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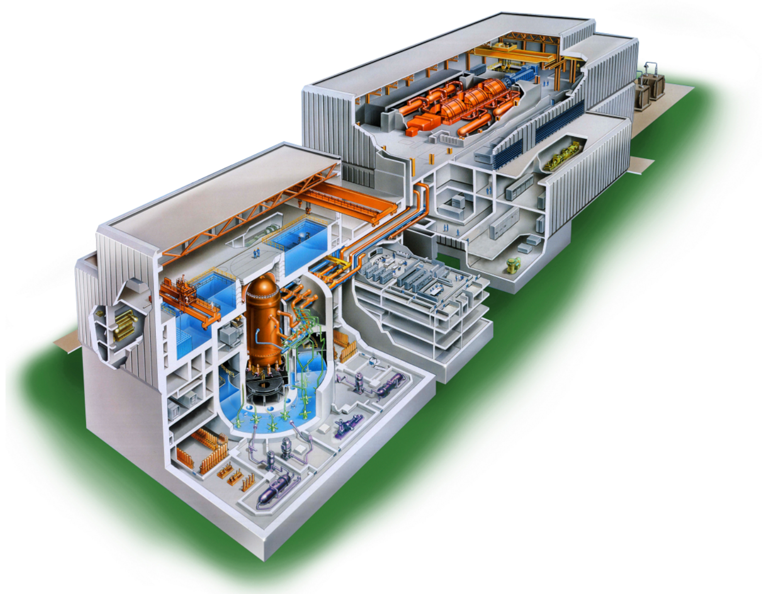

Chapter 3

Design of Structures, Components, Equipment and Systems

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3.0 Design of Structures, Components, Equipment and Systems

3.1 Conformance with NRC General Design Criteria

3.1.1 Summary Description

This section contains an evaluation of the principal design criteria of the ABWR Standard Plant as measured against the NRC General Design Criteria for Nuclear Power Plants, 10CFR50 Appendix A. The general design criteria, which are divided into six groups with the last criterion numbered 64, are intended to establish minimum requirements for the principal design criteria for nuclear power plants.

The NRC General Design Criteria were intended to guide the design of all water-cooled nuclear power plants; separate BWR-specific criteria are not addressed. As a result, the criteria are subject to a variety of interpretations. For this reason, there are some cases where conformance to a particular criterion is not directly measurable. In these cases, the conformance of the ABWR design to the interpretation of the criteria is discussed. For each criterion, a specific assessment of the plant design is made and a complete list of references is included to identify where detailed design information pertinent to that criterion is treated in this standard safety analysis report.

Based on the contents herein, the design of the ABWR design fully satisfies and is in compliance with the NRC General Design Criteria.

3.1.2 Evaluation Against Criteria

3.1.2.1 Group I—Overall Requirements

3.1.2.1.1 Criterion 1—Quality Standards and Records

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

3.1.2.1.1.2 Evaluation Against Criterion 1

Safety-related and non-safety-related structures, systems, and components are identified on Table 3.2-1. The total QA program is described in Chapter 17 and is applied to the safety-related items. The quality requirements for non-safety-related items are controlled by the QA program described in Chapter 17 in accordance with the functional importance of the item. The intent of the QA 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 program assures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. It also includes the observance of proper preoperational and operational testing and maintenance procedures, as well as the documentation of the foregoing by keeping appropriate records. The total QA program is responsive to and in conformance with the intent of the quality-related requirements of 10CFR50 Appendix B.

Structures, systems, and components are identified in Section 3.2 with respect to their location, service and their relationship to the safety-related or non-safety-related function to be performed. Recognized codes and standards are applied to the equipment per the safety classifications to assure meeting the required safety-related function.

Documents are maintained which demonstrate that all the requirements of the QA program are being satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are observed, specified materials are used, correct procedures are utilized, qualified personnel are provided, and the finished parts and components meet the applicable specifications for safe and reliable operation. These records are available so that any desired item of information is retrievable for reference. These records will be maintained during the life of the operating licenses.

The detailed QA program is in conformance with the requirements of Criterion 1.

For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
3.2	Classification of Structures, Components, and Systems

3.1.2.1.2 Criterion 2—Design Bases for Protection Against Natural Phenomena

3.1.2.1.2.1 Criterion 2 Statement

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without 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 combinations 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.

3.1.2.1.2.2 Evaluation Against Criterion 2

Since the ABWR design is designated as a standard plant, the design bases for safety-related (Subsection 3.1.2.1.1.2) structures, systems, and components, cannot accurately reflect the most severe of the natural phenomena that have been historically reported for each possible site and their surrounding areas. However, the envelope of site-related parameters which blankets the majority of potential sites in the contiguous United States is defined in Chapter 2. The design bases for these structures, systems, and components reflect this envelope of natural phenomena, including appropriate combinations of the effects of normal and accident conditions with this envelope. The design bases is not required to meet the requirements of Criterion 2.

Detailed discussions of the various phenomena considered and design criteria developed are presented in the following sections:

Chapter/Section	Title
2.0	Summary of Site Characteristics
3.2	Classification of Structures, Components, and Systems
3.3	Wind and Tornado Loadings
3.4	Water Level (Flood) Design
3.5	Missile Protection
3.7	Seismic Design
3.8	Design of Seismic Category I Structures
3.9	Mechanical Systems and Components

Chapter/Section	Title
3.10	Seismic Qualifications of Seismic Category I Instrumentation and Electrical Equipment
3.11	Environmental Qualification of Safety-Related Mechanical and Electrical Equipment
7.1	Table 7.1-2

3.1.2.1.3 Criterion 3—Fire Protection

3.1.2.1.3.1 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 whenever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

3.1.2.1.3.2 Evaluation Against Criterion 3

Fires in the plant are prevented or mitigated by the use of non-combustible 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 objectionable 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 a fire.

Combustible supplies, such as logs, records, manuals, etc., are limited in such areas as the control room to amounts required for current operation, thus limiting the effect of a fire or explosion.

The plant Fire Protection System includes the following provisions:

(1) Automatic fire detection equipment in those areas where fire danger is greatest.

(2) Extinguishing services which include automatic actuation with manual override as well as manually-operated fire extinguishers.

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

Chapter/Section	Title
3.8.2.6	Materials, Quality Control and Special Construction Techniques
7	Instrumentation and Control Systems
8	Electric Power
9.5	Fire Protection System
Appendix 9A	Fire Hazard Analysis
13	Conduct of Operations

3.1.2.1.4 Criterion 4—Environmental and Dynamic Effects Design Bases

3.1.2.1.4.1 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 (LOCAs). These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

3.1.2.1.4.2 Evaluation Against Criterion 4

Essential (see introduction to Section 3.6) structures, systems, and components are designed to accommodate the dynamic effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, and postulated pipe failure accidents, including LOCAs.

These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failure. The effects of missiles from sources external to the ABWR Standard Plant are also considered. Design requirements specify the time which each must survive the extreme environmental conditions following a LOCA. The design of these structures, systems, and components meets the requirements of Criterion 4.

Subsection 3.6.3 identifies the requirements for the piping that is to be excluded from postulation of pipe ruptures for design of the plant against dynamic effects from the associated pipe ruptures.

For further discussion, see the following sections:

Chapter/Section	Title
2.0	Summary of 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	Design of Seismic Category I Structures
3.11	Environmental Qualification of Safety-Related Mechanical and Electrical Equipment
5.2	Integrity of Reactor Coolant Pressure Boundary
6	Engineered Safety Features
7	Instrumentation and Control Systems
8	Electric Power

3.1.2.1.5 Criterion 5—Sharing of Structures, Systems, and Components

3.1.2.1.5.1 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 will not significantly impair the ability to perform the safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

3.1.2.1.5.2 Evaluation Against Criterion 5

Since the ABWR design is for a single-unit station, this criterion is not applicable.

3.1.2.2 Group II—Protection by Multiple Fission Product Barriers

3.1.2.2.1 Criterion 10—Reactor Design

3.1.2.2.1.1 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.

3.1.2.2.1.2 Evaluation Against Criterion 10

The reactor core components consist of fuel assemblies, control rods, incore 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 provide integrity over a complete range of power levels, including transient conditions. The core is sized with sufficient heat transfer area and coolant flow to ensure that fuel design limits are not exceeded under normal conditions or anticipated operational occurrences.

The Reactor Protection System (RPS) is designed to monitor certain reactor parameters, sense abnormalities, and scram the reactor, thereby preventing fuel design limits from being exceeded when trip points are exceeded. Scram trip setpoints are selected on operating experience and by the safety design basis. There is no case in which the scram trip setpoints allow the core to exceed the thermal-hydraulic safety limits. Power for the RPS is supplied by four independent uninterruptible AC power supplies. An alternate power source and battery are available for each bus. The reactor will scram on loss of power or hydraulic pressure.

An analysis and evaluation has been made of the effects upon core fuel following adverse plant operating conditions. The results of abnormal operational transients (Chapter 15) show that the minimum critical power ratio (MCPR) does not fall below the transient MCPR limit, thereby satisfying the transient design basis.

The reactor core and associated coolant, control, and protection systems are designed to assure that the specified acceptable 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
1.2	General Plant Description
4.2	Fuel Design System
4.3	Nuclear Design

4.4	Thermal and Hydraulic Design
5.4.1	Reactor Recirculation System
5.4.6	Reactor Core Isolation Cooling System
5.4.7	Residual Heat Removal System
7.2	Reactor Protection System
7.3	Engineered Safety Feature Systems—Instrumentation and Control
15	Accident Analyses

3.1.2.2.2 Criterion 11—Reactor Inherent Protection

3.1.2.2.2.1 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.

3.1.2.2.2.2 Evaluation Against Criterion 11

The reactor core is designed to have a reactivity response that regulates or damps changes in power level and spatial distributions of power production to a level consistent with safe and efficient operation.

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

(1) Fuel temperature or Doppler coefficient.

(2) Moderator void coefficient.

(3) Moderator temperature coefficient.

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

Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it; it contributes to system stability. Since the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator void coefficient to Doppler coefficient for optimum load-following capability. The BWR has an inherently large moderator-to-Doppler coefficient ratio which permits use of coolant flow rate for load following.

In a BWR, the moderator void coefficient is of importance during operation at power. Nuclear design requires the void coefficient inside the fuel channel to be negative. The negative void reactivity coefficient provides an inherent negative feedback during power transients. Because of the large negative moderator coefficient of reactivity, the BWR has a number of inherent advantages, such as:

(1) The use of coolant flow as opposed to control rods for load following.

(2) The inherent self-flattening of the radial power distribution.

(3) The ease of control.

(4) The spatial xenon stability.

The reactor is designed so that the moderator temperature coefficient is small and positive in the cold condition; however, the overall power reactivity coefficient is negative. Typically, the power coefficient at full power is about –0.04 $\Delta k/k/\ \Delta P/P$ at the beginning of life and about -0.03 $\Delta k/k/\ \Delta P/P$ at 10,000 MW d/t. These values are 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 tends to compensate for any rapid increase in reactivity in accord with Criterion 11.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
7.2	Reactor Protection System, Instrumentation and Control
7.3	Engineered Safety Feature Systems, Instrumentation and Control
7.7	Control Systems not Required for Safety
15	Accident Analyses

3.1.2.2.3 Criterion 12—Suppression of Reactor Power Oscillations

3.1.2.2.3.1 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.

3.1.2.2.3.2 Evaluation Against Criterion 12

The reactor core is designed to ensure that no power oscillation will cause fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient to the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large BWRs, under-damped, unacceptable power distribution behavior could only be expected to occur with power coefficients more positive than about $-0.01\ \Delta k/k/\ \Delta P/P$. Operating experience has shown large BWRs to be inherently stable against xenon induced power instability. The large negative coefficients provide:

(1) Good load following with well-damped behavior and little undershoot or overshoot in the heat transfer response.

(2) Load following with recirculation flow control.

(3) Strong damping of spatial power disturbances.

The RPS design provides protection from excessive fuel cladding temperatures and protects the reactor coolant pressure boundary (RCPB) from excessive pressures which threaten the integrity of the system. Local abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundance, power supply redundancy, and physical separation.

The reactor core and associated coolant, control, and protection systems are designed to suppress any power oscillations which could result in exceeding fuel design limits. These systems assure that Criterion 12 is met.

For further discussions, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
7.2	Reactor Protection System—Instrumentation and Control
7.3	Engineered Safety Feature Systems—Instrumentation and Control
7.7	Control Systems not Required for Safety
15	Accident Analyses

3.1.2.2.4 Criterion 13—Instrumentation and Control

3.1.2.2.4.1 Criterion 13 Statement

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for (1) normal operation, (2) anticipated operational occurrences, and (3) 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 RCPB, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3.1.2.2.4.2 Evaluation Against Criterion 13

The neutron flux in the reactor core is monitored by five 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 from 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 Flow Rate Subsystem (FRS) provides the control and reference signal for the APRM core flow-rate dependent trips. The Automated Traversing Incore Probe (ATIP) Subsystem provides a means for calibrating the LPRM Subsystem. 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 circuity from affecting the scram circuitry.

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and RCPB, the Leak Detection and Isolation System (LDS) initiates automatic isolation of appropriate pipelines whenever monitored variables exceed preselected operational limits.

The leakage limits for the Reactor Coolant System (Subsection 3.1.2.2.6.2) are established so that appropriate action can be taken to ensure the integrity of the RCPB. The monitored leakage rates are classified as identified and unidentified, which corresponds, respectively, to the flow to the equipment drain and floor drain sumps. The permissible total leakage rate limit to these sumps is based upon the makeup capabilities of various reactor component systems. High pump fill-up rate and pump-out rate are alarmed in the main control room. The unidentified leakage rate (Chapter 5) is less than the value that has been conservatively calculated to be a minimum leakage from a crack large enough to propagate rapidly, but which still allows time for identification and corrective action before integrity of the process barrier is threatened.

The Process Radiation Monitoring System monitors radiation levels of various processes and provides trip signals to the RPS and LDS 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.

Additional information on the instrumentation and controls is given in Chapter 7.

3.1.2.2.5 Criterion 14—Reactor Coolant Pressure Boundary

3.1.2.2.5.1 Criterion 14 Statement

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

3.1.2.2.5.2 Evaluation Against Criterion 14

The piping and equipment pressure parts within the RCPB (as defined by Section 50.2 of 10CFR50) are designed, fabricated, erected, and tested in accordance with 10CFR50.55a to provide a high degree of integrity throughout the plant lifetime. Section 3.2 classifies systems and components within the RCPB as Quality Group A. The design requirements and codes and standards applied to this quality group ensure high integrity in keeping with the safety-related function.

In order 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 ASME Boiler and

Pressure Vessel (B&PV) Code Section III, where applicable. Where RCPB piping penetrate 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 screwed joints. Welding procedures are employed which produce welds of complete fusion free of unacceptable defects. All welding procedures, welders, and welding machine operators used in producing press-containing welds are qualified in accordance with the requirements of ASME B&PV Code Section IX for the materials to be welded. Qualification 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 General Design Criteria.

The design, fabrication, erection, and testing of the RCPB assure an extremely low probability of failure of abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
3	Design of Structures, Components, Equipment, and Systems
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
5.4.1	Reactor Recirculation System
15	Accident Analyses
17	Quality Assurance

3.1.2.2.6 Criterion 15—Reactor Coolant System Design

3.1.2.2.6.1 Criterion 15 Statement

The Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOO).

3.1.2.2.6.2 Evaluation Against Criterion 15

The RCS, as identified in Section 5.1, consists mainly of the Nuclear Steam Supply Systems (NSSS), comprised of the reactor vessel and appurtenances, the Reactor Recirculation System (RCS) and the Nuclear Boiler System (NBS) including the main steamlines, feedwater lines and pressure-relief discharge system; the Reactor Core Isolation Cooling (RCIC) System; the Residual Heat Removal (RHR) System; and the Reactor Water Cleanup (CUW) System.

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 anticipated operational occurrences. As described in the evaluation of Criterion 13, instrumentation is provided to monitor essential variables to ensure 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 which provides sufficient margin to assure that the design conditions of the RCPB are not exceeded is the automatic initiation of the pressure relief system of the NBS upon receipt of an overpressure signal. To accomplish overpressure protection of the reactor pressure vessel system and RCPB, a number of pressure-operated relief valves are provided that can 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 an LOCA in which the vessel is not depressurized by the accident. The depressurization of the RCS in this situation allows operation of the low-pressure emergency core cooling systems 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.

The application of appropriate codes and standards and high quality requirements to the RCPB and the design features of its associated auxiliary, control, and protection systems assure that the requirements of Criterion 15 are satisfied.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
3	Design of Structure, Components, Equipment, and Systems
5.2.2	Overpressurization Protection
5.2.5	RCPB and Core Cooling Systems Leakage Detection

Chapter/Section	Title
5.3	Reactor Vessel
5.4.1	Reactor Recirculation System
7.3	Engineered Safety Feature Systems—Instrumentation and Control
15	Accident Analyses

3.1.2.2.7 Criterion 16—Containment Design

3.1.2.2.7.1 Criterion 16 Statement

Reactor containment and associated systems shall be provided to establish an essentially leaktight 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.

3.1.2.2.7.2 Evaluation Against Criterion 16

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

(1) A leaktight primary containment vessel (PCV) enclosing the reactor pressure vessel, the RCPB, and other branch connections of the reactor primary coolant system. The PCV is a cylindrical steel-lined reinforced concrete structure with a removable steel head and has upper and lower drywell zones, diaphragm floor (D/F) and annular suppression chamber (or wetwell zone) under upper drywell separated by the D/F.

(2) A suppression pool containing a large amount of water used to rapidly condense steam from a reactor vessel blowdown or from a break in a major pipe.

(3) Associated containment penetrations and isolation devices.

The drywell and wetwell zones condense the steam and contain fission product releases from the postulated design bases accident (i.e., the double-ended rupture of the largest pipe in the primary coolant system). The leaktight PCV prevents the release of fission products to the environment.

The secondary containment boundary of the reactor building, which completely encloses and structurally integrates the PCV, provides additional radiation shielding to protect operating personnel and the public and also protects the PCV from weather and external missiles.

Temperature and pressure in the PCV are limited following an accident by using the RHR System to condense steam in the containment atmosphere and to cool the suppression pool water.

The design of the containment systems meets the requirements of Criterion 16.

For further discussion, see the following sections.

Chapter/Section	Title
1.2	General Plant Description
3.8.2	Steel Components of the Reinforced Concrete Containment
6.2	Containment Systems
15	Accident Analyses

3.1.2.2.8 Criterion 17—Electric Power Systems

3.1.2.2.8.1 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 for 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 RCPB are not exceeded as a result of anticipated operational occurrences.

(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 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 (AC) power supplies and the other offsite electric power circuit to assure that specified acceptable fuel design limits and design conditions of the RCPB are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that the 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 (1) the loss of power generated by the nuclear power unit, (2) the loss of power from the transmission network, or (3) the loss of power from the onsite electric power supplies.

3.1.2.2.8.2 Evaluation Against Criterion 17

3.1.2.2.8.2.1 Onsite Electric Power System

There are three independent AC load groups provided to assure independence and redundancy of equipment function. These meet the safety requirements, assuming a single failure, since:

(1) Each load group is independently capable of isolation from the offsite power sources.

(2) Each load group has separate circuits to independent power sources.

For each of the three AC load groups, there are independent batteries which furnish DC load and control power for the corresponding divisions. An additional battery furnishes DC load and control power for the safety system logic and control (SSLC) Division IV bus.

The reactor protection instrumentation is powered from four independent AC/DC power sources.

The onsite electric power systems are designed to meet the requirements of Criterion 17. For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
3.10	Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment
3.11	Environmental Qualification of Safety-Related Mechanical and Electrical Equipment
8.3	Onsite Power Systems

3.1.2.2.8.2.2 Offsite Electric Power System

A part of the design of the offsite power systems is out of the scope of the ABWR design. A description of the offsite power system and the scope split between the ABWR Standard Plant design and the COL applicant design is defined in Subsection 8.2.1.1 and 8.2.1.2. The ABWR Standard Plant interfaces requirements are addressed in Subsection 8.2.3.

3.1.2.2.9 Criterion 18—Inspection and Testing of Electric Power Systems

3.1.2.2.9.1 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 periodically test:

(1) The operability and functional performance of the components of the systems such as onsite power sources, relays, switches, and buses.

(2) The operability of the systems as a whole and, under conditions as close to design as practical, the full operational 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.

3.1.2.2.9.2 Evaluation Against Criterion 18

The important power supply buses and associated normal preferred, alternate, and standby AC power supplies are arranged for periodic inspection and testing of each load group independently. The testing procedure includes a bus transfer from normal preferred power supply to alternate preferred power supply, and simulates a loss of preferred power (LOPP) signal or a LOCA signal to start the diesel generator bringing it to operating condition. Full load testing of the diesel generator can be performed by manually synchronizing the generator to the normal preferred power supply. These tests are performed periodically to prove the engineered safety system operability.

Design of the standby power systems provides testability in accordance with the requirements of Criterion 18. For further discussion, see the following sections:

Chapter/Section	Title
8.3	Onsite Power Systems
14	Initial Test Program

3.1.2.2.10 Criterion 19—Control Room

3.1.2.2.10.1 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.

(2) With a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

3.1.2.2.10.2 Evaluation Against Criterion 19

The control room contains the following equipment: (1) controls and necessary surveillance equipment for operation of the plant functions such as the reactor and its auxiliary systems, (2) engineered safety features, (3) turbine generator, (4) steam and power conversion systems, and (5) station electrical distribution boards.

The control room is located in a Seismic Category I Control Building. Safe occupancy of the control room during abnormal conditions is provided for in the design. Adequate shielding is provided to maintain tolerable radiation levels in the control room in the event of a design basis accident for the duration of the accident.

The control building ventilation system has redundant equipment and provides radiation detectors and smoke detectors with appropriate alarms and interlocks. The control room intake air can be filtered through high-efficiency particulate air/absolute (HEPA) and charcoal filters.

The control room is continuously occupied by qualified operating personnel under all operating and accident conditions. In the unlikely event that the control room must be vacated and access is restricted, instrumentation and controls are provided outside the control room which 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
1.2	General Plant Description
3.8.4	Other Seismic Category I Structures
7	Instrumentation and Control Systems

Chapter/Section	Title
7.4.1.4 and 7.4.2.4	Remote Shutdown System— Instrumentation and Controls
6.4	Habitability Systems
9.4.1	Control Building Ventilation System
9.5.1	Fire Protection System
12.3.2	Shielding
12.3.3	Ventilation

3.1.2.3 Group III—Protection and Reactivity Control System

3.1.2.3.1 Criterion 20—Protection System Functions

3.1.2.3.1.1 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.

(2) To sense accident conditions and initiate the operation of systems and components important to safety.

3.1.2.3.1.2 Evaluation Against Criterion 20

The Reactor Protection System (RPS) is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB barrier. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored variables of nuclear steam supply systems (Subsection 3.1.2.2.6.2) exceed pre-established limits of anticipated operational occurrences. 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 ride through power sources, sensors, transmitters, bypass circuity, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by neutron monitoring system variables, nuclear steam supply systems (NSSS) high pressure, high suppression pool temperature, turbine stop valve closure, turbine control valve fast closure, and reactor vessel low-water level 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. Control rod scram motion starts in about 290 milliseconds after the high flux setpoint is exceeded.

A fully withdrawn control rod traverses 60% of its full stroke in sufficient time to assure that acceptable fuel design limits are not exceeded.

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 initiate automatically the operation of other systems and components important to safety. Systems such as the Emergency Core Cooling System (ECCS) are initiated automatically to limit the extent of fuel damage following a LOCA. 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 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
1.2.1	Principal Design Criteria
4.6	Functional Design of Reactivity Control Systems
5.2.2	Overpressurization Protection
5.4.5	Main Steamline Isolation System
6.3	Emergency Core Cooling System
7.2	Reactor Protection System
7.3.1.1 and 7.3.2.1	Emergency Core Cooling Systems— Instrumentation and Control
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls

Chapter/Section	Title
7.6.1.2 and 7.6.2.2	Process Radiation Monitoring and System—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.2 Criterion 21—Protection System Reliability and Testability

3.1.2.3.2.1 Criterion 21 Statement

The protection system shall be designed for 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.

(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.

3.1.2.3.2.2 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, other monitored components scram the reactor if their 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 inservice 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 preestablished limits. This system is arranged as four separately powered divisions.

Each division has a logic which 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 test verifies the ability to de-energize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive 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 nuclear steam supply systems at most power levels. One control rod is tested at a time. Control rod mechanism overdrive demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control room instrumentation. More importantly, the HCU scram accumulator level is continuously monitored.

The main steamline isolation valves may be tested during full reactor operation. Individually, they can be closed to 90% of full-open position without affecting the reactor operation. If reactor power is reduced sufficiently, the isolation valves may be fully closed. During refueling operation, valve leakage rates can be determined.

The RHR System testing can be performed during normal operation. Main system pumps can be evaluated by taking suction from the suppression pool and discharging through test lines back to the suppression pool. System design and operating procedures also permit testing the supply valves of the three RHR lines. The lower pressure flooder mode can be tested after reactor shutdown. Each active component of the ECCS provided to operate in a design basis accident is designed to be operable for test purposes during normal operation of the nuclear system.

The high functional reliability, redundancy, and inservice testability of the protection system satisfy the requirements specified in Criterion 21.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.6	Functional Design of Reactivity Control Systems
5.4.5	Main Steamline Isolation Valve System
5.4.7	Residual Heat Removal System
6.2	Containment Systems
6.3	Emergency Core Cooling Systems

Chapter/Section	Title
7.2	Reactor Protection System
7.3.1.1 and 7.3.2.1	Emergency Core Cooling Systems—Instrumentation and Control
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls
7.6.1.2 and 7.6.2.2	Process Radiation Monitoring—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.3 Criterion 22—Protection System Independence

3.1.2.3.3.1 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.

3.1.2.3.3.2 Evaluation Against Criterion 22

Components of protection systems are designed so that the mechanical, thermal and radiological environment 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 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 an independent input for each actuator logic. When an essential 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, calibration 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
1.2.1	Principal Design Criteria
4.6	Functional Design of Reactivity Control Systems
5.4.5	Main Steamline Isolation Valve System
5.4.7	Residual Heat Removal System
6.3	Emergency Core Cooling Systems
7.2	Reactor Protection System
7.3.1.1 and 7.3.2.1	Emergency Core Cooling System—Instrumentation and Controls
7.3.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls
7.6.1.2 and 7.6.2.2	Process Radiation Monitoring— Instrumentation and Controls
15	Accident Analyses

3.1.2.3.4 Criterion 23—Protection System Failure Modes

3.1.2.3.4.1 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.

3.1.2.3.4.2 Evaluation Against Criterion 23

The RPS is designed to fail into a safe state. Use of an independent channel for each actuator logic allows the system to sustain any logic channel failure without preventing other sensors monitoring the same variable from initiating a scram. Any two-out-of-four logic channel trips initiate a scram. Intentional bypass for maintenance or testing causes the scram logic to revert to two-out-of-three. A failure of any one RPS input or subsystem component produces a trip in one channel. This condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function upon trip of another channel.

The environmental conditions in which the instrumentation and equipment of the RPS 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 failure modes of the RPS are such that it fails into a safe state as required by Criterion 23.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
7.2	Reactor Protection System

3.1.2.3.5 Criterion 24—Separation of Protection and Control Systems

3.1.2.3.5.1 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.

3.1.2.3.5.2 Evaluation Against Criterion 24

There is separation between the RPS and the process control systems. Logic channels 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 non-safety channel interface. Therefore, failure in the controls and instrumentation of process systems cannot induce failure of any portion of the protection system. Scram reliability is designed into the RPS and hydraulic control unit for the control rod drive. The scram signal and mode of operation override all other signals.

The systems that isolate the containment and reactor pressure vessel 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 essential variables.

Process radiation monitoring is provided on process liquid and gas lines that may serve as discharge routes for radioactive materials. Four instrumentation channels are used to prevent an inadvertent scram and isolation as a result of instrumentation malfunctions. The output trip signals from each channel are combined in such a way that two channels must signal high radiation to initiate scram and main steam isolation.

The protection system is separated from control systems as required in Criterion 24.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.6	Functional Evaluation of Reactivity Control Systems
6.3	Emergency Core Cooling Systems
7.2	Reactor Trip System
7.3.1.1 and 7.3.2.1	Emergency Core Cooling System— Instrumentation and Controls
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls
7.6.1.2 and 7.6.2.2	Process Radiation Monitoring—and Instrumentation and Controls
7.7.1.2 and 7.7.2.2	Rod Control and Information System—Instrumentation and Controls

3.1.2.3.6 Criterion 25—Protection System Requirements for Reactivity Control Malfunctions

3.1.2.3.6.1 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 system such as accidental withdrawal (not ejection or dropout) of control rods.

3.1.2.3.6.2 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 or the primary containment vessel pressure boundary. Any monitored variable which exceeds the scram setpoint will initiate an automatic scram and not impair the remaining variables from being monitored, and if one channel fails, the remaining portions of the RPS shall function.

The Rod Control and Information System (RCIS) is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the RCIS is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the RCIS circuitry from affecting the scram circuitry. Because only two control rods are controlled by an individual hydraulic control unit (HCU), a failure that results in continued energizing of an insert solenoid valve on an HCU can affect only 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
1.2.1	Principal Design Criteria
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
4.6	Functional Design of Reactivity Control Systems
7.2	Reactor Trip System

Chapter/Section	Title
7.7.1.2 and 7.7.2.2	Rod Control and Information System—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.7 Criterion 26—Reactivity Control System Redundancy and Capability

3.1.2.3.7.1 Criterion 26 Statement

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

3.1.2.3.7.2 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) powder. Positive insertion of these control rods is provided by means of the control rod drive electrical and hydraulic system. 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 operator-controlled insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram will not adversely affect the capability to maintain the core within fuel design limits.

The circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for 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 the fine motion control rod drives, FMCRDs) provide needed control rod insertion performance over the entire range of reactor pressure (i.e., from operating conditions to cold shutdown). The design of the control rod 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 rod worths. The operating procedures to accomplish such patterns are supplemented by the rod pattern control system, which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected 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. In the event that a reactor scram is necessary, the unlikely occurrence of a limited number of stuck rods will not hinder the capability of the control rod system to render the core subcritical.

A standby liquid control 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 a subcritical condition at any time during the core life. The reactivity control provided to reduce reactor power from rated power to a shutdown condition with the control rods withdrawn in the power pattern accounts for the reactivity effects of 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 control rod system is capable of holding the reactor core subcritical under cold conditions, even when the number of control rods of highest worth controlled by an hydraulic control unit is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing the fuel with burnable poison (Gd_2O_3) to control the high reactivity of fresh fuel.

The redundancy and capabilities of the reactivity control systems for the ABWR satisfy the requirements of Criterion 26.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.6	Functional Design of Reactivity Control Systems
7.3	Engineered Safety Feature Systems
7.4.1.2 and 7.4.2.2	Standby Liquid Control System—Instrumentation and Controls
7.7.1.2 and 7.7.2.2	Rod Control and Information System— Instrumentation and Controls

3.1.2.3.8 Criterion 27—Combined Reactivity Control Systems Capability

3.1.2.3.8.1 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 systems 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.

3.1.2.3.8.2 Evaluation Against Criterion 27

There is no credible event applicable to the ABWR which requires combined capability of the control rod system and poison additions. The ABWR design is capable of maintaining the reactor core subcritical, including allowance for a pair of stuck rods controlled by a hydraulic control unit (HCU), without addition of any poison to the reactor coolant. The primary reactivity control system for the ABWR during postulated accident conditions is the control rod 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 scram and manual control circuitry, individual HCU controlling a pair of control rods, and fail-safe design features built into the rod drive system. Response by the RPS is prompt and the total scram time is short.

In the unlikely event that more than one control rod fails to insert and the core cannot be maintained in a subcritical condition by control rods alone as the reactor is cooled down subsequent to initial shutdown, the Standby Liquid Control System (SLCS) can be actuated to insert soluble boron into the reactor core. The SLCS has sufficient capacity to ensure that the reactor can always be maintained subcritical; and, hence, only decay heat will be generated by the core which can be removed by the Residual Heat Removal (RHR) System, thereby ensuring that the core will always be coolable.

The design of the reactivity control systems assures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability to cool the core is maintained under all postulated accident conditions; thus, Criterion 27 is satisfied.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design

Chapter/Section	Title
4.6	Functional Design of Reactivity Control System
7.2	Reactor Trip System
7.4.1.2 and 7.4.2.2	Standby Liquid Control System—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.9 Criterion 28—Reactivity Limits

3.1.2.3.9.1 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 RCPB 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, steamline rupture, changes in reactor coolant temperature and pressure, and cold water addition.

3.1.2.3.9.2 Evaluation Against Criterion 28

The control rod system design incorporates 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 worths. The Rod Pattern Control System (RPCS) prevents withdrawal other than by the preselected rod withdrawal pattern. The RPCS function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low-power level operation control rod procedures.

The control rod mechanical design incorporates a passive brake and hydraulic inlet check valve which individually, prevents rapid rod ejection. The brake spring force holds the rod 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. Normal rod movement and the rod withdrawal rate is limited through the fine motion control motor.

The accident analysis (Chapter 15) evaluates the postulated reactivity accidents, as well as abnormal operational transients in detail. Analyses are included for rod dropout, steamline rupture, 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 RCPB. In addition, the integrity of the core, its support structures or other reactor pressure vessel internals is maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident analysis.

The design features of the reactivity control system which limit the potential amount and rate of reactivity increase ensure that Criterion 28 is satisfied for all postulated reactivity accidents.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
3.9.4	Control Rod Drive System
3.9.5	Reactor Pressure Vessel Internals
4.3	Nuclear Design
4.5.3	Control Rod Drive Housing Supports
4.6	Functional Design of Reactivity Control Systems
5.2.2	Overpressurization Protection
5.3	Reactor Vessel
5.4.4	Main Steamline Flow Restrictors
5.4.5	Main Steamline Isolation Valves
7.7.1.2 and 7.7.2.2	Rod Control and Information System—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.10 Criterion 29—Protection Against Anticipated Operational Occurrences

3.1.2.3.10.1 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.

3.1.2.3.10.2 Evaluation Against Criterion 29

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

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure. Components important to safety, such as control rod drives, MSIVs, RHR pumps, RCIC, etc., are testable during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analysis, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering the failure probabilities of individual components and the reliability effects during individual component testing on the portion of the system not undergoing test. The capability for inservice 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 anticipated operational occurrences satisfy the requirements of Criterion 29.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
5.4.5	Main Steamline Isolation Valve System
5.4.6	Reactor Core Isolation Cooling
5.4.7	Residual Heat Removal System

Chapter/Section	Title
6.2	Containment Systems
6.3	Emergency Core Cooling Systems
7.2	Reactor Trip System
7.3	Engineered Safety Feature Systems
15	Accident Analyses

3.1.2.4 Group IV—Fluid Systems

3.1.2.4.1 Criterion 30—Quality of Reactor Coolant Pressure Boundary

3.1.2.4.1.1 Criterion 30 Statement

Components which are part of the RCPB 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.

3.1.2.4.1.2 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 3.1.2.2.5.2). Accordingly, components which 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 is 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 Reactor Coolant System (RCS). The Leak Detection and Isolation System (LDS) consists of sensors and instruments to detect, annunciate, and, in some cases, isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. 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 make up the RCS the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal AC power with loss of feedwater supply, makeup capabilities are provided by the RCIC System. While the LDS provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The RCPB and the LDS are designed to meet requirements of Criterion 30.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
3.2	Classification of Structures, Components, and Systems
5.2.2	Overpressurization Protection
5.2.5	Detection of Reactor Coolant Leakage Through RCPB
5.3	Reactor Vessel
5.4.1	Reactor Recirculation Pumps
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls
7.7.1.1	Reactor Vessel Instrumentation
17	Quality Control System

3.1.2.4.2 Criterion 31—Fracture Prevention of Reactor Coolant Pressure Boundary

3.1.2.4.2.1 Criterion 31 Statement

The RCPB 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.

(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 effects of irradiation on material properties.

(3) Residual, steady-state, and transient stresses.

(4) Size of flaws.

3.1.2.4.2.2 Evaluation Against Criterion 31

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

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about $1x10^{17}$ nvt with neutron of energies in excess of 1.6022E-13J.

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 of 100% for the plant life time, the neutron fluence at the inner surface of the vessel causes a slight shift in the transition temperatures. 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 assume that NDT temperature shifts are accounted for in the reactor operation.

The RCPB is designed, maintained, and tested to provide adequate assurance that the boundary will behave 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
3	Design of Structures, Components, Equipment and Systems
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel

3.1.2.4.3 Criterion 32—Inspection of Reactor Coolant Pressure Boundary

3.1.2.4.3.1 Criterion 32 Statement

Components which are part of the RCPB 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.

3.1.2.4.3.2 Evaluation Against Criterion 32

The reactor pressure vessel design and engineering effort include provisions for inservice inspection. Removable plugs in the reactor shield wall and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system safety/relief valves, and on the main steam and feedwater systems 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 Inservice Inspection Plan, access provisions, and areas of restricted access.

Vessel material surveillance samples will be located within the reactor pressure vessel. The program will include specimens of the base metal, weld metal, and heat-affected zone metal.

The plant testing and inspection program ensure that the requirements of Criterion 32 will be met.

For further discussion, see the following sections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
5.2	Integrity of Reactor Coolant Pressure Boundary

3.1.2.4.4 Criterion 33—Reactor Coolant Makeup

3.1.2.4.4.1 Criterion 33 Statement

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

3.1.2.4.4.2 Response to Criterion 33

Means are provided for detecting reactor coolant leakage. The LDS consists of sensors and instruments to detect, annunciate, and, in some cases, isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. 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 predicted and experimentally determined behavior of cracks in pipes, the ability to make up reactor coolant leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal AC power containment with a loss of feedwater supply, makeup capabilities are provided by the RCIC System.

The plant is designed to provide ample reactor coolant makeup for protection against small leaks in the RCPB for anticipated operational occurrences and postulated accident conditions. The design of these systems meets the requirements of Criterion 33.

For further discussion, see the following sections:

Chapter/Section	Title
5.2.5	Detection of Reactor Coolant Leakage Through Reactor Coolant Pressure Boundary
5.4.6	Reactor Core Isolation Cooling System
6.3	Emergency Core Cooling Systems
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls

3.1.2.4.5 Criterion 34—Residual Heat Removal

3.1.2.4.5.1 Criterion 34 Statement

A system to remove residual heat shall be provided. The 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 RCPB are not exceeded.

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

3.1.2.4.5.2 Evaluation Against Criterion 34

The RHR System provides the means to remove decay heat and residual heat from the nuclear steam supply systems (NSSS) so that refueling and servicing of NSSS can be performed.

The major equipment of the RHR System consists of heat exchangers, main system pumps, and service water pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation.

Three independent loops are located in separate protected areas.

Both normal AC power and the auxiliary onsite power system provide power adequate to operate all the auxiliary loads necessary for plant operation. The power sources for the plant auxiliary power system are sufficient in number and of such electrical and physical independence that no single failure will prevent auxiliary systems from supporting two of the three RHR divisions.

The plant auxiliary buses supplying power to engineered safety features and RPS s and those auxiliaries required for safe shutdown are connected by appropriate switching to standby diesel-driven generators located in the plant. Each power source, up to the point of its connection to the auxiliary power buses, is capable of complete and rapid isolation from any other source.

Loads important to plant operation and safety are split and diversified between switchgear sections, and means are provided for detection and isolation of system faults.

The plant layout is designed to effect physical separation of essential bus sections, standby generators, switchgear, interconnections, feeders, power centers, motor control centers, and other system components.

Full capacity standby diesel generators are provided to supply a source of electrical power which is self-contained within the ABWR Standard Plant and is not dependent on external sources of supply. The standby generators produce AC power at a voltage and frequency compatible with the normal bus requirements for essential equipment within the plant. Each of the diesel generators has sufficient capacity to start and carry the essential loads it is expected to drive.

The RHR System is adequate to remove residual heat from the reactor core to assure fuel and RCPB design limits are not exceeded. Redundant reactor coolant circulation paths are available to and from the vessel and RHR System. Redundant onsite electric power systems are provided. The design of the RHR System, including its power supply, meets the requirements of Criterion 34.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
6.3	Emergency Core Cooling System
7.3.1.1.2 and 7.3.2.1	Emergency Core Cooling System—Instrumentation and Controls
8.2	Onsite Power Systems
9.2	Water Systems
15	Accident Analyses

3.1.2.4.6 Criterion 35—Emergency Core Cooling

3.1.2.4.6.1 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 LOCA at a rate such that:

(1) Fuel and clad damage that could interfere with continued effective core cooling is prevented.

(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.

3.1.2.4.6.2 Evaluation Against Criterion 35

The Emergency Core Cooling System (ECCS) consists of the following:

(1) High Pressure Core Flooder (HPCF) System

(2) Reactor Core Isolation Cooling (RCIC)System

(3) Low Pressure Flooder (LPFL) mode of the Residual Heat Removal System (RHR)

(4) Automatic Depressurization System (ADS)

The ECCS is designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the RCPB, including a complete and sudden circumferential rupture of the larger pipe connected to the reactor vessel.

The HPCF System consists of two subsystems, each having a single motor-driven pump, system piping, valves, controls, and instrumentation. The RCIC System consists of similar equipment except that it is a single system and the pump delivering high pressure flow is driven by steam turbine. The HPCF and RCIC Systems assure that the reactor core is adequately cooled to prevent excessive fuel clad temperatures for breaks in the NSSS which do not result in rapid depressurization of the reactor vessel. The HPCF or RCIC System continues to operate when reactor vessel pressure is below the pressure at which the RHR/LPFL System operation maintains core cooling. A source of water is available from either the condensate storage pool or the suppression pool.

The ADS functions to reduce the reactor pressure so that flow from RHR/LPFL enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the Nuclear Boiler System SRVs to relieve the high pressure steam to the suppression pool.

The HPCF System consists of a centrifugal pump that can be powered by normal auxiliary power of the standby AC power system, a core flooder sparger in the reactor vessel above the core, piping and valves to convey water from the condensate storage pool or the suppression pool to the sparger, and associated controls and instrumentation. In case of low-water level in the reactor vessel or high pressure in the drywell, the HPCF System automatically injects water into the vessel in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature.

In case of low-water level in the reactor or high pressure in the drywell, the LPFL mode of operation of the RHR System pumps and injects water into the reactor vessel in time to flood the core and prevent excessive fuel temperature. The RHR System is described in Subsection 3.1.2.4.5.2. Protection provided by RHR/LPFL extends to a small break where the ADS has operated to lower the reactor vessel pressure.

Results of the performance of the ECCS for the entire spectrum of liquid line breaks are discussed in Subsection 6.3.3. Peak cladding temperatures are well below the 1204°C design basis.

Also provided in Subsection 6.3.3 is an analysis to show that the ECCS conforms to the 10CFR50 Appendix K. This analysis shows complete compliance with the Criterion 35 with the following results:

(1) Peak clad temperatures are well below the 1204°C NRC acceptability limit.

(2) The amount of fuel cladding reacting with steam is nearly an order of magnitude below the 1% acceptability limit.

(3) The clad temperature transient is terminated while core geometry is still amenable to cooling.

(4) The core temperature is reduced and the decay heat can be removed for an extended period of time.

The redundancy and capability of the onsite electrical power systems for the ECCS are represented in the evaluation against Criterion 34.

The ECCS is adequate to prevent fuel and clad damage which could interfere with effective core cooling and to limit clad metal water reaction to a negligible amount. The design of the ECCS, including the power supply, meets the requirements of Criterion 35.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
5.4.6	Reactor Core Isolation Cooling System
6.3	Emergency Core Cooling System
7.3.1.1.2 and 7.3.2.1	Emergency Core Cooling System—Instrumentation and Controls
8.3	Onsite Power Systems
9.2	Water Systems
15	Accident Analyses

3.1.2.4.7 Criterion 36—Inspection of Emergency Core Cooling System

3.1.2.4.7.1 Criterion 36 Statement

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

3.1.2.4.7.2 Evaluation Against Criterion 36

The ECCS discussed in Criterion 35 includes inservice inspection considerations. The flooder spargers within the vessel are accessible for inspection during each refueling outage. Removable plugs in the reactor shield and/or panels in the insulation is provided on the ECCS piping up to and including the first isolation valve outside the drywell. Inspection of the ECCS is in accordance with the intent of ASME Code Section XI. Subsection 5.2.4 defines the Inservice Inspection Plan, access provisions, and areas of restricted access.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the containment can be visually inspected at any time. Components inside the containment can be inspected when the containment is open for access. When the reactor vessel is open for refueling or other purposes, the spargers and other internals can be inspected. Portions of the ECCS which are part of the RCPB are designed to specifications for inservice inspection to detect defects which might affect the cooling performance. Particular attention will be given to the reactor nozzles, and core flooder spargers. The design of the reactor vessel and internals for inservice inspection and the plant testing and inspection program ensures that the requirements of Criterion 36 will be met.

For further discussion, see the following sections:

Chapter/Section	Title
3.9.5	Reactor Pressure Vessel Internals
5.2.4	Inservice Inspection and Testing of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
6.3	Emergency Core Cooling Systems
6.6	Inservice Inspection of Class 2 and 3 Components

3.1.2.4.8 Criterion 37—Testing of Emergency Core Cooling System

3.1.2.4.8.1 Criterion 37 Statement

The ECCS 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.

(3) The operability and performance of the active components of the system.The operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

3.1.2.4.8.2 Evaluation Against Criterion 37

The ECCS consists of the HPCF System, RCIC System, LPFL mode of the RHR System, and the ADS. Each of these systems is provided with sufficient test connections and isolation valves to permit appropriate periodic pressure testing to assure the structural and leaktight integrity of its components.

Each of the ECCS is designed to permit periodic testing to assure the operability and performance of the active components of each system.

The pumps and valves of these systems will be tested periodically to verify operability. Flow rate tests will be conducted on the HPCF System, RCIC System, and RHR/LPFL System.

The ECCS will be subjected to tests to verify the performance of the full operational sequence that brings each system into operation. The operation of the associated cooling water systems is discussed in the evaluation of Criterion 46. It is concluded that the requirements of Criterion 37 are met.

For further discussion, see the following sections:

Chapter/Section	Title
5.2.2	Overpressurization Protection
6.3	Emergency Core Cooling Systems
7.3.1.1	Emergency Core Cooling Systems—Instrumentation and Controls
8.3.1	AC Power Systems
16	Technical Specifications

3.1.2.4.9 Criterion 38—Containment Heat Removal

3.1.2.4.9.1 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 acceptable 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.

3.1.2.4.9.2 Evaluation Against Criterion 38

The containment heat removal function is accomplished by the suppression pool cooling mode of the RHR System. Following a LOCA, the suppression pool cooling (SPC) mode limits the temperature within the wetwell by recirculating the suppression pool water and removing heat via the RHR System heat exchangers. This subsystem is initiated automatically when suppression pool temperature increases to a preset level. Suppression pool cooling can also be initiated manually. If a LOCA signal is present, the RHR System will function in the core cooling (LPFL) mode.

Following a LOCA, wetwell and drywell spray mode of the RHR System condenses steam within the drywell and wetwell zones of the containment by spraying suppression pool water cooled through the heat exchangers. Wetwell/drywell spray is started manually. The drywell spray mode is initiated by operator action post-LOCA in the presence of high drywell pressure. The wetwell spray mode can be manually initiated in the control room, unless an overriding LOCA signal for the LPFL is present. The wetwell spray mode does not depend on the operation of the suppression pool cooling mode.

The redundancy and capability of the offsite and onsite electrical power systems for the RHR System is presented in the evaluation against Criterion 34.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
6.2.2	Containment Heat Removal Systems
7.3	Engineered Safety Features—Instrumentation and Controls
8.3.1	AC Power Systems
9.2	Water Systems
15	Accident Analyses

3.1.2.4.10 Criterion 39—Inspection of Containment Heat Removal System

3.1.2.4.10.1 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.

3.1.2.4.10.2 Evaluation Against Criterion 39

Provisions are made to facilitate periodic inspections of active components and other important equipment of the containment heat removal systems. During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the containment can be visually inspected at any time and will be inspected periodically. Such components inside the containment will be tested and inspected during periodic outages. The testing frequencies of most components will be correlated with the component inspection.

The suppression pool is designed to permit appropriate periodic inspection. Space is provided outside the containment for inspection and maintenance.

The containment heat removal system is designed to permit periodic inspection of major components. This design meets the requirements of Criterion 39.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
6.2	Containment Systems
6.3	Emergency Core Cooling Systems
9.2	Water Systems

3.1.2.4.11 Criterion 40—Testing of Containment Heat Removal System

3.1.2.4.11.1 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.

(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.

3.1.2.4.11.2 Evaluation Against Criterion 40

The containment heat removal function is accomplished by a suppression pool cooling mode of the RHR System.

The RHR System is provided with sufficient test connections and isolation valves to permit periodic pressure and flow rate testing.

The pumps and valves of the RHR System will be operated periodically to verify operability. The cooling is initiated manually or automatically on sensed high temperature in the suppression pool, however, operation of the components is periodically verified. The operation of associated cooling water systems is discussed in the response to Design Criterion 46. It is concluded that the requirements of Criterion 40 are met.

For further discussion, see Subsection 6.2.2.

3.1.2.4.12 Criterion 41—Containment Atmosphere Cleanup

3.1.2.4.12.1 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 onsite electric power system operation (assuming offsite power is not available) and its safety function can be accomplished, assuming a single failure.

3.1.2.4.12.2 Evaluation Against Criterion 41

The quantity and quality of fission products released into the environment following postulated accidents is controlled by the Standby Gas Treatment System (SGTS) that has the redundancy and capability to filter and treat the gaseous effluent from the primary and the secondary containment.

Equipment for purging and inerting is provided to control the oxygen concentration of the inert volume of the ABWR primary containment atmosphere so that a flammable condition will not be created during an accident.

These systems have design provisions to ensure that the safety function is accomplished, assuming a single failure. These systems meet the requirements of Criterion 41. For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
6.2.5	Combustible Gas Control in Containment
6.5.1	Engineered Safety Features Filter System
6.5.3	Fission Product Control Systems
7	Instrumentation and Control System
8	Electric Power
9.5.9	Suppression Pool Cleanup System
15	Accident Analyses

3.1.2.4.13 Criterion 42— Inspection of Containment Atmosphere Cleanup System

3.1.2.4.13.1 Criterion 42 Statement

The Containment Atmosphere Cleanup System 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.

3.1.2.4.13.2 Evaluation Against Criterion 42

Except for components located in the primary containment and steam tunnel, all components of the fission product control system can be inspected during normal plant operation at power. The components within the containment and steam tunnel may be inspected during refueling and maintenance outages.

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

Chapter/Section	Title
1.2	General Plant Description
6.2.5	Combustible Gas Control in Containment
6.5.1	Engineered Safety Features Filter System
6.5.3	Fission Product Control Systems
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping
7	Instrumentation and Control System
8	Electric Power
9.5.9	Suppression Pool Cleanup System

3.1.2.4.14 Criterion 43—Testing of Containment Atmosphere Cleanup System

3.1.2.4.14.1 Criterion 43 Statement

The Containment Atmosphere Cleanup 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 systems such as fans, filters, dampers, pumps, and valves.

(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.

3.1.2.4.14.2 Evaluation Against Criterion 43

All active components of the fission product control system can be tested during normal plan operation at power.

Complete system operation can be tested during reactor shutdown.

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

Chapter/Section	Title
1.2	General Plant Description
6.2.5	Combustible Gas Control in Containment
6.5.3	Fission Product Control System
7	Instrumentation and Control System
8	Electric Power

3.1.2.4.15 Criterion 44—Cooling Water

3.1.2.4.15.1 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 systems operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

3.1.2.4.15.2 Evaluation Against Criterion 44

The system provided to transfer heat from safety-related equipment to the ultimate heat sink is the Reactor Building Cooling Water (RCW) System.

This system is operable either from offsite power or from onsite emergency power and is designed with suitable redundancy, isolation capability, and separation such that no single failure prevents a safe plant shutdown.

The design of this system meets the requirements of Criterion 44.

For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
9.2	Water Systems

3.1.2.4.16 Criterion 45— Inspection of Cooling Water System

3.1.2.4.16.1 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.

3.1.2.4.16.2 Evaluation Against Criterion 45

All important components in the ABWR Standard Plant scope are located in accessible locations to facilitate periodic inspection during normal plant operation. Suitable manholes, handholes, inspection ports, or other design and layout features are provided for this purpose.

These features meet the requirements of Criterion 45.

For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
9.2	Water Systems
14	Initial Test Program

3.1.2.4.17 Criterion 46—Testing of Cooling Water System

3.1.2.4.17.1 Criterion 46 Statement

The cooling water 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 the performance of the active components of the system.

(3) The operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

3.1.2.4.17.2 Evaluation Against Criterion 46

Redundancy and isolation are provided to allow periodic pressure and functional testing of the system as a whole including the functional sequence that initiates system operation. This also includes transfer between the offsite power supply and the onsite emergency diesel-generator power supply. At least one of the redundant systems is in service during normal plant operations.

The system design thus meets the requirements of Criterion 46.

For further discussion, see the following sections:

Chapter/Section	Section
1.2	General Plant Description
9.2	Water Systems
14	Initial Test Program
16	Technical Specifications

3.1.2.5 Group V—Reactor Containment

3.1.2.5.1 Criterion 50—Containment Design Basis

3.1.2.5.1.1 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 the 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.

(3) The conservatism of the calculational model and input parameters.

3.1.2.5.1.2 Evaluation Against Criterion 50

Design of the containment is based on the safe shutdown earthquake (SSE) postulated to occur at the site simultaneously with the design basis accident (DBA), which is defined as the worst LOCA pipe break having the consequences of maximum containment and drywell pressure and/or temperature. These conditions are coupled with the loss of offsite power.

The maximum pressure and temperature reached in the drywell and containment during this worst-case accident are shown in Chapter 6 to be well below the design pressure and temperature of the structures. This provides an adequate margin for uncertainties in potential energy sources.

The design of the containment system thus 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
6.2.1	Containment Functional Design
6.2.2	Containment Heat Removal System
15	Accident Analyses

3.1.2.5.2 Criterion 51—Fracture Prevention of Containment Pressure Boundary

3.1.2.5.2.1 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.

(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

(3) Size of flaws

3.1.2.5.2.2 Evaluation Against Criterion 51

The primary containment vessel (PCV) is a reinforced concrete structure with ferritic parts (the removable head, personnel locks, equipment hatches and penetrations), which are made of material that has a nil-ductility transition temperature of at least 17°C below the minimum service temperature.

The PCV is enclosed by and is integrated with the reinforced concrete reactor building. The pre-operational test program and the QA program ensure the integrity of the containment and its ability to meet all normal operating and accident requirements.

The containment design thus meets the requirements of Criterion 51.

For further discussion, see the following sections:

Chapter/Section	Title
3.8	Design of Seismic Category I Structures
17	Quality Assurance

3.1.2.5.3 Criterion 52—Capability for Containment Leakage Rate Testing

3.1.2.5.3.1 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.

3.1.2.5.3.2 Evaluation Against Criterion 52

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leak-rate tests during the plant lifetime. The testing program is conducted in accordance with 10CFR50 Appendix J.

The testing provisions provided and the test program meet the requirements of Criterion 52.

For further discussion, see the following sections:

Chapter/Section	Title
3.8.2.7	Testing and Inservice Inspection Requirements
6.2.6.1	Containment Integrated Leakage Test Rate

3.1.2.5.4 Criterion 53—Provisions for Containment Testing and Inspection

3.1.2.5.4.1 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.

(3) Periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

3.1.2.5.4.2 Evaluation Against Criterion 53

There are special provisions for conducting individual leakage rate tests on applicable penetrations. Penetrations are visually inspected and pressure tested for leaktightness at periodic intervals in accordance with 10CFR50 Appendix J.

The provisions made for protection testing meet the requirements of Criterion 53.

For further discussion, see the following sections:

Chapter/Section	Title
6.2.6.2	Containment Penetration Leakage Rate Test (Type B)
6.2.6.3	Containment Isolation Value Leakage Rate Test (Type C)

3.1.2.5.5 Criterion 54—Piping Systems Penetrating Containment

3.1.2.5.5.1 Criterion 54 Statement

Piping systems penetrating 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 periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

3.1.2.5.5.2 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.

3.1.2.5.6 Criterion 55—Reactor Coolant Pressure Boundary Penetrating Containment

3.1.2.5.6.1 Criterion 55 Statement

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

(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.

3.1.2.5.6.2 Evaluation Against Criterion 55

The RCPB as defined in 10CFR50 Section 50.2, consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, valves, and pipes which extend from the reactor pressure vessel up to and including the outermost isolation valves. The lines of the RCPB which penetrate the containment have suitable isolation valves capable of isolating the containment, thereby precluding any significant release of radioactivity.

The design of the isolation systems detailed in the following sections meets the requirements of Criterion 55.

For further discussion, see the following sections:

Chapter/Section	Title
5.2	Integrity of Reactor Coolant Pressure Boundary
6.2.4	Containment Isolation Systems
7	Instrumentation and Controls
15	Accident Analyses
16	Technical Specifications

3.1.2.5.7 Criterion 56—Primary Containment Isolation

3.1.2.5.7.1 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 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.

3.1.2.5.7.2 Evaluation Against Criterion 56

The manner in which the containment isolation system meets this requirement is detailed in the following sections:

Chapter/Section	Title
6.2.4	Containment Isolation Systems
7	Instrumentation and Controls
15	Accident Analyses
16	Technical Specifications

3.1.2.5.8 Criterion 57—Closed System Isolation Valves

3.1.2.5.8.1 Criterion 57 Statement

Each line that penetrates primary reactor containment and is neither part of the RCPB 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.

3.1.2.5.8.2 Evaluation Against Criterion 57

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

Details demonstrating conformance with Criterion 57 are provided in the following section:

Chapter/Section	Title
6.2.4	Containment Isolation Systems

3.1.2.6 Group VI—Fuel and Reactivity Control

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

3.1.2.6.1.1 Criterion 60 Statement

The nuclear power unit design shall include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. 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.

3.1.2.6.1.2 Evaluation Against Criterion 60

The ABWR is designed so that releases of radioactive materials, in either gaseous, liquid, or solid form, are minimized. Gaseous releases come primarily from the turbine condenser offgas and the ventilation systems along with the flue gas from any incineration of dry active waste (DAW). Activity in the condenser offgas is minimized by the advanced fuel design utilized in the ABWR and the improved water chemistry which reduces the corrosion potential of the primary system water. Noble gas and any iodine activity that enters the Turbine Offgas System is held up by ambient temperature charcoal beds. The Offgas System itself has undergone significant improvement to increase reliability. All ventilation system releases are through the plant stack. The plant stack and the major streams feeding the plant stack are monitored by the process radiation monitoring system so that suitable action may be taken to avoid releases in excess of the relevant regulatory limits.

The Radwaste System processes liquid and solid wastes. Because of the overall improved system and equipment design, improved reliability and capacity factor, and improved operations, the generation of solid radwaste is expected to be significantly less than in current operating plants. Processes are installed in radwaste to fully treat and solidify solid wastes, as required by applicable state and federal regulations. In addition, the ABWR Radwaste System can be operated in a mode where all non-detergent and non-chemical waste streams are treated to allow maximum recycle to the primary system (condensate storage tank). This mode of operation would minimize releases via the liquid or discharge pathway at the expense of some increase in solid waste generated. The optimal balance is best established during operations and is significantly affected by the overall plant water balance.

The Radwaste System has significant holdup capacity, both in waste collection tanks and in sample tanks containing processed water. This holdup or surge capacity provides the utility flexibility in operations when deciding when and how to release effluents to the environment.

For further discussion, see the following Sections: 11.2 through 11.4, “Radioactive Waste Management” and 11.5, “Process Radiation Monitoring.”

3.1.2.6.2 Criterion 61—Fuel Storage and Handling and Radioactivity Control

3.1.2.6.2.1 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 a capability to permit appropriate periodic inspection and testing of components important to safety

(2) With suitable shielding for radiation protection

(3) With appropriate containment, confinement, and filtering systems

(4) With a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal

(5) To prevent significant reduction in fuel storage coolant inventory under accident conditions

3.1.2.6.2.2 Evaluation Against Criterion 61

3.1.2.6.2.2.1 Fuel Storage and Handling System

Fuel storage pools have adequate water shielding for stored spent fuel. Adequate shielding for transporting fuel is also provided. Liquid level sensors are installed to detect low pool water level. Buildings are designed to meet Regulatory Guide 1.13 criteria. The fuel storage pools are designed with no penetrations below the water level that is needed for maintenance of adequate shielding at the operating floor and cooling. Check valves are used in pool circulation lines to prevent siphoning in the event of a break of such a line.

New fuel is contained in storage racks in the spent fuel pool. These storage racks preclude accidental criticality (see evaluation against Criterion 62).

The fuel storage and handling system is designed to assure adequate safety under normal and postulated accident conditions. The design of these systems meets the requirements of Criterion 61.

Per Regulatory Guide 1.143, the substructure of the radwaste building is designed as Seismic Category I and it is sufficient to contain the maximum liquid inventory expected to be in the building.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
6.2	Containment Systems
9.1	Fuel Storage and Handling
11	Radioactive Waste Management
12	Radiation Protection

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

3.1.2.6.3.1 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.

3.1.2.6.3.2 Evaluation Against Criterion 62

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in new and spent fuel storage is prevented by presence of fixed neutron absorbing material. Fuel elements are limited by rack design to only top-loaded fuel assembly positions. The new and spent fuel racks are Seismic Category I components.

The new and spent fuel is stored under water in the spent fuel pool. A full array of loaded fuel racks is designed to be subcritical, by at least 5% Δk. Neutron-absorbing material, as an integral part of the design, is employed to assure that the calculated k_{eff}, including biases and uncertainties, will 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 fuel handling equipment without first disengaging the fuel from the hoisting equipment.

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

The presence of fixed neutron-absorbing material in the new and spent 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

3.1.2.6.4 Criterion 63—Monitoring Fuel and Waste Storage

3.1.2.6.4.1 Criterion 63 Statement

Appropriate systems shall be provided in 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.

(2) Initiate appropriate safety actions.

3.1.2.6.4.2 Evaluation Against Criterion 63

The Fuel Pool Cooling and Cleanup (FPC) System removes decay heat from fuel storage pools. In addition, two loops of the RHR System can provide additional cooling of the spent fuel pool, as required. Fuel pool temperature and level are monitored as part of the FPC System. High pool temperature or low skimmer surge tank level would signal the need for providing additional cooling [e.g., adding a loop of RHR, or makeup water, e.g., from the makeup water system (condensate) connection, as appropriate]. Area radiation monitors are provided as part of the area radiation monitoring system which monitors the operating/refueling floor for high radiation levels.

Area radiation and tank and sump levels are monitored and alarmed to give indication of conditions which may result in excessive radiation levels in radioactive waste system areas. These systems satisfy the requirements of Criterion 63.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
9.1.3	Fuel Pool Cooling and Cleanup System
9.2.9	Makeup Water System (Condensate)
11	Radioactive Waste Management
12	Radiation Protection

3.1.2.6.5 Criterion 64—Monitoring Radioactivity Releases

3.1.2.6.5.1 Criterion 64 Statement

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA 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.

3.1.2.6.5.2 Evaluation Against Criterion 64

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

(1) Gaseous releases

(2) Liquid discharge

In addition, the containment atmosphere is monitored.

For further discussion of the same means and equipment used for monitoring reactivity releases, see the following sections:

Chapter/Section	Title
5.2.5	Reactor Coolant Pressure Boundary and Core Cooling Systems Leakage Detection
11	Radioactive Waste Management

3.2 Classification of Structures, Components, and Systems

ABWR Standard Plant structures, systems and components are categorized as nuclear safety-related or non-nuclear safety-related (see Table 3.2-1). The safety-related structures, systems and components, perform nuclear safety-related functions as defined here, and are classified in accordance with Subsection 3.2.3. In addition, specific design requirements are identified for the safety-related equipment commensurate with their safety classification (see Table 3.2-2 and 3.2-3).

A safety-related function is a direct or support function that is necessary to assure:

(1) The integrity of the reactor coolant pressure boundary.

(2) The capability to shut down the reactor and maintain it in a safe condition.

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guidelines exposures of 10CFR100.

3.2.1 Seismic Classification

ABWR Standard Plant safety-related structures, systems, and components, including their foundations and supports, that are required to perform nuclear safety-related functions during or after a safe shutdown earthquake (SSE) are designated as Seismic Category I.

All safety-related ABWR Standard Plant structures, components, and systems are classified as Seismic Category I, except those (e.g., pipe whip restraints), as noted on Table 3.2-1, which need not function during but shall remain functional after the event of an SSE. Also some non-safety-related structures, systems, and components are classified as Seismic Category I as noted on Table 3.2-1.

The Seismic Category I structures, systems and components are designed to withstand, without loss of function, the appropriate seismic loads (as discussed in Section 3.7) in combination with other appropriate loads.

The seismic classifications indicated in Table 3.2-1 meet the requirements of Regulatory Guide 1.29 except as otherwise noted in the table.

3.2.2 Quality Group Classifications

Quality group classifications as defined in NRC Regulatory Guide 1.26 are shown in Table 3.2-1 for all components under the heading, “Quality Group Classification”. Although not within the scope of Regulatory Guide 1.26 definitions, component supports, core support structures and primary containment boundary that are within the scope of ASME Code Section III, are assigned per Tables 3.2-2 and 3.2-3, a quality group classification as identified in Table 3.2-1.

Quality group classifications and design and fabrication requirements defined in Regulatory Guide 1.26 are indicated in Tables 3.2-1 and 3.2-3, respectively. Figure 6.2-38 depicts quality group classifications of the components in major systems.

3.2.3 Safety Classifications

Safety-related structures, systems, and components of the ABWR Standard Plant are classified for design requirements as Safety Class 1, Safety Class 2, or Safety Class 3 in accordance with their nuclear 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 Code Section III, depends upon that of the supported component.

The definitions of the safety classes in this section are based on Section 3.3 of ANS Standard 52.1, and examples of their broad application are given. 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.

Minimum design requirements for various safety-related classes are delineated in Tables 3.2-2 and 3.2-3. 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.

3.2.3.1 Safety Class 1

Safety Class 1 (SC-1) applies to all components of the reactor coolant pressure boundary (as defined in 10CFR50.2), 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 Code Section III.

Safety Class 1 components are identified in Table 3.2-1.

3.2.3.2 Safety Class 2

Safety Class 2 (SC-2) applies to pressure-retaining portions, and their supports, of 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 SC-1 and are designed and relied upon to accomplish the following nuclear 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., suppression pool)

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

(1) Those components of the control rod system which 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 25A or smaller that are part of the reactor coolant pressure boundary

Safety Class 2 structures, systems, and components are identified in Table 3.2-1.

3.2.3.3 Safety Class 3

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

(1) Provide for functions defined in SC-1 or -2 by means of equipment, or portions thereof, that is not within the scope of the ASME 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 or secondary containment) containing SC-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 SC-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 nuclear safety-related functions provided by SC-1, -2, or -3 equipment (e.g., provide heat removal for SC-1, -2, or -3 heat exchangers, provide lubrication of SC-2 or -3 pumps, provide fuel oil to the emergency diesel engine).

(10) Provide actuation or motive power for SC-1, -2, or -3 equipment.

(11) Provide information or controls to ensure capability for manual or automatic actuation of nuclear safety-related functions required of SC-1, -2, or -3 equipment.

(12) Supply or process signals or supply power required for SC-1, -2, or -3 equipment to perform their required nuclear safety-related functions.

(13) Provide a manual or automatic interlock function to ensure or maintain proper performance of nuclear safety-related functions required of SC-1, -2, or -3 equipment.

(14) Provide acceptable environments for SC-1, -2, or -3 equipment and operating personnel.

(15) Monitor plant variables that are identified requiring Category 1 electrical instrumentation in Table 1 of Regulatory Guide 1.97.

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 which restrict the rate of insertion of positive reactivity

(4) Secondary containment

(5) Service water systems required for the purpose of:

  (a) Removal of heat from SC-1, SC-2 or SC-3 equipment

  (b) Emergency core cooling

  (c) Post-accident heat removal from the suppression pool

(d) Providing cooling water needs for the functioning of emergency systems

(6) Initiating systems required to accomplish emergency core cooling, containment isolation and other safety-related functions

(7) Spent fuel pool

(8) Fuel supply for the onsite emergency electrical system

(9) Emergency equipment area cooling

(10) Compressed gas or hydraulic systems required to provide control or operation of safety-related systems

Safety Class 3 structures, systems and components of the ABWR design are identified in Table 3.2-1.

3.2.4 Correlation of Safety Classes with Industry Codes

The design of plant safety-related equipment is commensurate with the safety importance of the equipment. Hence, the various safety classes have a gradation of design requirements. The correlation of safety classes with other design requirements is summarized in Tables 3.2-2 and 3.2-3.

3.2.5 Non-Safety-Related Structures, Systems, and Components

3.2.5.1 Definition of Non-Nuclear Safety (NNS) Category

Structures, systems, and components that are not SC-1, -2, or -3, are non-nuclear safety-related (NNS) and are identified with “N” in the Safety Class column of Table 3.2-1.

Some NNS structures, systems and components have one or more selected but limited, requirements that are specified to ensure acceptable performance of specific NNS functions. The selected requirements are established on a case-by-case basis commensurate with the specific NNS function performed (see Table 3.2-2). The functions performed by this subset of NNS structures, systems, and components are:

(1) Process, extract, encase, or store radioactive waste.

(2) Ensure required cooling for the stored fuel (e.g., spent fuel pool cooling system).

(3) Provide cleanup of radioactive material from the reactor coolant system or the fuel storage cooling system.

(4) Monitor radioactive effluents to ensure that release rates or total releases are within limits established for normal operations and transient events.

(5) Resist failure that could prevent any SC-1, -2, or -3 equipment from performing its nuclear safety-related function (see Table 3.2-2).

(6) Structurally bear the load or protect NNS equipment providing any of the functions listed in this Subsection 3.2.5.1.

(7) Provide permanent shielding for protection of SC-1, -2, or -3 equipment or of onsite personnel.

(8) Provide operational, maintenance or post-accident recovery functions involving radioactive materials without undue risk to the health and safety of the public.

(9) Following a control room evacuation, provide an acceptable environment for equipment required to achieve or maintain a safe shutdown condition.

(10) Handle spent fuel, the failure of which could result in fuel damage such that significant quantities of radioactive material could be released from the fuel.

(11) Ensure reactivity control of stored fuel.

(12) Protect safety-related equipment necessary to attain or maintain safe shutdown following a fire.

(13) Monitor variables to:

- (a) Verify that plant operating conditions are within technical specification limits (e.g., emergency core cooling water storage tank level, safety-related cooling water temperature).
- (b) Indicate the status of protection system bypasses that are not automatically removed as a part of the protection system operation.
- (c) Indicate status of safety-related equipment.
- (d) Aid in determining the cause or consequences of events for post-accident investigation.

3.2.5.2 Design Requirements for NNS Structures, Systems and Components

The design requirements for NNS 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, the Seismic Category I, ASME Code Section III, or IEEE Class 1E requirements are specified for NNS equipment in Table 3.2-1. Generally, design requirements are based on applicable industry codes and standards. Where these are not available, accepted industry or engineering practice is followed.

3.2.5.3 Main Steam Line Leakage Path

The ABWR main steam leakage path utilizes the large volume and surface area in the main steam piping, bypass line, and condenser to hold up and plate out the release of fission products following postulated core damage. In this manner, the main steam piping, bypass line, and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after an SSE.

The main steamlines and all branch lines 65A nominal pipe size in diameter and larger, up to and including the first valve (including lines and valve supports) are designed by the use of an appropriate dynamic seismic system analysis to withstand the safe shutdown earthquake (SSE) design loads in combination with other appropriate loads, within the limits specified. The mathematical model for the dynamic seismic analyses of the main steamlines and branch line piping includes the turbine stop valves and piping to the turbine casing and the turbine bypass valves and piping to the condenser. The dynamic input loads for design of the main steamlines in the reactor building and the control building are derived from a time history model analysis or an equivalent method as described in Section 3.7.

Dynamic input loads for the design of the main steamlines in the turbine building are derived as follows: For locations on the basemat, the ARS shall be based upon Regulatory Guide 1.60 Response spectra normalized to 0.6g (i.e., 2 times ARS of the site envelope). For locations at the operating deck level (either operating deck or turbine deck), the ARS used shall be the same as used at the reactor building end of the main steam tunnel. Seismic Anchor motions shall be similarly calculated.

Figure 3.2-1 depicts the classification requirements for the main steamline leakage path as described below.

(1) Main steam piping from the reactor pressure vessel up to and including the outboard isolation valve is classified as QG A (SC-1) and Seismic Category I.

(2) Main steam piping beyond the outboard isolation valve up to the seismic interface restraint and connecting branch lines up to the first normally closed valve is classified as QG B (SC-2) and Seismic Category I.

(3) [*The main steamline from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally closed valve) is classified as QG B and inspected in accordance with applicable portions of the American Society of Mechanical Engineers (ASME) Section XI. This portion of the steamline is classified as non-Seismic Category I and analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B, 10CFR Part 50*

are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions.]*

The seismic interface restraint provides a structural barrier between the Seismic Category I portion of the main steamline in the reactor building and the non-Seismic Category I portions of the main steamline in the turbine building. The seismic interface restraint is located inside the Seismic Category I building. The classification of the main steamline in the turbine building as non-Seismic Category I is consistent with the classification of the turbine building.

At the interface between Seismic and non-Seismic Category I main steam piping system, the Seismic Category I dynamic analyses will be extended to either the first anchor point in the non-seismic system or to a sufficient distance in the non-seismic system so as to not degrade the validity of the Seismic Category I analysis.

(4) [*To ensure the integrity of the remainder of main steamline leakage path, the following requirements are met:*

(a) *The main steam piping between the turbine stop valve and the turbine inlet, the turbine bypass line from the bypass valve to the condenser, and the main steam drain line from the first valve to the condenser are not required to be classified as safety-related nor as Seismic Category I, but are analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions.*

(b) *The condenser anchorage is seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure.*]†

[*A plant-specific walkdown of non-seismically designed systems, structures, and components overhead, adjacent to, and attached to the main steamline leakage path (i.e., the main steam piping, the bypass line, and the main condenser) shall be conducted to confirm by inspection that the as-built main steam piping, bypass lines to the condenser, and the main condenser are not compromised by non-seismically designed systems, structures and components.*]*

3.2.6 Quality Assurance

Structures, systems, and components that perform nuclear safety-related functions conform to the quality assurance requirement of 10CFR50 Appendix B as shown in Table 3.2-1 under the heading, "Quality Assurance Requirements," and in Table 3.2-2. Some NNS structures, systems, and components meet the same requirements as noted on Table 3.2-1. The Quality Assurance Program is described in Chapter 17.

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* See Subsection 3.9.1.7.

† See Subsection 3.9.1.7.

Table 3.2-1 Classification Summary

The classification information is presented by System* in the following order:

Item No.	MPL Number†	Title
B Nuclear Boiler Supply System		
B1	B11	Reactor Pressure Vessel System‡
B2	B21	Nuclear Boiler System‡
B3	B31	Reactor Recirculation System
C Control and Instrument Systems		
C1	C11	Rod Control and Information System
C2	C12	Control Rod Drive System
C3	C31	Feedwater Control System
C4	C41	Standby Liquid Control System
C5	C51	Neutron Monitoring System‡
C6	C61	Remote Shutdown System
C7	C71	Reactor Protection System‡
C8	C81	Recirculation Flow Control System
C9	C82	Automatic Power Regulator System
C10	C85	Steam Bypass and Pressure Control System
C11	C91	Process Computer (Includes PMCS and PGCS)
C12	C93	Refueling Platform Control Computer
C13	C94	CRD Removal Machine Control Computer
D Radiation Monitoring Systems		
D1	D11	Process Radiation Monitoring System‡
D2	D21	Area Radiation Monitoring System
D3	D23	Containment Atmospheric Monitoring System‡

* Systems that are in and out of the ABWR Standard Plant scope are included in this table. See Subsection 1.1.2 for the identification of the site-specific elements outside the scope of the ABWR Standard Plant.

† Master Parts List Number designated for the system.

‡ These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety-related components and, conversely, some systems whose primary functions are non-safety-related contain components that have been designated safety-related.

Table 3.2-1 Classification Summary (Continued)

The classification information is presented by System* in the following order:

Item No.	MPL Number†	Title
E Core Cooling Systems		
E1	E11	Residual Heat Removal System‡
E2	E22	High Pressure Core Flooder System‡
E3	E31	Leak Detection and Isolation System‡
E4	E51	Reactor Core Isolation Cooling System‡
F Reactor Servicing Equipment		
F1	F11	Fuel Servicing Equipment
F2	F12	Miscellaneous Servicing Equipment
F3	F13	RPV Servicing Equipment
F4	F14	RPV Internal Servicing Equipment
F5	F15	Refueling Equipment
F6	F16	Fuel Storage Facility
F7	F17	Under-Vessel Servicing Equipment
F8	F21	CRD Maintenance Facility
F9	F22	Internal Pump Maintenance Facility
F10	F32	Fuel Cask Cleaning Facility
F11	F41	Plant Start-up Test Facility
F12	F51	Inservice Inspection Equipment
G Reactor Auxiliary Systems		
G1	G31	Reactor Water Cleanup System
G2	G41	Fuel Pool Cooling and Cleanup System
G3	G51	Suppression Pool Cleanup System

* Systems that are in and out of the ABWR Standard Plant scope are included in this table. See Subsection 1.1.2 for the identification of the site-specific elements outside the scope of the ABWR Standard Plant.

† Master Parts List Number designated for the system.

‡ These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety-related components and, conversely, some systems whose primary functions are non-safety-related contain components that have been designated safety-related.

Table 3.2-1 Classification Summary (Continued)

The classification information is presented by System* in the following order:

Item No.	MPL Number†	Title
H Control Panels		
H1	H11	Main Control Room Panels‡
H2	H12	Control Room Back Panels‡
H3	H14	Radioactive Waste Control Panels
H4	H21	Local Control Panels‡
H5	H22	Instrument Racks
H6	H23	Multiplexing System
H7	H25	Local Control Boxes
J Nuclear Fuel		
J1	J11	Fuel Assembly
J2	J12	Fuel Channel
K Radioactive Waste System		
K1	K17	Radwaste System
N Power Cycle Systems		
N1	N11	Turbine Main Steam System
N2	N21	Condensate, Feedwater and Condensate Air Extraction System
N3	N22	Heater, Drain and Vent System
N4	N25	Condensate Purification System
N5	N26	Condensate Filter Facility
N6	N27	Condensate Demineralizer
N7	N31	Main Turbine
N8	N32	Turbine Control System

* Systems that are in and out of the ABWR Standard Plant scope are included in this table. See Subsection 1.1.2 for the identification of the site-specific elements outside the scope of the ABWR Standard Plant.

† Master Parts List Number designated for the system.

‡ These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety-related components and, conversely, some systems whose primary functions are non-safety-related contain components that have been designated safety-related.

Table 3.2-1 Classification Summary (Continued)

The classification information is presented by System* in the following order:

Item No.	MPL Number†	Title
N9	N33	Turbine Gland Steam System
N10	N34	Turbine Lubricating Oil System
N11	N35	Moisture Separator Heater
N12	N36	Extraction System
N13	N37	Turbine Bypass System
N14	N38	Reactor Feedwater Pump Driver
N15	N39	Turbine Auxiliary Steam System
N16	N41	Generator
N17	N42	Hydrogen Gas Cooling System
N18	N43	Generator Cooling System
N19	N44	Generator Sealing Oil System
N20	N51	Exciter
N21	N61	Main Condenser
N22	N62	Offgas System
N23	N71	Circulating Water System
N24	N72	Condenser Cleanup System
P Station Auxiliary Systems		
P0	P10	Makeup Water System (Preparation)
P1	P11	Makeup Water System (Purified)
P2	P13	Makeup Water System (Condensate)
P3	P21	Reactor Building Cooling Water System‡
P4	P22	Turbine Building Cooling Water System
P5	P24	HVAC Normal Cooling Water System
P6	P25	HVAC Emergency Cooling Water System
P7	P32	Oxygen Injection System

* Systems that are in and out of the ABWR Standard Plant scope are included in this table. See Subsection 1.1.2 for the identification of the site-specific elements outside the scope of the ABWR Standard Plant.

† Master Parts List Number designated for the system.

‡ These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety-related components and, conversely, some systems whose primary functions are non-safety-related contain components that have been designated safety-related.

Table 3.2-1 Classification Summary (Continued)

The classification information is presented by System* in the following order:

Item No.	MPL Number†	Title
P8	P40	Ultimate Heat Sink
P9	P41	Reactor Service Water System
P10	P42	Turbine Service Water System
P11	P51	Station Instrument Air System
P12	P52	Instrument Air System
P13	P54	High Pressure Nitrogen Gas Supply System
P14	P61	Heating Steam and Condensate Water Return System
P15	P62	House Boiler
P16	P63	Hot Water Heating System
P17	P73	Hydrogen Water Chemistry System
P18	P74	Zinc Injection System
P19	P81	Breathing Air System
P20	P91	Sampling System (Includes PASS)
P21	P92	Freeze Protection System
P22	P95	Iron Injection System
R Station Electrical Systems		
R1	R10	Electrical Power Distribution System
R2	R11	Unit Auxiliary Transformer
R3	R13	Isolated Phase Bus
R4	R21	Non-Segregated Phase Bus
R5	R22	Metalclad Switchgear
R6	R23	Power Center
R7	R24	Motor Control Center
R8	R31	Raceway System
R9	R34	Grounding Wire

* Systems that are in and out of the ABWR Standard Plant scope are included in this table. See Subsection 1.1.2 for the identification of the site-specific elements outside the scope of the ABWR Standard Plant.

† Master Parts List Number designated for the system.

‡ These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety-related components and, conversely, some systems whose primary functions are non-safety-related contain components that have been designated safety-related.

Table 3.2-1 Classification Summary (Continued)

The classification information is presented by System* in the following order:

Item No.	MPL Number†	Title
R10	R35	Electrical Wiring Penetration
R11	R40	Combustion Turbine Generator
R12	R42	Direct Current Power Supply‡
R13	R43	Emergency Diesel Generator System‡
R14	R46	Vital AC Power Supply
R15	R47	Instrument and Control Power Supply
R16	R51	Communication System
R17	R52	Lighting and Servicing Power Supply
S Power Transmission Systems		
S1	S12	Reserve Auxiliary Transformer
T Containment and Environmental Control Systems		
T0	T10	Primary Containment System
T1	T11	Primary Containment Vessel
T2	T12	Containment Internal Structures
T3	T13	Reactor Pressure Vessel Pedestal
T4	T22	Standby Gas Treatment System‡
T5	T25	PCV Pressure and Leak Testing Facility
T6	T31	Atmospheric Control System
T7	T41	Drywell Cooling System
T8	T49	Flammability Control System
T9	T53	Suppression Pool Temperature Monitoring System‡
U Structures and Servicing Systems		
U1	U21	Foundation Work

* Systems that are in and out of the ABWR Standard Plant scope are included in this table. See Subsection 1.1.2 for the identification of the site-specific elements outside the scope of the ABWR Standard Plant.

† Master Parts List Number designated for the system.

‡ These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety-related components and, conversely, some systems whose primary functions are non-safety-related contain components that have been designated safety-related.

Table 3.2-1 Classification Summary (Continued)

The classification information is presented by System* in the following order:

Item No.	MPL Number†	Title
U2	U24	Turbine Pedestal
U3	U31	Cranes and Hoists
U4	U32	Elevator
U5	U41	Heating, Ventilating and Air Conditioning‡
U5.1	U42	Potable and Sanitary Water System
U6	U43	Fire Protection System
U7	U46	Floor Leakage Detection System
U8	U47	Vacuum Sweep System
U9	U48	Decontamination System
U10	U71	Reactor Building‡
U11	U72	Turbine Building‡
U12	U73	Control Building‡
U13	U74	Radwaste Building
U14	U75	Service Building
U15	U79	Control Building Annex
Y Yard Structures and Equipment		
Y1	Y31	Stack
Y2	Y52	Oil Storage and Transfer System
Y3	Y86	Site Security

* Systems that are in and out of the ABWR Standard Plant scope are included in this table. See Subsection 1.1.2 for the identification of the site-specific elements outside the scope of the ABWR Standard Plant.

† Master Parts List Number designated for the system.

‡ These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety-related components and, conversely, some systems whose primary functions are non-safety-related contain components that have been designated safety-related.

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Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B1 Reactor Pressure Vessel System							
1.	Reactor pressure vessel (RPV)	1	C	A	B	I	
2.	Reactor vessel support skirt and stabilizer	1	C	A	B	I	
3.	RPV appurtenances—reactor coolant pressure boundary portions (RCPB)	1	C	A	B	I	
4.	Lateral supports for CRD housing and in-core housing	1	C	A	B	I	
5.	Reactor internal structures, spargers, for feedwater, RHR shutdown cooling low pressure flooder, and high pressure core flooder systems (see Subsection 3.9.5)	2	C	—	B	I	
6.	Reactor internal structures—safety-related components (except spargers) including core support structures (See Subsection 3.9.5)	3	C	—	B	I	
7.	Reactor internal structures—non-safety-related components (See Subsection 3.9.5)	N	C	—	E	—	
8.	Not Used						
9.	Not Used						
10.	Not Used						
11.	Reactor Internal Pump Motor Casing (a part of RPV boundary)	1	C	A	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
B2 Nuclear Boiler System						
1. Vessels—level instrumentation condensing chambers	1	C	A	B	I	
2. Vessel-nitrogen accumulators (for ADS and SRVs)	3/N	C	C	B	I	
3. Piping including supports—safety/relief valve discharge and quencher	3	C	C	B	I	(h)
4. Piping including supports main steamline (MSL) and feedwater (FW) line up to and including the outermost isolation valve	1	C,M	A	B	I	
5. Piping including supports						
a. MSL (including branch lines to first valve) from outermost isolation valve up to and including seismic interface restraint	2	M	B	B	I	(r)
b. FW (including branch lines to first valve) from outermost isolation valve to and including the shutoff valve	2	M	B	B	I	(r)

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
6. Piping including supports—MSL (including branch lines to first valve) from the seismic interface restraint up to but not including the turbine stop valve and turbine bypass valve	N	M,T	B	E	—	(r)
7. Piping from FW shutoff valve to seismic interface restraint	N	M	D	E	I	(ee)
8. Not Used						
9. Not Used						
10. Pipe whip restraint—MSL/FW	3	M,C	—	B	—	
11. Piping including supports—other within outermost isolation valves						
a. RPV head vent	1	C	A	B	I	(g)
b. Main steam drains	1	C,M	A	B	I	(g)
12. Piping including supports—other beyond outermost isolation or shutoff valves						
a. RPV head vent beyond shutoff valves	N	C	C	E	—	
b. Main steam drains to first valve	2/N	M,T	B	B	I/—	(r)
c. Main steam drains beyond first valve	N	M, T	D	E	—	(r)

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
13. Piping including supports—instrumentation up to and beyond outermost isolation valves	2/N	C,SC	B/D	B/E	I/—	(g)
14. Safety/relief valves	1	C	A	B	I	
15. Valves—MSL and FW isolation valves, and other FW valves within containment	1	C,M	A	B	I	
16. Valves—FW, other beyond outermost isolation valves up to and including shutoff valves	2	M	B	B	I	(ee)
17. Valves—within outermost isolation valves						
a. RPV head vent	1	C	A	B	I	(g)
b. Main steam drains	1	C,M	A	B	I	(g)
18. Valves, other						
a. RPV head vent	3	C	C	B	I	
b. First main steam drain valves	2/N	M	B	B	I/—	(r)
c. Other main steam drain valves	N	M,T	D	E	—	(r)
19. Not Used						
20. Mechanical modules—instrumentation with safety-related function	3	C,M,SC	—	B	I	
21. Electrical modules with safety-related function	3	C,SC,X, RZ, M	—	B	I	(i)
22. Cable with safety-related function	3	C,SC,X RZ, M	—	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B3 Reactor Recirculation System						
1. Piping, Valves and all their supports—Purge System, heat exchanger and primary side of recirculation motor cooling system (RMCS)	2	C	B	B	I	(s)(g)
2. Pump motor cover, bolts and nuts	1	C	A	B	I	
3. Pump non-pressure retaining parts including motor, instruments, electrical cables, and seals	N	C, RZ	—	E	—	
C1 Rod Control and Information System						
1. Electrical Modules	N	RZ,X	D	E	—	
2. Cable	N	SC,RZ,X	D	E	—	
C2 CRD System						
1. Valves with no safety-related function (not part of HCU)	N	SC	D	E	—	
2. Piping including supports-insert line	2	C,SC	B	B	I	(j)
3. Piping-other (pump suction, pump discharge, drive header)	N	SC	D	E	—	(g)
4. Hydraulic control unit	2	SC	—	B	I	(k)
5. Fine motion drive motor	N	C	—	E	—	
6. CRD water pumps	N	SC	D	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
7.	Control Rod Drive	1/3	C	A/—	B	I	
8.	Electrical modules with safety-related function	3	C,SC	—	B	I	
9.	Cable with safety-related function	3	C,SC,X	—	B	I	
10.	ATWS Equipment associated with the Alternate Rod Insert (ARI) functions	N	SC	—	E	—	(cc)
C3 Feedwater Control System		N	C,T, X	—	E	—	
C4 Standby Liquid Control System							
1.	Standby liquid control tank including supports	2	SC	B	B	I	
2.	Pump including supports	2	SC	B	B	I	
3.	Pump motor	2	SC	—	B	I	
4.	Valves—injection	1	SC	A	B	I	
5.	Valves within injection valves	1	C,SC	A	B	I	
6.	Valves beyond injection valves	2	SC	B	B	I	
7.	Piping including supports within injection valves	1	C,SC	A	B	I	(g)
8.	Piping including supports beyond injection valves	2	SC	B	B	I	(g)
9.	Electrical equipment and devices	3/N	SC,X, RZ	—	B/E	I/—	
10.	Cable	3/N	SC,X, RZ	—	B/E	I/ —	
C5 Neutron Monitoring System							

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
1.	Electrical modules—SRNM, LPRM and APRM	3	SC,X	—	B	I	
2.	Cable—SRNM and LPRM	3	C,SC,X, RZ	—	B	I	
3.	Detector and tube assembly	2/3	C	B/C	B	I	
C6 Remote Shutdown System							
	This system utilizes components included under B2, E1, E4, G3, H4, and P2.						
1.	Electrical modules with safety-related functions	3	RZ	—	B	I	
2.	Cable with safety-related functions	3	RZ	—	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
C7 Reactor Protection System						
1. Electrical modules with safety-related functions	3	SC,X,T, RZ	—	B	I	
2. Cable with safety-related functions	3	SC,X,T, RZ,	—	B	I	
3. Not Used						
4. Not Used						
C8 Recirculation Flow Control System	N	X, RZ, XA	—	E	—	
C9 Automatic Power Regulator System	N	X	—	E	—	
C10 Steam Bypass and Pressure Control System	N	X	—	E	—	
C11 Process Computer (includes PMCS & PGCS)	N	X	—	E	—	
C12 Refueling Platform Control Computer	N	RZ	—	E	—	
C13 CRD Removal Machine Control Computer	N	SC	—	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
D1 Process Radiation Monitoring System (includes gaseous and liquid effluent monitoring)							
1.	Electrical modules—with safety-related functions (including monitors)	3	SC,X,RZ	—	B	I	
2.	Cable with safety-related functions	3	SC,X,RZ	—	B	I	
3.	Electrical Modules, other	N	T,SC,RZ,X, W	—	E	—	(u)
4.	Cables, other	N	T,SC,RZ,X, W	—	E	—	(u)
D2	**Area Radiation Monitoring System**	N	X,T,W, SC,RZ,H	—	E	—	
D3 Containment Atmospheric Monitoring System							
1.	Component with safety-related function	3	C,SC,X RZ	—	B	I	
Notes and footnotes are listed on pages 3.2-52 through 3.2-59							

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location [c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
E1 RHR System							
1.	Heat exchangers—primary side	2	SC	B	B	I	
2.	Deleted						
3.	Piping including supports within outermost isolation valves*	1	C,SC	A	B	I	(g)
4.	Containment spray piping including supports and spargers, within and including the outer most isolation valves 2	2	C,SC	B	B	1	
4a.	Piping including supports beyond outermost isolation valves	2	SC	B	B	I	(g)
5.	Main Pumps including supports	2	SC	B	B	I	
5a.	Pump suction strainers in suppression pool	2	C	B	B	I	(ii)
6.	Main Pump motors	2	SC	B	B	I	
7.	Valves—isolation, (LPFL line) including shutdown suction line isolation valves	1	C,SC	A	B	I	(g)
8.	Valves—isolation, other (pool suction valves and pool test return valves)	2	SC	B	B	I	(g)
9.	Valves beyond isolation valves	2	SC	B	B	I	(g)
10.	Jockey pumps and motors including supports	2	SC	B	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
11.	Valves to fire protection, Subsystems B and C (F100B/C, F103B/C, and F104B/C)	N	SC	—	E	—	
E2 High Pressure Core Flooder System							
1.	Reactor pressure vessel injection line and connected piping including supports within outermost isolation valve†	1/2	C,SC	A/B	B	I	(g)
2.	All other piping including supports‡	2	SC,O	B	B	I	(g)
3.	Main Pump	2	SC	B	B	I	
3a.	Pump suction strainers in suppression pool	2	C	B	B	I	(ii)
4.	Main Pump Motor	3	SC	—	B	I	
5.	Valves—other isolation and within the reactor pressure vessel injection line and connected lines	1	C,SC	A	B	I	(g)
6.	All other valves	2/3	SC	B/C	B	I	(g)
7.	Electrical modules with safety-related functions	3	C,SC,X RZ	—	B	I	
8.	Cable with safety-related functions	3	C,SC,X RZ	—	B	I	
E3 Leak Detection and Isolation System							
1.	Temperature sensors	3/N	C,SC,T,M	—	B/E	I/—	(z)
2.	Pressure transmitters	3	C,SC	—	B	I/—	(z)
3.	Differential pressure transmitters (flow)	3	C,SC	—	B	I/—	(z)
4.	Fission Product Monitor	N	SC	—	E	I	
5.	Isolation Valves	2/N	SC	B/C	B/E	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
6.	Instrument lines	3	C,SC	B	B	I	
7.	Sample lines[f]	2/N	C,SC	C/D/—	B/E	I/—	
8.	Flow transmitters	N	SC	—	E	—	
9.	Electrical modules	3/N	SC,RZ,X	—	B/E	I/—	
10.	Cables	3/N	SC,RZ,X	—	B/E	I/—	
E4 RCIC System							
1.	Piping including supports within outermost isolation valves	1	C,SC	A	B	I	
2.	Piping including supports—discharge line from vacuum pump to containment isolation valves, and discharge line from condensate pump to the first globe valve	N	SC	C	E	—	(g)
3.	Piping including supports beyond outermost isolation valves up to the turbine exhaust line to the suppression pool, including turbine inlet and outlet drain lines	2/3	C,SC	B/C	B	I	(g)
4.	RCIC Pump and piping including support, CST suction line from the first RCIC motorized valve, S/P suction line to the pump, discharge line up to the FW line “B” thermal sleeve	2	SC, M	B	B	I	(g)
4a.	Pump suction strainer in suppression pool	2	C	B	B	I	(ii)

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
	5. Other Pump motors (Support Systems)	N	SC	—	E	I	
	6. Valves—outer isolation and within	1	C,SC	A	B	I	(g)
	7. Valves—outside the PCV (except item 8)	2	SC	B	B	I	(g)
	8. Valves—beyond turbine inlet drain line second shutoff	N	SC	C	E	I	(g)
	9. Turbine including supports	2	SC	—	B	I	(m)
	10. Electrical modules with safety-related functions	3	SC,X,RZ	—	B	I	
	11. Cable with safety-related functions	3	C,SC,X, RZ	—	B	I	
	12. Other mechanical and electrical modules	N	SC,X	—	E	—	
F1	**Fuel Servicing Equipment**	N/2	SC	—/B	E/B	—	(x)
F2	**Miscellaneous Servicing Equipment**	N	SC,RZ	—	E	—	
F3	**RPV Servicing Equipment**	N/2	SC	—/B	E/B	—/I	(gg)
F4	**RPV Internal Servicing Equipment**	N	SC	—	E	—	
F5	**Refueling Equipment**						
	1. Refueling equipment machine assembly	N	SC	—	E	I	(bb)

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
F6 Fuel Storage Equipment						
1. Fuel and equipment storage racks—new and spent	N	SC	—	E	I	(bb)
2. Defective fuel container	N	SC	—	E	—	(bb)
3. Spent fuel pool liner	N	SC	—	E	I	
F7 Under-Vessel Servicing Equipment	N	SC	—	E	—	(bb)
F8 CRD Maintenance Facility	N	SC	—	E	—	
F9 Internal Pump Maintenance Facility	N	SC	—	E	—	
F10 Fuel Cask Cleaning Facility	N	SC	—	E	—	
F11 Plant Start-up Test Equipment	N	C,SC,M, RZ,X,T	—	E	—	
F12 Inservice Inspection Equipment	N	C,SC,M, RZ,X,T,U	—	E	—	
G1 Reactor Water Cleanup System						
1. Vessels including supports (filter/ demineralizer)	N	SC	C	E	—	
2. Regenerative heat exchangers including supports carrying reactor water	N	SC	C	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
3.	Cleanup recirculation pump, motors	N	SC	C	E	—	
4.	Piping including supports and valves within and including outermost containment isolation valves	1	C,SC	A	B	I	(g)
5.	Pump suction and discharge piping including supports and valves from containment isolation valves back to and including shut-off valve at feedwater line connection	N	SC,M	C	E	—	
6.	Piping including supports and valves leading to radwaste and main condenser	N	SC	C	E	—	
7.	Non-regenerative heat exchanger tube inside and piping including supports and valves carrying process water	N	SC	C	E	—	
8.	Non-regenerative heat exchanger shell and piping including supports carrying closed cooling water	N	SC	D	E	—	
9.	Filter/demineralizer precoat subsystem	N	SC	D	E	—	
10.	Filter demin holding pumps including supports—valves and piping including supports	N	SC	C	E	—	
11.	Sample station	N	SC	D	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
12.	Electrical modules and cable with no safety-related functions	N	SC,X	—	E	—	
13.	Electrical modules and cable for isolation valves	3	C,SC,RZ, X	—	B	I	
G2 Fuel Pool Cooling and Cleanup System							
1.	Vessels including supports—filter/ demineralizers	N	SC	D	E	—	
2.	Piping and valves including supports upstream of F/D outlet isolation valve	N	SC	D	E	—	
3.	Piping and valves including supports downstream of F/D inlet isolation valve	N	SC	D	E	—	
4.	Heat exchangers including supports	N	SC	C	E	I	
5.	Pumps including supports	N	SC	C	E	I	
6.	Pump motors	N	SC	—	E	—	
7.	Piping including supports and valves—cooling portion	N	SC	C	E	I	
8.	Makeup Water System (MUWC) connection including valves and supports	N	SC	C	E	I	
9.	RHR piping connections and valves including supports for safety-related makeup and supplemental cooling	3	SC	C	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
10.	SPCU piping connections and valves including supports	3	SC	C	B	I	
11.	Electrical modules and cables with no safety-related function	N	SC, X	—	E	—	
12.	Spent Fuel Pool Wide Range Level Instrumentation	3	SC, X	C	B	I	
G3 Suppression Pool Cleanup System							
1.	Isolation valves and piping including supports within outermost isolation valves	2	SC	B	B	I	
2.	Pump including supports	N	SC	C	E	I	
3.	Pump motor	N	SC	—	E	—	
4.	Piping and components beyond outermost-containment isolation valve including supports	N	SC	C	E	I	
5.	Not Used						
6.	Not Used						
7.	Electrical modules and Cables with no safety-related function	N	SC,X	—	E	—	
8.	Electrical modules and cables for isolation valves	3	SC,X,RZ	—	B	I	
H1 Main Control Room Panels							
1.	Panels	3/N	X	—	B/E	I/—	(aa)
2.	Electrical Modules with safety-related functions	3	X	—	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
3.	Cable with safety-related functions	3	X	—	B	I	
4.	Other mechanical and electrical modules	N	X	—	E	—	
H2	**Control Room Back Panels**						
1.	Panels	3/N	X	—	B/E	I/—	(aa)
2.	Electrical modules with safety-related function	3	X	—	B	I	
3.	Cable with safety-related function	3	X	—	B	I	
4.	Other mechanical and electrical modules	N	X	—	E	—	
H3	**Radioactive Waste Control Panels**	N	W	—	E	—	(p)
H4	**Local Control Panels**						
1.	Panels and Racks	3/N	RZ,SC,X	—	B/E	I/—	(aa)
2.	Electrical modules with safety-related functions	3	RZ,SC,X	—	B	I	
3.	Cable with safety-related functions	3	RZ,SC,X	—	B	I	
4.	Other mechanical and electrical modules	N	RZ,SC,X	—	E	—	
H5	**Instrument Racks**						
1.	Mechanical and electrical with safety-related functions	3	SC,RZ, X,W,M	—	B	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
2.	Other mechanical and selected modules	N	SC,RZ,X, T	—	E	—	
H6 Multiplexing System							
1.	Electrical module with safety-related functions (Essential)	3	RZ,X	—	B	I	
2.	Cable with safety-related functions (Essential)	3	RZ,X	—	B	I	
3.	Other electrical modules and cables (Non-essential)	N	SC,RZ,X, W	—	E	—	
H7 Local Control Boxes							
1.	Electrical modules with safety-related functions	3	SC,RZ,X, H,T,W,M	—	B	I	
2.	Other electrical modules	N	SC,RZ,X, H,T,W,M,	—	E	—	
J1 Fuel Assembly							
1.	Fuel assemblies	3	C,SC	—	B	I	
2.	Control Rods	3	C,SC	—	B	I	
3.	Loose Parts Monitoring System	N	C,SC	—	E	—	
J2 Fuel Channel		3	C,SC	—	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
K1 Radwaste System						
1. Drain piping including supports and valves—radioactive	N	C,H,SC,T, W,X	D	E	—	(p)
2. Not Used						
3. Not Used						
4. Piping including supports and valves forming part of containment boundary	2	C,SC	B	B	I	
5. Pressure vessels including supports	N	W	—	E	—	(p)
6. Atmospheric tanks including supports	N	C,SC,H,T, W	—	E	—	(p)
7. 0-103.42 kPaG Tanks and supports	N	W	—	E	—	(p)
8. Heat exchangers and supports	N	C,SC,W	—	E	—	(p)
9. Piping including supports and valves	N	C,SC,H, T,W	—	E	—	(p)
10. Other mechanical and electrical modules	N	ALL	D	E	—	(p)
11. ECCS equipment room sump backflow protection check valves	N	SC	C	B	I	
12. Control Building high water level sensors	3	X	—	B	I	
13. Electrical modules and cables with safety-related functions	3	C, SC, X, RZ	—	B	I	
N1 Turbine Main Steam System						
1. Not Used (see B2.5)						
2. Not Used (see B2.6)						

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
N2 Condensate, Feedwater and Condensate Air Extraction System						
1. Feedwater system components beyond seismic interface restraint	N	M,T	D	E	—	(ee)
N3 Heater, Drain and Vent System	N	T	—	E	—	
N4 Condensate Purification System	N	T	—	E	—	
N5 Condensate Filter Facility	N	T	—	E	—	
N6 Condensate Demineralizer	N	T	—	E	—	
N7 Main Turbine	N	T	—	E	—	
N8 Turbine Control System						
1. Turbine stop valve, turbine bypass valves, and the main steam leads from the turbine stop valve to the turbine casing	N	T	D	E	—	(l)(n)(o)(r)
N9 Turbine Gland Steam System	N	T	D	E	—	
N10 Turbine Lubricating Oil System	N	T	—	E	—	
N11 Moisture Separator Heater	N	T	—	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
N12 Extraction System	N	T	—	E	—	
N13 Turbine Bypass System						
1. Turbine bypass piping including supports up to the condenser	N	T	D	E	—	(r)
N14 Reactor Feedwater Pump Driver	N	T	—	E	—	
N15 Turbine Auxiliary Steam System	N	T	—	E	—	
N16 Generator	N	T	—	E	—	
N17 Hydrogen Gas Cooling System	N	T	—	E	—	
N18 Generator Cooling System	N	T	—	E	—	
N19 Generator Sealing Oil System	N	T	—	E	—	
N20 Exciter	N	T	—	E	—	
N21 Main Condenser	N	T	—	E	—	
N22 Offgas System	N	T	—	E	—	(p)(q)(v)
N23 Circulating Water System	N	T	D	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classi- fication[d]	Quality Assur- ance Require- ment[e]	Seismic Category[f]	Notes
N24 Condenser Cleanup Facility	N	T	—	E	—	
P0 Makeup Water System (Preparation)						
1. Demineralizer water storage tank including supports	N	O	D	E	—	
P1 Makeup Water System (Purified)						
1. Piping including supports and valves forming part of the containment boundary	2	C, SC	B	B	I	
2. Piping including supports and valves	N	SC,RZ,T, H,W,X, XA	D	E	—	
3. Other components	N	O	D	E	—	
P2 Makeup Water System (Condensate)						
1. Condensate storage tank including supports	N	O	D	E	—	(w)
2. Condensate header—piping including supports, level instrumentation and valves	2	SC	B	B	I	
3. Piping including supports and valves and other components	N	O	D	E	—	
P3 Reactor Building Cooling Water System						
1. Piping and valves forming part of primary containment boundary	2	SC,C	B	B	I	(g)

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
2.	Other safety-related piping including supports, pumps and valves	3	SC,RZ,X, XA	C	B	I	
3.	Electrical modules with safety-related functions	3	SC,RZ,X	—	B	I	
4.	Cable with safety-related functions	3	SC,X,C, RZ	—	B	I	
5.	Other mechanical and electrical modules	N	SC,C,X,T W,RZ	D	E	—	
P4	**Turbine Building Cooling Water System**	N	T	D	E	—	
P5	**HVAC Normal Cooling Water System**						
1.	Piping including supports and valves forming part of containment boundary	2	C,SC	B	B	I	
2.	Other mechanical and electrical modules	N	C,SC,RZ T,X, XA	—	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P6 HVAC Emergency Cooling Water System						
1. Chillers, pumps, valves, and piping, including supports	3	SC,X,RZ	C	B	I	
2. Electrical modules and cable with safety-related functions	3	RZ,X	—	B	I	
P7 Oxygen Injection System	N	T	—	E	—	
P8 Ultimate Heat Sink	3	O	C	B	I	
P9 Reactor Service Water System						
1. Safety-related piping including supports, piping and valves	3	U,O,X	C	B	I	
2. Electrical modules and cables with safety-related functions	3	RZ,U,O,X	—	B	I	
P10 Turbine Service Water System						
1. Non-safety-related piping including supports, piping and valves	N	P, O, T	—	E	—	
2. Electrical modules and cables with non-safety-related functions	N	P,O,T	—	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P11 Station Service Air System							
1.	Containment isolation including supports, valves and piping	2	C,SC	B	B	I	
2.	Other non-safety-related mechanical and electrical components	N	SC,RZ, X,T,H, W,C, XA	—	E	—	
P12 Instrument Air Service							
1.	Containment isolation including supports, valves and piping	2	C,SC	B	B	I	
2.	Other non-safety-related mechanical components	N	C,RZ,T,H, W,SC,X, XA	—	E	—	
3.	Other non-safety-related electrical components	N	SC,RZ,X, T,H, W,C	—	E	—	
P13 High Pressure Nitrogen Gas Supply Systems							
1.	Containment isolation including supports, valves and piping	2	C,SC	B	B	I	
2.	Gas bottles, piping and valves including supports with safety-related functions	3	SC,C	C	B	I	
3.	Electric modules with safety-related functions	3	SC,RZ,X	—	B	I	
4.	Cable with safety-related functions	3	SC,RZ,X	—	B	I	
5.	Other non-safety-related mechanical components	N	SC,RZ,X	C	E	—	
6.	Other non-safety-related electrical components	N	SC,RZ,X	—	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]			Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P14	**Heating Steam and Condensate Water Return System**		N	T,SC,W	—	E	—	
P15	**House Boiler**		N	T	—	E	—	
P16	**Hot Water Heating System**		N	T	—	E	—	
P17	**Hydrogen Water Chemistry System**		N	T	—	E	—	
P18	**Zinc Injection System**		N	T	—	E	—	
P19	**Breathing Air System**		N	C,SC,T	—	E	—	
P20	**Sampling System (Includes PASS)**		N	SC,RZ,T	—	E	—	
P21	**Freeze Protection System**		N	O	—	E	—	
P22	**Iron Injection System**		N	T	—	E	—	
R1	**Electrical Power Distribution System**							
	1.	120 VAC safety-related distribution equipment including inverters	3	SC,X, RZ,U	—	B	I	
	2.	Safety-related Motors	3	SC,C,X, RZ,U	—	B	I	
	3.	Safety-related Protective relays and control panels	3	SC,X,RZ, U	—	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
4. Safety-related Valve operators	3	SC,C, X, RZ,U	—	B	I	
R2 Unit Auxiliary Transformers						
1. Unit Auxiliary Transformers	N	O	—	E	—	
2. Safety-related Transformers	3	RZ	—	B	I	
R3 Isolated Phase Bus	N	O,T	—	E	—	
R4 Non-Segregated Phase Bus	N	O,T	—	E	—	
R5 Metalclad Switchgear						
1. Safety-related 6900 Volt switchgear	3	RZ	—	B	I	
R6 Power Center						
1. Safety-related 480 Volt power centers	3	RZ,U	—	B	I	
R7 Motor Control Center						
1. Safety-related 480 Volt motor control centers	3	X,RZ,U	—	B	I	
R8 Raceway System						
1. Safety-related control and power cables (including underground cable systems, cable splices, connectors and terminal blocks	3	SC, C, X, M,RZ,O,U	—	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
	2. Safety-related conduit and cable trays and their supports	3	SC,C,X M,RZ,O,U	—	B	I	
R9	**Grounding Wire**	N	ALL	—	—	—	
R10	**Safety-related Electrical Wiring Penetrations**	3	SC,C	—	B	I	
R11	**Combustion Turbine Generator**	N	T	—	E	—	
R12	**Safety-related Direct Current Power Supply**						
	1. 125 Volt batteries, battery racks, battery chargers, and distribution equipment	3	SC,X, RZ,U	—	B	I	
	2. Protective relays and control panels	3	SC,X,RZ, U	—	B	I	
	3. Motors	3	SC,X, RZ	—	B	I	
R13	**Emergency Diesel Generator System**						
	1. Starting air receiver tanks piping including supports from and including check valve and downstream piping including supports, valves, and compressors.	3	RZ	C	B	I	(y)
	2. Starting air compressor motors	3	RZ	—	B	I	
	3. Combustion air intake and exhaust system	3	RZ,O	C	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
4.	Safety-related piping including supports, valves—fuel oil system, diesel cooling water system, and lube oil system	3	RZ,O	C	B	I	
5.	Pump motors—fuel oil system, diesel cooling water system and lube oil system	3	RZ,O	—	B	I	
6.	Diesel generators	3	RZ	—	B	I	(y)
7.	Mechanical and electrical modules with safety-related functions	3	RZ,X	—	B	I	
8.	Cable with safety-related functions	3	RZ,O,X	—	B	I	
9.	Other mechanical and electrical modules	N	RZ,O	—	E	—	
R14	**Safety-related Vital AC Power Supply**	3	X	—	B	I	
R15	**Safety-related Instrument and Control Power Supply**	3	X	—	B	I	
R16	**Communication System**	N	SC,C,RZ, X	—	B	I	
R17	**Lighting and Servicing Power Supply**						
1.	Normal Lighting	N	ALL	—	E	—	
2.	Standby Lighting	3/N	ALL	C/—	B/E	I/—	(hh)
3.	DC Emergency Lighting	3/N	X,W,RZ,T	C/—	B/E	I/—	(hh)
4.	Guide Lamp Lighting	3/N	ALL (except C)	C/—	B/E	I/—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]			Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
S1	**Reserve Auxiliary Transformer**		N	O	—	E	—	
T0	**Primary Containment System**							
	1.	Suppression chamber/drywell vacuum breakers	2	C	B	B	I	
T1	**Primary Containment Vessel**							
	1.	Primary containment vessel (PCV)—reinforced concrete containment vessel (RCCV)	2	C	B	B	I	
	2.	Vent system (vertical flow channels and horizontal discharges	2	C	B	B	I	
	3.	PCV penetrations and drywell steel head	2	C	B	B	I	
	4.	Upper and lower drywell airlocks	2	C,SC	—	B	I	
	5.	Upper and lower drywell equipment hatches	2	C,SC	—	B	I	
	6.	Lower drywell access tunnels	2	C	—	B	I	
	7.	Suppression chamber access hatch	2	C,SC	—	B	I	
	8.	Safety-related instrumentation	3	C,SC	—	B	I	
T2	**Containment Internal Structures**							
	1.	RPV stabilizer truss (see B1.2)						

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
2.	Support structures and equipment for safety-related piping	3	C	—	B	I	
3.	Diaphragm Floor	3	C	—	B	I	
4.	L/D equipment and personnel tunnels	3	C	—	B	I	
5.	Miscellaneous Platforms	3	C	—	B	I	
T3	**RPV Pedestal**						
1.	RPV pedestal and shield wall	3	C	—	B	I	
T4	**Standby Gas Treatment System**						
1.	All equipment except deluge piping and valves	3	SC,RZ	C	B	I	
T5	**PCV Pressure and Leak Testing Facility**	N	SC	—	E	—	
T6	**Atmospheric Control System**						
1.	Nitrogen Storage Tanks	N	O	—	E	—	
2.	Vaporizers and controls	N	O	—	E	—	
3.	Piping including supports and valves forming part of containment boundary	2	SC	B	B	I	
4.	Piping including supports and valves beyond the first rupture disk up to and including the second rupture disk	3	SC	C	B	I	
5.	Electrical modules with safety-related functions	3	SC,X,RZ	—	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
6. Cables with safety-related function	3	SC,X,RZ	—	B	I	
7. Other non-safety-related mechanical and electrical components	N	SC,O,X	—	E	—	
T7 Drywell Cooling System						
1. Motors	N	C	—	E	—	
2. Fans	N	C	—	E	—	
3. Coils, cooling	N	C	—	E	—	
4. Other mechanical and electrical modules	N	C,X,RZ	—	E	—	
T8 Flammability Control System	2	SC	B	B	I	
T9 Suppression Pool Temperature Monitoring System						
1. Electrical modules with safety-related functions	3	C,X,SC, RZ	—	B	I	
2. Cable with safety-related functions	3	C,X,SC, RZ	—	B	I	
U1 Foundation Work	2/3	C,SC,RZ	—	B	I	
U2 Turbine Pedestal	N	T	—	E	—	
U3 Cranes and Hoists						
1. Reactor Building crane	N	SC	—	E	—	(x)
2. Refueling Platform	N	SC	—	E	—	(x)
3. Upper Drywell Servicing	N	C	—	E	I	
4. Lower Drywell Servicing	N	C	—	E	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
5. Main Steam Tunnel Servicing	N	M	—	E	—	
6. Special Service Rooms	N	SC,RZ,T, W,X, XA	—	E	—	
U4 Elevator	N	SC,RZ,X	—	E	—	
U5 Heating, Ventilation and Air Conditioning**						
1. Safety-related equipment††						
a. Fan-coil cooling units	3	SC,RZ,X	—	B	I	
b. Heating units—electrical or water	3	SC,RZ,X	—	B	I	
c. Blowers—Air supply or	3	SC,RZ,X	—	B	I	
d. Ductwork	3	SC,RZ,X	—	B	I	
e. Filters—Equipment areas	3	SC,RZ,X	—	B	I	
f. HEPA Filters, Charcoal Adsorbers—Control Rooms and Secondary Containment	3	SC,X	—	B	I	
g. Valves and Dampers—secondary containment isolation	3	SC,RZ	—	B	I	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
h. Other safety-related valves and dampers	3	RZ,X	—	B	I	
i. Electrical modules with safety-related functions	3	SC,RZ,X	—	B	I	
j. Cable with safety-related functions	3	SC,RZ, X	—	B	I	
2. Non-safety-related equipment‡‡						
a. HVAC mechanical or electrical components with non-safety-related functions	N	SC,RZ,H X,W,T, C,M, XA	—	E	—	
b. Non-safety-related fire protection valves and dampers	N	SC,RZ,H, X,W,T	—	E	—	(t)(u)
U5.1 Potable and Sanitary Water System						
1. Potable and sanitary water equipment	N	All (except SC,C, M)	—	E	—	
2. Drain piping including supports and valves—nonradioactive	N	All (except SC,C, M)	D	E	—	
U6 Fire Protection System						
1. Other piping including supports and valves	N	SC,X, RZ,H,T, W,O, XA	D	E	—	(t) (u)
2. Water storage tank	N	O	D	E	—	(t) (u)

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
3. Pumps	N	F	D	E	—	(t) (u)
a. Motor Driven	N	F	D	E	—	(t) (u)
b. Engine Driven	N	F	D	E	—	(t) (u)
4. Pump motors	N	F	—	E	—	(t) (u)
5. Electrical Modules	N	SC,X RZ,H, T,W,F	—	E	—	(t) (u)
6. Not Used						
7. Cables	N	SC,X,RZ, H,T,W,F	—	E	—	(t) (u)
8. Sprinklers or deluge water	N	H,W,SC, RZ,T,O	D	E	—	(t) (u)
9. Foam, reaction or deluge	N	RZ,T	—	E	—	(t) (u)
U7 Floor Leakage Detection System	N	SC,RZ	—	E	—	
U8 Vacuum Sweep System	N	C,SC	—	E	—	
U9 Decontamination System	N	C,SC,RZ T,W,S,X	—	E	—	
U10 Reactor Building	3	C,SC,RZ, M	—	B	I	
U11 Turbine Building	N	T	—	E	—	(v)
U12 Control Building	3	X	—	B	I	
U13 Radwaste Building						

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
	1. Structural walls and slabs above grade level (see Subsection 3H.3.3)	N	W	—	E	—	
	2. Radwaste Building Substructure	3	W	—	B	I	
U14	**Service Building**	N	H	—	E	—	
U15	**Control Building Annex**	N	XA	—	E	—	(u)
Y1	**Stack**	N	RZ	—	E	—	
Y2	**Diesel Generator Fuel Oil Storage and Transfer System**	3	O,RZ	—	B	I	
Y3	**Site Security**	N	ALL	—	E	—	

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

Table 3.2-1 Notes and Footnotes

* The RHR/ECCS low pressure flooder spargers are part of the reactor pressure vessel system, see Item B1.5.

