SOUTHERN COMPANY®

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UFSAR Formatting Legend

Color	Description
	Original Westinghouse AP1000 DCD Revision 19 content (part of plant-specific DCD)
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	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 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. Due to their inherent electrical isolation characteristics, fiber optic cables are exempt from electrical isolation qualification testing. 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 3.2.2.5.)

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. The turbine building first bay building structure, including Wall 11.2, is a seismic Category II structure as identified in Table 3.2-2. The turbine building first bay building structure provides tornado missile protection for openings in Wall 11 as described in Table 3.5-1. Wall 11.2 also protects Wall 11 from the dynamic effects of pipe failure events in the nonseismic portion of the turbine building.

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. As identified in Subsection 3.7.2, seismic Category II building structures are designed for the safe shutdown earthquake using the same methods and design stress limits as are used for seismic Category I structures. Seismic Category II building structures are also designed to withstand the design basis tornado loads, including missiles, in accordance with the loading combinations identified in Table 3.8.4-2.

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. These quality requirements are applicable to the seismic Category II turbine building first bay building structure.

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 to protect Class A, B, and C structures, systems, and components from events such as internal/external missiles, seismic, flooding, and the dynamic and environmental effects of pipe failures. Structures protecting equipment from nonseismic events are not required to be seismic Category I. An exception to this equipment Class C criterion is the equipment Class D barriers described in Table 3.5-1 that provide tornado missile protection. The seismic Category II turbine building first bay building structure also provides tornado missile protection for openings in Wall 11 as described in Table 3.5-1, and turbine building first bay building structure Wall 11.2 provides protection of Wall 11 from HELB loads resulting from postulated ruptures in main steam and main feedwater piping north of the turbine building first bay.

- 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. The tornado missile barriers in the turbine building first bay building structure identified in Table 3.5-1 are seismic Category II, equipment Class D.

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 Edition, “Power Piping,” including 1989 Addenda.

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(14)	3	1E	GROUP C	YES	YES(7)	(8)
D	NNS(2)	NA(9)(14)	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.
14. The seismic Category II tornado missile barriers located on or within the turbine building first bay and identified in Subsections 3.2.2.5 and 3.2.2.6, and in Table 3.5-1 provide tornado missile protection for openings in auxiliary building Wall 11.

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 56)
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	
BDS-PL-V009	Discharge to WLS Vent	D	NS	ANSI B16.34	
BDS-PL-V037	Discharge to WLS Valve	D	NS	ANSI B16.34	
BDS-PL-V046	WWS Contaminated Flow Connection	D	NS	ANSI B16.34	
Balance of 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	

Table 3.2-3 (Sheet 2 of 56)
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-V215	CCS Supply Containment Isolation Valve Test Connection - IRC	C	I	ASME III-3	
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-V220	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.	

Table 3.2-3 (Sheet 3 of 56)
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)					
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	
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 Thermal 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	

Table 3.2-3 (Sheet 4 of 56)
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-V096	Zinc Injection Containment Isolation Test Connection	B	I	ASME III-2	
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	Non-Stamped ASME VIII	Located in Annex Building

Table 3.2-3 (Sheet 5 of 56)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Standby Diesel Fuel Oil System (Continued)					
Balance of system components are Class E					
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-V241	DWS Containment Penetration Thermal Relief Valve	C	I	ASME III-3	
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

Table 3.2-3 (Sheet 6 of 56)
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-01	Spent Fuel Pool	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MT-02	Fuel Transfer Canal	C	I	ACI 349	ACI 349 Evaluation of Structural Boundary Only
FHS-MT-03	Refueling Cavity	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 Quick Opening Transfer Tube Closure	B	I	ASME III Class MC	
Balance of system components are Class E					
Fire Protection System (FPS)				Location: Various	
FPS-PL-V049	Fire Water Containment Test Connection Isolation	B	I	ASME III-2	
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	
FPS-PL-V702	FPS Containment Penetration Thermal Relief Valve	C	I	ASME III-3	
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

Table 3.2-3 (Sheet 7 of 56)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
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					
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	In-core Instrument Thimble Assembly Tubes	D	II	Manufacturer Std.	
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	D	I	NUREG-0554 supplemented by ASME NOG-1	
MHS-MH-02	Cask Handling Crane	D	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					

Table 3.2-3 (Sheet 8 of 56)
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 (PCS)			Location: Containment Shield Building and Auxiliary Building		
PCS-JE-FE001	PCCWST Flow Through L001A Element	C	I	ASME III-3	
PCS-JE-FE002	PCCWST Flow Through L001B Element	C	I	ASME III-3	
PCS-JE-FE003	PCCWST Flow Through L001C Element	C	I	ASME III-3	
PCS-JE-FE004	PCCWST Flow Through L001D Element	C	I	ASME III-3	
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
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-MY-Y01	PCCWST Screen 1	C	I	Manufacturer Std.	
PCS-MY-Y02	PCCWST Screen 2	C	I	Manufacturer Std.	
PCS-MY-Y03	PCCWST Screen 3	C	I	Manufacturer Std.	
PCS-MY-Y04	PCCWST Screen 4	C	I	Manufacturer Std.	
PCS-MY-Y05	PCCWST Screen 5	C	I	Manufacturer Std.	
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	

Table 3.2-3 (Sheet 9 of 56)
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-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-3	
PCS-PL-V016	PCCWST Drain Isolation Valve	C	I	ASME III-3	
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-V026	Makeup to Distribution Bucket Isolation Valve	C	I	ASME III-3	
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	

Table 3.2-3 (Sheet 10 of 56)
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-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	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	

Table 3.2-3 (Sheet 11 of 56)
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-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	
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	

Table 3.2-3 (Sheet 12 of 56)
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-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	
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	

Table 3.2-3 (Sheet 13 of 56)
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-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					
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.

Table 3.2-3 (Sheet 14 of 56)
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-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.
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-MY-Y11A	CMT A Upper Level Standpipe	A	I	ASME III-1	
PXS-MY-Y11B	CMT B Upper Level Standpipe	A	I	ASME III-1	
PXS-MY-Y12A	CMT A Upper Level Standpipe	A	I	ASME III-1	
PXS-MY-Y12B	CMT B Upper Level Standpipe	A	I	ASME III-1	
PXS-MY-Y13A	CMT A Lower Level Standpipe	A	I	ASME III-1	
PXS-MY-Y13B	CMT B Lower Level Standpipe	A	I	ASME III-1	
PXS-MY-Y14A	CMT A Lower Level Standpipe	A	I	ASME III-1	
PXS-MY-Y14B	CMT B Lower Level Standpipe	A	I	ASME III-1	
PXS-MY-Y21	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y22	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y23	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y24	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y25	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y26	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y27	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y28	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y29	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y30	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y31	IRWST Hood Vent Cover	C	I	Manufacturer Std.	

Table 3.2-3 (Sheet 15 of 56)
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-Y32	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y33	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y41	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y47	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y48	IRWST Hood Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y61	IRWST SG Wall Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y62	IRWST SG Wall Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y63	IRWST SG Wall Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y64	IRWST SG Wall Vent Cover	C	I	Manufacturer Std.	
PXS-MY-Y71	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y72	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y73	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y74	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y75	IRWST Overflow Weir Cover	C	I	Manufacturer Std.	
PXS-MY-Y76	IRWST Overflow Weir Cover	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	

Table 3.2-3 (Sheet 16 of 56)
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-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	
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 Check Valve	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	

Table 3.2-3 (Sheet 17 of 56)
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-V082A	CMT A Upper Level A Isolation 1	A	I	ASME III-1	
PXS-PL-V082B	CMT B Upper Level A Isolation 1	A	I	ASME III-1	
PXS-PL-V083A	CMT A Upper Level A Isolation 2	A	I	ASME III-1	
PXS-PL-V083B	CMT B Upper Level A Isolation 2	A	I	ASME III-1	
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	A	I	ASME III-1	
PXS-PL-V086B	CMT B Upper Level B Isolation 1	A	I	ASME III-1	
PXS-PL-V087A	CMT A Upper Level B Isolation 2	A	I	ASME III-1	
PXS-PL-V087B	CMT B Upper Level B Isolation 2	A	I	ASME III-1	
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	A	I	ASME III-1	
PXS-PL-V092B	CMT B Lower Level A Isolation 1	A	I	ASME III-1	
PXS-PL-V093A	CMT A Lower Level A Isolation 2	A	I	ASME III-1	
PXS-PL-V093B	CMT B Lower Level A Isolation 2	A	I	ASME III-1	
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	A	I	ASME III-1	
PXS-PL-V096B	CMT B Lower Level B Isolation 1	A	I	ASME III-1	
PXS-PL-V097A	CMT A Lower Level B Isolation 2	A	I	ASME III-1	
PXS-PL-V097B	CMT B Lower Level B Isolation 2	A	I	ASME III-1	
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	

Table 3.2-3 (Sheet 18 of 56)
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-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	
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	A	I	ASME III-1	Note 5
PXS-PL-V118B	Containment Recirculation B Isolation	A	I	ASME III-1	Note 5
PXS-PL-V119A	Containment Recirculation A Check	C	I	ASME III-3	
PXS-PL-V119B	Containment Recirculation B Check	C	I	ASME III-3	

Table 3.2-3 (Sheet 19 of 56)
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-V120A	Containment Recirculation A Isolation	A	I	ASME III-1	Note 5
PXS-PL-V120B	Containment Recirculation B Isolation	A	I	ASME III-1	Note 5
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	
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	

Table 3.2-3 (Sheet 20 of 56)
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-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-V170A	PRHR Flow Transmitter A Vent	B	I	ASME III-2	
PXS-PL-V170B	PRHR Flow Transmitter B Vent	B	I	ASME III-2	
PXS-PL-V171A	PRHR Flow Transmitter A Vent	B	I	ASME III-2	
PXS-PL-V171B	PRHR Flow Transmitter B Vent	B	I	ASME III-2	
PXS-PL-V201A	Accumulator A Leak Test	B	I	ASME III-2	
PXS-PL-V201B	Accumulator B Leak Test	B	I	ASME III-2	
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	

Table 3.2-3 (Sheet 21 of 56)
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-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	
PXS-PY-E01	DVI-B Inline Expansion Joint	C	I	ASME III-3	
PXS-PY-R01A	Core Makeup Tank A Orifice	A	I	ASME III-1	
PXS-PY-R01B	Core Makeup Tank B Orifice	A	I	ASME III-1	
PXS-PY-R02A	Accumulator Tank A Orifice	C	I	ASME III-3	
PXS-PY-R02B	Accumulator Tank B Orifice	C	I	ASME III-3	
Balance of system components are Class E					
Reactor Coolant System (RCS)				Location: Containment	
RCS-MB-01	Steam Generator 1	A	I	ASME III-1	
n/a	SG 1 Shell	B	I	ASME III-1	
n/a	SG 1 Channel Head Divider Plate	B	I	ASME III-1	
n/a	SG 1 Tube Bundle Support Assembly	C	I	Manufacturer Std	
n/a	SG 1 Steam Flow Limiting Venturi	B	I	ASME III-1	See Subsections 5.4.4.2 and 5.4.4.3 for additional requirements.
n/a	SG 1 Feedwater Distribution Ring Supports	C	I	Manufacturer Std	
RCS-MB-02	Steam Generator 2	A	I	ASME III-1	
n/a	SG 2 Shell	B	I	ASME III-1	
n/a	SG 2 Channel Head Divider Plate	B	I	ASME III-1	
n/a	SG 2 Tube Bundle Support Assembly	C	I	Manufacturer Std	
n/a	SG 2 Steam Flow Limiting Venturi	B	I	ASME III-1	See Subsections 5.4.4.2 and 5.4.4.3 for additional requirements.

Table 3.2-3 (Sheet 22 of 56)
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	SG 2 Feedwater Distribution Ring Supports	C	I	Manufacturer Std	
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	
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-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	
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	

Table 3.2-3 (Sheet 23 of 56)
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-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	
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	
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-V108A	Hot Leg 1 Sample Isolation	B	I	ASME III-2	
RCS-PL-V108B	Hot Leg 2 Sample Isolation	B	I	ASME III-2	

Table 3.2-3 (Sheet 24 of 56)
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-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-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-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	
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	

Table 3.2-3 (Sheet 25 of 56)
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-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-V227A	Pressurizer Level Reference Leg Tubing Instrumentation Root	B	I	ASME III-2	
RCS-PL-V227B	Pressurizer Level Reference Leg Tubing Instrumentation Root	B	I	ASME III-2	
RCS-PL-V227C	Pressurizer Level Reference Leg Tubing Instrumentation Root	B	I	ASME III-2	
RCS-PL-V227D	Pressurizer Level Reference Leg Tubing Instrumentation 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-V250	ADS Discharge Line Isolation	C	I	ASME III-3	
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-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-PY-B01	ADS Hydrostatic Test Spectacle Flange	C	I	ASME III-3	
RCS-PY-B02	ADS Hydrostatic Test Spectacle Flange	C	I	ASME III-3	
RCS-PY-B03	RCS Vacuum Pump Suction Spectacle Blind	C	I	ASME III-3	

Table 3.2-3 (Sheet 26 of 56)
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-PY-B04	RCS Vacuum Ejector Package Suction Spectacle Blind	C	I	ASME III-3	
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	
RCS-PY-R01A	Reactor Vessel Head Vent Flow Orifice A	C	I	ASME III-3	
RCS-PY-R01B	Reactor Vessel Head Vent Flow Orifice B	C	I	ASME III-3	
RCS-PY-Y01A	Pressurizer Level Reference Leg L225A Flex Hose	B	I	ASME III-2	
RCS-PY-Y01B	Pressurizer Level Reference Leg L225B Flex Hose	B	I	ASME III-2	
RCS-PY-Y01C	Pressurizer Level Reference Leg L225C Flex Hose	B	I	ASME III-2	
RCS-PY-Y01D	Pressurizer Level Reference Leg L225D Flex Hose	B	I	ASME III-2	
Gravity and Roof Drain Collection System (RDS)				Location: Various	
System components are Class E					
Normal Residual Heat Removal System (RNS)			Location: Containment and Auxiliary Building		
RNS-JE-FE001A	RNS Train A Discharge Flow Element	C	I	ASME III-3	
RNS-JE-FE001B	RNS Train B Discharge Flow Element	C	I	ASME III-3	
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	

Table 3.2-3 (Sheet 27 of 56)
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-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	
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	

Table 3.2-3 (Sheet 28 of 56)
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-V025	RNS Suction from IRWST - Bonnet Relief Isolation	C	I	ASME III-3	
RNS-PL-V026	RNS Suction from IRWST - Containment Isolation Test	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	
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	

Table 3.2-3 (Sheet 29 of 56)
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-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	
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	

Table 3.2-3 (Sheet 30 of 56)
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-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	
RNS-PY-R01A	Residual Heat Removal Discharge to DVI Line Orifice A	C	I	ASME III-3	
RNS-PY-R01B	Residual Heat Removal Discharge to DVI Line Orifice B	C	I	ASME III-3	
RNS-PY-R02A	Residual Heat Removal Pump Miniflow Orifice A	C	I	ASME III-3	
RNS-PY-R02B	Residual Heat Removal Pump Miniflow Orifice B	C	I	ASME III-3	
RNS-PY-R03	Residual Heat Removal to IRWST Line Orifice	C	I	ASME III-3	
RNS-PY-R04	Residual Heat Removal to Spent Fuel Pool Line Orifice	C	I	ASME III-3	
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.	
RXS-FR-E13	Control Rod Cluster E13	B	I	Manufacturer Std.	
RXS-FR-F02	Control Rod Cluster F2	B	I	Manufacturer Std.	

Table 3.2-3 (Sheet 31 of 56)
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-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-N05	Control Rod Cluster N05	B	I	Manufacturer Std.	
RXS-FR-N07	Control Rod Cluster N07	B	I	Manufacturer Std.	
RXS-FR-N09	Control Rod Cluster N09	B	I	Manufacturer Std.	
RXS-FR-N11	Control Rod Cluster N11	B	I	Manufacturer Std.	
RXS-FR-P06	Control Rod Cluster P06	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	
RXS-MI-22	Vortex Suppression Plate	D	II	Manufacturer Std.	
RXS-MI-23	Core Shroud Assembly	D	II	Manufacturer Std.	

Table 3.2-3 (Sheet 32 of 56)
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-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	

Table 3.2-3 (Sheet 33 of 56)
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-11B06/ B08/B10/C05/C07/ C09/C11/D04/ D06/D08/D10/ D12/E03/E05/E07/ E09/E11/E13/F02/ F04/F06/F08/F10/ F12/F14/G03/G05/ G07/G09/G11/G13/ H02/H04/H06/H08/ H10/H12/H14/J03/ J05/J07/J09/J11/ J13/K02/K04/K06/ K08/K10/K12/K14/ L03/L05/L07/L09/ L11/L13/M04/M06/ M08/M10/M12/ N05/N07/N09/N11/ P06/P08/P10	CRDM Latch Assemblies	C	I	Manufacturer Std.	
RXS-MV-11B06/ B08/B10/C05/C07/ C09/C11/D04/ D06/D08/D10/ D12/E03/E05/E07/ E09/E11/E13/F02/ F04/F06/F08/F10/ F12/F14/G03/G05/ G07/G09/G11/G13/ H02/H04/H06/H08/ H10/H12/H14/J03/ J05/J07/J09/J11/ J13/K02/K04/K06/ K08/K10/K12/K14/ L03/L05/L07/L09/ L11/L13/M04/M06/ M08/M10/M12/N05/ N07/N09/N11/P06/ P08/P10	CRDM Drive Rod Assemblies	D	NS	Manufacturer Std.	

Table 3.2-3 (Sheet 34 of 56)
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-11B06/ B08/B10/C05/C07/ C09/C11/D04/D06/ D08/D10/D12/E03/ E05/E07/E09/E11/ E13/F02/F04/F06/ F08/F10/F12/F14/ G03/G05/G07/G09/ G11/G13/H02/H04/ H06/H08/H10/H12/ H14/J03/J05/J07/ J09/J11/J13/K02/ K04/K06/K08/K10/ K12/K14/L03/L05/ L07/L09/L11/L13/ M04/M06/M08/ M10/M12/N05/ N07/N09/N11/P06/ P08/P10	CRDM Coil Stack Assemblies	D	NS	Manufacturer Std.	
RXS-MV-11B06L/ B08L/B10L/C05L/ C07L/C09L/C11L/ D04L/D06L/D08L/ D10L/D12L/E03L/ E05L/E07L/E09L/ E11L/E13L/F02L/ F04L/F06L/F08L/ F10L/F12L/F14L/ G03L/G05L/G07L/ G09L/G11L/G13L/ H02L/H04L/H06L/ H08L/H10L/H12L/ H14L/J03L/J05L/ J07L/J09L/J11L/ J13L/K02L/K04L/ K06L/K08L/K10L/ K12L/K14L/L03L/ L05L/L07L/L09L/ L11L/L13L/M04L/ M06L/M08L/ M10L/M12L/N05L/ N07L/N09/N11L/ P06L/P08L/P10L	CRDM Latch Housings	A	I	ASME III-1	

Table 3.2-3 (Sheet 35 of 56)
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-11B06R/ B08R/B10R/C05R/ C07R/C09R/C11R/ D04R/D06R/D08R/ D10R/D12R/E03R/ E05R/E07R/E09R/ E11R/E13R/F02R/ F04R/F06R/F08R/ F10R/F12R/F14R/ G03R/G05R/G07R/ G09R/G11R/G13R/ H02R/H04R/H06R/ H08R/H10R/H12R/ H14R/J03R/J05R/ J07R/J09R/J11R/ J13R/K02R/K04R/ K06R/K08R/K10R/ K12R/K14R/L03R/ L05R/L07R/L09R/ L11R/L13R/M04R/ M06R/M08R/ M10R/M12R/ N05R/N07R/N11R/ P06R/P08R/P10R	CRDM Rod Travel Housings	A	I	ASME III-1	
RXS-MY-Y01	IHP Lower Shroud Assembly	C	I	ASME-NF	
RXS-MY-Y51	Integrated Head Package Lifting Rig	C	II	NUREG-0612, ANSI N14.6	
RXS-MY-Y11	Incore Instrumentation QuickLoc Assembly 1	A	I	ASME III-1	
RXS-MY-Y12	Incore Instrumentation QuickLoc Assembly 2	A	I	ASME III-1	
RXS-MY-Y13	Incore Instrumentation QuickLoc Assembly 3	A	I	ASME III-1	
RXS-MY-Y14	Incore Instrumentation QuickLoc Assembly 4	A	I	ASME III-1	
RXS-MY-Y15	Incore Instrumentation QuickLoc Assembly 5	A	I	ASME III-1	
RXS-MY-Y16	Incore Instrumentation QuickLoc Assembly 6	A	I	ASME III-1	
RXS-MY-Y17	Incore Instrumentation QuickLoc Assembly 7	A	I	ASME III-1	
RXS-MY-Y18	Incore Instrumentation QuickLoc Assembly 8	A	I	ASME III-1	
Balance of system components are Class E					
Sanitary Drainage System (SDS)				Location: Various	
SDS-PL-V001	SDS MCR Vent Isolation Valve	C	I	ASME III-3	
SDS-PL-V002	SDS MCR Vent Isolation Valve	C	I	ASME III-3	
Balance of system components are Class D, P, and W					

Table 3.2-3 (Sheet 36 of 56)
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 (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	
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	Cask Loading Pit Level Transmitter Root Isolation Valve	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	

Table 3.2-3 (Sheet 37 of 56)
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-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	
SFS-PL-V117	Refueling Cavity Drain Line Test Connection	C	I	ASME III-3	
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	
SFS-PY-R04	SFS Cask Loading and Washdown Orifice	C	I	ASME III-3	
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, LT011 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V001B	LT005, LT013 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, LT012 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V003B	LT006, LT014 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, LT015, LT044 Root Isolation Valve	B	I	ASME III-2	

Table 3.2-3 (Sheet 38 of 56)
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-V005B	LT007, LT017, LT046 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, LT016, LT045 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V007B	LT008, LT018, LT047 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	
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-V020A	FT024 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V020B	FT025 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V021A	FT024 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V021B	FT025 Root Isolation Valve	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	

Table 3.2-3 (Sheet 39 of 56)
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-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	
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	

Table 3.2-3 (Sheet 40 of 56)
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-V047A	LT015, LT044 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V047B	LT017, LT046 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, LT045 Root Isolation Valve	B	I	ASME III-2	
SGS-PL-V049B	LT018, LT047 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-V066A	FT055 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V066B	FT056 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-V068A	FT055 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V068B	FT056 Root Isolation Valve	C	I	ASME III-3	
SGS-PL-V074A	SG Blowdown Isolation	B	I	ASME III-2	
SGS-PL-V074B	SG Blowdown Isolation	B	I	ASME III-2	
SGS-PL-V075A	SG Series Blowdown Isolation	C	I	ASME III-3	
SGS-PL-V075B	SG Series Blowdown Isolation	C	I	ASME III-3	

Table 3.2-3 (Sheet 41 of 56)
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-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-V100A	Startup Feedwater Drain Isolation Valve	C	I	ASME III-3	
SGS-PL-V100B	Startup Feedwater Drain Isolation Valve	C	I	ASME III-3	
SGS-PL-V101A	Main Feedwater Drain Isolation Valve	B	I	ASME III-2	
SGS-PL-V101B	Main Feedwater Drain Isolation Valve	B	I	ASME III-2	
SGS-PL-V102A	Startup Feedwater Vent Isolation Valve	C	I	ASME III-3	
SGS-PL-V102B	Startup Feedwater Vent Isolation Valve	C	I	ASME III-3	
SGS-PL-V103A	Main Feedwater Vent Isolation Valve	B	I	ASME III-2	
SGS-PL-V103B	Main Feedwater Vent Isolation Valve	B	I	ASME III-2	
SGS-PL-V104A	Main Feedwater Drain Isolation Valve	C	I	ASME III-3	
SGS-PL-V104B	Main Feedwater Drain Isolation Valve	C	I	ASME III-3	
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	

Table 3.2-3 (Sheet 42 of 56)
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-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-PL-V300	SG1 NR LT001 Upper Isol	B	I	ASME III-2	
SGS-PL-V301	SG1 NR LT001 Lower Isol	B	I	ASME III-2	
SGS-PL-V302	SG1 NR LT002 Upper Isol	B	I	ASME III-2	
SGS-PL-V303	SG1 NR LT002 Lower Isol	B	I	ASME III-2	
SGS-PL-V304	SG1 NR LT003 Upper Isol	B	I	ASME III-2	
SGS-PL-V305	SG1 NR LT003 Lower Isol	B	I	ASME III-2	
SGS-PL-V306	SG1 NR LT004 Upper Isol	B	I	ASME III-2	
SGS-PL-V307	SG1 NR LT004 Lower Isol	B	I	ASME III-2	
SGS-PL-V308	SG1 NR LT011 Lower Isol	B	I	ASME III-2	
SGS-PL-V309	SG1 NR LT012 Lower Isol	B	I	ASME III-2	
SGS-PL-V310	SG1 NR LT015 Lower Isol	B	I	ASME III-2	
SGS-PL-V311	SG1 NR LT016 Lower Isol	B	I	ASME III-2	
SGS-PL-V312	SG1 Main Steam FT020 Isol	B	I	ASME III-2	
SGS-PL-V313	SG1 Main Steam FT021 Isol	B	I	ASME III-2	
SGS-PL-V314	SG1 Main Steam FT024 Isol	B	I	ASME III-2	
SGS-PL-V320	SG2 NR LT005 Upper Isol	B	I	ASME III-2	
SGS-PL-V321	SG2 NR LT005 Lower Isol	B	I	ASME III-2	
SGS-PL-V322	SG2 NR LT006 Upper Isol	B	I	ASME III-2	
SGS-PL-V323	SG2 NR LT006 Lower Isol	B	I	ASME III-2	
SGS-PL-V324	SG2 NR LT007 Upper Isol	B	I	ASME III-2	
SGS-PL-V325	SG2 NR LT007 Lower Isol	B	I	ASME III-2	
SGS-PL-V326	SG2 NR LT008 Upper Isol	B	I	ASME III-2	
SGS-PL-V327	SG2 NR LT008 Lower Isol	B	I	ASME III-2	
SGS-PL-V328	SG2 NR LT013 Lower Isol	B	I	ASME III-2	
SGS-PL-V329	SG2 NR LT014 Lower Isol	B	I	ASME III-2	
SGS-PL-V330	SG2 NR LT017 Lower Isol	B	I	ASME III-2	
SGS-PL-V331	SG2 NR LT018 Lower Isol	B	I	ASME III-2	
SGS-PL-V332	SG2 Main Steam FT022 Isol	B	I	ASME III-2	
SGS-PL-V333	SG2 Main Steam FT023 Isol	B	I	ASME III-2	
SGS-PL-V334	SG2 Main Steam FT025 Isol	B	I	ASME III-2	
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 43 of 56)
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	
SGS-PY-R02A	SG 1 Main Steam Line Drain Plug-Resistant Orifice	C	I	ASME III-3	
SGS-PY-R02B	SG 2 Main Steam Line Drain Plug-Resistant Orifice	C	I	ASME III-3	
SGS-PY-Y01A	Flexible Hose from SG 1 to Upper Tap LT001/LT011	B	I	ASME III-2	
SGS-PY-Y01B	Flexible Hose from SG 2 to Upper Tap LT005/LT013	B	I	ASME III-2	
SGS-PY-Y02A	Flexible Hose from SG 1 to Upper Tap LT001	B	I	ASME III-2	
SGS-PY-Y02B	Flexible Hose from SG 2 to Upper Tap LT005	B	I	ASME III-2	
SGS-PY-Y03A	Flexible Hose from SG 1 to Upper Tap LT002/LT015	B	I	ASME III-2	
SGS-PY-Y03B	Flexible Hose from SG 2 to Upper Tap LT006/LT014	B	I	ASME III-2	
SGS-PY-Y04A	Flexible Hose from SG 1 to Upper Tap LT002	B	I	ASME III-2	
SGS-PY-Y04B	Flexible Hose from SG 2 to Upper Tap LT006	B	I	ASME III-2	
SGS-PY-Y05A	Flexible Hose from SG 1 to Upper Tap LT003/LT012/LT044	B	I	ASME III-2	
SGS-PY-Y05B	Flexible Hose from SG 2 to Upper Tap LT007/LT017/LT046	B	I	ASME III-2	
SGS-PY-Y06A	Flexible Hose from SG 1 to Upper Tap LT003	B	I	ASME III-2	
SGS-PY-Y06B	Flexible Hose from SG 2 to Upper Tap LT007	B	I	ASME III-2	
SGS-PY-Y07A	Flexible Hose from SG 1 to Upper Tap LT004/LT016/LT045	B	I	ASME III-2	
SGS-PY-Y07B	Flexible Hose from SG 2 to Upper Tap LT008/LT018/LT047	B	I	ASME III-2	
SGS-PY-Y08A	Flexible Hose from SG 1 to Upper Tap LT004	B	I	ASME III-2	

Table 3.2-3 (Sheet 44 of 56)
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-Y08B	Flexible Hose from SG 2 to Upper Tap LT008	B	I	ASME III-2	
SGS-PY-Y09A	Flexible Hose from SG 1 to Upper Tap LT011	B	I	ASME III-2	
SGS-PY-Y09B	Flexible Hose from SG 2 to Upper Tap LT013	B	I	ASME III-2	
SGS-PY-Y10A	Flexible Hose from SG 1 to Upper Tap LT012	B	I	ASME III-2	
SGS-PY-Y10B	Flexible Hose from SG 2 to Upper Tap LT014	B	I	ASME III-2	
SGS-PY-Y11A	Flexible Hose from SG 1 to Upper Tap LT015/LT044	B	I	ASME III-2	
SGS-PY-Y11B	Flexible Hose from SG 2 to Upper Tap LT017/LT046	B	I	ASME III-2	
SGS-PY-Y12A	Flexible Hose from SG 1 to Upper Tap LT016/LT045	B	I	ASME III-2	
SGS-PY-Y12B	Flexible Hose from SG 2 to Upper Tap LT018/LT047	B	I	ASME III-2	
SGS-PY-Y13A	Flexible Hose from SG 1 to Lower Tap FT024	B	I	ASME III-2	
SGS-PY-Y13B	Flexible Hose from SG 2 to Lower Tap FT025	B	I	ASME III-2	
SGS-PY-Y14A	Flexible Hose from SG 1 to Lower Tap FT021	B	I	ASME III-2	
SGS-PY-Y14B	Flexible Hose from SG 2 to Lower Tap FT023	B	I	ASME III-2	
SGS-PY-Y15A	Flexible Hose from SG 1 to Lower Tap FT020	B	I	ASME III-2	
SGS-PY-Y15B	Flexible Hose from SG 2 to Lower Tap FT022	B	I	ASME III-2	
SGS-PY-Y16A	Flexible Hose from Main Steam Line to Upper Tap FT020	B	I	ASME III-2	
SGS-PY-Y16B	Flexible Hose from Main Steam Line to Upper Tap FT022	B	I	ASME III-2	
SGS-PY-Y17A	Flexible Hose from Main Steam Line to Upper Tap FT021	B	I	ASME III-2	
SGS-PY-Y17B	Flexible Hose from Main Steam Line to Upper Tap FT023	B	I	ASME III-2	
SGS-PY-Y18A	Flexible Hose from Main Steam Line to Upper Tap FT024	B	I	ASME III-2	
SGS-PY-Y18B	Flexible Hose from Main Steam Line to Upper Tap FT025	B	I	ASME III-2	

Table 3.2-3 (Sheet 45 of 56)
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-Y19A	Flexible Hose from Main Steam Line to PT030	B	I	ASME III-2	
SGS-PY-Y19B	Flexible Hose from Main Steam Line to PT034	B	I	ASME III-2	
SGS-PY-Y20A	Flexible Hose from Main Steam Line to PT032	B	I	ASME III-2	
SGS-PY-Y20B	Flexible Hose from Main Steam Line to PT036	B	I	ASME III-2	
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 46 of 56)
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.	
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	MCR Pressure Boundary Penetration Cast Sleeve	C	I	AISC N690	CA52
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					

Table 3.2-3 (Sheet 47 of 56)
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 (VES)				Location: Auxiliary Building	
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 48 of 56)
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 49 of 56)
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-PY-N02	Eductor Bypass Isolation Discharge Silencer	C	I	ASME III-3	
VES-PY-R02	Eductor Bypass Flow Orifice	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	Relief Damper A	Note 1	I	ASME N509/N510	
VES-MD-D001B	Relief Damper B	Note 1	I	ASME N509/N510	

Table 3.2-3 (Sheet 50 of 56)
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-D002	MCR Air Filtration Line Supply Damper	Note 1	I	ASME AG-1 Section DA	
VES-MD-D003	MCR Air Filtration Line Supply Damper	Note 1	I	ASME AG-1 Section DA	
VES-MY-G001	Ducting Grill	Note 1	I	ASME N509/N510	
VES-MY-R001	Ducting Register	Note 1	I	ASME N509/N510	
VES-MY-R002	Ducting Register	Note 1	I	ASME N509/N510	
VES-MY-R003	Ducting Register	Note 1	I	ASME N509/N510	
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-V006	Containment Isolation Test Connection	C	I	ASME III-3	
VFS-PL-V007	RCS Ejector Discharge Isolation	C	I	ASME III-3	
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-V101	Containment Air Supply Line Test Connection	C	I	ASME III-3	
VFS-PL-V202	Containment Atmosphere to Filtration Units Isolation	C	I	ASME III-3	
VFS-PL-V587	Filtration Units to Containment Atmosphere Manual Isolation	C	I	ASME III-3	
VFS-PL-V800A	Vacuum Relief Containment Isolation A – ORC	B	I	ASME III-2	

Table 3.2-3 (Sheet 51 of 56)
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)					
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	
n/a	Fans, Ductwork	L or R	NS	SMACNA or ASME AG-1, Note 4	
Balance of system components are Class E, L and 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					

Table 3.2-3 (Sheet 52 of 56)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
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	
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-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	
VWS-MS-02	Air-Cooled Chiller 2	D	NS	ARI/ASME VIII	
VWS-MS-03	Air-Cooled Chiller 3	D	NS	ARI/ASME VIII	
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 Return Penetration	B	I	ASME III, 2	
VWS-PY-C02	Containment Chilled Water Supply Penetration	B	I	ASME III, 2	
VWS-PL-V053	VWS Containment Penetration Thermal Relief Valve	C	I	ASME III-3	

Table 3.2-3 (Sheet 53 of 56)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Central Chilled Water System (Continued)					
VWS-PL-V057	VWS Containment Penetration Thermal Relief Valve	C	I	ASME III-3	
VWS-PL-V058	Fan Coolers Supply Containment Isolation	B	I	ASME III-2	
VWS-PL-V062	Fan Coolers Supply Containment Isolation Check Valve	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					
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					

Table 3.2-3 (Sheet 54 of 56)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
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.	
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.	

Table 3.2-3 (Sheet 55 of 56)
AP1000 Classification of Mechanical and Fluid Systems, Components, and Equipment

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
Liquid Radwaste System (Continued)					
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.	
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	

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

Tag Number	Description	AP1000 Class	Seismic Category	Principal Construction Code	Comments
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.
5. PXS valves V118A/B and V120A/B are Class A valves in Class C piping. The basis for these valves is “safety injection, maintain reactor coolant inventory.”

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 portions of the annex and turbine buildings adjacent to the nuclear island are classified as seismic Category II and are designed to seismic Category I structure tornado loading. The acceptance criteria are based on ACI 349 for concrete structures, on AISC N690 for steel structures, including the supplemental requirements described in Subsections 3.8.4.4.1 and 3.8.4.5, and on AWS D1.1-2000 for weld design, qualification, fabrication, and inspection as described in Subsections 3.8.3.2 and 3.8.4.2. The structures are 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.

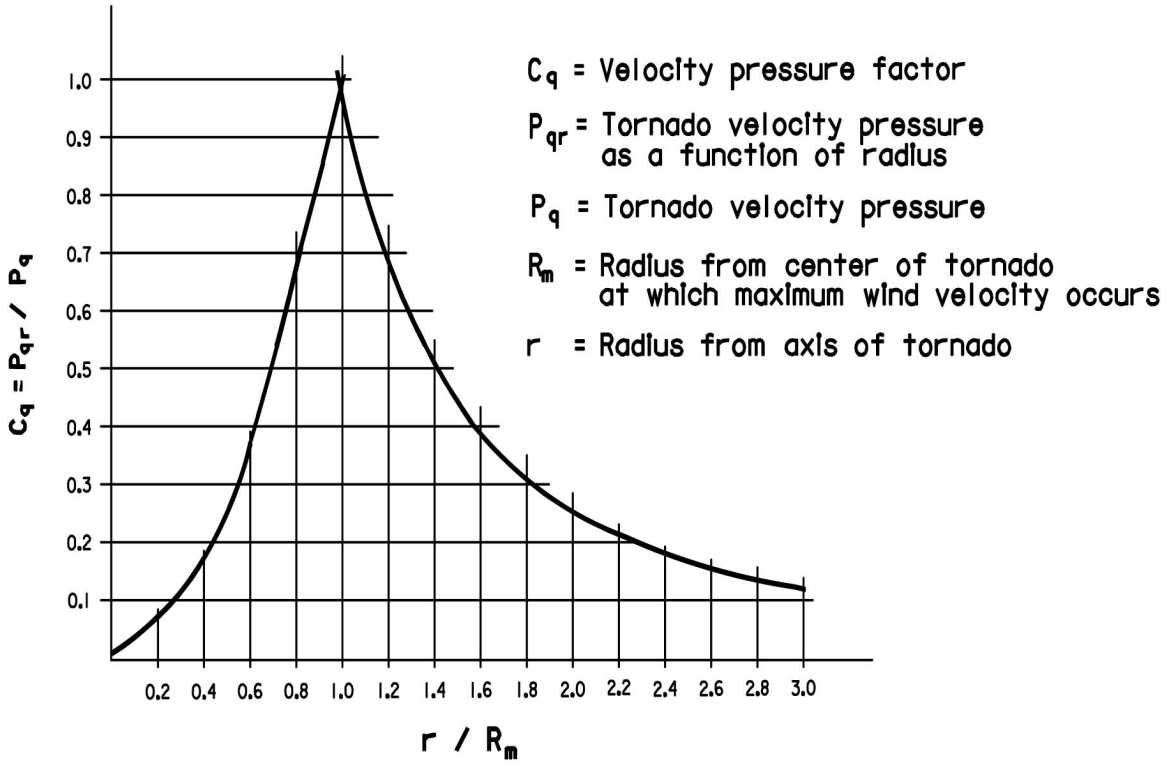


Figure 3.3-1
Velocity Pressure Variation with Radius from Center of Tornado

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 approximately 656,000 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 6.2.2.2.3. 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, weirs and elevated thresholds
- Watertight hatches
- 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. Watertight hatches are provided to close openings not required for hydrogen venting. The walls, floors, hatches, 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, lower 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. The northeast compartment is interconnected with the RNS valve room by a passageway. These two compartments are treated, in this discussion, as one floodable volume referred to 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 eight 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 and one normal residual heat removal 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″ or watertight hatches are provided to close openings not required for hydrogen venting 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 or watertight hatch is provided at 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 or demineralized water brought into the area in portable containers is used for battery washdown. Potable water is used for 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 seismically qualified fire protection system piping routed through levels 1, 2, 3, and 4 is the only piping in level 1 that is greater than 1 inch in diameter.

  Fire fighting activities or fire protection system piping through-wall cracks 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 water storage tank. A nominal volume of 18,000 gallons is provided for the fire protection system. A volume of approximately 26,200 gallons is conservatively assumed; this is the volume in the tank between the elevations of the fire protection system inlet and the tank overflow, and associated system piping. In the event of a through-wall crack in the fire protection system piping, the maximum water depth in the battery rooms would be less than 7.5 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 battery jars, which are located 7.5 inches above the floor. The terminal height is 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 primarily operated 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 and through-wall cracks in fire protection system piping. 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. The main control room contains a water heater tank and expansion tank Section in the men's toilet room to provide hot water for hand washing and dishwashing. To mitigate the effects of a postulated tank rupture, the water heater tank and expansion tank are enclosed in a seismic Category I structure and the room design is configured to contain the water and direct it to the floor drain, where it is gravity-drained to the sanitary drainage system (SDS).

  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. The only safety-related equipment on level 5 that could be affected by flooding is the potable water system main control room isolation valve located in the main mechanical HVAC equipment room, but it is positioned above the maximum flood height for this room.

  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. 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 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 bounding through-wall crack is postulated in the 4 inch recirculation piping. This leak is not isolable from the 756,700-gallon passive containment cooling system water storage tank above the valve room.

The PCS valve room contains floor drains with sufficient capacity to prevent water accumulation in the room following a postulated through-wall crack. Water from the through-wall crack flows into the floor drains out to the auxiliary building roof, then to the storm drain system. The passive containment cooling system isolation valves are located above the maximum flood level in the valve room, so they remain operable.

The PCCWST level instruments are relied upon to monitor PCCWST water level for operation of the PCS and to identify leakage from the PCS. 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."

NOTES:

1. HDPE DOUBLE-SIDED TEXTURED WATERPROOF MEMBRANE ON TOP OF FIRST LAYER OF MUDMAT AND ON OUTSIDE VERTICAL FACE OF AUXILIARY BUILDING WALL UP TO EL 100' -0" GRADE LEVEL (WITH PROTECTIVE SHIELD ON VERTICAL FACE)

2. MSE WALL TO BE DESIGNED WITH GEOREINFORCED MATERIALS AND 18" THICK LAYERS OF COMPACTED FREE DRAINING GRANULAR SOIL

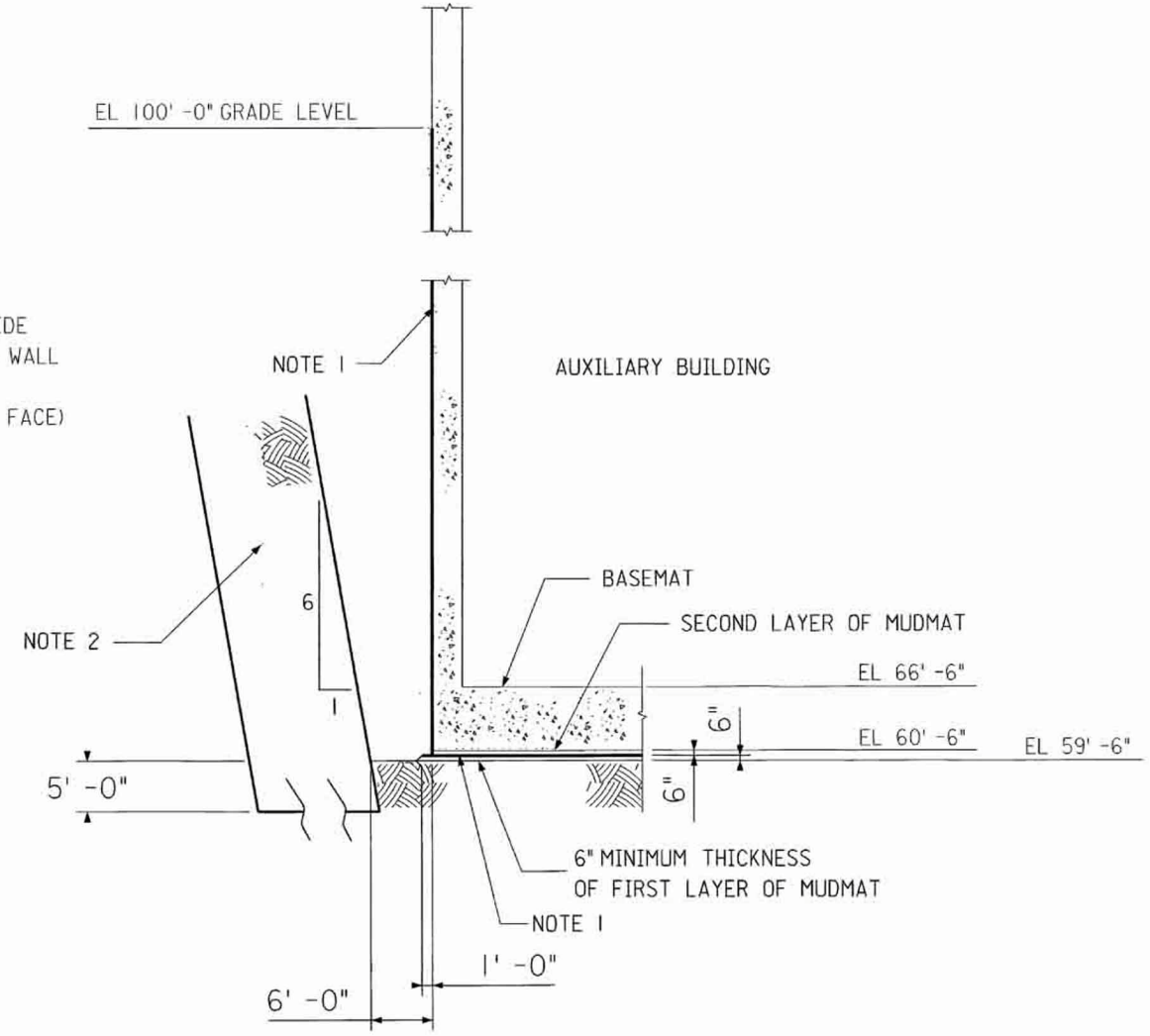


Figure 3.4-1
Typical Details of
Nuclear Island Waterproofing Below Grade

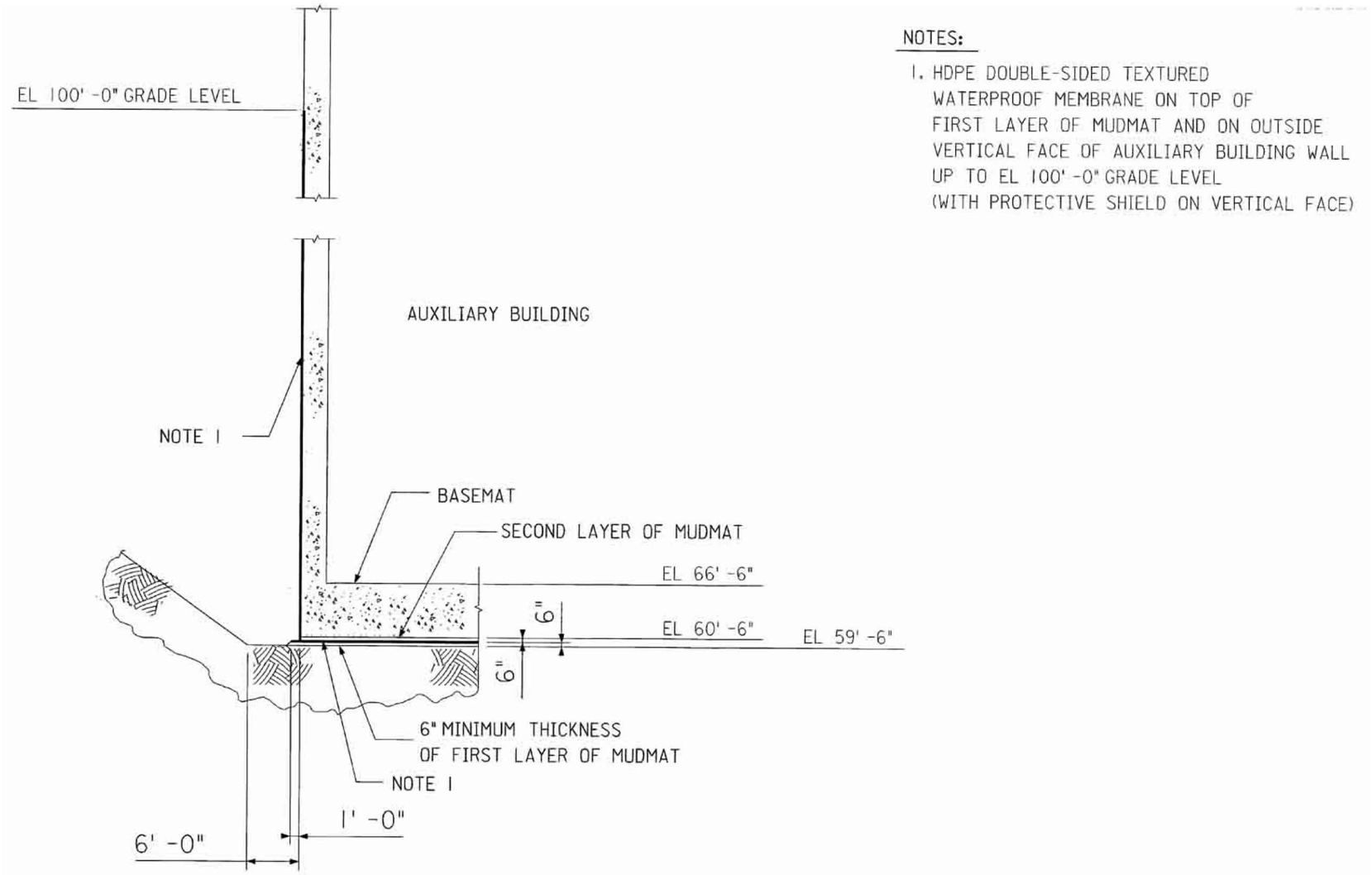


Figure 3.4-2
Typical Details of Nuclear Island Waterproofing Below Grade with Step Back

Figure 3.4-3 Not Used

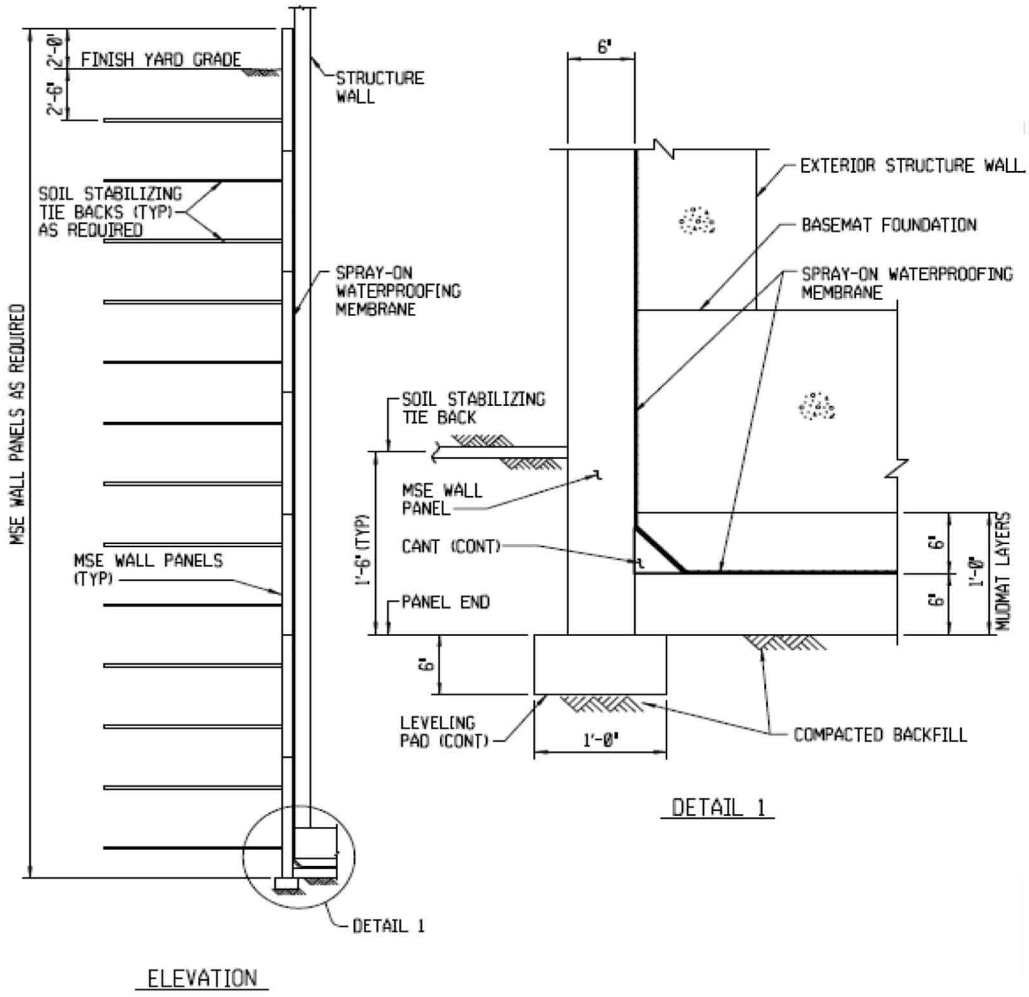


Figure 3.4-4
Typical Details of Membrane Corner Detail at Basemat and Exterior Wall

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 head. They are included as part of 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 3.5.1.1.2.1.

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 1×10^{-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.

Where necessary, adjacent structures and/or missile barriers are used to protect openings in the nuclear island building structures. Building structures credited in this evaluation to protect openings in seismic Category I nuclear island building structures are seismic Category I or seismic Category II (see Subsection 3.2.1.1.2). As identified in Subsection 3.7.2, seismic Category II building structures are designed for the safe shutdown earthquake using the same methods and design stress limits as are used for seismic Category I structures. Seismic Category II building structures are also designed to withstand the design basis tornado loads, including missiles, in accordance with the loading combinations identified in Table 3.8.4-2.

Evaluation of the openings in the north wall of the auxiliary building (Wall 11) considered the protection provided by the seismic Category II turbine building first bay building structure and associated missile barriers, consistent with the provisions in Subsections 3.2.1.1.1, 3.2.1.1.2, 3.2.2.5, and 3.2.2.6. The seismic Category II turbine building first bay building structure and the missile barriers provided over the turbine building first bay large openings are designed and analyzed to protect the openings in the north wall of the auxiliary building from the automobile missile described in Subsection 3.5.1.4. The turbine building first bay large opening missile barriers are located as follows:

- First bay compartment elevation 100'-0"
  - Four large opening missile barriers on Wall 11.2
- First bay compartment elevation 117'-6"
  - Two large opening missile barriers on Wall 11.2
  - One large opening missile barrier on Wall R
- First bay compartment elevation 135'-3"
  - One large opening missile barrier on Wall 11.2
  - One large opening missile barrier on Wall R
- First bay compartment elevation 148'-10"
  - One large opening missile barrier on Wall R

In accordance with the missile identification and protection criteria provided in Section 3.5, a realistic assessment of potential missile paths was conducted. Where the line of sight of a missile passing through a turbine building first bay building structure opening could potentially result in missile impact upon a Wall 11 opening, the missile was considered credible, and additional protection was provided or additional evaluation performed. As a result of this analysis, a missile barrier designed to stop the 8-inch artillery shell and 1-inch sphere missiles is provided within the interior of the turbine building first bay building structure, at elevation 117'-6", northeast of MSIV Compartment B. Where the 1-inch diameter solid steel sphere or 8-inch artillery shell missiles were determined to have a line of sight to a Wall 11 penetration, analysis was performed to confirm that there was no line of sight through the penetration, the contents of the penetration would prevent the missile from passing through Wall 11, or equipment required to achieve safe shutdown could not be adversely affected.

The steel tornado missile barriers located on or within the turbine building first bay are designed and analyzed in accordance with the barrier design procedures contained in Subsection 3.5.3 and applicable AISC N690 requirements.

The steel tornado missile barriers located on or within the turbine building first bay, identified in Table 3.5-1, are designed and analyzed in accordance with the barrier design procedures and ductility requirements contained in Subsection 3.5.3 and applicable AISC N690 requirements. The missile barriers are permanently anchored to the turbine building first bay or auxiliary building as applicable. Where anchored to concrete, the anchors conform with the anchorage requirements contained in Subsection 3.8.4.5.1. The configuration of the barriers is based on the lines of sight to be eliminated and the missile sizes pertinent to those lines of sight. The configuration also provides for functional requirements such as airflow, venting, and personnel access. Steel barrier designs are provided in both solid and grating-type configurations to address the required missile protection and functional requirements on a case-by-case basis. The missile protection provided for Wall 11 openings and penetrations is tabulated in Table 3.5-1. Protection of Wall 11 openings from external missiles is provided by the seismic Category II turbine building first bay building structure, the seismic Category II missile barriers located on or within the turbine building first bay building structure, the seismic Category I Wall 11 doors, and the seismic Category I Wall 11 penetrations and spare penetration covers within missile lines of sight. For some missile paths, lines of sight are eliminated for the 8-inch artillery shell and 1-inch sphere missiles due to the depth of the grating-type barrier design.

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:

$$\frac{t_s}{d} = 2.12 + 1.36\frac{x}{d} \qquad \text{for } 0.65 \le \frac{x}{d} \le 11.75$$

$$\frac{t_s}{d} = 7.91\left(\frac{x}{d}\right) - 5.06\left(\frac{x}{d}\right)^2 \qquad \text{for } \frac{x}{d} \le 0.65$$

$$\frac{t_p}{d} = 1.32 + 1.24\frac{x}{d} \qquad \text{for } 1.35 \le \frac{x}{d} \le 13.5$$

$$\frac{t_p}{d} = 3.19\left(\frac{x}{d}\right) - 0.718\left(\frac{x}{d}\right)^2 \qquad \text{for } \frac{x}{d} \le 1.35$$

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 3.5-1 (Sheet 1 of 2)
External Missile Protection Provided for Auxiliary Building Wall 11 Openings

Wall 11 Opening	Protected Room	Elevation	Missile[1]	Protection[2][3]
Room 12306 doorway	Valve/Piping Penetration Room	100'-0"	Sphere	Room 12306 Wall 11 door
			Artillery Shell	Room 12306 Wall 11 door
			Automobile	First bay building structure Wall 11.2 large opening missile barriers
Room 12404 vent	Lower MSIV Compartment B	117'-6"	Sphere	First bay building structure Wall 11.2 large opening missile barriers First bay interior missile barrier
			Artillery Shell	First bay building structure Wall 11.2 large opening missile barriers First bay interior missile barrier
			Automobile	First bay building structure Wall 11.2 large opening missile barriers
Room 12405 doorway	Lower VBS B&D Equipment Room	117'-6"	Sphere	Room 12405 Wall 11 door
			Artillery Shell	Room 12405 Wall 11 door
			Automobile	First bay building structure Wall 11.2 large opening missile barriers Wall R large opening missile barrier
Room 12406 vent	Lower MSIV Compartment A	117'-6"	Sphere	First bay building structure Wall 11.2 large opening missile barriers
			Artillery Shell	First bay building structure Wall 11.2 large opening missile barriers
			Automobile	First bay building structure Wall 11.2 large opening missile barriers Wall R large opening missile barrier
Room 12504 doorway	Upper MSIV Compartment B	135'-3"	Sphere	Room 12504 Wall 11 door
			Artillery Shell	Room 12504 Wall 11 door
			Automobile	First bay building structure Wall 11.2 large opening missile barrier
Room 12505 doorway	Upper VBS B&D Equipment Room	135'-3"	Sphere	Room 12505 Wall 11 door
			Artillery Shell	Room 12505 Wall 11 door
			Automobile	First bay building structure Wall 11.2 large opening missile barrier
Room 12506 doorway	Upper MSIV Compartment A	135'-3"	Sphere	Room 12506 Wall 11 door
			Artillery Shell	Room 12506 Wall 11 door
			Automobile	First bay building structure Wall R large opening missile barrier

Table 3.5-1 (Sheet 2 of 2)
External Missile Protection Provided for Auxiliary Building Wall 11 Openings

Wall 11 Opening	Protected Room	Elevation	Missile[1]	Protection[2][3]
Wall 11 penetrations	Various	Various	Sphere	First bay building structure Pipe sleeves Penetration contents Spare penetration covers
			Artillery Shell	First bay building structure Pipe sleeves Penetration contents Spare penetration covers
			Automobile	First bay building structure Wall 11.2 large opening missile barriers Wall R large opening missile barriers

Notes:

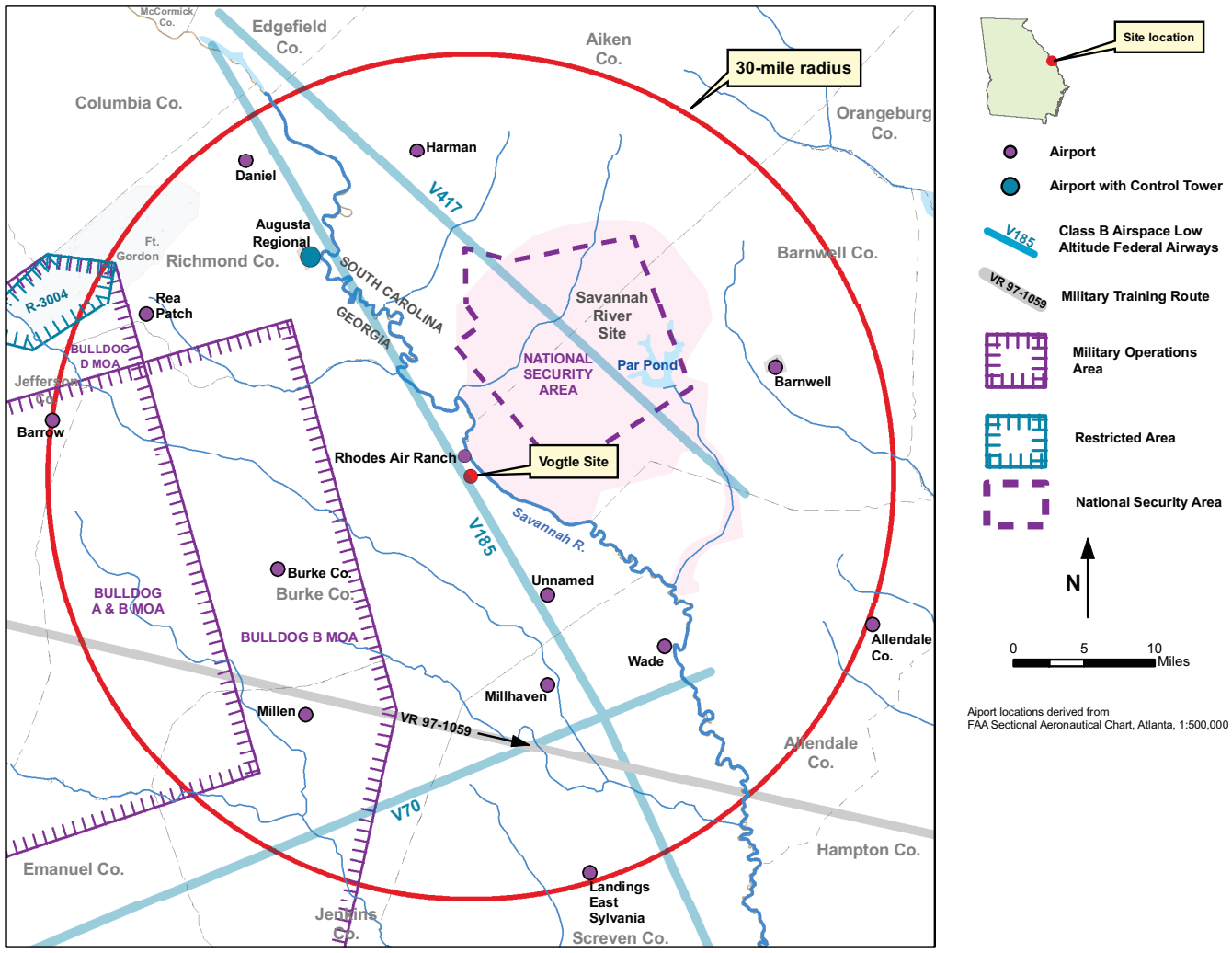
1. Tornado missiles are defined in Subsection 3.5.1.4.
2. Turbine building first bay is a seismic Category II building structure. Wall 11 doors, Wall 11 penetration sleeves, and Wall 11 spare penetration covers that provide tornado missile protection are equipment Class C. Barriers in the turbine building first bay building structure that provide tornado missile protection are equipment Class D.
3. The materials of construction for the missile barriers installed on or within the turbine building first bay and on Wall 11 shall be steel as specified below, or steel with equal or better material properties:
   - Wall 11.2 and Wall R large opening missile barriers
     - grating bar, baseplate - ASTM A240
     - perimeter frame - ASTM A36 or ASTM A992
     - anchors - ASTM F1554, Gr. 105 ·
   - First bay interior missile barrier
     - barrier plate - ASTM A240
     - barrier support frame - ASTM A500, Gr. B
     - anchors - ASTM A325 or ASTM A490, and ASTM F1554, Gr. 105 as required
   - Wall 11 missile doors
     - door plate - ASTM A240
     - embed plates - ASTM A572, Gr. 50
     - anchors - ASTM A 1064
   - Wall 11 spare penetration covers
     - ASTM A240
   - Pipe sleeves
     - material shall be as specified for their penetration function

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. However, the seismic Category II wall separating the first bay of the turbine building from the main area of the turbine building (Wall 11.2) is credited with protecting the north wall of the auxiliary building (Wall 11) from high energy line break (HELB) loads resulting from postulated ruptures in the main steam and main feedwater piping north of the turbine building first bay. 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.

Outside the containment, the wall separating the first bay of the turbine building from the main area of the turbine building (Wall 11.2) is credited with protecting the north wall of the auxiliary building (Wall 11) from HELB loads resulting from postulated ruptures in main steam and main feedwater piping north of the turbine building first bay. The wall is designed and analyzed for the loadings associated with the effects of postulated pipe ruptures in high energy lines north of the turbine building first bay in conjunction with the required load combinations identified in Subsection 3.8.4, and determined to meet the acceptance criteria contained in ACI 349-01 and AISC N690-1994.

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.*

*NRC Staff approval is required prior to implementing a change in this information.

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.*

- *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

*NRC Staff approval is required prior to implementing a change in this information.

– 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-3653 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.

- 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, considering Level A and B Service conditions, does not exceed either 0.4 (1.8 Sh + SA) or 0.4 (1.5 Sy + 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, 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.

[*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)

*NRC Staff approval is required prior to implementing a change in this information.

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 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.

*NRC Staff approval is required prior to implementing a change in this information.

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. RELAP5 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. American Welding Society (AWS), Structural Welding Code-Steel, AWS D1.1-2000 provides an acceptable alternative for AISC N690 weld requirements as described in Subsections 3.8.3.2 and 3.8.4.2.

- 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,

*NRC Staff approval is required prior to implementing a change in this information.

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.

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

*NRC Staff approval is required prior to implementing a change in this information.

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.

16. Information Systems Laboratories (2006), “RELAP5/MOD3.3 (Patch03) Code Manual Volume I to VIII, NUREG/CR-5535/Rev P3-Vol I to VIII,” prepared for U.S. Nuclear Regulatory Commission.

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)		•
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).

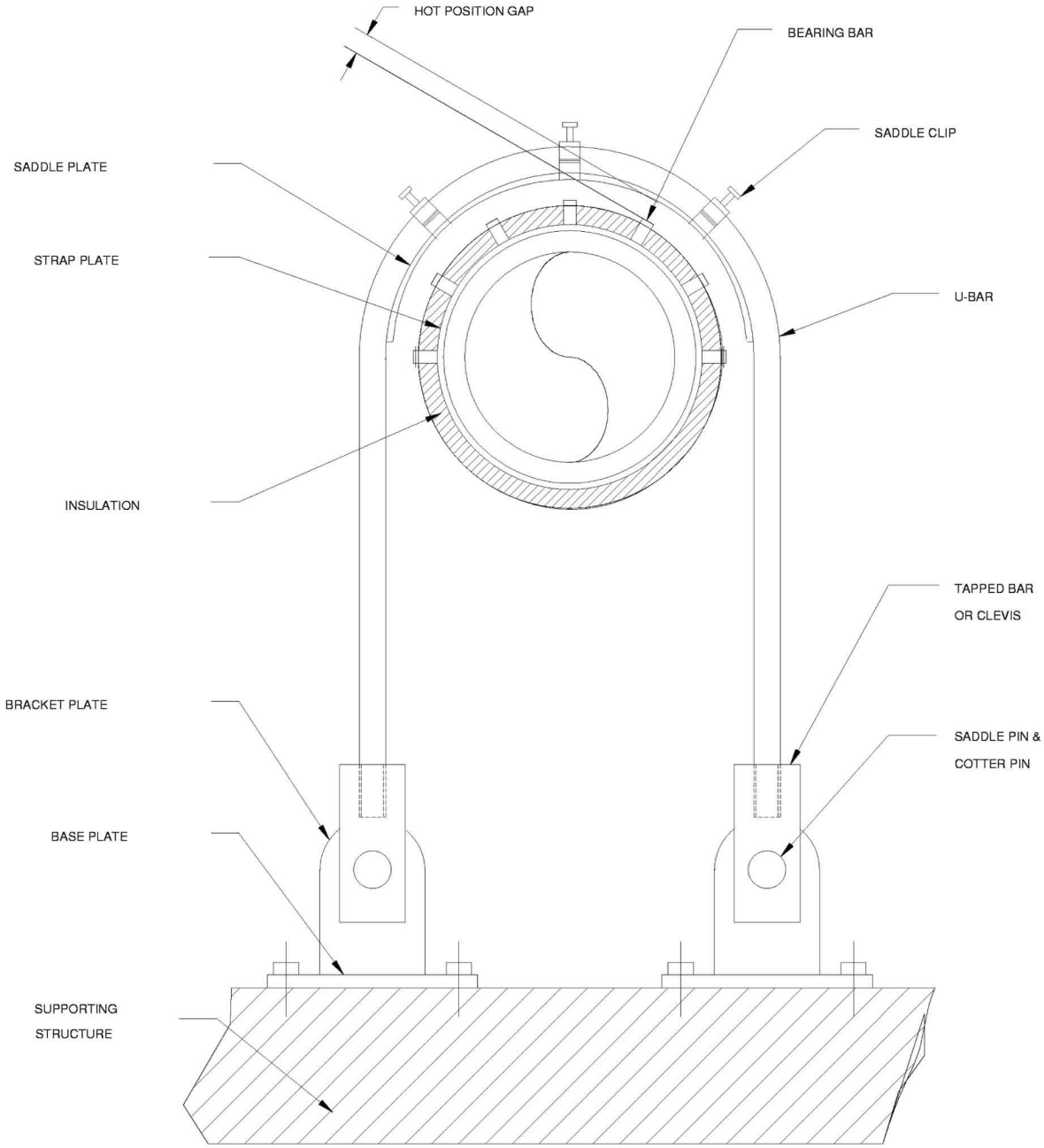


Figure 3.6-1
Typical U-Bar Restraint

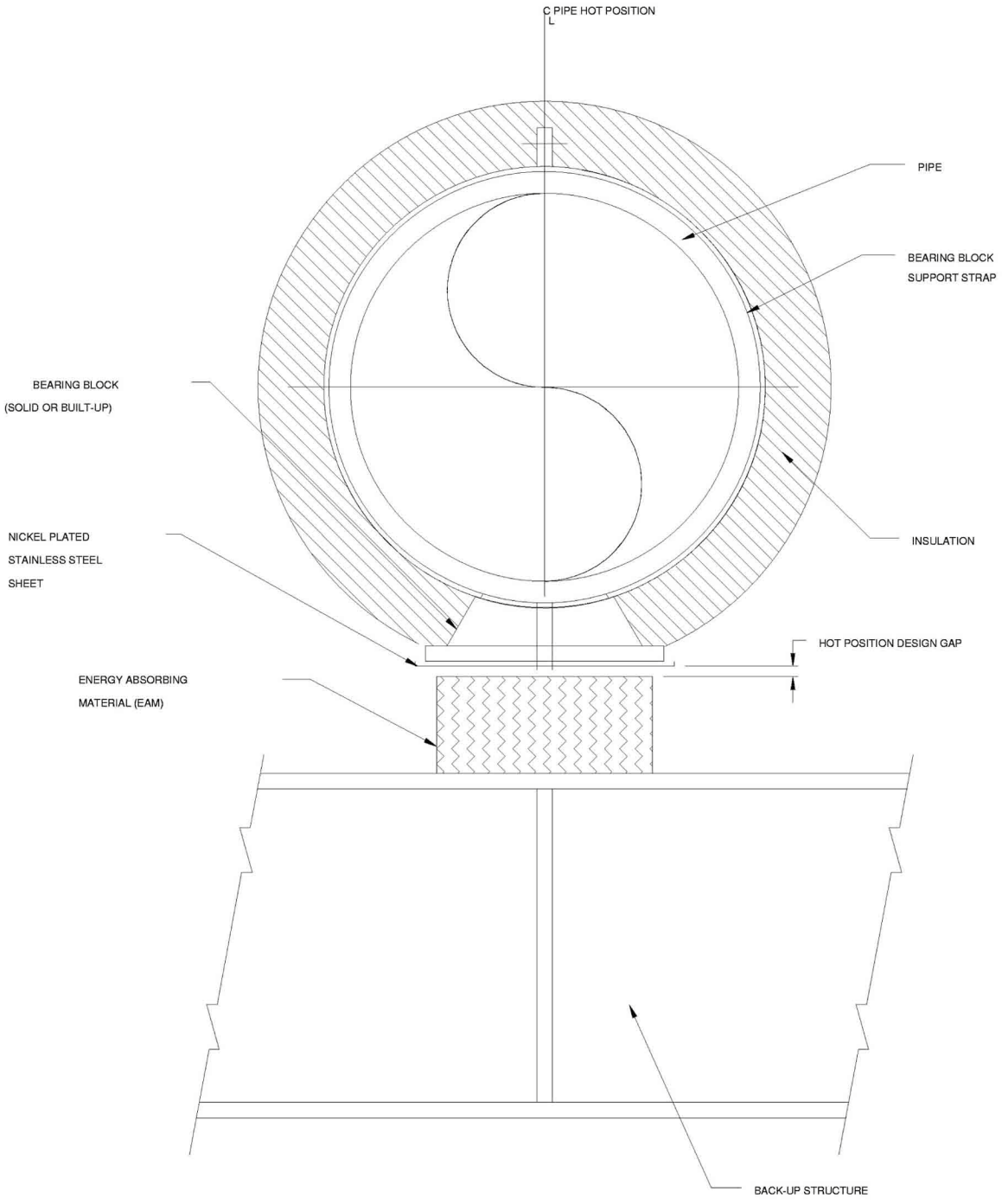
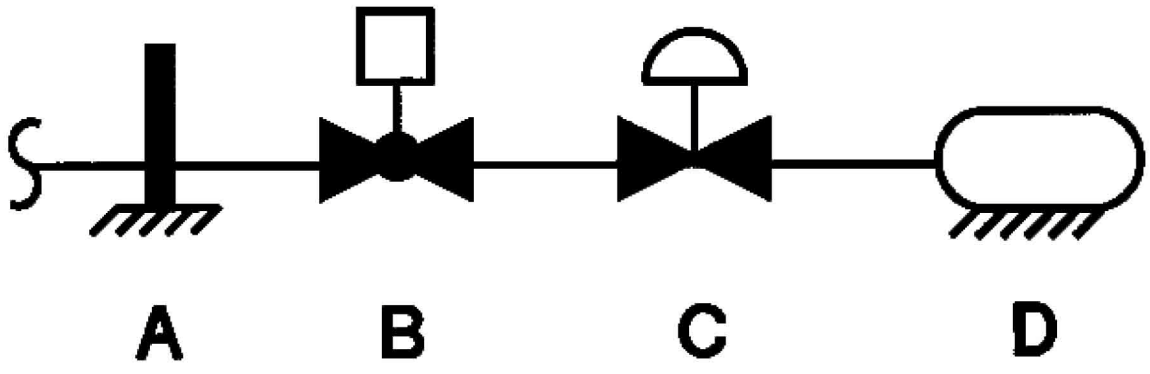


Figure 3.6-2
Typical Energy Absorbing Material Restraint



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

*NRC Staff approval is required prior to implementing a change in this information.

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, on AISC N690 for steel structures, including the supplemental requirements described in Subsections 3.8.4.4.1 and 3.8.4.5, and on AWS D1.1-2000 for weld design, qualification, fabrication, and inspection as described in Subsections 3.8.3.2 and 3.8.4.2.

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 an updated 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.2-5A with five masses at the mid-height of the bridge at elevation 236′-6″ and one mass for the trolley. The polar crane model includes the flexibility of the crane bridge girders and truck assembly. The containment shell's local flexibility is considered when combined with the global model. When fixed at the center of containment, the model shows fundamental frequencies of 4.6 hertz transverse to the bridge, 6.9 hertz vertically, and 9.4 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 modeling used to represent the structural configuration in the soil structure interaction analysis is shown in Figure 3.7.2-21.

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, AISC-N690 for steel features, including the supplemental requirements described in Subsections 3.8.4.4.1 and 3.8.4.5, and AWS D1.1-2000 for weld design, qualification, fabrication, and inspection as described in Subsections 3.8.3.2 and 3.8.4.2.

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

l 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.

$$k_1 \cdot \text{sign}(A_{NS}) \cdot (\alpha \cdot A_{NS}) + k_2 \cdot \text{sign}(A_{EW}) \cdot (\alpha \cdot A_{EW}) + k_3 \cdot A_{VT}$$

where,

k_i combination factors (±1.0, ±0.4, ±0.4)

sign(X) sign of variable X: X < 0 results -1; X ≥ 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 ≥ 0.01 and ≤ 0.10, if Rf ≤ 0.8 or if Rf is ≥ 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
- Raceway Systems
- 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.

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

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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 = *< l, (if RF > l, 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.*

$M_L = 0.8h^{0.6} D^2t\ Sy$ *for* $h < 1.45$

$M_L = D^2t\ Sy$ *for* $h > 1.45$

h = *Flexibility characteristic of elbow/bend*

D = *Outside diameter of elbow/bend*

t = *Thickness of elbow/bend*

R = *Bend radius of elbow/bend*

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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.*]*

3.7.3.14 Seismic Analyses for Reactor Internals

See Subsection 3.9.2 for the dynamic analyses of reactor internals.

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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 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.

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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

1. EPRI Report NP-5930, "A Criterion for Determining Exceedance of the Operating Basis Earthquake," July 1988.

2. Uniform Building Code, 1997.

3. ASCE Standard 4-98, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," American Society of Civil Engineers, September 1998.

4. ASME B&PV Code, Code Case N-411.

5. FEMA 356, "Prestandard and Commentary for the Seismic Rehabilitation of Buildings," Federal Emergency Management Agency, November 2000.

6. Not used.

7. Not used.

8. Not used.

9. Not used.

10. Hyde, S. J., J. M. Pandya, and K. M. Vashi, "Seismic Analysis of Auxiliary Mechanical Equipment in Nuclear Plants," Dynamic and Seismic Analysis of Systems and Components, ASME-PVP-65, American Society of Mechanical Engineers, Orlando, Florida, 1982.

11. Lin, C. W., and T. C. Esselman, "Equivalent Static Coefficients for Simplified Seismic Analysis of Piping Systems," SMIRT Conference 1983, Paper K12/9.

12. M. S. Yang, J. S. M. Leung, and Y. K. Tang, "Analysis of Piping Systems with Gapped Supports Using the Response Spectrum Method." Presented at the 1989 ASME Pressure Vessels and Piping Conference at Honolulu, July 23-27, 1989.

13. "Impact Response of Piping Systems with Gaps," P. H. Anderson and H. Loey, ASME Seismic Engineering, 1989, Volume 182.

14. "Independent Support Motion (ISM) Method of Modal Spectra Seismic Analysis," December 1989; by Task Group on Independent Support Motion as Part of the PVRC Technical Committee on Piping Systems Under the Guidance of the Steering Committee.

15. J. S. Meserole, A. Fortini, "Slosh Dynamics in a Toroidal Tank," Journal Spacecraft Vol. 24, Number 6, November-December 1987.

16. IEEE 344-1987, "Recommended Practices for Seismic Qualification of 1E Equipment for Nuclear Power Generating Stations."

17. EPRI Report TR-100082, "Standardization of the Cumulative Absolute Velocity," December 1991.

18. EPRI Report NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake," December 1989.

19. "Cable Tray and Conduit Raceway Seismic Test Program, Release 4," Report 1053-21.1-4, ANCO Engineers, Inc., December 15, 1978.

20. Not used.

21. Not used.

22. WCAP 7921AR, "Damping Values of Nuclear Power Plant Components," May 1974.

23. McGuire, R. K., G. R. Toro, and W. J. Silva, "Engineering Model of Earthquake Ground Motion for Eastern North America," Technical Report NP-6074, Electric Power Research Institute, 1988.

24. Boore, D. M. and G. M. Atkinson, "Stochastic prediction of ground motion and spectral response at hard-rock sites in eastern North America," Bull. Seism. Soc. Am., 77:2, pages 440-467.

25. Nuttli, O. W., Letter dated September 19, 1986 to J. B. Savy, Reproduced in: D. Bernreuter, J. Savy, R. Mensing, J. Chen, and B. Davis, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains: Questionnaires," U.S. Nuclear Regulatory Commission, Technical Report NUREG/CR-5250, UCID-21517, 7, Prepared by Lawrence Livermore National Laboratory.

26. Trifunac, M. and V. W. Lee, "Preliminary Empirical Model for Scaling Pseudo Velocity Spectra of Strong Earthquake Accelerations in Terms of Magnitude, Distance, Site Intensity and Recording Site Conditions," Report No. CE 85-04, University of Southern California, Department of Civil Engineering, 1985.

27. Bernreuter, D., J. Savy, R. Mensing, J. Chen, and B. Davis, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains: Questionnaires," U.S. Nuclear Regulatory Commission, Technical Report NUREG/CR-5250, UCID-21517, Lawrence Livermore National Laboratory, 1989.

28. McGuire, R. K., G. R. Toro, J. P Jacobson, T. F. O'Hara, and W. J. Silva, "Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," Technical Report NP-6395-D, Electric Power Research Institute, 1989.

29. Philippacoupoulos, A. J., "Recommendations for Resolution of Public Comments on USI A-40, Seismic Design Criteria," Brookhaven National Laboratory Report BNL-NUREG-52191, prepared for the U.S. Nuclear Regulatory Commission, and published as NUREG/CR-5347, 1989.

30. C. Chen, "Definition of Statistically Independent Time Histories," Journal of the Structural Division, ASCE, February 1975.

31. WCAP-9903, "Justification of the Westinghouse Equivalent Static Analysis Method for Seismic Qualification of Nuclear Power Plant Auxiliary Mechanical Equipment," August 1980.

32. Letter from James T. Wiggins to John J. Taylor, September 13, 1993.

33. Not used.

34. "Seismic Provisions for Structural Steel Buildings," American Institute of Steel Construction, April 1997 including Supplement No. 2, November 2000.

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

36. ANSYS Engineering Analysis Users Manual, Releases up to and including ANSYS 5.7, ANSYS Inc.

37. H. B. Seed, and I. M. Idriss, "Soil Moduli and Damping Factors for Dynamic Response Analysis," Report No. EERC-70-14, Earthquake Engineering Research Center, University of California, Berkeley, 1970.

38. EPRI TR-102293, "Guidelines for Determining Design Basis Ground Motions," 1993.

39. APP-GW-GLR-021, "AP1000 As-Built COL Information Items," Westinghouse Electric Company LLC.

40. International Building Code, 2006.

41. AISC 341-05, "Seismic Provisions for Structural Steel Buildings," March 9, 2005, including Supplement No. 1 dated November 16, 2005.

42. ASCE 7-05, "Minimum Design Loads for Buildings and Other Structures."

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

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	VES air storage	VES high pressure air bottles
12651	VAS Equipment Room	VFS containment isolation valves
12563	Spent fuel storage pit	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

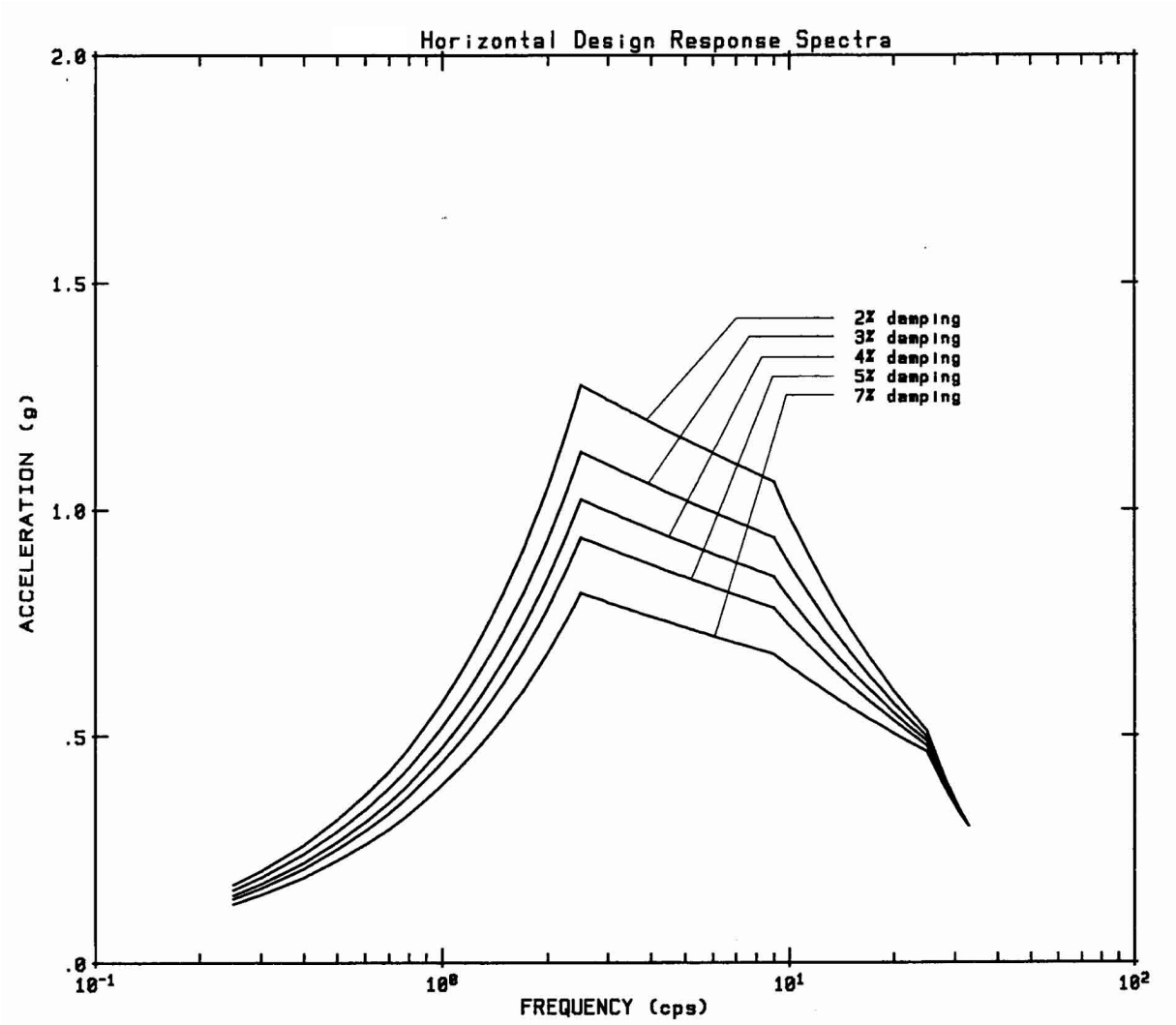


Figure 3.7.1-1
Horizontal Design Response Spectra
Safe Shutdown Earthquake

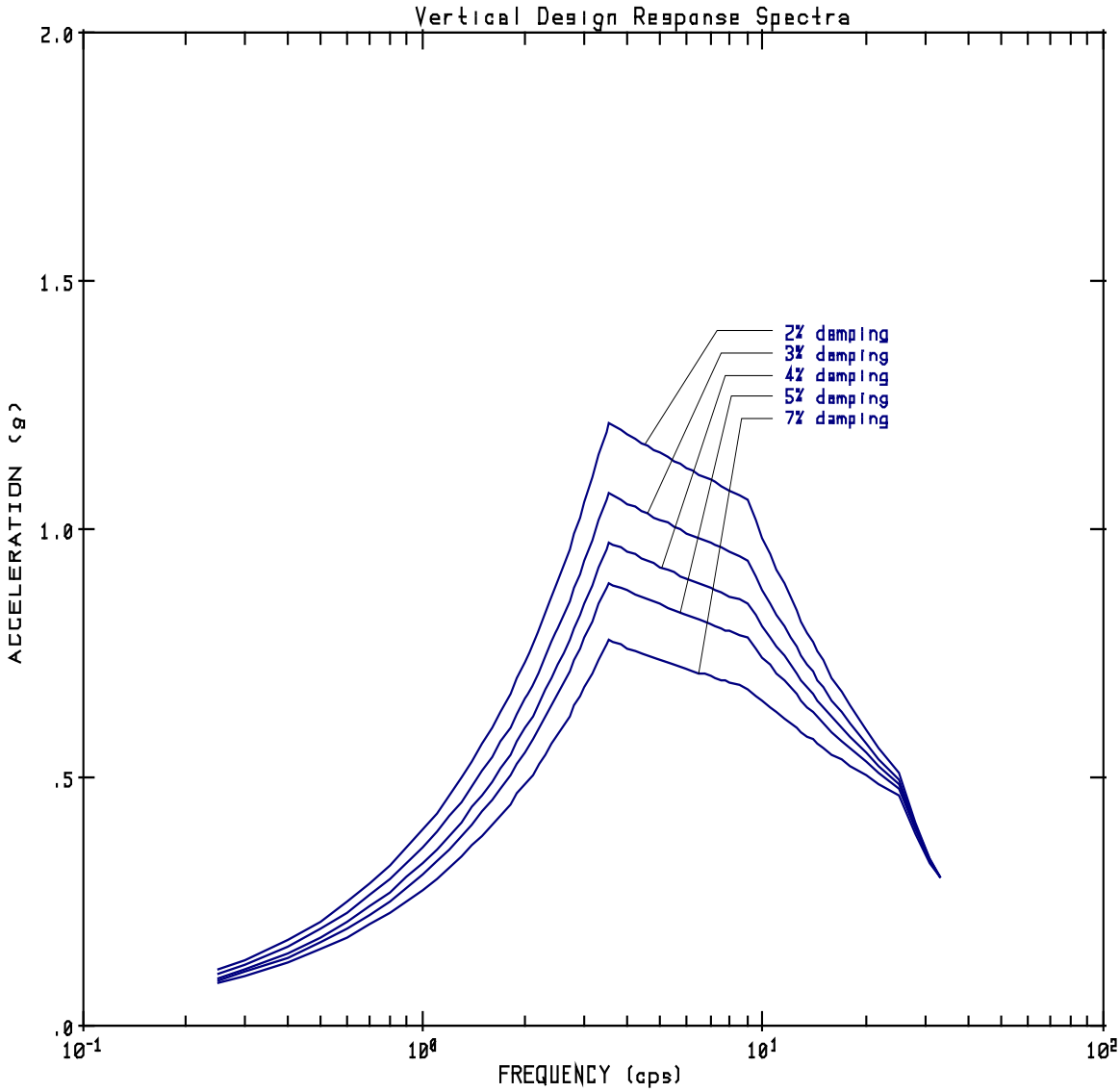


Figure 3.7.1-2
Vertical Design Response Spectra
Safe Shutdown Earthquake

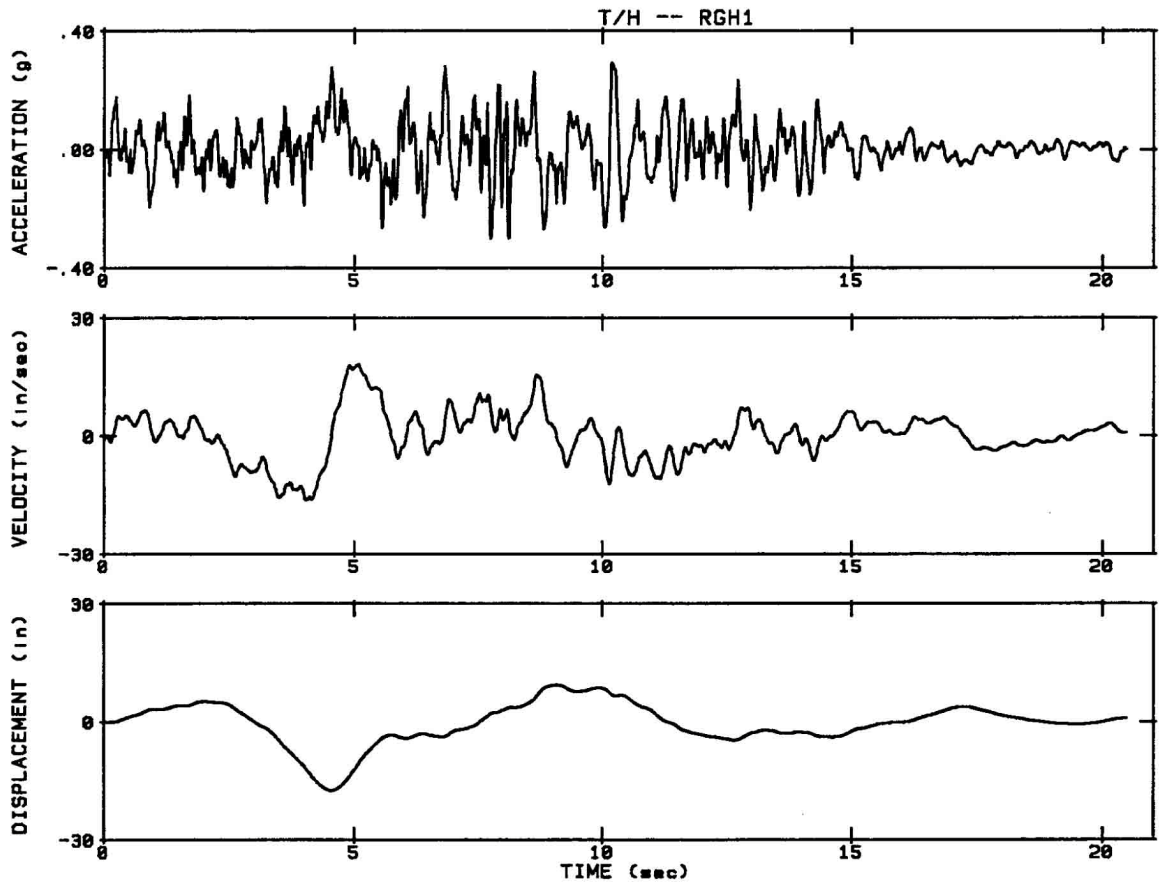


Figure 3.7.1-3
Design Horizontal Time History, "H1"
Acceleration, Velocity & Displacement Plots

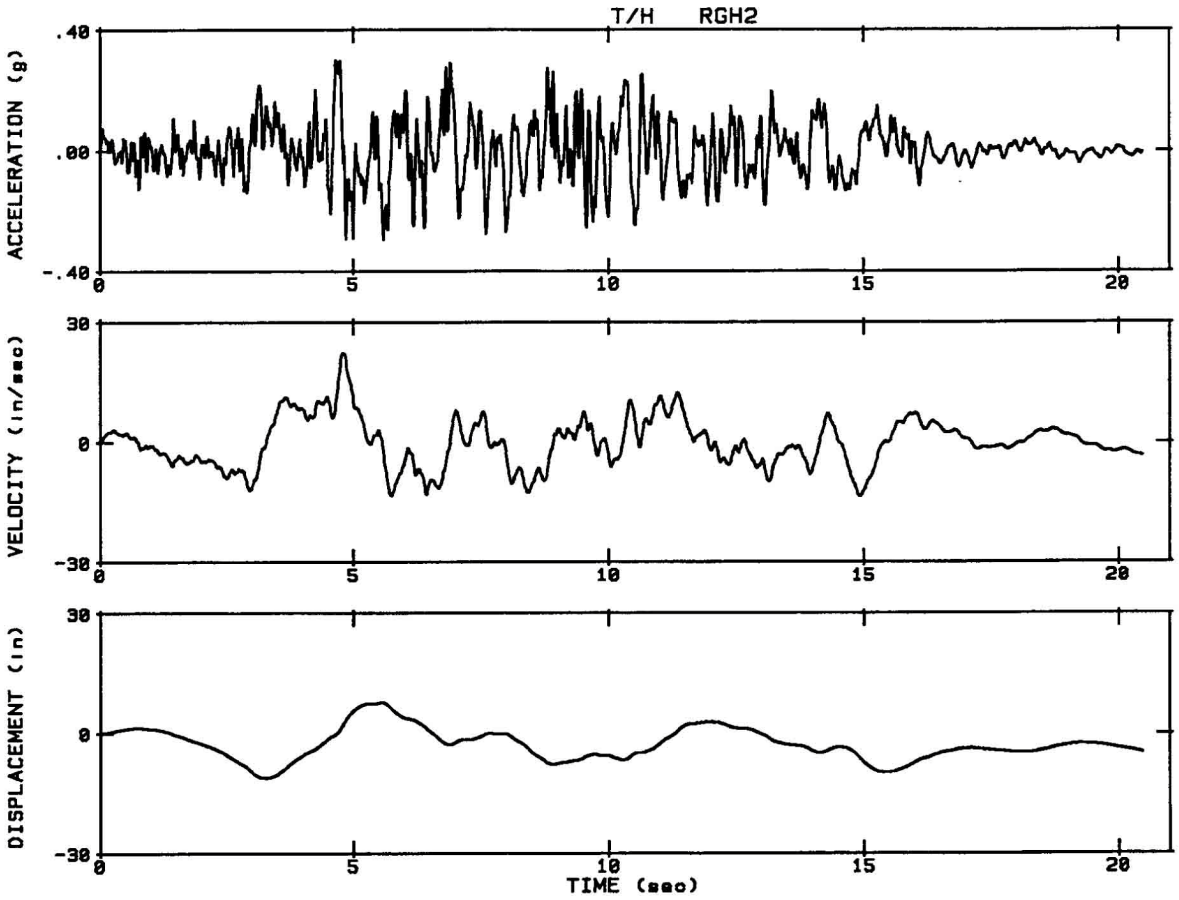


Figure 3.7.1-4
Design Horizontal Time History, "H2"
Acceleration, Velocity & Displacement Plots

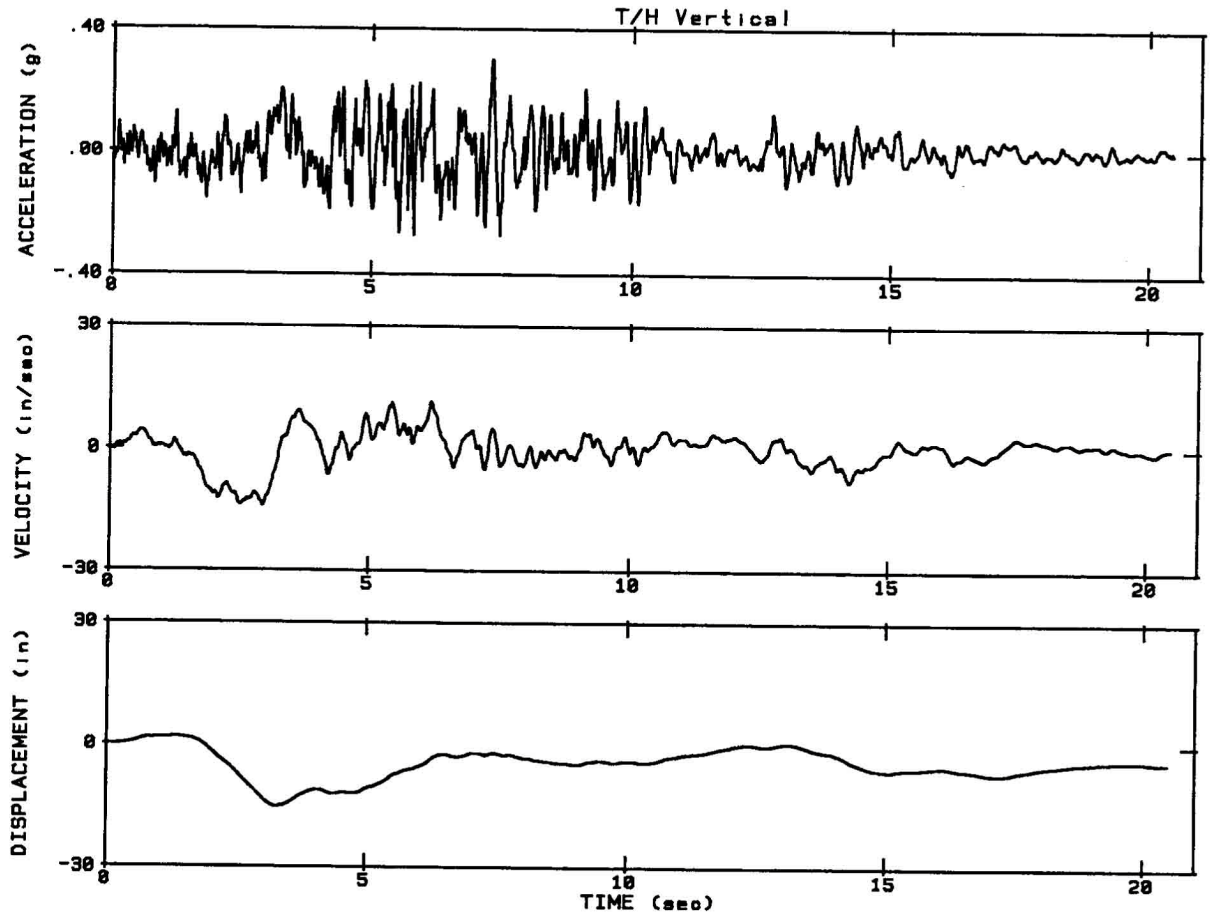


Figure 3.7.1-5
Design Vertical Time History
Acceleration, Velocity & Displacement Plots

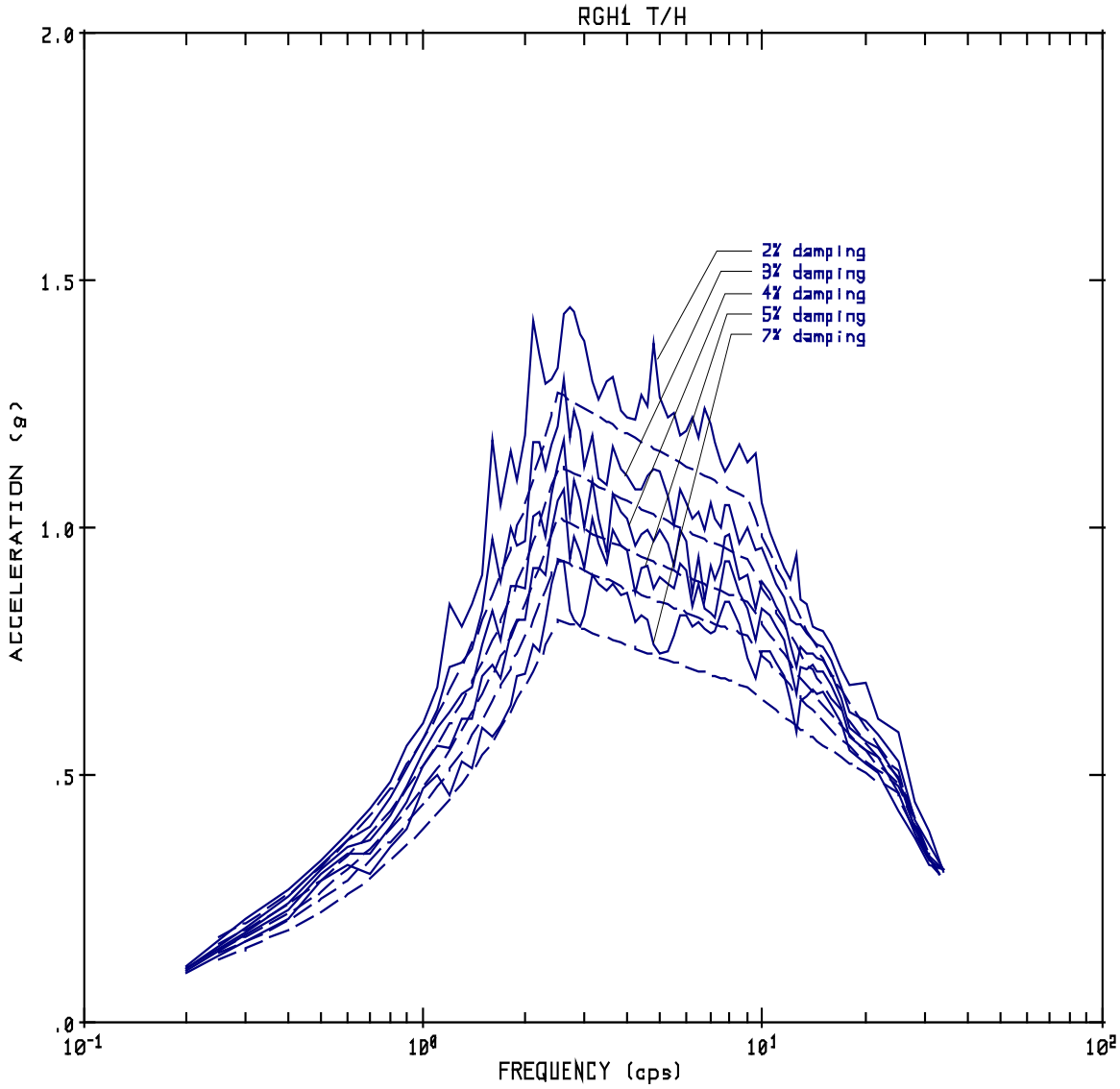


Figure 3.7.1-6
Acceleration Response Spectra of
Design Horizontal Time History, "H1"

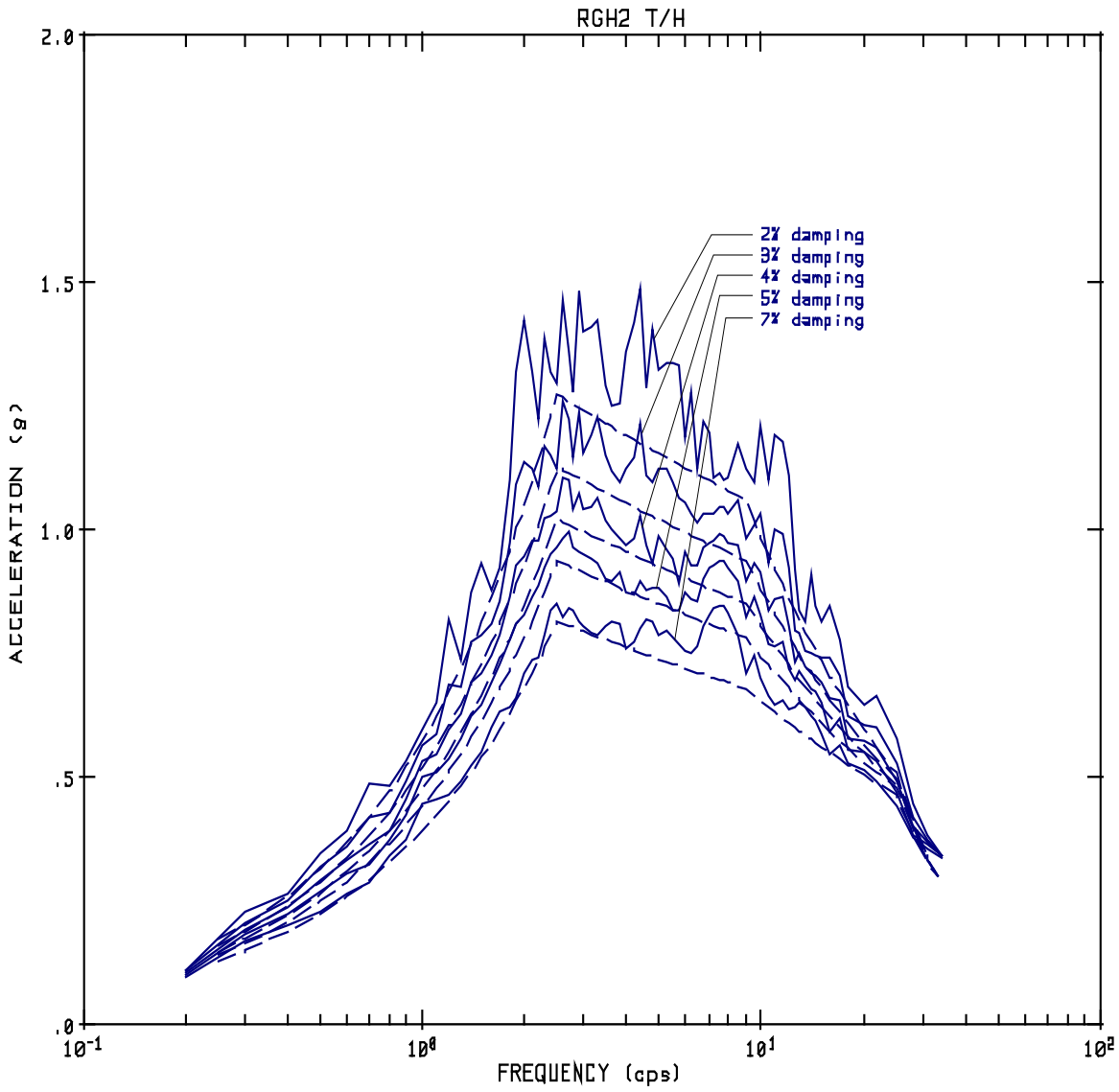


Figure 3.7.1-7
Acceleration Response Spectra of
Design Horizontal Time History, "H2"

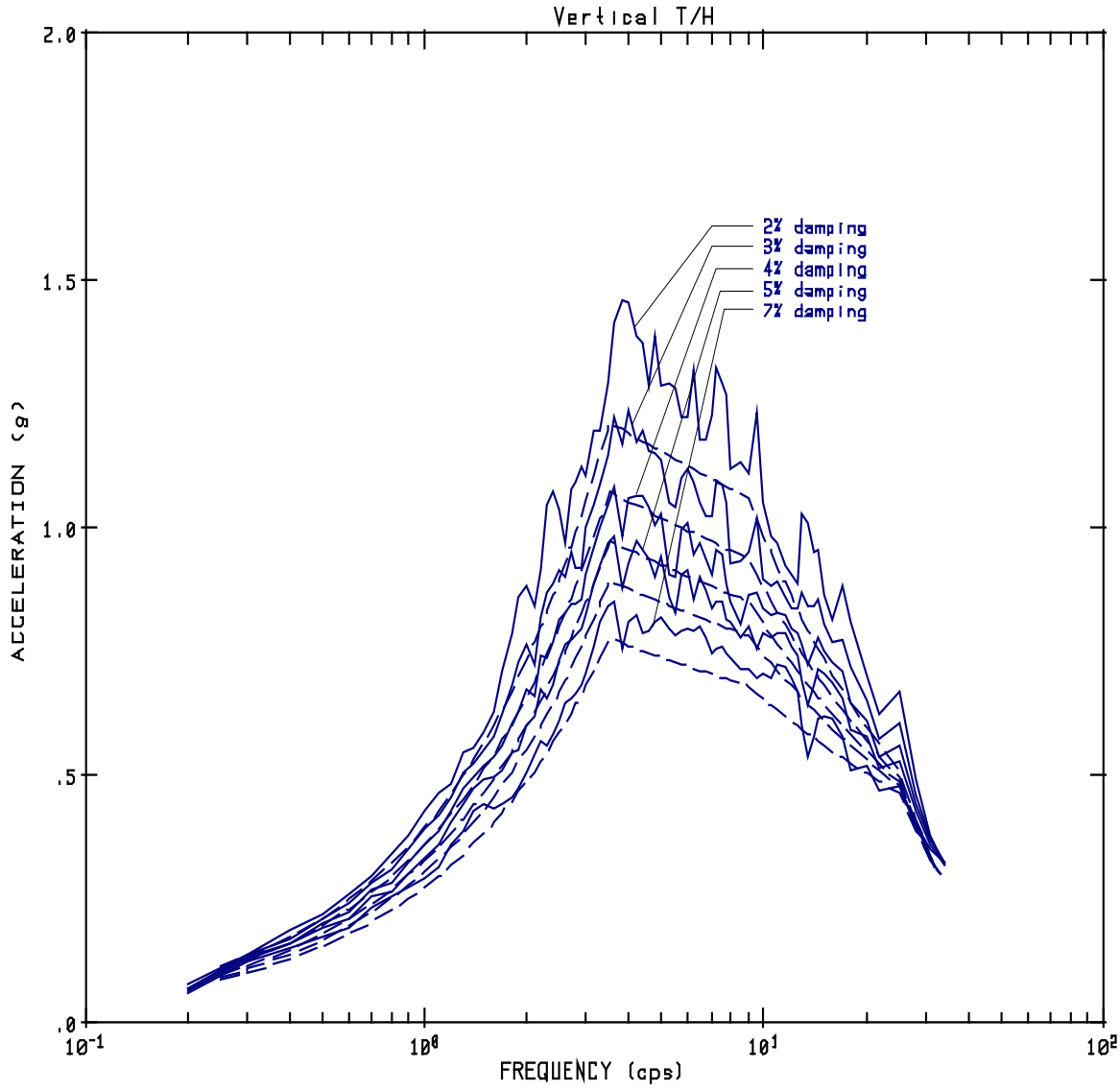


Figure 3.7.1-8
Acceleration Response Spectra of
Design Vertical Time History

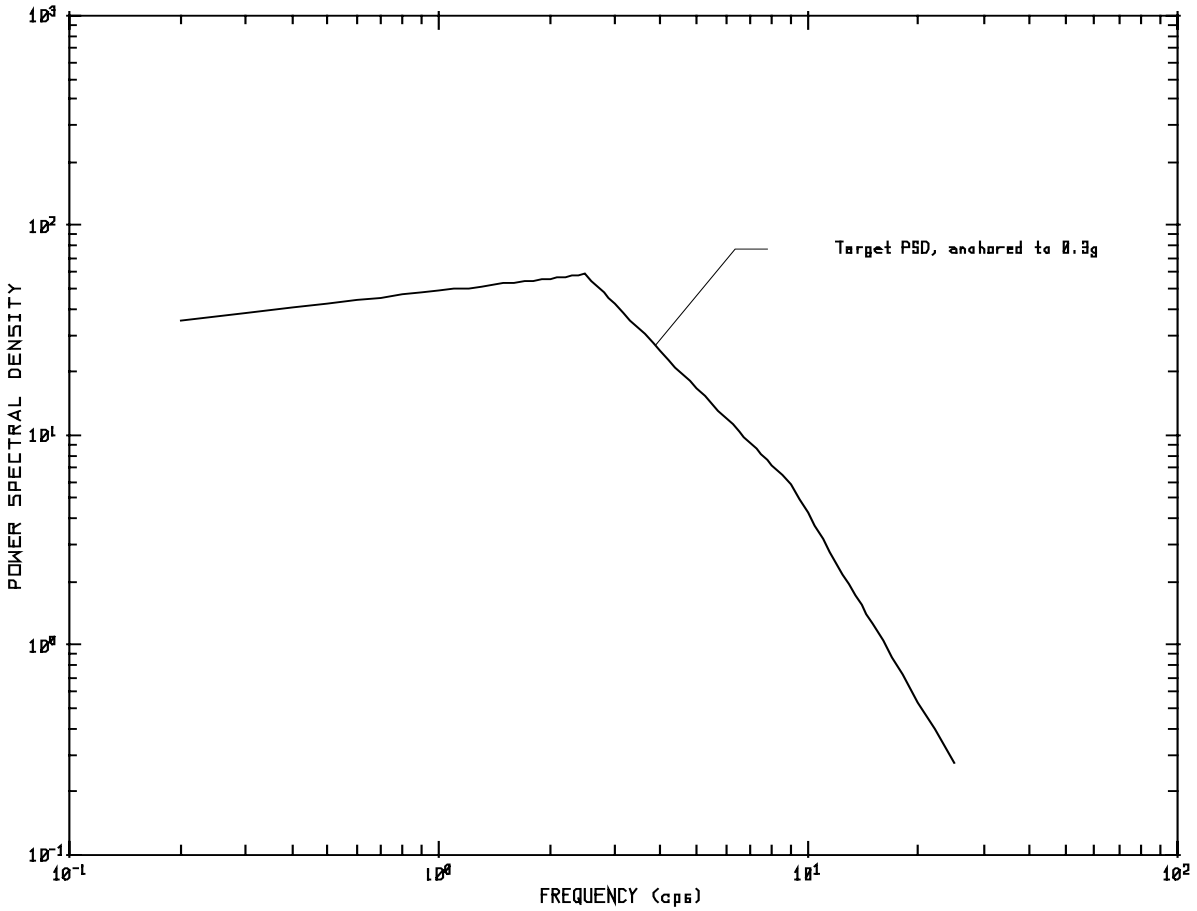


Figure 3.7.1-9
Minimum Power Spectral Density Curve
(Normalized to 0.3g)

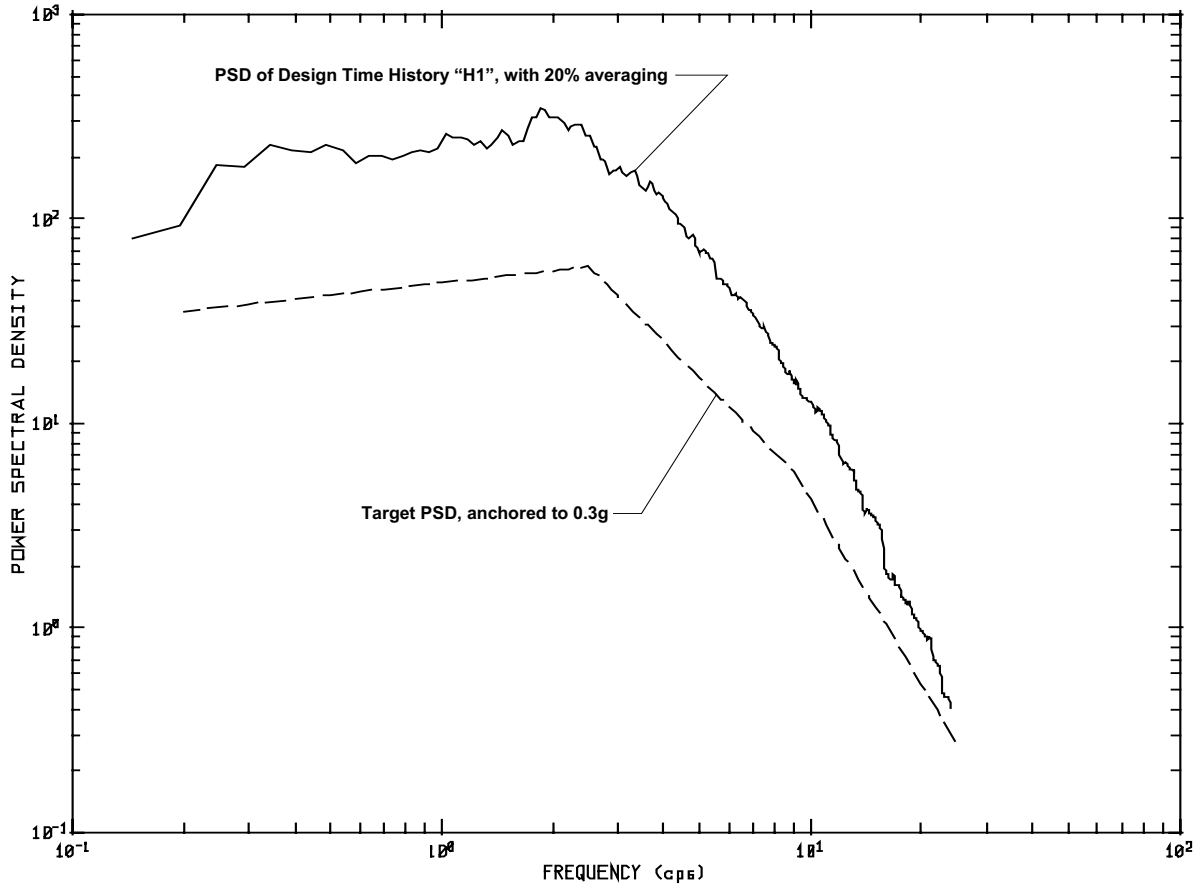


Figure 3.7.1-10
Power Spectral Density of
Design Horizontal Time History, "H1"

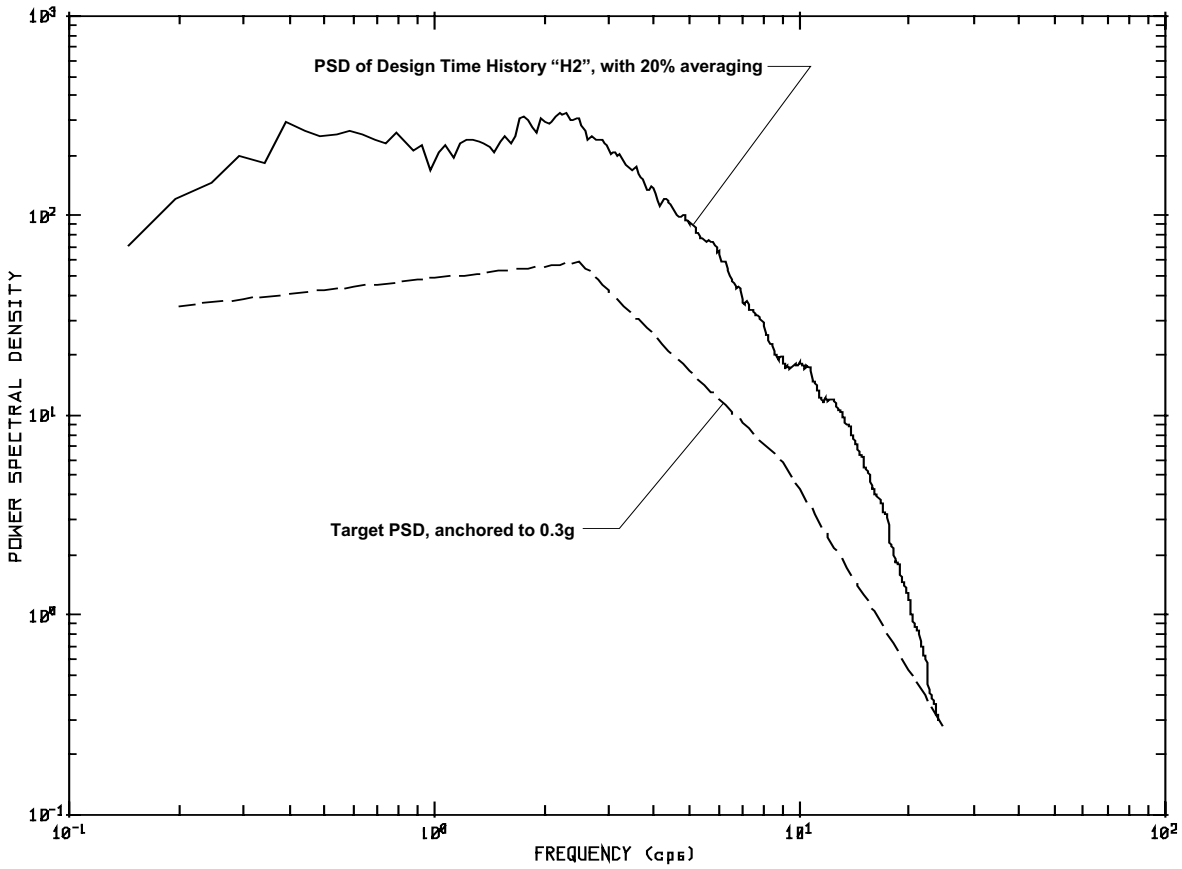


Figure 3.7.1-11
Power Spectral Density of
Design Horizontal Time History, "H2"

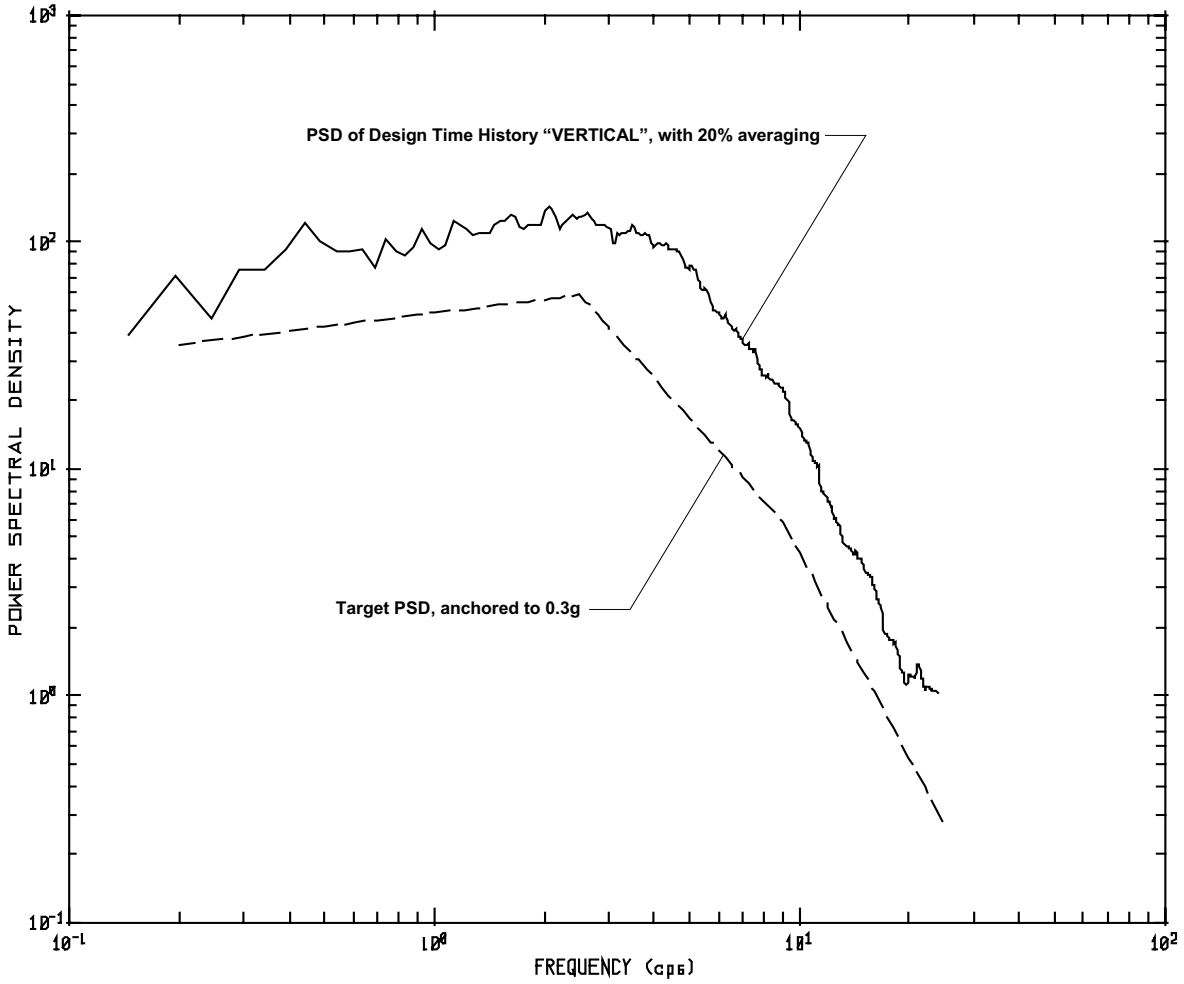


Figure 3.7.1-12
Power Spectral Density of
Design Vertical Time History

Figure 3.7.1-13 Not Used

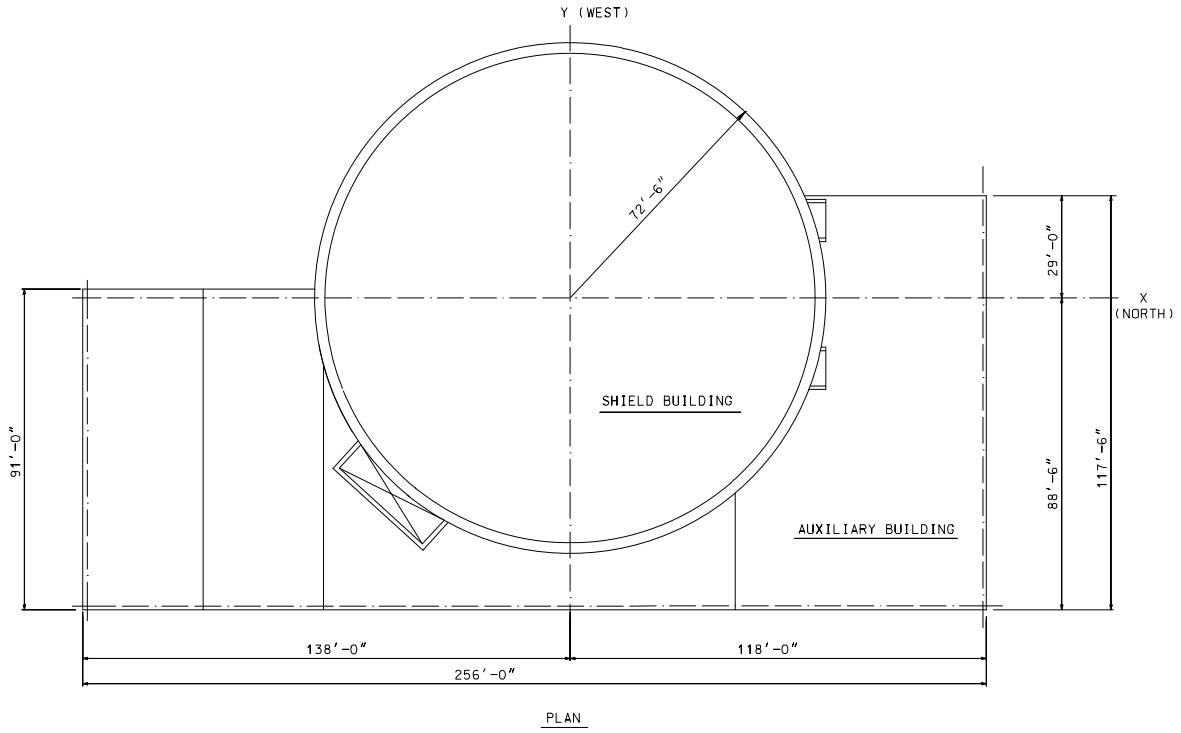
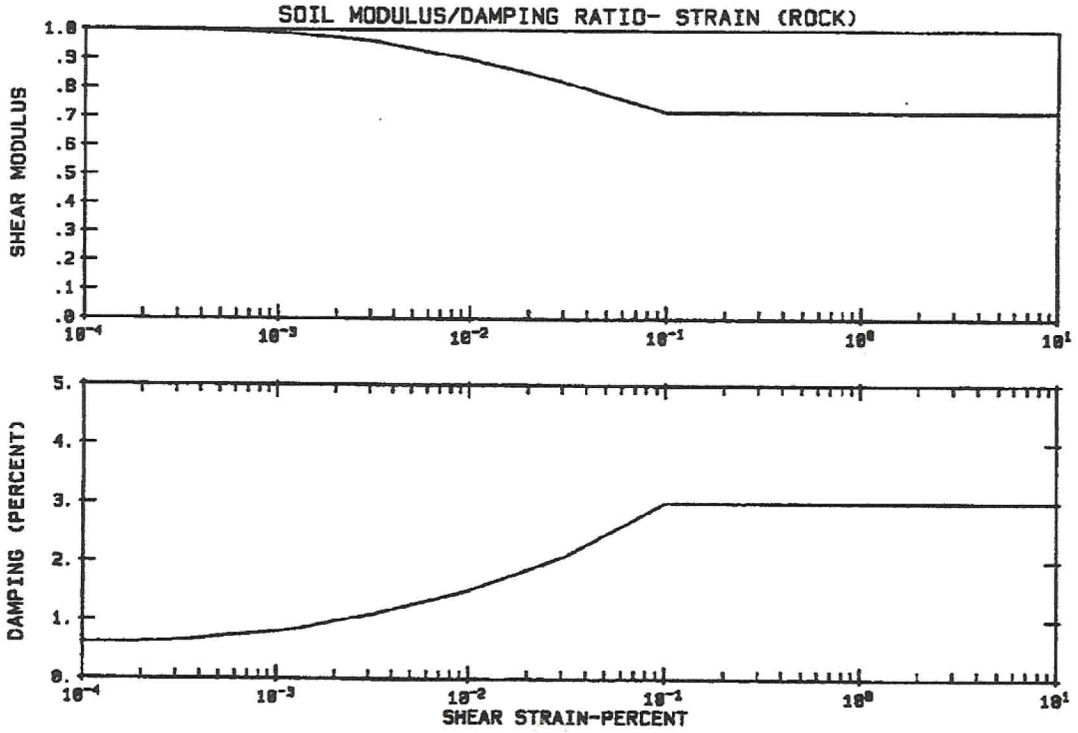


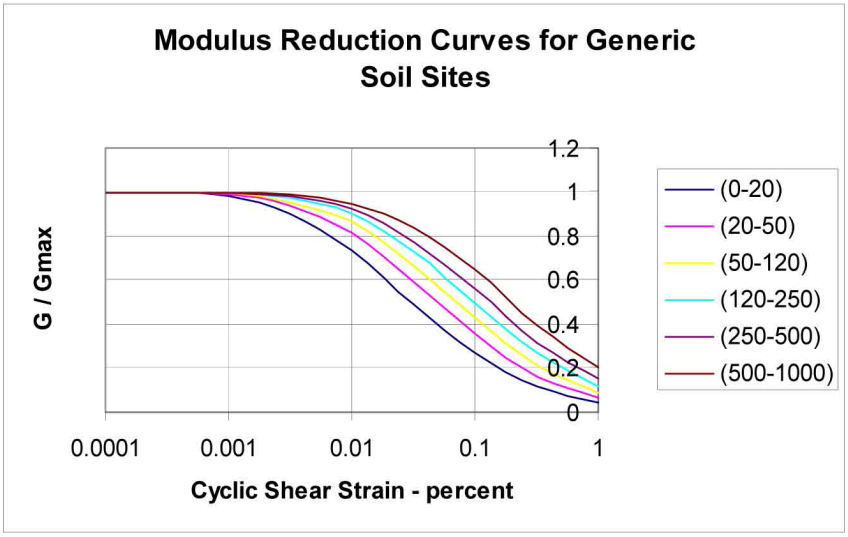
Figure 3.7.1-14
[*Nuclear Island Structures Dimensions*]*

*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)



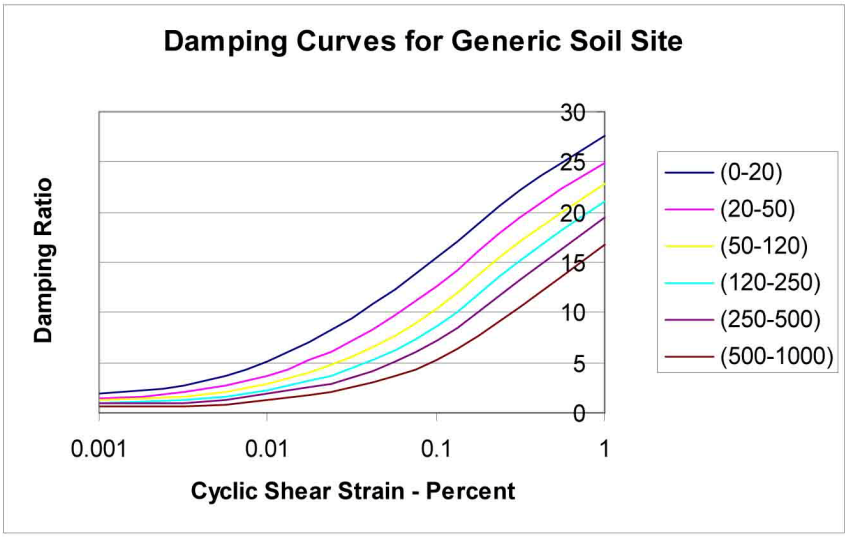
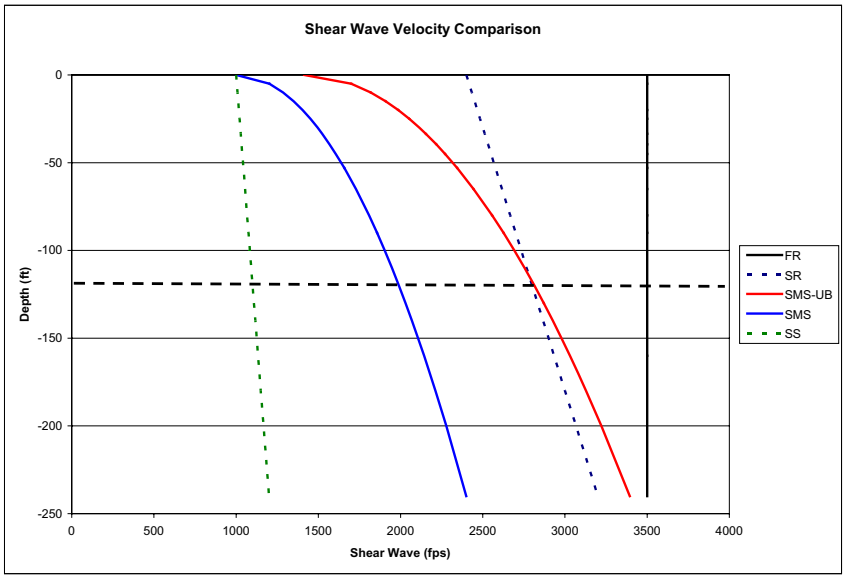
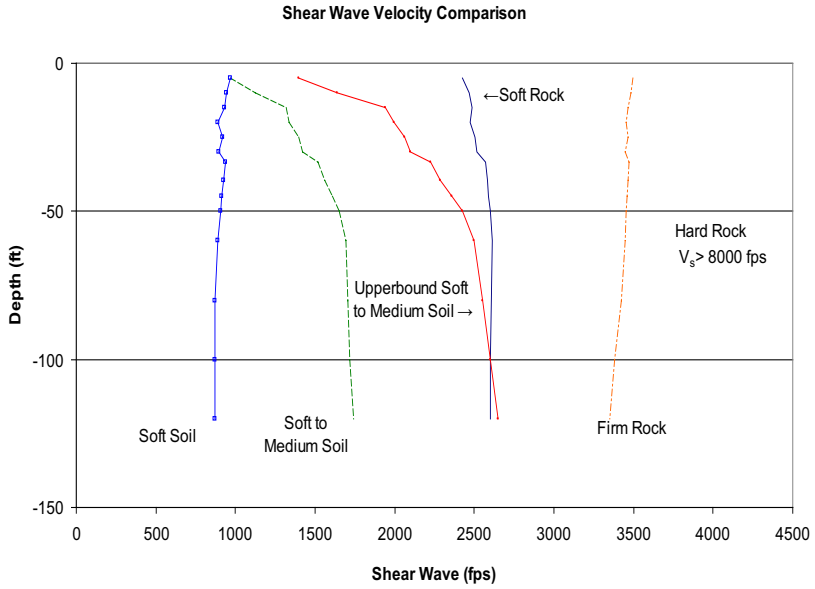


Figure 3.7.1-16
Strain Dependent Properties of Soil Material (Ref. 38)



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

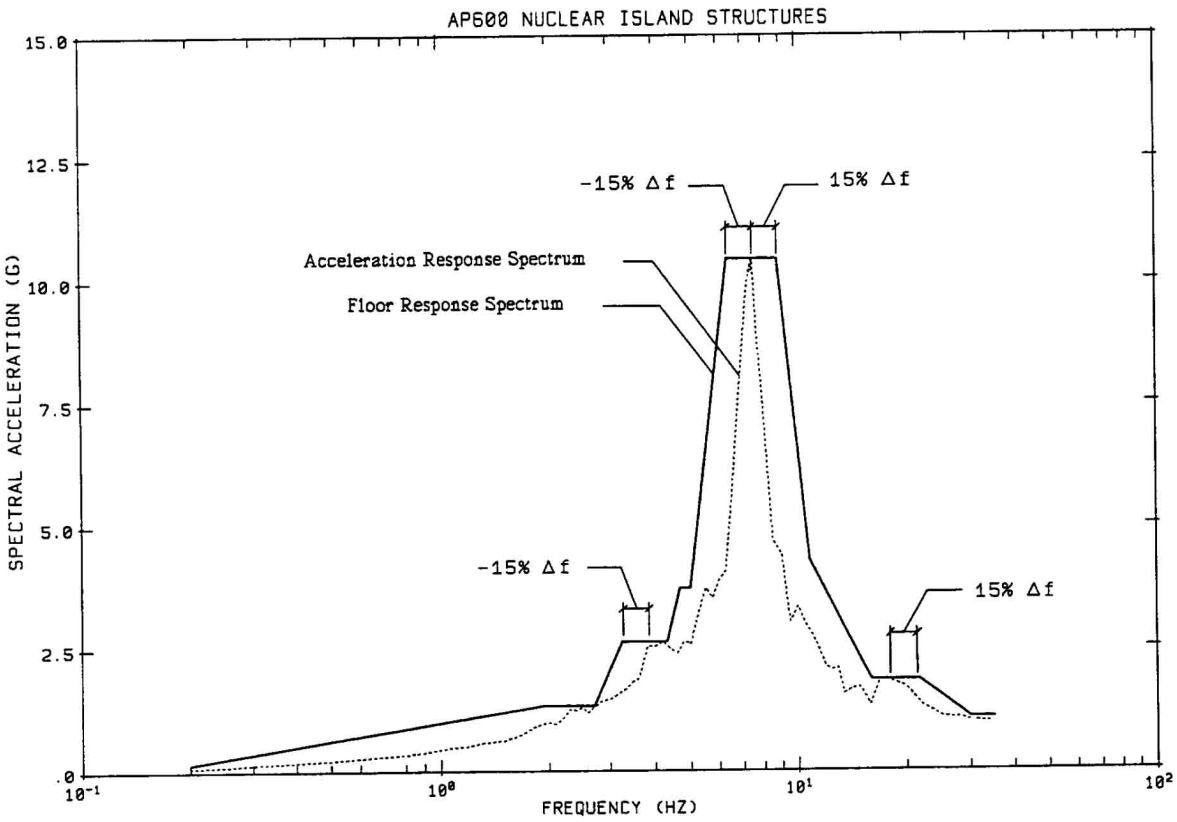


Figure 3.7.2-14
Typical Design Floor Response Spectrum

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

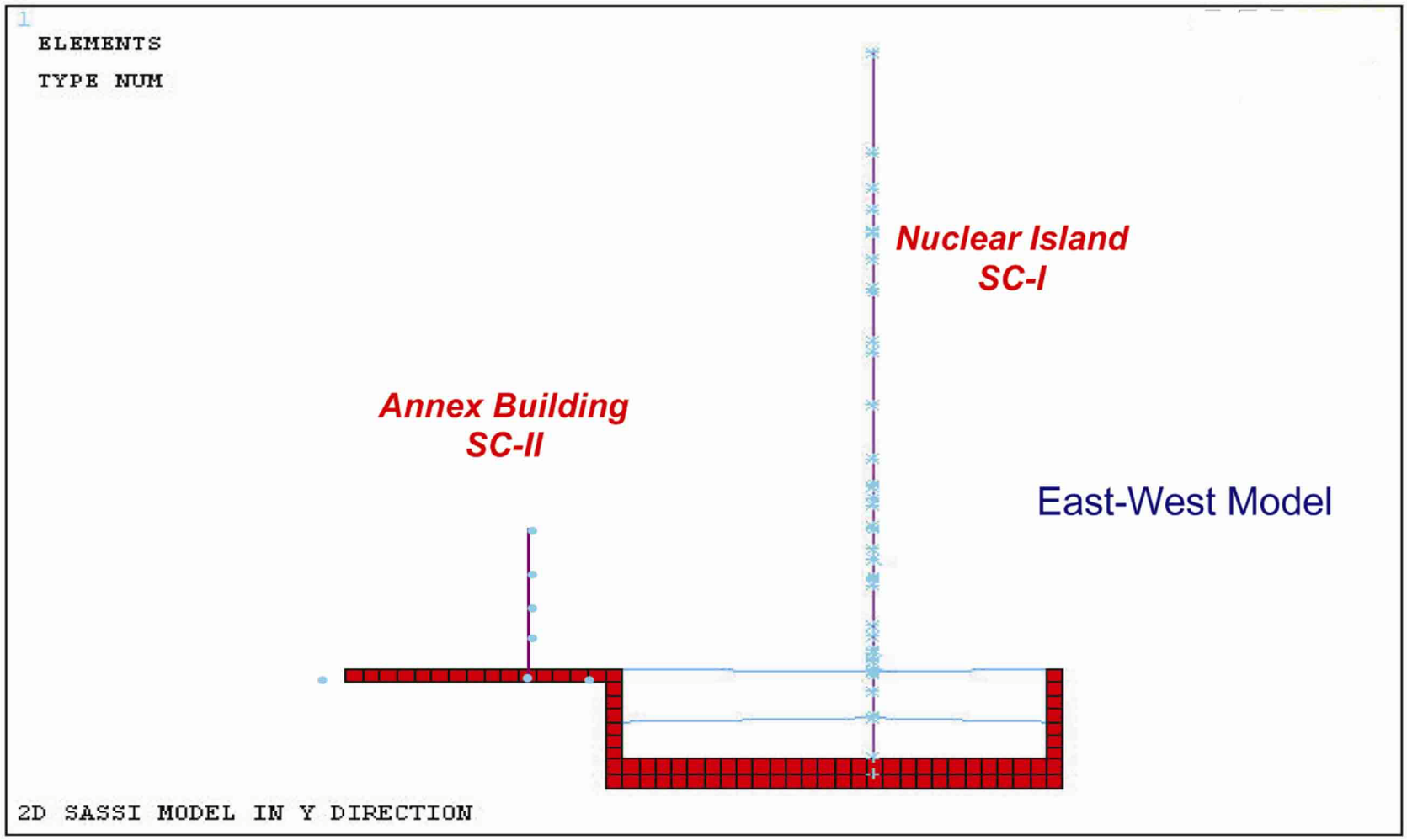


Figure 3.7.2-20
East-West 2D SASSI Model with Adjacent Buildings

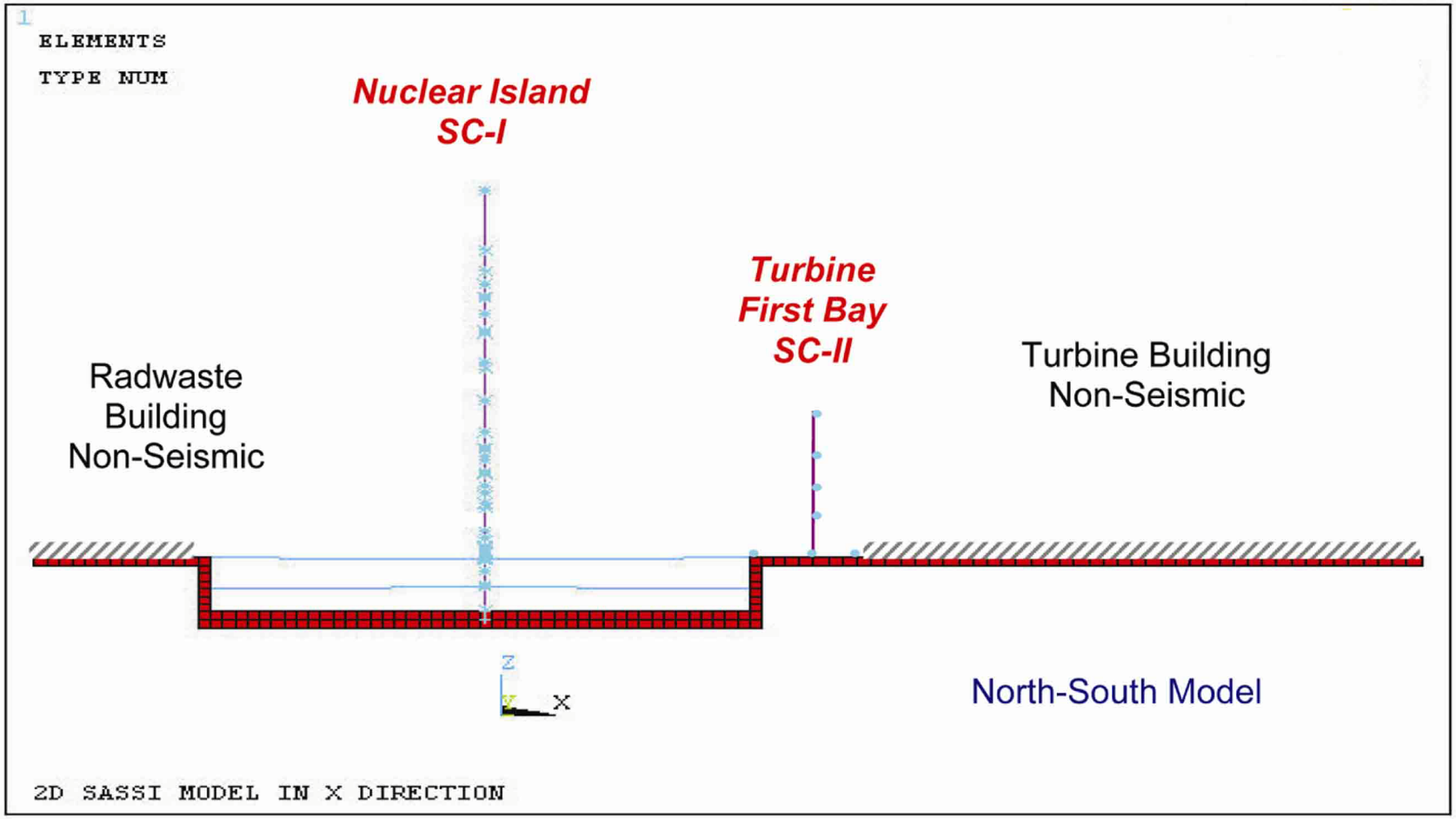


Figure 3.7.2-21
2D North-South SASSI Model with Adjacent Buildings

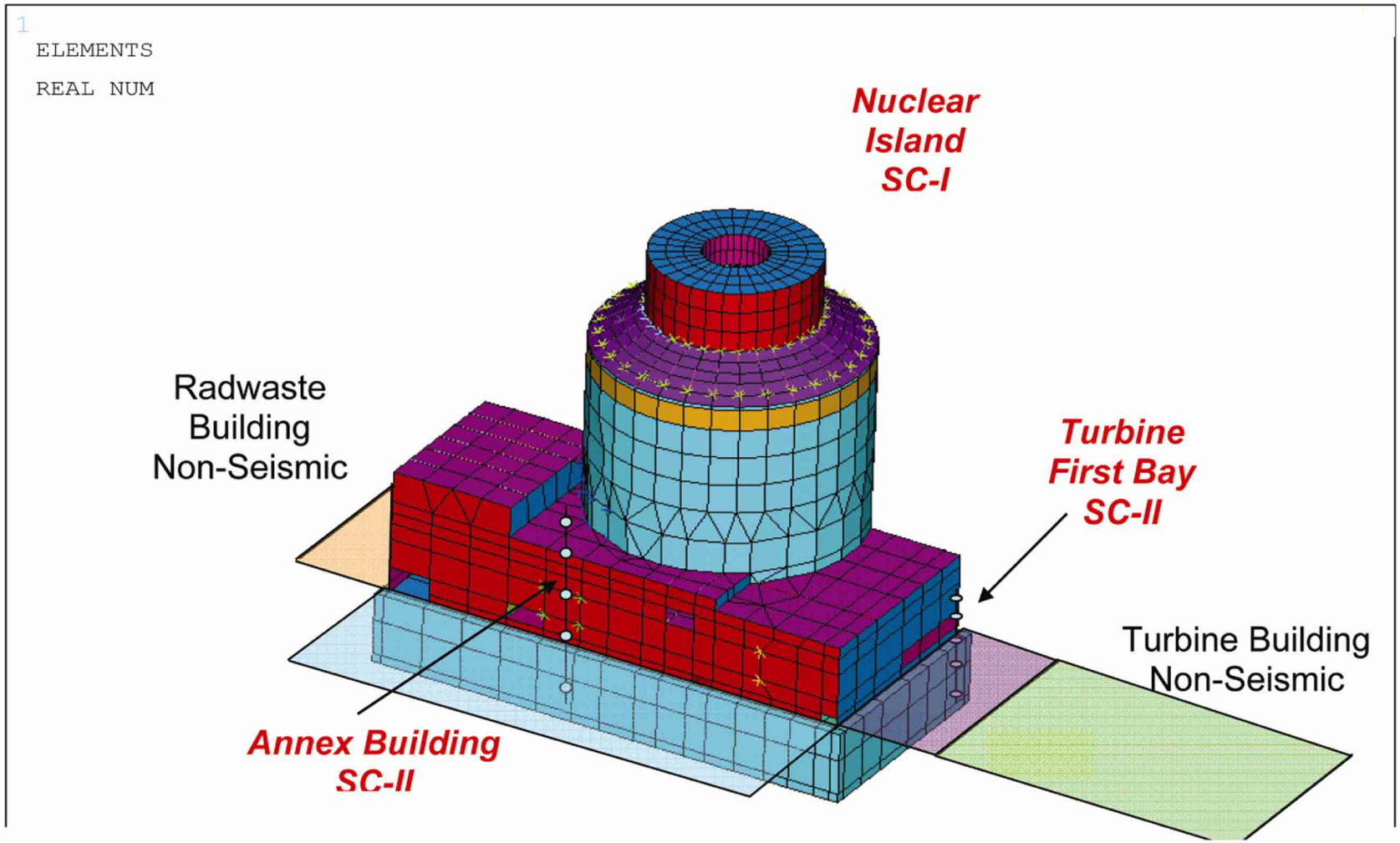


Figure 3.7.2-22
3D SASSI Model with Adjacent Buildings

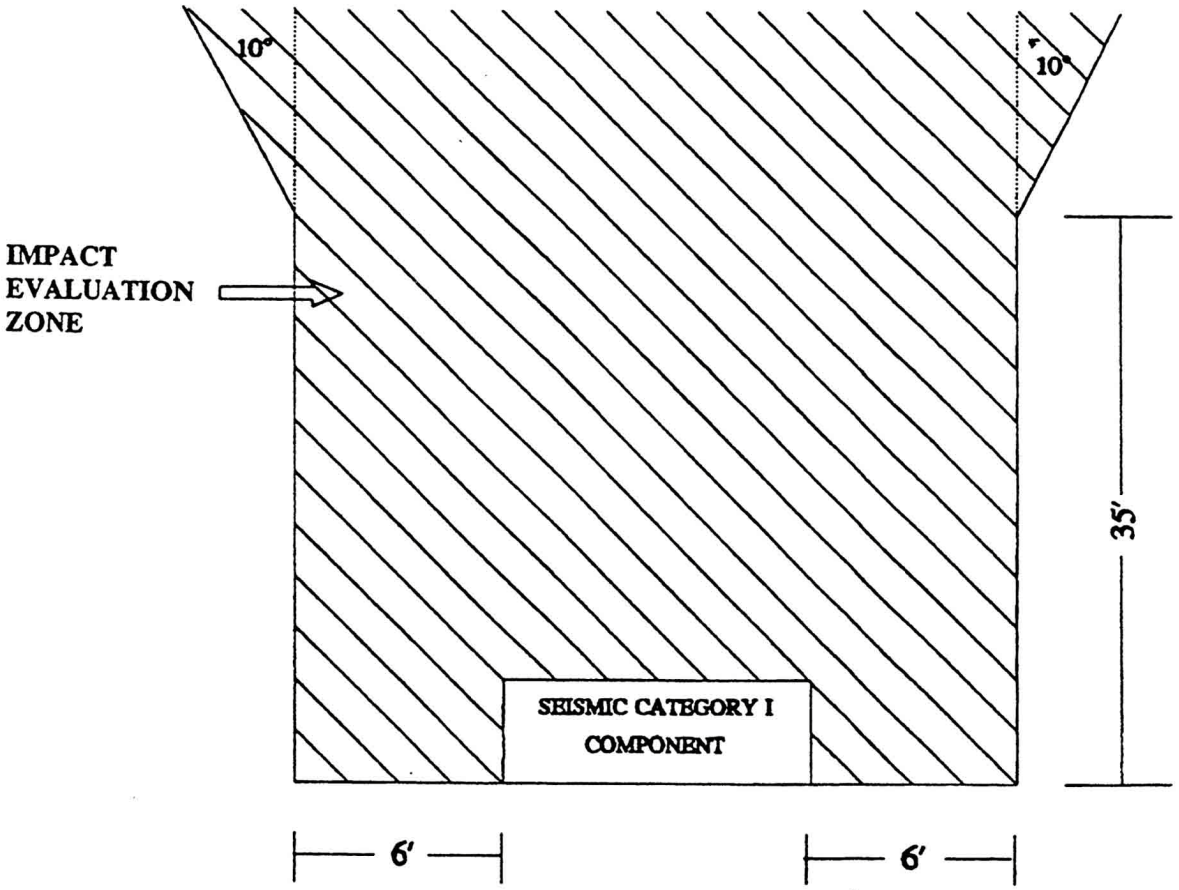


Figure 3.7.3-1
Impact Evaluation Zone

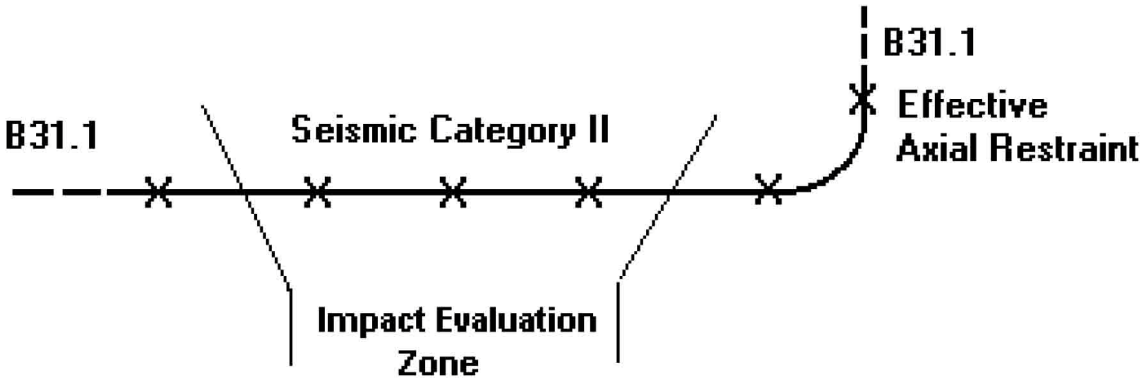


Figure 3.7.3-2
Impact Evaluation Zone and Seismic Supported Piping

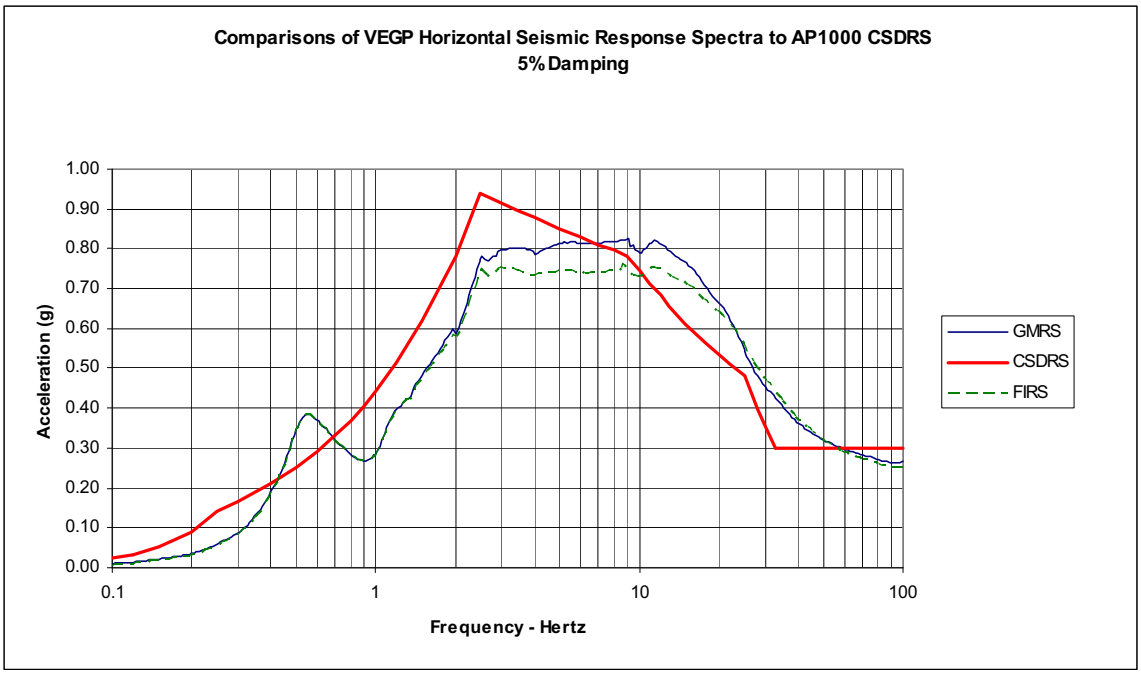


Figure 3.7-201
VEGP AP1000 Horizontal Spectra Comparison

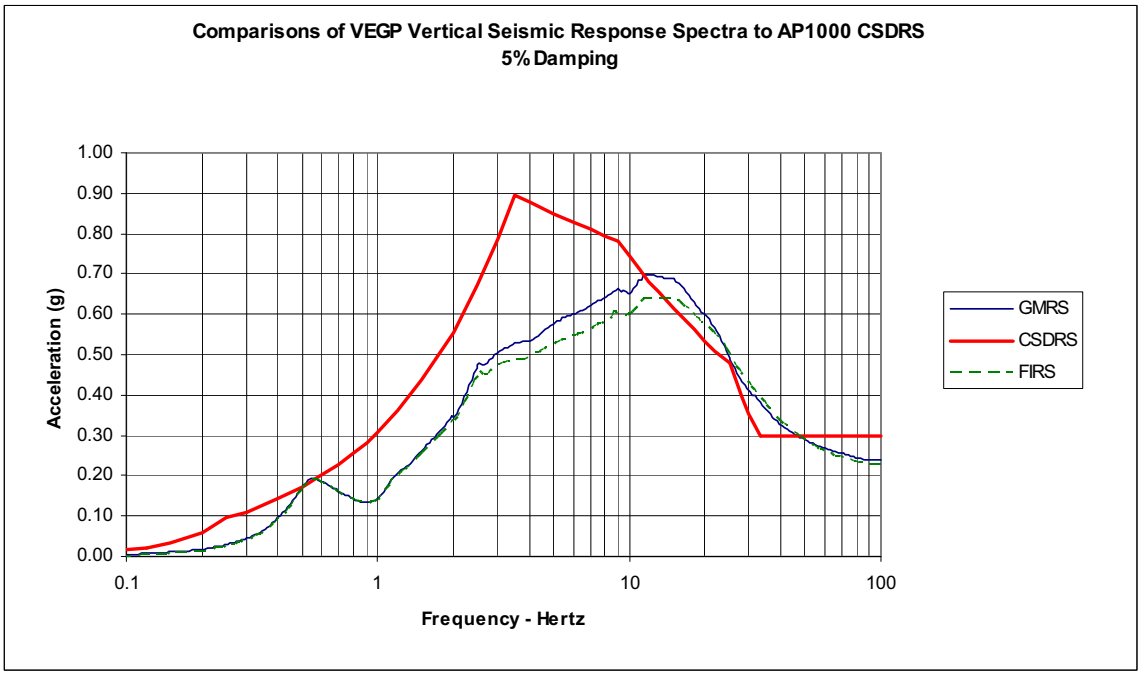


Figure 3.7-202
VEGP AP1000 Vertical Spectra Comparison