Chapter 3 Design of Structures, Components, Equipment, and Systems

3.1 Conformance with NRC General Design Criteria

This section contains an evaluation of the principal design criteria of the ESBWR Standard Plant as measured against the Nuclear Regulatory Commission (NRC) General Design Criteria (GDC) for Nuclear Power Plants, 10 CFR 50 Appendix A. The GDC are intended to establish minimum requirements for the principal design criteria for nuclear power plants.

The NRC GDC are intended to guide the design of water-cooled nuclear power plants; separate Boiling Water Reactor (BWR) specific criteria are not addressed. As a result, the criteria are subject to a variety of interpretations. For this reason, in some cases conformance to a particular criterion is not directly measurable. In these cases, the conformance of the ESBWR design to the interpretation of the criteria is discussed. For each criterion, the ESBWR design is specifically assessed and a complete list of references is included to identify where detailed design information pertinent to that criterion is treated in this Design Control Document (DCD).

3.1.1 Group I — Overall Requirements

3.1.1.1 Criterion 1 — Quality Standards and Records

Criterion 1 Statement

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions 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 shall 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 nuclear power unit licensee throughout the life of the unit.

Evaluation Against Criterion 1

Safety-related and nonsafety-related Structures, Systems, or Components (SSCs) are identified in Table 3.2-1. The quality assurance program is described in Chapter 17 and applies to the safety-related items. Nonsafety-related items are also controlled by the quality assurance program described in Chapter 17 in accordance with the functional importance of the item. The intent of the quality assurance program is to assure sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license

application. In addition, the quality assurance program assures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. The quality assurance program also includes the observance of proper preoperational and operational testing and maintenance procedures as well as the appropriate documentation. The quality assurance program is responsive to and in conformance with the intent of the quality-related requirements of 10 CFR 50 Appendix B.

SSCs are identified in Section 3.2 with respect to their location, service, and their relationship to the safety-related or nonsafety-related function to be performed. Applicable codes and standards are applied to the equipment commensurate with their safety-related function.

Documents are maintained to demonstrate that the requirements of the quality assurance program are satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are identified, correct materials are specified, correct procedures are utilized, qualified personnel are provided, and the finished parts and components meet the applicable specifications. These records are available so that any desired item of information is retrievable for reference. These records are maintained for the life of the operating licenses.

The quality program and records meet Criterion 1. For further discussion, see the following sections:

Chapter/Section	Title
3.2	Classification of Structures, Components, and Systems
3.9	Mechanical Systems and Components
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment
3.11	Environmental Qualification of Mechanical and Electrical Equipment
5	Reactor Coolant System and Connected Systems
6	Engineered Safety Features
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.1.5	Overhead Heavy Load Handling Systems
9.3	Process Auxiliaries
17	Quality Assurance

3.1.1.2 Criterion 2 — Design Bases for Protection Against Natural Phenomena

Criterion 2 Statement

Structures, systems, and components important to safety shall be designed to withstand the effect of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches

without loss of 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 combination of the effects of normal and accident conditions with the effects of the natural phenomena.
3. the importance of the safety functions to be performed.

Evaluation Against Criterion 2

The ESBWR design is designated as a standard plant, so the design bases for safety-related SSCs may not have been evaluated against the most severe of the natural phenomena that have been historically reported for each possible site and its surrounding area. The envelope of the site-related parameters, which encompass the majority of the potential sites in the contiguous United States is defined in Chapter 2. The design bases for safety-related SSCs reflect this envelope of natural phenomena including appropriate combinations of the effects of normal and accident conditions within this envelope.

The design bases for safety-related SSCs meet the requirements of Criterion 2. Detailed discussions of various phenomena considered and design criteria developed are presented in the following sections:

Chapter/Section	Title
2.0	Site Characteristics
3.2	Classification of Structures, Systems, and Components
3.3	Wind and Tornado Loadings
3.4	Water Level (Flood) Design
3.5	Missile Protection
3.7	Seismic Design
3.8	Seismic Category I Structures
3.9	Mechanical Systems and Components
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment
3.11	Environmental Qualification of Mechanical and Electrical Equipment
5.2	Integrity of Reactor Coolant Pressure Boundary
5.4.6	Isolation Condenser System
5.4.7	Residual Heat Removal System
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
6.2.2	Passive Containment Cooling System

6.2.4	Containment Isolation Function
6.3	Emergency Core Cooling Systems
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
8.1.5.2	Onsite Power
8.3	Onsite Power Systems
9	Auxiliary System
19A	Regulatory Treatment of Non-Safety Systems

3.1.1.3 Criterion 3 — Fire Protection

Criterion 3 Statement

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. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of the structures, systems, and components.

Evaluation Against Criterion 3

Fires in the plant are prevented or mitigated by the use of noncombustible and heat-resistant materials such as metal cabinets, metal wireways, high melting point insulation, and flame resistant markers for identification wherever practicable.

Cabling is suitably rated and cable tray loading is designed to avoid unacceptable internal heat buildup. Cable trays are suitably separated to avoid the loss of redundant channels of protective cabling if a fire occurs. The arrangement of equipment in reactor protection channels provides physical separation to limit the effects of fire.

Combustible supplies, such as logs, records, manuals, etc., are limited in such areas as the control room, thus limiting the potential of a fire.

The plant Fire Protection System (FPS) includes the following provisions:

- Automatic fire detection equipment in those areas where fire danger is greatest.
- A trained fire brigade.
- Suppression services which include suppression systems with automatic actuation with manual override as well as manually-operated fire extinguishers.

The design of the FPS meets the requirements of Criterion 3. For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
9.5.1	Fire Protection System
11.3	Gaseous Waste Management System

3.1.1.4 Criterion 4 — Environmental and Dynamic Effects Design Bases

Criterion 4 Statement

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 plant 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.

Evaluation Against Criterion 4

Safety-related SSCs are designed to accommodate the dynamic effects of, and to be compatible with, environmental conditions associated with normal operation, maintenance, and postulated pipe failure accidents including Loss-of-Coolant-Accidents (LOCA).

Safety-related SSCs are appropriately protected against dynamic effects including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failure. The effects of missiles originating outside the ESBWR Standard Plant are also considered. Design requirements specify the duration that safety-related SSCs must survive the environmental conditions following a LOCA.

The design of safety-related SSCs meets the requirements of Criterion 4. For further discussion, see the following sections:

Chapter/Section	Title
2.0	Site Characteristics
3.3	Wind and Tornado Loadings
3.4	Water Level (Flood) Design
3.5	Missile Protection

3.6	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping
3.8	Seismic Category I Structures
3.9	Mechanical Systems and Components
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment
3.11	Environmental Qualification of Mechanical and Electrical Equipment
4.6	Functional Design of Reactivity Control System
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
5.4.6	Isolation Condenser System
5.4.7	Residual Heat Removal System
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
6	Engineered Safety Features
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
8	Electric Power
9	Auxiliary System
10.2.1	Turbine Generator

3.1.1.5 **Criterion 5 — Sharing of Structures, Systems, and Components**

Criterion 5 Statement

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing does 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 units.

Evaluation Against Criterion 5

There are no shared SSCs because the ESBWR Standard Plant is a single-unit station; the requirements of Criterion 5 are met.

3.1.2 **Group II — Protection by Multiple Fission Product Barriers**

3.1.2.1 **Criterion 10 — Reactor Design**

Criterion 10 Statement

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.

Evaluation Against Criterion 10

The reactor core components consist of fuel assemblies, control rods, in-core ion chambers, neutron sources, and related items. The mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The fuel is designed to maintain integrity over a complete range of power levels, including anticipated operational occurence (AOO) transient conditions. The core is sized with sufficient heat transfer area and coolant flow to ensure that specified acceptable fuel design limits are not exceeded under normal conditions or AOOs.

The safety-related Reactor Protection System (RPS) is designed to monitor certain reactor parameters, sense abnormalities, and to scram the reactor, thereby preventing specified acceptable fuel design limits from being exceeded. Scram setpoints are based on safety design basis analyses and setpoint methodology. There is no normal operation or AOO condition from which the scram setpoints allow the reactor core to exceed the specified acceptable safety limits.

AOO analyses are presented in Chapter 15. The results show that the minimum critical power ratio does not fall below the safety limit minimum critical power ratio, thereby satisfying the transient design basis.

The reactor core and associated coolant, control, and protection systems are designed to assure that the specified fuel design limits are not exceeded during conditions of normal or abnormal operation and, therefore, meet the requirements of Criterion 10. For further discussion, see the following sections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
15	Safety Analysis

3.1.2.2 Criterion 11 — Reactor Inherent Protection

Criterion 11 Statement

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.

Evaluation Against Criterion 11

The reactor core is designed to have responses that regulate or dampen changes in power level and spatial distribution of power production to a level consistent with safe and efficient operation.

The inherent dynamic behavior of the core is characterized in terms of:

- Fuel temperature or Doppler reactivity coefficient
- Moderator void reactivity coefficient
- Moderator temperature reactivity coefficient

The combined effect of these coefficients in the power range is termed the power coefficient.

A negative Doppler reactivity coefficient is maintained for any operating condition. Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it; it contributes to system stability.

A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels is maintained for any operating condition. The negative void reactivity coefficient provides an inherent negative feedback during power transients. Because of the large negative moderator void reactivity coefficient, the ESBWR has a number of inherent advantages, such as:

- The inherent self-flattening of the radial power distribution
- The ease of control
- The spatial xenon stability

The reactor is designed so that the moderator temperature reactivity coefficient is negative above hot standby, and the overall power reactivity coefficient is negative, well within the range required for adequate damping of power and spatial xenon disturbances.

The reactor core and associated coolant system are designed so that in the power operating range, prompt inherent dynamic behavior compensates for any rapid increase in reactivity in accordance with Criterion 11. For further discussion, see the following sections:

Chapter/Section	Title
4.3	Nuclear Design

3.1.2.3 Criterion 12 — Suppression of Reactor Power Oscillations

Criterion 12 Statement

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.

Evaluation Against Criterion 12

The ESBWR is designed to be inherently stable, and in addition, it includes control and protection systems designed to ensure that power oscillations that could result in exceeding specified acceptable fuel design limits are reliably and readily detected and suppressed. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler reactivity coefficient, moderator reactivity void coefficient and moderator temperature reactivity coefficient.

The power reactivity coefficient is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Operating experience has shown large BWRs to be inherently stable against xenon induced power instability. The negative reactivity coefficients also provide:

- Good load following with well-damped behavior and little undershoot or overshoot in the heat transfer response.
- Strong damping of spatial power disturbances.

ESBWR stable operation is developed by establishing sufficiently high natural circulation flow through inherent design features such as shorter length fuel to reduce core pressure drop and the addition of a tall chimney above the core to promote natural circulation. Power fluctuations subject to coupled neutronic-thermal-hydraulic feedback are inherently damped under the high natural circulation flow operating conditions.

The Neutron Monitoring System in conjunction with the RPS design provides further protection from coupled neutronic-thermal-hydraulic instability. Core wide and local oscillations abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of this protection system is achieved through the combination of logic arrangement, trip channel redundancy, power supply redundancy, and physical separation.

The combination of inherently stable design and the instability detection and suppression systems assure that Criterion 12 is met. For further discussions, see the following sections:

Chapter/Section	Title
4.3	**Nuclear Design**
4.4	**Thermal and Hydraulic Design**
4D	**Stability Evaluation**
Table 7.1-1	**I&C Regulatory Requirements Applicability Matrix**

3.1.2.4 Criterion 13 — Instrumentation and Control

Criterion 13 Statement

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.

Evaluation Against Criterion 13

Modern proven BWR instrumentation and controls are provided in the ESBWR Standard Plant design. The neutron flux in the reactor core is monitored by four subsystems. The Startup Range Neutron Monitor (SRNM) Subsystem measures the flux from startup through 15% power (into the power range). The power range is monitored by many detectors which make up the Local Power Range Monitor (LPRM) Subsystem. The output of these detectors is used in many ways. The output of selected core-wide sets of detectors is averaged to provide a core-average neutron flux. This output is called the Average Power Range Monitor (APRM) Subsystem. The Automated Fixed In-core Probe (AFIP) Subsystem provides a means for calibrating the LPRM. Both the SRNM and APRM Subsystems generate scram trips to the RPS. They also generate rod-block trips.

The RPS protects the fuel barriers and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined setpoints are exceeded. Separation of the scram and normal rod control function prevents failures in the reactor manual control circuitry from affecting the scram circuitry. To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and reactor coolant pressure boundary (RCPB), the Leak Detection and Isolation System (LD&IS) initiates automatic isolation of appropriate pipelines whenever monitored variables exceed pre-selected operational limits.

The LD&IS provides instrumentation and controls to detect, annunciate and, in some cases, isolate the RCPB to ensure its integrity. Also see the evaluation of GDC 30.

The Process Radiation Monitoring System (PRMS) monitors radiation levels of various processes and provides trip signals to the LD&IS whenever pre-established limits are exceeded.

Adequate instrumentation has been provided to monitor system variables in the reactor core, RCPB, and reactor containment. Appropriate controls have been provided to maintain the variables in the operating range and to initiate the necessary corrective action in the event of abnormal operational occurrence or accident.

The design of instrumentation and control systems meets the requirements of Criterion 13. For further discussions, see the following sections:

Chapter/Section	Title
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
6.2	Containment Systems
6.3	Emergency Core Cooling Systems
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9	Auxiliary Systems

3.1.2.5 Criterion 14 — Reactor Coolant Pressure Boundary

Criterion 14 Statement

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.

Evaluation Against Criterion 14

The piping and equipment pressure parts within the RCPB (as defined by Section 50.2 of 10 CFR 50) are designed, fabricated, erected, and tested in accordance with 10 CFR 50.55a to provide a high degree of integrity throughout the plant lifetime. Systems and components within the RCPB are classified as Quality Group A (Subsection 3.2.2.1). The RCPB is protected from overpressure by means of pressure relieving devices. The design requirements and codes and standards applied to this quality group help ensure high integrity in keeping with the safety-related function.

To minimize the possibility of brittle fracture within the RCPB, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Section 5.2 describes the methods utilized to control toughness properties of the RCPB materials. Materials are to be impact tested in accordance with American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code Section III, where applicable. Where RCPB piping penetrates the containment, the fracture toughness temperature requirements of the RCPB materials apply.

Piping and equipment pressure parts of the RCPB are assembled and erected by welding unless applicable codes permit flanged or threaded joints. Welding procedures are employed which produce welds of complete fusion that are free of unacceptable defects. All welding procedures, welders, and welding machine operators used in producing pressure containing welds are qualified in accordance with the requirements of the ASME B&PV Code Section IX for the materials to be welded. Qualifications records, including the results of procedure and performance qualification tests and identification symbols assigned to each welder, are maintained.

Section 5.2 contains the detailed material and examination requirements for the piping and equipment of the RCPB prior to and after its assembly and erection. Leakage testing and surveillance is accomplished as described in the evaluation against Criterion 30 of the GDC.

The design, fabrication, erection, and testing of the RCPB help assure an extremely low probability of abnormal leakage, thus satisfying the requirements of Criterion 14. For further discussion, see the following sections:

Chapter/Section	Title
3	Design of Structures, Components, Equipment, and Systems
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
5.4.12	Reactor Coolant System High Point Vents
6.1	Design Basis Accident Engineered Safety Feature Materials
9.3.2	Process Sampling System

3.1.2.6 Criterion 15 — Reactor Coolant System Design

Criterion 15 Statement

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 any condition of normal operation, including anticipated operational occurrences.

Evaluation Against Criterion 15

The RCS consists mainly of the reactor vessel and appurtenances, and the Nuclear Boiler System (NBS) including the main steamlines, feedwater lines and pressure-relief discharge system. The Isolation Condenser System (ICS), and portions of the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System, Gravity Driven Cooling System (GDCS), and Control Rod Drive (CRD) System are also part of the RCS.

The auxiliary, control, and protection systems associated with the RCS act to provide sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. As described in the evaluation of Criterion 13, instrumentation is provided to monitor variables to verify that they are within prescribed operating limits. If the monitored variables exceed their predetermined settings, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

An example of the integrated protective action scheme is the Isolation Condenser System (ICS). Upon receipt of an overpressure signal, the ICS automatically initiates to assure that the design conditions of the RCPB are not exceeded. In addition to the ICS, overpressure protection of the Reactor Pressure Vessel (RPV) system and RCPB is provided by pressure-operated safety relief valves (SRVs) that discharge steam from the main steamlines to the suppression pool. The pressure relief system also provides for automatic depressurization of the RCS in the event of a

LOCA in which the vessel is not depressurized by the accident. The depressurization of the RCS in this situation allows operation of the GDCS to supply enough cooling water to adequately cool the core.

In a similar manner, other auxiliary, control, and protection systems provide assurance that the design conditions of the RCPB are not exceeded during any conditions of normal operation, including AOOs, so that Criterion 15 is met. For further discussion, see the following sections:

Chapter/Section	Title
3	Design of Structure, Components, Equipment, and Systems
5.2.2	Overpressure Protection
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
15	Safety Analyses

3.1.2.7 Criterion 16 — Containment Design

Criterion 16 Statement

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.

Evaluation Against Criterion 16

The Primary Containment System consists of the following major structures and components:

- A leaktight containment vessel encloses the RPV, the RCPB, and other branch connections of the reactor primary coolant system. The containment vessel is a reinforced concrete cylindrical structure with an internal leaktight steel liner providing the primary containment boundary. The containment vessel structure consists of the drywell top slab, cylindrical containment wall, suppression pool floor slab, RPV pedestal, and the basemat. A steel drywell head closes the opening in the top of the containment vessel for servicing and refueling the RPV. The upper drywell encloses the upper portion of the RPV, the major piping systems (main steam, feedwater, GDCS, and ICS lines, SRVs, Depressurization Valves [DPVs]), Drywell Cooling System (DCS), GDCS pools, and other miscellaneous systems. The lower drywell encloses the lower portion of the RPV and encloses the cooling system ducts, fine motion control rod drives (FMCRDs), and other miscellaneous systems as well as providing maintenance space below the RPV.
- The wetwell includes the suppression pool, horizontal vents and airspace above the suppression pool. The water volume in the suppression pool serves as a heat sink to condense the steam released during a LOCA or SRV discharge. The airspace volume in the wetwell serves as the

blowdown reservoir for the nitrogen displaced from the upper and lower drywells during a LOCA after it passes through the horizontal vents and suppression pool.

- Associated containment penetrations and isolation devices.

The drywell and wetwell condense the steam and contain fission product releases from the postulated design basis accident (DBA) (i.e., the double-ended rupture of the largest pipe in the RCS). The leaktight containment vessel prevents the release of fission products to the environment.

Temperature and pressure in the containment vessel are limited following an accident by using the Passive Containment Cooling System (PCCS), an Engineered Safety Feature (ESF) system to condense steam in the containment atmosphere. Additionally, the Isolation Condensers (ICs) and the RWCU/SDC system can assist in cooling reactor steam and reactor water coolants following an accident. The Fuel and Auxiliary Pools Cooling System (FAPCS) and RWCU/SDC can be used to cool the suppression pool water. Safety analyses demonstrate that important containment parameters are maintained within design limits for as long as required.

The design of the containment structure and associated systems meets the requirements of Criterion 16. For further discussion, see the following sections:

Chapter/Section	Title
3.8.1	Concrete Containment
6.2	Containment Systems
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.2.8 Criterion 17 — Electric Power Systems

Criterion 17 Statement

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 of each system (assuming 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 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 the other off-site 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.

Evaluation Against Criterion 17

Onsite Electric Power System — The onsite power system is divided into multiple trains at the Medium Voltage level(s). This arrangement allows for design and operational flexibility of the plant non-safety fluid and mechanical systems. Separate unit auxiliary and reserve auxiliary transformers provide both a normal preferred and alternate preferred feeds to each of the Medium Voltage power trains. The Medium Voltage trains are divided into two categories; Power Generation and Plant Investment Protection (PIP).

The Power Generation trains supply power to nonsafety-related loads required primarily for unit operation.

The PIP trains supply power to permanent nonsafety-related loads, which due to their specific functions, are generally required to remain operational at all times. The PIP trains may also be connected to the onsite non-safety Alternating Current (AC) power supplies (Standby Diesel Generators [SDGs]). The PIP trains also provide power to the four safety-related divisional isolation buses, which in turn provide AC power to the battery chargers, rectifiers, and isolation bus transformers.

In addition to the two SDGs, the design also includes two Ancillary Diesel Generators (ADGs). The ADGs provide electrical power to a subset of the loads on the PIP trains and loads that have been classified as Criterion B under the Regulatory Treatment of Non-Safety Systems Program (Section 19A.3).

Each division of the safety-related power distribution system is provided with physically separated and electrically independent batteries sized to supply normal and emergency power to the engineered safety systems in the event of loss of all other preferred AC power sources.

The onsite Direct Current (DC) power includes both safety-related and nonsafety-related systems. These DC systems include plant batteries and battery chargers and their DC load, the DC/AC inverters and the inverter loads.

The safety loads utilize safety-related AC power for systems required for safety. DC power from the four divisional safety-related batteries is converted to AC power by the safety-related DC/AC inverters to provide the necessary electrical power. The systems required for safety are:

- RPS
- Engineered Safety Features Systems
- ICS
- Standby Liquid Control (SLC) system
- Safety-related information systems

Offsite Electric Power System — The offsite power system consists of the set of electrical circuits and associated equipment that is used to interconnect the offsite transmission system with the plant main generator and the onsite electrical power distribution system.

The system includes the plant switchyard and the high voltage tie lines to the unit auxiliary and Reserve Auxiliary Transformer (RAT) disconnects at the switchyard side of the Unit Auxiliary Transformers (UATs) and RATs.

The offsite power system begins at the terminals on the transmission system side of the main generator circuit breakers and the switchyard side of the UAT and RAT disconnects, which connect to the offsite transmission systems.

Power is supplied to the plant from two electrically independent and physically separate offsite power sources as follows:

- “Normal Preferred” source through the UATs
- “Alternate Preferred” source through the RATs

During plant startup, normal or emergency shutdown, or during plant outages, the offsite power system serves to supply power from the offsite transmission system to the plant auxiliary and service loads. During normal operation, the offsite power system is used to transmit generated power to the offsite transmission system and to the plant auxiliary and service loads.

The design of the offsite power systems is outside the scope of the ESBWR Standard Plant design. However, offsite power system requirements that meet the requirements of Criterion 17 are provided in Section 8.2. The onsite electric power systems are designed to meet the requirements of Criterion 17.

The ESBWR DC onsite power systems are adequate to accomplish required safety-related functions under all postulated accident conditions. The ESBWR does not require AC power (other than that provided by the uninterruptible power supplies) to achieve safe shutdown or to perform any safety-related function. Therefore, onsite safety-related DC power systems are applicable for Regulatory Guide (RG) 1.93 conformance; onsite AC power systems and offsite AC power systems are excluded from conformance with RG 1.93.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.6	Isolation Condenser System
5.4.12	Reactor Coolant System High Point Vents
6.3	Emergency Core Cooling Systems
8.1.5.2	Onsite Power
	Offsite Power Systems
8.3	Onsite Power Systems
9.3.5	Standby Liquid Control System

3.1.2.9 Criterion 18 — Inspection and Testing of Electric Power Systems

Criterion 18 Statement

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.

Evaluation Against Criterion 18

All safety-related loads are normally supplied directly through DC to AC inverters. Capability is provided for testing each battery, rectifier, battery charger, and inverter without disrupting power to the safety-related loads.

Design of the safety-related power system provides testability in accordance with the requirements of Criterion 18. For further discussion, see the following sections:

Chapter/Section	Title
8.1.5.2	Onsite Power
	Offsite Power Systems
8.3	Onsite Power Systems

3.1.2.10 **Criterion 19 — Control Room**

Criterion 19 Statement

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.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

Evaluation Against Criterion 19

The control room contains the controls and necessary surveillance equipment for operation of the plant systems, including the reactor and its auxiliary systems, ESFs, turbine generator, steam and power conversion systems, and station electrical distribution.

The control room is located in the Control Building (CB). Safe occupancy of the control room during abnormal conditions is provided in the design. Adequate shielding is provided to maintain radiation levels in the control room within prescribed limits in the event of a DBA for the duration of the accident.

The control room ventilation system has redundant equipment and includes radiation, toxic gas and smoke detectors with appropriate alarms and interlocks. The control room intake air can be filtered through high efficiency particulate air/absolute and charcoal filters. If any of the above hazards exist at the normal control room ventilation intake, habitability is assured by the Control Room Habitability Area HVAC Subsystem (CRHAVS), which upon isolation of the control room provides a positive air purge through an Emergency Filter Unit (EFU).

The control room is continuously occupied by qualified operating personnel under both operating and accident conditions. In the unlikely event that the control room must be vacated and access is restricted, instrumentation and controls are provided by two divisional Remote Shutdown System (RSS) panels located outside the control room in the Reactor Building (RB). Either or both of the RSS panels can be utilized to safely perform a hot shutdown and a subsequent cold shutdown of the reactor.

The control room design meets the requirements of Criterion 19. For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
5.4.12	Reactor Coolant System High Point Vents
6.3	Emergency Core Cooling Systems
6.4	Control Room Habitability Systems
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.4.1	Control Building HVAC System
9.4.7	Electrical Building HVAC System
11.5	Process Radiation Monitoring System
12.3	Radiation Protection
12.3.3	Ventilation
18.1.1	Design Goals and Design Bases
19A	Regulatory Treatment of Non-Safety Systems

3.1.3 Group III — Protection and Reactivity Control Systems

3.1.3.1 Criterion 20 — Protection System Functions

Criterion 20 Statement

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.

Evaluation Against Criterion 20

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and RCPB barrier. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored variables of the nuclear steam supply system (NSSS) (Section 7.2) exceed pre-established limits of AOOs. Scram trip settings are

selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The RPS includes the uninterruptible power sources, sensors, transmitters, bypass circuitry, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by the Neutron Monitoring System (NMS) signals, nuclear boiler high pressure, and reactor vessel low and high water levels prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal hydraulic safety limits during abnormal operational transients. Response by the RPS is prompt and the total scram time is short.

In addition to the RPS, which provides for automatic shutdown of the reactor to prevent fuel damage, protection systems are provided to sense accident conditions and to initiate automatically the operation of other safety-related systems and components. Other systems automatically isolate the reactor vessel or the containment to prevent the release of significant amounts of radioactive materials from the fuel and the RCPB. The controls and instrumentation for the Emergency Core Cooling System (ECCS) and the isolation systems are initiated automatically when monitored variables exceed pre-selected operational limits.

The design of the protection system satisfies the functional requirements as specified in Criterion 20. For further discussion, see the following sections:

Chapter/Section	Title
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.3.2 Criterion 21 — Protection System Reliability and Testability

Criterion 21 Statement

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in 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.

Evaluation Against Criterion 21

RPS design provides assurance that, through redundancy, each channel has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass maintenance operation, calibration operation, or test to verify operational availability, impairs the ability of the system to perform its intended safety function. Additionally, the system design assures that when a

scram trip point is exceeded, there is a high scram probability. However, should a scram not occur from the RPS, the Alternate Rod Insertion (ARI) actuates when the trip points are exceeded. There is sufficient electrical and physical separation between channels and between logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

The RPS includes design features that permit in-service testing. This ensures the functional reliability of the system should the reactor variable exceed the corrective action setpoint.

The RPS initiates an automatic reactor shutdown if the monitored plant variables exceed pre-established limits. This system is arranged as four separately powered divisions. Each division has a logic that can produce an automatic trip signal. The logic scheme is a two-out-of-four arrangement.

The RPS can be tested during reactor operation. Manual scram testing is performed by operating one of the four manual scram controls; this tests one division. The total tests verify the ability to de-energize the scram pilot valve solenoids. Indicating lights verify that the actuator's contacts have opened. This capability for a thorough testing program significantly increases reliability.

CRD operability can be tested during normal reactor operation. Rod position indicators and in-core neutron detectors are used to verify control rod movement. Each control rod can be withdrawn one step and then reinserted to the original position without significantly perturbing the NSSS at most power levels. One control rod is tested at a time. Hydraulic supply subsystem pressure can be observed on control room instrumentation.

The high functional reliability, redundancy, and in-service testability of the protection system satisfy the requirements specified in Criterion 21. For further discussion, see the following sections:

Chapter/Section	Title
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.3.3 Criterion 22 — Protection System Independence

Criterion 22 Statement

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 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.

Evaluation Against Criterion 22

Components of the protection system are designed so that the mechanical, thermal and radiological environmental conditions resulting from any accident situation in which the components are required to function do not interfere with the operation of that function. The redundant sensors are electrically and physically separated. Only circuits of the same division are run in the same raceway. Multiplexed signals are carried out by fiber optic medium to assure control signal isolation.

The RPS is designed to permit maintenance and diagnostic work while the reactor is operating, without restricting the plant operation or hindering the output of safety functions. The flexibility in design afforded the protection system allows operational system testing by the use of independent input for each actuator logic. When a safety-related monitored variable exceeds its scram trip point, it is sensed by four independent sensors, each located in a separate instrumentation channel. A bypass of any single channel is permitted for maintenance operation, test, etc. This leaves three channels per monitored variable, each of which is capable of initiating a scram. Only two actuator logics must trip to initiate a scram. Thus, the two-out-of-four arrangement assures that a scram occurs as a monitored variable exceeds its scram setting.

The protection system meets the design requirements for functional and physical independence as specified in Criterion 22. For further discussion, see the following sections:

Chapter/Section	Title
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.3.4 Criterion 23 — Protection System Failure Modes

Criterion 23 Statement

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.

Evaluation Against Criterion 23

The Reactor Protection (trip) System is designed to fail into a safe state. Use of independent channels allows the system to sustain any logic channel failure without preventing other sensors monitoring the same variable from initiating a scram. With a two-out-of-four logic design, the trip of any two channels initiates a scram. Intentional bypass for maintenance or testing causes the scram logic to revert to two-out-of-three. A failure of any one reactor protection input or subsystem component produces a trip in one channel. This condition is insufficient to produce a reactor scram, and the system performs its protective function upon trip of another channel. Failure of inputs or subsystem components in two channels produces a reactor scram.

The environmental conditions in which the instrumentation and equipment of the reactor protection must operate were considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient conditions in which the instruments must operate.

The fail-safe design of the Reactor Protection (trip) System meets the requirements of Criterion 23. For further discussion, see the following sections:

Chapter/Section	Title
3.11	Environmental Qualification of Mechanical and Electrical Equipment
4.6	Functional Design of Reactivity Control System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.3.5 Criterion 24 — Separation of Protection and Control Systems

Criterion 24 Statement

The protection system shall be separated from 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 to assure that safety is not significantly impaired.

Evaluation Against Criterion 24

There is separation between the RPS and the process control systems. Logic channel and actuator logics of the RPS are not used directly for automatic control of process systems. Sensor outputs may be shared, but each signal is optically isolated before entering a redundant or nonsafety-related channel interface. Therefore, failure in the controls and instrumentation of process systems cannot induce failure of any portion of the protective system. Scram reliability is designed into the RPS and Hydraulic Control Unit (HCU) for the CRD. The scram signal and mode of operation override all other signals.

The systems that isolate containment and the RPV are designed so that any one failure, maintenance operation, calibration operation, or test to verify operational availability does not impair the functional ability of the isolation systems to respond to safety-related variables.

The protection system is separated from control systems as required in Criterion 24. For further discussion, see the following sections:

Chapter/Section	Title
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.3.6 Criterion 25 — Protection System Requirements for Reactivity Control Malfunctions

Criterion 25 Statement

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 control rods.

Evaluation Against Criterion 25

The RPS provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB. Any monitored variable, which exceeds the scram setpoint, initiates an automatic scram and does not impair the remaining variables from being monitored, and if one channel fails, the remaining portion shall function.

The Rod Control and Information System (RC&IS) is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry of the RC&IS is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor normal circuitry from affecting the scram circuitry. Because one or two control rods are controlled by an individual HCU, a failure that results in continued energizing of an insert solenoid valve on a HCU can affect, at most, two control rods. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one HCU or two control rods.

The design of the protection system assures that specified acceptable fuel limits are not exceeded for any single malfunction of the reactivity control systems as specified in Criterion 25. For further discussion, see the following sections:

Chapter/Section	Title
4.6	Functional Design of Reactivity Control System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.3.7 Criterion 26 — Reactivity Control System Redundancy and Capability

Criterion 26 Statement

Two independent reactivity control systems of different design principles shall be provided. One of these 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 acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Evaluation Against Criterion 26

Two independent reactivity control systems utilizing different design principles are provided. The normal method of reactivity control employs control rod assemblies, which contain boron carbide (B_4C), hafnium or other approved material. A SLC system is also provided.

Positive insertion of these control rods is provided redundantly by means of the CRD electrical and hydraulic systems. The control rods are capable of reliably controlling reactivity changes during normal operation (e.g., power changes, power shaping, xenon burnout, normal startup and shutdown) via electrical powered insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during AOOs via the hydraulic powered automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram does not adversely affect the capability to maintain the core within fuel design limits.

The CRD system is capable of maintaining the reactor core subcritical under cold conditions, even when the pair of the control rods of the highest worth controlled by a HCU is assumed to stick in the fully withdrawn position. This shutdown capability of the CRD system is made possible by designing the fuel with burnable poison (Gadolinium Oxide [Gd_2O_3]) to control the high reactivity of fresh fuel.

The circuitry for electrical powered insertion or withdrawal of control rods is completely independent of the circuitry for hydraulic powered reactor scram. This separation of the scram and normal rod control functions prevents failures in the reactor manual-control circuitry from affecting the scram circuitry. Two sources of energy (accumulator pressure and electrical power to the motors of FMCRDs) are available for control rod insertion over the entire range of reactor pressure (i.e., from operating conditions to cold shutdown). The design of the CRD system includes appropriate margin for malfunctions such as stuck rods in the unlikely event that they do occur. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance and, simultaneously, low individual control rod worth. The operating procedures to accomplish such patterns are supplemented by the RC&IS, which prevent rod withdrawals yielding a rod worth greater than permitted by the pre-selected rod withdrawal pattern. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved with the insertion of a small number of the many independent control rods.

A SLC system containing a neutron-absorbing sodium pentaborate solution is the independent backup system. This system has the capability to shut the reactor down from full power and maintain it in subcritical condition at any time during the core life. The reactivity control is provided to reduce reactor power from rated power to cold shutdown conditions, with the control rods withdrawn in the power pattern, accounting for the reactivity effects of the xenon decay, elimination of steam voids, change in water density due to the reduction in water temperature, Doppler effect in uranium, change in the neutron leakage from boiling to cold, and change in the rod worth as boron affects the neutron migration length.

The redundancy and capabilities of the reactivity control systems for the ESBWR satisfy the requirements of Criterion 26. For further discussion, see the following sections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
4.3	Nuclear Design
4.6	Functional Design of Reactivity Control System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.3.2	Process Sampling System
9.3.5	Standby Liquid Control System

3.1.3.8 Criterion 27 — Combined Reactivity Control Systems Capability

Criterion 27 Statement

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.

Evaluation Against Criterion 27

There is no credible event applicable to the ESBWR that requires combined capability of the CRD system and the SLC system. The ESBWR design is capable of maintaining the reactor core subcritical, including allowance for a pair of stuck rods controlled by a HCU, without addition of any poison to the reactor coolant. The primary reactivity control system for the ESBWR during postulated accident conditions is the CRD system. Abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through automatic insertion of control rods. High integrity of the protection system is achieved through the combination of logic arrangement, actuator redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by separation of individual HCUs controlling a pair of control rods and by fail-safe design features built into the CRD system. Response by the RPS is prompt and the total scram time is short.

In the very unlikely event that more than one control rod fails to insert and the core cannot be maintained subcritical by control rods alone, the SLC system can be actuated to insert soluble boron into the reactor core. The SLC system has sufficient capacity to ensure that the reactor can always be maintained subcritical; and, hence, only decay heat is generated by the core, which can be removed by the appropriate decay heat removal systems (e.g., ICS), thereby ensuring that the core is always coolable.

The design of the reactivity control systems ensure reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability to cool the core is

maintained under postulated accident conditions; thus, Criterion 27 is satisfied. For further discussion, see the following sections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
4.6	Functional Design of Reactivity Control System
6.3	Emergency Core Cooling Systems
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.3.5	Standby Liquid Control System

3.1.3.9 **Criterion 28 — Reactivity Limits**

Criterion 28 Statement

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 (1) 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.

Evaluation Against Criterion 28

The combined features of the CRD system and the RC&IS designs incorporate appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The RC&IS prevents any withdrawal other than the pre-selected rod withdrawal pattern. The RC&IS function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown and power operation control rod procedures.

The CRD mechanical design incorporates a passive brake and hydraulic inlet check valve that individually prevent rapid rod ejection. The brake spring holds the rod in position if there is a break in the FMCRD primary pressure boundary. The check valve prevents rod ejection if there is a failure of the scram insert line. The FMCRD includes a separation switch that detects when withdrawal of a stuck control rod is being attempted and stops rod motion. Normal rod movement and the rod withdrawal rate are limited through the fine motion control motor.

The Safety Analyses evaluate the postulated reactivity accidents, as well as abnormal operational transients, in detail. Analyses are included for steam line break, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions,

calculational models, sequences of events, and anticipated results of each postulated occurrence are covered in detail. The results of these analyses indicate that none of the postulated reactivity transients or accidents results in damage to the RPV internals, so that the capability to cool the core is not impaired.

The design features of the RC&IS, which limit the potential amount and rate of reactivity increase, ensure that Criterion 28 is satisfied for postulated reactivity accidents. For further discussion, see the following sections:

Chapter/Section	Title
4.6	Functional Design of Reactivity Control System
Table 7.7-1	I&C Regulatory Requirements Applicability Matrix

3.1.3.10 **Criterion 29 — Protection Against Anticipated Operational Occurrences**

Criterion 29 Statement

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.

Evaluation Against Criterion 29

The high functional reliability of the RPS and reactivity control system is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and in-service testability. These design features are discussed in detail in Criteria 21, 22, 23, 24, and 26.

A thorough program of in-service testing and surveillance maintains an extremely high reliability of timely response to AOOs.

Safety-related components, such as CRDs, RPS components, etc., are testable during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analyses, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability effects during individual component testing on the portion of the system not undergoing test. The capability for in-service testing ensures the high functional reliability of protection and reactivity control systems if a reactor variable exceeds the corrective action setpoint.

The capabilities of the protection and reactivity control systems to perform their safety functions in the event of AOOs satisfy the requirements of Criterion 29. For further discussion, see the following sections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
4.6	Functional Design of Reactivity Control System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.4 Group IV — Fluid Systems

3.1.4.1 Criterion 30 — Quality of Reactor Coolant Pressure Boundary

Criterion 30 Statement

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.

Evaluation Against Criterion 30

By utilizing conservative design practices and detailed quality control procedures, the pressure retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions (Subsection 3.1.2.5). Accordingly, components that comprise the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Chapter 5 and Table 3.2-1. Further product and process quality planning are provided as described in Chapter 17 to assure conformance with the applicable codes and standards, and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is treated in the response to Criterion 14.

Means are provided for detecting leakage in the RCPB. The LD&IS consists of sensors and instruments to detect, annunciate, and, in some cases, isolate the RCPB from potentially hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and increased airborne radioactivity. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines, and changes in reactor water level. The allowable leakage rates have been based on the predicted and experimentally determined behavior of cracks in pipes, the ability to makeup to the RCS, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments.

The RCPB and the LD&IS are designed to meet requirements of Criterion 30. For further discussion, see the following sections:

Chapter/Section	Title
3.2	Classification of Structures, Systems, and Components
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment
5.2	Integrity of Reactor Coolant Pressure Boundary
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection
5.3	Reactor Vessel
5.4.12	Reactor Coolant System High Point Vents
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.4.2 Criterion 31 — Fracture Prevention of Reactor Coolant Pressure Boundary

Criterion 31 Statement

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) 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 (1) material properties, (2) the effect of irradiation on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

Evaluation Against Criterion 31

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against non-ductile fracture. To minimize the possibility of brittle fracture failure of the RPV, the RPV is designed to meet the requirements of ASME Code Section III.

The Nil-Ductility Transition Temperature (NDTT) is defined as the temperature below which ferritic steel behaves in a brittle rather than ductile manner. The NDTT increases as a function of neutron exposure at integrated neutron exposures greater than about $1x10^{17}$ nvt with neutron energies in excess of 1 MeV.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the reactor vessel that serves to attenuate the fast neutron flux incident upon the reactor vessel wall. This annular volume contains the core shroud and reactor coolant. Assuming plant operation at rated power and availability 100% of the plant lifetime, the cumulative neutron fluence at the inner surface of the vessel causes a slight shift in the transition temperature. Expected shifts in transition temperature during design life as a result of environmental conditions,

such as neutron flux, are considered in the design. Operational limitations ensure that NDTT shifts are accounted for in the reactor operation.

The RCPB is designed, maintained, and tested to provide adequate assurance that the boundary behaves in a non-brittle manner throughout the life of the plant. Therefore, the RCPB is in conformance with Criterion 31. For further discussion, see the following sections:

Chapter/Section	Title
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
6.1	Design Basis Accident Engineered Safety Feature Materials

3.1.4.3 Criterion 32 — Inspection of Reactor Coolant Pressure Boundary

Criterion 32 Statement

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Evaluation Against Criterion 32

The RPV design and engineering effort includes provisions for in-service inspection. Access to the annulus between the shield wall and vessel is provided by removable shield plugs and panels in the insulation. These openings provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the NBS piping and valves extending out to and including the first isolation valve outside containment. Inspection of the RCPB is in accordance with ASME B&PV Code Section XI. Section 5.2 defines the In-service Inspection Plan, access provisions, and areas of restricted access.

Vessel material surveillance samples are located within the RPV. The program includes specimens of the base metal, weld metal, and heat affected zone metal.

The plant testing and inspection program ensures that the requirements of Criterion 32 are met. For further discussion, see the following sections:

Chapter/Section	Title
5.3	Reactor Vessel

3.1.4.4 Criterion 33 — Reactor Coolant Makeup

Criterion 33 Statement

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 on-site electric power system operation (assuming offsite power is not available) and for off-site 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.

Evaluation Against Criterion 33

For small breaks without vessel depressurization and with preferred power or onsite AC power available, coolant from nonsafety-related inventory is automatically provided to the vessel by the nonsafety-related CRD System. Safety-related makeup from **the initial inventory maintained in the ICs while they are in Hot Standby,** is also provided by ICS upon automatic startup to perform its primary decay heat removal function. With or without preferred power and with a loss of feedwater supply, safety-related makeup is provided by the automatic depressurization system (ADS) with GDCS operation. Safety-related makeup is provided for the complete range of break sizes by the GDCS. For small breaks where depressurization of the reactor vessel is necessary to achieve GDCS flow, the ADS function (SRVs and DPVs) of the NBS operates to fully depressurize the vessel. After vessel depressurization and GDCS coolant inventory injection, makeup for core boil-off is provided by safety-related PCCS function through steam condensation and return to the vessel via the GDCS.

This design combination of nonsafety-related and safety-related systems provides the plant with ample reactor coolant makeup for protection against small leaks in the RCPB in response to AOOs and postulated accidents. The requirements of Criterion 33 are met with these systems. For further discussion, see the following sections:

Chapter/Section	Title
5.4.6	Isolation Condenser System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.4.5 Criterion 34 — Residual Heat Removal

Criterion 34 Statement

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.

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.

Evaluation Against Criterion 34

The ICS provides the means to remove decay heat and residual heat from the NSSS at a rate such that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded.

The major equipment of the ICS consists of heat exchangers. The equipment is connected to the reactor by associated valves and piping, and controls and instrumentation are provided for proper system operation.

Simply opening one of a pair of redundant, diverse drain line valves actuates each ICS sub-loop. Three of the four ICS sub-loops are adequate operating alone to remove residual heat from the reactor core and to assure fuel and RCPB design limits are not exceeded following an NSSS isolation event. The ICS provides the capability to reliably remove decay heat and residual heat from the reactor as required by Criterion 34.

The design of the ICS meets the requirements of Criterion 34. For further discussion, see the following sections:

Chapter/Section	Title
5.4.6	Isolation Condenser System
5.4.7	Residual Heat Removal System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.4.6 Criterion 35 — Emergency Core Cooling

Criterion 35 Statement

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 (1) 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.

Evaluation Against Criterion 35

The ECCS consists of the following:

- ICS
- SLC system
- GDCS
- ADS

The ECCS is designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the RCPB, including the complete circumferential rupture of the largest pipe connected to the RPV. The ESBWR ECCS does not rely on pumps, offsite AC power, or SDGs to accomplish its safety function.

The ICS and GDCS provide flow to the annulus region of the reactor through their own nozzles. The SLC system provides coolant to the bypass region of the core.

GDCS provides gravity-driven flow from three separate water pools located within the drywell at an elevation above the active core region. It also provides water flow from the suppression pool to meet long-term post-LOCA core cooling requirements.

ICS provides water that accumulates in heat exchangers and condensate pipe when the system is in standby. The water flows into the vessel when the ICS is initiated. This capability is available over the entire range of reactor vessel pressures.

SLC injects borated water into the vessel in the event of low level in the vessel for the purposes of providing additional coolant volume.

The ADS provides reactor depressurization capability in the event of a pipe break that does not rapidly depressurize the reactor. The ADS is a function of the NBS and is accomplished through the combined use of squib-type permanently - opening DPVs and nitrogen operated SRVs.

The ADS operates as follows: when a confirmed low-low water level (Level 1) signal is received and sealed-in to the ECCS logic, a number of SRVs and DPVs actuate in a sequence described in Subsection 6.3.3. This sequence of SRV and DPV openings ensures that the RPV is depressurized rapidly to allow GDCS initiation prior to core uncovery.

Results of the performance of the ECCS for the entire spectrum of reactor pressure boundary line breaks are discussed in Subsection 6.3, which provides an analysis to show that the ECCS conforms to 10 CFR 50.46. This analysis shows complete compliance with Criterion 35 with the following results:

- Peak cladding temperatures are below the NRC acceptable limit.
- The amount of fuel cladding reacting with steam is well below the acceptable limit.
- The accident is terminated while the core is maintained in a coolable geometry.
- The core temperature is reduced and the decay heat can be removed for an extended period of time.
- The ESBWR ECCS is powered by the safety-related station batteries. The redundancy and capability of the onsite electrical power systems are presented in the evaluation against Criterion 17.

The design of the ECCS, including the power supply, meets the requirements of Criterion 35. For further discussion, see the following subsections:

Chapter/Section	Title
5.4.6	Isolation Condenser System
6.1	Design Basis Accident Engineered Safety Feature Materials
6.3	Emergency Core Cooling Systems
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.3.5	Standby Liquid Control System

3.1.4.7 Criterion 36 — Inspection of Emergency Core Cooling System

Criterion 36 Statement

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.

Evaluation Against Criterion 36

The ECCS discussed in Criterion 35 includes in-service inspection considerations. Removable plugs in the reactor shield wall and/or panels in the insulation are provided on the ECCS piping in the drywell.

During plant operations, the instrumentation valves, instrument piping, instrumentation, wiring, and other components that are outside the drywell can be visually inspected at any time. Components inside the drywell can be inspected when the drywell is open for access during outages. Portions of the ECCS, which are part of the reactor pressure boundary, are designed to specifications for

in-service inspection to detect defects, which might affect the cooling performance. Particular attention is given to the GDCS nozzles.

Design of the reactor vessel and internals for in-service inspection and the plant testing program ensure that the requirements of Criterion 36 are met. For further discussion see the following subsections:

Chapter/Section	Title
5.4.6	Isolation Condenser System
6.3	Emergency Core Cooling Systems
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping
9.3.5	Standby Liquid Control System

3.1.4.8 Criterion 37 — Testing of Emergency Core Cooling System

Criterion 37 Statement

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) 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.

Evaluation Against Criterion 37

Each of the ECCS subsystems (ICS, SLC, ADS, and GDCS) is designed to permit periodic testing to assure operability and performance of active components of each system.

The ADS DPVs, SLC and the GDCS valves cannot be tested during power operation; selected actuators are removed and test fired during refueling outages. The GDCS check valves can be functionally tested via dedicated test line connections every refueling outage. GDCS flow testing is conducted as part of preoperational testing. Provisions for flushing the GDCS injection lines and venturi within the GDCS injection nozzle are provided. The ECCS is subject to periodic tests to verify the logic sequence that initiates ADS, ICS, SLC, and the GDCS system. A periodic self-test of the logic circuitry is performed to verify operability.

The design of the ECCS subsystems meets the requirements of Criterion 37. For further discussions, see the following subsections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
5.4.6	Isolation Condenser System
6.3	Emergency Core Cooling Systems
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.3.5	Standby Liquid Control System

3.1.4.9 Criterion 38 — Containment Heat Removal

Criterion 38 Statement

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 LOCA 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 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.

Evaluation Against Criterion 38

The containment heat removal function is accomplished by the PCCS and associated support systems. The PCCS provides sufficient decay heat removal post-LOCA to assure that containment never exceeds its design pressure and temperature.

The PCCS consists of six independent steam condensers that are an integral part of the containment. Each PCCS condenser contains two heat exchanger modules that condense steam on the tubeside and transfer heat to water in the Isolation Condenser/Passive Containment Cooling System (IC/PCCS) pool which is vented to atmosphere. The IC/PCCS pool is positioned above, and outside, the ESBWR containment (drywell). To assure availability, no valves are employed, thus precluding inadvertent isolation of the PCCS condensers. Long-term effectiveness of the PCCS credits an active gas recirculation system, which uses in-line fans to pull drywell gas through the PCCS condensers. These manually actuated fans (one per train) are located on a branch from the vent line and discharge to the GDCS pool.

The PCCS condensers receive a steam-gas mixture supply directly from the drywell. PCCS flow is driven by the pressure difference created between the containment drywell and the suppression pool during a LOCA. The PCCS does not require power supplies, sensors, control logic, power-actuated devices or operator actions to function in the first 72 hours after a LOCA. During normal plant operation, the PCCS condensers are in “ready standby.” In order to ensure the 72 hours of passive operation of the PCCS, the pool cross-connect valves, which are part of the ICS, must open to allow water to flow from the equipment pool to the IC/PCCS inner expansion pools. These valves are controlled by the Q-DCIS.

The PCCS is designed to Quality Group B Requirements per Regulatory Guide (RG) 1.26. The system is designed as Seismic Category I per RG 1.29. The common pool that the PCCS condensers share with the ICs of the ICS is an ESF. This pool is designed such that no locally generated force (such as an IC tube rupture) can destroy its function. Protection requirements against mechanical damage, fire and flood apply to the common IC/PCCS pool.

The safety-related IC/PCCS pool subcompartments provide protection for the PCCS condensers to comply with 10 CFR 50, Appendix A, Criteria 2 and 4.

The PCCS condensers do not fail in a manner that damages the safety-related IC/PCCS pool because it is designed to withstand the induced dynamic loads, which are caused by combined seismic, DPV/SRV or LOCA conditions in addition to PCCS operating loads.

The PCCS provides the containment heat removal function required in Criterion 38. For further discussion, see the following subsections:

Chapter/Section	Title
6.2.2	Passive Containment Cooling System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.4.10 **Criterion 39 — Inspection of Containment Heat Removal System**

Criterion 39 Statement

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.

Evaluation Against Criterion 39

The PCCS condenser is an integral part of the containment (drywell) pressure boundary and is used to mitigate the consequences of an accident. Because of this function it is classified as an ESF. The PCCS is designed to ASME Code Section III, Class MC and Section XI, IWE requirements for design and accessibility of welds for in-service inspection to meet 10 CFR 50 Appendix A, Criterion 16. Ultrasonic testing of tube-to-header welds and eddy current testing of tubes can be done with the PCCS condenser in place.

The containment heat removal system is designed to permit periodic inspection of major components to meet the requirements of Criterion 39. For further discussion, see the following subsections:

Chapter/Section	Title
6.2.2	Passive Containment Cooling System
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

3.1.4.11 Criterion 40 — Testing of Containment Heat Removal System

Criterion 40 Statement

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) 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.

Evaluation Against Criterion 40

The PCCS accomplishes the containment heat removal function. The PCCS is an integral part of the containment boundary. It is designed to be periodically pressure tested as part of overall Containment Leakage Rate Testing Program (Subsections 6.2.6.1, 6.2.6.2 and 6.2.6.3) to demonstrate structural and leaktight integrity. Also, the PCCS loops can be isolated for individual pressure testing during maintenance or in-service inspection using various non-destructive examination methods.

Functional and operability testing of the PCCS is not needed because there are no active components of the system needed in the first 72 hours after a LOCA. Long-term effectiveness of the PCCS requires that the vent fans are manually actuated. Performance testing during power operation is not feasible; however, the performance capability of the PCCS is proven by full-scale PCCS condenser prototype tests at a test facility before their application to the plant containment system design. Performance is established for the range of in-containment environmental conditions following a LOCA. Integrated containment cooling tests have been completed on a full height, reduced section test facility, and the results have been correlated with TRACG computer program analytical predictions; this computer program is used to show acceptable containment performance.

The design of containment heat removal system testing meets the requirements of Criterion 40. For further discussion, see the following subsections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
6.2.2	Passive Containment Cooling System
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

3.1.4.12 Criterion 41 — Containment Atmosphere Cleanup

Criterion 41 Statement

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 on-site 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.

Evaluation Against Criterion 41

Fission products, hydrogen, oxygen, and other substances released from the reactor are contained within the low-leakage containment. Leakage from the containment after an accident is such that the dose guidelines of 10 CFR 52.47 are not exceeded. Containment leakage enters the RB or Turbine Building where it is assumed to be released to the environment. The threat posed by hydrogen and oxygen is addressed by maintaining the containment inerted with nitrogen during operation by the Containment Inerting System (CIS), and by designing certain components to withstand hydrogen combustion loads. Passive Autocatalytic Recombiners (PARs) are provided to recombine hydrogen and oxygen for long-term pressure control following an accident.

The containment integrity is assured for postulated accidents and requirements of Criterion 41 are met. For further discussion, see the following sections:

Chapter/Section	Title
6.1	Design Basis Accident Engineered Safety Feature Materials
6.2.5	Combustible Gas Control in Containment
6.5.4	Suppression Pool as a Fission Product Cleanup System
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.3.2	Process Sampling System

3.1.4.13 Criterion 42 — Inspection of Containment Atmosphere Cleanup Systems

Criterion 42 Statement

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.

Evaluation Against Criterion 42

Containment atmosphere control is provided by the Containment Inerting System (CIS). Except for components located in the containment, other components of the CIS are accessible for inspection during normal plant operation at power. The components within the containment may be inspected during refueling and maintenance outages.

The design of the CIS meets the requirements of Criterion 42. For further discussion, see the following sections:

Chapter/Section	Title
6.2.5	Combustible Gas Control in Containment
6.5.4	Suppression Pool as a Fission Product Cleanup System
6.6	Preservice and In-service Inspection and Testing of Class 2 and 3 Components and Piping
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.4.14 Criterion 43 — Testing of Containment Atmosphere Cleanup Systems

Criterion 43 Statement

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight 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.

Evaluation Against Criterion 43

Containment atmosphere control is provided by the CIS. The CIS is designed to be periodically tested.

The design of the CIS meets the requirements of Criterion 43. For further discussion, see the following sections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
6.2.3	Reactor Building Functional Design
6.2.5	Combustible Gas Control in Containment
6.5.4	Suppression Pool as a Fission Product Cleanup System
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix

3.1.4.15 Criterion 44 — Cooling Water

Criterion 44 Statement

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.

Evaluation Against Criterion 44

The ESBWR ultimate heat sink is the IC/PCCS pool. In the event of a DBA, heat is transferred to the IC/PCCS pool(s) through the ICS and the PCCS. The water in the IC/PCCS pool(s) is allowed to boil and the resulting steam is vented to the environment. The PCCS has no active components and requires no electrical motive power or control and instrumentation functions to perform its safety-related function of transferring heat to the ultimate heat sink. The initial IC/PCCS pool volume, combined with the Equipment Storage Pool and Reactor Well, provides sufficient water

volume for at least 72 hours after a LOCA without external make-up to the IC/PCCS pools. The pool cross-connect valves, which are part of the ICS, must open to allow water to flow from the equipment pool to the IC/PCCS inner expansion pools. These valves are controlled by the Q-DCIS. Because only one of the two valves on either side of the equipment pool needs to open, no credible single failure can prevent the IC/PCCS pools from performing their safety-related function.

The requirements of Criterion 44 for heat transfer to the ultimate heat sink are met. For further discussion, see the following sections:

Chapter/Section	Title
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9	Auxiliary System
9.2.5	Ultimate Heat Sink

3.1.4.16 Criterion 45 — Inspection of Cooling Water System

Criterion 45 Statement

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.

Evaluation Against Criterion 45

The IC/PCCS pool is located outside containment and is accessible for periodic inspections. During outages, the IC/PCCS pool compartments can be drained to permit inspection of the IC/PCCS pool components.

The features of the IC/PCCS pools meet the requirements of Criterion 45. For further discussion, see the following sections:

Chapter/Section	Title
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping
9	Auxiliary System
9.2.5	Ultimate Heat Sink
14	Initial Test Program

3.1.4.17 Criterion 46 — Testing of Cooling Water System

Criterion 46 Statement

The Cooling Water System shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural leaktight 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 the 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.

Evaluation Against Criterion 46

Redundancy and isolation are provided to allow periodic inspection of the IC/PCCS pool compartments. As discussed in the evaluation of Criterion 44, the IC/PCCS pools contain no active components aside from connections to the Equipment Storage Pool that open automatically to ensure adequate coolant is provided for at least the initial 72 hours following an accident. These connections are accessible during an outage to permit inspection. The periodic inspections described in the response to Criterion 45 verify system integrity (see the evaluation of Criterion 40).

The design of the IC/PCCS pools meets the requirements of Criterion 46. For further discussion, see the following sections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping
9	Auxiliary System
9.2.5	Ultimate Heat Sink

3.1.5 **Group V — Reactor Containment**

3.1.5.1 **Criterion 50 — Containment Design Basis**

Criterion 50 Statement

The reactor containment structure, including access openings, 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 peak conditions, such as energy in steam generators and, as required by Section 50.44, energy from metal-water and other chemical reactions that may result from degradation but not total failure of 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.

Evaluation Against Criterion 50

Design of the containment is based on consideration of the full spectrum of postulated accidents which would result in the release of reactor coolant to the containment. These accidents include liquid breaks, steam breaks, and partial breaks (both steam and liquid). The evaluation of the containment design is based on enveloping the results of this range of analyses, plus provision for appropriate margins. The most limiting short-term and long-term pressure and temperature responses are assessed to verify adequacy of the containment structure.

The design of the containment system meets the requirements of Criterion 50. For further discussion, see the following sections:

Chapter/Section	Title
3.7	Seismic Design
3.8	Design of Seismic Category I Structures
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
6.2.1	Containment Functional Design
8.1.5.2	Onsite Power
8.3	Onsite Power Systems

3.1.5.2 Criterion 51 — Fracture Prevention of Containment Pressure Boundary

Criterion 51 Statement

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.

Evaluation Against Criterion 51

The containment vessel is a reinforced concrete structure with ferritic parts, such as a liner and a removable head, which are made of materials that have a NDTT sufficiently below the minimum service temperature to assure that under operating, maintenance, testing, and postulated accident conditions the ferritic materials behave in a nonbrittle manner considering the uncertainties in determining the material properties, stresses and size of flaws.

The containment vessel is enclosed by and integrated with the reinforced concrete RB. The pre-operational test program and the quality assurance program ensure the integrity of the containment and its ability to meet all normal operating and accident requirements.

The containment design meets the requirements of Criterion 51. For further discussion, see the following sections:

Chapter/Section	Title
6.2	Containment Systems

3.1.5.3 Criterion 52 — Capability for Containment Leakage Rate Testing

Criterion 52 Statement

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.

Evaluation Against Criterion 52

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leak-rate tests. The testing program is conducted in accordance with 10 CFR 50 Appendix J.

The testing provisions provided and the test program meet the requirements of Criterion 52. For further discussion, see the following subsection:

Chapter/Section	Title
6.2.2	Passive Containment Cooling System
6.2.6	Containment Leakage Testing

3.1.5.4 Criterion 53 — Provisions for Containment Testing and Inspection

Criterion 53 Statement

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 leaktightness of penetrations which have resilient seals and expansion bellows.

Evaluation Against Criterion 53

There are special provisions for conducting individual leakage rates tests on applicable penetrations. Penetrations are visually inspected and pressure tested for leaktightness at periodic intervals in accordance with 10 CFR 50 Appendix J.

The provisions made for periodic testing meet the requirements of Criterion 53. For further discussion, see the following sections:

Chapter/Section	Title
6.2.1	Containment Functional Design
6.2.2	Passive Containment Cooling System
6.2.6	Containment Leakage Testing

3.1.5.5 Criterion 54 — Piping Systems Penetrating Containment

Criterion 54 Statement

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.

Evaluation Against Criterion 54

Piping systems penetrating containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections to allow periodic leak detection tests as necessary to determine if valve leakage is within acceptable limits.

The actuation test circuitry provides the means for testing isolation valve operability as necessary to determine if operability is within acceptable limits.

The design and provisions made for piping systems penetrating containment meet the requirements of Criterion 54. For further discussion, see the following subsections:

Chapter/Section	Title
3.9.6	Mechanical Systems and Components
5.4.6	Isolation Condenser System
6.2.4	Containment Isolation Function
6.2.6	Containment Leakage Testing
6.5.2	Fission Product Control Systems and Structures

3.1.5.6 Criterion 55 — Reactor Coolant Pressure Boundary Penetrating Containment

Criterion 55 Statement

Each line that is part of the reactor coolant pressure boundary and that 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 containment.
2. One automatic isolation valve inside and one locked closed isolation valve outside containment.
3. One locked closed 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.
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 the 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 inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Evaluation Against Criterion 55

The RCPB, as defined in 10 CFR 50, Section 50.2, consists of the RPV, pressure-retaining appurtenances attached to the vessel, valves and pipes which extend from the RPV up to and including the outermost isolation valves. The lines of the RCPB, which penetrate the containment, have isolation valves capable of isolating the containment, thereby precluding any significant release of radioactivity. Justification for the design of each RCPB line penetrating containment is provided in Subsection 6.2.4.

The manner in which RCPB lines that penetrate primary containment meet the requirements of Criterion 55 is discussed further in the following sections:

Chapter/Section	Title
5.4.5	Main Steamline Isolation System
5.4.6	Isolation Condenser System
6.2.4	Containment Isolation System

3.1.5.7 Criterion 56 — Primary Containment Isolation

Criterion 56 Statement

Each line that connects directly to the containment atmosphere and 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 instruments lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment.
2. One automatic isolation valve inside and one locked closed isolation valve outside containment.
3. One locked closed 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.
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 the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Evaluation Against Criterion 56

Lines penetrating containment and connecting directly to the containment atmosphere are isolatable by one of the methods specified in Criterion 56 or are exempted. A justification is provided for each containment penetration in Subsection 6.2.4.

The manner in which the containment isolation system meets the requirements of Criterion 56 is discussed further in the following sections:

Chapter/Section	Title
6.2.4	Containment Isolation System

3.1.5.8 Criterion 57 — Closed System Isolation Valves

Criterion 57 Statement

Each line that penetrates 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, or 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.

Evaluation Against Criterion 57

Each line that penetrates the containment and is not connected to the containment atmosphere and is not part of the RCPB has at least one isolation valve outside containment.

The manner in which lines that penetrate the containment boundary but are not part of the RCPB nor connect to the containment atmosphere meet the requirements of Criterion 57 is discussed further in the following subsection:

Chapter/Section	Title
6.2.4	Containment Isolation Systems

3.1.6 Group VI — Fuel and Radioactivity Control

3.1.6.1 Criterion 60 — Control of Releases of Radioactive Materials to the Environment

Criterion 60 Statement

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 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.

Evaluation Against Criterion 60

The ESBWR is designed so that releases of radioactive materials, in their gaseous, liquid, and solid form are minimized. Gaseous releases come primarily from the turbine condenser offgas and the ventilation systems. Noble gas and iodine activity that enters the turbine offgas system is held by ambient charcoal beds. Ventilation releases are through multiple plant stacks. The Turbine Building, Reactor Building/Fuel Building (RB/FB) and Radwaste Building stacks and the major streams feeding the plant stacks are monitored by the Process Radiation Monitoring System so that action may be taken to avoid releases in excess of regulatory limits.

The radwaste systems process liquid and solid wastes. Processes are provided to treat and package solid wastes, as required by applicable state and federal regulations. In addition, the ESBWR liquid radwaste system can be operated in a mode where non-detergent and non-chemical waste streams are treated to allow maximum recycle to the condensate storage tank. This mode of operation would minimize releases of radioactivity via the liquid or discharge pathway, but would increase solid waste generated.

The radwaste system has significant hold-up capacity, both in waste collection tanks and in sample tanks containing processed water. This hold-up or surge capacity provides the plant operator flexibility in operations when deciding when and how to release effluents to the environment.

The provisions made for controlling the release of radioactive material meet the requirements of Criterion 60. For further discussion, see the following sections:

Chapter/Section	Title
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
9	Auxiliary System
11	Radioactive Waste Management

3.1.6.2 Criterion 61 — Fuel Storage and Handling and Radioactivity Control

Criterion 61 Statement

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 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.

Evaluation Against Criterion 61

The spent fuel storage pool has adequate water shielding for stored spent fuel. Adequate shielding for transporting fuel is also provided in the buffer pool between the vessel and spent fuel storage pool. Liquid level sensors are installed to detect low pool water level. The RB/FB is designed to meet RG 1.13 criteria. The spent fuel storage pool is designed with no penetrations below the water level needed for adequate shielding at the operating floor. Anti-siphoning provisions protect against draining the spent fuel storage pool in the event of a line break.

New fuel storage racks are provided in the buffer pool adjacent to the vessel cavity. These storage racks preclude accidental criticality (see evaluation against Criterion 62). The new fuel storage racks do not require any special in-service inspection and testing for nuclear safety purposes.

The nonsafety-related FAPCS normally removes decay heat from fuel storage pools. Without the active cooling trains of the FAPCS, the safety-related method of cooling the spent fuel is to allow the spent fuel pools to boil. Sufficient pool water inventory is provided to permit boiling for several days without makeup. If required, makeup water is provided from on site sources for up to at least 7 days from the FPS. Safety-related FAPCS piping is used to transport makeup water to the spent fuel pool from FPS (for at least 7 days) and from a connecting point (also safety-related) in the yard area to portable water sources (See Subsection 9.1.3.2).

The fuel storage and handling system is designed to ensure adequate safety under normal and postulated abnormal conditions. (See Subsection 9.1.4.)

The design of these systems meets the requirements of Criterion 61. For further discussion, see the following sections:

Chapter/Section	Title
5.4.8	Reactor Water Cleanup/Shutdown Cooling System
9.1.1	New Fuel Storage
9.1.2	Spent Fuel Storage
9.1.3	Fuel and Auxiliary Pools Cooling System
9.1.4	Light Load Handling Systems
9.1.5	Overhead Heavy Load Handling System
9.4.2	Fuel Building HVAC Systems
11.4	Radioactive Waste Management System
12.3	Radiation Protection

3.1.6.3 Criterion 62 — Prevention of Criticality in Fuel Storage and Handling

Criterion 62 Statement

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

Evaluation Against Criterion 62

Fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality of new fuel stored in the buffer pool is prevented by physical separation. Criticality in the spent fuel storage pool is prevented by presence of fixed neutron absorbing material. The new and spent fuel racks are Seismic Category I components.

The spent fuel is stored under water in the spent fuel storage pool. A full array of loaded spent fuel racks is designed to be subcritical. Neutron-absorbing material, as an integral part of the design, is employed to assure that the calculated k_{eff}, including biases and uncertainties, does not exceed 0.95 under all normal and abnormal conditions. The abnormal conditions accounted for are an earthquake, accidental dropping of equipment, or impact caused by the horizontal movement of the fuel handling equipment without first disengaging the fuel from the hoisting equipment.

Refueling interlocks include circuitry that senses conditions of the refueling equipment and the control rods. These interlocks reinforce the operational procedures that prohibit making the reactor critical. The fuel handling system is designed to provide a safe, effective means of transporting and handling and is designed to minimize the possibility of mishandling or maloperation.

The presence of fixed neutron-absorbing material in the spent fuel storage, physical separation in the new fuel storage and the design of fuel handling systems precludes accidental criticality in accordance with Criterion 62. For further discussion, see the following section:

Chapter/Section	Title
9.1	Fuel Storage and Handling

3.1.6.4 Criterion 63 — Monitoring Fuel and Waste Storage

Criterion 63 Statement

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

Evaluation Against Criterion 63

Fuel pool temperature and level are monitored as part of the FAPCS. High pool temperature or low skimmer surge tank level would signal the need for providing additional cooling. Area radiation monitors are provided as part of the Area Radiation Monitoring System, which monitors the operating/refueling floors for high radiation levels.

The radwaste system has no active decay heat removal functions, since the decay heat from the activity in the inputs to radwaste is not sufficient to warrant concern. Radwaste Building area radiation monitors are provided to protect against excessive personal exposure, and monitoring shipping container activity and surface radiation levels to meet appropriate waste and transportation criteria.

The design of these systems meets the requirements of Criterion 63. For further discussion, see the following subsections:

Chapter/Section	Title
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.1.2	Spent Fuel Storage
9.1.3	Fuel and Auxiliary Pools Cooling System
9.3.2	Process Sampling System
11	Radioactive Waste Management System
12	Radiation Protection

3.1.6.5 Criterion 64 — Monitoring Radioactivity Releases

Criterion 64 Statement

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.

Evaluation Against Criterion 64

Means have been provided for monitoring radioactivity releases resulting from normal operations and AOOs and from postulated accidents. The following releases are monitored:

- Gaseous releases
- Liquid discharge

In addition, the containment atmosphere is monitored.

The design of these systems meets the requirements of Criterion 64. For further discussion of the means and equipment used for monitoring reactivity releases, see the following sections:

Chapter/Section	Title
Table 7.1-1	I&C Regulatory Requirements Applicability Matrix
9.3.2	Process Sampling System
11.3	Gaseous Waste Management System
11.4	Solid Waste Management System

3.1.7 COL Information

None.

3.2 Classification of Structures, Systems and Components

ESBWR structures, systems and components are categorized as safety-related (as defined in 10 CFR 50.2) or nonsafety-related. The safety-related structures, systems and components are those relied upon to remain functional during and following design basis events to ensure:

- The integrity of the reactor coolant pressure boundary (RCPB).
- The capability to shut down the reactor and maintain it in a safe condition.
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the applicable exposure limits set forth in 10 CFR 52.47(a)(2)(iv).

Safety-related structures, systems and components conform to the quality assurance requirements of Appendix B to 10 CFR 50. Nonsafety-Related structures, systems and components have quality

assurance requirements applied commensurate with the importance of the item's function. The quality assurance program is described in Chapter 17.

The ESBWR complies with 10 CFR 50, Appendix A, General Design Criterion (GDC) 2, as the safety-related structures, systems and components are designed to withstand the effects of earthquakes without loss of capability to perform their safety-related functions. Specific requirements for seismic design and quality group classifications are identified for these ESBWR items commensurate with their safety classification. Table 3.2-1 identifies these classifications for ESBWR structures, systems and components.

There are no site specific safety related or non-safety related RTNSS systems beyond the scope of the DCD.

3.2.1 **Seismic Classification**

The ESBWR meets the acceptance criteria of Standard Review Plan (SRP) 3.2.1 (Reference 3.2-1). Structures that must remain integral with systems and components (including their foundations and supports) that must remain functional or retain their pressure integrity in the event of a safe shutdown earthquake (SSE) are designated Seismic Category I. These include safety-related items and fuel storage racks.

The Seismic Category I structures, systems, and components are designed to withstand the appropriate seismic loads (as discussed in Section 3.7) in combination with other appropriate loads without loss of function or pressure integrity. The seismic classifications indicated in Table 3.2-1 are consistent with the guidelines of Regulatory Guide (RG) 1.29 (Reference 3.2-2).

Structures, systems and components that perform no safety-related function, but whose structural failure or interaction could degrade the functioning of a Seismic Category I item to an unacceptable level of safety or could result in incapacitating injury to occupants of the main control room, are designated Seismic Category II. These items are designed to structurally withstand the effects of an SSE. Seismic Category II structures, systems and components that are also classified as Regulatory Treatment of Non-Safety Systems (RTNSS) Criterion B in Tables 19A-2 and 19A-3 are required to remain functional following a seismic event.

Structures, systems, and components that are not categorized as Seismic Category I or II are designated Seismic Category NS.

Seismic Category NS structures and equipment are designed for seismic requirements in accordance with the International Building Code (IBC) (Reference 3.2-6). The building structures are classified as Category IV (Power Generating Stations) with an Occupancy Importance Factor of 1.5. Either of the methods permitted by the IBC, simplified analysis or dynamic analysis, is acceptable for determination of seismic loads on Seismic Category NS structures and equipment. Refer to Subsection 19A.8.3 for seismic design requirements applicable to Seismic Category NS

structures, systems and components designated as RTNSS, and to Table 19A-2 for a list of RTNSS structures, systems and components.

3.2.2 System Quality Group Classification

The ESBWR meets the acceptance criteria of SRP 3.2.2 (Reference 3.2-3). NRC RG 1.26 (Reference 3.2-4) describes a quality group classification method for fluid systems and relates it to industry codes. Items are classified by Quality Group A, B, C or D, as indicated in Table 3.2-3. Table 3.2-3 tabulates the design and fabrication requirements for each quality group, as defined in RG 1.26.

Table 3.2-1 shows the quality group classifications for ESBWR components. Core support structures and containment boundaries are not within the scope of RG 1.26 definitions, but are within the scope of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III. The quality group classifications assigned to core support structures and containment boundaries in Table 3.2-1 are in accordance with Tables 3.2-2 and 3.2-3.

Due to the use of many passive safety-related systems in ESBWR, the definitions of the quality groups provided in RG 1.26 can be somewhat misleading when trying to apply them directly to the ESBWR design. The following definitions in this section are consistent with the definitions in RG 1.26, but have been modified to more accurately describe their application to the ESBWR design.

3.2.2.1 Quality Group A

Quality Group A applies to pressure-retaining portions and supports of mechanical items that form part of the RCPB and whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability. These items are designed to meet the ASME B&PV Code, Section III. Remaining portions of the RCPB are classified in accordance with Subsection 3.2.2.2.

3.2.2.2 Quality Group B

Quality Group B applies to pressure-retaining portions and supports of containment and other mechanical items, requirements for which are within the scope of ASME B&PV Code, Section III. These items are not assigned to Quality Group A and are relied upon to accomplish one or more of the following safety-related functions:

- Maintain the pressure integrity of RCPB items that are not Quality Group A.
- During or following design basis accidents whose consequences could result in potential offsite exposures comparable to the limits of 10 CFR 52.47(a)(2)(iv). These items include those that:
  - Maintain the pressure integrity of the containment, containment isolation, or extension of containment.

- Maintain the pressure integrity of items that are (1) exterior to the containment; (2) communicate with the RCPB or containment interior; and (3) are not isolated normally, cannot be automatically isolated, or are not isolated following a design basis accident or anticipated operation occurrence (transient).
- Maintain the pressure integrity of items that provide emergency negative reactivity insertion (scram).

As defined in RG 1.26, the Quality Group B standards defined in Table 3.2-3 are applied to water- and steam-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves that are either part of the RCPB as defined in 10 CFR 50.2 but excluded from the requirements of 10 CFR 50.55a pursuant to paragraph (c)(2) of that section, or not part of the RCPB but part of:

a. Systems or portions of systems important to safety that are designated for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident fission product removal.
b. Systems or portions of systems important to safety that are designed for (1) reactor shutdown or (2) residual heat removal.
c. Those portions of the steam systems of boiling water reactors extending from the outermost containment isolation valve up to but not including the turbine stop and bypass valves and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation. Alternatively, for boiling water reactors containing a shutoff valve (in addition to the two containment isolation valves) in the main steamline and in the main feedwater line, Group B quality standards should be applied to those portions of the steam and feedwater systems extending from the outermost containment isolation valves up to and including the shutoff valve or the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation.
d. Systems or portions of systems that are connected to the RCPB and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.

Quality Group B may also be assigned to nonsafety-related equipment in some instances.

3.2.2.3 **Quality Group C**

Quality Group C applies to pressure-retaining portions and supports of items that are not assigned to Quality Group A or Quality Group B, but (1) are within the scope of the codes and standards defined on Table 3.2-3, and (2) are relied upon to accomplish safety-related functions.

As defined in RG 1.26, the Quality Group C standards defined in Table 3.2-3 are applied to water-, steam- and radioactive-waste-containing pressure vessels, heat exchangers (other than turbines

and condensers), storage tanks, piping, pumps, and valves not part of the RCPB or included in Quality Group B but part of:

a. Cooling water and auxiliary feedwater systems or portions of these systems important to safety that are designed for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, or (4) residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems). Portions of these systems that are required for their safety functions and that (1) do not operate during any mode of normal reactor operation and (2) cannot be adequately tested should be classified as Quality Group B.

b. Cooling water and seal water systems or portions of these systems important to safety that are designed for functioning on components and systems important to safety, such as reactor coolant pumps, diesels, and control room.

c. Systems or portions of systems that are connected to the RCPB and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.

d. Systems, other than radioactive waste management systems, not covered by items a. through c. above that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that exceed 5 mSv (0.5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2. For those systems located in Seismic Category I structures, only single component failures need be assumed.

Quality Group C may also be assigned to nonsafety-related equipment in some instances.

3.2.2.4 **Quality Group D**

Quality Group D applies to pressure-retaining portions and supports of items that are not assigned to Quality Group A, Quality Group B or Quality Group C but (1) are within the scope of the codes and standards defined on Table 3.2-3, and (2) are subject to one or more significant licensing requirements or commitments. These items include those that:

- Process, extract, encase, or store radioactive waste.
- Monitor radioactive effluents to ensure that release rates or total releases are within limits established for normal operation and design basis transients.
- Resist failure that could prevent any Quality Group A, Quality Group B or Quality Group C items from performing a safety-related function.
- Protect items necessary to attain or maintain safe shutdown following fire.

3.2.3 **Safety Classification**

Safety-related structures, systems, and components of the ESBWR Standard Plant are classified for design requirements as Safety Class 1, Safety Class 2, or Safety Class 3 in accordance with their safety importance. These safety classifications are identified on Table 3.2-1 for principal structures, systems, and components. Components within a system are assigned different safety classes depending upon their differing safety importance; a system may thus have components in more than one safety class. Safety classification for supports within the scope of ASME B&PV Code Section III depends upon that of the supported component.

This section provides definitions of the safety classes and gives examples of their broad application. Because of specific design considerations, these general definitions are subject to interpretation and exceptions. Table 3.2-1 identifies component classifications on a component-by-component basis for primary components.