† The ECCS high pressure core flooder spargers are part of the Reactor Pressure Vessel System, see Item B1.5.

‡ Pool suction piping, suction piping from condensate storage tank, test line to pool, pump discharge piping and return line to pool.

ƒ These sample lines are totally within containment and the fission product monitor provides no isolation function.

** Includes Reactor Building, Control Building, and Service Building thermal and radiological environmental control functions within the ABWR Standard Plant.

†† Controls environment in Main and Local control rooms, diesel-generator rooms, battery rooms, ECCS-RCIC, pump rooms within the ABWR Standard Plant.

‡‡ Controls environment in rooms or areas containing non-safety-related equipment within the ABWR Standard Plant.

a. A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, signal processors, and mechanical modules include turbines, strainers, and orifices. Safety-related motor control centers, power centers, metal clad switchgear, and remote mulitplexing units in the Reactor Building are located outside the Secondary Containment in the emergency electric equipment rooms. The specific location of many of the electrical modules in the Reactor Building are given on Table 9A.6-2.

b. 1, 2, 3, N = Nuclear safety-related function designation defined in Subsections 3.2.3 and 3.2.5.

c.
C = Primary Containment
H = Service Building
M = Reactor Building steam tunnel
O = Outside onsite
RZ = Reactor Building Clean Zone (balance portion of the reactor building outside the Secondary Containment Zone)
SC = Secondary Containment portion of the reactor building
T = Turbine Building
W = Radwaste Building
X = Control Building
F = Firewater Pump House*
U = Ultimate Heat Sink Pump House*
P = Power Cycle Heat Sink Pump House*
XA = Control Building Annex

d. A,B,C,D= Quality groups defined in Regulatory Guide 1.26 and Subsection 3.2.2. The structures, systems and components are designed and constructed in accordance with the requirements identified in Tables 3.2-2 and 3.2-3.

--- = Quality Group Classification not applicable to this equipment.

e. B = The quality assurance requirements of 10CFR50, Appendix B are applied in accordance with the quality assurance program described in Chapter 17.

E = Elements of 10CFR50, Appendix B are generally applied, commensurate with the importance of the equipment's function.

f. I = The design requirements of Seismic Category I structures and equipment are applied as described in Section 3.7, Seismic Design.

--- = The seismic design requirements for the safe shutdown earthquake (SSE) are not applicable to the equipment. However, the equipment that

* Pump House structures are out of the ABWR Standard Plant scope.

is not safety-related but which could damage Seismic Category I equipment if its structural integrity failed is checked analytically and designed to assure its integrity under seismic loading resulting from the SSE.

g. 1. Lines 25A and smaller which are part of the reactor coolant pressure boundary and are ASME Code Section III, Class 2 and Seismic Category I.

2. All instrument lines which are connected to the reactor coolant pressure boundary and are utilized to actuate and monitor safety systems shall be Safety Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

3. All instrument lines which are connected to the reactor coolant pressure boundary and are not utilized to actuate and monitor safety systems shall be Code Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

4. All other instrument lines:

i. Through the root valve the lines shall be of the same classification as the system to which they are attached.

ii. Beyond the root valve, if used to actuate a safety system, the lines shall be of the same classification as the system to which they are attached.

iii. Beyond the root valve, if not used to actuate a safety system, the lines may be Code Group D.

5. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system may be Code Group D.

6. All safety-related instrument sensing lines shall be in conformance with the criteria of Regulatory Guides 1.11 and 1.151.

h. Safety/Relief valve discharge line (SRVDL) piping to the quencher shall be Quality Group C and Seismic Category I. In addition, all welds in the SRVDL piping in the wetwell above the surface of the suppression pool shall be non-destructively examined to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 2.

SRVDL piping from the safety/relief valve to the quenchers in the suppression pool consists of two parts: the first part is located in the drywell and is attached at one end to the safety/relief valve and attached at its other end to the diaphragm floor penetration. This first part of the SRVDL is analyzed with the main steam piping as a complete system. The second part of the SRVDL is in the wetwell and extends from the penetration to the quenchers in the suppression pool. Because of the penetration on this part of the line, it is physically decoupled from the main steam piping and the first part of the SRVDL piping and is, therefore, analyzed as a separate piping system.

i. Electrical devices include components such as switches, controllers, solenoids, fuses, junction boxes, and transducers which are discrete components of a larger subassembly/module. Nuclear safety-related devices are Seismic Category I. Fail-safe devices are non-Seismic Category I.

j. The control rod driver insert lines from the drive flange up to and including the first valve on the hydraulic control unit are Safety Class 2, and non-safety-related beyond the first valve.

k. The hydraulic control unit (HCU) is a factory-assembled engineered module of valves, tubing, piping, and stored water which controls two control rod drives by the application of pressures and flows to accomplish rapid insertion for reactor scram.

   Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by Groups A, B, C, and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connection to conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments).

   The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but of the remaining parts and details. For example: (1) all welds are LP inspected; (2) all socket welds are inspected for gap between pipe and socket bottom; (3) all welding is performed by qualified welders; and (4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer standards and proven design techniques which are not explicitly defined within the codes for Quality Groups A, B, or C. This is supplemented by the QC technique described.

l. The turbine stop valve is designed to withstand the SSE and maintain its integrity.

m. The RCIC turbine is not included in the scope of standard codes. To assure that the turbine is fabricated to the standards commensurate with safety and performance requirements, General Electric has established specific design requirements for this component which are as follows:

1. All welding shall be qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code.
2. All pressure-containing castings and fabrications shall be hydrotested at 1.5 times the design pressure.
3. All high-pressure castings shall be radiographed according to:

ASTM E-94	
E-141	
E-142	Maximum feasible volume
E-446, 186 or 280	Severity level 3

4. As-cast surfaces shall be magnetic-particle or liquid-penetrant tested according to ASME Code, Section III, Paragraphs NB-2545, NC-2545, or NB-2546, and NC-2546.
5. Wheel and shaft forgings shall be ultrasonically tested according to ASTM A-388.
6. Butt welds in forgings shall be radiographed and magnetic particle or liquid penetrant tested according to the ASME Boiler and Pressure Vessel Code, Section III Paragraph NB-2575, NC-2575, NB-2545, NC-2545, NB-2546, NC-2546 respectively. Acceptance standards shall be in accordance with ASME Boiler and Pressure Vessel Code Section III, Paragraph NB-5320, NC-5320, NB-5340, NC-5340, NB-5350, NC-5350, respectively.
7. Notification shall be made on major repairs and records maintained thereof.
8. Record system and traceability shall be according to ASME Section III, NCA-4000.
9. Quality control and identification shall be according to ASME Section III, NCA-4000.
10. Authorized inspection procedures shall conform to ASME Section III, NB-5100 and NC-5100.

11. Non-destructive examination personnel shall be qualified and certified according to ASME Section III, NB-5500 and NC-5500.

n. 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 alternate to radiographic methods. Examination procedures and acceptance standards are at least equivalent to those defined in Paragraph 136.4, Nonboiler External Piping, ANSI B31.1.

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

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valve to turbine casing utilizes quality control procedures equivalent to those defined in GE Publication GEZ-4982A, General Electric Large Steam Turbine Generator Quality Control Program.

2. A certification obtained from the manufacturer of these valves and steam loads demonstrates that the quality control program as defined has been accomplished.

The following requirements shall be met in addition to the Quality Group D requirements:

1. All longitudinal and circumferential butt weld joints shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrate examination may be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1.

2. All fillet and socket welds shall be examined by either magnetic particle or liquid penetrant methods. All structural attachment welds to pressure retaining materials shall be examined by either magnetic particle or liquid penetrate methods. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1

3. All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

p. A quality assurance program meeting the guidance of Regulatory Guide 1.143 will be applied during design and construction.

q. Detailed seismic design criteria for the offgas system are provided in Subsection 11.3.8.

r. See Subsection 3.2.5.3.

s. The recirculation motor cooling system (RMCS) is classified Quality Group B and Safety Class 2 which is consistent with the requirements of 10CFR50.55a. The RMCS, which is part of the reactor coolant pressure boundary (RCPB) meets 10CFR50.55a (c)(2). Postulated failure of the RMCS piping cannot cause a loss of reactor coolant in excess of normal makeup (CRD return or RCIC flow), and the RMCS is not an engineered safety feature. Thus, in the event of a postulated failure of the RMCS piping during normal operation, the reactor can be shutdown and cooled down in an orderly manner, and reactor coolant makeup can be provided by a normal make up system (e.g., CRD return or RCIC system). Thus, per 10CFR50.55a(c)(2), the RMCS need not be classified Quality Group A or Safety Class 1, however, for plant availability, the system is designed, fabricated and constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class 1 criteria as specified in Subsection 3.9.3.1.4 and Figure 5.4-4.

t. A quality assurance program for the Fire Protection System meeting the guidance of Branch Technical Position CMEB 9.5-1 (NUREG-0800), is applied.

u. Special seismic qualification and quality assurance requirements are applied.

v. See Regulatory Guide 1.143, Paragraph C.5 for the offgas vault seismic requirements.

w. The condensate storage tank will be designed, fabricated, and tested to meet the intent of API Standard API 650. In addition, the specification for this tank will require: (1) 100% surface examination of the side wall to bottom joint and (2) 100% volumetric examination of the side wall weld joints.

x. The cranes and Safety Class 2 fuel servicing equipment are designed to hold up their loads and to maintain their positions over the units under conditions of SSE.

y. All off-engine components are constructed to the extent possible to the ASME Code, Section III, Class 3.

z. Components associated with safety-related function (e.g., isolation) are safety-related.

aa. Structures which support or house safety-related mechanical or electrical components are safety-related.

bb. All quality assurance requirements shall be applied to ensure that the design, construction and testing requirements are met.

cc. A quality assurance program, which meets or exceeds the guidance of Generic Letter 85-06, is applied to all non-safety-related ATWS equipment.

dd. Deleted.

ee. Figure 3.2-2 depicts the classification requirements for the feedwater system. At the interface between Seismic and non-Seismic Category I feedwater piping system, the Seismic Category I dynamic analyses will be extended to either the first anchor point in the non-seismic system or to sufficient distance in the non-seismic system so as not to degrade the validity of the Seismic Category I analysis.

ff. Deleted

gg. The Head Holding Pedestal is non-safety related and Seismic Category I. All other reactor vessel servicing equipment is non-seismic Category I.

hh. Light fixtures and bulbs are not seismically qualified but fixtures which receive Class 1E power are seismically supported (see Subsections 9.5.3.2.2.1 and 9.5.3.2.3.1).

ii. ASME BPV Code Section III Class 2 requirements are used as guidance for specification development of the design, fabrication, and inspection of the ECCS pump suction strainers, commensurate with the safety importance of the strainers. The strainers are not required to be ASME Code stamped and no ASME Certificate of Authorization is required (the strainers do not function as a pressure boundary and are attached to the end of the piping within the suppression pool). In addition, if required, the strainers may be supported from the suppression pool wall and floor.

Table 3.2-2 Minimum Design Requirements for an Assigned Safety Designation

	Minimum Design Requirements*			
Safety Designation†	Quality Group‡	Seismic Category ƒ	Electrical Classification**	Quality Assurance††
SC-1	A	I	—	B
SC-2	B	I	—	B
SC-3	C	I	1E	B
NNS	†	‡	ƒ	**

* For structural design requirements that are not covered here and in Table 3.2-3, see Section 3.8.

† Safety designations are defined in Subsections 3.2.3 and 3.2.5.

‡ Table 3.2-3 shows applicable codes and standards for components and structures in accordance with their quality group identified in Table 3.2-1.

Non-nuclear safety (NNS) related structures, systems and equipment that are not assigned a Quality Group in Table 3.2-1 are designed to requirements of applicable industry codes and standards (see Subsection 3.2.5.2).

Some NNS structures, systems, and components are optionally designed to Quality Group C or D requirements of Table 3.2-3, per Quality Group designation on Table 3.2-1.

ƒ Seismic Category I structures, systems and components meet design and analysis requirements of Subsection 3.7.

Some NNS structures, systems and components are optionally designed to Seismic Category I design criteria as noted on Table 3.2-1. Some safety-related components (e.g., Pipe whip restraints) have no safety-related function in the event of an SSE, and are not Seismic Category I.

** Safety-related electrical equipment and instrumentation are designated SC-3 and are designed to meet IEEE Class 1E (as well as Seismic Category I) design requirements.

Some NNS electrical equipment and instrumentation are optionally designed to IEEE Class 1E requirements as noted on Table 3.2-1.

†† Safety-related structures, systems and components meet the quality assurance requirements of 10CFR50, Appendix B, as described in Chapter 17.

Some NNS structures, systems, and components meet the QA requirements as noted on Table 3.2-1.

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

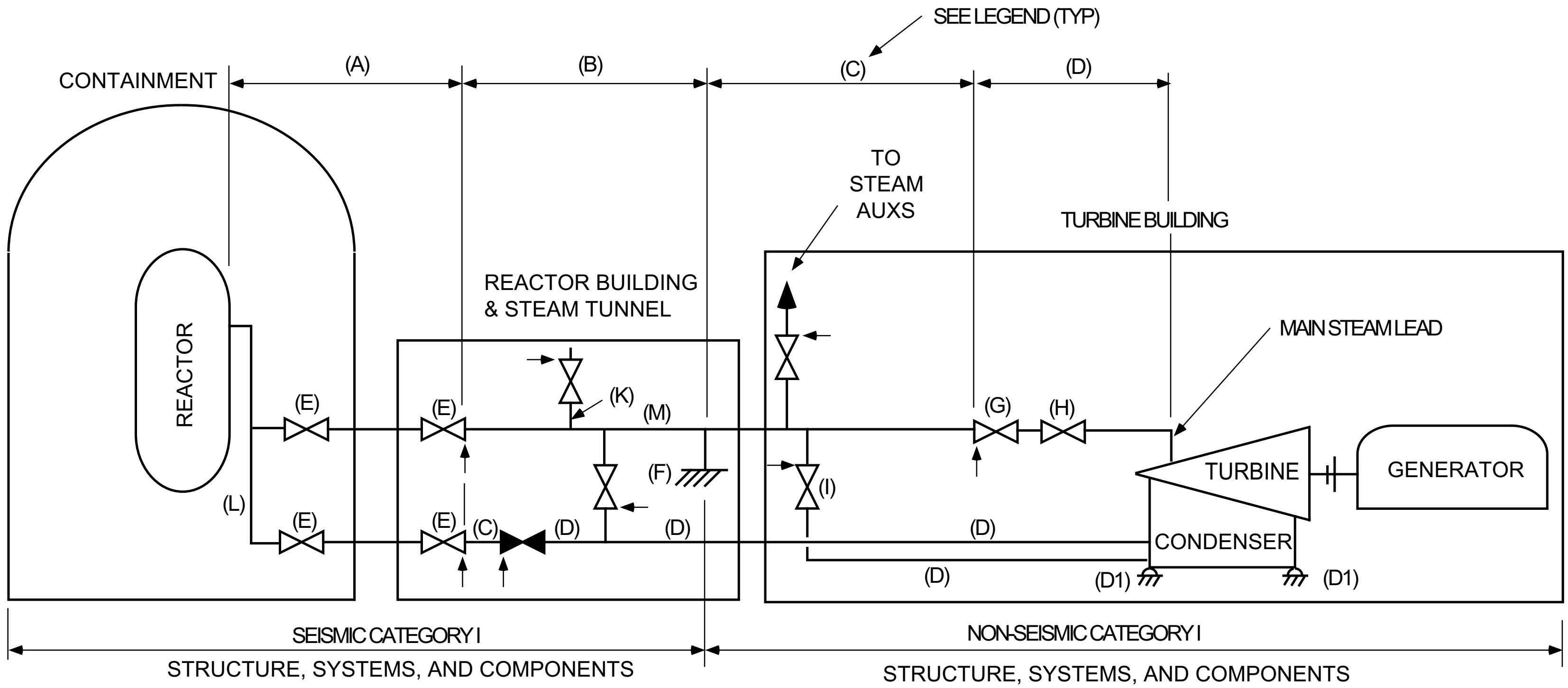
Quality Group Classif - ication	ASME Section III Code Classes*	Pressure Vessels and Heat Exchangers*	Pipes, Valves and Pumps	Storage Tanks 0-15 psig	Storage Tanks Atmospheric	ASME Section III Component Parts	Core Support Structures	Primary Containment Boundary
A	1	NCA and NB TEMA C	[*NCA and NB*]^f	—	—	[*NCA and NF*]^f	—	—
B	2	NCA and NC TEMA C	[*NCA and NC*]^f	NCA and NC	NCA and NC	[*NCA and NF*]^f	—	—
	CC and MC	—	—	—	—	—	—	NCA, CC and NE
	CS	—	—	—	—	—	NG	—
C	3	NCA and ND TEMA C	[*NCA and ND*]^f	NCA and ND	NCA and ND	[*NCA and NF*]^f	—	—
D	—	ASME Section VIII Div 1 TEMA C	Piping & Valves ANSI B31.1. Pumps†	API-620 or equivalent‡	API-650 AWWA-D100 ANSI B96.1 or equivalent	—	—	—

* Applicable Standards or Subsections of the ASME Code Section III.

† For pumps classified in Group D, ASME Code Section VIII, Division 1, shall be used as a guide in calculating the wall thickness for pressure-retaining parts ad in sizing the cover bolting.

‡ Tanks shall be designed to meet the intent of API, AWWA, and/or ANSI B96.1 Standards as applicable.

^{f} See Subsection 3.9.1.7. The change restriction is limited only to the applicability to design of piping and piping supports. See Table 1.8-21 for restriction to change of ASME Code Edition for design of piping and supports only.



LEGEND:

A. QUALITY GROUP A	G. TURBINE STOP VALVE
B. QUALITY GROUP B	H. TURBINE CONTROL VALVE
C. QUALITY GROUP B, NON-SEISMIC CATEGORY I, DYNAMICALLY ANALYZED	I. TURBINE BYPASS VALVE
D. QUALITY GROUP D, NON-SEISMIC CATEGORY I, DYNAMICALLY ANALYZED	J. MAIN STEAM LEAD
D1. NON-SEISMIC CATEGORY I, DYNAMICALLY ANALYZED	K. BRANCH LINE
E. ISOLATION VALVE	L. DRAIN LINE
F. SEISMIC INTERFACE RESTRAINT	M. STEAM LINE

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

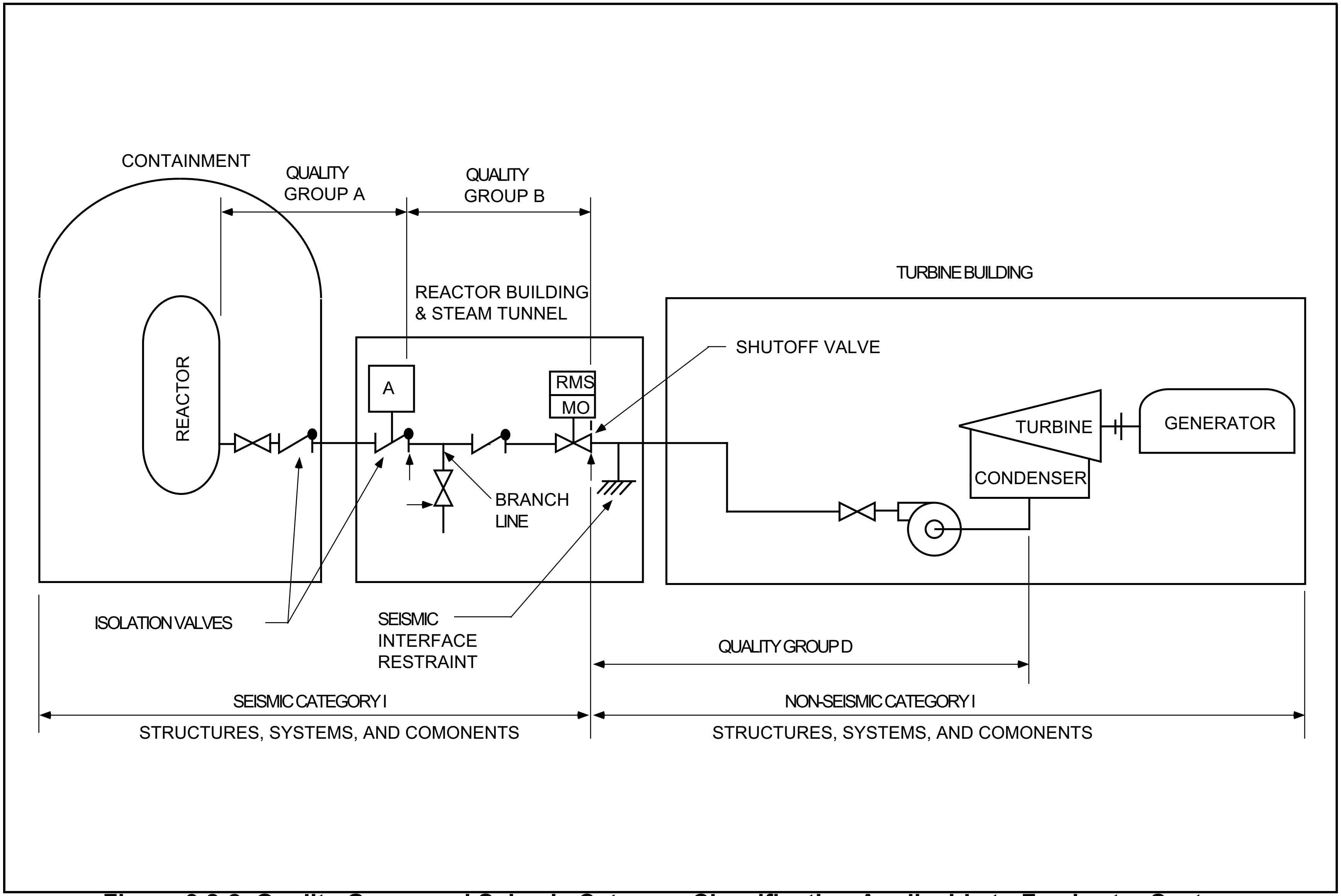


Figure 3.2-2 Quality Group and Seismic Category Classification Applicable to Feedwater System

3.3 Severe Wind and Extreme Wind (Tornado and Hurricane) Loadings

ABWR Standard Plant structures which are Seismic Category I are designed for severe and extreme wind phenomena.

3.3.1 Severe Wind Loads

3.3.1.1 Design Wind Velocity

Seismic Category I structures are designed to withstand a design wind velocity (fastest-mile) of 177 km/h with a recurrence interval of 50 years and 197 km/h with a recurrence interval of 100 years at an elevation of 10m above grade (see Subsection 3.3.3.1 and 3.3.3.3 for COL license information requirements).

3.3.1.2 Determination of Applied Forces

The design wind velocity is converted to velocity pressure using the formula given in Reference 3.3.1:

$$q_z = 4.94 \times 10^{-5} K_z (IV)^2$$

where K_z = The velocity pressure exposure coefficient which depends upon the type of exposure and height (z) above ground per Table 6 of Reference 3.3-1

I = The importance factor which depends on the type of structure; appropriate values of I are listed in Table 3.3-1

V = Design wind velocity (fastest-mile) with a recurrence interval of 50 years, in km/h, and

q_z = Velocity pressure in kPa

The design wind pressures and forces for buildings, components and cladding, and other structures at various heights above the ground are obtained, in accordance with Table 4 of Reference 3.3-1 by multiplying the velocity pressure by the appropriate pressure coefficients and gust factors. Gust factors are in accordance with Table 8 of Reference 3.3-1. Appropriate pressure coefficients are in accordance with Figures 2, 3a, 3b, 4, and Tables 9 and 11 through 16 of Reference 3.3-1. Reference 3.3-2 is used to obtain the effective wind pressures for cases which Reference 3.3-1 does not cover. Since the Seismic Category I structures are not slender or flexible, vortex-shedding analysis is not required and the above wind loading is applied as a static load.

Applied forces for the Reactor, Control and Radwaste Buildings are found in Appendices 3H.1, 3H.2 and 3H.3, respectively.

3.3.2 Extreme Wind Loads (Hurricanes and Tornados)

3.3.2.1 Applicable Design Parameters

Extreme wind loads include loads from design basis hurricane and design basis tornado.

The design basis hurricane is described by the following parameters:

(1) A maximum hurricane wind speed of 257 km/h (fastest-mile).

(2) The spectrum of hurricane generated missile and their pertinent characteristics as given in Table 2.0-1.

The design basis tornado is described by the following parameters:

(1) A maximum tornado wind speed of 483 km/h (fastest 1/4-mile) at a radius of 45.7m from the center of the tornado.

(2) A maximum translational velocity of 97 km/h.

(3) A maximum tangential velocity of 386 km/h, based on the translational velocity of 97 km/h.

(4) A maximum atmospheric pressure drop of 13.8 kPa with a rate of the pressure change of 8.3 kPa/s.

(5) The spectrum of tornado-generated missiles and their pertinent characteristics as given in Table 2.0-1.

See Subsection 3.3.3.2 for COL license information.

3.3.2.2 Determination of Forces on Structures

The procedures for transforming extreme hurricane wind loading into effective loads and distribution across the structures are in accordance with Reference 3.3-1.

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 procedure for transforming the tornado-generated missile impact into an effective or equivalent static load on structures is given in Subsection 3.5.3.1. The loading combinations of the individual tornado loading components and the load factors are in accordance with Reference 3.3-3. Per RG 1.221, in areas where effects of design basis tornado missiles do not bound the effects of site-specific hurricane missiles, the site-specific hurricane loadings should replace tornado loadings. These areas are limited to certain coastal regions along Southwestern Atlantic and Gulf of Mexico.

The reactor building and control building are not vented structures. The exposed exterior roofs and walls of these structures are designed for the 13.8 kPa pressure drop. Tornado dampers are provided on all air intake and exhaust openings. These dampers are designed to withstand a negative13.8 kPa pressure.

3.3.2.3 Effect of Failure of Structures, Systems or Components Not Designed for Tornado and Hurricane Loads

All safety-related systems and components are protected within tornado-resistant or hurricane-resistant structures.

See Subsection 3.3.3.4 for COL license information requirements.

3.3.3 COL License Information

3.3.3.1 Site-Specific Design Basis Wind

The site-specific design basis wind shall not exceed the design basis wind given in Table 2.0-1 (Subsection 2.2.1).

3.3.3.2 Site-Specific Design Basis Tornado and Hurricane

The site-specific design basis tornado and hurricane shall not exceed the design basis tornado and hurricane given in Table 2.0-1 (Subsection 2.2.1).

3.3.3.3 Effect of Remainder of Plant Structures, Systems and Components Not Designed for Wind Loads

All remainder of plant structures, systems and components not designed for wind loads shall be analyzed using the 1.11 importance factor or shall be checked that their mode of failure will not effect the ability of safety-related structures, systems or components performing their intended safety functions.

3.3.3.4 Effect of Remainder of Plant Structures, Systems, and Components Not Designed for Tornado and Hurricane Loads

All remainder of plant structures, systems, and components not designed for tornado loads and hurricane loads shall be analyzed for the site-specific loadings to ensure that their mode of failure will not effect the ability of the Seismic Category I ABWR Standard Plant structures, systems, and components to perform their intended safety functions. (See Subsection 3.3.2.3)

3.3.4 References

3.3-1 ANSI/ASCE 7-88, “Minimum Design Loads for Buildings and Other Structures.”

3.3-2 ASCE Paper No. 3269, “Wind Forces on Structures”, 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.”

Table 3.3-1 Importance Factor (I) for Wind Loads

Non-Safety-Related	Safety-Related
1.00	1.11

Notes:

(1) These values of (I) are based on Table 5 of Reference 3.3-1.

3.4 Water Level (Flood) Design

Criteria for the design basis for protection against external flooding of an ABWR plant site shall conform to the requirements of RG 1.59. The types and methods used for protecting the ABWR safety-related structures, systems and components from external flooding shall conform to the guidelines defined in RG 1.102. The design criteria for protection against the effects of compartment flooding shall conform to the requirements of ANSI/ANS-56.11 (Reference 3.4-2).

The design basis flood levels and ground water levels for the ABWR standard plant are shown in Table 3.4-1 as specified by Table 2.0-1. For those structures outside the scope of the ABWR Standard Plant (e.g., the ultimate heat sink pump house), the COL applicant will demonstrate the structures outside the scope meet the requirements of Table 2.0-1 and GDC 2 and the guidance of RG 1.102. See Subsections 3.4.3.1, 3.4.3.2, and 3.4.3.3 for COL license information requirements.

The impact of flooding was determined by the maximum flow rate and the volume of water available to feed the break. In many cases no active response was assumed to terminate the flow and the entire volume of available water was assumed to spill into the effected building. For flood water, sources outside the building (fire, service, and ground water) automatic or operator actions are required to terminate the flow.

All flood water is assumed to ultimately accumulate in the basement of the affected building. No credit is taken for sumps removing water. Safety-related equipment located in the basement is protected by locating it in flood protective compartments and/or installing it a minimum of 200 mm above the floor. Water from breaks inside flood protected compartments is prevented from entering other flood protected compartments but may flow to non-flood protected spaces (i.e. corridors). Water from breaks outside flood protected compartments is prevented from entering flood protected compartments.

Floods initiating on other elevations are prevented from effecting more than one division of safety-related equipment. Safety-related equipment is mounted a minimum of 200 mm above the floor and spray effects are confined to a single division. Design features limit the accumulation of water to depths less than 200 mm and the floors are designed to prevent water from seeping to lower elevations other than through predetermined pathways including drains, stair towers, and elevator shafts.

The plant design has been evaluated for all potential breaks and includes design features to limit damage from flooding to a single division. The ABWR can be safely shutdown following all potential floods.

3.4.1 Flood Protection

This section discusses the flood protection measures that are applicable to the standard ABWR plant safety-related structures, systems, and components for both external flooding and postulated flooding from plant component failures. These protection measures also apply to other structures that house systems and components important to safety which fall within the scope of specific plant.

A compliance review will be conducted of the as-built design against the assumptions and requirements that are the basis of the flood evaluation presented below and Appendix 19R. This as-built evaluation will be documented in a Flood Analysis Report. The report will include an assessment and disposition of any non-compliances that are found between the as-built facility and the information in this section and Appendix 19R. The criterion for determining appropriate disposition of any non-compliance(s) will be that the final as-built facility conforms with the following flood protection characteristics and the assessment in Appendix 19R.

3.4.1.1 Flood Protection Measures for Structures

The safety-related systems and components of the ABWR Standard Plant are located in the Reactor and Control Buildings which are Seismic Category I structures. Descriptions of these structures are provided in Subsections 3.8.4 and 3.8.5. The ABWR standard plant structures are protected as required (Table 3.4-1), against the postulated design basis flood level specified in Table 2.0-1. Postulated flooding from component failures in the building compartments is prevented from adversely affecting plant safety or posing any hazard to the public.

Table 3.4-1 identifies the exterior or access openings and penetrations that are provided with features for protection against floods.

3.4.1.1.1 Flood Protection from External Sources

The safety-related components located below the design basis flood level inside a Seismic Category I structure are shown in the Section 1.2 building arrangement drawings. All safety-related components located below the design flood level are protected using the hardened protection approach described below.

Seismic Category I structures remain protected for safe shutdown of the reactor during all flood conditions.

The safety-related systems and components are flood-protected either because they are located above the design flood level or are enclosed in reinforced concrete Seismic Category I structures which have the following features:

(1) Exterior walls below flood level are not less than 0.6m thick.

(2) Water stops are provided in all construction joints below flood level.

(3) Watertight penetrations and doors between Category I and non-Category I buildings are installed below flood level.

(4) Waterproof coating is applied to external surfaces exposed to flood level, and is extended a minimum of 150 mm along the penetration surfaces.

(5) Roofs are designed to prevent pooling of large amounts of water in accordance with RG 1.102.

(6) Tunnels below grade do not penetrate exterior walls.

Waterproofing of foundations and walls of Seismic Category I structures below grade is accomplished principally by the use of water stops at expansion and construction joints. In addition to water stops, waterproofing of the plant structures and penetrations that house safety-related systems and components is provided up to 8 cm above the plant ground level to protect the external surfaces from exposure to water.

The flood protection measures that are described above also guard against flooding from onsite storage tanks that may rupture. The largest is the condensate storage tank.

This tank is located between the Turbine Building and the radwaste building where there are no direct entries to these buildings. All plant entries start 30 cm above grade. Any flash flooding that may result from tank rupture will drain away from the site and cause no damage to site equipment.

Additional specific provisions for flood protection include administrative procedures to assure that all watertight doors and hatch covers are locked in the event of a flood warning. If local seepage occurs through the walls, it is controlled by sumps and sump pumps. Deterioration of exterior building wall penetration seals will be detectable by visual seepage. Corrective actions can be taken in a timely manner to control the problem. The COL applicant will review the use of penetration seals below grade and develop procedures as necessary to protect the plant against the effects of seal failure. This review will be included in the Flood Analysis (Subsection 3.4.1) and will contain information that shows that the ability to safely shutdown the reactor and maintain a safe, cold shutdown condition is maintained because either

(1) the penetration seals are highly reliable or

(2) emergency procedures are in place to guide the plant response to a postulated seal failure followed by flooding.

In the event of a flood, flood levels take a relatively long time to develop. This allows ample lead time to perform necessary emergency actions for all accesses that need to be protected.

3.4.1.1.2 Compartment Flooding from Postulated Component Failures

All piping, vessels, and heat exchangers with flooding potential in the Reactor Building are seismically qualified, and complete failure of a non-seismic tank or piping system is not applicable.

In accordance with Reference 3.4-2, leakage cracks are postulated in any point of moderate-energy piping larger than nominal 25A size; breaks are postulated in such piping if it is a high-energy piping. The leakage flow area from a crack is assumed to be a circular orifice with flow area equal to one-half of the pipe outside diameter multiplied by one-half of the pipe nominal wall thickness. Resulting leakage flow rates are approximated using Equation 3-2 from Reference 3.4-1 with a flow coefficient of 0.59 and a normal operating pressure in the pipe.

Water spray, foaming and flooding effects in the room of the pipe crack or break are considered by conservatively assuming that a division having a pipe failure within its rooms is lost from service to bring the reactor to a safe shutdown condition. In addition, the following provisions and assumptions are made to limit the effects to one division:

(a) Divisional flood walls located on the -8200 mm elevation of the Control and Reactor Buildings are 0.6m or thicker. Watertight doors and penetrations are provided in openings below the maximum flood level to prevent water seepage or flow,

(b) Doors and penetrations rated as 3 hour fire barriers are assumed to prevent water spray from crossing divisional boundaries,

(c) Floors are assumed to prevent water seepage to lower levels,

(d) Penetrations between floors for pipe, cable, HVAC duct, and other equipment will be designed to prevent water seepage to lower elevations from 200 mm of standing water through the use of seals or curbs,

(e) In general equipment access hatches shall prevent water seepage to lower elevations from 200 mm of standing water. Hatches to filter/demineralizer compartments may not be required to prevent water seepage, and

(f) Water from a pipe break is assumed to flow under non-water-tight doors and spread evenly over the available areas. Water sensitive safety-related equipment is raised 200 mm above the floor. The depth of water on the floor is limited to less than 200 mm utilizing available floor space and limiting the availability of water volumes. Furthermore, floor drains, stair towers, and elevator shafts provide drain paths to the building basements inside secondary containment.

The MSL tunnel area is instrumented with radiation and air temperature monitors that are used to automatically isolate the MSIVs upon detection of high abnormal limits.

However, in the event of worst case flooding involving a 550A nominal pipe size feedwater line break, the maximum flow rate from this high energy line break will not exceed 3.6 m^3/min over a 2-hour period. Refer to Table 15.6-16 for feedwater line leakage parameters. Water discharged from a postulated feedwater line break will be contained in the Seismic Category I structure of the MSL tunnel area and will not flood any safety-related equipment in the Reactor Building. The flooded area will be allowed to drain through normally closed floor drains in the tunnel area which are routed to the HCW sumps in the Reactor Building for collection and discharge.

No credit is taken for operation of the drain sump pumps, although they are expected to operate during some of the postulated flooding events.

After receiving a flood detection alarm, the operator has a 10-minute grace period to act in cases when flooding can be identified and terminated by a remote action from the control room. In cases involving visual inspection to identify the specific flooding source in the affected area (except ECCS areas) followed by a remote or local operator action, a minimum of 30 minutes is provided for the operator.

In all instances of compartment flooding, a single failure of an active component is considered for systems required to mitigate consequences of a particular flooding condition. The Emergency Core Cooling System (ECCS) rooms are also evaluated on the basis of a loss-of-coolant accident (LOCA) and a single active failure or a LOCA combined with a single passive failure 10 minutes or more after the LOCA.

Considering the above criteria and assumptions, analyses of piping failures and their consequences are performed to demonstrate the adequacy of the ABWR design. These analyses are provided separately for the Reactor, Control, Radwaste, Service, and Turbine Buildings.

Analyses of the worst flooding due to pipe and tank failures and their consequences are performed in this subsection for the Reactor Building, Control Building, Radwaste Building, Turbine Building and the Service Building. No credit is taken for safety-related equipment within these structures if the equipment becomes flooded. However, in accordance with Section 3.11, all safety-related equipment is qualified to high relative humidity.

There is no need of the COL applicant information requirements upon the remainder of the plant from effects of possible flooding in the ABWR Standard Plant buildings. Lines, such as storm drains and sanitary waste lines, interface with plant yard piping. However, provisions are made in these lines that, should the yard piping become plugged, crushed, or otherwise inoperable, they will vent onto the ground relieving any flooded condition.

3.4.1.1.2.1 Evaluation of Reactor Building Flood Events

Analysis of potential flooding within the Reactor Building is considered on a floor-by-floor basis. The potential consequences of the high energy breaks in the Reactor Building are evaluated in Subsection 6.2.3.3.1.

3.4.1.1.2.1.1 Evaluation of Floor 100 (B3F)

The HCW sumps in the basement floor are the collection location points for floods routed through floor drains in the building.

Worst case flooding on this floor level would result from leakage of the RHR 450 mm suction line between the containment wall and the system isolation valve (this applies also to the HPCF, RCIC, and SPCU suction lines, although in smaller line sizes). Leakage from this source may cause flooding of the affected RHR heat exchanger (Hx) room and the neighboring ECCS room of the same division at a rate of 1.04 m^3/min and may continue until equalization of water level occurs between these rooms with the suppression pool level. Flooding in these ECCS rooms may cause loss of functions for that particular divisional system loop. This will not impair the safe shutdown capability of the reactor system. Flooding of other divisions is prevented by watertight doors with open/close status indicator lights and alarms in MCR. Suction lines to other services always remain submerged. Other flooding incidents may result from failures of other piping systems penetrating the RHR Hx rooms for each division; these events, however, upon detection by sump alarms, are controllable by terminating flow with closure of valves and shutdown of pressure sources.

HPCF and RCIC Systems, while having the susceptibility to flood their respective compartments from the suppression pool, do so at lower rates than the RHR System. Failure, however, is guarded against by watertight doors so that flooding in one division does not propagate to other divisions.

A failure in the nondivisional area involving the failure of the 200A SPCU system suction line before the isolation valve will permit flooding of the fourth quadrant and corridors uncontrollably by system elements. The flooding rate, driven by suppression pool head, would not exceed 0.27 m^3/min and, depending upon the rate of manual repair, may permit 25–30 m^3 of water to escape into the fourth quadrant. Certain functions of the SPCU, CUW, and backwash systems may be lost, but because of watertight doors on divisional areas, no essential functions would be lost and plant integrity would not be in question.

Manual firefighting on this floor will bring in 0.57 m^3/min. Such activities would not create a water depth which would cause concern.

3.4.1.1.2.1.2 Evaluation of Floor 200 (B2F)

Flooding events on this floor may result from piping failures in the RHR pipeways, or from piping systems of the HPCF and RCIC. Maximum flooding would occur from a 250A RHR pressure line at a flow rate of 1.34 m^3/min. These lines are inside pipe chases. Hence, leakage from these lines accumulates on floor 100 (B3F).

In associated divisional compartments failures of 400A RCW pressure lines may also occur. Prior to system isolation, this would result in a flooding rate of 2.8 m^3/min. A total of 27 m^3 of water are spilled.

In the fourth quadrant, SPCU and CUW F/D valve rooms, holding pumps and HNH pump and Hx rooms, may experience flooding from various system line failures. The maximum flooding will be from a 200A line at pump pressure resulting in discharge of 0.92 m^3/min. A total of 27 m^3 is spilled.

The leakage may propagate between divisions but an area of 300 m^2 is available, so that depth of less than 200 mm is maintained. No water will damage safety-related equipment. Alarming and prompt operator isolation of these systems is then performed.

Firefighting on this floor will be accomplished manually with a flow of 0.57 m^3/min. Areas of activity are rather large so that this quantity of water (less than for other events above) presents no problem.

3.4.1.1.2.1.3 Evaluation of Floor 300 (B1F)

Primary flooding events on this floor are associated with pipeways and pipechases utilized by RHR, HPCF, and RCIC Systems. Maximum leakage is that postulated for a 250A RHR pressure line failure in rooms that are connected to rooms on floor 100 (B3F). Hence leakage from these breaks accumulates on floor 100 (B3F).

Flooding in the emergency electric rooms A, B and C and the remote shutdown rooms may occur from leakage or failures in the heating and ventilating chilled water supply or emergency HVAC cooling water system. These failures are limited in potential water release by line inventory and surge tank capacity and will not exceed 8 m^3, causing a total water depth of 4–5 cm.

Equipment is raised 200 mm to prevent loss of function due to flooding until response to the failure is made.

Firefighting activity in all areas of this level are carried out by manual means at a maximum rate of 0.57 m^3/min and no greater effects than those already considered will occur.

Failures in the CUW and SPCU systems filter/demineralizers and associated piping may occur as on Floor 200 but will spread over a comparable area or drain down to Floor 200 or 100 so that adequate time is available for detection and subsequent system isolation.

3.4.1.1.2.1.4 Evaluation of Floor 400 (1F)

Flooding from the RHR, HPCF, and RCIC systems may occur in valve rooms A, B and C. Maximum flooding is a failure of the 250A RHR pressure line with leakage flow of 1.34 m^3/min. These rooms are connected to floor 100 (B3F) by pipe chases. No accumulation is expected on this floor.

The floors in the valve rooms are open gratings, open to the divisional pipe chases which lead to the divisional ECCS rooms in the basement. The valve room floor is two meters higher than the access corridor such that flooding in the corridor can not drain to the divisional ECCS areas in the basement via the divisional pipe chases.

Emergency diesel generator A, B and C rooms contain cooling water piping to components of this system. Flooding may occur from failures of 200A RCW piping serving these cooling needs at a maximum rate of 0.9 m^3/min, which will fill the floor area and escape into the corridor, with potential cascading down the stairwell. The water will spread over the side areas on the lower floor while action to isolate the failed system takes place. Equipment is raised to 200 mm to prevent loss of function due to flooding until response to the failure is made.

Leakage of lubricating oil is also possible in the diesel generator rooms, but level indication provides a continuing control on this source. Even major leakage will be contained in the subject rooms due to the small inventory of fluid available.

Firefighting in the diesel generator area will be provided by a foam sprinkler system. Other firefighting will be by hoses but will be of smaller volumes than those considered, and will be of limited duration.

3.4.1.1.2.1.5 Evaluation of Floor 500 (2F)

This floor contains the following equipment areas:

(1) The emergency diesel generator A, B, and C equipment areas including fans, control panels, air storage tanks, and associated piping.

(2) The fuel pool cooling and cleanup system consisting of two circulating pumps, two heat exchangers, two filter demineralizers, instrumentation and associated valves and piping.

(3) The SGTS monitor room.

(4) The stack monitor room.

(5) The MSL tunnel area.

Flooding may occur from failure of 200A fuel pool cooling line at a maximum rate of 0.9 m^3/min, which will fill the floor area. The water will escape down stairwells or flow down the drain system to floor 100 (B3F). Due to limited inventory, water is limited to a few centimeters in depth. Safety-related equipment sensitive to water (i.e., electrical, control, and instrumentation) will be protected by raising them at least 200 mm above the floor.

Flooding may also occur inside the steam tunnel. This water volume will be kept inside the tunnel until the operators are ready to pump it to radwaste for treatment. No safety-related equipment will be effected by this break. All valve operators are well off the floor. They are expected to act prior to their immersion by any flood.

For item (1) above, the three DG equipment areas house the exhaust fans for heating and cooling these areas. Flooding can only occur from rupture of the chilled and hot water lines to the fan coils. However, any flooding is expected to be minimal and will be contained within the affected area. The three divisional DG equipment areas are separated and mechanically isolated from each other and water intrusion from a flooded compartment is unlikely.

For item (2) above, the FPC equipment is classified non-safety-related. The only safety-related piping that connects with the FPC System belongs to RHR, which is used to supplement the FPC cooling capability and also provide supplemental makeup water that may be needed to keep the fuel storage pool full to the rim. The FPC equipment is located in isolated compartments on the second floor of the Reactor Building below the fuel pool facilities. During normal plant operation, area flooding can only occur from rupture of the 300A line which will cause the loss of fuel pool cooling. This is not considered detrimental to safety, since any decrease in the level of the fuel storage pool that may result from water over heating will be made up by the RHR System (Refer to the response to Question 410.34 and 410.37 for the discussions on RHR safety-related make-up capability.)

The FPC pools are structurally designed for Seismic Category I and utilize stainless steel liners to prevent leakage out of the pool. Therefore, leakage from the fuel pool facilities is unlikely and will not impact the operational capability of other equipment.

For the areas identified under items (3) and (4) above, flooding is not possible because there are no water line connections to these rooms.

For item (5) above, the steamline tunnel area is isolated from other areas on this floor through sealed doors and firewalls. Any flooding in this area will be contained and will not propagate into other divisional areas.

3.4.1.1.2.1.6 Evaluation of Floor 600 (3F)

Flooding events at this floor level may involve fuel oil as well as water. Those divisional rooms associated with the emergency diesel generator fuel tank have the potential of leakage from the fuel storage tanks. These rooms must accommodate leakage of 11.4 m^3 for each division. Sunken volumes in the day tank rooms will successfully contain all the volume in the tanks. Leakage from these tanks will also be monitored through safety grade level indication and alarm equipment so that protracted leakage as well as gross leakage can be identified. The rooms are protected by a foam sprinkler system.

Water flooding from the cooling system may occur at about 0.15 m^3/min. If undetected for several hours water may begin cascading down the nearest stairwell but is prevented from damaging safety-related equipment by raising water sensitive equipment at least 200 mm above the floor.

In the SGTS areas, the room cooling equipment may cause flooding at a rate 0.15 m^3/min. Flooding may also occur from manual firefighting in equipment maintenance areas. Large floor areas permit spread of water at limited depth.

Flooding from the standby liquid control tanks is also possible. Maximum tank leak rate will be 0.1 m^3/min so that a response to tank level alarms within 10 minutes will limit the loss. The tanks are inside a diked area with 500 mm side walls which will contain the spill.

3.4.1.1.2.1.7 Evaluation of Floor 700 (M4F)

Flooding in the FMCRD panel rooms may occur from firefighting activities at an input rate of 0.57 m^3/min. Since these activities are manually controlled, any excessive depth of water will be noted and action taken to mitigate water intrusion to other areas.

Flooding on this level may also occur from room cooling systems or from firefighting efforts. Cooling system failures in air supply, exhaust or filter rooms may allow flooding at the rate of 0.3 m^3/min, which will flow out into adjacent corridor areas. If undetected for 10 minutes, the approximate 3 m^3 released may create a depth of a few centimeters over the available floor area; a very limited amount of water will cascade down the stairwells. Safety-related equipment will be raised at least 200 mm off the floor to minimize the impact of flooding.

Firefighting activities in this area would cause water inflow of 0.57 m^3/min under controlled conditions and expected water intrusion is no more than that above.

3.4.1.1.2.1.8 Evaluation of Floor 800 (4F)

Flooding on this floor can be caused by rupture of the RCW surge tanks A, B & C piping. However, each tank and its associated piping is located in a separate compartment which can be sealed off in the event of accidental flooding. Also, the use of pedestals for equipment installation of the RIP supply and exhaust fans and for the DG-C exhaust fans will guard against flooding this equipment.

Flooding in the main reactor hall may occur from reactor service operations. Firefighting water expended into this area would occur at a maximum rate of 0.57 m^3/min but will spread over the large service area available. Minor amounts of water may find the way to stairwells, but would not impede operations.

3.4.1.1.2.1.9 Flooding Summary Evaluation

Floor-by-floor analysis of potential pipe failure generated flooding events in the Reactor Building shows the following:

(1) Where extensive flooding occurs, such as due to suppression pool suction line failure in a basement compartment, propagation to other divisions is prevented by watertight doors. Flooding in one division is limited to that division and flood water cannot propagate to other divisions.

(2) Leakage of water from large circulating water lines, such as RCW lines, may flood rooms and corridors, but through sump alarms and leakage detection systems the control room is alerted and can control flooding by system isolation. Only a limited amount of water accumulation is expected; therefore, safety-related equipment will be kept at least 200 mm off the floor for their protection.

(3) Limited flooding that may occur from manual firefighting or from lines and tanks having limited inventory will cause only a limited amount of water accumulation; therefore, safety-related equipment will be kept at least 200 mm off the floor for their protection.

Therefore, within the Reactor Building, internal flooding events as postulated will not prevent the safe shutdown of the reactor.

3.4.1.1.2.2 Evaluation of Control Building Flooding Events

The Control Building is a seven-story building, which houses (in separate areas) the control room proper, control and instrument cabinets with power supplies, closed cooling water pumps and heat exchangers, mechanical equipment (HVAC and chillers) necessary for building occupation and environmental control for computer and control equipment, and the steam tunnel. The internal flooding events as postulated below involve flooding limited to one division only and will not prevent safe shutdown of the reactor.

The only high energy lines in the Control Building are the main steamlines and feedwater lines which pass through the steam tunnel connecting the Reactor Building to the Turbine Building. There are no openings into the Control Building from the steam tunnel. The tunnel is sealed at the Reactor Building end and open at the Turbine Building end. It consists of reinforced concrete with 1.6m thick walls, floor and ceiling. Any break in a mainsteam or a feedwater line will flood the steam tunnel with steam. The rate of blowdown will cause most of the steam to vent out of the tunnel into the Turbine Building. Water or steam cannot enter the Control Building. All water will flow into the reactor or turbine portions of the steam tunnel. See Subsection 3.6.1.3.2.3 for a description of the subcompartment pressurization analysis performed for the steam tunnel.

Moderate energy water services in the Control Building comprise 700A service water lines, 450A cooling water lines, 150 A cooling water lines to the chiller condenser, 150A fire protection lines, and 150A chilled water heater lines. Smaller lines supply drinking water, sanitary water and makeup for the chilled water system. All rooms are supplied with floor drains to route leakage to the basement floor so that control or computer equipment is not subjected to water.

Maximum flooding may occur from leakage in a 700A service water line at a maximum rate of 12.0 m^3/min. Alarms (two-out-of-four logic) have been installed inside the RCW/RSW heat exchanger room to warn operators of a flood. The first alarm is 400 mm above the basemat. It will warn the operators of flooding in a division. In the case of a RSW piping failure, a second set of alarms (two-out-of-four logic) are located 1500 mm above basemat. This alarm will only sound in the event of a RSW piping failure inside the Control Building. The level sensors are diverse and are powered from their respective divisional Class 1E power supply. These sensors send signals to the corresponding divisions of the RSW systems indicating flooding in that division of the C/B. This signal automatically closes isolation valves, stops the pumps, and alarms the operators in the MCR. The expected release of a service water leak is limited to line volume plus 1500 mm depth of water in a division. Water will be contained inside a division at the bottom level of the control building. A maximum of 5.0m of water is expected assuming 2 km of service water piping out to UHS pump house. Watertight doors will confine the water to a single division.

The failure of a cooling water line in the mechanical rooms of the Control Building may result in a leak of 0.6 m^3/min. Early detection by control room personnel will limit the extent of flooding. Total release from the chilled water system will be limited to line inventory and surge tank volume, spillage of more than 6 m^3 is unlikely. Elevation differences and separation of the mechanical functions from the remainder of the Control Building prevent propagation of the water to the control area.

Flooding events that may result from the failure of the fire fighting systems within the Control Building are directed to the basement by the floor drain system. Manual fire fighting in the Control Building with 2 hand held hoses at 0.57m^3/min each (1.14 m^3/min total) ultimately results in the accumulation of water in the basement. The accumulation of water from 1 hour of fire suppression will not affect water sensitive safety-related equipment in the basement which are located at least 400 mm above the floor. Even in the unlikely event that fire suppression activities extend beyond 1 hour there is a substantial period of time before 1 hour before safety-related equipment may be effected. Furthermore, the Division “A” RSW/RCW heat exchanger room in the basement is separated by water tight barriers from the fire water accumulation in the other two divisions and would remain free of water damage and enable the reactor to be shutdown safely.

On all floors, except the basement, water sensitive equipment, outside the control room, will be raised at least 200 mm off the floor to protect them in case of water intrusion due to manual firefighting or other flooding event on their floor. On the basemat the water sensitive components of the RCW pumps will be kept at least 400 mm off the floor for their protection.

The control room area utilizes a raised floor throughout the complex. The outside wall of the control room complex is sealed to prevent water in the corridor from entering the control room subfloor area. Water sources inside the control room are limited to drinking supplies and bathroom facilities. Drains located in the subfloor area will conduct water to the basement. Water seals in the drain lines will ensure the HVAC boundaries are maintained. Fire fighting activities can be accommodated by the large surface area and the floor drains.

There are no sprinkler systems in the Control Building. Hose and standpipes are located in the corridors. Service equipment rooms may build up limited water levels from either service water, cooling water, or chilled water leaks, but elevation differences prevent intrusion of water into control areas. Control room responses to those various levels of flooding may extend from system isolation and correction to reduction of plant load or shutdown, but control room capability is not compromised by any of the postulated flooding events.

3.4.1.1.2.3 Evaluation of Radwaste Building Flooding Events

The Radwaste Building is a reinforced concrete structure consisting of a Seismic Category I substructure 13.5 m below grade at the basemat top and a super-structure 15.7m above grade. This building does not contain safety-related equipment and is not contiguous with other plant structures except through the radwaste piping and tunnel. In case of a flood, the building substructure serves as a large sump which can collect and hold any leakage within the building. Also, the medium and large radwaste tanks are housed in sealed compartments which are designed to contain any spillage or leakage from tanks that may rupture. The piping that transfer the liquid waste from the other buildings to the Radwaste Building traverse through a tunnel which runs near (but does not penetrate) the Radwaste Building at an elevation of 1,500mm, 3m above the basement slab (Table 3.4-1). Seals are provided for all penetrations from the tunnel to prevent building to building flooding.

The structural design of this building is such that no internal flooding is expected or will occur under the worst case conditions from those tanks that are isolated by the Seismic Category I compartments.

Therefore, it can be concluded from the above analysis that there is no uncontrolled path of radioactive liquid from the Radwaste Building under the conditions of worst-case internal flooding.

3.4.1.1.2.4 Evaluation of Service Building Flooding Events

The Service Building is a non-seismic concrete structure consisting of four floors, two above and two below grade. It serves as the main security entrance to the plant and provides the controlled access corridors to the Control Building, the Turbine Building, and the Reactor Building. This building does not house any safety-related equipment.

Some of the connecting corridors to other buildings are below plant grade as indicated in Table 3.4-1. These passage ways are water tight to prevent seepage into the corridors. Also, the controlled access corridors employ watertight doors to guard against water leakage from the Service Building into structures that house safety-related equipment.

The only plant piping that runs through this building are those needed for fire protection, water services, HVAC heaters and chillers, and for draining the sumps. This building has floor drains and two sump pumps (HCW & HSD) for collecting and transferring the liquid waste. Under worst-case conditions, flooding from line ruptures is unlikely and can be contained from spreading to the structures that house safety- related equipment.

3.4.1.1.2.5 Evaluation of Turbine Building Flooding Events

Circulating Water System and Turbine Building Service Water System are the only systems large enough to fill the condenser pit; therefore, only these two systems can flood into adjacent buildings.

A failure in either of these systems will result in the total flooding of the Turbine Building up to grade. Water is prevented from crossing to other buildings by two means. The first is a normally closed alarmed door in the connecting passage between the Turbine Building and Service Building. The second is that the radwaste tunnel will be sealed at both ends to prevent water from either entering the tunnel or leaving the tunnel. A large hydrostatic head is prevented by a large non-water-tight truck door at grade to provide a release point for any water.

Because of the large size of the circulating water system, a leak will fill the condenser pit quickly. Monitors were added in the condenser pit of the Turbine Building to provide leak detection and an automatic means to shutdown the Circulating Water System in the event of flooding in the Turbine Building (Subsections 10.4.5.2.3 and 10.4.5.6).

3.4.1.2 Permanent Dewatering System

There is no permanent dewatering system provided for in the flood design.

3.4.2 Analytical and Test Procedures

Since the deign flood elevation is 30.5 cm below the finished plant grade, there is no dynamic force due to flood. The lateral hydrostatic pressure on the structures due to the design flood water level, as well as ground and soil pressures, are calculated.

Structures, systems, and components in the ABWR Standard Plant are designed and analyzed for the maximum hydrostatic and hydrodynamic forces in accordance with loads and load combinations indicated in Subsections 3.8.4.3 and 3.8.5.3 using well established methods based on the general principles of engineering mechanics. All Seismic Category I structures are in stable condition due to either moment or uplift forces which result from the proper load combinations including the design basis flood.

3.4.3 COL License Information

3.4.3.1 Flood Elevation

The COL applicant will ensure the design basis flood elevation for the ABWR Standard Plant structures will be 30.5 cm below grade (Section 3.4).

3.4.3.2 Ground Water Elevation

The COL applicant will ensure the design basis ground water elevation for the ABWR Standard Plant structures will be 61.0 cm below grade (Section 3.4).

3.4.3.3 Flood Protection Requirements for Other Structures

The COL applicant will demonstrate, for the structures outside the scope of the ABWR Standard Plant, that they meet the requirements of GDC 2 and the guidance of RG 1.102 (Subsection 3.4).

3.4.4 References

3.4-1 Crane Co., “Flow of Fluids Through Valves, Fittings, and Pipe,” Technical Paper No. 410, 1973.

3.4-2 ANSI/ANS 56.11, Standard, “Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants.”

3.4-3 Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants.”

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

Structure	Reactor Building	Service Building	Control Building	Radwaste Building	Turbine Building
Design Flood Level (mm)	11,695	11,695	11,695	11,695	11,695
Design Ground Water Level (mm)	11,390	11,390	11,390	11,390	11,390
Reference Plant Grade (mm)	12,000	12,000	12,000	12,000	12,000
Base Slab (mm)	–8,200	–2150 & 3500	–8,200	–1,500	5,300
Actual Plant Grade (mm)	12,000	12,000	12,000	12,000	12,000
Building Height (mm)	49,700	22,200	22,200	28,000	54,300
Penetrations Below Design Flood Level (Notes 1 through 4)	Refer to Table 6.2-9	None	RCW, RSW and miscellaneous lines, and electrical penetrations	None, except radwaste piping	Radwaste piping
Access Openings Below Design Flood Level (Notes 5 and 6)	Access way from S/B @ 4,800 mm	Area access way from R/B @ 3500 mm,	(Fig. 1.2-18)		
		Area access ways from C/B @ –2150mm, 3500mm, and 7900mm,	Hx area access from S/B @ –2150 mm, (Fig. 1.2-15) Area access from S/B @ 3500 mm, (Fig. 1.2-18) Area access from S/B @ 7900 mm (Fig. 1.2-15)		
		Area access way from T/B @ 3500mm	(Fig. 1.2-24)		Area access from S/B @ 5,300 mm

Notes:

(1) Watertight penetrations will be provided for all Reactor, Control, Turbine and Radwaste Building penetrations that are below grade.

(2) The safety-related and non-safety-related tunnels prevent the lines running through them from being exposed to outside ground flooding.

(3) Penetrations below design flood level will be sealed against any hydrostatic head resulting from the design basis flood, or from a moderate energy pipe failure in the tunnel or inside a connecting building.

(4) Waterproof sealant applied to the building exterior walls below flood level will also be extended a minimum of 150 mm along the penetration surfaces.

(5) Watertight doors (bulkhead type) are provided at all Reactor and Control Building access ways that are below grade.

(6) The figure shown best depicts the indicated accesses.

3.5 Missile Protection

The missile protection design basis for Seismic Category I structures, systems, and components is described in this section. A tabulation of safety-related structures, systems, and components (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 structures, systems, and components 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 have been designed to withstand missile effects, the possible missile loadings, and the procedures to which each barrier has been 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 have been 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 analyzed comply with Criterion 4 of 10CFR50 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 which could cause unacceptable plant consequences or violation of the guidelines of 10CFR100.

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

(1) 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 it is considered not to be statistically significant.

(2) 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).

(3) If the product of (P_1) and (P_2) is less than 10^{-7} per year, the missile is dismissed from further consideration.

(4) 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 x P_2 x P_3 = P_4) is less than 10^{-7} per year, the missile is dismissed.

(5) 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.

Protection of safety-related structures, systems, and components is afforded by one or more of the following practices:

(1) Location of the system or component in an individual missile-proof structure

(2) Physical separation of redundant systems or components of the system for the missile trajectory path or calculated range

(3) Provision of localized protection shields or barriers for systems or components

(4) Design of the particular structure or component to withstand the impact of the most damaging missile

(5) Provision of design features on the potential missile source to prevent missile generation

(6) Orientation of the potential missile source to prevent unacceptable consequences due to missile generation

The following criteria have been adopted to provide an acceptable design basis for the plant's capability to withstand the statistically significant missiles postulated inside the reactor building:

(1) No loss of containment function as a result of missiles generated internal to containment.

(2) Reasonable assurance that a safe plant shutdown condition can be achieved and maintained.

(3) Offsite exposure within the 10CFR100 guidelines for those potential missile damage events resulting in radiation activity release.

(4) The failure of non-safety-related equipment, components, or structures whose failure could result in a missile do not cause the failure of more than one division of safety-related equipment.

(5) No high energy lines are located near the standby-gas treatment charcoal vaults, the offgas charcoal storage vault, or the spent fuel pool.

The systems requiring protection are:

(1) Reactor coolant pressure boundary

(2) Residual Heat Removal System

(3) High Pressure Core Flooder System

(4) Reactor Core Isolation Cooling System

(5) Reactor Building Cooling Water System

(6) Automatic Depressurization System relief valves

(7) Standby diesel generator system

(8) CRD scram system (hydraulic and electrical)

(9) Fuel Pool Cooling and Cleanup System

(10) Remote shutdown panel

(11) Reactor Protection System

(12) All containment isolation valves

(13) HVAC emergency chilled water system

(14) HVAC systems required during operation of items (1) through (12)

(15) Electrical and control systems and wiring required for operation of items (1) through (14)

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

(1) 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.

(2) Components and equipment of the various systems are designed and built to the standards established by the ASME Code or other equivalent industrial standard. A stringent quality control program is also enforced during manufacture, testing, and installation.

(3) Volumetric and ultrasonic testing where required by code coupled with periodic inservice 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)

These missiles are considered to be those missiles resulting internally from plant equipment failures within the ABWR Standard Plant but outside containment.

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, and turbines and, in particular, components in systems normally functioning during power reactor operation, has been examined for any possible source of credible and significant missiles.

3.5.1.1.1.2 RCIC Steam Turbine

The RCIC steam turbine driving the pump is not a credible source of missiles. It is provided with mechanical overspeed protection as well as automatic governing; very extensive industrial and nuclear experience with this model of turbine has never resulted in a missile which penetrated the turbine casing.

3.5.1.1.1.3 Main Steam Turbine

Acceptance Criteria 1 of SRP Section 3.5.1.3 considers a plant with a favorable turbine generator placement and orientation and adhering to the guidelines of Regulatory Guide 1.115 adequately protected against turbine missile hazards. Further, this criterion specifies that exclusions of safety-related structures, systems or components from low trajectory turbine missile strike zones constitutes adequate protection against low trajectory turbine missiles. The turbine generator placement and orientation of the ABWR Standard Plant meets the guidelines of Regulatory Guide 1.115 as illustrated in Figure 3.5-2.

In addition, the COL applicant shall:

(1) Submit for NRC approval, within three years of obtaining an operating license, a turbine system maintenance program including probability calculations of turbine

missile generation based on the NRC approved methodology (such as Reference 3.5-9).

(2) Volumetrically inspect all low pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff.

(3) Meet the minimum requirement for the probability of turbine missile generation given in Table 3.5-1.

See Subsection 3.5.4.5 for COL license information.

3.5.1.1.1.4 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:

(1) The equipment design and manufacturing criteria mentioned previously result in (P_1) being less than 10^{-7} per year.

(2) 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.

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. Since the AC power supply frequency variation is limited to a narrow range, it is not likely they will attain an overspeed condition. At rated speed, if a piece such as a fan blade breaks off, it will not penetrate the casing. The issue of missile generation in rotating machinery is a general safety problem which is not limited to nuclear applications. The designers and manufacturers of these equipment consider this factor as a requirement in their design. Industrial experience and studies conducted on system components indicate that the probability of a missile escaping the casing is very low. GE has also conducted a study on potential missile generation from electrical machines (motors, exciters, generators), flexible couplings and fluid drives. One example where missile generation is significant is in fluid drives where the rotating part and housing diameters are big and the relative thickness of the housing is small, Reference 3.5-1. Based on the results from this study, it was concluded that the potential of a missile being generated and leaving the equipment housing is negligibly small.

3.5.1.1.2 Pressurized Components

3.5.1.1.2.1 Missile Characterization

Potential missiles which could result from the failure of pressurized components are analyzed in this subsection. These potential missiles may be categorized as contained fluid-energy missiles or stored strain-energy (elastic) missiles. These potential missiles have been conservatively evaluated against the design criteria in Subsection 3.5.1.

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 have been analyzed as piston-type missiles, while retaining bolts are examples of stored strain-energy missiles.

3.5.1.1.2.2 Missile Analyses

Pressurized components outside the containment capable of producing missiles have been reviewed. Although piping failures could result in significant dynamic effects if permitted to whip, they do not form missiles as such because the whipping section remains attached to the remainder of the whip. Since Section 3.6 addresses the dynamic effects associated with pipe breaks, 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 ASME Code Section III.

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

(1) Valve bonnets (large and small)

(2) Valve stems

(3) Pressure vessels

(4) Thermowells

(5) Retaining bolts

(6) Blowout panels

These are analyzed as follows:

(1) **Valve Bonnets**—Valves of ANSI 6.2 MPaGrating and above and constructed in accordance with ASME Code Section III 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 3.5-8).

Most valves of ANSI 4.1 MPa rating and below are valves with bolted bonnets. These type valves were analyzed for the safety factors against failure, and, coupled with the low historical incidents of complete severance failure, were determined to not be a potential missile source (Reference 3.5-8).

(2) **Valve Stems**—All the isolation valves installed in the reactor coolant systems have stems with a back seat which eliminates the possibility of ejecting valve stems even if the stem threads fail. Since 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 can be dismissed as a source of missiles.

(3) **Pressure Vessels**—Moderate energy vessels less than 1.9 MPa are not credible missile sources. The pneumatic system air bottles are designed for 17.2 MPa to ASME Code Section III requirements. These bottles are not considered a credible source of missiles for the following qualitative analysis:

(a) The bottles are fabricated from heavy-wall rolled steel.

(b) 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.

(c) The fill connection is protected by a permanent steel collar.

(d) The bottles are strapped in a rack to prevent them from toppling over. The rack is seismically designed to ASME Code Section III, Subsection NF, requirements.

(4) **Thermowells**—Thermowells are welded to socket connections which, in turn, are welded to the wall of the pipe. An analysis of a postulated failure of this weld has been performed. The following expression relates the missile displacement and velocity following the postulated failure:

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

where:

y	=	Distance traveled by the missile from the break (m)
W	=	Missile weight (kg)
A	=	Frontal area of missile (m^2)
u_{∞}	=	Asymptotic velocity of jet (m/s)
υ_{∞}	=	Asymptotic specific volume of jet (m^3/kg)

V = Velocity of missile (m/s)

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 3.5-2. 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:

$$Y^* = \frac{y}{(W/A)} \qquad (3.5\text{-}2)$$

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

$$f^* = \left(\frac{fl}{D}\right)_P \left(\frac{A_E}{Ap}\right)^2 \qquad (3.5\text{-}3)$$

where:

$\left(\frac{fl}{D}\right)_P$ = Equivalent loss coefficient between the broken pressurized component and fluid reservoir, dimensionless

A_E = Area of break, m^2

Ap = Area of pressurized component between break and fluid reservoir, m^2 (assumes $Ap \geq A_E$)

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. Based on ejection by steam at 7.2 MPa, the ejection velocity could reach 61 m/s, which is not sufficient to inflict significant damage to critical systems. (P_4) is, therefore, less than 10^{-7} per year.

(5) **Retaining Bolts**—Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and, thus, are of no concern as potential missiles.

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

(7) **Compartment Shielding Blocks**—Compartment shielding blocks exist in areas within secondary containment. The shielding blocks will be designed for any HELB load present.

3.5.1.1.3 Missile Barriers and Loadings

For local shields and barriers see the response to Question 410.9.

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.

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 becoming potential missiles. All reactor internal pumps are incapable of achieving an overspeed condition and the motors and impellers are incapable of escaping the casing and the reactor vessel wall, respectively.

All drywell cooler fans are designed such that their blades are incapable of leaving the case.

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 of containment. For example, the ADS accumulators are moderate energy vessels and are therefore not considered a credible missile source. One additional item is fine motion control rod drives (FMCRD) 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 have been considered as follows:

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

Non-Seismic Category I items and systems inside containment are considered as Follows:

(1) Cable Tray

All cable trays for both Class 1E and non-Class 1E circuits are seismically supported whether or not a hazard potential is evident.

(2) Conduit and Non-Safety Pipe

Non-Class 1E conduit is seismically supported if it is identified as a potential hazard to safety-related equipment. All ABWR Standard Plant non-safety related piping that is identified as a potential hazard is seismically analyzed per Subsection 3.7.3.13.

(3) Equipment for Maintenance

All other equipment, such as hoists, that is required during maintenance will either be removed prior to 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. See Subsection 3.5.4.6 for COL license information.

3.5.1.3 Turbine Missiles

See Subsection 3.5.1.1.1.3.

3.5.1.4 Missiles Generated by Natural Phenomena

The limiting natural phenomena hazard in the design of all structures required for safe shutdown of the nuclear power plant has been determined to be either design basis tornado-generated missiles or site-specific hurricane-generated missiles. The design basis tornado/hurricane for the ABWR Standard Plant is the maximum windspeed corresponding to a probability of l0E-7 per year. The other characteristics of tornados/hurricanes are summarized in Subsection 3.3.2.1. The design basis tornado/hurricane missiles are per SRP 3.5.1.4, Spectrum I. See Table 2.0-1.

Using the design basis tornado/hurricane and missile spectrum as defined above with the design of the Seismic Category I buildings, compliance with all of the positions of Regulatory Guide 1.117, “Tornado Design Classification,” Positions C.1 and C.2 is assured.

The SGTS charcoal absorber beds are housed in the tornado resistant reactor building and, therefore, are protected from the design basis tornado/hurricane missiles. The offgas system charcoal absorber beds are located deep within the Turbine Building and it is considered very unlikely that these beds could be ruptured as a result of a design basis tornado missile. These features assure compliance with Position C.3 of Regulatory Guide 1.117.

See Subsections 3.5.4.2 and 3.5.4.4 for COL license information requirements.

3.5.1.5 Site Proximity Missiles Except Aircraft

External missiles other than those generated by tornados or hurricanes are not considered as a design basis (i.e. < 10^{-7} per year).

3.5.1.6 Aircraft Hazards

Aircraft hazards are not a design basis event for the ABWR Standard Plant (i.e. ≤ 10^{--7}per year). See Subsection 3.5.4.3 for COL license information requirements.

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

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 be structures, systems, or components, must therefore 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 which are designed as tornado resistant. Since the tornado/hurricane missiles are the design basis missiles, the systems, structures, and components listed are considered to be adequately protected. Provisions are made to protect the charcoal delay tanks against tornado/hurricane missiles.

See Subsection 3.5.4.1 and 3.5.4.7 for COL license information requirements.

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. Structures and barriers that are designed for design basis tornado missile features described in Tier 1 and Tier 2 are also protected from design basis hurricane missiles. For areas where the effects of design basis tornado missiles do not bound site-specific hurricane missiles, the site-specific hurricane-generated missiles need to meet the criteria specified in RG. 1.221 (see COL information in Section 3.5.4.2). The following procedures are in accordance with Section 3.5.3 of NUREG-0800 (Standard Review Plan). Missile protection design features described in Tier 1 and Tier 2, even if indicated as tornado missile protection, also resist hurricane missiles. According to Table 2.0-1, the automobile missile is considered to impact at all altitudes less than 9.14 m (30 feet) above all plant grade levels within 0.8 km (0.5 mile) of the plant structures. For site-specific situations for which automobiles parked within 0.8 km of the plant may be at elevations higher than 9.14 m above plant grade level (e.g., elevated parking garage), a site-specific automobile missile evaluation must be performed by the COL applicant and addressed in the barrier design (see COL information in Section 3.5.4.2).

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. Composite barriers are not utilized in the ABWR Standard Plant for missile protection.

3.5.3.1.1 Concrete Structures and Barriers

Empirical equations, such as the modified Petry formula (Reference 3.5-3) or the TM 5-855-1 formula (Reference 3.5-4), may be used to estimate missile penetration into concrete. The resulting thickness of concrete required to prevent perforation, spalling, or scabbing should in no case be less than those for Region II listed in Table 1 of SRP 3.5.3 for protection against tornado missiles.

3.5.3.1.2 Steel Structure and Barriers

The Stanford equation (Reference 3.5-5) is applied for steel structures and barriers.

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 has been assumed that the impact is plastic with all of the initial momentum of the missile transferred to the structure or barrier and only a portion of the kinetic energy 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 similar to that in Reference 3.5-6 for rigid missiles, and the procedure in Reference 3.5-7 for deformable missiles.

3.5.4 COL License Information

3.5.4.1 Protection of Ultimate Heat Sink

Compliance with Regulatory Guide 1.27 as related to the ultimate heat sink and connecting conduits being capable of withstanding the effects of externally generated missiles shall be demonstrated (Subsection 3.5.2).

3.5.4.2 Missiles Generated by Site-Specific Natural Phenomena

The COL applicant shall identify missiles generated by site-specific natural phenomena that may be more limiting than those considered in the ABWR design and shall provide protection for the structures, systems, and components against such missiles. The COL applicant will provide this information to the NRC (Subsection 3.5.1.4 and Subsection 3.5.3).

3.5.4.3 Site Proximity Missiles and Aircraft Hazards

Analyses shall be provided that demonstrate that the probability of site proximity missiles (including aircraft) impacting the ABWR Standard Plant and causing consequences greater than 10CFR100 exposure guidelines is $\leq 10^{-7}$ per year (Subsection 3.5.1.6).

3.5.4.4 Impact of Failure of Out of ABWR Standard Plant Scope Non-Safety-Related Structures, Systems, and Components Due to a Design Basis Extreme Wind (Tornado and Hurricane)

An evaluation of all out of ABWR Standard Plant Scope non-safety-related structures, systems, and components (not housed in a tornado/hurricane protected structure) whose failure due to a design basis tornado/hurricane missile that could adversely impact the safety function of safety-related systems and components will be provided to the NRC by the COL applicant (Subsection 3.5.1.4).

3.5.4.5 Turbine System Maintenance Program

A turbine system maintenance program, including probability calculations of turbine missile generation meeting the minimum requirement for the probability of missile generation, shall be provided to the NRC (Subsection 3.5.1.1.1.3).

3.5.4.6 Maintenance Equipment Missile Prevention Inside Containment

The COL applicant will provide procedures to ensure that all equipment inside containment, such as hoists, that is required during maintenance will either be removed prior to 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 [Subsection 3.5.1.2.3 (3)].

3.5.4.7 Failure of Structures, Systems, and Components Outside ABWR Standard Plant Scope

Any failure of structures, systems and components outside ABWR Standard Plant scope which may result in external missile generation shall not prevent safety-related structures, systems and components from performing their intended safety function. The COL applicant will provide an evaluation of the adequacy of these designs for external missile protection for NRC review (Subsection 3.5.2).

3.5.5 References

3.5-1 K. Karim-Panahi et. al, “Recirculation MG Set Missile Generation Study”, PED-18-0389, March 1989. (Proprietary).

3.5-2 F. J. Moody, “Prediction of Blowdown Thrust and Jet Forces”, ASME Publication 69-HT-31, August 1969.

3.5-3 A. Amirikan, “Design of Protective Structures”, Bureau of Yards and Docks, Publication No. NAVDOCKS P-51, Department of the Navy, Washington, D.C., August 1960.

3.5-4 US Department of Army, Fundamentals of Protective Design for Conventional Weapons, TM 5-855-1, November 1986.

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

3.5-6 R. A. Williamson and R. R. Alvy, “Impact Effect of Fragments Striking Structural Elements”, Holmes and Narver, Inc., Revised November 1973.

3.5-7 J. D. Riera, “On the Stress Analysis of Structures Subjected to Aircraft Impact Forces”, Nuclear Engineering and Design, North Holland Publishing Co., Vol. 8, l968.

3.5-8 “River Bend Station Updated Safety Analysis Report”, Docket No. 50-458, Volume 6, pp. 3.5-4 and 3.5-5, August 1987.

3.5-9 NUREG-1048, “Safety Evaluation Report Related to the Operation of Hope Creek Generating Station”, Supplement No. 6, July 1986.

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

Criterion	Probability/Yr	Required Licensee Action
(A)	$P_1 < 10^{-4}$	Criterion (A) is the general, minimum 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 COL applicant 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 from the steam supply within 60 days, at which time the COL applicant 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 COL applicant is to meet Criterion (A) before returning the turbine to service.

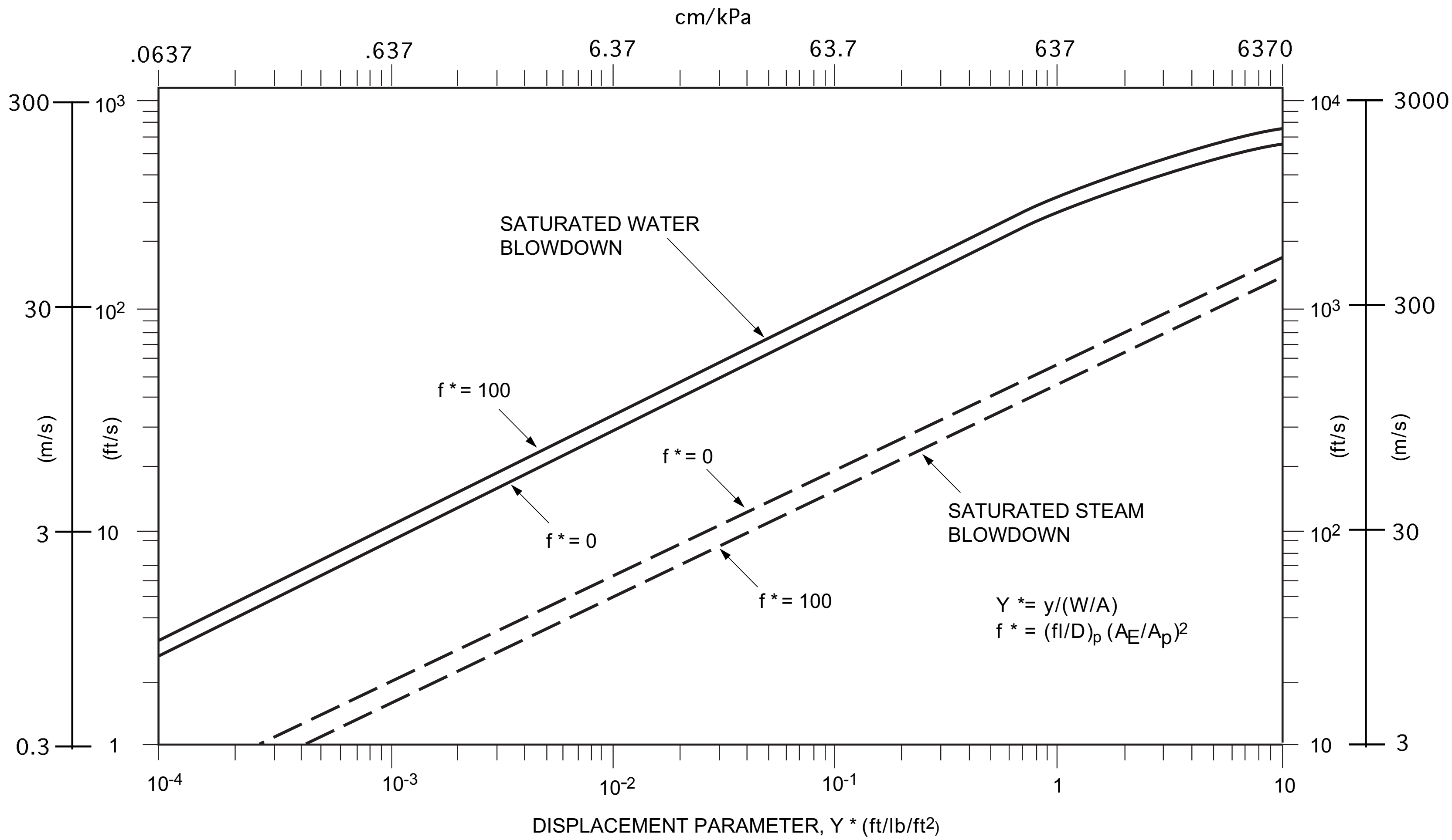
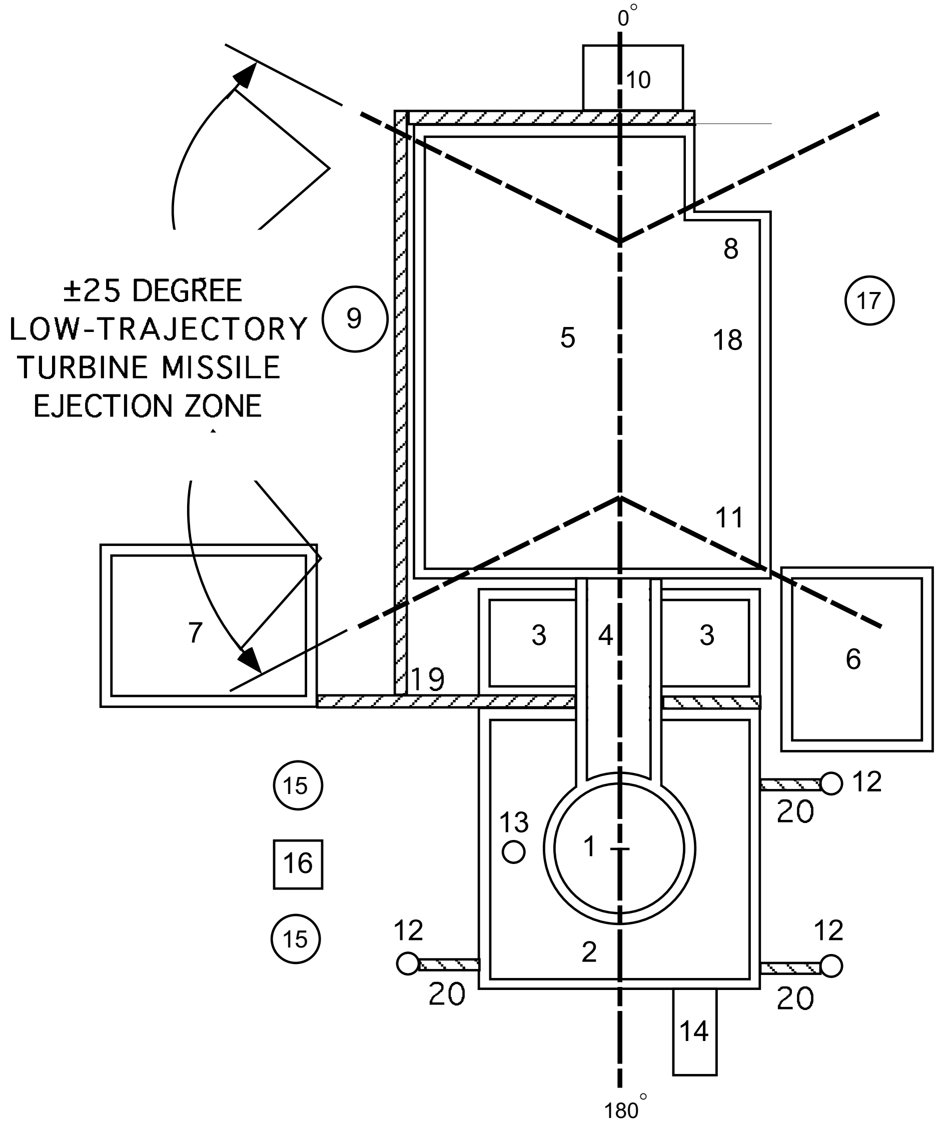


Figure 3.5-1 Missile Velocity and Displacement Characteristics Resulting from Saturated Steam and Water Blowdowns (7.2 MPaA Stagnation Pressure)



No.	FACILITY
1	REACTOR CONTAINMENT
2	REACTOR BUILDING
3	CONTROL BUILDING
4	MAIN STEAM/FEEDWATER TUNNEL
5	TURBINE BUILDING
6	SERVICE BUILDING
7	RADWASTE BUILDING
8	HOUSE BOILER
9	CONDENSATE STORAGE TANK
10	UNIT AUXILIARY TRANSFORMERS
11	NORMAL SWITCHGEAR
12	DIESEL OIL STORAGE TANK (3)
13	STACK
14	EQUIPMENT ENTRY LOCK
15	FIRE PROTECTION WATER STORAGE TANK (2)
16	FIRE PROTECTION PUMPHOUSE
17	BUNKER FUEL TANK
18	COMBUSTION TURBINE GENERATOR
19	RADWASTE TUNNELS RB, CB, TB
20	DG OIL TRANSFER TUNNEL (3)

Figure 3.5-2 ABWR Standard Plant Low-Trajectory Turbine Missile Ejection Zone

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 ABWR Standard Plant.

Subsections 3.6.1 and 3.6.2 describe the design bases and protective measures which ensure that the containment, essential systems, components and equipment, and other essential 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 the 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 throughwall 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 essential 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. There are also seven types of loads identified with a LOCA shown on Table 3.9-2.

Subsection 3.6.3 and Appendix 3E describe the implementation of the leak-before-break (LBB) evaluation procedures as permitted by the broad scope amendment to General Design Criterion 4 (GDC-4) published in Reference 3.6-1. It is anticipated, as mentioned in Subsection 3.6.5.2, that a COL applicant will apply to the NRC for approval of LBB qualification of selected piping by submitting a technical justification report. The approved piping, referred to in Tier 2 as the LBB piping, will be excluded from pipe breaks, which are required to be postulated by Subsections 3.6.1 and 3.6.2, for design against their potential dynamic effects. However, such piping are included in postulation of pipe cracks for their effects as described in Subsections 3.6.1.3.1, 3.6.2.1.5 and 3.6.2.1.6.2. It is emphasized that an LBB qualification submittal is not a mandatory requirement; a COL applicant has an option to select from none to all technically feasible piping systems for the benefits of the LBB approach. The decision may be made based upon a cost-benefit evaluation (Reference 3.6-6).

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside of Containment

This subsection sets forth the design bases, description, and safety evaluation for determining the effects of postulated piping failures in fluid systems both inside and outside the containment, and for including necessary protective measures.

3.6.1.1 Design Bases

3.6.1.1.1 Criteria

Pipe break event protection conforms to 10CFR50 Appendix A, General Design Criterion 4, “Environmental and Missile Design Bases.” The design bases for this protection are in compliance with NRC Branch Technical Positions (BTP) ASB 3-1 and MEB 3-1 included in Subsections 3.6.1 and 3.6.2, respectively, of NUREG-0800 (Standard Review Plan), except for the following:

(1) MEB 3-1, B.1.b(1)(a) Footnote 2 should read, “For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification (excluding earthquake loads).”

(2) MEB 3-1, B.1.b(1)(d) should read, “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 have been specified in the system’s Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) excluding earthquake loads should not exceed 0.8 (1.8 $S_h + S_A$).”

(3) MEB 3-1, B.1.C(1)(b) should read, “At intermediate locations where the maximum stress range as calculated by Equation (10) exceeds 2.4 Sm, and the stress range calculated by either Equation (12) or Equation (13) in Paragraph NB-3653 exceeds 2.4 Sm.”

MEB 3-1 describes an acceptable basis for selecting the design locations and orientations of postulated breaks and cracks in fluid systems piping. Standard Review Plan Sections 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 is 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.

3.6.1.1.2 Objectives

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

(1) Assure that the reactor can be shut down safely and maintained in a safe cold shutdown condition and that the consequences of the postulated piping failure are mitigated to acceptable limits without offsite power.

(2) Assure that containment integrity is maintained.

(3) Assure that the radiological doses of a postulated piping failure remain below the limits of 10CFR100.

3.6.1.1.3 Assumptions

The following assumptions are used to determine the protection requirements:

(1) Pipe break events may occur during normal plant conditions (i.e., reactor startup, operation at power, normal hot standby* or reactor cooldown to cold shutdown conditions but excluding test modes).

(2) 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.

(3) 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 in item (4) below. A SACF is a 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 single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure.

(4) Where the postulated piping failure is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential 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, provided the system is designed to Seismic Category I standards, is powered from both offsite

* Normal hot standby is a normally attained zero power plant operating state (as opposed to a hot standby initiated by a plant upset condition) where both feedwater and main condenser are available and in use.

and onsite sources, and is constructed, operated, and inspected to quality assurance, testing and inservice inspection standards appropriate for nuclear safety-related systems (e.g., the RHR System).

(5) If a pipe break event involves a failure of non-Seismic Category I piping, the pipe break event must not result in failure of essential systems, components, and equipment to shut down the reactor and mitigate the consequences of the pipe break event considering a SACF in accordance with items (3) and (4) above.

(6) If loss of offsite power is a direct consequence of the pipe break event (e.g., trip of the turbine-generator producing a power surge, which, in turn, trips the main breaker), then a loss of offsite power occurs in a mechanistic time sequence with a SACF. Otherwise, offsite power is assumed available with a SACF.

(7) Pipe whip shall be considered capable of causing circumferential and longitudinal breaks, individually, in impacted pipes of smaller nominal pipe size, irrespective of pipe wall thickness, and developing through-wall cracks in equal or larger nominal pipe sizes with equal or thinner wall thickness. Analytical or experimental data, or both, for the expected range of impact energies may be used to demonstrate the capability to withstand the impact without rupture; however, loss of function due to reduced flow in the impacted pipe should be considered.

(8) All available systems, including those actuated by operator actions, are available to mitigate the consequences of a postulated piping failure. In judging the availability of systems, account is taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, 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.

(9) Although a pipe break event outside the containment may require a cold shutdown, up to eight hours in hot standby (within RCIC capability) is allowed in order for plant personnel to assess the situation and make repairs.

(10) Pipe whip, with rapid motion of a 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 and rotating provided its movement can be defined and evaluated.

(11) 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.

(12) All walls, doors, floors, and penetrations which serve as divisional boundaries will be designed to withstand the worst case pressurizations associated with the postulated pipe failures inside primary containment.

All structural divisional separation walls will maintain their structural integrity after a postulated failure outside primary containment and within secondary containment. Divisional separation doors, penetration and floors are not required to maintain their structural integrity. Justification for divisional separation integrity is addressed in Subsections 3.4.1, 3.13 and 9.5.1.

3.6.1.1.4 Approach

To comply with the objectives previously described, the essential systems, components, and equipment are identified. The essential 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

High-energy lines are defined in Subsection 3.6.2.1.1 are listed in Table 3.6-3 for inside the containment and in Table 3.6-4 for outside the containment. Moderate-energy lines are defined in Subsection 3.6.2.1.2 and are defined in Table 3.6-5 for inside and Table 3.6-6 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 for primary containment subcompartments.

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

The control room is protected from high-energy line breaks. As such, there are no effects upon the habitability of the control room by a piping failure in the control building 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 Safety Evaluation

3.6.1.3.1 General

An analysis of pipe break events is performed to identify those essential 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 essential 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.

3.6.1.3.2 Protection Methods

3.6.1.3.2.1 General

The direct effects associated with a particular postulated break or crack must be mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties (in accordance with SRP 3.6.2), and equipment arrangements are considered in defining the following specific measure for protection against actual pipe movement and other associated consequences of postulated failures:

(1) 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.

(2) The precise method chosen depends largely upon limitations placed on the designer such as accessibility, maintenance, and proximity to other pipes.

3.6.1.3.2.2 Separation

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

However, due to the complexities of several divisions being adjacent to high-energy lines in the drywell and Reactor Building steam tunnel, specific break locations are determined in accordance with Subsection 3.6.2.1.4.3 for possible spatial separation. Care is taken to avoid concentrating essential equipment in the break exclusion zone allowed per Subsection 3.6.2.1.4.2. If spatial separation requirements (distance and/or arrangement described below) cannot be met based on the postulation of specific breaks, then barriers, enclosures, shields, or restraints are provided. These methods of protection are discussed in Subsections 3.6.1.3.2.3 and 3.6.1.3.2.4.

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:

(1) 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.

(2) Essential systems, components, and equipment at a distance greater than 9.14m from any high-energy piping are considered as meeting spatial separation requirements. No damage is assumed to occur due to jet impingement since the impingement, force becomes negligible beyond 9.14m. Likewise, a 9.14m evaluation zone is established for pipe breaks to assure protection against potential damage from a whipping pipe. Assurance that 9.14m represents the maximum free length is made in the piping layout.

(3) Essential systems, components, and equipment at a distance less than 9.14m from any high-energy piping are evaluated to see if damage could occur to more than one essential 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.

(4) If damage could occur to more than one division of a redundant essential system within 9.14m of any high-energy piping, then other protection measures in the form of barriers, shields, or enclosures (Subsection 3.6.1.3.2.3) or pipe whip restraints (Subsection 3.6.1.3.2.4) are used (see Subsection 3.6.5.1, Item (8) for COL license information requirements), as described above.

3.6.1.3.2.3 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 due to 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 steam tunnel is made of reinforced concrete 2m thick. A steam tunnel subcompartment analysis was performed for the postulated rupture of a main steamline and for a feedwater line (Subsection 6.2.3.3.1.3). The calculated peak pressure from a main steamline break was found to be 58.84 kPaG. The calculated peak pressure from a feedwater line break was found to be 26.48 kPaG. The steam tunnel is designed for the effects of an SSE coincident with HELB inside the steam tunnel. Under this conservative load combination, no failure in any portion of the steam tunnel was found to occur; therefore, a HELB inside the steam tunnel will not effect control room habitability.

The MSIVs and the feedwater isolation and check valves located inside the tunnel shall be 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 will be provided by the COL applicant (Subsection 3.6.5.1, Items 4 and 6).

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.

3.6.1.3.2.4 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 Subsections 3.6.2.1.4.3 and 3.6.2.1.4.4. After the restraints are located, the piping and essential 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 3.6.2.3.3.

3.6.1.3.3 Specific Protection Measures

(1) Nonessential 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 nonessential system or component could initiate or escalate the pipe break event in an essential system or component, or in another nonessential system whose failure could affect an essential system.

(2) For high-energy piping systems penetrating through the containment, isolation valves are located as close to the containment as possible.

(3) The pressure, water level, and flow sensor instrumentation for those essential systems, which are required to function following a pipe rupture, are protected.

(4) 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.

(5) For any postulated pipe rupture, the structural integrity of the containment structure 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.

(6) Safety/relief valves (SRVs) and the Reactor Core Isolation Cooling (RCIC) System steamline are located and restrained so that a pipe failure would not prevent depressurization.

(7) Separation is provided to preserve the independence of the low-pressure flooder (LPFL) systems.

(8) Protection for the FMCRD scram insert lines is not required since the motor operation of the FMCRD can adequately insert the control rods even with a complete loss of insert lines (Subsection 3.6.2.1.6.1).

(9) The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture do not preclude:

    (a) Accessibility to any areas required to cope with the postulated pipe rupture

    (b) Habitability of the control room

    (c) The ability of essential 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 is presented in this subsection and is supplemented in Appendix 3L. The location criteria and methods of analysis are 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 essential structures, systems, and components defined in the introduction of Section 3.6.

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.

3.6.2.1.1 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 3.6.1.1.3 (1)), are either in operation or are maintained pressurized under conditions where either or both of the following are met:

(1) Maximum operating temperature exceeds 93°C.

(2) Maximum operating pressure exceeds 1902.5 kPaG.

Table 3.6-3 lists the high-energy lines inside containment, and Table 3.6-4 lists the high-energy lines outside the containment.

3.6.2.1.2 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 3.6.1.1.3 (1)), are either in operation or are maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

(1) Maximum operating temperature is 93°C or less.

(2) Maximum operating pressure is 1902.5 kPaG 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. Table 3.6-5 lists the moderate-energy lines inside containment, and Table 3.6-6 lists the moderate-energy lines outside containment.

3.6.2.1.3 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 a moderate-energy fluid system, 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:

(1) All piping which is part of the reactor coolant pressure boundary and subject to reactor pressure continuously during station operation.

(2) All piping which is beyond the second isolation valve but subject to reactor pressure continuously during station operation.

(3) 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 a normally closed valve. Pump and valve bodies are also exempted from consideration of pipe break because of their greater wall thickness.

3.6.2.1.4 Locations of Postulated Pipe Breaks

Postulated pipe break locations are selected as follows:

3.6.2.1.4.1 Piping Meeting Separation Requirements

[*Based on the HELSA evaluation described in Subsection 3.6.1.3.2.2, 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 per the following subsections if special protection devices such as barriers or restraints are provided.*]*

3.6.2.1.4.2 Piping in Containment Penetration Areas

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

(1) The following design stress and fatigue limits of (a) through (e) are not exceeded. When meeting the limits of (a) and (d), earthquake loads are excluded (Subsection 3.6.1.1.1).

* See Subsection 3.9.1.7.

For ASME Code Section III, Class 1 Piping

(a) *The maximum stress range between any two loads sets (including the zero load set) does not exceed 2.4 S_m, and is calculated** *by Eq. (10) in NB-3653, ASME Code, Section III.*

If the calculated maximum stress range of Eq. (10) exceeds 2.4 S_m, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 meet the limit of 2.4 S_m.

(b) The cumulative usage factor is less than 0.1.

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

For ASME Code Section III, Class 2 Piping

(d) 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 ASME Code Section III.

(e) The maximum stress, as calculated by Eq. (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 which are deflection limited by whip restraints. The exceptions permitted in (c) 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 ANSI 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.

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* For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification.

(2) *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 limits of Item (1).*

(3) *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 inservice volumetric examination of longitudinal and circumferential welds.*

(4) *The length of these portions of piping is reduced to the minimum length practical.*

(5) *The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) does 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 limits of Item (1).*

(6) *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 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:*

  (a) *The design pressure and temperature are not less than the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions.*

  (b) *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.*

  (c) *The assemblies are subjected to a single pressure test at a pressure not less than its design pressure.*

  (d) *The assemblies do not prevent the access required to conduct the inservice examination specified in Item (7).*

(7) *A 100% volumetric inservice examination of all pipe welds would be conducted during each inspection interval as defined in IWA-2400, ASME Code Section XI.*]*

See COL license information requirements in Subsection 3.6.5.3.

* See Subsection 3.9.1.7.

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

[*With the exception of those portions of piping identified in Subsection 3.6.2.1.4.2, breaks in ASME Code Section III Class 1 Piping are postulated at the locations identified in (1), (2), and (3) in each piping and branch run. Earthquake loads are excluded from (2).*

(1) *At terminal ends**.

(2) At intermediate locations where the maximum stress range as calculated by Eq. (10) exceeds 2.4 Sm, and

The stress range calculated by either Eq. (12) or Eq. (13) in Paragraph NB-3653 exceeds 2.4 Sm.

(3) At intermediate locations where the cumulative usage factor exceeds 0.1.

As a result of piping re-analysis due to 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:

(a) The dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

(b) A change is required in pipe parameters such as major differences in pipe size, wall thickness, and routing.]†

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

[*With the exception of those portions of piping identified in Subsection 3.6.2.1.4.2, 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:*

(1) At terminal ends (Subsection 3.6.2.1.4.3, Paragraph (1)).

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* Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior. In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve.

† See Subsection 3.9.1.7.

(2) *At intermediate locations selected by one of the criteria below. Earthquake loads are excluded from criteria (b).*

(a) *At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.*

(b) *At each location where stresses calculated (Subsection 3.6.2.1.4.2, Paragraph (1) (d)) by the sum of Eqs. (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.*

As a result of piping re-analysis due to 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.]*

3.6.2.1.4.5 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 3.7.3.13.

3.6.2.1.4.6 Separating Structure with High-Energy Lines

If a structure separates a high-energy line from an essential 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 3.6.1.3.2.3, some structures that are identified as necessary by the HELSA evaluation (i.e., based on no specific break locations) are designed for worst-case loads.

3.6.2.1.5 Locations of Postulated Pipe Cracks

Postulated pipe crack locations are selected as follows:

3.6.2.1.5.1 Piping Meeting Separation Requirements

Based on the HELSA evaluation described in Subsection 3.6.1.3.2.2, 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 essential structures, systems and components.

3.6.2.1.5.2 High-Energy Piping

[*With the exception of those portions of piping identified in Subsection 3.6.2.1.4.2, leakage cracks are postulated for the most severe environmental effects as follows: {Earthquake loads are excluded from criteria (1) and (2)}*

(1) For ASME Code Section III, Class 1 piping, at axial locations where the calculated stress range [Subsection 3.6.2.1.4.2, Paragraph (1)(a)] by Equation (10) in NB-3653 exceeds 1.2 S_m.

(2) For ASME Code Section III Class 2 and 3 or non-ASME class piping, at axial locations where the calculated stress [Subsection 3.6.2.1.4.4, Paragraph (2)(b)] 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.

(3) *Non-ASME class piping which has not been evaluated to obtain stress information have leakage cracks postulated at axial locations that produce the most severe environmental effects.*]*

3.6.2.1.5.3 Moderate-Energy Piping

3.6.2.1.5.3.1 Piping in Containment Penetration Areas

[*Leakage cracks are not postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves, provided they meet the requirements of ASME Code Section III, NE-1120, and the stresses calculated (Subsection 3.6.2.1.4.4, Paragraph (2)(b)) by the sum of Equations (9) and (10) in ASME Code Section III, NC-3653 do not exceed 0.4 times the sum of the stress limits given in NC-3653.*]†

3.6.2.1.5.3.2 Piping in Areas Other Than Containment Penetration

(1) [Leakage cracks are postulated in piping located adjacent to essential structures, systems or components, except for (a), (b), and (c) below. Earthquake loads are excluded from the stress criteria of (a), (b), and (c).

(a) Where exempted by Subsections 3.6.2.1.5.3.1 and 3.6.2.1.5.4.

(b) For ASME Code Section III, Class 1 piping the stress range calculated by Eq. (10) in NB-3653 is less than 1.2 S_m.

(c) For ASME Code Section III Class 2 or 3 and non-ASME class piping, the stresses calculated [Subsection 3.6.2.1.4.4, Paragraph (2) (b)] by the sum of

* See Subsection 3.9.1.7.

† See Subsection 3.9.1.7.

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.

(2) *Leakage cracks, unless the piping system is exempted by Item (1) above, are postulated at axial and circumferential locations that result in the most severe environmental consequences.*

(3) *Leakage cracks are postulated in fluid system piping designed to nonseismic standards as necessary to meet the environmental protection requirements of Subsection 3.6.1.1.3.*]*

3.6.2.1.5.4 Moderate-Energy Piping in Proximity to High-Energy Piping

Moderate-energy fluid system piping or portions thereof that 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 throughwall 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 Subsection 3.6.2.1.5.3 are applied.

3.6.2.1.6 Types of Breaks and Cracks to be Postulated

3.6.2.1.6.1 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 3.6.2.1.4.

(1) No breaks are postulated in piping having a nominal diameter less than or equal to 25A. Instrument lines 25A and less nominal pipe or tubing size meet the provision of RG 1.11 (Table 3.2-1). Additionally, the 32A hydraulic control unit fast scram lines do not require special protection measure because of the following reasons:

   (a) The piping to the control rod drives from the hydraulic control units (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 on other HCU lines. As discussed in Subsection 3.6.1.1.3, Paragraph (7), a whipping pipe will only rupture an impacted pipe of smaller nominal pipe size or cause a throughwall crack in the same nominal pipe size but with thinner wall thickness.

   (b) The total amount of energy contained in the 32A nominal pipe size piping between normally closed scram insert valve on the HCU module and the ball-check valve in the control rod housing is small. In the event of a rupture of this line, the ball-check valve will close to prevent reactor vessel flow out of the break.

(c) Even if a number of the HCU lines ruptured, the control rod insertion function would not be impaired, since the electrical motor of the FMCRD would drive in the control rods.

(2) Longitudinal breaks are postulated only in piping having a nominal diameter equal to or greater than 100A.

(3) Circumferential breaks are only assumed at all terminal ends.

(4) At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Subsections 3.6.2.1.4.3 and 3.6.2.1.4.4, 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.

(5) 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.

(6) 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).

(7) 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.

(8) 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).

(9) 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.

3.6.2.1.6.2 Pipe Cracks

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

(1) Cracks are postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 25A.

(2) At axial locations determined per Subsection 3.6.2.1.5, the postulated cracks are oriented circumferentially to result in the most severe environmental consequences.

(3) 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.

(4) 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

3.6.2.2.1 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 which can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon the fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors.

The thrust time-histories acting at the break location and on the segments of the ruptured piping system shall be defined according to the following:

(1) Pipe segment forces are defined by the generalized equations in Paragraph 6.2 and Appendix A of ANS 58.2.

(2) Pipe segment forces are further defined according to the methods and procedures in “The Thermal-Hydraulics of a Boiling Water Nuclear Reactor”, by R.T. Lahey, Jr. and F.J. Moody. (Reference 3.6-7)

(3) Thrust forces acting at the rupture point are determined according to the simplified methods contained in Appendix B of ANS 58.2, and are assumed to occur at 102%.

When the pipe rupture analysis requires a complete-system dynamic analysis, as defined in Paragraph 6.3.1 of ANS 58.2, the pipe segment time-histories are calculated by a computer program such as MS-BRK in compliance with (1) and (2) above.

All thrust time-history calculations shall be based on the postulated rupture descriptions contained in Paragraph 4.2 of ANS 58.2.

When the pipe rupture analysis is performed by a simplified analysis with a portion of the pipe system, as defined in Paragraph 6.3.2 of ANS 58.2, the thrust time-histories acting at the break locations may be calculated manually in compliance with (3) above.

3.6.2.2.2 Pipe Whip Dynamic Response Analyses

An analysis shall be conducted of the postulated ruptured piping and pipe whip restraint system response to the fluid dynamic forces specified in Subsection 3.6.2.2.1 in accordance with the requirements of Paragraph 6.3 of ANS 58.2. The analysis shall be in sufficient detail to evaluate the potential for pipe whip, determine potential jet impingement targets, establish the pipe whip restraint and associated structural loads and demonstrate that following dynamic event the system would be capable of supporting fluid forces at steady state flow conditions.

The alternative analytical approaches described in Paragraphs 6.3.1 through 6.3.5 of ANS 58.2 are acceptable approaches for piping response calculation. Criteria for an acceptable design are:

(1) The piping stresses between the isolation valves are within the allowable limits specified in SRP 3.6.2 and BTP MEB 3-1, Paragraph B.1.b.

(2) The pipe whip restraint loads and displacements due to postulated break are within the allowable limits specified in SRP 3.6.2.

(3) Calculated loads or stresses for safety-related valves or equipment to which the ruptured piping is attached do not exceed the operability limits specified in Subsection 3.9.3.

Appendix 3L provides an acceptable procedure for evaluation of the piping-pipe whip restraint system due to the dynamic effect of fluid forces resulting from postulated pipe ruptures. The procedure in Appendix 3L covers the analytical approach for (1) a complete system dynamic analysis as defined in Paragraph 6.3.1 of ANS 58.2 using the ANSYS computer program, and (2) a simplified dynamic analysis as defined in Paragraph 6.3.2 of ANS 58.2 using the PDA computer program.

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 methods used to evaluate the jet effects resulting from the postulated breaks of high-energy piping are described in Appendices C and D of ANS 58.2 and presented in this subsection.

[*The criteria used for evaluating the effects of fluid jets on essential structures, systems, and components are as follows:*

(1) *Essential structures, systems, and components are not impaired so as to preclude essential functions. For any given postulated pipe break and consequent jet, those essential structures, systems, and components needed to safely shut down the plant are identified.*

(2) *Essential structures, systems, and components which are not necessary to safely shut down the plant for a given break are not protected from the consequences of the fluid jet.*

(3) *Safe shutdown of the plant due to 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.*

(4) *Offsite dose limits specified in 10CFR100 are complied with.*

(5) *Postulated breaks resulting in jet impingement loads are assumed to occur in high-energy lines at full (102%) power operation of the plant.*

(6) *Throughwall leakage cracks are postulated in moderate-energy lines and are assumed to result in wetting and spraying of essential structures, systems, and components.*

(7) *Reflected jets are considered only when there is an obvious reflecting surface (such as a flat plate) which directs the jet onto an essential equipment. Only the first reflection is considered in evaluating potential targets.*

(8) Potential targets in the jet path are considered at the calculated final position of the broken end of the ruptured pipe. This selection of potential targets is considered adequate due to the large number of breaks analyzed and the protection provided from the effects of these postulated breaks.

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

(1) The direction of the fluid jet is based on the arrested position of the pipe during steady-state blowdown.

(2) The impinging jet proceeds along a straight path.

(3) 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 3.6.2.2.1 and with jet characteristics shown in Figure 3.6-1.

(4) The jet impingement force is uniformly distributed across the cross-sectional area of the jet and only the portion intercepted by the target is considered.

(5) The break opening is assumed to be a circular orifice of cross-sectional flow area equal to the effective flow area of the break.

(6) The jet impingement force is equal to the steady-state value of the fluid blowdown force calculated by the methods described in Subsection 3.6.2.2.1.

(7) The distance of jet travel is divided into two or three regions. Region 1 (Figure 3.6-1) 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° (Figures 3.6-1).

(8) 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 which 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.

(9) The distance downstream from the break where the asymptotic area is reached (Region 3, Figure 3.6-1) is calculated for circumferential and longitudinal breaks.

(10) Both longitudinal and fully separated circumferential breaks are treated similarly. The value of friction loss used in the blowdown calculation is used for jet impingement also.

(11) 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:

(a) The jet is uniformly distributed around the periphery.

(b) The jet cross section at any cut through the pipe axis has the configuration depicted in Figure 3.6-1b and the jet regions are as therein delineated.

(c) The jet force F_j = total blowdown F.

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

$$P_j = \frac{F_j}{A_j} \tag{3.6-1}$$

where:

A_j = The cylindrical surface area of the jet at a radius equal to the distance from the centerline to the target, calculated in accordance with ANS-58.2, Appendix C.

(e) The pressure of the jet is then multiplied by the area of the target submerged within the jet.

(12) Target loads are determined using the following procedures:

(a) For both the fully separated circumferential break and the longitudinal break, the jet is studied by determining target locations vs. asymptotic distance and the target shape factor and load are calculated in accordance with ANS-58.2, Appendices C and D.

(b) For circumferential break with limited separation, the jet is analyzed by using the equations of ANS 58.2, Appendices C and D and determining respective target and asymptotic locations

(c) After determination of the total area of the jet at the target, the jet pressure 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 ANS-58.2. If the effective target area (A_{te}) *is less than the expanded jet area* ($A_te \le A_x$)*, 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. *The effective target area* (A_{te}) *for various geometries follows.*

(1) *Flat Surface—For a case where a target with physical area* A_t *is oriented at angle* θ *with respect to the jet axis and with no flow reversal, the effective target area* A_{te} *equals* A_t.

(2) 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. The effective target area A_{te} *is:*

$$A_{te} = (D_A)\,(D) \qquad (3.6\text{-}3)$$

where

D_A = *Diameter of the jet at the target interface, and*

D = *Pipe OD 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

$$A_{te} = A_x \qquad (3.6\text{-}4)$$

(3) 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}.]*

* See Subsection 3.9.1.7.

3.6.2.3.2 Pipe Whip Effects on Essential Components

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

Pipe whip (displacement) effects on essential 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, and conduits, etc.

3.6.2.3.2.1 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:

(1) [*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 Code Section III-imposed limits for essential components under faulted loading.*

(2) *If these components are required for safe shutdown or serve to protect the structural integrity of an essential component, limits to meet the ASME Code requirements for faulted conditions and limits to ensure required operability are met.*]*

The methods used to calculate the pipe whip loads on piping components in the same run as the postulated break are described in Section 3.6.2.2.2.

3.6.2.3.2.2 Pipe Displacement Effects on Essential Structures, Other Systems, and Components

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

(1) The effects on essential structures and barriers are evaluated in accordance with the barrier design procedures given in Subsection 3.5.3.

(2) 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.

* See Subsection 3.9.1.7.

(3) *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.*

(4) *Damage of unrestrained whipping pipe on essential structures, components, and systems other than the ruptured one is prevented by either separating high-energy systems from the essential systems or providing pipe whip restraints.*]*

3.6.2.3.3 Design Criteria and Load Combinations for Pipe Whip Restraint

The loading combinations and design criteria for pipe whip restraints are dependent on the type of restraint and the function it performs. Some restraints in the ABWR are designed to perform a dual function of supporting the pipe during operating conditions and also of controlling the motion of the pipe following a postulated rupture. However, most pipe whip restraints in the ABWR are single purpose restraints designed to control the motion of a broken pipe.

Figure 3.6-3 illustrates some acceptable pipe whip restraint designs. These designs include:

(1) **The U-bar restraint**—This is a single purpose, energy absorbing restraint designed for once-in-a-lifetime loading. The gap between the pipe and the restraint is relatively large to permit free thermal expansion of the pipe and does not provide support to maintain structural integrity of the pipe during any of the plant operating conditions. Most of the restraints used in the ABWR plant on large ASME Class 1 piping are U-bar restraints with stainless steel U-bars. This restraint is further defined in this Subsection and serves as the basis for the Appendix 3L procedure for evaluation of postulated ruptures in high-energy pipes. Although piping integrity does not depend on this single purpose pipe whip restraint, the restraint shall be designed to remain functional following an earthquake up to and including the SSE (Subsection 3.2.1). This pipe whip restraint is further illustrated in Figure 3.6-2.

(2) **Restraints with Crushable Material**—Pipe whip restraints with crushable material have the same design basis as the U-bar restraint. It is a single purpose, energy absorbing restraint with sufficient gap between the pipe and the restraint to allow free thermal expansion of the pipe. Restraints with crushable pads may not have lateral load capability so they must be provided in every direction in which the jet thrust from the ruptured pipe may occur. Figure 3.6-3 illustrates several acceptable pipe whip restraint designs using crushable material: the crushable ring, the honeycomb restraint, and the frame with a series of crushable rings.

(3) **Rigid Restraints**—Rigid pipe whip restraints are dual purpose, essentially elastic restraints that take the form of seismic guides, struts, and structural frames. Since rigid restraints are attached to the pipe or are separated from the pipe by very small gaps, they carry loads caused by thermal expansion, dead weight, seismic and other dynamic events during plant operation. Rigid restraints therefore serve a pressure

integrity function and are considered as pipe supports that must meet the requirements of ASME III, Subsection NF. They are modeled as spring elements in the static and dynamic analysis of the piping. Following a postulated pipe rupture these restraints carry the load from the jet thrust and control motion of a broken pipe. These restraints are designed to stop the pipe without exceeding ASME III, Subsection NF, Level D limits. The seismic guide provided on the main steam and feedwater pipe serves as a rigid pipe whip restraint performing a dual function.

The specific design objectives of pipe whip restraints are:

(1) Single purpose restraints shall in no way increase the reactor coolant pressure boundary stresses by their presence during any normal mode of reactor operation.

(2) The restraint system shall function to stop the movement of a ruptured pipe without allowing damage to critical components or missile development.

(3) The restraints should permit inservice inspection of the process piping.

For the purpose of design, the pipe whip restraints are designed for the following dynamic loads:

(1) Blowdown thrust of the pipe section that impacts the restraint.

(2) Dynamic inertia loads of the moving pipe section which is accelerated by the blowdown thrust and subsequent impact of the restraint.

(3) Non-linear design characteristics of the pipe whip restraints are included and verified by the pipe whip dynamic analysis described in Subsection 3.6.2.2.2 and Appendix 3L.

(4) Since single purpose pipe whip restraints are not contacted during normal plant operation, the postulated pipe rupture event is the only design loading condition.

(5) For unruptured pipe, dual purpose pipe whip restraints act as ASME III, Subsection NF pipe supports and must meet the Code requirements for service loads and load combinations for unruptured pipe specified in the design specification and summarized in Table 3.9-2. Following postulated pipe rupture, the restraint stress must not exceed ASME III, Subsection NF, Level D limits for pipe rupture loads acting in combination with loading for which service Level A limits are specified.

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:

(1) 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,

(2) 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,

(3) Representative or actual test data values are used in the design of components and structures including justifiable elevated strain rate-affected stress limits in excess of 10%, or

(4) 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 ABWR primary containment does not require guard pipes.

3.6.2.5 Material to be Supplied for the Operating License Review

See Subsection 3.6.5.1 for COL license information requirements.

3.6.3 Leak-Before-Break Evaluation Procedures

Per Regulatory Guide 1.70, the Safety Analysis Section 3.6 has traditionally addressed the protection measures against dynamic effects associated with the non-mechanistic or postulated ruptures of piping. The dynamic effects are defined in introduction to Section 3.6. Three forms of piping failure (full flow area circumferential and longitudinal breaks, and throughwall leakage crack) are postulated in accordance with Subsection 3.6.2 and Branch Technical Position MEB 3-1 of NUREG-0800 (Standard Review Plan) for their dynamic as well as environmental effects.

However, in accordance with the modified General Design Criterion 4 (GDC-4), effective November 27, 1987 (Reference 3.6-1), the mechanistic leak-before-break (LBB) approach, justified by appropriate fracture mechanics techniques, is recognized as an acceptable procedure under certain conditions to exclude design against the dynamic effects from postulation of breaks in high-energy piping. The LBB approach is not used to exclude postulation of cracks and associated effects as required by Subsections 3.6.2.1.5 and 3.6.2.1.6.2. It is anticipated, as mentioned by Subsections 3.6.5.2, that a COL applicant will apply to the NRC for approval of LBB qualification of selected piping. This approved piping, referred to in Tier 2 as the LBB-qualified piping, will be excluded from pipe breaks, which are required to be postulated by Subsections 3.6.1 and 3.6.2, for design against their potential dynamic effects.

The following subsections describe (1) certain design bases where the LBB approach is not recognized by the NRC as applicable for exclusion of pipe breaks, and (2) certain conditions which limit the LBB applicability. Appendix 3E provides guidelines for LBB applications describing in detail the following necessary elements of an LBB report to be submitted by a COL applicant for NRC approval: fracture mechanics methods, leak rate prediction methods, leak detection capabilities and typical special considerations for LBB applicability. Also included in Appendix 3E is a list of candidate piping systems for LBB qualification. The LBB application approach described in this subsection and Appendix 3E is consistent with that documented in Draft SRP 3.6.3 (Reference 3.6-4) and NUREG-1061 (Reference 3.6-5). See Subsection 3.6.5.2 for COL license information requirements.

3.6.3.1 Scope of LBB Applicability

The LBB approach is not used to replace existing regulations or criteria pertaining to the design bases of emergency core cooling system (Section 6.3), containment system (Section 6.2) or environmental qualification (Section 3.11). However, consistent with modified GDC-4, the design bases dynamic qualification of mechanical and electrical equipment (Section 3.10) may exclude the dynamic load or vibration effects resulting from postulation of breaks in the LBB-qualified piping. This is also reflected in a note to Table 3.9-2 for ASME components. The LBB-qualified piping may not be excluded from the design bases for environmental qualification unless the regulation permits it at the time of LBB qualification. For clarification, it is noted that the LBB approach is not used to relax the design requirements of the primary containment system that includes the primary containment vessel (PCV), vent systems (vertical flow channels and horizontal vent discharges), drywell zones, suppression chamber (wetwell), vacuum breakers, PCV penetrations, and drywell head.

3.6.3.2 Conditions for LBB Applicability

The LBB approach is not applicable to piping systems where operating experience has indicated particular susceptibility to failure from the effects of intergranular stress corrosion cracking (IGSCC), water hammer, thermal fatigue, or erosion. Necessary preventive or mitigation measures are used and necessary analyses are performed, as discussed below, to avoid concerns for these effects. Other concerns, such as creep, brittle cleavage type failure, potential indirect source of pipe failure, and deviation of as-built piping configuration, are also addressed.

(1) Degradation by erosion, erosion/corrosion and erosion/cavitation due to unfavorable flow conditions and water chemistry is examined. The evaluation is based on the industry experience and guidelines. Additionally, fabrication wall thinning of elbows and other fittings is considered in the purchase specification to assure that the code

minimum wall requirements are met. These evaluations demonstrate that these mechanisms are not potential sources of pipe rupture

(2) The ABWR plant design involves operation below 371°C in ferritic steel piping and below 427°C in austenitic steel piping. This assures that creep and creep-fatigue are not potential sources of pipe rupture.

(3) The design also assures that the piping material is not susceptible to brittle cleavage-type failure over the full range of system operating temperatures (that is, the material is on the upper shelf).

(4) The ABWR plant design specifies use of austenitic stainless steel piping made of material (e.g., nuclear grade or low carbon type) that is recognized as resistant to IGSCC. The major high-energy piping in the primary and secondary containments is made of carbon steel or ferritic steel, except for the austenitic stainless reactor water cleanup piping in the primary containment.

(5) A systems evaluation of potential waterhammer is made to assure that pipe rupture due to this mechanism is unlikely. Waterhammer is a generic termincluding various unanticipated high frequency hydrodynamic events such as steam hammer and water slugging. To demonstrate that waterhammer is not a significant contributor to pipe rupture, reliance on historical frequency of waterhammer events in specific piping systems, coupled with a review of operating procedures and conditions, is used for this evaluation. The ABWR design includes features such as vacuum breakers and jockey pumps coupled with improved operational procedures to reduce or eliminate the potential for waterhammer identified by past experience. Certain anticipated waterhammer events, such as a closure of a valve, are accounted for in the Code design and analysis of the piping.

(6) The systems evaluation also addresses a potential for fatigue cracking or failure from thermal/mechanical-induced fatigue. Based on past experience, the piping design avoids the potential for significant mixing of high- and low-temperature fluids or mechanical vibration. The startup and preoperational monitoring assures avoidance of detrimental mechanical vibration.

(7) Based on experience and studies by Lawrence Livermore Laboratory, potential indirect sources of indirect pipe rupture are remote causes of pipe rupture. Compliance with the snubber surveillance requirements of the technical specifications assures that snubber failure rates are acceptably low.

(8) Initial LBB evaluation is based on the design configuration and stress levels that are acceptably higher than those identified by the initial analysis. This evaluation is reconciled when the as-built configuration is documented and the Code stress evaluation is reconciled. It is assured that the as-built configuration does not deviate

significantly from the design configuration to invalidate the initial LBB evaluation, or a new evaluation, coupled with necessary configuration modifications, is made to assure applicability of the LBB procedure.

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 shall be performed. The as-built inspection shall confirm that systems, structures and components, 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 shall be performed.

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

3.6.5 COL License Information

3.6.5.1 Details of Pipe Break Analysis Results and Protection Methods

The following shall be provided by the COL applicant (Subsection 3.6.2.5):

(1) A summary of the dynamic analyses applicable to high-energy piping systems in accordance with Subsection 3.6.2.5 of Regulatory Guide 1.70. This shall include:

  (a) 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.

  (b) 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 MEB 3-1, as modified by Subsection 3.6.1.1.1.

(2) For failure in the moderate-energy piping systems listed in Tables 3.6-5 and 3.6-6, descriptions showing how safety-related systems are protected from the resulting jets, flooding and other adverse environmental effects.

(3) 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.

(4) The details of how the MSIV functional capability is protected against the effects of postulated pipe failures.

(5) 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).

(6) The details of how the feedwater line check and feedwater isolation valves functional capabilities are protected against the effects of postulated pipe failures.

(7) An inspection of the as-built high-energy pipe break mitigation features shall be performed. The pipe break analysis report or leak-before-break report shall document the results of the as-built inspection of the high-energy pipe break mitigation features (see Subsection 3.6.4, for a summary of the as-built inspection requirements).

(8) High-energy line separation analysis (HELSA) will be performed by the COL applicant to determine which high-energy lines meet the spatial separation requirements and which lines require further protection (see Subsection 3.6.1.3.2.2, for a summary of the HELSA requirements).

3.6.5.2 Leak-Before-Break Analysis Report

As required by Reference 3.6-1, an LBB analysis report shall be prepared for the piping systems proposed for exclusion from analysis for the dynamic effects due to postulated breaks in high energy piping systems. The report shall be prepared in accordance with the guidelines presented in Appendix 3E and submitted by the COL applicant to the NRC for approval (Subsection 3.6.3).

3.6.5.3 Inservice Inspection of Piping in Containment Penetration Areas

The COL applicant shall perform a 100% volumetric examination of circumferential and longitudinal pipe welds for those portions of piping within the break-exclusion region. The examination shall be performed in accordance with the requirements specified in Subsection 3.6.2.1.4.2 (7).

3.6.6 References

3.6-1 "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 to 41295, October 27, 1987

3.6-2 RELAP 3, "A Computer Program for Reactor Blowdown Analysis", IN-1321, issued June 1970, Reactor Technology TID-4500.

3.6-3 ANSI/ANS-58.2, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture."

3.6-4 "Standard Review Plan; Public Comments Solicited", Federal Register, Volume 52, No. 167, Notices, Pages 32626 to 32633, August 28, 1987.

3.6-5 NUREG-1061, Volume 3, "Evaluation of Potential for Pipe Breaks, Report of the U.S. NRC Piping Review Committee", November 1984.

3.6-6 Mehta, H. S., Patel, N.T. and Ranganath, S., "Application of the Leak-Before-Break Approach to BWR Piping", Report NP-4991, Electric Power Research Institute, Palo Alto, CA, December 1986.

3.6-7 Lahey, R.T. and Moody, F.J., "Thermal Hydraulics of a Boiling Water Nuclear Reactor", American Nuclear Society, 1977.

Table 3.6-1 Essential 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. Emergency Core Cooling Systems† (For LOCA events only)

   One of the following combinations is available (Table 6.3-3):

   (a) HPCF (B and C) + RCIC + RHR-LPFL (B and C) + ADS

   (b) HPCF (B and C) + RHR-LPFL (A and B and C) + ADS

   (c) HPCF (B or C) + RCIC + RHR-LPFL (A and either of B or C) + ADS
5. Core Cooling Systems (other than LOCA events)

   (a) HPCF (B or C) or RCIC

   (b) RHR-LPFL (A or B or C) + ADS

   (c) RHR shutdown Cooling Mode (two loops)

   (d) RHR Suppression Pool Cooling Mode (two loops)
6. Control Rod Drive (scram/rod insertion)
7. Flow Restrictors (passive)
8. Atmospheric Control (for LOCA event only)
9. Standby Gas Treatment‡ (for LOCA event only)
10. Control Room Environmental‡
11. The following equipment/systems or portions thereof required to assure the proper operation of those essential items listed in items 1 through 10:

    (a) Class 1E electrical systems, AC and DC (including diesel generator system‡, 6900, 480 and 120VAC, and 125VDC emergency buses‡, motor control centers‡, switchgear‡, batteries‡ and distribution systems)

    (b) Reactor Building Cooling Water‡ to the following:

       1. Room coolers
       2. Pump coolers
       3. Diesel generator jacket coolers
       4. Electrical switchgear coolers

    (c) Environmental Systems‡ (HVAC)

    (d) Instrumentation (including post-LOCA monitoring)

    (e) Fire Protection System‡

    (f) HVAC Emergency Cooling Water System‡

    (g) Process Sampling System‡

* The essential items listed in this table are protected in accordance with Subsection 3.6.1 consistent with the particular pipe break evaluated.

† See Section 6.3 for detailed discussion of emergency core cooling capabilities.

‡ Located outside containment but listed for completeness of essential shutdown requirements.

Table 3.6-2 Essential Systems, Components, and Equipment* for Postulated Pipe Failures Outside Containment

1. Containment Isolation System and Containment Boundary.
2. Reactor Protection System (SCRAM signals)
3. Core Cooling systems:
   (a) HPCF (B or C) or RCIC
   (b) RHR-LPFL (A or B or C) + ADS
   (c) RHR shutdown cooling mode (two loops)
   (d) RHR suppression pool cooling mode (two loops)
4. Flow Restrictors
5. Control Room Habitability
6. Spent Fuel Pool Cooling
7. Standby Gas Treatment
8. The following equipment/systems or portions thereof required to assure the proper operation of those essential items listed in items 1 through 7:
   (a) Class 1E electrical systems, AC and DC (including diesel generator system, 6900, 480 and 120VAC, and 125VDC emergency buses, motor control centers, switchgear, batteries, auxiliary shutdown control panel, and distribution systems).
   (b) Reactor Building Cooling water to the following:
      (1) Room coolers
      (2) Pump coolers (motors and seals)
      (3) Diesel generator auxiliary system coolers
      (4) Electrical switchgear coolers
      (5) RHR heat exchangers
      (6) FPC heat exchangers
      (7) HECW refrigerators
   (c) HVAC
   (d) Instrumentation (including post accident monitoring)
   (e) Fire Water System
   (f) HVAC Emergency Cooling Water System
   (g) Process Sampling System

* The essential items listed in this table are protected in accordance with Subsection 3.6.1 consistent with the particular pipe break evaluated.

Table 3.6-3 High-Energy Piping Inside Containment

Piping System
Main steam
Main steam drains
Steam supply to RCIC
Feedwater
Recirculation motor cooling
HPCF (RPV to first check valve)
RHR-LPFL (RPV to first check valve)
RHR (Suction from RPV to first normally closed gate valve)
Reactor Water Cleanup (suction from RHR and RPV drain supply to RPV head spray from first inlet valve)
RPV head spray (RPV to first check valve)
RPV vent (RPV to first closed valve)
Standby Liquid Control (from HPCF to first check valve)
CRD (Scram/rod insertion)
RPV bottom head drain lines (RPV to first closed valves)
Miscellaneous 80A and smaller piping

Table 3.6-4 High-Energy Piping Outside Containment

Piping System*
Main Steam
Main Steam Drains
Steam supply to RCIC Turbine
CRD (to and from HCU)
RHR (injection to feedwater from nearest check valves in the RHR lines)
Reactor Water Cleanup (to Feedwater lines and to first inlet valve to RPV head spray)
Reactor Water Cleanup (pumps suction and discharge)

* Fluid systems operating at high-energy levels less than 2% of the total time that the system operates as a moderate-energy fluid system are not included. These systems are classified moderate-energy systems, (i.e., HPCF, RCIC, SAM and SLCS).

Table 3.6-5 Moderate-Energy Piping Inside Containment

Residual Heat Removal System
Radioactive Waste System
Instrument/Service Air System
HVAC Cooling Water System
Reactor Building Cooling Water System

Table 3.6-6 Moderate-Energy Piping Outside Containment

Residual Heat Removal System (Piping beyond outermost isolation valve)
High Pressure Core Flooder System (Piping beyond outermost isolation valve)
Reactor Core Isolation Cooling System (Suction line from condensate storage pool beyond second shutoff valve, vacuum pump discharge line from vacuum pump to containment isolation valve)
Control Rod Drive System (Piping up to pump suction)
Standby Liquid Control System (Piping beyond injection valves)
Suppression Pool Cleanup System (Beyond containment isolation valve)
Fuel Pool Cooling and Cleanup System
Radioactive Waste System (Beyond isolation valve)
Instrument/Service Air System (Beyond isolation valve)
HVAC Cooling Water System
Makeup Water System (Condensate)
Reactor Building Cooling Water System
Turbine Building Cooling Water System
Atmospheric Control System (Beyond shutoff valve)

Table 3.6-7 Additional Criteria for Integrated Leakage Rate Test

(1) Those portions of fluids systems that are part of the reactor coolant pressure boundary, that are open directly to the primary reactor containment atmosphere under post-accident conditions and become an extension of the boundary of the primary reactor containment, shall be opened or vented to the containment atmosphere prior to or during the Type A test. Portions of closed systems inside containment that penetrate primary containment and are not relied upon for containment isolation purposes following a LOCA shall be vented to the containment atmosphere.
(2) All vented systems shall be drained of water to the extent necessary to ensure exposure of the system primary containment isolation valves to the containment air test pressure.
(3) Those portions of fluid systems that penetrate primary containment, that are external to containment and are not designed to provide a containment isolation barrier, shall be vented to the outside atmosphere as applicable, to assure that full post-accident differential pressure is maintained across the containment isolation barrier.
(4) Systems that are required to maintain the plant in a safe condition during the Type A test shall be operable in their normal mode and are not vented.
(5) Systems that are normally filled with water and operating under post-LOCA conditions need not be vented.

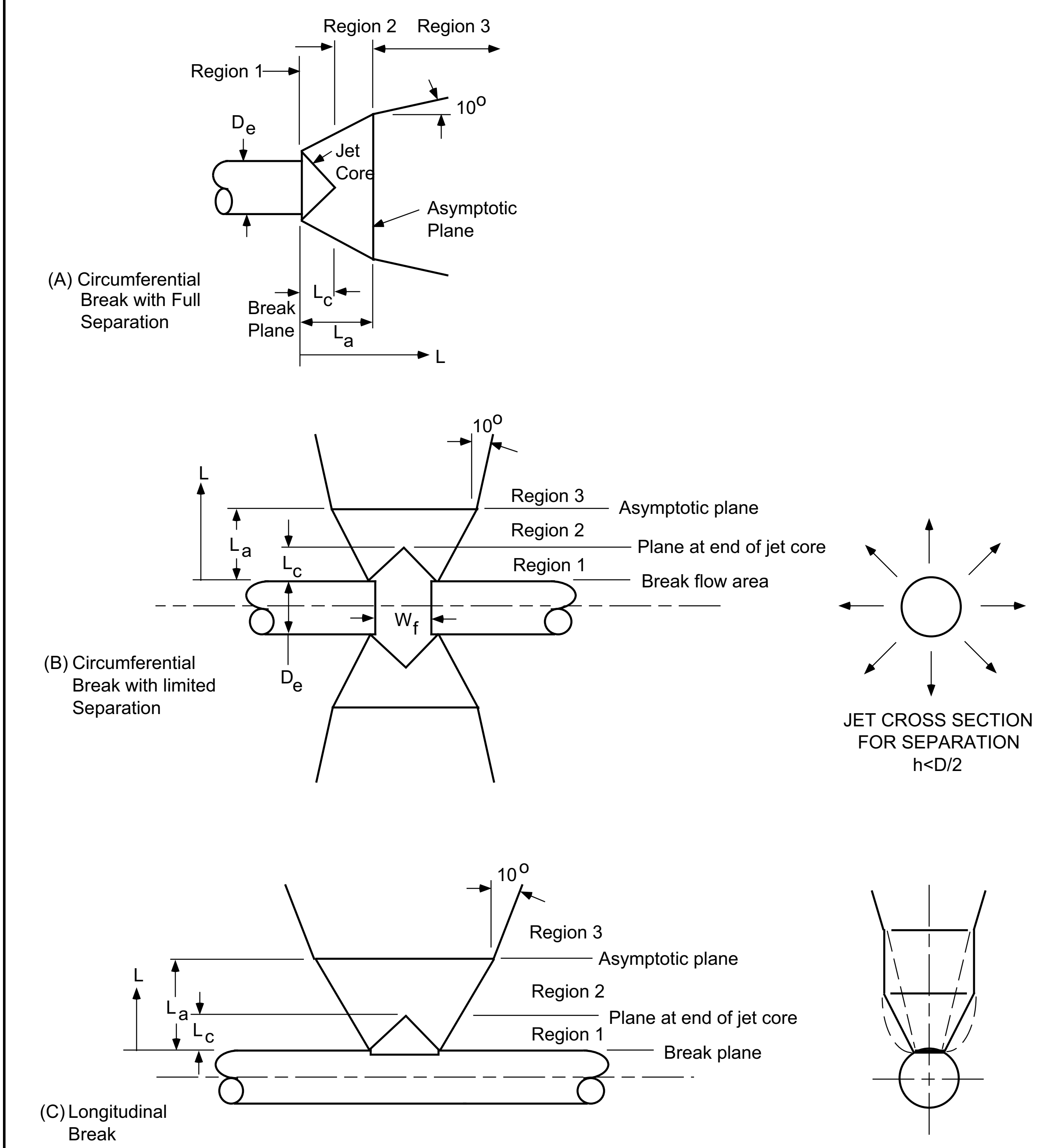

Figure 3.6-1 Jet Characteristics

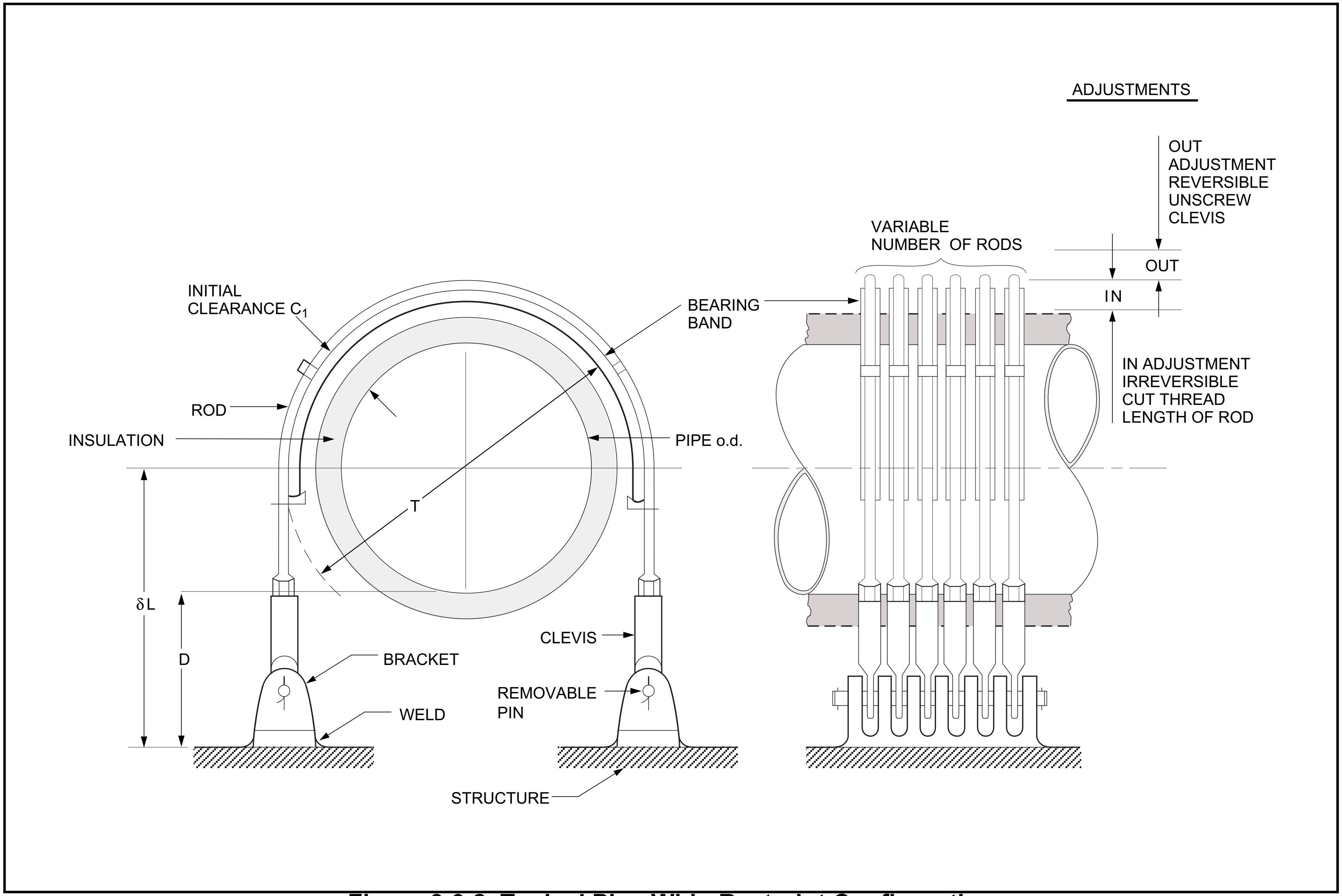

Figure 3.6-2 Typical Pipe Whip Restraint Configuration

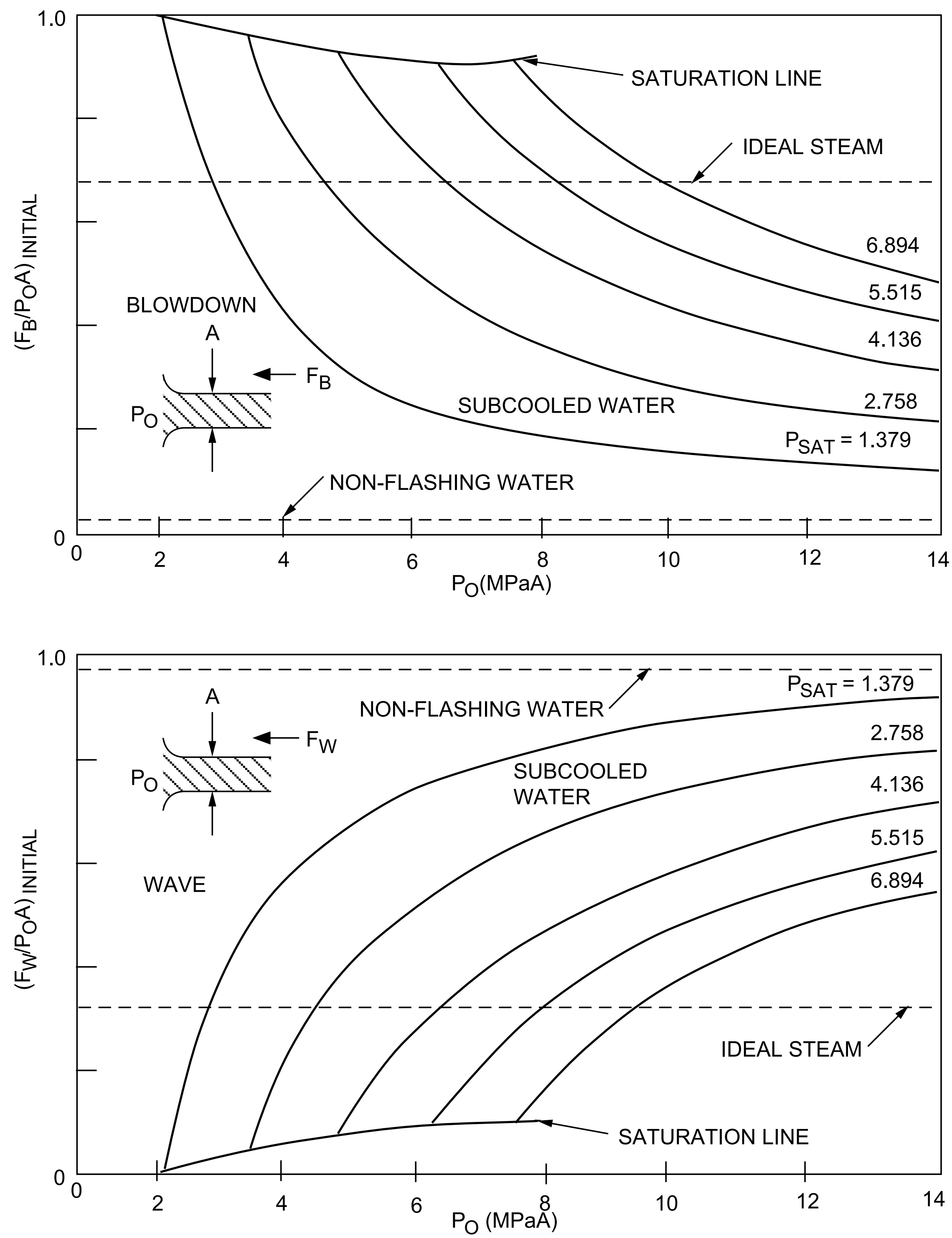

Figure 3.6-3 Initial Blowdown and Wave Forces

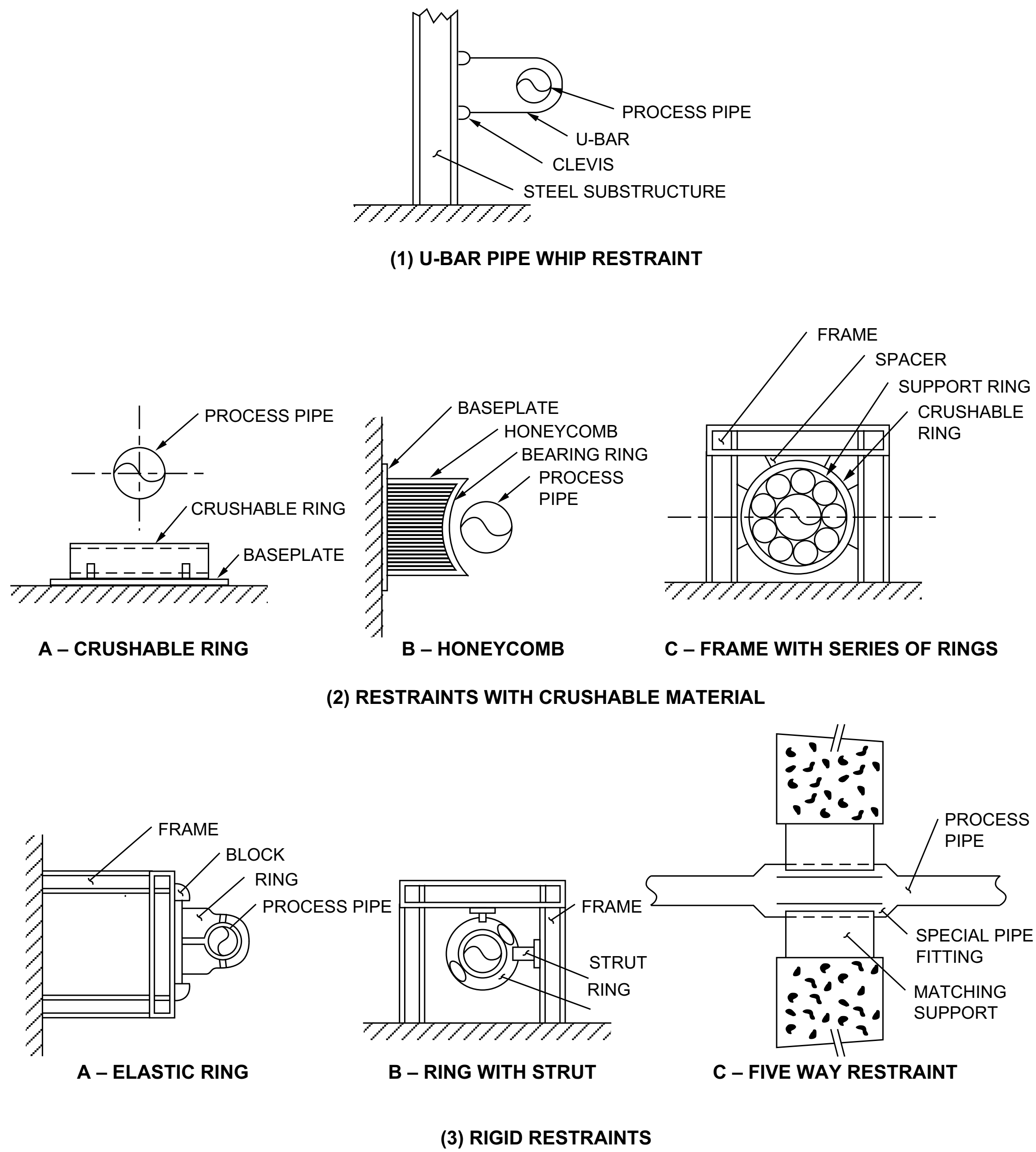

Figure 3.6-4 Acceptable Types of Pipe Whip Restraints

3.7 Seismic Design

All structures, systems, and equipment of the facility are defined as either Seismic Category I or non-Seismic Category I. The requirements for Seismic Category I identification are given in Section 3.2 along with a list of systems, components, and equipment which are so identified.

All structures, systems, components, and equipment that are safety-related, as defined in Section 3.2, are designed to withstand earthquakes as defined herein and other dynamic loads including those due to reactor building vibration (RBV) caused by suppression pool dynamics. Although this section addresses seismic aspects of design and analysis in accordance with Regulatory Guide 1.70, the methods of this section are also applicable to other dynamic loading aspects, except for the range of frequencies considered. The cutoff frequency for dynamic analysis is 33 Hz for seismic loads and 60 Hz for suppression pool dynamic loads. For piping systems with a fundamental frequency greater than 20 Hz, the cutoff frequency for dynamic analysis is 33 Hz for seismic loads and 100 Hz for suppression pool dynamic loads. The definition of rigid system used in this section is applicable to seismic design only.

A safe shutdown earthquake (SSE) is one 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 that earthquake which produces the maximum vibratory ground motion for which Seismic Category I systems and components are designed to remain functional. These systems and components are those necessary to ensure:

(1) The integrity of the reactor coolant pressure boundary.

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition.

(3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

The operating basis earthquake (OBE) is not a design requirement. The effects of low-level earthquake (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.

The seismic design for the SSE is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the required systems and components do not lose their capability to perform their safety-related function. This is referred to as the no-loss-of-function criterion and the loading condition as the SSE loading condition.

Not all safety-related components have the same functional requirements. For example, the reactor containment must retain capability to restrict leakage to an acceptable level. Therefore, based on present practice, elastic behavior of this structure under the SSE loading condition is ensured. On the other hand, there are certain structures, components, and systems that can suffer permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain contents and allow fluid flow.

Table 3.2-1 identifies the equipment in various systems as Seismic Category I or non-Seismic Category I.

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The design earthquake loading is specified in terms of a set of idealized, smooth curves called the design response spectra in accordance with Regulatory Guide 1.60.

Figure 3.7-1 shows the standard ABWR design values of the horizontal SSE spectra applied at the finished grade in the free field for damping ratios of 2.0, 3.0, 4.0, 5.0, and 7.0% of critical damping where the maximum horizontal ground acceleration is 0.3g. Figure 3.7-2 shows the standard ABWR design values of the vertical SSE spectra applied at the finished grade in the free field for damping ratios of 2.0, 3.0, 4.0, 5.0, and 7.0% of critical damping where the maximum vertical ground acceleration is 0.30g at 33Hz, same as the maximum horizontal ground acceleration.

The design spectra are constructed in accordance with Regulatory Guide 1.60. The normalization factors for the maximum values in two horizontal directions are 1.0 and 1.0 as applied to Figure 3.7-1. For vertical direction, the normalization factor is 1.0 as applied to Figure 3.7-2.

3.7.1.2 Design Time History

The design time histories are synthetic acceleration time histories generated to match the design response spectra defined in Subsection 3.7.1.1.

The earthquake acceleration time history components are identified as H1, H2, and V. The H1 and H2 are the two horizontal components mutually perpendicular to each other. Both H1 and H2 are based on the design horizontal ground spectra shown in Figure 3.7-1. The VT is the vertical component and it is based on the design vertical ground spectra shown in Figure 3.7-2. The SSE acceleration time histories of the three components are shown in Figures 3.7-3 through 3.7-5 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 Standard Review Plan (SRP) 3.7.1. The computed response spectra of 2%, 3%, 4%, 5% and 7% damping are compared with the corresponding design Regulatory Guide 1.60 spectra in Figures 3.7-6 through 3.7-10 for the H1 components, in Figures 3.7-11 through 3.7-15 for the H2 component, and Figures 3.7-16 through 3.7-20 for the VT 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 Spectra Density (PSD) requirement stipulated in Appendix A to SRP 3.7.1 The computed PSD functions envelop the target PSD of a maximum 0.3g acceleration with a wide margin in the frequency range of 0.3 Hz to 24 Hz as shown in Figures 3.7-24 and 3.7-25 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 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 with the following input coefficients for 1.0g peak ground acceleration:

$$
\begin{aligned}
So(f) &= 2288.51 cm^2/s^3 \, (f/3.5)^{0.2} \qquad && f \le 3.5 \text{ Hz} \\
&= 2288.51 \text{ } cm^2/s^3 \, (3.5/f)^{1.6} && 3.5 < f \le 9.0 \text{ Hz} \\
&= 504.98 \text{ } cm^2/s^3 \, (9.0/f)^{3.0} && 9.0 < f \le 16.0 \text{ Hz} \\
&= 89.88 \text{ } cm^2/s^3 \, (16.0/f)^{7.0} && 16.0 < f \text{ Hz}
\end{aligned}
\qquad (3.7\text{-}1)
$$

The PSD function of vertical component of the design time history (SSE with 0.3g PGA) is computed and subsequently averaged and smoothed using SRP 3.7.1 criteria. Similarly, the target PSD is computed for 0.3g maximum acceleration. The PSD of the design time history is compared with the target and 80% of target PSD in Figure 3.7-26. As shown in this figure, PSD of the vertical time history envelopes the target PSD with a wide margin. This comparison confirms the adequacy of energy content of the vertical time history.

The time histories of three spatial components are checked for statistic independence. The cross-correlation coefficient at zero time lag is 0.01351 between H1 and H2, 0.07037 between H1 and VT, and 0.07367 between H2 and VT. All of them are less than 0.16 as recommended in the reference of Regulatory Guide 1.92. Thus, H1, H2, and VT acceleration time histories are mutually statistically independent.

3.7.1.3 Critical Damping Values

The damping values for SSE analysis are presented in Table 3.7-1 for various structures and components. They are in compliance with Regulatory Guides 1.61 and 1.84, except for the damping values of cable trays and conduits.

The damping values shown in Table 3.7-1 and Figure 3.7-27 for cable trays and conduits are based on the results of over 2000 individual dynamic tests conducted by Bechtel/ANCO for a variety of raceway configurations (Reference 3.7-8). The damping value of cable tray systems (including supports) depends on the level of input motion and the amount of cable fill. In the acceleration range of interest to the ABWR design, the damping value is 7% for empty trays, and it increases to 20% for 50% to fully loaded trays. For trays loaded to less than 50% the damping value can be obtained by linear interpolation. The damping value of conduit systems (including supports) is 7% constant. For HVAC ducts and supports the damping value is 7% for companion angle or pocket lock construction and is 4% for welded construction.

3.7.1.4 Supporting Media for Seismic Category I Structures

The following ABWR Standard Plant Seismic Category I structures have concrete mat foundations supported on soil, rock or compacted backfill. The maximum value of the embedment depth below plant grade to the bottom of the base mat is given below for each structure:

(1) Reactor Building (including the enclosed primary containment vessel and reactor pedestal)—25.7m

(2) Control Building—23.2m

(3) Radwaste Building Substructure—16m

All of the above buildings have independent foundations. In all cases the maximum value of embedment is used for the dynamic analysis to determine seismic soil-structure interaction effects. The foundation support materials withstand the pressures imposed by appropriate loading combinations without failure. The total structural height of each building is described in Subsections 3.8.2 through 3.8.4. (see Subsection 3.8.5 for details of the structural foundations). The ABWR Standard Plant is designed for a range of soil conditions given in Appendix 3A.

3.7.1.4.1 Soil-Structure Interaction

When a structure is supported on a flexible foundation, the soil-structure interaction is taken into account by coupling the structural model with the soil medium. The finite-element representation is used for a broad range of supporting medium conditions. Detailed methodology and results of the soil-structure interaction analysis are provided in Appendix 3A.

3.7.2 Seismic System Analysis

This subsection applies to the design of Seismic Category I structures and the reactor pressure vessel (RPV). Subsection 3.7.3 applies to all Seismic Category I piping systems and equipment.

3.7.2.1 Seismic Analysis Methods

Analysis of Seismic Category I structures and the RPV is accomplished using the response spectrum or time-history approach. The time-history approach is made either in the time domain or in the frequency domain.

Either approach utilizes the natural period, mode shapes, and appropriate damping factors of the particular system toward the solution of the equations of dynamic equilibrium. The time-history approach may alternately utilize the direct integration method of solution. When the structural response is computed directly from the coupled structure-soil system, the time-history approach solved in the frequency domain is used. The frequency domain analysis method is described in Appendix 3A.

3.7.2.1.1 The Equations of Dynamic Equilibrium for Base Support Excitation

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped-mass, distributed-stiffness system are expressed in a matrix form as:

$$[M]\{\ddot{u}(t)\} + [c]\{\dot{u}(t)\} + [K]\{u(t)\} = \{P(t)\} \qquad (3.7\text{-}2)$$

where

$\{u(t)\}$	=	Time-dependent displacement vector of non-support points relative to the supports ($u_t(t) = u(t) + u_s(t)$)
$\{\dot{u}(t)\}$	=	Time-dependent velocity vector of non-support points relative to the supports
$\{\ddot{u}(t)\}$	=	Time-dependent acceleration vector of non-support points relative to the supports
$[M]$	=	Mass matrix
$[C]$	=	Damping matrix
$[K]$	=	Stiffness matrix
$\{P(t)\}$	=	Time-dependent inertia force vector ($-[M]\ \{\ddot{u}_s(t)\}$) acting at non-support points

The manner in which a distributed-mass, distributed-stiffness system is idealized into a lumped-mass, distributed-stiffness system of Seismic Category I structures and the RPV is shown in Figure 3.7-28 along with a schematic representation of relative acceleration; $\ddot{u}(t)$, support acceleration; $\ddot{u}_s(t)$ and total acceleration; $\ddot{u}_t(t)$.

3.7.2.1.2 Solution of the Equations of Motion by Modal Superposition

The technique used for the solution of the equations of motion is the method of modal superposition.

The set of homogeneous equations represented by the undamped free vibration of the system is:

$$[M]\{\ddot{u}(t)\} + [K]\{u(t)\} = \{0\}. \qquad (3.7\text{-}3)$$

Since the free oscillations are assumed to be harmonic, the displacements can be written as:

$$\{u(t)\} = \{\phi\}e^{i\omega t}. \qquad (3.7\text{-}4)$$

where

$\{\phi\}$	=	Column matrix of the amplitude of displacements $\{u\}$
ω	=	Circular frequency of oscillation
t	=	Time

Substituting Equation 3.7-4 and its derivatives in Equation 3.7-3 and noting that $e^{i\omega t}$ is not necessarily zero for all values of ωt yields:

$$[-\omega^2[M] + [K]]\{\phi\} = \{0\}. \qquad (3.7\text{-}5)$$

Equation 3.7-5 is the classic dynamic characteristic equation, with solution involving the eigenvalues of the frequencies of vibrations ω_i and the eigenvalues mode shapes, $\{\phi\}_i$, (i = 1, 2, …, n).

For each frequency ω_i, there is a corresponding solution vector $\{\phi\}_i$ determined to be within an arbitrary scalar factor Y_i known as the normal coordinate. It can be shown that the mode shape vectors are orthogonal with respect to the stiffness matrix [K] in the n-dimensional vector space.

The mode shape vectors are also orthogonal with respect to the mass matrix [M].

The orthogonality of the mode shapes can be used to effect a coordinate transformation of the displacements, velocities and accelerations such that the response in each mode is independent of the response of the system in any other mode. Thus, the problem becomes one of solving n independent differential equations rather than n simultaneous differential equations; and, since the system is linear, the principle of superposition holds and the total response of the system oscillating simultaneously in n modes may be determined by direct addition of the responses in the individual modes.

3.7.2.1.3 Analysis by Response Spectrum Method

The response spectrum method is based on the fact that the modal response can be expressed as a set of convolution integrals which satisfy the governing differential equations. The advantage of this form of solution is that, for a given ground motion, the only variables under the integral are the damping factor and the frequency. Thus, for a specified damping factor it is possible to construct a curve which gives a maximum value of the integral as a function of frequency.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in Subsection 3.7.2.7.

When the equipment is supported at two or more points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple support excitation analysis methods may be used where acceleration time histories or response spectra are applied to all the equipment attachment points. In some cases, the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors may be applied identically to all floors, provided there is no significant shift in frequencies of the spectra peaks.

3.7.2.1.4 Support Displacements in Multi-Supported Structures

In the preceding sections, analysis procedures for forces and displacements induced by time-dependent support displacement were discussed. In a multi-supported structure there are, in addition, time-dependent support displacements which produce additional displacements at nonsupport points and pseudo-static forces at both support and nonsupport points.

[*The governing equation of motion of a structural system which is supported at more than one point and has different excitations applied at each may be expressed in the following concise matrix form:*

$$\left[\begin{array}{c|c} M_a & O \\ \hline O & M_s \end{array}\right]\left\{\begin{array}{c} \ddot{U}_a \\ \hline \ddot{\overline{U}}_s \end{array}\right\} + \left[\begin{array}{c|c} C_{aa} & C_{as} \\ \hline C_{as} & C_{ss} \end{array}\right]\left\{\begin{array}{c} \dot{U}_a \\ \hline \dot{\overline{U}}_s \end{array}\right\} + \left[\begin{array}{c|c} K_{aa} & K_{as} \\ \hline K_{as} & K_{ss} \end{array}\right]\left\{\begin{array}{c} U_a \\ \hline \overline{U}_s \end{array}\right\} = \left\{\begin{array}{c} \overline{F}_a \\ \hline F_s \end{array}\right\} \qquad (3.7\text{-}6)$$

where

U_a = *Displacement of the active (unsupported) degrees of freedom*

$\bar{U}_s$	=	*Specified displacements of support points*
M_a and M_s	=	*Lumped diagonal mass matrices associated with the active degrees of freedom and the support points*
C_{aa} and K_{aa}	=	*Damping matrix and elastic stiffness matrix, respectively, expressing the forces developed in the active degrees of freedom due to the motion of the active degrees of freedom*
C_{ss} and K_{ss}	=	*Support forces due to unit velocities and displacement of the supports*
C_{as} and K_{as}	=	*Damping and stiffness matrices denoting the coupling forces developed in the active degrees of freedom by the motion of the supports and vice versa*
$\bar{F}_a$	=	*Prescribed external time-dependent forces applied on the active degrees of freedom*
F_s	=	*Reaction forces at the system support points*

Total differentiation with respect to time is denoted by (˙) in Equation 3.7-6. Also, the contributions of the fixed degrees of freedom have been removed in the equation. The procedure utilized to construct the damping matrix is discussed in Subsection 3.7.2.15. The mass and elastic stiffness matrices are formulated by using standard procedures.

Equation 3.7-6 can be separated into two sets of equations. The first set of equations can be written as:

$$[M_s]\{\ddot{\bar{U}}_s\} + [C_{ss}]\{\dot{\bar{U}}_s\} + [K_{ss}]\{\bar{U}_s\} + [C_{as}]\{\dot{U}_a\} + [K_{as}]\{U_a\} = \{F_s\}; \tag{3.7-7}$$

and the second set as:

$$[M_a]\{\ddot{U}_a\} + [C_{aa}]\{\dot{U}_a\} + [K_{aa}]\{U_a\} + [C_{as}]\{\dot{\bar{U}}_s\} + [K_{as}]\{\bar{U}_s\} = \{\bar{F}_a\}; \tag{3.7-8}$$

The timewise solution of Equation 3.7-8 can be obtained easily by using the standard normal mode solution technique. After obtaining the displacement response of the active degrees of freedom (U_a), Equation 3.7-7 can then be used to solve the support point reaction forces (F_s). Analysis can be performed using the time history method or response spectrum method.

Modal superposition is used to determine the solutions of the uncoupled form of Equation 3.7-7. The procedure is identical to that described in Subsection 3.7.2.1.2. Additional *requirements associated with the independent support motion response spectrum method of analysis are given in Subsection 3.7.3.8.1.10.*]*

3.7.2.1.5 Dynamic Analysis of Buildings

The time-history method either in the time domain or in the frequency domain is used in the dynamic analysis of buildings. As for the modeling, both finite-element and lumped-mass methods are used.

3.7.2.1.5.1 Description of Mathematical Models

A mathematical model reflects the stiffness, mass, and damping characteristics of the actual structural systems. One important consideration is the information required from the analysis. Consideration of maximum relative displacements among supports of Seismic Category I structures, systems, and components require that enough points on the structure be used. Locations of Seismic Category I equipment are taken into consideration. Buildings are mathematically modeled as a system of lumped masses located at elevations of mass concentrations such as floors.

In general, three-dimensional models are used for seismic analysis. In all structures, six degrees of freedom exist for all mass points (i.e., three translational and three rotational). However, in most structures, some of the dynamic degrees of freedom can be neglected or can be uncoupled from each other so that separate analyses can be performed for different types of motions.

Coupling between the two horizontal motions occurs when the center of mass, the centroid, and the center of rigidity do not coincide. The degree of coupling depends on the amount of eccentricity and the ratio of the uncoupled torsional frequency to the uncoupled lateral frequency. Since lateral/torsional coupling and torsional response can significantly influence floor accelerations, structures are in general designed to keep minimum eccentricities. However, for analysis of structures that possess unusual eccentricities, a model of the support building is developed to include the effect of lateral/torsional coupling.

3.7.2.1.5.1.1 Reactor Building and Reactor Pressure Vessel

The Reactor Building (R/B) complex includes: (1) the reinforced concrete containment vessel (RCCV), which includes the reactor shield wall (RSW), the reactor pedestal, and the reactor pressure vessel (RPV) and its internal components (2) the secondary containment zone having many equipment compartments, and (3) the clean zone. The building basemat is assumed to be rigid. Building elevations along the 0°–180° and 90°–270° sections are shown in Figures 3.7-29 and 3.7-30, respectively. The mathematical model is shown in Figure 3.7-31. The model X and Y axes correspond to the R/B 0°–180° and 90°–270° directions, respectively. The Z axis is along the vertical direction. The combined R/B model (Figure 3.7-31) basically consists of two uncoupled 2-D models in the X-Z and Y-Z planes, since the building is essentially of a

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

symmetric design with respect to its two principal directions in the horizontal plane. The double symmetry assumption of this stick model is justified by comparing its responses to that of a detailed 3-D finite element model at major elevations for the fixed base condition with embedded effect included. The results show that the two models are dynamically equivalent and the responses are in good agreement, except for the vertical responses of the building walls where the finite element results are higher at frequencies between 20 and 30 Hz. Therefore, with the exception noted, the stick model of a double symmetry representation (i.e., without eccentricities and torsional degrees of freedom) can be used to predict seismic response of the Reactor Building complex. To account for the variations in the building wall vertical responses, the results of the finite element analysis are enveloped with the results of SSI analyses using the stick model in establishing seismic design loads as described in Subsection 3A.10.2. The methods used to account for torsional effects to define design loads are given in Subsection 3.7.2.11.

The model shown in Figure 3.7-31 corresponds to the X-Z plane. The only differences in terms of schematic representation between the X-Z and Y-Z plane models is that the rotational spring between the RCCV top slab (Node 90) and the basemat top (Node 88) is presented only in the X-Z plane.

Each structure in the R/B complex is idealized by a center-lined stick model of a series of massless beam elements. Axial, flexural, and shear deformation effects are included in formulating beam stiffness terms. Coupling between individual structures is modeled by linear spring elements. Masses, including dead weights of structural elements, equipment weights and piping weights, are lumped to nodal points. The weights of water in the spent fuel storage pool and the suppression pool are also considered and lumped to appropriate locations.

The portions of the R/B outside the RCCV are box-type shear wall systems of reinforced concrete construction. The major walls between floor slabs are represented by beam elements of a box cross section. The shear rigidity in the direction of excitation is provided by the parallel walls. The bending rigidity includes the cross walls contribution. The R/B is fully integrated with the RCCV through floor slabs at various elevations. Spring elements are used to represent the slab in-plane shear stiffness in the horizontal direction. In the vertical direction a single mass point is used for each slab and it is connected to the walls and RCCV by spring elements. The spring stiffness is determined so that the fundamental frequency of the slab in the vertical direction is maintained.

The RCCV is a cylindrical structure with a flat top slab with the drywell opening, which, along with upper pool girders and R/B walls, form the upper pool. Mass points are selected at the R/B floor slab locations. Stiffnesses are represented by a series of beam elements. In the X-Z plane, a rotational spring element connecting the top slab and the basemat is used to account for the additional rotational rigidity provided by the integrated RCCV-pool girder-building walls system. The RCCV is also coupled to the RPV through the refueling bellows, and to the reactor pedestal through the diaphragm floor. Spring elements are used to account for these interactions. The lower drywell access tunnels spanning between the RCCV and the reactor pedestal are not modeled, since flexible rings are provided which are designed to reduce the coupling effects.

The RSW consists of two steel ring plates with concrete fill in between for shielding purposes. Concrete in the RSW does not contribute to stiffness; but its weight is included. The reactor pedestal is a cylindrical structure of a composite steel-concrete design. The total stiffness of the pedestal includes the full strength of the concrete core. Mass points are selected at equipment interface locations and geometrical discontinuities. In addition, intermediate mass points are chosen to result in more uniform mass distribution. The pedestal supports the reactor pressure vessel and it also provides lateral restraint to the reactor CRD housings below the vessel. The RSW is connected to the RPV by the RPV stabilizers which are modeled as spring elements.

The model of the RPV and its internal components is described in Subsection 3.7.2.3.2. This model (Figure 3.7-32) is coupled with the above-described R/B model for the seismic analysis.

3.7.2.1.5.1.2 Control Building

The Control Building dynamic model is shown in Figure 3.7-33. The Control Building is a box type shear wall system of reinforced concrete. The major walls between floor slabs are represented by beam elements of a box cross section. The shear rigidity in the direction of excitation is provided by the parallel walls. The bending rigidity includes the cross walls contribution. In the vertical direction a single mass point is used for each slab and is connected to the walls by spring elements. The spring element stiffness is determined so that the fundamental frequency of the slab in the vertical direction is maintained.

3.7.2.1.5.1.3 Radwaste Building

The Radwaste Building dynamic model is shown in Figure 3.7-34. The Radwaste Building is a box type shear wall system of reinforced concrete. The major walls between floor slabs are represented by beam elements of a box cross section. The shear rigidity in the direction of excitation is provided by the parallel walls. The bending rigidity includes the cross walls contribution. In the vertical direction a single mass point is used for each slab and is connected to the walls by spring elements. The spring element stiffness is determined so that the fundamental frequency of the slab in the vertical direction is maintained.

3.7.2.1.5.2 Rocking and Torsional Effects

Rocking effects due to horizontal ground movement are considered in the soil-structure interaction analysis as described in Appendix 3A. Whenever building response is calculated from a second step structural analysis, rocking effects are included as input simultaneously applied with the horizontal translational motion at the basemat. The torsional effect considered is described in Subsection 3.7.2.11.

3.7.2.1.5.3 Hydrodynamic Effects

For a dynamic system in which a liquid such as water is involved, the hydrodynamic effects on adjacent structures due to horizontal excitation are taken into consideration by including hydrodynamic mass coupling terms in the mass matrix. The basic formulas used for computing these terms are in Reference 3.7-4. In the vertical excitation, the hydrodynamic coupling effects are assumed to be negligible and the water mass is lumped to appropriate structural locations.

3.7.2.2 Natural Frequencies and Response Loads

The natural frequencies up to 33 Hz for the Reactor, Control and Radwaste Buildings are presented in Tables 3.7-2 through 3.7-6 for the fixed base condition.

Enveloped response loads at key locations in the Reactor Building complex and the control building due to SSE for the range of site conditions considered are presented in Appendix 3A. Response spectra at the major equipment elevations and support points are also given in Appendix 3A.

The design SSE loads for the Radwaste Building are given in Table 3H.3-1.

3.7.2.3 Procedure Used for Modeling

3.7.2.3.1 Modeling Techniques for Systems Other Than Reactor Pressure Vessel

An important step in the seismic analysis of systems other than the reactor pressure vessel is the procedure used for modeling. The techniques center around two methods. In the first method, the system is represented by lumped masses and a set of spring dashpots idealizing both the inertial and stiffness properties of the system. The details of the mathematical models are determined by the complexity of the actual structures and the information required for the analysis. For the decoupling of the subsystem and the supporting system, the following criteria equivalent to the SRP requirements are used:

(1) If $R_m \leq 0.01$, decoupling can be done for any R_f.

(2) If $0.01 \leq R_m \leq 0.1$, decoupling can be done if $R_f \leq 0.8$ or $R_f \geq 1.25$.

(3) If $R_m > 0.1$, an approximate model of the subsystem should be included in the primary system model.

where R_m and R_f are defined as:

R_m = Total mass of the supported system/mass that supports the subsystem

R_f = Fundamental frequency of the supported subsystem/frequency of the dominant support motion

If the subsystem is comparatively rigid in relation to the supporting system, and also is rigidly connected to the supporting system, it is sufficient to include only the mass of the subsystem at the support point in the primary system model. On the other hand, in case of a subsystem supported by very flexible connections (e.g., pipe supported by hangers), the subsystem need not be included in the primary model. In most cases, the equipment and components, which come under the definition of subsystems, are analyzed (or tested) as a decoupled system from the primary structure and the seismic input for the former is obtained by the analysis of the latter. One important exception to this procedure is the Reactor Coolant System, which is considered a subsystem but is usually analyzed using a coupled model of the Reactor Coolant System and primary structure.

In the second method of modeling, the structure of the system is represented as a two- or three-dimensional finite-element model using combinations of beam, plate, shell, and solid elements. The details of the mathematical models are determined by the complexity of the actual structures and the information required for the analysis.

3.7.2.3.2 Modeling of Reactor Pressure Vessel and Internals

The seismic loads on the RPV and reactor internals are based on coupled dynamic analysis with the Reactor Building. The mathematical model of the RPV and internals is shown in Figure 3.7-32. This model is coupled with the R/B model for dynamic analysis.

The RPV and internals mathematical model consists of lumped masses connected by elastic beam element members. Using the elastic properties of the structural components, the stiffness properties of the model are determined and the effects of axial, bending and shear are included.

Mass points are located at all points of critical interest such as anchors, supports, points of discontinuity, etc. In addition, mass points are chosen so that the mass distribution in various zones is as uniform as practicable and the full range of frequency of response of interest is adequately represented. Further, in order to facilitate hydrodynamic mass calculations, several mass points (fuel, shroud, vessel) are selected at the same elevation. The RPV and internals are quite stiff in the vertical direction. Vertical modes in the frequency range of interest are adequately obtained with few dynamic degrees of freedom. Therefore, vertical masses are distributed to a few key nodal points. The various lengths of CRD housing are grouped into the two representative lengths shown in Figure 3.7-32. These lengths represent the longest and shortest housing in order to adequately represent the full range of frequency response of the housings.

Not included in the mathematical model are the stiffness properties of light components, such as incore guide tubes and housings, sparger, and their supply headers. This is done to reduce the complexity of the dynamic model. For the seismic responses of these components, floor response spectra generated from system analysis are used.

The presence of a fluid and other structural components (e.g., fuel within the RPV) introduces a dynamic coupling effect. Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix which will serve to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. The details of the hydrodynamic mass derivation are given in Reference 3.7-4.

3.7.2.4 Soil-Structure Interaction

The soil model and soil-structure interaction analysis are described in Appendix 3A.

3.7.2.5 Development of Floor Response Spectra

In order to predict the seismic effects on equipment located at various elevations within a structure, floor response spectra are developed using a time-history analysis technique.

The procedure entails first developing the mathematical model assuming a linear system and then obtaining its natural frequencies and mode shapes. The dynamic response at the mass points is subsequently obtained by using a time-history approach.

Using the acceleration time-history response of a particular mass point, a spectrum response curve is developed and incorporated into a design acceleration spectrum to be utilized for the seismic analysis of equipment located at the mass point. Horizontal and vertical response spectra are computed for various damping values applicable for evaluation of equipment. Two orthogonal horizontal and one vertical earthquake component are input separately. Response spectra at selected locations are then generated for each earthquake component separately. They are combined using the square-root-of-the-sum-of-the-squares (SRSS) method to predict the total co-directional floor response spectrum for that particular frequency. This procedure is carried out for each site-soil case used in the soil-structure interaction analysis. Response spectra for all site-soil cases are finally combined to arrive at one set of final response spectra.

An alternate approach to obtain co-directional floor response spectra is to perform dynamic analysis with simultaneous input of various earthquake components if those components are statistically independent to each other.

The response spectra values are computed as a minimum either at frequency intervals as specified in Table 3.7.1-1 of SRP 3.7.1 or at a set of frequencies in which each frequency is within 10% of the previous one.

3.7.2.6 Three Components of Earthquake Motion

The three components of earthquake motion are considered in the building seismic analyses. To properly account for the responses of systems subjected to the three-directional excitation, a statistical combination is used to obtain the net response according to the SRSS criterion of Regulatory Guide 1.92. The SRSS method accounts for the randomness of magnitude and direction of earthquake motion. The SRSS criterion, applied to the responses associated with the three components of ground earthquake motion, is used for seismic stress computation for steel structural design, as well as for resultant seismic member force computations for reinforced concrete structural design.

The SRSS method of combination is used when the response analysis is performed using the time history method (separate analysis for each component), the response spectrum method, or the static coefficient method. If the time history method of analysis is performed separately for each of the components which are mutually statistically independent, the total response may alternatively be obtained by algebraically adding the codirectional responses calculated separately for each component at each time step. Furthermore, when the time history method is performed applying the three mutually statistically independent motions simultaneously, the combined response is obtained directly by solution of the equations of motion.

3.7.2.7 Combination of Modal Response

When the dynamic response analysis is performed using the response spectrum method, the methods of modal response combination delineated in Regulatory Guide 1.92 are used. The effects of high-frequency modes are considered in accordance with Appendix A to SRP 3.7.2.

3.7.2.8 Interaction of Non-Seismic Category I Structures, Systems and Components with Seismic Category I Structures, Systems and Components

The interfaces between Seismic Category I and non-Seismic Category I structures, systems and components are designed for the dynamic loads and displacements produced by both the Category I and non-Category I structures, systems and components. All non-Category I structures, systems and components will meet any one of the following requirements:

(1) The collapse of any non-Category I structure, system or component will not cause the non-Category I structure, system or component to strike a Seismic Category I structure, system or component.

(2) The collapse of any non-Category I structure, system or component will not impair the integrity of Seismic Category I structures, systems or components. This may be demonstrated by showing that the impact loads on the Category I structure, system or component resulting from collapse of an adjacent non-Category I structure, because of its size and mass, are either negligible or smaller than those considered in the design (e.g., loads associated with tornado/hurricane, including missiles).

(3) The non-Category I structures, systems or components will be analyzed and designed to prevent their failure under SSE conditions in a manner such that the margin of safety of these structures, systems or components is equivalent to that of Seismic Category I structures, systems or components.

The COL applicant will describe the process for completion of the design of balance-of-plant and non-safety related systems to minimize interactions and propose procedures for an inspection of the as-built plants for interactions. (See Subsection 3.7.5.4 for COL license information requirements).

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Floor response spectra calculated according to the procedure described in Subsection 3.7.2.5 are peak broadened to account for uncertainties associated with the material properties of the structure and soil and approximations in the modeling techniques used in the analysis. If no parametric variation studies are performed, the spectral peaks associated with each of the structural frequencies are broadened by ±15%. If a detailed parametric variation study is made, the minimum peak broadening ratio is ±10%. In lieu of peak broadening, the peak shifting method of Appendix N of ASME Section III, as permitted by Regulatory Guide 1.84, can be used.

3.7.2.10 Use of Constant Vertical Static Factors

Since all Seismic Category I structures and the RPV are subjected to a vertical dynamic analysis, no constant vertical static factors are utilized.

3.7.2.11 Methods Used to Account for Torsional Effects

Torsional effects for two-dimensional analytical models are accounted for in the following manner. The locations of the center of mass are calculated for each floor. The centers of rigidity and rotational stiffness are determined for each story. Torsion effects are introduced in each story by applying a rotational moment about its center of rigidity. The rotational moment is calculated as the sum of the products of the inertial force applied at the center of mass of each floor above and a moment arm equal to the distance from the center of mass of the floor to the center of rigidity of the story plus 5% of the maximum building dimension at the level under consideration. To be conservative, the absolute values of the moments are used in the sum. The torsional moment and story shear are distributed to the resisting structural elements in proportion to each individual stiffness. For the Reactor and Control Buildings, the actual eccentricities are negligible and the torsional moments are due to accidental torsion only.

The RPV model is axisymmetric with no built-in eccentricity. The effects of accidental torsion on the RPV are negligible since the torsion-induced shear stress is only 5% of the shear stress due to the direct shear force.

3.7.2.12 Comparison of Responses

The time history method of analysis is used for the Reactor and Control Buildings. A comparison of responses with the response spectrum method is therefore not required. The Radwaste Building is analyzed using the response spectrum method, since the time history method needed for the generation of floor response spectra is not necessary because there are no safety-related components inside the building.

3.7.2.13 Methods for Seismic Analysis of Category I Dams

The analysis of all Category I dams, if applicable for the site, taking into consideration the dynamic nature of forces (due to both horizontal and vertical earthquake loadings), the behavior of the dam material under earthquake loadings, soil structure interaction effects, and nonlinear stress-strain relations for the soil, will be used. Analysis of earth-filled dams, if applicable, includes an evaluation of deformations.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

Seismic loads are dynamic in nature. The method of calculating seismic loads with dynamic analysis and then treating them as static loads to evaluate the overturning of structures and foundation failures while treating the foundation materials as linear elastic is conservative. Overturning of the structure, assuming no soil slip failure occurs, can be caused only by the center of gravity of the structure moving far enough horizontally to cause instability.

Furthermore, when the combined effect of earthquake ground motion and structural response is strong enough, the structure undergoes a rocking motion pivoting about either edge of the base. When the amplitude of rocking motion becomes so large that the center of structural mass reaches a position right above either edge of the base, the structure becomes unstable and may tip over. The mechanism of the rocking motion is like an inverted pendulum and its natural period is long compared with the linear, elastic structural response. Thus, with regard to overturning, the structure is treated as a rigid body.

The maximum kinetic energy can be conservatively estimated to be:

$$E_s = \frac{1}{2}\sum_i m_i[(v_H)_i^2 + (v_V)_i^2] \qquad (3.7\text{-}9)$$

where (v_H) and (v_V) are the maximum values of the total lateral velocity and total vertical velocity, respectively, of mass m_i.

Values for $(v_H)_i$ and $(v_V)_i$ are computed as follows:

$$(v_H)_i^2 = (v_x)_i^2 + (v_H)_g^2 \qquad (3.7\text{-}10)$$

$$(v_V)_i^2 = (v_z)_i^2 + (v_V)_g^2 \qquad (3.7\text{-}11)$$

where $(v_H)_g$ and $(v_V)_g$ are the peak horizontal and vertical ground velocity, respectively, and $(v_x)_i$ and $(v_z)_i$ are the maximum values of the relative lateral and vertical velocity of mass m_i.

Letting m_o be total mass of the structure and base mat, the energy required to overturn the structure is equal to

$$E_o = m_o gh + W_p - W_b \qquad (3.7\text{-}12)$$

where h is the height to which the center of mass of the structure must be lifted to reach the overturning position, g is the gravity constant, and W_p and W_b are the energy components caused by the effect of embedment and buoyance, respectively. Because the structure may not be a symmetrical one, the value of h is computed with respect to the edge that is nearer to the center of mass. The structure is defined as stable against overturning when the ratio E_o to E_s is no less than 1.1 for the SSE in combination with other appropriate loads.

3.7.2.15 Analysis Procedure for Damping

In a linear dynamic analysis using a modal superposition approach, the procedure to be used to properly account for damping in different elements of a coupled system model is as follows:

(1) The structural percent critical damping of the various structural elements of the model is first specified. Each value is referred to as the damping ratio (C_j) of a particular component which contributes to the complete stiffness of the system.

(2) An eigenvalue analysis of the linear system model is performed. This results in the eigenvector matrices (ϕ_i) which are normalized and satisfy the orthogonality conditions:

$$\phi_i^T K \phi_i = \omega_i^2, \text{ and } \phi_i^T K \phi_j = 0 \text{ for } i \neq j \qquad (3.7\text{-}13)$$

where

K = Stiffness matrix

ω_i = Circular natural frequency associated with mode i

ϕ_i^T = Transpose of i^{th} mode eigenvector ϕ_i

Matrix ϕ contains all translational and rotational coordinates.

(3) Using the strain energy of the individual components as a weighting function, the following equation is derived to obtain a suitable damping ratio (β_i) for mode i:

$$\beta_i = \frac{1}{\omega_i^2} \sum_{j=1}^{N} [C_j(\phi_i^T K \phi_i)_j] \qquad (3.7\text{-}14)$$

where

β_i = Modal damping coefficient for i^{th} mode

N = Total number of structural elements

ϕ_i = Component of i^{th} mode eigenvector corresponding to j^{th} element

ϕ_i^T = Transpose of ϕ_i defined above

C_j = Percent critical damping associated with element j

K = Stiffness matrix of element j

ω_i = Circular natural frequency of mode i

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods

This subsection discusses the methods by which Seismic Category I subsystems and components are qualified to ensure the functional integrity of the specific operating requirements which characterize their Seismic Category I designation.

In general, one of the following five methods of seismically qualifying the equipment is chosen based upon the characteristics and complexities of the subsystem:

(1) Dynamic analysis

(2) Testing procedures

(3) Equivalent static load method of analysis

(4) A combination of (1) and (2)

(5) A combination of (2) and (3)

Equivalent static load method of subsystem analysis is described in Subsection 3.7.3.5.

Appropriate design response spectra are furnished to the manufacturer of the equipment for seismic qualification purposes. Additional information such as input time history is also supplied only when necessary.

When analysis is used to qualify Seismic Category I subsystems and components, the analytical techniques must conservatively account for the dynamic nature of the subsystems or components.

[*The dynamic analysis of Seismic Category I subsystems and components is accomplished using the response spectrum or time-history approach. Time-history analysis is performed using either the direct integration method of the modal superposition method. The time-history technique described in Subsection 3.7.2.1.1 generates time histories at various support elevations for use in the analysis of subsystems and equipment. The structural response spectra curves are subsequently generated from the time-history accelerations.*]*

At each level of the structure where vital components are located, three orthogonal components of floor response spectra (two horizontal and one vertical) are developed. The floor response spectrum is smoothed and envelopes all calculated response spectra from different site soil conditions. The response spectra are peak broadened ±15%.

For vibrating systems and their supports, two general methods are used to obtain the solution of the equations of dynamic equilibrium of a multi-degree-of-freedom model. The first is the method of modal superposition described in Subsection 3.7.2.1.2. The second method of dynamic analysis is the direct integration method. The solution of the equations of motion is obtained by direct step-by-step numerical integration. The numerical integration time step, Δt, must be sufficiently small to accurately define the dynamic excitation and to render stability and convergency of the solution up to the highest frequency of significance. [*The integration time step is considered acceptable when smaller time steps introduce no more than a 10% error in the total dynamic response. For most of the commonly used numerical integration methods (such as Newark β-method and Wilson θ-method), the maximum time step is limited to one-tenth of the smallest period of interest.*]* The smallest period of interest is generally the reciprocal of the analysis cutoff frequency.

[*When the time-history method of analysis is used, the time-history data is broadened plus and minus 15% of Δt in order to account for modeling uncertainties.*]* For loads such as safety-relief valve blowdown, tests have been performed which confirm the conservatism of the analytical results. Therefore, for these loads the calculated force time-histories are not broadened plus and minus 15% of Δt.

Piping modeling and dynamic analysis are described in Subsection 3.7.3.3.1.

When testing is used to qualify Seismic Category I subsystems and components, all the loads normally acting on the equipment are simulated during the test. The actual mounting of the equipment is also simulated or duplicated. Tests are performed by

supplying input accelerations to the shake table to such an extent that generated test response spectra (TRS) envelope the required response spectra.

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

For certain Seismic Category I equipment and components where dynamic testing is necessary to ensure functional integrity, test performance data and results reflect the following:

(1) Performance data of equipment which has been subjected to dynamic loads equal to or greater than those experienced under the specified seismic conditions.

(2) Test data from previously tested comparable equipment which has been subjected under similar conditions to dynamic loads equal to or greater than those specified.

(3) Actual testing of equipment in accordance with one of the methods described in Subsection 3.9.2.2 and Section 3.10.

3.7.3.2 Determination of Number of Earthquake Cycles

[*The SSE is the only design earthquake considered for the ABWR Standard Plant.*]* To account for the cyclic effects of the more frequent occurrences of lesser earthquakes and their aftershocks, [*the fatigue evaluation for ASME Code Class 1, 2, and 3 components and core support structures takes into consideration two SSE events with 10 peak stress cycles per event for a total of 20 full cycles of the peak SSE stress.*]* This is equivalent to the cyclic load basis of one SSE and five OBE events as currently recommended in the SRP 3.9.2. [*Alternatively, a number of fractional vibratory cycles equivalent to 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE-344.*]*

For equipment seismic qualification performed in accordance with IEEE-344 as endorsed by Regulatory Guide 1.100, the equivalent seismic cyclic loads are five 0.5 SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five 0.5 SSE events may be used in accordance with Appendix D of IEEE-344 when followed by one full SSE.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Modeling of Piping Systems

3.7.3.3.1.1 Summary

To predict the dynamic response of a piping system to the specified forcing function, the dynamic model must adequately account for all significant modes. Careful selection must be made of the proper response spectrum curves and proper location of anchors in order to separate Seismic Category I from non-Category I piping systems.

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

3.7.3.3.1.2 Selection of Mass Points

[*Mathematical models for Seismic Category I piping systems are constructed to reflect the dynamic characteristics of the system. The continuous system is modeled as an assemblage of pipe elements supported by hangers, guides, anchors, struts and snubbers. Pipe and hydrodynamic masses are lumped at the nodes and are connected by weightless elastic beam elements which reflect the physical properties of the corresponding piping segment. The node points are selected to coincide with the locations of large masses, such as valves, pumps and motors, and with locations of significant geometry change. All pipe-mounted equipment, such as valves, pumps and motors, are modeled with lumped masses connected by elastic beam elements which reflect the physical properties of the pipe-mounted equipment. The torsional effects of valve operators and other pipe-mounted equipment with offset centers of gravity with respect to the piping center line are included in the mathematical model. On straight runs, mass points are located at spacings no greater than the span which would have a fundamental frequency equal to the cutoff frequency stipulated in Section 3.7 when calculated as a simply supported beam with uniformly distributed mass.*]*

3.7.3.3.1.3 Selection of Spectrum Curves

In selecting the spectrum curve to be used for dynamic analysis of a particular piping system, a curve is chosen which most closely describes the accelerations existing at the end points and restraints of the system. The procedures for decoupling small branch lines from the main run of Seismic Category I piping systems when establishing the analytical models to perform seismic analysis are as follows:

(1) [*The small branch lines are decoupled from the main runs if the ratio of run to branch pipe moment of inertia is 25 to 1, or more.*]*

(2) The stiffness of all the anchors and its supporting steel is large enough to effectively decouple the piping on either side of the anchor for analytic and code jurisdictional boundary purposes. The RPV is very stiff compared to the piping system and, therefore, it is modeled as an anchor. Penetration assemblies (head fittings and penetration sleeve pipe) are very stiff compared to the piping system and are modeled as anchors.

The stiffness matrix at the attachment location of the process pipe (i.e., main steam, RHR supply and return, RCIC, etc.) head fitting is sufficiently high to decouple the penetration assembly from the process pipe. Previous analysis indicates that a satisfactory minimum stiffness for this attachment point is equal to the stiffness in bending and torsion of a cantilevered pipe section of the same size as the process pipe and equal in length to three times the process pipe outer diameter.

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* See Subsection 3.9.1.7.

For a piping system supported at two or more points located at different elevations in the building, the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple support excitation analysis methods may be used where distinct response spectra for each individual support are applied at corresponding piping attachment points. Finally, the worst single floor response spectrum selected from a set of floor response spectra obtained at various floors may be applied identically to all floors provided it envelops the other floor response spectra in the set.

3.7.3.3.1.4 A Dynamic Analysis of Seismic Category I, Decoupled Branch Pipe

The dynamic analysis of Seismic Category I, decoupled branch pipe is performed by either the equivalent static method or by one of the dynamic analysis methods described in Tier 2. In addition, small bore branch pipe may be designed and analyzed in accordance with a small bore pipe manual in accordance with the requirements of Subsection 3.7.3.8.1.9.

The response spectra used for the dynamic analysis or for determining the static input load when the equivalent static method is used will be selected as follows:

(1) The response spectra will be based on the building or structure elevation of the branch line connection to the pipe run and the elevation of the branch line anchors and restraints.

(2) The response spectra will not be less than the envelope of the response spectra used in the dynamic analysis of the run pipe.

(3) If the location of branch connection to the run pipe is more than three run pipe diameters from the nearest run pipe seismic restraint, amplification by the run pipe will be accounted for.

When the equivalent static analysis method is used, the horizontal and vertical load coefficients C_h and C_v applied to the response spectra accelerations will conform with Subsection 3.7.3.8.1.5.

The relative anchor motions to be used in either static or dynamic analysis of the decoupled branch pipe shall be as follows:

(1) The inertial displacements only, as determined from analysis of the run pipe, may be applied to the branch pipe if the relative differential building movements of the large pipe supports and the branch pipe supports are less than 0.16 cm.

(2) [*If the relative differential building movements of the large pipe supports and the branch pipe supports are more than 0.16 cm, motion of the restraints and anchors of the branch pipe must be considered in addition to the inertial displacement of the run pipe.*]*

3.7.3.3.1.5 Selection of Input Time-Histories

In selecting the acceleration time-history to be used for dynamic analysis of a piping system, the time-history chosen is one which most closely describes the accelerations existing at the piping support attachment points. For a piping system supported at two or more points located at different elevations in the building, the time-history analysis is performed using the independent support motion method where acceleration time histories are input at all of the piping structural attachment points.

3.7.3.3.1.6 Modeling of Piping Supports

[*Snubbers are modeled with an equivalent stiffness which is based on dynamic tests performed on prototype snubber assemblies or on data provided by the vendor. Struts are modeled with a stiffness calculated based on their length and cross-sectional properties. The stiffness of the supporting structure for snubbers and struts is included in the piping analysis model, unless the supporting structure can be considered rigid relative to the piping. The supporting structure can be considered as rigid relative to the piping as long as the criteria specified in Subsection 3.7.3.3.4 are met.*

Anchors at equipment such as tanks, pumps and heat exchangers are modeled with calculated stiffness properties. Mass effects will be included for equipment which have a fundamental frequency less than the dynamic analysis cutoff frequency. A simplified model of the equipment is included in the piping system model. Frame type pipe supports are modeled as described in Subsection 3.7.3.3.4.]*

3.7.3.3.1.7 Modeling of Special Engineered Pipe Supports

[*Modifications to the normal linear-elastic piping analysis methodology used with conventional pipe supports are required to calculate the loads acting on the supports and on the piping components when the special engineered supports, described in Subsection 3.9.3.4.1(6), are used. These modifications are needed to account for greater damping of the energy absorbers and the non-linear behavior of the limit stops. If these special devices are used, the modeling and analytical methodology will be in accordance with methodology accepted by the regulatory agency at the time of certification or at the time of application, per the discretion of the COL applicant. In addition, the information required by Regulatory Guide 1.84 will be provided to the regulatory agency.*]*

3.7.3.3.1.8 Response Spectra Amplification at Support Attachment Points

[*The drywell equipment and pipe support structure (DEPSS) should meet the criteria given in Subsection 3.7.3.3.4. If this criteria can not be met, the COL applicant will generate the ARS at piping attachment points considering the DEPSS as part of the structure using the dynamic*

* See Subsection 3.9.1.7.

analysis methods described in Subsection 3.7.2, or will analyze the piping systems considering the DEPSS as part of the pipe support.]*

3.7.3.3.2 Modeling of Equipment

For dynamic analysis, Seismic Category I equipment is represented by lumped-mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

(1) The number of modes of a dynamic system is controlled by the number of masses used; therefore, the number of masses is chosen so that all significant modes are included. The modes are considered as significant if the corresponding natural frequencies are less than 33 Hz and the stresses calculated from these modes are greater than 10% of the total stresses obtained from lower modes. This approach is acceptable provided at least 90% of the loading/inertia is contained in the modes used. Alternately, the number of degrees of freedom are taken more than twice the number of modes with frequencies less than 33 Hz.

(2) Mass is lumped at any point where a significant concentrated weight is located (e.g., the motor in the analysis of pump motor stand, the impeller in the analysis of pump shaft, etc).

(3) If the equipment has free-end overhang span with flexibility significant compared to the center span, a mass is lumped at the overhang span.

(4) When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to lower the natural frequencies of the equipment because the equipment frequencies are in the higher spectral range of the response spectra. Similarly, in the case of live loads (mobile) and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen to yield the lowest frequency content for the system. This ensures conservative dynamic loads, since the equipment frequencies are such that the floor spectra peak is in the lower frequency range. If not, the model is adjusted to give more conservative results.

3.7.3.3.3 Field Location of Supports and Restraints

The field location of seismic supports and restraints for Seismic Category I piping and piping system components is selected to satisfy the following two conditions:

(1) The location selected must furnish the required response to control stress within allowable limits.

* See Subsection 3.9.1.7.

(2) Adequate building strength and stiffness for attachment of the component supports must be available.

The final location of seismic supports and restraints for Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraining devices is made to assure that their location and characteristics are consistent with the dynamic and static analyses of the system.

3.7.3.3.4 Analysis of Frame Type Supports

[*The design loads on frame type pipe supports include (a) loads transmitted to the support by the piping response to thermal expansion, dead weight, and the inertia and anchor motion effects, (b) support internal loads caused by the weight, thermal and inertia effects of loads of the structure itself, and (c) friction loads caused by pipe sliding on the support.*]* To calculate the frictional force acting on the support, dynamic loads that are cyclic in nature need not be considered. [*The following static coefficients of friction will be used in the analysis: 0.80 for steel on steel, and 0.15 for lubricated slide plates.*]* To determine the response of the support structure to applied dynamic loads, the equivalent static load method of analysis described in Subsection 3.7.3.8.1.5 may be used. The loads transmitted to the support by the piping will be applied as static loads acting on the support.

As in the case of other supports, the forces the piping places on the frame-type support are obtained from an analysis of the piping. Nonlinear analysis methods to account for gaps between pipe and supports subjected to high frequency vibration loads, such as suppression pool loads, will not be used. In the analysis of the piping the stiffness of the frame-type supports shall be included in the piping analysis model, unless the support can be shown to be rigid. The frame-type supports may be modeled as rigid restraints, provided they are designed so the maximum service level D deflection in the direction of the applied load is less than 3.2 mm, and [*providing the total gap or diametrical clearance between the pipe and frame support is between 1.6 mm and 4.76 mm when the pipe is in either the hot or cold condition.*]* For a frame-type support to be considered rigid, it shall be at least 200 times as stiff as the piping. The piping stiffness is calculated using the following equation:

$$Kp = \frac{EI}{L^3}$$

E = Modulus of elasticity of pipe

I = Moment of inertia of pipe

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

L = The suggested pipe support spacing in Table NF-3611-1 of ASME Code Section III

3.7.3.4 Basis of Selection of Frequencies

Where practical, in order to avoid adverse resonance effects, equipment and components are designed/selected such that their fundamental frequencies are outside the range of 1/2 to twice the dominant frequency of the associated support structures. Moreover, in any case, the equipment is analyzed and/or tested to demonstrate that it is adequately designed for the applicable loads considering both its fundamental frequency and the forcing frequency of the applicable support structure.

All frequencies in the range of 0.25 to 33 Hz are considered in the analysis and testing of structures, systems, and components. These frequencies are excited under the seismic excitation.

If the fundamental frequency of a component is greater than or equal to 33 Hz, it is treated as seismically rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered as they represent very flexible structures and are not encountered in this plant.

The frequency range between 0.25 Hz and 33 Hz covers the range of the broad band response spectrum used in the design.

3.7.3.5 Use of Equivalent Static Load Methods of Analysis

See Subsection 3.7.3.8.1.5 for equivalent static load analysis method.

3.7.3.6 Three Components of Earthquake Motion

[*The total seismic response is predicted by combining the response calculated from the two horizontal and the vertical analysis.*

When the response spectrum method or static coefficient is used, the method for combining the responses due to the three orthogonal components of seismic excitation is as follows:

$$R_i = \left[\sum_{j=1}^{3} R_{ij}^2\right]^{1/2} \tag{3.7-15}$$

where

R_{ij} = *Maximum, coaxial seismic response of interest (e.g., displacement, moment, shear, stress, strain) in directions i due to earthquake excitation in direction j, (j = 1, 2, 3).*

R_i = *Seismic response of interest in i direction for design (e.g., displacement, moment, shear, stress, strain) obtained by the SRSS rule to account for the nonsimultaneous occurrence of the* R_{ij}'s.

When the time-history method of analysis is used and separate analyses are performed for each earthquake component, the total combined response for all three components shall be obtained using the SRSS method described above to combine the maximum codirectional responses from each earthquake component. The total response may alternatively be obtained, if the three component motions are mutually statistically independent, by algebraically adding the codirectional responses calculated separately for each component at each time step.

When the time-history analysis is performed by applying the three component motions simultaneously, the combined response is obtained directly by solution of the equations of motion. This method of combination is applicable only if the three component motions are mutually statistically independent.]*

3.7.3.7 Combination of Modal Response

When the response spectrum method of analysis is used, the modal responses for modes below the cutoff frequency (specified in Section 3.7) are combined in accordance with the methods given in Subsection 3.7.3.7.1. The responses associated with higher frequency modes (above cutoff frequency) are calculated and combined with the low frequency modal responses according to the procedure described in Subsection 3.7.3.7.2. These methods and procedures are applicable for seismic loads as well as for loads with higher frequency input such as suppression pool dynamic loads.

3.7.3.7.1 Modes Below the Cutoff Frequencies

[*When the response spectrum method of modal analysis is used, contributions from all modes, except the closely spaced modes (i.e., the difference between any two natural frequencies is equal to or less than 10%) are combined by the SRSS combination of modal responses. This is defined mathematically as:*

$$R = \sqrt{\sum_{i=1}^{N} (R_i)^2} \qquad (3.7\text{-}16)$$

where

R = *Combined response*

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

R_i = *Response to the i^{th} mode*

N = *Number of modes considered in the analysis*

Closely spaced modes are combined by the grouping method described in Regulatory Guide 1.92.

An alternate to the grouping method presented in Regulatory Guide 1.92 is the ten percent method as following:

$$R = \left[\sum_{i=1}^{N} R_i^2 + 2\Sigma |R_l R_m| \right]^{1/2} \quad (3.7\text{-}17)$$

where the second summation is to be done on all l and m modes whose frequencies are closely spaced to each other.

In lieu of the grouping method and the ten percent method, the double sum method may also be used. This method, as defined in Regulatory Guide 1.92, is mathematically:

$$R = \left(\sum_{k=1}^{N} \sum_{s=1}^{N} |R_k R_s| \varepsilon_{ks} \right)^{1/2} \quad (3.7\text{-}18)$$

where

R = *Representative maximum value of a particular response of a given element to a given component of excitation*

R_k = *Peak value of the response of the element due to the k^{th} mode*

N = *Number of significant modes considered in the modal response combination*

R_s = *Peak value of the response of the element attributed to s^{th} mode*

where

$$\varepsilon_{ks} = \left[1 + \left\{ \frac{(\omega'_k - \omega'_s)}{(\beta'_k \omega_k + \beta'_s \omega_s)} \right\}^2 \right]^{-1} \quad (3.7\text{-}19)$$

in which

$$\omega_k' = \omega_k[1 - \beta_k^2]^{1/2}$$

$$\beta_k' = \beta_k + \frac{2}{t_d \omega_k}$$

where ω_k *and* β_k *are the modal frequency and the damping ratio in the kth mode, respectively, and* t_d *is the duration of the earthquake.*]*

3.7.3.7.2 Methodologies Used to Account for High-Frequency Modes

[*Sufficient modes are to be included in the dynamic analysis to ensure that the inclusion of additional modes does not result in more than 10% increase in responses. To satisfy this requirement, the responses associated with high-frequency modes are combined with the low-frequency modal responses. High-frequency modes are those modes with frequencies greater than the dynamic analysis cutoff frequency specified in Section 3.7.*

For modal combination involving high-frequency modes, the following procedure applies:

- ***Step 1****—Determine the modal responses only for those modes that have natural frequencies less than that at which the spectral acceleration approximately returns to the ZPA of the input response spectrum (33 Hz for seismic). Combine such modes in accordance with the methods described in Subsection 3.7.3.7.1.*

- ***Step 2****—For each degree of freedom (DOF) included in the dynamic analysis, determine the fraction of DOF mass included in the summation of all of the modes included in Step 1. This fraction* d_i *for each* DOF_i *is given by:*

$$d_i = \sum_{n=1}^{N} S_n x \phi_{n,i} \qquad (3.7\text{-}20)$$

where

n	=	*Order of the mode under consideration*
N	=	*Number of modes included in Step 1*
$\phi_{n,i}$	=	*Mass-normalized mode shape for mode n and* DOF_i

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

S_n = Participation factor for mode n (see Eq. 3.7-23 for expression)

Next, determine the fraction of DOF mass not included in the summation of these modes:

$$e_i = |d_i - \delta_{ij}| \tag{3.7-21}$$

where δ_{ij} is the Kronecker delta, which is one if DOF_i is in the direction of the input motion and zero if DOF_i is a rotation or not in the direction of the input motion. If, for any DOF_i, the absolute value of this fraction e_i exceeds 0.1, one should include the response from higher modes with those included in Step 1.

- ***Step 3**—Higher modes can be assumed to respond in phase with the ZPA and, thus, with each other; hence, these modes are combined algebraically, which is equivalent to pseudo-static response to the inertial forces from these higher modes excited at the ZPA. The pseudo-static inertial force associated with the summation of all higher modes for each DOF_i is given by:*

$$P_i = ZPAxM_ixe_i \tag{3.7-22}$$

where P_i is the force or moment to be applied at DOF_i, and M_i is the mass or mass moment of inertia associated with DOF_i. The system is then statically analyzed for this set of pseudo-static inertial forces applied to all of the degrees of freedom to determine the maximum responses associated with high-frequency modes not included in Step 1.

- ***Step 4**—The total combined response to high-frequency modes (Step 3) is combined by the SRSS method with total combined response from lower-frequency modes (Step 1) to determine the overall peak responses.*

This procedure requires the computation of individual modal responses only for lower-frequency modes (below the ZPA). Thus, the more difficult higher-frequency modes need not be determined. The procedure ensures inclusion of all modes of the structural model and proper representation of DOF masses.

In lieu of the above procedure, an alternative method is as follows. Modal responses are computed for enough modes to ensure that the inclusion of additional modes does not increase the total response by more than 10%. Modes that have natural frequencies less than that at which the spectral acceleration approximately returns to the ZPA are combined in accordance with *Regulatory Guide 1.92. Higher-mode responses are combined algebraically (i.e., retain sign) with each other. The absolute value of the combined higher modes is then added directly to the total response from the combined lower modes.*]*

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

3.7.3.8 Analytical Procedure

3.7.3.8.1 Qualification by Analysis

3.7.3.8.1.1 General

The methods used in seismic analysis vary according to the type of subsystems and supporting structure involved. The following possible cases are defined along with the associated analytical methods used.

3.7.3.8.1.2 Rigid Subsystems with Rigid Supports

If all natural frequencies of the subsystem are greater than 33 Hz, the subsystem is considered rigid and analyzed statically as such. In the static analysis, the seismic forces on each component of the subsystem are obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate maximum floor acceleration.

3.7.3.8.1.3 Rigid Subsystems with Flexible Supports

If it can be shown that the subsystem itself is a rigid body (e.g., piping supported at only two points) while its supports are flexible, the overall subsystem is modeled as a single-degree-of-freedom subsystem consisting of an effective mass and spring.

The natural frequency of the subsystem is computed and the acceleration determined from the floor response spectrum curve using the appropriate damping value. A static analysis is performed using 1.5 times the acceleration value. In lieu of calculating the natural frequency, the peak acceleration from the spectrum curve may be used.

If the subsystem has no definite orientation, the excitation along each of three mutually perpendicular axes is aligned with respect to the system to produce maximum loading. The excitation in each of the three axes is considered to act simultaneously. The responses are combined by the SRSS method.

3.7.3.8.1.4 Flexible Subsystems

If the subsystem has more than two supports, it cannot be considered a rigid body and must be modeled as a multi-degree-of-freedom subsystem.

The subsystem is modeled as discussed in Subsection 3.7.3.3.1 in sufficient detail (i.e., number of mass points) to ensure that the lowest natural frequency between mass points is greater than 33 Hz. The mathematical model is analyzed using a time-history analysis technique or a response spectrum analysis approach. After the natural frequencies of the subsystem are obtained, a stress analysis is performed using the loads obtained from the dynamic analysis.

In a response spectrum dynamic analysis, modal responses are combined as described in Subsection 3.7.3.7. In a response spectrum or time-history dynamic analysis, responses due to the three orthogonal components of seismic excitation are combined as described is Subsection 3.7.3.6.

3.7.3.8.1.5 Static Analysis

A static analysis is performed in lieu of a dynamic analysis by applying the following forces at the concentrated mass locations (nodes) of the analytical model of the seismic subsystem:

(1) Horizontal static load, $F_h = C_h W$, in one of the horizontal principal directions.

(2) Equal static load, F_h, in the other horizontal principal direction.

(3) Vertical static load, $F_v = C_v W$:

where

C_h, C_v = Multipliers of the gravity acceleration, g, determined from the horizontal and vertical floor response spectrum curves, respectively. (They are functions of the period and the appropriate damping of the piping system)

W = Weight at node points of the analytical model

For special case analyses, C_h and C_v may be taken as:

(1) 1.0 times the zero-period acceleration of the response spectrum of subsystems described in Subsection 3.7.3.8.1.2.

(2) 1.5 times the value of the response spectrum at the determined frequency for subsystems described in Subsections 3.7.3.8.1.3 and 3.7.3.8.1.4.

(3) 1.5 times the peak of the response spectrum for subsystems described in Subsections 3.7.3.8.1.3 and 3.7.3.8.1.4.

An alternate method of static analysis which allows for simpler technique with added conservatism is acceptable. No determination of natural frequencies is made, but rather the response of the subsystem is assumed to be the peak of the appropriate response spectrum at a conservative and justifiable value of damping. The response is then multiplied by a static coefficient of 1.5 to take into account the effects of both multifrequency excitation and multimodal response.

3.7.3.8.1.6 Dynamic Analysis

[*The dynamic analysis procedure using the response spectrum method is provided as follows:*

(1) The number of node points and members is indicated. If a computer program is utilized, use the same order of number in the computer program input. The mass at

each node point, the length of each member, elastic constants, and geometric properties are determined.

(2) *The dynamic degrees of freedom according to the boundary conditions are determined.*

(3) *The dynamic properties of the subsystem (i.e., natural frequencies and mode shapes) are computed.*

(4) *Using a given direction of earthquake motion, the modal participation factors,* s_j*, for each mode are calculated:*

$$s_j = \frac{\sum_{i=1}^{N} M_i \phi'_{ij}}{\sum_{i=1}^{N} M_i \phi_{ij}^2} \qquad (3.7\text{-}23)$$

where

M_i = i^{th} *mass*

ϕ'_{ij} = *Component of* ϕ_{ij} *in the earthquake direction*

ϕ_{ij} = i^{th} *characteristic displacement in the* j^{th} *mode*

s_j = *Modal participation factor for the* j^{th} *mode*

N = *Number of masses.*

(5) *Using the appropriate response spectrum curve, the spectral acceleration,* r_a*, for the* j^{th} *mode as a function of the* j^{th} *mode natural frequency and the damping of the system is determined.*

(6) *The maximum modal acceleration at each mass point, i, in the model is computed as follows:*

$$a_{ij} = s_j r_{aj} \phi_{ij} \qquad (3.7\text{-}24)$$

where

a_{ij} = *Acceleration of the i^{th} mass point in the j^{th} mode.*

(7) *The maximum modal inertia force at the i^{th} mass point for the j^{th} mode is calculated from the equation:*

$$F_{ij} = M_i a_{ij} \tag{3.7-25}$$

(8) *For each mode, the maximum inertia forces are applied to the subsystem model, and the modal forces, shears, moments, stresses, and deflections are determined.*

(9) *The modal forces, shears, moments, stresses, and deflections for a given direction are combined in accordance with Subsection 3.7.3.7.*

(10) *Steps (5) through (9) are performed for each of the three earthquake directions.*

(11) *The seismic force, shear, moment, and stress resulting from the simultaneous application of the three components of earthquake loading are obtained in the following manner:*

$$R = \sqrt{R_x^2 + R_y^2 + R_z^2} \tag{3.7-26}$$

R = *Equivalent seismic response quantity (force, shear, moment, stress, etc.)*

$R_x R_y R_z$ = *Colinear response quantities due to earthquake motion in the x, y, and z directions, respectively*]*

3.7.3.8.1.7 Damping Ratio

The damping ratio percentage of critical damping of subsystems corresponds to Regulatory Guide 1.61 or 1.84 (ASME Code Case N-411-1). The damping ratio is specified in Table 3.7-1.

[*ASME Code Case N-411-1 damping cannot be used for analyzing linear energy absorbing supports designed in accordance with ASME Code Case N-420.*]*

[*Strain energy weighted modal damping can also be used in the dynamic analysis. Strain energy weighting is used to obtain the modal damping coefficient due to the contributions of elements with different damping properties in the model. The element damping values are specified in Table 3.7-1. Strain energy weighted modal damping is calculated as specified in Subsection 3.7.2.15.*]†

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

† See Subsection 3.9.1.7. The change restriction applies only to piping design.

In direct integration analysis, damping is input in the form of α and β damping constants, which give the percentage of critical damping, λ as a function of the circular frequency, ω.

$$\lambda = \frac{\alpha}{2\omega} + \frac{\beta\omega}{2} \tag{3.7-27}$$

3.7.3.8.1.8 Effect of Differential Building Movements

In most cases, subsystems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event. The movements may range from insignificant differential displacements between rigid walls of a common building at low elevations to relatively large displacements between separate buildings at a high seismicity site.

Differential endpoint or restraint deflections cause forces and moments to be induced into the system. The stress thus produced is a secondary stress. It is justifiable to place this stress, which results from restraint of free-end displacement of the system, in the secondary stress category because the stresses are self-limiting and, when the stresses exceed yield strength, minor distortions or deformations within the system satisfy the condition which caused the stress to occur.

[*The earthquake thus produces a stress-exhibiting property much like a thermal expansion stress and a static analysis can be used to obtain actual stresses. The differential displacements are obtained from the dynamic analysis of the building. The displacements are applied to the anchors and restraints corresponding to the maximum differential displacements which could occur. The static analysis is made three times: once for one of the horizontal differential displacements, once for the other horizontal differential displacement, and once for the vertical.*

The inertia (primary) and displacement (secondary) loads are dynamic in nature and their peak values are not expected to occur at the same time. Hence, the combination of the peak values of inertia load and anchor displacement load is quite conservative. In addition, anchor movement effects are computed from static analyses in which the displacements are applied to produce the most conservative loads on the components. Therefore, the primary and secondary loads are combined by the SRSS method.]*

3.7.3.8.1.9 Design of Small Branch and Small Bore Piping

(1) [*Small branch lines are defined as those lines that can be decoupled from the analytical model used for the analysis of the main run piping to which the branch lines attach. As allowed by Subsection 3.7.3.3.1.3, branch lines can be decoupled when the ratio of run to branch pipe moment of inertia is 25 to 1, or greater. In addition to the moment of inertia criterion for acceptable decoupling, these small branch lines shall be designed with no concentrated masses, such as valves, in the first one-half span length from the main run pipe; and with sufficient flexibility to prevent restraint of movement of the main run pipe. The small branch line is*

considered to have adequate flexibility if its first anchor or restraint to movement is at least one-half pipe span in a direction perpendicular to the direction of relative movement between the pipe run and the first anchor or restraint of the branch piping. A pipe span is defined as the length tabulated in Table NF-3611-1, Suggested Piping Support Spacing, ASME B&PV Code Section III, Subsection NF. For branches where the preceding criteria for sufficient flexibility cannot be met, the applicant will demonstrate acceptability by using an alternative criteria for sufficient flexibility, or by accounting for the effects of the branch piping in the analysis of the main run piping.

(2) For small bore piping defined as piping 50A and less nominal pipe size, and small branch lines 50A and less nominal pipe size, as defined in (1) above, it is acceptable to use small bore piping handbooks in lieu of performing a system flexibility analysis, using static and dynamic mathematical models, to obtain loads on the piping elements and using these loads to calculate stresses per equations in NB, NC, and ND3600 in ASME Code Section III, whenever the following are met:

(a) The small bore piping handbook at the time of application is currently accepted by the regulatory agency for use on equivalent piping at other nuclear power plants.

(b) When the small bore piping handbook is serving the purpose of the Design Report it meets all of the ASME requirements for a piping design report. This includes the piping and its supports.

(c) Formal documentation exists showing piping designed and installed to the small bore piping handbook (1) is conservative in comparison to results from a detail stress analysis for all applied loads and load combinations using static and dynamic analysis methods defined in Subsection 3.7.3, (2) does not result in piping that is less reliable because of loss of flexibility or because of excessive number of supports, (3) satisfies required clearances around sensitive components.

The small bore piping handbook methodology will not be applied when specific information is needed on (a) magnitude of pipe and fittings stresses, (b) pipe and fitting cumulative usage factors, (c) accelerations of pipe-mounted equipment, or locations of postulated breaks and leaks.

The small bore piping handbook methodology will not be applied to piping systems that are fully engineered and installed in accordance with the engineering drawings.]*

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

[*Supports for ASME Section III instrumentation lines are analyzed in accordance with Subsection 3.7.3 and are designed in accordance with Subsection 3.9.3.4.*]*

3.7.3.8.1.10 Multi–Supported Equipment and Components with Distinct Inputs

[*For multi-supported systems (equipment and piping) analyzed by the response spectrum method for the determination of inertial responses, either of the following two input motions is acceptable:*

(1) Envelope response spectrum of all support points for each orthogonal direction of excitation, or

(2) Independent support motion (ISM) response spectrum at each support for each orthogonal direction of excitation.

When the ISM response spectrum method of analysis is used, the following conditions should be met:

(1) ASME Code Case N-411-1 damping is not used.

(2) A support group is defined by supports which have the same time-history input. This usually means all supports located on the same floor, or portions of a floor, of a structure.

(3) The responses due to motions of supports in two or more different groups are combined by the SRSS procedure.

In lieu of the response spectrum analysis, the time-history method of analysis subjected to distinct support motions may be used for multi-supported systems.]*

3.7.3.8.2 Qualification by Design by Rule

For distributive systems such as cable trays, conduits, and HVAC ducts, an alternative to qualification by analysis described in Subsection 3.7.3.8.1 is the design by rule method approved by the NRC at the time of COL application.

3.7.3.9 Multi-Supported Equipment and Components with Distinct Inputs

The procedure and criteria for analysis are described in Subsections 3.7.2.1.3, 3.7.3.3.1.3, and 3.7.3.8.1.10.

3.7.3.10 Use of Constant Vertical Static Factors

All Seismic Category I subsystems and components are subjected to a vertical dynamic analysis with the vertical floor spectra or time histories defining the input. A static analysis is performed in lieu of dynamic analysis if the peak value of the applicable floor spectra times a factor of 1.5 is used in the analysis. A factor of 1.0 instead of 1.5 can be used if the equipment is simple enough such that it behaves essentially as a single degree of freedom system. If the fundamental frequency of a component in the vertical direction is greater than or equal to 33 Hz, it is treated as seismically rigid and analyzed statically using the zero period acceleration (ZPA) of the response spectrum.

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are included for Seismic Category I subsystems similar to that for the piping systems discussed in Subsection 3.7.3.3.1.2.

3.7.3.12 Buried Seismic Category I Piping and Tunnels

All Seismic Category I utilities (i.e. piping, conduits, or auxiliary system components) that are routed underground are installed in reinforced concrete tunnels in direct contact with soil. The design and analysis follows engineering process specified in SRP 3.7.3 for "Seismic Category I Buried Piping, Conduits and Tunnels." The following approaches are considered in the design and analysis:

(1) The inertial effects due to an earthquake upon tunnel systems will be adequately accounted for in the design and analysis. Buried reinforced concrete tunnel systems (including contained piping and cabling), are sufficiently flexible relative to the surrounding or underlying soil, and it is assumed that the systems will follow essentially the displacements and deformations that the soil would have if the systems were absent. The enclosure of the tunnel needs to be designed to provide adequate dynamic clearance to its housing piping/cabling to avoid direct transmission of seismic in-ground accelerations and seismic in-ground displacements. When applicable, procedures, which take into account the phenomena of wave travel and wave reflection in compacting soil displacements from the ground displacements, are employed and the effects due to dynamic soil pressure, local soil settlements, soil arching, etc., are also considered in the design and analysis.

(2) The design response spectra for the underground piping are the horizontal and vertical design spectra at the ground surface given in Figures 3.7-1 and 3.7-2. These design spectra are constructed in accordance with Regulatory Guide 1.60. The piping analysis is performed using one of the methods described in Subsection 3.7.3.

(3) Since all underground Seismic Category I piping is contained in tunnels, its design, analysis and construction will be in accordance with the ASME Section III of the Boiler and Pressure Vessel Code and all sections of Tier 2 pertaining to Seismic Category I piping supports will be applied.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

[*In certain instances, non-Seismic Category I piping may be connected to Seismic Category I piping at locations other than a piece of equipment which, for purposes of analysis, could be represented as an anchor. The transition points typically occur at Seismic Category I valves which may or may not be physically anchored. Since a dynamic analysis must be modeled from pipe anchor point to anchor point, two options exist:*

(1) Specify and design a structural anchor at the Seismic Category I valve and analyze the Seismic Category I subsystem; or, if impractical to design an anchor.

(2) Analyze the subsystem from the anchor point in the Seismic Category I subsystem through the valve to either the first anchor point in the non-Seismic Category I subsystem; or for a distance such that there are at least two seismic restraints in each of the three orthogonal directions.]*

Where small, non-Seismic Category I piping is directly attached to Seismic Category I piping, it can be decoupled from Seismic Category I piping per Subsection 3.7.3.3.1.3.

[*Furthermore, non-Seismic Category I piping (particularly high energy piping as defined in Section 3.6) is designed to withstand the SSE to avoid jeopardizing adjacent Seismic Category I piping if it is not feasible or practical to isolate these two piping systems.*]*

3.7.3.14 Seismic Analysis for Reactor Internals

The modeling of RPV internals is discussed in Subsection 3.7.2.3.2. The damping values are given in Table 3.7-1. The seismic model of the RPV and internals is shown in Figure 3.7-32.

3.7.3.15 Analysis Procedures for Damping

Analysis procedures for damping are discussed in Subsections 3.7.2.15 and 3.7.3.8.1.7.

3.7.3.16 Analysis Procedure for Non-Seismic Structures in Lieu of Dynamic Analysis

For the design of non-seismic Category I structures, the procedures described in the Uniform Building Code (UBC) seismic design criteria shall be followed.

* See Subsection 3.9.1.7.

Where a structure is required to be designed to withstand a SSE, the following limitations apply:

(1) The seismic zone shall be "Zone 3".

(2) The structure shall be classified as "Essential Facility"; thereby using appropriate importance factors for wind and seismic.

(3) For dual systems (i.e., shear wall with braced steel frame), one of the two systems must be designed to be capable of carrying all of the seismic or wind loading without collapse. No credit will be given for the other for resisting lateral loads.

3.7.3.17 Methods for Seismic Analysis of Above-Ground Tanks

The seismic analysis of Category I above-ground tanks considers the following items:

(1) At least two horizontal modes of combined fluid-tank vibration and at least one vertical mode of fluid vibration are included in the analysis, taking into consideration the SSI effects as appropriate. The horizontal response analysis includes at least one impulsive mode in which the response of the tank shell and roof is coupled together with the portion of the fluid contents that move in unison with the shell, and the fundamental sloshing (convective) mode.

(2) The fundamental natural horizontal impulsive mode of vibration of the fluid-tank system is estimated giving due consideration to the flexibility of the supporting medium and to any uplifting tendencies for the tank. The rigid tank assumption is not made unless it can be justified. The horizontal impulsive-mode spectral acceleration, S_{a1} is then determined using this frequency and the appropriate damping for the fluid-tank system. Alternatively, the maximum spectral acceleration corresponding to the relevant damping may be used.

(3) Damping values used to determine the spectral acceleration in the impulsive mode are based upon the system damping associated with the tank shell material as well as with the soil-structure interaction (SSI).

(4) In determining the spectral acceleration in the horizontal convective mode, S_{a2}, the fluid damping ratio is 0.5% of critical damping unless a higher value can be substantiated by experimental results.

(5) The maximum overturning moment, M_o, at the base of the tank is obtained by the modal and spatial combination methods discussed in Subsections 3.7.3.7 and 3.7.3.6, respectively. The uplift tension resulting from M_o, is resisted either by tying the tank to the foundation with anchor bolts, etc., or by mobilizing enough fluid weight on a thickened base skirt plate. The latter method of resisting M_o, when used, must be shown to be conservative.

(6) The seismically induced hydrodynamic pressures on the tank shell at any level are determined by the modal and spatial combination methods discussed in Subsections 3.7.3.7 and 3.7.3.6 respectively. The maximum hoop forces in the tank wall are evaluated with due regard for the contribution of the vertical component of ground shaking. The beneficial effects of soil-structure interaction may be considered in this evaluation. The hydrodynamic pressure at any level is added to the hydrostatic pressure at that level to determine the hoop tension in the tank shell.

(7) Either the tank top head is located at an elevation higher than the slosh height above the top of the fluid or else is designed for pressures resulting from fluid sloshing against this head.

(8) At the point of attachment, the tank shell is designed to withstand the seismic forces imposed by the attached piping. An appropriate analysis is performed to verify this design.

(9) The tank foundation is designed to accommodate the seismic forces imposed on it. These forces include the hydrodynamic fluid pressures imposed on the base of the tank as well as the tank shell longitudinal compressive and tensile forces resulting from M_o.

(10) In addition to the above, a consideration is given to prevent buckling of tank walls and roof, failure of connecting piping, and sliding of the tank.

3.7.4 Seismic Instrumentation

3.7.4.1 Location and Description or Seismic Instrumentation

State-of-the-art solid-state digital instrumentation that will enable the prompt processing of the data at the plant site should be used. A triaxial time-history accelerometer should be provided at each of the following locations.

- One at the finished grade in the free-field
- Three in the reactor building: one located on the foundation mat at elevation –8.2m, one at floor elevation 12.3m and one at the operating floor at elevation 31.7m.
- Two in the control building: one on the foundation mat at elevation –8.2m and one at elevation 7.9m.

3.7.4.2 Seismic Instrumentation Operability and Characteristics

The seismic instrumentation should operate during all modes of plant operation, including periods of plant shutdown. The maintenance and repair procedures should provide for keeping the maximum number of instruments in service during plant operation and shutdown.

The design should include provisions for inservice testing. The instruments should be capable of periodic channel checks during normal plant operation and the capability for in-place functional testing. The instrumentation on the foundation and at elevations within the same building or structure should be interconnected for common starting and common timing, and the instrumentation should contain provisions for an external remote alarm to indicate actuation. The pre-event memory of the instrumentation should be sufficient to record the onset of the earthquake. It should operate continuously during the period in which the earthquake exceeds the seismic trigger threshold and for a minimum of 5 seconds beyond the last trigger level signal. The instrument should be capable of a minimum of 25 minutes of continuous recording. The acceleration sensors should have a dynamic range of 1000:1 zero to peak (i.e. 0.00lg to 1.0g) and the frequency range should be 0.20 Hz to 50 Hz.

The seismic instrumentation system is triggered by the accelerometer signals. The actuating level should be adjustable for a minimum of 0.005g to 0.2g. The trigger is actuated whenever the acceleration exceeds 0.01g. The initial setpoint may be changed (but shall not exceed 0.02g) once sufficient plant operating data have been obtained which indicate that a different setpoint would provide better system operation.

The instrumentation should be capable of on-line digital recording all components accelerometer signals. The digitized rate of the recorder should be at least 200 samples per second, the frequency band width should be at least from 0.20 Hz to 50 Hz, and the dynamic range should be 1000:1. The instrumentation should be capable of using the recorded signal to calculate the standardized cumulative absolute velocity (CAV) and the 5% of critically damped response spectrum.

The instruments should be capable of having routine channel checks, functional tests, and calibrations. The CAV shutdown threshold of 0.16g-seconds should be calibrated with the October, 1987 Whittier California earthquake record or an equivalent calibration record provided for this purpose by the manufacturer of the instrumentation. In the event that an earthquake is recorded at the plant site, all calibrations including that of the CAV will be performed to demonstrate that the system was functioning properly at the time of the earthquake.

3.7.4.3 Control Room Operator Notification

Activation of the seismic trigger causes an audible and visual annunciation in the control room to alert the plant operator that an earthquake has occurred.

3.7.4.4 Comparison of Measured and Predicted Response

Within four hours after the earthquake, the 5% damped response spectrum and the CAV for each of the three components of recorded data in the free field will be obtained and evaluated to determine if the shutdown criteria defined in EPRI reports NP-5930 (Reference 3.7-5), and NP-6695 (Reference 3.7-7) have been exceeded. The plant will be shut down when the recorded motion in the free field in any of the three directions (two horizontal and one vertical) exceeds both the response spectrum limit and the CAV limit as follows:

(1) The response spectrum limit is exceeded if:

(a) At frequencies between 2 and 10 Hz, the recorded response spectral accelerations of 5% damping exceed 0.33 of the corresponding SSE values or 0.2g, whichever is greater, or

(b) At frequencies between 1 and 2 Hz, the recorded response spectral velocity of 5% damping exceed 0.33 of the corresponding SSE values or 15.24 cm/s, whichever is greater.

(2) The CAV limit is exceeded if the CAV value calculated according to the procedures in EPRI report TR-100082 (Reference 3.7-6) is greater than 0.16 g-sec.

3.7.4.5 Inservice Surveillance

Each of the seismic instruments will be demonstrated operable by the performance of the channel check, channel calibration, and channel functional test operations. The channel checks will be performed every two weeks for the first three months of service after startup. After the initial three-month period and three consecutive successful checks, the channel checks will be performed on a monthly basis. The channel calibration will be performed during each refueling. The channel functional test will be performed every six months.

3.7.5 COL License Information

3.7.5.1 Seismic Design Parameters

To confirm the seismic design adequacy, the COL applicant shall demonstrate that the standardized design is applicable to the site according to the procedure specified in Subsection 2.3.1.2.

3.7.5.2 Pre-Earthquake Planning and Post-Earthquake Actions

The COL applicant shall submit to the NRC as part of its application the procedures it plans to use for pre-earthquake planning and post-earthquake actions. The procedures shall implement the seismic instrumentation program specified in Subsection 3.7.4 and follow the guidelines recommended in EPRI report NP-6695 (Reference 3.7-8), with the following exceptions:

(1) Section 3.1. Short-Term Actions

(a) Item 3. “Evaluation of Ground Motion Records”—There is a time limitation of four hours within which the licensee shall determine if the shutdown criterion has been exceeded. After an earthquake has been recorded at the site, the licensee shall provide a response spectrum calibration record and CAV calibration record to demonstrate that the system was functioning properly.

(b) Item 4. “Decision on Shutdown”—Exceedance of the EPRI criterion as amended in Subsection 3.7.4.4 of Tier 2 or observed evidence of significant damage as defined by EPRI NP-6695 shall constitute a condition for mandatory shutdown unless conditions prevent the licensee from accomplishing an orderly shutdown without jeopardizing the health and safety of the public.

(c) Add item 7. “Documentation”—The licensee shall record the chronology of events and control room problems while the earthquake evaluation is in progress.

(2) Section 4.3.1. Immediate Operator Actions. Add to the checks listed in this section a prompt check of the neutron flux monitoring instruments for stability of the reactor.

(3) Section 4.3.4. Pre-Shutdown Inspection. Exceeding the EPRI criterion or evidence of significant damage should constitute a condition for mandatory plant shutdown.

(4) Section 4.3.4.1. Safe Shutdown Equipment. In addition to the safe shutdown systems on this list, containment integrity must be maintained following an earthquake. Since the containment isolation valves may have malfunctioned during the earthquake, inspection of the containment isolation system is necessary to assure continued containment integrity.

3.7.5.3 Piping Analysis, Modeling of Piping Supports

The COL applicant shall provide justification for methods used other than those described in Subsection 3.7.3.3.1.6 for determining pipe support stiffness used in the piping analysis. The justification should include verification that the pipe support stiffness values are representative of the types of supports used in the piping system. The alternative approach used to determine pipe supports stiffness values and its bases should be submitted to the NRC staff for review and approval before its use (Subsection 3.7.3.3.1.6).

3.7.5.4 Assessment of Interaction Due to Seismic Effects

The COL applicant will describe the process for completion of the design of balance-of-plant and non-safety related systems to minimize II/I interactions and propose procedures for an inspection of the as-built plant for II/I interactions. (Subsection 3.7.2.8)

3.7.6 References

3.7-1 Not Used.

3.7-2 Not Used.

3.7-3 Not Used.

3.7-4 L. K. Liu, “Seismic Analysis of the Boiling Water Reactor”, symposium on seismic analysis of pressure vessel and piping components, First National Congress on Pressure Vessel and Piping, San Francisco, California, May 1971.

3.7-5 EPRI NP-5930, “A Criterion for Determining Exceedance of the Operating Basis Earthquake”, July 1988.

3.7-6 EPRI TR-100082, “Standardization of Cumulative Absolute Velocity”, December 1991.

3.7-7 EPRI NP-6695, “Guidelines for Nuclear Plant Response to an Earthquake”, December 1989.

3.7-8 P. Koss, “Seismic Testing of Electrical Cable Support Systems”, Structural Engineers of California Conference, San Diego, September 1979.

Table 3.7-1 Damping for Different Materials

Item	Percent Critical Damping SSE
Reinforced concrete structures	7
Welded structural assemblies	4
Steel frame structures	4
Bolted or riveted structural assemblies	7
Equipment	3
piping systems*	
- diameter greater than 300A nominal	3
- diameter less than or equal to 300A nominal	2
Reactor pressure vessel, support skirt, shroud head and separator	4
Guide tubes and CRD housings	2
Fuel	6
Cable trays	20 (max) (see Figure 3.7-27)
Conduits	7
HVAC ductwork	
– companion angle	7
– pocket lock	7
– welded	4

* Damping values of ASME Code Case N-411-1, alternative damping values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1, may be used as permitted by Regulatory Guide 1.84. These damping values are applicable in analyzing piping response for Seismic and other dynamic loads filtering through building structures in high frequencies range beyond 33 Hz.

Table 3.7-2 Natural Frequencies of the Reactor Building Complex in X Direction (0°–180° Axis)—Fixed Base Condition

Mode No.	Frequency (HZ)
1	4.14
2	4.53
3	7.71
4	9.01
5	9.60
6	10.10
7	11.53
8	12.72
9	13.44
10	13.58
11	14.64
12	15.60
13	17.46
14	18.00
15	18.95
16	22.01
17	22.72
18	24.31
19	25.48
20	26.11
21	27.08
22	28.20
23	29.84
24	30.94
25	33.16

Table 3.7-3 Natural Frequencies of the Reactor Building Complex in Y Direction (90°–270° Axis)—Fixed Base Condition

Mode No.	Frequency (HZ)
1	3.92
2	4.52
3	7.71
4	8.68
5	9.60
6	9.84
7	11.53
8	12.72
9	13.25
10	13.53
11	14.16
12	16.06
13	18.00
14	18.95
15	21.22
16	22.62
17	22.88
18	25.44
19	25.93
20	26.79
21	27.80
22	28.54
23	30.59
24	33.13

Table 3.7-4 Natural Frequencies of the Reactor Building Complex in Z Direction (Vertical)—Fixed Base Condition

Mode No.	Frequency (HZ)
1	7.79
2	9.53
3	10.69
4	11.50
5	12.05
6	13.30
7	14.12
8	15.59
9	20.69
10	28.41
11	28.93
12	32.32

Table 3.7-5 Natural Frequencies of the Control Building—Fixed Base Condition

	Frequency (Hz)		
Mode No.	X (0°–180°)	Y (90°–270°)	Z (Vertical)
1	5.59	6.72	13.52
2	15.91	16.24	15.86
3	29.22	23.76	15.93
4	30.85	35.20	15.97
5			22.38

Table 3.7-6 Natural Frequencies of the Radwaste Building—Fixed Base Condition

Mode No.	Frequency (HZ)	Direction
1	6.52	Y Horiz
2	7.19	X Horiz
3	11.98	Y Horiz
4	14.74	Z Vert
5	16.24	X Horiz
6	16.72	Y Horiz
7	21.49	X Horiz
8	26.28	Y Horiz
9	29.57	X Horiz
10	31.40	Z Vert

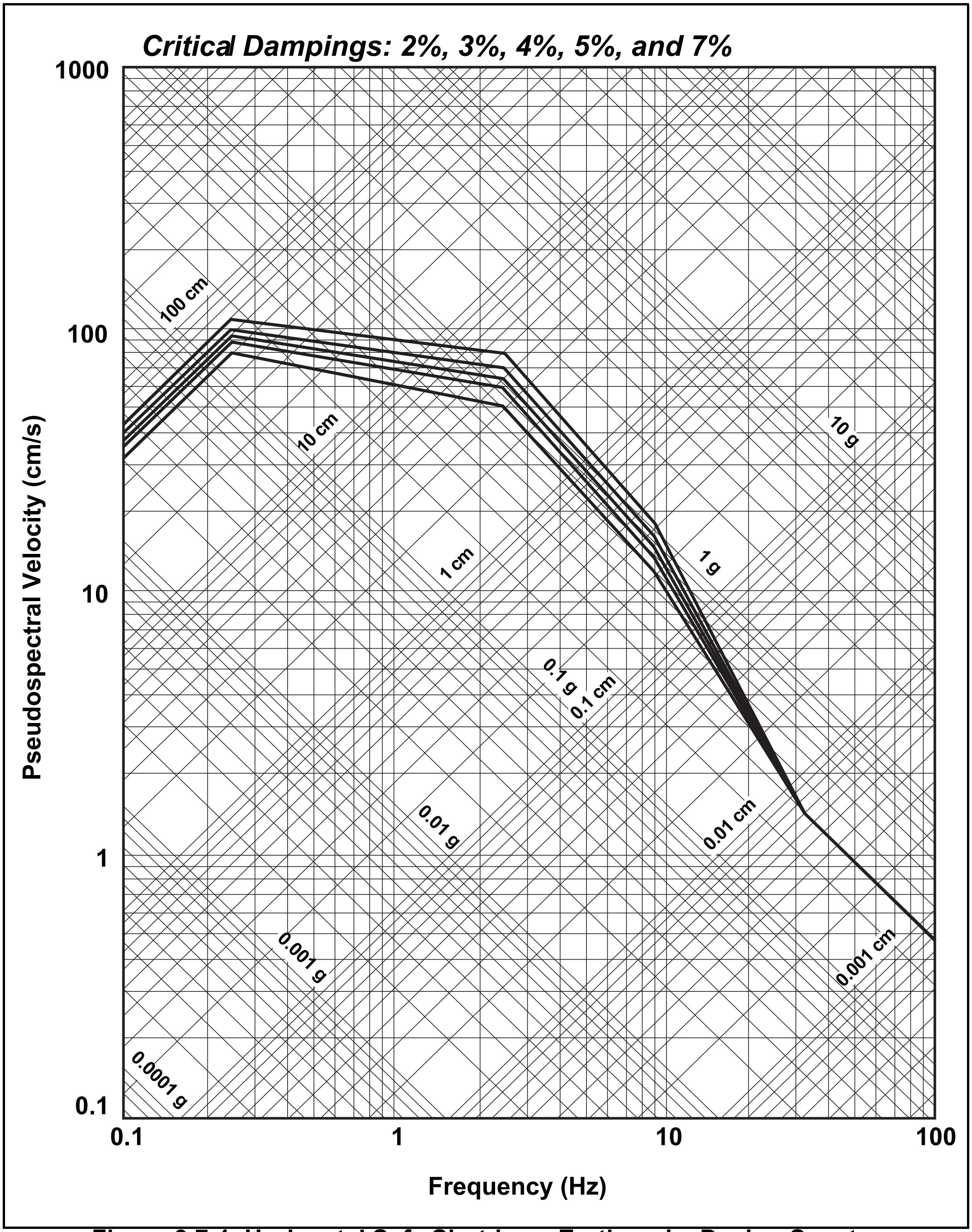

Figure 3.7-1 Horizontal Safe Shutdown Earthquake Design Spectra

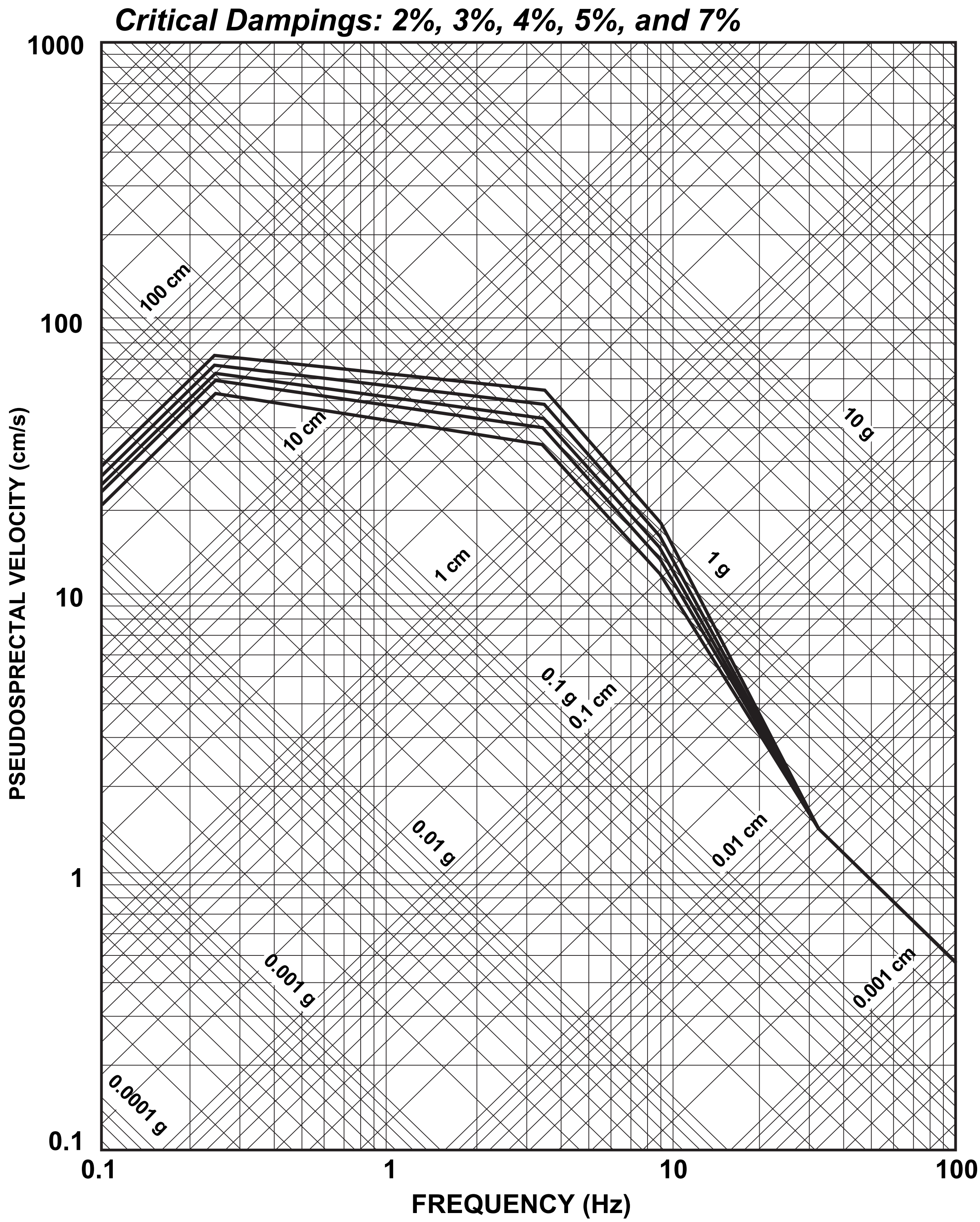

Figure 3.7-2 Vertical Safe Shutdown Earthquake Design Spectra

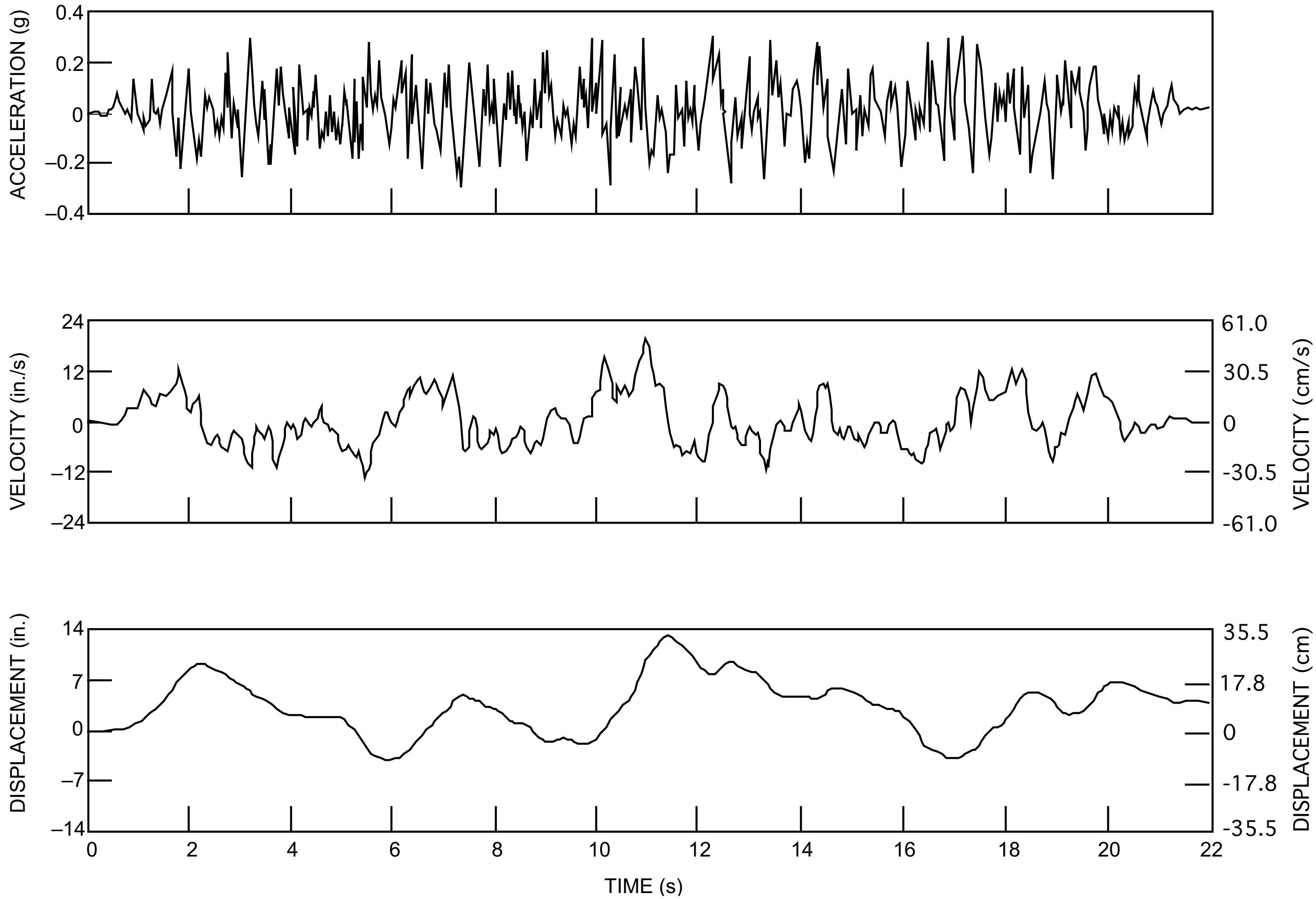

Figure 3.7-3 Horizontal, H1 Component Time History

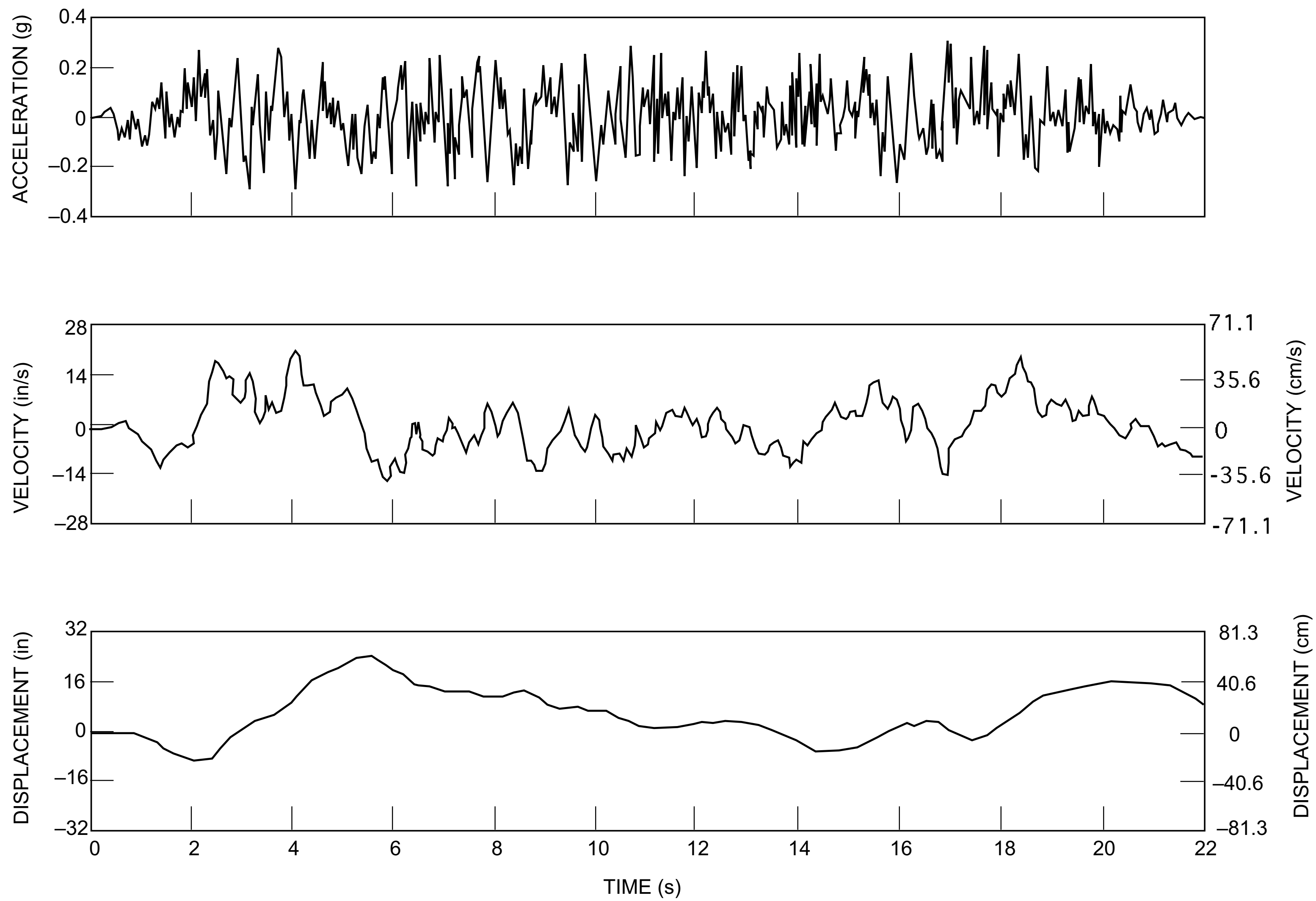

Figure 3.7-4 Horizontal, H2 Component Time History

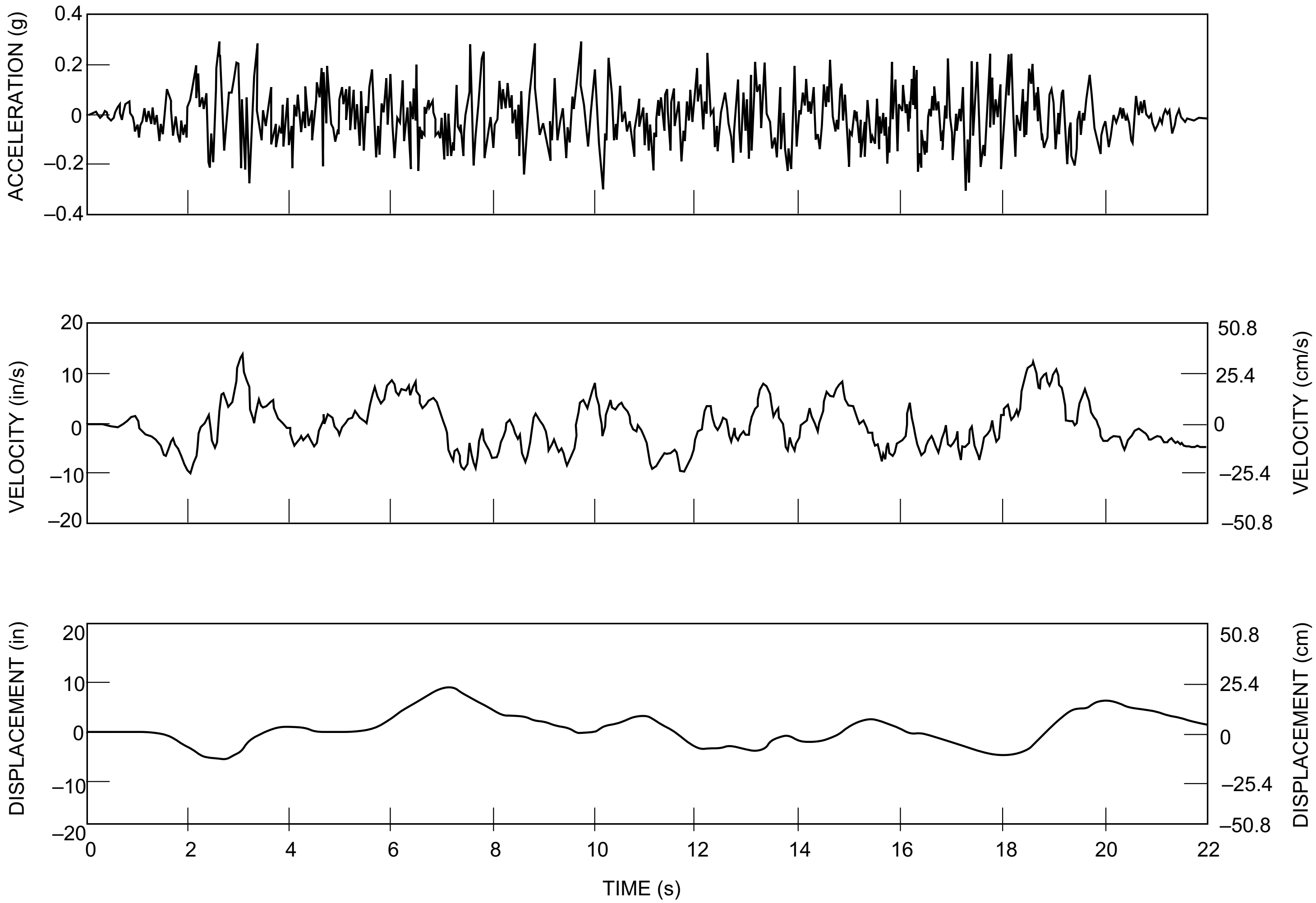

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Figure 3.7-5 Vertical, Component Time History

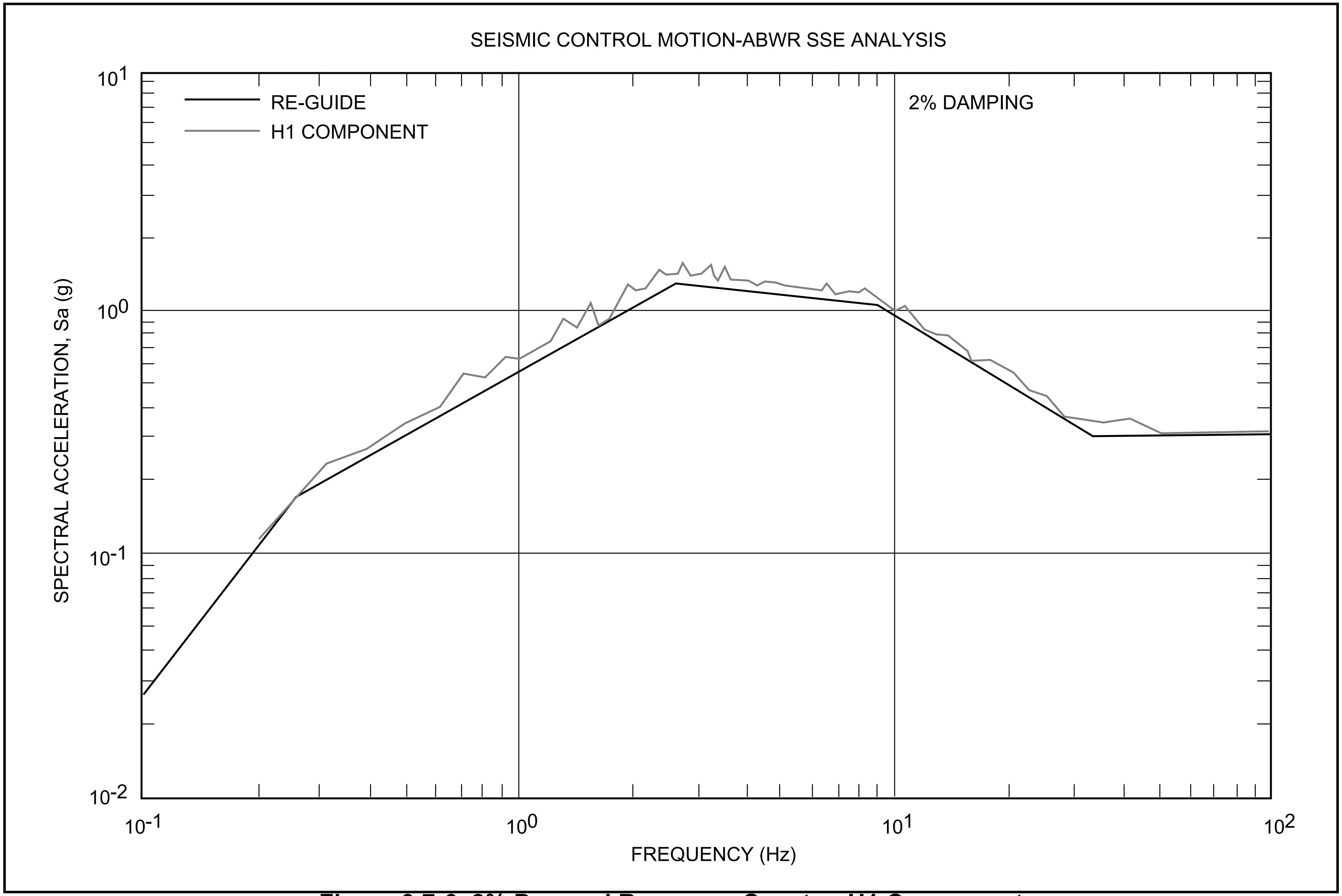

Figure 3.7-6 2% Damped Response Spectra, H1 Component

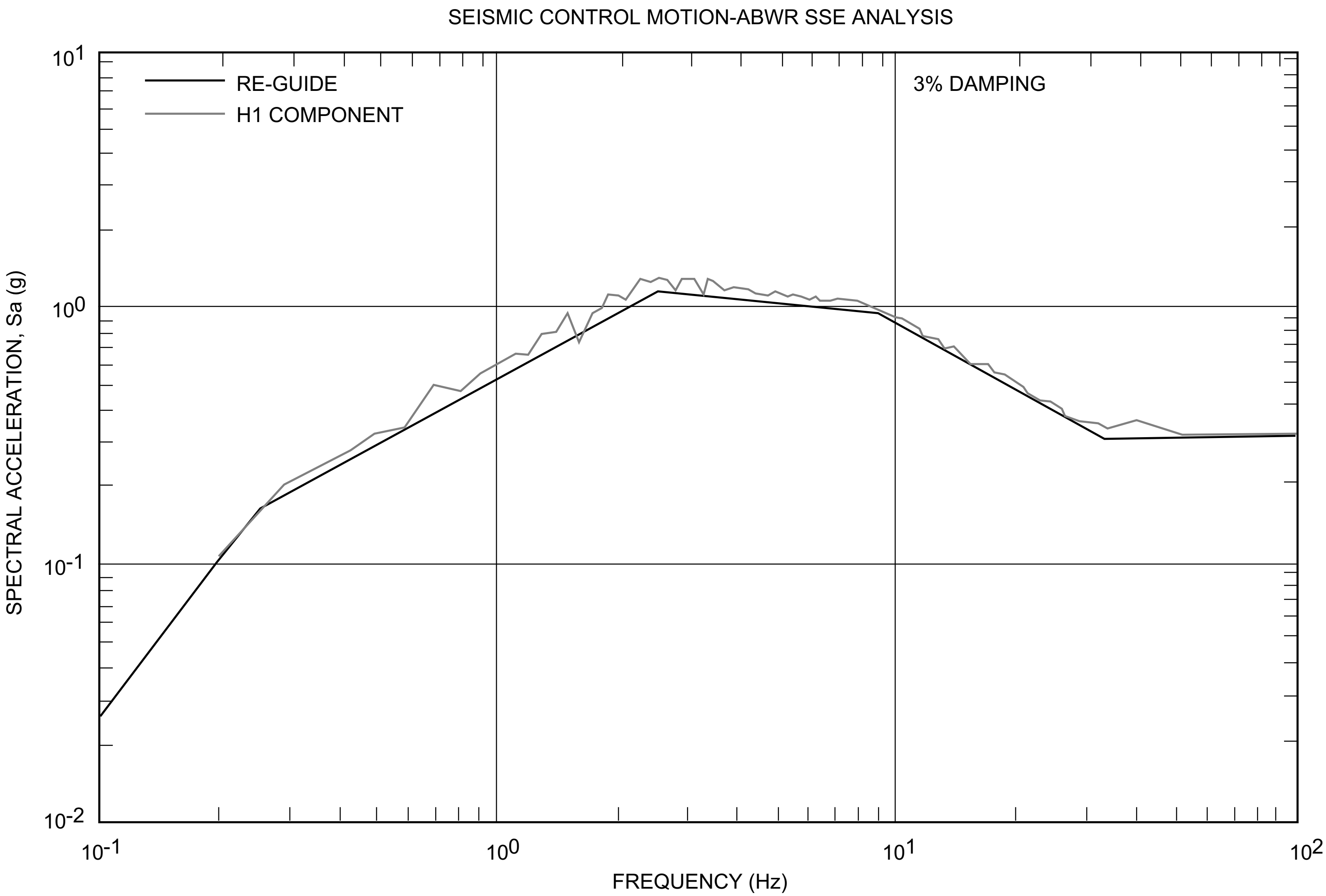

Figure 3.7-7 3% Damped Response Spectra, H1 Component

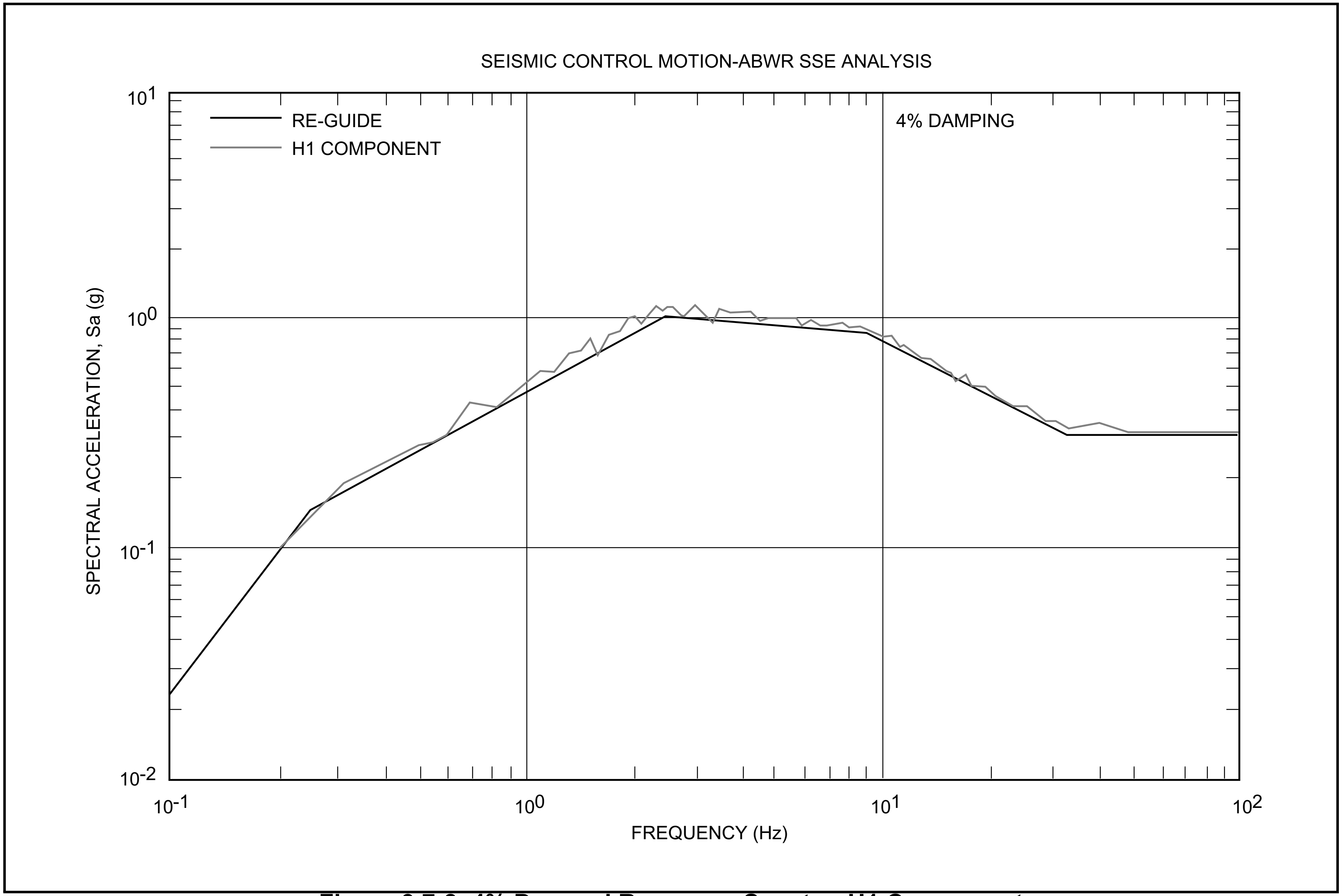

Figure 3.7-8 4% Damped Response Spectra, H1 Component

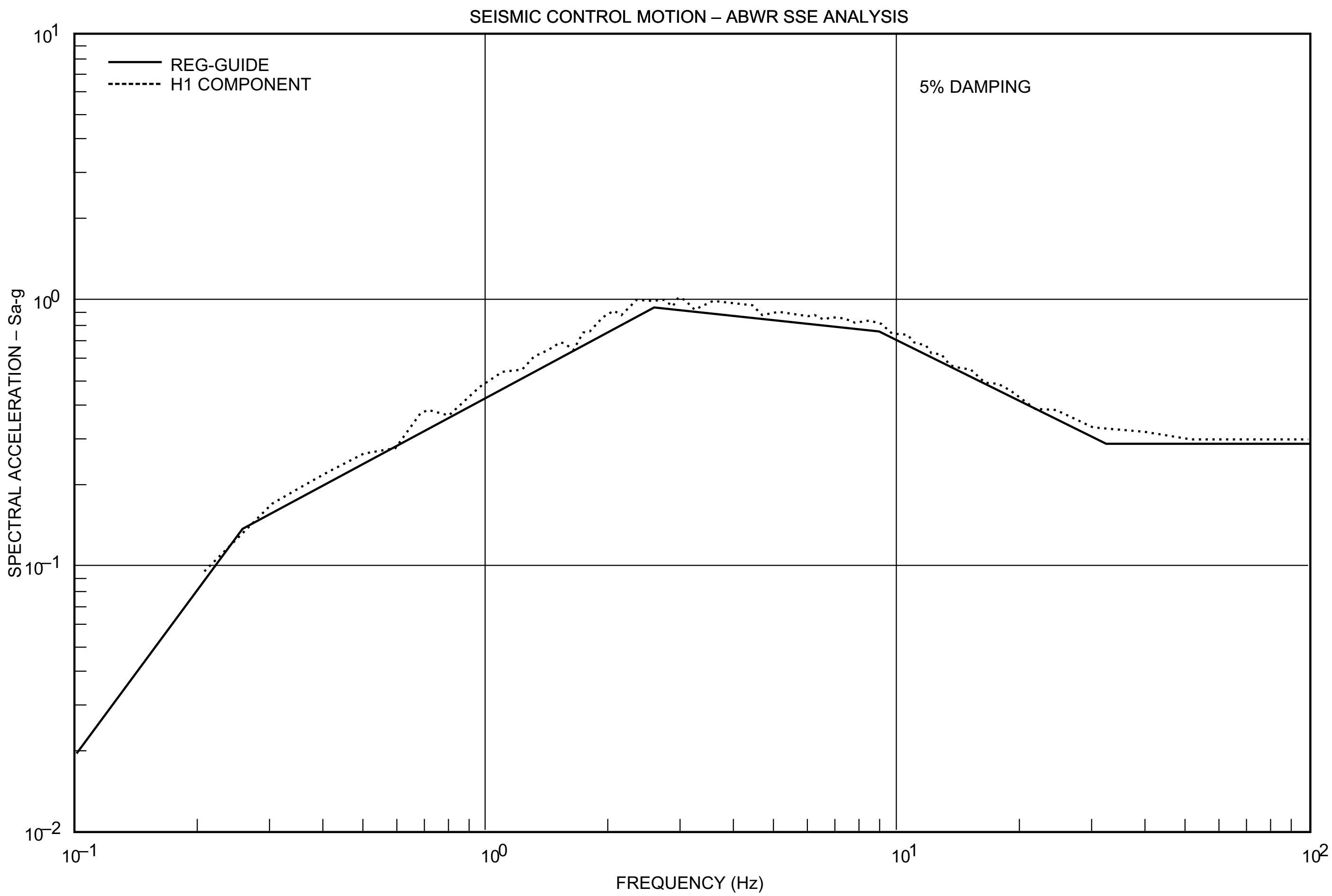

Figure 3.7-9 5% Damped Response Spectra, H1 Component

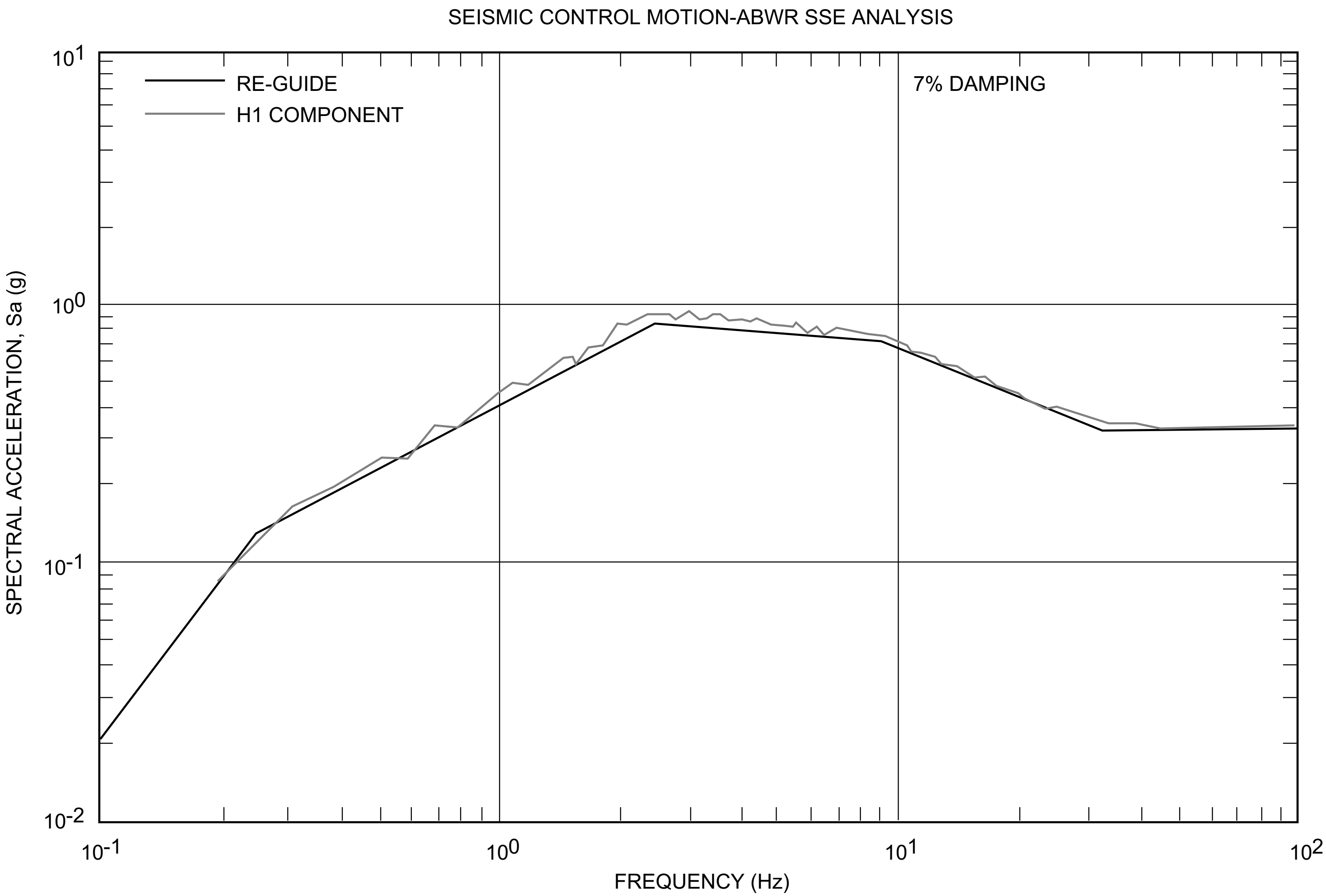

Figure 3.7-10 7% Damped Response Spectra, H1 Component

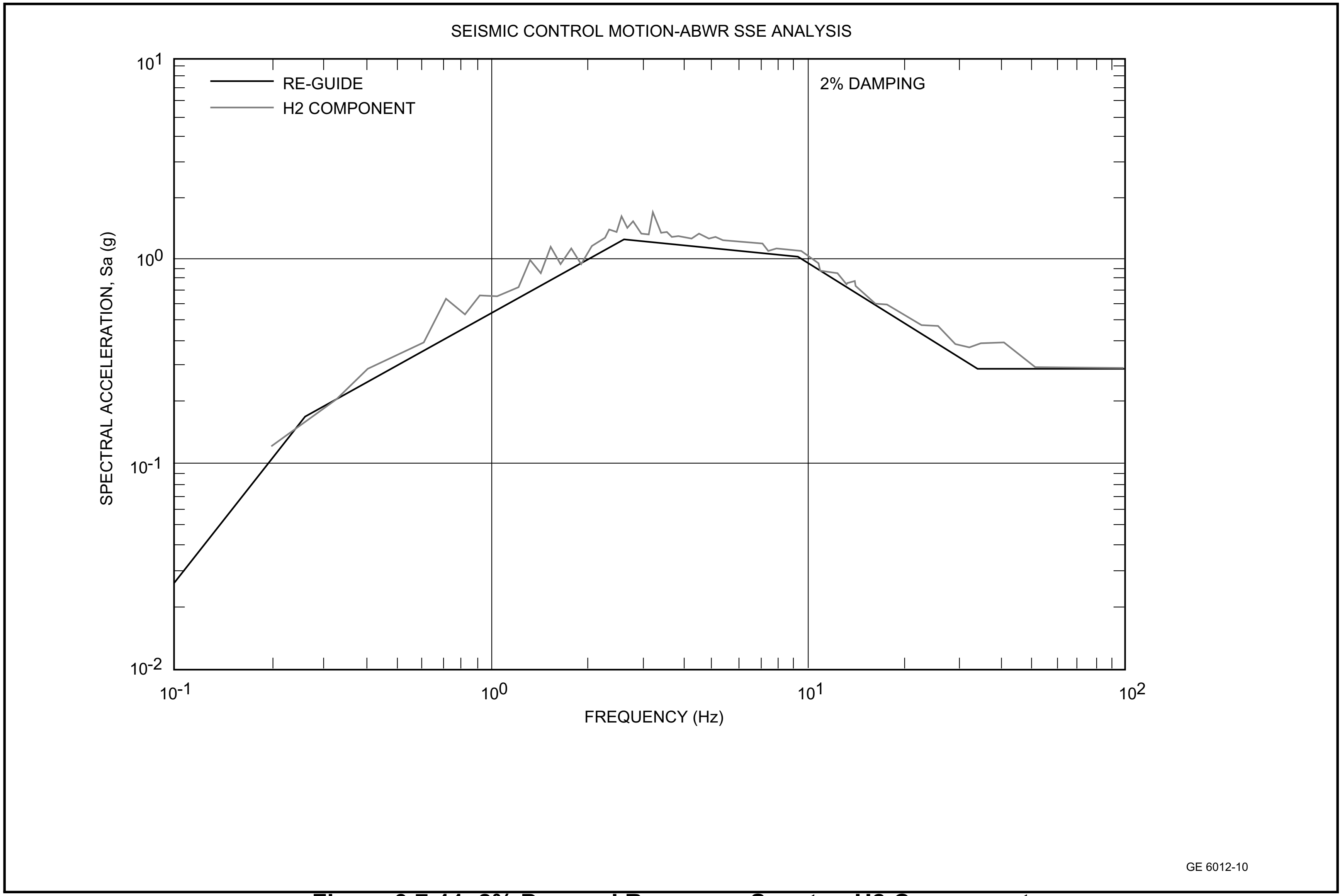

Figure 3.7-11 2% Damped Response Spectra, H2 Component

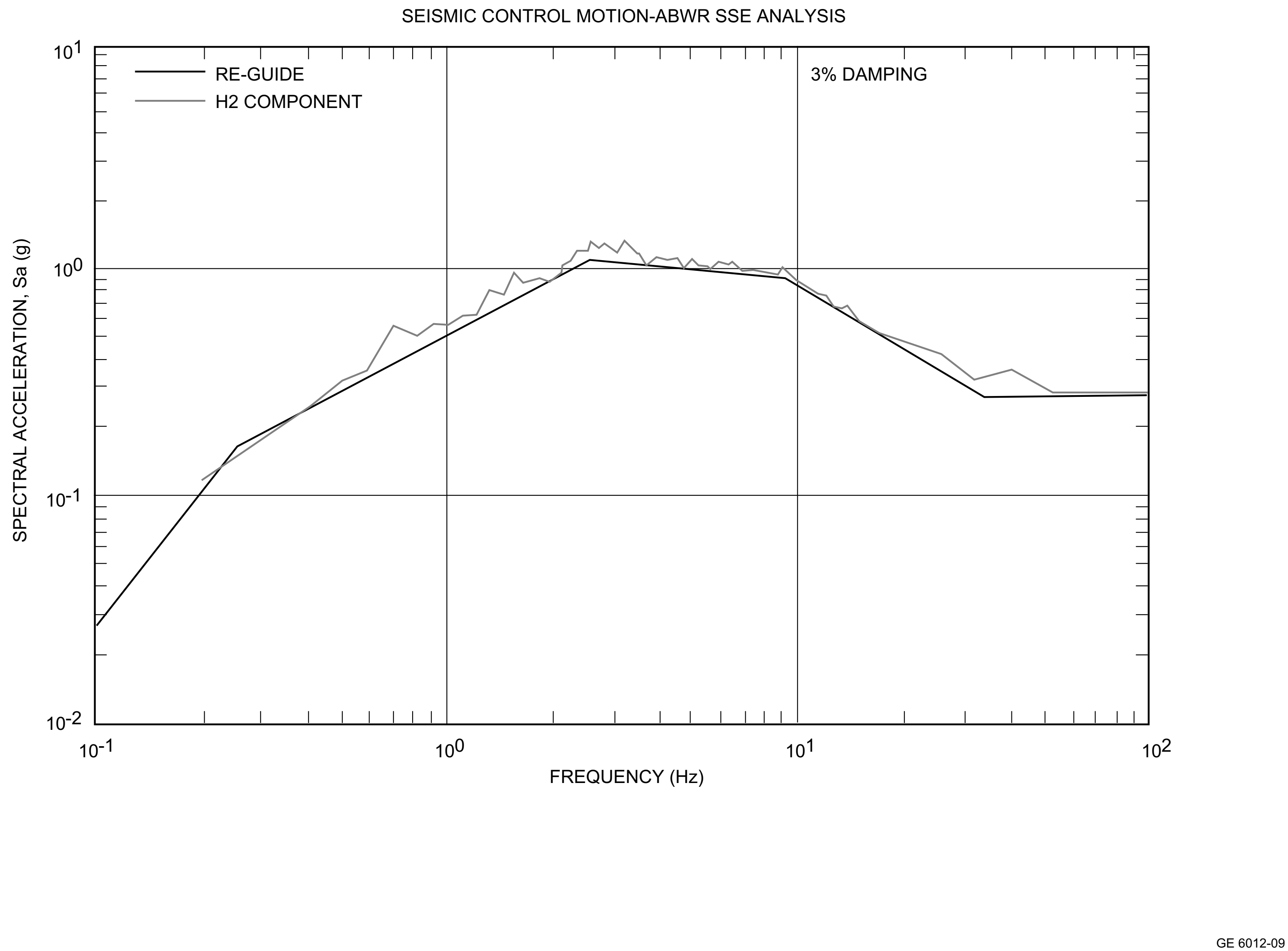

Figure 3.7-12 3% Damped Response Spectra, H2 Component

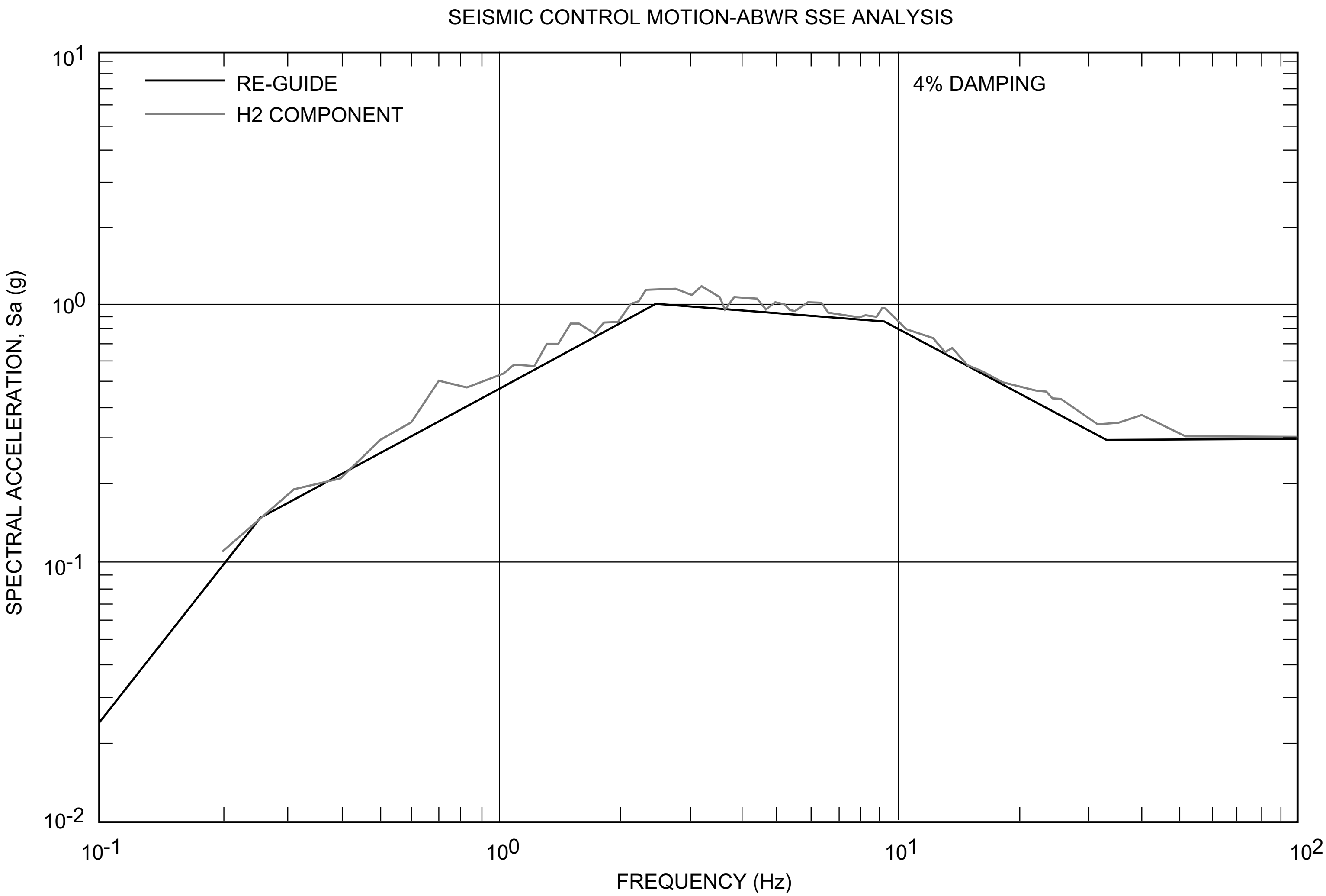

Figure 3.7-13 4% Damped Response Spectra, H2 Component

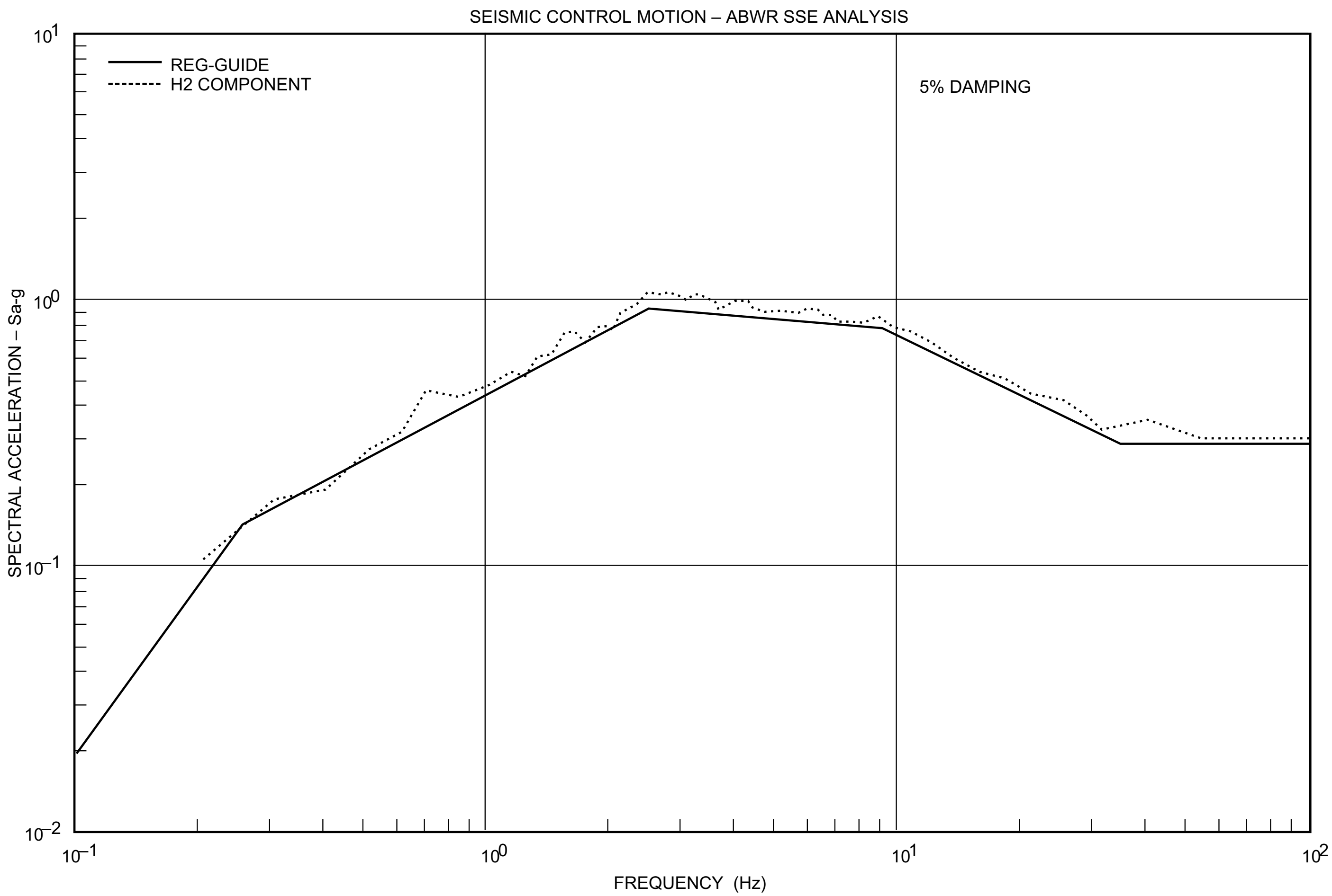

Figure 3.7-14 5% Damped Response Spectra, H2 Component

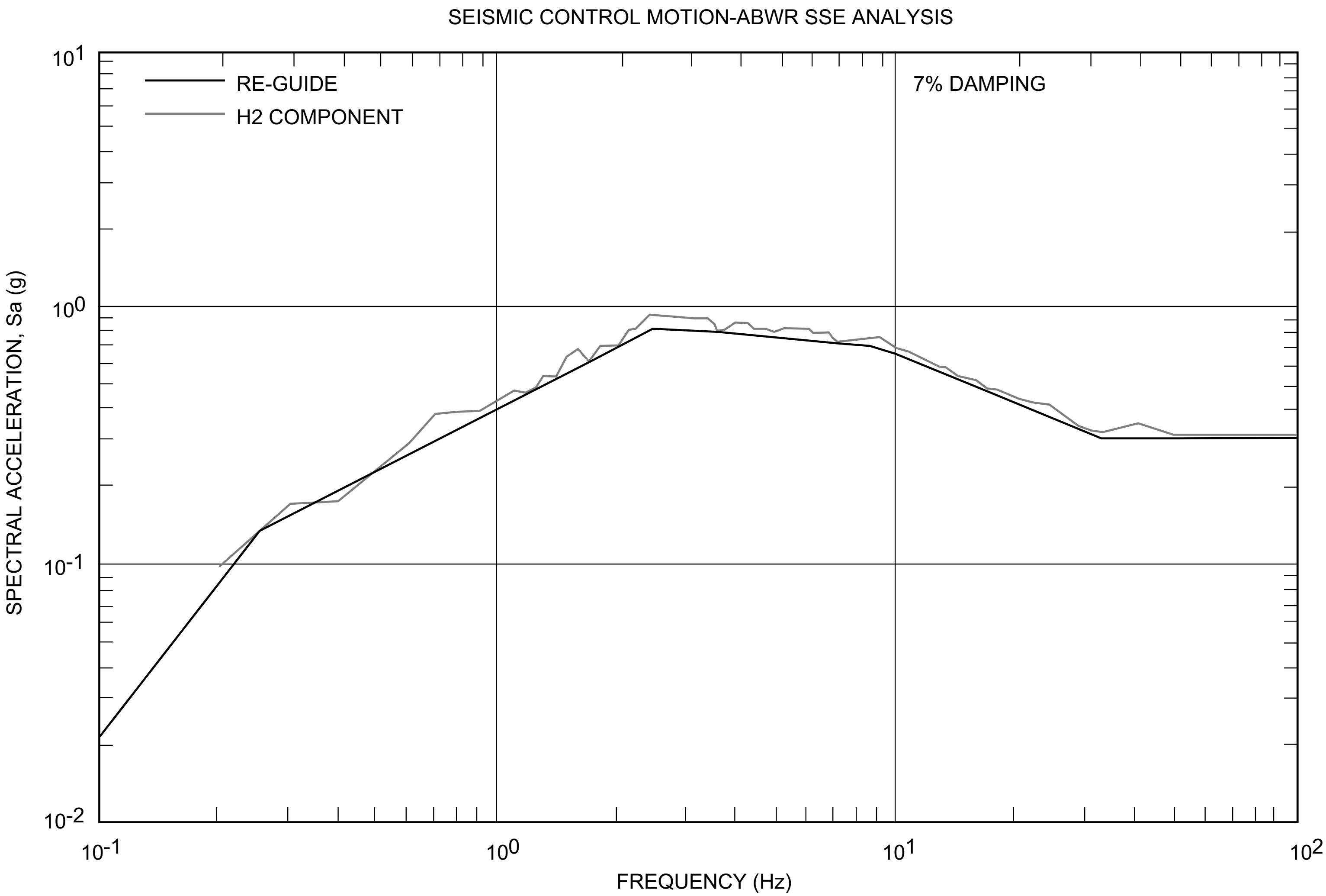

Figure 3.7-15 7% Damped Response Spectra, H2 Component

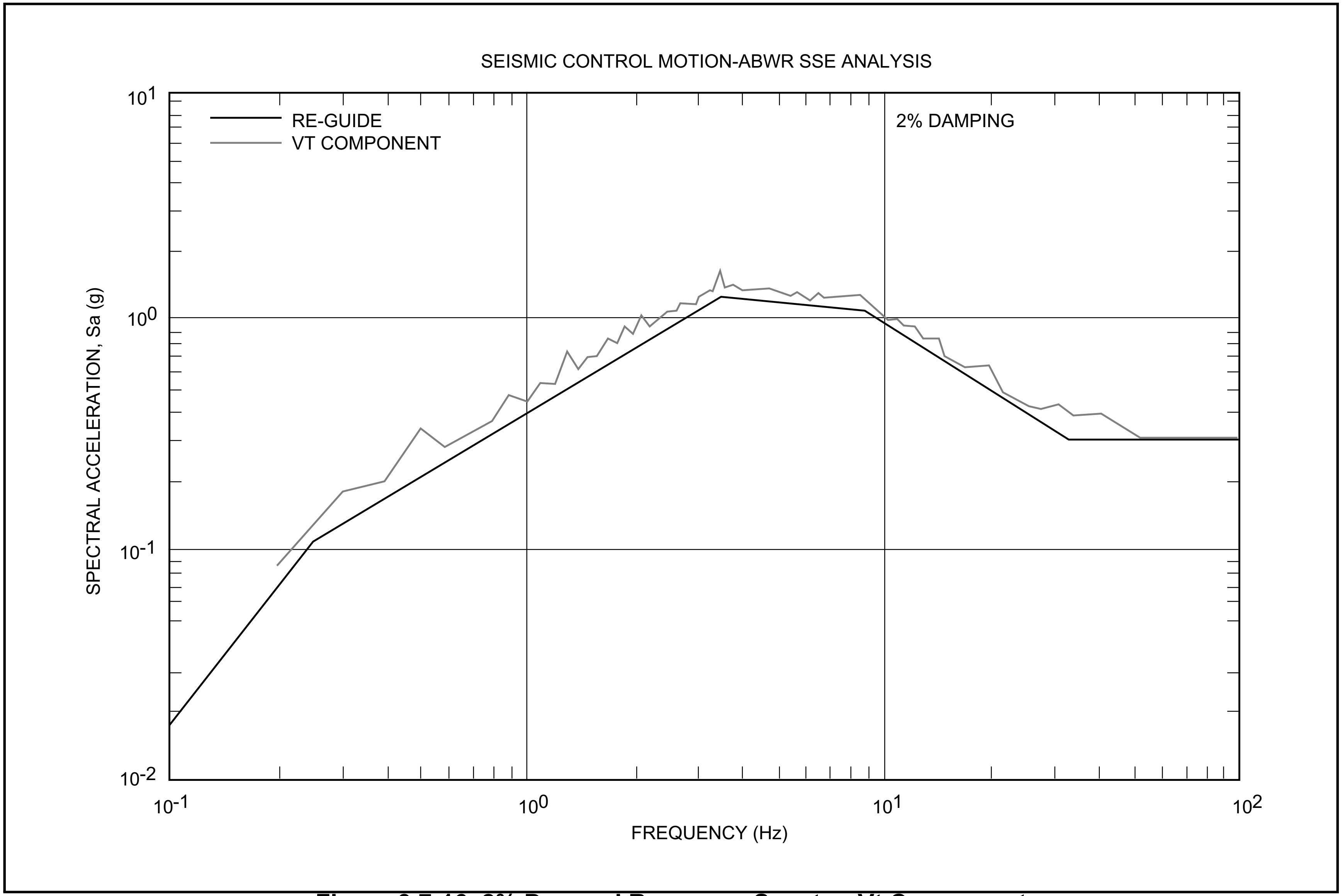

Figure 3.7-16 2% Damped Response Spectra, Vt Component

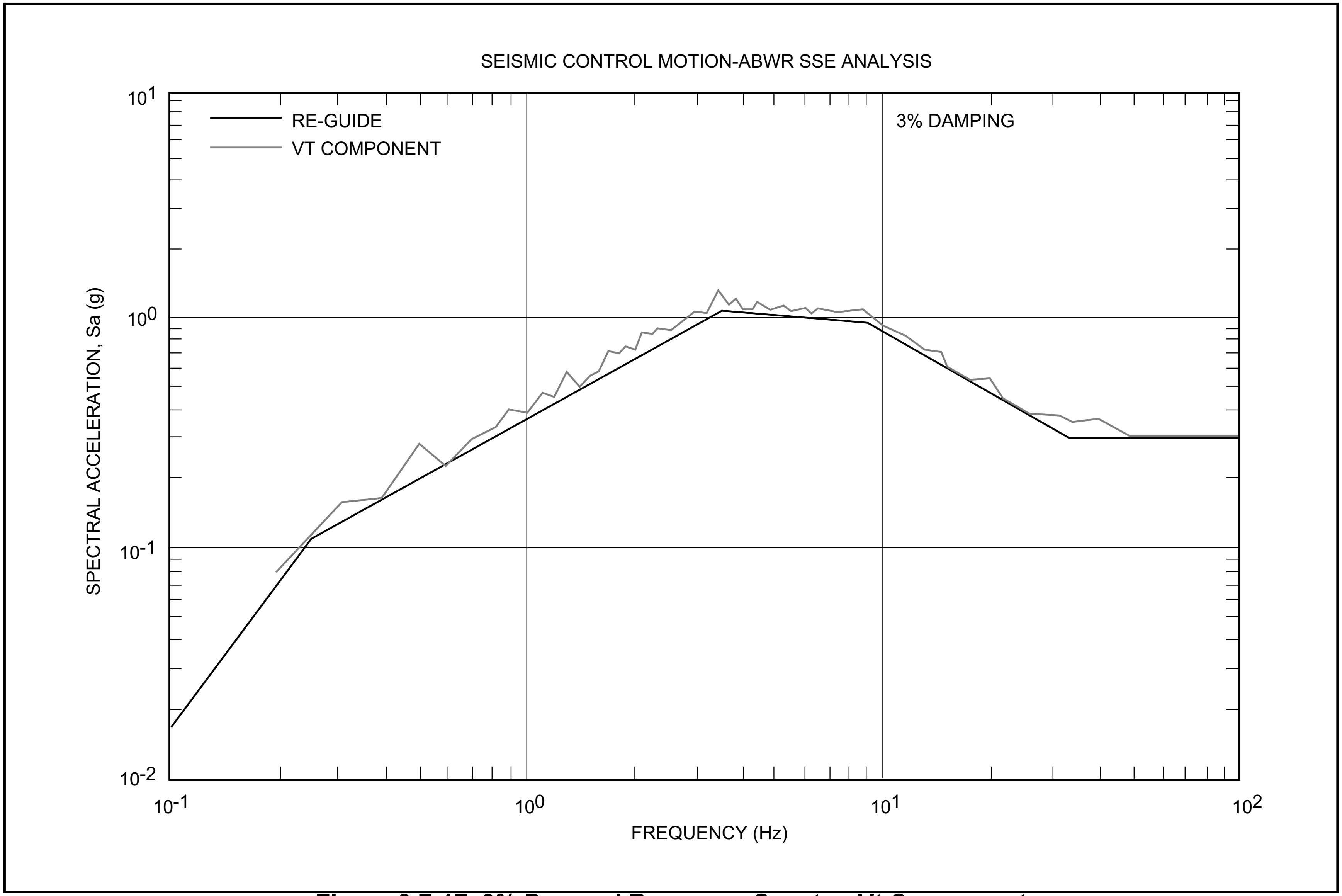

Figure 3.7-17 3% Damped Response Spectra, Vt Component

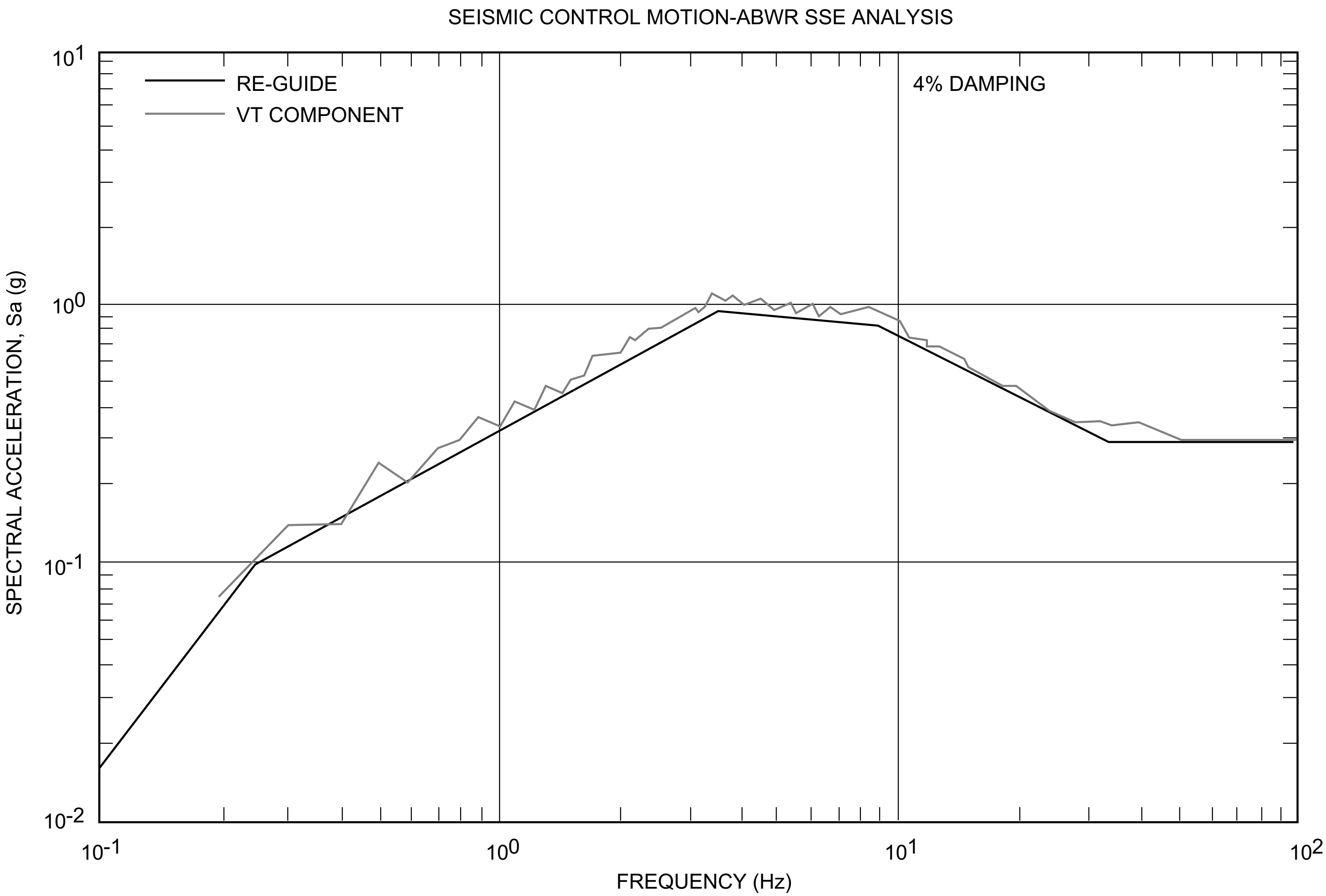

Figure 3.7-18 4% Damped Response Spectra, Vt Component

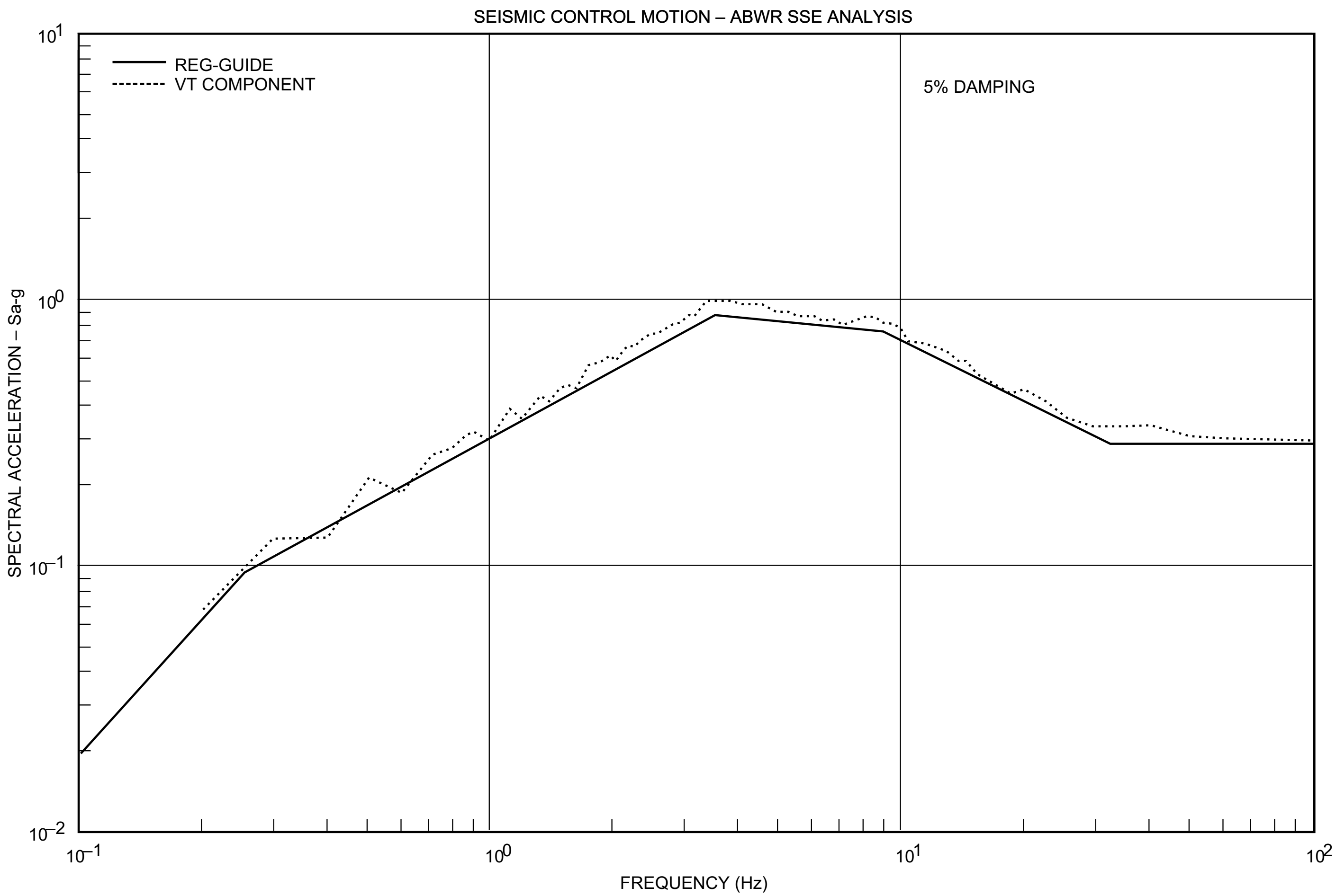

Figure 3.7-19 5% Damped Response Spectra, Vt Component

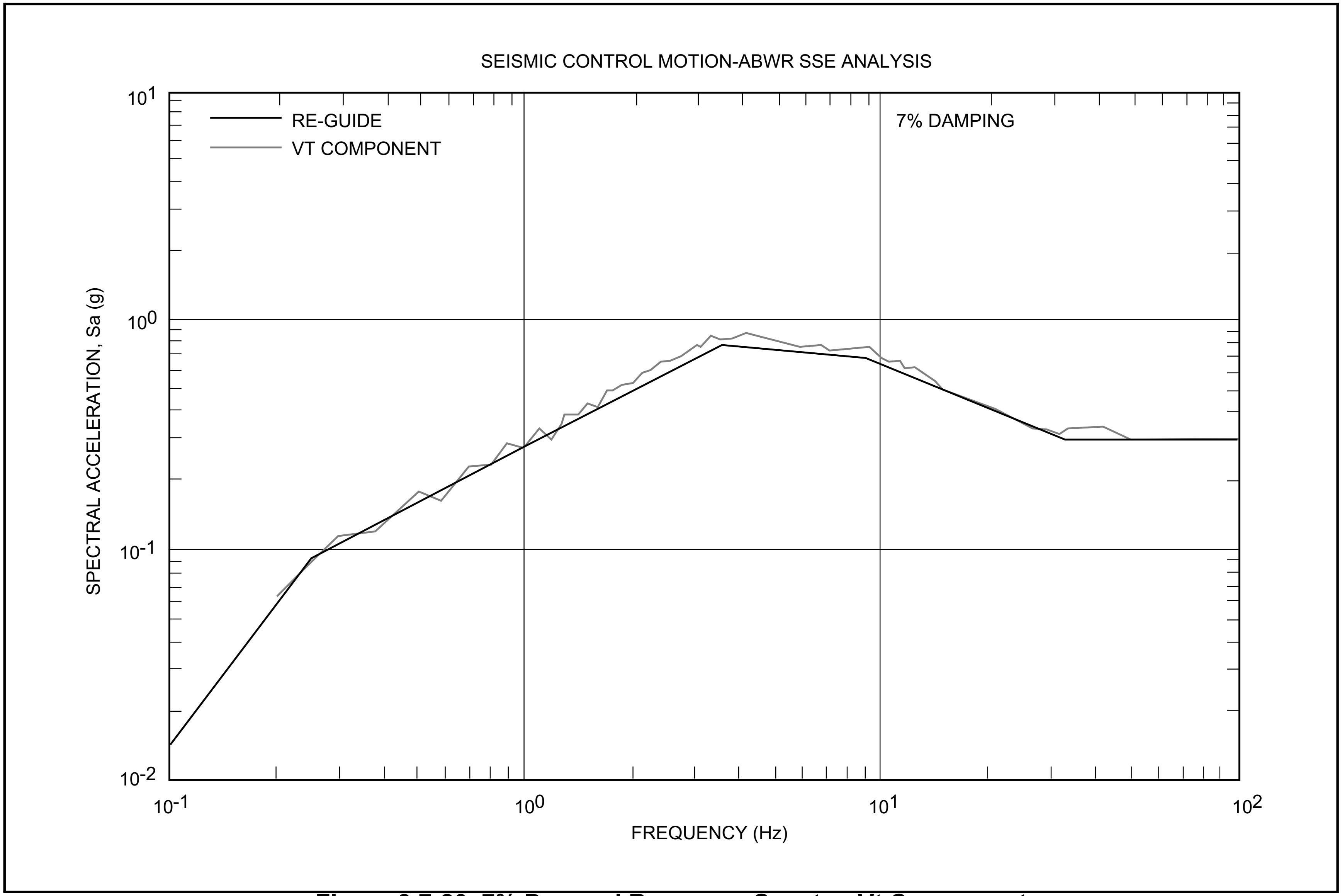

Figure 3.7-20 7% Damped Response Spectra, Vt Component

Figure 3.7-21 Not Used

Figure 3.7-22 Not Used

Figure 3.7-23 Not Used

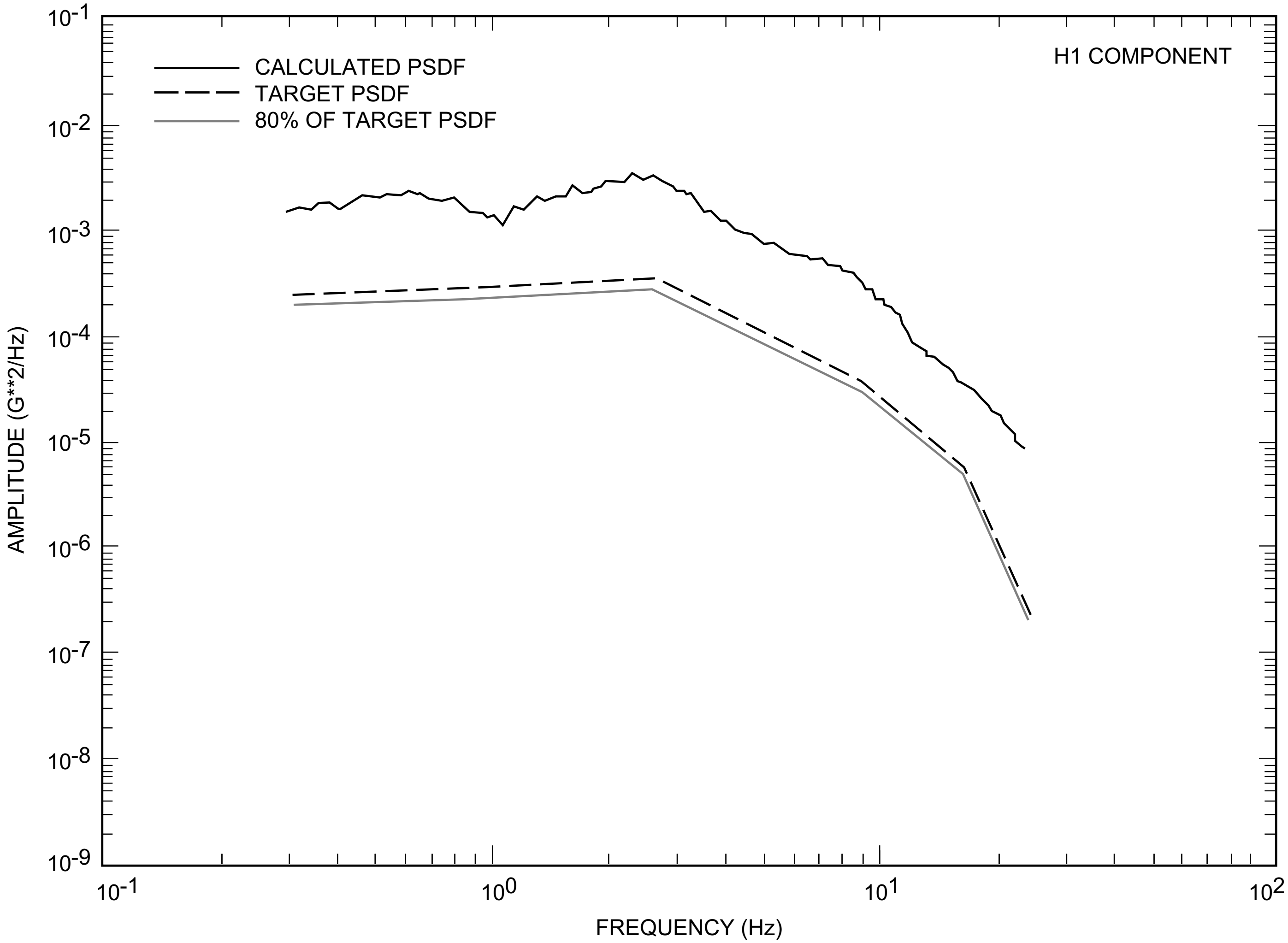

Figure 3.7-24 Power Spectral Density Function, H1 Component

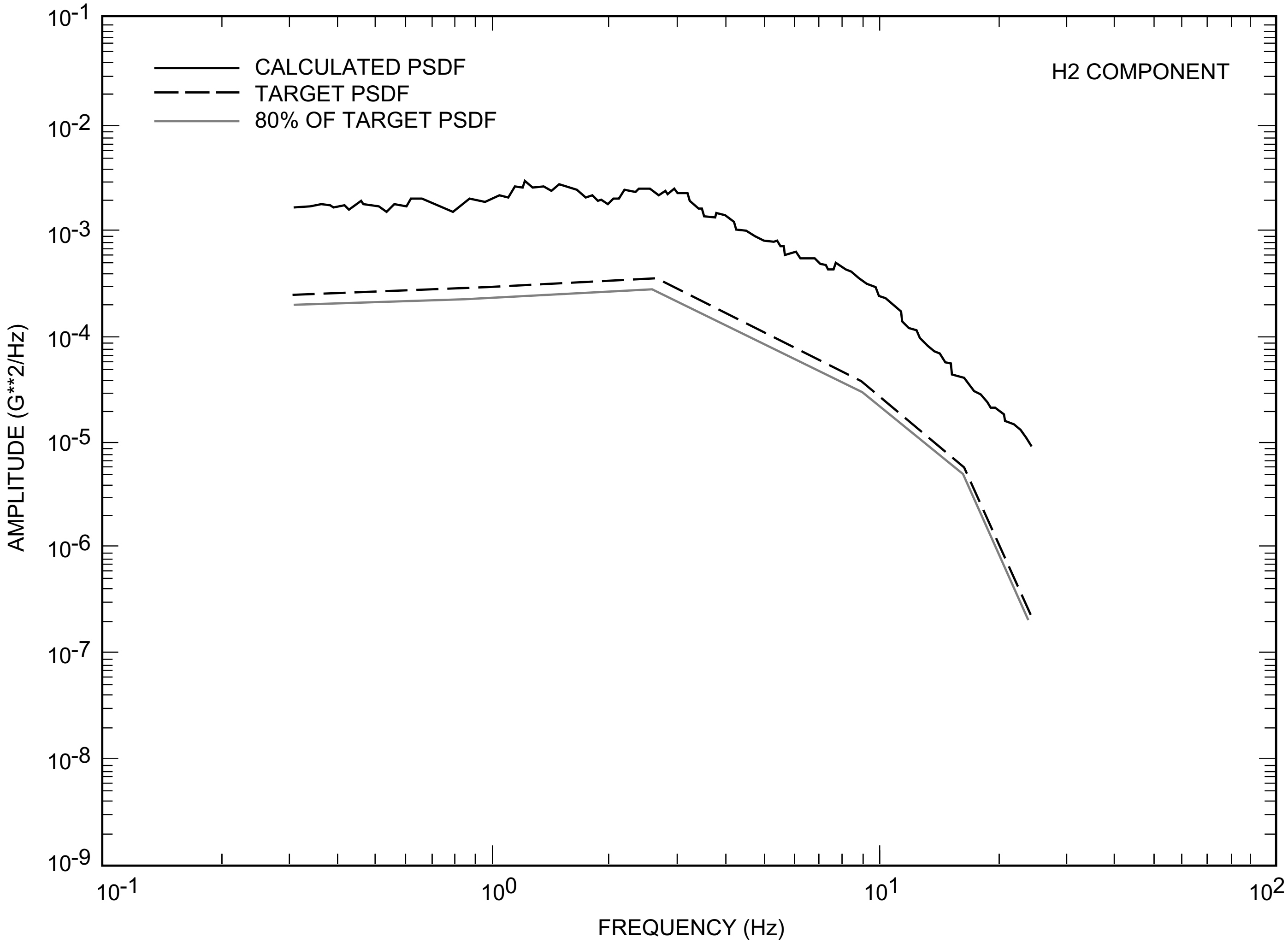

Figure 3.7-25 Power Spectral Density Function, H2 Component

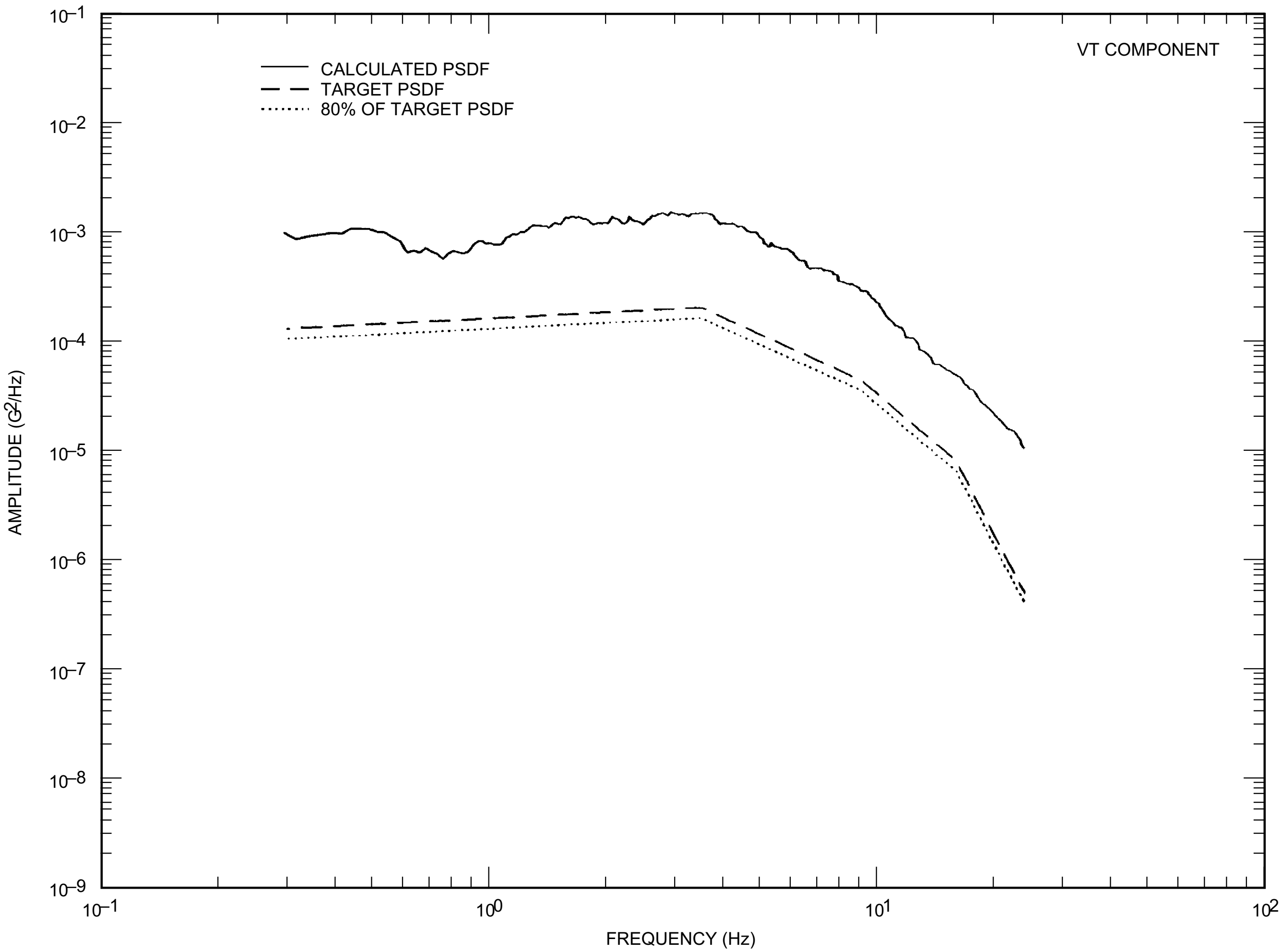

Figure 3.7-26 Power Spectral Density Function, VT Component

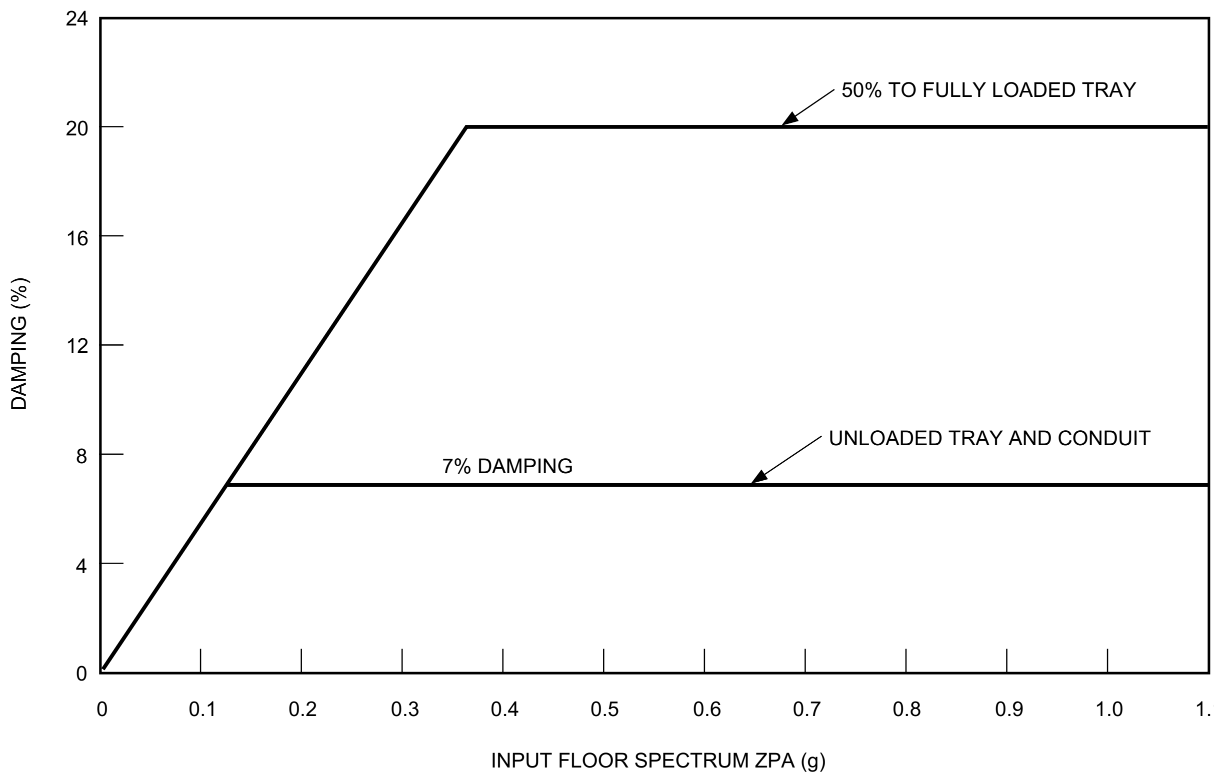

Figure 3.7-27 Damping Values for Electrical Raceway Systems

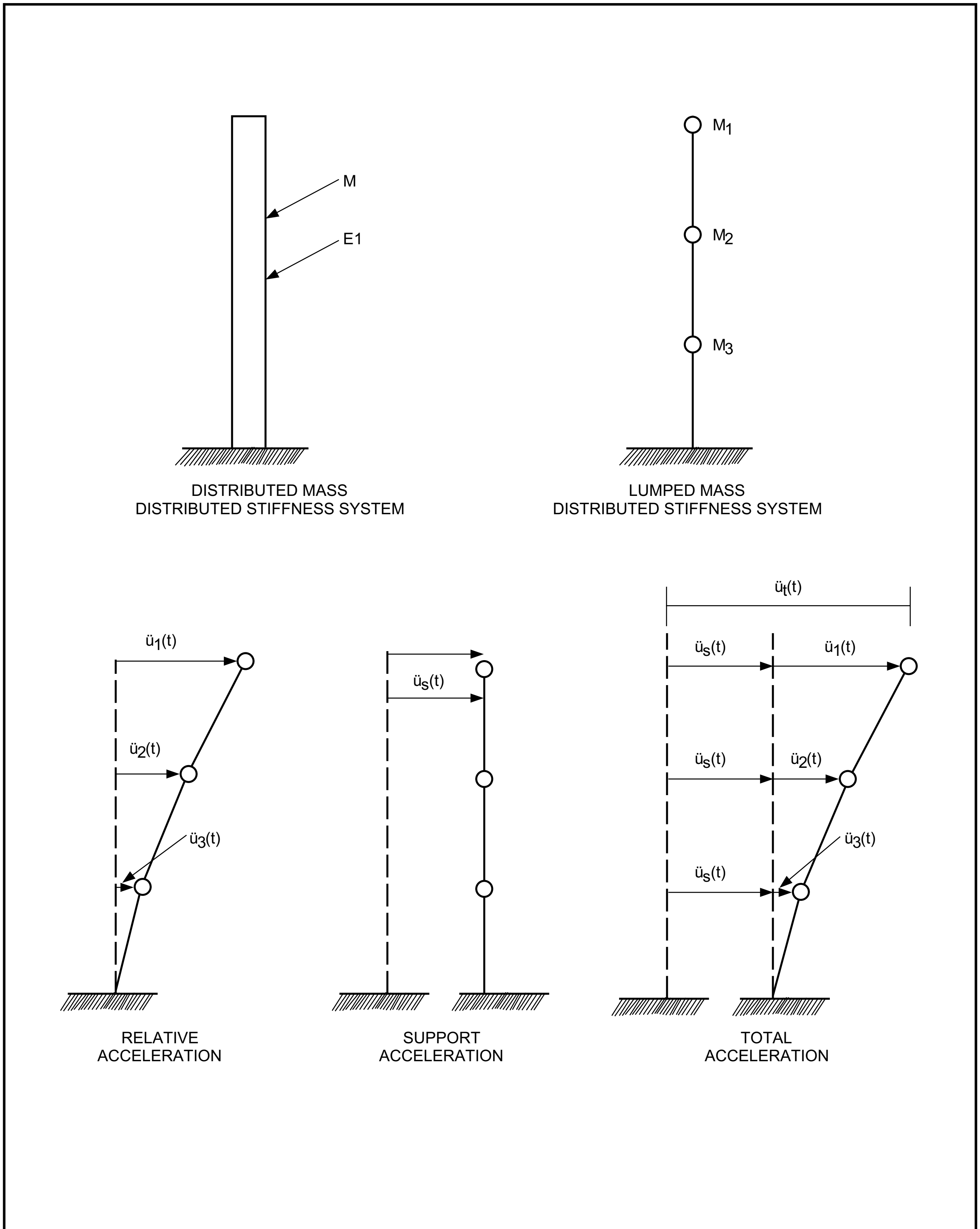

Figure 3.7-28 Seismic System Analytical Model

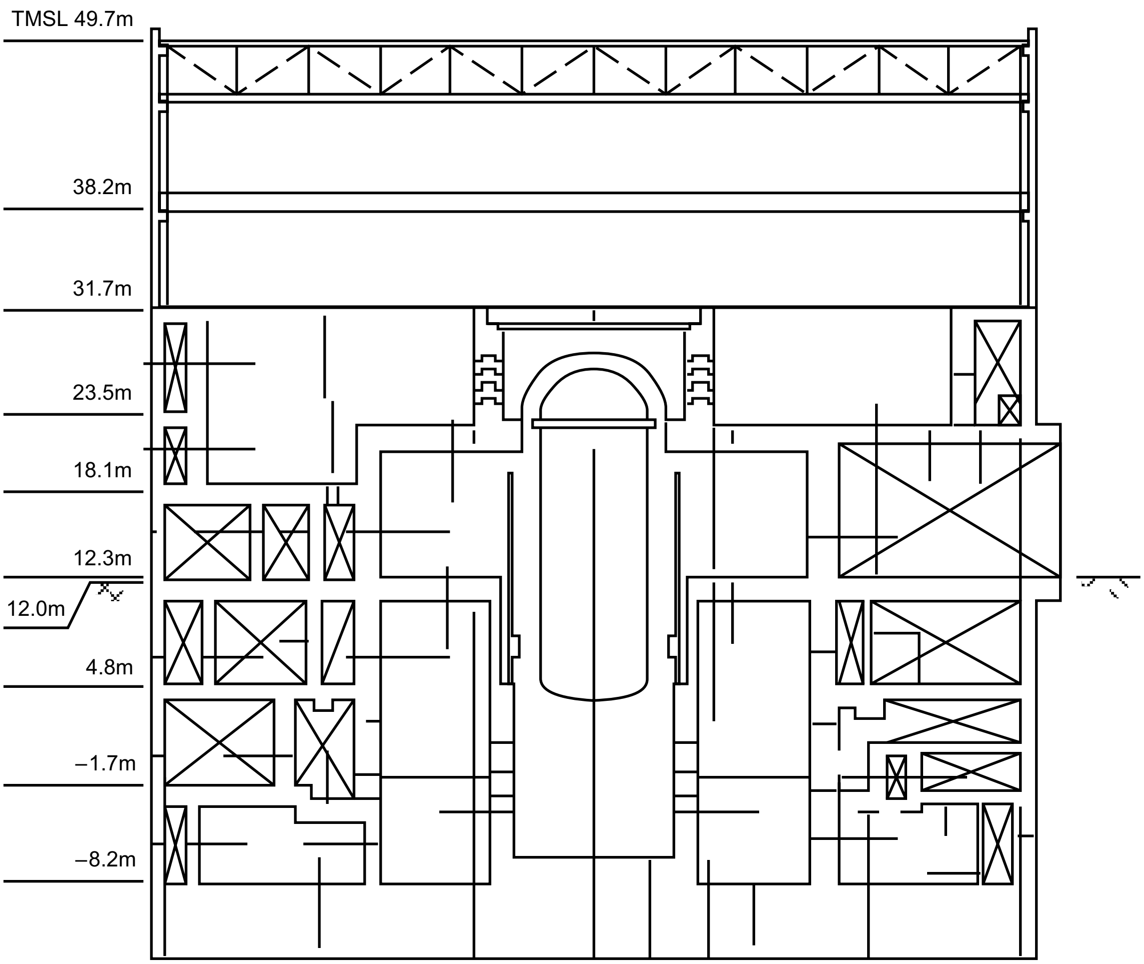

Figure 3.7-29 Reactor Building Elevation (0°–180° Section)

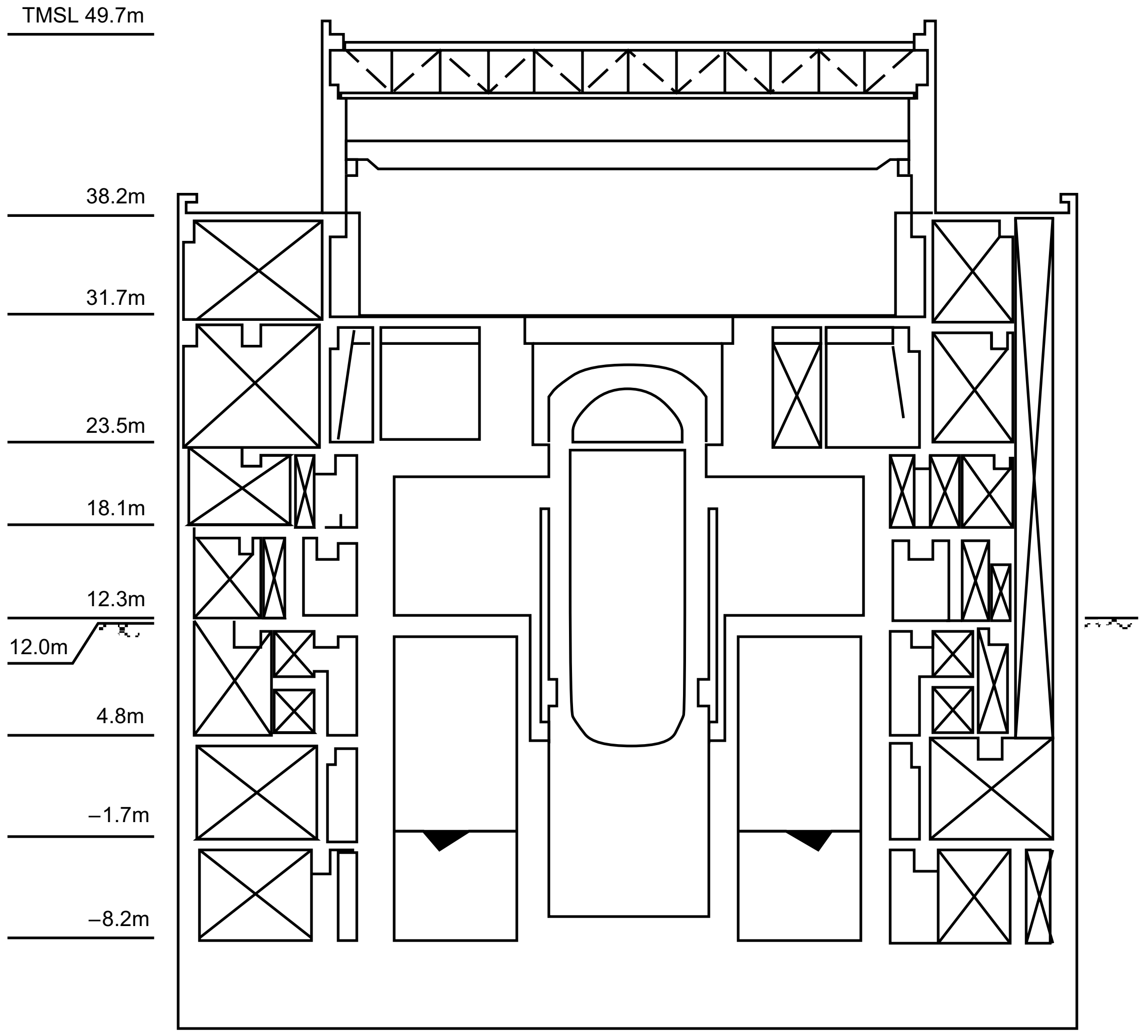

Figure 3.7-30 Reactor Building Elevation (90°–270° Section)

Figure 3.7-31 Reactor Building Model (see Figure 3A-8)

Figure 3.7-32 Reactor Pressure Vessel (RPV) and Internals Model (see Figure 3A-9)

Figure 3.7-33 Control Building Dynamic Model (see Figure 3A-27)

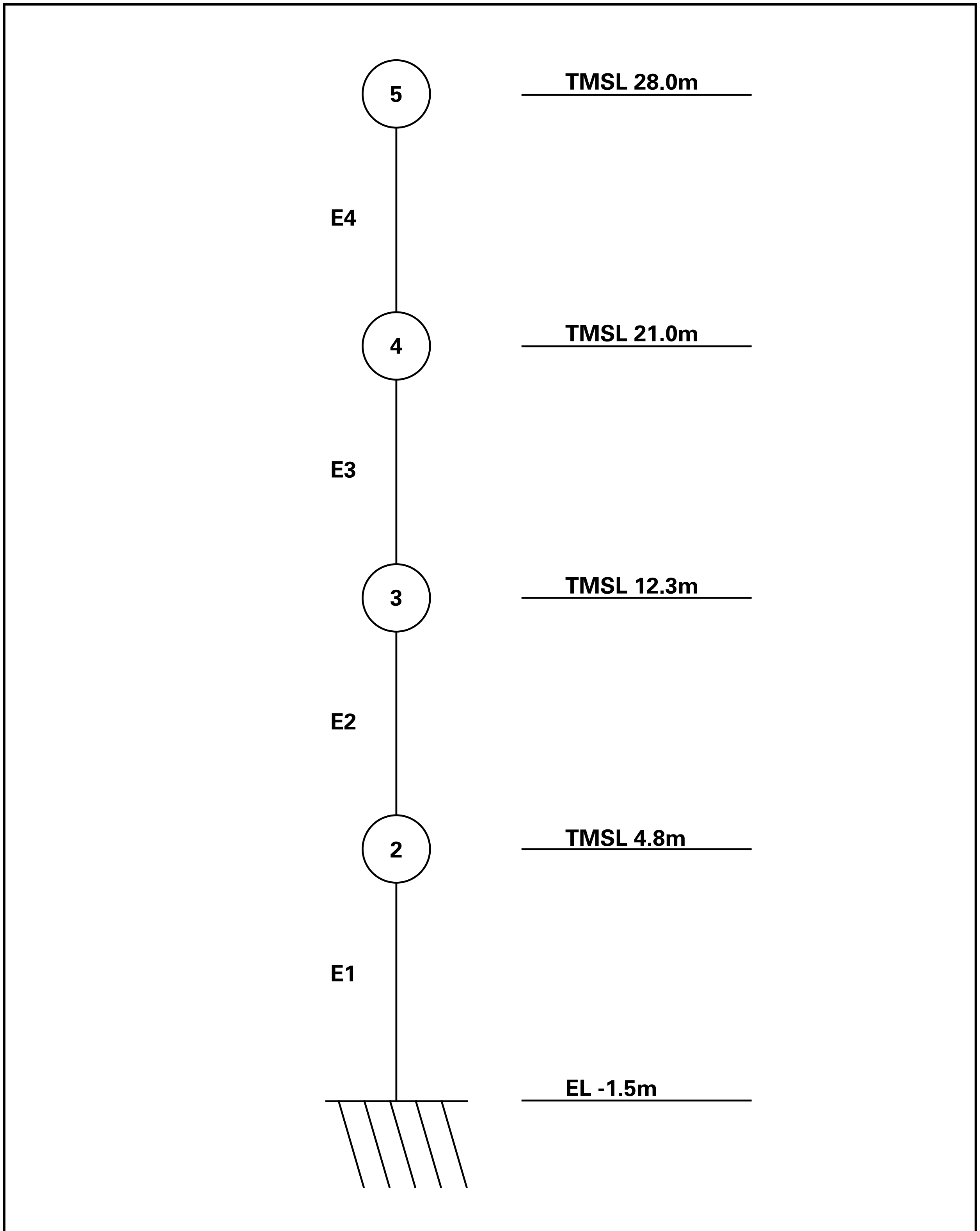

Figure 3.7-34 Radwaste Building Seismic Model

3.8 Seismic Category I Structures

A cathodic protection system is provided for the Seismic Category I structures as described in Section 8A.2. Its design is plant unique as it must be tailored to the site conditions.

3.8.1 Concrete Containment

The containment structure is designed to house the primary nuclear system and is part of the containment system, whose functional requirement is to confine the potential release of radioactive material in the event of a LOCA. This subsection describes the concrete containment structure. Steel components of the containment that resist pressure and are not backed by structural concrete are discussed in Subsection 3.8.2. A detailed description of the containment system is presented in Section 6.2.

3.8.1.1 Description of the Containment

3.8.1.1.1 Concrete Containment

The containment is shown in the summary report contained in Section 3H.1. This report contains a description of the containment, figures, loads, load combinations, concrete stresses, reinforcement stresses, and liner strains for the concrete containment vessel.

The structural system is a low-leakage reinforced concrete structure with an internal steel liner in the drywell and suppression chamber to serve as a leaktight membrane. The containment is a cylindrical shell structure which is divided by the diaphragm floor and the reactor pedestal into an upper drywell chamber, a lower drywell chamber and a suppression chamber. The top slab of the containment is an integral part of the fuel pool with the pool girders rigidly connected to the containment top slab and the reactor building walls. The Reactor Building (R/B) floors that abut the containment are integrated structurally with the concrete containment. The containment foundation mat is continuous with the R/B foundation mat. The containment wall, top slab, R/B floor slabs and foundation mat are constructed of cast-in-place, conventionally reinforced concrete.

The containment foundation mat is 5.5m thick. The foundation mat reinforcement consists of a top layer of reinforcement, a bottom layer of reinforcement, and vertical shear reinforcement. The bottom layer of reinforcement is arranged in a rectangular grid. The top layer of reinforcement is arranged in a rectangular grid at the center of the mat and then radiates outward in a polar pattern in order to avoid interference with the containment wall reinforcement.

The containment wall is a right, circular cylinder, 2m thick, with an inside radius of 14.5m and has a height of 29.5m measured from the top of the foundation mat to the bottom of the containment top slab. The main reinforcement in the wall consists of inside and outside layers of hoop and vertical reinforcement and radial bars for shear reinforcement.

Reinforcement is placed at major discontinuities in the wall, including the intersection of the wall and foundation mat, the vicinity of the wall intersection with the top slab, around major piping penetrations, the upper drywell equipment hatch and personnel airlock, the lower drywell equipment hatch and personnel airlock tunnels, and suppression chamber access hatch.

The containment top slab is nominally 2.2m thick. The slab thickness is increased to 2.4m beneath the fuel pool, steam dryer and steam separator pool, and around the drywell head opening.

The containment top slab main reinforcement consists of a top and bottom layer of reinforcement. The top layer of reinforcement is arranged in a rectangular grid. The bottom layer of reinforcement is arranged in a rectangular grid and then is bent near the containment wall into a radial pattern to avoid interference with the containment wall vertical reinforcement. Hoop reinforcement is provided in the area of the drywell head opening.

[*Table 2 of DCD/Introduction identifies the commitments of Code editions for design and construction of the concrete containment (Subsection 3.8.1) and buckling analysis of drywell head (Subsection 3.8.2.4.1.4), which, if changed, requires NRC Staff review and approval prior to implementation. The applicable portions of the Tier 2 sections and tables, identified on Table 2 of DCD/Introduction for this restriction, are italicized on the sections and tables themselves.*]*

3.8.1.1.2 Containment Liner Plate

The internal surface of the containment is lined with welded steel plate to form a leaktight barrier. The liner plate is fabricated from carbon steel, except that stainless steel plate or clad is used on wetted surfaces of the suppression chamber.

The liner plate is stiffened by use of structural sections and plates to carry the design loads and to anchor the liner plate to the concrete, as shown Figure 3.8-9. The liner plate is thickened locally and additional anchorage is provided at major structural attachments such as penetration sleeves, structural beam brackets, the RPV pedestal and the SRV quencher support connection to the basemat, and the diaphragm floor connection to the containment wall.

The erection of the liner is performed using standard construction procedures. The containment wall liner and top slab liner are used as a form for concrete placement. The liner on the bottom of the suppression chamber and lower drywell is placed after the foundation mat concrete is in place.

* See section 3.5 of DCD/Introduction.

3.8.1.2 Applicable Codes, Standards, and Specifications

The design, fabrication, construction, testing, and inservice inspection of the containment conforms to the applicable codes, standards, specifications, and regulations listed below, except where specifically stated otherwise.

3.8.1.2.1 Regulations

(1) Code of Federal Regulations, Title 10, Energy, Part 50, “Licensing of Production and Utilization Facilities.”

(2) Code of Federal Regulations (CFR), Title 10 - Energy, Part 100, Reactor Site Criteria, (10CFR100), including Appendix A thereto, “Seismic and Geologic Siting Criteria for Nuclear Power Plants.”

3.8.1.2.2 Construction Codes of Practice

[*American Society of Mechanical Engineers(ASME) Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CC.*]*

3.8.1.2.3 General Design Criteria, Regulatory Guides, and Industry Standards

(1) 10CFR50, Appendix A, “General Design Criteria for Nuclear Power Plants”, Criteria 1, 2, 4, 16 and 50. Conformance is discussed in Section 3.1.

(2) U.S. Nuclear Regulatory Commission (NRC) Regulatory Guides. Regulatory Guide 1.136, “Materials, Construction and Testing of Concrete Containment”.

(3) Industry Standards

Nationally recognized industry standards such as those published by the American Society for Testing and Materials (ASTM) and the American National Standards Institute (ANSI) as referenced by the Applicable Codes, Standards, and Regulations are used.

3.8.1.2.4 Containment Boundary

The jurisdictional boundary for application of Section III, Division 2 of the ASME Code to the concrete containment is shown in Figure 3H.1-2. The boundary extends to the:

(1) Outside diameter of the containment wall from the foundation mat to the containment top slab.

(2) The foundation mat within the outside diameter of the containment wall.

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* See Subsection 3.8.1.1.1.

(3) The containment top slab from the drywell head opening to the outside diameter of the containment wall.

(4) The intersection of the RPV pedestal on top of the basemat.

(5) The intersection of the diaphragm floor with the containment wall.

The concrete containment pressure boundary is limited to the cylindrical wall of the drywell and wetwell, and the drywell top slab.

They are included in ASME Code jurisdiction boundary for design, material, fabrication, inspection, testing, stamping, etc., requirements of the code. However, the fuel pool girders and any other structural components which are integral with the containment structure are treated the same as the containment only as far as loads and loading combinations are concerned in the design. Similarly, the R/B floor slabs that are integrated with the containment are not included in the ASME Code jurisdictional boundaries, but are treated the same as the containment only as far as loads and load combinations are concerned.

The reactor pedestal and diaphragm floor slab, which partition the containment into drywell and suppression chamber, are not part of the containment boundary. The reactor pedestal, steel structures filled with concrete, and the diaphragm floor slab are designed according to codes given in Subsections 3.8.3 and 3.8.4, respectively.

Those portions of the structure outside the indicated Code jurisdictional boundary will be designed, analyzed and constructed as indicated in Subsections 3.8.3, 3.8.4 and 3.8.5. The analytical models will include both the containment and Reactor Building and therefore will provide continuity in the analysis.

3.8.1.3 Loads and Load Combinations

The containment is analyzed and designed for all credible conditions of loading, including normal loads, preoperational testing loads, loads during severe environmental conditions, loads during extreme environmental conditions and loads during abnormal plant conditions.

3.8.1.3.1 Normal Loads

(1) D — Dead load of the structure and equipment plus any other permanent loads, including vertical and lateral pressures of liquids.

(2) L — Live loads, including any moveable equipment loads and other loads which vary in intensity and occurrence, such as forces exerted by the lateral pressure of soil.

Live Load L, includes floor area live loads, laydown loads, nuclear fuel and fuel transfer casks, equipment handling loads, trucks, railroad vehicles and similar items. The floor area live load shall be omitted from areas occupied by equipment whose weight is specifically included in dead load. Live load shall not be omitted under equipment where access is provided, for instance, an elevated tank on four legs.

The criteria for consideration of live loads for the designs of structural elements of the Reactor Building, Control Building and the Radwaste Building are provided in Subsections 3H.1.4.3.1, 3H.2.4.3.1 and 3H.3.4.3.1, respectively. The inertial properties include all tributary mass expected to be present in operating conditions at the time of earthquake. This mass includes dead load, stationary equipment, piping and appropriate part of live load established in accordance with the layout and mechanical requirements. In the ABWR design, 25% of full live load L (designated as L_o), is used in the load combinations that include seismic loads. This value of L_o is justifiable on the following consideration:

(a) Because of the overall light occupancy of the power plants during their operation, it is a general practice to use a minimum of L_o to be 25% of the full live load L.

(b) Section 9.3 of ASCE Standards 7-88 and Section 2334(a) of the 1991 Uniform Building Code specify that a minimum of 25% of the floor live loads should be considered for the computation of design seismic forces for storage and warehouse type occupancies. The variation in live load intensity and occurrence in operating nuclear plants is expected to be no higher than that for storage in warehouse occupancies. A 25% of full live loads is, therefore, equally applicable to the nuclear plants

However, the live load values used in the governing loading combination for design of local elements such as beams and slabs, are the full values. For example in the loading combination for RCCV shown in Subsection 3H.1.4.3.2.1 for load combination No. 15, the value of SSE is computed using 25% of the live load and in addition full value of live load is used for L for the design of structural components.

(3) T_o — Thermal effects and loads during normal operating, startup or shutdown conditions, including liner plate expansion, equipment and pipe reactions, and thermal gradients based on the most critical transient or steady- state thermal gradient.

(4) R_o — Pipe reactions during normal operating or shutdown conditions based on the most critical transient or steady-state conditions.

(5) P_o — Pressure loads resulting from the pressure difference between the interior and exterior of the containment, considering both interior pressure changes because of heating or cooling and exterior atmospheric pressure variations.

(6) Construction Loads — Loads which are applied to the containment from start to completion of construction. The definitions for D, L and T_o given above are applicable, but are based on actual construction methods and/or conditions.

(7) SRV — Safety/relief valve loads. Oscillatory dynamic pressure loadings resulting from discharge of safety/relief valves (SRVs) into the suppression pool. The development of these loads is with the methods described in Section 3B. The R/B vibration dynamic effects shall be included in the load combinations. The number and combinations of valves that will open during an RPV pressure transient are as follows:

(a) G_1 — Design pressure load on the suppression pool boundary resulting from discharge of one SRV into the suppression pool. First actuation and subsequent actuation shall be considered.

(b) G_2 — Design pressure load on the suppression pool boundary resulting from discharge of two adjacent SRVs, first actuation, into the suppression pool.

(c) G_{ALL} — Design pressure load on the suppression pool boundary resulting from discharge of all SRVs, first actuation, into the suppression pool.

(d) ADS — Design pressure load on the suppression pool boundary resulting from the SRV Automatic Depressurization System (ADS) discharge into the suppression pool.

3.8.1.3.2 Preoperational Testing Loads

(1) P_t — Test loads are loads which are applied during the structural integrity test.

(2) T_t — Thermal effects and loads during the structural integrity test.

3.8.1.3.3 Severe Environmental Loads

(1) W — Loads generated by the design wind specified for the plant site as defined in Section 3.3.

3.8.1.3.4 Extreme Environmental Loads

(1) E'— Safe shutdown earthquake (SSE) loads as defined in Section 3.7.

(2) W' — Loads generated by the tornado specified in Section 3.3.

3.8.1.3.5 Abnormal Plant Loads

(1) F_L — Hydrostatic load due to post-LOCA flooding of the containment for fuel recovery subsequent to a design basis accident.

(2) R_a — Pipe reactions (including R_o) from thermal conditions generated by a LOCA.

(3) T_a — Thermal effects (including T_o) and loads generated by a LOCA.

(4) P_a — Design accident pressure load within the containment generated by large break LOCA (LBL), based upon the calculated peak pressure with an appropriate margin.

(5) P_i — Design accident pressure load within the containment generated by an intermediate break LOCA (IBL).

(6) P_s — Design accident pressure load within the containment generated by a small break LOCA (SBL).

(7) Y — Local effects on the containment due to a LOCA. The local effects shall include the following:

(a) Y_r — Load on the containment generated by the reaction of a ruptured high-energy pipe during the postulated event of the DBA. The time-dependent nature of the load and the ability of the containment to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the effects of Y_r.

(b) Y_j — Load on the containment generated by the jet impingement from a ruptured high-energy pipe during the postulated event of the DBA. The time-dependent nature of the load and the ability of the containment to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the effects of Y_j.

(c) Y_m — The load on the containment resulting from the impact of a ruptured high-energy pipe during the DBA. The type of impact (e.g., example plastic or elastic), together with the ability of the containment to deform beyond yield, shall be considered in establishing the structural capacity necessary to resist the impact.

(8) CO — An oscillatory dynamic loading (condensation oscillation) on the suppression pool boundary due to steam condensation at the vent exits during the period of high steam mass flow through the vents following a LOCA.

(9) CHUG — An oscillatory dynamic loading (chugging) in the top vent and on the suppression pool boundary due to steam condensation inside the top vent or at the top vent exit during the period of low steam mass flow in the top vent following a LOCA.

(10) VLC — Loads from component response or direct fluid forces, on components located in the suppression pool, caused by the main vent line clearing phenomenon.

(11) PS — Pool swell bubble pressure (axisymmetrical and nonaxisymmetrical) on the suppression pool boundary due to a LOCA.

3.8.1.3.6 Load Combinations for the Containment Structure and Liner Plate

The containment structure is designed using the loads, load combinations, and load factors listed in Table 3.8-1.

Loads and load combinations listed in Table 3.8-1 shall be used for the design of the steel liner and liner anchors, but the load factor for all loads in the load combinations shall be 1.0.

3.8.1.4 Design and Analysis Procedures

This section describes the analytical and design procedures used in designing the containment.

3.8.1.4.1 Containment Cylindrical Wall, Top Slab, and Foundation Mat

3.8.1.4.1.1 Analytical Methods

The containment structure is analyzed by the use of the linear elastic finite element computer program STARDYNE described in Section 3C. The containment and Reactor Building layout utilizes an integrated structural system. The structure is idealized as a three-dimensional assemblage of beam elements, and iso-parametric membrane-bending plate elements.

Since the containment and Reactor Building are essentially symmetrical about the centerline of the plant parallel to the fuel pool girders and steam tunnel, only 180° of the structure is modeled. Boundary conditions are applied along the centerline of the plant that simulate the symmetry of the whole structure.

The foundation soil is simulated by a set of horizontal and vertical springs. The soil spring constraints are calculated based on using correction factors to account for the R/B embedment.

The containment and reactor building 180° finite element model is shown in Figure 3H.1-12.

3.8.1.4.1.1.1 Nonaxisymmetrical Loads

Nonaxisymmetrical loads imposed on the structure include the following, and are as defined in Subsection 3.8.1.3:

(1) Tornado wind

(2) Design wind

(3) Safe shutdown earthquake

(4) Local pipe rupture forces, including local compartmental pressures from ruptured pipes in compartments inside or outside the containment

(5) Pool swell bubble pressure

(6) SRV actuation in the suppression pool

(7) Loadings from embedded steel brackets in the wall and top slab

The containment structure is analyzed for the nonaxisymmetrical pool swell bubble pressure, and nonaxisymmetrical pressures from discharge.

An equivalent static analysis is performed for the nonaxisymmetrical pool swell bubble pressure loading using the peak pressures for this loading.

Input data for nonaxisymmetrical pool swell bubble pressures is described in Subsection 3.8.1.3.5.

Seismic inertial forces (two orthogonal horizontal and one vertical) based on the analysis described in Section 3.7 are applied to the finite element model as equivalent static forces. The resulting moments and forces at various sections of the containment structure are combined by the square-root-of-the-sum-of the-squares (SRSS) method.

The containment wall is shielded from the design wind by the Reactor Building, which completely encloses the structure. Forces from the design wind are transmitted directly to the containment wall through the R/B connections.

The actuation of SRVs results in dynamic loads on the suppression pool boundaries. These dynamic loads are formulated by applying a time function to the attentuated pressures in the suppression pool which result from single and multiple valve discharge. These attenuated pressures are calculated based upon the methodology presented in Appendix 3B. The time function is an oscillation of the bubble pressure within the suppression pool, which is shown in Appendix 3B. The magnitude of the pressure at any point within the pool decreases with time, with the duration of the load being less than 1 second. This pressure time history is represented in terms of an equivalent static load and then used as input for the structural analysis with a dynamic load factor.

The containment wall and containment basemat are extremely stiff steel-lined reinforced concrete structures which form the suppression pool boundary. Thus, effects of fluid-structure interaction upon the total containment building response due to dynamic suppression pool boundary loads are small. Suppression pool boundary loads, defined in Appendix 3B, are applied to the mathematical model as rigid wall loads. The mass of the suppression pool water has been lumped at those node points that form the suppression pool boundary.

3.8.1.4.1.1.2 Axisymmetrical Loads

Axisymmetric loads imposed on the containment structure include the following, and are as defined in Subsection 3.8.1.3:

(1) Structure dead load

(2) Surcharge loads from adjacent structures

(3) Hydrostatic load from probable maximum flood

(4) Hydrostatic load from normal site water table

(5) Hydrostatic load from post-LOCA flooding of the containment

(6) Local dead and live loads from embedded brackets, treated as axisymmetrical loads for overall structural response

(7) Dead and live loads from internal structures imposed on the foundation mat

(8) Normal operating thermal gradients

(9) Abnormal plant thermal gradients (including those from LBL, IBL and SBL)

(10) Preoperational test pressure

(11) Abnormal plant pressure loads (including those from LBL, IBL and SBL)

(12) Normal external pressure load

(13) Safety/relief valve actuation in suppression pool

(14) Pool swell bubble pressure

A LOCA and SRV actuation result in dynamic loads on the suppression pool boundaries. These hydrodynamic loads are formulated by applying a time function to the attenuated pressures in the suppression pool. The attenuated pressures are calculated based on the methodology presented in Appendix 3B. Once the pressure time histories are formulated, they are represented in terms of an equivalent static load and then used as input for the finite element analysis with a dynamic load factor.

3.8.1.4.1.1.3 Major Penetrations

The major penetrations in the containment wall include: (1) the upper drywell equipment and personnel hatches, (2) the lower drywell equipment and personnel tunnels and hatches, (3) the suppression chamber access hatch, and (4) the main steam and feedwater pipe penetrations. The state of stress and behavior of the containment wall around these openings is determined by the use of analytical numerical techniques. The analysis of the area around the penetrations consists of a three-dimensional finite element analysis with boundaries extending to a region where the discontinuity effects of the opening are negligible.

Displacements compatible with the global analysis of the containment are applied at these boundaries. The stresses and strains in the reinforcement, concrete and liner plate are obtained from the local finite element model. The analysis considers concrete cracking and thermal strains.

3.8.1.4.1.1.4 Variation of Physical Material Properties

In the design analysis of the containment, the physical properties of materials are based on the values specified in applicable codes and standards. Reconciliation evaluation will be performed if the as-built properties differ significantly from the design values.

3.8.1.4.1.2 Design Methods

The design of the containment structure is based on the membrane forces, shear forces and bending moments for the load combinations defined in Subsection 3.8.1.3.6. The membrane forces, shear forces and bending moments in selected sections are obtained by the computer program STARDYNE, as described in Subsection 3.8.1.4.1.1. The selected sections are shown in Figure 3H.1-21.

The Concrete Element Cracking Analysis Program (CECAP), described in Section 3B, is used to determine the extent of concrete cracking at these sections, and the concrete and rebar stresses and liner plate strains. The CECAP program models a single element of unit height, unit width, and depth equal to the thickness of the wall or slab. The calculations used in CECAP assume that the concrete is isotropic and linear elastic but with zero tensile strength. CECAP also can calculate the reduced thermal forces and moments due to concrete cracking. However, the redistribution of forces and moments is not calculated. To account for the concrete cracking effects on the redistribution of forces and moments, an iterative procedure described in Subsection 3.8.1.4.1.3 is used.

The input data for the CECAP program consist of the membrane forces, shear forces and bending moments calculated by the STARDYNE analysis. The areas of the reinforcing steel in terms of steel area to concrete cross-section ratio are based on the design shown in Section 3H. The evaluation of containment structural adequacy is shown in Subsection 3.8.1.5.

3.8.1.4.1.3 Concrete Cracking Considerations

The membrane forces, shear forces and bending moments in the containment structure subjected to loads are obtained by applying the STARDYNE computer program to the finite element model that was developed on the basis of an uncracked section. This model is called the uncracked model. In sizing the reinforcing steel or in calculating the rebar stresses, the concrete is not relied upon for resisting tension. Thus, those portions of structures which are either in membrane tension or in flexural tension are cracked to transfer loads from concrete to rebar. The CECAP program, described in Section 3C, is used for calculating the extent of concrete cracking and the stresses in the concrete and in the steel. Because of concrete cracking that leads to stiffness changes, the distribution of forces and moments is different from those calculated from the uncracked model described above. To determine the effects due to concrete cracking, an axisymmetric finite element model which takes into account the concrete cracking, was developed for applying the FINEL computer program to the SIT loadings. The FINEL program (Subsection 3C.4) performs the non-linear static analysis utilizing a stepwise linear iteration solution technique. Within each solution cycle, status of all elements is determined and their stiffness adjusted by the program prior to the next iteration cycle.

The procedures for the design and analysis of the liner plate and its anchorage system are in accordance with the provisions of the ASME Code Section III, Division 2, Subarticle CC-3600. The liner plate analysis considers deviations in geometry due to fabrication and erection tolerances. The strains and stresses in the liner and its anchors are within allowable limits defined by the ASME Code Section III, Division 2, Subarticle CC-3720.

3.8.1.4.1.4 Corrosion Prevention

Type 304L stainless steel or clad carbon steel plate will be used for the containment liner in the wetted areas of the suppression pool as protection against any potential pitting and corrosion on all wetted surfaces and at the water-to-air interface area.

The suppression pool contains air-saturated, stagnant, high purity water and is designed for a 60-year life. The amount of corrosion is based on the annual temperature profile of suppression pool water for a typical plant in southern states under normal operation (Figure 3.8-20). The following conditions can cause the pool temperature to rise above normal:

(1) Reactor core isolation mode: pool temperature can rise 17°C above normal for a total of 165 days during the 60-year lifetime

(2) Suppression pool cooling mode: pool temperature can rise 17°C above normal for a total of 540 days during the 60-year lifetime.

The corrosion allowance for Type 304L stainless steel in air-saturated water for any oxygen level and temperatures up to 316°C for 60 years is 0.12 mm. The major concern has involved the air/water interface area where pitting is most likely to occur. The 0.12 mm corrosion allowance is a small fraction of the stainless steel thickness, which will be a nominal 2.5 mm if clad carbon steel plate is used.

Water used to fill the suppression pool is either condensate or demineralized. No chemicals are added to the suppression pool water.

Observations made on suppression pool water quality over a period of several years indicate that periodic pool cleaning such as by underwater vacuuming will be required, as well as the use of the Suppression Pool Cleanup (SPC) System to maintain water quality standards. The SPC System (Subsection 9.5.9) also acts to maintain purity levels.

An ultrasonic thickness measurement program will be performed to detect any general corrosion at underwater positions. A visual examination for local pitting on the underwater portions of the steel containment will be made at refueling outages using underwater lighting and short focus binoculars. This covers 10% of the surface at the first refueling outage after the start of commercial operation, 5% additional surface approximately two to five years later and 5% at five-year intervals thereafter. If pits are detected at any examination, representative ones are ultrasonically tested and the depth of those large enough for measurement will be determined. Appropriate repairs can be made as required.

3.8.1.4.2 Ultimate Capacity of the Containment

An analysis is performed to determine the ultimate capacity of the containment. The results of this analysis are summarized in Appendix 19F.

3.8.1.5 Structural Acceptance Criteria

For evaluation of the adequacy of the containment structural design, the major allowable stresses of concrete and reinforcing steel for service load combinations and factored load combinations according to ASME Code Section III, Division 2 (except for tangential shear stress carried by orthogonal reinforcement for which a lower allowable is adopted for ABWR) are shown in Table 3.8-2.

The allowable tangential shear strength provided by orthogonal reinforcement without inclined reinforcement is limited to 3.9 MPa for factored load combinations. Inclined reinforcement is not used to resist tangential shear in the ABWR containment. The maximum tangential shear stress calculated for factored load combinations is 3.6 MPa. The maximum membrane shear strain value for governing loading combination is 0.00295 which is based on very conservative calculations with fully cracked concrete and without any consideration of the stiffness provided by steel liner.

The maximum tangential shear strain value for the seismic load alone, based on elastic analysis is 0.000355.

3.8.1.6 Material, Quality Control and Special Construction Techniques.

Materials used in construction of the containment are in accordance with Regulatory Guide 1.136 and ASME Code Section III, Division 2, Article CC-2000. Specifications covering all materials are in sufficient detail to assure that the structural design requirements of the work are met.

3.8.1.6.1 Concrete

All concrete materials are approved prior to start of construction on the basis of their characteristics in test comparisons using ASTM standard methods. Concrete aggregates and cement, conforming to the acceptance criteria of the specifications, are obtained from approved sources. Concrete properties are determined by laboratory tests. Concrete admixtures are used to minimize the mixing water requirements and increase workability. The specified compressive strength of concrete at 28 days, or earlier, is:

Structure	Specified Strength f_c' MPa
Containment	27.56
Foundation Mat	27.56

All structural concrete is batched and placed in accordance with Subarticle CC-2200 and Article CC-4000 of ASME Code Section III, Division 2.

(1) Cement

Cement is Type II conforming to the Specification for Portland Cement (ASTM C 150). The cement contains no more than 0.60% by weight of alkalies calculated as sodium oxide plus 0.658 percent by weight potassium oxide. Certified copies of material test reports showing the chemical composition and physical properties are obtained for each load of cement delivered.

For sites where concrete may come into contact with soils having more than 0.20% water soluble sulfate (as SO_4) of ground- water with a sulfate concentration exceeding 1500 ppm, only Type V cement shall be used unless other suitable means are employed to prevent sulfate attack and concrete deterioration.

(2) Aggregates

All aggregates conform to the Specification for Concrete Aggregates (ASTM C 33).

(3) Water

Water and ice for mixing is clean, with a total solids content of not more than 2000 ppm as measured by ASTM D-1888. The mixing water, including that contained as free water in aggregate, contains not more than 250 ppm of chlorides as Cl as determined by ASTM D-512. Chloride ions contained in the aggregate are included in calculating the total chloride ion content of the mixing water. The chloride content contributed by the aggregate is determined in accordance with ASTM D-1411.

(4) Admixtures

The concrete may also contain an air-entraining admixture and/or a water-reducing admixture. The air-entraining admixture is in accordance with the Specification of Air Entraining Admixtures for Concrete (ASTM C-260). It is capable of entraining 3 to 6% air, is completely water soluble, and is completely dissolved when it enters the batch. Superplasticizers, entraining from 1.5 to 4.5% air, may be used in concrete mixes (f ' = 34.42 MPa, maximum) for congested areas to improve workability and prevent the formation of voids around reinforcement. The water-reducing admixture conforms to the standard specification for Chemical Admixtures for Concrete (ASTM C-494), Types A and D. Type A is used when average ambient temperature for the daylight period is below 21.1^{o}C. Type D is used when average ambient air temperature for the daylight period is 21.1^{o}C and above. Pozzolans, if used, conform to Specification for Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete (ASTM C-618), except that the loss on ignition shall be limited to 6%. Admixtures containing more than 1% by weight chloride ions are not used.

(5) Concrete Mix Design

Concrete mixes are designed in accordance with ACI 211.1 (Recommended Practice for Selecting Proportions for Normal and Heavy Weight Concrete), using materials qualified and accepted for this work. Only mixes meeting the design requirements specified for concrete are used.

3.8.1.6.2 Reinforcing Steel

Reinforcing bars for concrete are deformed bars meeting requirements of the Specification for Deformed and Plain Billet Steel Bars for Concrete Reinforcement (ASTM A-615, Grade 60). Mill test reports, in accordance with ASTM A-615, are obtained from the reinforcing steel supplier to substantiate specification requirements.

The test procedures are in accordance with ASTM A-370, and acceptance standards are in accordance with ASTM A-615.

3.8.1.6.3 Splices of Reinforcing Steel

Sleeves for reinforcing steel mechanical splices conform to ASTM A-513, A-519 or A- 576 Grades 1008 through 1030. Certified copies of material test reports indicating chemical composition and physical properties are furnished by the manufacturer for each sleeve lot.

Placing and splicing of reinforcing bars is in accordance with Article CC-4300 and Subarticle CC-3530 of ASME Code Section III, Division 2.

3.8.1.6.4 Liner Plate and Appurtenances

The materials conform to all applicable requirements of ASME Code Section III, Division 2.

Steel plate is tested at the mill in full conformance to the applicable ASTM specifications, and certified mill test reports are supplied for review and approval. The plate is visually examined for laminations and pitting. Identity of the plate is maintained throughout fabrication.

Dimensional tolerances for the erection of the liner plate and appurtenances are detailed in the Construction Specification based on the structure geometry, liner stability, concrete strength, and the construction methods to be used.

3.8.1.6.5 Quality Control

Quality control procedures are established in the Construction Specification and implemented during construction and inspection. The Construction Specification covers the fabrication, furnishing, and installation of each structural item and specifies the inspection and documentation requirements to ensure that the requirements of ASME Code Section III, Division 2, and the applicable Regulatory Guides are met.

3.8.1.6.6 Welding Methods and Acceptance Criteria for Containment Vessel Lines and Appurtenances

Welding methods and acceptance criteria for the containment vessel liner and appurtenance are the same as those for the steel components of the concrete containment vessel (i.e., personnel air locks, equipment hatches, penetrations, and drywell head) given in Subsection 3.8.2.7.1.

3.8.1.7 Testing and Inservice Inspection Requirements

3.8.1.7.1 Structural Integrity Pressure Test

A structural integrity test of the containment structure will be performed by the COL applicant in accordance with Article CC-6000 of ASME Code Section III, Division 2 and Regulatory Guide 1.136, after completion of the containment construction. The test is conducted at 115% of the design pressure condition of 309.9 kPaG in both the drywell and suppression chamber, simultaneously. A pressure test for the design differential pressure condition of 172.6 kPaG between the drywell and the suppression chamber is also performed where the drywell pressure is greater than the suppression chamber pressure.

During these tests, the suppression chamber and spent fuel pool are filled with water to the normal operational water level. Deflection and concrete crack measurements are made to determine that the actual structural response is within the limits predicted by the design analysis.

In addition to the deflection and crack measurements, the first prototype containment structure is instrumented for the measurement of strains in accordance with the provisions of Subarticle CC-6230 of ASME Code Section III, Division 2. See Subsection 3.8.6.3 for COL license information.

3.8.1.7.2 Preoperational and Inservice Integrated Leak Rate Test

Preoperational and inservice integrated leak rate testing is discussed in Subsection 6.2.6.

3.8.2 Steel Components of the Reinforced Concrete Containment

3.8.2.1 Description of the Containment

The ABWR has a reinforced concrete containment vessel (RCCV) as described in Subsection 3.8.1. This section will describe the following steel components of the concrete containment vessel:

(1) Personnel Air Locks

(2) Equipment Hatches

(3) Penetrations

(4) Drywell Head

3.8.2.1.1 Description of Penetrations

The penetrations through the RCCV include the following.

3.8.2.1.1.1 Personnel Air Locks

Two personnel air locks with an inside diameter sufficient to provide 1850 mm high by 750 mm wide minimum clearance above the floor at the door way are provided. One of these air locks provides access to the upper drywell and the other provides access to the lower drywell via the access tunnel.

Lock and swing of the doors is by manual and automatic means. The locks extend radically outward from the RCCV into the Reactor Building and are supported by the RCCV only. The minimum clear horizontal distance not impaired by the door swing is 1850 mm.

Each personnel air lock has two pressure-seated doors interlocked to prevent simultaneous opening of both doors and to ensure that one door is completely closed before the opposite door can be opened. The design is such that the interlocking is not defeated by postulated malfunctions of the electrical system. Signals and controls that indicate the operational status of the doors are provided. Provision is made to permit temporary bypassing of the door interlock system during plant cold shutdown. The door operation is designed and constructed so either door may be operated from inside the containment vessel, inside the lock, or from outside the containment vessel.

The lock is equipped with a digital readout pressure transducer system to read inside and outside pressures. Quick-acting valves are provided to equalize the pressure in the air lock when personnel enter or leave the containment vessel. The personnel air locks have a double sealed flange with provisions to pressure test the space between the seals of the flange.

3.8.2.1.1.2 Equipment Hatch

Three equipment hatches are provided. One of these serves the upper drywell and the other serves the lower drywell via the access tunnel. The third equipment hatch provides personnel and equipment access to the suppression chamber airspace.

The equipment hatch covers have a double sealed flange with provisions to pressure test the space between the seals of the flange. A means for removing and handling the equipment hatch cover is provided. The hoisting equipment and hoisting guides are arranged to minimize contact between the doors and seals during opening and closing. The equipment hatch includes the electric-motorized hoist with pushbutton control stations, lifting slings, hoist supports, hoisting guides, access platforms, and ladders for access to the dogged position of the door and hoist, latches, seats, dogging devices, and tools required for operation and maintenance of the hatch.

The equipment hatches and covers are entirely supported by the RCCV. Figure 3.8-15 shows general details of the equipment hatch and cover.

3.8.2.1.1.3 Other Penetrations

The RCCV penetrations are categorized into two basic types. These types differ with respect to whether the penetration is subjected to a hot or cold operational environment.

The cold penetrations pass through the RCCV wall and are embedded directly in it. The hot penetrations do not come in direct contact with the RCCV wall but are provided with a thermal sleeve which is attached to the RCCV wall. The thermal sleeve is attached to the process pipe at distance from the RCCV wall to minimize conductive heat transfer to the RCCV wall.

Besides piping penetrations, several electrical penetrations also exist. A description of the various penetrations is given Chapter 8.

3.8.2.1.1.4 Drywell Head

A 10,300 mm diameter opening in the RCCV upper drywell top slab over the RPV is covered with a removable steel torispherical drywell head which is part of the pressure boundary. The drywell head is designed for removal during reactor refueling and for replacement prior to reactor operation using the reactor building crane. One pair of mating flanges is anchored in the drywell top slab and the other is welded integrally with the drywell head. Provisions are made for testing the flange seals without pressurizing the drywell. Figure 19F-4 shows the drywell head.

3.8.2.1.2 Boundaries

The boundaries of the steel components of the RCCV consist of those defined in Paragraph NE-1132, ASME Code Section III, Division 1.

3.8.2.2 Applicable Codes, Standards, and Specifications

3.8.2.2.1 Codes and Standards

In addition to the codes and standards specified in Subsection 3.8.1.2.2, the following codes and standards apply:

(1) American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components, Subsection NE, Class MC.

(2) AISC Manual for Steel Construction.

3.8.2.2.2 Code Classification

The steel components of the RCCV are classified as Class MC in accordance with Subarticle NA-2130, ASME Code Section III.

3.8.2.2.3 Code Compliance

[*The steel components within the boundaries defined in Subsection 3.8.2.1.2, are designed, fabricated, erected, inspected, examined, and tested in accordance with Subsection NE, Class MC Components and Articles NA-4000 and NA-5000 of ASME Code Section III, Division 1.*]*

Structural steel attachments beyond the boundaries established for the steel components of the RCCV are designed, fabricated, and constructed according to the AISC Manual for Steel Construction.

3.8.2.3 Loads and Load Combinations

The applicable loads and load combinations are described in Subsection 3.8.1.3.

3.8.2.4 Design and Analysis Procedures

The steel components of the RCCV are designed in accordance with the General Design Rules of Subarticles NE-3100 (General Design), NE-3200 (Design by Analysis), and NE-3300 (Vessel Design) of ASME Code Section III. For the configurations and loadings which are not explicitly treated in Subarticle NE-3130, the design is in accordance with the applicable Subarticles designated in paragraphs (b) and (d) of Subarticles NE-3130 of ASME Code Section III.

The design of nonpressure-resisting parts is performed in accordance with the general practices of the AISC Manual of Steel Construction.

* See Subsection 3.8.1.1.1 as applied in Subsection 3.8.2.4.1.4.

3.8.2.4.1 Description

Following are individual descriptions of the design and analysis procedures required to verify the structural integrity of critical areas present within the steel components of the RCCV.

3.8.2.4.1.1 Personnel Air Locks

The personnel air lock consists of four main sections: doors, bulkheads, main barrel, and reinforcing barrel with collar. The personnel air locks are supported entirely by the RCCV wall. The lock barrel is welded directly to the containment liner penetration through the RCCV wall. The personnel lock and penetration through the RCCV wall will be analyzed using a finite element computer program. The discontinuity stresses induced by the combination of external, dead, and live loads, including the effects of earthquake loadings, are evaluated. The required analyses and limits for the resulting stress intensities are in accordance with Subarticles NE-3130 and NE-3200 of ASME Code Section III, Division 1.

The piping system and components inside the personnel air locks are Class 2 and are designed in accordance with ASME Code Section NC.

3.8.2.4.1.2 Equipment Hatches

An equipment hatch assembly consists of the equipment hatch cover and the equipment hatch body ring which is imbedded in the RCCV wall and connects to the RCCV liner.

A finite-element analysis model will be used to determine the stresses in the body ring and hatch cover of the equipment hatch. The equipment analysis and the stress intensity limits are in accordance with Subarticles NE-3130 and NE-3200 of ASME Code Section III. The hatch cover with the bolted flange is designed in accordance with Subarticle NE-3326 of ASME Code Section III.

3.8.2.4.1.3 Penetrations

Piping penetrations and electrical penetrations are subjected to various combinations of piping reactions, mechanical, thermal and seismic loads transmitted through the RCCV wall structure. The resulting forces due to various load combinations are combined with the effects of external and internal pressures.

The stresses in the penetrations are evaluated using a finite-element model for stress analysis. For penetrations subjected to cyclic loads, the peak stress intensities are also evaluated. The required analysis and associated stress intensity limits are in accordance with Subarticles NE-3130 and NE-3200 of ASME Code Section III, Division 1.

3.8.2.4.1.4 Drywell Head

The drywell head, consisting of shell, flanged closure and drywell-head anchor system, will be analyzed using a finite-element stress analysis computer program. The stresses, including discontinuity stresses induced by the combination of external pressure or internal pressure, dead load, live load, thermal effects and seismic loads, are evaluated. The required analyses and limits for the resulting stress intensities are in accordance with Subarticles NE-3130 and NE-3200 of ASME Code Section III, Division 1.

[*The compressive stress within the knuckle region caused by the internal pressure and the compression in other regions caused by other loads are limited to the allowable buckling stress values in accordance with Subarticle NE-3222 of ASME Code Section III, Division 1*]*, or Code Case N-284.

3.8.2.5 Structural Acceptance Criteria

The structural acceptance criteria for the steel components of the RCCV (i.e., the basis for establishing allowable stress values, the deformation limits, and the factors of safety) are established by and in accordance with ASME Code Section III, Subsection NE.

In addition to the structural acceptance criteria, the RCCV is designed to meet minimum leakage rate requirements discussed in Section 6.2. Those leakage requirements also apply to the steel components of the RCCV.

The combined loadings designated under “Normal”, “Construction”, “Severe Environmental”, “Extreme Environmental”, “Abnormal”, “Severe Environmental” and “Abnormal/Extreme Environmental” in Table 3.8-1 are categorized according to Level A, B, C and D service limits as defined in NE-3113. The resulting primary and local membrane, bending, and secondary stress intensities, including compressive stresses, are calculated and their corresponding allowable limit is in accordance with Subarticle NE-3220 of ASME Code Section III.

In addition, the stress intensity limits for testing, design and post-LOCA flooding conditions are summarized in Table 3.8-3.

Stability against compression buckling is assured by an adequate factor of safety.

The allowable stress limits used in the design and analysis of nonpressure-resisting components are in accordance with Subsection 3.8.2.2.1 (2).

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The steel components of the RCCV locks, hatches, penetrations, and drywell head will be fabricated from the following materials:

(1) Plate (SA-516 grade 70, SA-240 type 304L, SA-516 grade 60 or 70 purchased to SA-264)

* See Subsection 3.8.1.1.1.

(2) Pipe (seamless SA-333 grade 1 or 6 or SA-106 grade B or SA-312 type 304L)

(3) Forgings (SA-350 grade LFl or LF2)

(4) Bolting (SA-320-L43 or SA-193-B7 bolts with SA-194-7 or A325 or A490 nuts)

(5) Castings (SA-216, grade WCB or SA-352, grade LCB, A27, or 7036)

(6) Cold finished steel (A108 grade 1018 to 1050)

(7) Bar and machine steel (A576, carbon content not less than 0.3%)

(8) Clad (SA-240 type 304L)

The structural steel materials located beyond the containment vessel boundaries are as follows:

(1) Carbon steel (A36 or SA-36)

(2) Stainless steel extruded shapes (SA-479)

The materials meet requirements as specified in Subarticle NE-2000 of ASME Code Section III. The lowest service metal temperature is -1.1°C.

3.8.2.7 Testing and Inservice Inspection Requirements

Leakage of the containment vessel, including the steel components, is described in Subsection 3.8.1.7.

3.8.2.7.1 Welding Methods and Acceptance Criteria

Welding activities shall be performed in accordance with requirements of Section III of the ASME Code. The required nondestructive examination and acceptance criteria are provided in Table 3.8-8.

3.8.2.7.2 Shop Testing Requirements

The shop tests of the personnel air locks include operational testing and an overpressure test. After completion of the personnel air locks tests (including all latching mechanisms and interlocks), each lock is given an operational test consisting of repeated operating of each door and mechanism to determine whether all parts are operating smoothly without binding or other defects. All defects encountered are corrected and retested. The process of testing, correcting defects, and retesting is continued until no defects are detectable.

For the operational test, the personnel air locks are pressurized with air to the maximum permissible code test pressure. All welds and seals are observed for visual signs of distress or noticeable leakage. The lock pressure is then reduced to design pressure and a thick bubble solution is applied to all welds and seals and observed for bubbles or dry flaking as indications of leaks. All leaks and questionable areas are clearly marked for identification and subsequent repair.

During the overpressure testing, the inner door is blocked with holddown devices to prevent unseating of the seals. The internal pressure of the lock is reduced to atmospheric pressure and all leaks are repaired. Afterward, the lock is again pressurized to the design pressure with air and all areas suspected or known to have leaked during the previous test are retested by the bubble technique. This procedure is repeated until no leaks are discernible.

3.8.3 Concrete and Steel Internal Structures of the Concrete Containment

3.8.3.1 Description of the Internal Structures

The functions of the containment internal structures include (1) support of the reactor vessel radiation shielding, (2) support of piping and equipment, and (3) formation of the pressure suppression boundary. The containment internal structures are constructed of reinforced concrete and structural steel. The containment internal structures include the following:

(1) Diaphragm floor

(2) Reactor pedestal

(3) Reactor shield wall

(4) Drywell and equipment pipe support structure

(5) Miscellaneous platforms

(6) Lower drywell equipment tunnel

(7) Lower drywell personnel tunnel

Figures 3.8-17 and 3.8-18 and Figures 1.2-2 through 1.2-13 show an overview of the containment including the internal structures.

The summary report contained in Section 3H.1 contains the figures for the reactor pedestal and the diaphragm slab. Including but not limited to structural steel details, reinforcement details, loads, load combinations, concrete stresses, reinforcement stresses, liner stresses, and structural shell stresses.

3.8.3.1.1 Diaphragm Floor

The diaphragm floor serves as a barrier between the drywell and the suppression chamber. It is a reinforced concrete circular slab, with an outside diameter of 14.5m , and a thickness of 1.2m.

The diaphragm floor is supported by the reactor pedestal and the containment wall. The connection of the diaphragm floor to the containment wall is a fixed support. The diaphragm floor connection to the reactor pedestal is a hinged support. The diaphragm floor is penetrated by 18-508 mm diameter sleeves for the SRV lines.

A 6.4 mm thick, carbon steel liner plate is provided on the bottom of the diaphragm floor, and is anchored to it. The liner plate serves as a form during construction and prevents the bypass flow of steam from the upper drywell to the suppression chamber air space during a LOCA.

3.8.3.1.2 Reactor Pedestal

A composite steel and concrete pedestal provides support for the reactor pressure vessel, the reactor shield wall, the diaphragm floor, access tunnels, horizontal vents, and the lower drywell access platforms. The pedestal consists of two concentric steel shells tied together by vertical steel diaphragms. The regions formed by the steel shells and the vertical diaphragms, except the vents and the vent channels, are filled with concrete. There are ten drywell connecting vent (DCV) channels connecting the upper drywell to the lower drywell and the horizontal vents.

The wetted portion of the exterior surface of the reactor pedestal steel shell in the suppression chamber is clad with stainless steel to provide corrosion protection. The extent of the cladding and the reactor pedestal configuration is provided in
Figure 1.2-2.

3.8.3.1.3 Reactor Shield Wall

The reactor shield wall is supported by the reactor pedestal and surrounds the reactor pressure vessel. Its function is to attenuate radiation emanating from the reactor vessel. In addition, the reactor shield wall provides structural support for the reactor vessel stabilizer, the reactor vessel insulation and the drywell equipment and pipe support structure. Openings are provided in the shield wall to permit the routing of necessary piping to the RPV and to permit in-service inspection of the RPV and piping.

The shield wall is shaped as a right cylinder. The shield wall consists of two concentric steel cylindrical shells joined together by horizontal and vertical steel plate diaphragms. Full depth stiffeners are provided in the reactor shield wall at the attachment locations of major pipe supports, pipewhip restraints and beam supports. The annular region between the outer and inner shells is filled with concrete. The arrangement of the reactor shield wall is provided in Figure 1.2-3.

3.8.3.1.4 Drywell Equipment and Pipe Support Structure

The drywell equipment and pipe support structure (DEPSS) consists of various structural components such as beams and columns. Built-up box shapes are used for beams and columns that must resist torsion and biaxial bending. The beams span between the reactor shield wall and the vertical support columns which are anchored to the diaphragm floor. The DEPSS provides support for piping, pipe whip restraints, mechanical equipment, electrical equipment and general access platforms and stairs.

3.8.3.1.5 Other Internal Structures

3.8.3.1.5.1 Miscellaneous Platforms

Miscellaneous platforms are designed to allow access and to provide support for equipment and piping. The platforms consist of steel beams and grating.

3.8.3.1.5.2 Lower Drywell Equipment Tunnel

A steel tunnel is provided at azimuth 180° for equipment access to the lower dry- well from the Reactor Building. The tunnel has an inside diameter of 4.3m, is 20 mm in thickness, and has a flanged closure at the R/B end. The wetted portion of the tunnel is stainless steel or carbon steel with stainless steel cladding. The tunnel is attached rigidly to the containment wall at one end and the reactor pedestal at the other end and is partially submerged in the suppression pool at normal water level. The tunnel has one or two flexible rings to accommodate differential displacement of the containment wall and reactor pedestal. The configuration of the tunnel and the connection details at the containment wall and reactor pedestal are shown in Figure 1.2-2. Fine motion control rod drive (FMCRD) piping is routed through the tunnel. The tunnel permits entry from the R/B into the lower drywell without exposure to the suppression chamber atmosphere.

3.8.3.1.5.3 Lower Drywell Personnel Tunnel

The lower drywell personnel tunnel is located at azimuth 0° and is similar to the lower drywell equipment tunnel described in Subsection 3.8.3.1.5.2. However, it has a personnel lock at the R/B end. The arrangement and details of the tunnel are shown in Figure 1.2-2.

3.8.3.2 Applicable Codes, Standards, and Specifications

The design of the concrete and steel internal structures of the containment conform to the applicable codes, standards, and specifications and regulations listed in Table 3.8-4 except where specifically stated otherwise.

Structure or Component	Specific Reference Number
Diaphragm Floor	13
Reactor Pedestal	1-12, 15-20
Reactor Shield Wall	1-12, 15-20
DEPSS	15-20
Miscellaneous platforms	15-20

Structure or Component	**Specific Reference Number**
L/D Equipment Tunnel	15-20
L/D Personnel Tunnel	15-20

[*Table 3 of DCD/Introduction identifies the commitments on use of ACI 349 Code, which, if changed, requires NRC Staff review and approval prior to implementation. The applicable portions of the Tier 2 sections and tables, identified on Table 3 of DCD/Introduction for this restriction, are italicized on the sections and tables themselves.*]*

[*Table 4 of DCD/Introduction identifies the commitments on use of Standard ANSI/AISC N690, which, if changed, requires NRC review and approval prior to implementation. The applicable portions of the Tier 2 sections and tables, identified on Table 4 of DCD/Introduction for this restriction, are italicized on the sections and tables themselves.*]†

3.8.3.3 Loads and Load Combinations

3.8.3.3.1 Load Definitions

The loads and applicable load combinations for which the structure is designed depend on the conditions to which the particular structure is subjected.

The containment internal structures are designed in accordance with the loads described in Subsection 3.8.1.3. These loads and the effects of these loads are considered in the design of all internal structures as applicable. The loads within the loading combinations are combined using the absolute sum technique. (Those loads which are defined as reversible in algebraic sign are combined in such a way as to produce the maximum resultant stresses in the structure. All other loads are combined in accordance with their direction of application to the structure.) The loads are defined in Subsection 3.8.1.3 except as follows:

(1) P_o—Pressure loads resulting from the normal operating pressure difference between the drywell (upper and lower) and the suppression chamber of the containment.

(2) Construction Loads—Loads which are applied to the containment internal structures from start to completion of construction. The definitions for D, L and T_o are applicable, but are based on actual construction methods and/or conditions.

* See section 3.5 of DCD/Introduction.

† See section 3.5 of DCD/Introduction.

(3) RV2—Loads from component response or direct fluid forces, on components located in the suppression pool, caused by SRV air cleaning loads.

(4) RBV—Loads due to reactor building vibrations caused by an SRV and LOCA event.

(5) AP—Loads and pressures directly on the reactor shield wall and loads from component response or direct steam flow forces on components located in the reactor vessel shield wall annulus region, caused by a rupture of a pipe within the reactor vessel shield wall annulus region.

(6) SL—Loads from component response or direct fluid forces, on components located in the sloshing zone of a pool or component, caused by the sloshing phenomenon from any dynamic event.

3.8.3.3.2 Load Combination

The load combinations and associated acceptance criteria for concrete and steel internal structures of the containment are listed in Tables 3.8-5 and 3.8-6, respectively.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Diaphragm Floor

The design and analysis procedures used for the diaphragm floor are similar to those used for the containment structure. The diaphragm slab is included in the finite- element model described in Subsection 3.8.1.4.1.1.

3.8.3.4.2 Reactor Pedestal

The reactor pedestal is included in the finite-element model described in Subsection 3.8.1.4.1.1.

The design and analysis is based on the elastic method. All loads are resisted by the integral action of the inner and outer steel shells. The concrete placed in the annulus between the inner and outer shells acts to distribute loads between the steel shells, and provides stability to the compression elements of the pedestal.

3.8.3.4.3 Reactor Shield Wall

The design and analysis procedures used for the reactor shield wall are similar to those used for the reactor pedestal described in Subsection 3.8.3.4.2.

3.8.3.4.4 Drywell Equipment and Pipe Support Structure

The drywell equipment and pipe support structure (DEPSS) is designed using the AISC working stress methods for steel safety-related structures for nuclear facilities (ANSI/AISC N690). The DEPSS is designed to support the deadweight of non-safety-related equipment and support safety-related and non-safety-related piping. The non-safety-related equipment are the drywell cooling coils and fans. The safety-related items include safety-relief valves, mainsteam isolation valves, ECCS isolation valves, and feedwater check valves. In addition the DEPSS provides access platforms such that all of these pieces of equipment can be accessed, inspected, and removed from the drywell if necessary. The DEPSS is a 2 level, 3D space frame, consisting of columns, radial beams, circumferential beams, and steel grating. The DEPSS is shown in Figures 1.2-3, 1.2-3a, 1.2-13a, 1.2-13b, and 1.2-13c.

The DEPSS provides piping systems within the drywell a stable platform for pipe support, and pipe whip restraints. It is designed in accordance with ANSI/AISC-N690. In addition, the criteria given in Subsection 3.7.3.3.4 is applied to the DEPSS. If the criteria can not be met, the COL applicant will generate the ARS at piping attachment points considering the DEPSS as part of the structure using the dynamic analysis methods described in Subsection 3.7.2, or will analyze the piping systems treating the DEPSS as part of pipe support.

Those beams and columns supporting pipe supports will carry piping dynamic loads without buckling and while remaining elastic. Those beams and columns supporting pipe whip restraints allow inelastic deformations due to pipe rupture loads.

All safety-related items which the inelastic beam deformations may effect are evaluated to verify that no required safety function would be compromised.

3.8.3.4.5 Other Internal Structures

The design and analysis procedures used for other internal structures are similar to those used for the drywell equipment and pipe support structure as described in Subsection 3.8.3.4.4.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Drywell Equipment and Pipe Support Structure

[*The structural acceptance criteria for the DEPSS are in accordance with ANSI/AISC-N690.*]*

3.8.3.5.2 Other Internal Structures

[*The structural acceptance criteria for other internal concrete or steel structures are in accordance with ACI-349 and ANSI/AISC-N690, respectively.*]*

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Diaphragm Floor

The materials, quality control, and construction techniques used for the diaphragm floor and liner plate are the same as those used for the containment wall and liner plate in Subsection 3.8.1.6.

* See Subsection 3.8.3.2.

3.8.3.6.2 Reactor Pedestal

The materials conform to all applicable requirements of ANSI/AISC N690 and ACI 349 and comply with the following:

Item	Specification
Inner and outer shells (excluding the portions submerged in the suppression pool)	ASTM A441 or A572
Internal stiffeners	ASTM A441 or A572
Concrete fill	f_c'= 27.56 MPa
Outer shell submerged in the suppression pool	ASTM A533, Type B, Class 2 with SA-240 Type 304 L clad

3.8.3.6.3 Reactor Shield Wall

The materials conform to all applicable requirements of ANSI/ASIC N690 and ACI 349 and comply with the following:

Item	Specification
Inner and outer shells	ASTM A441 or A572
Internal stiffeners	ASTM A441or A572
Concrete fill	f_c'= 27.56 MPa minimum

3.8.3.6.4 Drywell Equipment and Pipe Support Structure

The materials conform to all applicable requirements of ANSI/AISC N690 and comply with the following:

Item	Specification
Structural steel and connections	ASTM A36
High strength structural steel plates	ASTM A572 or A441
Bolts, studs, and nuts (dia. > 19 mm)	ASTM A325
Bolts, studs, and nuts (dia. ≤ 19 mm)	ASTM A307

3.8.3.6.5 Other Internal Structures

The materials conform to all applicable requirements of ANSI/AISC N690 and comply with the following:

Item	Specification
Miscellaneous platforms	Same as Subsection 3.8.3.6.4
Lower drywell equipment tunnel	ASTM A533, Type B, Class 2 with SA-240 Type 304 L clad
Lower drywell personnel tunnel	ASTM A533, Type B, Class 2 with SA-240 Type 304 L clad
Lower drywell floor fill material	A material other than limestone concrete

3.8.3.7 Testing and Inservice Inspection Requirements

A formal program of testing and inservice inspection is not planned for the internal structures except the diaphragm floor, reactor pedestal, and lower drywell access tunnels. The other internal structures are not directly related to the functioning of the containment system; therefore, no testing or inspection is performed.

Testing and inservice inspection of the diaphragm floor, reactor pedestal and lower drywell access tunnels are discussed in Subsection 3.8.1.7.

3.8.3.8 Welding Methods and Acceptance Criteria for Structural and Building Steel

Welding activities shall be accomplished in accordance with written procedures and shall meet the requirements of the American Institute of Steel Construction (AISC) Manual of Steel Construction. The visual acceptance criteria shall be as defined in American Welding Society (AWS) Structural Welding Code D1.1 and Nuclear Construction Issue Group (NCIG) Standard, “Visual Weld Acceptance Criteria for Structural Welding at Nuclear Plants”, NCIG-01.

3.8.4 Other Seismic Category I Structures

Other Seismic Category I structures which constitute the ABWR Standard Plant are the Reactor Building, Control Building and Radwaste Building substructure. Figure 1.2-1 shows the spatial relationship of these buildings. The only other structure in close proximity to these structures is the Turbine Building. It is structurally separated from the other ABWR Standard Plant buildings.

The Seismic Category I structures within the ABWR Standard Plant, other than the containment structures, that contain high-energy pipes are the Reactor Building and Control Building. The steam tunnel walls protect the R/B and C/B from potential impact by rupture of the high-energy pipes. These buildings are designed to accommodate the guard pipe support forces.

The R/B, steam tunnel, Residual Heat Removal (RHR) System, Reactor Water Cleanup (CUW) System, and Reactor Core Isolation Cooling (RCIC) System rooms are designed to handle the consequences of high-energy pipe breaks. The RHR, RCIC, and CUW rooms are designed for differential compartment pressures, with the associated temperature rise and jet force. Steam generated in the RHR compartment from the postulated pipe break exits to the steam tunnel through blowout panels. The steam tunnel is vented to the Turbine Building (T/B) through the seismic interface restraint structure (SIRS). The steam tunnel, which contains several pipelines (e.g., main steam, feedwater, RHR), is also designed for a compartment differential pressure with the associated temperature changes and jet force.

Seismic Category I masonry walls are not used in the design. The ABWR Standard Plant does not contain Seismic Category I pipelines buried in soil.

The COL applicant will identify all Seismic Category I structures. See Subsection 3.8.6.4 for COL license information.

3.8.4.1 Description of the Structures

3.8.4.1.1 Reactor Building Structure

The Reactor Building (R/B) is constructed of reinforced concrete. The R/B has four stories above the ground level and three stories below. Its shape is a rectangle of 59.6m by 56.6m and a height of about 57.9m from the top of the basemat.

The Reinforced Concrete Containment Vessel (RCCV) in the center of the R/B encloses the Reactor Pressure Vessel (RPV). The RCCV supports the upper pool and is integrated with the R/B structure from the basemat up through the elevation of the RCCV top slab. The interior floors of the R/B are also integrated with the RCCV wall. The R/B has slabs and beams which join the exterior wall. Columns support the floor slabs and beams. The fuel pool girders are integrated with the RCCV top slab and with R/B wall-columns. The R/B is a shear wall structure designed to accommodate all seismic loads with its walls. Therefore, frame members such as beams or columns are designed to accommodate deformations of the walls in case of earthquake conditions.

The summary report for the Reactor Building is in Section 3H.1. This report contains a description of the Reactor Building, the loads, load combinations, reinforcement stresses, and concrete stresses at locations of interest. In addition, the report contains reinforcement details for the basemat, seismic walls, and fuel pool girders.

3.8.4.1.2 Control Building

The Control Building (C/B) is located between the Reactor Building and the Turbine Building (see Section 1.2).

The C/B houses the essential electrical, control and instrumentation equipment, the control room for the Reactor and Turbine Buildings, the C/B HVAC equipment, R/B cooling water pumps and heat exchangers, the essential switchgear, essential battery rooms, and the steam tunnel.

The C/B is a Seismic Category I structure that houses control equipment and operation personnel and is designed to provide missile and tornado/hurricane protection. The C/B is constructed of reinforced concrete. The C/B has two stories above the ground level and four stories below. Its shape is a rectangle of 56m by 24m, and a height of about 30.4m from the top of the basemat.

The C/B is a shear wall structure designed to accommodate all seismic loads with its walls and the connected floors. Therefore, frame members such as beams or columns are designed to accommodate deformations of the walls in case of earthquake conditions.

The summary report for the control building is in Section 3H.2. This report contains a description of the control building, the loads, load combinations, reinforcement stresses, and concrete reinforcement details for the basemat, seismic walls, steam tunnel, and floors.

3.8.4.1.3 Radwaste Building Substructure

The Radwaste Building (RWB) Substructure is shown in Section 1.2.

The Radwaste Building is a reinforced concrete structure 60.4m by 41.2m and a height of 29.5m from the top of the basemat. The building consists of a below grade substructure consisting of walls (1.2m thick) and slabs of reinforced concrete forming a rigid box structure which serves as a container to hold radioactive waste in case of an accident. This substructure is located below grade to increase shielding capability and to maximize safety. It is supported on a separate foundation mat whose top is 13.5m below grade. In addition, a reinforced concrete superstructure 15.7m high extends above grade floor level and houses the balance of the radwaste equipment.

The RWB Substructure houses the high and low conductivity tanks, clean-up phase separators, spent resin storage tanks, a concentrated waste storage tank, distillate tank and associated filters, and pumps for the radioactive liquid and solid waste treatment systems.

Although the radwaste superstructure is not a Seismic Category I structure, its major structural concrete walls, slabs, columns and roof are designed to resist Seismic Category I loads.

The summary report for the radwaste building is in Section 3H.3. This report contains a description of the radwaste building, the loads, load combinations, reinforcement stresses, concrete stresses, rebar stresses and required rebar splices at locations of interest. In addition, the report contains reinforcement details for the basemat, seismic walls, and floors.

3.8.4.1.4 Seismic Category I Cable Trays, Cable Tray Supports, Conduit, and Conduit Supports

Electrical cables are carried on continuous horizontal and vertical runs of steel trays supported at intervals by structural steel frames. The tray locations and elevations are predetermined based on the requirements of the electrical cable network. Generally, several trays of different sizes are grouped together and connected to a common support.

The support frame spacing is determined by allowable tray spans, which are governed by rigidity and stress. The frames may be ceiling-supported, or wall-supported, or a combination of both. Various types of frames form a support system with transverse and longitudinal bracing to the nearest wall or ceiling to take the seismic loads.

3.8.4.1.5 Seismic Category I HVAC Ducts and Supports

HVAC ducts are supported at intervals by structural steel frames. The duct locations and elevations are predetermined based on the requirements of the HVAC system.

The support frame spacing is determined by allowable tray spans, which are governed by rigidity and stress. The frames may be ceiling-supported, or wall-supported, or a combination of both. Various types of frames form a support system with transverse and longitudinal bracing to the nearest wall or ceiling to take the seismic loads.

3.8.4.1.6 Seismic Category I Buried Piping, Conduit and Tunnels

Seismic Category I buried piping, conduit, and tunnels shall be designed and analyzed per SRP 3.7.3.

3.8.4.2 Applicable Codes, Standards, and Specifications

3.8.4.2.1 Reactor Building

The major portion of the Reactor Building is not subjected to the abnormal and severe accident conditions associated with a containment. A listing of applicable documents follows:

(1) [*ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures (as modified by Table 3.8-10).*]*

(2) [*ANSI/AISC-N690, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities" (as modified by Table 3.8-9).*]†

* See Subsection 3.8.3.2.

(3) “ASME Boiler and Pressure Vessel Code Section III”, Subsection NE, Division 1, Class MC (for design of main steam tunnel embedment piping anchorage in the R/B and C/B only).

(4) “AWS Structural Welding Code”, AWS D1.1.

(5) “AWS Structural Welding Code”, AWS D12.1.

(6) NRC publications TID 7024 (“Nuclear Reactors and Earthquakes”) and TID 25021 (“Summary of Current Seismic Design Practice for Nuclear Reactor Facilities”).

(7) The inservice inspection requirements for the fuel pool liners in the Reactor Building are in conformance with “ASME Code Section III”, Division 2.

(8) NRC Regulatory Guides:

- (a) Regulatory Guide 1.10 - “Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures”
- (b) Regulatory Guide 1.15 - “Testing of Reinforcing Bars for Category I Concrete Structures”
- (c) Regulatory Guide 1.28 - “Quality Assurance Program Requirements” (Design and Construction)
- (d) Regulatory Guide 1.29 - “Seismic Design Classification”
- (e) Regulatory Guide 1.31 - “Control of Stainless Steel Welding”
- (f) Regulatory Guide 1.44 - “Control of the Use of Sensitized Stainless Steel“
- (g) Regulatory Guide 1.55 - “Concrete Placement in Category I Structures”
- (h) Regulatory Guide 1.60 - “Design Response Spectra for Seismic Design of Nuclear Power Plants”
- (i) Regulatory Guide 1.61 - “Quality Assurance Requirements for the Design of Nuclear Power Plants”
- (j) Regulatory Guide 1.69 - “Concrete Radiation-Shields for Nuclear Power Plants”
- (k) Regulatory Guide 1.76 - “Design Basis Tornado”
- (l) Regulatory Guide 1.142 - “Safety-Related Concrete Structures for Nuclear Power Plants” (Other than Reactor Vessels and Containment)

† See Subsection 3.8.3.2.

(m) Regulatory Guide 1.94 - “Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants”

(n) Regulatory Guide 1.221 - “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants”

(9) ANSI:

(a) ANSI/ASCE 7 - “Minimum Design Loads for Buildings and Other Structures”

(b) ANSI N5.12 - “Protective Coatings (Paint) for the Nuclear Industry”

(c) NQA-1 - “Quality Assurance Program Requirements for Nuclear Facilities and NQA-1a, Addenda to ANSI/ASME NQA-1”

(d) Not Used

(e) Not Used

(f) ANSI N45.4 - “Leakage-Rate Testing of Containment Structures for Nuclear Reactors”

(g) ANSI N101.2 - “Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities”

(h) ANSI N101.4 - “Quality Assurance for Protective Coatings Applied to Nuclear Facilities”

(10) Steel Structures Painting Council Standards

(a) SSPC-PA-1 - “Shop, Field and Maintenance Painting”

(b) SSPC-PA-2 - “Measurement of Paint Film Thickness with Magnetic Gages”

(c) SSPC-SP-1 - “Solvent Cleaning”

(d) SSPC-SP-5 - “White Metal Blast Cleaning”

(e) SSPC-SP-6 - “Commercial Blast Cleaning”

(f) SSPC-SP-10 - “Near-White Blast Cleaning”

(11) ACI-ASCE Committee 326 - “Shear and Diagonal Tension, ACI Manual of Concrete Practice, Part 2.”

(12) Applicable ASTM Specifications for Materials and Standards.

(13) “AASHTO Standard Specifications for Highway Bridges for truck loading area.”

3.8.4.2.2 Control Building

[*Refer to Subsection 3.8.4.2.1.*]*

Add NRC Rules and Regulations Title 10, Chapter 1, Code of Federal Regulations, Part 73.2 and 73.55.

3.8.4.2.3 Radwaste Building Substructure

[*The RWB Substructure shall be designed using the same codes and standards as the reactor building. Refer to Subsection 3.8.4.2.1 for a complete list.*]*

In addition, the non-Seismic Category I reinforced concrete portion of the superstructure is designed according to the seismic provisions of the uniform building code.

3.8.4.2.4 Seismic Category I Cable Tray, Cable Tray Supports, Conduit and Conduit Supports

(1) ANSI/AISC-N690, "Specification for Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facility."

(2) AISI SG-673 - "Cold formed Steel Design Manual."

(3) "ANSI/NEMA FB1, Fittings and Supports for Conduit and Cable Assemblies."

3.8.4.2.5 Seismic Category I HVAC Ducts and Supports

(1) ASME/ANSI AG-1, "Code on Nuclear Air and Gas Treatment."

(2) ANSI/AISC-N690, "Specification for Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facility."

3.8.4.2.6 Welding and Weld Acceptance Criteria

3.8.4.2.6.1 Welding of Electrical Cable Tray and Conduit Supports

Welding activities shall be accomplished in accordance with the AWS Structural Welding Code, Dl.l. The weld visual acceptance criteria shall be as defined in AWS Structural Welding Code Dl.1 and NCIG-01.

3.8.4.2.6.2 Welding of Heating Ventilation and Air Conditioning Supports

Welding activities shall be accomplished in accordance with the AWS Structural Welding Code, Dl.l. The weld visual acceptance criteria shall be as defined in AWS Structural Welding Code Dl.1 and NCIG-01.

* See Subsection 3.8.3.2.

3.8.4.2.6.3 Welding of Refuel Cavity and Spent Fuel Pool Liners

Welding activities shall be accomplished in accordance with the AWS Structural Welding Code, Dl.l. The welded seams of the liner plate shall be spot radiographed where accessible, liquid penetrant and vacuum box examined after fabrication to ensure that the liner does not leak. The acceptance criteria for these examinations shall meet the acceptance criteria stated in Subsection NE-5200 of Section III of the ASME Code.

3.8.4.3 Loads and Load Combinations

3.8.4.3.1 Reactor Building

The temperature and pressure loads caused by a LOCA do not occur on the Reactor Building (R/B). The R/B ventilation system is designed to keep the building within operating design conditions.

3.8.4.3.1.1 Loads and Notations

Loads and notations are as follows:

D = Dead load of structure plus any other permanent load.

L = Conventional floor or roof live loads, movable equipment loads, and other variable loads such as construction loads. The following live loads are used:

- Concrete floors and slabs (including roofs) — 9.8 kPa.
- Stairs, stair platforms, grating floors, and platforms — 4.90 kPa.
- Concrete roofs, live or snow load (not concurrent) —2.452 kPa.
- Construction live load on floor framing in addition to dead weight of floor — 2.452 kPa*.

R_o = Pipe reactions during normal operating or shutdown conditions based on the most critical transient or steady-state condition.

R_a = Pipe reactions under thermal conditions generated by the postulated break and including $R_{o.}$

Y_r = Equivalent static load on a structure generated by the reaction on the broken high-energy pipe during the postulated break and including a calculated dynamic factor to account for the dynamic nature of the load.

* If the actual construction live load is greater than this value, a design check of the structures will be made.

Y_j = Jet impingement equivalent static load on a structure generated by the postulated break and including a calculated dynamic factor to account for the dynamic nature of the load.

Y_m = Missile impact equivalent static load on a structure generated by or during the postulated break, like pipe whipping, and including a calculated dynamic factor to account for the dynamic nature of the load.

W = Wind force (Subsection 3.3.1).

W_t = Tornado load (Subsection 3.3.2) (tornado-generated missiles are described in Subsection 3.5.1.4, and barrier design procedures in Subsection 3.5.3). The design basis tornado missile loads bound those of the design basis hurricane.

P_a = Internal negative pressure of 13.73 kPaD due to tornado; accident pressure at main steam tunnel piping embedment.

B = Uplift forces created by the rise of the ground water table.

F = Internal pressures resulting from flooding of compartments.

E' = Safe shutdown earthquake (SSE) loads as defined in Section 3.7.

T_o = Thermal effects — load effects induced by normal thermal gradients existing through the R/B wall and roof. Both summer and winter operating conditions are considered. In all cases, the conditions are considered of long enough duration to result in a straight line temperature gradient. The temperatures are as follows:

(1) Summer operation:

(a) Air temperature inside building — 49°C

(b) Exterior temperature — 46°C

(2) Winter operation:

(a) Air temperature inside building — 21.1°C

(b) Exterior temperature — (-) 40°C

(3) Winter shutdown

(a) Air temperature inside building - 46°C

(b) Exterior temperature — (-) 40°C

For all cases, as-constructed temperature is 15.6°C.

T_a = Thermal effects (including T_o) which may occur during a design accident at 74°C maximum 30 minutes after LOCA.

U = For concrete structures, the section strength required to resist design loads based on the strength design method described in ACI 318.

H = Loads caused by static or seismic earth pressures.

For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

3.8.4.3.1.2 Load Combinations for Concrete Members

For the load combinations in this subsection, where any load reduces the effects of other loads, the corresponding coefficient for that load shall be taken as 0.9, if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the coefficient for that load shall be taken as zero.

(1) Normal operating conditions — The strength design method is used and the following load combinations are satisfied:

$$U = 1.4\ D + 1.7\ L + 1.3\ T_o + 1.7\ R_o + 1.7\ H + 1.4\ B$$

$$U = 1.4\ D + 1.7\ L + 1.3\ T_o + 1.7\ R_o + 1.7\ H + 1.7\ W$$

For fluid pressure F, replace 1.7 H by 1.7 F in the second equation above.

(2) Abnormal/extreme environmental conditions — The strength design method is used and the following load combinations are satisfied:

$$U = D + L + T_o + R_o + H + B$$

$$U = D + L + T_o + R_o + H + E'$$

$$U = D + L + T_o + R_o + H$$

$$U = D + L + T_o + R_o + H + W_t$$

$$U = D + L + T_a + R_a + 1.5\ P_a + H$$

$$U = D + L + T_a + R_a + P_a + H + E' + (Y_r + Y_j + Y_m)$$

3.8.4.3.1.3 Load Combinations for Steel Members

(1) Normal operating conditions — the elastic working stress design method is used for the following load combinations:

$$S = D + L$$

$$S = D + L + W$$

Since thermal stresses due to T_o and R_o are present and are secondary and self-limiting in nature, the following combinations are also satisfied:

$$1.5\ S = D + L + T_o + R_o$$

$$1.5\ S = D + L + T_o + W$$

In all these load conditions, both cases of L having its full value or being completely absent are checked.

(2) Abnormal/extreme environmental conditions — The elastic working stress design method is used and the following load combinations are satisfied:

$$1.6\ S = D + L + T_o + R_o + E'$$

$$1.6\ S = D + L + T_o + R_o + W_t$$

$$1.6\ S = D + L + T_o + R_o + W_t + H$$

$$1.6\ S = D + L + T_a + R_a + P_a$$

$$1.6\ S = D + L + T_a + R_a + E' + P_a + (Y_j + Y_r + Y_m)$$

In all these load combinations, both cases of L having its full value or being completely absent are checked.

3.8.4.3.2 Control Building and Radwaste Building Substructure

Refer to the loads, notations, and combinations established in Subsection 3.8.4.3.1, except that fluid pressure F, accident pressure P_a, and pipe break loads Y_r, Y_j, Y_m do not exist and the live loads are as follows:

– All concrete floors — 19.61 kPa

– Stairs, stair platforms, grating floors, and platforms — 4.90 kPa

– Roof live or snow load (non concurrent) —2.452 kPa

– Construction live load on floor framing in addition to dead weight of floor — 2.452 kPa*

T_o = thermal effects. As-constructed temperature is 15.6°C. The temperatures inside the building are as follows:

Operating Conditions

Control Room:	summer	23.9°C
	winter	21.1°C
HVAC Room:	summer	35°C
	winter	15.6°C
Other Areas:	summer	23.9°C
	winter	23.9°C

Shutdown condition

Control Room:	summer	26.7°C
	winter	10°C
HVAC Room:	summer	40°C
	winter	10°C
Other Areas:	summer	32.2°C
	winter	10°C

3.8.4.3.3 Seismic Category I Cable Tray, Cable Tray Supports, and Conduit Supports

Loads used in dynamic analysis for tray and conduit supports are the following:

D + L = 1.12 kg/cm used for 46 cm tray, 0.74 kg/cm used for 30 cm and narrower tray

Dynamic loads = SSE plus other RBV dynamic loads.

Load combinations used in dynamic analysis for the tray and conduit supports are the following:

D + L

D + L + SSE + RBV

Where D, L, SSE, and RBV are defined in Subsection 3.8.4.3.1.1.

* If the actual construction live load is greater than this value, a design check of the structures will be made.

3.8.4.3.4 Seismic Category I HVAC Ducts and Supports

Loads and load combinations used for dynamic analysis for HVAC ducts and supports are the following:

D + L + Po

D + L + Po + SSE + RBV

Where D, L, SSE, and RBV are defined in Subsection 3.8.4.3.1.1, and Po is the internal pressure of the HVAC duct.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Reactor Building, Control Building, and Radwaste Building Substructure

[*The Reactor Building, Control Building and Radwaste Building Substructure will be designed in accordance with ACI-349 for concrete structures and ANSI/AISC-N690 specification for steel structures.*]*

The Reactor Building, Control Building, and Radwaste Building Substructure are analyzed using the computer codes listed in Appendix 3C.

The foundation for Category I structures is contained in the summary reports for their respective buildings. The reactor building foundations is contained in Section 3H.1, the control building foundation is in Section 3H.2, and the radwaste building foundation is in Section 3H.3. This summary report contains a section detailing safety factors against sliding, over turning, and floatation.

3.8.4.4.2 Seismic Category I Cable Tray, Cable Tray Supports, and Conduit Supports

All seismic Category I cable trays and conduit supports are designed by one of the methods discussed in Subsection 3.7.3 or by design by rule methods as approved by the NRC. Design by rule methods will be based on documented performance of conduit and cable trays during prior qualification tests or analysis or exposure to natural seismic disturbances. If an analysis is performed it will use one of the codes listed in Appendix 3C.

3.8.4.4.3 Seismic Category I HVAC Ducts and Supports

All seismic Category I HVAC duct and duct supports are designed by one of the methods discussed in Subsection 3.7.3 or by design by rule methods approved by the NRC. Design by rule methods will be based on documented performance of HVAC ducts during prior qualification tests or analysis or exposure to natural seismic disturbances. If an analysis is performed it will use one of the codes listed in Appendix 3C.

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* See Subsection 3.8.3.2.

3.8.4.4.3.1 Cable Tray Supports

Wherever possible, the supporting frames for a tray or group of trays are designed to have adequate rigidity to avoid causing additional amplification of seismic acceleration transmitted by the building structures. Where rigidity cannot be achieved without an excessive increase in support member size, the design of the supports is then based on the amplified seismic load obtained from the floor response spectra.

Thus, two methods are used in design and analysis of cable tray supports.

(1) Rigid Support with Flexible Tray — In this method, trays are modeled as flexible elastic systems and analyzed by the response spectrum method. The resulting reactions are used for the design of the supports.

(2) Flexible Support with Flexible Tray — In this method, the composite system of trays and supports is modeled and analyzed by computer as a multidegree of freedom elastic system. The support motions can be prescribed by the appropriate floor response spectrum. The resulting responses are used to obtain design loads for the supports.

3.8.4.4.3.2 Conduit Supports

The design and analysis of conduit supports are basically the same as for cable tray supports. Since conduits are more flexible and have comparatively less dead load, a rigid support approach is used as described in method (1) of cable tray support design.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 Reactor Building

3.8.4.5.1.1 General Criteria

The first criterion is that the Reactor Building shall provide biological shielding for plant personnel and the public outside of the site boundary. This criterion dictates the minimum wall and roof thicknesses.

The second criterion is that the Reactor Building shall protect the reinforced concrete containment from environmental hazards such as tornado, hurricane and other site proximity-generated missiles. The shielding thicknesses are sufficient for this purpose.

The Reactor Building provides a means for collection of fission product leakage from the reinforced concrete containment following an accident.

The Reactor Building SGTS is designed to keep the compartments surrounding the reinforced concrete containment at a negative pressure even after a LOCA. In order to achieve a maximum in-leakage rate of 50% per day under a pressure differential of 6 mm of water, the reinforcing steel is designed to remain elastic during the SSE load combinations.

3.8.4.5.1.2 Structural and Materials Criteria

[*Structural acceptance criteria are defined in ANSI/AISC-N690 and ACI 349 Codes*.]*

Refer to the materials criteria established in Subsection 3.8.4.2.1 for the strength and materials requirements for the reinforced concrete Reactor Building.

3.8.4.5.2 Control Building

[*Structural acceptance criteria are defined in ANSI/AISC-N690 and ACI 349 Codes.*]* In no case does the allowable stress exceed 0.9 F_y, where F_y is the minimum specified yield stress. The design criteria preclude excessive deformation of the Control Building. The clearances between adjacent buildings are sufficient to prevent impact during a seismic event. The extreme wind (tornado and hurricane) load analysis for this building is the same as the analysis for the Reactor Building.

3.8.4.5.3 Radwaste Building Substructure

[*Structural acceptance criteria are defined in ANSI/AISC-N690 and ACI 349 Codes.*]* In no case does the allowable stress exceed 0.9 F_y, where F_y is the minimum specified yield stress. The design criteria preclude excessive deformation of the Reactor Building. The clearances between adjacent buildings are sufficient to prevent impact during a seismic event.

3.8.4.5.4 Seismic Category I Cable Trays and Conduit Supports

Structural acceptance criteria if the analysis option is selected are defined in ANSI/AISC-N690 Code. In no case does the allowable stress exceed 0.9 F_y where F_y is the minimum specified yield stress.

3.8.4.5.5 Seismic Category I HVAC Duct and Supports

The structural acceptance criteria for HVAC ducts if the analysis option is selected will be in accordance with ANSI/ASME AG-1 Code. The HVAC supports will be in accordance with the ANSI/AISC-N690 code.

3.8.5 Foundations

This section describes foundations for all Seismic Category I structures of the ABWR Standard Plant.

3.8.5.1 Description of the Foundations

The foundations of the Reactor Building and Control Building are reinforced concrete mat foundations.

These two foundation mats are separated from each other by a separation gap of 2m wide to minimize the structural interaction between the buildings.

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* See Subsection 3.8.3.2.

The Reactor Building foundation is a rectangular reinforced concrete mat 56.6m by 59.6m and 5.5m thick. The foundation mat is constructed of cast-in-place conventionally reinforced concrete. It supports the Reactor Building, the containment structure, the reactor pedestal, and other internal structures. The top of the foundation mat is 20.2m below grade.

The containment structure foundation, defined as within the perimeter or the exterior surface of the containment structure, is integral with the Reactor Building foundation. The containment foundation mat details are discussed in Subsection 3.8.1.1.1.

The Control Building foundation is rectangular reinforced concrete mat 24m by 56m by 3.0m. The top of the foundation mat is 20.2m below grade.

The Radwaste Building foundation is a rectangular reinforced concrete mat 60.4m by 41.2m and 2.5m thick. The top of the Radwaste Building mat is 13.5m below grade. The foundation mat is constructed of cast-in-place conventionally reinforced concrete. It supports the Radwaste Building structure.

The foundation for Category 1 structures is contained in the summary reports for their respective buildings. The Reactor Building foundation is contained in Section 3H.1, the Control Building foundation is in Section 3H.2, and the Radwaste Building foundation is in Section 3H.3. This summary report contains a section detailing safety factors against sliding, over turning, and floatation.

3.8.5.2 Applicable Codes, Standards and Specifications

[*The applicable codes, standards, specifications and regulations are discussed in Subsection 3.8.1.2 for the containment foundation and in Subsection 3.8.4.2 for the other Seismic Category I foundations.*]*

3.8.5.3 Loads and Load Combinations

The loads and load combinations for the containment foundation mat are given in Subsection 3.8.1.3. The loads and load combinations for the other Seismic Category I structure foundations are given in Subsection 3.8.4.3.

The loads and load combinations for all Seismic Category I foundations examined to check against sliding and overturning due to earthquakes, winds and tornados, and against flotation due to floods are listed in Table 3.8-7.

* See Subsection 3.8.3.2.

3.8.5.4 Design and Analysis Procedures

The foundations of Seismic Category I structures are analyzed using well-established methods where the transfer of loads from the foundation mat to the supporting foundation media is determined by elastic methods.

Bearing walls and columns carry all the vertical loads from the structure to the foundation mat. Lateral loads are transferred to shear walls by the roof and floor diaphragms. The shear walls then transmit the loads to the foundation mat.

The design of the mat foundations for the structures of the plant involves primarily determining shear and moments in the reinforced concrete and determining the interaction of the substructure with the underlying foundation medium. For a mat foundation supported on soil or rock, the main objectives of the design are (1) to maintain the bearing pressures within allowable limits, particularly due to overturning forces, and (2) to ensure that there is adequate frictional and passive resistance to prevent sliding of the structure when subjected to lateral loads.

The design loads considered in analysis of the foundations are the worst resulting forces from the superstructures and loads directly applied to the foundation mat due to static and dynamic load combinations.

The capability of the foundation to transfer shear with waterproofing will be evaluated. See Subsection 3.8.6.1 for COL license information requirements.

The standard ABWR design is developed using a range of soil conditions as detailed in Appendix 3A. The variations of physical properties of the site-specific subgrade materials will be determined (Subsection 3.8.6.2). Settlement of the foundations, and differential settlement between foundations for the site-specific foundations medium, will be calculated, and safety-related systems (i.e., piping, conduit, etc.) will be designed for the calculated settlement of the foundations. The effect of the site-specific subgrade stiffness and calculated settlement on the design of the Seismic Category I structures and foundations will be evaluated. See Subsection 3.8.6.2 for COL license information requirements.

A detailed description of the analytical and design methods for the Reactor Building foundation mat including the containment foundation, is included in Appendix 3H.

3.8.5.5 Structural Acceptance Criteria

The main structural criteria for the containment portion of the foundation are adequate strength to resist loads and sufficient stiffness to protect the containment liner from excessive strain. The acceptance criteria for the containment portion of the foundation mat are presented in Subsection 3.8.1.5. The structural acceptance criteria for the Reactor Building foundations are described in Subsection 3.8.4.5.

The calculated and allowable factors of safety of the ABWR structures for overturning, sliding, and flotation are shown in Appendix 3H for each foundation mat evaluated according to the following procedures.

The factor of safety against overturning due to earthquake loading is determined by the energy approach described in Subsection 3.7.2.14.

The factor of safety against sliding is defined as:

$$FS = (F_s + F_p)/(F_d + F_h)$$

where F_s and F_p are the shearing and sliding resistance, and passive soil pressure resistance, respectively. F_d is the maximum lateral seismic force including any dynamic active earth pressure, and F_h is the maximum lateral force due to all loads except seismic loads.

The factor of safety against flotation is defined as:

$$FS = F_{DL}/F_B$$

where F_{DL} is the downward force due to dead load and F_B is the upward force due to buoyancy.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The foundations of Seismic Category I structures are constructed of reinforced concrete using proven methods common to heavy industrial construction. For further discussion, see Subsection 3.8.1.6.

3.8.5.7 Testing and Inservice Inspection Requirements

A formal program of testing and inservice inspection is not planned and is not required for the Seismic Category I structures of the ABWR.

3.8.6 COL License Information

3.8.6.1 Foundation Waterproofing

The capability of foundations to transfer shear loads where foundation waterproofing is used will be evaluated (Subsection 3.8.5.4).

3.8.6.2 Site Specific Physical Properties and Foundation Settlement

Physical properties of the site-specific subgrade medium shall be determined and the settlement of foundations and structures, including Seismic Category I, will be evaluated (Subsection 3.8.5.4).

3.8.6.3 Structural Integrity Pressure Result

Each COL applicant will perform the structural integrity test (SIT) of the ABWR containment in accordance with Subsection 3.8.1.7.1. Additionally, the first ABWR containment is considered as a prototype and its SIT performed accordingly. The details of the test and the instrumentation, as required for such a test, will be provided by the first COL applicant for NRC review and approval.

3.8.6.4 Identification of Seismic Category I Structures

The COL applicant will identify all Seismic Category I Structures (Subsection 3.8.4).

3.8.6.5 Loads Associated with Post-DBA Suppression Pool Water Level

The COL applicants will confirm that the suppression pool water level used in the containment loads evaluation is based on the maximum predicted post-accident suppression pool water level rise that can occur concurrent with each of the defined containment loads (Appendix 3B). This load will then be used to update the associated analyses in Section 3.8, Appendix 3G and Appendix 3H.

3.8.6.6 Seismic Category I Buried Piping, Conduits, and Tunnels

The COL applicant shall provide Design and Analysis report for Seismic Category I buried piping, conduits and tunnels per SRP 3.7.3 (Subsection 3.7.3.12).

Table 3.8-1 Load Combinations, Load Factors and Acceptance Criteria for the Reinforced Concrete Containment[*] [†] [‡]

Load Combination Description	No.	D	L	P_t	P_o	P_a	P_i	P_s	T_t	T_o	T_a	E’	W	W’	R_o	R_a	Y[**]	F_L	1^v	SRV[f] ADS	SRV[f] G_{ALL}	LOCA	Acceptance Criteria
Service																							
Test	1	1.0	1.0	1.0					1.0														S
Construction	2	1.0	1.0							1.0			1.0										S
Normal	3	1.0	1.0		1.0					1.0					1.0				1.0		1.0		S
Factored																							
Severe	4	1.0	1.3		1.0					1.0					1.0				1.0		1.0		U
Environmental	5	1.0	1.3		1.0					1.0			1.5		1.0				1.0		1.0		U
Extreme	6	1.0	1.0		1.0					1.0		1.0			1.0				1.0		1.0		U
Environmental	7	1.0	1.0		1.0					1.0				1.0	1.0				1.0				U
Abnormal	8	1.0	1.0			1.5					1.0					1.0			1.0			Note[††]	U
	8a	1.0	1.0				1.5				1.0					1.0			1.0	1.0		Note[††]	U
	8b	1.0	1.0					1.5			1.0					1.0			1.0	1.0		Note[††]	U
	9	1.0	1.0			1.0					1.0					1.25			1.0			Note[††]	U
	9a	1.0	1.0				1.0				1.0					1.25			1.0	1.0		Note[††]	U
	9b	1.0	1.0					1.0			1.0					1.25			1.0	1.0		Note[††]	U
	10	1.0	1.0			1.25					1.0					1.0			1.25			Note[††]	U
	10a	1.0	1.0				1.25				1.0					1.0			1.25	1.25		Note[††]	U
	10b	1.0	1.0					1.25			1.0					1.0			1.25	1.25		Note[††]	U

Table 3.8-1 Load Combinations, Load Factors and Acceptance Criteria for the Reinforced Concrete Containment* † ‡ (Continued)

Load Combination		Load Condition																		SRV^f			Acceptance Criteria
Description	No.	D	L	P_t	P_o	P_a	P_i	P_s	T_t	T_o	T_a	E’	W	W’	R_o	R_a	Y**	F_L	1^v	ADS	G_{ALL}	LOCA	
Abnormal/	11	1.0	1.0			1.25					1.0					1.0			1.0			Note††	U
Severe	11a	1.0	1.0				1.25				1.0					1.0			1.0	1.0		Note††	U
Environmental	11b	1.0	1.0					1.25			1.0					1.0			1.0	1.0		Note††	U
	12	1.0	1.0			1.25					1.0		1.25			1.0			1.0			Note††	U
	12a	1.0	1.0				1.25				1.0		1.25			1.0			1.0	1.0		Note††	U
	12b	1.0	1.0					1.25			1.0		1.25			1.0			1.0	1.0		Note††	U
	13	1.0	1.0							1.0								1.0					U
	14	1.0	1.0							1.0			1.0					1.0					U
Abnormal/	15	1.0	1.0			1.0					1.0	1.0				1.0	1.0		1.0			Note††	U
Extreme	15a	1.0	1.0				1.0				1.0	1.0				1.0	1.0		1.0	1.0		Note††	U
Environmental	15b	1.0	1.0					1.0			1.0	1.0				1.0	1.0		1.0	1.0		Note††	U

* The loads are described in Subsection 3.8.1.3 and acceptance criteria in Subsection 3.8.1.5.

† For any load combination, if the effect of any load component (other than D) reduces the combined load, then the load component is deleted from the load combination.

‡ Since P_a, P_i, P_s, T_a, SRV and LOCA are time-dependent loads, their effects will be superimposed accordingly.

f The sequence of occurrence of SRV loads is given in Appendix 3B.
1^v (G_1 or G_2), ADS and G_{ALL} are not concurrent, when they are indicated in the load combination.

** Y includes Y_j, Y_m and Y_r.

†† LOCA loads, CO, CHUG, VLC and PS are time-dependant loads. The sequence of occurrence is given in Appendix 3B. The load factor for LOCA loads shall be the same as the corresponding pressure load P_a, P_i or P_s.

Table 3.8-2 Major Allowable Stresses in Concrete and Reinforcing Steel

	Concrete		Reinforcing Steel	
	Compression	Tangential Shear	Tension	
Service Load Combination	16.54 MPa	(1) Provided by concrete $\nu_c = 0$	206.8 MPa	
		(2) Provided by orthogonal reinforcement $\nu_{so} = 1.2\sqrt{f'_c} = 1.96$ MPa	310.3 MPa	(For test pressure case)
Factored Load Combination	23.44 MPa	(1) Provided by concrete $\nu_c = 0$	372.4 Mpa	
		(2) Provided by orthogonal reinforcement $\nu_{so} = 2.4\sqrt{f'_c} = 3.92$ MPa		

Table 3.8-3 Stress Intensity Limits

	Primary Stresses			
	Gen. Mem P_m	Local Mem. P_L	Bending & Local Mem. $P + P_L$	Primary & Secondary Stresses $P_L + P + Q$
Test Condition	0.75 Sy	1.15 Sy	1.15 Sy	N/A
Design Condition	1.0 Sm*	1.5 Sm	1.5 Sm	N/A
Post-LOCA Flooding	The larger of 1.2 Smc or 1.0 Sy	The larger of 1.8 Smc or 1.5 Sy	The larger of 1.8 Smc or 1.5 Sy	3 Sm*

* The allowable stress intensity Sm is the Sm listed in Table I-10.0 and Sy is the yield strength listed in Table I-2.0 of Appendix I of ASME Code Section III.

Table 3.8-4 Codes, Standards, Specifications, and Regulations Used in the Design and Construction of Seismic Category I Internal Structures of the Containment

Specification Reference Number	Specification or Standard Designation	Title
1	ACI 301	Specifications for Structural Concrete for Builders
2	ACI 307	Recommended Practice for Concrete Formwork
3	ACI 305	Recommended Practice for Hot Weather Concreting
4	ACI 211.1	Recommended Practice for Selecting Proportions for Normal Weight Concrete
5	ACI 315	Manual of Standard Practice for Detailing Reinforced Normal Weight Concrete
6	ACI 306	Recommended Practice for Cold Weather Concreting
7	ACI 309	Recommended Practice for Consolidation of Concrete
8	ACI 308	Recommended Practice for Curing Concrete
9	ACI 212	Guide for use of Admixtures in Concrete
10	ACI 214	Recommended Practice for Evaluation of Compression Test results of Field Concrete
11	ACI 311	Recommended Practice for Concrete Inspection
12	ACI 304	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
13	[*ACI 349*	*Code Requirements for Nuclear Safety-Related Concrete Structures (as modified by Table 3.8-10)*][*]
14	[*ACI 359*	*ASME Boiler and Pressure Vessel Code, Section III, Division 2, Concrete Reactor Vessels and Containments*][†]
15	[*ANSI/AISCN690*	*Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities (as modified by Table 3.8-9)*][*]
16	AWS D1.1	Structural Welding Code
17	NCIG-02	Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants
18	ANSI/ASME NQA-1-1986	Quality Assurance Program Requirements for Nuclear Facilities
19	Not Used	
20	NRC Regulatory Guide 1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

Table 3.8-4 Codes, Standards, Specifications, and Regulations Used in the Design and Construction of Seismic Category I Internal Structures of the Containment (Continued)

Specification Reference Number	Specification or Standard Designation	Title
21	NRC Regulatory Guide 1.136	Materials for Concrete Containments (Article CC-2000 of the Code for Concrete Reactor Vessels and Containments)
22	NRC Regulatory Guide 1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)

Explanation of Abbreviation

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ANSI	American National Standards Institute
ASME	American Society for Mechanical Engineers
AWS	American Welding Society
NCIG	Nuclear Construction Issues Group
NRC	Nuclear Regulatory Commission

[*] See Subsection 3.8.3.2.

[†] See Subsection 3.8.1.1.1.

Table 3.8-5 Load Combination, Load Factors and Acceptance Criteria for Reinforced Concrete Structures Inside the Containment * †

Load Combination		Load Condition																SRV*ƒ*			Acceptance Criteria‡	
Description	No.	D	L	P_t	P_o	P_a	P_i	P_s	T_t	T_o	T_a	E'	W	W'	R_o	R_a	Y**	1^v	ADS	G_{ALL}	LOCA	
Test	1	1.0	1.0	1.0					1.0													S
Normal	3	1.0	1.0		1.0					1.0					1.0			1.0		1.0		S
	3a	1.4	1.7		1.0										1.7			1.7		1.7		U
	3b	1.05	1.3		1.0					1.3					1.3			1.3		1.3		U
Severe	4a	1.4	1.7		1.0										1.7			1.7		1.7		U
Environmental	4b	1.05	1.3		1.0					1.3					1.3			1.3		1.3		U
	5a	1.4	1.7		1.0								1.7		1.7			1.7		1.7		U
	5b	1.05	1.3		1.0					1.3			1.3		1.3			1.3		1.3		U
Extreme	6	1.0	1.0		1.0					1.0		1.0			1.0			1.0		1.0		U
Environmental	7	1.0	1.0		1.0					1.0				1.0	1.0			1.0		1.0		U
Abnormal	8	1.0	1.0			1.5					1.0					1.0		1.25			Note††	U
	8a	1.0	1.0				1.5				1.0					1.0		1.25	1.25		Note††	U
	8b	1.0	1.0					1.5			1.0					1.0		1.25	1.25		Note††	
Abnormal/	11	1.0	1.0			1.25					1.0					1.0	1.0	1.0			Note††	U
Severe	11a	1.0	1.0				1.25				1.0					1.0	1.0	1.0	1.0		Note††	U
Environmental	11b	1.0	1.0					1.25			1.0					1.0	1.0	1.0	1.0		Note††	U
Abnormal/	15	1.0	1.0			1.0					1.0	1.0				1.0	1.0	1.0			Note††	U
Extreme	15a	1.0	1.0				1.0				1.0	1.0				1.0	1.0	1.0	1.0		Note††	U
Environmental	15b	1.0	1.0					1.0			1.0	1.0				1.0	1.0	1.0	1.0		Note††	U

* The Loads are described in Subsection 3.8.3.3

† (Same as Note †, Table 3.8-6)

‡ S=required strength to resist service loads per ASME Code Section III, Div. 2
U=required strength to resist factored loads per ACI 349

ƒ (Same as Note *ƒ*, Table 3.8-1)

** (Same as Note ** Table 3.8-1)

†† (Same as Note ††, Table 3.8-1)

Table 3.8-6 Load Combination, Load Factors and Acceptance Criteria for Steel Structures Inside the Containment * †

Load Combination Description	No.	Load Condition D	L	P_o	P_a	P_l	P_s	T_o	T_a	E'	W	W'	R_o	R_a	Y**	1^v	SRV^f ADS	G_{ALL}	LOCA	Acceptance Criteria‡
Normal	1	1.0	1.0	1.0																S
	2	1.0	1.0	1.0				1.0						1.0		1.0		1.0		S††
Severe	1	1.0	1.0	1.0							1.0					1.0		1.0		S
Environmental	4	1.0	1.0	1.0												1.0		1.0		S
	5	1.0	1.0	1.0				1.0			1.0		1.0			1.0		1.0		S††
	6	1.0	1.0	1.0				1.0					1.0			1.0		1.0		S††
Extreme	7	1.0	1.0	1.0				1.0				1.0	1.0			1.0		1.0		1.6S
Environmental	8	1.0	1.0	1.0				1.0		1.0			1.0			1.0		1.0		1.6S
Abnormal	9	1.0	1.0		1.0				1.0					1.0		1.0			Note‡‡	1.6S
	9a	1.0	1.0			1.0			1.0					1.0		1.0	1.0		Note‡‡	1.6S
	9b	1.0	1.0				1.0		1.0					1.0		1.0	1.0		Note‡‡	1.6S
Abnormal/	10	1.0	1.0		1.0				1.0					1.0	1.0	1.0			Note‡‡	1.6S
Severe	10a	1.0	1.0			1.0			1.0					1.0	1.0	1.0	1.0		Note‡‡	1.6S
Environmental	10b	1.0	1.0				1.0		1.0					1.0	1.0	1.0	1.0		Note‡‡	1.6S
Abnormal/	11	1.0	1.0		1.0				1.0	1.0				1.0	1.0	1.0			Note‡‡	1.7S
Extreme	11a	1.0	1.0			1.0			1.0	1.0				1.0	1.0	1.0	1.0		Note‡‡	1.7S
Environmental	11b	1.0	1.0				1.0		1.0	1.0				1.0	1.0	1.0	1.0		Note‡‡	1.7S

* (Same as Note *, Table 3.8-5)

† Since P_a, T_a, SRV, and LOCA are time-dependent loads, their effects will be superimposed accordingly.

‡ Allowable Elastic Working Stress (S) is the allowable stress limit specified in Part 1 of ANSI/AISC N690.

ƒ (Same as Note ƒ, Table 3.8-1)

** (Same as Note **, Table 3.8-1)

†† For primary plus secondary stress, the allowable limits are increased by a factor of 1.5.

‡‡ (Same as Note ††, Table 3.8-1)

Table 3.8-7 Load Combinations for Foundation Design

Load Combination No.	Load Condition							
	D	**L**	**H**	**F**	**F'**	**E'**	**W**	**W'**
1	1.0	1.0	1.0	1.0				
2	1.0		1.0	1.0			1.0	
3	1.0	1.0	1.0	1.0		1.0		
4	1.0		1.0	1.0				1.0
5	1.0				1.0			

Nomenclature:

D	Dead Load
F	Buoyant Force of Design Ground Water
F'	Buoyant Force of Design Basis Flood
H	Lateral Earth Pressure
L	Live Load
E'	Basic SSE Seismic Load
W	Wind Load
W'	Tornado Wind

Note:

Load combinations 1 and 3 shall be evaluated for two cases where:

1. Live load is considered to have its full value, and
2. Live load is considered completely absent.
3. For extreme wind loads, the design basis tornado winds and missile loads bound those of the design basis hurricane.

Table 3.8-8 Welding Activities and Weld Examination Requirements for Containment Vessel (1)(2)(3)

Component	Weld Type	NDE Requirements
Containment	Category A. Butt welds (Long'l)	RT
Containment	Category B, Butt welds (Circ.)	RT
Containment	Category C, Butt welds	RT
Containment	Category C, Nonbutt welds	UT or MT of PT
Containment	Category D, Butt welds	RT
Containment	Category D, Nonbutt welds	UT or MT of PT
Containment	Structural attachment welds a.) Butt welds b.) Nonbutt welds	 RT UT or MT or PT
Special Welds	Weld metal cladding	PT

NOTES:

(1) The required confirmation that facility welding activities are in compliance with the requirements will include the following third-party verifications:

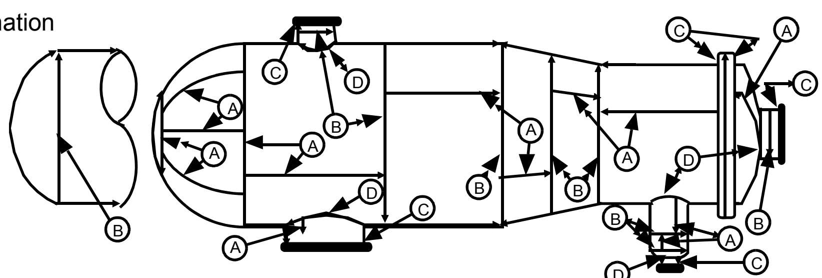
- (a) Facility welding specifications and procedures meet the applicable ASME Code requirements;
- (b) Facility welding activities are performed in accordance with the applicable ASME Code requirements;
- (c) Welding activities related records are prepared, evaluated and maintained in accordance with the ASME Code requirements;
- (d) Welding processes used to weld dissimilar base metal and welding filler metal combinations are compatible for the intended applications;
- (e) The facility has established procedures for qualifications of welders and welding operators in accordance with the applicable ASME Code requirements;
- (f) Approved procedures are available and use for preheating and post heating of welds, and those procedures meet the applicable requirements of the ASME Code;
- (g) Completed welds are examined in accordance with the applicable examination method required by the ASME code.

(2) Radiographic film will be reviewed and accepted by the licensee's nondestructive examination (NDE), Level III examiner prior to final acceptance.

(3) The NDE requirements for containment vessels will be as stated in subarticle NE-5300 of Section III of the ASME Code.

LEGEND:

RT – Radiographic Examination
MT – Magnetic Particle Examination
PT – Liquid Penetrant Examination
UT – Ultrasonic Examination



Categories A, B, C, and D Welded Joint Typical Locations

Table 3.8-9 Staff Position on the Use of Standard ANSI/AISC N690 Nuclear Facilities-Steel Safety-Related Structures

[*The use of the Standard ANSI/AISC N690 for the design, fabrication and erection of safety-related structures in ABWR is acceptable when supplemented by the following provisions.*

(1) In Section Q1.0.2, the definition of secondary stress should apply to stresses developed by temperature loading only.

(2) Add the following notes to Section Q1.3:

"When any load reduces the effects of other loads, the corresponding coefficient for that load should be taken as 0.9, if it can be demonstrated that the load is always present or occurs simultaneously with other loads. Otherwise, the coefficient for that load should be taken as zero."

"Where the structural effects of differential settlement are present, they should be included with the dead load 'D'."

For structures or structural components subjected to hydrodynamic loads resulting from LOCA and/or SRV actuation, the consideration of such loads should be as indicated in the Appendix to SRP Section 3.8.1. Any fluid structure interaction associated with these hydrodynamic loads and those from the postulated earthquake(s) should be taken into account."

(3) The stress limit coefficients (SLC) for compression in Table Q1.5.7.1 should be as follows:

1.3 instead of 1.5 stated in footnote (c) in load combinations 2, 5 and 6.
1.4 instead of 1.6 in load combinations 7, 8 and 9.
1.6 instead of 1.7 in load combination 11.

(4) Add the following note to Section Q1.5-8:

"For constrained (rotation and/or displacement) members supporting safety-related structures, systems or components the stresses under load combinations 9, 10 and 11 should be limited to those allowed in Table Ql.5.7.1 as modified by provision 3 above. Ductility factors of Table Ql.5.8.1 (or provision 5 below) should not be used in these cases."

(5) For ductility factors 'μ' in Sections Q1.5.7.2 and Q1.5.8, substitute provisions of Appendix A, II.2 of SRP Section 3.5.3 in lieu of Table Q1.5.8.1.

(6) In load combination 9 of Section Q2.1, the load factor applied to load P_a should be 1.5/1.1 = 1.37, instead of 1.25.]*

(7) Sections Q1.24 and Ql.25 should be supplemented with the following requirements regarding painting of structural steel:

(a) Shop painting to be in accordance with Section M3 of AISC LRFD Specification.
(b) All exposed areas after installation to be field painted (or coated) in accordance with the applicable portion of Section M3 of AISC, LRFD Specification.
(c) The quality assurance requirements for painting (or coating) of structural steel to be in accordance with ANSI N101.4 as endorsed by Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants."

* See Subsection 3.8.3.2.

Table 3.8-10 Staff Position on Steel Embedments

[*The use of Appendix B to ACI 349 for the design of steel embedments for safety-related concrete structures in ABWR is acceptable when supplemented by the following provisions.*

(1) Section B.4.2 - Tension and Figures B.4.1 and B.4.2.

This section and the figures specify that the tensile strength of concrete for any anchorage can be calculated by a 45 degree failure cone theory. The staff has disseminated the German test data questioning the validity of the 45 degree failure cone theory to licensees, A/Es, bolt manufacturers, and the code committee members in its meetings with them. The data indicated that the actual failure cone was about 35 degree and the use of the 45 degree cone theory could be unconservative for anchorage design, especially for anchorage of groups of bolts. The Code Committee, having gone through some research of its own, recently agreed with the staff's position. Changes to this section are in the making by the Code Committee. In the meantime, the staff position on issues related to this Section is to ensure adoption of design approaches consistent with the test data through case by case review.

(2) Section B.5.1.1 - Tension

This section states a criterion for ductile anchors. The criterion is that the design pullout strength (force) of the concrete as determined in Section B.4.2 exceeds the minimum specified tensile strength (force) of the steel anchor,. Any anchor that meets this criterion is qualified as a ductile anchor and, thus, a low safety factor can be used. The staff believes that the criterion is deficient in two areas. One is that the design pullout strength of the concrete so calculated is usually higher than the actual strength, which has been stated in Section B.4.2 above. The other is that anchor steel characteristics are not taken into consideration. For example, the Drillco Maxi-Bolt Devices, Ltd. claims that its anchors are ductile anchors and, thus, can use a low safety factor. The strength of the Maxi-Bolt is based on the yield strength of the anchor steel, which is 723.9 MPa. The embedment length of the anchor, which is used to determine the pullout strength of the concrete, is based on the minimum specified tensile strength of the anchor steel of 861.8 MPa. The staff believes that the 19% margin (125/105) for the embedment length calculation is insufficient considering the variability of parameters affecting the concrete cone strength. The staff also questions the energy absorption capability (deformation capability after yield) of such a high strength anchor steel. Therefore, in addition to the position taken with regard to Section B.4.2 above, the staff will review vendor or manufacturer specific anchor bolt behaviors to determine the acceptable design margins between anchor bolt strengths and their corresponding pullout strengths based on concrete cones.

Section B.5.1.1(a) - Lateral bursting concrete strength

This section states that the lateral bursting concrete strength is determined by the 45 degree concrete failure cone assumption. Since this assumption is wrong and likely to be replaced as stated before, the staff believes that the lateral bursting concrete strength determination is also wrong and needs to be replaced. The staff will review the anchor bolts and lateral bursting force created by the pulling of anchor bolts against test data to determine if adequate reinforcement against lateral bursting force need to be provided on a case by case basis.

(3) Section B.5.1.2.1 - Anchor, Studs, or Bars

This section states that the concrete resistance for shear can be determined by a 45 degree half-cone to the concrete free surface from the centerline of the anchor at the shearing surface. Since the 45 degree concrete failure cone for tension has been found to be incorrect, the staff believes that the use of the 45 degree half-cone for shear should be re-examined. In the meantime, the staff will review the adequacy of shear capacity calculation of concrete cones on a case by case basis with emphasis on methodology verification through vendor specific test data.

Table 3.8-10 Staff Position on Steel Embedments

(4) Section B.5.1.2.2(c) - Shear Lugs

This section states that the concrete resistance for each shear lug in the direction of a free edge shall be determined based on the 45 degree half-cone assumption to the concrete free surface from the bearing edge of the shear lug. This is the same assumption as used in Section B.5.1.2.1 and the staff has the same comment as stated in that section. Therefore, the staff position related to the design of shear lugs is to perform case-by-case reviews. The staff review will emphasize methodology verification through sdpecific test data.

(5) Section B.7.2 - Alternative design requirements for expansion anchors

This section states that the design strength of expansion anchors shall be 0.33 times the average tension and shear test failure loads, which provides a safety factor of 3 against anchor failure. The staff position on safety factor for design against anchor failure is 4 for wedge anchors and 5 for shell anchors unless a lower safety factor can be supported by vendor specific test data.

(6) Anchors in tension zone of supporting concrete

When anchors are located within a tensile zone of supporting concrete, the anchor capacity reduction due to concrete cracking shall be accounted for in the anchor design.][*]

[*] See Subsection 3.8.3.2.

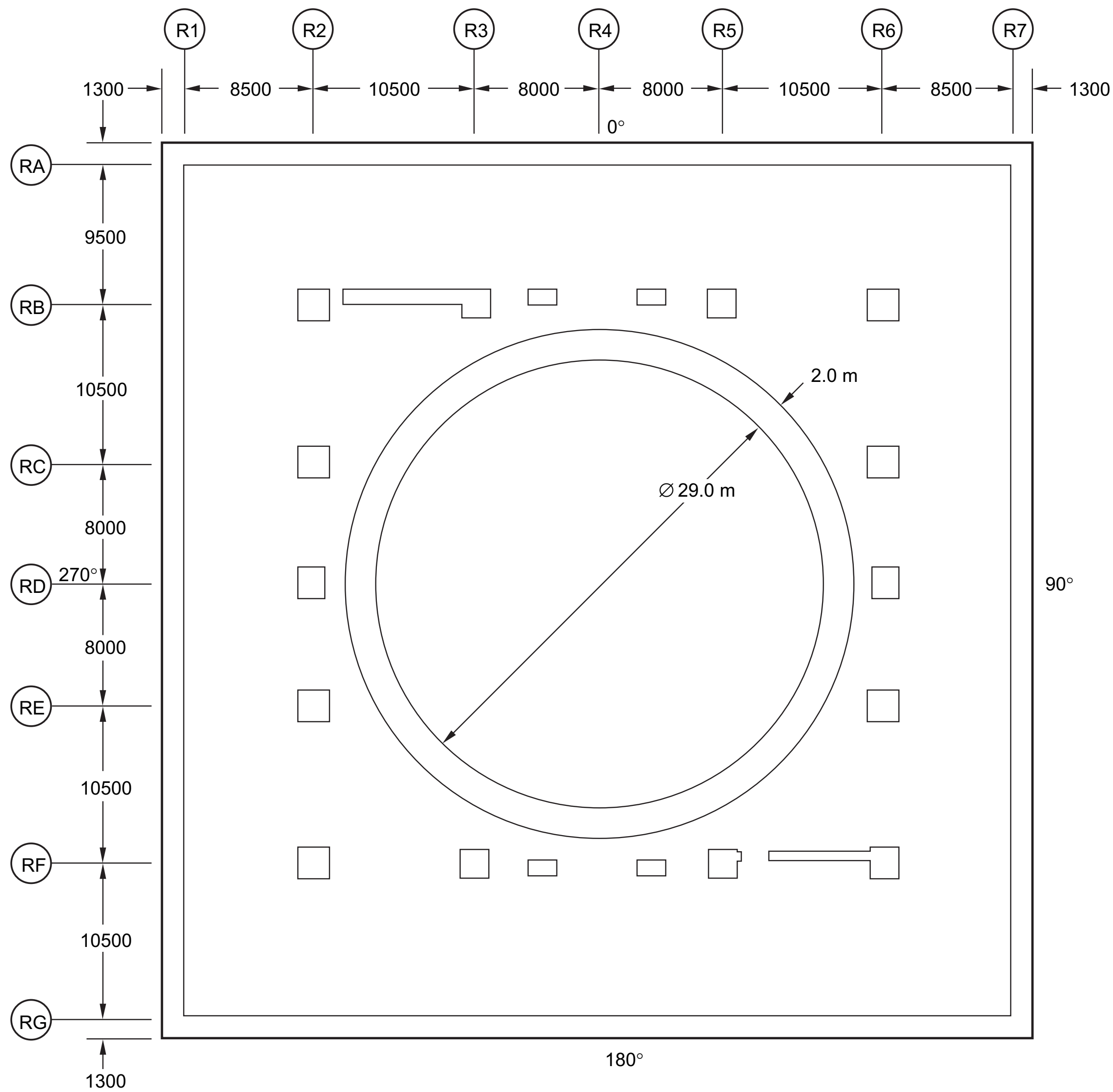


Figure 3.8-1 Reactor Building Arrangement Floor B2F Elevation –1700 mm

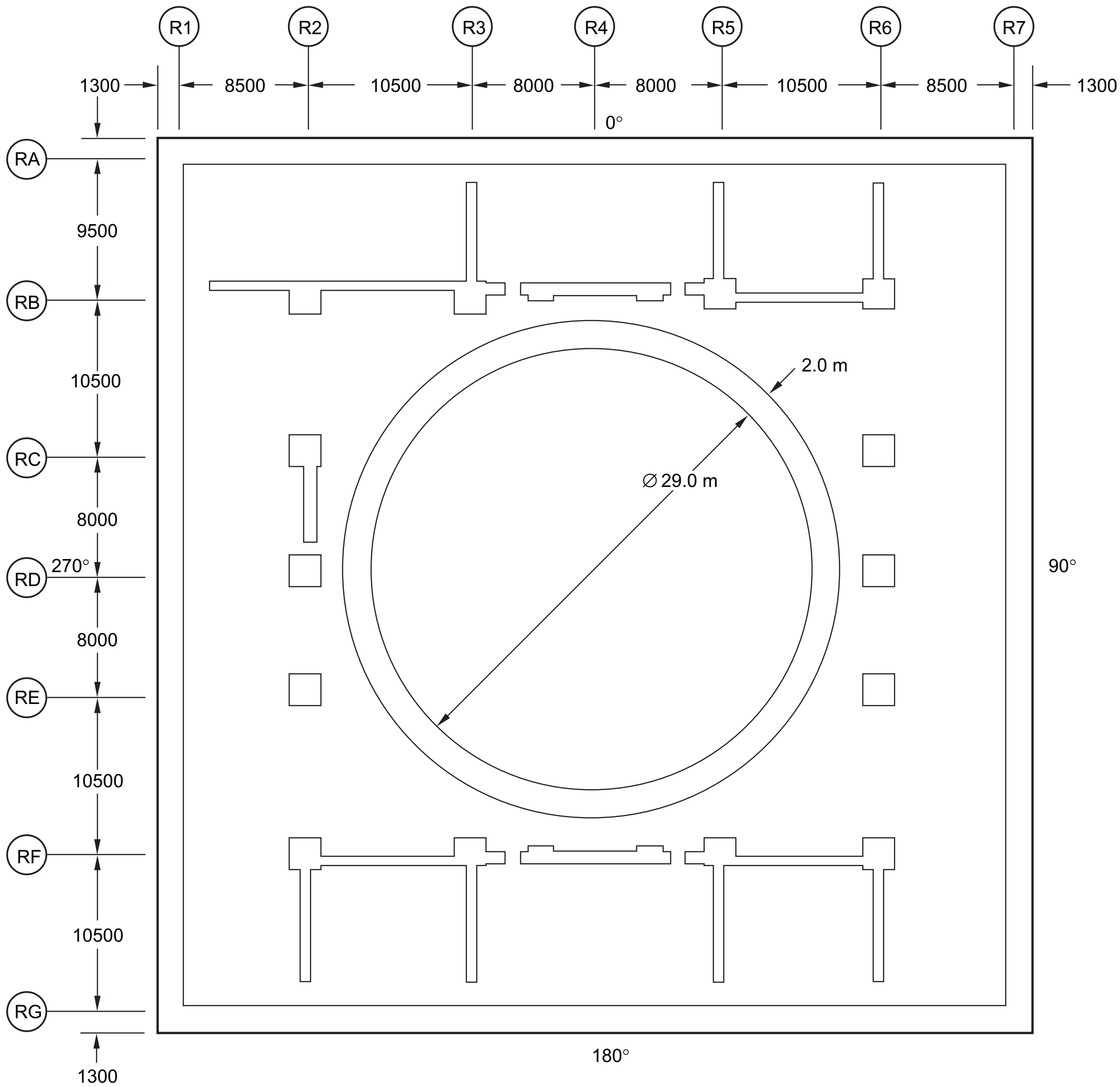


Figure 3.8-2 Reactor Building Arrangement Floor B3F Elevation –8200 mm

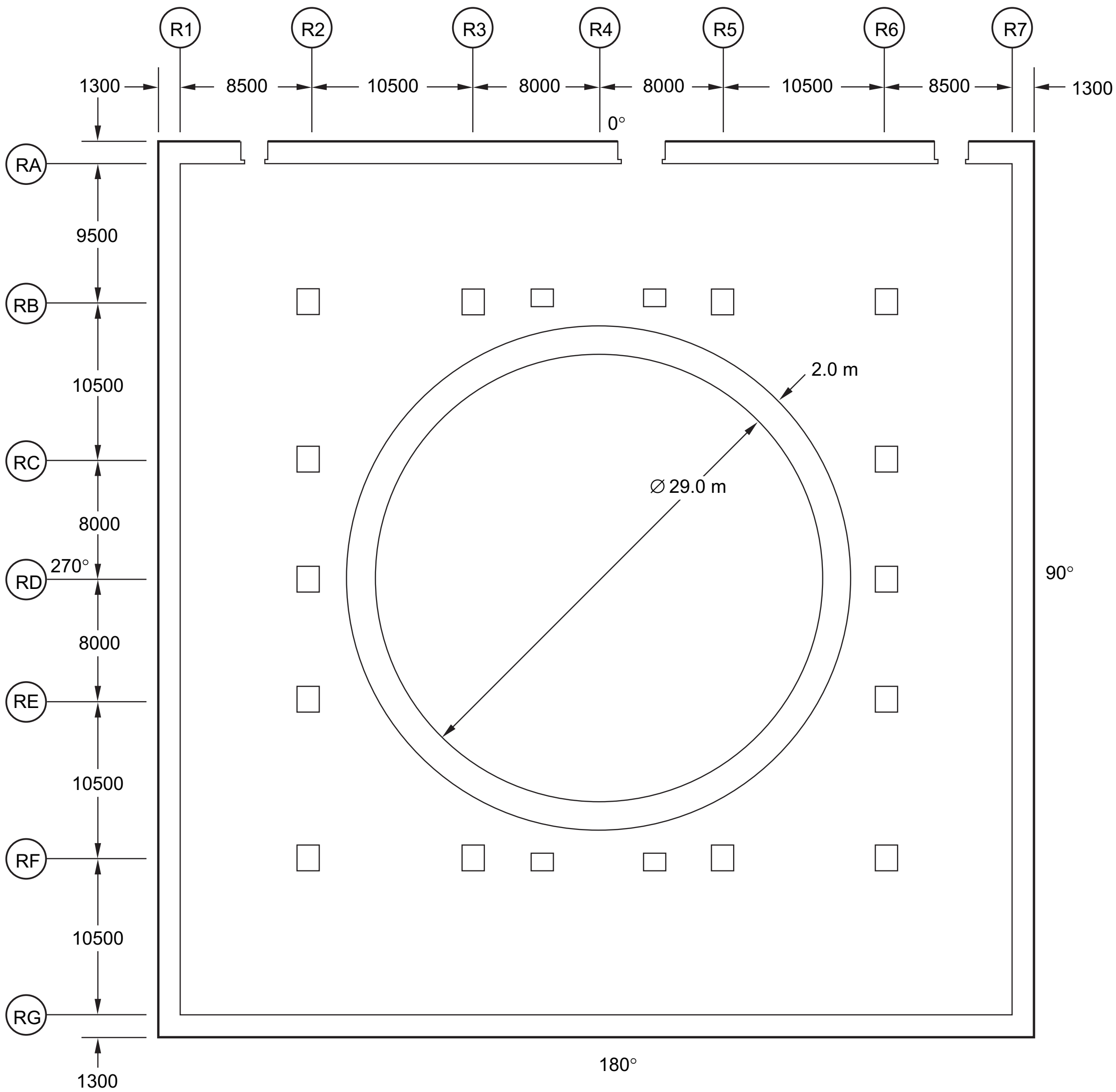


Figure 3.8-3 Reactor Building Arrangement Floor B1F Elevation 4800 mm

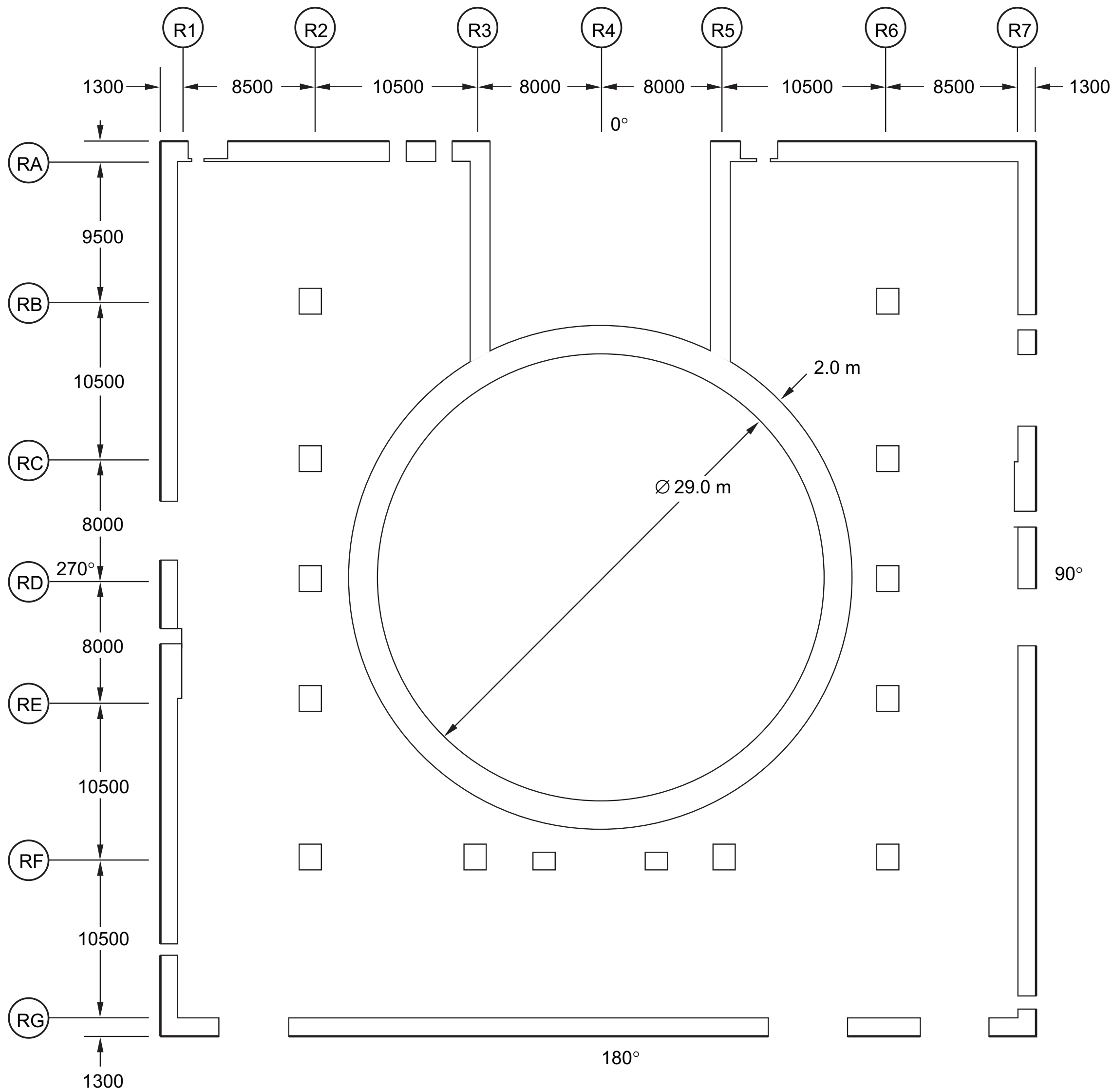


Figure 3.8-4 Reactor Building Arrangement Floor 1F Elevation 12300 mm

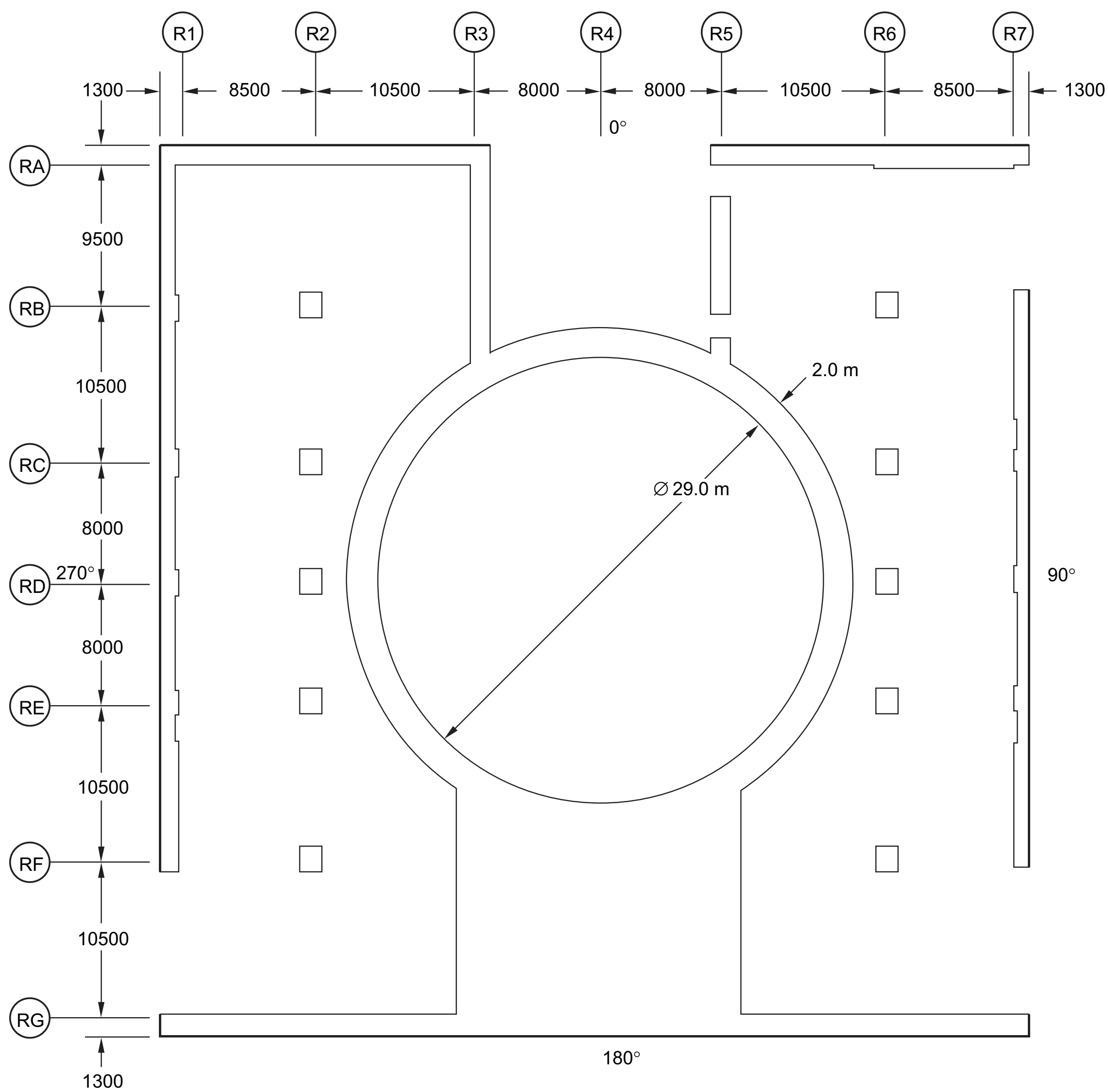


Figure 3.8-5 Reactor Building Arrangement Floor 2F Elevation 18100 mm

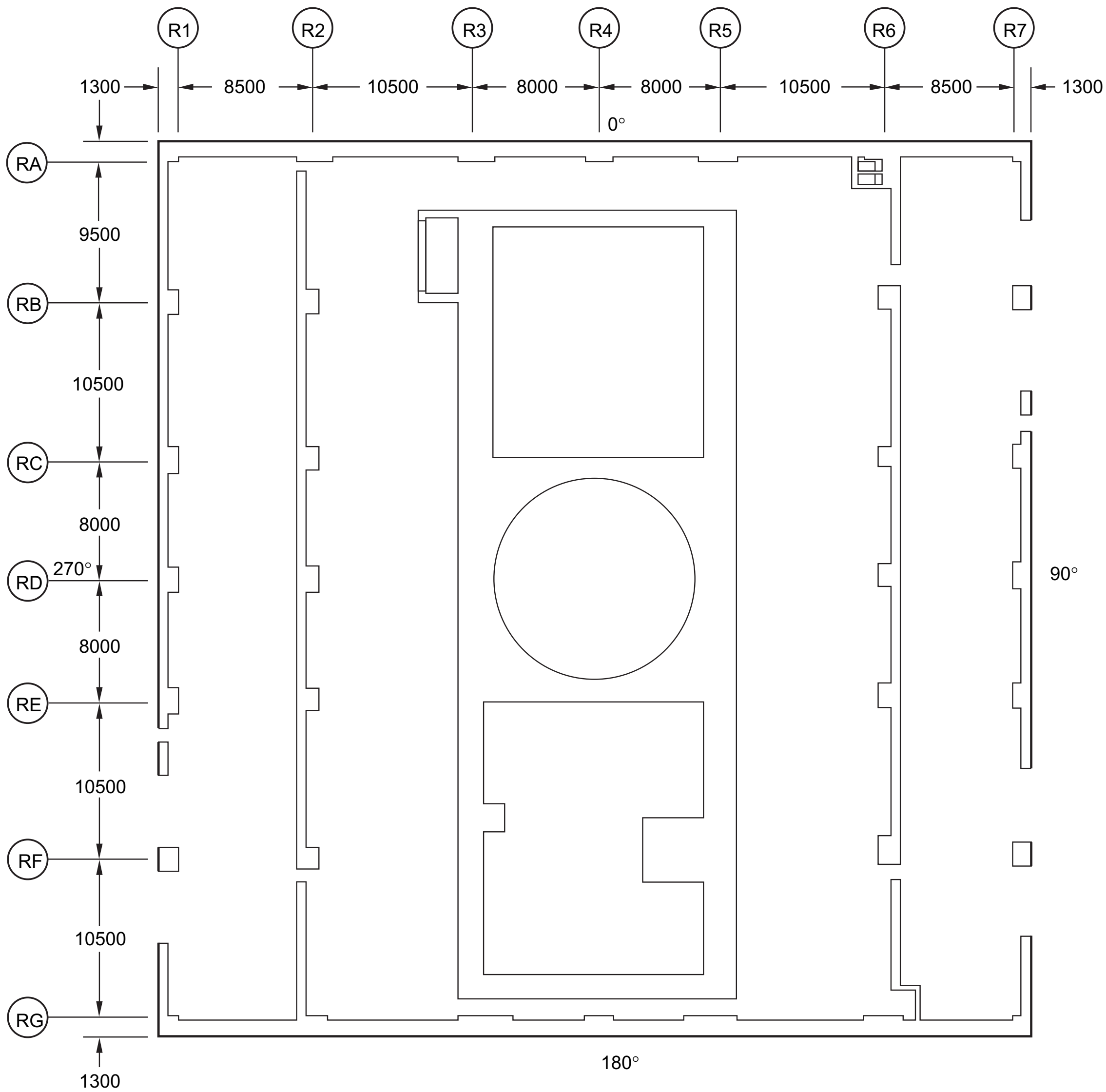


Figure 3.8-6 Reactor Building Arrangement Floor 3F Elevation 23500 mm

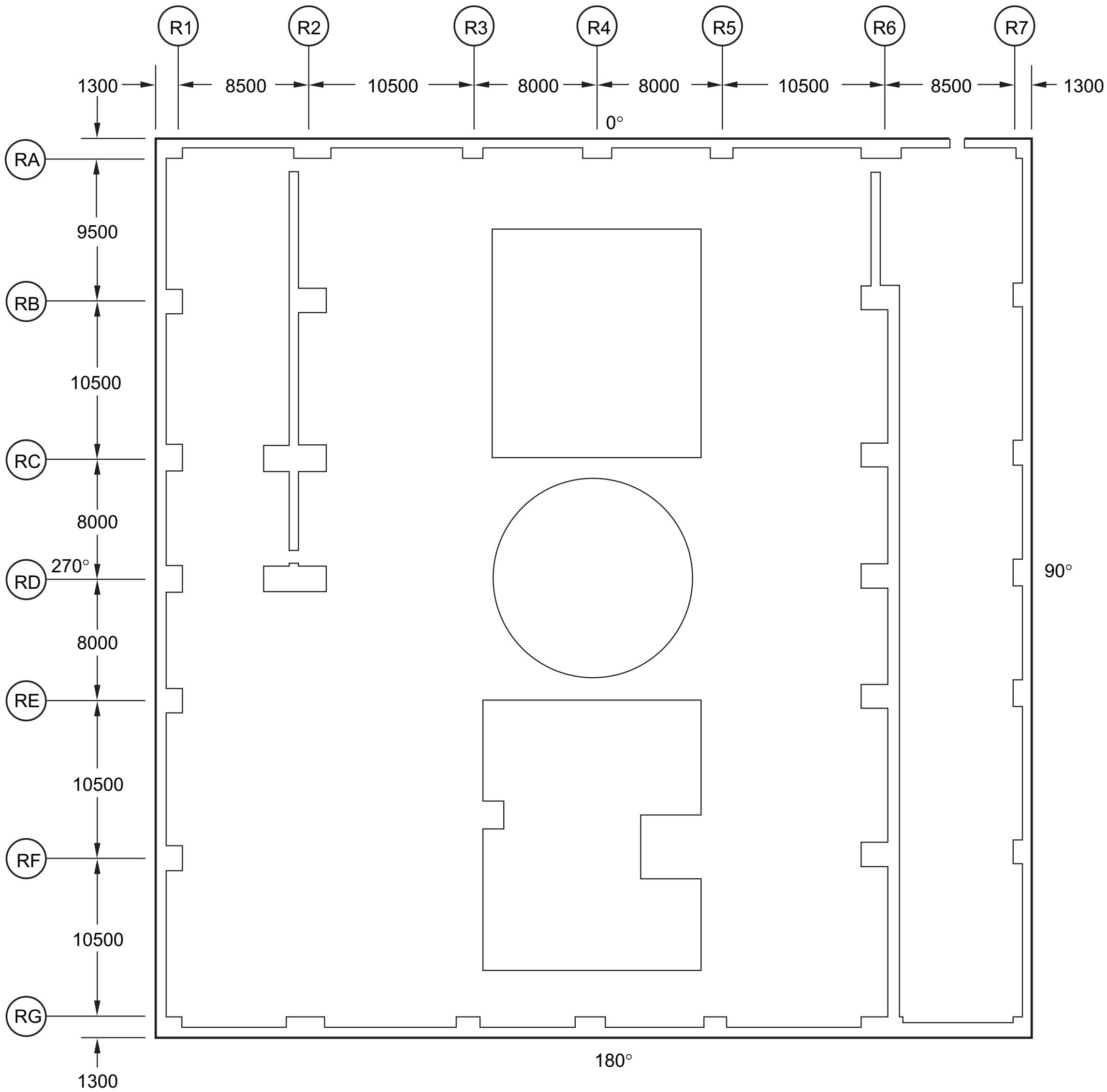


Figure 3.8-7 Reactor Building Arrangement Floor 4F Elevation 31700 mm

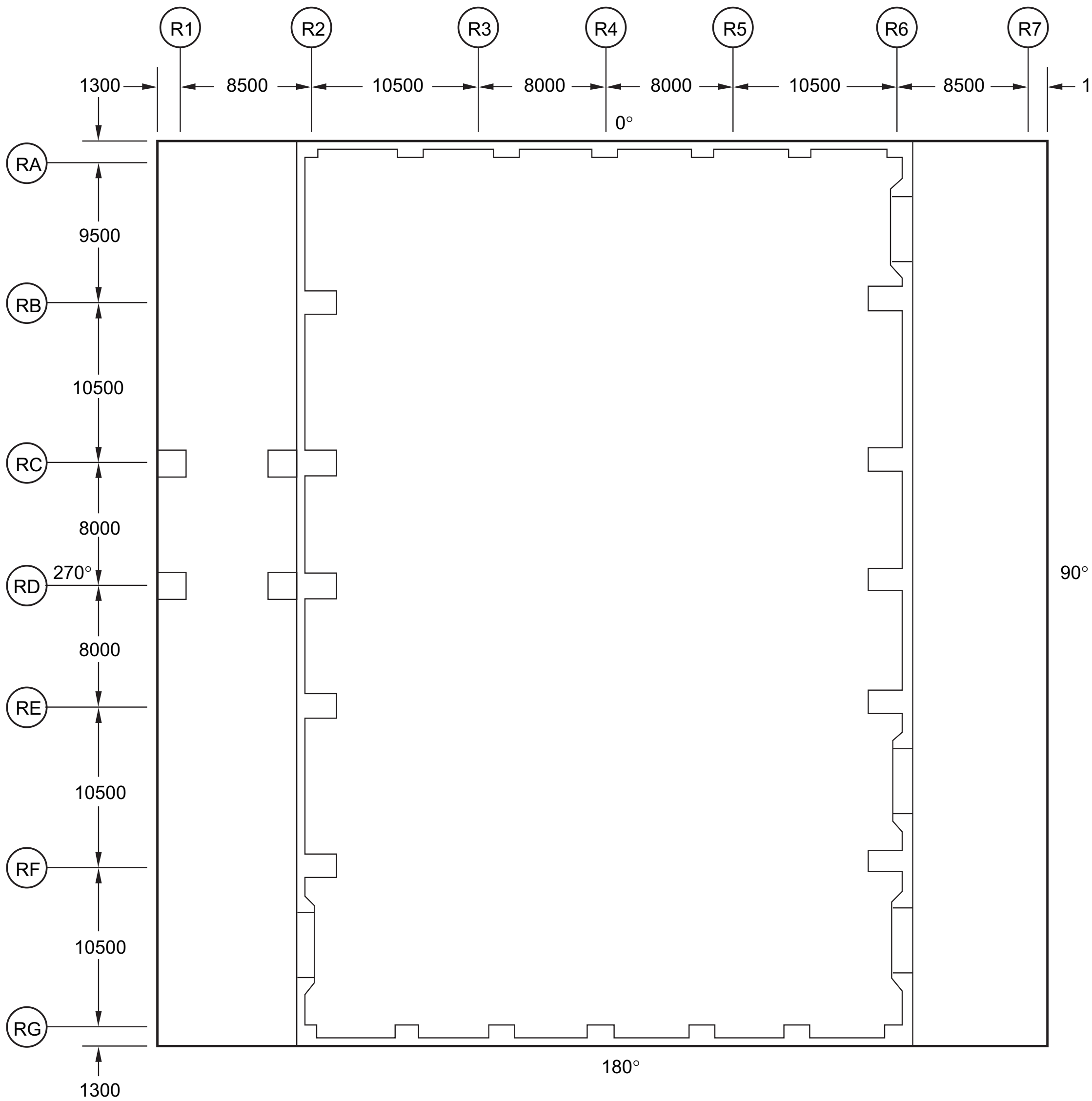


Figure 3.8-8 Reactor Building Arrangement Elevation 38200 mm

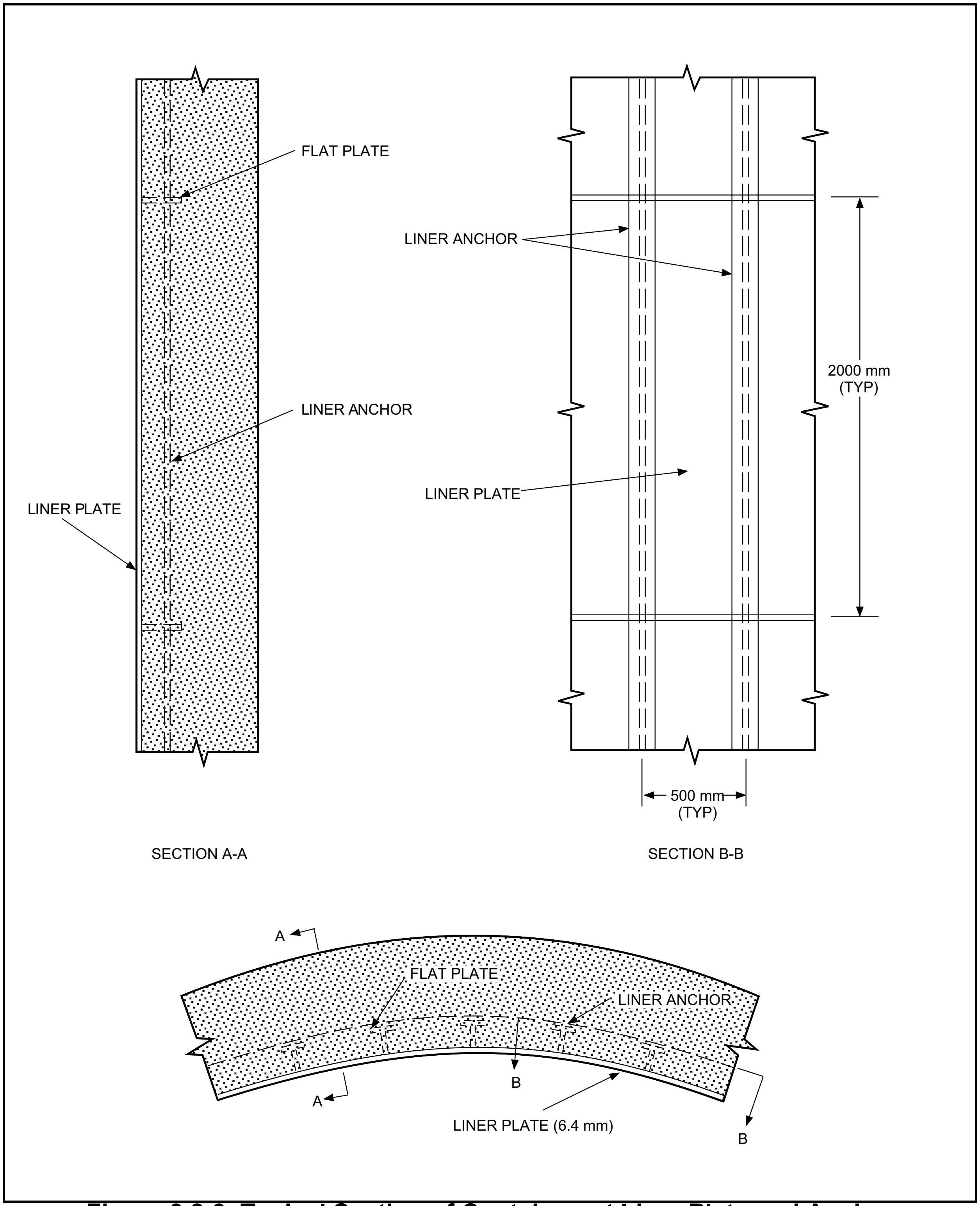


Figure 3.8-9 Typical Section of Containment Liner Plate and Anchor

Figure 3.8-10 Not Used

Figure 3.8-11 Not Used

Figure 3.8-12 Not Used

Figure 3.8-13 Not Used

Figure 3.8-14 Not Used

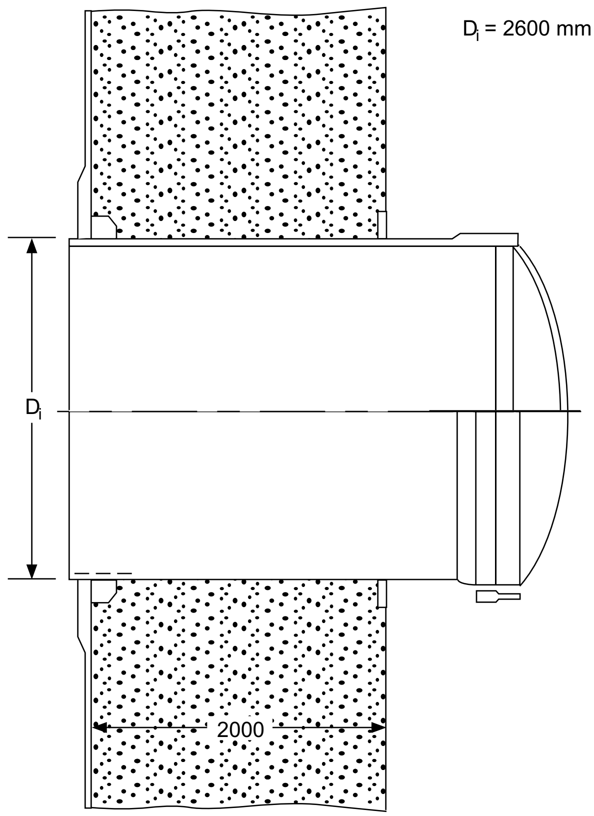


Figure 3.8-15 Reactor Building—Containment Upper Drywell Equipment Hatch

Figure 3.8-16 Not Used

1. DRYWELL HEAD
2. RCCV WALL
3. DRYWELL HEAD FLANGE
4. RPV STABILIZER
5. PENETRATION
6. SHIELDING LIP
7. VERTICAL CHANNELS
8. LINER PLATE
9. SUPPRESSION CHAMBER ACCESS HATCH
10. DIAPHRAGM FLOOR (D/F)
11. REACTOR SHIELD WALL
12. U/D EQUIPMENT HATCH
13. U/D PERSONNEL LOCK
14. DRYWELL CONNECTING VENTS
15. LOWER DRYWELL (L/D)
16. UPPER DRYWELL (U/D)
17. REACTOR PRESSURE VESSEL
18. SUPPRESSION CHAMBER AIRSPACE
19. EQUIPMENT PLATFORM
20. L/D EQUIPMENT TUNNEL AND HATCH
21. L/D PERSONNEL TUNNEL AND LOCK
22. SUPPRESSION POOL
23. RPV SKIRT
24. RPV SUPPORT BRACKET
25. RPV PEDESTAL
26. SRVDL QUENCHER
27. QUENCHER SUPPORT
28. HORIZONTAL VENT
29. BOTTOM LINER PLATE
30. L/D SEAL PLATE
31. RETURN LINE
32. RCCV TOP SLAB
33. BASEMAT
34. REACTOR WELL
35. REFUELING SEAL
36. L/D FLOODERS

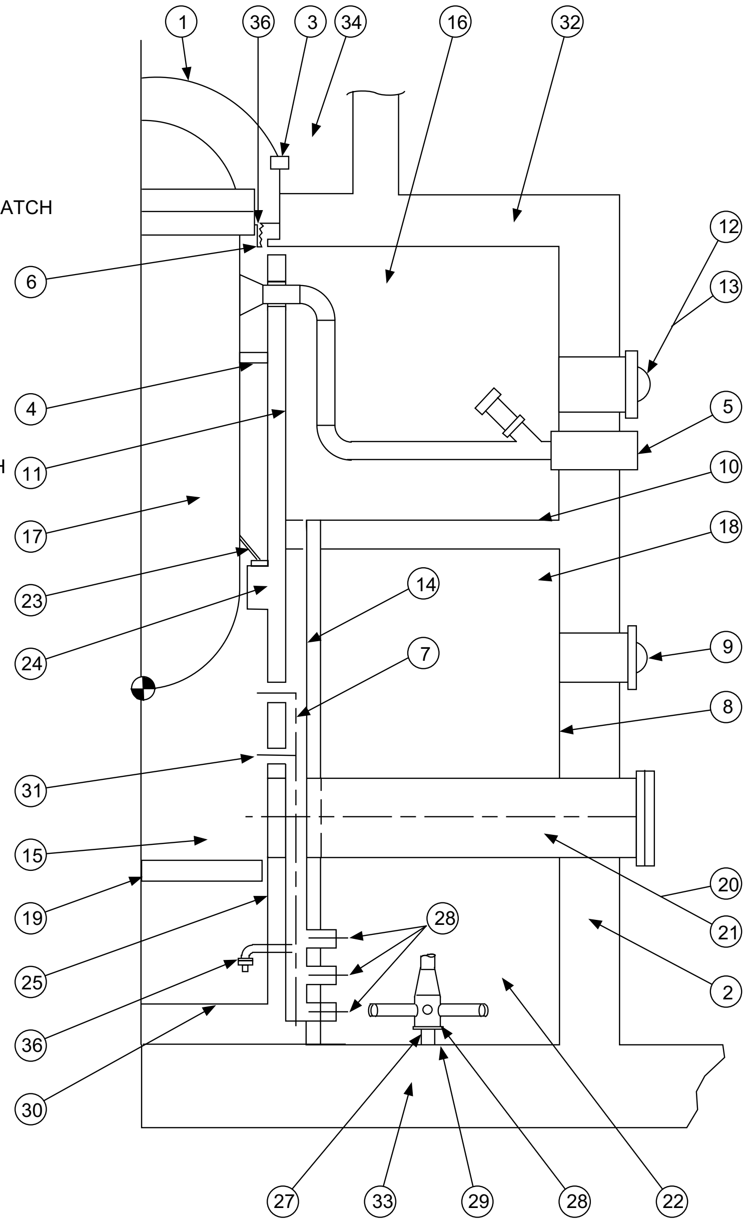


Figure 3.8-17 Reactor Building RCCV Internal Structures Nomenclature

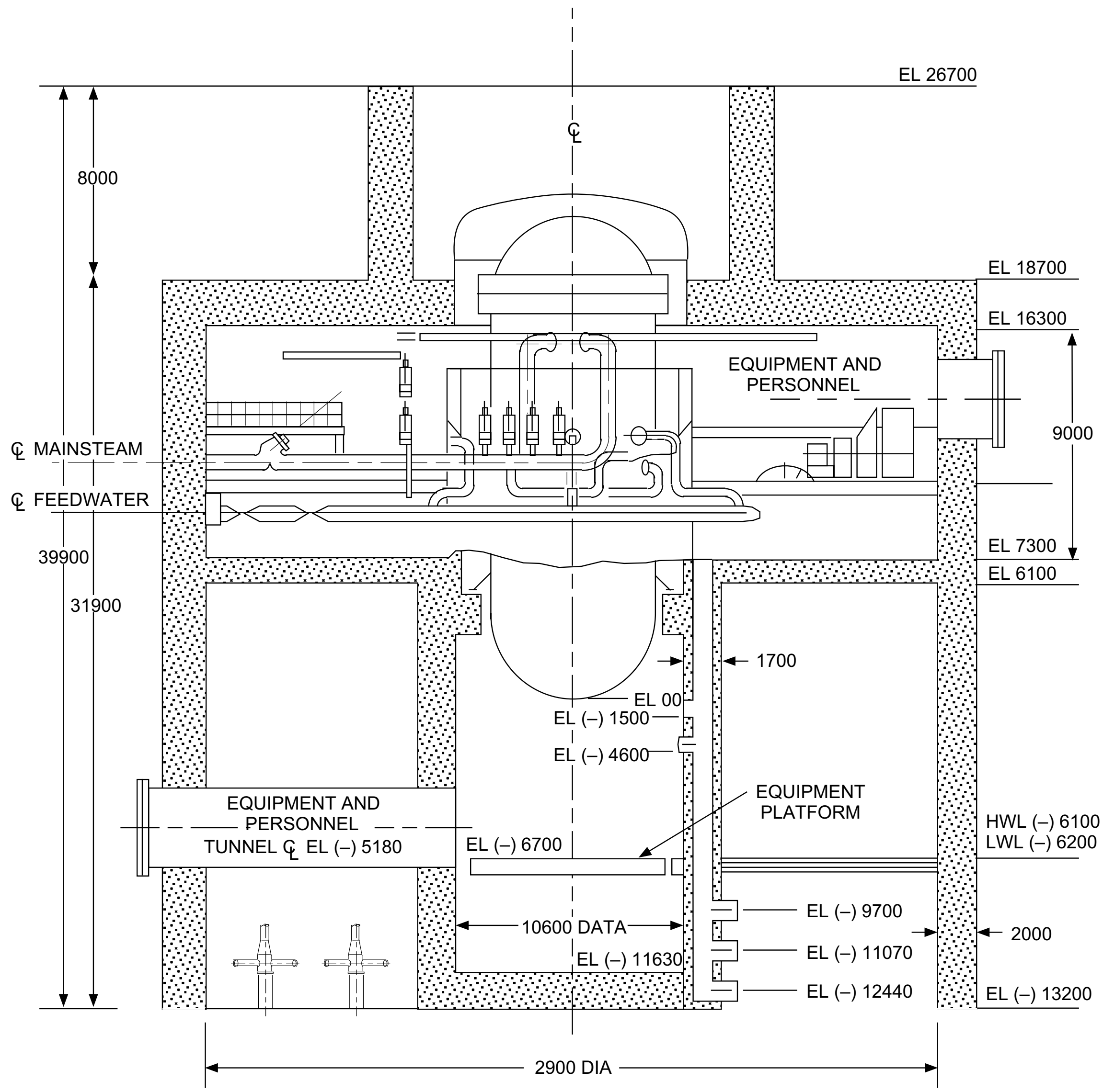


Figure 3.8-18 Reactor Building—RCCV Configuration

Figure 3.8-19 Not Used

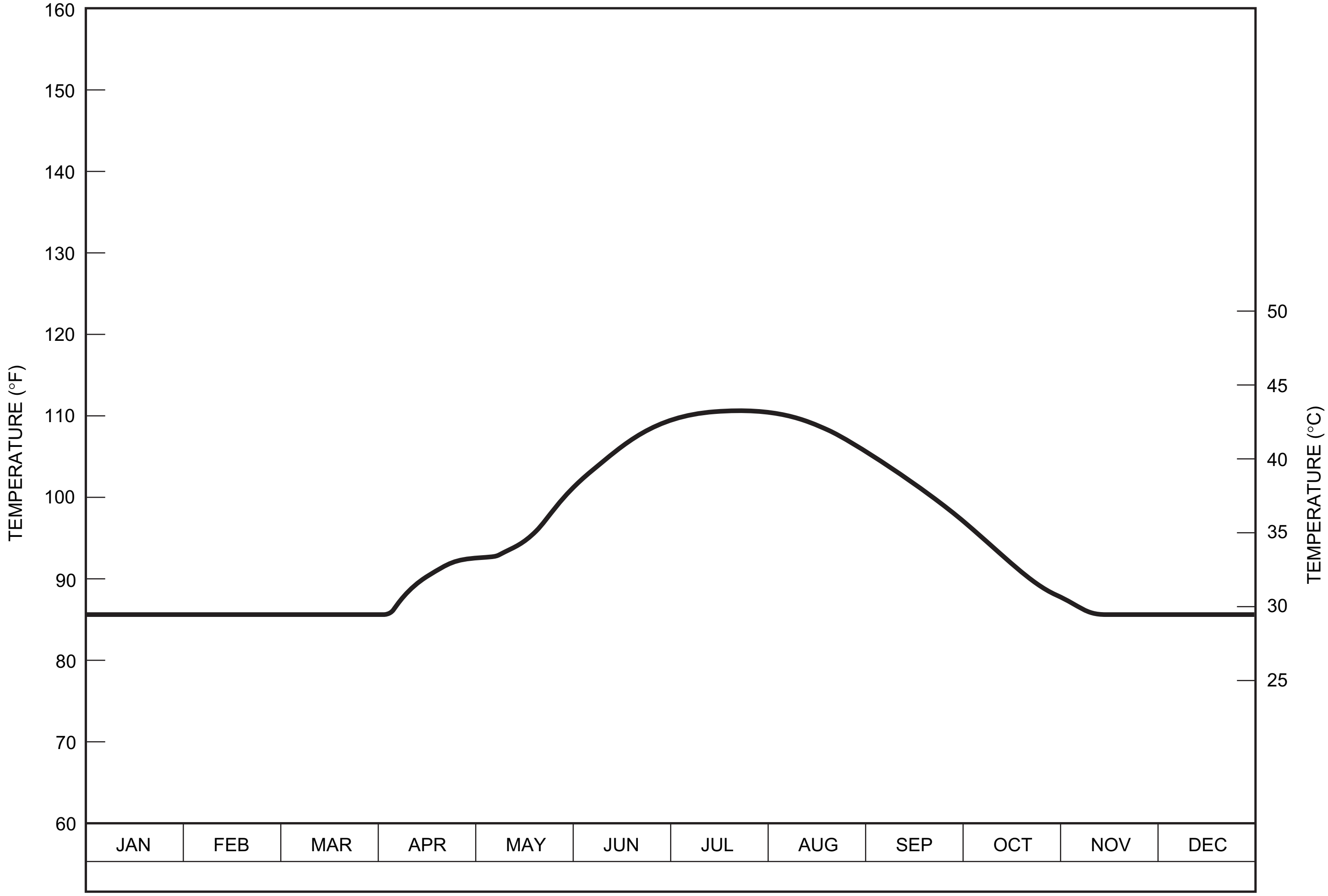


Figure 3.8-20 Annual Temperature Profile of Suppression Pool Water During Normal Operation of a Typical Plant in Southern States

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

The plant events affecting the mechanical systems, components and equipment are summarized in Table 3.9-1 in two groups: (1) plant operating events during which thermal-hydraulic transients occur, and (2) dynamic loading events due to accidents, earthquakes and certain operating conditions. The number of cycles associated with each event for the design of the reactor pressure vessel (RPV), for example, are listed in Table 3.9-1. The plant operating conditions are identified as normal, upset, emergency, faulted, or testing as defined in Subsection 3.9.3.1.1. Appropriate Service Levels (A, B, C, D or testing), as defined in ASME Code Section III, are designated for design limits. The design and analysis of safety-related piping and equipment using specific applicable thermal-hydraulic transients which are derived from the system behavior during the events listed in Table 3.9-1 are documented in the design specification and/or stress report of the respective equipment. Table 3.9-2 shows the loading combinations and the standard acceptance criteria.

3.9.1.2 Computer Programs Used in Analyses

The computer programs used in the analyses of the major safety-related components are described in Appendix 3D.

The computer programs used in the analyses of Seismic Category I components are maintained either by GE or by outside computer program developers. In either case, the quality of the programs and their computed results are controlled. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems.

The updates to Appendix 3D will be provided to indicate any additional programs used or the later version of the described programs, and the method of their verification.

[*Computer codes used for piping system dynamic analysis shall be benchmarked against the NRC Benchmark Problems for ABWR, defined in Reference 3.9-11. The results of piping dynamic analysis shall be compared with the results of the Benchmark Problems provided in Reference 3.9-11. The piping results to be compared and evaluated and the acceptance criteria or range of acceptable values are specified in Reference 3.9-11. Any deviations from these values as well as justification for such deviations shall be documented and submitted to the NRC staff for review and approval before initiating the final certified piping analysis.*]*

* See Subsection 3.9.1.7.

3.9.1.3 Experimental Stress Analysis

The following subsections list those NSSS components for which experimental stress analyses are performed in conjunction with analytical evaluations. The experimental stress analysis methods used are in compliance with the provisions of Appendix II of ASME Code Section III.

3.9.1.3.1 Piping Snubbers and Restraints

The following components have been tested to verify their design adequacy:

(1) Piping seismic snubbers

(2) Pipe whip restraints

Descriptions of the snubber and whip restraint tests are contained in Subsection 3.9.3.4 and Section 3.6, respectively.

3.9.1.3.2 Fine Motion Control Rod Drive (FMCRD)

Experimental data was used in developing the hydraulic analysis computer program called FMCRD01. The output of FMCRD01 is used in the dynamic analysis of both ASME Code and non-Code parts. Pressures used in the analysis of these parts are also determined during actual testing of prototype control rod drives.

3.9.1.4 Considerations for the Evaluation of Faulted Condition

All Seismic Category I equipment are evaluated for the faulted (Service Level D) loading conditions identified in Tables 3.9-1 and 3.9-2. In all cases, the calculated actual stresses are within the allowable Service Level D limits. The following subsections address the evaluation methods and stress limits used for the equipment and identify the major components evaluated for faulted conditions. Additional discussions of faulted analysis can be found in Subsections 3.9.2.5, 3.9.3, and 3.9.5.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits (e.g., as clearance limits) are satisfied.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Fine Motion Control Rod Drive

The fine motion control rod drive (FMCRD) major components that are part of the reactor coolant pressure boundary (RCPB) are analyzed and evaluated for the faulted conditions in accordance with ASME Code Section III, Appendix F.

3.9.1.4.1.2 Hydraulic Control Unit

The hydraulic control unit (HCU) is analyzed and tested for withstanding the faulted condition loads. Dynamic tests establish the “g” loads in horizontal and vertical directions as the HCU capability for the frequency range that is likely to be experienced in the plant. These tests also insure that the scram function of the HCU can be performed under these loads. Dynamic analyses of the HCU with the mounting beams are performed to assure that the maximum faulted condition loads remain below the HCU capability.

3.9.1.4.2 Reactor Pressure Vessel Assembly

The reactor pressure vessel (RPV) assembly includes: (1) the RPV boundary out to and including the nozzles and housings for FMCRD, internal pump and incore instrumentation; (2) support skirt; and (3) the shroud support, including legs, cylinder, and plate. The design and analyses of these three parts comply with Subsections NB, NF, and NG, respectively, of ASME Code Section III. For faulted conditions, the reactor vessel is evaluated using elastic analysis. For the support skirt and shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly.

3.9.1.4.3 Core Support Structures and Other Safety Reactor Internal Components

The core support structures and other safety class reactor internal components are evaluated for faulted conditions. The bases for determining the faulted loads for seismic events and other dynamic events are given in Section 3.7 and Subsection 3.9.5, respectively. The allowable Service Level D limits for evaluation of these structures are provided in Subsection 3.9.5.

3.9.1.4.4 RPV Stabilizer and FMCRD—and Incore Housing Restraints (Supports)

The calculated maximum stresses meet the allowable stress limits stated in Table 3.9-1 and 3.9-2 under faulted conditions for the RPV stabilizer and support for the FMCRD housing and incore housing for faulted conditions. These supports restrain the components during earthquake, pipe rupture or other Reactor Building vibration events.

3.9.1.4.5 Main Steam Isolation Valve, Safety/Relief Valve and Other ASME Class 1 Valves

Elastic analysis methods and standard design rules, as defined in ASME Code Section III, are utilized in the analysis of the pressure boundary, Seismic Category I, ASME Class 1 valves. The Code-allowable stresses are applied to assure integrity under applicable loading conditions, including faulted condition. Subsection 3.9.3.2.4 discusses the operability qualification of the major active valves, including MSIV and the main steam SRV for seismic and other dynamic conditions. The allowable stresses for various operating conditions, including faulted, for active ASME Class 1 valves are provided in Footnote 7 of Table 3.9-2.

3.9.1.4.6 ECCS and SLC Pumps, RRS and RHR Heat Exchangers, RCIC Turbine, and RRS Motor

The ECCS (RHR, RCIC and HPCF) pumps, SLC pumps, RHR heat exchangers, and RCIC turbine are analyzed for the faulted loading conditions. The ECCS and SLC pumps are active ASME Class 2 components. The allowable stresses for active pumps are provided in a footnote to Table 3.9-2.

The RCPB components of the Reactor Recirculation System (RRS) pump motor cover and Recirculation Motor Cooling (RMC) Subsystem heat exchanger are ASME Class 1 and Class 2, respectively, and are analyzed for the faulted loading conditions. All equipment stresses are within the elastic limits.

3.9.1.4.7 Fuel Storage and Refueling Equipment

Storage, refueling, and servicing equipment which is important to safety is classified as essential components per the requirements of 10CFR50 Appendix A. This equipment and other equipment which, in case of a failure, would degrade an essential component is defined in Subsection 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite-element analysis to generate loadings. This analysis utilizes appropriate floor response spectra and combines loads at frequencies up to 33 Hz for seismic loads and up to 60 Hz for other dynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI or Industrial Standards, AISC allowables.

3.9.1.4.8 Fuel Assembly (Including Channel)

GE BWR fuel assembly (including channel) design bases, and analytical and evaluation methods, including those applicable to the faulted conditions, are the same as those contained in References 3.9-1 and 3.9-2.

3.9.1.4.9 ASME Class 2 and 3 Vessels

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 vessels. The equivalent allowable stresses using elastic techniques are obtained from NC/ND-3300 and NC-3200 of ASME Code Section III. These allowables are above elastic limits.

3.9.1.4.10 ASME Class 2 and 3 Pumps

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 pumps. The equivalent allowable stresses for nonactive pumps using elastic techniques are obtained from NC/ND-3400 of ASME Code Section III. These allowables are above elastic limits. The allowables for active pumps are provided in footnote 12 to Table 3.9-2.

3.9.1.4.11 ASME Class 2 and 3 Valves

Elastic analysis methods and standard design rules are used for evaluating faulted loading conditions for Class 2 and 3 valves. The equivalent allowable stresses for nonactive valves using elastic techniques are obtained from NC/ND-3500 of ASME Code Section III. These allowables are above elastic limits. The allowables for active valves are provided in footnote 12 of Table 3.9-2.

3.9.1.4.12 ASME Class 1, 2 and 3 Piping

Elastic analysis methods are used for evaluating faulted loading conditions for Class 1, 2, and 3 piping. The equivalent allowable stresses using elastic techniques are obtained from NB/NC/ND-3600 of the ASME Code Section III. The allowables for functional capability are provided in footnote 9 of Table 3.9-2.

3.9.1.5 Inelastic Analysis Methods

Inelastic analysis is only applied to ABWR components to demonstrate the acceptability of three types of postulated events. Each event is an extremely low-probability occurrence and the equipment affected by these events would not be reused. These three events are:

(1) Postulated gross piping failure

(2) Postulated blowout of a reactor internal recirculation (RIP) motor casing due to a weld failure

(3) Postulated blowout of a control rod drive (CRD) housing due to a weld failure

The loading combinations and design criteria for pipe whip restraints utilized to mitigate the effects of postulated piping failures are provided in Subsection 3.6.2.3.3.

In the case of the RIP motor casing failure event, there are specific restraints applied to mitigate the effects of the failure. The mitigation arrangement consists of lugs on the RPV bottom head to which are attached two long rods for each RIP. The lower end of each rod engages two lugs on the RIP motor/cover. The use of inelastic analysis methods is limited to the middle slender body of the rod itself. The attachment lugs, bolts and clevises are shown to be adequate by elastic analysis. The selection of stainless steel for the rod is based on its high ductility assumed for energy absorption during inelastic deformation.

The mitigation for a failure in the CRD housing welded attachment (see Subsection 4.6.1.2.2.9) is by somewhat different means than are those of the RIP, in that the components with regular functions also function to mitigate the weld failure effect. The components are specifically:

(1) Core plate

(2) Control rod guide tube (CRGT)

(3) CRD housing

(4) CRD outer tube and middle flange, and

(5) Bayonet connector of CRD (internal CRD blowout support)

Only the cylindrical body of the CRGT deforms inelastically with other components deforming elastically in energy absorption. All other components are evaluated elastically, except the CRGT base, which is evaluated by the limit load method but is assumed conservatively in load prediction to absorb less energy by deforming elastically.

In elastic analyses for the latter two events, together with the criteria used for evaluation, are consistent with the procedures described in Subsection 3.6.2.3.3 for a pipe whip restraint. Figure 3.9-6 shows the stress-strain curve used for the inelastic restraints. The component evaluations are consistent with the elastic or limit load method of Appendix F of the ASME Code Section III.

3.9.1.6 Welding Methods and Acceptance Criteria for ASME Code Welding and Welding of Non-ASME Pressure Retaining Piping.

3.9.1.6.1 ASME Code Welding

Welding activities for pressure boundary and core support structure shall be performed in accordance with the requirements of Section III or Section VIII as applicable, of the ASME Code. The required nondestructive examination and acceptance criteria are stated in Table 3.9-10. Component supports shall be fabricated and examined in accordance with the requirements of Subsection NF of Section III of the ASME Code and NCIG-01 (Reference 3.9-12).

3.9.1.6.2 Welding of Non-ASME Pressure Retaining Piping

Welding activities involving non-ASME pressure retaining piping shall be accomplished in accordance with written procedures and shall meet the requirements of the ANSI B31.1 Code. The weld acceptance criteria shall be as defined for the applicable nondestructive examination method described in ANSI B31.1 Code.

3.9.1.7 Piping Design Acceptance Criteria

[*Table 7 of DCD/Introduction identifies the piping design acceptance criteria, which, if changed, requires NRC Staff review and approval prior to implementation. The applicable portions of the Tier 2 sections, tables, and figures identified on Table 7 of DCD/Introduction for this restriction, are italicized on the sections, tables, and figures themselves.*]*

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The overall test program is divided into two phases: the preoperational test phase and the initial startup test phase. Piping vibration, thermal expansion and dynamic effects testing will be performed during both of these phases, as described in Chapter 14. Subsections 14.2.12.1.51, 14.2.12.2.10 and 14.2.12.2.11 relate the specific role of this testing to the overall test program.

* See section 3.5 of DCD/Introduction.

Discussed below are the general requirements for this testing. It should be noted that, because one goal of the dynamic effects testing is to verify the adequacy of the piping support system, such components are addressed in the subsections that follow. However, the more specific requirements of the design and testing of the piping support system are described in Subsection 3.9.3.4.1.

3.9.2.1.1 Vibration and Dynamic Effects Testing

The purpose of these tests is to confirm that the piping, components, restraints and supports of specified high- and moderate-energy systems have been designed to withstand the dynamic effects of steady-state flow-induced vibration (FIV) and anticipated operational transient conditions. The general requirements for vibration and dynamic effects testing of piping systems are specified in Regulatory Guide 1.68, “Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors”. More specific vibration testing requirements are defined in ANSI/ASME OM3, “Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems”. Preparation of detailed test specifications will be in full accordance with this standard and will address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in more detail below.

3.9.2.1.1.1 Measurement Techniques

There are essentially three methods available for determining the acceptability of steady state and transient vibration for the affected systems. These three measurement techniques are (1) visual observation, (2) local measurements, or (3) remotely monitored/recorded measurements. The technique used in each case will depend on such factors as (1) the safety significance of the particular system, (2) the expected mode and/or magnitude of the vibration, (3) the assessability of the system during designated testing conditions, or (4) the need for a time history recording of the vibratory behavior. Typically, the systems where vibration has the greatest safety implication will be subject to more rigorous testing and precise instrumentation requirements and, therefore, will require remote monitoring techniques. Local measurement techniques, such as the use of a hand-held vibrometer, are more appropriate in cases where it is expected that the vibration will be less complex and of lessor magnitude. Many systems that are assessable during the preoperational test phase and that do not show significant intersystem interactions will fall into this category. Visual observations are utilized where vibration is expected to be minimal and the need for a time history record of transient behavior is not anticipated. However, unexpected visual observations or local indications may require that a more sophisticated technique be used. Also, the issue of assessability should be considered. Application of these measurement techniques is detailed in the appropriate testing specification consistent with the guidelines contained in ANSI/ASME OM3.

3.9.2.1.1.2 Monitoring Requirements

As described in Subsections 14.2.12.1.51, 14.2.12.2.10 and 14.2.12.2.11, all safety-related piping systems will be subjected to steady-state and transient vibration measurements. The scope of such testing shall include safety-related instrumentation piping and attached small-bore piping (branch piping). Special attention should be given to piping attached to pumps, compressors, and other rotating or reciprocating equipment. Monitoring location selection considerations should include the proximity of isolation valves, pressure or flow control valves, flow orifices, distribution headers, pumps and other elements where shock or high turbulence may be of concern. Location and orientation of instrumentation and/or measurements will be detailed in the appropriate test specification. Monitored data should include actual deflections and frequencies as well as related system operating conditions. The time duration of data recording should be sufficient to indicate whether the vibration is continuous or transient. Steady-state monitoring should be performed at critical conditions such as minimum or maximum flow, or abnormal combinations or configurations of system pumps or valves. Transient monitoring should include anticipated system and total plant operational transients where critical piping or components are expected to show significant response. Steady-state conditions and transient events to be monitored will be detailed in the appropriate testing specifications consistent with OM3 guidelines.

3.9.2.1.1.3 Test Evaluation and Acceptance Criteria

The piping response to test conditions shall be considered acceptable if the review of the test results indicates that the piping responds in a manner consistent with predictions of the stress report and/or that piping stresses are within ASME Code Section III (NB-3600) limits. Acceptable limits are determined after the completion of piping systems stress analysis and are provided in the piping test specifications.

To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. For steady-state and transient vibration, the pertinent acceptance criteria are usually expressed in terms of maximum allowable displacement/deflection. Visual observation should only be used to confirm the absence of significant levels of vibration and not to determine acceptability of any potentially excessive vibration. Therefore, in some cases, other measurement techniques will be required with appropriate quantitative acceptance criteria.

There are typically two levels of acceptance criteria for allowable vibration displacements/deflections. Level I criteria are bounding type criteria associated with safety limits, while Level 2 criteria are stricter criteria associated with system or component expectations. For steady-state vibration, the Level 1 criteria are based on the endurance limit (68.6 MPaG) to assure no failure from fatigue over the life of the plant. The corresponding Level 2 criteria are based on one half the endurance limit (34.3 MPaG). For transient vibration, the Level 1 criteria are based on either the ASME Code Section III upset primary stress limit or the applicable snubber load capacity. Level 2 criteria are based on a given tolerance about the expected deflection value.

3.9.2.1.1.4 Reconciliation and Corrective Actions

During the course of the tests, the remote measurements will be regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements should be monitored at more frequent intervals. The test will be held or terminated as soon as criteria are violated. As soon as possible after the test hold or termination, appropriate investigative and corrective actions will be taken. If practicable, a walkdown of the piping and suspension system should be made in an attempt to identify potential obstructions or improperly operating suspension components. Hangers and snubbers should be positioned such that they can accommodate the expected deflections without bottoming out or extending fully. All signs of damage to piping supports or anchors shall be investigated.

Instrumentation indicating criteria failure shall be checked for proper operation and calibration, including comparison with other instrumentation located in the proximity of the excessive vibration. The assumptions used in the calculations that generated the applicable limits should be verified against actual conditions and discrepancies noted should be accounted for in the criteria limits. This may require a reanalysis at actual system conditions.

Should the investigation of instrumentation and calculations fail to reconcile the criteria violations, physical corrective actions may be required, including (1) identification and reduction or elimination of offending forcing functions, (2) detuning of resonant piping spans by appropriate modifications, (3) addition of bracing, or (4) changes in operating procedures to avoid troublesome conditions. Any such modifications will require retest to verify vibrations have been sufficiently reduced.

3.9.2.1.2 Thermal Expansion Testing

A thermal expansion preoperational and startup testing program performed through the use of visual observation and remote sensors has been established to verify that normal unrestrained thermal movement occurs in specified safety-related high- and moderate-energy piping systems. The purpose of this program is to ensure the following:

(1) The piping system during system heatup and cooldown is free to expand and move without unplanned obstruction or restraint in the x, y, and z directions.

(2) The piping system does shakedown after a few thermal expansion cycles.

(3) The piping system is working in a manner consistent with the assumption of the stress analysis.

(4) There is adequate agreement between calculated values and measured values of displacements.

(5) There is consistency and repeatability in thermal displacements during heatup and cooldown of the systems.

The general requirements for thermal expansion testing of piping systems are specified in Regulatory Guide 1.68, “Preoperational and Initial Startup Testing Programs for Water-Cooled Power Reactors.” More specific requirements are defined in ANSI/ASME OM7, “Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems.” Detailed test specifications will be prepared in full accordance with this standard and will address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points, and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in more detail below.

3.9.2.1.2.1 Measurement Techniques

Verification of acceptable thermal expansion of specified piping systems can be accomplished by several methods. One method is to physically walkdown the piping system and verify by visual observation that free thermal movement is unrestrained. This might include verification that piping supports such as snubbers and spring hangers are not fully extended or bottomed out and that the piping (including branch lines and instrument lines) and its insulation is not in hard contact with other piping or support structures. Another method would involve local measurements, using a hand-held scale or ruler, against a fixed reference or by recording the position of a snubber or spring can. A more precise method would be using permanent or temporary instrumentation that directly measures displacement, such as a lanyard potentiometer, that can be monitored via a remote indicator or recording device. The technique to be used will depend on such factors as the amount of movement predicted and the assessability of the piping.

Measurement of piping temperature is also of importance when evaluating thermal expansion. This may be accomplished either indirectly via the temperature of the process fluid or by direct measurement of the piping wall temperature, and such measurements may be obtained either locally or remotely. The choice of technique used shall depend on such considerations as the accuracy required and the assessability of the piping.

3.9.2.1.2.2 Monitoring Requirements

As described in Subsections 14.2.12.1.51 and 14.2.12.2.10, all safety-related piping shall be included in the thermal expansion testing program. Thermal expansion of specified piping systems should be measured at both the cold and hot extremes of their expected operating conditions. Physical walkdowns and recording of hanger and snubber positions should also be conducted where possible considering assessability and local environmental and radiological conditions in the hot and cold states. Displacements and appropriate piping/process temperatures shall be recorded for those systems and conditions specified. Sufficient time shall have passed before taking such measurements to ensure the piping system is at a steady-state condition. In selecting locations for monitoring piping response, consideration shall be given to the maximum responses predicted by the piping analysis. Specific consideration should also be given to the first run of pipe attached to component nozzles and pipe adjacent to structures requiring a controlled gap.

3.9.2.1.2.3 Test Evaluation and Acceptance Criteria

To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. Limits of thermal expansion displacements are established prior to start of piping testing to which the actual measured displacements are compared to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limit values by more than the specified tolerance, the piping system is responding in a manner consistent with the predictions and is therefore acceptable. The piping response to test conditions shall be considered acceptable if the review of the test results indicates that the piping responds in a manner consistent with the predictions of the stress report and/or that piping stresses are within ASME Code Section III (NB-3600) limits. Acceptable thermal expansion limits are determined after the completion of piping systems stress analysis and are provided in the piping test specifications. Level 1 criteria are bounding criteria based on ASME Code Section III stress limits. Level 2 criteria are stricter criteria based on the predicted movements using the calculated deflections plus a selected tolerance.

3.9.2.1.2.4 Reconciliation and Corrective Actions

During the course of the tests, the remote measurements will be regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements should be monitored at more frequent intervals. The test will be held or terminated as soon as criteria are violated. As soon as possible after the test hold termination, appropriate investigative and corrective actions will be taken. If practicable, a walkdown of the affected piping and suspension system should be made in an attempt to identify potential obstruction to free piping movement. Hangers and snubbers should be positioned within their expected cold and hot settings. All signs of damage to piping or supports shall be investigated.

Instrumentation indicating criteria failure shall be checked for proper operation and calibration, including comparison with other instrumentation located in the proximity of the out-of-bounds movement. Assumptions, such as piping temperature, used in the calculations that generated the applicable limits should be compared with actual test conditions. Discrepancies noted should be accounted for in the criteria limits, including possible reanalysis.

Should the investigation of instrumentation and calculations fail to reconcile the criteria violations or should the visual inspection reveal an unintended restraint, physical corrective actions may be required. This might include (1) complete or partial removal of an interfering structure, (2) replacing, readjusting or repositioning piping system supports, (3) modifying the pipe routing, or (4) modifying system operating procedures to avoid the temperature conditions that resulted in the unacceptable thermal expansion.

3.9.2.1.3 Thermal Stratification in Feedwater Piping

This special test is part of the startup program to monitor the conditions and effects of temperature stratification that may exist during certain operating conditions in:

(1) The feedwater piping header inside and outside the containment.

(2) The short horizontal runs of the riser piping inside the containment where feedwater enters the vessel nozzles.

Stratification in the feedwater piping can occur during plant startup when hot CUW is added to the cold feedwater line. The hot CUW flows on top of the colder water in the feedwater line and does not mix with the cold water until mixing of the two streams occurs at the outer swing check isolation valve. Stratification for this condition can thus affect only the feedwater piping outside containment.

A second condition of plant operation which can cause stratification in the feedwater piping is when the plant is in hot standby condition following a scram. After a scram, the temperature of the entire feedwater line is hot when cold water is introduced to make up for decay heat boiloff in the RPV. The colder water flows along the bottom of the large diameter horizontal feedwater pipe at low flow rate, creating stratification. The temperature difference between the top and bottom of the pipe will decrease along the pipe in the direction of flow, but stratification could still exist in the feedwater piping inside the containment, since the swing check valves are not effective in mixing the cold water flowing along the bottom of the pipe.

The test program will consist of measurement of:

(1) Temperature around the circumference of the feedwater pipe at various locations inside and outside the containment.

(2) Strains at points of highest stress inside the containment.

(3) Measurements of pipe displacements and movements inside and outside the containment due to pipe bowing because of stratification.

This test will be performed in accordance with the general requirements of Regulatory Guide 1.68 and the more specific requirements in ANSI/ASME OM7. Detailed test procedures will be prepared in accordance with these documents. The development and specifications of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in Subsection 3.9.2.1.2.

The feedwater thermal stratification test is not required if the applicant can show that a test performed at a previous plant meets the requirements of this paragraph and is applicable to this plant.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment (Including Other RBV Induced Loads)

This subsection describes the criteria for dynamic qualification of safety-related mechanical equipment and associated supports, and also describes the qualification testing and/or analyses applicable to the major components on a component-by-component basis. Seismic and other events that may induce Reactor Building Vibration (RBV) are considered (Appendix 3B). In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit (e.g., ECCS pumps). These modules are generally discussed in this subsection and Subsection 3.9.3.2, rather than providing discussion of the separate electrical parts in Section 3.10. Electrical supporting equipment such as control consoles, cabinets, and panels are discussed in Section 3.10.

3.9.2.2.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety function during and after the application of a dynamic load is demonstrated by tests and/or analysis. The analysis is performed in accordance with Section 3.7. Selection of testing, analysis, or a combination of the two, is determined by the type, size, shape, and complexity of the equipment being considered. When practical, the equipment operability is demonstrated by testing. Otherwise, operability is demonstrated by mathematical analysis.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or static bend test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or static bend testing is also used to show that there are no natural frequencies below 33 Hz for seismic loads and 60 Hz for other RBV loads*. If a natural frequency lower than 33 Hz in the case of seismic loads and 60 Hz in the case of other RBV-induced loads is discovered, dynamic tests and/or mathematical analyses may be used to verify operability and structural integrity at the required dynamic input conditions.

[*When the equipment is qualified by dynamic test, the response spectrum or time history of the attachment point is used in determining input motion.*]†

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Dynamic load conditions are simulated by testing using random vibration input or single frequency input (within equipment capability)

* The 60 Hz frequency cutoff for dynamic analysis of suppression pool dynamic loads is the minimum requirement based on a generic Reference 3.9-8, using the missing strain energy method, performed for representative BWR equipment under high-frequency input loadings.

† See Section 3.10.

over the frequency range of interest. [*Whichever method is used, the input amplitude during testing envelopes the actual input amplitude expected during the dynamic loading condition.*]†

[*The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.*]†

Equipment having an extended structure, such as a valve operator, may be analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a combination of testing and analysis in accordance with IEEE-344 is used to determine operational capability at maximum equivalent dynamic load conditions.

3.9.2.2.1.1 Random Vibration Input

When random vibration input is used, the actual input motion envelopes the appropriate floor input motion at the individual modes. However, single frequency input such as sine beats can be used, provided one of the following conditions are met:

(1) The characteristics of the required input motion are dominated by one frequency.

(2) The anticipated response of the equipment is adequately represented by one mode.

(3) The input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra will envelop the corresponding response spectra of the individual modes.

3.9.2.2.1.2 Application of Input Modes

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

3.9.2.2.1.3 Fixture Design

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

3.9.2.2.1.4 Prototype Testing

Equipment testing is conducted on prototypes of the equipment to be installed in the plant.

3.9.2.2.2 Qualification of Safety-Related Mechanical Equipment

The following subsections discuss the testing or analytical qualification of the safety-related major mechanical equipment and other ASME Code Section III equipment, including equipment supports.

3.9.2.2.2.1 CRD and CRD Housing

The qualification of the CRD housing (with enclosed CRD) is done analytically, and the stress results of their analysis establish the structural integrity of these components. Preliminary dynamic tests are conducted to verify the operability of the control rod drive during a dynamic event. A simulated test, imposing dynamic deflection in the fuel channels up to values greater than the expected seismic response, is performed with the CRD demonstrated functioning satisfactorily.

The test was conducted in two phases due to facility limitations. The seismic test facility cannot be pressurized while shaking; therefore, the charging pressure of the hydraulic control unit is reduced to simulate the backpressure that is applied in the reactor. The appropriate adjustment was determined by first running scram tests with the full reactor pressure and with peak transient pressure. Then with the test vessel at atmospheric pressure, the scram tests were repeated with reduced charging pressures until the scram performance matched that of the pressurized tests. This was repeated for the peak pressure also. The seismic tests were then performed with the appropriate pressure adjustments for the conditions being tested. The tests were run for various vibration levels with fuel channel deflections being the independent variable. The test facility was driven to vibration levels that produced various channel deflections up to 41 mm and the scram curves recorded. The 41 mm channel deflection is several times the channel deflection calculated for the actual seismic condition. The correlation of the test with analysis is via the channel deflection not the housing structural analysis, since scramability is controlled by channel deflection, not housing deflection.

3.9.2.2.2.2 Core Support (Fuel Support and CR Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and other RBV events is performed to show that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.2.2.3 Hydraulic Control Unit (HCU)

The HCU is analyzed for the seismic and other RBV loads faulted condition, and the maximum stress on the HCU frame is calculated to be below the maximum allowable for the faulted condition. As discussed in Subsection 3.9.1.4.1.2, the faulted condition loads are calculated to be below the HCU maximum capability.

3.9.2.2.2.4 Fuel Assembly (Including Channel)

GE BWR fuel channel design bases, analytical methods, and seismic considerations are similar to those contained in References 3.9-1 and 3.9-2. The resulting combined acceleration profiles, including fuel lift for all normal/upset and faulted events are to be shown less than the respective design basis acceleration profiles.

3.9.2.2.2.5 Reactor Internal Pump and Motor Assembly

The reactor internal pump (RIP) and motor assembly, including its appurtenances and support, is classified as Seismic Category I, but not active, and is designed to withstand the seismic forces, including other RBV loads. The qualification of the assembly is done analytically and with a dynamic test.

3.9.2.2.2.6 ECCS Pump and Motor Assembly

A prototype ECCS (RHR and HPCF) pump motor assembly is qualified for seismic and other RBV loads via a combination of dynamic analysis and dynamic testing. The complete motor assembly is qualified via dynamic testing in accordance with IEEE-344. The qualification test program includes demonstration of startup capability as well as operability during dynamic loading conditions (see Subsection 3.9.3.2.1.4 for details).

The pump and motor assemblies, as units operating under seismic and other RBV load conditions, are qualified by dynamic analysis, and results of the analysis indicate that the pump and motor are capable of sustaining the above loadings without exceeding the allowable stresses (see Subsections 3.9.3.2.1.1 and 3.9.3.2.1.2 for details).

3.9.2.2.2.7 RCIC Pump and Turbine Assembly

The RCIC pump construction is a horizontal, multistage type and is supported on a pedestal. The RCIC pump assembly is qualified analytically by static analysis for seismic and other RBV loadings, as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are less than the allowable (Subsection 3.9.3.2.2).

The RCIC turbine is qualified for seismic and other RBV loads via a combination of static analysis and dynamic testing (Subsection 3.9.3.2.1.5). The turbine assembly consists of rigid masses (wherein static analysis is utilized) interconnected with control levers and electronic control systems, necessitating final qualification via dynamic testing. Static loading analyses are employed to verify the structural integrity of the turbine assembly and the adequacy of bolting under operating, seismic, and other RBV loading conditions. The complete turbine assembly is qualified via dynamic testing in accordance with IEEE-344. The qualification test program includes a demonstration of startup capability, as well as operability during dynamic loading conditions. Operability under normal load conditions is assured by comparison to the operability of similar turbines in operating plants.

3.9.2.2.2.8 Standby Liquid Control Pump and Motor Assembly

The SLCS positive displacement pump and motor assembly, which is mounted on a common base plate, is qualified analytically by static analysis of seismic and other RBV loadings, as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are less than the allowables (Subsection 3.9.3.2.2).

3.9.2.2.2.9 RMC and RHR Heat Exchangers

A three-dimensional finite-element model is developed for each of the Recirculation Motor Cooling (RMC) and Residual Heat Removal (RHR) System heat exchangers and supports. The model is used to dynamically analyze the heat exchanger and its supports using the response spectrum analysis method, and to verify that the heat exchanger and supports can withstand seismic and other RBV loads. The same model is used to statically analyze and evaluate the nozzles due to the effect of the external piping loads and dead weight in order to ensure that nozzle load criteria and limits are met. Critical location stresses are evaluated and compared with the allowable stress criteria. The results of the analysis demonstrate that the stresses at all investigated locations are less than their corresponding allowable values.

3.9.2.2.2.10 Standby Liquid Control Storage Tank

The Standby Liquid Control Storage Tank (SLCST) is a cylindrical tank, with approximate dimensions of 3.05m diameter and 4.9m height, bolted to the concrete floor. The SLCST is qualified for seismic and other RBV loads by analysis for:

(1) Stresses in the tank bearing tank plate

(2) Bolt stresses

(3) Sloshing loads imposed at the sloshing natural frequency

(4) Minimum wall thickness

(5) Buckling

The results of this analysis confirm that the calculated stresses at all investigated locations are less than their corresponding allowable values.

3.9.2.2.2.11 Main Steam Isolation Valves

The main steam isolation valves (MSIVs) are qualified for seismic and other RBV loads. The fundamental requirement of the MSIV following an SSE or other faulted RBV loadings is to close and remain closed after the event. This capability is demonstrated by the test and analysis as outlined in Subsection 3.9.3.2.4.1.

3.9.2.2.2.12 Standby Liquid Control Valve (Injection Valve)

The motor-operated standby liquid control valve is qualified by type test to IEEE-344 for seismic and other RBV loads. The qualification test as discussed in Subsection 3.9.3.2.4.3 demonstrates the ability to remain operable after the application of horizontal and vertical dynamic loading in excess of the required response spectra. The valve and motor assemblies are qualified by dynamic analysis, and the results of the analysis indicate that the valve is capable of sustaining the dynamic loads without overstressing the pressure retaining components.

3.9.2.2.2.13 Main Steam Safety/Relief Valves

Due to the complexity of the structure and the performance requirements of the valve, the total assembly of the SRV (including electrical and pressure devices) is tested at dynamic accelerations equal to or greater than the combined SSE and other RBV loadings determined for the plant. Tests and analyses demonstrate the satisfactory operation of the valves during and after the test (Subsection 3.9.3.2.4.2).

3.9.2.2.2.14 Fuel Pool Cooling and Cleanup System Pump and Motor Assembly

A static analysis is performed on the pump and motor assembly of the Fuel Pool Cooling and Cleanup (FPC) System. This analysis shows that the pump and motor will continue to operate if subjected to a combination of SSE, other RBV, and normal operating loads. Analysis also ensures that pump running clearances, which include deflection of the pump shaft and pump pedestal, are met during seismic and other RBV loadings.

3.9.2.2.2.15 Other ASME III Equipment

Other equipment, including associated supports, is qualified for seismic and other RBV loads to ensure its functional integrity during and after the dynamic event. The equipment is tested, if necessary, to ensure its ability to perform its specified function before, during, and following a test.

Dynamic load qualification is done by a combination of test and/or analysis (Subsection 3.9.2.2.1). Natural frequency, when determined by an exploratory test, is in the form of a single-axis continuous-sweep frequency search using a sinusoidal steady-state input at the lowest possible amplitude which is capable of determining resonance. The search is conducted on each principal axis with a minimum of two continuous sweeps over the frequency range of interest at a rate no greater than one octave per minute. If no resonances are located, then the equipment is considered as rigid and single frequency tests at every one third octave frequency interval are acceptable. Also, if all natural frequencies of the equipment are greater than 33 Hz for seismic loads and 60 Hz for other RBV loads, the equipment may be considered rigid and analyzed statically as such. In this static analysis, the dynamic forces on each component are obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate floor acceleration. The dynamic stresses are then added to the operating stresses and a determination made of the adequacy of the strength of the equipment. The search for the natural frequency is done analytically if the equipment shape can be defined mathematically and/or by prototype testing.

If the equipment is a rigid body while its support is flexible, the overall system can be modeled as a single-degree-of-freedom system consisting of a mass and a spring. The natural frequency of the system is computed; then the acceleration is determined from the floor response spectrum curve using the appropriate damping value. A static analysis is then performed using this acceleration value. In lieu of calculating the natural frequency, the peak acceleration from the spectrum curve is used. The critical damping values for welded steel structures from Table 3.7-1 are employed.

In case the equipment cannot be considered as a rigid body, it can be modeled as a multi-degree-of-freedom system. It is divided into a sufficient number of mass points to ensure adequate representation. The mathematical model can be analyzed using the modal analysis technique or direct integration of the equations of motion. Specified structural damping is used in the analysis unless justification for other values can be provided. A stress analysis is performed using the appropriate inertial forces or equivalent static loads obtained from the dynamic analysis of each mode.

For a multiple-degree-of-freedom modal analysis, the modal response accelerations can be taken directly from the applicable floor response spectrum. The maximum spectral values within ±10% band of the calculated frequencies of the equipment are used for computation of modal dynamic response inertial loading. The total dynamic stress is obtained by combining the modal stresses. The dynamic stresses are added to the operating stresses using the loading combinations stipulated in the specific equipment specification and then compared with the allowable stress levels.

If the equipment being analyzed has no definite orientation, the worst possible orientation is considered. Furthermore, equipment is considered to be in its operational configuration (i.e., filled with the appropriate fluid and/or solid). The investigation ensures that the point of maximum stress is considered. Lastly, a check is made to ensure that partially filled or empty equipment do not result in higher response than the operating condition. The analysis includes an evaluation of the effects of the calculated stresses on mechanical strength, alignment, electrical performance (microphonics, contact bounce, etc.), and noninterruption of function. Maximum displacements are computed and interference effects determined and justified.

Individual devices are tested separately, when necessary, in their operating condition. Then the component to which the device is assembled is tested with a similar but inoperative device installed upon it.

The equipment, component, or device to be tested is mounted on the vibration generator in a manner that simulates the final service mounting. If the equipment is too large, other means of simulating the service mounting are used. Support structures such as air conditioning units, consoles, racks, etc., could be vibration tested without the equipment and/or devices being in operation, provided they are performance tested after the vibration test. However, the components are in their operational configuration during the vibration test. The goal is to determine that, at the specified vibratory accelerations, the support structure does not amplify the forces beyond that level to which the devices have been qualified.

Equipment could alternatively be qualified by presenting historical performance data which demonstrates that the equipment satisfactorily sustains dynamic loads which are equal to or greater than those specified for the equipment, and that the equipment performs a function equal to or better than that specified for it.

Equipment for which continued function is not required after a seismic and other RBV loads event, but whose postulated failure could produce an unacceptable influence on the performance of systems having a primary safety function, is evaluated. Such equipment is qualified to the extent required to ensure that an SSE, including other RBV loads, in combination with normal operating conditions, would not cause unacceptable failure. Qualification requirements are satisfied by ensuring that the equipment in its functional configuration, complete with attached appurtenances, remains structurally intact and affixed to the interface. The structural integrity of internal components is not required; however, the enclosure of such components is required to be adequate to ensure their confinement. Where applicable, fluid or pressure boundary integrity is demonstrated. With a few exceptions, simplified analytical techniques are adequate.

Historically, it has been shown that the main cause for equipment damage during a dynamic excitation has been the failure of its anchorage. Stationary equipment is designed with anchor bolts or other suitable fastening strong enough to prevent overturning or sliding. The effect of friction on the ability to resist sliding is neglected. The effect of upward dynamic loads on overturning forces and moments is considered. Unless specifically specified otherwise, anchorage devices are designed in accordance with the requirements of ASME Code Section III, Division 1, Subsection NF, or the AISC Manual of Steel Construction and ACI 318.

Dynamic design data are provided in the form of acceleration response spectra for each floor area of the equipment. Dynamic data for the ground or building floor to which the equipment is attached is used. For the case of equipment having supports with different dynamic motions, the most severe floor response spectrum is applied to all of the supports.

Refer to Subsections 3.9.3.2.3.1.4 and 3.9.3.2.5.1.2 for additional information on the dynamic qualification of active pumps and valves, respectively.

3.9.2.2.2.16 Supports

Subsections 3.9.3.4 and 3.9.3.5 address analyses or tests that are performed for component supports to assure their structural capability to withstand seismic and other dynamic excitations.

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The major reactor internal components within the vessel are subjected to extensive testing coupled with dynamic system analyses to properly evaluate the resulting flow-induced vibration (FIV) phenomena during normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Special analyses of the response signals measured for reactor internals of many similar designs are performed to obtain the parameters which determine the amplitude and model contributions in the vibration responses. These studies provide useful predictive information for extrapolating the results from tests of components with similar designs to components of different designs. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

(1) Dynamic analyses of major components and subassemblies are performed to identify vibration modes and frequencies. The analysis models used for Seismic Category I structures are similar to those outlined in Subsection 3.7.2.

(2) Data from previous plant vibration measurements are assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among BWRs of differing size and design.

(3) Parameters are identified which are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates and structural parameters such as natural frequency and significant dimensions.

(4) Correlation functions of the variable parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response mode.

(5) Predicted vibration amplitudes for components of the prototype plant are obtained from these correlation functions based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analyses of item (1).

The dynamic modal analysis forms the basis for interpretation of the preoperational and initial startup test results (Subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of ±68.6 MPa.

Vibratory loads are continuously applied during normal operation and the stresses are limited to ±68.6 MPa to prevent fatigue failure. Prediction of vibration amplitudes, mode shapes, and frequencies of normal reactor operations are based on statistical extrapolation of actual measured results on the same or similar components in reactors now in operation.

The dynamic loads due to flow-induced vibration from the feedwater jet impingement have no significant effect on the steam separator assembly. Analyses are performed to show that the impingement feedwater jet velocity is below the critical velocity. Also, it can be shown that the excitation frequency of the steam separator skirt is very different from the natural frequency of the skirt.

The calculated stresses due to hydrodynamic forces during core flooding operation are small and considered negligible when compared to the design-allowable stresses. Locations for which calculations were made include the weld joints, elbows, and rings.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Reactor internals vibration measurement and inspection programs are conducted during preoperational and initial startup testing in accordance with guidelines of Regulatory Guide 1.20 for prototype reactor internals. A flanged nozzle is provided in the top head of the reactor pressure vessel for bolting of the flange associated with the prototype test instrumentation. These programs are conducted in the three phases described as follows:

(1) **Preoperational Tests Prior to Fuel Loading:** Steady-state test conditions include balanced recirculation system operation and unbalanced operation over the full range of flow rates up to rated flow. Transient flow conditions include single and multiple pump trips from rated flow. This subjects major components to a minimum of 10^6 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements are obtained during this test and a close visual inspection of internals is conducted before and after the test.

(2) **Precritical Testing with Fuel:** This vibration measurement series is conducted with the reactor assembly complete but prior to reactor criticality. Flow conditions include balanced, unbalanced, and transient conditions as for the first test series. The purpose of this series is to verify the anticipated effects of the fuel on the vibration response of internals. Previous vibration measurements in BWRs (Reference 3.9-3) have shown that the fuel adds damping and reduces vibration amplitudes of major internal structures; thus, the first test series (without fuel) is a conservative evaluation of the vibration levels of these structures.

(3) **Initial Startup Testing:** Vibration measurements are made during reactor startup at conditions up to 100% rated flow and power. Balance, unbalanced, and transient conditions of recirculation system operation will be evaluated. The primary purpose of this test series is to verify the anticipated effect of two-phase flow on the vibration response of internals. Previous vibration measurements in BWRs (Reference 3.9-3) have shown that the effect of the two-phase flow is to broaden the frequency response spectrum and diminish the maximum response amplitude of the shroud and core support structures.

Vibration sensor types may include strain gauges, displacement sensors (linear variable transformers), and accelerometers.

Accelerometers are provided with double integration signal conditioning to give a displacement output. Sensor locations include the following:

(1) Top of shroud head, lateral acceleration(displacement)

(2) Top of shroud, lateral displacement

(3) Control rod drive housings, bending strain

(4) Incore housings, bending strain

(5) Core flooder internal piping, bending strain

In addition to these components, vibration of the core spray sparger is measured during preoperational testing of that system at the designated prototype.

In all prototype plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data are recorded on magnetic tape and provision is made for selective online analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the dominant vibration modes of each component by the test engineer using frequency, phase, and amplitude information for the component dynamic analyses. Comparison of measured vibration amplitudes to predicted and allowable amplitudes is then to be made on the basis of the analytically obtained normal mode which best approximates the observed mode.

The visual inspections conducted prior to and following preoperational testing are for vibration, wear, or loose parts. At the completion of preoperational testing, the reactor vessel head and the shroud head are removed, the vessel is drained, and major components are inspected on a selected basis. The inspections cover the shroud, shroud head, core support structures, recirculation internal pumps, the peripheral control rod drive, and incore guide tubes. Access is provided to the reactor lower plenum for these inspections.

The analysis, design and/or equipment that are to be utilized in a facility will comply with Regulatory Guide 1.20 as explained below.

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. The vibration assessment program meets the requirements of Criterion 1, Quality Standards and Records 10CFR50 Appendix A and Section 50.34 (Contents of Applications; Technical Information) of 10CFR50. This Regulatory Guide is applicable to the core support structures and other reactor internals.

Vibration testing of reactor internals is performed on all GE-BWR plants. At the time of original issue of Regulatory Guide 1.20, test programs for compliance were instituted for the then designed reactors. The first ABWR plant is considered a prototype and is instrumented and subjected to preoperation and startup flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage. Subsequent plants which have internals similar to those of the prototypes are also tested in compliance with the requirements of Regulatory Guide 1.20. GE is committed to confirm satisfactory vibration performance of internals in these plants through preoperational flow testing followed by inspection for evidence of excessive vibration. Extensive vibration measurements in prototype plants, together with satisfactory operating experience in all BWR plants, have established the adequacy of reactor internal designs. GE continues these test programs for the generic plants to verify structural integrity and to establish the margin of safety.

See Subsection 3.9.7.1 for COL license information pertaining to the reactor internals vibration testing program.

3.9.2.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

The faulted events that are evaluated are defined in Subsection 3.9.5.2.1. The loads that occur as a result of these events and the analysis performed to determine the response of the reactor internals are as follows:

(1) **Reactor Internal Pressures—**The reactor internal pressure differentials (Figure 3.9-1a) due to assumed break of main steam or feedwater line are determined by analysis as described in Subsection 3.9.5.2.2. In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces during an accident, a comparison is made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods are determined from a comprehensive vertical dynamic model of the RPV and internals with 12 degrees of freedom. Besides the real masses of the RPV and core support structures, account is made for the water inside the RPV.

(2) **External Pressure and Forces on the Reactor Vessel—**An assumed break of the main steamline, the feedwater line or the RHR line at the reactor vessel nozzle results in jet reaction and impingement forces on the vessel and asymmetrical pressurization

of the annulus between the reactor vessel and the shield wall. These time-varying pressures are applied to the dynamic model of the reactor vessel system. Except for the nature and locations of the forcing functions, the dynamic model and the dynamic analysis method are identical to those for seismic analysis as described below. The resulting loads on the reactor internals, defined as LOCA loads, are shown in Table 3.9-3.

(3) **Safety/Relief Valve Loads (SRV Loads)—**The discharge of the SRVs results in reactor building vibration (RBV) due to suppression pool dynamics (Appendix 3B). The response of the reactor internals to the RBV is also determined with the dynamic model and dynamic analysis method described below for seismic analysis.

(4) **LOCA Loads—**The assumed LOCA also results in RBV due to suppression pool dynamics (Appendix 3B), and the responses of the reactor internals are again determined with the dynamic model and dynamic analysis method used for seismic analysis. Various types of LOCA loads are identified on Table 3.9-2.

(5) **Seismic Loads—**The theory, methods, and computer codes used for dynamic analysis of the reactor vessel, internals, attached piping and adjoining structures are described in Section 3.7 and Subsection 3.9.1.2. Dynamic analysis is performed by coupling the lumped-mass model of the reactor vessel and internals with the building model to determine the system natural frequencies and mode shapes. The relative displacement, acceleration, and load response is then determined by either the time-history method or the response-spectrum method. The load on the reactor internals due to faulted event SSE are obtained from this analysis.

The above loads are considered in combination as defined in Table 3.9-2. The SRV, LOCA (SBL, IBL or LBL) and SSE loads defined in Table 3.9-2 are all assumed to act in the same direction. The peak colinear responses of the reactor internals to each of these loads are added by the square-root-of-the-sum-of-the-squares (SRSS) method. The resultant stresses in the reactor internal structures are directly added with stress resulting from the static and steady-state loads in the faulted load combination, including the stress due to peak reactor internal pressure differential during the LOCA. The reactor internals satisfy the stress deformation and fatigue limits as defined in Subsection 3.9.5.3.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Prior to initiation of the instrumented vibration measurement program for the prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test will be analyzed in detail.

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic and other Reactor Building vibration (RBV) events for the design of safety-related ASME Code components (except containment components, which are discussed in Section 3.8).

This section discusses the ASME Class 1, 2, and 3 equipment and associated pressure-retaining parts and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of major equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for ASME Class 1, 2, and 3 equipment are covered in Subsection 3.9.1.1. Seismic-related loads and dynamic analyses are discussed in Section 3.7. The suppression pool-related RBV loads are described in Appendix 3B. Table 3.9-2 presents the combinations of dynamic events to be considered for the design and analysis of all ABWR ASME Code Class 1, 2, and 3 components, component supports, core support structures and equipment. Specific loading combinations considered for evaluation of each specific equipment are derived from Table 3.9-2 and are contained in the design specifications and/or design reports of the respective equipment.

Piping loads due to the thermal expansion of the piping and thermal anchor movements at supports are included in the piping load combinations. All operating modes are evaluated and the maximum moment ranges are included in the fatigue evaluation. [*Piping systems with maximum operating temperatures of less than or equal to 65°C are not required to be analyzed for thermal expansion loading.*]*

[*Low-pressure piping systems that interface with the reactor coolant pressure boundary will be designed with a pipe wall thickness calculated for a pressure equal to 0.4 times the reactor coolant system pressure but no less than that of a schedule 40 pipe.*]* See Appendix 3M for additional information on intersystem LOCA.

Thermal stratification of fluids in a piping system is one of the specific operating conditions included in the loads and load combinations contained in the piping design specifications and design reports. It is known that stratification can occur in the feedwater piping during plant startup and when the plant is in hot standby conditions following scram (Subsection 3.9.2.1.3).

* See Subsection 3.9.1.7.

[*If, during design or startup, evidence of thermal stratification is detected in any other piping system, then stratification will be evaluated. If it cannot be shown that the stresses in the pipe are low and that movement due to bowing is acceptable, then stratification will be treated as a design load. In general, if temperature differences between the top and bottom of the pipe are less than 27°C, it may be assumed that design specification and stress reports need not be revised to include stratification. The piping design reports shall be inspected to confirm that the piping systems have been designed for thermal stratification in accordance with the requirements of this paragraph.*]*

Under thermally stratified flow conditions, it has been observed that a relatively thin dynamic interface region exists, which oscillates in a wave pattern. This results in undulation in the hot-to-cold interface region which produces thermal striping on the inside of the pipe wall. Thermal striping stresses are the result of differences between the pipe inside surface temperatures which vary with time due to the interface oscillation and the average through-wall temperatures. The results of the feedwater piping thermal striping stress analysis confirm that the feedwater thermal striping fatigue usage is minimal per the ASME Code, Section III, fatigue evaluation requirement; therefore, thermal striping fatigue effects are negligible.

[*Supplement 3 to Bulletin 88-08 involves the development of potential cyclic stratified flow and associated thermal striping that may occur because of possible leakage past the valve disk and out the valve stem packing gland. This flow stratification and striping may occur when the pressure on the upstream side of the valve is less than the RPV system pressure during normal operation. Sections of the RHR and HPCF Systems could be susceptible to unacceptable thermal stresses, owing to this phenomenon. To address the potential problem described in Supplement 3 to Bulletin 88-08, for the affected piping sections, it is required that either (1) the gate valve in each of the unisolable piping sections be located at a distance equal or greater than 25 pipe diameters from the RPV nozzle or (2) stress analysis be performed to show that stresses and fatigue from potential stratification and thermal striping are acceptable per ASME Code. Their requirements are incorporated in the design by note 32 of the RHR P&ID Figure 5.4-10 and note 29 of the HPCF P&ID Figure 6.3-7.*]*

The design life for the ABWR Standard Plant is 60 years. A 60-year design life is a requirement for all major plant components with reasonable expectation of meeting this design life. However, all plant operational components and equipment, except the reactor vessel, are designed to be replaceable, design life not withstanding. The design life requirement allows for refurbishment and repair, as appropriate, to assure that the design life of the overall plant is achieved. [*In effect, essentially all piping systems, components and equipment are designed for a 60-year design life. Many of these components are classified as ASME Class 2 or 3 or Quality Group D. In the event that any non-Class 1 components are subjected to cyclic loadings, including operating vibration loads and thermal transient effects, of a magnitude and/or duration so severe that the 60-year design life can not be assured by required Code calculations. COL applicants will identify these components and either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. Components excluded from this requirement are:*

(1) Tees where mixing of hot and cold fluids occurs and thermal sleeves have been provided in accordance with the P&IDs.

(2) *Feedwater piping outside containment that is designed so cyclic loadings and stresses are no more severe than experienced by Class 1 piping inside the containment.*]*

Severe thermal transients that will be evaluated for possible effect on plant life are temperature rate changes faster than 830°C/h, when the total fluid temperature change is greater than 55°C.

[*The safety relief valve (SRV) discharge piping in the wetwell and the SRV Quenchers are subjected to severe thermal transients during SRV blowdown events. Therefore, the COL applicant will perform ASME Class 1 fatigue analyses of the ASME Class 3 SRV discharge piping in the wetwell and the SRV Quenchers.*]* The purpose of these fatigue evaluations is to confirm that the fatigue stresses are less than their allowables. The fatigue evaluations will include the SRV blowdown thermal transient loads, thermohydraulic loads, Safe Shutdown earthquake loads and the reactor building vibration loads due to SRV blowdown. Environmental effects will be considered in the fatigue analysis in accordance with the requirements for ASME Section III Class 1 carbon steel piping specified in Subsection 3.9.3.1.1.7.

The SRV discharge piping in the wetwell will be analyzed for SRV blowdown thermal stresses due to a step change in temperature inside the pipe from 32°C to 166°C. In order to minimize piping thermal stresses, no shear lugs will be welded to the SRV discharge piping.

The fatigue analysis of the Quencher will be performed in accordance with ASME Section III, Subsection NB-3200. The quencher will be analyzed for heat transfer transient during SRV blowdown where there is a step change in temperature inside the quencher from 20°C to 166°C, and the outside of the quencher remains at 20°C. The fatigue evaluation will also include the SRV discharge pipe applied thermal loads, thermohydraulic transient loads, safe shutdown earthquake (SSE) loads and the SRV blowdown reactor building vibration loads.

See Subsection 3.9.7.2 for COL license information requirements.

3.9.3.1.1 Plant Conditions

All events that the plant will or might credibly experience during a reactor year are evaluated to establish design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence as discussed in Subsection 3.9.3.1.1.5) and correlated to service levels for design limits defined in ASME B&PV Code Section III as shown in Tables 3.9-1 and 3.9-2.

3.9.3.1.1.1 Normal Condition

Normal conditions are any conditions in the course of system startup, operation in the design power range, normal hot standby (with condenser available), and system shutdown other than upset, emergency, faulted, or testing.

* See Subsection 3.9.1.7.

3.9.3.1.1.2 Upset Condition

An upset condition is any deviation from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include system operational transients (SOT) which result from (1) any single operator error or control malfunction, (2) fault in a system component requiring its isolation from the system, (3) a loss of load or power. Hot standby with the main condenser isolated is an upset condition.

3.9.3.1.1.3 Emergency Condition

An emergency condition includes deviations from normal conditions which require shutdown for correction of the condition(s) or repair of damage in the reactor coolant pressure boundary (RCPB). Such conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, infrequent operational transients (IOT) caused by one of the following: (1) a multiple valve blowdown of the reactor vessel; (2) LOCA from a small break or crack (SBL) which does not depressurize the reactor systems, does not actuate automatically the ECCS operation, nor results in leakage beyond normal makeup system capacity, but which requires the safety functions of isolation of containment and shutdown and may involve inadvertent actuation of automatic depressurization system (ADS); (3) improper assembly of the core during refueling; or (4) improper or sudden start of one recirculation pump. Anticipated transient without scram (ATWS) or reactor overpressure with delayed scram (Tables 3.9-1 and 3.9-2) is an IOT classified as an emergency condition.

3.9.3.1.1.4 Faulted Condition

A faulted condition is any of those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events, such as LOCAs, that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These events are the most drastic that must be considered in the design and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: (1) a control rod drop accident; (2) a fuel-handling accident; (3) a main steamline or feedwater line break; (4) the combination of any small/intermediate break LOCA (SBL or IBL) with the safe shutdown earthquake, and a loss of offsite power; or (5) the safe shutdown earthquake plus large break LOCA (LBL) plus a loss of offsite power.

The IBL classification covers those breaks for which the ECCS operation will occur during the blowdown, and which results in reactor depressurization. The LBL classification covers the sudden, double-ended severance of a main steamline inside or outside the containment that results in transient reactor depressurization, or any pipe rupture of equivalent flow cross-sectional area with similar effects.

3.9.3.1.1.5 Correlation of Plant Condition with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation identifies the appropriate plant conditions and assigns the appropriate ASME Section III service levels for any hypothesized event or sequence of events.

Plant Condition	ASME Code Service Level	Event Encounter Probability per Reactor Year
Normal (planned)	A	1.0
Upset (moderate probability)	B	$1.0 > P \geq 10^{-2}$
Emergency (low probability)	C	$10^{-2} > P \geq 10^{-4}$
Faulted (extremely low probability)	D	$10^{-4} > P > 10^{-6}$

3.9.3.1.1.6 Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment and piping (Subsection 3.2.3) shall be capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could deteriorate its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment and piping shall be capable of accomplishing its safety functions as required by the event, but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

For active Class 2 and 3 pumps and active Class 1, 2 and 3 valves, specific stress criteria to meet the functional requirements are identified in a footnote to Table 3.9-2. For piping the specific stress criteria for functional requirements are identified in footnote 9 of Table 3.9-2.

3.9.3.1.1.7 Environmental Effects on Fatigue Evaluation of Carbon Steel Piping

[*Environmental effects on the fatigue design of ASME Section III Class 1 carbon steel piping will be evaluated in accordance with GE document, 408HA414 (Reference 3.9-9). Additional fatigue evaluations for environmental effects are not required for any of the following conditions: (a) water temperature is below 245°C, (b) fittings, such as elbows and tees, that are conservatively designed and analyzed using the ASME Section III stress indices and (c) for transients having total cycle times of 10 seconds or less and no tensile hold time, provided that the oxygen content of the water does not exceed 0.3 ppm.*

Environmental effects are considered by increasing the local peak stress through four factors used as multipliers to the stress indices. The four factors are: (1) the notch factor, (2) the mean stress factor, (3) the environmental correction factor, and (4) the butt weld strength reduction factor.]*

3.9.3.1.2 Reactor Pressure Vessel Assembly

The reactor pressure vessel assembly consists of the RPV, vessel support skirt, and shroud support.

The reactor pressure vessel, vessel support skirt, and shroud support are constructed in accordance with ASME Code Section III. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The RPV assembly components are classified as ASME Class 1. Complete stress reports on these components are prepared in accordance with ASME Code requirements. NUREG-0619 (Reference 3.9-5) is also considered for feedwater nozzle and other such RPV inlet nozzle design.

The stress analysis is performed on the reactor pressure vessel (RPV), vessel support skirt, and shroud support for various plant operating conditions (including faulted conditions) by using the elastic methods except as noted in Subsection 3.9.1.4.2. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are discussed in Subsection 3.9.5.

3.9.3.1.3 Main Steam (MS) System Piping

The piping systems extending from the reactor pressure vessel to and including the outboard MSIV are constructed in accordance with ASME Code Section III, Class 1 criteria. Stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of ASME Code Section III.

The MS System piping extending from the outboard MSIV to the turbine stop valve is constructed in accordance with ASME Code Section III, Class 2 criteria.

Turbine stop valve (TSV) closure in the main steam (MS) piping system results in a transient that produces momentary unbalanced forces acting on the MS piping system. Upon closure of the TSV, a pressure wave is created and it travels at sonic velocity toward the reactor vessel through each MS line. Flow of steam into each MS line from the reactor vessel continues until the steam compression wave reaches the reactor vessel. Repeated reflections of the pressure wave at the reactor vessel and the TSV produce time varying pressures and velocities, throughout the MS lines.

The analysis of the MS piping TSV closure transient consists of a stepwise time-history solution of the steam flow equation to generate a time-history of the steam properties at numerous

* See Subsection 3.9.1.7.

locations along the pipe. Reaction loads on the pipe are determined at each elbow. These loads are composed of pressure-times-area, momentum change and fluid-friction terms.

The time-history direct integration method of analysis is used to determine the response of the MS piping system to TSV closure. The forces are applied at locations on the piping system where steam flow changes direction, thus causing momentary reactions. The resulting loads on the MS piping are combined with loads due to other effects specified in Subsection 3.9.3.1.

3.9.3.1.4 Recirculation Motor Cooling (RMC) Subsystem

The RMC System piping loop between the recirculation motor casing and the heat exchanger is constructed in accordance with ASME Code Section III, Subsection NB-3600. Stresses are calculated on an elastic basis and evaluated in accordance with NB-3600.

3.9.3.1.5 Recirculation Pump Motor Pressure Boundary

The motor casing of the recirculation internal pump is a part of and welded into an RPV nozzle and is constructed in accordance with requirements of an ASME Code Section III, Class 1 component. The motor cover is a part of the pump/motor assembly and is constructed as an ASME Class 1 component. These pumps are not required to operate during the SSE or after an accident.

3.9.3.1.6 Standby Liquid Control (SLC) Tank

The standby liquid control tank is constructed in accordance with the requirements of a ASME Code Section III, Class 2 component.

3.9.3.1.7 RRS and RHR Heat Exchangers

The primary and secondary sides of the RRS are constructed in accordance with the requirements of an ASME Code Section III, Class 1 and Class 2 component, respectively. The primary and secondary side of the RHR System heat exchanger is constructed as ASME Class 2 and Class 3 component, respectively.

3.9.3.1.8 RCIC Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine is designed and evaluated and fabricated following the basic guidelines of ASME Code Section III for Class 2 components.

3.9.3.1.9 ECCS Pumps

The RHR, RCIC, and HPCF pumps are constructed in accordance with the requirements of an ASME Code Section III, Class 2 component.

3.9.3.1.10 Standby Liquid Control (SLC) Pump

The SLC System pump is constructed in accordance with the requirements for ASME Code Section III, Class 2 components.

3.9.3.1.11 Standby Liquid Control (SLC) Valve (Injection Valve)

The SLC System injection valve is constructed in accordance with the requirements for ASME Code Section III, Class 1 components.

3.9.3.1.12 Main Steam Isolation and Safety/Relief Valves

The MSIVs and SRVs are constructed in accordance with ASME Code Section III, Subsection NB-3500, requirements for Class 1 components.

3.9.3.1.13 Safety/Relief Valve Piping and Quencher

The SRV discharge piping in the drywell extending from the relief valve discharge flange to the diaphragm floor penetration and the SRV discharge piping in the wetwell, extending from the diaphragm floor penetration to and including the quencher, is constructed in accordance with ASME Code Section III requirements for Class 3 components. In addition, all welds in the SRVDL piping in the wetwell above the surface of the suppression pool shall be non-destructively examined to the requirements of ASME Code Section III, Class 2.

3.9.3.1.14 Reactor Water Cleanup (CUW) System Pump and Heat Exchangers

The CUW pump and heat exchangers (regenerative and nonregenerative) are not part of a safety system and are non-Seismic Category I equipment. ASME Code Section III for Class 3 components is used as a guide in constructing the CUW System pump and heat exchanger components.

3.9.3.1.15 Fuel Pool Cooling and Cleanup System Pumps and Heat Exchangers

The pumps and heat exchangers are constructed in accordance with the requirements for ASME Code Section III Class 3 components.

3.9.3.1.16 ASME Class 2 and 3 Vessels

The Class 2 and 3 vessels (all vessels not previously discussed) are constructed in accordance with ASME Code Section III. The stress analysis of these vessels is performed using elastic methods.

3.9.3.1.17 ASME Class 2 and 3 Pumps

The Class 2 and 3 pumps (all pumps not previously discussed) are designed and evaluated in accordance with ASME Code Section III. The stress analysis of these pumps is performed using elastic methods. See Subsection 3.9.3.2 for additional information on pump operability.

3.9.3.1.18 ASME Class 1, 2 and 3 Valves

The Class 1, 2, and 3 valves (all valves not previously discussed) are constructed in accordance with ASME Code Section III.

All valves and their extended structures are designed to withstand the accelerations due to seismic and other RBV loads. The attached piping is supported so that these accelerations are not exceeded. The stress analysis of these valves is performed using elastic methods. See Subsection 3.9.3.2 for additional information on valve operability.

3.9.3.1.19 ASME Class 1, 2 and 3 Piping

The Class 1, 2 and 3 piping (all piping not previously discussed) is constructed in accordance with ASME Code Section III. For Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of ASME Code Section III. For Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NC/ND-3600 of the Code.

[*The effects of displacement-limited, seismic anchor motions (SAM) due to an SSE shall be evaluated for ASME Class 1, 2 and 3 piping design in accordance with footnote 6 of Table 3.9-1.*]*

3.9.3.1.20 As-Built Stress Reports for ASME Class 1, 2 and 3 Piping Systems

[*For ASME Class 1, 2 and 3 piping systems, the as-built piping system shall be reconciled with the as-designed piping system. An as-built inspection of the pipe routing, location and orientation, the location, size, clearances and orientation of piping supports, and the location and weight of pipe mounted equipment shall be performed. This inspection will be performed by reviewing the as-built drawings containing verification stamps, and by performing a visual inspection of the installed piping system. The piping configuration and component location, size, and orientation shall be within the tolerances specified in the certified as-built piping stress report. The tolerances to be used for reconciliation of the as-built piping system with the as-designed piping system are provided in Reference 3.9-10. A reconciliation analysis using the as-built and as-designed information shall be performed. The certified as-built Stress Report shall document the results of the as-built reconciliation analysis.*]*

3.9.3.1.21 Pipe-Mounted Equipment Allowable Loads

[*The piping design reports shall document that the pipe applied loads on attached equipment; such as valves, pumps, tanks and heat exchangers, are less than the equipment vendor's specified allowable loads.*]*

* See Subsection 3.9.1.7.

3.9.3.1.22 ASME Class 1, 2 and 3 Piping System Clearance Requirements

[*ASME Class 1, 2 and 3 piping systems shall be designed to provide clearance from structures, systems, and components where necessary for the accomplishment of the structure, system, or component's safety function as specified in the representative structure or system design description. The maximum calculated piping system deflections under service conditions shall be verified that they do not exceed the minimum clearance between the piping system and nearby structures, systems, or components. The certified design stress report shall document that the clearance requirements have been met.*]*

3.9.3.2 Pump and Valve Operability Assurance

Active mechanical (with or without electrical operation) equipment are Seismic Category I and each is designed to perform a mechanical motion for its safety-related function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include active pumps and valves in fluid systems such as the RHR System, ECCS, and MS system.

This subsection discusses operability assurance of active ASME Code Section III pumps and valves, including motor, turbine or operator that is a part of the pump or valve (Subsection 3.9.2.2). The COL applicant must ensure that specific environmental parameters are properly defined and enveloped in the methodology for its specific plant and implemented in its equipment qualification program.

Safety-related valves and pumps are qualified by testing and analysis and by satisfying the stress and deformation criteria at the critical locations within the pumps and valves. Operability is assured by meeting the requirements of the programs defined in Subsection 3.9.2.2, design and qualification requirements Subsection 3.9.6, Sections 3.10 and 3.11, and the following subsections.

Section 4.4 of GE's Environmental Qualification Program (Reference 3.9-6) applies to this subsection, and the seismic qualification methodology presented therein is applicable to mechanical as well as electrical equipment.

3.9.3.2.1 ECCS Pumps, Motors and Turbine

Dynamic qualification of the ECCS (RHR, RCIC and HPCF) pumps with motor or turbine assembly is also described in Subsections 3.9.2.2.2.6 and 3.9.2.2.2.7.

3.9.3.2.1.1 Consideration of Loading, Stress, and Acceleration Conditions in the Analysis

In order to avoid damage to the ECCS pumps during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, other RBV loads, and dynamic system loads are limited to the material elastic limit. A three-dimensional finite-element model of the pump and associated motor (see Subsections 3.9.3.2.2 and 3.9.3.2.1.5 for RCIC pump and turbine, respectively) and its support is developed and analyzed using the response spectrum and the dynamic analysis method. [*The same is analyzed due to static nozzle loads, pump thrust loads, and dead weight. Critical location stresses are compared with the allowable stresses and the critical location deflections with the allowables, and accelerations are checked* *to evaluate operability. The average membrane stress σm for the faulted condition loads is limited to 1.2S or approximately 0.75 σ_y (σ_y = yield stress), and the maximum stress in local fibers (σm + bending stress σb) is limited to 1.8S or approximately 1.1 σ_y. The maximum faulted event nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.*]*

Performing these analyses with the conservative loads stated and with the restrictive stress limits as allowables assures that critical parts of the pump and associated motor or turbine will not be damaged during the faulted condition and that the operability of the pump for post-faulted condition operation will not be impaired.

3.9.3.2.1.2 Pump/Motor Operation During and Following Dynamic Loading

Active ECCS pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor and the nature of the random short duration loading characteristics of the dynamic event prevent the rotor from becoming seized. The seismic and other RBV loadings can be predicted to require only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed; therefore, the pump is expected to operate at the design speed during the faulted event loads.

The functional ability of the active pumps after a faulted condition is assured, since only normal operating loads and steady-state nozzle loads exist. For the active pumps, the faulted condition loads are greater than the normal condition loads only due to the SSE and other RBV transitory loads. These faulted events are infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be no worse than the normal plant operating limits. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions be limited to the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.1.3 ECCS Pumps

All active ECCS (RHR, RCIC and HPCF) pumps are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include: (1) hydrostatic tests of pressure-retaining parts of 125% of the design pressure; (2) seal leakage tests; and (3) performance tests while the pump is operated with flow to determine total developed head, minimum and maximum head and net positive suction head (NPSH) requirements. Also monitored during these operating tests are bearing temperatures (except water cooled bearings) and vibration levels. Both are shown to be below specified limits. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests,

* See Section 3.10.

and the required periodic inservice inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, these pumps are analyzed for operability during a faulted condition by assuring that (1) the pump will not be damaged during the dynamic (SSE and LOCA) event, and (2) the pump will continue operating despite the dynamic loads.

3.9.3.2.1.4 ECCS Motors

Qualification of the Class 1E motors used for the ECCS motors complies with IEEE-323. The qualification of all motor sizes is based on completion of a type test, followed up with review and comparison of design and material details, and seismic and other RBV loads analyses of production units, ranging from 447 to 2610 kW, with the motor used in the type test. All manufacturing, inspection, and routine tests by motor manufacturer on production units are performed on the test motor.

The type test is performed on a 932 kw vertical motor in accordance with IEEE-323, first simulating a normal operation during the design life, then subjecting the motor to a number of vibratory tests, and then to the abnormal environmental condition possible during and after a LOCA. The test plans for the type test are as follows:

(1) Thermal aging of the motor electrical insulation system (which is a part of the stator only) is based on extrapolation in accordance with the temperature life characteristic curve from IEEE-275 for the insulation type used on the ECCS motors. The amount of aging equals the total estimated operation days at maximum insulation surface temperature.

(2) Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma during normal and abnormal conditions.

(3) The normal operational induced current vibration effect on the insulation system is simulated by 1.5g horizontal vibration acceleration at current frequency for one hour duration.

(4) The dynamic load deflection analysis on the rotor shaft is performed to ensure adequate rotation clearance, and is verified by static loading and deflection of the rotor for the type test motor.

(5) Dynamic load aging and testing is performed on a biaxial test table in accordance with IEEE-344. During this test, the shake table is activated to simulate the maximum design limit for the SSE and other RBV loads with as many motor starts and operation combinations consistent with the plant events of Table 3.9-1 and the ECCS inadvertent injections and tests planned over the life of the plant.

(6) An environmental test simulating a LOCA condition with a duration of 100 days is performed with the test motor fully loaded, simulating pump operation. The test consists of startup and six hours operation at 100°C ambient temperature and 100% steam environment. Another startup and operation of the test motor after one hour standstill in the same environment is followed by sufficient operation at high humidity and temperature based on extrapolation in accordance with the temperature life characteristic curve from IEEE-275 for the insulation type used on the ECCS motors.

3.9.3.2.1.5 RCIC Turbine

The RCIC turbine is qualified by a combination of static analysis and dynamic testing as described in Subsection 3.9.2.2.2.7. The turbine assembly consists of rigid masses (wherein static analysis is utilized) interconnected with control levers and electronic control systems, necessitating final qualification by dynamic testing. Static loading analysis has been employed to verify the structural integrity of the turbine assembly, and the adequacy of bolting under operating and dynamic conditions. The complete turbine assembly is qualified via dynamic testing, in accordance with IEEE-344. The qualification test program includes demonstration of startup capability, as well as operability during dynamic loading conditions. Operability under normal load conditions is assured by comparison to operability of similar turbines in operating plants.

3.9.3.2.2 SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies with natural frequencies well above 33 Hz. With this fact verified, each equipment assembly is qualified by the static analysis for seismic and other RBV loads. This qualification assures structural loading stresses within Code limitations, and verifies operability under seismic and other RBV loads (Subsections 3.9.2.2.2.8 and 3.9.2.2.2.7).

3.9.3.2.3 Other Active Pumps

The active pumps not previously discussed are ASME Class 2 or 3 and Seismic Category I. They are designed to perform their function including all required mechanical motions during and after a dynamic (seismic and other RBV) loads event and to remain operative during the life of the plant.

The program for the qualification of Seismic Category I components conservatively demonstrates that no loss of function results either before, during, or after the occurrence of the combination of events for which operability must be assured. No loss of function implies that the pressure boundary integrity will be maintained, that the component will not be caused to operate improperly, and that components required to respond actively will respond properly as appropriate to the specific equipment. In general, operability assurance is established during and after the dynamic loads event for active components.

3.9.3.2.3.1 Procedures

Procedures have been established for qualifying the mechanical portions of Seismic Category I pumps such as the body which forms a fluid pressure boundary, including the suction and discharge nozzles, shaft and seal retainers, impeller assembly (including blading, shaft, and bearings for active pumps), and integral supports.

All active pumps are qualified for operability by first being subjected to rigid tests both prior to installation and after installation in the plant. Electric motors for active pumps and instrumentation, including electrical devices which must function to cause the pump to accomplish its intended function, are discussed separately in Subsection 3.9.3.2.5.1.3.

3.9.3.2.3.1.1 Hydrostatic Test

All seismic-active pumps shall meet the hydrostatic test requirements of ASME Code Section III according to the class rating of the given pump.

3.9.3.2.3.1.2 Leakage Test

The fluid pressure boundary is examined for leaks at all joints, connections, and regions of high stress such as around openings or thickness transition sections while the pump is undergoing a hydrostatic test or during performance testing. Leakage rates that exceed the rates permitted in the design specification are eliminated and the component retested to establish an observed leakage rate. The actual observed leakage rate, if less than permitted, is documented and made a part of the acceptable documentation package for the component.

3.9.3.2.3.1.3 Performance Test

The pump is demonstrated capable of meeting all hydraulic requirements while operating with flow at the total developed head, minimum and maximum head, NPSH, and other parameters as specified in the equipment specification.

Bearing temperature (except water cooled bearings) and vibration levels are also monitored during these operating tests. Both are shown to be below specified levels.

3.9.3.2.3.1.4 Dynamic Qualification

The safety-related active pumps are analyzed for operability during dynamic loading event by assuring that the pump is not damaged during the seismic event and the pump continues operating despite the dynamic loads.

A test or dynamic analysis is performed for a pump to determine the dynamic seismic and other RBV load from the applicable floor response spectra.

Response spectra for the horizontal vibration are used in two orthogonal horizontal directions simultaneously with the response spectra for the vertical vibration. The effects from the three simultaneous accelerations are combined by the square-root-of-the-sum-of-the-squares method. [*The pump is demonstrated by test or analysis that the faulted condition nozzle loads do not impair the operability of the pumps during or following the faulted condition.*]* Components of the pump are considered essentially rigid when having a natural frequency above 33 Hz. A static shaft deflection analysis of the motor rotor is performed with the conservative SSE accelerations acting in horizontal and vertical direction simultaneously.

The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The allowable rotor clearances are limited by the deflection which would cause the rotor to just make contact with the stator. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE and dynamic system loads are limited to the material elastic limit.

[*The average membrane stress (σ_m) for the faulted conditions loads is limited to 1.2S or approximately 0.75 σ_y (σ_y = yield stress), and the maximum stress in local fibers (σm + bending stress σb) is limited to 1.8S or approximately 1.1 σ_y. The maximum dynamic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.*]†

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis.

In completing the seismic qualification procedures, the pump motor and all components vital to the operation of the pump are independently qualified for operation during the maximum seismic event by IEEE-344.

[*If the testing option is chosen, sine-beat testing for electrical equipment is performed by satisfying one or more of the following requirements to demonstrate that multi-frequency response is negligible or that the sine-beat input is of sufficient magnitude to conservatively account for this effect.*

(1) The equipment response is basically due to one mode.

(2) The sine-beat response spectra envelope the floor response spectra in the region of significant response.

(3) *The floor response spectra consist of one dominant mode and has a peak at this frequency.*]*

The degrees of cross coupling in the equipment shall determine if a single or multi-axis test is required. Multi-axis testing is required if there is considerable cross coupling. If coupling is very light, then single axis testing is justified. Or, if the degree of coupling can be determined,

* See Section 3.10.

† See Section 3.10.

then single-axis testing can be used with input sufficiently increased to include the effect of coupling on the response of the equipment.

The combined stresses of the support structures are designed to be within the limits of ASME Code Section III, Subsection NF, component Support Structures and/or other comparable limits of industry standards such as the AISC Specification for Buildings, plus Addenda for building support structures.

An analysis or test is accomplished which conservatively demonstrates structural integrity and/or functionality of the equipment supports.

The impeller, shaft, and bearings for active pumps are analyzed to determine adequacy while operating with the seismic and other RBV loading effects applied in addition to the applicable operating loads including nozzle loads. Functional requirements are partially demonstrated by a suitable analysis which conservatively shows the following:

(1) The stresses in the shaft do not exceed the minimum yield strength of the material used for its construction.

(2) The deflections of the shaft and/or impeller blades do not cause the impeller assembly to seize.

(3) The bearing temperature does not attain limits which may allow stresses in the bearing or bearing support to exceed minimum yield strength levels or jeopardize lubrication.

3.9.3.2.3.2 Documentation

All of the preceding requirements (Subsection 3.9.3.2.3.1) are satisfied to demonstrate that functionality is assured for active pumps. The documentation is prepared in a format that clearly shows that each consideration has been properly evaluated and tests have been validated by a designated quality assurance representative. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.2.4 Major Active Valves

Some of the major safety-related active valves (Table 6.2-2, 6.2-3 and 3.2-1) discussed in this subsection for illustration are the MSIVs and SRVs, and SLC valves and HPCF valves (motor-operated). These valves are designed to meet ASME Code Section III requirements and perform their mechanical motion in conjunction with a dynamic (SSE and other RBV) load event. These valves are supported entirely by the piping
(i. e., the valve operators are not used as attachment points for piping supports)
(Subsection 3.9.3.4.1). The dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2.4.1 Main Steam Isolation Valve

The typical Y-pattern MSIVs described in Subsection 5.4.5.2 are evaluated by analysis and test for capability to operate under the design loads that envelop the predicted loads during a DBA and SSE.

The valve body is designed, analyzed and tested in accordance with ASME Code Section III Class 1 requirements. The MSIVs are modeled mathematically in the MS System analysis. The loads, amplified accelerations and resonance frequencies of the valves are determined from the overall steamline analysis. The piping supports (snubbers, rigid restraints, etc.) are located and designed to limit amplified accelerations of and piping loads in the valves to the design limits.

As described in Subsection 5.4.5.3, the MSIV and associated electrical equipment (wiring, solenoid valves, and position switches) are dynamically qualified to operate during an accident condition.

3.9.3.2.4.2 Main Steam Safety/Relief Valve

The typical SRV design described in Subsection 5.2.2.4.1 is qualified by type test to IEEE-344 for operability during a dynamic event. Structural integrity of the configuration during a dynamic event is demonstrated by both Code (ASME Class 1) analysis and test.

(1) Valve is designed for maximum moments on inlet and outlet which may be imposed when installed in service. These moments are resultants due to dead weight plus dynamic loading of both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge.

(2) A production SRV is demonstrated for operability during a dynamic qualification (shake table) type test with moment and “g” loads applied greater than the required equipment’s design limit loads and conditions.

A mathematical model of this valve is included in the MS System analysis, as with the MSIVs. This analysis assures that the equipment design limits are not exceeded.

3.9.3.2.4.3 Standby Liquid Control Valve (Injection Valve)

The typical SLC injection valve design is qualified by type test to IEEE-344. The valve body is designed, analyzed and tested per ASME Code Section III Class 1. The qualification test demonstrates the ability to remain operable after the application of the horizontal and vertical dynamic loading exceeding the predicted dynamic loading.

3.9.3.2.4.4 High Pressure Core Flooder Valve (Motor-Operated)

The typical HPCF valve body design, analysis and testing is in accordance with the requirements of the ASME Code Section III Class 1 or 2 components. The Class 1E electrical motor actuator is qualified by type test in accordance with IEEE-382, as discussed in Subsection 3.11.2. A mathematical model of this valve is included in the HPCF piping system analysis. The analysis results are assured not to exceed the horizontal and vertical dynamic acceleration limits acting simultaneously for a dynamic (SSE and other RBV) event, which is treated as an emergency condition. Subsection 3.9.3.2.5 discusses the operability qualification of the HPCF valve for seismic and other dynamic loads.

3.9.3.2.5 Other Active Valves

Other safety-related active valves are ASME Class 1, 2 or 3 and are designed to perform their mechanical motion during dynamic loading conditions. The operability assurance program ensures that these valves will operate during a dynamic seismic and other RBV event.

3.9.3.2.5.1 Procedures

Qualification tests accompanied by analyses are conducted for all active valves. Procedures for qualifying electrical and instrumentation components which are depended upon to cause the valve to accomplish its intended function are described in Subsection 3.9.3.2.5.1.3.

3.9.3.2.5.1.1 Tests

Prior to installation of the safety-related valves, the following tests are performed: (1) shell hydrostatic test to ASME Code Section III requirements; (2) back seat and main seat leakage tests; (3) disc hydrostatic test; (4) functional tests to verify that the valve will open and close within the specified time limits when subject to the design differential pressure; and (5) operability qualification of valve actuators for the environmental conditions over the installed life. Environmental qualification procedures for operation follow those specified in Section 3.11. The results of all required tests are properly documented and included as a part of the operability acceptance documentation package.

3.9.3.2.5.1.2 Dynamic Load Qualification

The functionality of an active valve during and after a seismic and other RBV event is demonstrated by test or by a combination of analysis and test. The qualification of electrical and instrumentation components controlling valve actuation is discussed in Subsection 3.9.3.2.5.1.3. The valves are designed using either stress analyses or the pressure temperature rating requirements based upon design conditions. A test or analysis of the extended structure is performed for the expected dynamic loads acting on the extended structure. See Subsection 3.9.2.2 for further details.

The maximum stress limits allowed in these analyses confirm structural integrity and are the limits developed and accepted by the ASME for the particular ASME Class of valve analyzed.

[*The stress limits for operability are provided in footnote 12 of Table 3.9-2.*]*

* See Section 3.10.

Dynamic load qualification is accomplished in the following way:

(1) All the active valves are typically designed to have a fundamental frequency which is greater than the high frequency asymptote (ZPA) of the dynamic event. This is shown by suitable test or analysis.

(2) The actuator and yoke of the valve system is statically loaded to an amount greater than that due to a dynamic event. The load is applied at the center of gravity of the actuator alone in the direction of the weakest axis of the yoke. The simulated operational differential pressure is simultaneously applied to the valve during the static deflection tests.

(3) The valve is then operated while in the deflected position (i.e., from the normal operating position to the safe position). The valve is verified to perform its safety-related function within the specified operating time limits.

(4) Motor operators and other electrical appurtenances necessary for operation are qualified as operable during a dynamic event by appropriate qualification tests prior to installation on the valve. Alternately, the valve including the motor operator and all other accessories is qualified by a shake table test.

The piping, stress analysis, and pipe support design maintain the motor operator accelerations below the qualification levels with adequate margin of safety.

If the fundamental frequency of the valve, by test or analysis, is less than that for the ZPA, a dynamic analysis of the valve is performed to determine the equivalent acceleration to be applied during the static test. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations have been determined using the same conservatism contained in the horizontal and vertical accelerations used for rigid valves. The adjusted acceleration is then used in the static analysis and the valve operability is assured by the methods outlined in Steps (2) through (4), using the modified acceleration input. Alternatively, the valve including the actuator and all other accessories is qualified by shake table test.

Valves which are safety-related but can be classified as not having an overhanging structure, such as check valves and pressure-relief valves, are considered as follows:

3.9.3.2.5.1.2.1 Active Check Valves

Due to the particular simple characteristics of the check valves, the active check valves are qualified by a combination of the following tests and analysis:

(1) Stress analyses including the dynamic loads where applicable

(2) In-shop hydrostatic tests

(3) In-shop seat leakage test

(4) Periodic in-situ valve exercising and inspection to assure the functional capability of the valve

3.9.3.2.5.1.2.2 Active Pressure-Relief Valves

The active pressure-relief valves (RVs) are qualified by the following procedures. These valves are subjected to test and analysis similar to check valves, stress analyses including the dynamic loads, in-shop hydrostatic seat leakage, and performance tests. In addition to these tests, periodic in-situ valve inspections, as applicable, and periodic valve removal, refurbishment, performance testing, and reinstallation are performed to assure the functional capability of the valve. Tests of the RV under dynamic loading conditions demonstrate that valve actuation can occur during application of the loads. The tests include pressurizing the valve inlet with nitrogen and subjecting the valve to accelerations equal to or greater than the dynamic event (SSE plus other RBV) loads.

3.9.3.2.5.1.2.3 Qualification of Electrical and Instrumentation Components Controlling Valve Actuation

A practical problem arises in attempting to describe tests for devices (relays, motors, sensors, etc.) as well as for complex assemblies such as control panels. It is reasonable to assume that a device, as an integral part of an assembly, can be subjected to dynamic loads tests while in an operating condition and its performance monitored during the test. However, in the case of complex panels, such a test is not always practical. In such a situation, the following alternate approach is recommended.

The individual devices are tested separately in an operating condition and the test levels recorded as the qualification levels of the devices. The panel, with similar devices installed but inoperative, is vibration tested to determine if the panel response accelerations, as measured by accelerometers installed at the device attachment locations, are less than the levels at which the devices were qualified. Note that the purpose of installing the nonoperating devices is to assure that the panel has the structural characteristics it will have when in use. If the acceleration levels at the device locations are found to be less than the levels to which the device is qualified, then the total assembly is considered qualified. Otherwise, either the panel is redesigned to reduce the acceleration level to the device locations and retested, or the devices is requalified to the higher levels.

3.9.3.2.5.2 Documentation

All of the preceding requirements (Subsection 3.9.3.2.5.1) are satisfied to demonstrate that functionality is assured for active valves. The documentation is prepared in a format that clearly shows that each consideration has been properly evaluated and tests have been validated by a designated quality assurance representative. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.3 Design and Installation of Pressure Relief Devices

3.9.3.3.1 Main Steam Safety/Relief Valves

SRV lift in a main steam (MS) piping system results in a transient that produces momentary unbalanced forces acting on the MS and SRV discharge piping system for the period from opening of the SRV until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the MS and discharge piping following the relatively rapid opening of the SRV cause this piping to vibrate.

The analysis of the MS and discharge piping transient due to SRV discharge consists of a stepwise time-history solution of the fluid flow equation to generate a time history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification and the value of ASME Code flow rating increased by a factor to account for the conservative method of establishing the rating. As a conservative approach, it is assumed that all SRVs mounted on a MS line actuate simultaneously. Simultaneous actuation of all SRVs is considered to induce maximum stress in the MS piping. Further, a subsequent actuation condition rather than initial actuation for all SRVs is conservatively assumed. This is a conservative approach, considering that all SRVs will not actuate simultaneously with subsequent actuation condition in the SRV piping, because individual SRVs have different relief set pressure values. These features should preclude simultaneous subsequent actuation of all SRVs. The methodology for calculating hydrodynamic loading on SRV discharge piping due to subsequent SRV actuations is consistent with previously approved methodology for earlier BWR (Mark II/III) designs.The effect of subsequent valve actuation is included by assuming hot SRV discharge pipe condition before valve actuation which results in higher loads on the piping. SRV loads are calculated assuming initial SRV pipe metal temperature to be 149°C for the pipe in the drywell region and 93°C for the pipe in the wetwell region, consistent with that used for earlier BWRs. These temperature values are based on measured data from in-plant SRV blowdown tests. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum-change, and fluid-friction terms.

The method of analysis applied to determine response of the MS piping system, including the SRV discharge line, to relief valve operation is time-history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the SRV, the main steamline, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. In accordance with Tables 3.9-1 and 3.9-2, the Code stress limits for service levels corresponding to load combination classification as normal, upset, emergency, and faulted are applied to the main steam and discharge pipe.

3.9.3.3.2 Other Safety/Relief Valves

An SRV is identified as a pressure relief valve or vacuum breaker. SRVs in the reactor components and subsystems are described and identified in Subsection 5.4.13.

The operability assurance program discussed in Subsection 3.9.3.2.5 applies to safety/relief valves. The qualification of active relief valves is specifically outlined in Subsection 3.9.3.2.5.1.2.2.

ABWR SRVs (safety valves with auxiliary actuating devices and pilot operated valves) are designed and manufactured in accordance with ASME Code Section III Division 1 requirements. Specific rules for pressure relieving devices are as specified in Article NB-7000 and NB-3500 (pilot-operated and power-actuated pressure relief valves).

The design of ABWR SRVs incorporates SRV opening and pipe reaction load considerations required by ASME Section III, Appendix O, including the additional criteria of SRP Section 3.9.3, Paragraph II.2 and those identified under Subsection NB-3658 for pressure and structural integrity. SRV operability is demonstrated either by dynamic testing or analysis of similarly tested valves or a combination of both in compliance with the requirements of SRP Subsection 3.9.3.

3.9.3.3.3 Rupture Disks

There are no rupture disks in the ABWR plant design that must function during and after a dynamic event (SSE including other RBV loads) at design basis conditions. However, the rupture disk in the containment overpressure protection system may operate following severe accident seismic conditions.

3.9.3.4 Component Supports

The design of bolts for component supports is specified in ASME Code Section III, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

Moreover, on equipment which is to be, or may be, mounted on a concrete support, sufficient holes for anchor bolts are provided to limit the anchor bolt stress to less than 68.6 MPa on the nominal bolt area in shear or tension.

Concrete anchor bolts (including under-cut type anchor bolts) which are used for pipe support base plates will be designed to the applicable factors of safety defined in IE Bulletin 79-02, “Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts”, Revision 2, November 8, 1979.[*Justification shall be provided for the use of safety factors for concrete* *anchor bolts other than those specified in IE Bulletin 79-02. This justification shall be submitted to the NRC staff for review and approval prior to the installation of the concrete anchor bolts. Pipe support base plate flexibilities are accounted for in the calculation of concrete anchor bolt loads, in accordance with IE Bulletin 79-02.*]†

3.9.3.4.1 Piping

[*Supports and their attachments for essential ASME Code Section III, Class 1,2, and 3 piping are designed in accordance with Subsection NF* up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF.*]† The loading combinations for the various operating conditions correspond to those used for design of the supported pipe. The component loading combinations are discussed in Subsection 3.9.3.1. The stress limits are per ASME III Code Section, Subsection NF and Appendix F. Supports are generally designed either by load rating method per Code Paragraph NF-3280 or by the stress limits for linear supports per paragraph NF-3143. The critical buckling loads for the Class 1 piping supports subjected to faulted loads, which are more severe than normal, upset and emergency loads, are determined by using the methods discussed in Appendices F and XVII of the Code. To avoid buckling in the piping supports, the allowable loads are limited to two-thirds of the determined critical buckling loads.

[*Maximum calculated static and dynamic deflections at support locations are checked to confirm that the support has not rotated beyond the vendor's recommended cone of action or the recommended arc of loading.*]†

[*Supports for ASME Code Section III instrumentation lines are designed and analyzed in accordance with ASME Code Section III, Subsection NF.*]†

The design of all supports for non-nuclear piping satisfies the requirements of ANSI/ASME B31.1, Power Piping Code, Paragraphs 120 and 121.

For the major active valves identified in Subsection 3.9.3.2.4, the valve operators are not used as attachment points for piping supports.

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* [*Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83.*]†

† See Subsection 3.9.1.7.

The design criteria and dynamic testing requirements for the ASME III Subsection NF piping supports are as follows:

(1) **Piping Supports—**All piping supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed.[*All piping supports are designed in accordance with the rules of Subsection NF of the ASME Code up to the building structure interface as defined by the jurisdictional boundaries in Subsection NF.*]*

(2) **Spring Hangers—**The operating load on spring hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the operating load at both their hot and cold load settings. Spring hangers provide a specified down travel and up travel in excess of the specified thermal movement. Deflections due to dynamic loads are checked to confirm that they do not bottom out.

(3) **Snubbers—[***Mechanical and hydraulic type snubbers will be used when required as shock arrestors for nuclear safety-related piping systems. Snubbers are designed in accordance with ASME Section III, Subsection NF, Component Standard Supports.*]* Snubbers consist of a velocity-limiting or acceleration-limiting cylinder pinned to a pipe clamp at the pipe end and pinned to a clevis attached to the building structure at the other end. Snubbers operate as structural supports during dynamic events such as an earthquake, but during normal operation act as passive devices which accommodate normal expansions and contractions without resistance. The operating loads on snubbers are the loads caused by dynamic events (e.g., seismic, RBV due to LOCA and SRV discharge, discharge through a relief valve line or valve closure) during various operating conditions. Snubbers restrain piping against response to the vibratory excitation and to the associated differential movement of the piping system support anchor points. The criteria for locating snubbers and ensuring adequate load capacity, the structural and mechanical performance parameters used for snubbers and the installation and inspection considerations for the snubbers are as follows:

(a) Required Load Capacity and Snubber Location

[*The loads calculated in the piping dynamic analysis, described in Subsection 3.7.3.8, cannot exceed the snubber load capacity for design, normal, upset, emergency and faulted conditions.*

For hydraulic snubbers with load ratings greater than 222.4 kN, dynamic cyclic load tests will be conducted to verify the performance of the control valve. These hydraulic snubbers will be subjected to dynamic cyclic load tests at loads greater than or equal to one-half the calculated safe shutdown earthquake load on the snubbers.]†

* See Subsection 3.9.1.7.

† See Subsection 3.9.1.7.

Snubbers are generally used in situations where dynamic support is required because thermal growth of the piping prohibits the use of rigid supports. The snubber locations and support directions are first decided by estimation so that the stresses in the piping system will have acceptable values. The snubber locations and support directions are refined by performing the dynamic analysis of the piping and support system as described above in order that the piping stresses and support loads meet the Code requirements.

[*The pipe support design specification requires that snubbers be provided with position indicators to identify the rod position.*]* This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

(b) I*nspection, Testing, Repair and/or Replacement of Snubbers*

[The pipe support design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the testing, maintenance, and repair of the snubber. It also contains inspection points and the period of inspection.

The pipe support design specification requires that hydraulic snubbers be equipped with a fluid level indicator so that the level of fluid in the snubber can be ascertained easily.

The spring constant achieved by the snubber supplier for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are the same, then the snubber location and support direction become confirmed. If the spring constants are not in agreement, they are brought in agreement, and the system analysis is redone to confirm the snubber loads. This iteration is continued until all snubber load capacities and spring constants are reconciled.]*

(c) Snubber Design and Testing

To assure that the required structural and mechanical performance characteristics and product quality are achieved, the following requirements for design and testing are imposed by the design specification:

(i) [*The snubbers are required by the pipe support design specification to be designed in accordance with all of the rules and regulations of ASME Code Section III, Subsection NF. This design requirement includes analysis for the normal, upset, emergency, and faulted loads. These* *calculated loads are then compared against the allowable loads to make sure that the stresses are below the code allowable limit.*

(ii) The snubbers are tested to insure that they can perform as required during the seismic and other RBV events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. The following test requirements are included:

— *Snubbers are subjected to force or displacement versus time loading at frequencies within the range of significant modes of the piping system.*

— *Dynamic cyclic load tests are conducted for large bore hydraulic snubbers to determine the operational characteristics of the snubber control valve.*

— *Displacements are measured to determine the performance characteristics specified.*

— *Tests are conducted at various temperatures to ensure operability over the specified range.*

— *Peak test loads in both tension and compression are required to be equal to or higher than the rated load requirements.*

— *The snubbers are tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test.*]*

(d) Snubber Installation Requirements

[*An installation instruction manual is required by the pipe support design specification. This manual is required to contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing which contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.*]†

(e) Snubber Pre-service Examination

* See Subsection 3.9.1.7.

† See Subsection 3.9.1.7.

[*The pre-service examination plan of all snubbers covered by the Chapter 16 technical specifications will be prepared. This examination will be made after snubber installation but not more than 6 months prior to initial system pre-operational testing. The pre-service examination will verify the following:*

(i) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(ii) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

(iii) Snubbers are not seized, frozen or jammed.

(iv) Adequate swing clearance is provided to allow snubber movements.

(v) If applicable, fluid is to be at recommended level and not leaking from the snubber system.

(vi) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational tests exceeds 6 months because of unexpected situations, re-examination of Items i, iv, and v will be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements will be repaired or replaced and re-examined in accordance with the above criteria.]*

(4) **Struts—**Struts are defined as ASME Section III, Subsection NF, Component Standard Supports. They consist of rigid rods pinned to a pipe clamp or lug at the pipe and pinned to a clevis attached to the building structure or supplemental steel at the other end. Struts, including the rod, clamps, clevises, and pins are designed in accordance with ASME Code Section III, Subsection NF-3000.

Struts are passive supports, requiring little maintenance and inservice inspection, and will normally be used instead of snubbers where dynamic supports are required and the movement of the pipe due to thermal expansion and/or anchor motions is small. Struts will not be used at locations where restraint of pipe movement to thermal expansion will significantly increase the secondary piping stress ranges or equipment nozzle loads. Increases of thermal expansion loads in the pipe and nozzles will normally be restricted to less than 20%.

Because of the pinned connections at the pipe and structure, struts carry axial loads only. The design loads on struts may include those loads caused by thermal expansion, dead weight, and the inertia and anchor motion effects of all dynamic loads. As in the case of other supports, the forces on struts (obtained from an analysis) do not exceed the design loads for various operating conditions.

(5) **Frame Type (Linear) Pipe Supports**—Pipe Supports: Frame type supports are linear supports as defined as ASME Section III, Subsection NF, Component Standard Supports. They consist of frames constructed of structural steel elements that are not attached to the pipes. They act as guides to allow axial and rotational movement of the pipe, but also as rigid restraints to lateral movement in either one or two directions. [*Frame type pipe supports are designed in accordance with ASME Code Section III, Subsection NF-3000.*]*

Frame type supports are passive supports, requiring little maintenance and inservice inspection, and will normally be used instead of struts when they are more economical or where environmental conditions are not suitable for the ball bushings at the pinned connections of struts. Similar to struts, frame type supports will not be used at locations where restraint of pipe movement to thermal expansion will significantly increase the secondary piping stress ranges or equipment nozzle loads. Increases of thermal expansion loads in the pipe and nozzles will normally be restricted to less than 20%.

The design loads on frame type pipe supports include those loads caused by thermal expansion, dead weight, and the inertia and anchor motion effects of all dynamic loads. As in the case of other supports, the forces on frame type supports are obtained from an analysis and assured not to exceed the design loads for various operating conditions.

(6) **Special Engineered Pipe Supports**—In an effort to minimize the use and application of snubbers, there may be instances where special engineered pipe supports can be used where either struts or frame-type supports cannot be applied. Examples of special engineered supports are Energy Absorbers and Limit Stops.

(a) **Energy Absorbers** are linear energy absorbing support parts designed to dissipate energy associated with dynamic pipe movements by yielding. [*When energy absorbers are used they will be designed to meet the requirements of ASME Section III, Code Case N-420, Linear Energy Supports for Subsection NF, Classes 1, 2, and 3 Construction, Section III, Division 1.The information required by Regulatory Guide 1.84 will be provided to the regulatory agency.*]† The restrictions on location and application of struts and frame-type supports, discussed in (4) and (5) above, are also applicable to energy absorbers since energy absorbers allow thermal movement of the pipe only in its design directions.

* See Subsection 3.9.1.7.

† See Subsection 3.9.1.7.

(b) **Limit Stops** are passive seismic pipe support devices consisting of limit stops with gaps sized to allow for thermal expansion while preventing large seismic displacements.[*Limit stops are linear supports as defined as ASME Section III, Subsection NF, and are designed in accordance with ASME Code Section III, Subsection NF-3000.*]* They consist of either special component standard supports with a configuration, size and end to end dimensions similar to snubbers, or box frames constructed of structural steel elements that are not attached to the pipe. The box frames allow free movement in the axial direction but limits large displacement in the lateral direction.

If these special devices are used, the modeling and analytical methodology will be in accordance with the methodology accepted by the regulatory agency at the time of the certification or at the time of application, per the direction of the COL applicant.

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

The ABWR RPV support skirt is designed as an ASME Code Class 1 component per the requirements of ASME Code Section III, Subsection NF*. The loading conditions and stress criteria are given in Tables 3.9-1 and 3.9-2, and the calculated stresses meet the Code allowable stresses in the critical support areas for various plant operating conditions. The stress level margins assure the adequacy of the RPV support skirt. An analysis for buckling shows that the support skirt complies with Subparagraph F-1332.5 of ASME III, Appendix F, and the loads do not exceed two-thirds of the critical buckling strength of the skirt. The permissible skirt loads at any elevation, when simultaneously applied, are limited by the following interaction equation:

$$\left(\frac{P}{P_{crit}}\right) + \left(\frac{q}{q_{crit}}\right) + \left(\frac{\tau}{\tau_{crit}}\right) < \left(\frac{1}{S.F.}\right) \quad (3.9\text{-}1)$$

where:

q	=	Longitudinal load
P	=	External pressure
τ	=	Transverse shear stress

* Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet “Standard for Steel Support Design”, analyses will be performed in accordance with torsional analysis methods such as: “Torsional Analysis of Steel Members, USS Steel Manual”, Publication T114-2/83.

S.F. = Safety factor
= 3.0 for design, testing, service levels A & B
= 2.0 for Service Level C
= 1.5 for Service Level D.

3.9.3.4.3 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as a Safety Class 1 linear type component support in accordance with the requirements of ASME Code Section III, Subsection NF. The stabilizer provides a reaction point near the upper end of the RPV to resist horizontal loads due to effects such as earthquake, pipe rupture and RBV. The design loading conditions and stress criteria listed in Tables 3.9-1 and 3.9-2 show that calculated stresses meet the Code allowable stresses in the critical support areas for various plant operating conditions.

3.9.3.4.4 Floor-Mounted Major Equipment (Pumps, Heat Exchangers, and RCIC Turbine)

Since the major active valves are supported by piping and not tied to building structures, valve “supports” do not exist (Subsection 3.9.3.4.1).

The HPCF, RHR, RCIC, SLC, FPCCU, SPCU, and CUW pumps; RCW, RHR, CUW, and FPCCU heat exchangers; and RCIC turbine are all analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases, the load stresses in the critical support areas are within ASME Code allowables.

Seismic Category I active pump supports are qualified for dynamic (seismic and other RBV) loads by testing when the pump supports together with the pump meet the following test conditions:

(1) Simulate actual mounting conditions.

(2) Simulate all static and dynamic loadings on the pump.

(3) Monitor pump operability during testing.

(4) The normal operation of the pump during and after the test indicates that the supports are adequate (any deflection or deformation of the pump supports which precludes the operability of the pump is not accepted).

(5) Supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Dynamic qualification of component supports by analysis is generally accomplished as follows:

(1) Stresses at all support elements and parts such as pump holddown and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to

be within the allowable limits as specified in ASME Code Section III, Subsection NF.

(2) For normal and upset conditions, the deflections and deformations of the supports are assured to be within the elastic limits, and to not exceed the values permitted by the designer based on design verification tests. This ensures the operability of the pump.

(3) For emergency and faulted plant conditions, the deformations do not exceed the values permitted by the designer to ensure the operability of the pump. Elastic/plastic analyses are performed if the deflections are above the elastic limits.

3.9.3.5 Other ASME III Component Supports

The ASME III component supports and their attachments (other than those discussed in preceding subsection) are designed in accordance with Subsection NF of ASME Code Section III* up to the interface with the building structure. The building structure component supports are designed in accordance with the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings. The loading combinations for the various operating conditions correspond to those used to design the supported component. The component loading combinations are discussed in Subsection 3.9.3.1. Active component supports are discussed in Subsection 3.9.3.2. The stress limits are per ASME Code Section III, Subsection NF and Appendix F. The supports are evaluated for buckling in accordance with ASME Code Section III.

3.9.4 Control Rod Drive (CRD)

The Control Rod Drive (CRD) System is equipped with electro-hydraulic fine motion control rod drives (FMCRD), hydraulic control units (HCU), the condensate supply system, and power for FMCRD motors. The system extends inside the RPV to the coupling interface with the control rod blades.

3.9.4.1 Descriptive Information on CRD System

Descriptive information on the FMCRDs, as well as the entire CRD System, is contained in Section 4.6.

* Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet “Standard for Steel Support Design”, analyses will be performed in accordance with torsional analysis methods such as: “Torsional Analysis of Steel Members, USS Steel Manual”, Publication T114-2/83.

3.9.4.2 Applicable CRD System Design Specification

The CRD System is designed to meet the functional design criteria outlined in Section 4.6 and consists of the following:

(1) Fine motion control rod drive

(2) Hydraulic control unit

(3) Hydraulic power supply (pumps)

(4) Electric power supply (for FMCRD motors)

(5) Interconnecting piping

(6) Flow and pressure valves

(7) Instrumentation and electrical controls

Those components of the CRD System forming part of the primary pressure boundary are designed according to ASME Code Section III Class 1 requirements.

The quality group classification of the components of the CRD System is outlined in Table 3.2-1 and they are designed to the codes and standards, per Table 3.2-2, in accordance with their individual quality groups.

Pertinent aspects of the design and qualification of the CRD System components are discussed in the following locations: transients in Subsection 3.9.1.1, faulted conditions in Subsection 3.9.1.4, seismic testing in Subsection 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The ASME III Code components of the CRD System are evaluated analytically and the design loading conditions, and stress criteria are as given in Tables 3.9-1 and 3.9-2. For the non-Code components, the ASME III Code requirements are used as guidelines and experimental testing is used to determine the CRD performance under all possible conditions as described in Subsection 3.9.4.4.

3.9.4.4 CRD Performance Assurance Program

The following CRD tests are described in Subsection 4.6.1.

(1) Development tests

(2) Factory quality control tests

(3) Functional tests

(4) Operational tests

(5) Acceptance tests

(6) Surveillance tests

3.9.5 Reactor Pressure Vessel Internals

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel (RPV) internals, including core support structures.

Certain reactor internals support the core, flood the core during a loss-of-coolant accident (LOCA) and support safety-related instrumentation. Other RPV internals direct coolant flow, separate steam, hold material surveillance specimens, and support instrumentation utilized for plant operation.

3.9.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and incore nuclear instrumentation) are:

(1) Core Support Structures

Shroud

Shroud support (including the internal pump deck)

Core plate (and core plate hardware)

Top guide

Fuel supports (orificed fuel supports and peripheral fuel supports)

Control rod guide tubes

Non-pressure boundary portion of control rod drive housings

(2) Reactor Internals

Shroud head* and steam separators assembly*

Steam dryers assembly*

Feedwater spargers

* These are non-nuclear safety category components as defined in Subsection 3.2.5.1. In Subsection 3.9.5, such components are called non-safety class components, and the safety-related internals (Safety Class 2 or 3) are called safety class components.

RHR/ECCS low pressure flooder spargers

ECCS high pressure core flooder spargers and piping

Core differential pressure lines

RPV vent and head spray assembly

Internal pump differential pressure lines*

Incore guide tubes and stabilizers

Non-pressure boundary portion of in-core housings

Surveillance sample holders*

A general assembly drawing of the important reactor components is shown in Figures 5.3-2a and 5.3-2b and Table 5.3-2.

The floodable inner volume of the reactor pressure vessel can be seen in Figure 3.9-2. It is the volume up to the level of the core flooder sparger.

The design arrangement of the reactor internals, such as the shroud, steam separators and guide tubes, is such that one end is unrestricted and thus free to expand.

The ECCS core flooder couplings incorporate vertically-oriented slip-fit joints to allow free thermal expansion.

3.9.5.1.1 Core Support Structures

The core support structures consist of those items listed in Subsection 3.9.5.1(1) and are Safety Class 3 as defined in Section 3.2. These structures form partitions within the reactor vessel to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies. Figures 3.9-2 and 3.9-3 show the reactor vessel internal flow paths.

3.9.5.1.1.1 Shroud

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus. The volume enclosed by this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide plate below. The central portion of the shroud surrounds the active fuel and forms the longest section of the assembly.

This section is bounded at the top by the top guide plate and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the RPV shroud support. The shroud provides the horizontal support for the core by supporting the core plate and top guide.

3.9.5.1.1.2 Shroud Support

The RPV shroud support is designed to support the shroud, and includes the internal pump deck that locates and supports the pumps. The pump discharge diffusers penetrate the deck to introduce the coolant to the inlet plenum below the core. The RPV shroud support is a horizontal structure welded to the vessel wall to provide support to the shroud, pump diffusers, and core and pump deck differential pressure lines. The structure is a ring plate welded to the vessel wall and to a vertical cylinder supported by vertical stilt legs from the bottom head.

3.9.5.1.1.3 Core Plate

The core plate consists of a circular stainless steel plate with round openings and is stiffened with a rim and beam structure. The core plate provides lateral support and guidance for the control rod guide tubes, incore flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core plate.

The entire assembly is bolted to a support ledge in the lower portion of the shroud.

3.9.5.1.1.4 Top Guide

The top guide consists of a circular plate with square openings for fuel with a cylindrical side forming an upper shroud extension and having a top flange for attaching the shroud head. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Holes are provided in the bottom of the support intersections to anchor the incore flux monitors and startup neutron sources. The top guide is mechanically attached to the top of the shroud.

3.9.5.1.1.5 Fuel Supports

The fuel supports (Figure 3.9-4) are of two basic types: peripheral supports and orificed fuel supports. The peripheral fuel supports are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support supports one fuel assembly and has an orifice designed to assure proper coolant flow to the peripheral fuel assembly. Each orificed fuel support holds four fuel assemblies vertically upward and horizontally and has four orifices to provide proper coolant flow distribution to each rod-controlled fuel assembly. The orificed fuel supports rest on the top of the CRGTs, which are supported laterally by the core plate. The control rods pass through cruciform openings in the center of the orificed fuel support. This locates the four fuel assemblies surrounding a control rod. A control rod and the four adjacent fuel assemblies represent a core cell (Section 4.4).

3.9.5.1.1.6 Control Rod Guide Tubes

The control rod guide tubes (CRGTs) located inside the vessel extend from the top of the CRD housings up through holes in the core plate. Each guide tube is designed as the guide for the lower end of a control rod and as the support for an orificed fuel support. This locates the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD housing, which, in turn, transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. The control rod guide tubes (CRGTs) also contain holes, near the top of the control rod guide tube and below the core plate, for coolant flow to the orificed fuel supports.

3.9.5.1.2 Reactor Internals

The reactor internals consist of those items listed in Subsection 3.9.5.1(2). These components direct and control coolant flow through the core or support safety-related and non-safety-related function.

3.9.5.1.2.1 Shroud Head and Steam Separators Assembly

The shroud head and standpipes/steam separators are non-safety class internal components. The assembly is discussed here to describe the coolant flow paths in the reactor pressure vessel. The shroud head and steam separators assembly includes the upper flanges and bolts, and forms the top of the core discharge mixture plenum together with the separators and their connecting standpipes. The discharge plenum provides a mixing chamber for the steam/water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are supported on and attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam/water mixture rising through the standpipe passes vanes that impart a spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus. The assembly is removable from the reactor pressure vessel as a single unit on a routine basis.

3.9.5.1.2.2 Reactor Internal Pump (RIP)/Diffusers

The pump assembly (impeller, diffuser and pump shaft) is a non-safety class component and is discussed here to describe coolant flow paths (Figure 3.9-3) in the vessel. The pump provides a means for forced circulation of the reactor coolant through the core, including the mixing of feedwater and annulus water from the steam separators and distribution of this fluid to the vessel lower plenum and up through the lower grid to the core.

The pump assemblies are mounted vertically into pump nozzles arranged in an equally-spaced ring pattern on the bottom head of the RPV and are located inside the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the pump assemblies is covered in detail in Subsection 5.4.1. Each pump consists of three major hardware sections: an internal pump (IP) section; a recirculation motor (RM) section; and a stretch tube section (Figure 5.4-1).

The IP section of the RIP is located inside the RPV, in an opening through the RPV pump deck—the latter being the horizontal ring-plate enclosing the bottom of the downcomer annulus and thus separating the lower pressure annulus region from the higher-pressure lower plenum region. The IP, in turn, is comprised of a vertical axis single-stage, mixed-flow impeller driven from underneath by a pump shaft, with the impeller being encircled by a diffuser assembled into the pump deck opening.

The RM section of the RIP is located underneath, and at the periphery of, the RPV bottom head inside a pressure-retaining housing termed the motor casing. The motor casing itself is not part of the RM, but is instead a part of and welded into an RPV nozzle (pump nozzle). The motor casing thus comprises part of the RCPB and is a Safety Class 1 component.

The principal element of the stretch tube section is a thin-walled tube configured as a hollow bolt fitting around the pump shaft and within the pump nozzle. It has an external lip (bolt head) at its upper end and an external threaded section at this lower end. A stretch tube provides tight clamping of the IP diffuser to the gasketed, internal-mount end of the RPV pump nozzle, for all extremes of thermal transients and pump operating conditions.

3.9.5.1.2.3 Steam Dryer Assembly

The steam dryer assembly is a non-safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryer removes moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus.

The steam dryer assembly consists of multiple banks of dryer units mounted on a common structure which is removable from the RPV as an integral unit. The assembly includes the dryer banks, dryer supply and discharge ducting, drain collecting trough, drain piping, and a skirt which forms a water seal extending below the separator reference zero elevation. Upward and radial movement of the dryer assembly under the action of blowdown and seismic loads are limited by reactor vessel internal stops which are arranged to permit differential expansion growth of the dryer assembly with respect to the reactor pressure vessel. The assembly is arranged for removal from the vessel as an integral unit on a routine basis.

3.9.5.1.2.4 Feedwater Spargers

These are Safety Class 2 components. They are discussed here to describe coolant flow paths in the vessel and their safety function. Each of two feedwater lines is connected to three spargers via three RPV nozzles. One line is utilized by the RCIC System, the other by the RHR shutdown cooling system. During the ECCS mode, the two groups of spargers support diverse type of flooding of the vessel. The RCIC System side supports high pressure flooding and the RHR System side supports low pressure flooding, as required during the ECCS operation.

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger in two halves is fitted to each feedwater nozzle via a tee and is shaped to conform to the curve of the vessel wall. The sparger tee inlet is connected to the RPV nozzle safe end by a double thermal sleeve arrangement, with all connections made by full penetration welds. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense steam in the region above the downcomer annulus and to subcool water flowing to the recirculation internal pumps.

3.9.5.1.2.5 RHR/ECCS Low Pressure Flooder Spargers

These are Safety Class 2 components. The design features of these two spargers of the RHR shutdown cooling system are similar to those of the six feedwater spargers, three of which belonging to one feedwater line support additionally the same RHR (and ECCS) function. During the ECCS mode, these spargers support low pressure flooding of the vessel. The feedwater spargers are described in Subsection 3.9.5.1.2.4.

Two lines of the RHR shutdown cooling system enter the reactor vessel through the two diagonally opposite nozzles and connect to the spargers. The sparger tee inlet is connected to the RPV nozzle safe end by a thermal sleeve arrangement with all connections made by full penetration welds.

3.9.5.1.2.6 ECCS High Pressure Core Flooder Spargers and Piping

The core flooder spargers and piping are Safety Class 2. The spargers and piping are the means for directing high pressure ECCS flow to the upper end of the core during accident conditions.

Each of two High Pressure Core Flooder (HPCF) System lines enters the reactor vessel through a diagonally opposite nozzle in the same manner as an RHR low pressure flooder line, except that the curved sparger including the connecting tee is routed around the inside of and is supported by the cylindrical portion of the top guide. A flexible coupling is interposed between the sparger tee inlet and the sleeved inlet connector inside the nozzle. The two spargers are supported so as to accommodate thermal expansion.

3.9.5.1.2.7 RPV Vent and Head Spray Assembly

This is designed as a Safety Class 1 component. However, only the nozzle portion of the assembly is a reactor coolant pressure boundary, and the assembly function is not a safety-related operation. The reactor water cleanup return flow to the reactor vessel, via feedwater lines, can be diverted partly to a spray nozzle in the reactor head in preparation for refueling cooldown. The spray maintains saturated conditions in the reactor vessel head volume by condensing steam being generated by the hot reactor vessel walls and internals. The head spray subsystem is designed to rapidly cool down the reactor vessel head flange region for refueling and to allow installation of steamline plugs before vessel floodup for refueling.

The head vent side of the assembly passes steam and noncondensable gases from the reactor head to the steamlines during startup and operation. During shutdown and filling for hydrotesting, steam and noncondensable gases may be vented to the drywell equipment sump while the connection to the steamline is blocked. When draining the vessel during shutdown, air enters the vessel through the vent.

3.9.5.1.2.8 Core and Internal Pump Differential Pressure Lines

These lines comprise the core flow measurement subsystem of the Recirculation Flow Control System (RFCS) and provide two methods of measuring the ABWR core flow rates. The core DP lines (Safety Class 3) and internal pump DP lines (non-safety class) enter the reactor vessel separately through reactor bottom head penetrations. Four pairs of the core DP lines enter the head in four quadrants through four penetrations and terminate immediately above and below the core plate to sense the pressure in the region outside the bottom of the fuel assemblies and below the core plate during normal operation.

Similarly, four pairs of the internal pump DP lines terminate above and below the pump deck and are used to sense the pressure across the pump during normal pump operation. Each pair is routed concentrically through a penetration and upward along a shroud support leg in the lower plenum.

3.9.5.1.2.9 Incore Guide Tubes and Stabilizers

These are Safety Class 3 components. The guide tubes protect the incore instrumentation from flow of water in the bottom head plenum and provide a means of positioning fixed detectors in the core, as well as a path for insertion and withdrawal of the calibration monitors (ATIP, Automated Traversing Incore Probe Subsystem). The incore flux monitor guide tubes extend from the top of the incore flux monitor housing to the top of the core plate. (The power range detectors for the power range monitoring units and the dry tubes for the startup range neutron monitoring and average power range monitoring (SRNM) detectors are inserted through the guide tubes). The local power range monitor (PRNM) detector assemblies and the dry tubes for the startup range monitoring (SRNM) assemblies are inserted through the guide tube.

Two levels of stainless steel stabilizer latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The stabilizers are connected to the shroud and shroud support. The bolts are tack-welded after assembly to prevent loosening during reactor operation.

3.9.5.1.2.10 Surveillance Sample Holders

This is a non-safety class component. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The holders have brackets that are attached to the inside of the reactor vessel wall and are located in the active core beltline region. The radial and azimuthal positions are chosen to expose the specimens to the same environment and the maximum neutron fluxes experienced by the reactor vessel wall.

3.9.5.2 Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design bases (Subsection 3.9.5.3.1) must be satisfied by core support structures and safety-related internal components reveals four significant faulted events:

(1) **Feedwater Line Break—**A break in a feedwater line between the reactor vessel and the primary containment penetration; (the accident results in significant annulus pressurization and reactor building vibration (RBV) due to suppression pool dynamics).

(2) **Steamline Break Accident—**A break in one main steamline between the reactor vessel nozzle and the main steam isolation valve (the accident results in significant pressure differentials across some of the structures within the reactor and reactor building vibration due to suppression pool dynamics).

(3) **Earthquake—**subjects the core support structures and reactor internals to significant forces as a result of ground motion and consequent RBV.

(4) **Safety/relief valve discharge—**RBV due to suppression pool dynamics and structural feedback.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents show that the loads affecting core support structures and other safety-related reactor internals are less severe than those affected by the four postulated events.

The faulted conditions for the reactor pressure vessel internals are discussed in Subsection 3.9.1.4. Loading combinations and analyses for safety-related reactor internals, including core support structures, are discussed in Subsections 3.9.3.1, 3.9.5.3.5, and 3.9.5.3.6.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the reactor vessel following the main steamline break between the vessel nozzle and main steam isolation valve. The analytical model of the vessel consists of nine nodes which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates and pressures in the various regions of the reactor. Figure 3.9-5 shows the nine reactor nodes. The computer code used is the GE Short-Term Thermal-Hydraulic Model described in Reference 3.9-4. This model has been approved for use in ECCS conformance evaluation under 10CFR50 Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are therefore not described in Reference 3.9-4. These additional features are as follows:

(1) The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steamline.

(2) The flow path between the bypass region and the shroud head is more accurately modeled, since the fuel assembly pressure differential is influenced by flashing in the guide tubes and bypass region for a steamline break. In the ECCS analysis, the momentum equation is solved in this flow path but its irreversible loss coefficient is conservatively set at an arbitrary low value.

(3) The enthalpies in the guide tubes and the bypass are calculated separately since the fuel assembly pressure differential is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3 Feedwater Line and Main Steamline Break

3.9.5.2.3.1 Accident Definition

Both a feedwater line break (the largest liquid line break) and a Main Steamline (MSL) break (the largest steamline break) upstream of the MSIV are considered in determining the design basis accident for the safety-related reactor internals, including the core support structures.

The feedwater line break is the same as the design basis LOCA described in Subsection 6.2.1.1.3.3.1. A sudden, complete circumferential break is assumed to occur in one feedwater line. The pressure differentials on the reactor internals and core support structures are in all cases lower than those for the MSL break.

The analysis for the MSL break assumes a sudden, complete circumferential break of one main steamline at the reactor vessel nozzle, downstream of the limiting flow area (Subsection 6.2.1.1.3.3.2).

The steamline break accident produces significantly higher pressure differential across the reactor internal structures than does the feedwater line break. This results from the higher reactor depressurization rate associated with the steamline break. Therefore, the steamline break is the design basis accident for internal pressure differentials.

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant, the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. On the other hand, the core power affects both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core and, consequently, the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the RPV and, thus, the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

To ensure that calculated pressure differences bound those which could be expected if a steamline break should occur, an analysis is conducted at a low power high-recirculation flow condition in addition to the standard safety analysis condition at high power, rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (i.e., the drive flow necessary to achieve rated core flow at rated power).

This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, because the reactor will generally operate at or near full power; second, because high core flow is neither required nor desirable at such a reduced power condition.

Table 3.9-3 summarizes the maximum pressure differentials. Case 1 is the safety analysis condition; Case 2 is the low power high-flow condition.

3.9.5.2.4 Seismic and Other Reactor Building Vibration Events

The loads due to earthquake and other reactor building vibration (RBV) acting on the structure within the reactor vessel are based on a dynamic analysis described in Sections 3.7, 3.8, and Subsection 3.9.2.5. Dynamic analysis is performed by coupling the lumped-mass model of the reactor vessel and internals with the building model to determine the system natural frequencies and mode shapes. The relative displacement, acceleration, and load response is then determined by either the time-history method or the response-spectrum method.

3.9.5.3 Design Bases

3.9.5.3.1 Safety Design Bases

The reactor internals, including core support structures, shall meet the following safety design bases:

(1) The reactor vessel nozzles and internals shall be so arranged as to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

(2) Deformation of internals shall be limited to assure that the control rods and core standby cooling systems can perform their safety-related functions.

(3) Mechanical design of applicable structures shall assure that safety design bases (1) and (2) are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.3.2 Power Generation Design Bases

The reactor internals, including core support structures, shall be designed to the following power generation design bases:

(1) The internals shall provide the proper coolant distribution during all anticipated normal operating conditions to full power operation of the core without fuel damage.

(2) The internals shall be arranged to facilitate refueling operations.

(3) The internals shall be designed to facilitate inspection.

3.9.5.3.3 Design Loading Categories

The basis for determining faulted dynamic event loads on the reactor internals is shown in Sections 3.7, 3.8 and Subsections 3.9.2.5, 3.9.5.2.3 and 3.9.5.2.4. Table 3.9-2 shows the load combinations used in the analysis.

Core support structures are Seismic Category I and ASME Code Class CS structures, which meet the stress limits of the ASME Code Section III, Subsection NG. For these components and the safety-related internals, Level A, B, C, and D service limits are applied to the normal, upset, emergency, and faulted loading conditions, respectively, as defined in the design specification. Stress intensity and other design limits are discussed in Subsections 3.9.5.3.5 and 3.9.5.3.6

3.9.5.3.4 Response of Internals Due to Steamline Break Accident

As described in Subsection 3.9.5.2.3.2, the maximum pressure loads acting on the reactor internal components result from steamline break upstream of the main steam isolation valve and, on some components, the loads are greatest with operation at the minimum power associated with the maximum core flow (Table 3.9-3, Case 2). This has been substantiated by the analytical comparison of liquid versus steamline breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that, although possible, it is not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be as listed under Case 1 in Table 3.9-3.

3.9.5.3.5 Stress and Fatigue Limits for Core Support Structures

The design and construction of the core support structures are in accordance with ASME Code Section III, Subsection NG.

3.9.5.3.6 Stress, Deformation, and Fatigue Limits for Safety Class and Other Reactor Internals (Except Core Support Structures)

For safety class reactor internals, the stress deformation and fatigue criteria listed in Tables 3.9-4 through 3.9-7 are based on the criteria established in applicable codes and standards for similar equipment, by manufacturer's standards, or by empirical methods based on field experience and testing. For the quantity SF_{min} (minimum safety factor) appearing in those tables, the following values are used:

Service Level	Service Condition	SF_{min}
A	Normal	2.25
B	Upset	2.25
C	Emergency	1.5
D	Faulted	1.125

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined to determine if adequate clearances exist during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in Subsection 3.9.2.5.

The design criteria, loading conditions, and analyses that provide the basis for the design of the safety class reactor internals other than the core support structures meet the guidelines of NG-3000 and are constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

The design requirements for equipment classified as non-safety (other) class internals (e.g., steam dryers and shroud heads) are specified with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where Code design requirements are not applicable, accepted industry or engineering practices are used.

3.9.6 Testing of Pumps and Valves

Inservice testing of safety-related pumps and valves will be performed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1, 6, and 10. Table 3.9-8 lists the inservice testing parameters and frequencies for the safety-related pumps and valves. The reason for each code defined testing exception or justification for each code exemption request is noted in the description of the affected pump or valve. Valves having a containment isolation function are also noted in the listing. Inservice inspection is discussed in Subsection 5.2.4 and Section 6.6.

Details of the inservice testing program, including test schedules and frequencies, will be reported in the inservice inspection and testing plan to be provided by the applicant referencing the ABWR design. The plan will integrate the applicable test requirements for safety-related pumps and valves, including those listed in the technical specifications, Chapter 16, and the containment isolation system, Subsection 6.2.4. For example, the periodic leak testing of the reactor coolant pressure isolation valves (See Appendix 3M for design changes made to prevent intersystem LOCAs) in Table 3.9-9 will be performed in accordance with Chapter 16 Surveillance Requirement SR 3.6.1.5.10. This plan will include baseline pre-service testing to support the periodic inservice testing of the components. Depending on the test results, the plan will provide a commitment to disassemble and inspect the safety-related pumps and valves when limits of the OM Code are exceeded, as described in the following paragraphs. The primary elements of this plan, including the requirements of Generic Letter 89-10 for motor operated valves, are delineated in the subsections to follow. (See Subsection 3.9.7.3 for COL license information requirements.)

3.9.6.1 Testing of Safety-Related Pumps

For each pump, the design basis and required operating conditions (including tests) under which the pump will be required to function will be established. These designs (design basis and required operating) conditions include flow rate and corresponding head for each system mode of pump operation and the required operating time for each mode, acceptable bearing vibration levels, seismic/dynamic loads, fluid temperature, ambient temperature, and pump motor minimum voltage.

The COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. For each size, type, and model the COL applicant will perform testing encompassing design conditions that demonstrate acceptable flow rate and corresponding head, bearing vibration levels, and pump internals wear rates for the operating time specified for each system mode of pump operation. From these tests the COL applicant will also develop baseline (reference) hydraulic and vibration data for evaluating the acceptability of the pump after installation. The COL applicant will ensure that the pump specified for each application is not susceptible to inadequate minimum flow rate and inadequate thrust bearing capacity. With respect to minimum flow pump operation, the sizing of each minimum recirculation flow path is evaluated to assure that its use under all analyzed conditions will not result in degradation of the pump. The flow rate through minimum recirculation flow paths can also be periodically measured to verify that flow is in accordance with the design specification.

The ABWR safety-related pumps and piping configurations accommodate in-service testing at a flow rate at least as large as the maximum design flow for the pump application. The safety-related pumps are provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. These pumps can be disassembled for evaluation when Part 6 testing results in a deviation which falls within the “required action range.” The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related pumps, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3(1) for COL license information requirements.)

It is demonstrated that the ECCS pumps (including mechanical seal) can perform specified functions under all design basis conditions including post-LOCA debris loading conditions. Demonstration of acceptable performance for as-built ECCS pumps is validated under QME-1 2007, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants as endorsed by RG 1.100 Revision 3.

3.9.6.2 Testing of Safety-Related Valves

3.9.6.2.1 Check Valves

(1) Design and Qualification

For each check valve with an active safety-related function, the design basis and required operating conditions (including testing) under which the check valve will be required to perform will be established.

The COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model the COL applicant will ensure the design adequacy of the check valve under design (design basis and required operating) conditions. These design conditions include all the required system operating cycles to be experienced by the valve (numbers of each type of cycle and duration of each type cycle), environmental conditions under which the valve will be required to function, severe transient loadings expected during the life of the valve such as waterhammer or pipe break, life-time expectation between major refurbishments, sealing and leakage requirements, corrosion requirements, operating medium with flow and velocity definition, operating medium temperature and gradients, maintenance requirements, vibratory loading, planned testing and methods, test frequency and periods of idle operation. The design conditions may include other requirements as identified during detailed design of the plant systems. This testing of each size, type and model shall include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. The COL applicant will ensure proper check valve application including selection of the valve size and type based on the system flow conditions, installed location of the valve with respect to sources of turbulence, and correct orientation of the valve in the piping (i.e., vertical vs horizontal) as recommended or required by the manufacturer. The COL applicant will ensure that valve design features, material, and surface finish will accommodate non-intrusive diagnostic testing methods available in the industry or as specified. The COL applicant will also ensure that flow through the valve is determinable from installed instrumentation and that the valve disk positions are determinable without disassembly such as by use of non-intrusive diagnostic methods. Valve internal parts are designed with self-aligning features for purpose of assured correct installation. The COL applicant will compare the maximum loading on the check valve under design basis and the required operating conditions to the allowable structural capability limits for the individual parts of the check valve. The qualification acceptance criteria noted above will include baseline data developed during qualification testing and will be used for verifying the acceptability of the check valves after installation.

(2) Pre Operational Testing

The COL applicant will test each check valve in the open and/or close direction, as required by the safety function, under all normal operating system conditions. To the extent practical, testing of the valves as described in this section will be performed under fluid temperature conditions that would exist during a cold shutdown as well as under fluid temperature conditions that would be experienced by the valve during other modes of plant operation. The testing will identify the flow needed to open the valve to the full-open position. The testing will include the effects of rapid pump starts and stops as required by expected system operating conditions. The testing will include any other reverse flow conditions that may be required by expected system operating conditions. The COL applicant will examine the disk movement during valve testing and verify the leak-tightness of valve when fully closed. By using methods such as non-intrusive diagnostic equipment, the COL applicant will examine the open valve disk stability under the flow conditions during normal and other required system operating conditions.

The parameters and acceptance criteria for demonstrating that the above functional performance requirements have been met are as follows:

(a) During all test modes that simulate expected system operating conditions, the valve disk fully opens or fully closes as expected based on the direction of the differential pressure across the valve.

(b) Leak-tightness of valve when fully closed is within established limits, as applicable.

(c) Valve disk positions are determinable without disassembly.

(d) Valve testing must verify free disk movement whenever moving to and from the seat.

(e) The disk is stable in the open position under normal and other required system operating fluid flow conditions.

(f) The valve is correctly sized for the flow conditions specified, i.e., the disk is in full open position at normal full flow operating condition.

(g) Valve design features, material, and surfaces accommodate non-intrusive diagnostic testing methods available in the industry or as specified.

(h) Piping system design features accommodate all the applicable check valve testing requirements as described in Table 3.9-8.

All ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design conditions. Inservice testing will incorporate the use of advance non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves. The Part 10 tests will be performed, and check valves that fail to exhibit the required performance can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related check valves, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience (see Subsection 3.9.7.3 for COL license information requirements).

3.9.6.2.2 Motor-Operated Valves

For each motor-operated valve assembly (MOV) with an active safety related function, the design basis and required operating conditions (including testing) under which the MOV will be required to perform are established for the development and implementation of the design, qualification and preoperational testing.

[*Table 5 of DCD/Introduction identifies the commitments of design, qualification, and preoperational testing for MOVs, which, if changed, requires NRC Staff review and approval prior to implementation. The applicable portions of the Tier 2 sections and tables, identified on Table 5 of DCD/Introduction for this restriction, are italicized on the sections and tables themselves.*]*

(1) [*Design and Qualifications*

The COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model the COL applicant will determine the torque and thrust (as applicable to the type of MOV) requirements to operate the MOV and will ensure the adequacy of the torque and thrust that the motor-operator can deliver under

* See Section 3.5 of DCD/Introduction.

design (design basis and required operating) conditions. The COL applicant will also test each size, type, and model under a range of differential pressure and flow conditions up to the design conditions. These design conditions include fluid flow, differential pressure (including pipe break), system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and maximum stroke time requirements. This testing of each size, type and model shall include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. From this testing the COL applicant will demonstrate that the results of testing under in situ or installed conditions can be used to ensure the capability of the MOV to operate under design conditions. The COL applicant will ensure that the structural capability limits of the individual parts of the MOV will not be exceeded under design conditions. The COL applicant will ensure that the valve specified for each application is not susceptible to pressure locking and thermal binding.]*

(2) [*Pre-operational Testing*

The COL applicant will test each MOV in the open and close directions under static and maximum achievable conditions using diagnostic equipment that measures torque and thrust (as applicable to the type of MOV), and motor parameters. The COL applicant will test the MOV under various differential pressure and flow to maximum achievable conditions and perform a sufficient number of tests to determine the torque and thrust requirements at design conditions. The COL applicant will determine the torque and thrust requirements to close the valve for the position at which there is diagnostic indication of hard seat contact. The determination of design torque and thrust requirements will be made for such parameters as differential pressure, fluid flow, undervoltage, temperature and seismic dynamic effects for MOVs that must operate during these transients. The design torque and thrust requirements will be adjusted for diagnostic equipment inaccuracies. For the point of control switch trip, the COL applicant will determine any loss in torque produced by the actuator and thrust delivered to the stem for increasing differential pressure and flow conditions (referred to as load sensitive behavior). The COL applicant will compare the design torque and thrust requirements to the control switch trip torque and thrust subtracting margin for load sensitive behavior, control switch repeatability, and degradation. The COL applicant will measure the total thrust and torque delivered by the MOV under static and dynamic conditions (including diagnostic equipment inaccuracy and control switch repeatability) to compare to the allowable structural capability limits for the individual parts of the MOV. The COL applicant will test for proper control room position indication of the MOV.

* See Subsection 3.9.6.2.2.

The parameters and acceptance criteria for demonstrating that the above functional performance requirements have been met are as follows:

(a) *As required by the safety function: the valve must fully open; the valve must full close with diagnostic indication of hard seat contact.*

(b) *The control switch settings must provide adequate margin to achieve design requirements including consideration of diagnostic equipment inaccuracy, control switch repeatability, load sensitive behavior, an margin for degradation.*

(c) *The motor output capability at degraded voltage must equal or exceed the control switch setting including consideration of diagnostic equipment inaccuracy, control switch repeatability, load sensitive behavior and margin for degradation.*

(d) *The maximum torque and thrust (as applicable for the type MOV) achieved by the MOV including diagnostic equipment inaccuracies and control switch repeatability must not exceed the allowable structural capability limits for the individual parts of the MOV.*

(e) *The remote position indication testing must verify that proper disk position is indicated in the control room.*

(f) *Stroke time measurements taken during valve opening and closing must meet minimum and maximum stroke time requirements.*]*

The inservice testing of MOVs will rely on diagnostic techniques that are consistent with the state of the art and which will permit an assessment of the performance of the valve under actual loading. Periodic testing per GL89-10 Paragraphs D and J will be conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design basis conditions. The COL applicant will determine the optimal frequency of this periodic verification. The frequency and test conditions will be sufficient to demonstrate continuing design basis and required operating capability. See Subsection 3.9.7.3 for COL license information requirements. The Code provides criteria limits for the test parameters identified in Table 3.9-8 for code inservice testing.

A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related MOVs, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3 for COL license information requirements.)

* See Subsection 3.9.6.2.2.

3.9.6.2.3 Power Operated Valves

(1) Design and Qualification

For each power-operated (includes pneumatic- hydraulic-, piston-, and solenoid-operated) valve assembly (POV) with an active safety-related function, the design basis and required operating conditions (including testing) under which the POV will be required to perform will be established.

The COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model the COL applicant will determine the force (as applicable to the type of POV) requirement to operate the POV and will ensure the adequacy of the force that the operator can deliver under design (design basis and required operating) conditions. The COL applicant will also test each size, type, and model under a range of differential pressure and flow conditions up to the design conditions. These design conditions include fluid flow, differential pressure (including pipe break), system pressure, fluid temperature, ambient temperature, minimum air supply system (or accumulator) pressure, spring force, and minimum and maximum stroke time requirements. This testing of each size, type and model shall include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. From this testing, the COL applicant will demonstrate that the results of testing under in-situ conditions can be used to ensure the capability of the POV to operate under design conditions. The COL applicant will ensure that the structural capability limits of the assembly and the individual parts of the POV will not be exceeded under design conditions. The COL applicant will ensure that packing adjustment limits are specified for the valve for each application such that it is not susceptible to stem binding.

(2) Pre-operational Testing

The COL applicant will test each POV in the open and close directions under static and maximum achievable conditions using diagnostic equipment that measures or provides information to determine total friction, stroke time, seat load, spring rate, and travel under normal pneumatic or hydraulic pressure (as applicable to the type of POV), and minimum pneumatic or hydraulic pressure. The COL applicant will test the POV under various differential pressure and flow up to maximum achievable conditions and perform a sufficient number of tests to determine the force requirements at design conditions. The COL applicant will determine the force requirements to close the valve for the position at which there is a diagnostic indication of full valve closure (as required for the safety function of the applicable valves). The determination of design force requirements will be made for such parameters as differential pressure, fluid flow, minimum pneumatic or hydraulic pressure, power supply, temperature, and seismic/dynamic effects for POVs that must operate during these transients. The design force requirements will be adjusted for diagnostic equipment inaccuracies.

The COL applicant will measure the total force delivered by the POV under static and dynamic conditions (including diagnostic equipment inaccuracies) to compare to the allowable structural capability limits for the assembly and individual parts of the POV. The COL applicant will test for proper control room position indication of the POV.

The parameters and acceptance criteria for demonstrating that the above functional performance requirements have been met are a follows:

(a) As required by the safety function, the valve must fully open and/or the valve must fully close with diagnostic indication of hard seat contact.

(b) The assembly must demonstrate adequate margin to achieve design requirements including consideration of diagnostic equipment inaccuracies and margin for degradation.

(c) The assembly must demonstrate adequate output capability of the power-operator at minimum pneumatic or hydraulic pressure of electrical supply (or loss of motive force for fail-safe positioning) with consideration of diagnostic equipment inaccuracies and margin for degradation.

(d) The maximum force (as applicable for the type of POV) achieved by the POV including diagnostic equipment inaccuracies must not exceed the allowable structural capability limits for the assembly and individual part of the POV.

(e) The remote position indication testing must verify that proper disk position is indicated in the control room and other remote locations relied upon by operators in any emergency situation.

(f) Stroke-time measurements taken during valve opening and closing must meet minimum and maximum stroke-time requirements.

(g) For solenoid-operated valves (SOVs), the Class 1E electrical requirements are to be verified. The SOV should be verified to be capable of performing for design requirements for energized or denergized and rated appropriately for the electrical power supply amperage and voltage.

(h) Provide leak-tight seating which must meet specified maximum leakage rate, or meet leakage rate to ensure an overall containment maximum leakage.

All ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the POVs under design conditions. Inservice testing will incorporate the use of advance non-intrusive techniques to periodically assess degradation and the performance characteristics of the POVS. The Part 10 tests will be performed, and valves that fail to exhibit the required performance can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related POVs including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3 for COL license information requirements.)

3.9.6.2.4 Isolation Valve Leak Tests

The leaktight integrity will be verified for each valve relied upon to provide a leaktight function. These valves include:

(1) Pressure isolation valves—valves that provide isolation of pressure differential from one part of a system from another or between systems.

(2) Temperature isolation valves—valves whose leakage may cause unacceptable thermal loading on supports or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps.

(3) Containment isolation valves—valves that perform a containment isolation function in accordance with the Evaluation Against Criterion 54, Subsection 3.1.2.5.5.2, including valves that may be exempted from Appendix J, Type C, testing but whose leakage may cause loss of suppression pool water inventory.

Leakage rate testing for valve group (1) is addressed in Subsection 3.9.6. Valve groups (2) and (3) will be tested in accordance with Part 10, Paragraph 4.2.2.3. The fusible plug valves that provide a lower drywell flood for severe accidents are described in Subsection 9.5.12. The valves are safety-related due to the function of retaining suppression pool water as shown in Figure 9.5-3. The fusible plug valve is a nonreclosing pressure relief device and the Code requires replacement of each at a maximum of 5-year intervals.

3.9.7 COL License Information

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The first COL applicant will provide, at the time of application, the results of the vibration assessment program for the ABWR prototype internals. These results will include the following information specified in Regulatory Guide 1.20.

RG 1.20	Subject
C.2.1	Vibration Analysis Program
C.2.2	Vibration Measurement Program
C.2.3	Inspection Program
C.2.4	Documentation of Results

NRC review and approval of the above information on the first COL applicant's docket will complete the vibration assessment program requirements for prototype reactor internals.

In addition to the information tabulated above, the first COL applicant will provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals.

Subsequent COL applicants need only provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals (Subsection 3.9.2.4).

3.9.7.2 ASME Class 2 or 3 or Quality Group D Components with 60-Year Design Life

COL applicants will identify ASME Class 2 or 3 or Quality Group D components that are subjected to cyclic loadings, including operating vibration loads and thermal transients effects, of a magnitude and/or duration so severe the 60-year design life cannot be assured by required Code calculations and, if similar designs have not already been evaluated, either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads (Subsection 3.9.3.1.)

3.9.7.3 Pump and Valve Testing Program

COL applicants will provide plant specific environmental parameters for the equipment qualification program in accordance with Subsection 3.9.3.2.

COL applicants will provide a plan for the detailed pump and valve inservice testing and inspection program. This plan will:

(1) Include baseline pre-service testing to support the periodic inservice testing of the components required by technical specifications. Provisions are included to disassemble and inspect the pump, check valves, POVs, and MOVs within the Code and safety-related classification as necessary, depending on test results (Subsections 3.9.6, 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3).

(2) Provide a study to determine the optimal frequency of the periodic verification of the continuing MOV capability for design basis conditions (Subsections 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3).

The COL applicant will include the design qualification test, inspection and analysis criteria in Subsections 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2 and 3.9.6.2.3 in the development of the respective safety-related pump and valve design specifications.

The COL applicant will address the design, qualification, and preoperational testing for MOVs as discussed in Subsection 3.9.6.2.2 prior to plant startup.

3.9.7.4 Audit of Design Specification and Design Reports

COL applicants will make available to the NRC staff design specification and design reports required by ASME Code for vessels, pumps, valves and piping systems for the purpose of audit (Subsection 3.9.3.1).

The COL applicant shall ensure that the piping system design is consistent with the construction practices, including inspection and examination methods, of the ASME Code edition and addenda as endorsed in 10CFR50.55a in effect at the time of application.

The COL applicant shall identify ASME Code editions and addenda other than those listed in Tables 1.8-21 and 3.2-3, that will be used to design ASME Code Class 1, 2 and 3 pressure retaining components and supports. The applicable portions of the ASME Code editions and addenda shall be identified to the NRC staff for review and approval with the COL application (Subsection 3.9.3.1).

3.9.8 References

3.9-1 “BWR Fuel Channel Mechanical Design and Deflection”, NEDE-21354-P, September 1976.

3.9-2 “BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings”, NEDE-21175-P, November 1976.

3.9-3 NEDE-24057-P (Class III) and NEDE-24057 (Class I) Assessment of Reactor Internals. Vibration in BWR/4 and BWR/5 Plants, November 1977. Also NEDO-24057-P, Amendment 1, December 1978, and NEDE-2-P 24057 Amendment 2, June 1979.

3.9-4 “General Electric Company, Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K”, NEDE-20566P, Proprietary Document, November 1975.

3.9-5 “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking”, NUREG-0619.

3.9-6 ["*General Electric Environmental Qualification Program", NEDE-24326-1-P, Proprietary Document, January 1983*.]*

3.9-7 [*Functional Capability of Piping Systems, U.S. Nuclear Regulatory Commission, NUREG-1367, November 1992.*]†

3.9-8 "Generic Criteria for High Frequency Cutoff of BWR Equipment", NEDE-25250, Proprietary Document, January 1980.

3.9-9 [*General Electric Company, Plain Carbon Steels, 408HA414, Rev. 1*.]†

3.9-10 [*EPRI NP-5639, "Guidelines for Piping System Reconciliation", May 1988*.]†

3.9-11 [*NUREG/CR-6049, Piping Benchmark Problems for the GE ABWR, August 1993*.]†

3.9-12 NCIG-01, Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, Revision 2.

* See Section 3.10 and Appendix 3K. This reference is same as Reference 3.11-2 (Subsection 3.11.7).

† See Subsection 3.9.1.7.

Table 3.9-1 Plant Events

A. Plant Operating Events	ASME Code Service Limit[1]	No. of Cycles/Events[2]
1. Boltup[2]	A	45
2. Hydrostatic Test (two test cycles for each boltup cycle)	Testing	90
3. Startup (56°C/h Heatup Rate)[3]	A	260
4. Daily and Weekly Reduction to 50% Power[2]	A	18,000
5. Control Rod Pattern Change[2]	A	600
6. Loss of Feedwater Heaters	B	120
7. Scram:		
a. Turbine Generator Trip, Feedwater On, and Other Scrams	B	125
b. Loss of Feedwater Flow, Loss of Auxiliary Power	B	139
c. Turbine Bypass, Single Safety or Relief Valve Blowdown	B	8
8. Reduction to 0% Power, Hot Standby, Shutdown (56°C/h Cooldown Rate)[3]	A	252
9. Refueling Shutdown with Head Spray and Unbolt[2]	A	45
10. Scram:		
a. Reactor Overpressure with Delayed Scram (Anticipated Transient Without Scram, ATWS)	C	1[4]
b. Automatic Blowdown	C	1[4]
11. Improper or Sudden Start of Recirculation Pump with Cold Bottom Head or Hot Standby—Drain Shut Off—Pump Restart	C	1[4]

See next page for footnotes

Table 3.9-1 Plant Events (Continued)

B. Dynamic Loading Events[5]	ASME Code Service Limit[1]	No. of Cycles/Events[2]
12. [*Safe Shutdown Earthquake (SSE) Event at Rated Power Operating Conditions*	*B*[6]	*2 Events* [7] *10 Cycles/event*]*
13. Safe Shutdown Earthquake (SSE) at Rated Power Operating Conditions	D[8]	1 Cycle[4]
14. Turbine Stop Valve Full Closure (TSVC) [9] During Event 7a and Testing	B	320 Events 3 Cycles/ Event
15. Safety/Relief Valve (SRV) Actuations During Events 7a and 7b		
— One plus Two Adjacent	B	2536 Events[10]
— All plus Automatic Depressurization System	B	264 Events[10]
16. Loss-of-Coolant Accident (LOCA)		
Small Break LOCA (SBL) or	D[8]	1[4]
Intermediate Break LOCA (IBL) or	D[8]	1[4]
Large Break LOCA (LBL)	D[8]	1[4]

1 These ASME Code Service Limits apply to ASME Code Class 1, 2 and 3 components, component supports and Class CS structures. Different limits apply to Class MC and CC containment vessels and components, as discussed in Section 3.8.

2 Some events apply to reactor pressure vessel (RPV) only. The number of events/cycles applies to RPV as an example.

3 Bulk average vessel coolant temperature change in any one-hour period.

4 The annual encounter probability of a single event is $<10^{-2}$ for a Level C event and $<10^{-4}$ for a Level D event (Subsection 3.9.3.1.1.5).

5 Table 3.9-2 shows the evaluation basis combination of these dynamic loadings.

6 [*The effects of displacement-limited, seismic anchor motions (SAM) due to SSE shall be evaluated for safety-related ASME Code Class 1, 2, and 3 components and component supports to ensure their functionality during and following an SSE. The SAM effects shall include relative displacements of piping between building floors and slabs, at equipment nozzles, at piping penetrations and at connections of small diameter piping to large diameter piping. See Table 3.9-2 and Note 7 of Table 3.9-2 for stress limits to be used to evaluate the SAM effects.*]*

7 Use 20 peak SSE cycles for evaluation of ASME Class I components and core support structures for Service Level B fatigue analysis. Alternatively, an equivalent number of fractional SSE cycles may be used in accordance with Subsection 3.7.3.2.

8 Appendix F or other appropriate requirements of the ASME Code are used to determine the service Level D limits, as described in Subsection 3.9.1.4.

9 Applicable to main steam piping system only.

10. The number of Reactor Building vibratory load cycles on the reactor vessel and internal components is 19,600 cycles of varying amplitude during the 264 events of SRV actuation. The number of Reactor Building vibratory load cycles on the piping systems inside the containment is 2536 events of single SRV actuation, with 3 stress cycles per event and 264 events of SRV actuation of all valves or the Automatic Depressurization System valves, with 3 stress cycles per event.

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

Table 3.9-2 Load Combinations and Acceptance Criteria for Safety-Related, ASME Code Class 1, 2 and 3 Components, Component Supports, and Class CS Structures

Plant Event	Service Loading Combination [1 2 3]	ASME Service Level[4]
1. Normal Operation (NO)	N	A
2. Plant/System Operating Transients (SOT)	(a) N + TSVC (b) N + SRV[5]	B B
3. [*NO + SSE*	*N + SSE*	*B]*[*6 7]
4. Not Used		
5. Infrequent Operating Transient (IOT), ATWS	N[8] + SRV[5]	C[9]
6. SBL	N + SRV[10] + SBL[11]	C[9]
7. SBL or IBL + SSE	N + SBL (or IBL)[11] + SSE + SRV[10]	D[9 12]
8. LBL + SSE	N + LBL[11] + SSE	D[9 12]
9. NLF	N + SRV[5]+ TSVC	D[9]

1 See Legend on the following pages for definition of terms. See Table 3.9-1 for plant events and cycles information.
The service loading combination also applies to Seismic Category I Instrumentation and electrical equipment (Section 3.10).

2 For vessels and pumps, loads induced by the attached piping are included as identified in their design specification.
For piping systems, water (steam) hammer loads are included as identified in their design specification.

3 The method of combination of the loads is in accordance with NUREG-0484, Revision 1.

4 The service levels are as defined in appropriate subsection of ASME Section III, Division 1.

5 The most limiting load combination case among SRV(1), SRV(2) and SRV (ALL). For main steam and branch piping evaluation, additional loads associated with relief line clearing and blowdown into the suppression pool are included.

6 Applies only to fatigue evaluation of ASME Code Class I components and core support structures. See Dynamic Loading Event No. 12, Table 3.9-1, and Note 7 of Table 3.9-1 for number of cycles.

7 [*For ASME Code 1,2 and 3 piping the following changes and additions to ASME Code Section III Subsections NB-3600, NC-3600 and ND-3600 are necessary and shall be evaluated to meet the following stress limits:*

(a) ASME Code Class 1 Piping:

$$S_{SAM} = C_2 \frac{D_0}{2I} M_c \leq 6.0Sm$$

where: S_{SAM} *is the nominal value of seismic anchor motion stress*

M_c *is the combined moment range equal to the greater of (1) the resultant range of thermal and thermal anchor movements plus one-half the range of the SSE anchor motion, or (2) the resultant range of moment due to the full range of the SSE anchor motions alone.*

C_2, D_0 *and* I *are defined in ASME Code Subsection NB-3600*

SSE inertia and seismic anchor motion loads shall be included in the calculation of ASME Code Subsection NB-3600 equations (10) and (11).

(b) ASME Code Class 2 and 3 Piping:

$$S_{SAM} = i\frac{M_c}{Z} \leq 3.0S_h \quad (\leq 2.0S_y)$$

where: S_{SAM} *and* M_c *are as defined in (a) above, and*

i and Z are defined in ASME Code Subsections NC/ND-3600

SSE inertia and seismic anchor motion loads shall not be included in the calculation of ASME Code Subsections NC/ND-3600 Equation (9), Service Levels A and B and Equations (10) and (11).]*

8 The reactor coolant pressure boundary is evaluated using in the load combination the maximum pressure expected to occur during ATWS.

9 [*All ASME Code Class 1,2 and 3 Piping systems which are essential for safe shutdown under the postulated events are designed to meet the requirements of NUREG-1367 (Reference 3.9-7).*]* Piping system dynamic moments can be calculated using an elastic response spectrum or time history analysis.

10 The most limiting load combination case among SRV(1), SRV(2) and SRV (ADS). See Note (5) for main steam and branch piping.

11 The piping systems that are qualified to the leak-before-break criteria of Subsection 3.6.3 are excluded from the pipe break events to be postulated for design against LOCA dynamic effects, viz., SBL, IBL and LBL.

12 [*For active Class 2 and 3 pumps (and active Class 1,2 and 3 valves), the stresses are limited by criteria:* $\sigma m < 1.2S$ *(or 0.75 Sy), and (*σm *or* σL*) +* $\sigma b \leq 1.8S$ *(or 1.1 Sy), where the notations are as defined in the ASME Code, Section III, Subsections NB and NC or ND, respectively.*]†

* See Subsection 3.9.1.7.

† See Section 3.10.

Load Definition Legend:

Normal (N)	—	Normal and/or abnormal loads associated with the system operating conditions, including thermal loads, depending on acceptance criteria.
SOT	—	System Operational Transient (Subsection 3.9.3.1).
IOT	—	Infrequent Operational Transient (Subsection 3.9.3.1).
ATWS	—	Anticipated Transient Without Scram.
TSVC	—	Turbine stop valve closure induced loads in the main steam piping and components integral to or mounted thereon.
RBV Loads	—	Dynamic loads in structures, systems and components because of reactor building vibration (RBV) induced by a dynamic event.
NLF	—	Non-LOCA faulted.
SSE	—	RBV loads induced by safe shutdown earthquake.
SRV(l), SRV(2)	—	RBV loads induced by SRV discharge of one or two adjacent valves, respectively.
SRV(ALL)	—	RBV loads induced by actuation of all SRVs which activate within milliseconds of each other (e.g., turbine trip operational transient).
SRV(ADS)	—	RBV loads induced by the actuation of SRVs associated with the Automatic Depressurization System, which actuate within milliseconds of each other during the postulated small or intermediate break LOCA, or SSE.
LOCA	—	The loss-of-coolant accident associated with the postulated pipe failure of a high-energy reactor coolant line. The load effects are defined by $LOCA_1$ through $LOCA_7$. LOCA events are grouped in three categories, SBL, IBL or LBL, as defined here.
$LOCA_1$	—	Pool swell (PS) drag/fallback loads on essential piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
$LOCA_2$	—	Pool swell (PS) impact loads acting on essential piping and components located above the suppression pool water upper surface.

$LOCA_3$ — (a) Oscillating pressure-induced loads on submerged essential piping and components during main vent clearing (VLC), condensation oscillations (C)0, or chugging (CHUG), or

(b) Jet impingement (JI) load on essential piping and components as a result of a postulated IBL or LBL event.

Piping and components are defined essential, if they are required for shutdown of the reactor or to mitigate consequences of the postulated pipe failure without offsite power (see introduction to Subsection 3.6).

$LOCA_4$ — RBV load from main vent clearing (VLC).

$LOCA_5$ — RBV loads from condensation oscillations (CO).

$LOCA_6$ — RBV loads from chugging (CHUG).

$LOCA_7$ — Annulus pressurization (AP) loads due to a postulated line break in the annulus region between the RPV and shieldwall. Vessel depressurization loads on reactor internals (Subsection 3.9.2.5) and other loads due to reactor blowdown reaction and jet impingement and pipe whip restraint reaction from the broken pipe are included with the AP loads.

SBL — Loads induced by small break LOCA (Subsections 3.9.3.1.1.3 and 3.9.1.1.4); the loads are: $LOCA_3$(a), $LOCA_4$ and $LOCA_6$. See Note (11).

IBL — Loads induced by intermediate break LOCA (Subsection 3.9.3.1.1.4); the loads are: $LOCA_3$(a) or $LOCA_3$(b), $LOCA_4$, $LOCA_5$ and $LOCA_6$. See Note (11).

LBL — Loads induced by large break LOCA (Subsection 3.9.1.1.4); the loads are: $LOCA_1$ through $LOCA_7$. See Note (11).

Table 3.9-3 Pressure Differentials Across Reactor Vessel Internals

Reactor Component[1]		Maximum Pressure Differences Occurring During a Steamline Break (kPaD)	
		Case 1[2]	Case 2[3]
1.	Core plate and guide tube	184.08	162.01
2.	Shroud support ring and lower shroud (beneath the core plate)	242.04	260.67
3.	Shroud head (at marked elevation)	77.97	149.65
4.	Upper shroud (just below top guide)	90.22	152.4
5.	Core averaged power fuel bundle (bulge at bottom of bundle)	98.07	89.63
5.	Core averaged power fuel bundle (collapse at bottom of top guide)	81.40	79.34
6.	Maximum power fuel bundle (bulge at bottom of bundle)	111.8	96.50
7.	Top guide	42.76	64.72
8.	Steam Dryer	47.56	74.53
–	Shroud head to water level, from points (a) to (b), irreversible pressure drop	92.38	159.85
–	Shroud head to water level, from points (a) to (b), elevation pressure drop	10.40	15.20

1 Item numbers in this column correspond to the location (node) numbers identified in Figure 3.9-5.

2 Instantaneous break initiated at 102% rated core power, 102.4% rated steam flow, and 111.1% rated recirculation flow.

3 Instantaneous break initiated at 54.5% rated core power, 49.8% rated steam flow, and 114.8% rated recirculation flow.

Table 3.9-4 Deformation Limit for Safety Class Reactor Internal Structures Only

Either One of (Not Both)	General Limit
a. Permissible Deformation, DP / Analyzed Deformation Causing Loss of Function, DL	$\leq \frac{0.9}{SF_{Min}}$
b. Permissible Deformation, DP / Experiment Deformation Causing Loss of Function, DE	$\leq \frac{1.0}{SF_{Min}}$ [1]

Where:

DP = Permissible deformation under stated conditions of Service levels A, B, C or D (normal, upset, emergency or fault)

DL = Analyzed deformation which could cause a system loss of functions[2]

DE = Experimentally determined deformation which could cause a system loss of function

SF_{Min}= Minimum safety factor (see Subsection 3.9.5.3.6)

1 Equation will not be used unless supporting data are provided to the NRC by General Electric.

2 "Loss of Function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they will be specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, reactor internal pump wear, or excess leakage of any component.

Table 3.9-5 Primary Stress Limit for Safety Class Reactor Internal Structures Only

	Any One of (No More than One Required)	General Limit
a.	$\dfrac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresses, PN}}$	$\leq \dfrac{2.25}{SF_{Min}}$
b.	$\dfrac{\text{Permissible Load, LP}}{\text{Largest lower bound limit load, CL}}$	$\leq \dfrac{1.5}{SF_{Min}}$
c.	$\dfrac{\text{Elastic evaluated primary stress, PE}}{\text{Conventional ultimate strength at temperature, US}}$	$\leq \dfrac{0.75}{SF_{Min}}$
d.	$\dfrac{\text{Elastic-plastic evaluated nominal primary stress, EP}}{\text{Conventional ultimate strength at temperature, US}}$	$\leq \dfrac{0.9}{SF_{Min}}$
e.	$\dfrac{\text{Permissible Load, LP}}{\text{Plastic instability load, PL}}$	$\leq \dfrac{0.9}{SF_{Min}}$ [1]
f.	$\dfrac{\text{Permissible Load, LP}}{\text{Ultimate load from fracture analysis, UF}}$	$\leq \dfrac{0.9}{SF_{Min}}$ [1]
g.	$\dfrac{\text{Permissible Load, LP}}{\text{Ultimate load or loss of function loaf from test, LE}}$	$\leq \dfrac{1.0}{SF_{Min}}$ [1]

Where:

PE = Primary stresses evaluated on an elastic basis. The effective membrane stresses are to be averaged through the load carrying section of interest. The simplest average bending, shear or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.

PE = Permissible primary stress levels under service level A or B (normal or upset) conditions under ASME Boiler and Pressure Vessel Code, Section III.

CL = Lower bound limit load with yield point equal to 1.5 Sm, where Sm is the tabulated value of allowable stress at temperature of the ASME III code or its equivalent. The “lower bound limit load” is herein defined as that produced from the analysis of an ideally plastic (non-strain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yield to the uniaxial case.

US = Conventional ultimate strength at temperature or loading which would cause a system malfunction, whichever is more limiting.

Table 3.9-5 Primary Stress Limit for Safety Class Reactor Internal Structures Only (Continued)

EP = Elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress curve, which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.

PL = Plastic instability loads. The “Plastic Instability Load” is defined here as the load at which any load bearing section begins to diminish its cross-sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.

UF = Ultimate load from fracture analyses. For components which involve sharp discontinuities (local theoretical stress concentration) the use of a “Fracture Mechanics” analysis where applicable utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where “Fracture Mechanics” may be applied are for fillet welds or end of fatigue life crack propagation.

LE = Ultimate load or loss of function load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

SF_{Min}=Minimum safety factor (Subsection 3.9.5.3.6).

1 Do not use unless supporting data are provided to the NRC by General Electric.

Table 3.9-6
Buckling Stability Limit for Safety Class Reactor Internal Structures Only

	Any One Of (No More Than one Required)	General Limit
a.	$\frac{\text{Permissible Load, LP}}{\text{Service Level A (normal) permissible load, PN}}$	$\leq \frac{2.25}{SF_{Min}}$
b.	$\frac{\text{Permissible Load, LP}}{\text{Stability analysis load, SL}}$	$\leq \frac{0.9}{SF_{Min}}$
c.	$\frac{\text{Permissible Load, LP}}{\text{Ultimate buckling collapse load from test, SET}}$	$\leq \frac{1.0}{SF_{Min}}$ [1]

Where:

LP = Permissible load under stated conditions of service levels A, B, C or D (normal, upset, emergency or faulted).

PN = Applicable service level A (normal) event permissive load.

SL = Stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity on column members.

SET= Ultimate buckling collapse load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

SF_{Min}=Minimum safety factor (Subsection 3.9.5.3.6)

1 Equation C will not be used unless supporting data are provided to the NRC by General Electric.

Table 3.9-7 Fatigue Limit for Safety Class Reactor Internal Structures Only

Summation of fatigue damage usage following Miner hypotheses[1]:	
Cumulative Damage in Fatigue	**Limit for Service Levels A & B (Normal and Upset Conditions)**
Design fatigue cycle usage from analysis using the method of the ASME Code	≤ 1.0

1 Miner, M. A., "Cumulative Damage in Fatigue," Journal of Applied Mechanics, Vol. 12, ASME, Vol. 67, pp A159-A164, September 1945.

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves[1]

MPL	System	Pump Page No.	Valve Page No.
B21	Nuclear Boiler		3.9-99
B31	Reactor Recirculation		3.9-102
C12	Control Rod Drive		3.9-103
C41	Standby Liquid Control	3.9-97	3.9-103
C51	Neutron Monitoring (ATIP)		3.9-104
D23	Containment Atmospheric Monitoring		3.9-104
E11	Residual Heat Removal	3.9-97	3.9-105
E22	High Pressure Core Flooder	3.9-97	3.9-110
E31	Leak Detection & Isolation		3.9-112
E51	Reactor Core Isolation Cooling	3.9-97	3.9-113
G31	Reactor Water Cleanup		3.9-118
G41	Fuel Pool Cooling & Cleanup		3.9-119
G51	Suppression Pool Cleanup		3.9-121
K17	Radwaste		3.9-121
P11	Makeup Water (Purified)		3.9-121
P21	Reactor Building Cooling Water	3.9-97	3.9-121
P24	HVAC Normal Cooling Water		3.9-127
P25	HVAC Emergency Cooling Water	3.9-97	3.9-127
P41	Reactor Service Water	3.9-97	3.9-130
P51	Service Air		3.9-131
P52	Instrument Air		3.9-131
P54	High Pressure Nitrogen Gas Supply		3.9-131
T22	Standby Gas Treatment		3.9-132
T31	Atmospheric Control		3.9-134
T49	Flammability Control		3.9-137
U41	Heating, Ventilating and Air Conditioning		3.9-138
Y52	Oil Storage and Transfer	3.9-97	3.9-139

See page 3.9-139 for notes.

1 This table responds to NRC Questions 210.47, 210.48 and 210.49 regarding provisions for inservice testing of safety-related pumps and valves within the scope of the ABWR Standard Plant in accordance with the ASME Code. The information is presented separately for each system for the MPL number.

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves[1] (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Test Param (b)	Test Freq. (f)	Tier 2 Fig. (g)
System Pumps						
C41-C001	2	Standby Liquid Control System Pump	2	Pd,Vd, Q	3 mo	9.3-1
E11-C001	3	Residual Heat Removal System Pump	2	Pd, Pi, Q, Vv	3 mo	5.4-10 (Sh. 3, 4, 6)
E11-C002	3	Residual Heat Removal System fill pump (i1)	2	Pd,Pi, Vv	E10	5.4-10 (Sh. 3, 4, 6)
E22-C001	2	High Pressure Core Flooder pump	2	Pd,Pi, Q,Vv	3 mo	6.3-7(Sh. 2)
E51-C001	1	Reactor Core Isolation Cooling pump	2	N,Pd,Pi Q,Vv	3 mo	5.4-8(Sh. 1)
P21-C001	6	Reactor Building Cooling Water pump	3	Pd, Pi, Q, Vv	E10	9.2-1 (Sh. 1, 4, 7)
P25-C001	6	HVAC Emergency Cooling Water System pump	3	Pd, Pi, Q, Vv	E10	9.2-3 (Sh. 1, 2, 3)
P41-C001	6	Reactor Service Water System pump	3	Pd, Pi, Q, Vv	E10	9.2-7 (Sh. 1, 2, 3)
Y52-C001	6	Standby D/G Fuel Oil Transfer Pump	3	Pd, Pi, Q, Vv	3 mo	9.5-6

1 This table responds to NRC Questions 210.47, 210.48 and 210.49 regarding provisions for inservice testing of safety-related pumps and valves within the scope of the ABWR Standard Plant in accordance with the ASME Code. The information is presented separately for each system for the MPL number.

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
B21 Nuclear Boiler System Valves								
F001	2	Feedwater line motor-operated valve (MOV)	2	B	P	S	RO	5.1-3 sh. 4
F002	2	Upstream (First) FW line check valve (h3)	2	A,C	A	L,S	RO	5.1-3 sh. 4
F003	2	FW line outboard check valve-air-operated (AO) (h1)	1	A,C	I,A	L,P, S	RO	5.1-3 sh. 4
F004	2	FW line inboard check valve (h1)	1	A,C	I,A	L,S	RO	5.1-3 sh. 4
F005	2	FW line inboard maintenance valve	1	B	P		E1	5.1-3 sh. 4
F006	2	RWCU (or CUW) System injection line check valve (h3)	2	A,C	A	L,S	RO	5.1-3 sh. 4
F007	2	RWCU (or CUW) System injection line MOV	2	B	P	S	E1	5.1-3 sh. 4
F008	4	Inboard main Steam isolation valve. (MSIV)(h1)	1	A	I,A	L,P S	RO 3 mo	5.1-3 sh. 3
F009	4	Outboard Main Steam isolation valve (MSIV)(h1)	1	A	I,A	L,P S	RO 3 mo	5.1-3 sh. 3
F010	18	Safety/Relief Valve (SRV) (h1), (h2)	1	A,C	A	R P,S	5 yr RO	5.1-3 sh. 2
F011	1	MSL bypass/drain line inboard isolation valve (h1)	1	A	I,A	L,P S	RO 3 mo	5.1-3 sh. 3
F012	1	MSL bypass/drain line outboard isolation valve (h1)	1	A	I,A	L,P S	RO 3 mo	5.1-3 sh. 3
F013	1	MSL warmup line valve	2	B	P		E1	5.1-3 sh. 3
F016	1	MSL downstream drain line header valve	2	B	P		E1	5.1-3 sh. 3
F017	1	MSL downstream drain line header bypass	2	B	A	P S	RO 3 mo	5.1-3 sh. 3
F018	1	RPV non-condensable gas removal line	1	B	P		E1	5.1-3 sh. 2
F019	1	RPV head vent inboard shutoff valve (h1)	1	B	A	P,S	RO	5.1-3 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F020	1	RPV head vent outboard shutoff valve (h1)	1	B	A	P,S	RO	5.1-3 sh. 2
F021	18	SRV discharge line vacuum breaker(h1)	3	C	A	R,S	RO	5.1-3 sh. 2
F022	18	SRV discharge line vacuum breaker (h1)	3	C	A	R,S	RO	5.1-3 sh. 2
F024	4	Inboard MSIV nitrogen supply line check valve (h1)	3	C	A	S	RO	5.1-3 sh. 3
F025	4	Outboard MSIV air supply line check valve (h1)	3	C	A	S	RO	5.1-3 sh. 3
F026	8	SRV ADS pneumatic supply line check valve (h1)	3	A,C	A	L,S	RO	5.1-3 sh. 2
F029	18	SRV pneumatic supply check valve (h1)	3	C	A	S	RO	5.1-3 sh. 2
F031	2	Inboard valve on the outboard FW line check valve test line	2	B	P		E1	5.1-3 sh. 4
F033	4	Inboard shutoff valve on the outboard MSIV test line	2	B	P		E1	5.1-3 sh. 3
F035	1	Inboard test line valve for the MSL bypass/drain valve	2	B	P		E1	5.1-3 sh. 3
F039	2	Inboard test line valve for the inboard FW line check valve	2	B	P		E1	5.1-3 sh. 4
F040	2	Outboard test line valve for the FW line check valve	2	B	P		E1	5.1-3 sh. 4
F500	2	Inboard test line valve for the first FW line check valve	2	B	P		E1	5.1-3 sh. 4
F503	2	Outboard drain line valve for the FW line check valve	2	B	P		E1	5.1-3 sh. 4
F508	4	Inboard MSIV accumulator A001 drain valve	3	B	P		E1	5.1-3 sh. 3
F509	4	Outboard MSIV accumulator A002 drain valve	3	B	P		E1	5.1-3 sh. 3
F510	8	SRV ADS accumulator A003 drain valve	3	B	P		E1	5.1-3 sh. 2
F511	18	SRV accumulator A004 drain valve	3	B	P		E1	5.1-3 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F700	4	Manual Isolation valve—RPV water level instrument reference leg line	2	B	P		E1	5.1-3 sh. 5,6
F701	4	Excess flow check valve—RPV water level instrument reference leg line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 5,6
F702	4	Manual isolation valve—RPV narrow range water level instrument sensing line	2	B	P		E1	5.1-3 sh. 5,6
F703	4	Excess flow check valve—RPV narrow range water level instrument sensing line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 5,6
F704	4	Manual isolation valve—RPV wide range water level instrument sensing line	2	B	P		E1	5.1-3 sh. 5,6
F705	4	Excess flow check valve—RPV wide range water level instrument sensing line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 5,6
F706	1	Root valve—Reactor well water level instrument sensing line	2	B	P		E1	5.1-3 sh. 5
F709	1	Manual isolation valve—RPV shutdown range water level instrument reference leg line	2	B	P		E1	5.1-3 sh. 2
F710	1	Excess flow check valve—RPV shutdown range water level instrument reference leg line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 2
F711	1	Manual isolation valve—RPV head seal leakage instrument line	2	B	P		E1	5.1-3 sh. 8
F712	1	Excess flow check valve to RPV head seal leakage instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 8
F713	4	Manual isolation valve—RPV above pump deck instrument line	2	B	P		E1	5.1-3 sh. 7
F714	4	Excess flow check valve—RPV above pump deck instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 7

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F715	4	Manual isolation valve—RPV below pump deck instrument line	2	B	P		E1	5.1-3 sh. 7
F716	4	Excess flow check valve—RPV below pump deck instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 7
F717	4	Manual Isolation valve—RPV above core plate instrument line	2	B	P		E1	5.1-3 sh. 7
F718	4	Excess flow check valve—RPV above core plate instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 7
F719	4	Manual isolation valve—RPV below core plate instrument line	2	B	P		E1	5.1-3 sh. 7
F720	4	Excess flow check valve—RPV below core plate instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 7
F723	4	Manual isolation valve—MSL flow restrictor instrument line	2	B	P		E1	5.1-3 sh. 2
F724	4	Excess flow check valve—MSL flow restrictor instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 2
F725	4	Manual isolation valve—MSL flow restrictor instrument line	2	B	P		E1	5.1-3 sh. 2
F726	4	Excess flow check valve—MSL flow restrictor instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 2
F727	2	MSL PX instrument line inboard root valve	2	B	P		E1	5.1-3 sh. 3
B31 Reactor Recirculation Internal Pump Valves								
F008	10	Excess flow check valve RIP pump motor purge water line (h3)	2	A,C	I,A	L,S	RO	5.4-4 sh. 2
F010	10	RIP pump motor purge water supply line valve	2	B	P		E1	5.4-4 sh. 1
F011	10	RIP inflatable pressurized water line inboard valve	2	B	P		E1	5.4-4 sh. 1

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F013	10	RIP seal equalizing line valve	2	B	P		E1	5.4-4 sh. 1
F015	10	Manual maintenance valve—RIP pump motor purge water line	2	B	P		E1	5.4-4 sh. 2
F500	10	RIP cooling water HX vent line inboard valve	2	B	P		E1	5.4-4 sh. 1
F502	10	RIP drain line inboard valve	2	B	P		E1	5.4-4 sh. 1
F505	10	RIP cooling water HX shell drain line inboard valve	2	B	P		E1	5.4-4 sh. 1
C12 Control Rod Drive System Valves								
F719	4	Root valve charging line header pressure instrument line	2	B	P		E1	4.6-8 sh. 2
F720	4	Root valve charging line header pressure instrument line	2	B	P		E1	4.6-8 sh. 2
C41 Standby Liquid Control System Valves								
F001	2	SLCS storage tank outlet line MOV	2	B	A	P S	RO 3 mo	9.3-1
F002	2	SLCS pump suction line maintenance valve	2	B	P		E1	9.3-1
F003	2	SLCS pump discharge line relief valve	2	C	A	R	5 yr	9.3-1
F004	2	SLCS pump discharge line check valve	2	C	A	S	3 mo	9.3-1
F005	2	SLCS pump discharge line maintenance valve	2	B	P		E1	9.3-1
F006	2	SLCS pump injection valve MOV	2	A	I,A	L,P S	2 yr 3 mo	9.3-1
F007	1	SLCS injection line outboard check valve (h5)	2	A,C	I,A	L,S	2 yrs	9.3-1
F008	1	SLCS injection line inboard check valve (h1)	2	A,C	I,A	L,S	2 yr	9.3-1
F018	1	SLCS storage tank sample line inboard shutoff valve	2	B	P		E1	9.3-1

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F025	1	SLCS injection line test/vent line inboard valve	2	B	P		E1	9.3-1
F500	1	SLCS pump suction line drain line	2	B	P		E1	9.3-1
F501	2	SLCS pump discharge line drain line valve	2	B	P		E1	9.3-1
C51 Neutron Monitoring System Valves								
J004	3	Isolation valve assembly: ATIP ball valve	2	A	I,A	L,P S	RO 3 mo	7.6-1 SH. 3
		Index shear valve	2	A,D	A	X	RO	7.6-1 SH. 3
J011	3	Purge isolation valve	2	A,C	P	L,P	2yr	7.6-1 sh. 3
D23 Containment Atmospheric Monitoring System Valves								
F001	2	CAMS drywell pressure instrument line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F004	2	CAMS drywell sample line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F005	2	CAMS drywell return line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F006	2	CAMS wetwell sample line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F007	2	CAMS wetwell return line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F008	2	CAMS rack drain line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F009	2	CAMS drywell pressure instrument line outboard isolation valve	2	B	P		E1	7.6-7 sh. 2
F010	2	CAMS drywell sample line outboard valve	2	B	P		E1	7.6-7 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F011	2	CAMS drywell return line outboard valve	2	B	P		E1	7.6-7 sh. 2
F012	2	CAMS wetwell sample line outboard valve	2	B	P		E1	7.6-7 sh. 2
F013	2	CAMS wetwell return line outboard valve	2	B	P		E1	7.6-7 sh. 2
F014	2	CAMS rack drain line outboard containment isolation valve	2	B	P		E1	7.6-7 sh. 2
E11 Residual Heat Removal System Valves								
F001	3	Suppression pool suction valve	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh.3,4,6
F002	3	RHR pump discharge line check valve	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F003	3	RHR pump discharge line maintenance valve	2	B	P		E1	5.4-10 sh.3,4,6
F004	3	Heat Exchanger flow control valve	2	B	A	P S	2 yr 3 mo	5.4-10 sh.3,4,6
F005	1	RPV injection valve, Loop A (h6)	2	A	A	L,P S	RO CS	5.4-10 sh. 3
F005	2	RPV injection valve, Loop B & C (h6)	1	A	I,A	L,P S	RO CS	5.4-10 sh. 5,7
F006	1	RPV injection line check valve, Loop A	2	A,C	A	L,P S	RO 3 mo	5.5-10 sh. 3
F006	2	RPV injection line check valve, Loop B & C	1	A,C	I,A	L,P S	RO 3 mo	5.4-10 sh. 5,7
F007	2	RPV injection line inboard maint. valve	1	B	P		E1	5.4-10 sh. 5,7
F008	3	Suppression pool return line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh. 3,4,6
F009	3	Shutdown Cooling suction line maintenance valve	1	B	P		E1	5.4-10 sh. 2
F010	3	Shutdown Cooling suction line inboard isolation valve (h6)	1	A	I,A	L,P S	RO CS	5.4-10 sh. 2
F011	3	Shutdown Cooling suction line outboard isolation valve (h6)	1	A	I,A	L,P S	RO CS	5.4-10 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F012	3	Shutdown Cooling suction line adm. valve	2	B	A	P S	2 yr, 3 mo	5.4-10 sh.3,4,6
F013	3	Heat exchanger bypass flow control valve	2	B	A	P S	2 yr, 3 mo	5.4-10 sh.3,4,6
F014	2	Fuel Pool Cooling supply line inboard MOV	2	B	A	P S	2 yr, 3 mo	5.4-10 sh. 5,7
F015	2	Fuel Pool Cooling supply line outboard MOV	2	B	A	P S	2 yr, 3 mo	5.4-10 sh. 5,7
F016	2	Gate valve-line from Fuel Pool Cooling (FPC)	2	B	A	S	3 mo	5.4-10 sh. 2
F017	2	Drywell spray line inboard valve	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh. 5,7
F018	2	Drywell spray line outboard valve	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh. 5,7
F019	2	Wetwell spray line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh. 5,7
F020	3	RHR pump min flow bypass line check valve	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F021	3	RHR pump min flow bypass line MOV	2	A	I,A	L,P S	2 yr 3 mo	5.4-10 sh.3,4,6
F022	3	Discharge line fill pump suction line valve	2	B	P		E1	5.4-10 sh.3,4,6
F023	3	Fill pump discharge line check valve	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F024	3	Fill pump discharge line stop check valve	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F025	3	Fill pump minimum flow line globe valve	2	B	P		E1	5.4-10 sh.3,4,6
F026	3	RHR pump suction to High Conductivity Waste (HCW)	2	B	P		E1	5.4-10 sh.3,4,6
F027	3	Bypass line around the check valve MPL E11-F002	2	B	P		E1	5.4-10 sh.3,4,6
F028	3	Heat exchanger outlet line relief valve	2	C	A	R	5 yr	5.4-10 sh.3,4,6
F029	3	Inboard reactor well drain line valve	2	B	P		E1	5.4-10 sh.3,4,6

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F030	3	Drain to radwaste valve	2	B	P		E1	5.4-10 sh.3,4,6
F031	3	Outboard reactor well drain line valve (to SP)	2	A	I,P	L,P	RO	5.4-10 sh.3,4,6
F032	3	Shutoff valve—line from MUWC	2	B	P		E1	5.4-10 sh.3,4,6
F033	3	Check valve in the line from MUWC	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F034	1	RPV injection line vent/test line inboard valve, Loop A	2	B	P		E1	5.4-10 sh. 3
F034	2	RPV injection line vent/test line inboard valve, Loop B&C	1	B	P		E1	5.4-10 sh. 5,7
F036	1	Press equal valve around check valve E11-F006, Loop A	2	A	P		E1	5.4-10 sh. 3
F036	2	Press equal valve around check valve E11-F006, Loop B&C	1	A	P		E1	5.4-10 sh. 5,7
F037	3	Shutdown cooling suction line test line	1	A	P		E1	5.4-10 sh. 2
F039	3	Relief valve around the MOV MPL E11-F011	1	C	A	R	5 yr	5.4-10 sh. 2
F040	3	Shutoff valve—line from MUWC	2	B	P		E1	5.4-10 sh. 2
F041	3	Check valve—line from Make-Up Water Condenser (MUWC)	2	C	A	S	3 mo	5.4-10 sh. 2
F042	3	Shutdown Cooling Mode suction line relief valve	2	C	A		E1	5.4-10 sh.3,4,5
F043	3	HX outlet to the Sampling System (SS) test inboard valve	2	B	P		E1	5.4-10 sh.3,6,7
F045	1	HX outlet to the PASS—inboard valve	2	B	A	P S	2 yr 3 mo	5.4-10 sh. 3
F046	1	HX outlet to the PASS—outboard valve	2	B	A	P S	2 yr 3 mo	5.4-10 sh. 3
F047	2	Shutoff—line from MUWC	2	B	P		E1	5.4-10 sh. 5,7
F048	2	Check valve—line from MUWC	2	C	P		E1	5.4-10 sh. 5,7

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F049	2	Drywell spray line vent & test line inboard valve	2	B	P		E1	5.4-10 sh. 5,7
F051	3	Fill pump discharge line relief valve	2	C	A	R	5 yr	5.4-10 sh.3,4,6
F052	1	Drain line for the suppression pool	2	B	P		E1	5.4-10 sh. 4
F101	2	AC independent water addition input valve	2	B	A	S	3 mo	5.4-10 sh. 5,7
F102	2	AC independent water addition input valve	2	B	A	S	3 mo	5.4-10 sh. 5,7
F500	3	Heat exchanger inlet drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6
F502	3	HX outlet line drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6
F504	3	RPV injection line vent line inboard valve	2	B	P		E1	5.4-10 sh.3,4,7
F506	1	RPV injection line drain line inboard valve	2	B	P		E1	5.4-10 sh. 3
F506	2	RPV injection line drain line inboard valve	1	B	P		E1	5.4-10 sh. 5,7
F508	3	Shutdown Cooling suction line vent line valve	2	B	P		E1	5.4-10 sh. 2
F509	2	Vent valve—FPC return line	2	B	P		E1	5.4-10 sh. 5,7
F511	2	Drywell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F513	2	Drywell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F515	2	Wetwell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F517	3	RHR pump min flow line drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6
F700	3	RHR pump suction line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F701	3	RHR pump suction line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F702	3	RHR pump discharge line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F704	3	RHR pump discharge line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F706	3	RHR pump discharge line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F707	3	RHR pump discharge line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F708	3	FT MPL E11-FT008 instrument line inboard root valve	2	B	P		E1	5.4-10 sh.3,4,6
F709	3	FT MPL E11-FT008 instrument line outboard root valve	2	B	P		E1	5.4-10 sh.3,4,6

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F710	3	FT MPL E11-FT008 instrument line inboard root valve	2	B	P		E1	5.4-10 sh.3,4,6
F711	3	FT MPL E11-FT008 instrument line outboard root valve	2	B	P		E1	5.4-10 sh.3,4,6
F712	3	Shutdown Cooling Mode suction line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F713	3	Fill pump suction line instrument line valve	2	B	P		E1	5.4-10 sh.3,4,6
F714	1	Discharge to radwaste flow instrument line	2	B	P		E1	5.4-10 sh. 4
F716	1	Discharge to radwaste flow instrument line	2	B	P		E1	5.4-10 sh. 4
F718	3	Fill pump discharge line check valve test point	2	B	P		E1	5.4-10 sh. 3,4,6
F720	3	Fill pump discharge line check valve test point	2	B	P		E1	5.4-10 sh. 3,4,6
E22 High Pressure Core Flooder System Valves								
F001	2	Condensate Storage Tank (CST) suction line MOV	2	B	A	P S	2 yr 3 mo	6.3-7 sh. 2
F002	2	CST suction line check valve	2	C	A	S	3 mo	6.3-7 sh. 2
F003	2	HPCF System injection valve (h6)	1	A	I,A	L,P S	RO CS	6.3-7 sh. 1
F004	2	HPCF System inboard check valve	1	A,C	I,A	L,P S	RO 3 mo	6.3-7 sh. 1
F005	2	Pump discharge line inboard maintenance valve	1	B	P		E1	6.3-7 sh. 1
F006	2	Suppression pool suction line MOV	2	A	I,A	L,P, S	RO 3 mo	6.3-7 sh. 2
F007	2	Suppression pool suction line check valve	2	C	A	S	3 mo	6.3-7 sh. 2
F008	2	Test return line inboard valve	2	B	A	P S	2 yr 3 mo	6.3-7 sh. 2
F009	2	Test return line outboard valve	2	A	I,A	L,P S	RO 3 mo	6.3-7 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F010	2	Pump minimum flow bypass line MOV	2	A	I,A	L,P S	RO 3 mo	6.3-7 sh. 2
F011	2	Bypass line shutoff valve around check valve E22-F002	2	B	P		E1	6.3-7 sh. 2
F012	2	HPCI pump suction line drain line to HCW	2	B	P		E1	6.3-7 sh. 2
F014	2	Pump discharge line fill line outboard check valve	2	C	A	S	3 mo	6.3-7 sh. 1
F015	2	Pump discharge line fill line outboard check valve	2	C	A	S	3 mo	6.3-7 sh. 1
F017	2	Pump discharge line test and vent line inboard valve	1	A	P		E1	6.3-7 sh. 1
F019	2	Pressure equalizing valve around check valve E22-F004	1	A	P		E1	6.3-7 sh. 1
F020	2	Suppression pool suction line relief valve	2	C	A	R	5 yr	6.3-7 sh. 2
F021	2	Pump discharge check valve	2	C	A	S	3 mo	6.3-7 sh. 2
F022	2	Suppression pool suction line test line valve	2	B	P		E1	6.3-7 sh. 2
F023	2	Pump discharge line test line valve	2	B	P		E1	6.3-7 sh. 2
F500	2	Pump discharge line high point vent inboard valve	2	B	P		E1	6.3-7 sh. 1
F502	2	Pump discharge line drywell test line inboard valve	2	B	P		E1	6.3-7 sh. 1
F700	2	Pump suction line pressure instrument line root valve	2	B	P		E1	6.3-7 sh. 2
F701	2	Pump suction line pressure instrument line root valve	2	B	P		E1	6.3-7 sh. 2
F702	2	Pump discharge line pressure instrument line inboard valve	2	B	P		E1	6.3-7 sh. 2
F704	2	Pump discharge line pressure instrument line inboard valve	2	B	P		E1	6.3-7 sh. 2
F705	2	Pump discharge line pressure instrument line outboard valve	2	B	P		E1	6.3-7 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F706	2	Pump discharge line flow instrument line inboard valve	2	B	P		E1	6.3-7 sh. 1
F707	2	Pump discharge line flow instrument line outboard valve	2	B	P		E1	6.3-7 sh. 1
F708	2	Pump discharge line flow instrument line inboard valve	2	B	P		E1	6.3-7 sh. 1
F709	2	Pump discharge line flow instrument line outboard valve	2	B	P		E1	6.3-7 sh. 1
E31 Leak Detection and Isolation System Valves								
F001	1	Drywell fission product monitoring line maintenance valve	2	B	P		E1	5.2-8 sh. 9
F002	1	Drywell fission product monitoring line inboard isolation valve	2	A	I,A	L,P S	RO 3 mo	5.2-8 sh. 9
F003	1	Drywell fission product monitoring line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	5.2-8 sh. 9
F004	1	Drywell fission product monitoring line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	5.2-8 sh. 9
F005	1	Drywell fission product monitoring line inboard isolation valve	2	A	I,A	L,P S	RO 3 mo	5.2-8 sh. 9
F006	1	Drywell fission product monitoring line maintenance valve	2	B	P		E1	5.2-8 sh. 9
F701	4	RCIC instrument line manual maintenance valve	2	B	P		E1	5.2-8 sh. 6
F702	4	RCIC instrument line isolation excess flow check valve(h3)	2	A, C	I,A	L,S	RO	5.2-8 sh. 6
F703	4	RCIC instrument line manual maintenance valve	2	B	P		E1	5.2-8 sh. 6
F704	4	RCIC instrument line isolation excess flow check valve (h3)	2	A, C	I, A	L,S	RO	5.2-8 sh. 6
E51 Reactor Core Isolation Cooling System Valves								
F001	1	Condensate Storage Tank (CST) suction line MOV	2	B	A	P S	2 yr 3 mo	5.4-8 sh. 1

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F002	1	CST suction line check valve	2	C	A	S	3 mo	5.4-8 sh. 1
F003	1	RCIC pump discharge line check valve	2	C	A	P S	2 yr 3 mo	5.4-8 sh. 1
F004	1	RCIC System injection valve (h6)	2	A	A	L,P S	RO CS	5.4-8 sh. 1
F005	1	RCIC System discharge line testable check valve	2	C	A	L,P S	RO 3 mo	5.4-8 sh. 1
F006	1	Suppression Pool (CSP) suction line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 1
F007	1	Suppression Pool (CSP) suction line check valve	2	C	A	S	3 mo	5.4-8 sh. 1
F008	1	RCIC System suppression pool test return line MOV	2	A	A	P S	2 yr 3 mo	5.4-8 sh. 1
F009	1	RCIC System suppression pool test return line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 1
F010	1	RCIC System minimum flow bypass line check valve	2	C	A	P S	2 yr 3 mo	5.4-8 sh. 1
F011	1	RCIC System minimum flow bypass line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 1
F012	1	RCIC turbine accessories cooling water line MOV	2	B	A	P S	2 yr 3 mo	5.4-8 sh. 3
F013	1	RCIC turbine accessories cooling water line PCV	2	B	A		E1	5.4-8 sh. 3
F015	1	Barometric condenser condensate pump discharge line valve	2	B	P		E1	5.4-8 sh. 3
F016	1	Barometric condenser condensate pump discharge line check valve	2	C	P	P S	2 yr 3 mo	5.4-8 sh. 3
F017	1	RCIC pump suction line relief valve	2	C	A	R	5 yr	5.4-8 sh. 1
F018	1	Valve in the bypass line around check valve E51-F003	2	B	P		E1	5.4-8 sh. 1
F019	1	Pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F020	1	Pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1
F021	1	Pump discharge line fill line shutoff valve	2	B	P		E1	5.4-8 sh. 1
F022	1	Pump discharge line fill line check valve	2	C	A	S	3 mo	5.4-8 sh. 1
F023	1	Pump discharge line fill line check valve	2	C	A	S	3 mo	5.4-8 sh. 1
F024	1	Pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1
F025	1	Pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1
F026	1	Valve in pressure equalizing line around E51-F005	2	B	P		E1	5.4-8 sh. 1
F027	1	Suppression Pool (S/P) suction line test line valve	2	B	P		E1	5.4-8 sh. 1
F028	1	Minimum flow bypass line test line valve	2	B	P		E1	5.4-8 sh. 1
F029	1	Minimum flow bypass line test line valve	2	B	P		E1	5.4-8 sh. 1
F030	1	Turbine accessories cooling water line relief valve	2	C	A	R	5 yr	5.4-8 sh. 3
F031	1	Barometric condenser condensate discharge line AOV to HCW	2	B	P		E1	5.4-8 sh. 3
F032	1	Barometric condenser condensate discharge line AOV to HCW	2	B	P		E1	5.4-8 sh. 3
F033	1	Discharge line fill line bypass line shutoff valve	2	B	P		E1	5.4-8 sh. 3
F034	1	Barometric condenser condensate pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 3
F035	1	Steam supply line isolation valve	1	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 2
F036	1	Steam supply line isolation valve	1	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F037	1	Steam admission valve	2	B	A	P, S	2 yr 3 mo	5.4-8 sh. 2
F038	1	Turbine exhaust line check valve (h3)	2	A, C	I,A	L S	2 yr RO	5.4-8 sh. 2
F039	1	Turbine exhaust line MOV	2	A	I,A	L,P S	2 yr 3 mo	5.4-8 sh. 1
F044	1	Steam admission valve bypass line maintenance valve	2	B	P		E1	5.4-8 sh. 2
F045	1	Steam admission valve bypass line MOV	2	B	A	P S	2 yr 3 mo	5.4-8 sh. 2
F046	1	Barometric condenser vacuum pump discharge line check valve (h3)	2	A, C	I, A	L,S	RO	5.4-8 sh. 1
F047	1	Barometric condenser vacuum pump discharge line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 1
F048	1	Steam supply line warm-up line valve	1	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 2
F049	1	Steam supply line test line valve	2	B	P		E1	5.4-8 sh. 2
F050	1	Steam supply line test line valve	2	B	P		E1	5.4-8 sh. 2
F051	1	Turbine exhaust line drain line valve	2	B	P		E1	5.4-8 sh. 3
F052	1	Turbine exhaust line drain line valve	2	B	P		E1	5.4-8 sh. 3
F053	1	Turbine exhaust line test line valve	2	B	P		E1	5.4-8 sh. 1
F054	1	Turbine exhaust line vacuum breaker (h1)	2	C	A	R	RO	5.4-8 sh. 1
F055	1	Turbine exhaust line vacuum breaker (h1)	2	C	A	R	RO	5.4-8 sh. 1
F056	1	Steam supply line drain pot drain line test line valve	2	B	P		E1	5.4-8 sh. 1
F057	1	Steam supply line drain pot drain line test drain line	2	B	P		E1	5.4-8 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F059	1	Barometric condenser vacuum pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1
F500	1	Pump discharge line vent line valve	2	B	P		E1	5.4-8 sh. 1
F501	1	Pump discharge line vent line valve	2	B	P		E1	5.4-8 sh. 1
F502	1	Pump discharge line drain line valve	2	B	P		E1	5.4-8 sh. 1
F503	1	Pump discharge line drain line valve	2	B	P		E1	5.4-8 sh. 1
F700	1	Pump suction line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F701	1	Pump suction line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F702	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F703	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F704	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F705	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F706	1	Pump discharge line flow instrument root valve	2	B	P		E1	5.4-8 sh. 1
F707	1	Pump discharge line flow instrument root valve	2	B	P		E1	5.4-8 sh. 1
F708	1	Pump discharge line flow instrument root valve	2	B	P		E1	5.4-8 sh. 1
F709	1	Pump discharge line flow instrument root valve	2	B	P		E1	5.4-8 sh. 1

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F710	1	Pump discharge line pressure instrument root valve	2	B	P		E1	5.4-8 sh. 1
F711	1	Pump discharge line pressure instrument root valve	2	B	P		E1	5.4-8 sh. 1
F712	1	Turbine accessories cooling water line instrument root valve	2	B	P		E1	5.4-8 sh. 3
F713	1	Turbine accessories cooling water line instrument root valve	2	B	P		E1	5.4-8 sh. 3
F714	1	Turbine accessories cooling water line instrument root valve	2	B	P		E1	5.4-8 sh. 3
F716	1	Steam supply line pressure instrument root valve	2	B	P		E1	5.4-8 sh. 2
F717	1	Steam supply line pressure instrument root valve	2	B	P		E1	5.4-8 sh. 2
F718	1	Steam supply line drain pot instrument root valve	2	B	P		E1	5.4-8 sh. 2
F719	1	Steam supply line drain pot instrument root valve	2	B	P		E1	5.4-8 sh. 2
F720	1	Steam supply line drain pot instrument root valve	2	B	P		E1	5.4-8 sh. 2
F721	1	Steam supply line drain pot instrument root valve	2	B	P		E1	5.4-8 sh. 2
F722	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F723	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F724	1	Turbine exhaust pressure between rupture disk instrument root valve	2	B	P		E1	5.4-8 sh. 3
F725	1	Turbine exhaust pressure between rupture disk instrument root valve	2	B	P		E1	5.4-8 sh. 3
G31 Reactor Water Cleanup System Valves								

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F001	1	Line inside containment from RHR system maintenance valve	1	B	P		E1	5.4-12 sh. 1
F002	1	CUW System suction line inboard isolation valve	1	A	I,A	L,P S	RO 3 mo	5.4-12 sh. 1
F003	1	CUW System suction line outboard isolation valve	1	A	I,A	L,P S	RO 3 mo	5.4-12 sh. 1
F017	1	CUW System RPV head spray line outboard isolation valve (h3)	1	A	I,A	L,P S	RO CS	5.4-12 sh. 1
F018	1	CUW System RPV head spray line inboard check valve (h1)	1	A, C	I,A	L, S	RO	5.4-12 sh. 1
F019	1	CUW System bottom head drain line maintenance valve	1	B	P		E1	5.4-12 sh. 1
F026	1	CUW System suction line shutoff valve	1	B	P	P,S	RO	5.4-12 sh. 1
F050	1	Test line off the suction line outboard isolation valve G31-F003	2	B	P		E1	5.4-12 sh. 1
F058	1	Test line off RPV head spray line outboard isolation valve	2	B	P		E1	5.4-12 sh. 1
F060	1	RPV bottom head drain line sample line test line valve	2	B	P		E1	5.4-12 sh. 1
F070	1	RPV bottom head drain line sample line maintenance valve	2	B	P		E1	5.4-12 sh. 1
F071	1	RPV bottom head drain line sample line inboard valve	2	A	I,A	L,P S	RO 3 mo	5.4-12 sh. 1
F072	1	RPV bottom head drain line sample line outboard valve	2	A	I,A	L,P S	RO 3 mo	5.4-12 sh. 1
F500	1	CUW System bottom head drain line drain valve	2	B	P		E1	5.4-12 sh. 1
F501	1	CUW System bottom head drain line drain valve	2	B	P		E1	5.4-12 sh. 1
F700	2	CUW System suction line FE upstream instrument manual maintenance valve	2	B	P		E1	5.4-12 sh. 1

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F701	2	CUW System suction line FE downstream instrument manual maintenance valve	2	B	P		E1	5.4-12 sh. 1
F702	2	CUW System suction line FE upstream instrument excess flow check valve (h3)	2	A, C	I,A	L,S,P	RO	5.4-12 sh. 1
F703	2	CUW System suction line FE downstream instrument excess flow check valve (h3)	2	A, C	I,A	L,S,P	RO	5.4-12 sh. 1
G41 Fuel Pool Cooling and Cleanup Valves								
F015	2	FPC system heat exchanger outlet line maintenance valve	3	B	P		E1	9.1-1 sh. 2
F016	1	FPC system discharge line to spent fuel pool check valve	3	C	A	S	3 mo	9.1-1 sh. 2
F017	1	FPC system discharge line to spent fuel pool maintenance valve	3	B	P		E1	9.1-1 sh. 2
F018	1	FPC system discharge line to spent fuel pool check valve	3	C	A	S	3 mo	9.1-1 sh. 2
F019	2	FPC system discharge line to spent fuel pool valve	3	B	P		E1	9.1-1 sh. 1
F020	2	FPC system discharge line to spent fuel pool check valve	3	C	A	S	3 mo	9.1-1 sh. 1
F022	1	FPC system discharge line to reactor well maintenance valve	3	B	P		E1	9.1-1 sh. 2
F023	1	FPC system discharge line to reactor well check valve (h7)	3	C	A	S	RO	9.1-1 sh. 2
F091	1	FPC system supply line from SPCU check valve	3	C	A	S	3 mo	9.1-1 sh. 2
F093	1	FPC system RHR return line valve to FPC	3	B	P		E1	9.1-1 sh. 2
F094	1	FPC system RHR return line check valve to FPC (h7)	3	C	A	S	RO	9.1-1 sh. 2
F095	1	FPC system discharge line to spent fuel pool sample line	3	B	P		E1	9.1-1 sh. 2
F506	1	FPC system line valve from RHR-to-FPC line to LCW	3	B	P		E1	9.1-1 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
G51 Suppression Pool Cleanup System Valves								
F001	1	SPCU suction line inboard isolation valve	2	A	I,A	L,P S	RO 3 mo	9.5-1
F002	1	SPCU suction line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	9.5-1
F006	1	SPCU return line isolation valve	2	A	I,A	L, P S	RO 3 mo	9.5-1
F007	1	SPCU return line isolation valve	2	A	I,A	L,P S	RO 3 mo	9.5-1
K17 Radwaste System Valves								
F003	1	Drywell LCW sump pump inboard discharge line isolation valve	2	A	I,A	L,P S	RO 3 mo	11A.2-2 sh. 29
F004	1	Drywell LCW sump pump outboard discharge line isolation valve	2	A	I,A	L,P S	RO 3 mo	11A.2-2 sh. 29
F103	1	Drywell HCW sump pump inboard discharge line isolation valve	2	A	I,A	L,P S	RO 3 mo	11A.2-2 sh. 30
F104	1	Drywell HCW sump pump outboard discharge line isolation valve	2	A	I,A	L,P S	RO 3 mo	11A.2-2 sh. 30
P11 Makeup Water (Purified) System Valves								
F141	1	Outboard isolation valve	2	A	I,P	L	RO	9.2-5 sh. 2
F142	1	Inboard isolation valve	2	A, C	I,P	L	RO	9.2-5 sh. 2
P21 Reactor Building Cooling Water System Valves								
F001	6	Pump discharge line check valve	3	C	A	S	E2	9.2-1 sh. 1,4,7
F002	6	Pump discharge line maintenance valve	3	B	P		E1	9.2-1 sh. 1,4,7
F003	9	Heat exchanger inlet line valve	3	B	P		E1	9.2-1 sh. 1,4,7
F004	9	Heat exchanger outlet line MOV	3	B	P	P	2 yr	9.2-1 sh. 1,4,7
F005	3	Cold water line to hot/cold water blender	3	B	P		E1	9.2-1 sh. 1,4,7
F006	3	Hot/cold water blender valve—cold water	3	B	A	S	E2	9.2-1 sh. 1,4,7

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F007	3	Hot/cold water blender outlet line valve	3	B	P		E1	9.2-1 sh. 1,4,7
F008	3	Hot/cold water blender cold water bypass line	3	B	P		E1	9.2-1 sh. 1,4,7
F009	3	Hot water line to hot/cold water blender	3	B	P		E1	9.2-1 sh. 1,4,7
F010	3	Hot/cold water blender valve—hot water	3	B	A	S	E2	9.2-1 sh. 1,4,7
F011	3	Hot/cold water blender hot water bypass line	3	B	P		E1	9.2-1 sh. 1,4,7
F012	3	Cooling water supply line to RHR System maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F013	3	Cooling water return line from RHR System MOV	3	B	A	P S	2 yr 3 mo	9.2-1 sh. 2,5,8
F014	3	Cooling water return line from RHR Hx maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F015	6	Pump suction line maintenance valve	3	B	P		E1	9.2-1 sh. 1,4,7
F016	3	Surge tank outlet line to RCW pump suction	3	B	P		E1	9.2-1 sh. 2,5,8
F017	3	Surge tank makeup water line from SPCU	3	B	P		E1	9.2-1 sh. 2,5,8
F018	3	Surge tank makeup water line from SPCU	3	B	P	P	2 yr	9.2-1 sh. 2,5,8
F019	3	Surge tank makeup water from MUWP	3	B	P	P	2 yr	9.2-1 sh. 2,5,8
F020	3	Surge tank makeup water line from MUWP	3	B	P		E1	9.2-1 sh. 2,5,8
F021	2	Chemical addition tank inlet line valve	3	B	P		E1	9.2-1 sh. 1,4
F022	2	Chemical addition tank outlet line valve	3	B	P		E1	9.2-1 sh. 1,4
F024	6	Cooling water supply line to HECW refrigerator maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F025	6	Cooling water supply line to HECW refrigerator PCV	3	B	A	S	E2	9.2-1 sh. 2,5,8

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F026	6	Cooling water supply line to HECW refrigerator maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F027	6	Cooling water line to HECW refrigerator bypass line	3	B	P		E1	9.2-1 sh. 2,5,8
F028	6	Cooling water return line from HECW refrigerator	3	B	P		E1	9.2-1 sh. 2,5,8
F029	2	Cooling water supply line to FPC Hx	3	B	P		E1	9.2-1 sh. 2,5
F030	2	Cooling water return line from FPC Hx	3	B	P		E1	9.2-1 sh. 2,5
F031	2	Cooling water supply line to FPC pump room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F032	2	Cooling water return line from FPC pump room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F033	2	Cooling water line to PCV Atmospheric Monitoring System clr	3	B	P		E1	9.2-1 sh. 2,5
F034	2	Return line from PCV Atmospheric Monitoring System clr	3	B	P		E1	9.2-1 sh. 2,5
F035	2	Cooling water supply line to SGTS room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F036	2	Cooling water return line from SGTS room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F037	2	Cooling water supply line to FCS room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F038	2	Cooling water return line from FCS room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F039	3	Cooling water supply line to RHR equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2,5,8
F040	3	Cooling water return line from RHR equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2,5,8
F041	3	Cooling water supply line to RHR pump motor	3	B	P		E1	9.2-1 sh. 2,5,8
F042	3	Cooling water return line from RHR pump motor	3	B	P		E1	9.2-1 sh. 2,5,8
F043	3	Cooling water supply line to RHR pump mechanical seals	3	B	P		E1	9.2-1 sh. 2,5,8

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F044	3	Cooling water return line from RHR pump mechanical seals	3	B	P		E1	9.2-1 sh. 2,5,8
F045	1	Cooling water supply line to RCIC equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2
F046	1	Cooling water supply line from RCIC equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2
F047	2	Cooling water supply line to HPCF equipment room air conditioner	3	B	P		E1	9.2-1 sh. 5,8
F048	2	Cooling water supply line from HPCF equipment room air conditioner	3	B	P		E1	9.2-1 sh. 5,8
F049	2	Cooling water supply line to HPCF pump motor bearing	3	B	P		E1	9.2-1 sh. 5,8
F050	2	Cooling water return line from HPCF pump motor bearing	3	B	P		E1	9.2-1 sh. 5,8
F051	2	Cooling water supply line to HPCF pump mechanical seals	3	B	P		E1	9.2-1 sh. 5,8
F052	2	Cooling water return from HPCF pump mechanical seals	3	B	P		E1	9.2-1 sh. 5,8
F053	2	Surge tank outlet line to HECW System	3	B	P		E1	9.2-1 sh. 2,5,8
F055	6	Cooling water return line from Emergency Diesel Generator	3	B	A	P S	2 yr 3 mo	9.2-1 sh. 5,8
F056	3	Cooling water return line from Emergency Diesel Generator maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F057	2	Cooling water line to PCV Atmospheric Monitoring System air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F058	2	Return line from PCV Atmospheric Monitoring System air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F061	3	Cooling water line Emergency Diesel Generators	3	B	P		E1	9.2-1 sh. 2,5,8
F071	6	Cooling water supply line–to non-essential coolers	3	B	P		E1	9.2-1 sh. 2,5,8
F072	6	Cooling water supply line–to non-essential coolers	3	B	A	P S	2 yr 3 mo	9.2-1 sh. 2,5,8

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F075	2	Cooling water supply line to PCV outboard isolation valve (h3)	2	A	I,A	L,P S	RO, CS	9.2-1 sh. 3,6
F076	2	Cooling water supply line to PCV inboard check isolation valve (h1)	2	A, C	I,A	L,S	RO	9.2-1 sh. 3,6
F080	2	Cooling water return line from PCV inboard isolation valve (h1)	2	A	I,A	L,P,S	RO	9.2-1 sh. 3,6
F081	2	Cooling water return line from PCV outboard isolation valve (h3)	2	A	I,A	L,P S	RO, CS	9.2-1 sh. 3,6
F083	3	Cooling water return line from non-essential coolers (h4)	3	C	A	S	RO	9.2-1 sh. 2,5,8
F084	3	Cooling water return line from containment bypass line	3	B	P		E1	9.2-1 sh. 2,5,8
F175	3	Cooling water supply to RHR System Hx pressure relief valve	3	C	A	R	5 yr	9.2-1 sh. 2,5,8
F195	2	Cooling water supply line to FPC heat exchanger	3	B	A	P S	2 yr 3 mo	9.2-1 sh. 2,5
F220	9	Bypass line around RCW System outlet line MOV	3	B	P		E1	9.2-1 sh. 1,4,7
F251	2	Cooling water supply line to PCV test line	2	B	P		E1	9.2-1 sh. 3,6
F252	2	Cooling water return line from PCV test line	2	B	P		E1	9.2-1 sh. 3,6
F501	9	Heat exchanger shell side vent line	3	B	P		E1	9.2-1 sh. 1,4,7
F502	9	Heat exchanger shell side drain line	3	B	P		E1	9.2-1 sh. 1,4,7
F503	3	Surge tank drain line to SD	3	B	P		E1	9.2-1 sh. 2,5,8
F601	3	Cooling water supply line to RHR System drain line to SD	3	B	P		E1	9.2-1 sh. 2,5,8
F602	3	Cooling water supply line to RHR System drain line to HCW	3	B	P		E1	9.2-1 sh. 2,5,8
F603	3	Cooling water return line from RHR Hx drain line to SD	3	B	P		E1	9.2-1 sh. 2,5,8
F604	3	Cooling water return line from RHR Hx drain line to HCW	3	B	P		E1	9.2-1 sh. 2,5,8

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F701	6	Pump discharge line pressure instrument line	3	B	P		E1	9.2-1 sh. 1,4,7
F702	9	Hx discharge line sample line valve	3	B	P		E1	9.2-1 sh. 1,4,7
F703	3	Cooling water supply line pressure instrument line	3	B	P		E1	9.2-1 sh. 1,4,7
F704	3	Cooling water supply line sample valve	3	B	P		E1	9.2-1 sh. 1,4,7
F705	3	Cooling water supply line elbow tap instrument root valve	3	B	P		E1	9.2-1 sh. 1,4,7
F706	3	Cooling water supply line elbow tap instrument root valve	3	B	P		E1	9.2-1 sh. 1,4,7
F707	3	Cooling water supply line to RHR System FT instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F708	3	Cooling water supply line to RHR System FT instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F709	3	Cooling water return line from RHR Hx sample valve	3	B	P		E1	9.2-1 sh. 2,5,8
F710	6	Pump suction line PX instrument root valve	3	B	P		E1	9.2-1 sh. 1,4,7
F711	6	Pump suction line pressure instrument root valve	3	B	P		E1	9.2-1 sh. 1,4,7
F712	3	Surge tank level instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F713	3	Surge tank level instrument line root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F714	3	Surge tank level instrument line root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F717	3	Cooling water line to DG instrument line	3	B	P		E1	9.2-1 sh. 2,5,8
F718	3	Return water line from DG instrument line	3	B	P		E1	9.2-1 sh. 2,5,8
F719	3	Cooling water line to DG instrument line	3	B	P		E1	9.2-1 sh. 2,5,8
F720	3	Return water line from DG instrument line	3	B	P		E1	9.2-1 sh. 2,5,8

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F721	3	Cooling water supply line to non-essential coolers FT instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F722	3	Cooling water supply line to non-essential coolers FT instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
P24 HVAC Normal Cooling Water System Valves								
F053	1	HNCW supply line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	9.2-2
F054	1	HNCW supply line inboard isolation check valve (h1)	2	A, C	I,A	L,S	RO	9.2-2
F141	1	HNCW return inboard isolation valve (h1)	2	A	I,A	L,P,S	RO	9.2-2
F142	1	HNCW return outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	9.2-2
P25 HVAC Emergency Cooling Water System Valves								
F001	6	Pump discharge line check valve	3	C	P	S	E2	9.2-3 sh. 1,2,3
F002	6	Pump discharge line maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F003	6	Refrigerator outlet line maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F004	2	Maintenance valve at HECW supply to MCR cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F005	2	HECW supply to MCR cooler Temperature Control Valve (TCV)	3	B	A	S	E2	9.2-3 sh. 1,2,3
F006	2	Maintenance valve at HECW supply to MCR cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F007	6	Maintenance valve at HECW supply to MCR cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F008	6	Maintenance valve at HECW return from MCR cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F009	6	Pump suction line maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F010	2	TCV bypass at HECW discharge to MCR cooler	3	B	P		E1	9.2-3 sh. 1,2,3

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F011	3	Pump suction line/discharge line PCV maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F012	3	Pump suction line/discharge line PCV	3	B	A	S	E2	9.2-3 sh. 1,2,3
F013	3	Pump suction line/discharge line PCV maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F014	3	Pump suction line/discharge line PCV bypass line	3	B	P		E1	9.2-3 sh. 1,2,3
F015	3	Maintenance valve at HECW supply to C/B Essential Electrical Equipment Room Cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F016	3	HECW supply to C/B Essential Electrical Equipment Room cooler TCV	3	B	A	S	E2	9.2-3 sh. 1,2,3
F017	3	Maintenance valve at HECW supply to C/B Essential Electrical Equipment Room Cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F018	6	HECW supply to C/B Essential Electrical Equipment Room cooler maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F019	6	Maintenance valve at HECW return from C/B Essential Electrical Equipment Room Cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F020	3	TCV bypass valve at HECW supply to C/B Essential Electrical Equipment Room cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F021	3	Maintenance valve at HECW supply to DG zone cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F022	3	HECW supply to DG zone cooler TCV	3	B	A	S	E2	9.2-3 sh. 1,2,3
F023	3	Maintenance valve at HECW supply to DG zone cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F024	6	Maintenance valve at HECW supply to DG zone cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F025	6	Maintenance valve at HECW return from DG zone cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F026	3	TCV bypass valve at HECW supply to DG zone cooler	3	B	P		E1	9.2-3 sh. 1,2,3

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F030	3	Chemical addition tank return valve from HECW	3	B	P		E1	9.2-3 sh. 1,2,3
F031	3	Chemical addition tank feed valve to HECW	3	B	P		E1	9.2-3 sh. 1,2,3
F050	2	Make-up Water Purified (MUWP) line to pump suction check valve	3	C	A	S	E2	9.2-3 sh. 1,2,3
F070	6	Pump discharge line drain valve	3	B	P		E1	9.2-3 sh. 1,2,3
F400	6	Pump drain line valve	3	B	P		E1	9.2-3 sh. 1,2,3
F401	6	Pump bearing cooling water needle valve	3	B	P		E1	9.2-3 sh. 1,2,3
F402	3	Refrigerator outlet line sample line valve	3	B	P		E1	9.2-3 sh. 1,2,3
F700	6	Pump discharge line pressure instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3
F701	6	FE P25-FE003 upstream instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3
F702	6	FE P25-FE003 downstream instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3
F703	6	Pump suction pressure instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3
F704	6	Pump suction/discharge line Δp instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
P41 Reactor Service Water System Valves								
F001	6	Pump discharge line check valve	3	C	A	S	E2	9.2-7 sh. 1,2,3
F002	6	Pump discharge line maintenance valve	3	B	P		E1	9.2-7 sh. 1,2,3
F003	9	Service water inlet valve to RCW System heat exchanger	3	A	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F004	6	Service water inlet valve to service water strainer	3	B	P	P	2 yr	9.2-7 sh. 1,2,3
F005	9	Service water outlet valve from RCW heat exchanger	3	A	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F006	6	Service water strainer blowout valve	3	B	P	P	2 yr	9.2-7 sh. 1,2,3
F007	9	Supply line from Potable Water check valve	3	C	P		E1	9.2-7 sh. 1,2,3
F008	9	Supply line from Potable Water check valve	3	C	P		E1	9.2-7 sh. 1,2,3
F009	9	Supply valve from Potable Water System	3	B	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F010	9	RCW Hx tube side (service water side) relief valve	3	C	P	R	5 yr	9.2-7 sh. 1,2,3
F011	9	Bypass line around RCW Hx outlet line outlet valve MOV P41-F005	3	B	P		E1	9.2-7 sh. 1,2,3
F012	9	Service water sampling valve	3	B	P		E1	9.2-7 sh. 1,2,3
F013	6	Service water strainer outlet valve	3	A	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F014	3	Common service water strainer outlet valve	3	A	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F015	3	Discharge line to discharge canal MOV	3	A	A	P S	E1 E2	9.2-7 sh. 1,2,3
F501	9	RCW Hx shell side drain valve to SWSD	3	B	P		E1	9.2-7 sh. 1,2,3
F502	9	RCW Hx shell side vent valve to SWSD	3	B	P		E1	9.2-7 sh. 1,2,3
F503	9	RCW Hx shell side drain valve to SWSD	3	B	P		E1	9.2-7 sh. 1,2,3

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F504	9	RCW Hx shell side vent valve to SWSD	3	B	P		E1	9.2-7 sh. 1,2,3
F701	6	Pump discharge line pressure instrument line	3	B	P		E1	9.2-7 sh. 1,2,3
F702	3	Service water supply pressure instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
F703	6	ΔP across service water strainer upstream instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
F704	6	ΔP across service water strainer downstream instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
F705	9	Service water ΔP across RCW Hx upstream instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
F706	9	Service water ΔP across RCW Hx downstream instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
P51 Service Air System Valves								
F131	1	Outboard isolation manual valve	2	A	I,P	L	RO	9.3-7
F132	1	Inboard isolation manual valve	2	A	I,P	L	RO	9.3-7
P52 Instrument Air System Valves								
F276	1	Outboard isolation valve (h3)	2	A	I,A	L,P,S	RO	9.3-6
F277	1	Inboard isolation check valve (h3)	2	A,C	I,A	L,P,S	RO	9.3-6
P54 High Pressure Nitrogen Gas Supply System Valves								
F002	4	Nitrogen bottles N_2 supply line valve	3	B	P		E1	6.7-1
F003	2	Nitrogen bottles N_2 supply line MOV	3	B	A	P S	2 yr 3 mo	6.7-1
F004	2	N_2 bottle supply line PCV maintenance valve	3	B	P		E1	6.7-1
F005	2	N_2 bottle supply line PCV	3	B	A		E1	6.7-1
F006	2	N_2 bottle supply line PCV maintenance valve	3	B	P		E1	6.7-1
F007	2	Safety grade N_2 supply line isolation valve	2	A	I,A	L,P S	RO 3 mo	6.7-1
F008	2	Safety grade N_2 supply line isolation check valve (h1)	2	A,C	I,A	L, S	RO	6.7-1
F009	8	Safety grade N_2 supply line to SRV	3	B	P		E1	6.7-1

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F010	2	Bypass line around the N_2 bottle supply line PCV	3	B	P		E1	6.7-1
F011	2	N_2 bottle supply line relief valve	3	C	A	R	5 yr	6.7-1
F012	2	MOV at safety/non-safety boundary	3	A	A	P S	2 yr 3 mo	6.7-1
F200	1	Non-safety N2 supply line isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.7-1
F209	1	Non-safety N2 supply line isolation check valve	2	A,C	I,A	L,S	RO	6.7-1
T22 Standby Gas Treatment System Valves								
F001	2	Fuel handling floor inlet butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 1
F002	2	Filter train inlet butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 1
F003	2	Filter train exhaust gravity damper	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F004	2	Filter train exhaust butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F005	2	Cooling fan butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F006	2	Filter train R112 injection line valve	3	B	P		E1	6.5-1 sh. 2,3
F007	2	Filter train DOP injection line valve to pre HEPA filter	3	B	P		E1	6.5-1 sh. 2,3
F008	2	Filter train DOP sampling line valve downstream of pre HEPA	3	B	P		E1	6.5-1 sh. 2,3
F009	2	Filter train DOP sampling line valve downstream of pre HEPA	3	B	P		E1	6.5-1 sh. 2,3
F010	2	Filter train DOP injection line valve downstream of charcoal absorbent	3	B	P		E1	6.5-1 sh. 2,3
F011	2	Filter train DOP sampling line valve downstream of charcoal absorbent	3	B	P		E1	6.5-1 sh. 2,3
F012	2	Filter train DOP sampling line valve downstream of after HEPA	3	B	P		E1	6.5-1 sh. 2,3
F014	2	SGTS sample line valve	3	B	P		E1	6.5-1 sh. 2,3

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F015	2	PRM discharge to stack valve	3	B	P		E1	6.5-1 sh. 2,3
F500	2	Filter unit vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F501	2	Filter unit drain line valve	3	B	P		E1	6.5-1 sh. 2,3
F504	2	Filter unit vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F505	2	Exhaust fan vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F506	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F507	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F508	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F509	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F510	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F511	2	Exhaust stack drain line valve	3	B	P		E1	6.5-1 sh. 2,3
F700	2	Filter unit demister dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F701	2	Filter unit demister dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F705	2	Filter train prefilter dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F706	2	Filter train prefilter dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F707	2	Filter train preHEPA dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F708	2	Filter train preHEPA dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F709	2	Filter train charcoal absorber dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F710	2	Filter train charcoal absorber dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F711	2	Filter train after HEPA dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F712	2	Filter train after HEPA dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F713	2	Filter train exhaust flow instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F714	2	Filter train exhaust flow instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
T31 Atmospheric Control System Valves								
F001	1	Purge supply line outboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F002	1	Drywell purge line supply inboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F003	1	Wetwell purge supply line inboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F004	1	Drywell purge exhaust line inboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F005	1	Drywell purge exhaust line bypass line valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F006	1	Wetwell purge exhaust line inboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F007	1	Wetwell overpressure line valve (h2)	2	A	I,P	L, P S	2 yr RO	6.2-39 sh. 1
F008	1	Containment exhaust line to SGTS (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F009	1	Containment exhaust line to R/B HVAC (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F010	1	Wetwell overpressure line valve (h2)	2	A	I,P	L, P S	2 yr RO	6.2-39 sh. 1
F011	1	Containment exhaust line to SGTS (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F025	1	Purge supply line from outboard containment isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F039	1	N_2 makeup line from outboard containment isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F040	1	N_2 makeup line from to drywell inboard isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F041	1	N_2 makeup line from to wetwell inboard isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F044	8	Drywell/wetwell vacuum breaker valve	2	C	A	P,R	RO	6.2-39 sh. 2
F050	1	Purge supply line from test line valve	2	B	P		E1	6.2-39 sh. 1
F051	1	Purge exhaust line test line valve	2	B	P		E1	6.2-39 sh. 1
F054	1	Makeup line test line valve	2	B	P		E1	6.2-39 sh. 1
F055	1	Drywell personnel air lock hatch test line valve	2	B	P		E1	6.2-39 sh. 2
F056	1	Wetwell personnel air lock hatch test line valve	2	B	P		E1	6.2-39 sh. 2
F057	1	Overpressure protection test line valve	2	B	P		E1	6.2-39 sh. 1
F058	1	Overpressure protection test line valve	3	B	P		E1	6.2-39 sh. 1
F059	1	Overpressure protection test line valve	3	B	P		E1	6.2-39 sh. 1
F700	1	FE instrument line valve	2	B	P		E1	6.2-39 sh. 1
F701	1	FE instrument line valve	2	B	P		E1	6.2-39 sh. 1
F702	1	FE instrument line valve	2	B	P		E1	6.2-39 sh. 1
F703	1	FE instrument line valve	2	B	P		E1	6.2-39 sh. 1
F730	1	Drywell pressure instrument line isolation valve	2	B	P		E1	6.2-39 sh. 2
F731	1	Drywell pressure instrument line isolation valve	2	A	I,P	L,P	RO	6.2-39 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F732	2	Drywell pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F733	2	Drywell pressure instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F734	4	Drywell pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F735	4	Drywell pressure instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F736	2	Wetwell pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F737	2	Wetwell pressure instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F738	4	Suppression pool water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F739	4	Suppression pool water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F740	4	Suppression pool water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F741	4	Suppression pool water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F742	2	Suppression pool water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F743	2	Suppression pool water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F744	2	Suppression pool water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F745	2	Suppression pool water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F800	2	Drywell water level instrument line isolation valve	2	B	P		E1	6.2-39 sh. 2
F801	2	Drywell water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F802	2	Drywell water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F803	2	Drywell water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F804	2	DW/WW differential pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F805	2	DW/WW differential pressure instrument isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
D001	1	Wetwell overpressure rupture disk	2	D	I,P	Rplc.	5 yr	6.2-39 sh. 1
D002	1	Wetwell rupture disk	2	D	I,P	Rplc.	5 yr	6.2-39 sh. 1
T49 Flammability Control System Valves								
F001	2	Inlet line from drywell inboard isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-40
F002	2	Inlet line from drywell outboard isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-40
F003	2	Flow control valve for the FCS inlet line from drywell	3	B	A	P S	2 yr 3 mo	6.2-40
F004	2	Blower bypass line flow control valve	3	B	A	P S	2 yr 3 mo	6.2-40
F005	2	Blower discharge line to wetwell check valve (h9)	3	C	A	S	RO	6.2-40
F006	2	Discharge line to wetwell outboard isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-40
F007	2	Discharge line to wetwell inboard isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-40
F008	2	Cooling water supply line from the RHR System MOV	3	B	A	P S	2 yr 3 mo	6.2-40
F009	2	Cooling water supply line maintenance valve	3	B	P		E1	6.2-40
F010	2	Cooling water supply line admission MOV	3	B	A	P S	2 yr 3 mo	6.2-40
F013	2	Inlet line from drywell drain line valve	3	B	P		E1	6.2-40
F014	2	Blower drain line valve	3	B	P		E1	6.2-40
F015	1	Blower discharge line to wetwell pressure relief valve	2	A,C	I,A	R L	5 yr RO	6.2-40
F016	2	Blower discharge line to wetwell pressure relief line check valve (h3)	2	A,C	I,A	L, S	RO	6.2-40
F501	2	Inlet line from drywell test line valve	2	B	P		E1	6.2-40

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F502	2	Discharge line to wetwell test line valve	2	B	P		E1	6.2-40
F504	2	Blower suction line test line valve	3	B	P		E1	6.2-40
F505	2	Blower discharge line test line valve	3	B	P		E1	6.2-40
F506	2	Drain line to low conductivity waste (LCW) valve	3	B	P		E1	6.2-40
F507	2	Cooling water supply line test line valve	3	B	P		E1	6.2-40
F701	2	FE T49-FE002 upstream instrument line root valve	3	B	P		E1	6.2-40
F702	2	FE T49-FE002 downstream instrument line root valve	3	B	P		E1	6.2-40
F703	2	Blower suction line pressure instrument line root valve	3	B	P		E1	6.2-40
F704	2	FE T49-FE004 upstream instrument line root valve	3	B	P		E1	6.2-40
F705	2	FE T49-FE004 downstream instrument line root valve	3	B	P		E1	6.2-40
U41 Heating, Ventilating and Air Conditioning System Valves								
F001	2	Secondary containment supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F002	2	Secondary containment exhaust isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F003	3	Secondary Containment divisional supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F004	3	Secondary Containment divisional exhaust isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F007	4	MCR area HVAC bypass line isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
F008	4	MCR area HVAC supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
F009	4	MCR area HVAC emergency HVAC supply	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
F010	4	MCR area HVAC exhaust isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
Y52 Oil Storage Transfer System Valves								

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F001	6	D/G transfer pump discharge line check valve	3	C	A	S	3 mo	9.5-6
F002	3	D/G transfer pump discharge line relief valve	3	C	A	R	5 yr	9.5-6
F003	3	D/G transfer pump discharge line ball (plug) valve	3	B	P		E1	9.5-6
F004	3	D/G fuel oil day tank return to storage tank valve	3	B	P		E1	9.5-6
F501	3	D/G transfer pump discharge line drain valve	3	B	P		E1	9.5-6
F502	3	D/G transfer pump discharge line vent valve	3	B	P		E1	9.5-6

Notes:

(a) 1, 2, or 3—Safety Classification, Subsection 3.2.3.

(b) Pump test parameters per ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM--1987, Part 6:
N - Speed
Pd- Discharge Pressure
Pi - Inlet Pressure
Q - Flow Rate
Vd -Peak-to-peak vibration displacement
Vv -Peak vibration velocity

(c) A, B, C or D—Valve category per ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1 and 10.

(d) Valve function:
I - Primary containment isolation, Subsection 6.2.4
A or P -Active or passive per ASME Code in (c) above (Part 10, Paragraph 1.3).

(e) Valve test parameters per ASME Code in (c) above:
L - Leakage rate (Part 10, Paragraph 4.2.2, Tier 2 Table 6.2-7 for valves with function I in (d) above)

P - Local position verification (Part 10, Paragraph 4.1)
R - Relief valve test including visual examination, set pressure and seat tightness testing (Part 10, Paragraph 4.3.1 and Part 1, Paragraph 1.3.3 and 1.3.4)
S - Stroke exercise Category A or B (Part 10, Paragraphs 4.2.1.1, 4.2.1.2)
Category C (Part 10, Paragraphs 4.3.2.1, 4.3.2.2, 4.3.2.4)
X - Explosive charge test (Part 10, Paragraph 4.4.1)

(f) Pump or valve test exclusions, alternatives and frequency per ASME code in (b) or (c) above or Appendix I:
CS -Cold shutdown
RO - Refueling outage and/or no case greater than two years.
E1 -Used for operating convenience (i.e., passive vent, drain, instrument test, maintenance valves, or a system control valve). Tests are not required (Part 10, Paragraph 1.2).
E2 -In regular use. Test frequency is not required provided the test parameters are analyzed and recorded at an operation interval not exceeding three months.
Category A or B, Stroke (Part 10, Paragraph 4.2.1.5).
Category C, Stroke (Part 10, Paragraph 4.3.2.3).
E3 -Operability test every six months. Set pressure and leak test every refueling outage. (Part 1, Paragraph 1.3.4.3).
E10 -In regular use. Test frequency is not required provided the test parameters are recorded at least once every three months of operation (Part 6, Paragraph 5.3)
E11 -Lacking required fluid inventory. Test shall be performed at least once every two years with required fluid inventory provided (Part 6, Paragraph 5.5).

(g) Piping and instrument symbols and abbreviations are defined in Figure 1.7-1. Figure page numbers are shown in parenthesis.

(h) Reasons for code defined testing exceptions (Part 10, Paragraphs 4.2.1.2, 4.3.2.2).
(h1) Inaccessible inerted containment and/or steam tunnel radiation during power operations.
(h2) Avoids valve damage and impacts on power operations.
(h3) Avoids impacts on power operations.
(h4) A temporary crosstie is necessary to carry the ongoing cooling loads. A permanent crosstie would violate divisional separation.
(h5) Avoids cold/hot water injection to RPV during power operations.
(h6) Maintain pressure isolation during normal operation.
(h7) Inventory available only during refueling outage.
(h8) Not Used
(h9) Test connection size is insufficient for full flow test during operation. Therefore, test part stroke during plant operation and full stroke during refueling outage. A test

connection size which would be sufficient for full flow tests would pressurize the secondary containment beyond specified limits, thus affecting power operation.

(i) Summary justification for code exemption request (Part 6, Paragraph 5.2, or Part 10, Paragraph 6.2).

(i1) The piping is maintained full by a small fraction of the pump's flow capacity. These pumps may be a constant speed centrifugal type with a cooling by-pass loop. Normal operation will be near minimum flow in the flat or constant region of the pressure/flow performance curve. Therefore, a flow measurement would not be useful. The pumps will be designed and analyzed to withstand low flow operation without significant degradation.

Table 3.9-9 Reactor Coolant System Pressure Isolation Valves

Standby Liquid Control System	
C41-F006 A,B	Injection Valves
C41-F008	Inboard Check Valve
Residual Heat Removal System	
E11-F005 A,B,C	Injection Valve Loops A,B&C
E11-F006 A,B,C	Testable Check Valve A,B&C
E11-F010 A,B,C	Shutdown Cooling Inboard Suction Isolation Valve Loops A,B&C
E11-F011 A,B,C	Shutdown Cooling Outboard Suction Isolation Valve Loops A,B&C
High Pressure Core Flooder System	
E22-F003 B,C	Injection Valve Loops B&C
E22-F004 B,C	Testable Check Valve Loops B&C
Reactor Core Isolation Cooling System	
E51-F004	Injection Valve
E51-F005	Testable Check Valve

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds

Component	Weld Type	NDE Requirements
Class 1 Components (1) (2) (3)		
Vessel	Category A (Longitudinal)	RT plus MT or PT
Vessel, Pipe, Pump, Valve	Category B (Circumferential	RT plus MT or PT
Pipe, Pump, Valve	Butt weld Fillet and socket welds	RT plus MT or PT MT or PT
Vessels (9)	Category C and similar welds	RT plus MT or PT, RT must be multiple exposure
	Partial penetration and fillet welds	MT or PT on all accessible surfaces
Vessels (9) & Branched Connections	Category D	
	a) Butt welds, all	RT plus MT or PT
	b) Corner welded nozzles	RT plus MT or PT
	c) Corner welded branch and piping connection exceeding 100A nominal diameter	RT plus MT or PT
	d) Corner welds branch and piping 100A and less	MT or PT
	e) Weld buildup deposits at openings	UT plus a, b, c above if connected to nozzle or pipe
	f) Partial penetration	MT or PT progressive and final surface
	g) Oblique full penetration branch and piping connections	RT or UT plus MT or PT, In addition, UT of weld, fusion zone, and parent metal beneath attachment surface.
General	Fillet, partial penetration, socket welds	MT or PT
General	Structural attachment welds	MT or PT
Special Welds	1) Specially designed seals	MT or PT
	2) Weld metal cladding	PT
	3) Hard surfacing	PT
	a) Valves 100A or less	None
	4) Tube-tube sheet welds	PT
	5) Brazed joints	VT

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Class 2 Components (1) (2) (4)		
Vessel	Category A (Longitudinal) a) Either of the members exceeds 4.8 mm b) Each member 4.8 mm or less	 RT MT, PT, or RT
Pipe, Pump, Valve	Longitudinal	RT
Vessel	Category B (Circumferential a) Either of the members exceeds 4.8 mm b) Each member 4.8 mm or less	 RT MT, PT, or RT
Pipe, Pump, Valve	Circumferential a) Either of the members exceeds 4.8 mm b) Each member 4.8 mm or less	 RT MT or PT
Vessels (9) and Similar Joints in Other Components	Category C a) Corner joints, either of the members exceeds 4.8 mm b) Each member 4.8 mm or less c) Partial penetration and fillet welds	 RT MT, PT, or RT MT or PT
Vessels (9) and Similar Welds in Other Components	Category D a) Full penetration joints when either members exceed 4.8 mm of thickness b) Full penetration corner joints when either member exceeds 4.8 mm c) Both members 4.8 mm or less d) Partial penetration and fillet weld joints	 RT MT or PT MT or PT MT or PT
Branch Con. and Nozzles in Pipe, Valve Pump	a) Nominal size exceed 100A b) Nominal size100A or smaller	RT MT or PT (External and accessible internal surfaces)

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Vessels Designed to NC-3200	Category A	RT
	Category B	RT
	Category C, Butt weld	RT
	Category C, Full penetration corner	UT or RT
	Category C, Partial penetration corner and fillet welds	MT or PT both sides
	Category D, Full penetration (6)	
	Category D, Partial penetration	
	Fillet, partial penetration, socket, and	RT
	structural attachment welds	MT or PT both sides
		MT or Pt
Special Welds	a) Specially designed seals	MT or PT
	b) Weld metal cladding	MT or PT
	c) Hard surfacing	PT
	d) Hard surfacing for valves with inlet connection 100A nominal pipe size or less	None
	e) Tube-tube sheet welds	PT
	f) Brazed joints	VT
Storage Tanks (Atmospheric)	a) Side joints	RT
	b) Roof and roof-to-sidewall	VT
	c) Bottom joints	Vacuum box testing of at least 20.6 kPaG
	d) Bottom to sidewall	Vacuum box plus MT or PT
	e) Nozzle to tank side	MT or PT
	f) Nozzle to roof	VT
	g) Joints in nozzles	RT
	h) Others	Similar welds in vessels
Storage Tanks (0-103.42 kPaG)	a) Sidewall	RT
	b) Roof	RT
	c) Roof-to-sidewall	RT, if not possible, MT or PT
	d) Bottom & bottom-to-side	Vacuum box method plus MT or PT
	e) Nozzle tank	MT or PT
	f) Joints to nozzles	RT
	g) Others	Same as similar vessel joints

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Class 3 Components (1) (2) (5)		
Vessels	Category A (Longitudinal)	
	1) a) Thickness exceeding the limits of Table ND.5211.2-1	RT
	b) Welds based on joint efficiency permitted by ND.3351.1	RT
	c) