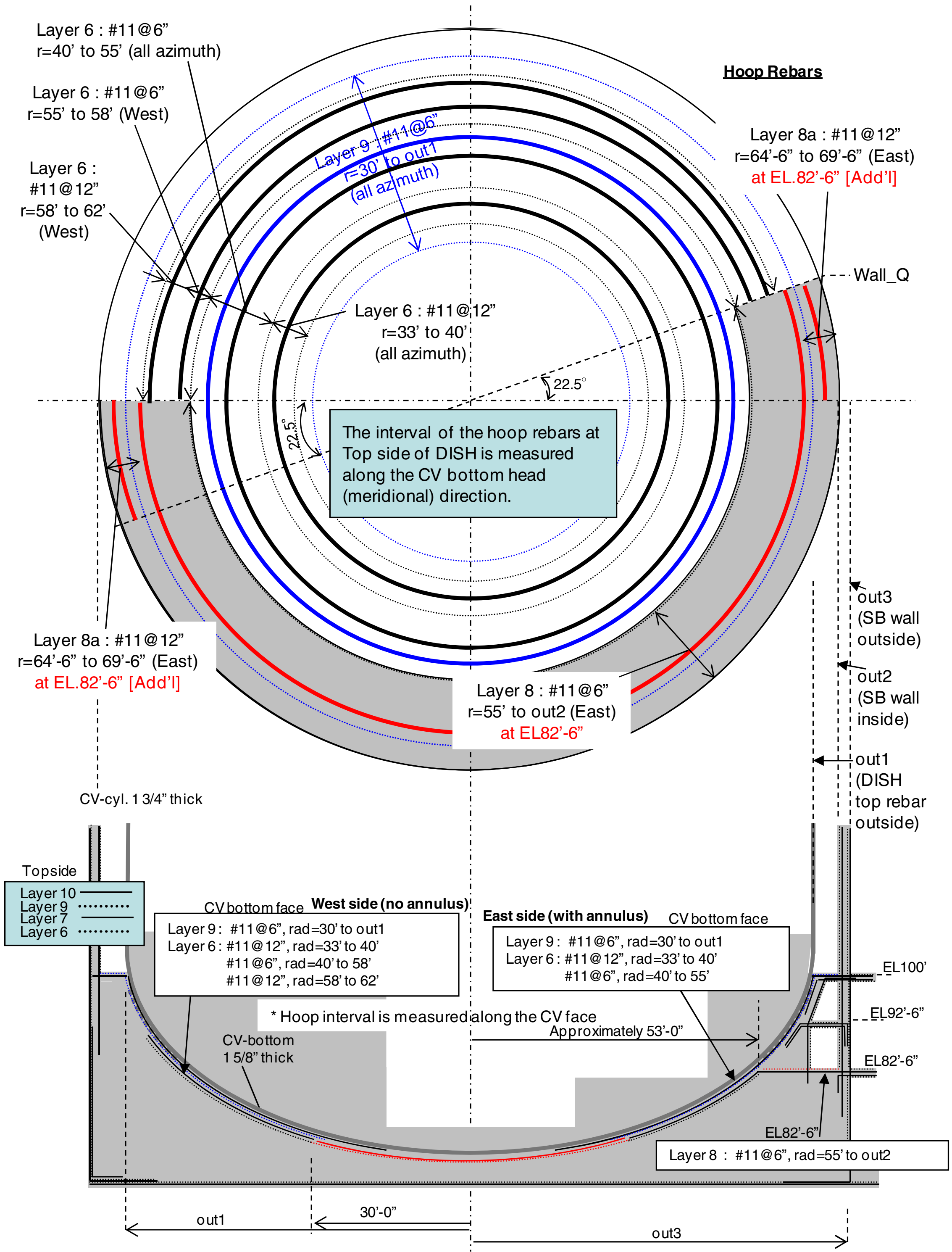
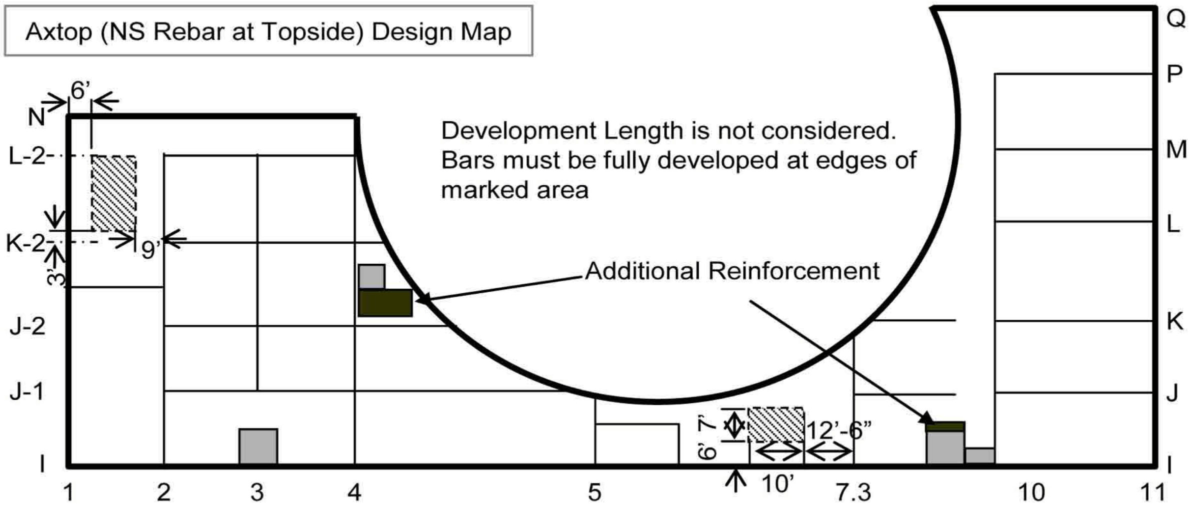
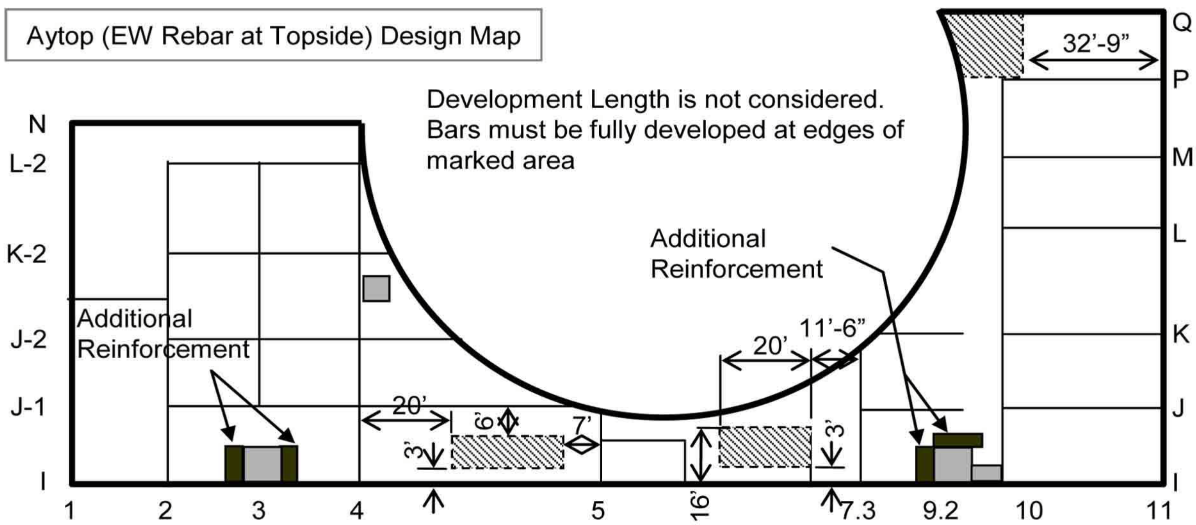


Figure 3.8.5-3 (Sheet 2 of 7)
Circumferential Reinforcement, Top Side of DISH



Rebar arrangement	: #14@12"	: #14@12"+#9@12"	: #11@12"
Location	Layer 4		Top side of the 2' & 3' thick Pits

Figure 3.8.5-3 (Sheet 3 of 7)
Longitudinal Reinforcement Map,
Top Side in NS Direction



Rebar arrangement	: #14@12" : #14@12"+#9@12"	: #11@12"
Location	Layer 5	Top side of the 2' & 3' thick Pits

Figure 3.8.5-3 (Sheet 4 of 7)
Longitudinal Reinforcement Map,
Top Side in EW Direction

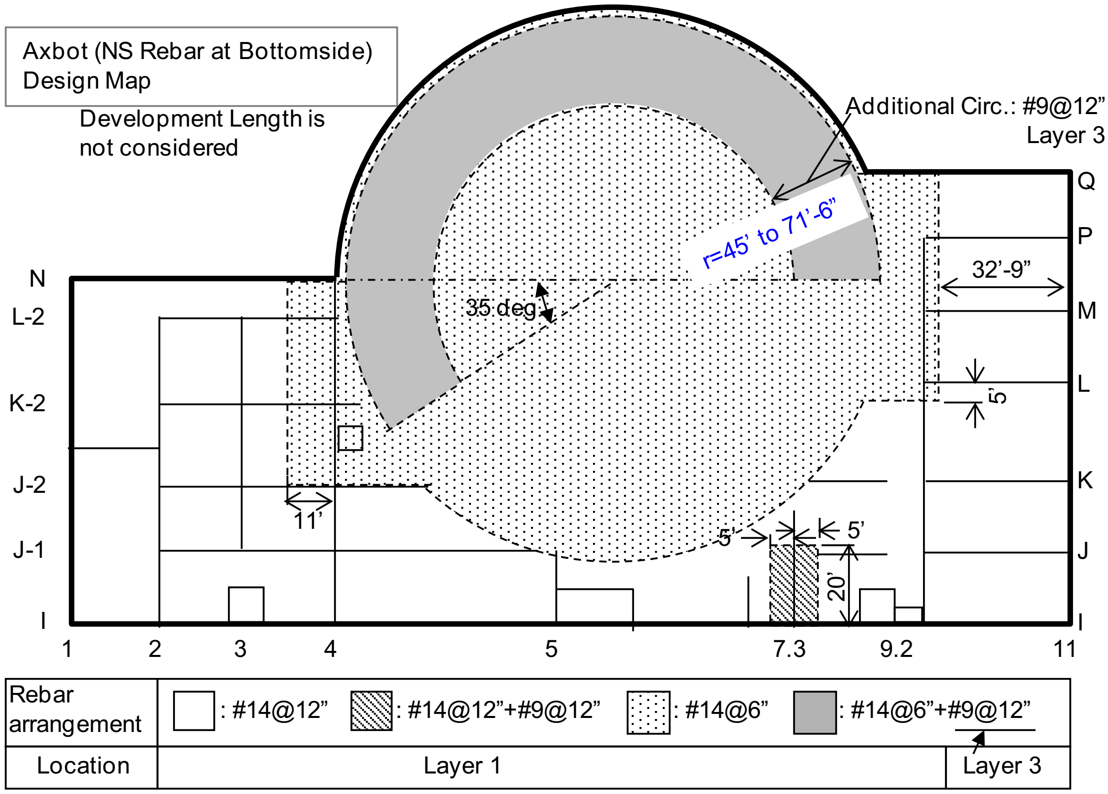


Figure 3.8.5-3 (Sheet 5 of 7)
Longitudinal Reinforcement,
Bottom Side of DISH and 6′ Basemat (NS)

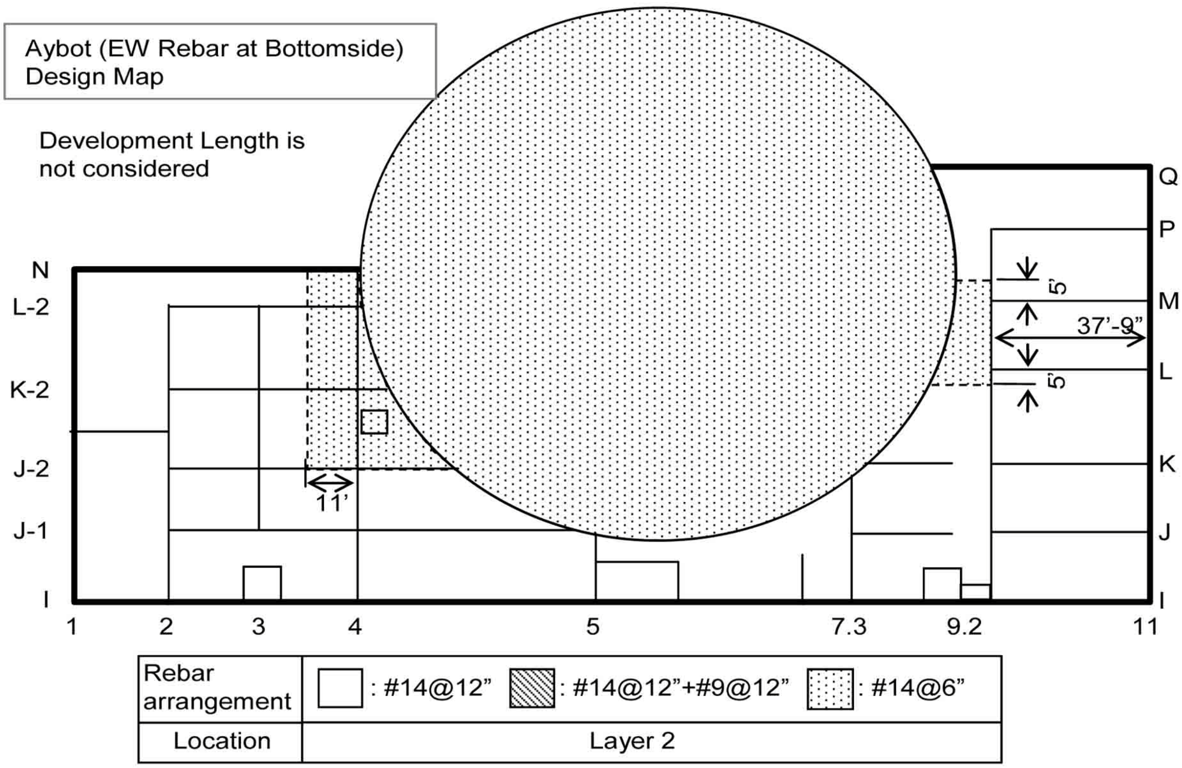


Figure 3.8.5-3 (Sheet 6 of 7)
Longitudinal Reinforcement,
Bottom Side of DISH and 6′ Basemat (EW)

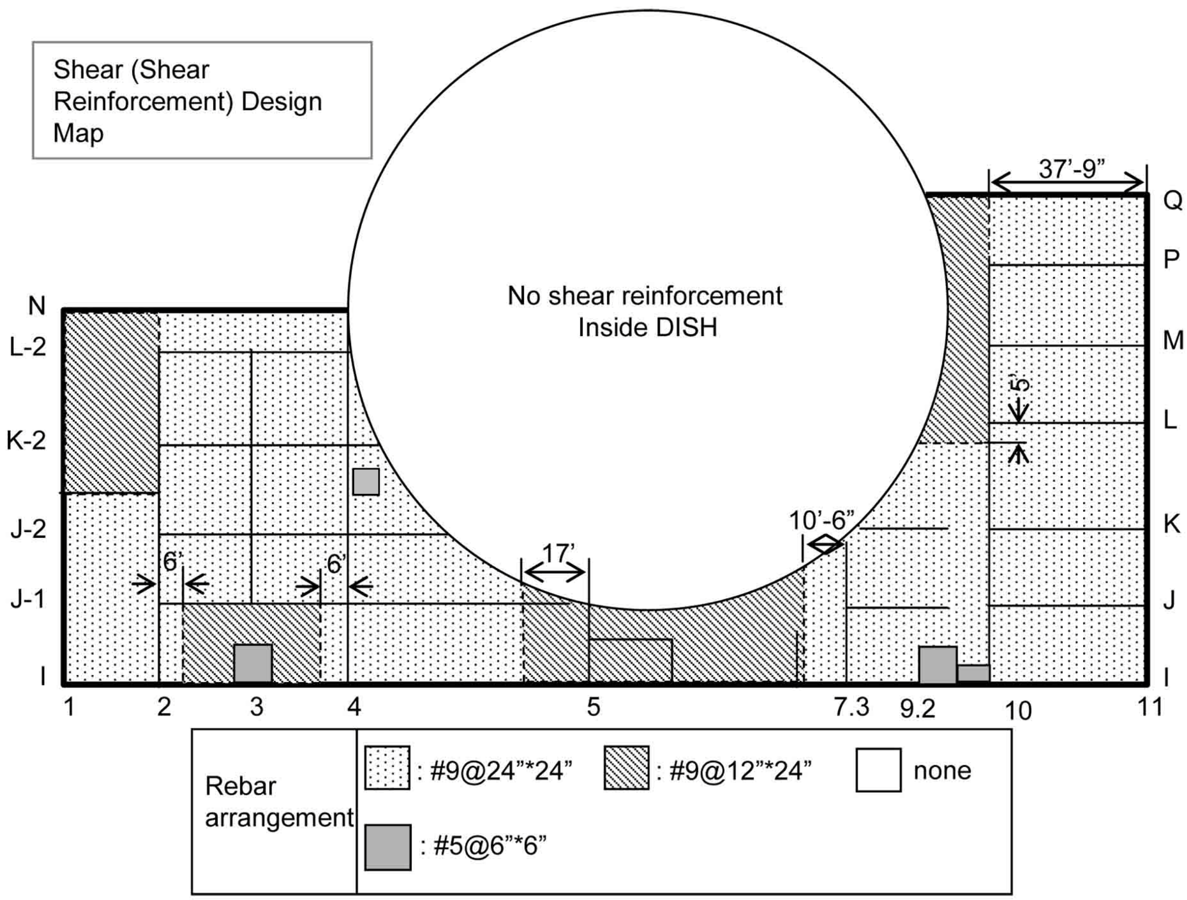


Figure 3.8.5-3 (Sheet 7 of 7)
Shear Reinforcement Map