Minimum classification requirements (i.e., quality group, seismic category, electrical classification and quality assurance) that are applicable to the various safety-related classes are delineated in Table 3.2-2. Table 3.2-3 identifies the applicable industry codes and standards for the various quality groups defined above in Subsection 3.2.2. Where possible, reference is made to accepted industry codes and standards which define design requirements commensurate with the safety-related function(s) to be performed. In cases where industry codes and standards have no specific design requirements, the sections that summarize the requirements to be implemented in the design are indicated.

Structures, systems and components that have no safety-related function are classified as nonsafety-related and designated N.

3.2.3.1 **Safety Class 1**

Safety Class 1 applies to all components of the RCPB (as defined in 10 CFR 50.2), and their supports, whose failure could cause a loss of reactor coolant at a rate in excess of the normal makeup system, and which are within the scope of the ASME B&PV Code Section III.

Safety Class 1 structures, systems and components are identified in Table 3.2-1. All Safety Class 1 structures, systems and components are subject to 10 CFR 50 Appendix B quality assurance requirements. Safety Class 1 structures, systems and components that are pressure-retaining components belong to Quality Group A as defined in Subsection 3.2.2.1.

3.2.3.2 **Safety Class 2**

Safety Class 2 applies to pressure-retaining portions, and their supports, of the primary containment and to other mechanical equipment, requirements for which are within the scope of the ASME Code Section III, that are not included in Safety Class 1 and are designed and relied upon to accomplish the following safety-related functions:

1. Provide primary containment radioactive material holdup or isolation.

2. Provide emergency heat removal for the primary containment atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere.
3. Introduce emergency negative reactivity to make the reactor subcritical.
4. Ensure emergency core cooling where the equipment provides coolant directly to the core (e.g., emergency core cooling systems).
5. Provide or maintain sufficient reactor coolant inventory for emergency core cooling (e.g., GDCS pools).

Safety Class 2 includes the pressure-retaining portions of the following:

1. Those control rod drive system components that are necessary for emergency negative reactivity insertion.
2. Emergency core cooling systems.
3. Primary containment vessel.
4. Post-accident containment heat removal systems.
5. Pipes having a nominal pipe size of 25 mm (1 inch) or smaller that are part of the RCPB.

Safety Class 2 structures, systems, and components are identified in Table 3.2-1. All Safety Class 2 structures, systems and components are subject to 10 CFR 50 Appendix B quality assurance requirements. Safety Class 2 structures, systems and components that are pressure-retaining components belong to Quality Group B (as a minimum) as defined in Subsection 3.2.2.2.

3.2.3.3 **Safety Class 3**

Safety Class 3 applies to those structures, systems, and components, not included in Safety Class 1 or 2, that are designed and relied upon to accomplish the following safety-related functions:

1. Provide for functions defined in Safety Class 1 or 2 by means of equipment, or portions thereof, that is not within the scope of the ASME B&PV Code Section III.
2. Provide secondary containment radioactive material holdup, isolation, or heat removal.
3. Except for primary containment boundary extension functions, ensure hydrogen concentration control of the primary containment atmosphere to acceptable limits.
4. Remove radioactive material from the atmosphere of confined spaces outside primary containment (e.g., control room) containing Safety Class 1, 2, or 3 equipment.
5. Maintain geometry within the reactor to ensure core reactivity control or core cooling capability.
6. Structurally bear the load or protect Safety Class 1, 2, or 3 equipment in accordance with the requirements.
7. Provide radiation shielding for the control room or offsite personnel.

8. Provide inventory of cooling water and shielding for stored spent fuel.
9. Ensure safety-related functions provided by Safety Class 1, 2, or 3 equipment (e.g., provide heat removal for Safety Class 1, 2, or 3 heat exchangers, provide lubrication of Safety Class 2 or 3 pumps).
10. Provide actuation or motive power for Safety Class 1, 2, or 3 equipment.
11. Provide information or controls to ensure capability for manual or automatic actuation of safety-related functions required of Safety Class 1, 2, or 3 equipment.
12. Supply or process signals or supply power required for Safety Class 1, 2, or 3 equipment to perform their required safety-related functions.
13. Provide a manual or automatic interlock function to ensure or maintain proper performance of safety-related functions required of Safety Class 1, 2, or 3 equipment.
14. Provide acceptable environments for Safety Class 1, 2, or 3 equipment and operating personnel.
15. Monitor plant variables that require Category 1 electrical instrumentation to meet the requirements of RG 1.97 (Reference 3.2-8).

Safety Class 3 includes the following:

1. Reactor protection system.
2. Electrical and instrumentation auxiliaries necessary for operation of the safety-related systems and components.
3. Systems or components that restrict the rate of insertion of positive reactivity.
4. Initiating systems required to accomplish emergency core cooling, containment isolation and other safety-related functions.
5. Spent fuel pool.
6. Batteries for the onsite emergency electrical system.
7. Emergency equipment area cooling.
8. Compressed gas or hydraulic systems required to provide control or operation of safety-related systems.

Safety Class 3 structures, systems and components are identified in Table 3.2-1. All Safety Class 3 structures, systems and components are subject to 10 CFR 50 Appendix B quality assurance requirements. Safety Class 3 structures, systems and components that are pressure-retaining components belong to Quality Group C (as a minimum) as defined in Subsection 3.2.2.3.

3.2.3.4 **NonSafety-Related**

Structures, systems and components that do not fall into Safety Classes 1, 2 or 3 are classified as “Nonsafety-Related,” which is abbreviated as “N” in Table 3.2-1.

The design requirements for Nonsafety-Related equipment are specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate.

Where appropriate or required by specific regulations, Seismic Category I requirements are specified for Nonsafety-Related equipment in Table 3.2-1. Generally, design requirements for Nonsafety-Related equipment are based on applicable industry codes and standards as summarized in Table 3.2-3. Where these are not available, accepted industry or engineering practice is followed.

Nonsafety-related structures, systems and components that are classified Seismic Category I or II and Quality Group B or C are subject to ASME B&PV Code Section III requirements (including N stamping) and ASME B&PV Code Section XI inspection requirements.

3.2.4 COL Information

None.

3.2.5 References

Note: Detailed references for all Regulatory Guides and Industry Codes and Standards referred to in Tables 3.2-1 through 3.2-3 can be found in Tables 1.9-21 and 1.9-22.

3.2-1 USNRC, “Seismic Classification,” NUREG-0800, SRP 3.2.1.

3.2-2 USNRC, “Seismic Design Classification,” Regulatory Guide 1.29.

3.2-3 USNRC, “System Quality Group Classification.” NUREG-0800, SRP 3.2.2.

3.2-4 USNRC, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Regulatory Guide 1.26.

3.2-5 (Deleted)

3.2-6 International Building Code – 2003 by International Code Council, Inc.

3.2-7 (Deleted)

3.2-8 NRC Regulatory Guide 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.”

3.2-9 Global Nuclear Fuel, “GESTAR II General Electric Standard Application for Reactor Fuel,” NEDE-24011-P-A-16, Class III (GE Proprietary) and NEDO-24011-A-16, Class I (Non-proprietary), Revision 16, October 2007.

Table 3.2-1 Classification Summary (Sheet 1 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
B NUCLEAR STEAM SUPPLY SYSTEMS						
B11 Reactor Pressure Vessel System						
1. Reactor pressure vessel	1	CV	A	Q	I	
2. Reactor vessel appurtenances – reactor coolant pressure boundary (RCPB) portions	1	CV	A	Q	I	
3. Control Rod Drive housing and in-core housing	1	CV	A	Q	I	
4. Control rods	2	CV	—	Q	I	
5. Standby Liquid Control (SLC) system header and spargers	2	CV	—	Q	I	
6. Reactor vessel support and stabilizer	1	CV	A	Q	I	
7. Other safety-related reactor internals, including core support structures (Subsection 3.9.5)	3	CV	B	Q	I	
8. Reactor internals – Nonsafety-Related components (Subsection 3.9.5)	N	CV	—	S	II	(5) c
B21 Nuclear Boiler System (NBS)						
1. Level instrumentation condensing chambers	1	CV	A	Q	I	
2. Safety relief valves (SRVs) and depressurization valves (DPVs)	1	CV	A	Q	I	
3. Safety relief discharge piping (including supports)	3	CV	C	Q	I	
4. Nitrogen accumulators (for ADS and manual actuation of SRVs)	3	CV	C	Q	I	
5. Piping and valves (including supports) for main steamlines (MSL) and feedwater (FW) lines up to and including the outermost containment isolation valves	1	CV, RB	A	Q	I	

Table 3.2-1 Classification Summary (Sheet 2 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
6. Piping (including supports) for MSL from outermost isolation valve to and including seismic interface restraint	2	RB	B	Q	I	Seismic interface restraints are located inside the seismic category I building.
7. Deleted.						
8. Piping and valves (including supports) for FW from outermost isolation valve to the seismic interface restraint	2	RB	B	Q	I	
9. Pipe whip restraints	3	CV, RB	—	Q	I or II	**Pipe Whip Restraints** —Pipe Whip Restraints are required on the MSL and FW piping.
10. Main steam drain piping and valves (including supports) within outermost containment isolation valves	1	CV, RB	A	Q	I	(7)
11. RPV head vent piping and valves (including supports) to the main steamline and to the second isolation valve	1	CV	A	Q	I	
12. Piping (including supports) for main steam drains inboard of outermost MSL isolation valves from outermost containment isolation valves up to and including the seismic restraints	N	RB	B	S	I	(5) a
13. Piping and valves (including supports) for main steam drains beyond outermost MSL isolation valves up to and including second drain isolation valve and associated restricting orifice or seismic restraint	N	TB	B	S	II	(5) c
14. Piping (including supports) for safety-related instrumentation up to but excluding the process instrument, and for nonsafety-related instrumentation up to and including the first instrument isolation valve	2	CV, RB	B	Q	I	(7)

Table 3.2-1 Classification Summary (Sheet 3 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
15. Piping and valves (including supports) for nonsafety-related instrumentation downstream of first instrument isolation valve	N	CV, RB	D	N	NS	(7)
16. Other mechanical modules with safety-related function	3	CV, RB, CB	—	Q	I	
17. Other electrical modules, cable, and instrumentation with safety-related function	3	CV, RB, CB	—	Q	I	
18. Components (piping, valves, fittings) for the above-valve-seat main steam drain piping from downstream of the seismic restraint, and also for the main steam low-point drain piping from the second drain isolation valve, to the condenser nozzle connection.	N	TB	D	S	NS	(5) c Analyzed to demonstrate structural integrity under SSE conditions.
19. Electrical modules, cables and instrumentation supporting diverse protection functions	N	CV, RB, TB	—	S	II	(5) c, (5) i, (5) j
B32 Isolation Condenser System (ICS)						
1. Steam supply line piping and valves (including supports) from the reactor up to and including the venturis outside containment and purge line returning to main steamline	1	CV, RB	A	Q	I	
2. Isolation condenser and piping outside containment from the supply line venturis to the condensate return line tee.	2	RB	B	Q	I	
3. Condensate return line piping and valves (including supports) from the reactor to the tee connection outside containment	1	CV, RB	A	Q	I	
4. Vent piping and valves (including supports) to suppression pool	2	CV, RB	B	Q	I	

Table 3.2-1 Classification Summary (Sheet 4 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
5. Electrical modules and cable with safety-related function	3	CV, RB	—	Q	I	
6. Pneumatic accumulators	3	CV, RB	C	Q	I	
7. Electrical modules and cables supporting diverse protection functions	N	CV, RB	—	S	II	(5) c, (5) i, (5) j
8. Pool cross-connect valves	3	RB	C	Q	I	
9. Electrical modules and cables supporting ICS lower header temperature monitoring	N	RB	—	S	II	(5)c
C CONTROL AND INSTRUMENT SYSTEMS						
C11 Rod Control and Information System (RC&IS)	N	RB, CB	—	S / N	NS	(5) j
C12 Control Rod Drive (CRD) System						
1. CRD primary pressure boundary	1	CV	A	Q	I	
2. CRD internals	3	CV	—	Q	I	
3. Hydraulic control unit (HCU)	2	RB	—	Q	I	(8)
4. Piping including supports – insert line	2	CV, RB	B	Q	I	
5. High pressure makeup piping including supports, from and including the check valve and test valve in the common line, isolation valves and isolation bypass valves up to the connection to RWCU/SDC	2	RB	B	Q	I	CRD piping classification is consistent with piping to which it connects.
6. Piping and valves with no safety-related function (pump suction, pump discharge, drive header, and other piping not part of HCU)	N	RB	D	S	II	(5) c, (7) , (5) k – for other risk-significant equipment
7. CRD water pumps	N	RB	D	S	II	(5) c
8. Fine motion drive motor	N	CV	—	S	II	(5) c
9. Electrical modules, solenoids, and cable with safety-related function	3	CV, RB, CB	—	Q	I	

Table 3.2-1 Classification Summary (Sheet 5 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
10. Electrical modules and cables supporting anticipated transients without scram (ATWS) Alternate Rod Insertion (ARI) and diverse protection functions	N	RB	—	S	II	(5) c, (5) f, (5) i, (5) j
C21 Leak Detection and Isolation System (LD&IS)						
1. Electrical modules (temperature sensors, pressure transmitters, etc.) and cable with safety-related function	3	CV, RB, CB	—	Q	I	
2. Other electrical modules and cable with no safety-related function	N	CV, RB, CB	—	N	NS	
C31 Feedwater Control System (FWCS)						
1. Electrical modules and cables supporting ATWS and diverse protection functions	N	TB, CB, EB	—	S	NS	(5) f, (5) j
2. Other equipment	N	CV, TB, RB, CB, EB	—	N	NS	
C41 Standby Liquid Control (SLC) System						
1. Standby liquid control accumulator including supports and vents	2	RB	B	Q	I	
2. Valves – injection	1	RB	A	Q	I	
3. Piping and valves (including supports) between injection valves and reactor vessel	1	CV, RB	A	Q	I	(7)
4. Piping and valves (including supports) upstream of injection valves and downstream of automatic N_2 makeup valve	2	RB	B	Q	I	(7)
5. N_2 gas bottles and associated piping up to automatic N_2 makeup valve	N	RB, SB	—	N	NS	
6. Electrical modules and cable with safety-related function	3	RB, CB	—	Q	I	

Table 3.2-1 Classification Summary (Sheet 6 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
7. Electrical modules and cables supporting diverse protection functions	N	RB, CB	—	S	II	(5) c, (5) j, (5) f – for ATWS equipment, (5) i – for RTNSS equipment
8. Electrical modules and cable – others	N	RB, CB	—	N	NS	
9. Piping and valves used for poison solution fill/makeup from the fill/makeup isolation valve downstream to the accumulators	2	RB	B	Q	I	
10. Other equipment used for poison fill/makeup, sampling and mixing	N	RB	—	N	NS	
C51 Neutron Monitoring System (NMS)						
1. Detector and tube assembly – primary pressure boundary	2	CV	B	Q	I	
2. Detector and tube assembly – internals	3	CV	C	Q	I	
3. Electrical modules and cable – SRNM, LPRM, APRM and OPRM	3	CV, CB, RB	—	Q	I	
4. Electrical modules and cables supporting diverse protection functions	N	CV, RB, CB	—	S	II	(5) c, (5) j
C61 Remote Shutdown System (RSS)						
1. Safety-related panels	3	RB	—	Q	I	
2. Nonsafety-Related panels	N	RB	—	S	II	(5) c
C62 NonSafety-Related Distributed Control and Information System (DCIS)						

Table 3.2-1 Classification Summary (Sheet 7 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
1. Electrical modules and cable with no safety-related function	N	ALL	—	S / N	II/NS	(5) c, (5) i Components whose failure can potentially adversely affect Seismic Category I components (e.g., in main control room) are required to be Seismic Category II and Safety-Related Classification S. Otherwise the components are Seismic Category NS and Safety-Related Classification N.
2. Performance Monitoring and Control Subsystem equipment	N	CB	—	S	II	(5) c
C63 Safety-Related DCIS						
1. Electrical modules and cables with safety-related function	3	RB, CB	—	Q	I	
C71 Reactor Protection System (RPS)	3	CB, TB, RB	—	Q	I	
C72 Diverse Protection System	N	CB, RB	—	S	NS	(5) f, (5) i, (5) j
C74 Safety System Logic and Control (SSLC)	3	RB, CB	—	Q	I	
C82 Plant Automation System	N	CB	—	N	NS	
C85 Steam Bypass and Pressure Control (SB&PC) System	N	CB	—	N	NS	
D RADIATION MONITORING SYSTEMS						
D11 Process Radiation Monitoring System (PRMS)						
1. Radiation monitors and sensors with safety-related function	3	RB, CB, FB	—	Q	I	
2. Fission product monitoring piping and valves (including supports) forming part of the containment boundary	2	CV, RB	B	Q	I	
3. Electrical modules and cable with safety-related function	3	CV, RB, CB, FB	—	Q	I	
4. Fission product monitoring system (other portions)	N	CV, RB, CB	—	N	NS	

Table 3.2-1 Classification Summary (Sheet 8 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
5. Other electrical modules and cable with no safety-related function	N	ALL	—	N	NS	
D21 Area Radiation Monitoring System (ARMS)	N	ALL, except CV	—	N	NS	
E CORE COOLING SYSTEMS						
E50 Gravity-Driven Cooling System (GDCS)						
1. Piping and valves (including supports) connected with the reactor vessel, including the squib valves, and up to and including the check valves upstream of the squib valves	1	CV	A	Q	I	
2. Piping and valves (including supports) from the check valves upstream of the squib valves to the suppression pool and GDCS pools	2	CV	B	Q	I	
3. Piping and valves (including supports) from the GDCS pools to the lower drywell	2	CV	B	Q	I	
4. Safety-related electrical modules, components and cables	3	CV, RB, CB	—	Q	I	
5. GDCS pool splash guard and perforated plate	3	CV	—	Q	I	
6. Nonsafety-Related electrical modules, components and cable	N	CV, RB, CB	—	S	II	(5) c, (5) i, (5) j, (5) k – for deluge function temperature sensors
F REACTOR SERVICING EQUIPMENT						
F11 Fuel Servicing Equipment						
1. Fuel Preparation Machine	N	FB	—	S	I	(5) a
2. New Fuel Inspection Stand	N	FB	—	S	II	(5) c
3. All Other Equipment	N	FB, RB	—	N	NS	

Table 3.2-1 Classification Summary (Sheet 9 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
F12 Miscellaneous Servicing Equipment	N	FB, RB	—	N	NS	
F13 Reactor Pressure Vessel Servicing Equipment						
1. RPV head holding pedestal	N	RB	—	S	I	(5) c
2. All other RPV servicing equipment	N	RB	—	N	NS	
F14 RPV Internal Servicing Equipment	N	RB	—	N	NS	
F15 Refueling Equipment						
1. Fuel Handling Machine	N	FB	—	S	I	(5) a
2. Refueling Machine	N	RB	—	S	I	(5) a
3. (Deleted)						
F16 Fuel Storage Racks						
1. Fuel storage racks - new and spent	N	RB, FB	—	S	I	(5) a
F17 Under-RPV Servicing Equipment	N	CV	—	N	NS	
F21 CRD Maintenance Facility	N	FB	—	N	NS	
F32 Fuel Cask Cleaning Facility	N	FB	—	N	NS	
F41 Plant Startup and Test Equipment	N	CV, RB, CB, TB, FB	—	N	NS	
F42 Fuel Transfer System (FTS)						
1. Transfer tube assembly from interface with upper fuel pool, through building to lower spent fuel pool terminus equipment, including drain connection	N	RB, FB	D	S	I	(5) a
2. Remaining equipment	N	RB, FB	D/—	S / N	II/NS	(5) c See Figure 9.1-2 for clarification of seismic classification boundaries. Seismic Category II items are Safety-Related Classification S.
G DECAY HEAT REMOVAL NETWORK						

Table 3.2-1 Classification Summary (Sheet 10 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
G21 Fuel and Auxiliary Pools Cooling System (FAPCS)						
1. Piping and valves including supports between containment isolation valves (including valves) for – Suppression pool return line – GDCS pool suction line – GDCS pool return line – Drywell spray discharge line	2	CV, RB	B	Q	I	
2. Piping between inboard manual valve and second outboard containment isolation valve on suppression pool suction line, as well as the low pressure coolant injection (LPCI) piping between the RWCU/SDC interface and the second isolation valve.	2	CV, RB	B	Q	I	
3. Independent line (including piping, valves, and supports) for safety-related makeup to IC/PCCS and spent fuel pools from piping connections at grade level in reactor yard area and to the fire protection system.	3	OO, RB, FB	C	Q	I	
4. GDCS pool interconnecting pipes	3	CV	C	Q	I	
5. Piping and components outside containment needed for fuel pool cooling, suppression pool cooling, LPCI and drywell spray modes of operation including skimmer lines and all components in the cooling and cleanup trains.	N	RB, FB	B	S	II	(5) b, (5) c, (5) i – for RTNSS equipment, (5) k – for other risk-significant equipment
6. Suppression pool suction line inside containment between inboard manual valve and its termination point (including suction strainers)	N	CV	B	S	I	(5) b, (5) c, (5) i – for RTNSS equipment

Table 3.2-1 Classification Summary (Sheet 11 of 30)

	Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
7.	Piping and valves inside containment between inboard containment isolation valves and their termination points inside containment for: – Suppression pool return line – Drywell spray discharge line	N	CV	C	S	I	(5) b, (5) c, (5) i – for RTNSS equipment
8.	Piping and valves inside containment between inboard containment isolation valves and their termination points inside containment for: – GDCS pool suction line – GDCS pool return line	N	CV	D	S	II	(5) c
9.	IC/PCCS pools active cooling and cleanup subsystem piping, and components.	N	RB	D	S	II	(5) c
10.	Auxiliary pools skimmer lines, and auxiliary pool return lines between isolation valves and terminus points, and all piping and mechanical components associated with pool liner leak detection.	N	RB, FB	D	N	NS	
11.	Instrument sensing lines for the following parameters – IC/PCCS pool water level – Spent fuel pool level – Buffer pool level	3	RB, FB	C	Q	I	
12.	Electrical modules and cables with safety-related function (containment isolation, LPCI isolation)	3	RB, CB, CV, FB	—	Q	I	
13.	Electrical modules and cables with nonsafety-related function	N	RB, CB, FB	—	S	II	(5) c, (5) i – for RTNSS equipment, (5) j – for diverse protection equipment, (5) k – for other risk-significant equipment
14.	Control and instrumentation required for safety-related functions	3	RB, CB	—	Q	I	

Table 3.2-1 Classification Summary (Sheet 12 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
15. Controls and instrumentation required for nonsafety-related functions	N	RB, FB, CB	—	S	II	5) c, (5) i – for RTNSS equipment, (5) j – for diverse protection equipment, (5) k – for other risk-significant equipment
G31 Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System						
1. Piping including supports and valves in the mid-vessel pump suction line from the RPV up to but excluding the flow control valve and up to but excluding the last check valve in the Train A post-LOCA return line to the RPV	1	CV, RB	A	Q	I	(7)
2. Piping including supports and valves from feedwater lines to and including shutoff valves	2	RB	B	Q	I	(7) RWCU/SDC piping classification is consistent with piping to which it connects.
3. Vessels including supports (demineralizer)	N	RB	C	S	I	(5) b, (5) c
4. Regenerative heat exchangers (including supports) carrying reactor water	N	RB	C	S	I	(5) b, (5) c
5. Cleanup recirculation pump, motors	N	RB	C	S	I	(5) b, (5) c
6. Other piping including supports and valves from and including the flow control valve in the mid-vessel suction line and from and including the first motor-operated valve in the bottom head suction line to but excluding the motor-operated shutoff valves at feedwater line connections	N	RB	C	S	I	(5) b, (5) c, (7)
7. Nonregenerative heat exchanger tube side and piping (including supports and valves) carrying process water	N	RB	C	S	I	(5) b, (5) c
8. Nonregenerative heat exchanger shell and piping (including supports and valves) carrying cooling water	N	RB	D	S	I	(5) c
9. Sample station	N	RB	D	S	I	(5) c

Table 3.2-1 Classification Summary (Sheet 13 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
10. Electrical modules, cable and instrumentation with safety-related function	3	RB, CB	—	Q	I	
11. Electrical modules, cable and instrumentation with no safety-related function	N	RB, CB	—	S	II	(5) c, (5) j
12. Overboard line piping outside reactor building	N	TB	D	S	II	(5) b, (5) c
13. Cross-tie piping including supports and valves in the post-accident containment heat removal return line to FAPCS up to and including the spectacle flange	N	RB	C	S	I	(5) b, (5) c
14. Cross-tie piping including supports and valves in the post-accident containment heat removal return line to FAPCS from but excluding the spectacle flange up to and including the first downstream manual isolation valve	N	RB	C	S	II	(5) b, (5) c
15. Cross-tie piping including supports and valves in the post-accident containment heat removal suction line from FAPCS from and including the first upstream manual isolation valve up to but excluding the spectacle flange	N	RB	C	S	II	(5) b, (5) c
16. Cross-tie piping including supports and valves in the post-accident containment heat removal suction line from FAPCS from and including the spectacle flange to the mid-vessel suction line	N	RB	C	S	I	(5) b, (5) c
17. Cross-tie piping including supports and valves up to and including the last check valve for post-accident return flow to mid vessel suction line	N	RB	C	S	I	(5) b, (5) c

Table 3.2-1 Classification Summary (Sheet 14 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
18. Piping including supports and valves in the bottom head suction line from the RPV up to but excluding the first motor-operated valve, and up to and including the outboard isolation valve in the branch line to the sample station	1	CV, RB	A	Q	I	(7)
H CONTROL PANELS						
H11 Main Control Room Panels						
1. Panels, electrical modules, and cable with safety-related function	3	CB	—	Q	I	**Control Panels** — Panels and associated structures that support or house safety-related mechanical or electrical components are safety-related.
2. Panels, electrical modules, and cable with no safety-related function	N	CB	—	S	II	(5) c
H12 MCR Back Room Panels						
1. Panels, electrical modules, and cable with safety-related function	3	CB	—	Q	I	**Control Panels** — Panels and associated structures that support or house safety-related mechanical or electrical components are safety-related.
2. Panels, electrical modules, and cable with no safety-related function	N	CB	—	S	II	(5) c
H14 Radwaste Control Room Panels	N	RW	—	S	NS	(5) d
H21 Local Panels and Racks						
1. Panels, electrical modules, and cable with safety-related function	3	ALL	—	Q	I	**Control Panels** – Panels and associated structures that support or house safety-related mechanical or electrical components are safety-related.
2. Panels, electrical modules, and cable with no safety-related function	N	ALL	—	N	NS	
J NUCLEAR FUEL						
J10 Core and Fuel Services	No physical items to be classified					

Table 3.2-1 Classification Summary (Sheet 15 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
J11 Nuclear Fuel	3	CV, RB, FB	—	Q	I	Nuclear fuel and channels are designed in accordance with NRC-approved methodology as described in chapters 4, 15 and Reference 3.2-9.
J12 Fuel Channel	3	CV, RB, FB	—	Q	I	See note for J11.
K RADIOACTIVE WASTE MANAGEMENT SYSTEMS						
K10 Liquid Waste Management System (LWMS)						
1. Mechanical modules (including supports)	N	RB, RW	D	S	NS	(5) d
2. Electrical modules and cabling	N	RB, RW	—	S	NS	(5) d
K20 Solid Waste Management System (SWMS)						
1. Mechanical modules (including supports)	N	RB, RW	D	S	NS	(5) d
2. Electrical modules and cabling	N	RB, RW	—	S	NS	(5) d
K30 Offgas System (OGS)	N	TB	D	S	NS	(5) d
N POWER CYCLE SYSTEMS						
N11 Turbine Main Steam System (TMSS)						
1. TMSS consists of the piping (including supports) for the MSL from the seismic interface restraint (or seismic guide) to the turbine stop valves (TSVs), turbine bypass valves and the connecting branch lines up to and including their isolation valves.	N	TB	B	S	II	(5) a **Main Steamlines** – TMSS lines are designed to ASME Section III Code, Class 2. Lines smaller than 63.5 mm (2.5 inches) are NS. Also see Figure 3.2-1.
2. Other mechanical and electrical modules	N	TB	D	N	NS	
N21 Condensate and Feedwater System (C&FS)						
1. Main feedwater line beyond seismic interface restraint	N	TB	D	N	NS	See Figure 3.2-2

Table 3.2-1 Classification Summary (Sheet 16 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
2. Electrical modules, cable and instrumentation associated with diverse protection functions	N	TB	—	S	NS	(5) j
N22 Heater Drain and Vent System (HDVS)	N	TB	—	N	NS	
N25 Condensate Purification System (CPS)	N	TB	D	N	NS	
N31 Main Turbine						
1. TSVs, turbine control valves (TCVs) and main steam leads from the TSVs to the turbine casing	N	TB	D	N	NS	(9)
2. All other system components	N	TB	—	N	NS	
N32 Turbine Generator Control System (TGCS)						
1. Electrical modules and cables associated with diverse protection functions	N	TB	—	S	NS	(5) j
2. All other components	N	TB	—	N	NS	
N33 Turbine Gland Seal System (TGSS)	N	TB	D	N	NS	
N34 Turbine Lube Oil System (TLOS)	N	TB	—	N	NS	
N35 Moisture Separator Reheater (MSR)	N	TB	—	N	NS	
N36 Extraction System	N	TB	—	N	NS	
N37 Turbine Bypass System (TBS)	N	TB	D	S	NS	(5) c Analyzed to demonstrate structural integrity under SSE loading conditions. TMSS lines up to the turbine bypass valves are designed to ASME Section III Code, Class 2. Lines smaller than 63.5 mm (2.5 inches) are NS. Also see Figure 3.2-1.

Table 3.2-1 Classification Summary (Sheet 17 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
N38 Turbine Hydraulics	N	TB	—	N	NS	
N39 Turbine Auxiliary Steam System (TASS)	N	TB	—	N	NS	
N41 Generator	N	TB	—	N	NS	
N42 Hydrogen Gas Control System (HGCS)	N	TB	—	N	NS	
N43 Stator Cooling Water System (SCWS)	N	TB	—	N	NS	
N44 Generator Lube and Seal Oil System (GLSOS)	N	TB	—	N	NS	
N45 Hydrogen and Carbon Dioxide Bulk Gas Storage System	N	OO	—	N	NS	
N51 Generator Excitation System (GES)	N	TB	—	N	NS	
N61 Main Condenser and Auxiliaries						See Figure 3.2-1.
1. Condenser anchorage	N	TB	—	S	NS (see note)	(5) c The condenser anchorage is seismically analyzed for SSE.
2. Condenser air removal system	N	TB	D	N	NS	
3. All other main condenser and auxiliaries components	N	TB	—	N	NS	
N71 Circulating Water System (CIRC)	N	TB, OO	D	N	NS	
P STATION AUXILIARY SYSTEMS						
P10 Makeup Water System (MWS)						
1. Piping and valves (including supports) forming part of the containment boundary	2	CV, RB	B	Q	I	
2. Piping and valves inside containment or inside Reactor Building	N	CV, RB	D	S	II	(5) c

Table 3.2-1 Classification Summary (Sheet 18 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
3. Other mechanical and electrical modules	N	OO, RW, RB, CB, SF	D	N	NS	
P21 Reactor Component Cooling Water System (RCCWS)						
1. Piping and valves inside Reactor and Fuel Buildings	N	RB, FB	D	S	II	(5) c, (5) i
2. Other mechanical and electrical modules	N	TB, RB, FB, EB	D	S / N	NS	(5) i – for RTNSS equipment
P22 Turbine Component Cooling Water System (TCCWS)	N	TB	D	N	NS	
P25 Chilled Water System (CWS)						
1. Piping and valves (including supports) forming part of the containment boundary	2	CV, RB	B	Q	I	
2. Piping and valves inside containment and Reactor Building	N	CV, RB	D	S	II	(5) c, (5) i
3. Other mechanical and electrical modules	N	TB, RB, CB, FB, EB, RW	D	S / N	NS	(5) i – for RTNSS equipment
P30 Condensate Storage and Transfer System (CS&TS)						
1. Mechanical modules, including piping and valves, in Reactor Building	N	RB	D	S	II	(5) c
2. Other mechanical modules, including piping, valves, and condensate storage tank	N	OO, RW, TB	D	N	NS	
3. Electrical modules and cable	N	RB	—	N	NS	
P32 Oxygen Injection System (OIS)	N	TB	—	N	NS	
P33 Process Sampling System (PSS)	N	RB, OO, TB, RW	D	N	NS	(7)
P41 Plant Service Water System (PSWS)						

Table 3.2-1 Classification Summary (Sheet 19 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
1. Mechanical and electrical modules, including piping and valves (including supports)	N	SF, OO, TB	D	S / N	NS	(5) i – for RTNSS equipment
P51 Service Air System (SAS)						
1. Piping and valves (including supports) forming part of the containment boundary	2	CV, RB	B	Q	I	
2. Other system components	N	ALL	D	N	NS	
P52 Instrument Air System (IAS)	N	ALL	D	N	NS	
P54 High Pressure Nitrogen Supply System (HPNSS)						
1. Piping and valves (including supports) forming part of the containment boundary	2	CV, RB	B	Q	I	
2. Other Nonsafety-Related mechanical modules	N	RB	D	N	NS	
3. Other Nonsafety-Related electrical modules	N	RB, CB	—	N	NS	
4. Nitrogen storage bottles	N	RB	—	N	NS	
P62 Auxiliary Boiler System (ABS)	N	OL	—	N	NS	
P73 Hydrogen Water Chemistry System	N	TB	—	N	NS	The site-specific plant design includes the HWCS. See Subsection 9.3.9 for further details.
P74 Zinc Injection System	N	TB	D	N	NS	The site-specific plant design does not include the Zinc Injection System.
R STATION ELECTRICAL SYSTEMS						
R10 Electrical Power Distribution System (EPDS)						
1. Main transformers	N	OO	—	N	NS	
2. Main generators	N	TB	—	N	NS	
3. Reserve and unit auxiliary transformers	N	OO	—	N	NS	
4. Isolated phase bus duct	N	OO, TB	—	N	NS	

Table 3.2-1 Classification Summary (Sheet 20 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
5. Non-segregated bus duct	N	OO, EB	—	N	NS	
6. Metal clad switchgear	N	RB, EB, TB, OL	—	N	NS	
7. Power centers	N	RB, EB, FB, TB, OL	—	N	NS	
8. Motor control centers	N	RB, EB, FB, CB, TB, OL	—	N	NS	
9. (Deleted)						
10. Other cable and supports with no safety function	N	CV, CB, RB, EB, TB, OL	—	N	NS	
R11 Medium Voltage Distribution System						
1. Power Generation (PG) buses	N	EB, TB, CP, OO	—	N	NS	
2. Plant Investment Protection (PIP) buses	N	EB, RB, TB, SF, ADB, RW, OO	—	S	NS	(5) i
R12 Low Voltage Distribution System						
1. Components supporting distribution of power from Ancillary Diesel Generators to RTNSS Criterion B structures, systems and components	N	ADB, RB, FWSC, CB, SB	—	S	II	(5) c, (5) h
2. Components supporting distribution of power from Ancillary Diesel Generators to RTNSS non-Criterion B structures, systems and components	N	ADB, CB, SB, EB	—	S	NS	(5) i
3. Nonsafety-related components designated as risk significant but not RTNSS	N	RB, EB	—	S	NS	(5) k
4. 4.All other components	N	ALL	—	N	NS	

Table 3.2-1 Classification Summary (Sheet 21 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
R13 Uninterruptible AC Power Supply						
1. Electrical modules and cable with safety-related function	3	CV, CB, RB	—	Q	I	
2. Other electrical modules and cable with no safety function	N	CV, RB, CB, EB, TB, OL	—	S / N	NS	(5) h, (5) i – for RTNSS equipment, (5) k – for other risk-significant equipment
R15 Lighting and Servicing Power Supply						
1. Lighting	N	ALL	—	N	NS	Components of the lighting systems associated with safety-related systems and emergency exit lighting are supported to Seismic Category I requirements.
2. Emergency lighting in main control room and remote shutdown system rooms	N	CB, RB	—	S	I	(5) c, (5) h Safety-related power is provided through isolation devices. The seismic classification applies to the supports for the lighting fixtures, not to the bulbs and fixtures.
R16 Direct Current Power Supply						
1. Electrical modules and cable with safety-related function	3	RB	—	Q	I	
2. Other electrical modules and cable with no safety function	N	EB, RB, TB, RW, SF, CP, OO, OL	—	S / N	NS	(5) h, (5) i – for RTNSS equipment, (5) k – for other risk-significant equipment
R21 Standby AC Power Supply						
1. Ancillary diesel generators and their support equipment	N	ADB	—	S	II	(5) c, (5) h
2. Standby diesel generators and their supporting equipment	N	EB	—	S	NS	(5) i
R31 Raceway System						
1. Conduit, cable trays and supports with safety-related function	3	CV, CB, RB, FB, TB	—	Q	I	

Table 3.2-1 Classification Summary (Sheet 22 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
2. Other electrical modules with no safety function	N	CV, CB, RB, EB, TB, OL	—	S / N	NS	(5) h, (5) i – for RTNSS equipment
3. Electrical penetrations	3	CV, RB	—	Q	I	
R41 Plant Grounding System	N	OO	—	N	NS	
R51 Communication System	N	ALL	—	S / N	NS	(5) c System components are mounted to Seismic Category II requirements in safety-related areas.
S POWER TRANSMISSION SYSTEMS						
S21 Switch Yard	N	OO	—	N	NS	
T CONTAINMENT AND ENVIRONMENTAL CONTROL SYSTEMS						
T10 Containment System						
1. Upper and lower drywell airlocks and equipment hatches, wetwell access hatch, and safety-related instrumentation	2	CV	B	Q	I	
2. Wetwell/drywell vacuum breakers	2	CV	B	Q	I	
3. Vacuum Breaker “Closed” Proximity Instrumentation	3	CV	—	Q	I	
4. Vacuum Breaker “Open” Proximity Instrumentation.	3	CV	—	Q	I	
5. Vacuum Breaker Isolation Valves	2	CV	B	Q	I	
6. Refueling bellows	N	CV	—	S	I	(5) c
7. Vacuum Breaker/Isolation Valve Temperature Sensor Instrumentation	3	CV	—	Q	I	
8. Basemat Internal Melt Arrest Coolability (BiMAC) device	N	CV	—	S	NS	(5) i
9. GDCS pool spillover pipes	N	CV	—	S	II	(5) c
T11 Containment Vessel						
1. Drywell head	2	CV	B	Q	I	

Table 3.2-1 Classification Summary (Sheet 23 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
2. Reinforced Concrete Containment Vessel (RCCV)	2	CV	B	Q	I	
3. Reactor pedestal (Part of RCCV)	2	CV	B	Q	I	
4. Portion of basemat under pedestal	2	CV	B	Q	I	
T12 Containment Internal Structures						
1. Reactor vessel support brackets and stabilizer support	3	CV	—	Q	I	
2. Support structures for safety-related piping, including supports and equipment	3	CV	—	Q	I	
3. Reactor shield wall	3	CV	—	Q	I	
4. Diaphragm floor	3	CV	—	Q	I	
5. GDCS pools	3	CV	—	Q	I	
6. Vent Wall	3	CV	—	Q	I	
T15 Passive Containment Cooling System (PCCS)						
1. All components other than vent fans and vent fan piping	2	CV	B	Q	I	
2. Vent fans and vent fan piping	N	CV	B	S	II	(5) b, (5) c, (5) h
3. Vent Line Catalyst Module	2	CV	B	Q	I	
T31 Containment Inerting System						
1. Piping and valves (including supports) forming part of the containment boundary	2	RB	B	Q	I	
2. Electrical modules and cables with safety-related function	3	RB, CB	—	Q	I	
3. Other mechanical modules (including nitrogen storage tanks, and vaporizers), piping, valves, and electrical modules and cables with no safety function	N	RB, OO	—	N	NS	

Table 3.2-1 Classification Summary (Sheet 24 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
4. Hardened containment vent line to RB/FB stack	N	RB	—	S	NS	(5) k
T41 Drywell Cooling System (DCS)	N	CV	—	S	II	(5) c
T49 Passive Autocatalytic Recombiner System (PARS)						
1. PARS units	N	CV	-	S	I	(5) c, (5) h
2. Igniters in lower drum of PCCS condensers	N	CV	-	S	II	(5)c
3. Electrical penetration for igniters	3	CV, RB	-	Q	I	
T62 Containment Monitoring System						
1. Mechanical components involved in containment isolation function	2	CV, RB	B	Q	I	
2. Other safety-related mechanical components	3	CV, RB, CB	C	Q	I	
3. Safety-related electrical modules, cables and instrumentation	3	CV, RB, CB	—	Q	I	
4. Electrical modules, cables and instrumentation supporting diverse protection functions	N	CV, RB, CB	—	S	II	(5) c, (5) j
5. Other nonsafety-related portions of system	N	CV, RB, CB	—	N	NS	
T64 Environmental Monitoring System	N	OL	—	N	NS	
U STRUCTURES AND SERVICING SYSTEMS						
U31 Cranes, Hoists, and Elevators						
1. Reactor building cranes, fuel building crane	N	RB, FB	—	S	I	(5) a **Cranes** — The reactor building and fuel building cranes are designed to maintain their position and hold up their loads under conditions of an SSE.

Table 3.2-1 Classification Summary (Sheet 25 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
2. Upper and lower drywell servicing hoists and cranes	N	CV	—	S	II	(5) c
3. Main steam tunnel servicing hoists and cranes	N	OL	—	S	II	(5) c
4. Special service rooms hoists and cranes	N	RB, TB, FB, RW	—	S or N	II or NS	(5) c Components must be seismic category II and Safety-Related Classification S if they can potentially damage safety-related equipment.
5. Elevators	N	RB, TB, FB, CB, RW, EB	—	N	NS	
U36 Electrical Building HVAC	N	EB	—	S / N	NS	(5) i – for RTNSS equipment
U37 Service Building HVAC	N	SB	—	N	NS	
U38 Radwaste Building HVAC	N	RW	—	S	NS	(5) d
U39 Turbine Building HVAC	N	TB	—	S / N	NS	(5) i – for RTNSS equipment
U40 Reactor Building HVAC						
1. Building isolation dampers	3	RB	—	Q	I	
2. Controls associated with the isolation dampers	3	RB	—	Q	I	
3. (Deleted)						
4. Other system components	N	RB	—	S	II	(5) c, (5) i – for RTNSS equipment
U41 Other Building HVAC	N	OL	—	N	NS	
U42 Potable Water and Sanitary Waste System	N	CB, SB, EB, RB, OO, TB, OL	—	N	NS	
U43 Fire Protection System (FPS)						
1. Non-seismic yard piping and valves including supports (includes secondary piping in Turbine and other Buildings supplied by yard piping)	N	OO, OL, TB, EB, RW, SB	D	S	NS	(5) e

Table 3.2-1 Classification Summary (Sheet 26 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
2. Seismic Category I piping and valves including supports providing source of makeup water to IC/PCCS and fuel pools	N	FWSC, OL	D	S	I	(5) c, (5) e, (5) h
3. Seismic Category II piping and valves including supports (includes balance of primary piping and valves)	N	FWSC, OL, RB, CB, FB	D	S	II	(5) c, (5) e
4. Primary firewater storage tanks	N	FWSC	D	S	I	(5) c, (5) e, (5) h
5. Secondary firewater storage	N	OO	D	S	NS	(5) e
6. (Deleted)						
7. Primary diesel-driven fire pump	N	FWSC	D	S	I	(5) c, (5) e, (5) h
8. Primary motor-driven fire pump	N	FWSC	D	S	II	(5) c, (5) e, (5) h
9. Other primary pumps	N	FWSC	D	S	II	(5) c, (5) e
10. Primary diesel fire pump fuel tank	N	FWSC	—	S	I	(5) c, (5) e, (5) h
11. Other pumps and motors	N	OO	D	S	NS	(5) e
12. Electrical modules and cables for RB preaction sprinklers	N	RB	—	S	NS	(5) e
13. All other electrical modules and cables	N	ALL	—	S	NS	(5) e
14. (Deleted)						
15. Sprinklers	N	RB, TB, RW, SB, EB, OL	D	S	NS	(5) e
16. Foam, preaction or deluge	N	EB, TB, ADB, OO	—	S	NS	(5) e
U44 Sanitary Waste Discharge System	N	CB, SB, EB, RB, OO	—	N	NS	
U50 Equipment and Floor Drain System						
1. Piping and valves forming part of the containment boundary	2	CV, RB	B	Q	I	

Table 3.2-1 Classification Summary (Sheet 27 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
2. Drain piping and valves, including supports, in Seismic Category I buildings	N	RB, FB	D	S	II	(5) c
3. Drain piping and valves, including supports, in other buildings	N	ALL except RB, FB	D	N	NS	
4. Other mechanical and electrical modules	N	ALL	—	N	NS	
U51 Oily Waste Drain System	N	TB	—	N	NS	
U61 Auxiliary Boiler Building Structure	N	OO	—	N	NS	
U62 Auxiliary Boiler Building HVAC System	N	OL	—	N	NS	
U63 Firewater Service Complex Structure	N	FWSC	—	S	I	(5) c, (5) e, (5) h
U64 Firewater Service Complex HVAC System	N	FWSC	—	S	II	(5) c, (5) e
U65 Other Building Structures						
1. (Deleted)						
2. Other buildings	N	OO, OL	—	N	NS	
U66 Access Tunnel Structures	N	OL	—	S	II	(5) c
U67 Radwaste Tunnel	N	OL	—	S	NS	(5) d Structural acceptance and material criteria for the Radwaste Tunnel are in accordance with RG 1.143, Safety Classification RW-IIa.
U68 Ancillary Diesel Building Structure	N	ADB	—	S	II	(5) c, (5) h
U69 Ancillary Diesel Building HVAC System	N	ADB	—	S	II	(5) c, (5) h
U71 Reactor Building Structure						
1. Main building	3	RB	—	Q	I	

Table 3.2-1 Classification Summary (Sheet 28 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
2. Stair towers, equipment removal access shaft and elevator shafts	N	RB	—	S	II	(5) c
3. Equipment storage pool, reactor well and buffer pool liners, and pool gates	3	RB	—	Q	I	
4. Reactor Building pressure relief devices	3	RB	—	Q	I	
U72 Turbine Building Structure	N	TB	—	S	II	(5) c
U73 Control Building Structure						
1. Main building	3	CB	—	Q	I	
2. Stair towers and elevator shaft	N	CB	—	S	II	(5) c
U74 Radwaste Building Structure	N	RW	—	S	NS	(5) d
U75 Service Building Structure	N	SB	—	S	II	(5) c
U77 Control Building HVAC						
1. Ducts, valves, and dampers (including supports) supporting safety-related areas	3	CB	—	Q	I	
2. Other ducts, valves and dampers (including supports)	N	CB	—	N	NS	
3. Electrical modules and cable with safety-related function	3	CB	—	Q	I	
4. Control Room air handling units and the air conditioning for their coils	N	CB	—	S	II	(5) c, (5) h
5. Other Nonsafety-Related equipment	N	CB	—	N	NS	
6. Emergency Filter Unit	3	CB	—	Q	I	
7. Safety-Related DCIS (Q-DCIS) room coolers	N	CB	—	S	II	(5) c, (5) h
U78 Cold Machine Shop	N	OO	—	N	NS	
U80 Electrical Building Structure	N	EB	—	S	NS	(5) i – Structure houses RTNSS C equipment
U81 Seismic Monitoring System	N	ALL	—	N	NS	
U84 Service Water Building Structure	N	SF	—	S	NS	(5) i – Structure houses RTNSS C equipment

Table 3.2-1 Classification Summary (Sheet 29 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
U85 Service Water Building HVAC	N	SF	—	N	NS	
U91 Administration Building Structure	N	OL	—	N	NS	
U93 Training Center	N	OL	—	N	NS	
U95 Hot Machine Shop	N	OO	—	N	NS	
U97 Fuel Building Structure						
1. Main building	3	FB	—	Q	I	
2. HVAC penthouse, stair towers and elevator shaft	N	FB	—	S	II	(5) c
3. Spent fuel pool liner and pool gates	3	FB	—	Q	I	
4. Fuel Building pressure relief devices	3	FB	—	Q	I	
U98 Fuel Building HVAC						
1. Building isolation dampers	3	FB	—	Q	I	
2. Ducting penetrating fuel building boundary	3	FB	—	Q	I	
3. Controls associated with the isolation dampers	3	FB	—	Q	I	
4. Other system components	N	FB	—	S	II	(5) c, (5) i – for RTNSS equipment
W INTAKE STRUCTURE AND SERVICING EQUIPMENT						
W12 Intake and Discharge Structures	N	OO	—	N	NS	
W24 Cooling Tower	N	OO	—	N	NS	
W32 Screen Cleaning Facility	N	OO	—	N	NS	
W33 Screens, Racks, and Rakes	N	OO	—	N	NS	
W41 Intake Structure Power Supply	N	OO	—	N	NS	
Y YARD STRUCTURES AND EQUIPMENT						
Y12 Roads and Walkways	N	OO	—	N	NS	

Table 3.2-1 Classification Summary (Sheet 30 of 30)

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	Safety-Related Classification[5]	Seismic Category[6]	Notes
Y21 Tanks and Equipment Pads	N	OO	—	N	NS	Some tanks in the yard area belong to other systems (e.g., firewater storage tank in U43) and have different classifications.
Y41 Station Water System	N	OO	—	N	NS	
Y46 Cathodic Protection System	N	OO	—	N	NS	
Y47 Meteorological Observation System	N	OO	—	N	NS	
Y51 Yard Miscellaneous Drain System	N	OO	—	N	NS	
Y52 Oil Storage and Transfer System						
1. System components supporting operation of ancillary diesel generators	N	ADB	—	S	II	(5) c, (5) h
2. All other system components	N	OO	—	N	NS	
Y53 Chemical Storage and Transfer System	N	OO	—	N	NS	
Y71 Piping Duct						Classification of individual piping ducts matches the highest classification of the pipe they carry.
1. Concrete Trench/Tunnel for Seismic Category I and II FPS Piping	N	OL	--	S	I/II	(5) c
2. Other Piping Duct	N	OL	--	N	NS	
Y72 Cable Duct						Classification of individual cable ducts matches the highest classification of the cables they carry.
1. Concrete duct banks between RB and CB	3	OL	--	Q	I	
2. Concrete duct banks between ancillary diesel building and other structures	N	OL	--	S	II	(5) c
3. Other Cable Duct	N	OL	--	N	NS	
Y86 Site Security	N	ALL	—	N	NS	

Notes:

1. Principal components: A module is an assembly of interconnected components that constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, and signal processors; and mechanical modules include turbines, strainers, and orifices.
2. Safety Class: 1, 2, 3 or N are designations for safety-related or nonsafety-related as discussed in Subsection 3.2.3.
3. Location codes:

ALL	=	All locations	ADB	=	Ancillary Diesel Building
CV	=	Containment Vessel	RW	=	Radwaste Building
CB	=	Control Building	CP	=	Circulating Water Pump House
RB	=	Reactor Building	SF	=	Service Water Building
OO	=	Outdoors Onsite	TB	=	Turbine Building
OL	=	Any Other Location	EB	=	Electrical Building
FB	=	Fuel Building	SB	=	Service Building
FWSC	=	Firewater Service Complex			

4. Quality group classifications: A, B, C, or D are quality groups defined in Regulatory Guide 1.26, as discussed in Subsection 3.2.2. The principal components are classified, designed, and constructed in accordance with the requirements identified in Tables 3.2-2 and 3.2-3. The designation “—” indicates that the quality groups A through D are not applicable to the associated principal component.
5. Safety-Related Classification: The designation “Q” indicates that the quality assurance requirements of 10 CFR 50, Appendix B, are applied in accordance with the quality assurance program described in Chapter 17. The designation “S” indicates that special quality assurance requirements are applied, commensurate with the importance of the item's function for one or more of the following reasons:
    a. Nonsafety-related structures, systems and components for which 10 CFR 50 Appendix B quality assurance requirements are to be fully applied.
    b. Nonsafety-related structures, systems and components required to be designed in accordance with Quality Group B or C requirements from RG 1.26. See note (4).
    c. Nonsafety-related structures, systems and components required to be designed in accordance with special seismic design requirements, such as Seismic Category I or II requirements. See note (6).
    d. Nonsafety-related structures, systems and components required to be designed in accordance with Radioactive Waste Management requirements from RG 1.143 for

Category RW-IIa (see Subsection 3.7.2.8.2 for further details on the design of the Radwaste Building and structures, systems and components housed inside the Radwaste Building). A quality assurance program meeting the guidance of NRC Regulatory Guide 1.143, as applied to radioactive waste management systems, is described in Chapter 17. The Radioactive Waste Management System components conform to Regulatory Guide 1.143 Table 1. For radwaste processing systems, Regulatory Guide 1.143 Table 1 modifies Regulatory Guide 1.26 Table 1 Quality Group D. This modification is acceptable per Standard Review Plan 3.2.2 Appendix C Note (9). Applicable portions of Regulatory Guide 1.143 Table 1 are reprinted in Chapter 11, Table 11.2-1. Exceptions to RG 1.143 requirements for the design of the radwaste building are defined in Chapter 2, Table 2.0-1.

e. Nonsafety-related structures, systems and components required to be designed in accordance with Fire Protection requirements from 10 CFR 50.48 and RG 1.189. A quality assurance program meeting the guidance of NRC Branch Technical Position SPLB 9.5-1 (NUREG-0800) is applied to the protection system. Also, special seismic qualification requirements are applied.

f. Nonsafety-related structures, systems and components required to be designed in accordance with ATWS requirements from 10 CFR 50.62. A quality assurance program that meets or exceeds the guidance of NRC Generic Letter 85-06 is applied to all nonsafety-related ATWS equipment.

g. Nonsafety-related structures, systems and components required to be designed in accordance with Station Blackout requirements from 10 CFR 50.63 and RG 1.155.

h. Nonsafety-related structures, systems and components required to be designed in accordance with RTNSS Criterion B requirements as specified in Appendix 19A.

i. Nonsafety-related structures, systems and components assigned to RTNSS criteria other than Criterion B that are required to be designed in accordance with RTNSS requirements as specified in Appendix 19A.

j. Nonsafety-related structures, systems and components associated with the performance of Diverse I&C functions that are required to be designed in accordance with a quality assurance program that meets or exceeds the guidance of NRC Generic Letter 85-06 as specified in Subsection 7.8.3.

k. Nonsafety-related structures, systems and components designated as risk significant, but that are not designated as RTNSS.

The designation “N” indicates that standard nonsafety-related quality assurance requirements are applied. See Subsection 17.1.22 for further details on the safety-related classification system.

6. Seismic category: The designations “I” or “II” indicate that the design requirements of Seismic Category I or II structures and equipment are applied as described in Subsection 3.2.1 and Section 3.7, Seismic Design. Structures and equipment that are not designated “I” or “II” are designated “NS.”

7. Small Piping and Instrument Lines — Lines 25 mm (one inch) and smaller in diameter that are part of the RCPB are Quality Group B and meet the requirements of the ASME B&PV Code, Section III, Class 2 and Seismic Category I, with the exceptions noted below:

   Instrument lines that are connected to the RCPB and are used to actuate or monitor safety-related systems are Quality Group B from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation. Instrument lines that are connected to the RCPB and are not used to actuate and monitor safety-related systems are nonsafety-related and Quality Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation. Other instrument lines meet the following requirements:

   - Through the root valve: the lines are the same classification as the system to which they are attached.
   - Beyond the root valve, if used to actuate a safety-related system: the lines are the same classification as the system to which they are attached.
   - Beyond the root valve, if not used to actuate a safety-related system: the lines may be Quality Group D.

   Sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system may be Quality Group D.

   Safety-related instrument lines comply with the guidance of NRC Regulatory Guide 1.151.

8. HCU for CRD system — Each HCU is a factory-assembled, engineered module of valves, tubing, piping, and stored water that controls two CRDs by the application of pressure and flow to accomplish rapid insertion for reactor scram.

   Although each HCU is field installed as a unit and connected to process piping, many of its internal parts differ markedly from process piping and components because of the more complex functions of the HCUs. Thus, although the codes and standards invoked by the different quality groups (A, B, C and D) apply to the interfaces between the HCUs and connections to conventional piping components (e.g., pipe nipples, fittings, hand valves, etc.), they are not considered applicable to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments).

   However, the design and construction specifications for the HCUs do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels. For example: (1) all welds are inspected using liquid penetrant, (2) all socket

welds are inspected for gaps between the pipe and socket bottom, (3) all welding is performed by qualified welders, and (4) all work is performed in accordance with written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group permit the use of manufacturer's standards and proven design techniques that are not explicitly defined within the codes for Quality Groups A, B or C. This is supplemented by appropriate quality control (QC) techniques.

9. Main Turbine —Turbine steam leads from the stop valves to the inlet nozzles, including stop and control valves, are Quality Group D and designed to withstand the SSE and maintain its pressure-retaining integrity.

   All cast pressure-retaining parts of a size and configuration for which volumetric methods are effective are examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards is used as an alternative to radiographic methods. Examination procedures and acceptance standards are at least equivalent to those defined in Paragraph 136.4, Nonboiler External Piping, ASME B31.1.

   The following qualifications are met with respect to the certification requirements:

   a. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steamlines from turbine control valve to turbine casing uses quality control procedures at least equivalent to those defined in GE Publication GEZ-4982A, General Electric Large Steam Turbine Generator Quality Control Program.
   b. A certification obtained from the manufacturer of these valves and steam lines demonstrates that the quality control program as defined has been accomplished.

   The following requirements are applied in addition to the Quality Group D requirements:

   a. All longitudinal and circumferential butt weld joints are radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination may be substituted. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ASME B31.1.
   b. All fillet and socket welds, and all structural attachment welds to pressure-retaining materials are examined by either magnetic particle or liquid penetrant methods. Examination procedures and acceptance standards are at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ASME B31.1.
   c. All inspection records are maintained for the life of the plant. These records include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

Table 3.2-2 Minimum Safety Class Requirements

Safety Class	Minimum Design Requirements for Specific Safety Class: Quality Group	ASME B&PV Section III Code Class	Seismic Category[1]	Electrical Classification[2]	Quality Assurance[4]
1	A	1	I	N/A	10 CFR 50 Appendix B
2	B	2, CC, MC or CS	I	N/A	10 CFR 50 Appendix B
3	C	3	I	Class 1E	10 CFR 50 Appendix B
N	D[3]	N[5]	II or NS	Non-Class 1E	—

Notes:

1. Seismic Category I structures, systems, and components meet the design and analysis requirements of Section 3.7. Some safety-related items (e.g., pipe whip restraints) have no safety-related function in the event of an SSE and are Seismic Category II.
2. Safety-related electrical equipment and instrumentation meet the design requirements of Institute of Electrical and Electronics Engineers (IEEE) Class 1E (as well as Seismic Category I). Some nonsafety-related electrical equipment and instrumentation are optionally designed to IEEE Class 1E requirements as noted in Table 3.2-1.
3. Some nonsafety-related structures, systems, and components are optionally designed to Quality Group B or C requirements, as designated in Table 3.2-1. Nonsafety-Related structures, systems, and components that are not assigned a quality group are designed to requirements of applicable industry codes and standards (see Subsection 3.2.3.4).
4. Safety-related (Safety Class 1, 2 and 3) structures, systems, and components meet the quality assurance requirements of 10 CFR 50, Appendix B, as described in Chapter 17. Nonsafety-Related (N) structures, systems and components meet quality assurance requirements as defined in the quality assurance program that are commensurate with the importance of the equipment's function. Structures, systems and components designated as Safety-Related Class S in Table 3.2-1 have special quality assurance requirements consistent with the portions of Note (5) that are referred to in the Notes column. See Subsection 17.1.22 for further details.
5. Nonsafety-related reactor internal structures subject to the requirements of ASME B&PV Code Section III, Division 1, Subsection NG, are assigned to Class IS.

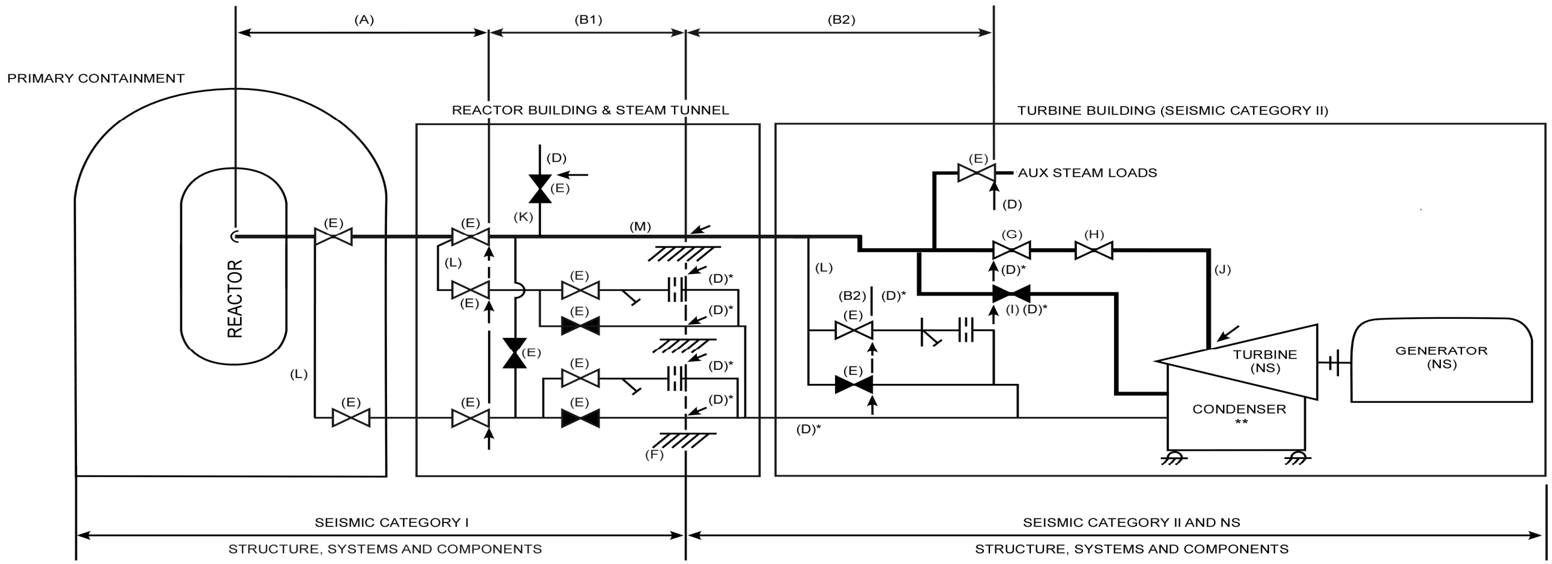
Table 3.2-3 Quality Group Designations – Codes and Industry Standards

Quality Group Classification	ASME BPVC Section III Code Classes	Pressure Vessels and Heat Exchangers[4]	Pipes, Valves, and Pumps	Storage Tanks (0-103 kPaG) 0-15 psig	Storage Tanks Atmospheric	ASME BPVC Section III Component Supports	Non-ASME BPVC Section III Component Supports	Core Support Structures and Reactor Internals	Containment Boundary
A	1	NCA and NB TEMA C	NCA and NB	—	—	NCA and NF	—	—	—
B	2	NCA and NC TEMA C	NCA and NC	NCA and NC	NCA and NC	NCA and NF	—	—	—
	CC[1] and MC	—	—	—	—	—	—	—	NCA, CC[1], and NE
	CS	—	—	—	—	—	—	NCA and NG	
C	3	NCA and ND TEMA C	NCA and ND	NCA and ND	NCA and ND	NCA and NF	—	—	—
D	—	ASME BPVC Sect. VIII Division 1 TEMA C	ASME B31.1 for piping and valves[2]	API-620 or equivalent[3]	API-650 AWWA-D100 ASME B96.1 or equivalent[3]	—	Manufacturer's Standards, e.g., ASME B31.1, AISC	—	—

Notes:

1. RCCV is designed to Subsection CC in ASME BPVC, Section III, Division 2.
2. For pumps classified in Quality Group D, the ASME B&PV Code, Section VIII, Division 1 is used as a guide in determining the wall thickness for pressure-retaining parts and in sizing the cover bolting.
3. Tanks are designed to meet the intent of American Petroleum Institute (API), American Water Works Association (AWWA), and/or ASME B96.1 standards, as applicable.
4. For heat exchangers, both the ASME Code and Tubular Exchanger Manufacturers' Association (TEMA) C must be taken into account.

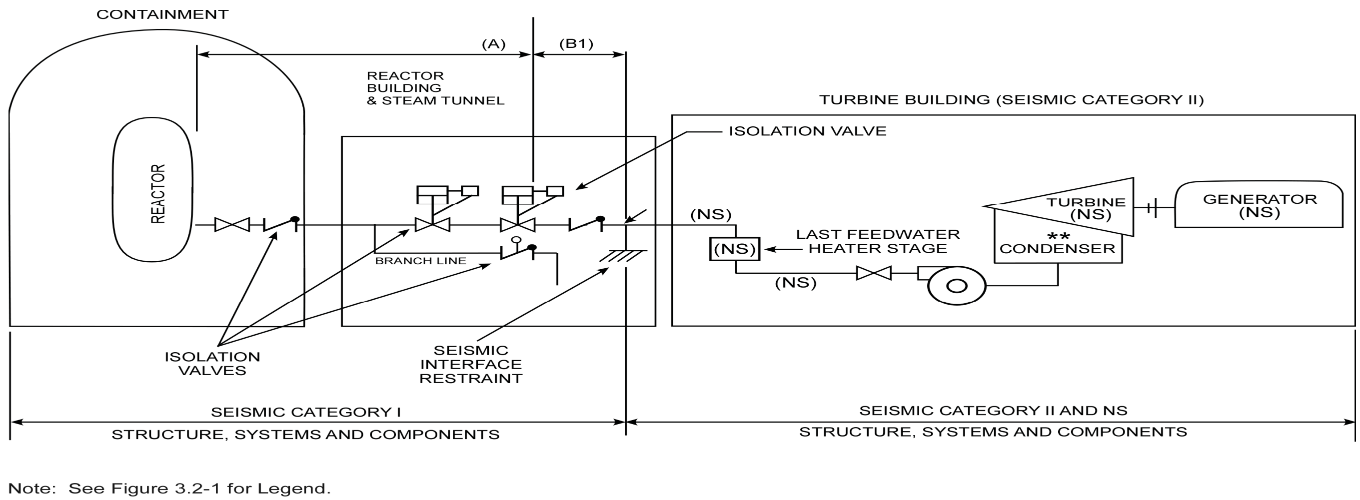
Figure 3.2-1 Quality Group and Seismic Category Classification Applicable to Power Conversion System



LEGEND:

A.	QUALITY GROUP A	I.	TURBINE BYPASS VALVE	* ANALYZED TO DEMONSTRATE STRUCTURAL INTEGRITY UNDER SSE LOADING CONDITIONS FOR MSIV LEAKAGE PATH
B 1.	QUALITY GROUP B	J.	MAIN STEAM LEAD	
B 2.	QUALITY GROUP B, SEISMIC CATEGORY II	K.	INSTRUMENT BRANCH LINE	** CONDENSER SUPPORTS AND ANCHORS ARE DESIGNED TO MAINTAIN CONDENSER INTEGRITY FOLLOWING SSE FOR MSIV LEAKAGE HOLDUP VOLUME.
D.	QUALITY GROUP D	L.	DRAIN LINE	
E.	ISOLATION VALVE	M.	STEAM LINE	
F.	SEISMIC INTERFACE RESTRAINT	NS	SEISMIC CATEGORY NS	
G.	TURBINE STOP VALVE	↙	CLASSIFICATION CHANGE	
H.	TURBINE CONTROL VALVE			

Figure 3.2-2 Quality Group and Seismic Category Classification Applicable to Feedwater System



Note: See Figure 3.2-1 for Legend.

3.3 Wind and Tornado Loadings

Seismic Category I structures are designed for tornado and extreme wind phenomena. Seismic Category II structures are designed for extreme and tornado wind (excluding tornado missiles).

3.3.1 **Wind Loadings**

As discussed in Standard Review Plan (SRP) 3.3.1, the design wind velocity and its recurrence interval, the velocity variation with height, and the applicable gust factors are used in defining the input parameters for the structural design criteria appropriate to account for wind loadings. The procedures that are utilized to transform the design wind velocity into an effective pressure applied to structures take into consideration the geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures.

The design of structures that must withstand the effects of the design wind load consider the relevant requirements of General Design Criterion 2 concerning natural phenomena. The wind used in the design includes the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data has been accumulated. Appropriate consideration has been given for the design wind velocity and its recurrence interval, the velocity variation with height, the applicable gust factors, and the bases for determining these site-related parameters. The procedures utilized to transform the wind velocity into an effective pressure applied to structures and parts and portions of structures, are as delineated in Reference 3.3-1.

3.3.1.1 **Design Wind Velocity and Recurrence Interval**

Seismic Category I and II structures are designed to withstand the design wind velocity listed in Table 2.0-1. The recurrence interval listed in Table 2.0-1 is equivalent to an importance factor of 1.15 based on Category IV building.

Seismic Category NS buildings that house RTNSS equipment are designed to withstand hurricane Category 5 wind velocity at 87.2 m/s (195 mph), 3-second gust, instead of wind speed listed in Table 2.0-1.

3.3.1.2 **Determination of Applied Forces**

The design wind velocity is converted to velocity pressure in accordance with Reference 3.3-1 with Exposure Category D.

The design wind velocity for use in the ESBWR is listed in Table 2.0-1. Reference 3.3-2 is used to obtain the effective wind pressures for geometric and physical cases that Reference 3.3-1 does not cover.

3.3.1.3 Effect of Failures of Structures or Components Not Designed for Wind Loads

Safety-related systems and components are protected within wind-resistant structures. The remainder of plant structures and components not designed for extreme wind loads are arranged or designed such that their failures do not adversely affect the ability of any Seismic Category I structures, systems, and components to perform their safety-related function(s).

3.3.2 Tornado Loadings

As discussed in SRP 3.3.2, the design of structures that have to withstand the effects of the design basis tornado are in conformance with the requirements of General Design Criterion 2.

3.3.2.1 Applicable Design Parameters

The design basis tornado and applicable missiles are described in Table 2.0-1.

3.3.2.2 Determination of Forces on Structures

The procedures of transforming the tornado loading into effective loads and the distribution across the structures are in accordance with Reference 3.3-3. The velocity pressure used meets the SRP 3.3.2 discussion. The procedure for transforming the tornado-generated missile impact into an effective or equivalent static load on structures is given in Subsection 3.5.3. The loading combinations of the individual tornado loading components and the load factors are in accordance with SRP 3.3.2.

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 = tornado wind load

W_p = tornado differential pressure load

W_m = tornado missile load

The Reactor Building, Fuel Building, and Control Building are not vented (enclosed) structures. The exposed exterior roofs and walls of these structures are designed for the full pressure drop. Tornado dampers are provided on Control Building EFU air intake openings. These dampers are designed to withstand the full negative pressure drop.

All Control Room Habitability Area ventilation penetrations for outside air intake and exhaust openings are provided with tornado protection. In addition, the Control Building Heating, Ventilation and Air Conditioning System outside air intake and return/exhaust openings are provided with tornado protection.

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

Safety-related systems and components are protected within tornado-resistant structures. The remainder of plant structures and components not designed for tornado loads are arranged or designed such that their failures do not adversely affect the ability of any Seismic Category I structures, systems and components to perform their safety-related function(s). Any nonsafety-related, non-seismic structure postulated to fail under tornado loading is located at least a distance of its height above grade from Seismic Category I structures. The Radwaste Building (RW) is designed for tornado wind loads. Refer to Table 2.0-1 for tornado wind speed, radius, pressure drop and rate of pressure drop for the RW. Refer to RG 1.143 for tornado missile spectrum for the RW.

3.3.3 References

3.3-1 American Society of Civil Engineers, “Minimum Design Loads for Buildings and Other Structures,” ASCE Standard 7-2002, Committee A. 58.1, American National Standards Institute.

3.3-2 American Society of Civil Engineers, “Wind Forces on Structures,” ASCE Paper No. 3269, Transactions of the American Society of Civil Engineers,” Vol. 126, Part II, 1961.

3.3-3 Bechtel Topical Report BC-TOP-3-A, Revision 3, “Tornado and Extreme Wind Design Criteria for Nuclear Power Plants,” August 1974.

3.4 Water Level (Flood) Design

Design of the plant flood protection includes all structures, systems and components (SSCs) whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity to assure conformance with the requirements of General Design Criterion 2.

3.4.1 Flood Protection

This section describes the plant flood protection for all SSCs whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity to assure conformance with the requirements of General Design Criterion 2. The analysis identifies the safety-related SSCs that must be protected against flooding from both external and internal causes to:

- Demonstrate the capabilities of structures housing safety-related systems or equipment to withstand flood considerations; that is, the relationship between structure elevation and flood elevation including waves and wind effects as described in Table 2.0-1.
- Assess the adequacy of the isolation of redundant safety-related systems or equipment subject to flooding, including possible inleakage sources, such as: (i) cracks in structures not designed to withstand seismic events and (ii) exterior or access openings or penetrations in structures located at a lower elevation than the flood level and associated wave activity.

The analysis also includes consideration of flooding from internal sources of safety-related SSCs from failure of tanks, vessels, and piping. The flooding analysis also considers the water-related effects of piping failures, while dynamic effects are addressed in Section 3.6.

The flood protection measures meet specific GDC and regulatory guides. The plant design for protection of SSCs from the effects of flooding considers the relevant requirements of General Design Criterion 2, “Design Bases for Protection Against Natural Phenomena,” and 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” Section IV.(c) as related to protecting safety-related SSCs from the effects of floods, water waves and other design conditions. The design meets the guidelines of Regulatory Guide (RG) 1.59 with regard to the methods utilized for establishing the probable maximum flood (PMF), probable maximum precipitation (PMP), seiche and other pertinent hydrologic considerations; and the guidelines of RG 1.102 regarding the means utilized for protection of safety-related SSCs from the effects of the PMF and PMP. If safety-related structures need to be protected from below-grade groundwater seepage by means of a permanent dewatering system, then the system would be designed as a safety-related system and would meet the single failure criterion requirements. However, the ESBWR does not require a safety-related permanent dewatering system. The design criteria for protection against the effects of compartment flooding meet ANSI/ANS 56.11 (Reference 3.4-1), “Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants”. This subsection discusses the flood protection design and operational

measures that are applicable to the plant Seismic Category I SSCs and addresses both external flooding and postulated internal flooding from plant piping failures, fire fighting, and other sources.

3.4.1.1 Flood Protection Summary

The safety-related systems and components of the ESBWR standard plant are located in the Seismic Category I structures that provide protection against external flood and groundwater damage. External flood design considerations for safety-related systems and components are provided for the postulated flood and groundwater levels and conditions described in Tables 2.0-1 and 3.4-1.

The Seismic Category I structures that house safety-related systems and equipment and that offer flood protection are described in Section 3.8. All exterior access openings are above flood level and exterior penetrations below design flood and groundwater levels are appropriately sealed.

The internal flood analysis evaluates whether a single pipe failure, a fire fighting event or other flooding source, as described in Subsection 3.4.1.4, could prevent safe reactor shutdown. In all cases, system components are located above the flood level or are capable of operating in a flooded environment. Appropriate means are provided to prevent flooding compartments that house redundant system trains or divisions. Some of the mechanisms used to minimize flooding are structural barriers or compartments; curbs and elevated thresholds, at least 200 mm (8 in) high; and a leak detection system. See Subsection 3.4.1.3 for further discussion.

3.4.1.2 Flood Protection From External Sources

Safety-related systems and components are protected from exterior sources (e.g., floods, groundwater) because they are located above design flood level or because they are enclosed in groundwater protected concrete structures.

The Seismic Category I structures that may be subjected to the design basis flood are designed to withstand the flood level and groundwater level stated in Table 2.0-1. This is done by locating the design plant grade elevation at least 300 mm (1 ft.) above the design flood level and by incorporating structural provisions into the plant design to protect the SSCs from the postulated flood and groundwater conditions.

These provisions include:

- Walls below flood level designed to withstand hydrostatic loads.
- Water stops provided in all expansion and construction joints below design basis maximum flood and groundwater levels.
- Waterproofing of external surfaces below design basis maximum flood and groundwater levels.
- Water seals at pipe penetrations below design basis maximum flood and groundwater levels.
- Roofs designed to prevent pooling of large amounts of water in accordance with RG 1.102.

- No exterior access openings below grade.

The flood protection measures that are described above are not only for external natural floods but also guard against flooding from onsite storage tank rupture. Such tanks are designed and constructed to minimize the risk of catastrophic failure and are located to allow drainage without damage to site facilities.

Because plant grade is above design flood level, the Seismic Category I structures remain accessible during postulated flood events (See Table 3.4-1). Thus, no emergency actions are required due to flooding to ensure the safe operation of the ESBWR plant.

3.4.1.3 **Internal Flooding Evaluation Criteria**

All safety-related components that affect the safe shutdown of the plant are located in the Reactor Building (RB) and Control Building (CB). Redundant systems and components are physically separated from each other and from nonsafety-related systems. If the failure of a system results in one division being inoperable, a redundant division is available to perform the safe shutdown of the plant. Protective features used to mitigate or eliminate the consequences of internal flooding are:

- Structural enclosures or barriers
- Curbs and sills
- Leakage detection components
- Drainage systems

The internal flooding analysis, besides identifying flooding sources, equipment in each area, and effect on safety-related equipment and maximum flood levels, also considers the following criteria:

- A flooding alarm in the main control room is followed by operator action within 30 minutes to identify the flooding source.
- Fire fighting events are considered assuming that fuel inventory for the fire is limited to a 1-hour event, during which two 7.9 l/s (125 gpm) fire hoses are in service.
- A single active failure of flood mitigating systems is assumed, following the initiating events, as required in ANSI/ANS 56.11 (Reference 3.4-1).
- No credit is taken for the drainage system or operation of the drain sump pumps for flooding mitigation, although they are expected to operate during some of the postulated flooding events.
- The free surface considered in each flooding zone is reduced by at least 10% due to space utilization by components located in that zone.

As established in Section 3.6, the moderate energy piping leakage failure is assumed to be a circular opening with a flow area equal to one-half of the outside pipe diameter multiplied by one-half of the pipe nominal wall thickness. Resulting leakage flow rates are calculated using normal operating pressure in the pipe.

The Fire Protection System (FPS) headers from the FPS pumps are routed outside Seismic Category I buildings. Floors are assumed to prevent water seepage to lower levels.

Spray damage is avoided by appropriate location of equipment or pipe or by providing protection from water spray. Doors and penetrations rated as 3 hour barriers are assumed to prevent water spray from crossing divisional boundaries.

All safety-related equipment within the Containment that must operate during or after a design basis accident is qualified for LOCA environmental conditions. Flooding associated with the postulated failure of any moderate energy pipe is within the bounds of the LOCA qualification. Consequently, no detailed evaluation of this less severe event is required to verify the effect on safety-related equipment or safe plant shutdown capability as a result of moderate energy piping failures in the Containment.

3.4.1.4 **Evaluation of Internal Flooding**

Leakage from pipe breaks and cracks, fire hose discharges and other flooding sources are collected by the floor drainage system, stair towers and elevator shafts and discharged to appropriate sumps. The flood level is evaluated taking into consideration the flow paths described above.

The RB and CB drain collection system and sumps are designed and separated so that drainage from a flooded compartment containing equipment for a train or division does not flow to compartments containing equipment for another system train or division. Zones that are isolated by watertight doors provide physical separation. Watertight doors between flood divisions have open/close sensors with status indication and alarms in the main control room. The location of the zones prevents two redundant trains from being affected by the flooding at the same time.

The following flooding sources are considered in the analysis:

- High energy piping breaks and cracks
- Moderate energy piping through-wall cracks
- Pump mechanical seal failures
- Storage tank ruptures
- Actuation of the FPS
- Flow from upper elevations and nearby areas

Through-wall cracks are considered in seismically supported, moderate energy piping as well as breaks and through-wall cracks in non-seismically supported moderate energy piping in the flooding analysis.

The analysis is performed based on the criteria and assumptions provided in Section 3.6 and ANS-56.11 (Reference 3.4-1). 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.

No breaks are assumed for piping with nominal diameters of 25 mm (1 in.) or less. For flooding analysis, in case of storage tank rupture, it is assumed that the entire tank inventory is drained.

Safety-related equipment and equipment necessary for safe shutdown is located above the maximum flood level or is qualified for flood conditions. Accordingly, flooding due to moderate energy pipe failure or fire fighting or other flooding sources does not affect any safety-related equipment and the ability to safely shut down the plant.

3.4.1.4.1 **Control Building**

There are no tanks or high-energy piping in the CB and the more relevant moderate-energy fluid system piping, i.e. FPS and Chilled Water System (CWS), is seismically qualified. The main source of floodwater is from the fire protection standpipe hose stations. A nominal volume of 57 m^3 (15,000 gal) is provided for the FPS considering two 7.9 l/s (125 gpm) fire hoses are in service for one (1) hour. This results in a flooding level in the lowest floor of the CB of 40 cm (16 in) in the corridors, stair towers and elevator rooms, assuming that the water propagates into these rooms by flowing through embedded drains and under the doors. This maximum water depth is below the Distributed Control and Information System (DCIS) room floor elevation; see Figure 1.2-2 (Rooms 3110, 3120, 3130 and 3140).

To prevent flooding in the Control Room Habitability Area (CRHA) emergency ventilation equipment from failures of liquid carrying systems in the Heating, Ventilation and Air Conditioning rooms, the water is routed by the installation of 300 mm (12 in) high curbs in the access doors, chases and other floor openings, as well as by normally closed isolation valves in the drain lines and elevated thresholds in the access doors to CRHA emergency ventilation equipment, to discharge the potential flooding water to the building stairwells.

In addition, for further protection, the DCIS room access doors are watertight. Normally closed valves are installed in the drain pipes of the DCIS rooms. Moreover, the access doors from the access tunnel to the CB at Elevation -2000 are watertight.

Therefore, the separation of electrical trains in independent zones, along with measures to direct the water to drains, maintains the safety function of the systems housed in the CB.

There is no flooding hazard in the main control room because the potential flood water from chilled water portion inside CRHA envelope is detected and isolated.

3.4.1.4.2 **Reactor Building**

The potential sources of water in the RB include the Reactor Component Cooling Water System (RCCWS); CWS; Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system; Control Rod Drive (CRD) system, including the CRD pump suction from the Condensate Storage and Transfer

System (CS&TS) and Condensate and Feedwater System (C&FS); FPS; Fuel and Auxiliary Pools Cooling System (FAPCS); Makeup Water System (MWS); and Standby Liquid Control (SLC) system.

The large number of pools in the ESBWR is contained within thick concrete walls designed for maximum hydrostatic loads combined with seismically induced hydrodynamic loads. GDCS pools inside containment are similarly contained within robust structural members designed for hydrostatic loads combined with seismically induced hydrodynamic loads. These pools are not considered as potential sources of flood.

The piping of the RCCWS, CWS, CRD pump suction (from CS&TS/C&FS), MWS, and FPS are seismically analyzed. These are moderate energy fluid systems and therefore only through-wall pipe cracks are considered.

The maximum flooding volume expected is from a through-wall pipe crack in the FPS or in the FAPCS suction lines from the suppression pool. The flooding volume from either of these sources is greater than flooding due to any failure in high and moderate energy piping or tanks.

The maximum volume of the suppression pool for flooding is limited to the difference between the maximum level and the anti-siphoning provision in the suction line elevation.

This results in a flood level of 20 cm (8 in) in the RB lower elevation. This maximum flood level is lower than the CRD Hydraulic Control Unit (HCU) room elevation, see Figure 1.2-1 (rooms 1110, 1120, 1130 and 1140). Other safety-related components in the lower elevation are located above the maximum flood level. Therefore, no flood in this RB elevation could affect the safety-related equipment or plant's safe shutdown capability.

For further protection, the HCU room access doors and the access doors to the RB at El. -1000 are watertight.

The SLC system accumulators for Division 1 and 2 are located in fully independent rooms in El. 17500 of the RB. Therefore, SLC system high energy pipe break or tank failure flooding of one division cannot affect the other.

Flooding in the electrical rooms is limited to the actuation of the FPS. The separation of the electrical trains in independent zones, along with measures to direct the water to drains, maintains the safety function of the systems housed in the RB.

The main steam tunnel contains the main steam and main feedwater piping and their isolation valves. In the event of a feedwater pipe break or leak in the main steam tunnel, water is drained to the Turbine Building (TB). The safety-related components in the main steam tunnel are located above the maximum flood level or are designed to function when flooded.

3.4.1.4.3 **Adjacent Flooding Events**

- **Turbine Building**. – There are no components in the TB that could affect the safe shutdown of the reactor.

  The TB is subject to flooding from a variety of potential sources including the Circulating Water System (CIRC), C&FS, PSWS, RCCWS, TCCWS, CWS and FPS.

  The bounding flooding source for the TB is a CIRC pipe or expansion joint failure. Level switches are located in the TB to limit flooding in the TB in the event of a failure in the CIRC (see Subsection 10.4.5.6). In any case, flooding in the TB could not affect the RB or CB because a 1.5 m (4.9 ft.) high flooding barrier is provided in the access tunnel to the RB and CB (see Figure 1.2-13). A hypothetical massive flooding in the TB would run out of the building to the yard through relief panels.

- **Fuel Building (FB) –** There are no safety-related components in the FB that could be affected by flooding in the FB. The FPS, CWS, RCCWS, FAPCS, MWS and CS&TS (including Condensate Storage Tank) are the primary sources of flooding in the FB. In any case, flooding in the FB could not affect the RB because the connection points in the lower elevation are watertight.

- **Radwaste Building –** The Radwaste Building (RW) does not contain safety-related equipment. The radwaste tunnel and other connections with the CB and RB are designed to prevent flooding from spreading in the RW to CB or RB. The primary sources of flooding in the RW are the LWMS, the building drain systems, RWCU/SDC, FAPCS, CPS, CS&TS, CWS and FPS. In case of flooding, the building substructure serves as a large sump that can collect and hold any leakage within the building.

- **Electrical Building (EB) –** There are no safety-related components in the EB. The flooding water in a nonsafety-related diesel generator room is discharged outside via the equipment access door.

  The primary sources of flooding in the EB are the FPS, CWS and RCCWS (nonsafety-related diesel generator rooms). The main source of floodwater is due to an FPS piping failure. A flooding barrier is provided at the Nuclear Island access tunnel EB access door. In addition, for further protection the access doors to the RB and CB are watertight.

3.4.2 **Analysis Procedures**

The following paragraphs describe the design of Seismic Category I structures to withstand the effects of the maximum external flood and highest groundwater levels specified for the plant. The maximum flood and highest groundwater levels are considered in defining the input design parameters for the structural design to account for flood and groundwater loadings. Because the ESBWR standard plant is located at a site where the flood level is below the finished ground level around Seismic Category I structures, the dynamic phenomena associated with a flooding event,

such as currents, wind waves, and their hydrodynamic effects, are not considered. The values for the maximum flood and maximum groundwater level parameters are provided in Table 2.0-1. The procedures utilized to transform the static and dynamic effects of the maximum flood and highest groundwater levels, that is, the design flood level and design ground water level, into effective loads applied to Seismic Category I structures are discussed in this subsection.

The design of ESBWR structures complies with the relevant requirements of GDC 2 concerning natural phenomena. The values for site parameters used in the design of Seismic Category I structures were selected to envelope the actual characteristics at most sites. Table 2.0-1 ensures that the site meets the following:

1. The highest flood and highest groundwater levels, including any dynamic effects, are the most severe 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. The highest flood or highest groundwater level for the plant is below the finished ground level as shown in Table 3.4-1.
3. Because the highest flood level of the plant is below the finished ground level, only hydrostatic effects need to be considered. The hydrostatic pressure associated with the design flood level or with the design groundwater level is considered as a structural load on the basemat and basement walls. Uplift or floating of the structure is considered and the total buoyancy force is based on the hydrostatic pressure due to the design flood level, excluding wave action, or the design groundwater level. The lateral, overturning and upward hydrostatic pressures acting on the side walls and on the foundation slab, respectively, are also considered in the structural design of these elements.

Because the design flood elevation is below the finished ground level (Table 3.4-1), there are no dynamic forces due to flood. The lateral hydrostatic pressures on the structures due to the design flood level, as well as ground water and soil pressure, are factored into the structural design in accordance with SRP 3.4.2. See Appendix 3G, Design Details and Evaluation Results of Seismic Category I Structures.

3.4.3 COL Information

None.

3.4.4 References

3.4-1 ANSI/ANS 56.11-1988, “Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants”.

Table 3.4-1 Structures, Penetrations and Access Openings Designed for Flood Protection

	Reactor & Fuel Buildings mm (ft.)	Control Building mm (ft.)
Building Elevation	El. 52700 (172.9)	El. 13500 (44.3)
Design Plant Grade	El. 4650 (15.3)	El. 4650 (15.3)
Finished Ground Level Grade	El. 4500 (14.8)	El. 4500 (14.8)
Design Flood Level	El. 4350 (14.3)	El. 4350 (14.3)
Design Groundwater Level	El. 4040 (13.3)	El. 4040 (13.3)
Top of Basemat	El. –11500 (-37.7)	El. -7400 (-24.3)
Penetrations Below Design Flood Level	Sealed	Sealed
Access Openings Below Design Flood Level	None (except at RB access to CB at tunnel)	None (except at CB access to RB at tunnel)

3.5 Missile Protection

The missile protection design basis for Seismic Category I structures, systems and components (SSCs) is described in this section. A tabulation of SSCs (both inside and outside containment), their location, seismic category, and quality group classification is given in Table 3.2-1. General arrangement drawings showing locations of the SSCs are presented in Section 1.2.

Missiles considered are those that could result from a plant-related failure or incident including failures within and outside of containment, environmental-generated missiles and site-proximity missiles. The structures, shields, and barriers that are designed to withstand missile effects, the possible missile loadings, and the procedures to which each barrier is designed to resist missile impact are described in detail.

3.5.1 **Missile Selection and Description**

Components and equipment are designed to have a low potential for generation of missiles as a basic safety precaution. In general, the design that results in reduction of missile-generation potential promotes the long life and usability of a component and is well within permissible limits of accepted codes and standards.

Seismic Category I structures are analyzed and designed to be protected against a wide spectrum of missiles. For example, failure of certain rotating or pressurized components of equipment is considered to be of sufficiently high probability and to presumably lead to generation of missiles. However, the generation of missiles from other equipment is considered to be of low enough probability and is dismissed from further consideration. Tornado-generated missiles and missiles resulting from activities particular to the site are also discussed in this section. The missile protection criteria to which the plant has been designed consider Criterion 4 of 10 CFR 50 Appendix A, General Design Criteria for Nuclear Power Plants.

Potential missiles that have been identified are listed and discussed in later subsections.

After a potential missile has been identified, its statistical significance is determined. A statistically significant missile is defined as a missile that could cause unacceptable plant consequences or violation of the limits of 10 CFR 52.47(a)(2)(iv).

The examination of potential missiles and their consequences is done in the following manner to determine statistically significant missiles:

- If the probability of occurrence of the missile, P_1, is determined to be less than 10^{-7} per year, the missile is dismissed from further consideration because at that likelihood it is considered not to be a statistically significant risk.
- If P_1 is found to be greater than 10^{-7} per year, it is examined for its probability of impacting a design target P_2.

- If the product of P_1 and P_2 is less than 10^{-7} per year, the missile is dismissed from further consideration.
- If the product of P_1 and P_2 is greater than 10^{-7} per year, the missile is examined for its damage probability P_3. If the combined probability (i.e., $P_1 \times P_2 \times P_3 = P_4$) is less than 10^{-7} per year, the missile is dismissed.
- Finally, measures are taken to design acceptable protection against missiles with P_4 greater than 10^{-7} per year to reduce P_1, P_2, and/or P_3, so that P_4 is less than 10^{-7} per year.

Many practices used in the fabrication, construction and inspection of equipment as well as conservative design criteria result in very robust components that are inherently missile resistant. These practices are used in making the design missile-proof.

Protection of SSCs is afforded by one or more of the following practices:

- Location of the system or component in an individual missile-proof structure.
- Physical separation of redundant systems or components of the system from the missile trajectory path or calculated range.
- Provision of localized protection shields or barriers for systems or components.
- Design of the particular structure or component to withstand the impact of the most damaging missile.
- Provision of design features on the potential missile source to prevent missile generation.
- Orientation of the potential missile source to prevent unacceptable consequences caused by missile generation.

The following criteria are adopted to provide an acceptable design basis for the plant's capability to withstand the statistically significant missiles postulated inside the Reactor Building:

- No loss of containment function as a result of missiles generated internal to containment.
- Reasonable assurance that a safe plant shutdown condition can be achieved and maintained.
- Offsite exposure within the 10 CFR 52.47(a)(2)(iv) limits for those potential missile damage events resulting in radiation activity release.
- The failure of nonsafety-related equipment, components, or structures whose failure could result in a missile, do not cause failure of more than one division of safety-related equipment.
- No high energy lines are located near Off-Gas Charcoal Bed Adsorbers (located in the Turbine Building).

The systems requiring protection are as follows:

1. Reactor coolant pressure boundary
2. Automatic Depressurization System relief valves

3. Passive Containment Cooling System
4. Isolation Condenser
5. Gravity Driven Cooling System
6. Control Rod Drive scram system (hydraulic and electrical)
7. Reactor Protection System
8. All containment isolation valves
9. Electrical and control systems and wiring required for operation of items (1) through (8)
10. Remote shutdown panel

The following general criteria are used in the design, manufacture, and inspection of equipment:

- All pressurized equipment and sections of piping that may periodically become isolated under pressure are provided with pressure-relief valves acceptable under ASME Code Section III. The valves ensure that no pressure buildup in equipment or piping sections exceeds the design limits of the materials involved.
- Components and equipment of the various systems are designed and built to the standards established by the ASME Code or other equivalent industrial standards. A stringent quality control program is also enforced during manufacture, testing, and installation.
- Volumetric and ultrasonic testing, where required by code, coupled with periodic in-service inspections of materials used in components and equipment, add further assurance that any material flaws that could permit the generation of missiles are detected.

3.5.1.1 **Internally Generated Missiles (Outside Containment)**

This subsection addresses SSCs provided to support the reactor facility, and that require protection from internally generated missiles (outside containment) to ensure conformance with the requirements of General Design Criterion 4. The design addresses concerns for missiles that could result from in-plant component overspeed failures and high-pressure system ruptures as discussed in SRP 3.5.1.1, when applicable.

3.5.1.1.1 **Rotating Equipment**

3.5.1.1.1.1 **Missile Characterization**

Equipment within the general categories of pumps, fans, blowers, diesel generators, compressors, turbines, and, in particular, components in systems normally functioning during power reactor operation, are examined for any possible source of credible and significant missiles.

3.5.1.1.1.2 **Main Steam Turbine**

The main turbine has a favorable turbine generator placement and orientation relative to placement of the containment. The arrangement adheres to the guidelines presented in Regulatory Guide

(RG) 1.115 and meets position C1 therein. The ESBWR turbine generator placement and orientation are shown in Figure 3.5-2. See Subsection 10.2.4 for additional evaluation.

Regulatory Treatment of Non-Safety Systems (RTNSS), Category B functions, as listed in DCD Tables 19A-2 and 19A-3, and structures, systems and components listed in Regulatory Guide 1.117 Appendix including the Independent Fuel Storage Installation are not within the low-trajectory turbine missile strike zone as defined in Regulatory Guide 1.115 and shown in Figure 3.5-2. Therefore barriers to protect this equipment, or the safety-related equipment, listed in Section 10.2, from low-trajectory turbine missile strikes are not required.

Favorable turbine generator placement and orientation, combined with quality assurance in design and fabrication, maintenance and inspection programs as provided in Section 10.2, and overspeed protection systems, provide an acceptably small risk from turbine missiles. The probability of turbine missile generation, P_1, is less than 1 x 10^{-5} per year. This is less than the required value provided in Table 3.5-1. See Section 10.2 for a discussion of turbine missile analysis, turbine inspection, test and maintenance program, and turbine missile probability calculations.

3.5.1.1.1.3 Other Missile Analysis

No remaining credible missiles meet the significance criteria of having a probability, P_4, greater than 10^{-7} per year for rotating or pressurized equipment, because either:

- The equipment design and manufacturing criteria mentioned previously result in P_1 being less than 10^{-7} per year.
- Sufficient physical separation (barriers and/or distance) of safety-related and redundant equipment exists so that the combined probability, P_1 x P_2, is less than 10^{-7} per year.

The configuration of components is robust as required by ASME Code.

These conclusions are arrived at by noting that pumps, fans, and the like are AC powered. Their speed is governed by the frequency of the AC power supply. Because the AC power supply frequency variation is limited to a narrow range, it is not likely that these components could attain an overspeed condition. At rated speed, if a component's piece such as a fan blade breaks off, it would not penetrate the casing. As an example, a typical containment high purge exhaust fan used in previous applications has been analyzed for a thrown blade at rated speed conditions using an analytical expression from Reference 3.5-2. It is determined, based on the maximum thickness this blade could penetrate, that the blade would not escape the fan casing and consequently P_1 is less than 10^{-7} per year.

3.5.1.1.2 Pressurized Components

3.5.1.1.2.1 Missile Characterization

Potential missiles that could result from the failure of pressurized components are addressed in this subsection. These potential missiles are categorized as contained fluid energy missiles or

stored-energy (elastic) missiles. These potential missiles are conservatively evaluated against the design criteria in the following subsections.

Examples of potential contained fluid-energy missiles are valve bonnets, valve stems, and retaining bolts. Valve bonnets are considered jet-propelled missiles and have been analyzed as such. Valve stems are analyzed as piston-type missiles, while retaining bolts are examples of stored strain-energy missiles.

3.5.1.1.2.2 **Missiles Analyses**

Pressurized components outside the containment capable of producing missiles have been reviewed. Although piping failures could result in dynamic effects if permitted to whip, they do not form missiles as such because the whipping section remains attached to the remainder of the pipe. Section 3.6 addresses the dynamic effects associated with pipe breaks, so pipes are not included here as potential internal missiles.

All pressurized equipment and sections of piping that may periodically become isolated under pressure are provided with pressure-relief valves acceptable under the ASME Code, Section III.

The only remaining pressurized components considered to be potentially capable of producing missiles are as follows:

- Valve bonnets (large and small)
- Valve stems
- Pressure vessels
- Thermowells
- Retaining bolts
- Blowout panels

3.5.1.1.2.2.1 **Valve Bonnets**

Valves of ANSI 900 Pressure Class and above are constructed in accordance with the ASME Code, Section III and are pressure-seal bonnet-type valves. Valve bonnets are prevented from becoming missiles by limiting stresses in the bolting to those defined by the ASME Code and by designing flanges in accordance with applicable code requirements. Safety factors involved against failure of these type bonnets are sufficiently high that these pressure seal-type valves are not considered a potential missile source (Reference 3.5-3).

Most valves of ANSI 600 Pressure Class rating and below are valves with bolted bonnets. These type valves are analyzed for the safety factors against failure, and, coupled with the low historical incidents of complete severance failure, are determined to not be a potential missile source (Reference 3.5-3).

3.5.1.1.2.2.2 Valve Stems

Isolation valves installed in the reactor coolant systems have stems with back seats, which eliminates the possibility of ejecting valve stems even if the stem threads fail. Because a double failure of highly reliable components would be required to produce a valve stem missile, the overall probability of occurrence is less than 10^{-7} per year. Hence, valve stems are dismissed as a source of missiles.

3.5.1.1.2.2.3 Pressure Vessels

Moderate energy vessels less than 1.9 MPaG (275 psig) are not credible missile sources. The pneumatic system air bottles and components are designed for 17.2 MPaG (2500 psig) and the standby liquid control accumulator tank is designed for 17.2 MPaA (2500 psia) to the ASME Code, Section III requirements. These bottles are not considered a credible source of missiles for the following qualitative analysis:

- The bottles are fabricated from heavy-wall rolled steel.
- The operating orientation is vertical with the ends facing concrete slabs. The bottles are topped with steel covers thick enough to preclude penetration by a missile.
- The fill connection and critical parts are protected by a permanent steel collar.
- The bottles are strapped in a rack to prevent them from toppling over. The rack is seismically designed to the ASME Code, Section III, Subsection NF requirements.

3.5.1.1.2.2.4 Thermowells

Thermowells are welded to sockolet connections, which in turn are welded to the wall of the pipe. An analysis of a postulated failure of this weld is performed as follows. The following expression relates the missile displacement and velocity following the postulated failure:

$$\frac{y}{W/A} = \upsilon_\infty\left[\ln\left(\left(\frac{1}{1 - V/u_\infty}\right) - \frac{V}{u_\infty}\right)\right]$$, where: (Reference 3.5-1)

y	=	distance traveled by the missile from the break
W	=	missile weight
A	=	frontal area of missile
u_∞	=	asymptotic velocity of jet
υ_∞	=	asymptotic specific volume of jet
V	=	velocity of missile

Inherently, the water and steam velocities are equal (i.e., a unity velocity ratio) in a saturated water blowdown. The jet asymptotic velocity (u_∞) and the jet asymptotic specific volume are determined by the methods described by Reference 3.5-4. The corresponding velocity-displacement

relationships for missiles resulting from saturated water and saturated steam blowdowns are presented in Figure 3.5-1 The ordinate is the missile velocity, V, and the abscissa is the displacement parameter, Y*, given by:

$$Y^* = \frac{y}{(W/A)}$$

Included in Figure 3.5-1 is the influence of different values on the friction parameter, f*, defined by:

$$f^* = \left(\frac{f1}{D}\right)_P \left(\frac{A_E}{A_P}\right)^2$$

where:

$\left(\frac{f1}{D}\right)_p$	=	equivalent loss coefficient between the broken pressurized component and fluid reservoir, dimension-less
A_E	=	area of break,
A_p	=	area of pressurized component between break and fluid reservoir (assumes $A_p > A_E$)

As illustrated in Figure 3.5-1, the effect of friction on the velocity-displacement relationship is reasonably small. It can be conservatively assumed that the most extreme friction condition persists with f* = 100 for the case of saturated water blowdown and f* = 0 for the case of saturated steam blowdown.

A typical thermowell weighs about 0.91 kg (2 lbs). Based on ejection by steam at 7.2 MPa (1044 psia), the ejection velocity could reach 61 m/s (200 ft/s), which is not sufficient to inflict significant damage to critical systems. P_4 is therefore less than 10^{-7} per year.

3.5.1.1.2.2.5 Retaining Bolts

Nuts, bolts, nut and bolt combinations, and nut and stud combinations have a small amount of stored energy and are of no concern as potential missiles.

3.5.1.1.2.2.6 Blowout Panels

Blowout panels are hinged to prevent them from becoming missiles. Guard rails for personnel protection are provided where required by the swing pattern. Thus by design, P_2 is less than 10^{-7} per year.

3.5.1.1.3 Missile Barriers and Loadings

Credit is taken in some cases of rotating and pressurized components generating missiles for missile-consequence mitigation by structural walls and slabs. These walls and slabs are designed to withstand internal missile effects; the applicable seismic category and quality group classification

are listed in Section 3.2. Penetration of structural walls by internally generated missiles is not considered credible.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Internal missiles are those resulting from plant equipment failures within the containment. Potential missile sources from both rotating equipment and pressurized components are considered, when applicable.

3.5.1.2.1 Rotating Equipment

By an analysis similar to that in Subsection 3.5.1.1.1, it is concluded that no items of rotating equipment inside the containment have the capability of generating potential missiles.

3.5.1.2.2 Pressurized Components

Identification of potential missiles and their consequences outside containment are specified in Subsection 3.5.1.1.2. The same conclusions are drawn for pressurized components inside containment. For example, the ADS accumulators are moderate energy vessels and are therefore not considered a credible missile source. Another group of items is Fine Motion Control Rod Drives (FMCRDs) under the reactor vessel. The FMCRD mechanisms are not credible missiles. The FMCRD housings are designed (Section 4.6) to prevent any significant nuclear transient in the event of a drive housing break.

3.5.1.2.3 Evaluation of Potential Gravitational Missiles Inside Containment

Gravitational missiles inside the containment are considered as follows:

Seismic Category I systems, components, and structures are not potential gravitational missile sources.

Non-Seismic items and systems inside containment are considered as follows:

- Cable Trays - All cable trays for both safety-related and nonsafety-related circuits are seismically supported whether or not a hazard potential is evident.
- Conduit and Nonsafety-Related Pipe - Nonsafety-related conduit is seismically supported if it is identified as a potential hazard to safety-related equipment. All nonsafety-related piping that is identified as a potential hazard is seismically analyzed per Subsection 3.7.3.8.
- Equipment for Maintenance - All other equipment, such as a hoist, that is required during maintenance is either removed during operation, moved to a location where it is not a potential hazard to safety-related equipment, or seismically restrained to prevent it from becoming a missile. This is ensured by plant procedures as described in Section 13.5.

3.5.1.3 Turbine Missiles

See Subsection 3.5.1.1.1.2.

3.5.1.4 **Missiles Generated by Natural Phenomena**

This subsection considers possible hazards due to missiles generated by the design basis tornado, flood, and any other natural phenomena identified in Section 3.5.

Tornado generated missiles are determined to be the limiting natural phenomena hazard in the design of all structures required for safe shutdown of the nuclear power plant. Because tornado missiles are used in the design basis, they envelop missiles generated by less intense phenomena such as extreme winds. See Reference 3.5-8.

The design basis tornado and missile spectrum as defined in Table 2.0-1 is included in the design of Seismic Category I buildings, and is in compliance with positions C1 and C2 of RG 1.76, "Design Basis Tornado for Nuclear Power Plants," positions C1, C2, and C3 of RG 1.117, "Tornado Design Classification," Position C2 of RG 1.13, "Spent Fuel Storage Facility Design Basis," and positions C2 and C3 of RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants."

Since Seismic Category I buildings are designed to resist tornado missiles for their full height (See Table 2.0-1), their resistance to missiles is independent of site topography.

Non-tornado resistant building superstructures are constructed from materials such as reinforced concrete block, and/or structural steel with metal siding and roof deck. Potential missiles or debris from these materials, resulting from failure of superstructure or from items blown off, when subjected to winds of tornado intensity, are not considered to generate missiles more severe than the Spectrum I missiles of SRP 3.5.1.4 in accordance with Reference 3.5-8.

3.5.1.5 **Site Proximity Missiles (Except Aircraft)**

The site is selected such that the probability of occurrence of the site proximity missile (except aircraft) is less than 10^{-7} occurrences per year. Site-specific missile sources are addressed in Section 2.2. The site proximity missile has been dismissed from further consideration because at that likelihood it is considered not to be a statistically significant risk.

3.5.1.6 **Aircraft Hazards**

The probability of aircraft hazards impacting the ESBWR Standard Plant and causing consequences greater than 10 CFR 52.47(a)(2)(iv) exposure limits is < about 10^{-7} per year.Site-specific aircraft hazard analysis and the site-specific critical areas are addressed in Section 2.2.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

This subsection discusses the SSCs to be protected from externally generated missiles and includes all safety-related SSCs on the plant site that are provided to support the reactor facility.

The sources of external missiles, which could affect the safety of the plant, are identified in Subsection 3.5.1. Certain items in the plant are required to safely shut down the reactor and maintain it in a safe condition assuming an additional single failure. These items, whether they are structures, systems or components, must all be protected from externally generated missiles.

These items are the safety-related items listed in Table 3.2-1; appropriate safety classes and equipment locations are given in this table. All of the safety-related systems listed are located in buildings that are designed as tornado resistant. Because the tornado missiles are the design basis missiles, the SSCs listed are adequately protected. Provisions are made to protect the Off-Gas Charcoal Bed Adsorbers, Seismic Category I portions of the Fire Protection System (FPS) and components of Fuel and Auxiliary Pool Cooling System that transport makeup water to Spent Fuel Pool and Isolation Condenser/Passive Containment Cooling Pools from the FPS against tornado missiles.

3.5.3 Barrier Design Procedures

The procedures by which structures and barriers are designed to resist the missiles described in Subsection 3.5.1 are presented in this section. The following procedures are in accordance with Subsection 3.5.3 of NUREG-0800 (Standard Review Plan) and ensure that the design of structures, shields, and barriers that must withstand the effects of environmental and natural phenomena meet the relevant requirements of GDC 2 and GDC 4.

3.5.3.1 Local Damage Prediction

The prediction of local damage in the impact area depends on the basic material of construction of the structure or barrier (i.e., concrete or steel). The corresponding procedures are presented separately.

3.5.3.1.1 Concrete Structures and Barriers

Sufficient thickness of concrete is provided to prevent perforation, spalling or scabbing of the barriers in the event of missile impact. The (modified) National Defense Research Committee formula (Reference 3.5-5) is applied analytically for missile penetration in concrete. To prevent perforation, ACI-349 Appendix C Section C.7 is used. The resulting thickness of concrete required to prevent perforation, spalling or scabbing is no less than that for Region I listed in Table 1 of SRP 3.5.3.

3.5.3.1.2 **Steel Structures and Barriers**

The Stanford equation (Reference 3.5-6) is applied for steel structures and barriers. Composite barriers are not utilized in the ESBWR Standard Plant for missile protection.

3.5.3.2 **Overall Damage Prediction**

The overall response of a structure or barrier to missile impact depends largely upon the location of impact (e.g., near mid-span or near a support), dynamic properties of the structure/barrier and missile, and on the kinetic energy of the missile. In general, it is assumed that the momentum of the missile is transferred to the structure or barrier and only a portion of the kinetic energy is absorbed as strain energy within the structure or barrier.

After demonstrating that the missile does not perforate the structure or barrier, an equivalent static load concentrated at the impact area is determined. The structural response to this load, in conjunction with other appropriate design loads, is evaluated using an analysis procedure provided in Reference 3.5-7.

The maximum allowable ductility ratios for steel and reinforced concrete given in ANSI/AISC N690-1994 including Supplement 2 (Reference Number 15 of Table 3.8-6) and RG 1.142 respectively are met.

3.5.3.3 **Impact of Failure of Nonsafety-Related Structures, Systems and Components**

Any non-seismic structure postulated to fail under tornado wind loads is located at least a distance of its height above grade from Seismic Category I structures. Per Subsection 3.5.2, Offgas Charcoal Bed Adsorbers are provided with missile protection.

3.5.4 **COL Information**

None.

3.5.5 **References**

3.5-1 USNRC, “Safety Evaluation Report of U.S. ABWR,” NUREG-1503, July 1994.

3.5-2 C. V. Moore, “The Design of Barricades for Hazardous Pressure Systems,” Nuclear Engineering and Design, Vol. 5, 1967.

3.5-3 “River Bend Station Updated Safety Analysis Report,” Docket No. 50-458, Volume 6, Pages 3.5-4 and 3.5-5, August 1987.

3.5-4 F. J. Moody, “Prediction of Blowdown Thrust and Jet Forces,” ASME Publication 69-HT-31, August 1969.

3.5-5 R. P. Kennedy, “A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects,” Nuclear Engineering and Design, Volume 37, Issue 2, May 1976.

3.5-6 Oak Ridge National Laboratory, W. B. Cottrell and A. W. Savolainen, “U. S. Reactor Containment Technology,” ORNL-NSIC-5, Vol. 1, Chapter 6.

3.5-7 Bechtel Power Corporation, “Design of Structures for Missile Impact”, Topical Report, BC-TOP-9A, Revision 2, September 1974.

3.5-8 J. R. McDonald, “Rationale for Wind-borne Missile Criteria for DOE facilities”, Sept. 1999 (UCRL-CR-135687 S/C B505188).

3.5-9 (Deleted)

Table 3.5-1 Requirement for the Probability of Missile Generation for ESBWR Standard Plant

Criterion	Probability/Yr	Required Licensee Action
(A)	$P_1 < 10^{-4}$	Criterion (A) is the general reliability requirement for loading the turbine and bringing the system on line.
(B)	$10^{-4} < P_1 < 10^{-3}$	If Criterion (B) is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the applicant, referencing the ESBWR design, is to take action to reduce P_1 to meet Criterion (A) before returning the turbine to service.
(C)	$10^{-3} < P_1 < 10^{-2}$	If Criterion (C) is reached during operation, the turbine is to be isolated within 60 days, at which time the applicant, referencing the ESBWR design, is to take action to reduce P_1 to meet Criterion (A) before returning the turbine to service.
(D)	$10^{-2} < P_1$	If Criterion (D) is reached at any time during the operation, the turbine is to be isolated from the steam supply within 6 days, at which time the applicant, referencing the ESBWR design, is to take action to reduce P_1 to meet Criterion (A) before returning the turbine to service.

Figure 3.5-1 Missile Velocity and Displacement Characteristics Resulting from Saturated Steam and Water Blowdowns (7.2 MPa (1044 psia) Stagnation Pressure)

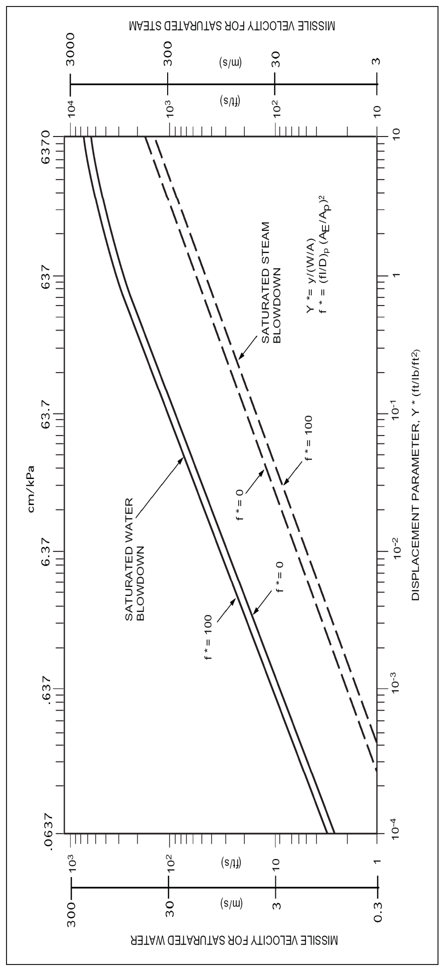
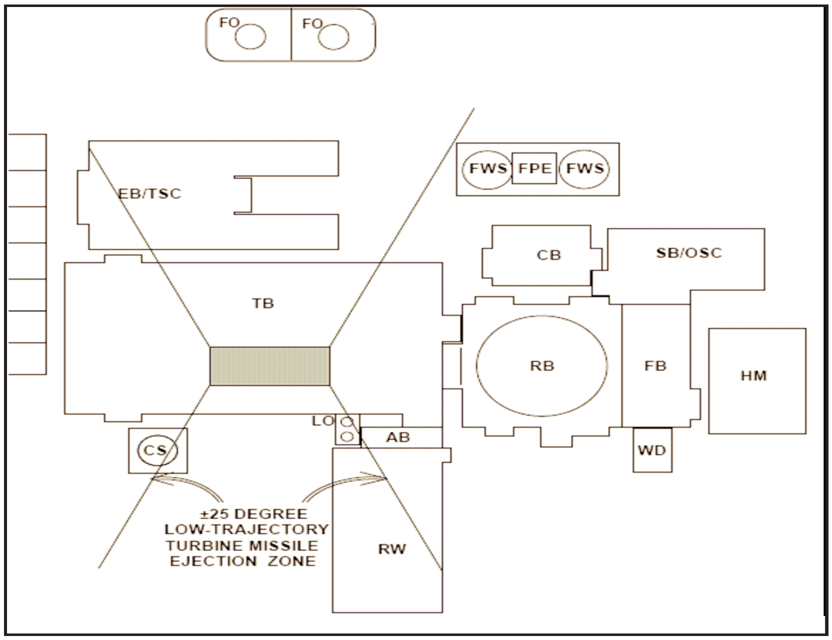


Figure 3.5-2 ESBWR Standard Plant Low-Trajectory Turbine Missile Strike Zone



See Figure 1.1-1 for nomenclature.

See Figure 1.1-1 for nomenclature.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

This section deals with the structures, systems, components and equipment in the ESBWR Standard Plant.

Subsections 3.6.1 and 3.6.2 describe the design bases and protective measures which ensure that (1) the containment, (2) safety-related systems, components and equipment, and (3) other safety-related structures are adequately protected from the consequences associated with a postulated rupture of high-energy piping or crack of moderate-energy piping both inside and outside the containment.

Before delineating the criteria and assumptions used to evaluate the consequences of piping failures inside and outside of containment, it is necessary to define a pipe break event and a postulated piping failure:

- Pipe Break Event—Any single postulated piping failure occurring during normal plant operation and any subsequent piping failure and/or equipment failure that occurs as a direct consequence of the postulated piping failure.
- Postulated Piping Failure—Longitudinal or circumferential break or rupture postulated in high-energy fluid system piping or through-wall leakage crack postulated in moderate-energy fluid system piping. The terms used in this definition are explained in Subsection 3.6.2.

Structures, systems, components and equipment that are required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power, are defined as safety-related and are designed to Seismic Category I requirements.

The dynamic effects that may result from a postulated rupture of high-energy piping include (1) missile generation, (2) pipe whipping, (3) pipe break reaction forces, (4) jet impingement forces, (5) compartment, subcompartment, and cavity pressurizations, (6) decompression waves within the ruptured pipes, and (7) eight types of loads identified with a Loss-of-Coolant-Accident. (See Subsection 3.8.1.3.5.)

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Inside and Outside of Containment

In accordance with NUREG-0800, SRP 3.6.1 and SRP 3.6.2, the plant is designed for protection against piping failures inside and outside containment to assure that such failures do not cause the loss of needed functions of safety-related systems and to ensure that the plant can be safely shut down in the event of such failures. The design includes consideration of high energy and moderate energy fluid system piping located inside and outside of containment. Where such a system penetrates containment, consideration starts with the first isolation valve outside of containment.

3.6.1.1 Design Bases

Criteria

Pipe break event protection conforms to 10 CFR 50 Appendix A, General Design Criterion 4, Environmental and Dynamic Effect Design Bases, as it relates to safety-related Structures, Systems, or Components (SSCs) being designed to accommodate the dynamic effects of postulated pipe rupture, including the effects of pipe whipping and discharging fluids. The design bases for this protection are in compliance with NRC Branch Technical Position (BTP) SPLB 3-1 (Formerly BTP ASB 3-1), and BTP 3-4 included in Subsections 3.6.1 and 3.6.2, respectively, of NUREG-0800 (Standard Review Plan). BTP 3-4 describes an acceptable basis for selecting the design locations and orientations of postulated breaks and cracks in fluid systems piping. Standard Review Plan Subsections 3.6.1 and 3.6.2 describe acceptable measures that could be taken for protection against the breaks and cracks and for restraint against pipe whip that may result from breaks.

The design of the containment structure, component arrangement, pipe runs, pipe whip restraints, and compartmentalization are done in consonance with the acknowledgment of protection against dynamic effects associated with a pipe break event. Analytically sized and positioned pipe whip restraints are engineered to preclude damage based on the pipe break evaluation.

Objectives

Protection against pipe break event dynamic effects is provided to fulfill the following objectives:

- Assure that the reactor can be shut down safely and maintained in a safe shutdown condition and that the consequences of the postulated piping failure are mitigated to acceptable limits with Loss of Preferred Power (LOPP).
- Assure that containment integrity is maintained.
- Assure that the radiological doses of a postulated piping failure remain below the limits of 10 CFR 52.47(a)(2)(iv).

Assumptions

The following assumptions are used to determine the protection requirements:

- Pipe break events may occur during normal plant conditions (i.e., reactor startup, operation at power, normal hot standby (Reference 3.6-1) or reactor cooldown to a cold shutdown condition but excluding test modes).
- A pipe break event may occur simultaneously with a seismic event; however, a seismic event does not initiate a pipe break event. This applies to Seismic Category I and non-Seismic Category I piping (seismically analyzed).

- A Single Active Component Failure (SACF) is assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor, except as noted below. A SACF is the malfunction or loss of function 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, or electrical malfunction but not the loss of component structural integrity. The direct consequences of a SACF are considered to be a part of the single active failure. The SACF is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure.
- Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy safety-related system (i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure), single active failure of components in the other train or trains of that system only are not assumed. This applies, provided the system is designed to Seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing and in-service inspection standards appropriate for safety-related systems.
- If a pipe break event involves a failure of non-Seismic Category I piping, the pipe break event must not result in failure of safety-related systems, components and equipment to shut down the reactor and mitigate the consequences of the pipe break event considering a SACF.
- If LOPP is a direct consequence of the pipe break event (e.g., trip of the turbine-generator producing a power surge that, in turn, trips the main breaker), then a LOPP occurs in a mechanistic time sequence with a SACF. Otherwise, preferred power is assumed available with a SACF.
- A whipping pipe is not capable of rupturing impacted pipes of equal or greater nominal pipe diameter, but may develop through-wall cracks in equal or larger nominal pipe sizes with thinner wall thickness.
- All available systems, including those actuated by operators, are able to mitigate the consequences of a failure. In judging the availability of systems, account is taken of the postulated failure and its direct consequences such as unit trip and LOPP, and of the assumed SACF and its direct consequences. The feasibility of carrying out operator actions is judged on the basis of ample time and adequate access to equipment being available for the proposed actions.
- Although a pipe break event outside the containment may require a cold shutdown, up to eight hours in hot standby is allowed in order for plant personnel to assess the situation and make repairs.
- Pipe whip with rapid motion of pipe resulting from a postulated pipe break occurs in the plane determined by the piping geometry and causes movement in the direction of the jet reaction. If

unrestrained, a whipping pipe with a constant energy source forms a plastic hinge and rotates about the nearest rigid restraint, anchor, or wall penetration. If unrestrained, a whipping pipe without a constant energy source (i.e., a break at a closed valve with only one side subject to pressure) is not capable of forming a plastic hinge or rotating about the hinge, provided its movement can be defined and evaluated.

- The fluid internal energy associated with the pipe break reaction can take into account any line restrictions (e.g., flow limiter) between the pressure source and break location and absence of energy reservoirs, as applicable.
- All walls, doors and penetrations which serve as divisional boundaries are designed to withstand the worst case pressurizations associated with the postulated pipe failures inside primary containment. All structural divisional separation walls are designed to maintain their structural integrity after a postulated failure outside containment and within Reactor Building. Divisional separation doors, penetrations and floors are not required to maintain their structural integrity. Justification for divisional separation integrity is addressed in Subsections 3.4.1, 6.2.3 and 9.5.1.

Approach

To comply with the objectives previously described, the safety-related systems, components, and equipment are identified. The safety-related systems, components, and equipment, or portions thereof, are identified in Table 3.6-1 for piping failures postulated inside the containment and in Table 3.6-2 for outside the containment.

3.6.1.2 Description

The lines identified as high and moderate-energy per Subsection 3.6.2.1 are listed in Table 3.6-3 for inside the containment and in Table 3.6-4 for outside the containment. Pressure response analyses are performed for the subcompartments containing high-energy piping. A detailed discussion of the line breaks selected, vent paths, room volumes, analytical methods, pressure results, etc., is provided in Section 6.2.

The effects of pipe whip, jet impingement, spraying, and flooding on the required functions of safety-related systems, components, and equipment, or portions thereof, inside and outside the containment, are considered.

In particular, there are no high-energy lines near the control room. As such, there are no effects upon the habitability of the control room by a piping failure in the control room or elsewhere either from pipe whip, jet impingement, or transport of steam. Further discussion on control room habitability systems is provided in Section 6.4.

3.6.1.3 **Design Evaluation**

General

An analysis of pipe break events is performed to identify those safety-related systems, components, and equipment that provide protective actions required to mitigate, to acceptable limits, the consequences of the pipe break event.

Pipe break events involving high-energy fluid systems are evaluated for the effects of pipe whip, jet impingement, flooding, room pressurization, and other environmental effects such as temperature. Pipe break events involving moderate-energy fluid systems are evaluated for wetting from spray, flooding, and other environmental effects.

By means of the design features such as separation, barriers, and pipe whip restraints, a discussion of which follows, adequate protection is provided against the effects of pipe break events for safety-related items to an extent that their ability to shut down the plant safely or mitigate the consequences of the postulated pipe failure would not be impaired.

General Protection Methods

The direct effects associated with a particular postulated break or crack are mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the following specific measure for protection against actual pipe movement and other associated consequences of postulated failures:

- Protection against the dynamic effects of pipe failures is provided in the form of pipe whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation.
- The precise method chosen depends largely upon limitations placed on the designer such as accessibility, maintenance, and proximity to other pipes.

Protection Methods by Separation

The plant arrangement provides physical separation to the extent practicable to maintain the independence of redundant safety-related systems (including their auxiliaries) in order to prevent the loss of safety function caused by any single postulated event. Redundant trains (e.g., A and B trains) and divisions are located in separate compartments to the extent possible. Physical separation between redundant safety-related systems with their related auxiliary supporting features, therefore, is the basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

Because of the complexities of several divisions being adjacent to high-energy lines in the drywell, specific break locations are determined in accordance with Subsection 3.6.2.1 for possible spatial separation. Care is taken to avoid concentrating safety-related equipment in the break exclusion zone allowed according to Subsection 3.6.2.1. If spatial separation requirements (distance and/or

arrangement to prevent damage) cannot be met based on the postulation of specific breaks, then barriers, enclosures, shields, or restraints are provided. These methods of protection are discussed below.

For other areas where physical separation is not practical, the following High Energy Line Separation Analysis (HELSA) evaluation is done to determine which high-energy lines meet the spatial separation requirement and which lines require further protection:

- For the HELSA evaluation, no particular break points are identified. Cubicles or areas through which the high-energy lines pass are examined in total. Breaks are postulated at any point in the piping system.
- Safety-related systems, components, and equipment at a distance greater than 9.1 m (30 ft) from any high energy piping are considered as meeting spatial separation requirements. No damage is assumed to occur on account of jet impingement, because the impingement force becomes negligible beyond 9.1 m (30 ft). Likewise, a 9.1 m (30 ft) evaluation zone is established for pipe breaks to assure protection against potential damage from a whipping pipe. Assurance that 9.1 m (30 ft) represents the maximum free length is made in the piping layout.

  Exception to the 9.1 m (30 ft) rule is the main steam line break (750 mm [30 inch] diameter pipe) for which 19.1 m (63 ft approx) is used as the evaluation zone for the jet interaction and structural integrity determination to any SSCs.
- Safety-related systems, components, and equipment at a distance less than 9.1 m (30 ft) from any high-energy piping are evaluated to see if damage could occur to more than one safety-related division, preventing safe shutdown of the plant. If damage occurred to only one division of a redundant system, the requirement for redundant separation is met. Other redundant divisions are available for safe shutdown of the plant and no further evaluation is performed.

  Exception to the 9.1 m (30 ft) rule is the main steam line break (750 mm [30 inch] diameter pipe), for which the distance of 19.1 m (63 ft approx) is used as requirement for the above evaluation.
- If damage could occur to more than one division of a redundant safety-related system within 9.1 m (30 ft) of any high energy piping, other protection in the form of barriers, shields, or enclosures is used. Exception to the 9.1 m (30 ft) rule is the main steam line break (750 mm [30 inch] diameter pipe), for which the distance of 19.1 m (63 ft approx) is used as the requirement for the evaluation. Pipe whip restraints are used if protection from whipping pipe is not possible by barriers and shields. These methods of protection are discussed below.

Barriers, Shields, and Enclosures

Protection requirements are met through the protection afforded by the walls, floors, columns, abutments, and foundations in many cases. Where adequate protection is not already present

because of spatial separation or existing plant features, additional barriers, deflectors, or shields are identified as necessary to meet the functional protection requirements.

Barriers or shields that are identified as necessary by the use of specific break locations in the drywell are designed for the specific loads associated with the particular break location.

The Main Steam Isolation Valves (MSIV) and the feedwater isolation and check valves located inside the main steam tunnel are designed for the effects of a line break. The details of how the MSIV and feedwater isolation and check valves functional capabilities are protected against the effects of these postulated pipe failures are provided as part of the pipe break evaluation report.

Barriers or shields that are identified as necessary by the HELSA evaluation (i.e., based on no specific break locations) are designed for worst-case loads. The closest high-energy pipe location and resultant loads are used to size the barriers.

Pipe Whip Restraints

Pipe whip restraints are used where pipe break protection requirements could not be satisfied using spatial separation, barriers, shields, or enclosures alone. Restraints are located based on the specific break locations determined in accordance with Subsection 3.6.2.1. After the restraints are placed, the piping and safety-related systems are evaluated for jet impingement and pipe whip. For those cases where jet impingement damage could still occur, barriers, shields, or enclosures are utilized.

The design criteria for restraints are given in Subsection 3.6.2.3.

Specific Protection Measures

- Nonsafety-related systems and system components are not required for the safe shutdown of the reactor, nor are they required for the limitation of the offsite release in the event of a pipe rupture. However, while none of this equipment is needed during or following a pipe break event, pipe whip protection is considered where a resulting failure of a nonsafety-related system or component could initiate or escalate the pipe break event in a safety-related system or component, or in another nonsafety-related system whose failure could affect a safety-related system.
- For high energy piping systems penetrating through the containment, isolation valves are located as close to the containment as possible.
- The pressure, water level, and flow sensor instrumentation for those safety-related systems required to function following a pipe rupture are protected.
- High-energy fluid system pipe whip restraints and protective measures are designed so that a postulated break in one pipe could not, in turn, lead to a rupture of other nearby pipes or components, if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break.

- For any postulated pipe rupture, the structural integrity of the containment is maintained. In addition, for those postulated ruptures classified as a loss of reactor coolant, the design leaktightness of the containment fission product barrier is maintained.
- Safety relief valves (SRVs) are located and restrained so that a pipe failure would not prevent depressurization.
- Protection for the FMCRD scram insert lines is not required, because the motor operation of the FMCRD can adequately insert the control rods even with a complete loss of insert lines (Subsection 3.6.2.1.3).
- The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture do not preclude:
  - accessibility to any areas required to cope with the postulated pipe rupture.
  - habitability of the control room.
  - the ability of safety-related instrumentation, electric power supplies, components, and controls to perform their safety-related function.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Information concerning break and crack location criteria and methods of analysis for dynamic effects are discussed in this Subsection in accordance with NUREG-0800, SRP 3.6.2. This includes location criteria and methods of analysis needed to evaluate the dynamic effects associated with postulated breaks and cracks in high and moderate-energy fluid system piping inside and outside of the primary containment. This information provides the basis for the requirements for the protection of safety-related structures, systems, and components defined in the introduction of Section 3.6, which includes meeting the requirements of GDC 4 as it relates to safety-related SSCs being designed to accommodate the dynamic effects of postulated pipe rupture, including postulation of pipe rupture locations; break and crack characteristics; dynamic analysis of pipe-whip; and jet impingement loads.

The plant meets the relevant requirements of GDC 4 as follows:

1. Criteria defining postulated pipe rupture locations and configurations inside containment are in accordance with BTP 3-4. For the piping system with reactor water, if the environmental fatigue is included in accordance with Regulatory Guide (RG) 1.207, the fatigue usage limit should be ≤ 0.40 as the criterion instead of ≤ 0.10 for determining pipe break locations.
2. Protection against postulated pipe ruptures outside containment is provided in accordance with BTP 3-4.
3. Detailed acceptance criteria covering pipe-whip dynamic analysis, including determination of the forcing functions of jet thrust and jet impingement are in accordance with Section III of

SRP 3.6.2. The general bases and assumptions of the analysis are in accordance with BTP 3-4.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

The following subsections establish the criteria for the location and configuration of postulated breaks and cracks.

Definition of High-Energy Fluid Systems

High-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.6.1.1), are either in operation or are maintained pressurized under conditions where either or both of the following are met:

- maximum operating temperature exceeds 93.3°C (200°F)
- maximum operating pressure exceeds 1.9 MPaG (275 psig)

Definition of Moderate-Energy Fluid Systems

Moderate-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.6.1.1), are either in operation or are maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

- maximum operating temperature is 93.3°C (200°F) or less
- maximum operating pressure is 1.9 MPaG (275 psig) or less

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system function but, for the major operational period, qualify as moderate-energy fluid systems. An operational period is considered short if the total fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2% of the total time that the system operates as a moderate-energy fluid system.

Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or a sudden longitudinal split without pipe severance, and is postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe failures are limited to postulation of cracks in piping and branch runs; these cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe. High-energy fluid systems are also postulated to have cracks for conservative environmental conditions in a confined area where high and moderate-energy fluid systems are located.

The following high-energy piping systems (or portions of systems) are considered as potential candidates for a postulated pipe break during normal plant conditions and are analyzed for potential damage resulting from dynamic effects:

- all piping which is part of the reactor coolant pressure boundary (RCPB) and subject to reactor pressure continuously during station operation.
- all piping which is beyond the second isolation valve but subject to reactor pressure continuously during station operation.
- all other piping systems or portions of piping systems considered high-energy systems.

Portions of piping systems that are isolated from the source of the high-energy fluid during normal plant conditions are exempted from consideration of postulated pipe breaks. This includes portions of piping systems beyond normally closed valves. Pump and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness.

3.6.2.1.1 ***Locations of Postulated Pipe Breaks***

Postulated pipe break locations are selected as follows:

Piping Meeting Separation Requirements

Based on the HELSA evaluation described in Subsection 3.6.1.3, the high-energy lines which meet the spatial separation requirements are generally not identified with particular break points. Breaks are postulated at all possible points in such high-energy piping systems. However, in some systems break points are particularly specified according to the following subsections if special protection devices such as barriers or restraints are provided.

Piping in Containment Penetration Areas

No pipe breaks or cracks are postulated in those portions of piping from the containment wall penetration to and including the inboard or outboard isolation valves which meet the following requirements in addition to the requirement of the ASME Code, Section III, Subarticle NE-1120:

- *The following design stress and fatigue limits are not exceeded:*

  For ASME Code, Section III, Class 1 Piping

  - *The maximum stress range between any two load sets (including the zero load set) does not exceed 2.4 S_m, and is calculated by Equation 10 in NB-3653, ASME Code, Section III. If the calculated maximum stress range of Equation 10 exceeds 2.4 S_m, the stress ranges calculated by both Equation 12 and Equation 13 in paragraph NB-3653 meet the limit of 2.4 S_m.*
  - *The cumulative usage factor is less than 0.1.*

For the piping system with reactor water, if the environmental fatigue is included in accordance with RG 1.207, the fatigue usage limit should be ≤ 0.40 as the criterion instead of ≤ 0.10 for determining pipe break locations.

- *The maximum stress as calculated by Equation 9 in NB-3652 under the loadings resulting from a postulated piping failure beyond those portions of piping, does not exceed the lesser of 2.25 S_m and 1.8 S_y except that, following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stress, provided a plastic hinge is not formed and operability of the valves with such stresses is assured in accordance with the requirements identified in Subsection 3.9.3. Primary loads include those that are deflection limited by whip restraints.*

For ASME Code, Section III, Class 2 Piping

- *The maximum stress as calculated by the sum of Equations 9 and 10 in Paragraph NC-3653, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits are specified in the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion), excluding an earthquake event, does not exceed 0.8(1.8 S_h + S_A). The S_h and S_A are allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.*
- *The maximum stress, as calculated by Equation 9 in NC-3653 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping, does not exceed the lesser of 2.25 S_h and 1.8 S_y.*

Primary loads include those that are deflection limited by whip restraints. The exceptions permitted above may also be applied provided that, when the piping between the outboard isolation valve and the restraint is constructed in accordance with the Power Piping Code ASME Code B31.1, the piping is either of seamless construction with full radiography of all circumferential welds, or all longitudinal and circumferential welds are fully radiographed.

- *Welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the above mentioned code limits.*
- *The number of circumferential and longitudinal piping welds and branch connections are minimized. Where penetration sleeves are used, the enclosed portion of fluid system piping is seamless construction and without circumferential welds unless specific access provisions are made to permit in-service volumetric examination of longitudinal and circumferential welds.*
- *The length of these portions of piping are reduced to the minimum length practical.*
- *The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) do not require welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings may be used) except where such welds are 100% volumetrically*

examinable in service and a detailed stress analysis is performed to demonstrate compliance with the above mentioned code limits.

- *Sleeves provided for those portions of piping in the containment penetration areas are constructed in accordance with the rules of Class MC, Subsection NE of the ASME Code, Section III, where the sleeve is part of the containment boundary. In addition, the entire sleeve assembly is designed to meet the following requirements and tests:*
  - *The design pressure and temperature are not less than the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions.*
  - *The Level C stress limits in NE-3220, ASME Code, Section III, are not exceeded under the loadings associated with containment design pressure and temperature in combination with the safe shutdown earthquake (SSE).*
  - *The assemblies are subjected to a single pressure test at a pressure not less than its design pressure.*
  - *The assemblies do not prevent the access required to conduct the in-service examination specified below.*
- *A 100% volumetric in-service examination of all pipe welds is conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI.*

ASME Code Section III Class 1 Piping in Areas Other Than Containment Penetration

With the exception of those portions of piping identified above, breaks in ASME Code, Section III, Class 1 piping are postulated at the following locations in each piping and branch run:

- *At terminal ends including the locations shown in Figure 3.6-3.*
- *At intermediate locations where the maximum stress range as calculated by Equation 10 in NB-3653, ASME Code, Section III exceeds 2.4 S_m, and either Equation 12 or Equation 13 in Paragraph NB-3653 exceeds 2.4 S_m.*
- *At intermediate locations where the cumulative usage factor exceeds 0.1. As a result of piping reanalysis caused by differences between the design configuration and the as-built configuration, the highest stress or cumulative usage factor locations may be shifted; however, the initially determined intermediate break locations need not be changed unless one of the following conditions exists:*
  - *The dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.*
  - *A change is required in pipe parameters, such as major differences in pipe size, wall thickness, and routing.*

For the piping system with reactor water, if the environmental fatigue is included in accordance with RG 1.207, the fatigue usage limit should be ≤ 0.40 as the criteria instead of ≤ 0.10 for determining pipe break locations.

ASME Code Section III Class 2 and 3 Piping in Areas Other Than Containment Penetration

With the exceptions of those portions of piping identified above, breaks in ASME Code, Section III, Class 2 and 3 piping are postulated at the following locations in those portions of each piping and branch run:

- *At terminal ends.*
- *At intermediate locations selected by one of the following criteria:*
  - *At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve.*
  - *At one location at each extreme of the piping run adjacent to the protective structure for piping that contains no fittings, welded attachments, or valves.*
  - *At each location where stresses calculated by the sum of Equations 9 and 10 in NC/ND-3653, ASME Code, Section III, exceed 0.8 times the sum of the stress limits given in NC/ND-3653.*

Piping will be designed to minimize the stresses and fatigue usage factors such that intermediate pipe break locations are avoided.

As a result of piping reanalysis caused by differences between the design configuration and the as-built configuration, the highest stress locations may be shifted; however, the initially determined intermediate break locations may be used unless a redesign of the piping resulting in a change in the pipe parameters (diameter, wall thickness, routing) is required, or the dynamic effects from the new (as-built) intermediate break location are not mitigated by the original pipe whip restraints and jet shields.

For complex piping systems such as those containing arrangements of headers and parallel piping running between headers, the pipe breaks are postulated pursuant to the applicable criteria identified in this subsection and in conformance with BTP 3-4.

The terminal end pipe break locations for high energy lines inside and outside containment are provided in Table 3.6-5 and 3.6-6. The high energy line breaks at the containment penetration outside of the containment penetration zone are provided in Table 3.6-7. Terminal end break locations in piping systems on both sides of the containment penetration are shown in Figure 3.6-3.

Non-ASME Class Piping

Breaks in seismically analyzed non-ASME Class (not ASME Class 1, 2, or 3) piping are postulated according to the same requirements for ASME Class 2 and 3 piping above. Separation and

interaction requirements between seismically analyzed and non-seismically analyzed piping are met as described in Subsection 3.7.3.8.

Separating Structure With High-Energy Lines

If a structure separates a high-energy line from a safety-related component, the separating structure is designed to withstand the consequences of the pipe break in the high-energy line at locations that the aforementioned criteria require to be postulated. However, as noted in Subsection 3.6.1.3, some structures which are identified as necessary by the HELSA evaluation (i.e., based on no specific break locations) are designed for worst-case loads.

3.6.2.1.2 ***Locations of Postulated Pipe Cracks***

Postulated pipe crack locations are selected as follows:

Piping Meeting Separation Requirements

Based on the HELSA evaluation described in Subsection 3.6.1.3, the high- or moderate-energy lines, which meet the separation requirements, are not identified with particular crack locations. Cracks are postulated at all possible points that are necessary to demonstrate adequacy of separation or other means of protections provided for safety-related SSCs.

High-Energy Piping

With the exception of those portions of piping identified above, leakage cracks are postulated for the most severe environmental effects as follows:

- *For ASME B&PV Code, Section III Class 1 piping, at axial locations where the calculated stress range by Equation 10 and either Equation 12 or Equation 13 in NB-3653 exceeds 1.2 S_m.*
- *For ASME B&PV Code, Section III Class 2 and 3 or non-ASME class piping, at axial locations where the calculated stress by the sum of Equations 9 and 10 in NC/ND-3653 exceeds 0.4 times the sum of the stress limits given in NC/ND-3653.*
- *Non-ASME class piping, which has not been evaluated to obtain stress information, has leakage cracks postulated at axial locations that produce the most severe environmental effects.*

Moderate-Energy Piping in Containment Penetration Areas

Leakage cracks are not postulated in those portions of piping from the containment wall to and including the inboard or outboard isolation valves, provided (1) they meet the requirements of the ASME B&PV Code, Section III, NE-1120, and (2) the stresses calculated by the sum of Equations 9 and 10 in ASME B&PV Code, Section III, NC-3653 do not exceed 0.4 times the sum of the stress limits given in NC-3653.

Moderate-Energy Piping in Areas Other Than Containment Penetration

- *Leakage cracks are postulated in piping located adjacent to safety-related SSCs, except:*
  - *Where exempted above.*
  - *For ASME B&PV Code, Section III, Class 1 piping the stress range calculated by Equation 10 and either Equation 12 or Equation 13 in NB-3653 is less than 1.2 S_m.*
  - *For ASME B&PV Code, Section III, Class 2 or 3 and non-ASME class piping, the stresses calculated by the sum of Equations 9 and 10 in NC/ND-3653 are less than 0.4 times the sum of the stress limits given in NC/ND-3653.*
- *Leakage cracks, unless the piping system is exempted above, are postulated at axial and circumferential locations that result in the most severe environmental consequences.*
- *Leakage cracks are postulated in fluid system piping designed to non-seismic standards as necessary to meet the environmental protection requirements of Subsection 3.6.1.1.*

Moderate-Energy Piping in Proximity to High-Energy Piping

Moderate-energy fluid system piping or portions thereof which are located within a compartment of confined area involving considerations for a postulated break in high-energy fluid system piping, are acceptable, without postulation of through-wall leakage cracks, except where a postulated leakage crack in the moderate-energy fluid system piping results in more severe environmental conditions than the break in the proximate high-energy fluid system piping, in which case the provisions of this subsection are applied.

3.6.2.1.3 **Types of Breaks and Cracks to be Postulated**

Pipe Breaks

The following types of breaks are postulated in high-energy fluid system piping at the locations identified by the criteria specified in Subsection 3.6.2.1.1.

- No breaks are postulated in piping having a nominal diameter less than or equal to 25 mm (1 inch). Instrument lines 25 mm (1 in) and less nominal pipe or tubing size meet the provision of RG 1.11. Additionally, the 32 mm (1.25 in) nominal diameter hydraulic control units (HCU) fast scram lines do not require special protection measures because of the following reasons:
  - The piping to the control rod drives (CRDs) from the HCUs are located in the containment under reactor vessel, and in the Reactor Building away from other safety-related equipment; therefore, should a line fail, it would not affect any safety-related equipment but only impact other HCU lines. As discussed in Subsection 3.6.1.1, a whipping pipe can only rupture an impacted pipe of smaller nominal pipe size or cause a through-wall crack in the same nominal pipe size but with thinner wall thickness.

  - The total amount of energy contained in the 32 mm (1.25 in) nominal diameter piping between the normally closed scram insert valve on the HCU module and the ball-check valve in the control rod housing is smaller than 6 kJ per meter (1348.85 ft. lbf per foot) of the 32 mm (1.25 in) line. In the event of a rupture of this line, the ball-check valve would close to prevent reactor vessel flow out of the break.
  - Even if a number of the HCU lines ruptured, the control rod insertion function would not be impaired, because the electrical motor of the fine motion control drive would drive in the control rods.
- Longitudinal breaks are postulated only in piping having a nominal diameter equal to or greater than 100 mm (4 in).
- Circumferential breaks are only assumed at all terminal ends.
- At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Subsection 3.6.2.1.1, consideration is given to the occurrence of either a longitudinal or circumferential break. Examination of the state of stress in the vicinity of the postulated break location is used to identify the most probable type of break. If the maximum stress range in the longitudinal direction is greater than 1.5 times the maximum stress range in the circumferential direction, only the circumferential break is postulated. Conversely, if the maximum stress range in the circumferential direction is greater than 1.5 times the stress range in the longitudinal direction, only the longitudinal break is postulated. If no significant difference between the circumferential and longitudinal stresses is determined, then both types of breaks are considered.
- Where breaks are postulated to occur at each intermediate pipe fitting, weld attachment, or valve without the benefit of stress calculations, only circumferential breaks are postulated.
- For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibility, pipe whip is assumed to occur in the plane defined by the piping geometry and configuration for circumferential breaks and out of plane for longitudinal breaks and to cause piping movement in the direction of the jet reactions. Structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis are considered in determining the piping movement limit (alternatively, circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections).
- For a circumferential break, the dynamic force of the jet discharged at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are used, as applicable, in the reduction of the jet discharge.

- Longitudinal breaks in the form of axial split without pipe severance are postulated in the center of the piping at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping configuration and produces out-of-plane bending. Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).
- For longitudinal breaks, the dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account as applicable in the reduction of jet discharge.

Pipe Cracks

The following criteria are used to postulate through-wall leakage cracks in high- or moderate-energy fluid systems or portions of systems:

- Cracks are postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 25 mm (1 in).
- At axial locations determined per Subsection 3.6.2.1.2, the postulated cracks are oriented circumferentially to result in the most severe environmental consequences.
- Crack openings are assumed as a circular orifice of area equal to that of a rectangle having dimensions one-half-pipe-diameter in length and one-half-pipe-wall thickness in width.
- The flow from the crack opening is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments, based on a conservatively estimated time period to effect corrective actions.

3.6.2.2 Analytic Methods to Define Blowdown Forcing Functions and Response Models

Analytic Methods to Define Blowdown Forcing Functions

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces for various piping systems are presented as follows:

The criteria that are used for calculation of fluid blowdown forcing functions include:

- Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically

limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).

- The dynamic force of the jet discharge at the break location is based on the cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally-determined thrust coefficient. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.
- All breaks are assumed to attain full size within one millisecond after break initiation.

Blowdown forcing functions are determined by the method specified in Appendix B of ANSI/ANS-58.2 (Reference 3.6-4).

Pipe Whip Dynamic Response Analyses

The prediction of time-dependent and steady thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from ruptured pipe is used in design and evaluation of dynamic effects of pipe breaks. A discussion of the analytical methods employed to compute these blowdown loads is given above. Following is a discussion of analytical methods used to account for this loading.

The criteria used for performing the pipe whip dynamic response analyses include the following:

- A pipe whip analysis is performed for each postulated pipe break. However, a given analysis can be used for more than one postulated break location if the blowdown forcing function, piping and restraint system geometry, and piping and restraint system properties are conservative for other break locations.
- The analysis includes the dynamic response of the pipe in question and the pipe whip restraints, which transmit loading to the support structures.
- The analytical model adequately represents the mass/inertia and stiffness properties of the system.
- Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration and to cause pipe movement in the direction of the jet reaction.
- Piping within the broken loop is no longer considered part of the RCPB. Plastic deformation in the pipe is considered as a potential energy absorber. Limits of strain are imposed which are similar to strain levels allowed in restraint plastic members. Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences do not result in direct damage to any safety-related system or component.
- Components, such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety-related function and components whose failure would not

further escalate the consequences of the accident are not designed to meet ASME Code-imposed limits for safety-related components under faulted loading. However, if these components are required for safe shutdown or serve to protect the structural integrity of a safety-related component, limits to meet the Code requirements for faulted conditions and limits to ensure required operability would be met.

Analyses for pipe whip restraint selection using the pipe dynamic analysis computer program and a pipe break modeling program (ANSYS) are performed as described in Appendix 3D, which predicts the response of a pipe subjected to the thrust force occurring after a pipe break. The program treats the situation in terms of generic pipe break configuration which involves a straight, uniform pipe fixed at one end and subjected to a time-dependent thrust force at the other end. A typical restraint used to reduce the resulting deformation is also included at a location between the two ends. Nonlinear and time-independent stress strain relationships are used to model the pipe and the restraint. Using a plastic-hinge concept, bending of the pipe is assumed to occur only at the fixed end and at the location supported by the restraint.

Effects of pipe shear deflection are considered negligible. The pipe-bending moment-deflection (or rotation) relation used for these locations is obtained from a static nonlinear cantilever-beam analysis. Using the moment-rotation relation, nonlinear equations of motion of the pipe are formulated using energy considerations and the equations are numerically integrated in small time steps to yield time-history of the pipe motion.

The piping stresses in the containment penetration areas are calculated by the ANSYS computer program, a program as described in Appendix 3D. The program is used to perform the non-linear analysis of a piping system for time varying displacements and forces due to postulated pipe breaks.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Jet Impingement Analyses and Effects on Safety-Related Components

The criteria used for evaluating the effects of fluid jets on safety-related SSCs are as follows:

- Safety-related SSCs are not impaired so as to preclude safety-related functions. For any given postulated pipe break and consequent jet, those safety-related SSCs needed to safely shut down the plant are identified.
- Safety-related SSCs which are not necessary to safely shut down the plant for a given break are not protected from the consequences of the fluid jet.
- Safe shutdown of the plant caused by postulated pipe ruptures within the RCPB is not aggravated by sequential failures of safety-related piping and the required emergency cooling system performance is maintained.
- Offsite doses comply with 10 CFR 52.47(a)(2)(iv).

- Postulated breaks resulting in jet impingement loads are assumed to occur in high-energy lines at 102% power operation of the plant.
- Through-wall leakage cracks are postulated in moderate-energy lines and are assumed to result in wetting and spraying of safety-related SSCs.
- Reflected jets are considered only when there is an obvious reflecting surface (such as a flat plate) which directs the jet onto safety-related equipment. Only the first reflection is considered in evaluating potential targets.
- Potential targets, or portions of targets adjacent to the jet boundary, are assumed to be impinged upon when reasonable variations in jet geometry or pipe movement are considered.

The analytical methods used to determine which targets could be impinged upon by a fluid jet and the corresponding jet impingement load include:

- The direction of the fluid jet is based on the arrested position, including reasonable variations of the broken pipe end movement during steady-state blowdown.
- The impinging jet proceeds along a straight path.
- The total impingement force acting on any cross-sectional area of the jet is time and distance invariant with a total magnitude equivalent to the steady-state fluid blowdown force given in Subsection 3.6.2.2 and with jet characteristics shown in Figure 3.6-1.
- The jet impingement thrust force on the target is calculated for certain cases according to ANSI/ANS 58.2 (Reference 3.6-4).
- For cases where the magnitude of a jet thrust force is only important for pipe reaction load, a detailed jet evaluation is not necessary. Simple load calculation may be applicable for a pipe break where the absence of an energy reservoir upstream or downstream of the break does not result in a continuous jet blowdown. A detailed jet impingement analysis is not significant for smaller pipe breaks if the design or analysis of larger size pipe break loads envelops these pipe break jet impingement loads affecting the same target. The jet shield, barrier and an enclosure designed for a large pipe break will bound smaller pipe jet impingement and pipe whip effects, and loads calculated based on simplified method may be sufficient to justify these cases.
- On a case by case basis a quantitative analysis approach to determine the dynamic jet force is necessary where jet characteristics such as, jet nonlinearity, turbulence, feedback amplification, and jet reflection are deemed significant in the jet modeling. For this purpose, other dynamic analysis method is appropriate such as computational fluid dynamic analysis. This method of analysis is capable of defining parameters associated with the jet flow properties, ambient condition, and surface profile of the interacting targets. The resulting force time history and jet pressures on target surface are obtained from such computational fluid dynamics analysis. The detailed jet analysis evaluation method is described later as analysis Steps 1, 2, and 3 in this subsection.

- The break opening is assumed to be a circular orifice of cross-sectional flow area equal to the effective flow area of the break.
- The jet impingement force is equal to the steady-state value of the fluid blowdown force calculated by the methods described in Subsection 3.6.2.2.
- The distance of jet travel is divided into two or three regions. Region 1 (Figure 3.6-1, items a, b and c) extends from the break to the asymptotic area. Within this region the discharging fluid flashes and undergoes expansion from the break area pressure to the atmospheric pressure. In Region 2 the jet expands further. For partial-separation circumferential breaks, the area increases as the jet expands. In Region 3, the jet expands at a half angle of 10 degrees (Figure 3.6-1, items a and c).
- The analytical model for estimating the asymptotic jet area for subcooled water and saturated water assumes a constant jet area. For fluids discharging from a break that are below the saturation temperature at the corresponding room pressure or have a pressure at the break area equal to the room pressure, the free expansion does not occur.
- The distance downstream from the break where the asymptotic area is reached (Region 2 in Figure 3.6-1) is calculated for circumferential and longitudinal breaks.
- Both longitudinal and fully separated circumferential breaks are treated similarly. The value of fL/D (where, f = friction factor, L/D = pipe length to pipe diameter ratio) used in the blowdown calculation is also used for jet impingement.
- Circumferential breaks with partial (i.e., h<D/2) separation between the two ends of the broken pipe not significantly offset (i.e., no more than one pipe wall thickness lateral displacement) are more difficult to quantify. For these cases, the following assumptions are made.
  - The jet is uniformly distributed around the periphery.
  - The jet cross-section at any cut through the pipe axis has the configuration depicted in Figure 3.6-1, item b. The jet regions are also shown.
  - The jet force F_j = total blowdown.

  The pressure at any point intersected by the jet (P_j) is:

$$P_j = \frac{F_s}{A_R} \tag{3.6-1}$$

where

A_R = the total 360° area of the jet at a radius equal to the distance from the pipe centerline to the target

F_s = Steady State blowdown force

The jet pressure at the target is calculated by:

$$P_1 = \frac{F_j}{A_x} \tag{3.6-2}$$

where

P_1 = incident pressure

A_x= area of the expanded jet at the target intersection.

Target shape factors are included in accordance with ANSI/ANS 58.2 (Reference 3.6-4).

If the effective target area (A_{te}) is less than the expanded jet area ($A_{te} < A_x$), the target is fully submerged in the jet and the impingement load is equal to (P_1) (A_{te}). If the effective target area is greater than the expanded jet area ($A_{te} > A_x$), the target intercepts the entire jet and the impingement load is equal to (P_1) (A_x) = F_j. Where applicable, the net load on the target is determined to be equal to the impingement load (F_j) times the appropriate shape factor of the target surface on which the jet is being impinged upon. The effective target area (A_{te}) for various geometries is as follows:

- Flat Surface — For a case where a target with physical area A_t is oriented at angle ϕ with respect to the jet axis and with no flow reversal, the effective target area A_{te} is:

$$A_{te} = (A_t)(\sin\phi) \tag{3.6-3}$$

- Pipe Surface — As the jet hits the convex surface of the pipe, its forward momentum is decreased rather than stopped; therefore, the jet impingement load on the impacted area is expected to be reduced. For conservatism, no credit is taken for this reduction and the pipe is assumed to be impacted with the full impingement load. However, where shape factors are justifiable, they may be used. The effective target area A_{te} is:

$$A_{te} = (D_A)(D) \tag{3.6-4}$$

where

D_A = diameter of the jet at the target interface

D = pipe outside diameter of target pipe for a fully submerged pipe

When the target (pipe) is larger than the area of the jet, the effective target area equals the expanded jet area

$$A_{te} = A_x \tag{3.6-5}$$

- For all cases, the jet area (A_x) is assumed to be uniform and the load is uniformly distributed on the impinged target area A_{te}.
- Where applicable, on a case-by-case basis, detailed structural analysis of protective devices for safety-related components necessary to achieve and maintain stable shutdown of the plant is performed due to the jet load impact. The analysis steps involved are as follows:

  Step 1: Thermal Hydraulic Analysis. A thermal hydraulic analysis of the pipe break is performed to calculate the mass flow rate and pipe reaction force time history through the break, along with the fluid conditions at the break. RELAP5 or TRACG is used for this analysis. The hydrodynamic model is a one-dimensional transient two-phase model with the capability for modeling non-condensable components in the steam phase and/or a soluble component in the water phase. The calculation scheme is based on the conservation of mass, momentum and energy among the control volumes and junctions for each phase, the state equations and constitutive relations (steam generation, wall heat transfer, etc.).

  The hydrodynamic model is based on the use of fluid control volumes and junctions to represent the spatial character of the flow. Velocities are located at the junctions and are associated with mass and energy flow between control volumes. The control system provides the capability to evaluate simultaneous algebraic and ordinary differential equations. The capability is primarily intended to simulate control systems typically used in hydrodynamic systems.

  A ruptured (circumferential break) pipe geometry is modeled as the control volumes and the required fluid parameters are provided as the input with the appropriate boundary conditions. A thermal hydraulic transient system analysis as a series of control volumes connected by junctions is carried out. RELAP5 or TRACG solves one-dimensional mass, momentum and energy equations for volumes assumed to contain homogeneous or non-homogeneous (as the case may be) fluid with the vapor and liquid phases in thermodynamic equilibrium.

  This analysis results include the mass flow rate time history through the break and the pipe reaction force time history among other desired output.

  Step 2: ANSYS Computational Fluid Dynamics Analysis. This program uses CFX, solver version 11.0 or the solver Fluent V6.3 included in ANSYS. Using the mass flow rate derived from the thermal hydraulic analysis and considering a worst case pipe displaced configuration (a pipe position that would cause maximum jet impact to the target structure) and defining the target location and its surface geometry in the computational fluid dynamics program as input, the computational fluid dynamics analysis provides results such as the time history of the force on the target. The computational fluid dynamics analysis captures the flow effects associated with the jet unsteadiness, nonlinearity, feedback amplification and jet reflections.

  The technical report in Appendix B of Reference 3.6-10 documents a methodology for evaluating blowdown forces created by jet impingement as a result of a high energy line break. The report also documents the benchmarking of the methodology against experimental data and

implements the methodology for an ESBWR Main Steamline break. The analysis methodology documented in this report will be used for the ESBWR to develop pipe break models during the detailed design phase that closely represent the geometry of the building volume and equipment being modeled.

Step 3: ANSYS Finite Element Analysis (FEA) Method. This program is used to model the target structure by FEA method. Using force time history as the input load resulting from the computational fluid dynamics analysis on the target structure, the transient dynamic analysis is performed. This dynamic time history analysis addresses the resonance (if any) with the input forcing function. To account for the uncertainty in the resonance frequencies of the target structure finite element model, input force time histories that are shifted in 2.5% increments spanning a ±10% uncertainty are applied to the structural FEA model, ensuring that the worst-case structural response is computed and used to assess structural integrity.

3.6.2.3.2 **Pipe Whip Effects on Safety-Related Components**

This subsection provides the criteria and methods used to evaluate the effects of pipe displacements on safety-related structures, systems, and components following a postulated pipe rupture.

Pipe whip (displacement) effects on safety-related structures, systems, and components can be placed in two categories: (1) pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run that the break occurs in; and (2) pipe whip or controlled displacements onto external components such as building structure, other piping systems, cable trays, conduits, etc.

Pipe Displacement Effects on Components in the Same Piping Run

The criteria for determining the effects of pipe displacements on inline components are as follows:

- Components such as vessel safe ends and valves which are attached to the broken piping system and do not serve a safety function or failure of which would not further escalate the consequences of the accident need not be designed to meet ASME B&PV Code Section III-imposed limits for safety-related components under faulted loading.
- If these components are required for safe shutdown or serve to protect the structural integrity of a safety-related component, limits to meet the ASME B&PV Code requirements for faulted conditions and limits to ensure required operability are met.

  The operability qualification of active pipe mounted components is described in Subsection 3.9.3.
- The methods used to calculate the pipe whip loads on piping components in the same run as the postulated break are described in Subsection 3.6.2.2 under paragraph titled “Pipe Whip Dynamic Response Analyses”.

Pipe Displacement Effects on Safety-Related Structures, Systems, and Components

The criteria and methods used to calculate the effects of pipe whip on external components consist of the following:

- The effects on safety-related structures and barriers are evaluated in accordance with the barrier design procedures given in Subsection 3.5.3.
- If the whipping pipe impacts a pipe of equal or greater nominal pipe diameter and equal or greater wall thickness, the whipping pipe does not rupture the impacted pipe. Otherwise, the impacted pipe is assumed to be ruptured.
- If the whipping pipe impacts other components (valve actuators, cable trays, conduits, etc.), it is assumed that the impacted component is unavailable to mitigate the consequences of the pipe break event.
- Damage of unrestrained whipping pipe on safety-related structures, components, and systems other than the ruptured one is prevented by either separating high-energy systems from the safety-related systems or providing pipe whip restraints.

3.6.2.3.3 **Loading Combinations and Design Criteria for Pipe Whip Restraint**

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry loads for an extremely low-probability gross failure in a piping system carrying high-energy fluid. Piping integrity does not depend on the pipe whip restraints for any piping design loading combination, including an earthquake, but the pipe whip restraints are required to remain functional following an earthquake up to and including the SSE (Subsection 3.2.1). When the piping integrity is lost because of a postulated break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance. The pipe whip restraints (i.e., those devices which serve only to control the movement of a ruptured pipe following gross failure) could be subjected to a once-in-a-lifetime loading. For the purpose of the pipe whip restraint design, the pipe break is considered to be a faulted condition (Subsection 3.9.3.1) and the structure to which the restraint is attached is analyzed and designed accordingly. The pipe whip restraints are non-ASME B&PV Code components; however, the ASME Code requirements may be used in the design selectively to assure its safety-related function if ever needed. Other methods (i.e., testing) with a reliable database for design and sizing of pipe whip restraints can also be used.

The pipe whip restraints utilize energy absorbing U-rods to attenuate the kinetic energy of a ruptured pipe. A typical pipe whip restraint is shown in Figure 3.6-2. The principal feature of these restraints is that they are installed with several inches of annular clearance between them and the process pipe. This allows for installation of normal piping insulation and for unrestricted pipe thermal movements during plant operation. Select critical locations inside the primary containment are also monitored during hot functional testing to provide verification of adequate clearances prior to plant operation. The specific design objectives for the restraints are:

- The restraints in no way increase the RCPB stresses by their presence during any normal mode of reactor operation or condition.
- The restraint system functions to stop the movement of a pipe failure (gross loss of piping integrity) without allowing damage to critical components or missile development.
- The restraints provide minimum hindrance to in-service inspection of the process piping.
- For the purpose of design, the pipe whip restraints are designed for the following dynamic loads:
- Blowdown thrust of the pipe section that impacts the restraint.
- Dynamic inertia loads of the moving pipe section, which is accelerated by the blowdown thrust and subsequent impact on the restraint.
- Design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in Subsection 3.6.2.2.
- Because the pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

Strain rate effects and other material property variations have been considered in the design of the pipe whip restraints. The material properties utilized in the design have included one or more of the following methods:

- Applicable code minimum or specification yield and ultimate strength values for the affected components and structures are used for both the dynamic and steady-state events.
- Not more than a 10% increase in minimum code or specification strength values is used when designing components or structures for the dynamic event, and code minimum or specification yield and ultimate strength values are used for the steady-state loads.
- Representative or actual test data values are used in the design of components and structures including justifiably elevated strain rate-affected stress limits in excess of 10%.
- Representative or actual test data are used for any affected component(s) and the minimum code or specification values are used for the structures for the dynamic and the steady-state events.

3.6.2.4 **Guard Pipe Assembly Design**

The ESBWR does not require guard pipes.

*3.6.2.5 **Pipe Break Analysis Results and Protection Methods***

The following information shall be provided in a pipe break evaluation report that will be completed in conjunction with closure of Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) Tier 1, Table 3.1-1 related to pipe break analysis report:

- *A summary of the dynamic analyses applicable to high-energy piping systems in accordance with Subsection 3.6.2.5 of RG 1.70. This shall include the following:*

  - *Sketches of applicable piping systems showing the location, size and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers.*
  - *A summary of the data developed to select postulated break locations including calculated stress intensities, cumulative usage factors and stress ranges as delineated in BTP 3-4.*
- *For failure in the moderate-energy piping systems, descriptions showing how safety-related systems are protected from the resulting jets, flooding and other adverse environmental effects.*
- *Identification of protective measures provided against the effects of postulated pipe failures for protection of each of the systems listed in Tables 3.6-1 and 3.6-2.*
- *The details of how the MSIV functional capability is protected against the effects of postulated pipe failures.*
- *Typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures include their enclosure in suitably designed structures or compartments (including any additional drainage system or equipment environmental qualification needs).*
- *The details of how the feedwater line check and feedwater isolation valves functional capabilities are protected against the effects of postulated pipe failures.*

3.6.2.6 **Analytic Methods to Define Blastwave Interaction to SSCs**

SSCs are evaluated for the blast wave effects. The blast wave occurs as a result of a pipe rupture that creates a rapid wave propagation of air surrounding the break due to the differential pressure between the rupture of a pressurized fluid in pipe and the ambient air. The blast effects are evaluated from all break types such as for the circumferential and longitudinal breaks for high and moderate energy piping systems. The wave propagation of the blast wave is dependent on the following conditions:

Blast Wave Due to a Pipe Rupture Occurring in an Open Space

The blast wave in an open space is considered as spherically expanding wave front. The blast wave pressure intensity is determined based on the pressure difference between pipe internal pressure prior to the pipe break and surrounding air at the break point, and the pressure attenuation occurs based on the radius cubed of the spherically expanding wave front.

Blast Wave Due to a Pipe Rupture Occurring in an Enclosed Space

Blast wave in an enclosed space experiences the propagation of shock wave and reflected wave effects. As the shock wave continues to propagate outward along the enclosed surface, a front known as the Mach front is formed by the interaction of the incident wave and the reflected wave. The reflected wave represents the incident wave that has been reinforced by the surrounding surface. Computational fluid dynamic analysis models analyze these phenomena and blast intensities farther from a pipe break location are determined.

Appendix A of Reference 3.6-10 provides a report that evaluates a blast wave induced by a high-energy line break at the feedwater nozzle inside containment. The blast wave propagates into the annular region between the RPV and the shield wall, and reflects between the boundaries of the annulus. This report establishes that a two-dimensional approximation of the annulus is conservative by comparing two-dimensional pressure amplitudes with those computed using a three-dimensional model. The report also establishes that the mesh discretization used is conservative by comparing pressures and velocities to those from a model generated with a coarser mesh. The methodology used in this report is representative of the methodology that will be used for breaks for which a blast wave calculation is performed.

3.6.3 (Deleted)

3.6.3.1 (Deleted)

3.6.3.2 (Deleted)

3.6.4 As-built Inspection of High-Energy Pipe Break Mitigation Features

An as-built inspection of the high-energy pipe break mitigation features is performed. The as-built inspection confirms that SSCs that are required to be functional during and following an SSE are protected against the dynamic effects associated with high-energy pipe breaks. An as-built inspection of pipe whip restraints, jet shields, structural barriers and physical separation distances is also performed.

For pipe whip restraints and jet shields, the location, the orientation, size and clearances to allow for thermal expansion are inspected. The locations of structures, identified as pipe break mitigation features, are inspected. Where physical separation is considered to be a pipe break mitigation feature, the assumed separation distances are confirmed during inspection.

3.6.5 COL Information

3.6.5-1-A (Deleted)

3.6.6 References

3.6-1 USNRC, “Modification of General Design Criterion 4, Requirements for Protection Against Dynamic Effects of Postulated Pipe Rupture,” Federal Register, Volume 52, No. 207, Rules and Regulations, Pages 41288 through 41295, October 27, 1987.

3.6-2 (Deleted)

3.6-3 (Deleted)

3.6-4 ANSI/ANS-58.2-1988 “Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture.”

3.6-5 (Deleted)

3.6-6 (Deleted)

3.6-7 (Deleted)

3.6-8 10 CFR 50 “Domestic licensing of production and utilization facilities.”

3.6-9 (Deleted)

3.6-10 GE Hitachi Nuclear Energy “ESBWR Safety Analysis - Additional Information,” NEDE-33440P, Revision 2, Class III (Proprietary), March 2010; NEDO-33440, Revision 2, Class I (Non-proprietary), March 2010.

Table 3.6-1 Safety-Related Systems, Components, and Equipment for Postulated Pipe Failures Inside Containment

1.	Reactor Coolant Pressure Boundary (up to and including the outboard isolation valves)
2.	Containment Isolation System and Containment Boundary (including liner plate)
3.	Reactor Protection System (scram signals)
4.	Control Rod Drive System (scram/rod insertion)
5.	Flow restrictors (passive)
6.	Passive Containment Cooling System
7.	Gravity-Driven Cooling System (including Fuel and Auxiliary Pools Cooling System interconnecting lines)
8.	Isolation Condenser System
9.	Standby Liquid Control System
10.	The following equipment/systems or portions thereof required to assure the proper operation of those safety-related items listed in items 1 through 9. (a) Safety-related electrical systems (b) Instrumentation (c) Process Sampling System

Table 3.6-2 Safety-Related Systems, Components, and Equipment for Postulated Pipe Failures Outside Containment

1.	Containment Isolation System and Containment Boundary
2.	Reactor Protection System (scram signals)
3.	Control Rod Drive System (scram/rod insertion)
4.	Flow restrictors
5.	Isolation Condenser System and Passive Containment Cooling System (Fuel and Auxiliary Pools Cooling System make-up lines included)
6.	Standby Liquid Control System
7.	The following equipment/systems or portions thereof required to assure the proper operation of those safety-related items listed in items 1 through 6, and GDCS function.
	(a) Safety-related Power Supply Systems (DC, Uninterruptible AC)
	(b) Instrumentation
	(c) Process Sampling System

Table 3.6-3 High and Moderate Energy Piping Inside Containment

High Energy Piping Inside Containment

1.	Nuclear Boiler System
2.	Control Rod Drive System (to and from HCU)
3.	Reactor Water Cleanup and Shutdown Cooling System (suction and RPV drain lines)
4.	Isolation Condenser System
5.	Gravity-Driven Cooling System Injection Lines (from RPV to isolation valves)
6.	Standby Liquid Control System Lines

Moderate Energy Piping Inside Containment

1.	Gravity Driven Cooling System
2.	Passive Containment Cooling System
3.	Fuel and Auxiliary Pools Cooling System
4.	Chilled Water System
5.	High Pressure Nitrogen Supply System
6.	Service Air System
7.	Equipment and Floor Drain System

Table 3.6-4 High and Moderate Energy Piping Outside Containment

High Energy Piping Outside Containment

1.	Reactor Water Cleanup and Shutdown Cooling System
2.	Nuclear Boiler System Lines in Steam Tunnel
3.	Control Rod Drive System (from CRD pumps to HCU and to FW lines and from HCU to containment penetrations)
4.	Standby Liquid Control Lines
5.	Isolation Condenser System Lines

Moderate Energy Piping Outside Containment

1.	Containment Inerting System
2.	Fuel and Auxiliary Pools Cooling System
3.	Chilled Water System
4.	Control Rod Drive System (pump suction line only)
5.	Makeup Water System
6.	Fire Protection System
7.	Service Air System
8.	High Pressure Nitrogen Supply System
9.	Instrument Air System
10.	Equipment and Floor Drain System
11.	Passive Containment Cooling System

Table 3.6-5 Terminal Pipe End Breaks at RPV Nozzles – High Energy Piping Systems

Terminal Pipe End Breaks for Systems	Location	System Condition	Jet Type	Analysis Method (Note 7)	Rupture Restraint Device Required (Note 6)
30" Main Steam Nozzle (Note 2)	RPV (Four nozzles)	Steam	Compressible, supersonic, turbulent, unsteady and expanding Quality: superheated steam	Jet by CFD Target by FEA	Note 3
12" FW Nozzle	RPV (Six nozzles) (Note 1)	Saturated Water	Compressible (mildly), expanding Quality: subcooled (some flashing can occur)	Enveloped by 12" RWCU analysis	Note 3
12" RWCU Nozzle	RPV (Two nozzles) (Note 1)	Saturated Water	Compressible (mildly), expanding Quality: subcooled (some flashing can occur)	Jet by CFD Target by FEA	Note 3
2" RWCU Drain Nozzle	RPV (Four nozzles located on bottom head of the RPV) (Note 1)	Saturated Water	Compressible, expanding Quality: subcooled (some flashing can occur)	Enveloped by 12" RWCU analysis	Note 4
18" IC Nozzle (Note 2)	RPV (Four nozzles) (Note 1)	Steam	Compressible, supersonic, turbulent, unsteady and expanding Quality: superheated steam	Jet by CFD Target by FEA	Note 3
8" IC Return Nozzle	RPV (Four nozzles) (Note 1)	Saturated Water	Compressible, expanding Quality: subcooled (some flashing can occur)	Enveloped by 12" RWCU analysis	Note 4
6" GDCS Nozzle (Note 2)	RPV (Eight nozzles) (Note 1)	Saturated Water	Compressible, expanding Quality: subcooled (some flashing can occur)	Enveloped by 12" RWCU analysis	Note 4
6" GDCS Equalizing Nozzle (Note 2)	RPV (Four nozzles) (Note 1)	Saturated Water	Compressible, expanding Quality: subcooled (some flashing can occur)	Enveloped by 12" RWCU analysis	Note 4
2" Stand-by Liquid Control Nozzle	RPV (Two nozzles) (Note 1)	Low Temp. Water	Compressible, expanding Quality: subcooled	Enveloped by 12" RWCU analysis	Note 4
2" RPV Level Inst. System (RVLIS) Piping (4 nozzles)	RPV (Four nozzles) (Note 1)	Steam	Compressible, supersonic, expanding Quality: superheated steam	Enveloped by 12" RWCU analysis	Note 4
2" Head Vent Nozzle	RPV (One nozzle) (Note 1)	Steam	Compressible, Supersonic, Expanding Quality: Super heated Steam	Enveloped by 12" RWCU analysis	Note 4
1-1/4" CRD Pipe at CRD Housing	269 Housings (On bottom shell of the RPV)	Low Temp. Water	Compressible, Non-expanding Quality: Subcooled	N/A	Note 5

Notes:

1. The terminal end location is within the Annulus formed by the RPV and Shield wall.
2. The nozzle has Venturi.
3. Rupture restraint device is required.
4. Rupture restraint function can be achieved by stiff pipe support structural hardware.
5. Rupture restraint device is not required.
6. The use of pipe restraints is subject to the final results of the high energy line break evaluations.
7. The analysis methods listed are used for forward flow cases from the reactor vessel and for reverse flow cases. CFD/FEA analyses include consideration of jet reflections.

Table 3.6-6 Terminal Pipe End Breaks Outside Containment High Energy Piping Systems (Sheet 1 of 2)

Terminal Pipe End Breaks for Systems	Pipe Break Locations	Building	System Condition	Jet Type	Analysis Method (Note 5)	Rupture Restraint Device Required (Note 4)
30” Main Steam Pipe	At header near Turbine Stop Valve	Turbine Building	Enveloped by 12” RWCU analysis Steam	Compressible, supersonic, expanding, turbulent, and unsteady Quality: superheated steam	Enveloped by 30” Main Steam Nozzle CFD analysis (Table 3.6-5)	Note 2
24” FW Pipe	At FW Heater nozzles Number of heaters = 6 (all in concrete wall enclosures)	Turbine Building	Saturated Water	Compressible, expanding Quality: subcooled	Jet by CFD Target by FEA	Note 2
6” & 8” RWCU Piping	At Regenerative Heat Exchanger (in a room)	Reactor Bldg.	Hot Water (for Regenerative Heat Exchanger inlet) Low Temp. Water (for Regenerative Heat Exchanger inlet)	Compressible, expanding (for Regenerative Heat Exchanger inlet), non-expanding for outlet (for Regenerative Heat Exchanger inlet) Quality: subcooled	Scale load based on 12” RWCU Nozzle CFD analysis (Table 3.6-5)	Note 2
12” RWCU Piping	At Non-Regenerative Heat Exchanger (in a room)	Reactor Bldg.	Low Temp. Water	Compressible, non-expanding Quality: subcooled	Results from 12” RWCU Nozzle CFD analysis (Table 3.6-5) are used	Note 2
8” and 12” RWCU Pump nozzles	RWCU pumps inlet (in a room)	Reactor Bldg.	Low Temp. Water	Compressible, non-expanding Quality: subcooled	Results from 12” RWCU Nozzle CFD analysis (Table 3.6-5) are used	Note 1
8” and 12” RWCU Pump	RWCU pumps outlet (in a room)	Reactor Bldg.	Low Temp. Water	Compressible, non-expanding Quality: subcooled	Results from 12” RWCU Nozzle CFD analysis (Table 3.6-5) are used	Note 1
6” RWCU piping	RWCU Demineralizer tank inlet & outlet	Reactor Bldg.	Low Temp. Water	Compressible, non-expanding Quality: subcooled	Scale load based on 12” RWCU Nozzle CFD analysis (Table 3.6-5) (Note 6)	Note 1

Table 3.6-6 Terminal Pipe End Breaks Outside Containment High Energy Piping Systems (Sheet 2 of 2)

Terminal Pipe End Breaks for Systems	Pipe Break Locations	Building	System Condition	Jet Type	Analysis Method (Note 5)	Rupture Restraint Device Required (Note 4)
8" IC Piping with ≈ 3"Dia. Venturi	At Inlet of Isolation Condenser in IC/PCCS Pool submerged in the water	Reactor Bldg.	Hot Water	Heat Exchanger nozzles submerged in the pool (jetting will not occur)	None required	Note 3
4" IC Piping	At Outlet of Isolation Condenser in IC/PCCS Pool submerged in the water	Reactor Bldg.	Hot Water	Heat Exchanger nozzles submerged in the pool (jetting will not occur)	None required	Note 3
3" Stand-by Liquid Control Piping	At SLC Tank Outlet (in a room)	Reactor Bldg.	Low Temp. Water	Compressible, non-expanding Quality: subcooled	Hand Calculation using DLF of 2.0 for loads (Note 6)	Note 1
1-1/4" CRD Piping (269 Lines)	At HCU (Hydraulic Control Units)	Reactor Bldg.	Low Temp. Water	Compressible, non-expanding Quality: subcooled	N/A (see Subsection 3.6.2.1.3)	Note 3

Notes:

1. This break is located in a separate room & has no other safety-related components. The pipe whip and jet interactions are limited within its system and components. The need for pipe rupture device is determined during the detailed design phase.
2. Rupture restraint device is required.
3. Rupture restraint device is not required.
4. The use of pipe restraints is subject to the final results of the high energy line break evaluations.
5. Unless otherwise indicated, the analysis methods listed are used for forward flow cases from the reactor vessel; reverse flow cases are not performed. CFD/FEA analyses include consideration of jet reflections.
6. The reverse flow case is also evaluated for these pipe end breaks, using the same analysis method listed for the forward flow case.

Table 3.6-7 Terminal End Breaks at Containment Penetrations (Inside and Outside the Drywell) (Sheet 1 of 2)

Penetration Number	Description	Pipe Dia, mm (in) (Note 6)	System Condition	Jet Type	Analysis Method (Note 8)	Rupture Restraint Device Required (Note 5 & Note 7)
B21-MPEN-0 001 through 4	Main Steam Line A through D	750 (30)	Steam	Same as in Tables 3.6-5 and 3.6-6.	Results from 30" Main Steam Nozzle (Table 3.6-5) are used; Target by FEA	Note 1
B21-MPEN-0 006 & 7	Feedwater Line A & B	550 (22)	Saturated Water	Same as in Tables 3.6-5 and 3.6-6	Jet by CFD Target by FEA	Note 1
B21-MPEN-0 005	Main Steam Drain Header	100 (4)	Steam/Hot Water (*)	Compressible, supersonic, turbulent, unsteady and expanding Quality: superheated steam	Hand Calculation using DLF of 2.0 for loads; Target by hand calculation	Note 1
B32-MPEN-0 001 through 4	IC Train A, B, C & D Steam Supply Line	350 (14)	Steam	Compressible, supersonic, turbulent, unsteady and expanding Quality: superheated steam	Results from 18" IC Nozzle (Table 3.6-5) are used; Target by FEA	Note 2
B32-MPEN-0 005 through 8	IC Train A, B, C & D Condensate Return	200 (8)	Hot water	Compressible, expanding Quality: subcooled (some flashing can occur)	Scale load based on 12" RWCU Nozzle CFD analysis (Table 3.6-5); Target by FEA	Note 3
C12-MPEN-0 001 through 12	FMCRD: Hydraulic Lines	32 (1.25)	Low Temp. Water	Compressible & non-expanding Quality: Sub-cooled	N/A (see Subsection 3.6.2.1.3)	Note 4
C41-MPEN-0 001 & 2	SLC (Train A & B)	80 (3)	Low Temp. Water	- Compressible, Expanding Quality: Sub-cooled (inside Cont.) - Compressible, non-Expanding Quality: Low temp water (outside)	Hand Calculation using DLF of 2.0 for loads; Target by hand calculation	Note 3
G31-MPEN-0 001 & 2	RWCU	300 (12)	Hot water	Compressible (mildly) Expanding Quality: Subcooled (Some flashing can occur)	Results from 12" RWCU Nozzle CFD analysis (Table 3.6-5) are used; Target by FEA	Note 1

Table 3.6-7 Terminal End Breaks at Containment Penetrations (Inside and Outside the Drywell) (Sheet 2 of 2)

Penetration Number	Description	Pipe Dia, mm (in) (Note 6)	System Condition	Jet Type	Analysis Method (Note 8)	Rupture Restraint Device Required (Note 5 & Note 7)
G31-MPEN-0003 & 4	RPV Bottom Drain Line	150 (6)	Hot Water	Compressible, Expanding Quality: sub-cooled (Some flashing can occur)	Scale load based on 12" RWCU Nozzle CFD analysis (Table 3.6-5); Target by FEA	Note 1

Notes:

1. Rupture restraint device is required on piping (inside and outside the penetration) near isolation valve.
2. Rupture restraint device is required inside the drywell side of the penetration only. This line penetrates the upper drywell through penetration into the IC/PCCS pool.
3. Rupture restraint function can be achieved by stiff pipe support structural hardware.
4. Rupture restraint device is not required.
5. See Figure 3.6-3 (Typical) for pipe break location.
6. Pipe diameter may be reduced at the containment penetration.
7. The use of pipe restraints is subject to the final results of the high energy pipe break evaluations.
8. The analysis methods listed are used for forward flow cases from the reactor vessel and for reverse flow cases. CFD/FEA analyses include consideration of jet reflections.

(*) – System is functional during plant startup only.

Legend:

B21 – System identification

MPEN-0001 – Mechanical Penetration 0001.

CFD – Computational fluid dynamics

FEA – Finite element analysis

DLF - Dynamic load factor

Figure 3.6-1 Jet Characteristics

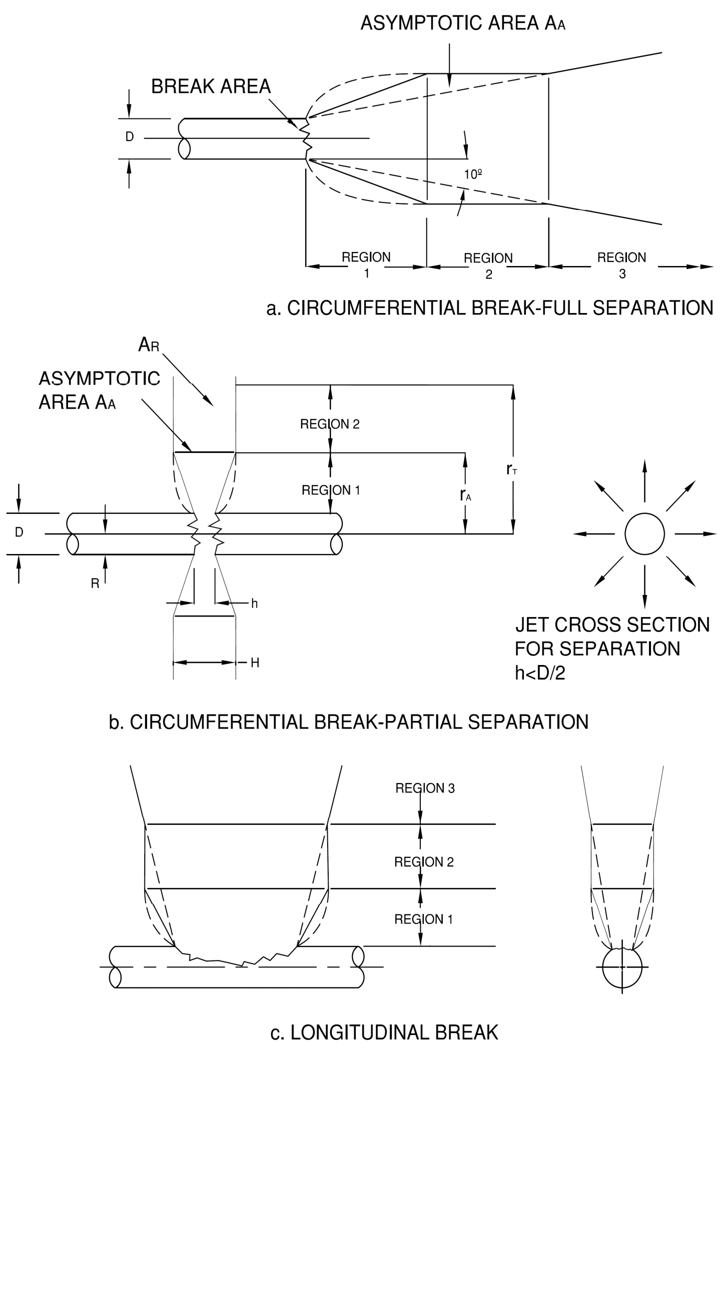


Figure 3.6-2 Typical Pipe Whip Restraint Configuration

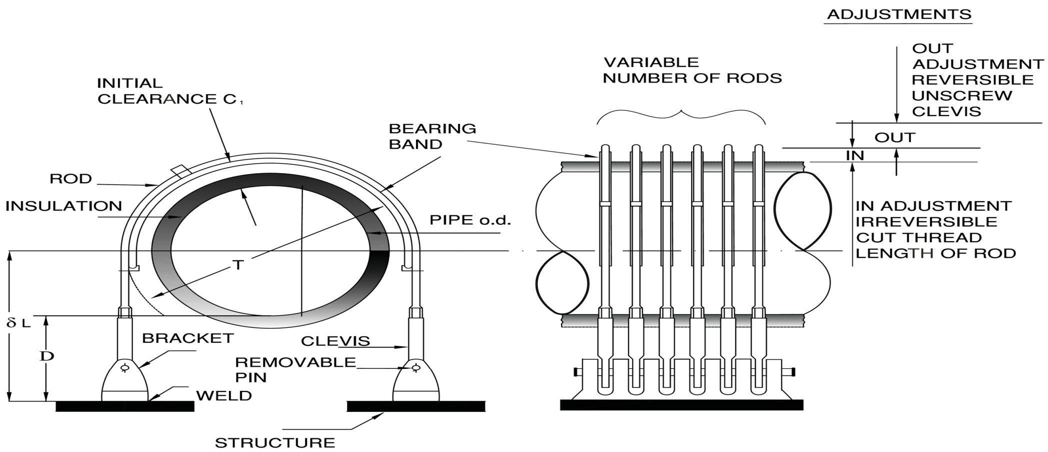
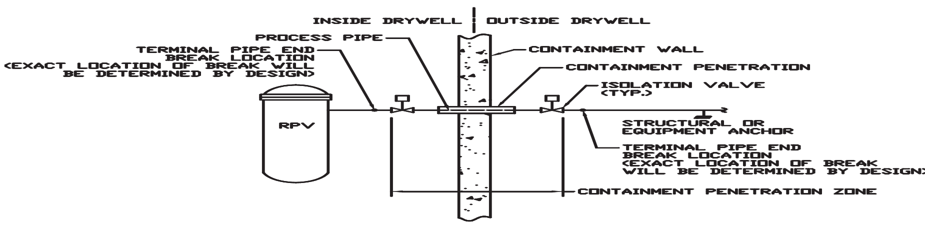


Figure 3.6-3 Typical Terminal End Break at Containment Penetration



3.7 Seismic Design

For seismic design purposes, all structures, systems, and components of the ESBWR standard plant are classified into Seismic Category I, Seismic Category II, or Seismic Category NS in accordance with the requirements to withstand the effects of the SSE as described in Section 3.2. For those Seismic Category I and Seismic Category II structures, systems and components (SSCs) in the Reactor Building (RB) complex, the effects of other dynamic loads caused by Reactor Building vibration (RBV) caused by suppression pool dynamics are also considered in the design. Although this section addresses seismic aspects of design and analysis in accordance with Regulatory Guide (RG) 1.70, the methods of this section are also applicable to RBV dynamic loadings, unless noted otherwise.

The SSE is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. It is the earthquake that produces the maximum vibratory ground motion for which Seismic Category I SSCs are designed to remain functional and within applicable stress, strain, and deformation limits. These systems and components are those necessary to ensure the following:

- The integrity of the reactor coolant pressure boundary.
- The capability to shut down the reactor and maintain it in a safe condition.
- The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable exposure limits set forth in 10 CFR 100 (10 CFR 52.47(a)(2)(iv)).

In response during an earthquake (up to SSE), the ESBWR will shut down and maintain safety using the Automatic Depressurization System and Gravity Driven Cooling System as described in the Probabilistic Risk Assessment. In this case, depressurization is accomplished in part with depressurization valves that remain open in order for the Gravity Driven Cooling System and the Passive Containment Cooling System to perform their safety functions.

Seismic Category II includes all plant SSCs which perform no safety-related function, and whose continued function is not required, but whose structural failure or interaction could degrade the functioning of a Seismic Category I structure, system or component to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room. Thus, this category includes the SSCs whose structural integrity, not their operational performance, is required. Seismic Category II SSCs that are also classified as RTNSS Criterion B in Tables 19A-2 and 19A-3 are required to remain functional following a seismic event. *The methods of seismic analysis and design acceptance criteria for Seismic Category II SSCs are the same as Seismic Category I; however, the procurement, fabrication and construction requirements for Seismic Category II SSCs are in accordance with industry practices. Seismic Category II items are those corresponding to position C.2 of RG 1.29.*

The operating basis earthquake (OBE) is a design requirement. For the ESBWR OBE, ground motion is chosen to be one-third of the SSE ground motion. Therefore, no explicit response or design analysis is required to show that OBE design requirements are met. This is consistent with Appendix S to 10 CFR 50. The effects of low-level earthquakes (lesser magnitude than the SSE) on fatigue evaluation and plant shutdown criteria are addressed in Subsections 3.7.3.2 and 3.7.4.4, respectively.

3.7.1 **Seismic Design Parameters**

As discussed in Standard Review Plan (SRP) 3.7.1, structures that are safety-related and that must withstand the effects of earthquakes are designed to the relevant requirements of GDC 2 and comply with Appendix S to 10 CFR 50 concerning natural phenomena. Standardized plants envelop the most severe earthquakes that affected a great number of sites where a nuclear plant may be located, with sufficient margin considering the limits of accuracy, quantity and period of time during which historical data have been accumulated. Seismic design parameters considered for ESBWR comprise two site conditions, generic sites and early site permit (ESP) sites. Two sites, Clinton (Reference 3.7-3) and Grand Gulf (Reference 3.7-4), have received ESPs. NRC is currently reviewing an ESP application from North Anna (Reference 3.7-2). A review of the three site conditions reveals that Clinton and Grand Gulf are bounded by the envelope of generic site and North Anna conditions. North Anna ESP site is therefore selected for further consideration in conjunction with generic sites for site enveloping seismic design of the ESBWR Standard Plant.

3.7.1.1 **Design Ground Motion**

The ESBWR standard plant SSE design ground motion is rich in both low and high frequencies. The low-frequency ground motion follows RG 1.60 ground spectra anchored to 0.3 g. The high-frequency ground motion matches the North Anna ESP site-specific spectra as representative of most severe rock sites in the Eastern US. These two ground motions are considered separately in the basic design. To verify the basic design the two separate inputs are further enveloped to form a single ground motion as the design basis ground motion for ESBWR. The single envelope design ground response spectra of 5% damping, also termed certified seismic design response spectra (CSDRS), are shown in Figures 2.0-1 and 2.0-2 for horizontal and vertical direction, respectively. They are defined as free-field outcrop spectra at the foundation level (bottom of the base slab) of the Reactor Building/Fuel Building (RB/FB) and Control Building (CB) structures. Application of design ground motion at the foundation level is a conservative approach for deeply embedded foundations as compared to the compatible free-field motion deconvoluted from the free ground surface motion at the finished grade. The ESBWR RB and CB foundations are embedded at depths of 20 m (66 ft.) and 14.9 m (49 ft.), respectively. The Fuel Building (FB) shares a common foundation mat with the RB. For the Firewater Service Complex (FWSC), which is essentially a surface founded structure, the CSDRS is 1.35 times the values shown in Figures 2.0-1 and 2.0-2

and is defined as free-field outcrop spectra at the foundation level (bottom of the base slab) of the FWSC structure. The ESBWR CSDRS are higher than RG 1.60 spectra anchored to 0.1 g peak ground acceleration (PGA) at the foundation level, meeting Appendix S to 10 CFR Part 50 regulations for 0.1 g minimum PGA for the horizontal component of the SSE at the foundation level in the free-field. The development of design ground motion is delineated in the following subsections.

Figure 3.7.1-228 and Figure 3.7.1-229 provide the Certified Seismic Design Response Spectra (CSDRS), which envelopes the site-specific design ground motions (FIRS) developed in Subsection 3.7.1.1.4 for the Reactor Building/Fuel Building (RB/FB) and Control Building (CB) and is used for design of the ESBWR RB/FB and CB. Figure 3.7.1-238 provides the Fire Water Service Complex (FWSC) CSDRS, which envelopes the site-specific design ground motions (FIRS) for the FWSC and is used for design of the FWSC.

For the Fermi 3 RB/FB and CB, site-specific soil-structure interaction (SSI) analyses were performed to address the following conditions:

- Partial embedment in the Bass Islands Group bedrock of the RB/FB and CB Seismic Category I structures, as shown on Figure 2.5.4-202 and Figure 2.5.4-203, to confirm that the design is applicable for this case.
- To demonstrate that the requirements for the backfill surrounding Seismic Category I structures above the top of bedrock can be neglected for RB/FB and CB with the RB/FB and CB partially embedded in the bedrock at the Fermi 3 site.

Figure 3.7.1-228, Figure 3.7.1-229, and Figure 3.7.1-238 show that the FIRS developed in Subsection 3.7.1.1.4 are enveloped by the CSDRS in both horizontal and vertical directions for the RB/FB, CB, and FWSC. Therefore, the Fermi 3 site-specific SSI analyses were not performed to address any exceedance of the CSDRS; rather, the Fermi 3 site-specific SSI analyses were performed to address the two Fermi 3 site-specific conditions outlined above for the RB/FB and CB.

The FWSC is a surface founded structure in Subsection 3.7.1.1, and there are no embedded walls for the FWSC. Therefore, the requirements for backfill surrounding Seismic Category I structures are not applicable to the FWSC embedded basemat. The FWSC is founded on fill concrete which meets the Table 2.0-1 requirements underneath Seismic Category I structures. Therefore, there is no site-specific SSI analysis performed for the FWSC.

The RB/FB and CB site-specific SSI analyses developed hazard-consistent seismic input for site response and SSI analyses consistent with Interim Staff Guidance DC/COL-ISG-017 and a Nuclear Energy Institute (NEI) developed white paper (Reference 3.7.1-201). The RB/FB and CB design ground motions for the SSI analyses (herein called the enhanced SCOR FIRS) were based on the outcrop FIRS developed in Subsection 3.7.1.1.4 for the full soil column (Soil Column Outcrop Response [SCOR]). Because site-specific SSI analyses are performed for Fermi 3, the site-specific

Safe Shutdown Earthquake (SSE) applicable for plant shut down purposes is the lower of the two enhanced SCOR FIRS for the RB/FB or CB. The SSE is defined at the foundation level to match the UFSAR.

The Operating Basis Earthquake (OBE) is one-third of the SSE. These definitions of the SSE and OBE are used in conjunction with the criteria specified in Subsection 3.7.4.4 to determine whether a plant shutdown is required following a seismic event.

3.7.1.1.1 **Low-Frequency Ground Motion**

The ground response spectra for low-frequency ground motion are developed in accordance with RG 1.60 anchored to 0.3 g and specified at the foundation level in the free field for generic sites. The 0.3 g SSE design response spectra for various damping ratios are shown in Figures 3.7-1 and 3.7-2 for the horizontal and vertical motions, respectively. The horizontal response spectra are equally applicable in two orthogonal horizontal directions.

Seismic input motions in the form of time histories are generated to envelop the design response spectra. The generic site 0.3 g SSE acceleration time histories for two horizontal components (H1 and H2) and vertical component are shown in Figures 3.7-3 through 3.7-5, respectively, together with corresponding velocity and displacement time histories. Each time history has a total duration of 22 seconds.

These time histories satisfy the spectrum-enveloping requirement stipulated in the NRC SRP 3.7.1. The computed response spectra for 2%, 3%, 4%, 5% and 7% damping are compared with the corresponding design RG 1.60 spectra in Figures 3.7-6 though 3.7-10 for the H1 component, in Figures 3.7-11 through 3.7-15 for the H2 component, and in Figures 3.7-16 through 3.7-20 for the vertical component. The response spectra are computed at frequency intervals suggested in Table 3.7.1-1 of SRP 3.7.1 plus three additional frequencies at 40, 50, and 100 Hz.

The time histories of the two horizontal components also satisfy the power spectral density (PSD) requirement stipulated in Appendix A to SRP 3.7.1. The computed PSD functions envelop the target PSD of a maximum 0.3 g acceleration with a wide margin in the frequency range of 0.3 Hz to 24 Hz as shown in Figures 3.7-21 and 3.7-22 for the H1 and H2 components, respectively. In these figures, the curve labeled as 80% of the target PSD is the minimum PSD requirement.

The target PSD compatible with the RG 1.60 vertical spectrum is not specified in Appendix A to SRP 3.7.1. Using the same methodology on which the minimum PSD requirement of Appendix A to SRP 3.7.1 for the RG 1.60 horizontal spectrum is based, the vertical target PSD compatible with the RG 1.60 vertical spectrum is derived using the following approach (Reference 3.7.1-215):

1. Establish initial candidate PSD.
2. Calculate several time histories using the PSD, each with a different phase function.
3. Calculate 2% critically damped pseudovelocity response spectrum (PSV) of each time history.

4. Compare the suite of PSVs from (3) to a target PSV.
5. If the average of the suite of PSVs does not fit (this is a visual fit) the target PSV, adjust form of PSD and go to Step (2).
6. Obtain the final PSD.

This vertical target PSD with the following input coefficients for 1.0 g PGA, is defined as $S_o(f)$ at frequency f:

$$S_0(f) = 2289\ cm^2/s^3(f/3.5)^{0.2} \quad f \le 3.5\ Hz$$
$$= 2289\ cm^2/s^3(3.5/f)^{1.6} \quad 3.5 < f \le 9.0\ Hz$$
$$= 505\ cm^2/s^3(9.0/f)^{3.0} \quad 9.0 < f \le 16.0\ Hz$$
$$= 89.9\ cm^2/s^3(16.0/f)^{7.0} \quad f > 16.0\ Hz$$

The PSD function for the vertical component of the design time history (SSE with 0.3 g PGA) is computed and subsequently averaged and smoothed using SRP 3.7.1 criteria. Similarly, the target PSD is computed for 0.3 g maximum acceleration. The PSD of the design time history is compared with the target and 80% of target PSD in Figure 3.7-23. As shown in this figure, PSD of the vertical time history envelops the target PSD by a wide margin. This comparison confirms the adequacy of energy content for the vertical time history.

The time histories of three spatial components are checked for statistical independency. The cross-correlation coefficient at zero time lag is 0.0135 between H1 and H2, 0.0704 between H1 and vertical, and 0.0737 between H2 and vertical. The cross-correlation coefficients are less than 0.16 as recommended in RG 1.92. Thus, H1, H2, and vertical acceleration time histories are mutually statistically independent.

The 0.3 g RG 1.60 input motion is considered in the basic design seismic analysis for generic uniform sites using the DAC3N computer code.

3.7.1.1.2 **High-Frequency Ground Motion**

The high-frequency ground motion is North Anna site-specific developed in the ESP application. The ESBWR foundation elevations at North Anna ESP site are EL. 205 ft. (62.484 m) for RB/FB and EL. 222 ft. (67.666 m) for CB. Since the low frequency parts of North Anna SSE ground spectra are enveloped by the 0.3 g RG 1.60 generic site spectra with large margins, only the high frequency

part is explicitly taken into account. The high frequency SSE ground spectra and compatible time histories at elevations of CB and RB/FB foundation level are shown in Figures 3.7-24 to 3.7-35.

Data	CB Base	RB/FB Base
Horizontal H1 target spectrum	Figure 3.7-24	Figure 3.7-30
Horizontal H1 time histories	Figure 3.7-25	Figure 3.7-31
Horizontal H2 target spectrum	Figure 3.7-26	Figure 3.7-32
Horizontal H2 time histories	Figure 3.7-27	Figure 3.7-33
Vertical target spectrum	Figure 3.7-28	Figure 3.7-34
Vertical time histories	Figure 3.7-29	Figure 3.7-35

The spectrum figures are associated with 5% damping. The PGA values, corresponding to the spectral acceleration at 100 Hz of the target spectra, are 0.492 g at the CB base and 0.469 g at the RB/FB base in both horizontal and vertical directions. The time histories are generated under the spectral matching criteria given in NUREG/CR-6728 and the cross-correlations between the three individual components are all less than the 0.16 requirement. Since a more stringent matching criteria of NUREG/CR-6728 is used, a separate PSD check per SRP 3.7.1.II.1 is not required.

The high-frequency input ground motion thus defined is considered in the basic design seismic analysis for North Anna ESP site condition using the DAC3N computer code.

3.7.1.1.3 **Single Envelope Ground Motion**

The single envelope ground response spectra are constructed to envelope the low-frequency 0.3 g RG 1.60 spectra (Subsection 3.7.1.1.1) and the high-frequency North Anna site-specific spectra (Subsection 3.7.1.1.2). The smoothed target spectra of 5% damping are shown in Table 3.7-2 and in Figures 2.0-1 and 2.0-2. The spectral values up to and including 9 Hz and 10 Hz in the horizontal and vertical directions, respectively, are based on 0.3 g RG 1.60 spectra. At higher frequencies, the spectral values closely match those of the envelope of North Anna ESP spectra at ESBWR RB/FB and CB foundations as a representative ground motion for Eastern US sites founded on rock. Note that there has never been recorded a seismic event containing simultaneously very high low-frequency excitations and very high high-frequency motions. Therefore, this envelope is very conservative in terms of energy content and is used to verify the basic design previously discussed.

A single set of three orthogonal, statistically independent time histories is generated to match the target spectra in accordance with NUREG/CR-6728 criteria. The computed response spectra are compared with the corresponding target spectra in Figures 3.7-38 through 3.7-40 for H1, H2 and vertical components, respectively. Spectral matching tests for 5% damping only is consistent with the recommendations of NUREG/CR-6728 (Reference 3.7.1-218) for specifying ground-motions in terms of 5% spectra. Use of 5% only is considered sufficient because there is a strong correlation among the response-spectral ordinates at damping ratios from 1 to 20%. Thus, if a time history

matches the 5% target, it is likely to match the targets at other damping ratios. Because the more stringent matching criterion of NUREG/CR-6728 is used, a separate PSD check per SRP 3.7.1.II.1 is not required. Tests referenced in NUREG/CR-6728 indicate that the response-spectrum tests are sufficient.

The acceleration time histories are shown in Figures 3.7-41 *through 3.7-43, together with corresponding velocity and displacement time histories. Each time history has a total duration of 40 seconds with time steps of 0.005 seconds. The strong motion duration is 7.8 seconds for H1, 12 seconds for H2 and 8.9 seconds for vertical. The cross-correlations between the three individual components are all less than the 0.16 requirement.*

The single envelope ground motion is considered in the design basis seismic analysis for all generic uniform sites using DAC3N and SASSI2000 computer codes and for layered sites using SASSI2000 computer code.

3.7.1.1.4 **Fermi 3 Site-Specific Ground Motions**

In the Fermi 3 site-specific SSI analyses (Section 3.7.2), the RB/FB and CB are modeled as partially embedded structures that penetrate into the Bass Islands Group bedrock. The elevation of the top of the Bass Islands Group bedrock is 168.1 m (551.7 ft) NAVD 88. The engineered granular backfill surrounding the RB/FB and CB above the Bass Islands Group bedrock is not included in the Fermi 3 site-specific licensing basis SSI analyses to demonstrate that the requirements for the backfill surrounding Seismic Category I structures above the top of bedrock are not required. To confirm that the engineered granular backfill does not adversely impact Seismic Category I structures, site-specific SSI analyses were also performed that included the engineered granular backfill above the top of the Bass Islands Group bedrock.

A consistent set of site-specific seismic inputs were developed to perform the Fermi 3 site-specific SSI analyses with and without engineered granular backfill above the top of the Bass Islands Group bedrock. In order to consider the potential influence of engineered granular backfill above the top of the Bass Islands Group bedrock in the SSI analyses, FIRS were developed for the Fermi 3 site as a SCOR at the RB/FB and CB foundation levels (herein called SCOR FIRS). The SCOR FIRS were enhanced using the procedure described in Section 5.2.1 of Interim Staff Guidance (ISG) DC/COL-ISG-017 and Section3.2.3 of the NEI developed white paper (Reference 3.7.1-201) to ensure hazard-consistent seismic inputs for the site-specific SSI analyses when compared to the Performance-Based Surface Response Spectra (PBSRS) at the finished ground level grade at Elevation 179.6 m (589.3 ft) NAVD 88. To ensure hazard-consistent seismic inputs for the site-specific licensing basis SSI analyses when compared to the Ground Motion Response Spectra (GMRS) at the top of the Bass Islands Group bedrock at Elevation 168.1 m (551.7 ft) NAVD 88, the general procedure described in Section 5.2.1 of the Interim Staff Guidance DC/COL-ISG-017 and Section 3.2.3 of the NEI developed white paper (Reference 3.7.1-201) was repeated without the engineered granular backfill. The SCOR FIRS are enhanced (herein called the enhanced SCOR

FIRS) to ensure hazard-consistent seismic inputs for the site-specific SSI analyses with and without the engineered granular backfill above the top of the Bass Islands Group bedrock when compared to the PBSRS and GMRS, and are used as seismic inputs for Fermi 3 site-specific SSI analyses for the RB/FB and CB.

Development of the enhanced SCOR FIRS and ground motion time histories in three directions (two horizontal and one vertical) compatible with the enhanced SCOR FIRS for the RB/FB and CB are discussed in the following subsections. Development of the FWSC FIRS and the deterministic profiles for the site-specific SSI analyses with and without engineered granular backfill above the top of the Bass Islands Group bedrock are also discussed in the following subsections. The deterministic profiles for the site-specific SSI analyses are the same for all deterministic profiles below the top of the Bass Islands Group bedrock.

3.7.1.1.4.1 **Full Soil Column and FWSC Ground Motions**

The process described in Section 3.2.3 of the NEI developed white paper (Reference 3.7.1-201) for development of a SCOR FIRS requires the development of the PBSRS at the finished ground level grade. The SCOR FIRS at the RB/FB and CB foundation levels are then computed as outcropping motions from the full soil column site response analysis. The method used to develop the site-specific PBSRS at the finished ground level grade for the Fermi 3 site is the same as that used in Subsection 2.5.2.5 to develop the GMRS with the exception that the soil column is extended to the finished ground level grade instead of being truncated at the top of the Bass Islands Group bedrock. The method in Subsection 2.5.2 employs Approach 2B outlined in NUREG/CR-6728 (Reference 3.7.1-202) to develop hazard-consistent surface spectra at the ground surface and foundation levels from the generic hard rock Uniform Hazard Response Spectra (UHRS) presented in Subsection 2.5.2.4. As described in Subsection 2.5.2.5, the following steps are involved in this approach:

1. Characterize the dynamic properties of the subsurface materials.
2. Randomize these properties to represent their uncertainty and variability across the site.
3. Based on the deaggregation of the rock hazard, define the distribution of magnitudes contributing to the controlling earthquakes for high-frequency (HF) and low-frequency (LF) ground motions and define the response spectra appropriate for each of the deaggregation earthquakes (DEs).
4. Match appropriate rock site time histories to the DE response spectra to be used as input at the base of the subsurface profiles.
5. Compute mean site amplification functions for the HF and LF controlling earthquakes based on the weighted average of the amplification function for the DEs.

6. Scale the response spectra for the controlling earthquakes (defined in the same manner as the reference earthquakes [REs]) and the rock UHRS by the mean amplification functions to obtain surface motions.
7. Envelop these scaled spectra to obtain surface motions at the finished ground level grade that are hazard consistent with the generic Central and Eastern United States (CEUS) hard rock hazard levels.

The Fermi 3 site-specific PBSRS at the finished ground level grade and the SCOR FIRS were developed by repeating the analysis steps presented above using the full soil column. Analysis steps 1 and 2 are described in Subsection 3.7.1.1.4.1.1. Analysis steps 3 and 4 are based on the rock hazard for the Fermi 3 site and are the same as those performed for the GMRS in Subsection 2.5.2.4 and Subsection 2.5.2.5. The input rock acceleration time histories for the site response analyses presented in Subsection 2.5.2.5 are used in the full soil column analysis without modification. Step 5 is described in Subsection 3.7.1.1.4.1.2. Steps 6 and 7 are described in Subsection 3.7.1.1.4.2.

In addition to the PBSRS and SCOR FIRS for the RB/FB and CB, FIRS are needed at the base of the FWSC. Subsection 3.7.1.1 states that the FWSC is essentially a surface founded structure. Therefore, the FIRS for the FWSC were developed as a Truncated Soil Column Response (TSCR) in accordance with Section 3.2.1 of the NEI developed white paper (Reference 3.7.1-201). The method used to develop the site-specific FWSC FIRS at the Fermi 3 site is the same as that used above to develop the PBSRS with the exception that the soil column is extended from the top of the Bass Islands Group bedrock to the bottom of the FWSC foundation using fill concrete with a mean compressive strength of 31 MPa (4,500 psi) (Subsection 2.5.4.5.4.2). The elevation of the bottom of the FWSC foundation basemat is 177.3 m (581.6 ft) NAVD 88, which is about 2.35 m (7.7 ft) below the finished ground level grade.

3.7.1.1.4.1.1 **Dynamic Properties for the Full Soil Column and FWSC Profile**

The PBSRS is defined at Elevation 179.6 m (589.3 ft) NAVD 88, the finished ground level grade for the Fermi 3 site. This elevation will be achieved by excavating and removing the existing overburden to the top of the Bass Islands Group bedrock at Elevation 168.1 m (551.7 ft) NAVD 88, and backfilling with engineered granular backfill to the finished ground level grade. This process results in an average engineered granular backfill thickness of approximately 11.5 m (37.6 ft). Beneath the FWSC, the existing overburden will also be excavated to the top of the Bass Islands Group bedrock, but will be backfilled with fill concrete instead of engineered granular backfill to the bottom of the FWSC foundation basemat. Additional fill concrete will also be placed between the embedded foundation walls of Seismic Category I structures and the adjacent bedrock. Subsection 2.5.2.5.1 discusses the development of the dynamic engineering properties for the in-situ bedrock material. The dynamic engineering properties for the in-situ bedrock material used in the site response analysis for computing the PBSRS at the finished ground level grade, the

SCOR FIRS for the RB/FB and CB, and the FWSC FIRS are the same as those in Subsection 2.5.2.5.1. Table 3.7.1-201 through Table 3.7.1-204 provide the dynamic engineering properties for the in-situ bedrock material below layer number 10.

Above the top of the Bass Islands Group bedrock, the shear wave velocity (V_S) for the engineered granular backfill is estimated in order to define lower-range (LR), intermediate-range (IR), and upper-range (UR) site response analysis profiles for the PBSRS and SCOR FIRS to represent the range in fill material properties that may be used. The values of V_S are based on empirical relationships for angular-grained material from Richart et al. (Reference 3.7.1-203) and for sandy and gravelly soils from Menq (Reference 3.7.1-204).

The empirical relationship for shear wave velocity from Richart et al. (Reference 3.7.1-203) for angular-grained material is:

$$V_S = [159 - (53.5)e](\overline{\sigma_0})^{0.25} \qquad \text{[Eq. 1]}$$

Where:

V_S is the shear wave velocity in ft/sec

e is the void ratio

$\overline{\sigma_0}$ is the average effective confining pressure in lb/ft^2 defined as

$$\overline{\sigma_0} = \frac{1}{3}(\sigma'_V + 2\sigma'_H) \qquad \text{[Eq. 2]}$$

Where:

σ'_V is the effective vertical stress in lb/ft^2

σ'_H is the effective horizontal stress in lb/ft^2 with $\sigma'_H = k_0\sigma'_V$

k_0 is the at-rest earth pressure coefficient defined as

$k_0 k_0 = \sin\phi' \sin\phi'$

ϕ' is the effective angle of internal friction of the soil

A range of material properties were considered in estimating two shear wave velocity profiles using the empirical relationship for angular-grained material from Richart et al. (Reference 3.7.1-203). Moist unit weights for granular soils up to 22.9 kN/m^3 (146 pcf) were considered, and a lower range value of 18.7 kN/m^3 (119 pcf) was selected to evaluate below the requirement of 20 kN/m^3 (125 pcf). The void ratio was estimated assuming an average condition of 50 percent saturation. The ϕ value was allowed to vary from the requirement of 35 degrees up to 50 degrees, the maximum value for dense coarse-grained material in Reference 3.7.1-205. The two shear wave velocity profiles used the following material properties to estimate an UR and LR of shear wave velocities:

- A void ratio of 0.18, a ϕ' of 50 degrees to calculate k_0, and a soil unit weight of 22.9 kN/m^3 (146 pcf) to calculate the average effective confining pressure for an UR estimate (a constant

effective lateral earth pressure of 500 psf is used for the UR estimate to a depth of 15.7 feet to account for compaction).

- A void ratio of 0.54, a ϕ' of 35 degrees to calculate k_0, and a soil unit weight of 18.7 kN/m^3 (119 pcf) to calculate the average effective confining pressure for a LR estimate.

The empirical relationship for small-strain shear modulus (G_{max}) from Menq (Reference 3.7.1-204) for sandy and gravelly soils is:

$$G_{max} = C_{G3} \times C_u^{b1} \times e^x \times \left(\frac{\overline{\sigma_0}}{P_a}\right)^{n_G} \qquad \text{[Eq. 3]}$$

Where:

C_{G3} is a coefficient equal to 67.1 MPa (1400 ksf)

C_U is the uniformity coefficient of the granular soil

$b1$ is an empirical coefficient equal to -0.20

e is the void ratio

x is a value dependent on the median grain size at 50 percent passing (D_{50}), in mm, defined as:

$$X = -1 - \left(\frac{D_{50}}{20}\right)^{0.75} \qquad \text{[Eq. 4]}$$

$\overline{\sigma_0}$ is the average effective confining pressure

P_a is atmospheric pressure

n_G is a value dependent on C_U defined as $n_G = 0.48 \times C_u^{0.09}$

The shear wave velocity is then estimated from G_{max} using the relationship (Reference 3.7.1-206):

$$V_S = \sqrt{\frac{G_{max}}{\rho}} \qquad \text{[Eq. 5]}$$

Where:

ρ is the soil density in units of mass per volume

Again, a range of values were used to estimate four different profiles of G_{max} and shear wave velocity for the engineered granular backfill using the empirical relationship from Menq (Reference 3.7.1-204). The values of C_U and D_{50} for two of the profiles were based on the MDOT 21A and 21AA gradations (Reference 3.7.1-207). The C_U values for the two other profiles (C_U = 3 and 200) were based on the larger range of values from reconstituted granular samples presented in Menq (Reference 3.7.1-204) and a range of void ratios from 0.18 to 0.54 were considered. The larger range of C_U values used a D_{50} of 8 mm (0.3 inches) based on the average of the MDOT

gradations. The four G_{max} and shear wave velocity profiles with the empirical relationship from Menq (Reference 3.7.1-204) used the following material properties:

- A C_U of 52.31, a D_{50} of 12.7 mm (0.5 inches), a unit weight of 22.9 kN/m^3 (146 pcf), and a void ratio of 0.3 (including effects of compaction).
- A C_U of 71.43, a D_{50} of 3.3 mm (0.13 inches), a unit weight of 22.9 kN/m^3 (146 pcf), and a void ratio of 0.26 (including effects of compaction).
- A C_U of 3, a D_{50} of 8 mm (0.3 inches), a unit weight of 18.7 kN/m^3 (119 pcf), and a void ratio of 0.54.
- A C_U of 200, a D_{50} of 8 mm (0.3 inches), a unit weight of 22.9 kN/m^3 (146 pcf), and a void ratio of 0.18 (including effects of compaction).

The LR profile represents the smallest shear wave velocity for each depth interval from the six shear wave velocity profiles described above for the empirical relationships of Richart et al. (Reference 3.7.1-203) and Menq (Reference 3.7.1-204). The UR profile represents the largest shear wave velocity for each depth interval from the six shear wave velocity profiles described above for the empirical relationships of Richart et al. (Reference 3.7.1-203) and Menq (Reference 3.7.1-204). The IR shear wave velocity profile represents the average of the LR and UR shear wave velocity profiles. Figure 3.7.1-201 shows the three estimated shear wave velocity profiles (LR, IR, and UR) for the engineered granular backfill used as input to the site response analysis for computing the PBSRS and SCOR FIRS. A range of values for the engineered granular backfill is used to assess the potential variability of the fill shear wave velocities in the full soil column site response analyses. The three shear wave velocity profiles for the engineered granular backfill are provided in layers 1 through 10 in Table 3.7.1-201, Table 3.7.1-202, and Table 3.7.1-203 for the LR, IR, and UR profiles, respectively. As stated in Subsection 2.5.2.5.1, a single velocity profile is appropriate for the in situ material at the Fermi 3 site; therefore, the velocity profile does not change below the engineered granular backfill. The groundwater table is assumed to be at the maximum historical groundwater elevation of 175.6 m (576.11 ft) NAVD 88 (Subsection 2.4.1.2) for estimating the shear wave velocities of the engineered granular backfill.

The FWSC is to be founded on fill concrete with shear dowels (i.e., steel reinforcing) extending to the top of the Bass Islands Group bedrock. The site response analysis profile for the FWSC FIRS was constructed by placing approximately 9.1 m (29.9 ft) of fill concrete between the bottom of the FWSC foundation basemat and the top of the Bass Islands Group bedrock. The estimated shear wave velocity of the fill concrete is 2,180 m/s (7,140 ft/s) based on an unconfined compressive strength of 31 MPa (4,500 psi) (see Subsection 2.5.4.5.4.2). Figure 3.7.1-202 shows the shear wave velocity profile used as input to the site response analysis for computing the FWSC FIRS. The shear wave velocity profile for the fill concrete beneath the FWSC is provided in layers 1 through 10 in Table 3.7.1-204. The dynamic engineering properties for the in-situ bedrock material used in the site response analysis are provided in Table 3.7.1-204 below layer number 10.

3.7.1.1.4.1.1.1 Density

Unit weights for the LR, IR, and UR site response analysis profiles are provided in Table 3.7.1-201, Table 3.7.1-202, and Table 3.7.1-203, respectively, for engineered granular backfill and bedrock. A range of values for the engineered granular backfill is used to assess the potential variability of density in the full soil column site response analyses.

Table 3.7.1-204 presents the unit weights for the fill concrete and the bedrock beneath the FWSC. The fill concrete beneath the FWSC basemat is assumed to have a unit weight of 22.8 kN/m^3 (145 pcf).

3.7.1.1.4.1.1.2 Shear Modulus Reduction and Damping

The upper 11.5 m (37.6 ft) of the Fermi 3 full soil column site response analysis profile consists of engineered granular backfill. The modulus reduction and damping relationships for the engineered granular backfill used in the site response analyses are based on the EPRI generic sand curves (Reference 3.7.1-208) and the relationship of Menq (Reference 3.7.1-204). The LR modulus reduction and damping relationship was developed using the methodology of Menq (Reference 3.7.1-204) and the gradation properties of MDOT 21A and 21AA aggregates (Reference 3.7.1-207), and represent the largest considered shear modulus reduction and damping for the engineered granular backfill. Specifically, the LR modulus reduction and damping curves represent a C_U of 71.43 and a D_{50} of 3.3 mm, which corresponds to the parameters of the shear wave velocity profiles that controlled most of the LR shear wave velocity profile. The LR modulus reduction and damping curves also used the LR unit weight of 18.7 kN/m^3 (119 pcf) to define effective confining stresses to represent the 0 to 6.1 m (20 ft) and 6.1 m to 11.5 m (20 to 37.6 ft) depth ranges. Modulus reduction and damping curves based on the relationship of Menq (Reference 3.7.1-204) were developed for these depth ranges to allow comparison to the EPRI generic sand curves for the 0 to 6.1 m (20 ft) and 6.1 m to 15.2 m (20 to 50 ft) depth ranges (Reference 3.7.1-208). The EPRI generic sand curves were used as the UR shear modulus reduction and damping curves, since they produced the least shear modulus reduction and damping. The IR shear modulus reduction and damping curves were developed by averaging the UR and LR curves.

The shear modulus reduction and damping relationships of Menq (Reference 3.7.1-204) are preferred over other recently published relationships since they incorporate the influence of gradation parameters based on tested samples of nonplastic sandy and gravelly soils that are anticipated to be similar to the engineered granular backfill. Similarly, the EPRI generic sand curves (Reference 3.7.1-208) were considered for the Fermi 3 site since they provide modulus reduction and damping relationships suitable for generic site response studies in Eastern North America, and are intended to represent soils in the general range of gravelly sand to low plasticity silty clays or sandy clays. The use of both the Menq (Reference 3.7.1-204) modulus reduction and damping

relationships and the EPRI generic sand curves (Reference 3.7.1-208) resulted in a range of modulus reduction and damping curves used to establish the LB and UB profiles.

Figure 3.7.1-203 presents the modulus reduction and damping relationships for the 0 to 6.1 m (20 ft) and 6.1 m to 11.5 m (20 to 37.6 ft) depth ranges. The damping ratio curves were limited to a maximum of 15 percent damping for site response analyses as recommended in Appendix E of Regulatory Guide (RG) 1.208. The modulus reduction and damping relationship assigned to the various layers of the engineered granular backfill in the LR, IR, and UR site response analysis profiles are listed in Table 3.7.1-201, Table 3.7.1-202, and Table 3.7.1-203, respectively.

The fill concrete with shear dowels (i.e., vertical steel reinforcement) between the bottom of the FWSC foundation basemat and the top of the Bass Islands Group bedrock is anticipated to have a damping of 4 percent based on the OBE damping value for reinforced concrete consistent with RG 1.61. However, a sensitivity analysis was completed using a lower damping of 0.5 percent due to the low effective dynamic stresses in the fill concrete and the relatively high shear wave velocities. This sensitivity analysis indicated a small difference of only 0.4 to 3 percent between the responses computed using damping values of 0.5 and 4 percent for the concrete fill; therefore, the lower damping value of 0.5 percent was used to develop the FWSC FIRS, as its use produced slightly higher response. The planned fill concrete with shear dowels is anticipated to remain essentially linear under the anticipated ground motion levels. Thus, the shear modulus reduction values for the fill concrete were set to 0.9999 for strain levels less than 3 percent and to 0.999 at higher strain levels.

Below the engineered granular backfill and fill concrete, the remaining portion of the full soil column and FWSC site response analysis profiles consists of dolomite and claystone bedrock, as discussed in Subsection 2.5.2.5.1.2. The bedrock is expected to remain essentially linear at low to moderate levels of shaking. Damping within the in-situ dolomite and claystone bedrock is characterized by a high-frequency attenuation parameter κ that ranges from 0.001 and 0.003 seconds (Subsection 2.5.2.5.1). The values of κ established in Subsection 2.5.2.5.1 were used to develop the site response analysis for the Fermi 3 site. As part of the development of the SSI inputs, the representation of damping in the in-situ bedrock was simplified from the seven different damping layers indicated in UFSAR Table 2.5.2-213 and Table 2.5.2-214 to the four different damping layers listed in UFSAR Table 3.7.1-201, Table 3.7.1-202, and Table 3.7.1-203. Sensitivity studies indicated that this simplification in the number of layers produces less than 0.1 percent difference in the mean amplification functions.

3.7.1.1.4.1.1.3. Randomization of Dynamic Properties

Site response analyses for the full soil column and FWSC profiles were conducted using randomized dynamic soil properties following the methods described in Subsection 2.5.2.5.1.3. The randomized dynamic properties included shear wave velocity, modulus reduction, and damping.

Additionally, the locations of velocity layer boundaries were randomized to vary uniformly within the range of layer thickness observed in the site borings.

Sixty randomized V_S profiles were generated for each of the LR, IR, and UR site response analysis profiles (a total of 180 randomized V_S profiles for development of the PBSRS and SCOR FIRS). Sixty randomized V_S profiles were also generated for the FWSC site response analysis profile. The statistics of the randomized profiles are summarized by comparing to the input target values for median velocity and standard deviation (sigma) of $\ln(V_S)$ for the LR, IR, UR, and FWSC profiles. As an example of this process, Figure 3.7.1-204 to Figure 3.7.1-206 show the 60 randomized velocity profiles and the statistics of the randomized shear wave velocity profiles for the IR site response analysis profile.

The modulus reduction and damping relationships associated with the LR, IR, and UR full column site response analysis profiles were also randomized as shown on Figure 3.7.1-207, Figure 3.7.1-208, and Figure 3.7.1-209, respectively. The standard deviation in the modulus reduction and damping were set so that the randomized relationships fell within recommended bounds provided by Silva (Reference 3.7.1-209). The damping ratio curves were limited to a maximum of 15 percent damping as recommended in Appendix E of RG 1.208.

The shear modulus reduction curve for the fill concrete described in Subsection 3.7.1.1.4.1.1.2 was also randomized using a standard deviation of 0.01 to maintain a value near unity. The randomized values of shear modulus reduction were greater than 0.9999 at all strain levels, which is consistent with shear wave velocities of at least 99.5 percent of the initial value and the interpretation that the fill concrete will behave as an essentially linear material.

The damping in the sedimentary bedrock beneath the engineered granular backfill was computed using the randomized sedimentary bedrock layer velocities and thicknesses, and the selected values of κ described in Subsection 2.5.2.5.1.

3.7.1.1.4.1.2. **Site Ampflication Functions**

A process similar to the description for developing the GMRS site amplification functions in Subsection 2.5.2.5.3 was repeated to produce mean site amplification functions for the PBSRS at the finished ground level grade, the SCOR FIRS for the RB/FB and CB foundation levels, and the FWSC FIRS.

Figure 3.7.1-210 shows the site response logic tree used to compute the controlling earthquake or RE mean amplification function and the weights assigned to the bedrock damping values and the subsurface profiles. This logic tree is similar to Figure 2.5.2-276; however, Figure 3.7.1-210 also includes the LR, IR, and UR site response analysis profiles to assess uncertainty in the dynamic properties of the engineered granular backfill. For each DE, mean amplification functions were computed using three bedrock damping values (κ) and the LR, IR, and UR profiles. The results obtained for the three DEs are then combined to produce a weighted mean amplification function for the RE. The weights assigned to the results for each DE are given in Table 2.5.2-212.

The mean amplification functions were then smoothed to remove small dips and peaks considered artifacts of the finite number of analyses. Linear interpolation in logarithmic space (log-log interpolation) was used to smooth the HF and LF amplification function above 1 Hz and below 7 Hz, respectively.

Figure 3.7.1-211 shows the mean PBSRS site amplification functions at the finished ground level grade for the $10^{-4}$and 10^{-5} exceedance levels of input ground motion. Both the unsmoothed and smoothed PBSRS site amplification functions are presented. Because of the non-linear behavior of the engineered granular backfill, the site amplification is sensitive to the level of input ground motion for most frequencies.

The SCOR site amplification functions at the RB/FB and CB foundation levels were obtained from the results of the site response analyses for the full soil column profile. Again, the mean amplification functions were smoothed to remove small features considered artifacts of the analyses. Figure 3.7.1-212 and Figure 3.7.1-213 show both the mean and smoothed SCOR site amplification functions for 10^{-4} and 10^{-5} exceedance levels of input ground motions at the RB/FB and CB foundation levels, respectively. The amplification functions for the RB/FB and CB SCOR FIRS show little sensitivity to the two different levels of motions because both foundations are founded in the same bedrock unit that has a relatively high and uniform shear wave velocity.

The site amplification functions at the FWSC foundation level used a logic tree similar to Figure 2.5.2-276. For each DE, mean amplification functions were computed using three bedrock damping values (κ). The results obtained for the three DEs are then combined to produce a weighted mean amplification function for the RE. The weights assigned to the results for each DE are given in Table 2.5.2-212. Mean amplification functions for the FWSC were determined for 10^{-4} and 10^{-5} exceedance levels of input ground motions at the FWSC foundation level. Again, the mean amplification functions were smoothed to remove small features considered artifacts of the analyses.

To incorporate the effect of the lateral contrast in dynamic properties between the fill concrete beneath the FWSC basemat and the adjacent engineered granular backfill, two dimensional (2D) site response analyses were completed using an equivalent linear, 2D finite element program for modeling the seismic response of soil masses. The program QUAD4MU (Reference 3.7.1-210), an updated version of QUAD4M (Reference 3.7.1-211) and the original program QUAD4 (Reference 3.7.1-212), was used to complete the 1D and 2D site response analyses. The 2D analyses were compared to equivalent one-dimensional (1D) analyses that assumed a uniform layer of concrete above the top of the Bass Islands Group bedrock to construct 2D/1D spectral ratios. Figure 3.7.1-214 presents the mean 2D/1D response spectral ratio for the 10^{-4} and 10^{-5} exceedance levels of input ground motions for both FWSC foundation dimensions (20 m [66 ft] by 52 m [171 ft]) and the IR engineered granular backfill properties. Figure 3.7.1-214 also presents smoothed envelopes used in the initial development of the 2D-to-1D spectral ratio envelopes. The smoothed envelope was then modified as needed to include the effects of the IR, LR, and UR

engineered granular backfill properties. Figure 3.7.1-215 presents the final 2D/1D response spectral ratio envelopes for the 10^{-4} and 10^{-5} exceedance levels of input ground motions. These 2D/1D response spectral ratio envelopes indicate that the limited extent of the fill concrete beneath the FWSC produces an increase in the mean site amplification functions above 5 Hz compared to that obtained from 1D site response. The increase is generally greater for the 10^{-5} exceedance level of input ground motion than for the10^{-4} exceedance level of input ground motions. The 10^{-4} and 10^{-5} exceedance levels of input ground motions produce 2D/1D response spectral ratio envelopes with an increase of up to 45 percent and 90 percent, respectively. Figure 3.7.1-216 shows the smoothed mean site amplification functions for 10^{-4} and 10^{-5} exceedance levels of input ground motions at the FWSC foundation level for the 1D site response compared with those incorporating the 2D effects due to the limited extent of the fill concrete.

3.7.1.1.4.2 **Surface Hazard Spectra for PBSRS, SCOR FIRS, and FWSC FIRS**

The surface UHRS at the finished ground level grade and at the RB/FB, CB, and FWSC foundation levels were constructed following the procedures described in Subsection 2.5.2.6 for the GMRS at the top of the Bass Islands Group bedrock. The appropriate site amplification functions were used to scale the generic hard rock UHRS and the LF and HF RE spectra to obtain the surface UHRS at the finished ground level grade and for the RB/FB, CB, and FWSC foundation levels. For the generic hard rock UHRS, the HF amplification function is used for frequencies above 5 Hz and the LF amplification function is used for frequencies below 2.5 Hz. Frequencies between 2.5 and 5 Hz use a weighted combination of the HF and LF amplifications.

Figure 3.7.1-217 shows the surface spectra for the scaled LF and HF RE, the scaled hard rock UHRS, and the envelop spectrum for the 10^{-4} exceedance level ground motions at the finished ground level grade using the mean PBSRS site amplification functions. The amplification functions and the corresponding surface spectrum show a slight dip in the frequency range of 5 to 20 Hz. The dip was conservatively removed in constructing the enveloped surface UHRS for the 10^{-4} exceedance level ground motions. As a result, the final spectra will be conservative in the frequency range of approximately 5 to 20 Hz.

Figure 3.7.1-218 shows the surface spectra for the scaled LF and HF RE, the scaled hard rock UHRS, and the envelop spectrum for the 10^{-4} exceedance level ground motions at the RB/FB foundation level using the mean SCOR site amplification functions. The SCOR UHRS at the RB/FB foundation level was developed using the same process described for the surface UHRS at the finished ground level grade, including smoothing through the dip in the spectrum between approximately 4 and 25 Hz.

Similar operations were performed to develop the UHRS for the 10^{-4} exceedance level ground motions at the CB and FWSC foundation levels and the 10^{-5} exceedance level motions at the finished ground level grade surface and the RB/FB, CB, and FWSC foundation levels. Since the amplification functions for the FWSC FIRS included the 2D effect of the fill concrete, the UHRS at

the FWSC foundation level includes the 2D effect of the limited extent of the fill concrete discussed in Subsection 3.7.1.1.4.1.2.

3.7.1.1.4.3 PBSRS at the Finished Ground Level Grade

3.7.1.1.4.3.1 Horizontal PBSRS

Development of the horizontal PBSRS at the finished ground level grade follows the same processes for development of the GMRS provided in RG 1.208 and Subsection 2.5.2.6. Figure 3.7.1-219 shows the 10^{-4} and 10^{-5} horizontal UHRS and the resulting horizontal PBSRS at the finished ground level grade. As described in Subsection 2.5.2.6.2.1, RG 1.208 specifies two approaches for calculating performance based ground motion response spectra, a design factor (DF) times the 10^{-4} UHRS and the minimum value of 0.45 times the 10^{-5} UHRS when the ratio of the 10^{-5} UHRS to the 10^{-4} UHRS exceeds 4.2. Both results are shown on Figure 3.7.1-219. The final PBSRS is the envelope of the two. Table 3.7.1-205 presents the resulting horizontal PBSRS values.

3.7.1.1.4.3.2 Vertical PBSRS

The vertical GMRS developed in Subsection 2.5.2.6 used vertical to horizontal (V/H) spectral ratios recommended by NUREG/CR-6728 (Reference 3.7.1-202) for CEUS bedrock sites. The vertical PBSRS at the finished ground level grade was also constructed using V/H spectral ratios; however, the full soil column profile consists of a thin layer of soil over bedrock. This profile is somewhat different than the generic rock conditions for which the V/H ratios shown on Figure 2.5.2-285 were developed. At present, there are no published V/H ratios for ground motions in the CEUS for the conditions represented by the full soil column profile, a profile with a thin soil layer over bedrock. Therefore, the V/H ratios for the vertical PBSRS were developed by examining differences between bedrock and shallow soil site V/H ratios for Western US (WUS) data and using the differences to adjust the CEUS hard rock V/H values.

The WUS V/H ratios recommended in NUREG/CR-6728 (Reference 3.7.1-202) were based on ground motion relationships for a generic bedrock site classification. More recently, Campbell and Bozorgnia (Reference 3.7.1-213) developed empirical ground motion prediction equations for bedrock sites that contained explicit categorization for firm bedrock (V_{S30} 830 m/s ± 339 m/s [2,720 ft/s ± 1,110 ft/s]) and soft rock (V_{S30} 421 m/s ± 109 m/s [1,380 ft/s ± 358 ft/s]) sites, where V_{S30} is the average shear wave velocity in the upper 30 m (100 ft). The soft bedrock V/H ratios are used to indicate the potential behavior of a shallow stiff soil site. The results obtained using Campbell and Bozorgnia (Reference 3.7.1-213) suggest that the peak in the V/H ratios for soft bedrock shifts slightly towards lower frequencies compared to the peak for firm bedrock sites. The V/H ratios are also lower on soft bedrock for frequencies less than about 3 Hz.

The Pacific Earthquake Engineering Research (PEER) Center's Next Generation Attenuation (NGA) Project (Reference 3.7.1-214) developed an extensive database of strong motion records

from active tectonic environments. The records from Reference 3.7.1-214 were analyzed by Gülerce and Abrahamson (Reference 3.7.1-215) to develop a model for V/H ratios based on V_{S30} values. In order to compare the model of Gülerce and Abrahamson (Reference 3.7.1-215) to the site categories of Campbell and Bozorgnia (Reference 3.7.1-213), V/H ratios were computed using the Gülerce and Abrahamson (Reference 3.7.1-215) model for V_{S30} values of 830 m/s (2,720 ft/s) and 421 m/s (1,380 ft/s). These V_{S30} values corresponded to the firm rock and soft rock categories of Campbell and Bozorgnia (Reference 3.7.1-213). The result suggests a trend similar to the Campbell and Bozorgnia (Reference 3.7.1-213) result.

Figure 3.7.1-220 shows V/H spectral ratios as a function of frequency used for generating the vertical PBSRS at the finished ground level grade, and the V/H spectral ratios recommended by NUREG/CR- 6728 (Reference 3.7.1-202) for CEUS bedrock sites with a PGA between 0.2 g and 0.5 g. The V/H spectral ratios used for generating the vertical PBSRS are based on the V/H spectral ratios recommended by NUREG/CR-6728 (Reference 3.7.1-202) for CEUS bedrock sites with a shift in the frequencies above 10 Hz to represent the shift in the peak V/H spectral ratios towards lower frequencies in the Campbell and Bozorgnia (Reference 3.7.1-213) and Gülerce and Abrahamson (Reference 3.7.1-215) comparisons. Additionally, at frequencies below 9 Hz, the V/H spectral ratio is reduced slightly to reflect the differences observed in the Campbell and Bozorgnia (Reference 3.7.1-213) and Gülerce and Abrahamson (Reference 3.7.1-215) comparisons. The resulting vertical PBSRS is listed in Table 3.7.1-205 along with the values of V/H. Figure 3.7.1-221 shows the horizontal and vertical PBSRS (5 percent damping) at the finished ground level grade.

3.7.1.1.4.3.3 **Deterministic Profiles for SSI Analyses**

Three deterministic profiles, the best estimate (BE), lower bound (LB), and upper bound (UB), were developed from the full soil column site response analysis following the requirements of Standard Review Plan (SRP) 3.7.2 and guidance from the Interim Staff Guidance DC/COL-ISG-017. These profiles were based on the statistics of the iterated soil properties for the randomized full soil column profile described in Subsection 3.7.1.1.4.1.1.3, and include the engineered granular backfill above the top of the Bass Islands Group bedrock. The 60 randomized full soil column profiles were developed for each of the nine alternative sets of dynamic properties (three alternative sets of engineered granular backfill properties and three alternative sets of damping values in the in-situ bedrock). These 540 profiles were then used in site response analyses with the time histories matched to the 10^{-4} and 10^{-5} HF and LF DEL, DEM, and DEH response spectra. The results of these calculations produced a total of 1,620 profiles of strain-compatible dynamic properties for each of the 10^{-4} and 10^{-5} HF and LF exceedance levels of input motion. Each strain-compatible profile was assigned a weight equal to the product of the weights on the corresponding branches of the site response logic tree shown in Figure 3.7.1-210 times 1/60 for each randomization case. The resulting values of shear wave velocity and damping in each soil layer were then ranked in increasing order and the empirical 16^{th}, 50^{th}, and 84^{th} percentile values were identified for the four loading levels.

The deterministic BE profile with engineered granular backfill above the top of the Bass Islands Group bedrock was set equal to values interpolated between the median iterated soil properties for the 10^{-4} and 10^{-5} exceedance level ground motions using linear interpolation based on the PGA values for the 10^{-4} and 10^{-5} UHRS and the PGA for the PBSRS. The 50^{th} percentile properties for the HF and LF input motions were averaged to produce the BE profile. The resulting subsurface layers and the corresponding strain compatible dynamic engineering properties for the full soil column BE profile are listed in Table 3.7.1-206.

The deterministic LB profile with engineered granular backfill above the top of the Bass Islands Group bedrock was set equal to the 16^{th} percentile of the distribution of randomized soil properties interpolated between the 10^{-4} and 10^{-5} exceedance level ground motions, and the deterministic UB profile with engineered granular backfill above the top of the Bass Islands Group bedrock was set equal to the 84^{th} percentile of the distribution of randomized soil properties interpolated between the 10^{-4} and 10^{-5} exceedance level ground motions. To maximize the range of values, the minimum values from the LF and HF ground motions were used for the 16^{th} percentile and the maximum of the LF and HF ground motions were used for the 84^{th} percentile. The range in the UB and LB shear wave velocities was increased where necessary to maintain the minimum variation from the shear modulus for the deterministic BE profile (G_{BE}) required in SRP 3.7.2. The minimum variation is defined by a multiplicative factor of 1 plus the minimum coefficient of variation (COV) in shear modulus such that G_{UB} is greater than or equal to the G_{BE} x ($1+COV_{min}$) and G_{LB} is less than or equal to G_{BE} / ($1+COV_{min}$). SRP 3.7.2 specifies that the minimum COV for well studied sites is 0.5 and for sites less well investigated the minimum COV should be at least 1.0. The in-situ subsurface materials have been well investigated at the Fermi 3 site and a COV of 0.5 was used to establish the minimum variation in G between the LB, BE, and UB profiles in these materials. However, properties of the engineered granular backfill are estimates based on a range of possible characteristics. Therefore, a minimum COV of 1.0 was used in establishing the minimum variation in G between the LB, BE, and UB profiles in the engineered granular backfill. Table 3.7.1-207 and Table 3.7.1-208 list the resulting subsurface layers and the corresponding strain compatible dynamic engineering properties for the LB and UB deterministic profiles, respectively, with engineered granular backfill above the top of the Bass Islands Group bedrock.

Figure 3.7.1-222 shows the full soil column LB, BE, and UB subsurface shear wave velocity profiles with engineered granular backfill above the top of the Bass Islands Group bedrock for the Fermi 3 site. The corresponding damping ratios were obtained from the statistics of the iterated profiles assuming negative correlation between shear wave velocity (V_S) and damping: that is, the 16^{th} percentile damping for the full soil column UB profile and the 84^{th} percentile damping for the full soil column LB profile. The compression wave velocities were based on the shear wave velocities in the LB, BE, and UB shear wave velocity profiles with engineered granular backfill above the top of the Bass Islands Group bedrock, the recommend Poisson's ratios in Table 2.5.4-202, and the relationship from Kramer (Reference 3.7.1-206) presented as follows:

$$\frac{V_P}{V_S} = \sqrt{\frac{2-2v}{1-2v}} \qquad \text{[Eq. 6]}$$

Where:

V_P is the compression wave velocity

V_S is the shear wave velocity

v is the Poisson's ratio

The bedrock and portions of the engineered granular backfill are below the groundwater table at the Fermi 3 site. The compression wave velocities in the bedrock exceeded the 1,460 m/s (4,790 ft/sec) the compression wave velocity of water from the UFSAR; therefore, a minimum compression wave velocity of 1,460 m/s (4,790 ft/sec) for the bedrock below the groundwater table was not imposed. In the engineered granular backfill, the compression wave velocities were less than the minimum value of 1,460 m/s (4,790 ft/sec) below the anticipated groundwater table. However, the compression wave velocities were not increased to the minimum value of 1,460 m/s (4,790 ft/sec) in the engineered granular backfill. Instead, the compression wave velocities were increased to the lower value of either 1,460 m/s (4,790 ft/sec) or the compression wave velocity that resulted in a maximum Poisson's ratio of 0.48 for the corresponding LB, BE, and UB shear wave velocity.

Figure 3.7.1-223 shows the LB, BE, and UB subsurface shear wave velocity profiles without engineered granular backfill above the top of the Bass Islands Group bedrock for the Fermi 3 site near the RB/FB and CB. Table 3.7.1-209, Table 3.7.1-210, and Table 3.7.1-211 present the BE, LB, and UB deterministic profiles without the engineered granular backfill above the top of the Bass Islands Group bedrock. The deterministic profiles without the engineered granular backfill above the top of the Bass Islands Group bedrock are the same as the deterministic profiles for the full soil column below the top of the Bass Islands Group bedrock.

3.7.1.1.4.4 SCOR FIRS for the RB/FB and CB

The process described in Subsection 3.7.1.1.4.3 was used to develop the SCOR FIRS at the RB/FB and CB foundation levels. The SCOR FIRS are shown on Figure 3.7.1-224 and Figure 3.7.1-225 for the RB/FB and CB, respectively. The spectral accelerations for the RB/FB and CB SCOR FIRS are provided in Table 3.7.1-212 and Table 3.7.1-213, respectively. Also shown on Figure 3.7.1-224 and Figure 3.7.1-225 are the ESBWR CSDRS. The SCOR FIRS for the RB/FB and CB are enveloped by the ESBWR CSDRS.

Since the RB/FB and CB foundation levels are within the bedrock units, the vertical SCOR FIRS were generated using the V/H spectral ratios for hard rock recommended by NUREG/CR-6728 (Reference 3.7.1-202) for CEUS bedrock sites. The recommended CEUS hard rock V/H spectral ratios for 0.2 g ≤ PGA ≤ 0.5 g are shown on Figure 3.7.1-220. Because the vertical PBSRS was based on modified V/H spectral ratios for a PGA between 0.2 g and 0.5 g, use of the rock V/H spectral ratios for this PGA range to develop the vertical SCOR FIRS maintains consistent vertical

to horizontal spectral ratios between the PBSRS and SCOR FIRS. Table 3.7.1-212 and Table 3.7.1-213 provide the V/H ratios used to develop the vertical SCOR FIRS for the RB/FB and CB.

The SCOR FIRS were compared to appropriate site-independent spectral shapes scaled to the minimum PGA of 0.1g specified in 10 CFR Part 50, Appendix S. The RB/FB and CB are to be founded on relatively hard rock. The median rock spectral shape defined in NUREG/CR-0098 (Reference 3.7.1-216) has been used in NUREG-1407 (Reference 3.7.1-217) to specify ground motions for safety evaluations of CEUS nuclear power plants. Figure 3.7.1-226 and Figure 3.7.1-227 show that the SCOR FIRS without enhancement for the RB/FB and CB, respectively, do not envelop the median rock site spectral shape from NUREG/CR-0098 (Reference 3.7.1-216) scaled to 0.1g PGA between about 0.8 and 2 Hz. Alternatively, as discussed in NUREG/CR-6926 (Reference 3.7.1-218), NUREG/CR-6728 (Reference 3.7.1-202) developed appropriate spectral shapes for ground motions on CEUS rock sites. The CEUS rock site spectral relationships presented in NUREG/CR-6728 (Reference 3.7.1-202) were used to develop rock spectral shapes for the DEs presented in Table 3.7.1-212 and to construct the single enveloping spectral shape presented on Figure 3.7.1-226 and Figure 3.7.1-227. Figure 3.7.1-226 and Figure 3.7.1-227 show that the SCOR FIRS without enhancement for the RB/FB and CB, respectively, envelop the enveloping CEUS rock site spectral shapes scaled to the minimum PGA of 0.1g in NUREG/CR-6728 (Reference 3.7.1-202). The comparisons shown on Figure 3.7.1-226 and Figure 3.7.1-227 show that a small enhancement of the SCOR FIRS is needed in the frequency range of 0.8 to 2 Hz in order to envelop NUREG/CR-0098 (Reference 3.7.1-216) broad band spectral shape scaled to the minimum PGA of 0.1g.

Figure 3.7.1-226 and Figure 3.7.1-227 also compare the SCOR FIRS without enhancement to the horizontal spectral shape from RG 1.60 scaled to the minimum PGA of 0.1g. The RG 1.60 spectral shape is based on statistical analyses that included recordings on soil sites and, therefore, produces higher ground motion levels at intermediate to low frequencies. The SCOR FIRS without enhancement do not envelop the RG 1.60 shape scaled to a PGA of 0.1g in the frequency range of about 0.2 to 3 Hz. To conservatively meet the requirements of the minimum ground motions specified in 10 CFR Part 50, Appendix S, the SCOR FIRS were initially enhanced in the frequency range of about 0.2 to 3 Hz to envelop the RG 1.60 spectrum scaled to a PGA of 0.1g. The resulting initially enhanced SCOR FIRS also envelop the median rock spectral shape defined in NUREG/CR-0098 (Reference 3.7.1-216). Figure 3.7.1-226 and Figure 3.7.1-227 present the initially enhanced SCOR FIRS for the RB/FB and CB, respectively, the RG-1.60 spectrum scaled to the minimum PGA of 0.1g, and the ESBWR CSDRS. The horizontal initially enhanced SCOR FIRS for the RB/FB and CB envelop or equal the appropriate site-independent spectral shapes scaled to the minimum PGA of 0.1g, meet the requirements specified in Appendix S of 10 CFR Part 50 and SRP 3.7.1, and are enveloped by the horizontal ESBWR CSDRS.

Interim Staff Guidance DC/COL-ISG-017 and the NEI developed white paper (Reference 3.7.1-201) state that time histories matched to the outcrop FIRS should be convolved from the foundation level

up to the finished ground level grade using the full soil column LB, BE, and UB subsurface profiles, and that the resulting envelope of the three surface spectra from the time histories should envelop the PBSRS at the finished ground level grade. This comparison was made by matching the seed time histories using the methods discussed in Subsection 3.7.1.1.5 to the initially enhanced SCOR FIRS. The matched time histories compatible with the initially enhanced SCOR FIRS were then input at the appropriate foundation level into the three deterministic profiles (LB, BE, and UB) for the full soil column with engineered granular backfill above the top of the Bass Islands Group bedrock (Table 3.7.1-206, Table 3.7.1-207, and Table 3.7.1-208), and convolved to the PBSRS level at the finished ground level grade with SHAKE analyses (Reference 3.7.1-222). Comparison of the resulting envelope of the three surface spectra from the horizontal time histories and the horizontal PBSRS showed that the resulting envelope did not envelop the horizontal PBSRS at frequencies below 0.2 Hz. Comparison of the resulting envelope of the three surface spectra from the vertical time histories and the vertical PBSRS showed that the envelope did not envelop the vertical PBSRS at frequencies below 0.2 Hz and at frequencies between about 1.5 Hz and 5 Hz.

Interim Staff Guidance DC/COL-ISG-017 and the NEI developed white paper (Reference 3.7.1-201) do not describe the development of hazard consistent seismic input for site response and SSI analyses for partially embedded structures. However, since the time histories matched to the SCOR FIRS will be used in the SSI analyses without engineered granular backfill above the top of the Bass Islands Group bedrock, the time histories were also convolved from the foundation level up to the top of the Bass Islands Group bedrock using the deterministic profiles without engineered granular backfill above the top of the Bass Islands Group bedrock in Table 3.7.1-209, Table 3.7.1-210, and Table 3.7.1-211. The matched time histories compatible with the initially enhanced SCOR FIRS were input at the appropriate foundation level into the three deterministic profiles (LB, BE, and UB) without engineered granular backfill above the top of the Bass Islands Group bedrock and convolved to the GMRS level at the top of the Bass Islands Group bedrock with SHAKE analyses (Reference 3.7.1-222). Comparison of the resulting envelope of the three surface spectra from the horizontal and vertical time histories and the horizontal and vertical GMRS showed that the resulting envelope fell below the GMRS at frequencies above 4 Hz. Comparison with the GMRS is considered appropriate since the GMRS represents a surface response spectra at the top of the deterministic profiles without engineered granular backfill above the top of the Bass Islands Group bedrock. The result of comparison of the surface spectra with the GMRS is similar to the comparison results for the PBSRS using the deterministic profiles with engineered granular backfill above the top of the Bass Islands Group bedrock.

The initially enhanced SCOR FIRS were then enhanced further by increasing the overall level of ground motion in the frequency ranges identified during the comparison of the resulting envelope of the three surface spectra from the time histories to the PBSRS and the GMRS. Figure 3.7.1-228 and Figure 3.7.1-229 show the horizontal SCOR FIRS without any enhancement and the horizontal enhanced SCOR FIRS for the RB/FB and CB, respectively. Also shown on Figure 3.7.1-228 and

Figure 3.7.1-229 are the horizontal ESBWR CSDRS. The enhanced horizontal SCOR FIRS for the RB/FB and CB are enveloped by the horizontal ESBWR CSDRS.

Time histories matched to the enhanced SCOR FIRS were then convolved from the foundation level up to the finished ground level grade using the full soil column LB, BE, and UB deterministic profiles for comparison to the PBSRS at the finished ground level grade. Figure 3.7.1-230 and Figure 3.7.1-231 show the comparison of the PBSRS at the finished ground level grade with the envelope of the surface response spectra obtained from SHAKE analyses (Reference 3.7.1-222) using the LB, BE, and UB full soil column deterministic profiles with engineered granular backfill above the top of the Bass Islands Group bedrock and the matched time histories compatible with the RB/FB and CB enhanced SCOR FIRS, respectively. The envelope of the three response spectra at the ground surface exceeds the PBSRS at the finished ground level grade for each component of motion, satisfying Interim Staff Guidance DC/COL-ISG-017 and the NEI developed white paper (Reference 3.7.1-201). The time histories matched to the enhanced SCOR FIRS were also convolved from the foundation level up to the top of the Bass Islands Group bedrock using the LB, BE, and UB deterministic profiles without the engineered granular backfill above the top of the Bass Islands Group bedrock for comparison to the GMRS. Figure 3.7.1-232 and Figure 3.7.1-233 show the comparison of the GMRS with the envelope of the surface response spectra obtained from SHAKE analyses (Reference 3.7.1-222) using the deterministic profiles without the engineered granular backfill above the top of the Bass Islands Group bedrock and the matched time histories compatible with the RB/FB and CB enhanced SCOR FIRS, respectively. The envelope of the three response spectra at the top of the Bass Islands Group bedrock exceeds the GMRS for each component of motion. The horizontal RB/FB and CB enhanced SCOR FIRS values are provided in Table 3.7.1-214 and Table 3.7.1-215, respectively.

The vertical SCOR FIRS was also enhanced in the identified frequency ranges. Figure 3.7.1-228 and Figure 3.7.1-229 also show the vertical SCOR FIRS and the vertical enhanced SCOR FIRS for the RB/FB and CB, respectively. Also shown on Figure 3.7.1-228 and Figure 3.7.1-229 are the vertical ESBWR CSDRS. The vertical enhanced SCOR FIRS for the RB/FB and CB are enveloped by the vertical ESBWR CSDRS. Vertical component time histories matched to the vertical enhanced SCOR FIRS were convolved from the foundation level up to the finished ground level grade using the deterministic profiles with engineered granular backfill above the top of the Bass Islands Group bedrock for comparison to the PBSRS at the finished ground level grade. Figure 3.7.1-234 and Figure 3.7.1-235 show the comparison of the vertical PBSRS at the finished ground level grade with the envelope of the surface response spectra obtained from SHAKE analyses (Reference 3.7.1-222) using the deterministic profiles with engineered granular backfill above the top of the Bass Islands Group bedrock and the matched time histories compatible with the vertical RB/FB and CB enhanced SCOR FIRS, respectively. The envelope of the three response spectra at the ground surface exceeds the vertical PBSRS at the finished ground level grade for each component of motion, satisfying Interim Staff Guidance DC/COL-ISG-017 and the NEI developed

white paper (Reference 3.7.1-201). The vertical component time histories matched to the vertical enhanced SCOR FIRS were also convolved from the foundation level up to the top of the Bass Islands Group bedrock using the deterministic profiles without engineered granular backfill above the top of the Bass Islands Group bedrock for comparison to the vertical GMRS. Figure 3.7.1-236 and Figure 3.7.1-237 show the comparison of the vertical GMRS with the envelope of the surface response spectra obtained from SHAKE analyses (Reference 3.7.1-222) using the deterministic profiles without engineered granular backfill above the top of the Bass Islands Group bedrock and the matched time histories compatible with the vertical RB/FB and CB enhanced SCOR FIRS, respectively. The envelope of the three response spectra at the top of the Bass Islands Group bedrock exceeds the GMRS for each component of motion. The vertical RB/FB and CB enhanced SCOR FIRS values are provided in Table 3.7.1-214 and Table 3.7.1-215, respectively.

Table 3.7.1-214 and Table 3.7.1-215 provide the PGA values – listed as the 100 Hz values – for the horizontal RB/FB and CB enhanced SCOR FIRS. The PGA values for the horizontal RB/FB and CB enhanced SCOR FIRS are higher than the minimum 0.1g requirement of SRP 3.7.1.

3.7.1.1.4.5 **FWSC FIRS**

The process described in Subsection 3.7.1.1.4.4 was used to develop the FIRS at the FWSC foundation level. The horizontal FWSC FIRS is shown on Figure 3.7.1-238. The spectral accelerations for the horizontal FWSC FIRS are provided in Table 3.7.1-216. The FWSC FIRS includes the 2D effect of the fill concrete by incorporating the 2D/1D response spectral ratio envelopes discussed in Subsection 3.7.1.1.4.1.2 in the amplification functions for the FWSC. Also shown on Figure 3.7.1-238 is the curve for 1.35 times the ESBWR CSDRS, which is the appropriate comparison for the FWSC FIRS, as described in Subsection 3.7.1.1. The FWSC FIRS is enveloped by 1.35 times the ESBWR CSDRS.

Since the FWSC foundation level is on top of fill concrete and bedrock units with relatively high shear wave velocities, the vertical FWSC FIRS was generated using the V/H spectral ratios for hard rock recommended by NUREG/CR-6728 (Reference 3.7.1-202) for CEUS bedrock sites. The recommended CEUS hard rock V/H spectral ratios for 0.2 g ≤ PGA ≤ 0.5 g are shown on Figure 3.7.1-220. The vertical FWSC FIRS is shown on Figure 3.7.1-238. The spectral accelerations and V/H ratios for the vertical FWSC FIRS are provided in Table 3.7.1-216.

3.7.1.1.5 **Site-Specific Design Ground Motion Time Histories**

Sets of three orthogonal time histories (two horizontal and one vertical component) were generated to match the horizontal and vertical enhanced SCOR FIRS (Subsection 3.7.1.1.4.4) for the RB/FB and CB, respectively, in accordance with the criteria of NUREG/CR-6728 (Reference 3.7.1-202). The selected seed time histories are those of the 1999 Chi-Chi Taiwan Earthquake, TAP078 recording, chosen from the CEUS record library provided in NUREG/CR-6728 (Reference 3.7.1-202). These time histories represent a distant recording of a large magnitude

(moment magnitude 7.6) earthquake, consistent with the large contribution of the New Madrid source to the hazard at the Fermi 3 site. Details of this record are provided in Table 3.7.1-217.

A single set of time histories (two horizontal and one vertical component) was developed for both the RB/FB and CB foundation levels following the requirements of Option 1, Approach 2 of SRP 3.7.1 Section II (Acceptance Criteria), Revision 3. Per paragraph 2(d) of Approach 2, in lieu of the power spectrum density (PSD) requirement, the requirement that the computed 5 percent damped response spectrum of the time history does not exceed the target response spectrum at any frequency by more than 30 percent was met at frequencies between 0.1 and 50 Hz. A few frequencies above 50 Hz do exceed the target spectrum by more than 30 percent; however, a check of the PSD for frequencies above 50 Hz that exceed the target spectrum by more than 30 percent is not required for CEUS sites by Appendix B of SRP 3.7.1. Table 3.7.1-218 presents the cross correlation coefficients between each combination of time history components (two horizontal and one vertical). The cross correlation coefficients all fall below the criteria of 0.16 in SRP 3.7.1 Section II (Acceptance Criteria), Revision 3.

Spectral matching was performed using the time-domain spectral matching procedure proposed by Lilhanand and Tseng (Reference 3.7.1-219) and later modified by Abrahamson (Reference 3.7.1-220). Figure 3.7.1-239 through Figure 3.7.1-244 show the comparison of the response spectrum in the two horizontal and one vertical direction for the following:

- The enhanced SCOR FIRS at the RB/FB and CB levels
- 1.3 times (30 percent greater) the enhanced SCOR FIRS
- 0.9 times (10 percent less) the enhanced SCOR FIRS
- Response spectrum for the spectrally matched time history

The response spectra for the spectrally-matched time histories were calculated for comparison with the enhanced SCOR FIRS at 301 spectral frequency points (or 100 frequencies per spectral frequency decade). As shown in Figure 3.7.1-239 through Figure 3.7.1-244, the 5 percent damped response spectra of the spectrally-matched time histories are within the range of 0.9 to 1.3 times the enhanced SCOR FIRS at frequencies between 0.1 and 50 Hz. Therefore, the criteria of Option 1, Approach 2 of SRP 3.7.1 Section II (Acceptance Criteria), Revision 3, are satisfied.

The time step and duration of the matched time histories are 0.005 seconds and 80 seconds, respectively. The duration of the time histories for Arias Intensity to rise from 5 to 75 percent is greater than the minimum 6 second duration identified in SRP 3.7.1, Section II (Acceptance Criteria), Revision 3, and are consistent with the characteristic earthquake duration of NUREG/CR-6728 (Reference 3.7.1-202). Details of the matched time histories including the PGA, peak ground velocity (PGV), and peak ground displacement (PGD) are presented in Table 3.7.1-219. Figure 3.7.1-245 to Figure 3.7.1-250 present the matched time histories (outcropping motions) compatible with the RB/FB and CB enhanced SCOR FIRS at the foundation levels. The duration and the value of PGV/PGA (Table 3.7.1-219) are generally consistent with the

characteristic values reported in NUREG/CR-6728 (Reference 3.7.1-202); however, the values of PGA*PGD/PGV^2 are larger. The hard rock UHRS for the Fermi 3 site represents a combination of hazard from large, distant earthquakes and smaller, closer earthquakes. Thus, it is expected that the PGA is enriched to represent smaller magnitude, closer earthquakes. Spectral matching of the time histories to response spectra extended to a period of 10 seconds also enriches the PGD values, leading to an increase in the values of PGA*PGD/PGV^2.

To demonstrate that there are no significant gaps in power for the spectrally-matched time histories, power spectral densities (PSD) were calculated for the frequency range of 0.3 to 50 Hz following the guidance in SRP 3.7.1, Appendix B. The equivalent stationary duration, T_D, used to calculate the PSD for the spectrally-matched time histories was established in general accordance with SRP 3.7.1 and the additional guidance in NUREG/CR-5347 (Reference 3.7.1-221). Figure 3.7.1-251 presents the normalized Arias intensities for the horizontal components, H1 and H2, of the spectrally matched time histories compatible with the RB/FB enhanced SCOR FIRS. The normalized Arias intensity plots for the spectrally matched time histories compatible with the CB enhanced SCOR FIRS are not presented since they are essentially identical to Figure 3.7.1-251.

The PSD was evaluated using the following two approaches for computing the Fourier amplitudes:

- Using the full duration of the spectrally matched time histories.
- Using only the portion of the spectrally matched time histories corresponding to the equivalent stationary duration.

Appendix B of NUREG/CR-5347 (Reference 3.7.1-221) indicates that T_D is estimated by identifying the portion of the time history where the slope (power) of a cumulative energy plot (represented by normalized Arias intensity) is nearly constant and near maximum. Figure 3.7.1-251 provides a range of estimates of constant power slopes for the two horizontal RB/FB components.

The Fermi 3 FIRS represent the combined effects of two distinct earthquakes, a nearby moderate magnitude earthquake and a distant large earthquake (New Madrid). The seed time history for spectral matching was selected to represent the long duration expected in a distant recording of a large magnitude earthquake. As illustrated by the spectrally matched acceleration and velocity time histories on Figure 3.7.1-245 and Figure 3.7.1-246, the time histories exhibit non-stationarity that results in high frequency energy more prominent in the early portion of the records and low frequency energy more prominent in the latter portion of the records. Because of the long duration and non-stationarity of the recording, longer values of T_D are needed to better represent the energy content of the recordings. Therefore, the time for the Arias intensities to rise from 0 to 100 percent is used to establish T_D instead of the more commonly used time to rise from 5 to 75 percent Arias intensity. This time was established by extending the constant power slopes to 0 percent and 100 percent Arias intensity, as shown on Figure 3.7.1-251, and the time between the intersection with the 0 and 100 percent Arias intensity levels was used to establish a value of T_D.

Use of the full duration of the spectrally matched time history records to compute the Fourier amplitudes for the PSD captures the full frequency content of the records, which is consistent with the SSI analyses that also use the full duration time history records. Values of T_D of 30 seconds for the H1 component and 31.5 seconds for the H2 component were selected from the range of estimated values for the PSD calculation using the full duration of the spectrally matched time histories. The resulting PSDs for the horizontal spectrally matched time histories are shown on Figure 3.7.1-252, Figure 3.7.1-253, Figure 3.7.1-254, and Figure 3.7.1-255 for the RB/FB H1 component, the RB/FB H2 component, the CB H1 component, and CB H2 component, respectively. As demonstrated in these figures, with the full duration considered, the spectrally matched time histories have no significant gaps in power over the frequency range of 0.3 to 50 Hz.

Appendix B of SRP 3.7.1 indicates the PSD is calculated using the portion of the spectrally matched time history corresponding to T_D. The effect of using only the T_D portion of the time histories to compute the PSD is illustrated on Figure 3.7.1-252 through Figure 3.7.1-255. On each figure, the different PSD are calculated using the portion of the spectrally matched time history windowed to the different T_D values shown on Figure 3.7.1-251. Outside of the T_D window, a two second duration cosine taper was applied to reduce the time history amplitude to zero.

For the RB/FB H1 component shown on Figure 3.7.1-252, the PSD for the windowed time histories are similar to the PSD computed using the full duration time history. There is a decrease in amplitude at frequencies above 30 Hz and below 1 Hz. The observed decreases reflect the fact that some of the energy content at these frequencies occurs outside of the selected T_D time window. For the shortest T_D of 30 seconds, a narrow dip in power occurs in the low frequency range near 0.4 Hz. However, the PSD for the windowed time histories are generally similar to the PSD using the full duration time history and show no significant gaps in power.

Figure 3.7.1-253 shows the results for the RB/FB H2 component. The PSD for the windowed time histories are also similar to the PSD computed using the full duration time history. There is a decrease in amplitude at frequencies above 25 Hz, and between about 0.7 and 1 Hz, again reflecting that some of the energy content at these frequencies occurs outside of the selected T_D time window. As was the case for the H1 component, the PSD for the windowed time histories are generally similar to the PSD using the full duration time history and show no significant gaps in power.

Figure 3.7.1-254 and Figure 3.7.1-255 show the corresponding comparisons for the CB H1 and CB H2 components, respectively. The results are similar to those shown for the corresponding RB/FB components.

In summary, the PSDs computed using the full duration, spectrally matched time histories show that there are no significant gaps in power over the frequency range of 0.3 to 50 Hz. PSDs computed using the windowed portion of the spectrally matched time histories corresponding to T_D also show no significant gaps in power. There is a narrow dip in power near 0.4 Hz using the window

corresponding to a T_D of 30 seconds for the H1 component for both the RB/FB and CB. However, extending TD to 32, 34, or 36 seconds eliminates this narrow dip. The PSD computed using the windowed time histories show a decrease in power compared to PSD computed using the full duration time histories above 25 Hz. This difference indicates a degree of non-stationarity in the time histories, but does not produce significant gaps in the frequency range of 0.3 to 50 Hz.

In accordance with Interim Staff Guidance DC/COL-ISG-017 and the NEI developed white paper (Reference 3.7.1-201), the spectrally-matched time histories compatible with the RB/FB and CB enhanced SCOR FIRS were then input as outcropping motions at the foundation level into the LB, BE, and UB deterministic profiles without engineered granular backfill above the top of the Bass Islands Group bedrock to compute the resulting in-column motions at the RB/FB and CB foundation levels using the program SHAKE (Reference 3.7.1-222). A total of 18 SHAKE analyses were performed using combinations of the LB, BE, and UB deterministic profiles without engineered granular backfill above the top of the Bass Islands Group bedrock, the three time history components (two horizontal and one vertical components) and the two foundation levels (RB/FB and CB). The SHAKE analyses were performed using the LB, BE, and UB deterministic profiles in Table 3.7.1-209, Table 3.7.1-210, and Table 3.7.1-211 without iteration of soil properties to generate in-column motions at the foundation levels for input into the Fermi 3 site specific SSI analysis without engineered granular backfill above the top of the Bass Islands Group bedrock.

In-column motions at the foundation levels were also generated for the LB, BE, and UB deterministic profiles with engineered granular backfill above the top of the Bass Islands Group bedrock in Table 3.7.1-206, Table 3.7.1-207, and Table 3.7.1-208. The SHAKE analyses were performed using the spectrally-matched time histories compatible with the RB/FB and CB enhanced SCOR FIRS and without iteration of soil properties to generate 18 additional in-column motions at the foundation levels for the Fermi 3 site specific SSI analysis with engineered granular backfill above the top of the Bass Islands Group bedrock.

To evaluate the energy present at different frequencies in the 36 in-column acceleration time histories, power spectra were computed for each of the time histories. The cumulative power was then calculated from 0 to 100 Hz to determine what percentage of power is below 50 Hz in the in-column acceleration time histories. As an example, Figure 3.7.1-256 presents the power spectrum and cumulative power plots for the horizontal (H1 and H2) in-column acceleration time history compatible with the BE deterministic profile without engineered granular backfill above the top of the Bass Islands Group bedrock. Table 3.7.1-220 presents the percentage of the cumulative power below 50 Hz for each in-column acceleration time history. The horizontal components include between 92 and 100 percent of the total power at frequencies below 50 Hz. The vertical components include between 88 and 95 percent of the total power at frequencies below 50 Hz.

3.7.1.2 **Percentage of Critical Damping Values**

Damping values of various structures and components are shown in Table 3.7-1 for use in SSE dynamic analysis. These damping values are consistent with RG 1.61 Revision 1 SSE damping.

For ASME Section III, Division 1 Class 1, 2, and 3, and ASME B31.1 piping systems, the damping values of Table 3.7-1 or alternative damping values specified in Figure 3.7-37 are used. The damping values shown in Table 3.7-1 are applicable to all modes of a structure or component constructed of the same material. Damping values for systems composed of subsystems with different damping properties are obtained using the procedures described in Subsection 3.7.2.13.

Table 3.7.1-206 through Table 3.7.1-208 provide the damping ratios for subsurface material properties used in Fermi 3 site-specific SSI analyses with engineered granular backfill above the top of the Bass Islands Group bedrock for the RB/FB and CB. Table 3.7.1-209 through Table 3.7.1-211 provide the damping ratios for subsurface material properties used in Fermi 3 site-specific SSI analyses without engineered granular backfill above the top of the Bass Islands Group bedrock for the RB/FB and CB. The damping ratios in Table 3.7.1-206 through Table 3.7.1-211 are the same for layers below the top of the Bass Islands Group bedrock at an elevation of 168.1 m (551.7 ft) NAVD 88.

3.7.1.3 **Supporting Media for Category I Structures**

The Seismic Category I structures have concrete mat foundations supported on soil, rock or compacted backfill. *The embedment depth, dimensions of the structural foundation, and total structural height for each structure are given in Subsection 3.8.5.1. The soil conditions considered for soil-structural interaction analysis are described in* Appendix 3A.

Subsection 2.5.4 provides site-specific properties of subsurface materials.

Subsection 2.5.4 provides engineering properties of subsurface materials at the Fermi 3 site. The design groundwater elevation assumed for development of the deterministic profiles is provided in Subsection 3.7.1.1.4.1.1. Table 3.7.1-209 through Table 3.7.1-211 provide the strain compatible dynamic engineering properties of subsurface material for the deterministic profiles used for the Fermi 3 site-specific SSI analyses for the RB/FB and CB without the engineered granular backfill above the top of the Bass Islands Group bedrock. Table 3.7.1-206 through Table 3.7.1-208 provide the strain compatible dynamic engineering properties of subsurface material for the deterministic profiles used for the Fermi 3 site-specific SSI analyses for the RB/FB and CB with the engineered granular backfill above the top of the Bass Islands Group bedrock. The LB, BE, and UB profiles with and without engineered granular backfill above the top of the Bass Islands Group bedrock are identical with the exception of approximately 11.5 m (37.6 ft) of engineered granular fill material above the top of the Bass Islands Group bedrock. Figure 3.7.1-223 shows the LB, BE, and UB subsurface shear wave velocity profiles for the Fermi 3 site-specific SSI analysis without the engineered granular backfill above the top of the Bass Islands Group bedrock. Figure 3.7.1-222

shows the LB, BE, and UB subsurface shear wave velocity profiles for the Fermi 3 site-specific SSI analysis with the engineered granular backfill above the top of the Bass Islands Group bedrock.

3.7.1.4 **References**

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Table 3.7.1-201 Full Soil Column Site Response Analysis Profile: Lower Range

Layer Number	Thickness (ft.)	Shear wave Velocity (fps)	Unit Weight (kips/ft.^{3})	Material Curves	Soil/Rock Type
Finished Ground Level Grade, Top of Profile Elevation 589.3 ft.					
1	2.9	397	0.119	MENQ 0-20 feet	Backfill
2	2.9	525	0.119	MENQ 0-20 feet	Backfill
3	4.2	578	0.119	MENQ 0-20 feet	Backfill
4	3.2	626	0.119	MENQ 0-20 feet	Backfill
5	2.5	653	0.119	MENQ 0-20 feet	Backfill
6	4.3	679	0.119	MENQ 0-20 feet	Backfill
7	5.0	718	0.119	MENQ 20-50 feet	Backfill
8	5.0	755	0.119	MENQ 20-50 feet	Backfill
9	3.45	789	0.119	MENQ 20-50 feet	Backfill
10	4.15	821	0.119	MENQ 20-50 feet	Backfill
11	9.7	6650	0.150	Linear, k layer 1	Bass Islands
12	10	6650	0.150	Linear, k layer 1	Bass Islands
13	10	6650	0.150	Linear, k layer 1	Bass Islands
14	10	6650	0.150	Linear, k layer 1	Bass Islands
15	11	6650	0.150	Linear, k layer 1	Bass Islands
16	12	6650	0.150	Linear, k layer 1	Bass Islands
17	12	6650	0.150	Linear, k layer 1	Bass Islands
18	15	4600	0.150	Linear, k layer 2	Bass Islands
19	20	3350	0.150	Linear, k layer 3	Salina F
20	20	3350	0.150	Linear, k layer 3	Salina F
21	20	3350	0.150	Linear, k layer 3	Salina F
22	21	3350	0.150	Linear, k layer 3	Salina F
23	21	4050	0.150	Linear, k layer 3	Salina F
24	21	4050	0.150	Linear, k layer 3	Salina F
25	10	5600	0.150	Linear, k layer 4	Salina E
26	20	9450	0.150	Linear, k layer 4	Salina E
27	21	9450	0.150	Linear, k layer 4	Salina E
28	21	9450	0.150	Linear, k layer 4	Salina E
29	21	9450	0.150	Linear, k layer 4	Salina E
30	45	9000	0.160	Linear, k layer 4	Salina C
31	45	9000	0.160	Linear, k layer 4	Salina C
Halfspace		9300	0.169	0.1 % Damping	Salina B

Table 3.7.1-202 Full Soil Column Site Response Analysis Profile: Intermediate Range

Layer Number	Thickness (ft.)	Shear wave Velocity (fps)	Unit Weight (kips/ft.^{3})	Material Curves	Soil/Rock Type
Finished Ground Level Grade, Top of Profile Elevation 589.3 ft.					
1	2.9	533	0.1325	Intermediate 0-20 feet	Backfill
2	2.9	623	0.1325	Intermediate 0-20 feet	Backfill
3	4.2	680	0.1325	Intermediate 0-20 feet	Backfill
4	3.2	739	0.1325	Intermediate 0-20 feet	Backfill
5	2.5	772	0.1325	Intermediate 0-20 feet	Backfill
6	4.3	806	0.1325	Intermediate 0-20 feet	Backfill
7	5.0	854	0.1325	Intermediate 20-50 feet	Backfill
8	5.0	901	0.1325	Intermediate 20-50 feet	Backfill
9	3.45	944	0.1325	Intermediate 20-50 feet	Backfill
10	4.15	984	0.1325	Intermediate 20-50 feet	Backfill
11	9.7	6650	0.150	Linear, k layer 1	Bass Islands
12	10	6650	0.150	Linear, k layer 1	Bass Islands
13	10	6650	0.150	Linear, k layer 1	Bass Islands
14	10	6650	0.150	Linear, k layer 1	Bass Islands
15	11	6650	0.150	Linear, k layer 1	Bass Islands
16	12	6650	0.150	Linear, k layer 1	Bass Islands
17	12	6650	0.150	Linear, k layer 1	Bass Islands
18	15	4600	0.150	Linear, k layer 2	Bass Islands
19	20	3350	0.150	Linear, k layer 3	Salina F
20	20	3350	0.150	Linear, k layer 3	Salina F
21	20	3350	0.150	Linear, k layer 3	Salina F
22	21	3350	0.150	Linear, k layer 3	Salina F
23	21	4050	0.150	Linear, k layer 3	Salina F
24	21	4050	0.150	Linear, k layer 3	Salina F
25	10	5600	0.150	Linear, k layer 4	Salina E
26	20	9450	0.150	Linear, k layer 4	Salina E
27	21	9450	0.150	Linear, k layer 4	Salina E
28	21	9450	0.150	Linear, k layer 4	Salina E
29	21	9450	0.150	Linear, k layer 4	Salina E
30	45	9000	0.160	Linear, k layer 4	Salina C
31	45	9000	0.160	Linear, k layer 4	Salina C
Halfspace		9300	0.169	0.1 % Damping	Salina B

Table 3.7.1-203 Full Soil Column Site Response Analysis Profile: Upper Range

Layer Number	Thickness (ft.)	Shear wave Velocity (fps)	Unit Weight (kips/ft.^{3})	Material Curves	Soil/Rock Type
Finished Ground Level Grade, Top of Profile Elevation 589.3 ft.					
1	2.9	670	0.146	EPRI 0-20 feet	Backfill
2	2.9	722	0.146	EPRI 0-20 feet	Backfill
3	4.2	781	0.146	EPRI 0-20 feet	Backfill
4	3.2	852	0.146	EPRI 0-20 feet	Backfill
5	2.5	892	0.146	EPRI 0-20 feet	Backfill
6	4.3	932	0.146	EPRI 0-20 feet	Backfill
7	5.0	990	0.146	EPRI 20-50 feet	Backfill
8	5.0	1046	0.146	EPRI 20-50 feet	Backfill
9	3.45	1098	0.146	EPRI 20-50 feet	Backfill
10	4.15	1147	0.146	EPRI 20-50 feet	Backfill
11	9.7	6650	0.150	Linear, k layer 1	Bass Islands
12	10	6650	0.150	Linear, k layer 1	Bass Islands
13	10	6650	0.150	Linear, k layer 1	Bass Islands
14	10	6650	0.150	Linear, k layer 1	Bass Islands
15	11	6650	0.150	Linear, k layer 1	Bass Islands
16	12	6650	0.150	Linear, k layer 1	Bass Islands
17	12	6650	0.150	Linear, k layer 1	Bass Islands
18	15	4600	0.150	Linear, k layer 2	Bass Islands
19	20	3350	0.150	Linear, k layer 3	Salina F
20	20	3350	0.150	Linear, k layer 3	Salina F
21	20	3350	0.150	Linear, k layer 3	Salina F
22	21	3350	0.150	Linear, k layer 3	Salina F
23	21	4050	0.150	Linear, k layer 3	Salina F
24	21	4050	0.150	Linear, k layer 3	Salina F
25	10	5600	0.150	Linear, k layer 4	Salina E
26	20	9450	0.150	Linear, k layer 4	Salina E
27	21	9450	0.150	Linear, k layer 4	Salina E
28	21	9450	0.150	Linear, k layer 4	Salina E
29	21	9450	0.150	Linear, k layer 4	Salina E
30	45	9000	0.160	Linear, k layer 4	Salina C
31	45	9000	0.160	Linear, k layer 4	Salina C
Halfspace		9300	0.169	0.1 % Damping	Salina B

Table 3.7.1-204 FWSC Foundation Input Response Spectrum Site Response Analysis Profile

Layer Number	Thickness (ft.)	Shear wave Velocity (fps)	Unit Weight (kips/ft.^{3})	Material Curves	Soil/Rock Type
FWSC FIRS Profile, Top of Profile Elevation 581.6 ft					
1	1.9	7140	0.145	Fill concrete	N/A
2	1.9	7140	0.145	Fill concrete	N/A
3	3.2	7140	0.145	Fill concrete	N/A
4	2.2	7140	0.145	Fill concrete	N/A
5	1.5	7140	0.145	Fill concrete	N/A
6	3.3	7140	0.145	Fill concrete	N/A
7	4.0	7140	0.145	Fill concrete	N/A
8	4.3	7140	0.145	Fill concrete	N/A
9	3.5	7140	0.145	Fill concrete	N/A
10	4.2	7140	0.145	Fill concrete	N/A
11	9.7	6650	0.150	Linear, k layer 1	Bass Islands
12	10	6650	0.150	Linear, k layer 1	Bass Islands
13	10	6650	0.150	Linear, k layer 1	Bass Islands
14	10	6650	0.150	Linear, k layer 1	Bass Islands
15	11	6650	0.150	Linear, k layer 1	Bass Islands
16	12	6650	0.150	Linear, k layer 1	Bass Islands
17	12	6650	0.150	Linear, k layer 1	Bass Islands
18	15	4600	0.150	Linear, k layer 2	Bass Islands
19	20	3350	0.150	Linear, k layer 3	Salina F
20	20	3350	0.150	Linear, k layer 3	Salina F
21	20	3350	0.150	Linear, k layer 3	Salina F
22	21	3350	0.150	Linear, k layer 3	Salina F
23	21	4050	0.150	Linear, k layer 3	Salina F
24	21	4050	0.150	Linear, k layer 3	Salina F
25	10	5600	0.150	Linear, k layer 4	Salina E
26	20	9450	0.150	Linear, k layer 4	Salina E
27	21	9450	0.150	Linear, k layer 4	Salina E
28	21	9450	0.150	Linear, k layer 4	Salina E
29	21	9450	0.150	Linear, k layer 4	Salina E
30	45	9000	0.160	Linear, k layer 4	Salina C
31	45	9000	0.160	Linear, k layer 4	Salina C
Halfspace		9300	0.169	0.1 % Damping	Salina B

Table 3.7.1-205 Horizontal and Vertical PBSRS at the Finished Ground Level Grade with Associated V/H Ratios (Sheet 1 of 2)

Period (sec)	Frequency (Hz)	Horizontal PBSRS (g)	V/H	Vertical PBSRS (g)
0.010	100.00	0.2368	1.0000	0.2368
0.017	60.241	0.2575	1.0883	0.2802
0.020	50.000	0.2833	1.1329	0.3210
0.025	40.000	0.3221	1.1289	0.3636
0.030	33.333	0.3465	1.0971	0.3802
0.033	30.303	0.3616	1.0491	0.3794
0.040	25.000	0.4166	0.9674	0.4030
0.042	23.810	0.4248	0.9542	0.4054
0.044	22.727	0.4328	0.9400	0.4069
0.046	21.739	0.4406	0.9258	0.4079
0.048	20.833	0.4481	0.9124	0.4089
0.050	20.000	0.4555	0.8997	0.4098
0.055	18.182	0.4683	0.8725	0.4086
0.060	16.667	0.4803	0.8517	0.4091
0.065	15.385	0.4908	0.8319	0.4083
0.070	14.286	0.5008	0.8138	0.4075
0.075	13.333	0.5102	0.8022	0.4093
0.080	12.500	0.5192	0.7929	0.4117
0.085	11.765	0.5278	0.7842	0.4140
0.090	11.111	0.5361	0.7762	0.4161
0.095	10.526	0.5440	0.7686	0.4181
0.10	10.000	0.5516	0.7615	0.4201
0.11	9.0910	0.5661	0.7485	0.4237
0.12	8.3330	0.5796	0.7368	0.4270
0.13	7.6920	0.5923	0.7262	0.4301
0.14	7.1430	0.6043	0.7165	0.4330
0.15	6.6670	0.6157	0.7076	0.4357
0.16	6.2500	0.6266	0.6994	0.4382
0.17	5.8820	0.6370	0.6918	0.4406
0.18	5.5560	0.6469	0.6846	0.4429
0.19	5.2630	0.6565	0.6780	0.4451
0.20	5.0000	0.6657	0.6717	0.4472
0.22	4.5450	0.6817	0.6602	0.4501
0.24	4.1670	0.6849	0.6500	0.4452
0.26	3.8460	0.6813	0.6500	0.4428
0.28	3.5710	0.6639	0.6500	0.4315
0.30	3.3330	0.6372	0.6500	0.4142
0.32	3.1250	0.6039	0.6500	0.3926
0.34	2.9410	0.5663	0.6500	0.3681
0.36	2.7780	0.5289	0.6500	0.3438
0.38	2.6320	0.4990	0.6500	0.3244
0.40	2.5000	0.4660	0.6500	0.3029
0.42	2.3810	0.4409	0.6500	0.2866
0.44	2.2730	0.4160	0.6500	0.2704
0.46	2.1740	0.3915	0.6500	0.2545
0.48	2.0830	0.3677	0.6500	0.2390

Table 3.7.1-205 Horizontal and Vertical PBSRS at the Finished Ground Level Grade with Associated V/H Ratios (Sheet 2 of 2)

Period (sec)	Frequency (Hz)	Horizontal PBSRS (g)	V/H	Vertical PBSRS (g)
0.50	2.0000	0.3469	0.6500	0.2255
0.55	1.8180	0.3082	0.6500	0.2004
0.60	1.6670	0.2737	0.6500	0.1779
0.65	1.5380	0.2438	0.6500	0.1584
0.70	1.4290	0.2191	0.6500	0.1424
0.75	1.3330	0.1997	0.6500	0.1298
0.80	1.2500	0.1824	0.6500	0.1186
0.85	1.1760	0.1685	0.6500	0.1095
0.90	1.1110	0.1573	0.6500	0.1023
0.95	1.0530	0.1468	0.6500	0.0954
1.0	1.0000	0.1373	0.6500	0.0892
1.1	0.9090	0.1234	0.6500	0.0802
1.2	0.8330	0.1128	0.6500	0.0733
1.3	0.7690	0.1024	0.6500	0.0666
1.4	0.7140	0.0966	0.6500	0.0628
1.5	0.6670	0.0918	0.6500	0.0597
1.6	0.6250	0.0875	0.6500	0.0569
1.7	0.5880	0.0839	0.6500	0.0546
1.8	0.5560	0.0817	0.6500	0.0531
1.9	0.5260	0.0797	0.6500	0.0518
2.0	0.5000	0.0778	0.6500	0.0506
2.2	0.4550	0.0719	0.6500	0.0467
2.4	0.4170	0.0669	0.6500	0.0435
2.6	0.3850	0.0626	0.6500	0.0407
2.8	0.3570	0.0589	0.6500	0.0383
3.0	0.3330	0.0556	0.6500	0.0361
3.2	0.3130	0.0527	0.6500	0.0342
3.4	0.2940	0.0505	0.6500	0.0328
3.6	0.2780	0.0485	0.6500	0.0315
3.8	0.2630	0.0467	0.6500	0.0304
4.0	0.2500	0.0451	0.6500	0.0293
4.2	0.2380	0.0436	0.6500	0.0283
4.4	0.2270	0.0422	0.6500	0.0274
4.6	0.2170	0.0409	0.6500	0.0266
4.8	0.2080	0.0397	0.6500	0.0258
5.0	0.2000	0.0386	0.6500	0.0251
5.5	0.1820	0.0352	0.6500	0.0229
6.0	0.1670	0.0324	0.6500	0.0211
6.5	0.1540	0.0300	0.6500	0.0195
7.0	0.1430	0.0280	0.6500	0.0182
7.5	0.1330	0.0262	0.6500	0.0170
8.0	0.1250	0.0244	0.6500	0.0158
8.5	0.1180	0.0228	0.6500	0.0148
9.0	0.1110	0.0214	0.6500	0.0139
10	0.1000	0.0190	0.6500	0.0124

Table 3.7.1-206 Deterministic Profile with Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock: Best Estimate

Layer	Thickness (ft)	Total Depth (ft)	Unit Weight (pcf)	Shear Wave Velocity (ft/sec)	Damping Ratio (%)	Compression Wave Velocity (ft/sec)	Elevation of Layer Base (ft NAVD 88)
Profile with Engineered Granular Backfill, Top of Profile Elevation 589.3 ft NAVD 88							
1	2.9	2.9	132.5	505	3.52	944	586.4
2	2.9	5.8	132.5	520	5.88	973	583.5
3	4.2	10.0	132.5	517	7.58	968	579.3
4	3.2	13.2	132.5	537	8.53	1004	576.1
5	2.5	15.7	132.5	536	9.00	2734	573.6
6	4.3	20.0	132.5	571	9.50	2910	569.3
7	5.0	25.0	132.5	639	7.82	3257	564.3
8	5.0	30.0	132.5	675	7.52	3443	559.3
9	3.5	33.5	132.5	717	7.33	3657	555.8
10	4.1	37.6	132.5	761	7.17	3880	551.7
11	9.5	47.1	150	6510	0.97	12923	542.2
12	1.8	48.9	150	6697	0.97	13294	540.4
13	8.4	57.3	150	6712	0.97	13324	532.0
14	8.3	65.6	150	6740	0.97	13381	523.7
15	2.1	67.7	150	6687	0.97	13275	521.6
16	9.7	77.4	150	6658	0.97	13217	511.9
17	11.1	88.5	150	6593	0.97	13088	500.8
18	12.0	100.5	150	6560	0.97	13024	488.8
19	12.1	112.6	150	6600	0.97	13102	476.7
20	15.0	127.6	150	4573	1.36	9777	461.7
21	20.3	147.9	150	3403	1.87	8014	441.4
22	20.0	167.9	150	3455	1.87	8136	421.4
23	20.0	187.9	150	3390	1.87	7982	401.4
24	21.0	208.9	150	3328	1.87	7838	380.4
25	21.0	229.9	150	4091	1.87	9633	359.4
26	21.1	251.0	150	4166	1.87	9810	338.3
27	10.1	261.1	150	5551	0.71	11410	328.2
28	20.2	281.3	150	9462	0.71	17702	308.0
29	21.0	302.3	150	9432	0.71	17645	287.0
30	21.0	323.3	150	9507	0.71	17786	266.0
31	20.3	343.6	150	9325	0.71	17445	245.7
32	45.0	388.6	160	8956	0.71	16201	200.7
33	45.2	433.8	160	8978	0.71	16243	155.5
34	halfspace	433.8	169	9113	0.10	16757	155.5

Table 3.7.1-207 Deterministic Profile with Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock: Lower Bound

Layer	Thickness (ft.)	Total Depth (ft)	Unit Weight (pcf)	Shear Wave Velocity (ft/sec)	Damping Ratio (%)	Compression Wave Velocity (ft/sec)	Elevation of Layer Base (ft NAVD 88)
Profile with Engineered Granular Backfill, Top of Profile Elevation 589.3 ft NAVD 88							
1	2.9	2.9	119	341	6.22	638	586.4
2	2.9	5.8	119	365	10.26	683	583.5
3	4.2	10.0	119	316	12.78	592	579.3
4	3.2	13.2	119	304	13.65	570	576.1
5	2.5	15.7	119	326	13.53	1663	573.6
6	4.3	20.0	119	296	13.94	1509	569.3
7	5.0	25.0	119	364	12.35	1854	564.3
8	5.0	30.0	119	388	12.24	1979	559.3
9	3.5	33.5	119	416	12.05	2119	555.8
10	4.1	37.6	119	436	11.73	2224	551.7
11	9.5	47.1	150	5315	1.77	10552	542.2
12	1.8	48.9	150	5468	1.77	10855	540.4
13	8.4	57.3	150	5480	1.77	10879	532.0
14	8.3	65.6	150	5503	1.77	10925	523.7
15	2.1	67.7	150	5460	1.77	10839	521.6
16	9.7	77.4	150	5436	1.77	10792	511.9
17	11.1	88.5	150	5383	1.77	10686	500.8
18	12.0	100.5	150	5356	1.77	10634	488.8
19	12.1	112.6	150	5389	1.77	10698	476.7
20	15.0	127.6	150	3734	2.43	7983	461.7
21	20.3	147.9	150	2779	3.03	6544	441.4
22	20.0	167.9	150	2793	3.03	6576	421.4
23	20.0	187.9	150	2768	3.03	6517	401.4
24	21.0	208.9	150	2718	3.03	6400	380.4
25	21.0	229.9	150	3254	3.03	7663	359.4
26	21.1	251.0	150	3273	3.03	7708	338.3
27	10.1	261.1	150	4532	1.28	9316	328.2
28	20.2	281.3	150	7726	1.28	14454	308.0
29	21.0	302.3	150	7701	1.28	14407	287.0
30	21.0	323.3	150	7763	1.28	14522	266.0
31	20.3	343.6	150	7614	1.28	14244	245.7
32	45.0	388.6	160	7312	1.28	13228	200.7
33	45.2	433.8	160	7331	1.28	13262	155.5
34	halfspace	433.8	169	7441	0.10	13682	155.5

Table 3.7.1-208 Deterministic Profile with Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock: Upper Bound

Layer	Thickness (ft.)	Total Depth (ft)	Unit Weight (pcf)	Shear Wave Velocity (ft/sec)	Damping Ratio (%)	Compression Wave Velocity (ft/sec)	Elevation of Layer Base (ft NAVD 88)
Profile with Engineered Granular Backfill, Top of Profile Elevation 589.3 ft NAVD 88							
1	2.9	2.9	146	750	2.26	1403	586.4
2	2.9	5.8	146	750	3.63	1403	583.5
3	4.2	10.0	146	750	4.73	1403	579.3
4	3.2	13.2	146	776	4.97	1451	576.1
5	2.5	15.7	146	831	5.30	4237	573.6
6	4.3	20.0	146	853	5.58	4350	569.3
7	5.0	25.0	146	946	3.94	4790	564.3
8	5.0	30.0	146	1024	3.89	4790	559.3
9	3.5	33.5	146	1033	4.06	4790	555.8
10	4.1	37.6	146	1127	3.76	4790	551.7
11	9.5	47.1	150	7972	0.50	15827	542.2
12	1.8	48.9	150	8202	0.50	16282	540.4
13	8.4	57.3	150	8220	0.50	16318	532.0
14	8.3	65.6	150	8255	0.50	16388	523.7
15	2.1	67.7	150	8189	0.50	16258	521.6
16	9.7	77.4	150	8154	0.50	16188	511.9
17	11.1	88.5	150	8074	0.50	16029	500.8
18	12.0	100.5	150	8035	0.50	15951	488.8
19	12.1	112.6	150	8083	0.50	16046	476.7
20	15.0	127.6	150	5601	0.71	11975	461.7
21	20.3	147.9	150	4187	0.95	9859	441.4
22	20.0	167.9	150	4231	0.95	9965	421.4
23	20.0	187.9	150	4151	0.95	9776	401.4
24	21.0	208.9	150	4370	0.95	10291	380.4
25	21.0	229.9	150	5190	0.95	12222	359.4
26	21.1	251.0	150	5260	0.95	12386	338.3
27	10.1	261.1	150	6798	0.37	13974	328.2
28	20.2	281.3	150	11589	0.37	21681	308.0
29	21.0	302.3	150	11552	0.37	21611	287.0
30	21.0	323.3	150	11644	0.37	21784	266.0
31	20.3	343.6	150	11421	0.37	21366	245.7
32	45.0	388.6	160	10968	0.37	19842	200.7
33	45.2	433.8	160	10996	0.37	19893	155.5
34	halfspace	433.8	169	11161	0.10	20523	155.5

Table 3.7.1-209 Deterministic Profile without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock: Best Estimate

Layer	Thickness (ft.)	Total Depth (ft)	Unit Weight (pcf)	Shear Wave Velocity (ft/sec)	Damping Ratio (%)	Compression Wave Velocity (ft/sec)	Elevation of Layer Base (ft NAVD 88)
Profile with Engineered Granular Backfill, Top of Profile Elevation 551.7 ft NAVD 88							
1	9.5	47.1	150	6510	0.97	12923	542.2
2	1.8	48.9	150	6697	0.97	13294	540.4
3	8.4	57.3	150	6712	0.97	13324	532.0
4	8.3	65.6	150	6740	0.97	13381	523.7
5	2.1	67.7	150	6687	0.97	13275	521.6
6	9.7	77.4	150	6658	0.97	13217	511.9
7	11.1	88.5	150	6593	0.97	13088	500.8
8	12.0	100.5	150	6560	0.97	13024	488.8
9	12.1	112.6	150	6600	0.97	13102	476.7
10	15.0	127.6	150	4573	1.36	9777	461.7
11	20.3	147.9	150	3403	1.87	8014	441.4
12	20.0	167.9	150	3455	1.87	8136	421.4
13	20.0	187.9	150	3390	1.87	7982	401.4
14	21.0	208.9	150	3328	1.87	7838	380.4
15	21.0	229.9	150	4091	1.87	9633	359.4
16	21.1	251.0	150	4166	1.87	9810	338.3
17	10.1	261.1	150	5551	0.71	11410	328.2
18	20.2	281.3	150	9462	0.71	17702	308.0
19	21.0	302.3	150	9432	0.71	17645	287.0
20	21.0	323.3	150	9507	0.71	17786	266.0
21	20.3	343.6	150	9325	0.71	17445	245.7
22	45.0	388.6	160	8956	0.71	16201	200.7
23	45.2	433.8	160	8978	0.71	16243	155.5
24	halfspace	433.8	169	9113	0.10	16757	155.5

Table 3.7.1-210 Deterministic Profile without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock: Lower Bound

Layer	Thickness (ft.)	Total Depth (ft)	Unit Weight (pcf)	Shear Wave Velocity (ft/sec)	Damping Ratio (%)	Compression Wave Velocity (ft/sec)	Elevation of Layer Base (ft NAVD 88)
Profile with Engineered Granular Backfill, Top of Profile Elevation 551.7 ft NAVD 88							
1	9.5	47.1	150	5315	1.77	10552	542.2
2	1.8	48.9	150	5468	1.77	10855	540.4
3	8.4	57.3	150	5480	1.77	10879	532.0
4	8.3	65.6	150	5503	1.77	10925	523.7
5	2.1	67.7	150	5460	1.77	10839	521.6
6	9.7	77.4	150	5436	1.77	10792	511.9
7	11.1	88.5	150	5383	1.77	10686	500.8
8	12.0	100.5	150	5356	1.77	10634	488.8
9	12.1	112.6	150	5389	1.77	10698	476.7
10	15.0	127.6	150	3734	2.43	7983	461.7
11	20.3	147.9	150	2779	3.03	6544	441.4
12	20.0	167.9	150	2793	3.03	6576	421.4
13	20.0	187.9	150	2768	3.03	6517	401.4
14	21.0	208.9	150	2718	3.03	6400	380.4
15	21.0	229.9	150	3254	3.03	7663	359.4
16	21.1	251.0	150	3273	3.03	7708	338.3
17	10.1	261.1	150	4532	1.28	9316	328.2
18	20.2	281.3	150	7726	1.28	14454	308.0
19	21.0	302.3	150	7701	1.28	14407	287.0
20	21.0	323.3	150	7763	1.28	14522	266.0
21	20.3	343.6	150	7614	1.28	14244	245.7
22	45.0	388.6	160	7312	1.28	13228	200.7
23	45.2	433.8	160	7331	1.28	13262	155.5
24	halfspace	433.8	169	7441	0.10	13682	155.5

Table 3.7.1-211 Deterministic Profile without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock: Upper Bound

Layer	Thickness (ft.)	Total Depth (ft)	Unit Weight (pcf)	Shear Wave Velocity (ft/sec)	Damping Ratio (%)	Compression Wave Velocity (ft/sec)	Elevation of Layer Base (ft NAVD 88)
Profile with Engineered Granular Backfill, Top of Profile Elevation 551.7 ft NAVD 88							
1	9.5	47.1	150	7972	0.50	15827	542.2
2	1.8	48.9	150	8202	0.50	16282	540.4
3	8.4	57.3	150	8220	0.50	16318	532.0
4	8.3	65.6	150	8255	0.50	16388	523.7
5	2.1	67.7	150	8189	0.50	16258	521.6
6	9.7	77.4	150	8154	0.50	16188	511.9
7	11.1	88.5	150	8074	0.50	16029	500.8
8	12.0	100.5	150	8035	0.50	15951	488.8
9	12.1	112.6	150	8083	0.50	16046	476.7
10	15.0	127.6	150	5601	0.71	11975	461.7
11	20.3	147.9	150	4187	0.95	9859	441.4
12	20.0	167.9	150	4231	0.95	9965	421.4
13	20.0	187.9	150	4151	0.95	9776	401.4
14	21.0	208.9	150	4370	0.95	10291	380.4
15	21.0	229.9	150	5190	0.95	12222	359.4
16	21.1	251.0	150	5260	0.95	12386	338.3
17	10.1	261.1	150	6798	0.37	13974	328.2
18	20.2	281.3	150	11589	0.37	21681	308.0
19	21.0	302.3	150	11552	0.37	21611	287.0
20	21.0	323.3	150	11644	0.37	21784	266.0
21	20.3	343.6	150	11421	0.37	21366	245.7
22	45.0	388.6	160	10968	0.37	19842	200.7
23	45.2	433.8	160	10996	0.37	19893	155.5
24	halfspace	433.8	169	11161	0.10	20523	155.5

Table 3.7.1-212 Horizontal and Vertical RB/FB SCOR FIRS at Elevation 523.7 (ft) NAVD 88 with Associated V/H Ratios (Sheet 1 of 3)

Period (sec)	Frequency (Hz)	Horizontal SCOR FIRS (g)	V/H Ratio	Vertical SCOR FIRS (g)
0.010	100.00	0.2092	1.0000	0.2092
0.017	60.241	0.3830	1.1374	0.4356
0.020	50.000	0.4543	1.1244	0.5108
0.025	40.000	0.5075	1.0426	0.5291
0.030	33.333	0.5302	0.9675	0.5129
0.033	30.303	0.5362	0.9400	0.5041
0.040	25.000	0.5487	0.8800	0.4829
0.042	23.810	0.5435	0.8681	0.4718
0.044	22.727	0.5386	0.8569	0.4615
0.046	21.739	0.5339	0.8461	0.4518
0.048	20.833	0.5295	0.8355	0.4424
0.050	20.000	0.5253	0.8255	0.4336
0.055	18.182	0.5111	0.8069	0.4124
0.060	16.667	0.4984	0.7984	0.3979
0.065	15.385	0.4834	0.7906	0.3822
0.070	14.286	0.4763	0.7834	0.3732
0.075	13.333	0.4698	0.7769	0.3650
0.080	12.500	0.4638	0.7708	0.3575
0.085	11.765	0.4583	0.7651	0.3506
0.090	11.111	0.4531	0.7597	0.3442
0.095	10.526	0.4482	0.7547	0.3383
0.10	10.000	0.4437	0.7500	0.3327
0.11	9.0910	0.4353	0.7500	0.3265
0.12	8.3330	0.4278	0.7500	0.3209
0.13	7.6920	0.4211	0.7500	0.3158
0.14	7.1430	0.4149	0.7500	0.3112
0.15	6.6670	0.4092	0.7500	0.3069
0.16	6.2500	0.4040	0.7500	0.3030
0.17	5.8820	0.3991	0.7500	0.2994
0.18	5.5560	0.3946	0.7500	0.2960
0.19	5.2630	0.3904	0.7500	0.2928
0.20	5.0000	0.3864	0.7500	0.2898
0.22	4.5450	0.3791	0.7500	0.2844
0.24	4.1670	0.3711	0.7500	0.2783
0.26	3.8460	0.3614	0.7500	0.2710
0.28	3.5710	0.3451	0.7500	0.2588
0.30	3.3330	0.3240	0.7500	0.2430

Table 3.7.1-212 Horizontal and Vertical RB/FB SCOR FIRS at Elevation 523.7 (ft) NAVD 88 with Associated V/H Ratios (Sheet 2 of 3)

Period (sec)	Frequency (Hz)	Horizontal SCOR FIRS (g)	V/H Ratio	Vertical SCOR FIRS (g)
0.32	3.1250	0.3041	0.7500	0.2281
0.34	2.9410	0.2836	0.7500	0.2127
0.36	2.7780	0.2652	0.7500	0.1989
0.38	2.6320	0.2490	0.7500	0.1868
0.40	2.5000	0.2323	0.7500	0.1742
0.42	2.3810	0.2201	0.7500	0.1650
0.44	2.2730	0.2081	0.7500	0.1561
0.46	2.1740	0.1963	0.7500	0.1472
0.48	2.0830	0.1860	0.7500	0.1395
0.50	2.0000	0.1766	0.7500	0.1325
0.55	1.8180	0.1600	0.7500	0.1200
0.60	1.6670	0.1466	0.7500	0.1099
0.65	1.5380	0.1358	0.7500	0.1018
0.70	1.4290	0.1265	0.7500	0.0949
0.75	1.3330	0.1186	0.7500	0.0889
0.80	1.2500	0.1113	0.7500	0.0835
0.85	1.1760	0.1054	0.7500	0.0790
0.90	1.1110	0.1000	0.7500	0.0750
0.95	1.0530	0.0947	0.7500	0.0710
1.0	1.0000	0.0907	0.7500	0.0680
1.1	0.9090	0.0850	0.7500	0.0638
1.2	0.8330	0.0799	0.7500	0.0599
1.3	0.7690	0.0755	0.7500	0.0567
1.4	0.7140	0.0728	0.7500	0.0546
1.5	0.6670	0.0710	0.7500	0.0533
1.6	0.6250	0.0691	0.7500	0.0518
1.7	0.5880	0.0673	0.7500	0.0505
1.8	0.5560	0.0658	0.7500	0.0494
1.9	0.5260	0.0644	0.7500	0.0483
2.0	0.5000	0.0632	0.7500	0.0474
2.2	0.4550	0.0591	0.7500	0.0443
2.4	0.4170	0.0553	0.7500	0.0415
2.6	0.3850	0.0525	0.7500	0.0394
2.8	0.3570	0.0499	0.7500	0.0374
3.0	0.3330	0.0476	0.7500	0.0357
3.2	0.3130	0.0454	0.7500	0.0341
3.4	0.2940	0.0436	0.7500	0.0327

Table 3.7.1-212 Horizontal and Vertical RB/FB SCOR FIRS at Elevation 523.7 (ft) NAVD 88 with Associated V/H Ratios (Sheet 3 of 3)

Period (sec)	Frequency (Hz)	Horizontal SCOR FIRS (g)	V/H Ratio	Vertical SCOR FIRS (g)
3.6	0.2780	0.0419	0.7500	0.0314
3.8	0.2630	0.0403	0.7500	0.0302
4.0	0.2500	0.0389	0.7500	0.0292
4.2	0.2380	0.0376	0.7500	0.0282
4.4	0.2270	0.0364	0.7500	0.0273
4.6	0.2170	0.0353	0.7500	0.0265
4.8	0.2080	0.0343	0.7500	0.0257
5.0	0.2000	0.0333	0.7500	0.0250
5.5	0.1820	0.0304	0.7500	0.0228
6.0	0.1670	0.0280	0.7500	0.0210
6.5	0.1540	0.0259	0.7500	0.0194
7.0	0.1430	0.0241	0.7500	0.0181
7.5	0.1330	0.0226	0.7500	0.0169
8.0	0.1250	0.0210	0.7500	0.0158
8.5	0.1180	0.0197	0.7500	0.0148
9.0	0.1110	0.0185	0.7500	0.0138
10	0.1000	0.0164	0.7500	0.0123

Table 3.7.1-213 Horizontal and Vertical CB SCOR FIRS at Elevation 540.4 (ft) NAVD 88 with Associated V/H Ratios (Sheet 1 of 3)

Period (sec)	Frequency (Hz)	Horizontal SCOR FIRS (g)	V/H Ratio	Vertical SCOR FIRS (g)
0.010	100.00	0.2084	1.0000	0.2084
0.017	60.241	0.3818	1.1374	0.4343
0.020	50.000	0.4531	1.1244	0.5095
0.025	40.000	0.5063	1.0426	0.5279
0.030	33.333	0.5291	0.9675	0.5119
0.033	30.303	0.5358	0.9400	0.5037
0.040	25.000	0.5498	0.8800	0.4838
0.042	23.810	0.5445	0.8681	0.4727
0.044	22.727	0.5395	0.8569	0.4623
0.046	21.739	0.5348	0.8461	0.4525
0.048	20.833	0.5304	0.8355	0.4431
0.050	20.000	0.5261	0.8255	0.4343
0.055	18.182	0.5118	0.8069	0.4130
0.060	16.667	0.4991	0.7984	0.3984
0.065	15.385	0.4840	0.7906	0.3827
0.070	14.286	0.4769	0.7834	0.3736
0.075	13.333	0.4703	0.7769	0.3654
0.080	12.500	0.4643	0.7708	0.3578
0.085	11.765	0.4587	0.7651	0.3509
0.090	11.111	0.4534	0.7597	0.3445
0.095	10.526	0.4485	0.7547	0.3385
0.10	10.000	0.4440	0.7500	0.3330
0.11	9.0910	0.4355	0.7500	0.3267
0.12	8.3330	0.4280	0.7500	0.3210
0.13	7.6920	0.4212	0.7500	0.3159
0.14	7.1430	0.4150	0.7500	0.3112
0.15	6.6670	0.4093	0.7500	0.3070
0.16	6.2500	0.4040	0.7500	0.3030
0.17	5.8820	0.3991	0.7500	0.2994
0.18	5.5560	0.3946	0.7500	0.2959
0.19	5.2630	0.3903	0.7500	0.2927
0.20	5.0000	0.3863	0.7500	0.2898
0.22	4.5450	0.3790	0.7500	0.2843
0.24	4.1670	0.3710	0.7500	0.2782
0.26	3.8460	0.3614	0.7500	0.2710
0.28	3.5710	0.3452	0.7500	0.2589
0.30	3.3330	0.3241	0.7500	0.2431

Table 3.7.1-213 Horizontal and Vertical CB SCOR FIRS at Elevation 540.4 (ft) NAVD 88 with Associated V/H Ratios (Sheet 2 of 3)

Period (sec)	Frequency (Hz)	Horizontal SCOR FIRS (g)	V/H Ratio	Vertical SCOR FIRS (g)
0.32	3.1250	0.3042	0.7500	0.2282
0.34	2.9410	0.2838	0.7500	0.2128
0.36	2.7780	0.2654	0.7500	0.1990
0.38	2.6320	0.2491	0.7500	0.1868
0.40	2.5000	0.2324	0.7500	0.1743
0.42	2.3810	0.2202	0.7500	0.1651
0.44	2.2730	0.2082	0.7500	0.1562
0.46	2.1740	0.1964	0.7500	0.1473
0.48	2.0830	0.1861	0.7500	0.1396
0.50	2.0000	0.1767	0.7500	0.1325
0.55	1.8180	0.1600	0.7500	0.1200
0.60	1.6670	0.1466	0.7500	0.1100
0.65	1.5380	0.1358	0.7500	0.1018
0.70	1.4290	0.1265	0.7500	0.0949
0.75	1.3330	0.1185	0.7500	0.0889
0.80	1.2500	0.1113	0.7500	0.0834
0.85	1.1760	0.1053	0.7500	0.0790
0.90	1.1110	0.1000	0.7500	0.0750
0.95	1.0530	0.0947	0.7500	0.0710
1.0	1.0000	0.0907	0.7500	0.0680
1.1	0.9090	0.0850	0.7500	0.0638
1.2	0.8330	0.0799	0.7500	0.0599
1.3	0.7690	0.0755	0.7500	0.0567
1.4	0.7140	0.0728	0.7500	0.0546
1.5	0.6670	0.0710	0.7500	0.0533
1.6	0.6250	0.0690	0.7500	0.0518
1.7	0.5880	0.0673	0.7500	0.0505
1.8	0.5560	0.0658	0.7500	0.0494
1.9	0.5260	0.0644	0.7500	0.0483
2.0	0.5000	0.0632	0.7500	0.0474
2.2	0.4550	0.0591	0.7500	0.0443
2.4	0.4170	0.0553	0.7500	0.0415
2.6	0.3850	0.0525	0.7500	0.0394
2.8	0.3570	0.0499	0.7500	0.0374
3.0	0.3330	0.0476	0.7500	0.0357
3.2	0.3130	0.0454	0.7500	0.0341
3.4	0.2940	0.0436	0.7500	0.0327

Table 3.7.1-213 Horizontal and Vertical CB SCOR FIRS at Elevation 540.4 (ft) NAVD 88 with Associated V/H Ratios (Sheet 3 of 3)

Period (sec)	Frequency (Hz)	Horizontal SCOR FIRS (g)	V/H Ratio	Vertical SCOR FIRS (g)
3.6	0.2780	0.0419	0.7500	0.0314
3.8	0.2630	0.0403	0.7500	0.0302
4.0	0.2500	0.0389	0.7500	0.0292
4.2	0.2380	0.0376	0.7500	0.0282
4.4	0.2270	0.0364	0.7500	0.0273
4.6	0.2170	0.0353	0.7500	0.0265
4.8	0.2080	0.0343	0.7500	0.0257
5.0	0.2000	0.0333	0.7500	0.0250
5.5	0.1820	0.0304	0.7500	0.0228
6.0	0.1670	0.0280	0.7500	0.0210
6.5	0.1540	0.0259	0.7500	0.0194
7.0	0.1430	0.0241	0.7500	0.0181
7.5	0.1330	0.0226	0.7500	0.0169
8.0	0.1250	0.0210	0.7500	0.0158
8.5	0.1180	0.0197	0.7500	0.0148
9.0	0.1110	0.0185	0.7500	0.0138
10	0.1000	0.0164	0.7500	0.0123

Table 3.7.1-214 Enhanced Horizontal and Vertical RB/FB SCOR FIRS at Elevation 523.7 (ft) NAVD 88 (Sheet 1 of 3)

Period (sec)	Frequency (Hz)	Horizontal Enhanced SCOR FIRS (g)	Vertical Enhanded SCOR FIRS (g)
0.010	100.00	0.2217	0.2301
0.017	60.241	0.4060	0.4792
0.020	50.000	0.4816	0.5619
0.025	40.000	0.5380	0.5821
0.030	33.333	0.5620	0.5642
0.033	30.303	0.5684	0.5545
0.040	25.000	0.5817	0.5312
0.042	23.810	0.5761	0.5190
0.044	22.727	0.5709	0.5077
0.046	21.739	0.5660	0.4970
0.048	20.833	0.5613	0.4867
0.050	20.000	0.5568	0.4770
0.055	18.182	0.5417	0.4742
0.060	16.667	0.5283	0.4717
0.065	15.385	0.5244	0.4694
0.070	14.286	0.5207	0.4672
0.075	13.333	0.5174	0.4653
0.080	12.500	0.5142	0.4634
0.085	11.765	0.5113	0.4617
0.090	11.111	0.5086	0.4601
0.095	10.526	0.5060	0.4586
0.10	10.000	0.5036	0.4571
0.11	9.0909	0.4991	0.4544
0.12	8.3333	0.4950	0.4520
0.13	7.6923	0.4913	0.4498
0.14	7.1429	0.4879	0.4478
0.15	6.6667	0.4848	0.4459
0.16	6.2500	0.4818	0.4441
0.17	5.8824	0.4791	0.4425
0.18	5.5556	0.4765	0.4409
0.19	5.2632	0.4741	0.4394
0.20	5.0000	0.4718	0.4381
0.22	4.5455	0.4676	0.4355
0.24	4.1667	0.4638	0.4332
0.26	3.8462	0.4363	0.4320
0.28	3.5714	0.4124	0.4309

Table 3.7.1-214 Enhanced Horizontal and Vertical RB/FB SCOR FIRS at Elevation 523.7 (ft) NAVD 88 (Sheet 2 of 3)

Period (sec)	Frequency (Hz)	Horizontal Enhanced SCOR FIRS (g)	Vertical Enhanded SCOR FIRS (g)
0.30	3.3333	0.3912	0.4145
0.32	3.1250	0.3724	0.3923
0.34	2.9412	0.3556	0.3726
0.36	2.7778	0.3404	0.3549
0.38	2.6316	0.3184	0.3389
0.40	2.5000	0.3130	0.3197
0.42	2.3810	0.3007	0.3024
0.44	2.2727	0.2894	0.2868
0.46	2.1739	0.2790	0.2726
0.48	2.0833	0.2694	0.2597
0.50	2.0000	0.2605	0.2479
0.55	1.8182	0.2409	0.2224
0.60	1.6667	0.2243	0.2014
0.65	1.5385	0.2100	0.1839
0.70	1.4286	0.1976	0.1690
0.75	1.3333	0.1867	0.1562
0.80	1.2500	0.1770	0.1451
0.85	1.1765	0.1684	0.1354
0.90	1.1111	0.1607	0.1290
0.95	1.0526	0.1537	0.1232
1.0	1.0000	0.1473	0.1179
1.1	0.9091	0.1362	0.1087
1.2	0.8333	0.1268	0.1010
1.3	0.7692	0.1187	0.0938
1.4	0.7143	0.1117	0.0875
1.5	0.6667	0.1056	0.0821
1.6	0.6250	0.1001	0.0774
1.7	0.5882	0.0952	0.0731
1.8	0.5556	0.0909	0.0694
1.9	0.5263	0.0869	0.0660
2.0	0.5000	0.0833	0.0629
2.2	0.4545	0.0770	0.0576
2.4	0.4167	0.0717	0.0531
2.6	0.3846	0.0672	0.0493
2.8	0.3571	0.0632	0.0461
3.0	0.3333	0.0597	0.0432

Table 3.7.1-214 Enhanced Horizontal and Vertical RB/FB SCOR FIRS at Elevation 523.7 (ft) NAVD 88 (Sheet 3 of 3)

Period (sec)	Frequency (Hz)	Horizontal Enhanced SCOR FIRS (g)	Vertical Enhanded SCOR FIRS (g)
3.2	0.3125	0.0566	0.0407
3.4	0.2941	0.0539	0.0385
3.6	0.2778	0.0514	0.0365
3.8	0.2632	0.0492	0.0347
4.0	0.2500	0.0471	0.0331
4.2	0.2381	0.0450	0.0316
4.4	0.2273	0.0431	0.0303
4.6	0.2174	0.0414	0.0291
4.8	0.2083	0.0397	0.0280
5.0	0.2000	0.0382	0.0269
5.5	0.1818	0.0350	0.0247
6.0	0.1667	0.0323	0.0227
6.5	0.1538	0.0299	0.0211
7.0	0.1429	0.0279	0.0197
7.5	0.1333	0.0262	0.0185
8.0	0.1250	0.0246	0.0174
8.5	0.1176	0.0233	0.0165
9.0	0.1111	0.0221	0.0156
10	0.1000	0.0200	0.0142

Table 3.7.1-215 Enhanced Horizontal and Vertical CB SCOR FIRS at Elevation 540.4 (ft) NAVD 88 (Sheet 1 of 3)

Period (sec)	Frequency (Hz)	Horizontal Enhanced SCOR FIRS (g)	Vertical Enhanded SCOR FIRS (g)
0.010	100.00	0.2209	0.2292
0.017	60.241	0.4047	0.4777
0.020	50.000	0.4803	0.5604
0.025	40.000	0.5367	0.5807
0.030	33.333	0.5608	0.5631
0.033	30.303	0.5680	0.5541
0.040	25.000	0.5827	0.5322
0.042	23.810	0.5772	0.5200
0.044	22.727	0.5719	0.5086
0.046	21.739	0.5669	0.4978
0.048	20.833	0.5622	0.4875
0.050	20.000	0.5577	0.4777
0.055	18.182	0.5425	0.4749
0.060	16.667	0.5290	0.4723
0.065	15.385	0.5250	0.4700
0.070	14.286	0.5213	0.4678
0.075	13.333	0.5179	0.4658
0.080	12.500	0.5147	0.4639
0.085	11.765	0.5118	0.4622
0.090	11.111	0.5090	0.4605
0.095	10.526	0.5064	0.4590
0.10	10.000	0.5039	0.4575
0.11	9.0909	0.4994	0.4548
0.12	8.3333	0.4953	0.4523
0.13	7.6923	0.4915	0.4501
0.14	7.1429	0.4881	0.4480
0.15	6.6667	0.4849	0.4461
0.16	6.2500	0.4819	0.4443
0.17	5.8824	0.4792	0.4426
0.18	5.5556	0.4766	0.4410
0.19	5.2632	0.4741	0.4396
0.20	5.0000	0.4718	0.4382
0.22	4.5455	0.4676	0.4356
0.24	4.1667	0.4637	0.4332
0.26	3.8462	0.4363	0.4320
0.28	3.5714	0.4123	0.4309

Table 3.7.1-215 Enhanced Horizontal and Vertical CB SCOR FIRS at Elevation 540.4 (ft) NAVD 88 (Sheet 2 of 3)

Period (sec)	Frequency (Hz)	Horizontal Enhanced SCOR FIRS (g)	Vertical Enhanded SCOR FIRS (g)
0.30	3.3333	0.3912	0.4145
0.32	3.1250	0.3724	0.3923
0.34	2.9412	0.3556	0.3726
0.36	2.7778	0.3404	0.3549
0.38	2.6316	0.3184	0.3389
0.40	2.5000	0.3130	0.3197
0.42	2.3810	0.3007	0.3024
0.44	2.2727	0.2894	0.2868
0.46	2.1739	0.2790	0.2726
0.48	2.0833	0.2694	0.2597
0.50	2.0000	0.2605	0.2479
0.55	1.8182	0.2409	0.2224
0.60	1.6667	0.2243	0.2014
0.65	1.5385	0.2100	0.1839
0.70	1.4286	0.1976	0.1690
0.75	1.3333	0.1867	0.1562
0.80	1.2500	0.1770	0.1451
0.85	1.1765	0.1684	0.1354
0.90	1.1111	0.1607	0.1290
0.95	1.0526	0.1537	0.1232
1.0	1.0000	0.1473	0.1179
1.1	0.9091	0.1362	0.1087
1.2	0.8333	0.1268	0.1010
1.3	0.7692	0.1187	0.0938
1.4	0.7143	0.1117	0.0875
1.5	0.6667	0.1056	0.0821
1.6	0.6250	0.1001	0.0774
1.7	0.5882	0.0952	0.0731
1.8	0.5556	0.0909	0.0694
1.9	0.5263	0.0869	0.0660
2.0	0.5000	0.0833	0.0629
2.2	0.4545	0.0770	0.0576
2.4	0.4167	0.0717	0.0531
2.6	0.3846	0.0672	0.0493
2.8	0.3571	0.0632	0.0461
3.0	0.3333	0.0597	0.0432

Table 3.7.1-215 Enhanced Horizontal and Vertical CB SCOR FIRS at Elevation 540.4 (ft) NAVD 88 (Sheet 3 of 3)

Period (sec)	Frequency (Hz)	Horizontal Enhanced SCOR FIRS (g)	Vertical Enhanded SCOR FIRS (g)
3.2	0.3125	0.0566	0.0407
3.4	0.2941	0.0539	0.0385
3.6	0.2778	0.0514	0.0365
3.8	0.2632	0.0492	0.0347
4.0	0.2500	0.0471	0.0331
4.2	0.2381	0.0450	0.0316
4.4	0.2273	0.0431	0.0303
4.6	0.2174	0.0414	0.0291
4.8	0.2083	0.0397	0.0280
5.0	0.2000	0.0382	0.0269
5.5	0.1818	0.0350	0.0247
6.0	0.1667	0.0323	0.0227
6.5	0.1538	0.0299	0.0211
7.0	0.1429	0.0279	0.0197
7.5	0.1333	0.0262	0.0185
8.0	0.1250	0.0246	0.0174
8.5	0.1176	0.0233	0.0165
9.0	0.1111	0.0221	0.0156
10	0.1000	0.0200	0.0142

Table 3.7.1-216 Horizontal and Vertical FWSC FIRS at Elevation 581.6 (ft) NAVD 88 with Associated V/H Ratios (Sheet 1 of 3)

Period (sec)	Frequency (Hz)	Horizontal FWSC FIRS (g)	V/H Ratio	Vertical FWSC FIRS (g)
0.010	100.00	0.2290	1.0000	0.2290
0.017	60.241	0.4282	1.1374	0.4871
0.020	50.000	0.5789	1.1244	0.6509
0.025	40.000	0.7887	1.0426	0.8224
0.030	33.333	0.9961	0.9675	0.9638
0.033	30.303	1.0524	0.9400	0.9893
0.040	25.000	1.0316	0.8800	0.9078
0.042	23.810	0.9953	0.8681	0.8640
0.044	22.727	0.9559	0.8569	0.8191
0.046	21.739	0.9141	0.8461	0.7735
0.048	20.833	0.8687	0.8355	0.7258
0.050	20.000	0.8429	0.8255	0.6958
0.055	18.182	0.7880	0.8069	0.6358
0.060	16.667	0.7613	0.7984	0.6078
0.065	15.385	0.7395	0.7906	0.5846
0.070	14.286	0.7150	0.7834	0.5602
0.075	13.333	0.6930	0.7769	0.5383
0.080	12.500	0.6729	0.7708	0.5187
0.085	11.765	0.6547	0.7651	0.5009
0.090	11.111	0.6379	0.7597	0.4846
0.095	10.526	0.6224	0.7547	0.4698
0.10	10.000	0.6081	0.7500	0.4561
0.11	9.0910	0.5824	0.7500	0.4368
0.12	8.3330	0.5598	0.7500	0.4199
0.13	7.6920	0.5398	0.7500	0.4049
0.14	7.1430	0.5220	0.7500	0.3915
0.15	6.6670	0.5059	0.7500	0.3794
0.16	6.2500	0.4913	0.7500	0.3685
0.17	5.8820	0.4779	0.7500	0.3584
0.18	5.5560	0.4657	0.7500	0.3493
0.19	5.2630	0.4544	0.7500	0.3408
0.20	5.0000	0.4439	0.7500	0.3330
0.22	4.5450	0.4251	0.7500	0.3189
0.24	4.1670	0.4087	0.7500	0.3065
0.26	3.8460	0.3941	0.7500	0.2956
0.28	3.5710	0.3865	0.7500	0.2899
0.30	3.3330	0.3689	0.7500	0.2767

Table 3.7.1-216 Horizontal and Vertical FWSC FIRS at Elevation 581.6 (ft) NAVD 88 with Associated V/H Ratios (Sheet 2 of 3)

Period (sec)	Frequency (Hz)	Horizontal FWSC FIRS (g)	V/H Ratio	Vertical FWSC FIRS (g)
0.32	3.1250	0.3471	0.7500	0.2603
0.34	2.9410	0.3232	0.7500	0.2424
0.36	2.7780	0.3002	0.7500	0.2251
0.38	2.6320	0.2789	0.7500	0.2092
0.40	2.5000	0.2564	0.7500	0.1923
0.42	2.3810	0.2391	0.7500	0.1793
0.44	2.2730	0.2236	0.7500	0.1677
0.46	2.1740	0.2085	0.7500	0.1564
0.48	2.0830	0.1956	0.7500	0.1467
0.50	2.0000	0.1843	0.7500	0.1383
0.55	1.8180	0.1653	0.7500	0.1240
0.60	1.6670	0.1500	0.7500	0.1125
0.65	1.5380	0.1380	0.7500	0.1035
0.70	1.4290	0.1281	0.7500	0.0961
0.75	1.3330	0.1198	0.7500	0.0899
0.80	1.2500	0.1118	0.7500	0.0838
0.85	1.1760	0.1059	0.7500	0.0794
0.90	1.1110	0.1007	0.7500	0.0755
0.95	1.0530	0.0952	0.7500	0.0714
1.0	1.0000	0.0910	0.7500	0.0682
1.1	0.9090	0.0852	0.7500	0.0639
1.2	0.8330	0.0802	0.7500	0.0602
1.3	0.7690	0.0759	0.7500	0.0569
1.4	0.7140	0.0730	0.7500	0.0548
1.5	0.6670	0.0713	0.7500	0.0535
1.6	0.6250	0.0694	0.7500	0.0520
1.7	0.5880	0.0675	0.7500	0.0506
1.8	0.5560	0.0660	0.7500	0.0495
1.9	0.5260	0.0646	0.7500	0.0485
2.0	0.5000	0.0634	0.7500	0.0475
2.2	0.4550	0.0593	0.7500	0.0445
2.4	0.4170	0.0555	0.7500	0.0416
2.6	0.3850	0.0528	0.7500	0.0396
2.8	0.3570	0.0502	0.7500	0.0376
3.0	0.3330	0.0478	0.7500	0.0359
3.2	0.3130	0.0457	0.7500	0.0343
3.4	0.2940	0.0438	0.7500	0.0328

Table 3.7.1-216 Horizontal and Vertical FWSC FIRS at Elevation 581.6 (ft) NAVD 88 with Associated V/H Ratios (Sheet 3 of 3)

Period (sec)	Frequency (Hz)	Horizontal FWSC FIRS (g)	V/H Ratio	Vertical FWSC FIRS (g)
3.6	0.2780	0.0418	0.7500	0.0314
3.8	0.2630	0.0404	0.7500	0.0303
4.0	0.2500	0.0391	0.7500	0.0293
4.2	0.2380	0.0380	0.7500	0.0285
4.4	0.2270	0.0369	0.7500	0.0277
4.6	0.2170	0.0358	0.7500	0.0268
4.8	0.2080	0.0348	0.7500	0.0261
5.0	0.2000	0.0339	0.7500	0.0255
5.5	0.1820	0.0312	0.7500	0.0234
6.0	0.1670	0.0288	0.7500	0.0216
6.5	0.1540	0.0267	0.7500	0.0200
7.0	0.1430	0.0249	0.7500	0.0187
7.5	0.1330	0.0234	0.7500	0.0176
8.0	0.1250	0.0219	0.7500	0.0164
8.5	0.1180	0.0205	0.7500	0.0154
9.0	0.1110	0.0193	0.7500	0.0145
10	0.1000	0.0173	0.7500	0.0130

Table 3.7.1-217 Seed Time History Recording Details

Earthquake	Station	Component	Filter Corners: High-Pass (Hz)	Filter Corners: Low-Pass (Hz)	Record Parameters: PGA (g)	Record Parameters: PGV (cm/sec)	Record Parameters: PGD (cm)	Record Parameters: Duration(1) (sec)
1999 Chi-Chi, Taiwan **M** 7.6	TAP078 R = 131 km	TAP078-N	0.04	40	0.088	13.0	5.6	25.8
		TAP078-W	0.02	40	0.094	10.7	5.0	30.1
		TAP078-V	0.03	33	0.063	8.6	8.3	30.5

Note:

1. Duration is defined as the time interval between the time history points at which 5 and 75 percent of the normalized Arias intensity (total energy measure) has been recorded.

Table 3.7.1-218 Cross Correlation Coefficients for the Matched Time Histories

Building	Components	Cross Correlation Coefficient
RB/FB	H1 – H2	-0.01
	H1 – V	0.02
	H2 – V	0.00
CB	H1 – H2	-0.02
	H1 – V	0.02
	H2 – V	0.00

Table 3.7.1-219 Matched Time History (Outcrop Motions) Parameters

Response Spectrum	Component	Record Parameters					
		PGA (g)	PGV (cm/sec)	PGD (cm)	Duration (sec)	PGV/PGA (cm/sec/g)	PGAxPGD/ $(PGV)^2$
RB/FB Enhanced SCOR FIRS	Horizontal 1	0.24	17.80	12.11	24.65	74.03	9.01
	Horizontal 2	0.24	17.28	12.57	29.21	73.50	9.71
	Vertical	0.24	14.06	9.38	30.89	58.17	11.25
CB Enhanced SCOR FIRS	Horizontal 1	0.24	18.63	12.07	24.52	77.24	8.23
	Horizontal 2	0.23	16.03	12.16	29.15	69.92	10.65
	Vertical	0.24	14.21	8.78	31.05	58.84	10.31

Note:

PGA – Peak ground acceleration (100 Hz)

PGV – Peak ground velocity

PGD – Peak ground displacement

Duration is defined as the time interval between the time history points at which 5 and 75 percent of the normalized Arias intensity (total energy measure) has been recorded.

Table 3.7.1-220 Cumulative Power below 50 Hz for In-Column Acceleration Time Histories with and without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock

Structure	Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock	In-Column Time History Component	Cumulative Power Below 50 Hz – Deterministic Profile BE	LB	UB
RB/FB	Without Backfill	Vertical	92%	94%	91%
		Horizontal – H1	100%	99%	99%
		Horizontal – H2	99%	98%	98%
	With Backfill	Vertical	93%	95%	91%
		Horizontal – H1	100%	99%	99%
		Horizontal – H2	99%	98%	98%
CB	Without Backfill	Vertical	88%	89%	88%
		Horizontal – H1	96%	97%	95%
		Horizontal – H2	93%	94%	92%
	With Backfill	Vertical	90%	91%	92%
		Horizontal – H1	96%	97%	96%
		Horizontal – H2	93%	94%	93%

Figure 3.7.1-201 Shear Wave Velocity Profiles for Site Response Analysis: Intermediate Range, Lower Range, and Upper Range Values

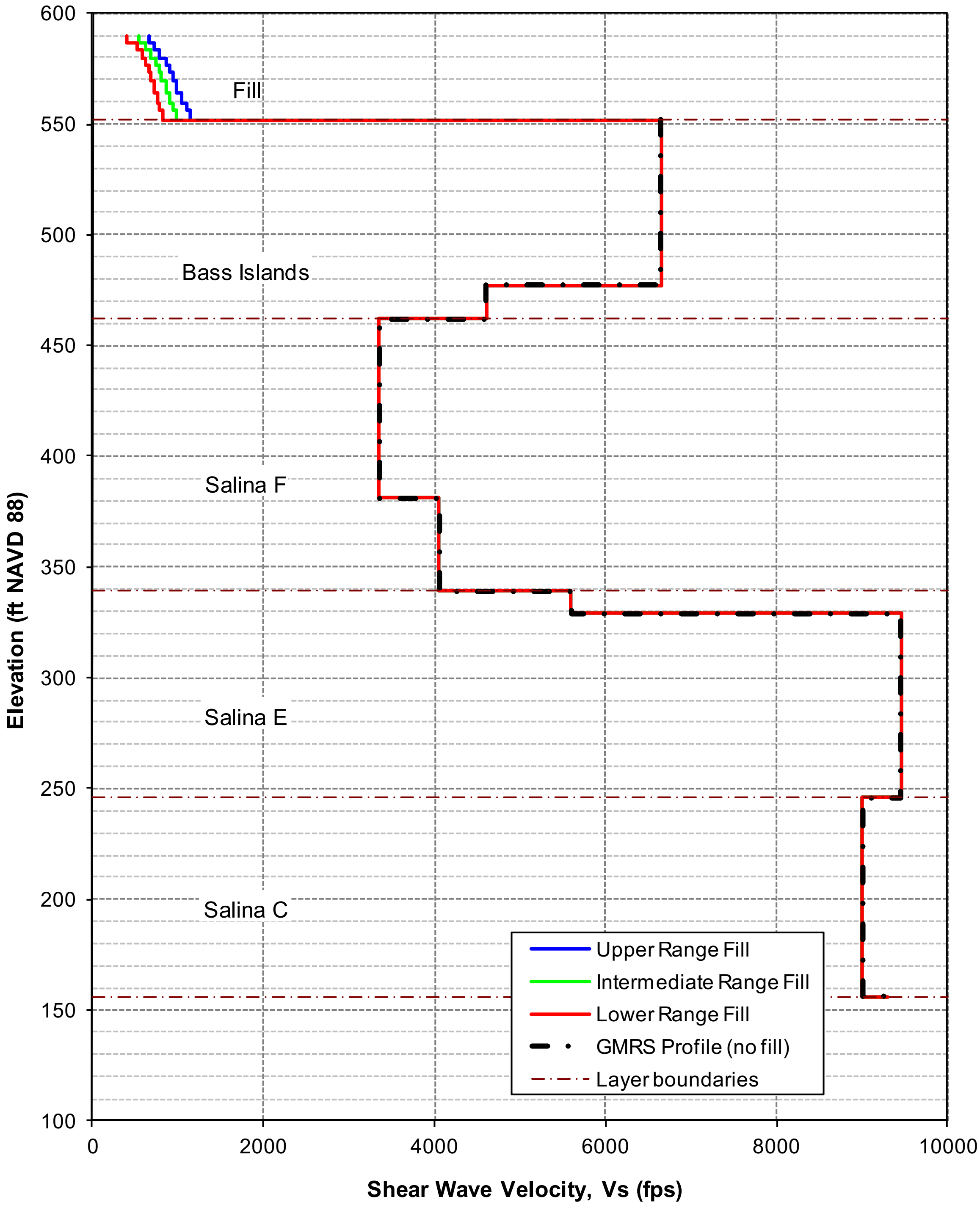


Figure 3.7.1-202 Shear Wave Velocity Profiles for Site Response Analysis: FWSC Shear Wave Velocity Profile

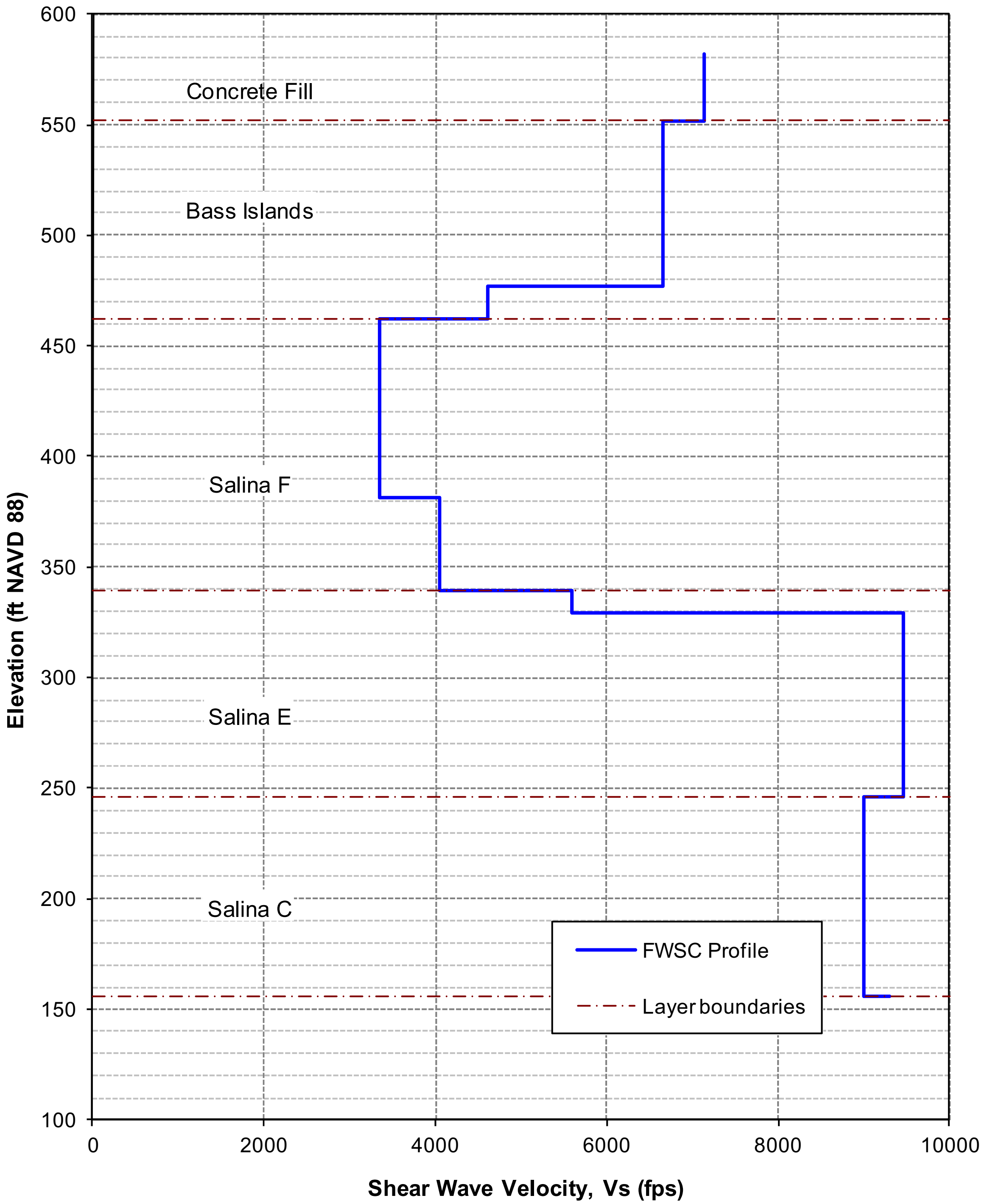


Figure 3.7.1-203 Modulus Reduction and Damping Relationships Used for the Engineered Granular Backfill Material

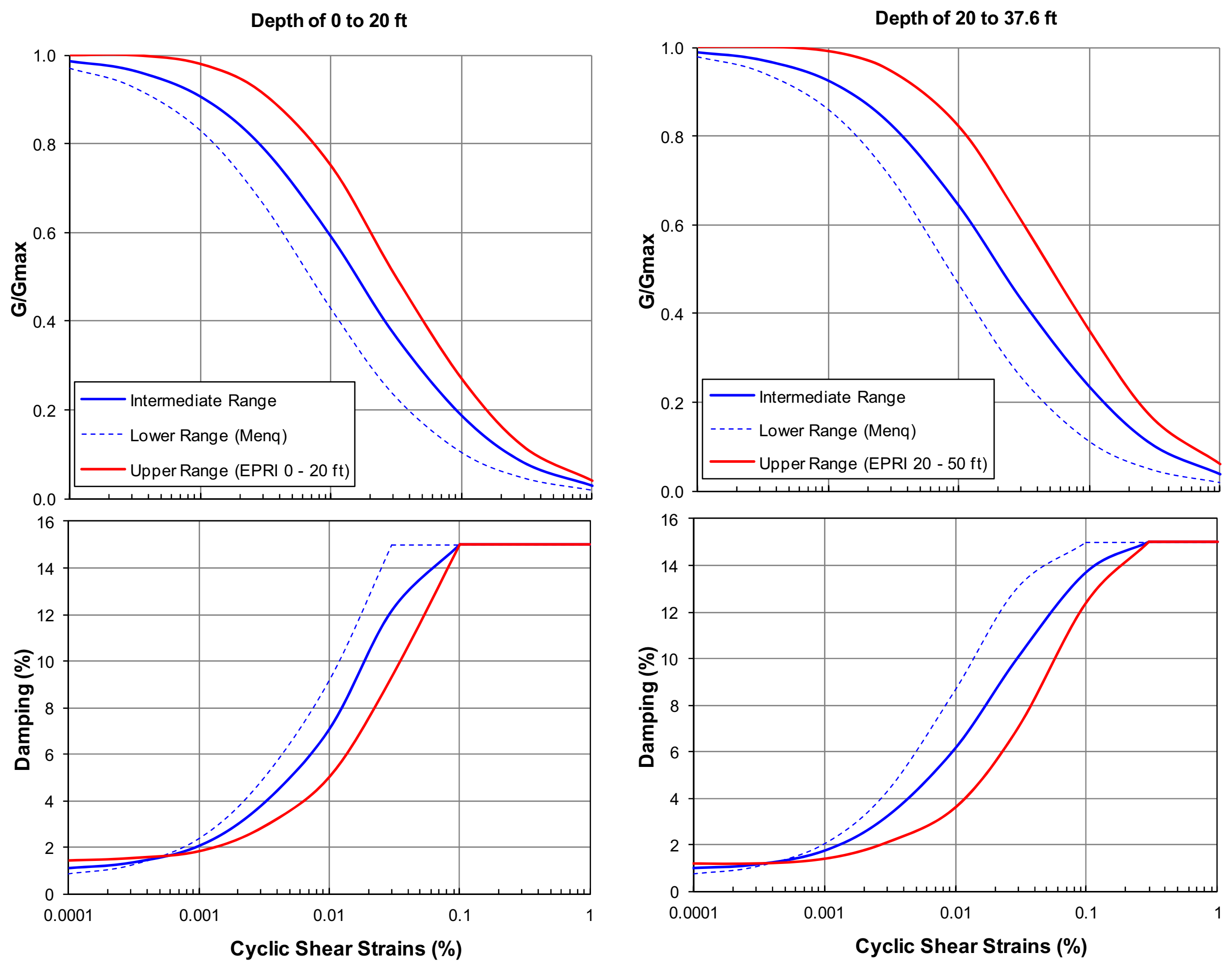


Figure 3.7.1-204 Randomized Shear Wave Velocity Profiles 1-30 for the Intermediate Range Site Response Analysis Profile

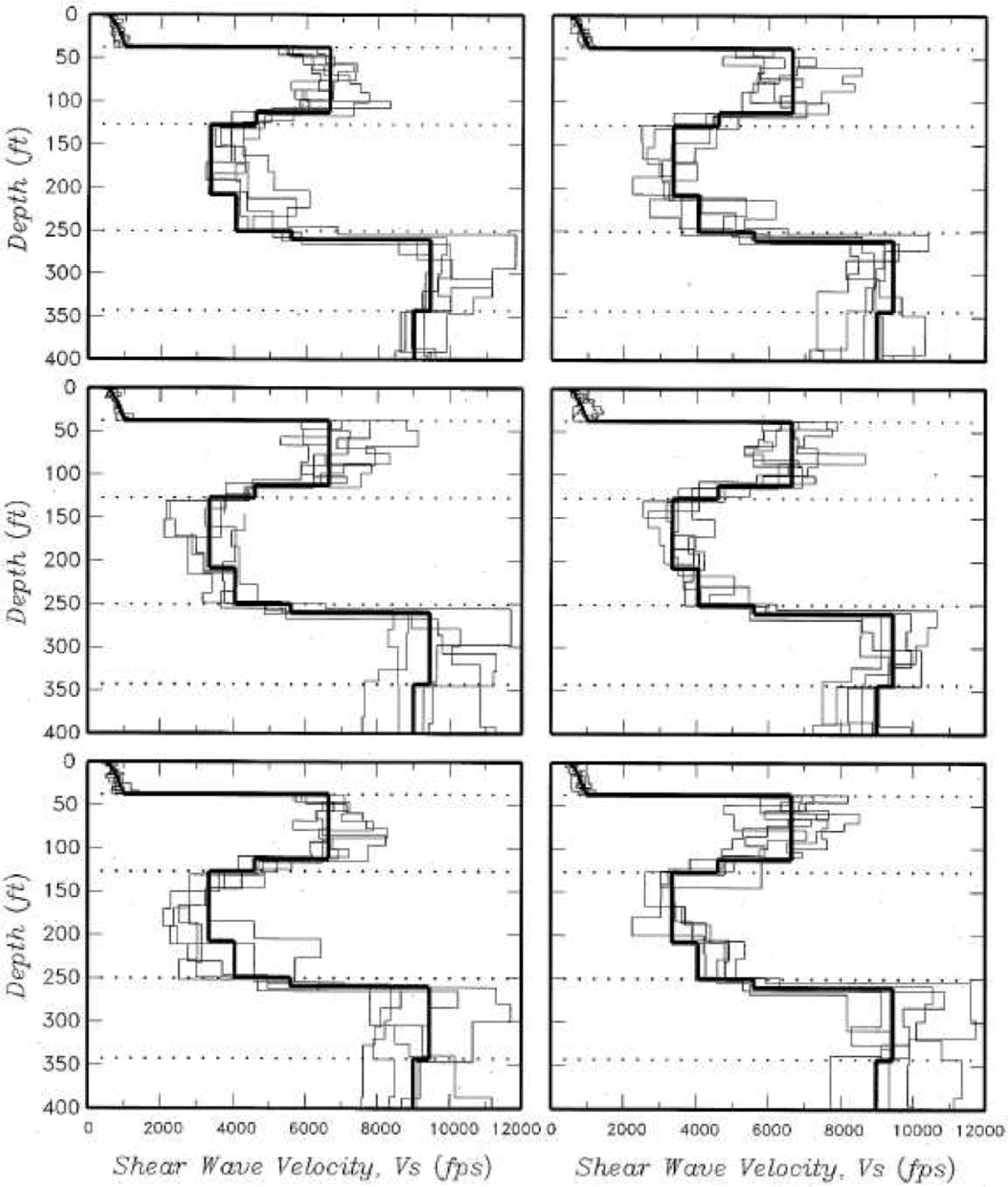


Figure 3.7.1-205 Randomized Shear Wave Velocity Profiles 31-60 for the Intermediate Range Site Response Analysis Profile

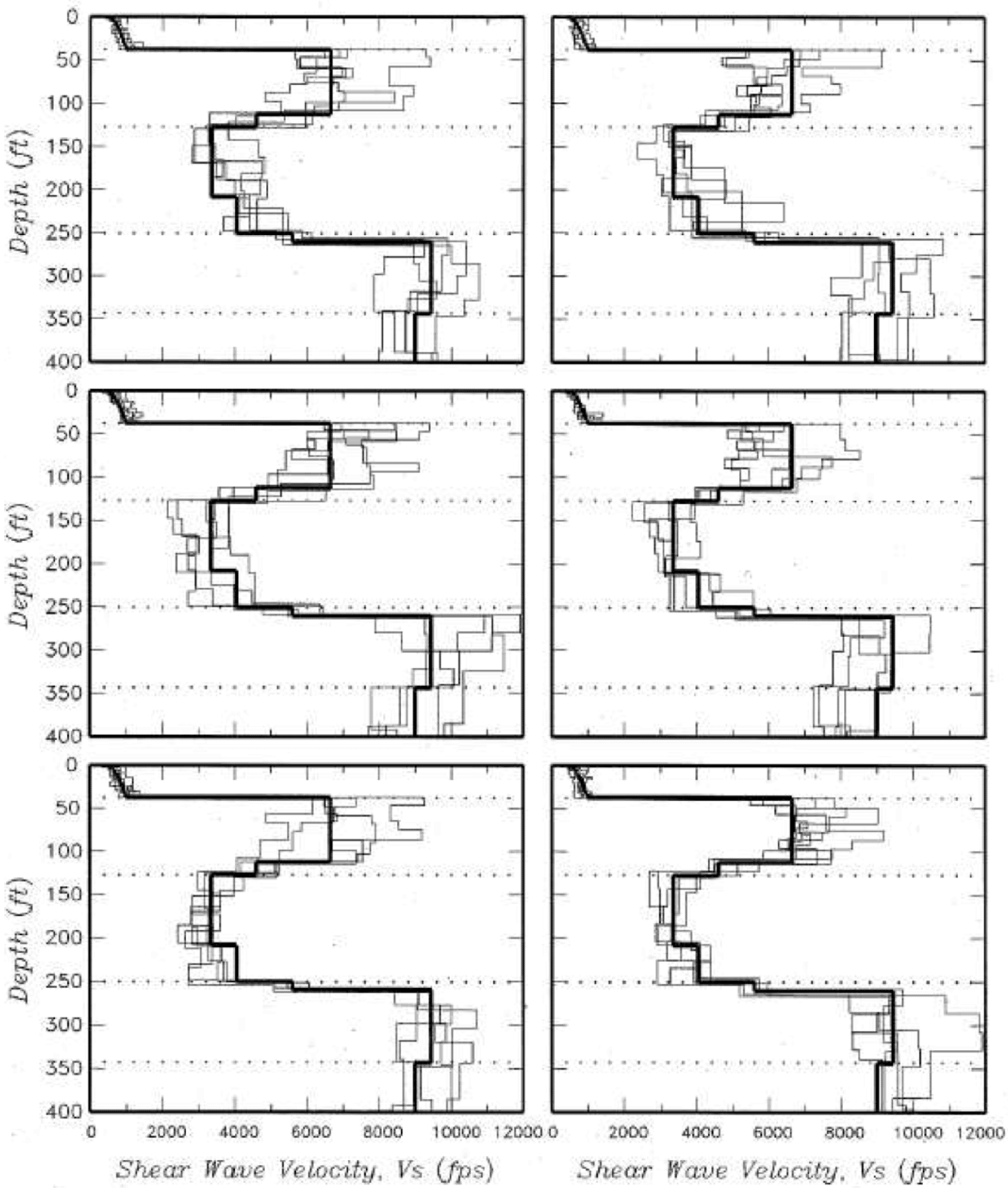


Figure 3.7.1-206 Statistics of Randomized Shear Wave Velocity Profiles for the Intermediate Range Site Response Analysis Profile

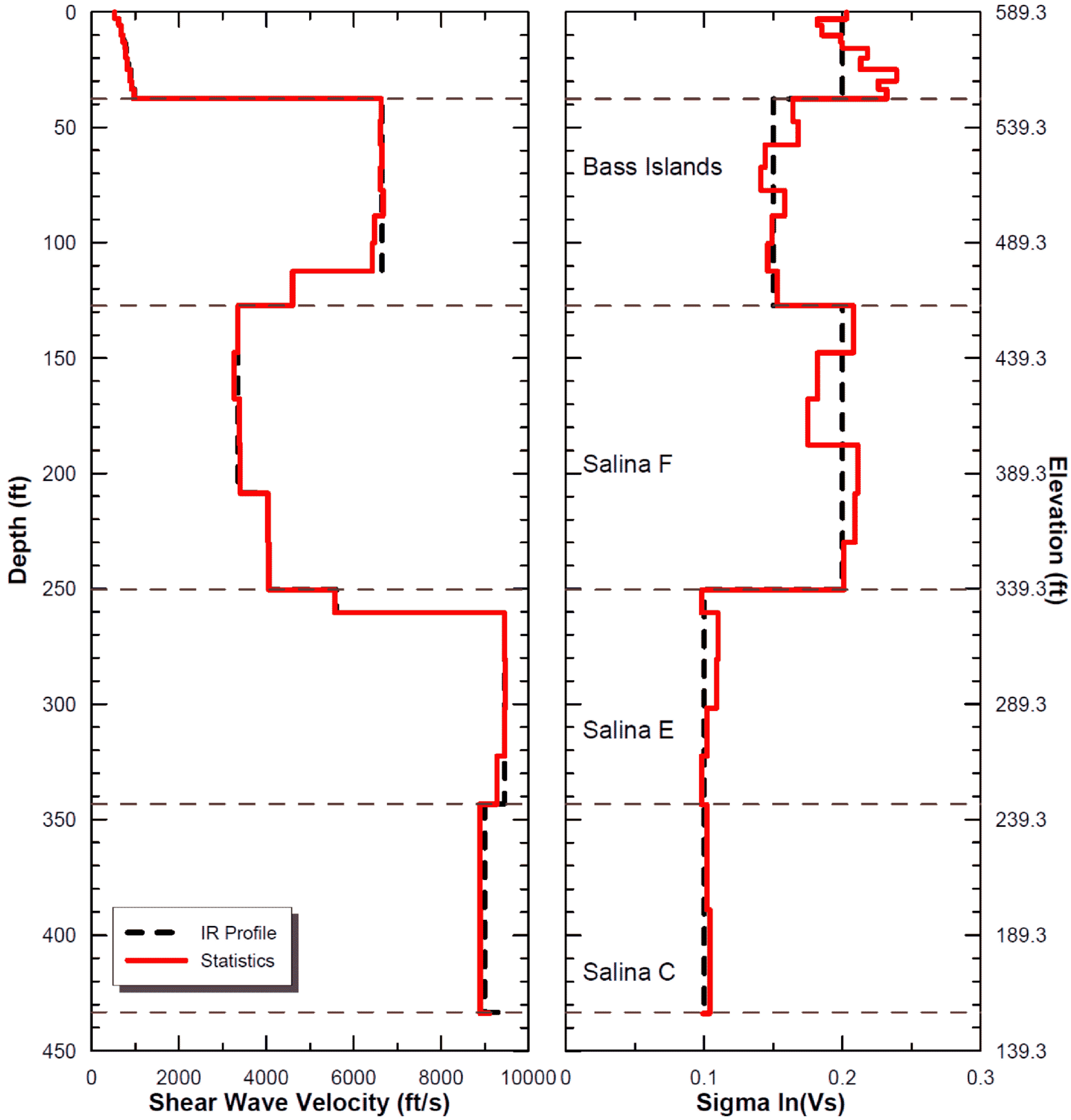
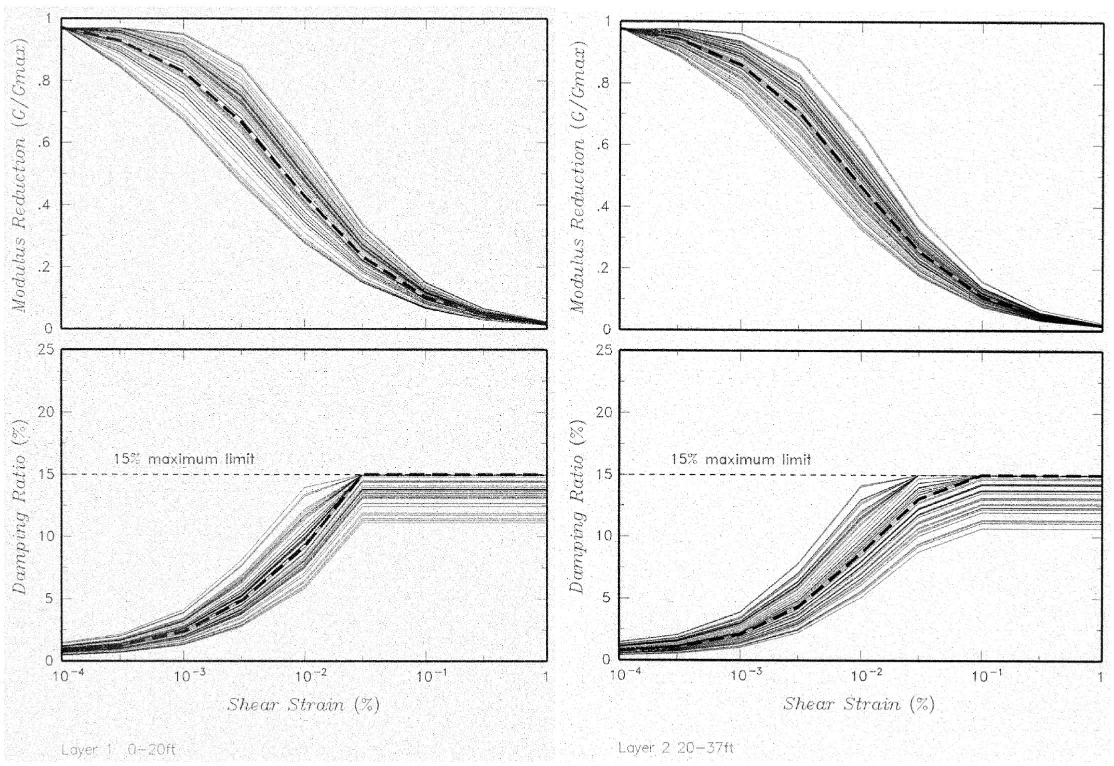


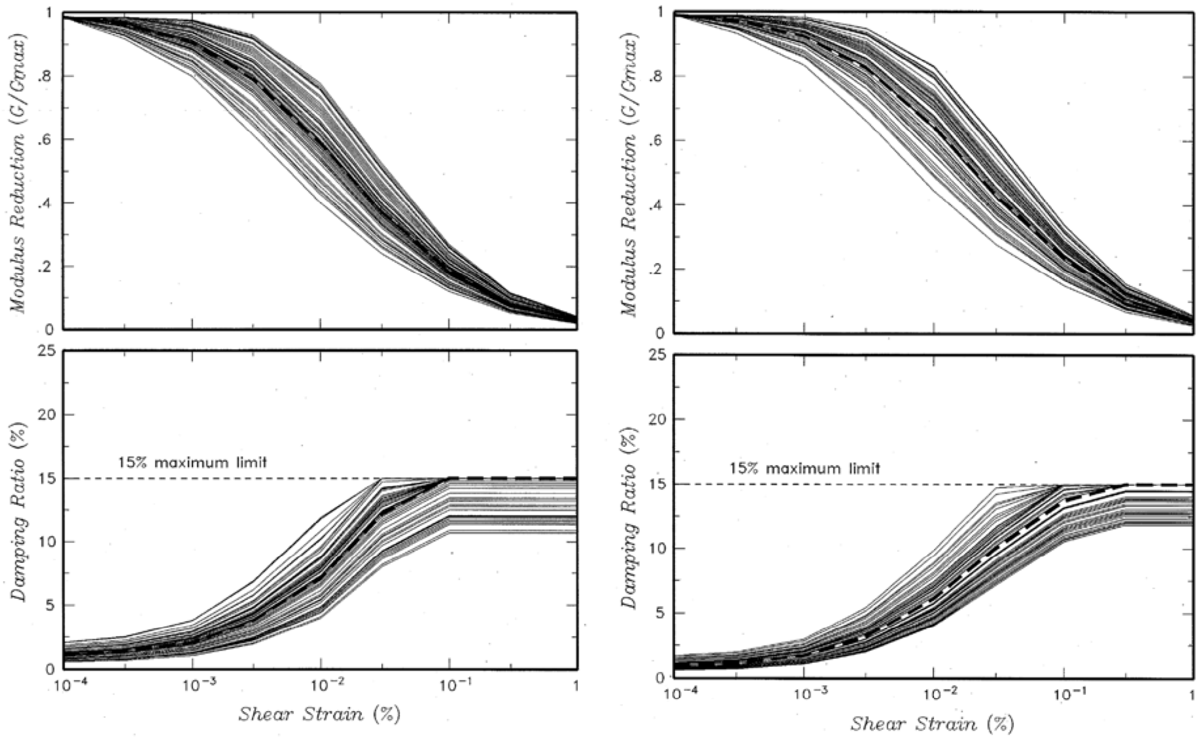
Figure 3.7.1-207 Randomized Shear Modulus Reduction and Damping Relationships Used for LR Engineered Granular Backfill Material



0 to 20 feet depth

20 to 37.6 feet depth

Figure 3.7.1-208 Randomized Shear Modulus Reduction and Damping Relationships Used for IR Engineered Granular Backfill Material



0 to 20 feet depth

20 to 50 feet depth

Figure 3.7.1-209 Randomized Shear Modulus Reduction and Damping Relationships Used for UR Engineered Granular Backfill Material

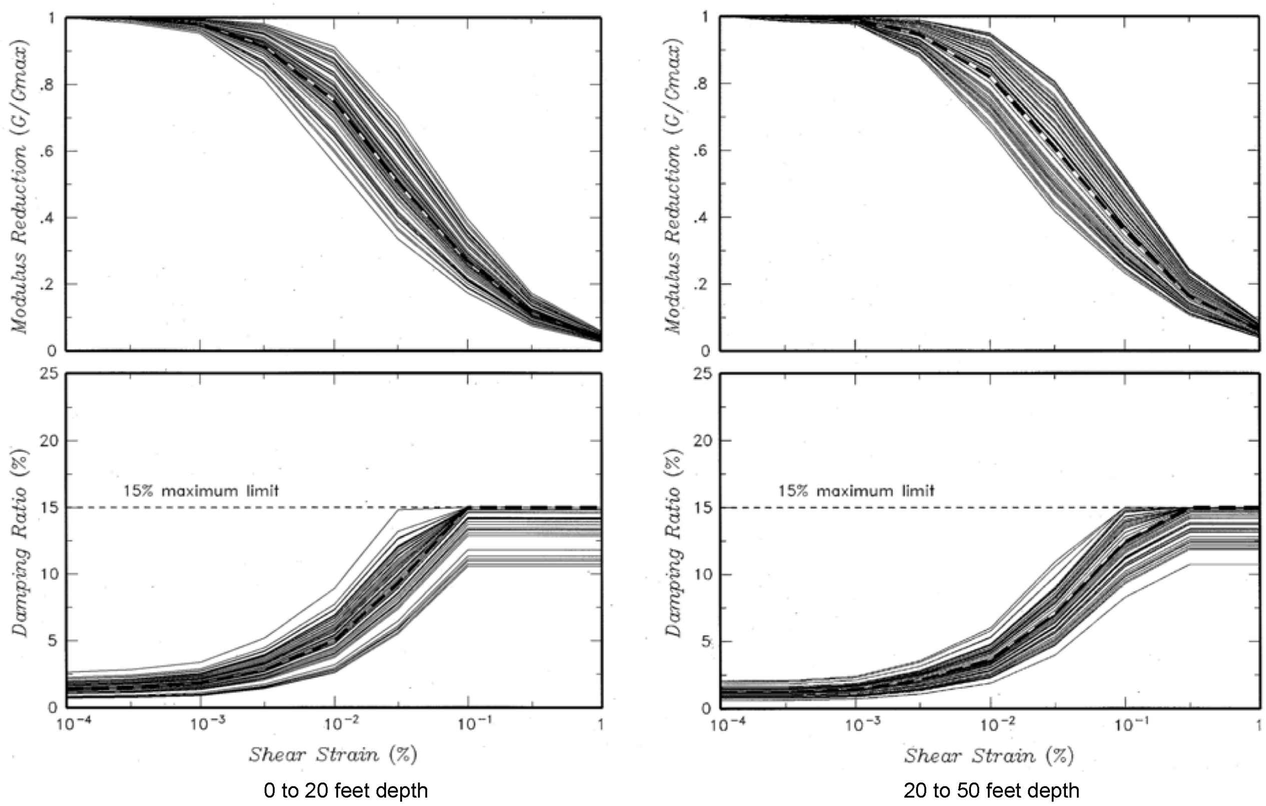


Figure 3.7.1-210 Site Response Logic Tree for Full Soil Column Profile

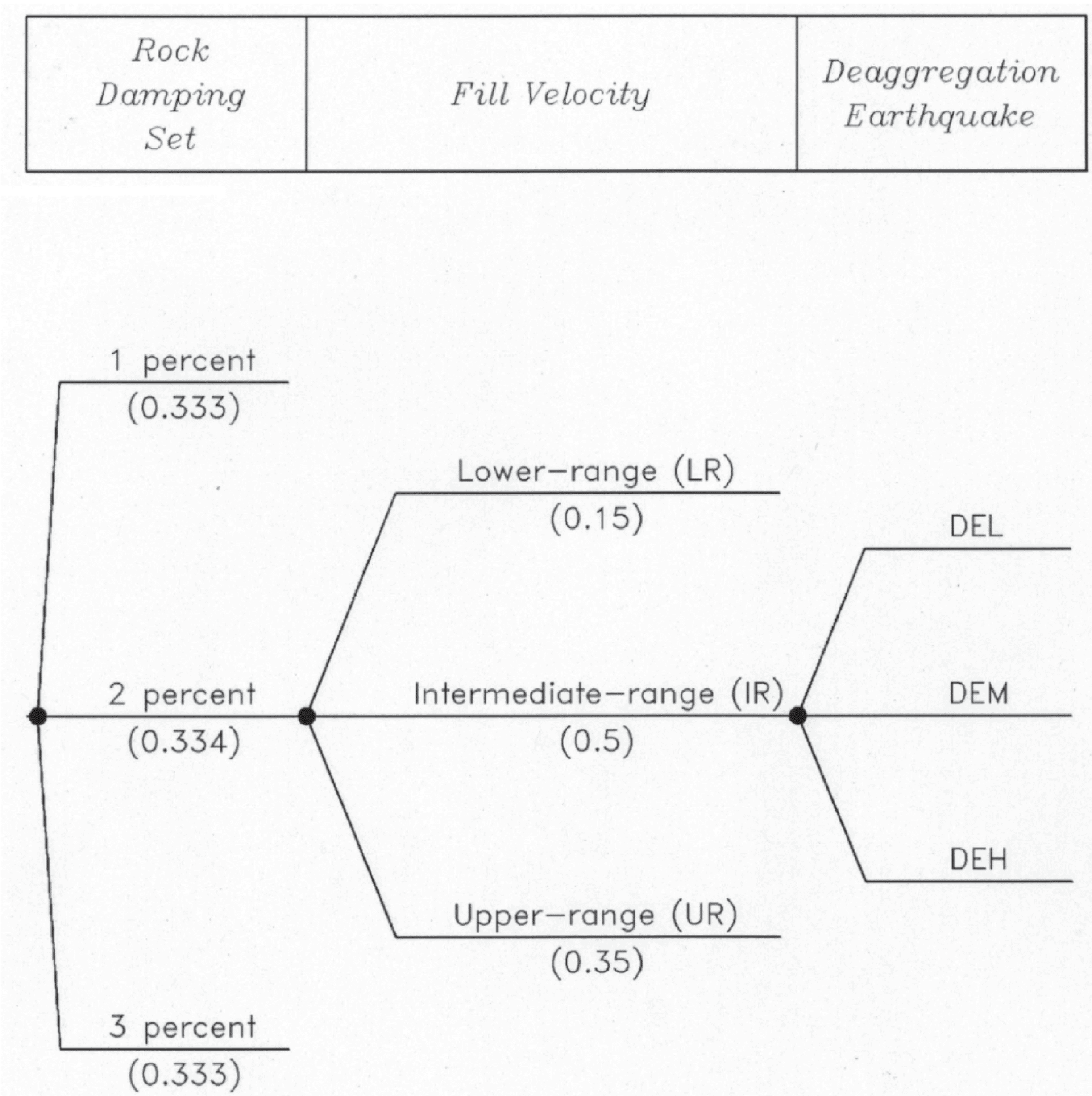


Figure 3.7.1-211 PBSRS Amplification Functions for the Fermi 3 Site

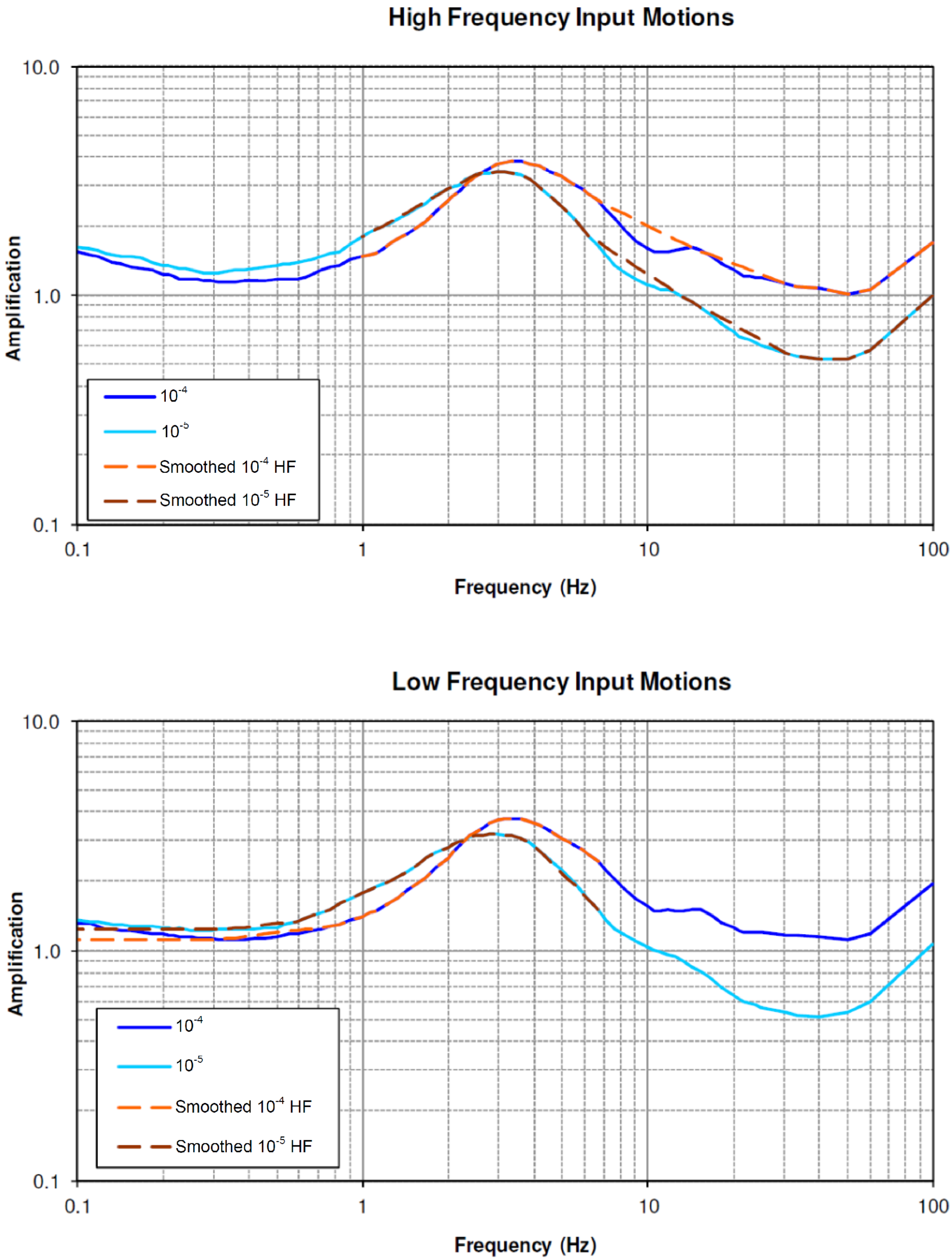
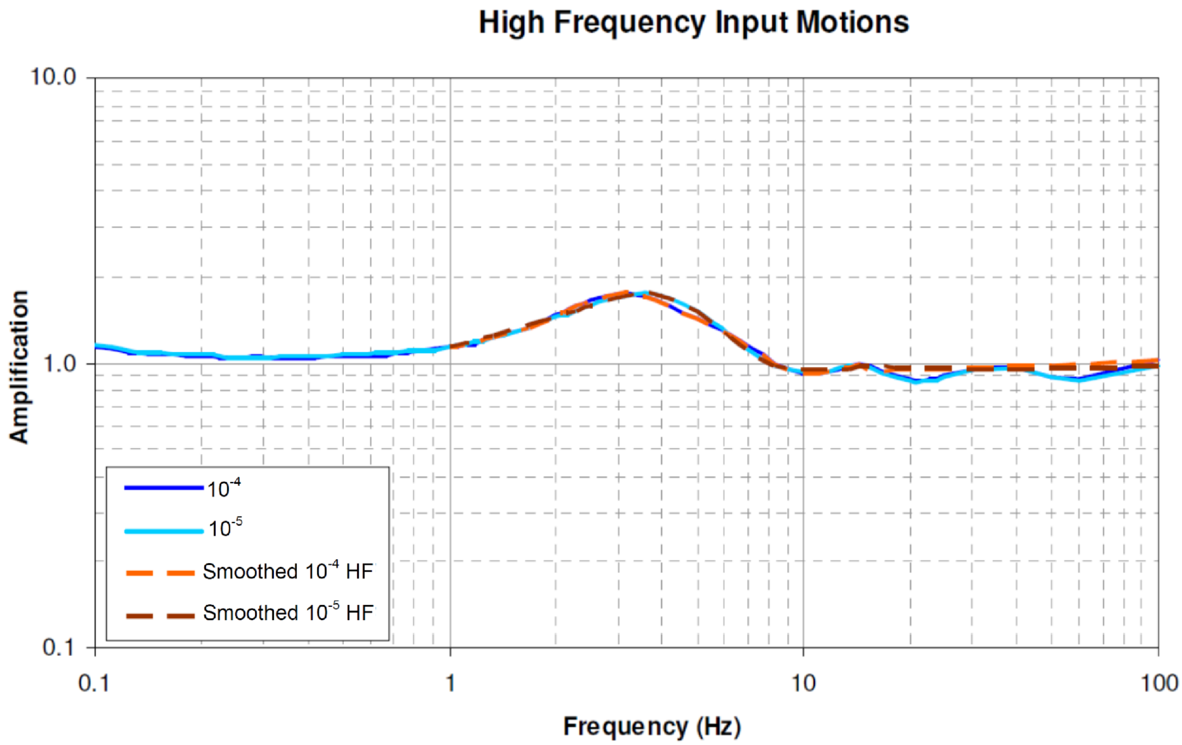


Figure 3.7.1-212 RB/FB SCOR FIRS Amplification Functions for the Fermi 3 Site



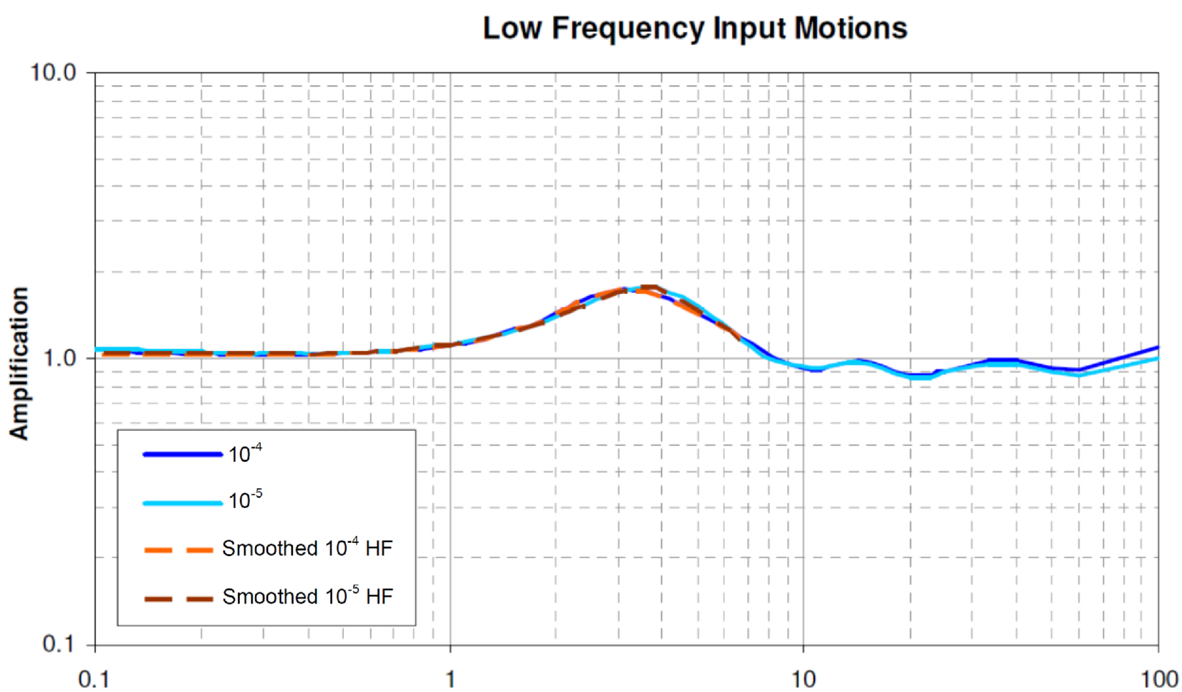
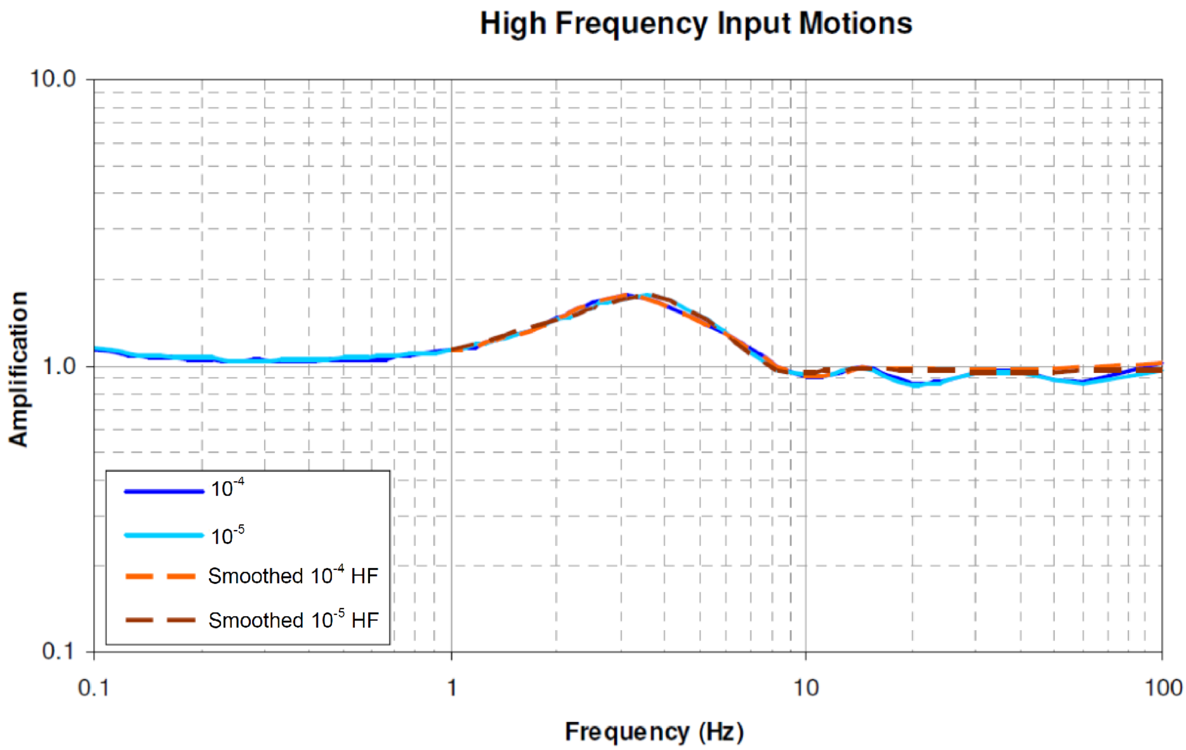


Figure 3.7.1-213 CB SCOR FIRS Amplification Functions for the Fermi 3 Site



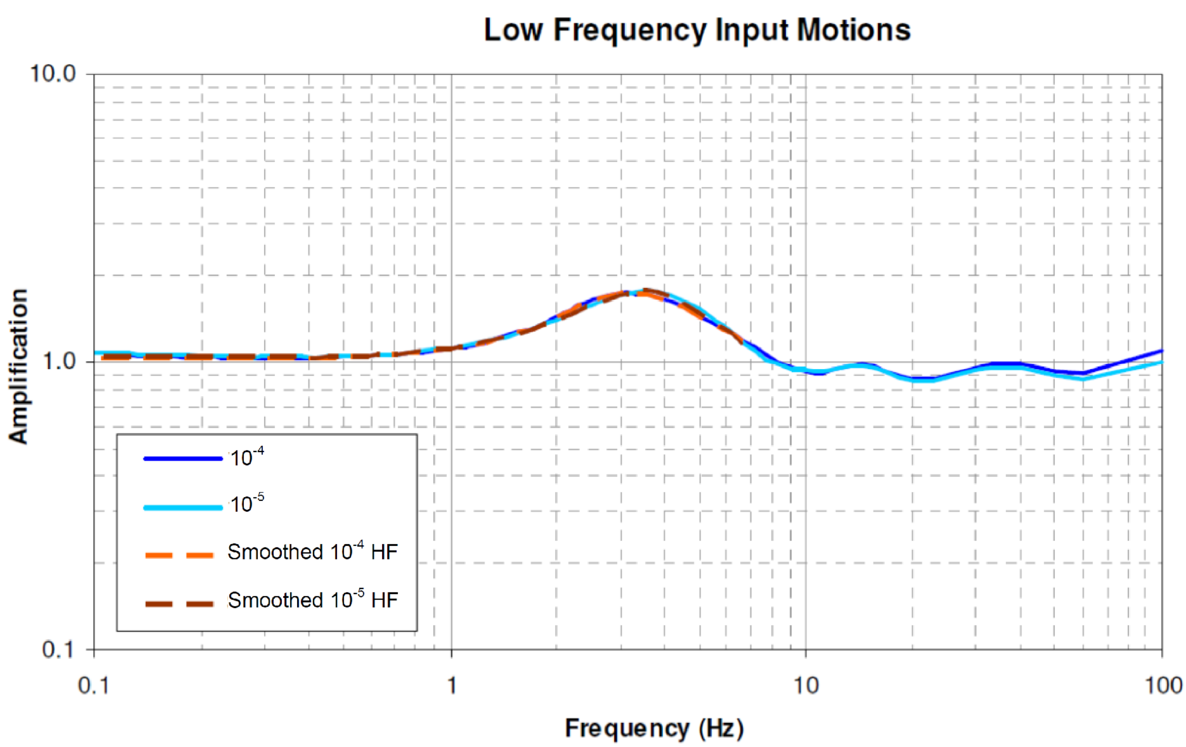


Figure 3.7.1-214 Example of FWSC 2D/1D Response Spectral Ratios for Fill Concrete Based on the 10^{-4} and 10^{-5} Exccedance Levels of Input Ground Motion

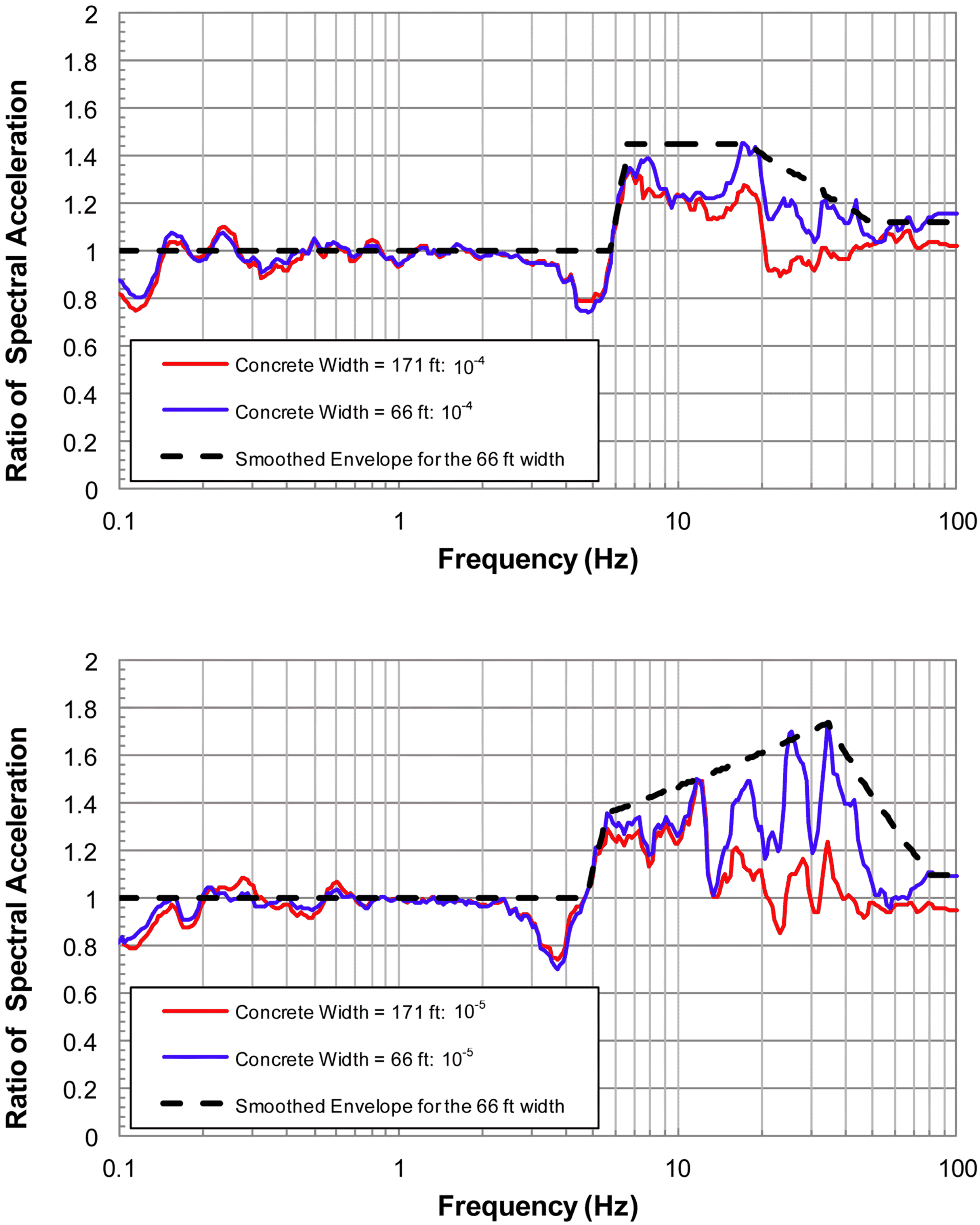


Figure 3.7.1-215 FWSC 2D/1D Response Spectral Ratios for Fill Concrete

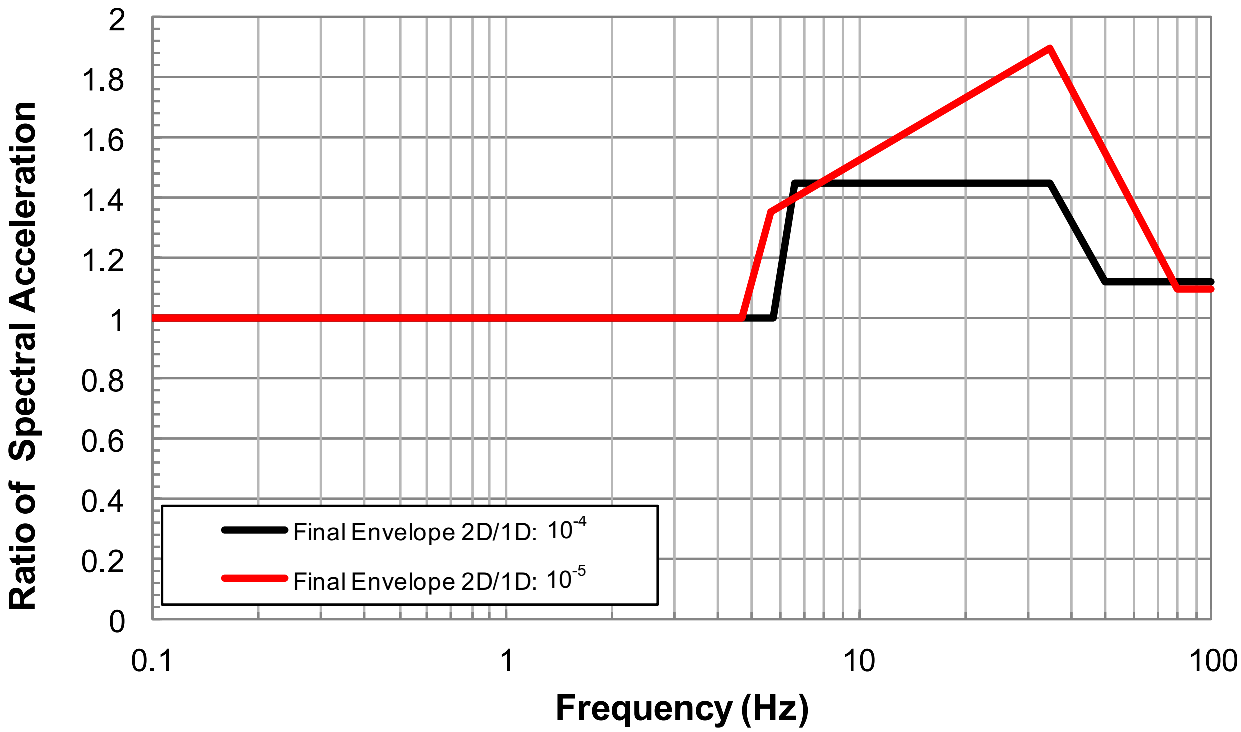


Figure 3.7.1-216 FWSC FIRS Amplification Functions for the Fermi 3 Site

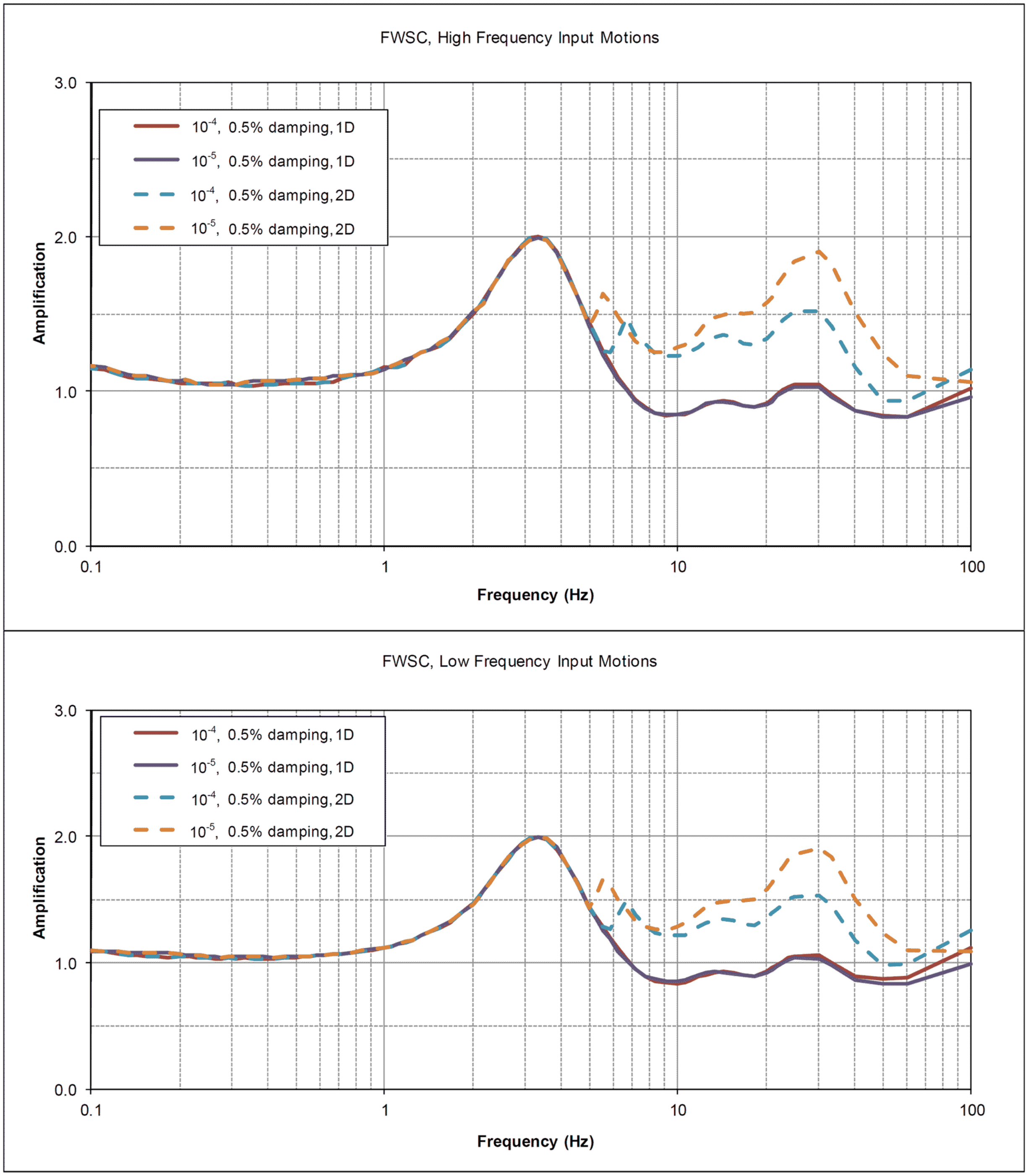


Figure 3.7.1-217 Development of 10^{-4} Surface UHRS at the Finished Ground Level Grade for the Full Soil Column Profile

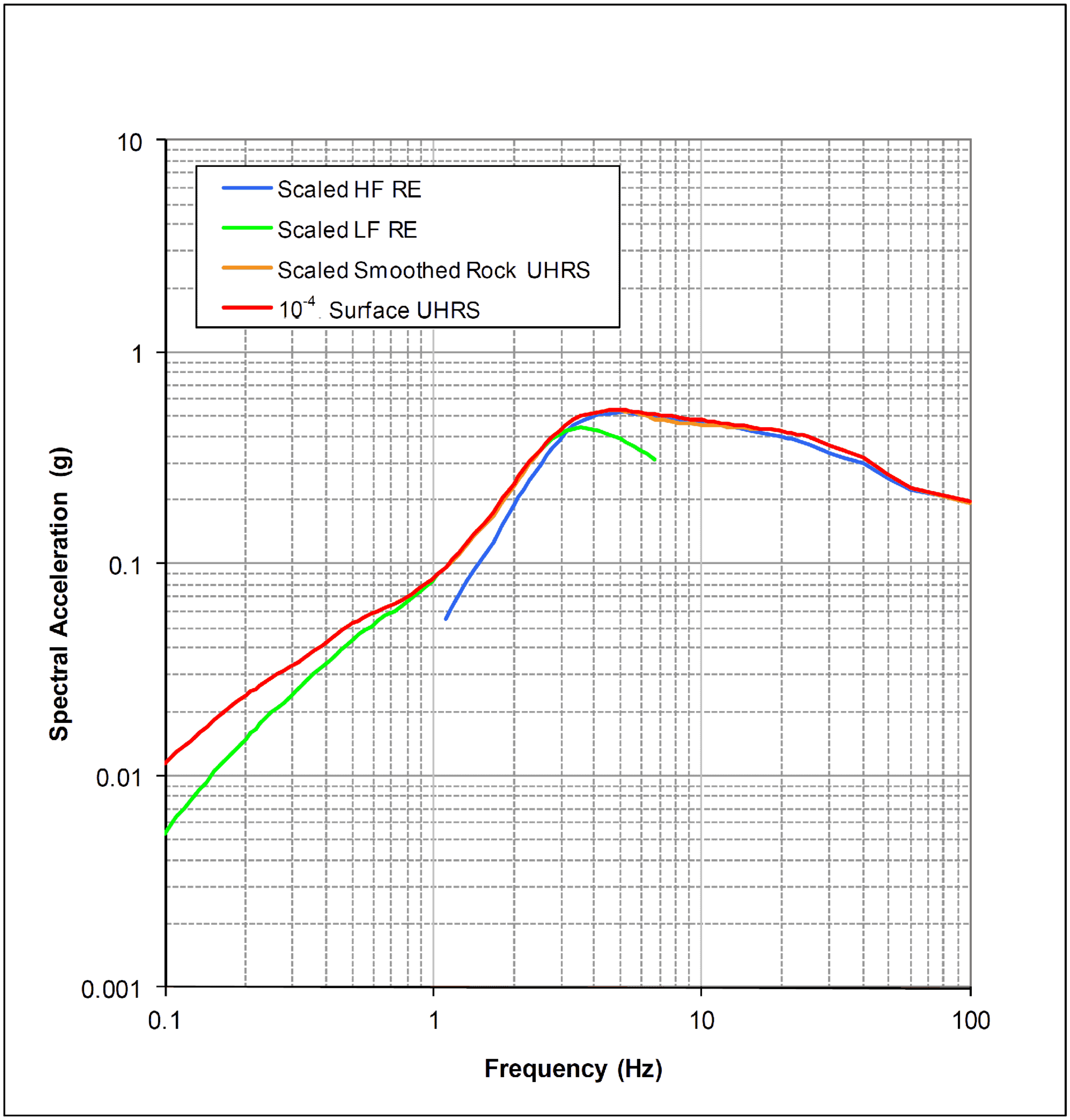


Figure 3.7.1-218 Development of 10^{-4} SCOR UHRS at the RB/FB Foundation Level for the Full Soil Column Profile

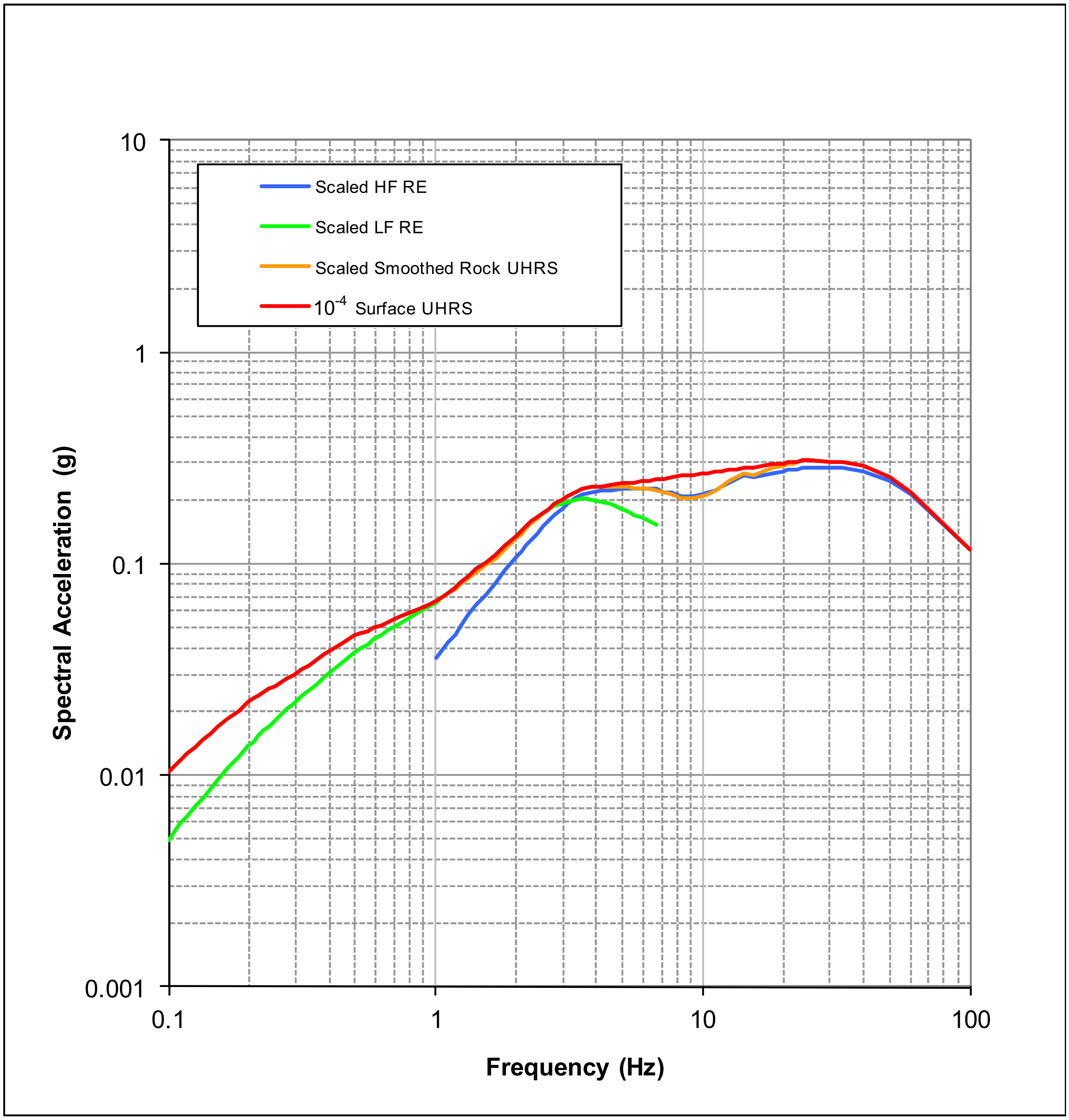


Figure 3.7.1-219 Development of the Horizontal PBSRS for the Fermi 3 Site

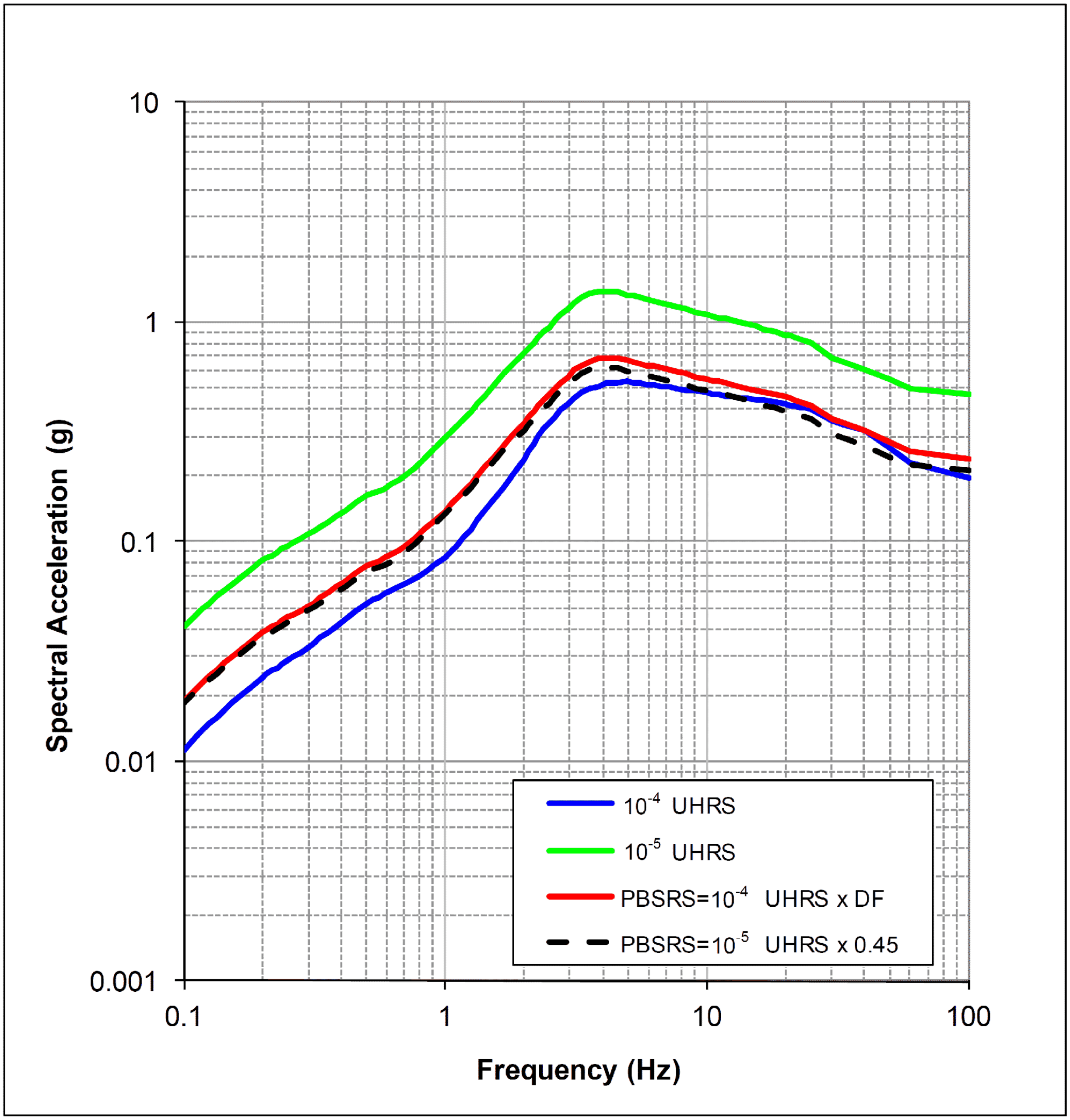


Figure 3.7.1-220 Vertical to Horizontal Spectral ratios Developed for the Fermi 3 Site Full Soil Column Profile

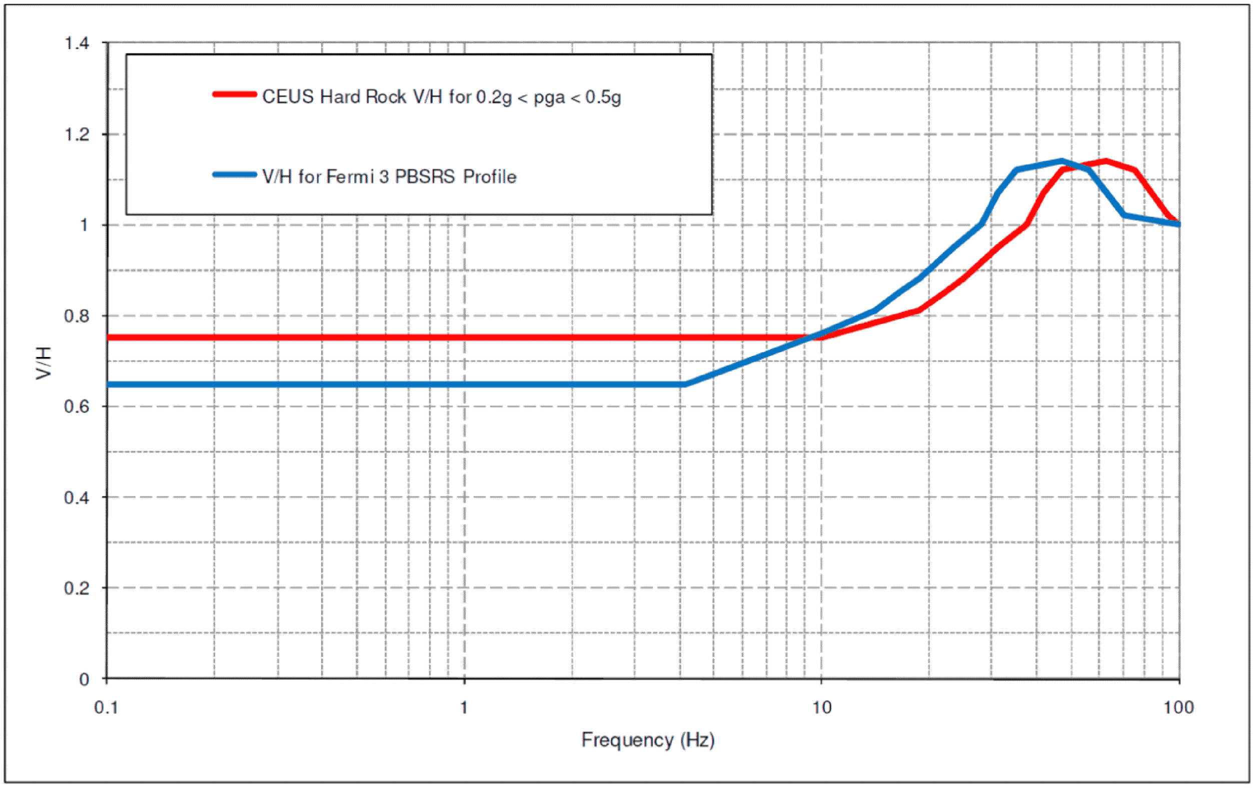


Figure 3.7.1-221 Horizontal and Vertical Fermi 3 PBSRS at Finished Ground Level Grade (5 Percent Damping)

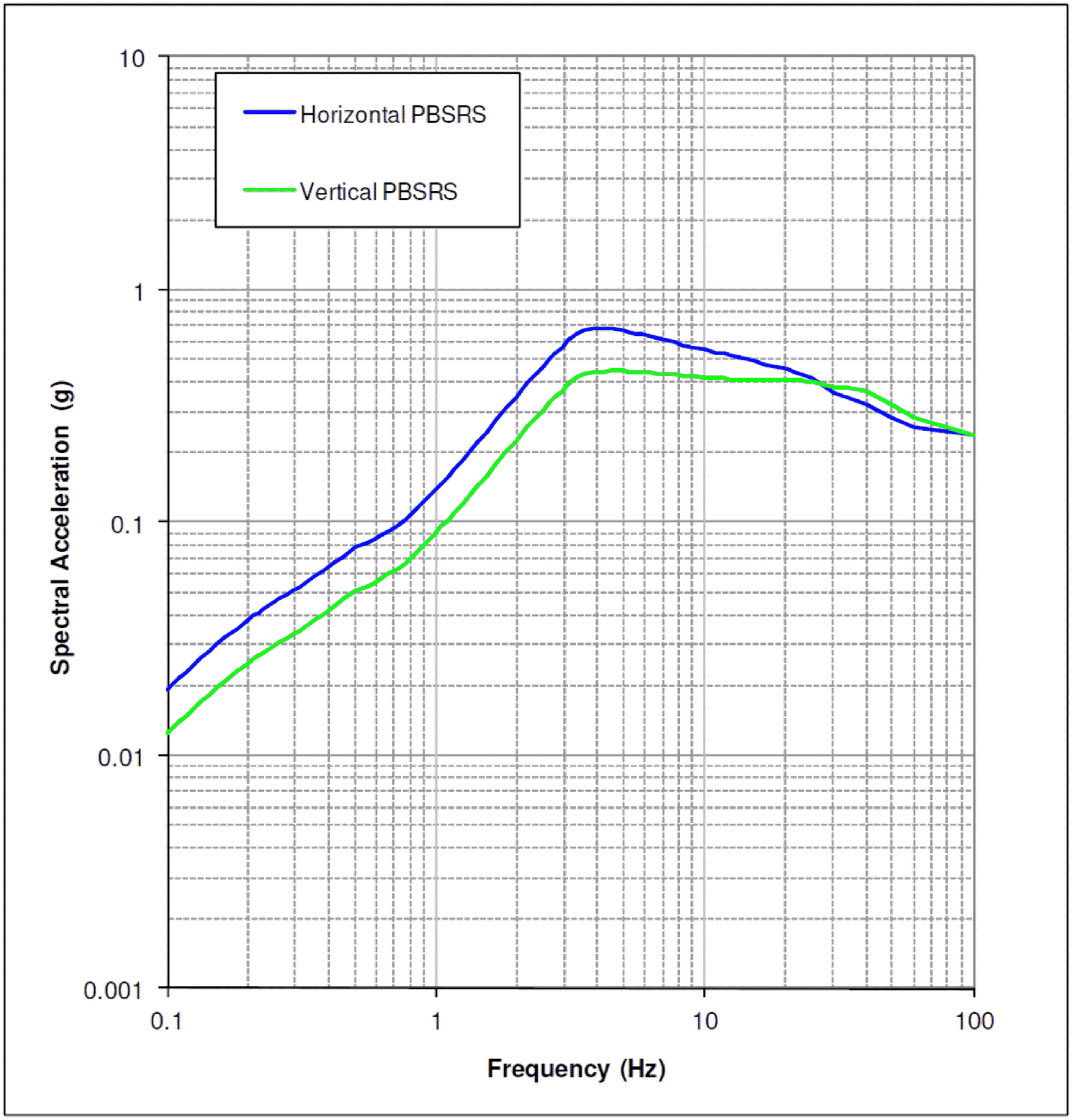


Figure 3.7.1-222 Deterministic Shear Wave Velocity Profiles for the Full Soil Column with Engineered Granular Backfill Above the Top of the Bass Islands Group Bedrock

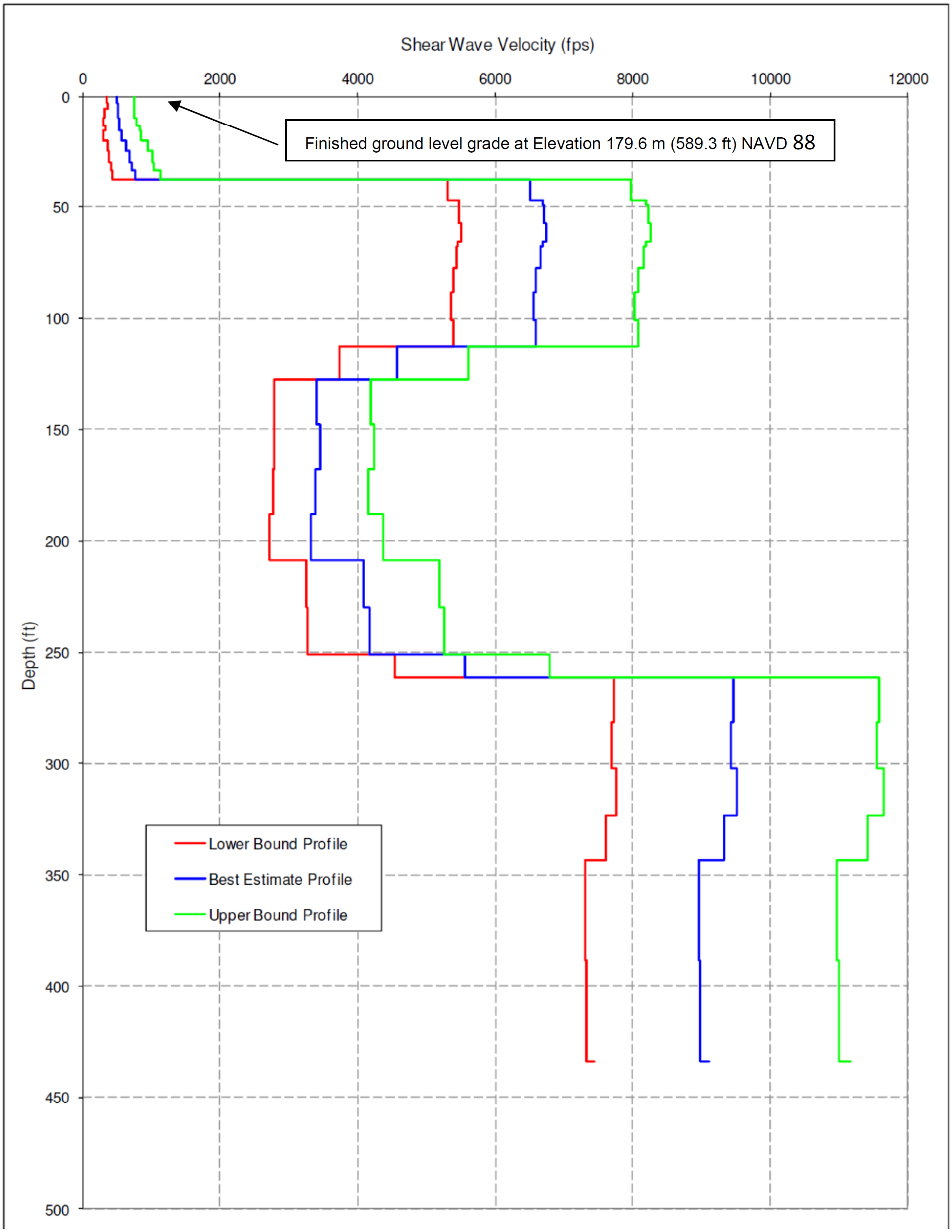


Figure 3.7.1-223 Deterministic Shear Wave Velocity Profiles for the Soil Column without Engineered Granular Backfill Above the Top of the Bass Islands Group Bedrock

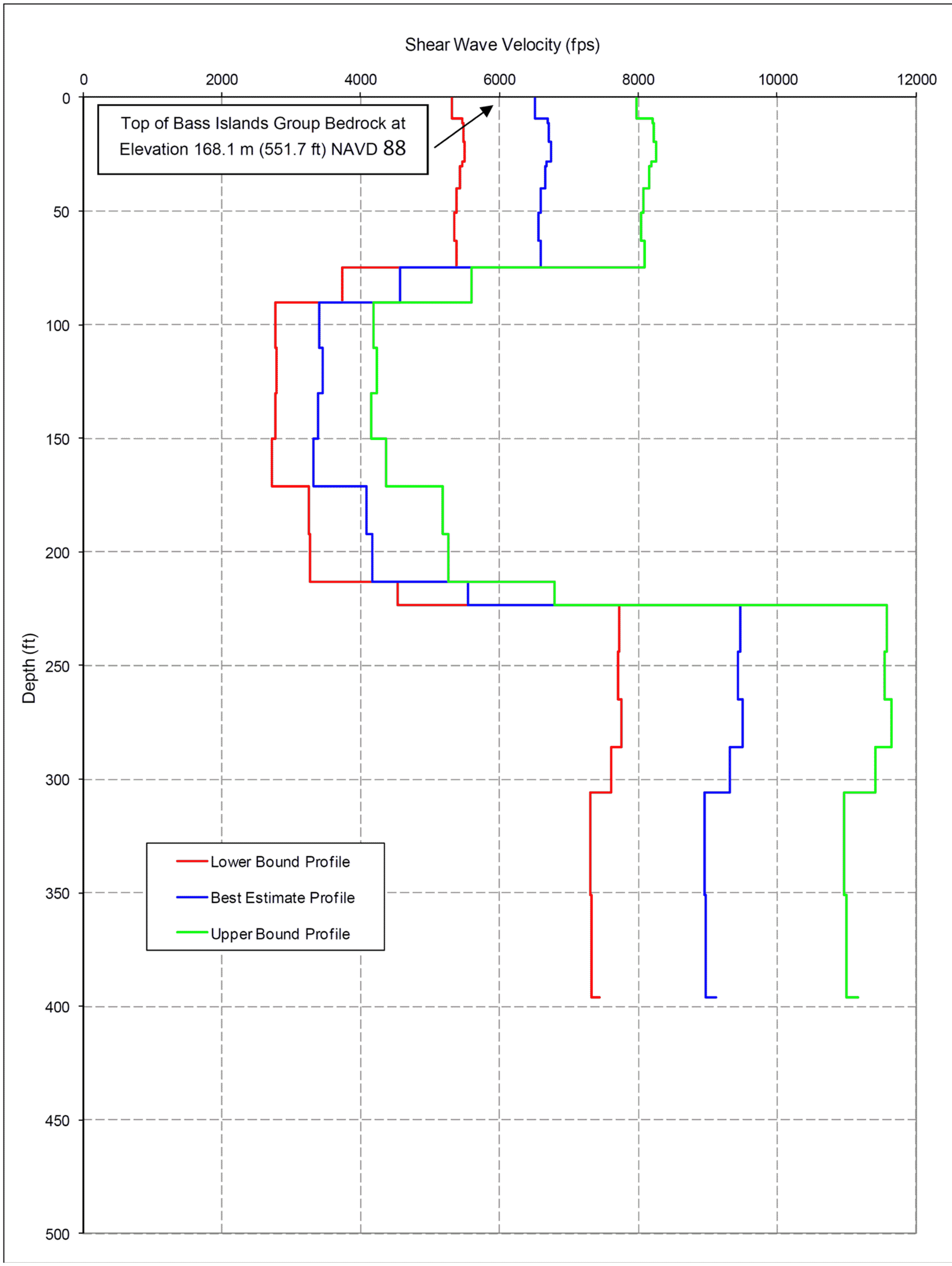


Figure 3.7.1-224 Fermi 3 RB/FB SCOR FIRS (5 Percent Damping)

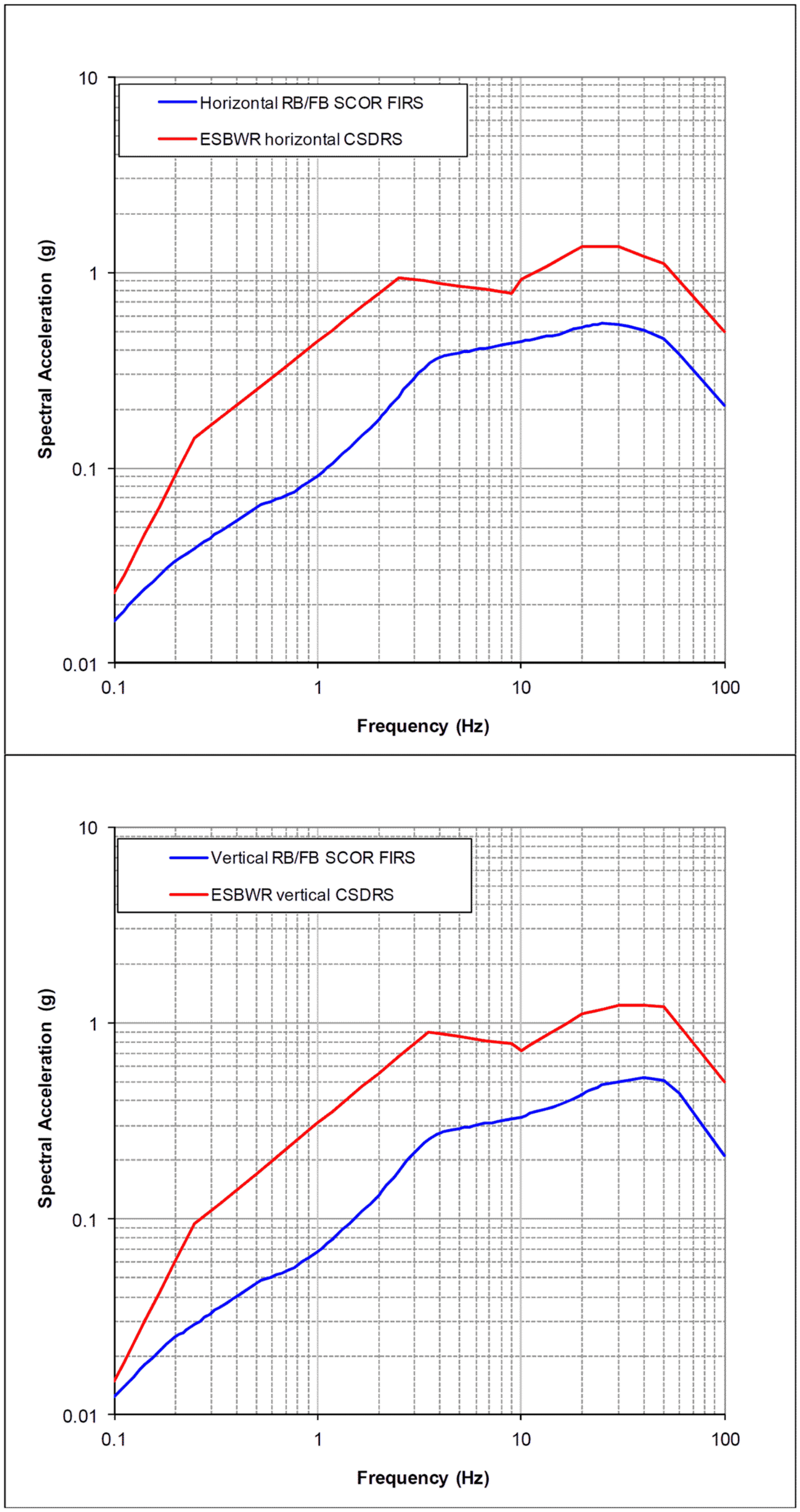


Figure 3.7.1-225 Fermi 3 CB SCOR FIRS (5 Percent Damping)

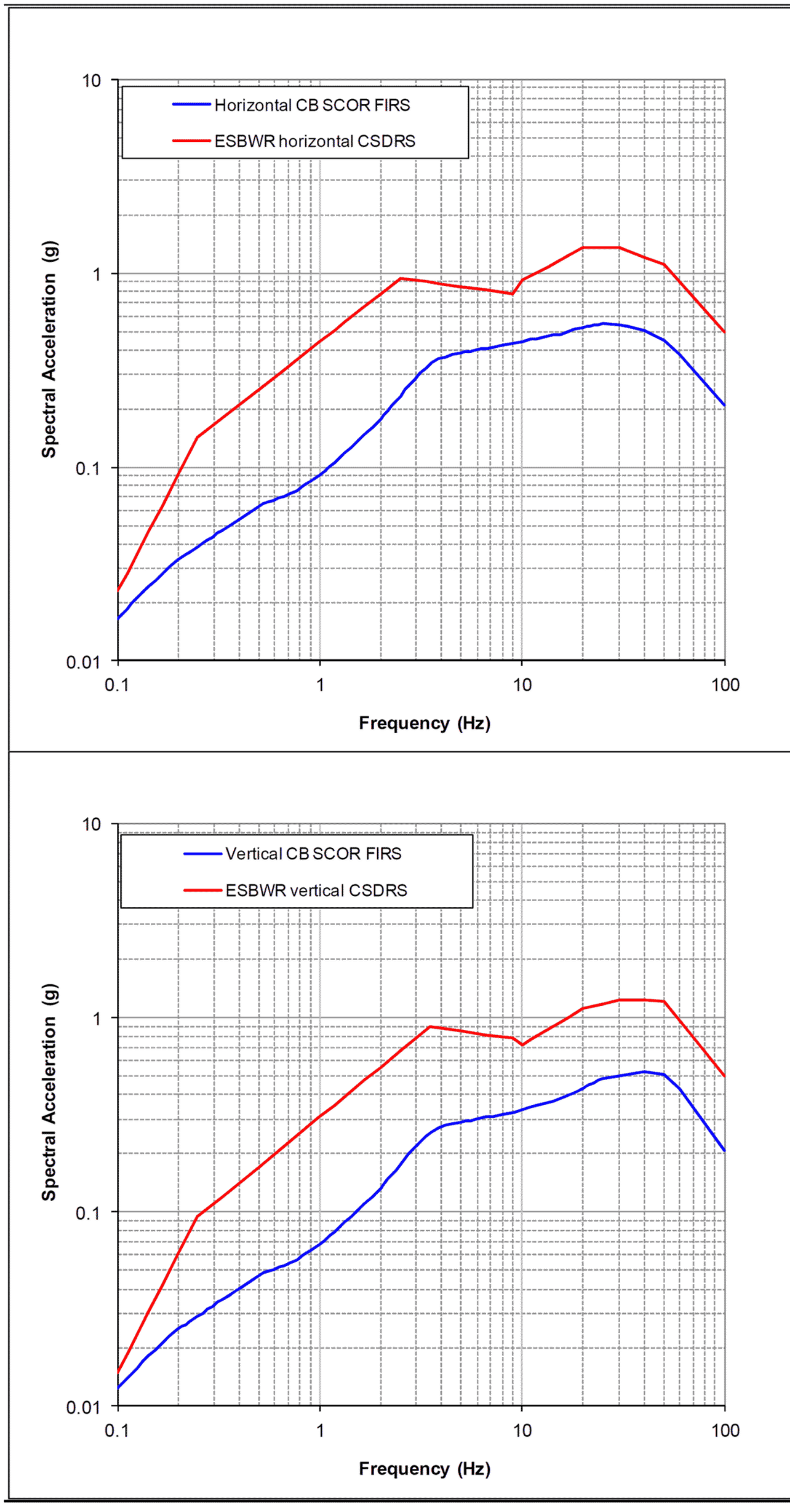


Figure 3.7.1-226 Fermi 3 Horizontal RB/FB SCOR FIRS and Initially Enhanced SCOR FIRS with the NUREG/CR-0098 Median Rock Spectral Shape, Enveloping NUREG/CR-6728 CEUS Spectral Shape, and RG 1.60 Spectral Shape, all Scaled to a Minimum PGA of 0.1 g (5 Percent Damping)

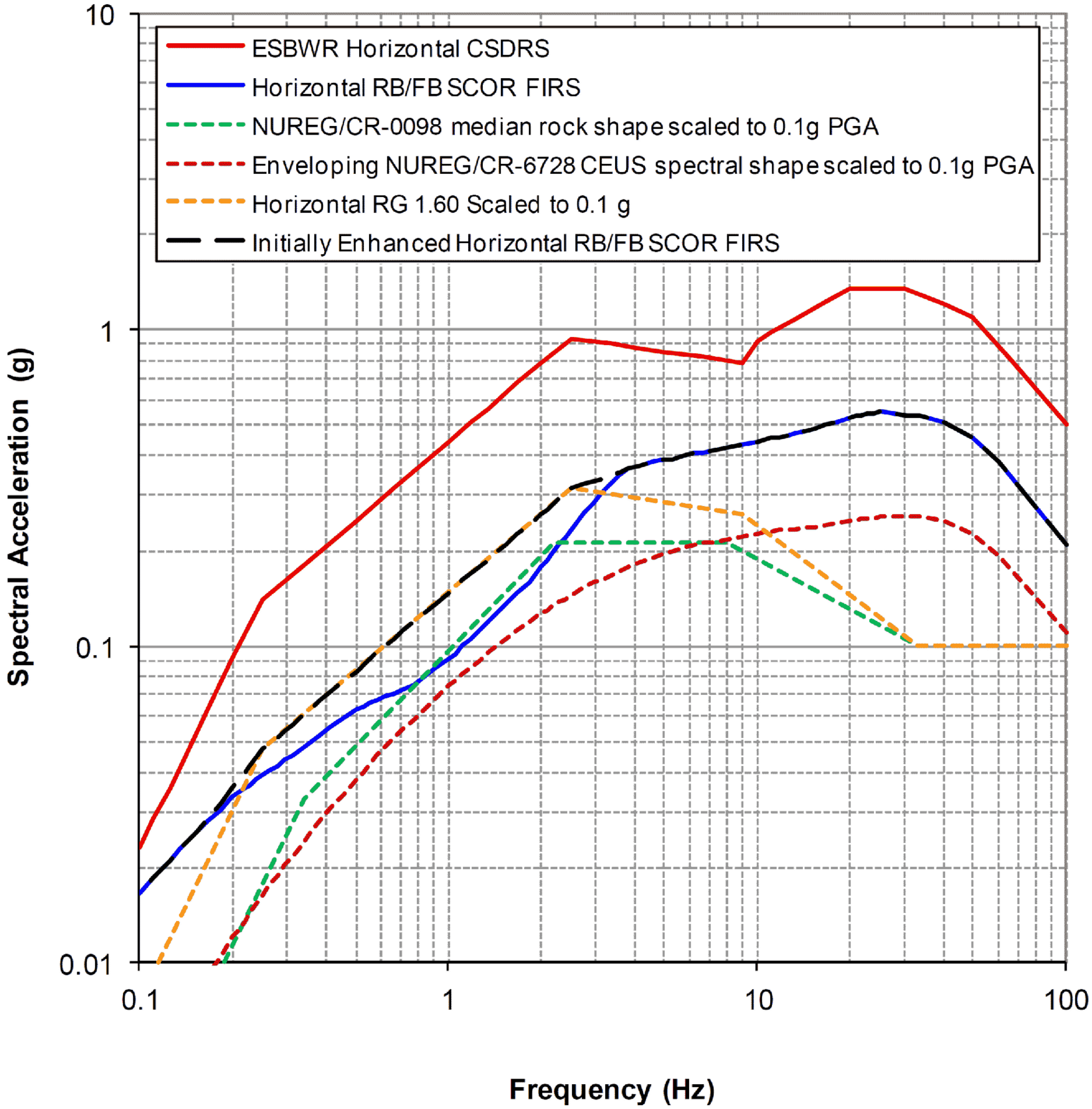


Figure 3.7.1-227 Fermi 3 Horizontal CB SCOR FIRS and Initially Enhanced SCOR FIRS with the NUREG/CR-0098 Median Rock Spectral Shape, Enveloping NUREG/CR-6728 CEUS Spectral Shape, and RG 1.60 Spectral Shape, all Scaled to a Minimum PGA of 0.1 g (5 Percent Damping)

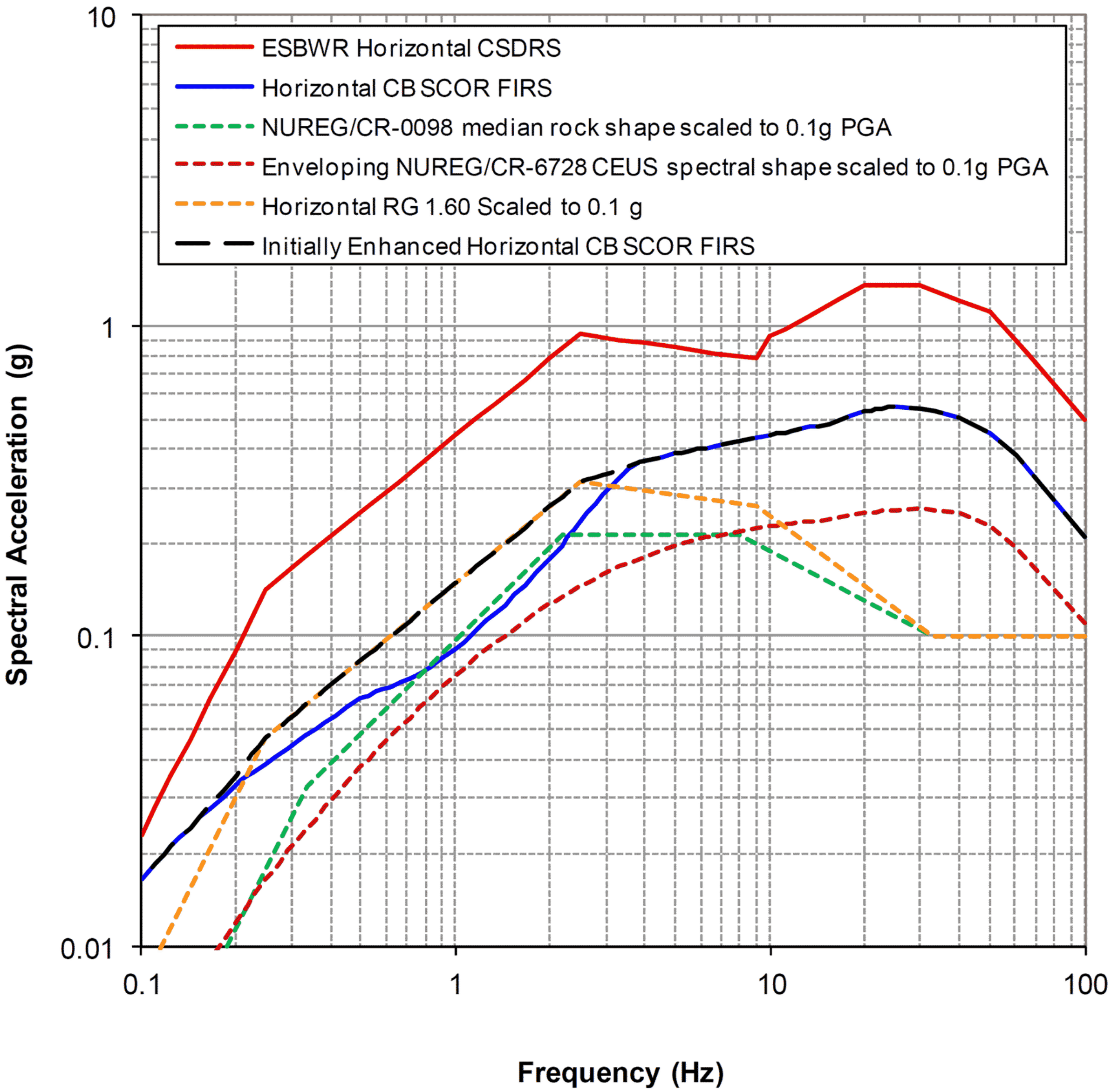


Figure 3.7.1-228 Fermi 3 Horizontal and Vertical RB/FB SCOR FIRS and Enhanced RB/FB SCOR FIRS (5 Percent Damping)

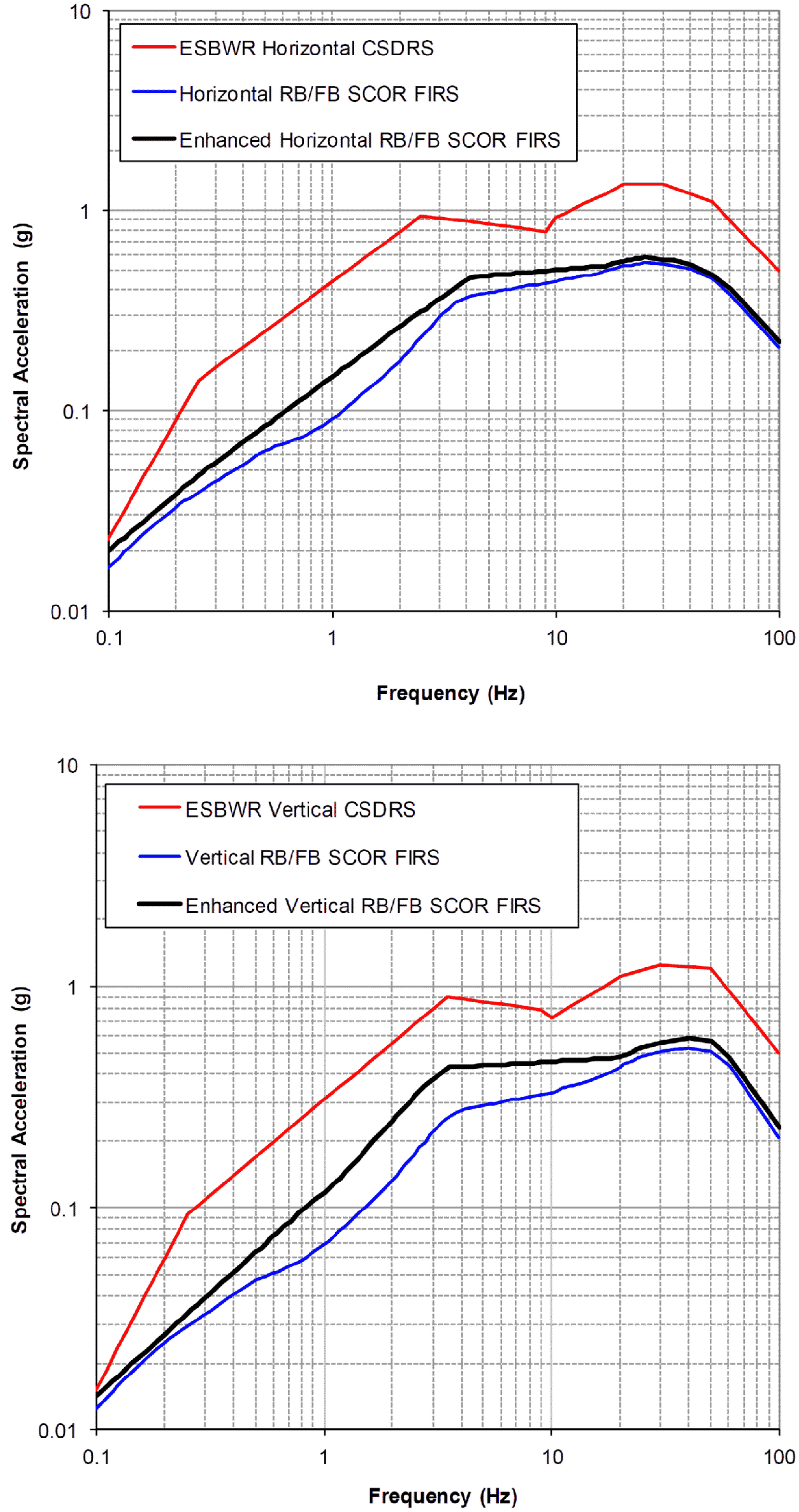
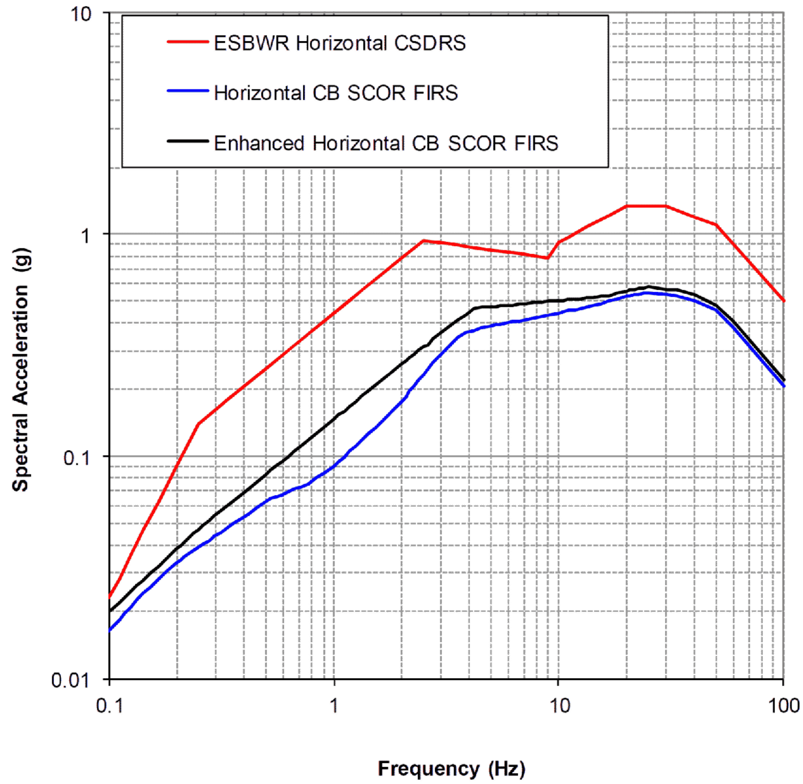


Figure 3.7.1-229 Fermi 3 Horizontal CB SCOR FIRS and Enhanced CB SCOR FIRS (5 Percent Damping)



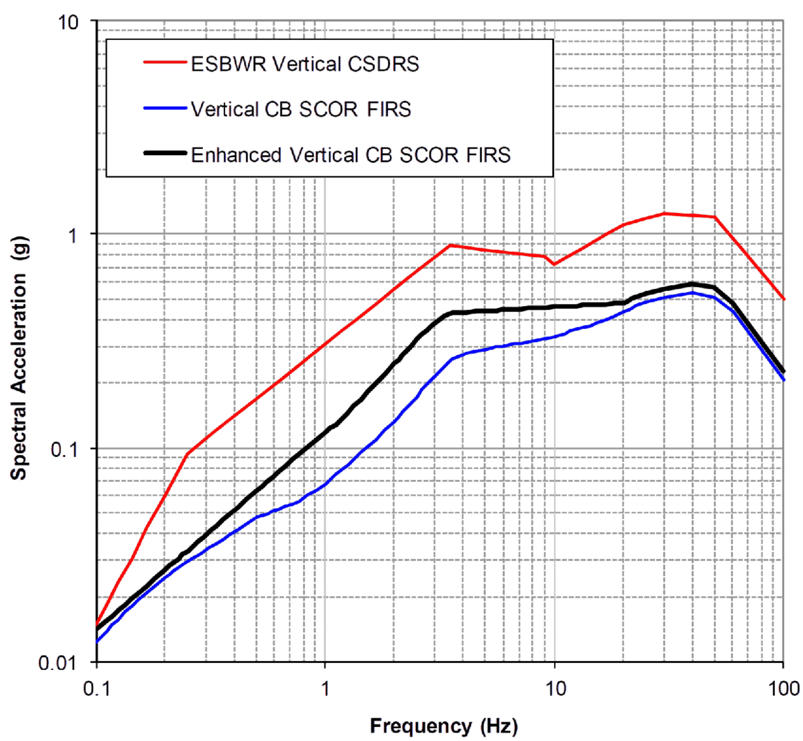


Figure 3.7.1-230 Comparison of the Envelope of the Response Spectra of Computed Horizontal Component Surface Motions for Deterministic Profiles with Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock Using the RB/FB Enhanced SCOR FIRS Input Motions with the Horizontal PBSRS

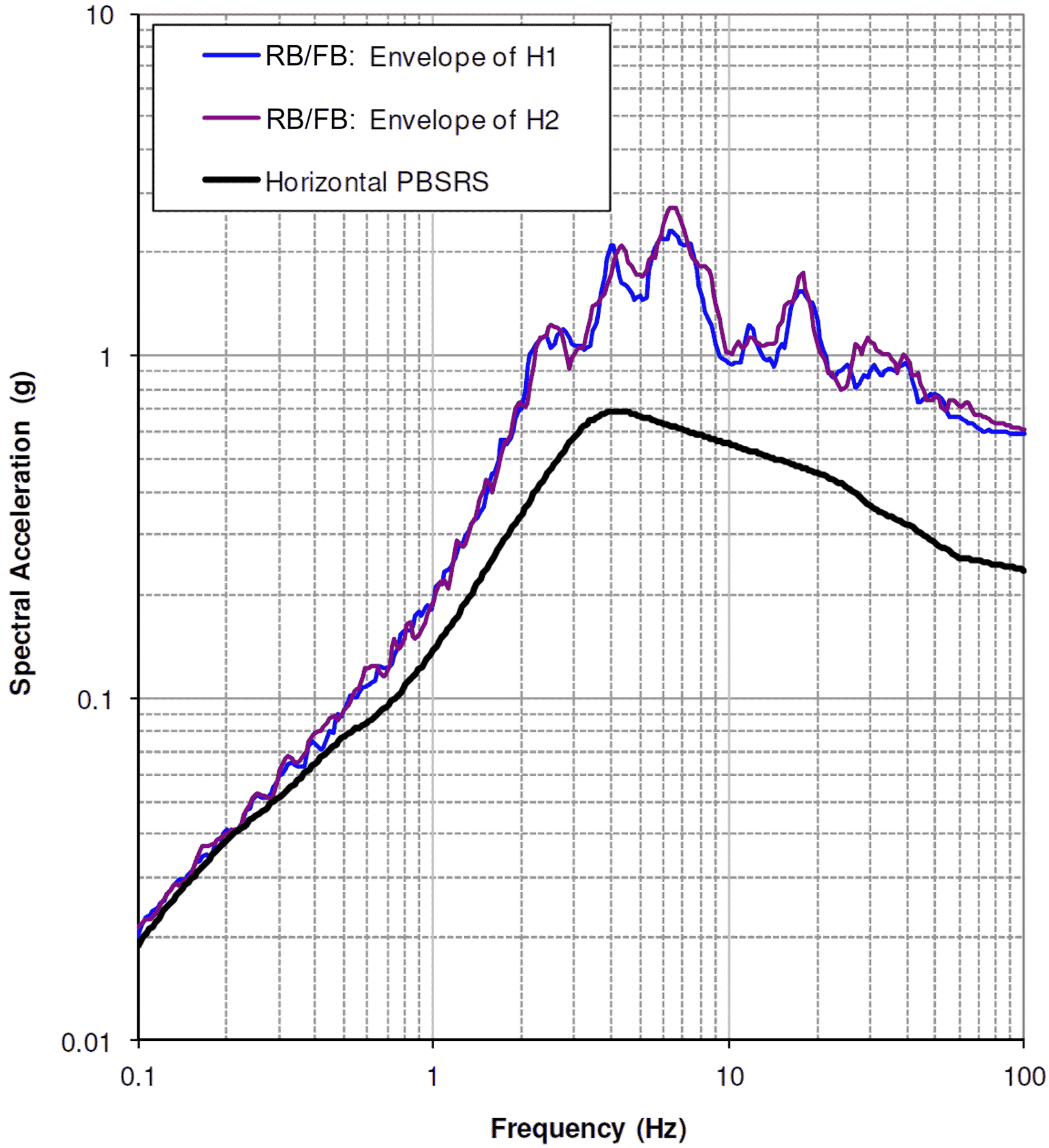


Figure 3.7.1-231 Comparison of the Envelope of the Response Spectra of Computed Horizontal Component Surface Motions for Deterministic Profiles with Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock Using the CB Enhanced SCOR FIRS Input Motions with the Horizontal PBSRS

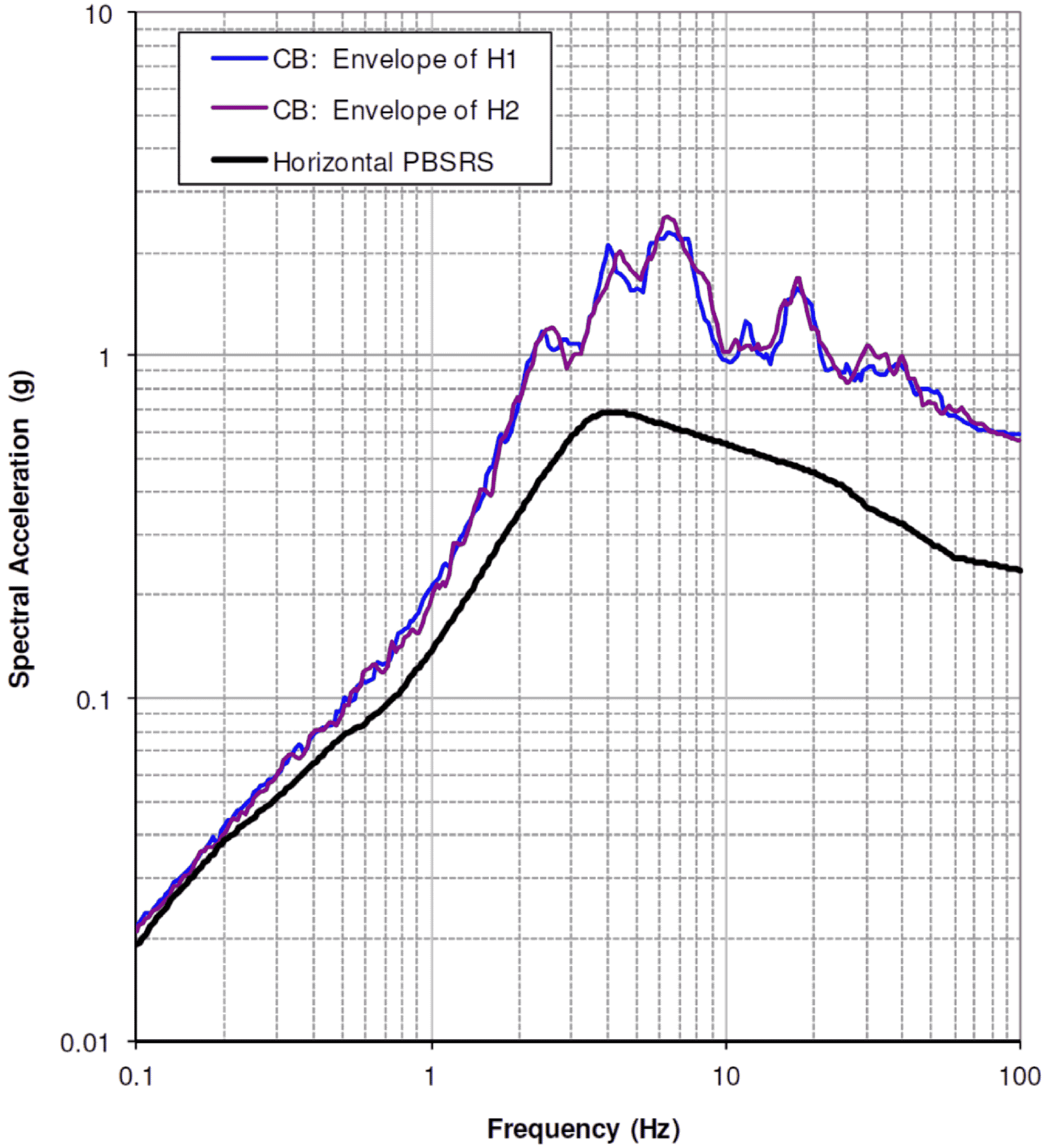


Figure 3.7.1-232 Comparison of the Envelope of the Response Spectra of Computed Horizontal Component Surface Motions for Deterministic Profiles without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock Using the RB/FB Enhanced SCOR FIRS Input Motions with the Horizontal GMRS

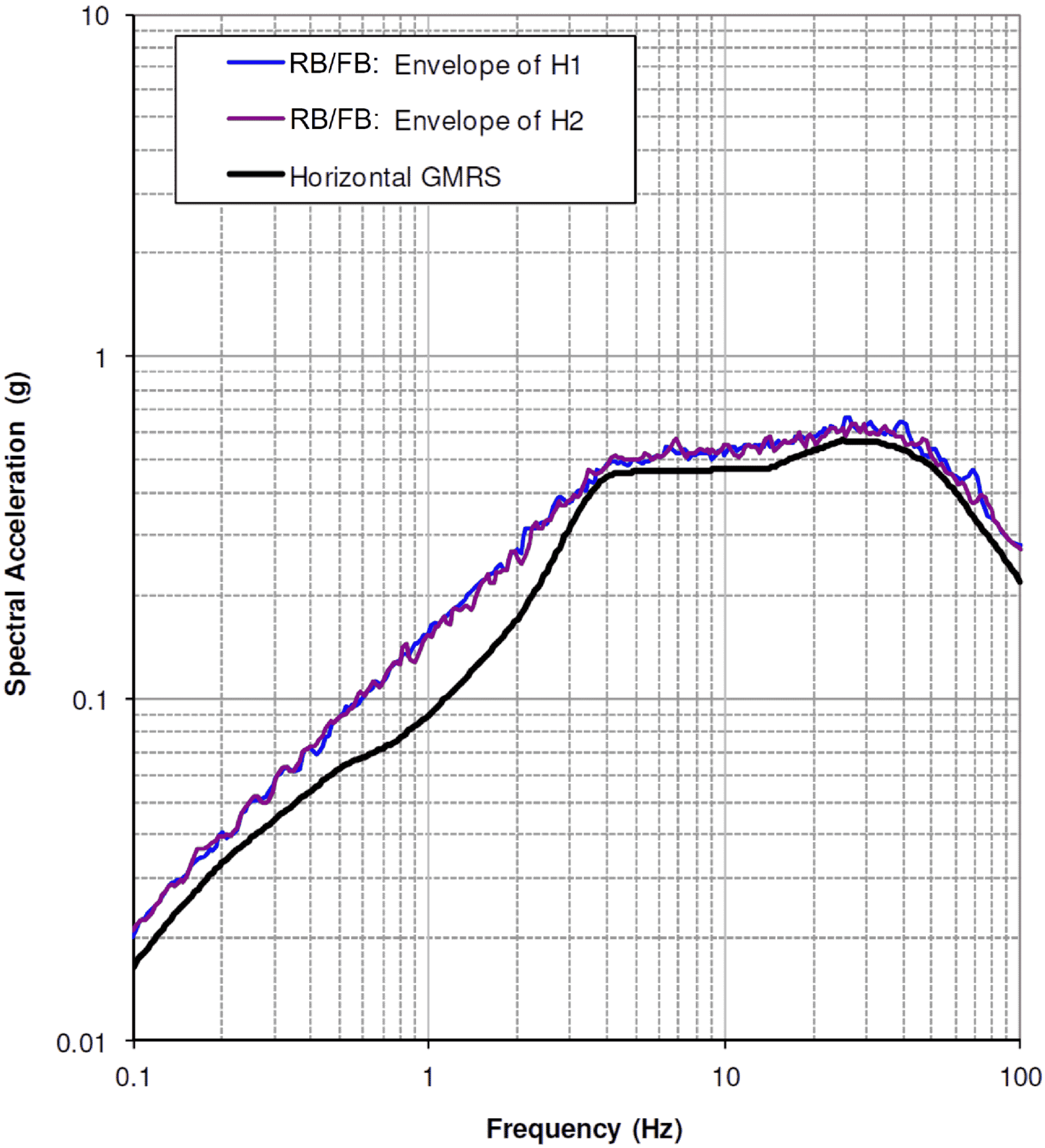


Figure 3.7.1-233 Comparison of the Envelope of the Response Spectra of Computed Horizontal Component Surface Motions for Deterministic Profiles without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock Using the CB Enhanced SCOR FIRS Input Motions with the Horizontal GMRS

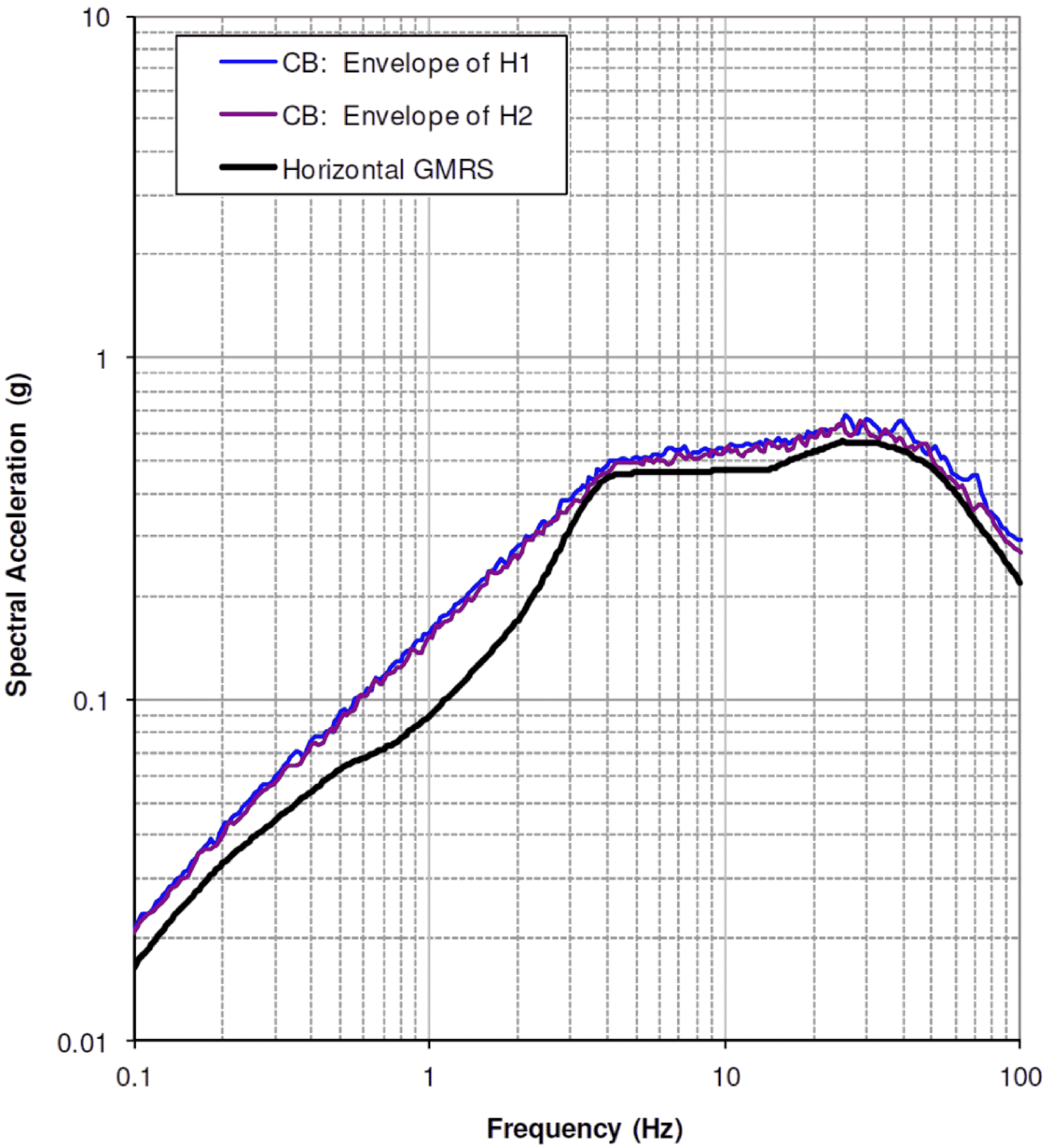


Figure 3.7.1-234 Comparison of the Envelope of the Response Spectra of Computed Vertical Component Surface Motions for Deterministic Profiles With Engineered Granular Backfill Above the top of the Bass Islands Group Bedrock Using the RB/FB Enhanced SCOR FIRS Input Motions with the Vertical PBSRS

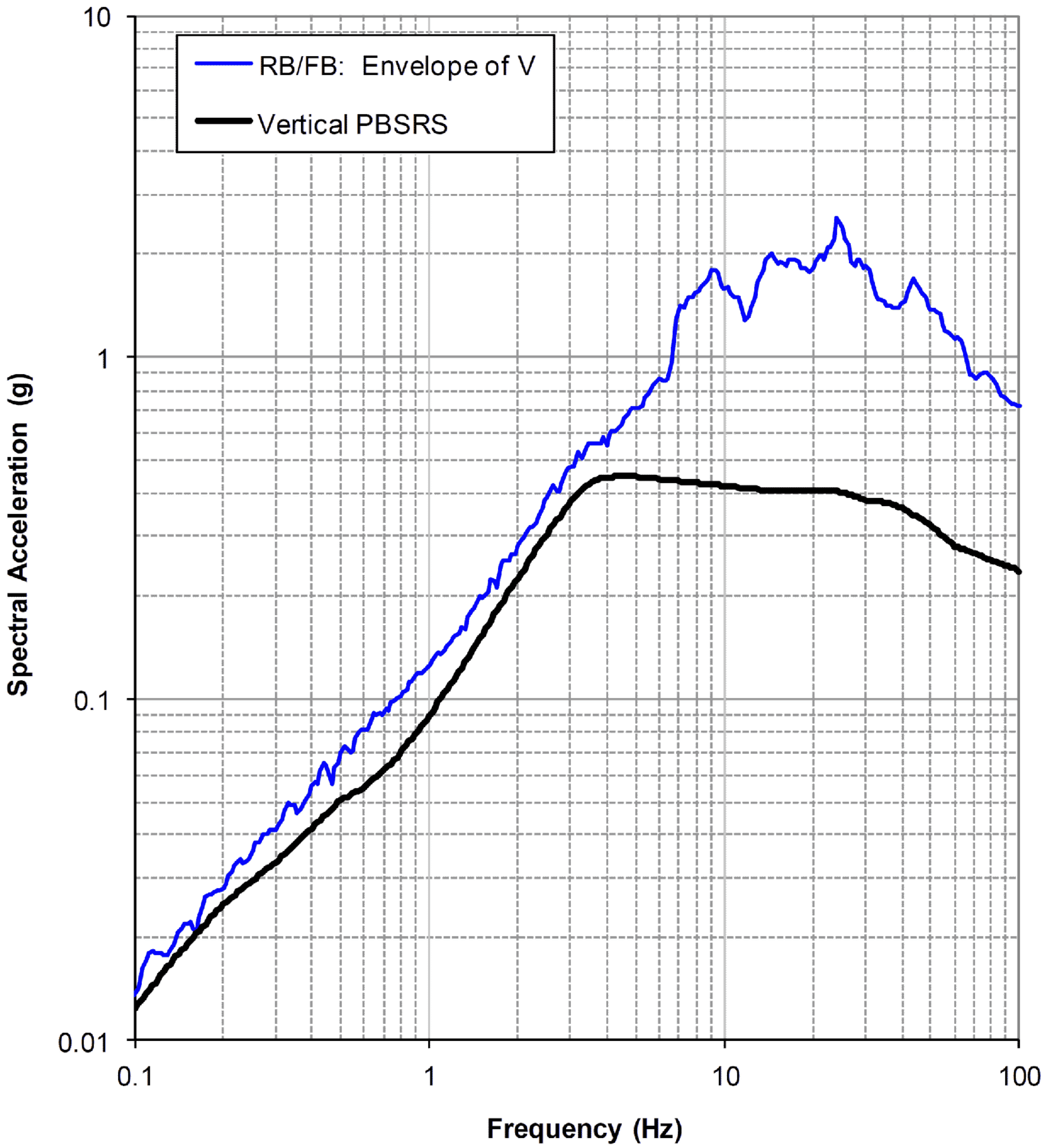


Figure 3.7.1-235 Comparison of the Envelope of the Response Spectra of Computed Vertical Component Surface Motions for Deterministic Profiles With Engineered Granular Backfill Above the top of the Bass Islands Group Bedrock Using the CB Enhanced SCOR FIRS Input Motions with the Vertical PBSRS

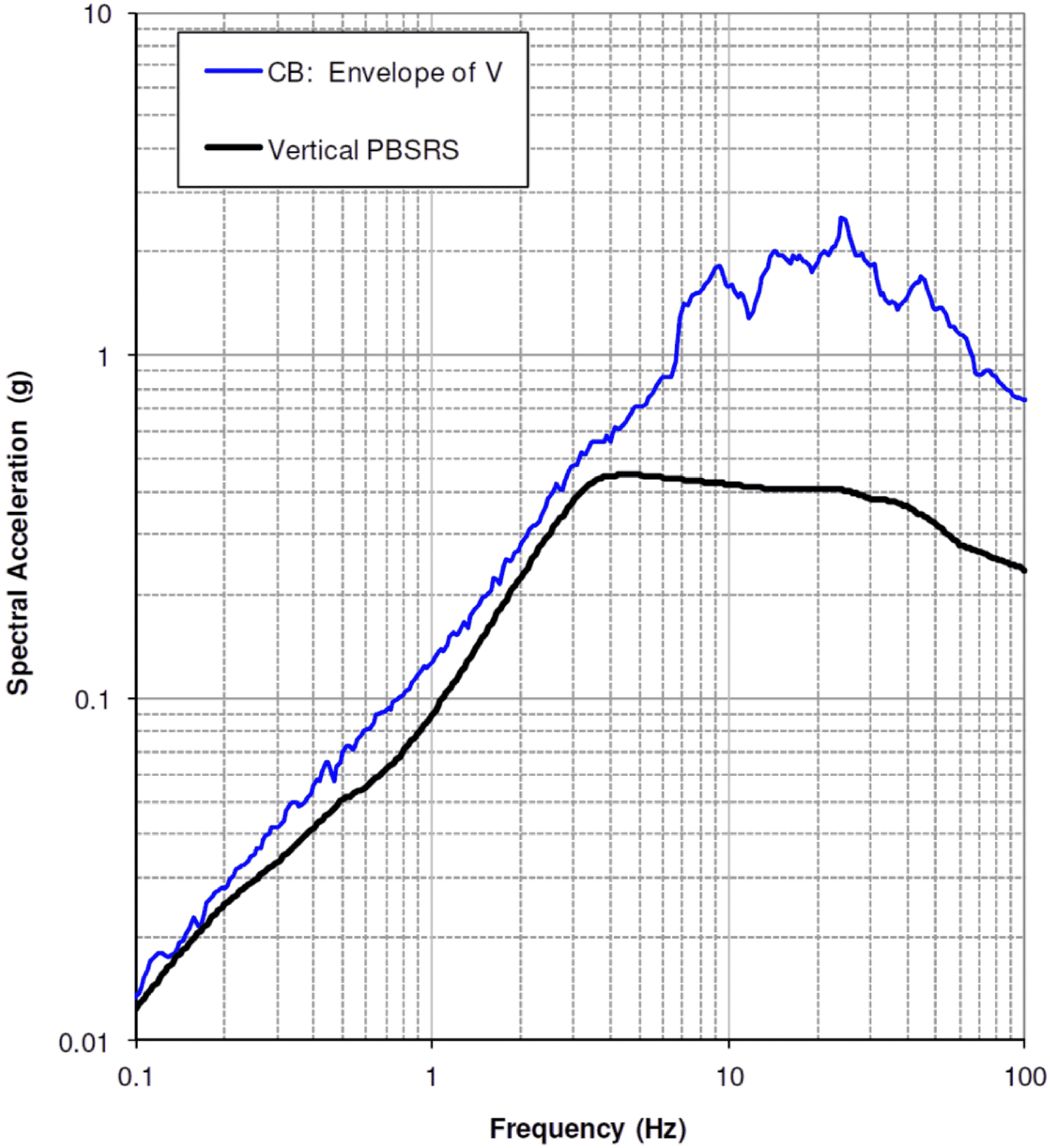


Figure 3.7.1-236 Comparison of the Envelope of the Response Spectra of Computed Vertical Component Surface Motions for Deterministic Profiles without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock Using the RB/FB Enhanced SCOR FIRS Input Motions with the Vertical GMRS

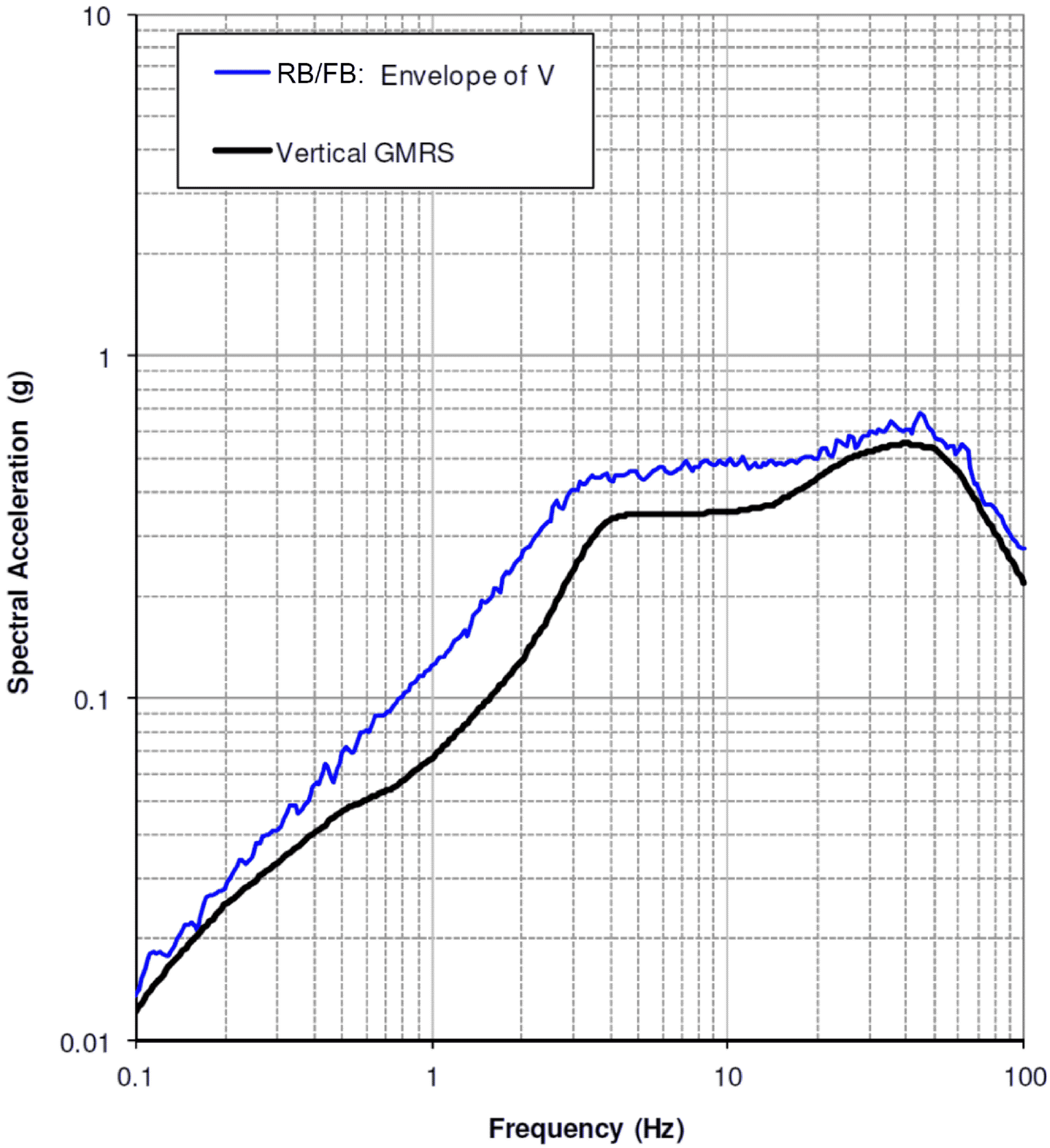


Figure 3.7.1-237 Comparison of the Envelope of the Response Spectra of Computed Vertical Component Surface Motions for Deterministic Profiles without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock Using the CB Enhanced SCOR FIRS Input Motions with the Vertical GMRS

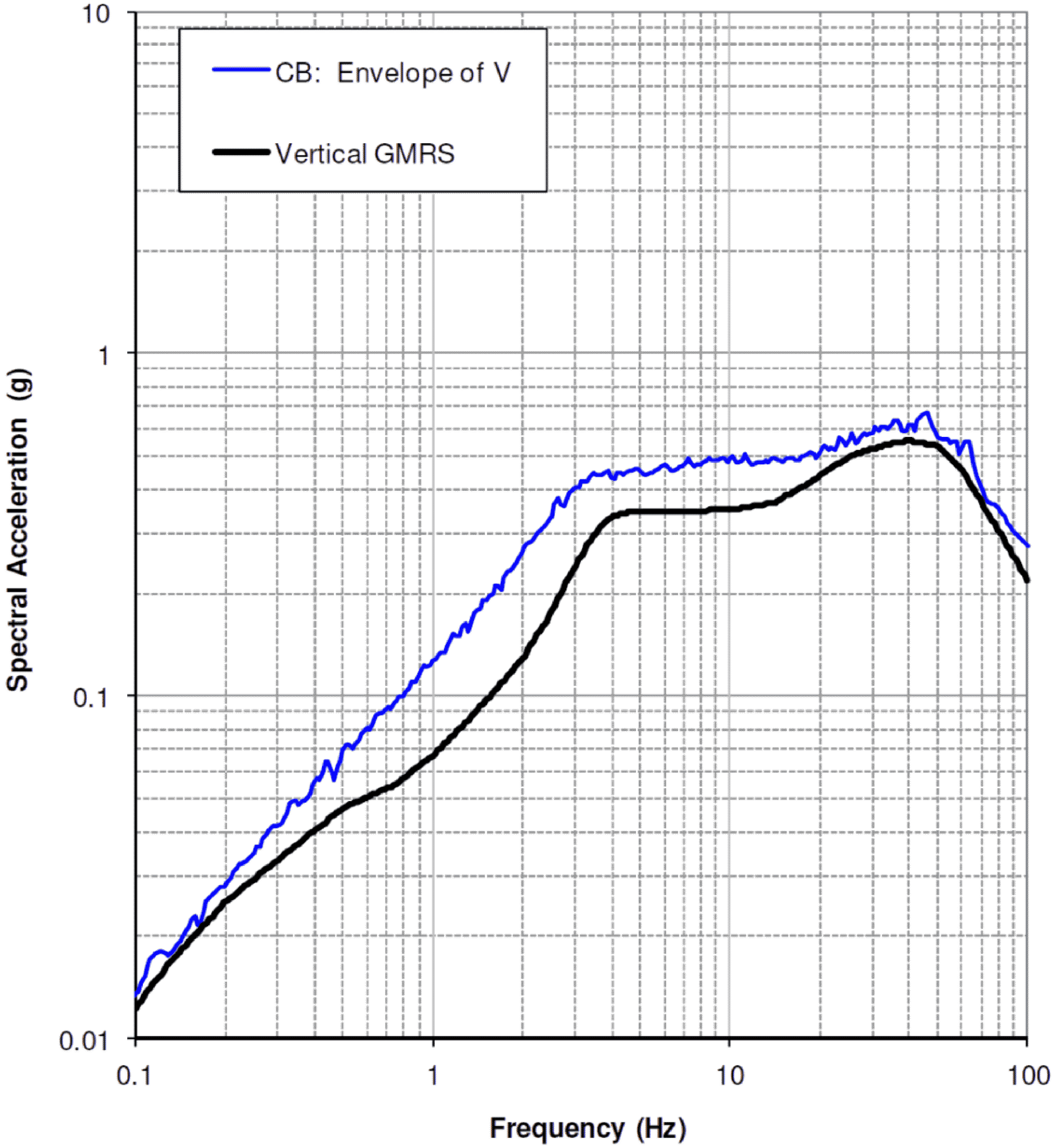
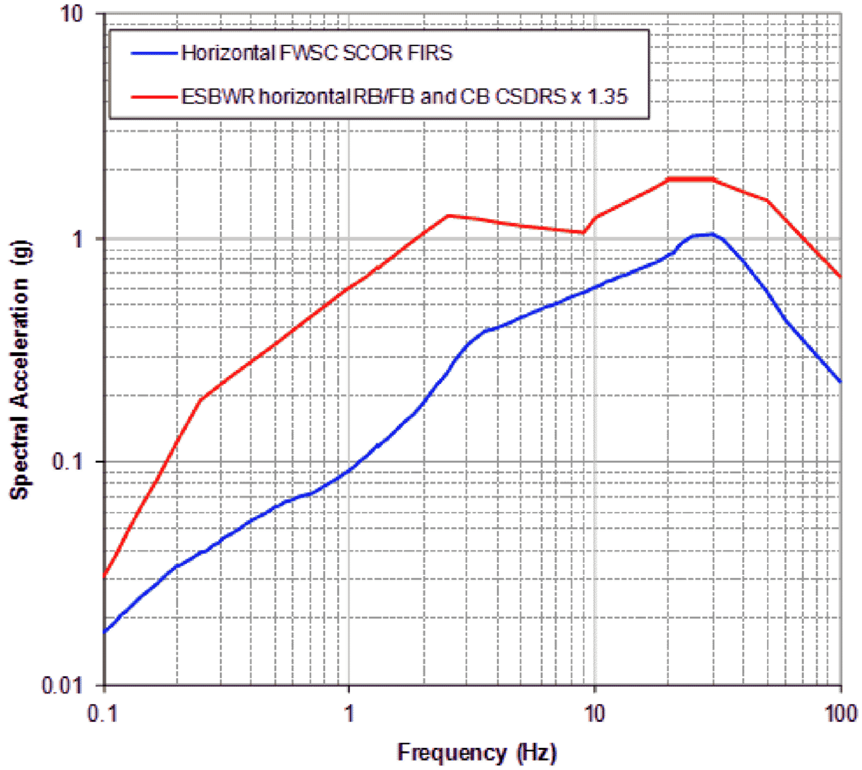


Figure 3.7.1-238 Fermi 3 FWSC FIRS (5 Percent Damping)



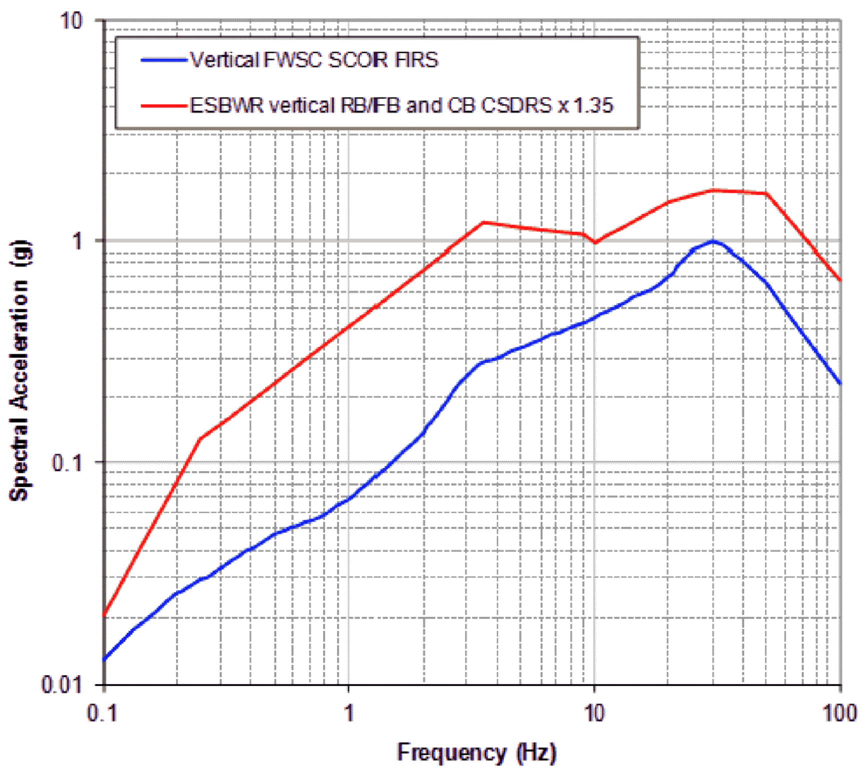


Figure 3.7.1-239 Response Spectrum for Spectrally Matched Horizontal (H1) Component for the Fermi 3 RB/FB Enhanced SCOR FIRS

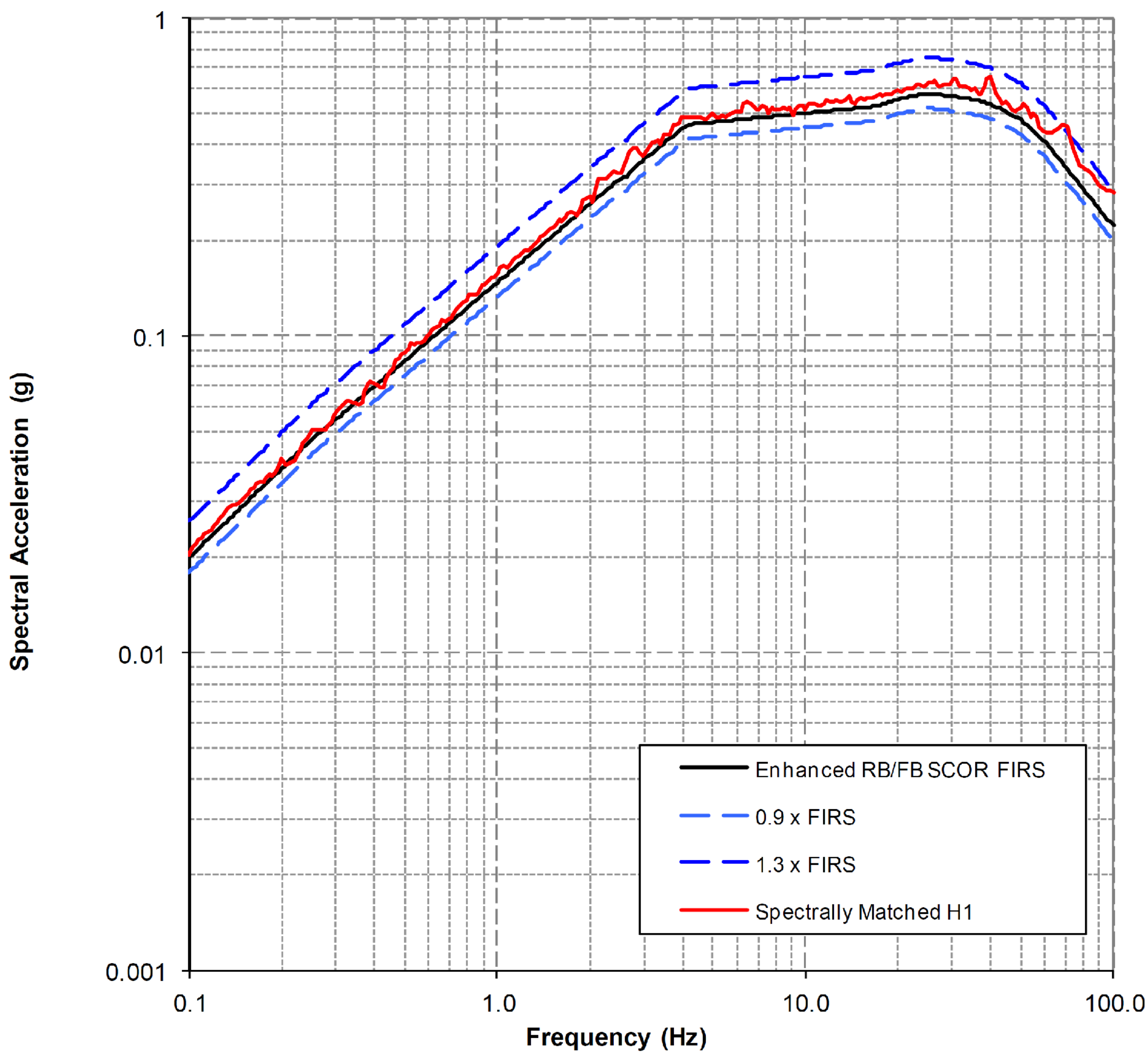


Figure 3.7.1-240 Response Spectrum for Spectrally Matched Horizontal (H2) Component for the Fermi 3 RB/FB Enhanced SCOR FIRS

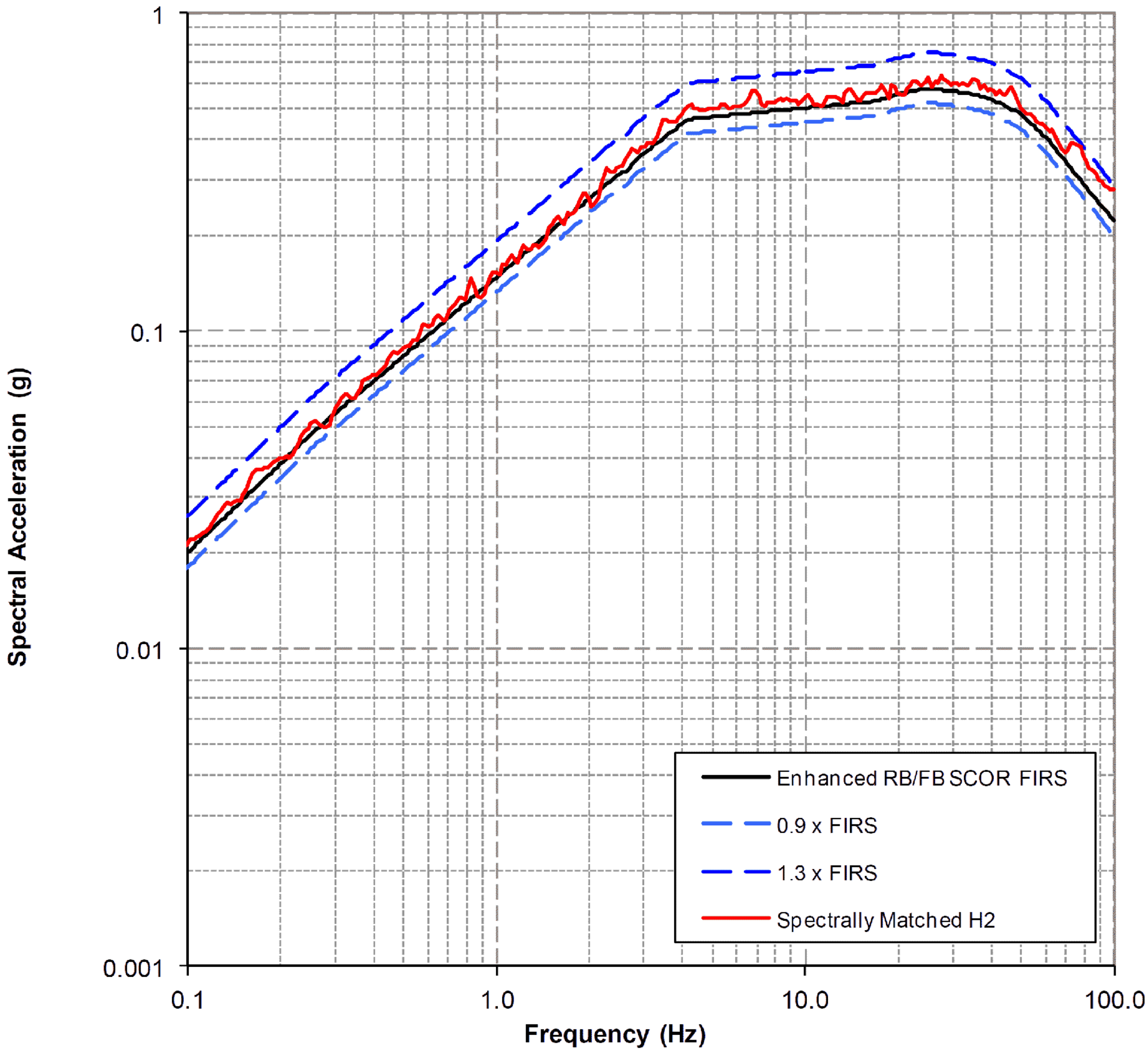


Figure 3.7.1-241 Response Spectrum for Spectrally Matched Vertical (V) Component for the Fermi 3 RB/FB Enhanced SCOR FIRS

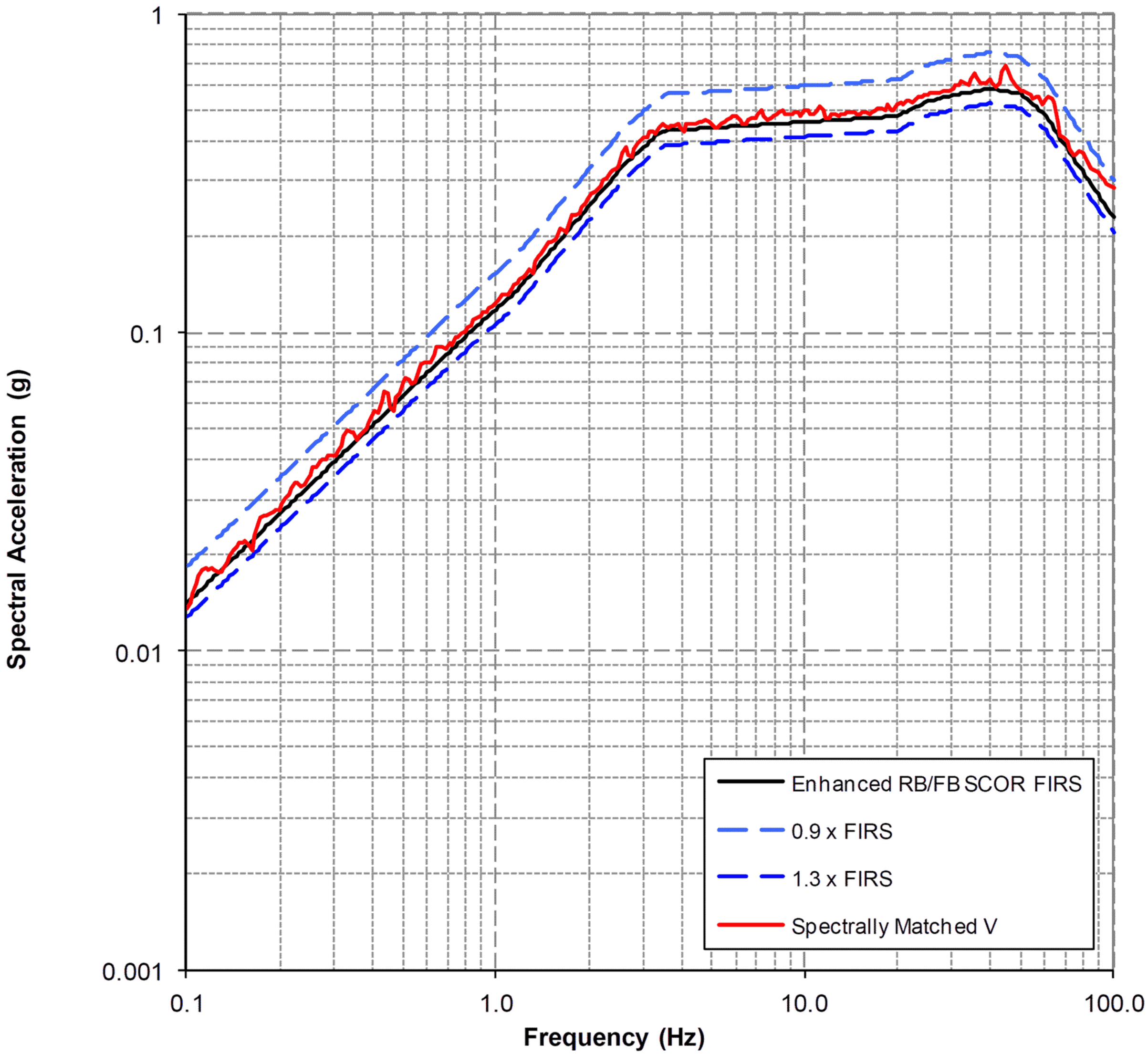


Figure 3.7.1-242 Response Spectrum for Spectrally Matched Horizontal (H1) Component for the Fermi 3 CB Enhanced SCOR FIRS

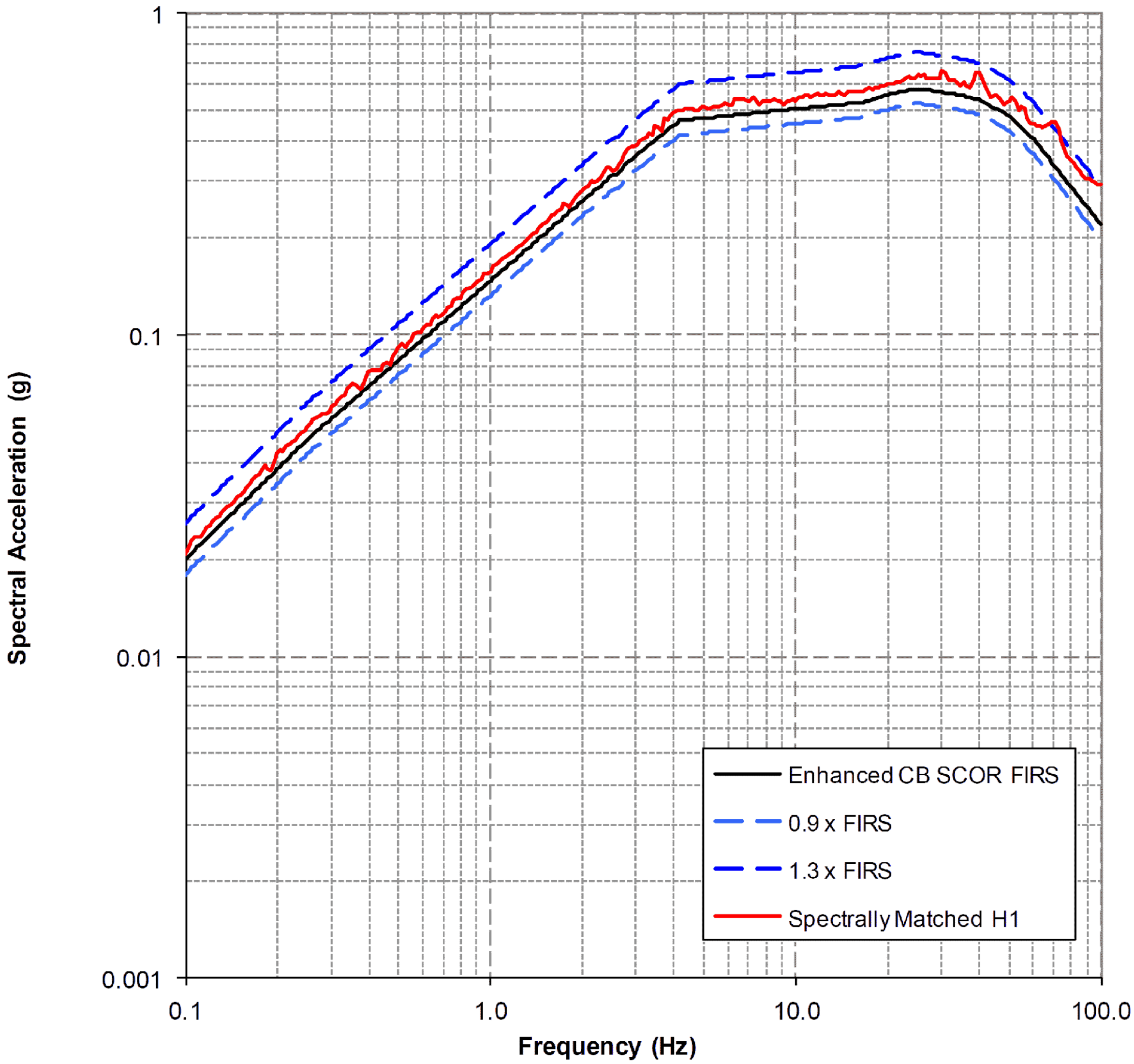


Figure 3.7.1-243 Response Spectrum for Spectrally Matched Horizontal (H2) Component for the Fermi 3 CB Enhanced SCOR FIRS

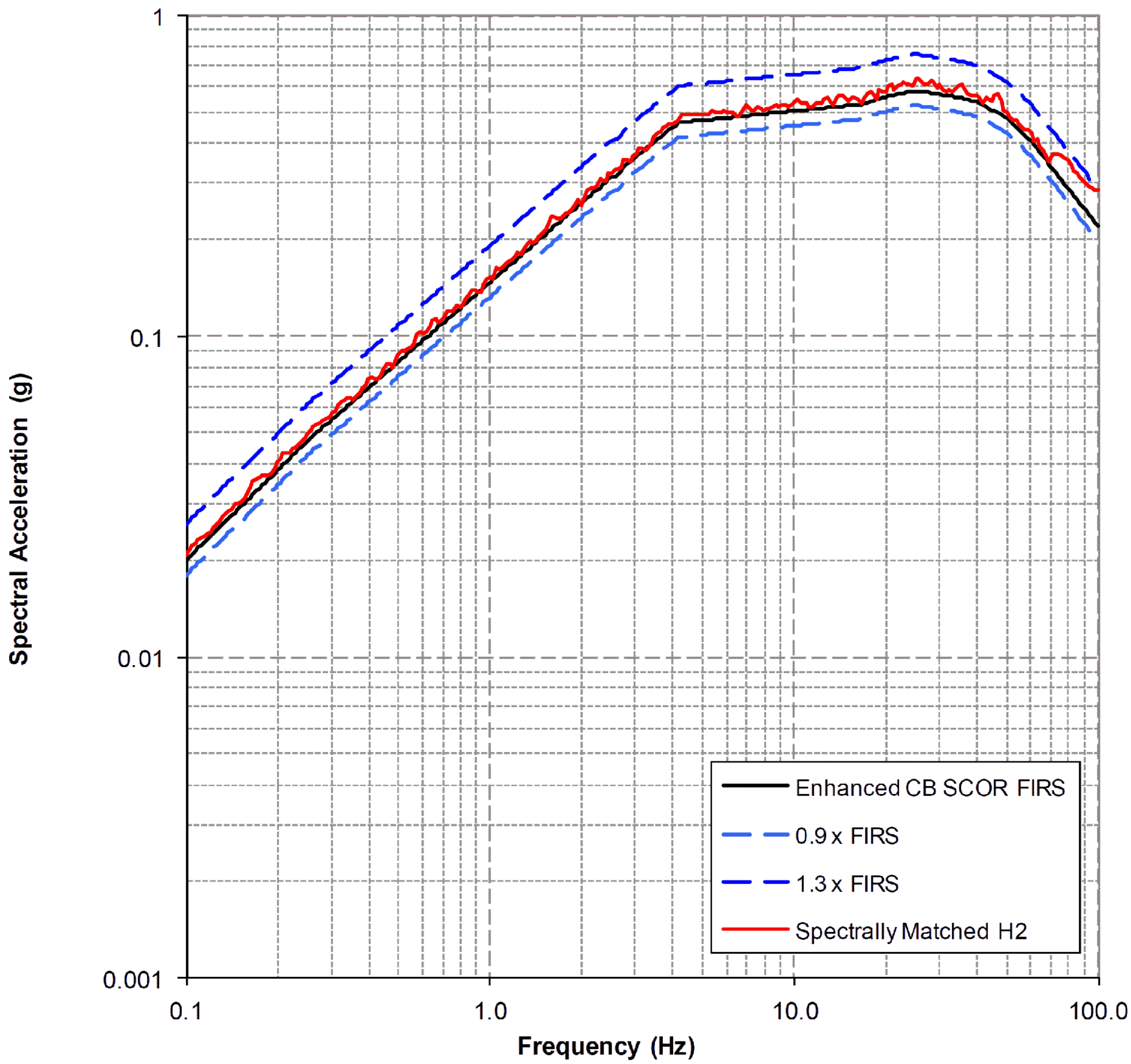


Figure 3.7.1-244 Response Spectrum for Spectrally Matched Vertical (V) Component for the Fermi 3 CB Enhanced SCOR FIRS

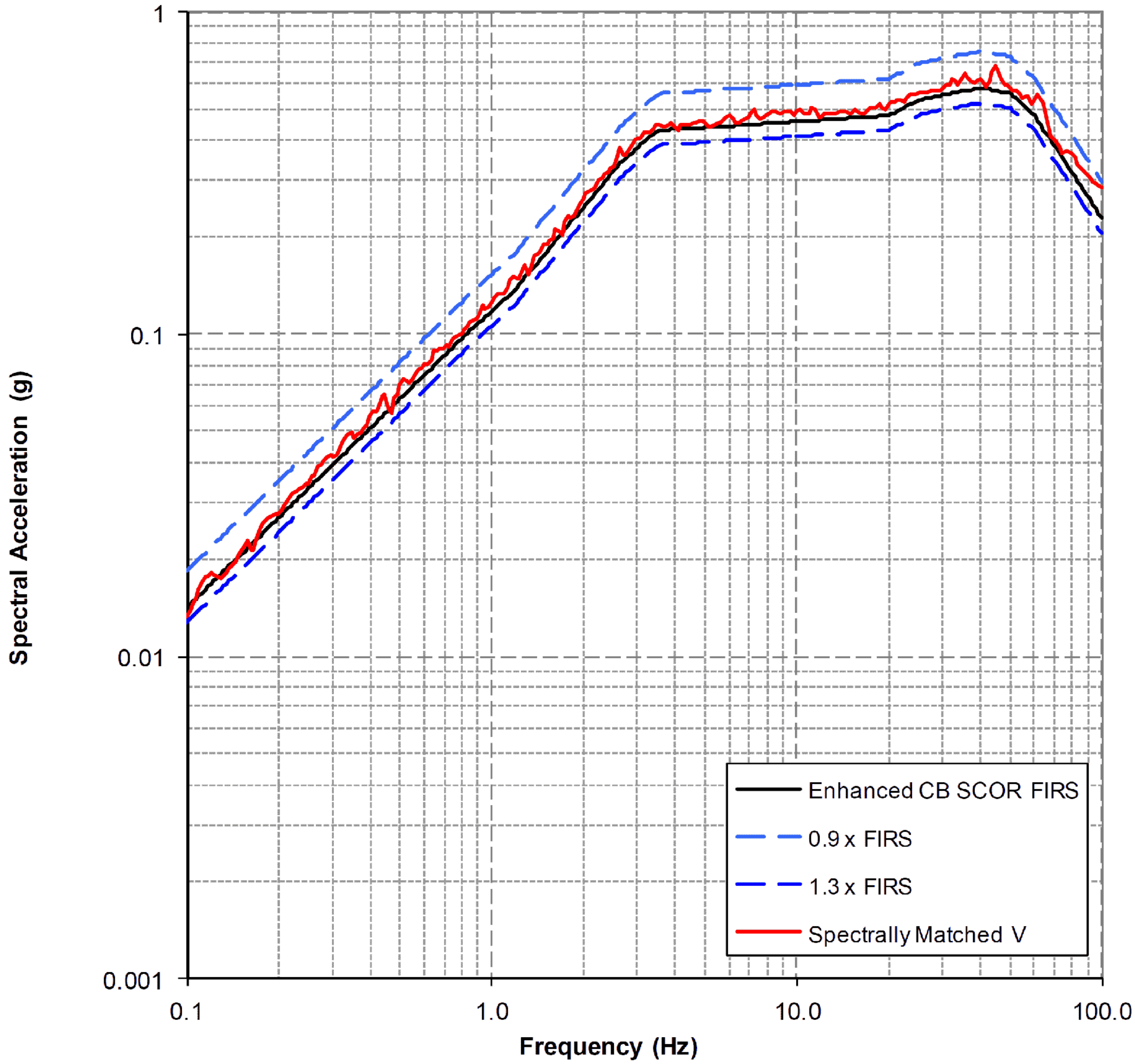


Figure 3.7.1-245 Acceleration, Velocity, and Displacement Time Histories for the Spectrally Matched Horizontal (H1) Component Compatible with the Fermi 3 RB/FB Enhanced SCOR FIRS

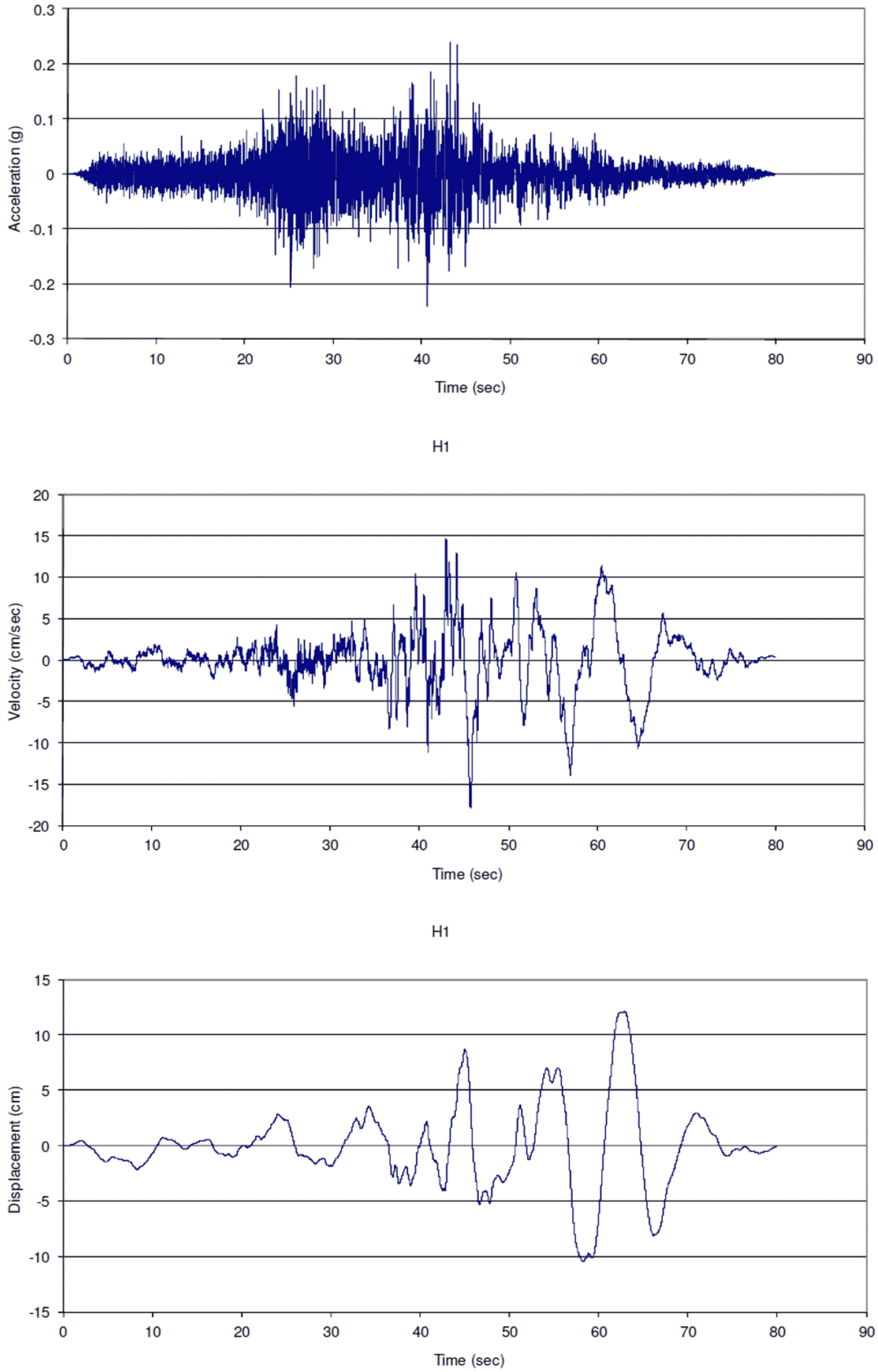


Figure 3.7.1-246 Acceleration, Velocity, and Displacement Time Histories for the Spectrally Matched Horizontal (H2) Component Compatible with the Fermi 3 RB/FB Enhanced SCOR FIRS

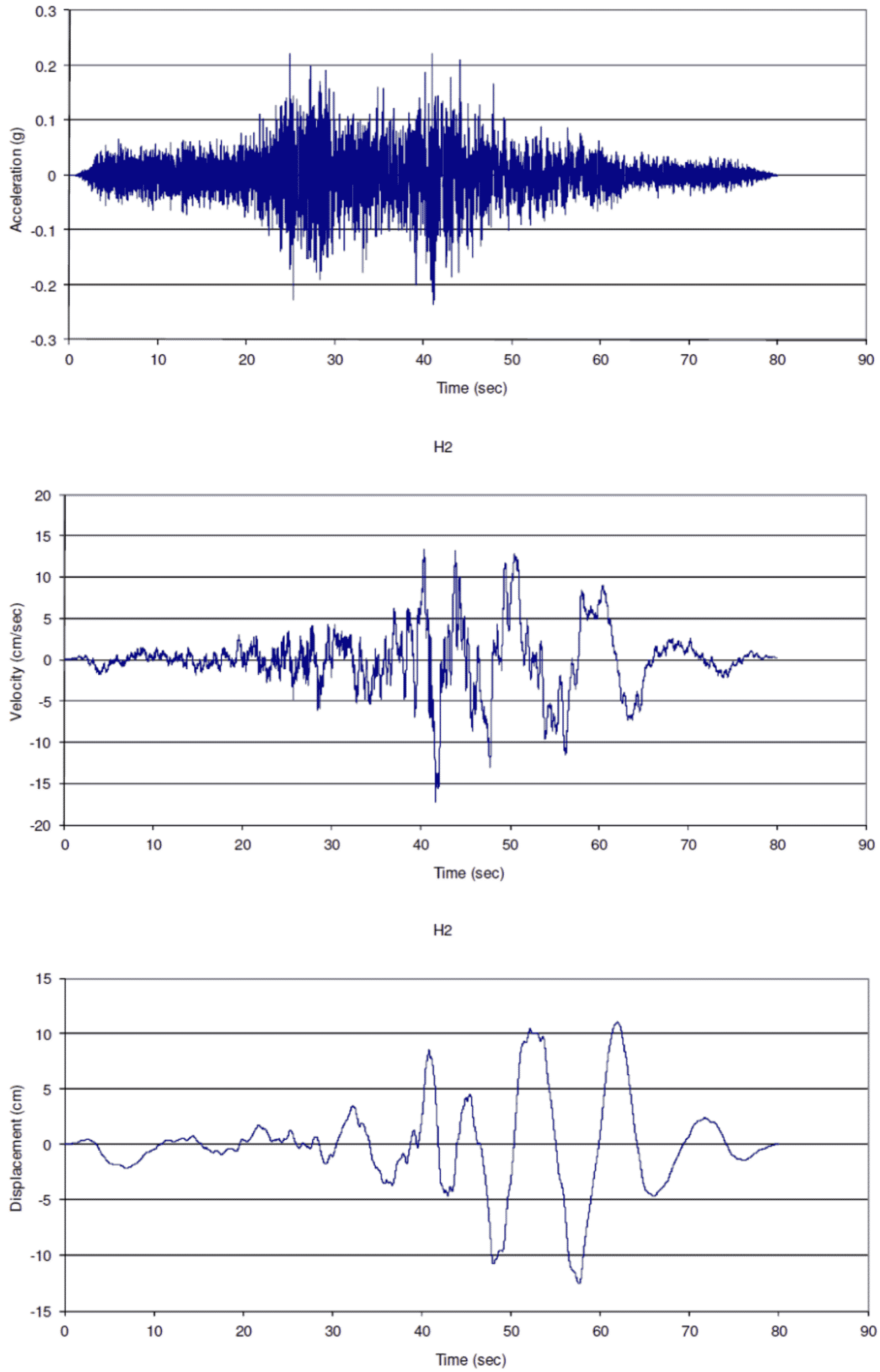


Figure 3.7.1-247 Acceleration, Velocity, and Displacement Time Histories for the Spectrally Matched Vertical (V) Component Compatible with the Fermi 3 RB/FB Enhanced SCOR FIRS

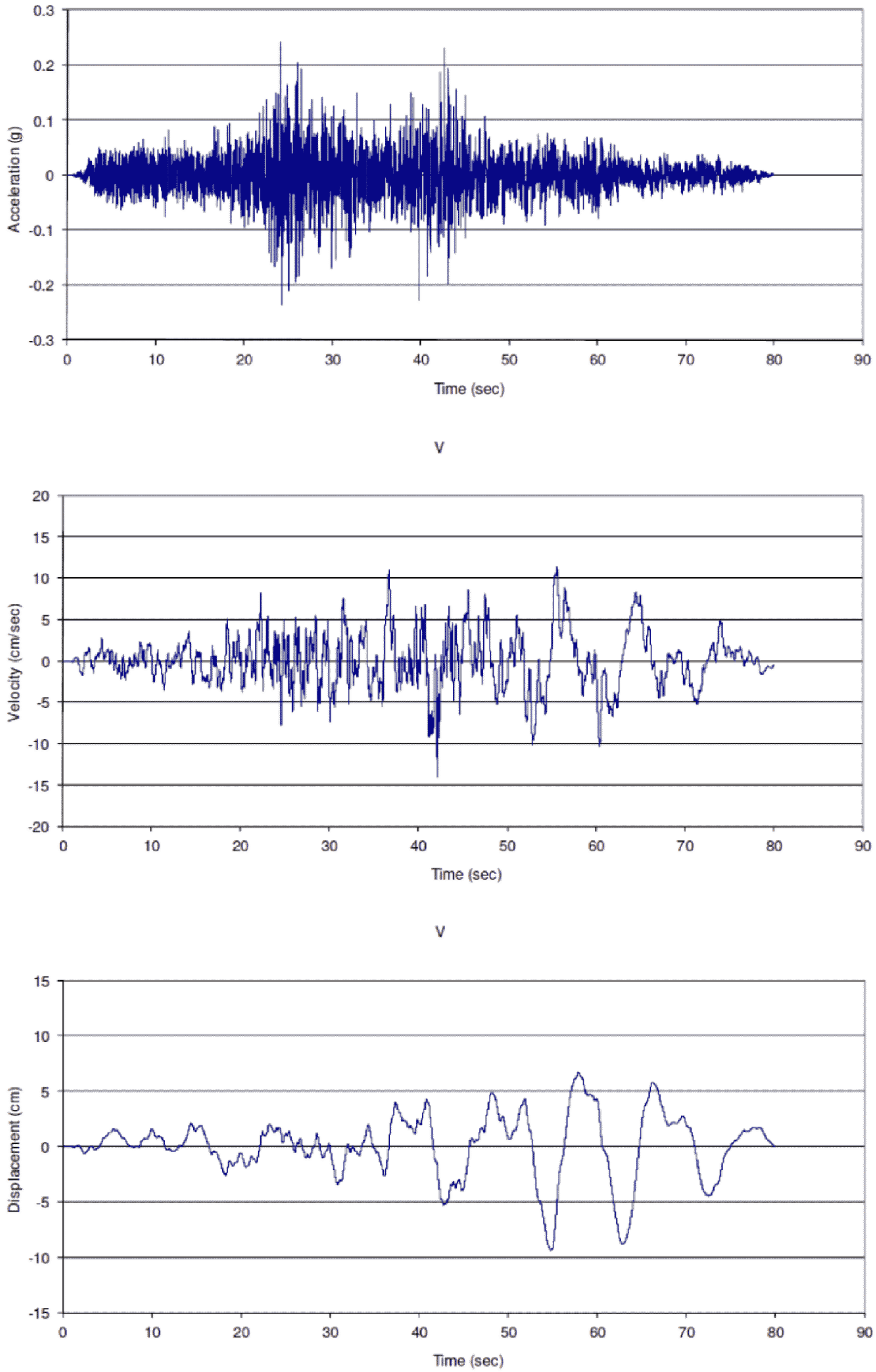


Figure 3.7.1-248 Acceleration, Velocity, and Displacement Time Histories for the Spectrally Matched Horizontal (H1) Component Compatible with the Fermi 3 CB Enhanced SCOR FIRS

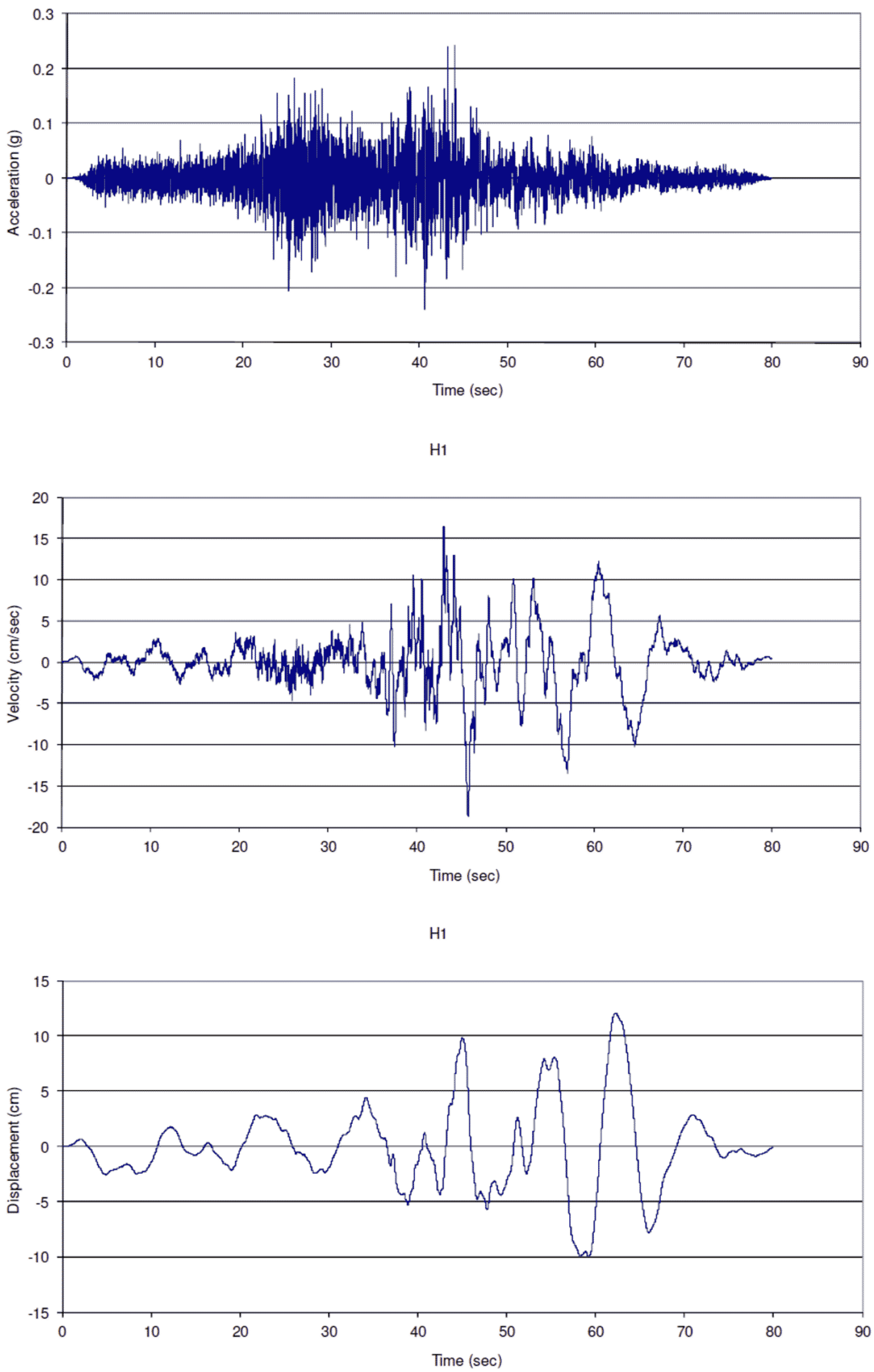


Figure 3.7.1-249 Acceleration, Velocity, and Displacement Time Histories for the Spectrally Matched Horizontal (H2) Component Compatible with the Fermi 3 CB Enhanced SCOR FIRS

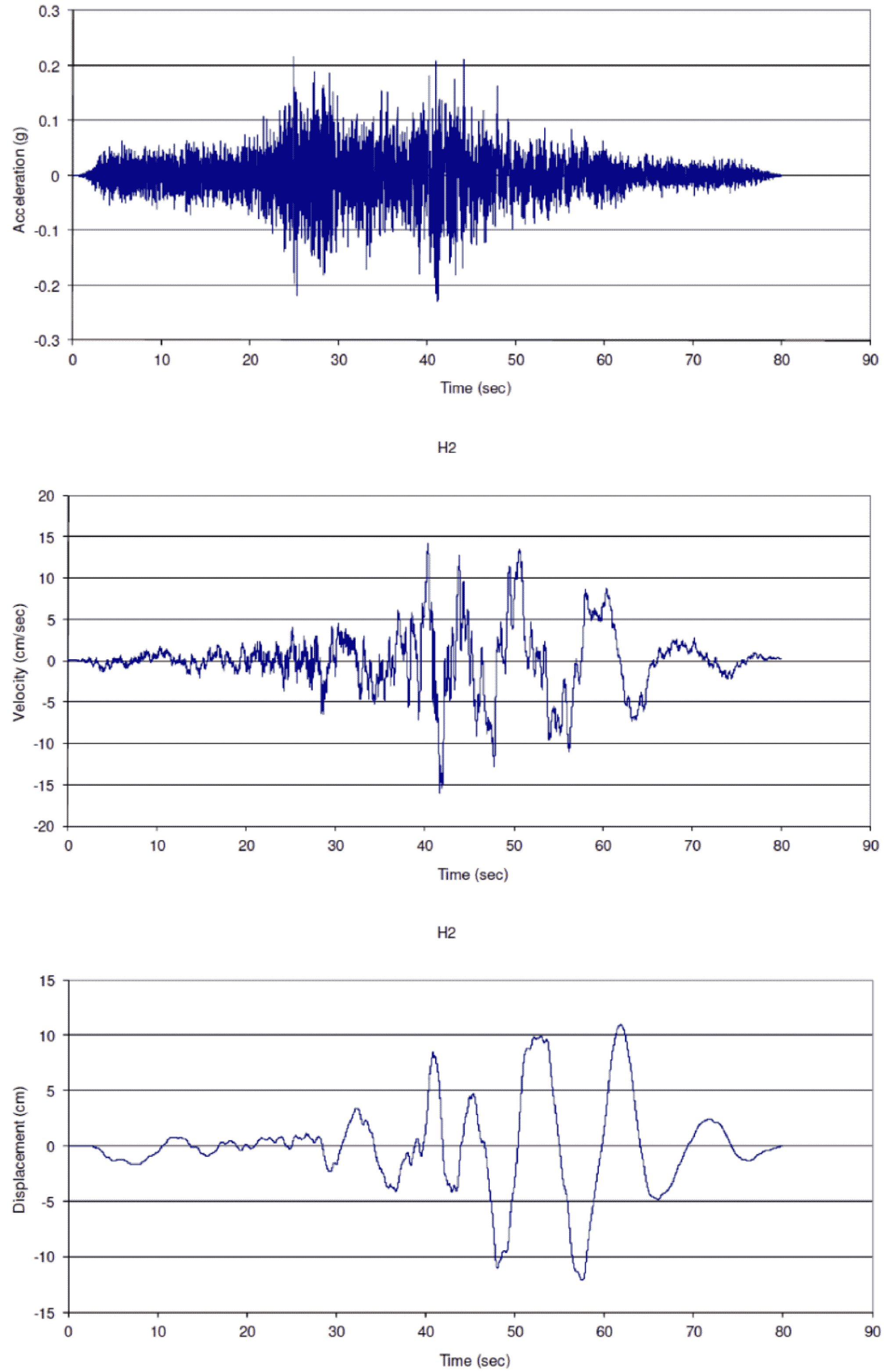


Figure 3.7.1-250 Acceleration, Velocity, and Displacement Time Histories for the Spectrally Matched Vertical (V) Component Compatible with the Fermi 3 CB Enhanced SCOR FIRS

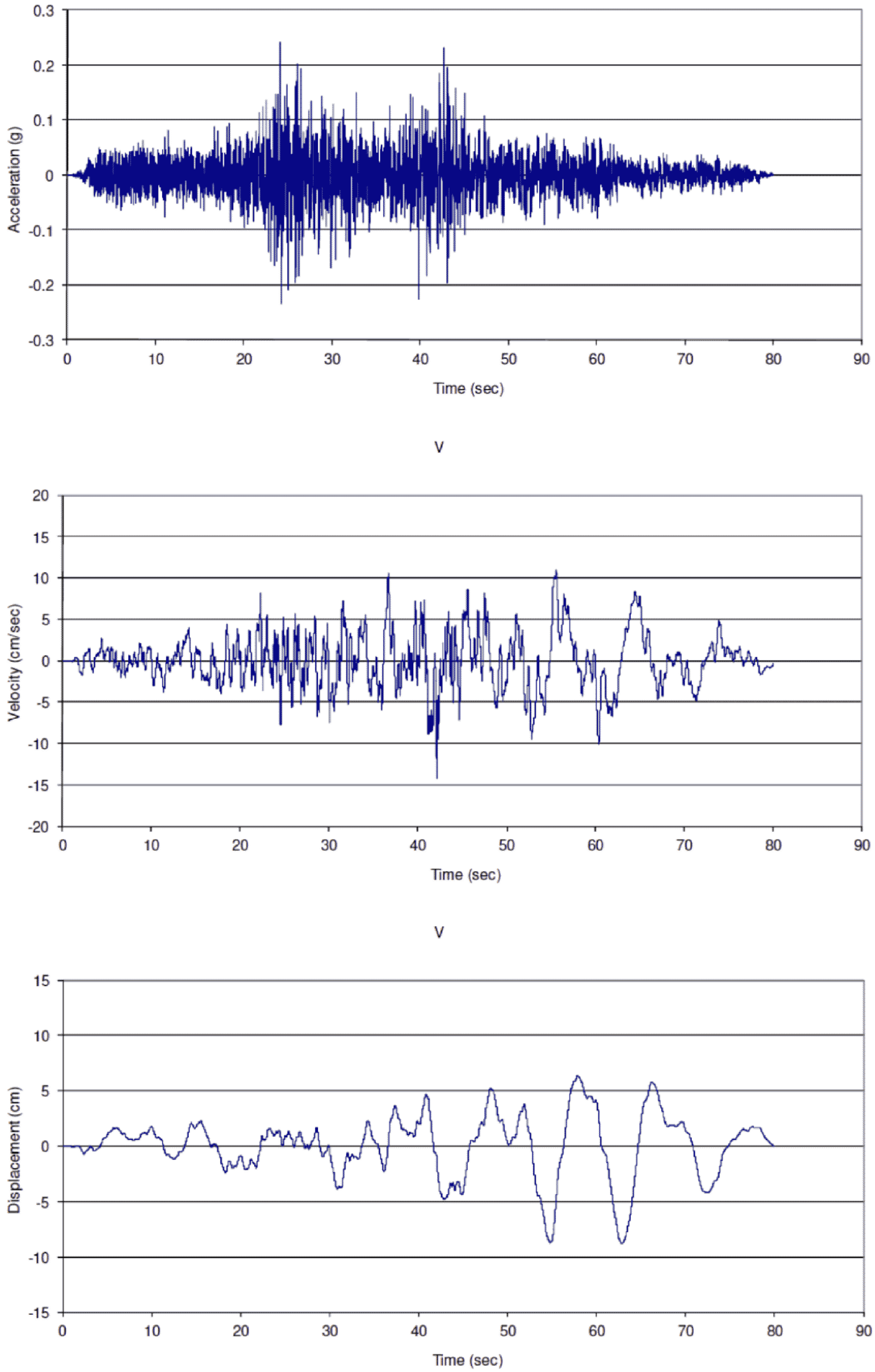
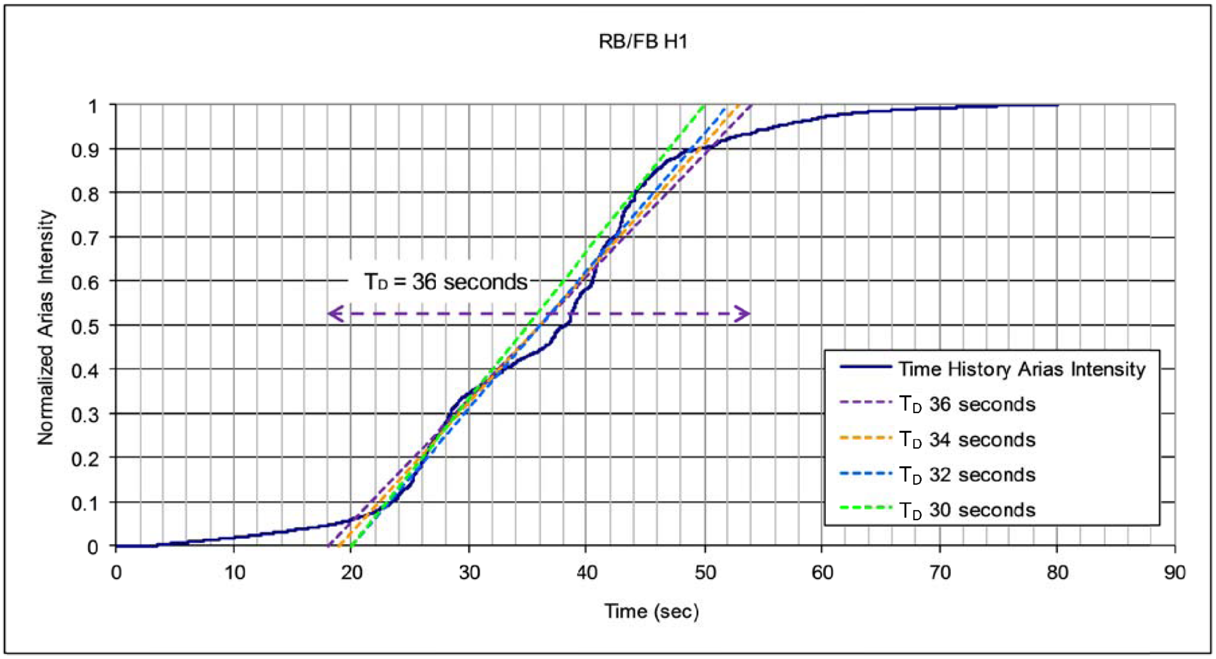


Figure 3.7.1-251 Normalized Arias Intensity and Estimates of Equivalent Stationary Duration for Calculating the PSD for the Spectrally Matched Horizontal (H1 and H2) Components Compatible with the Fermi 3 RB/FB Enhanced SCOR FIRS



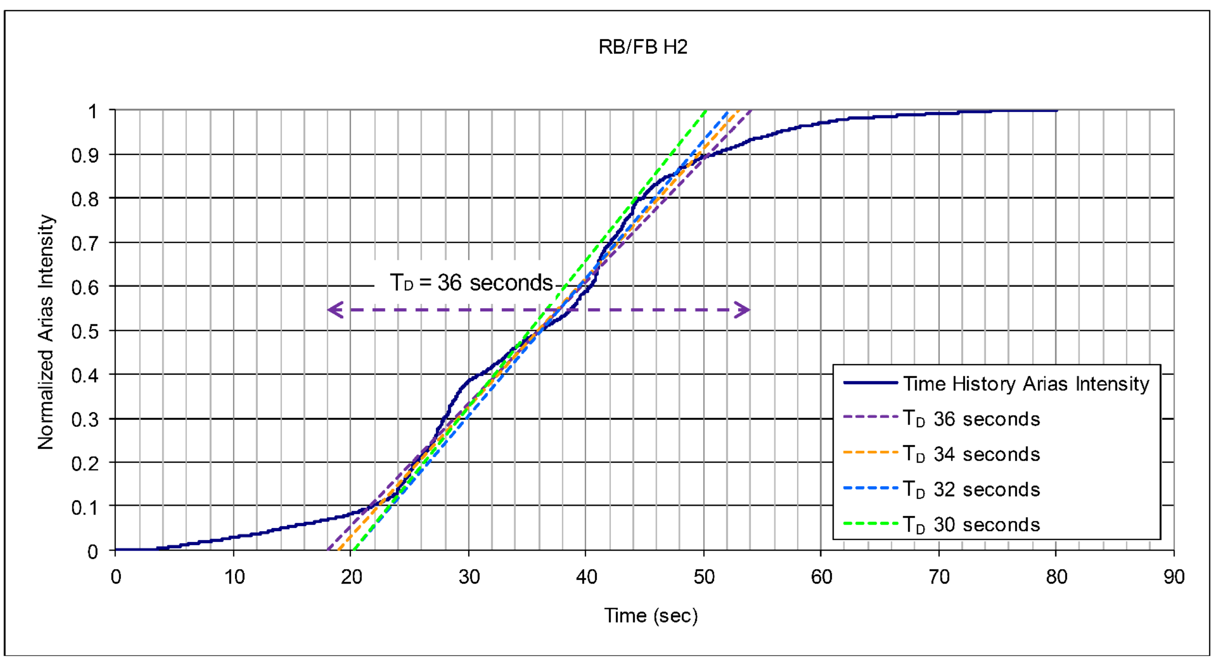


Figure 3.7.1-252 PSD Computed for the RB/FB H1 Component Spectrally Matched Time History (Enhanced SCOR FIRS) Using Full Duration Time Histories and Time Histories Windowed to an Equivalent Stationary Duration, T_D

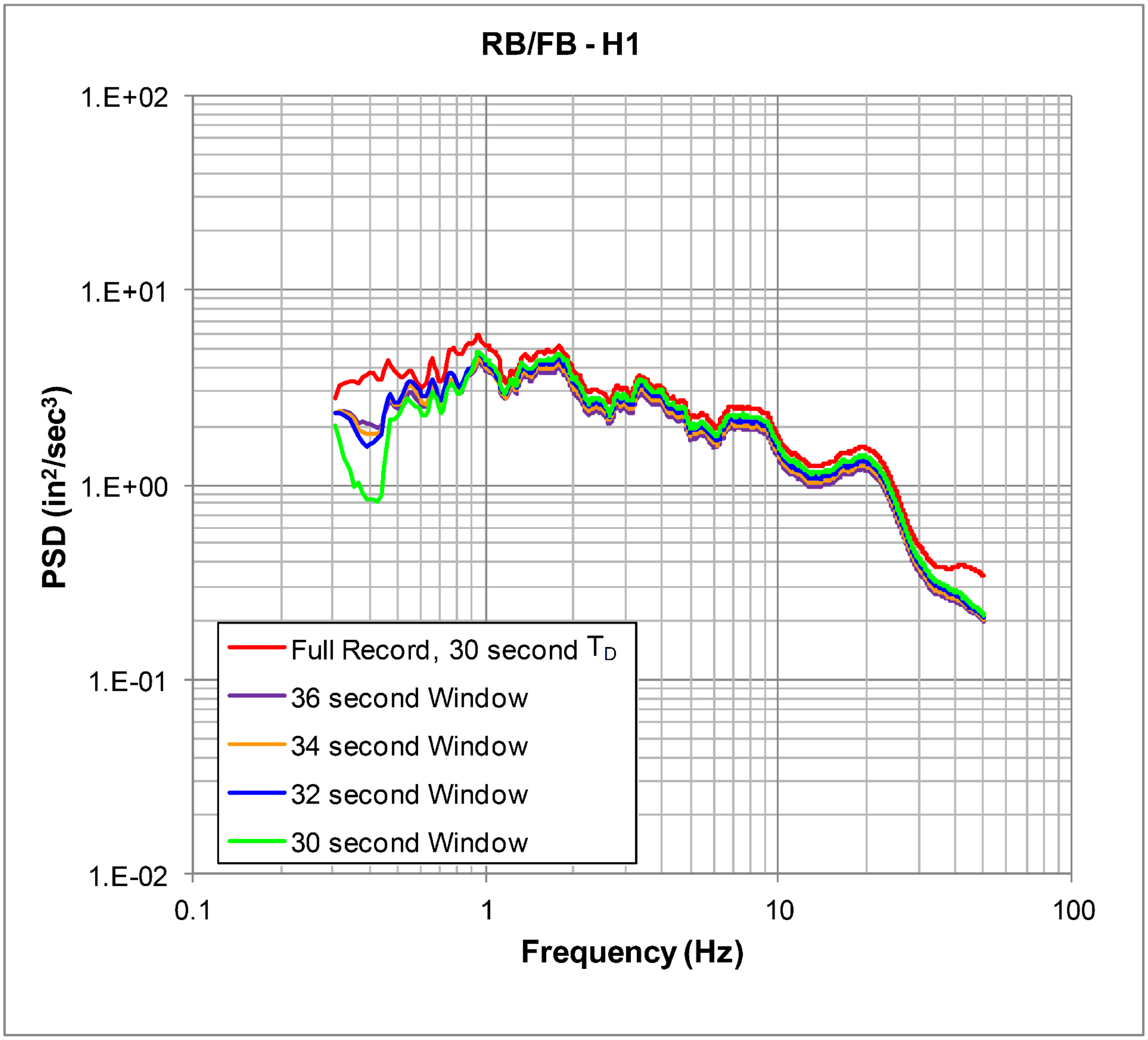


Figure 3.7.1-253 PSD Computed for the RB/FB H2 Component Spectrally Matched Time History (Enhanced SCOR FIRS) Using Full Duration Time Histories and Time Histories Windowed to an Equivalent Stationary Duration, T_D

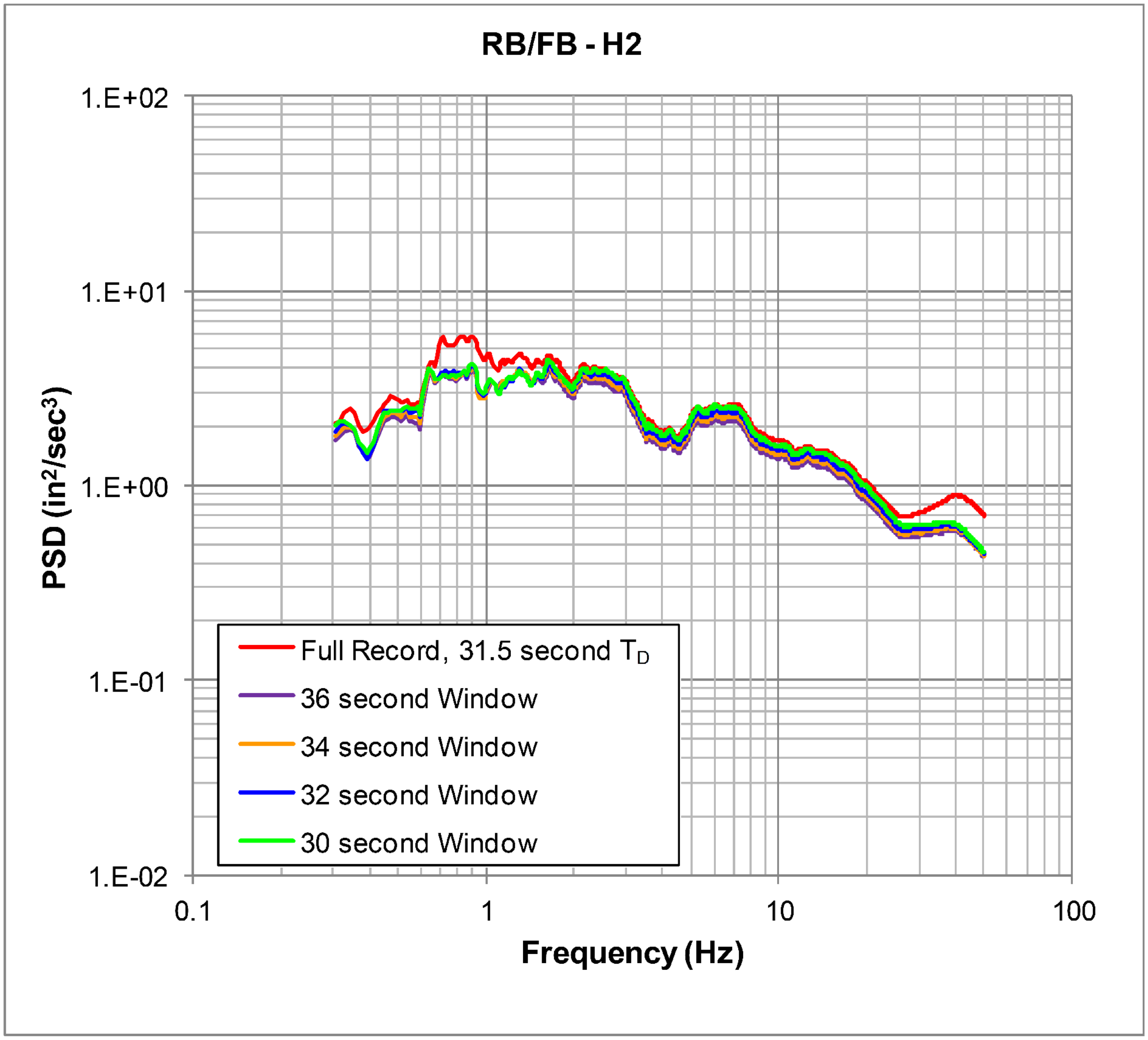


Figure 3.7.1-254 PSD Computed for the CB H1 Component Spectrally Matched Time History (Enhanced SCOR FIRS) Using Full Duration Time Histories and Time Histories Windowed to an Equivalent Stationary Duration, T_D

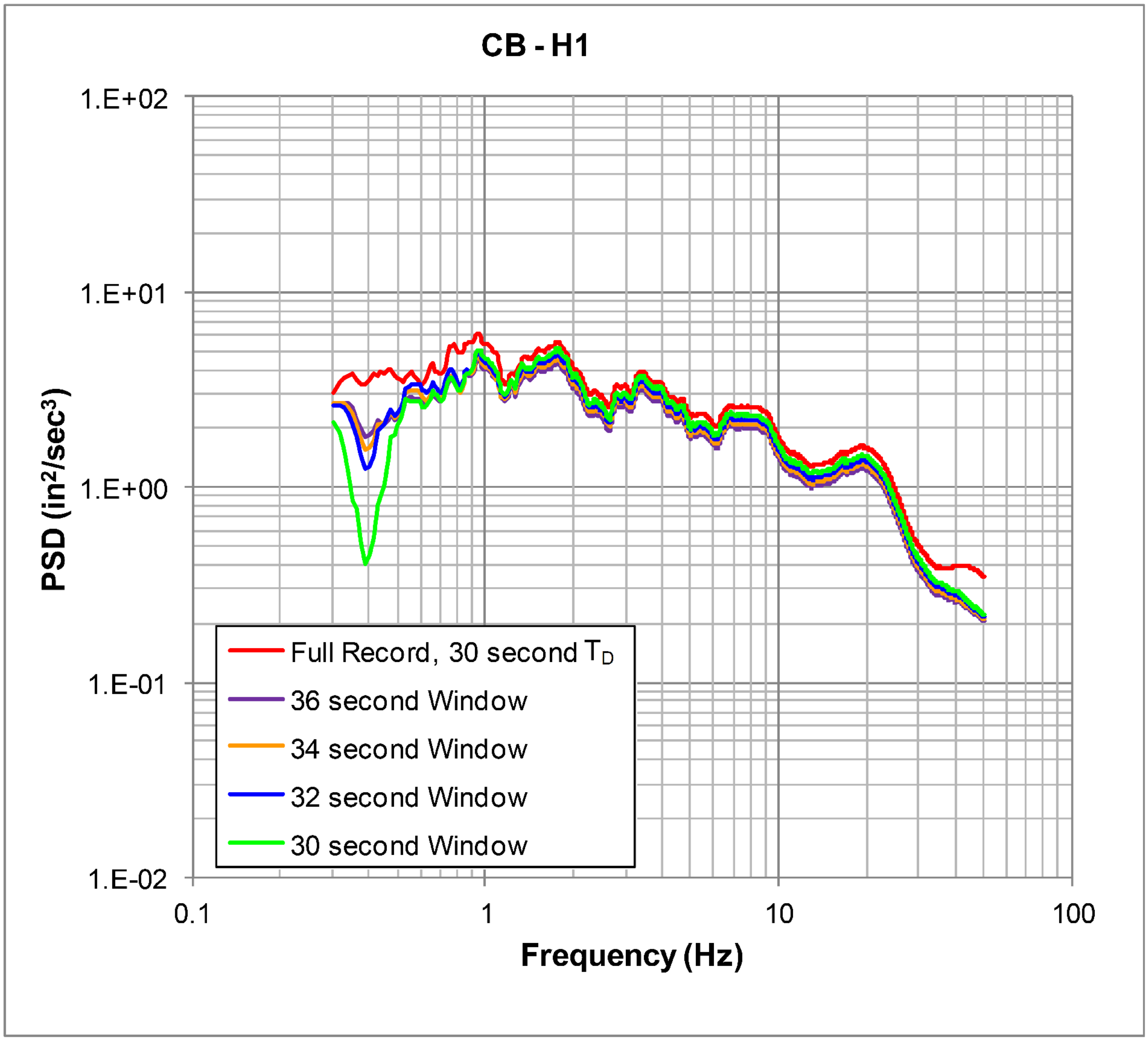


Figure 3.7.1-255 PSD Computed for the CB H2 Component Spectrally Matched Time History (Enhanced SCOR FIRS) Using Full Duration Time Histories and Time Histories Windowed to an Equivalent Stationary Duration, T_D

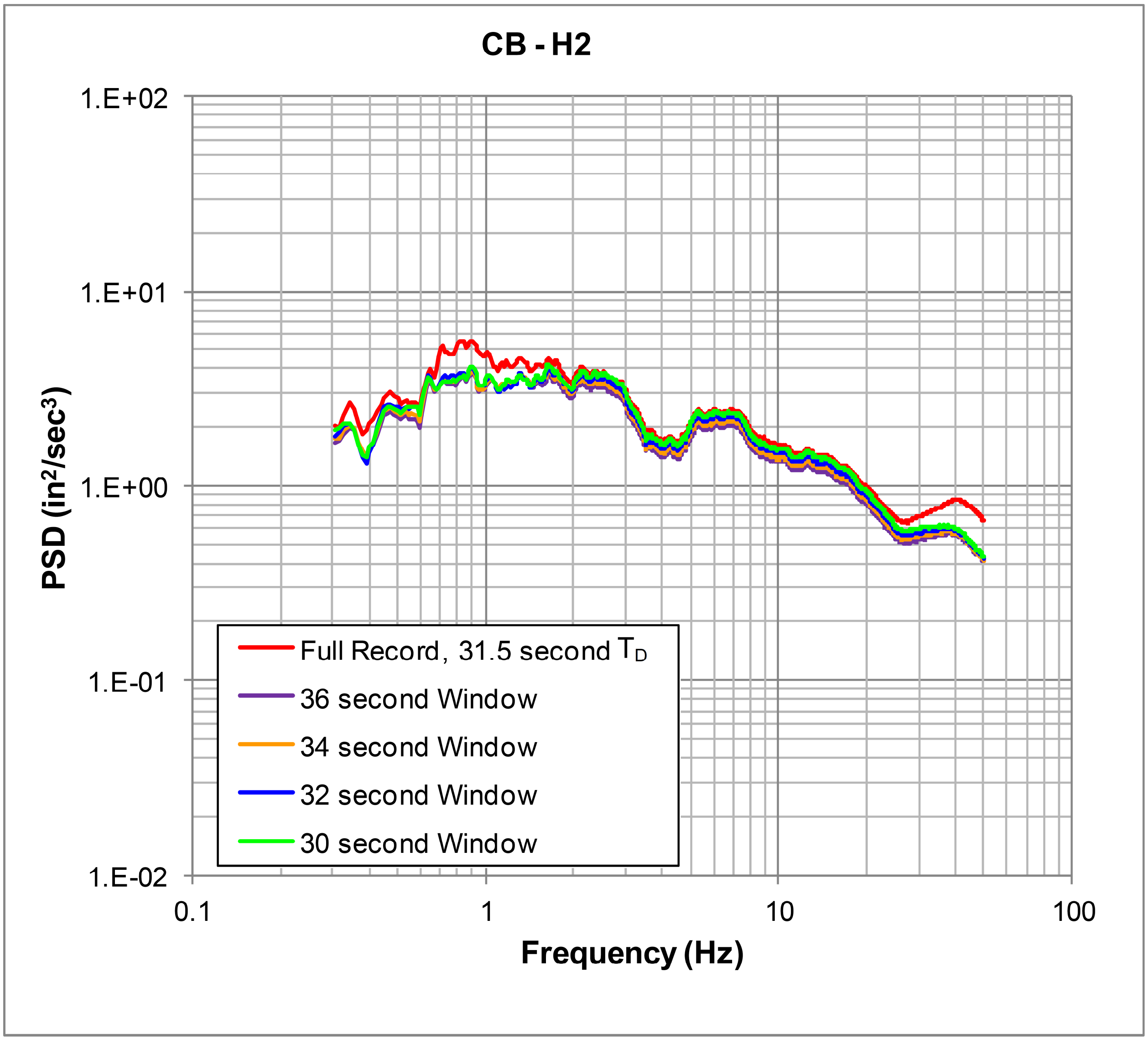


Figure 3.7.1-256 Power Spectra and Cumulative Power Plots for the Horizontal In-Column Acceleration Time Histories (H1 and H2) Compatible with the BE Deterministic Profile without Engineered Granular Backfill above the Top of the Bass Islands Group Bedrock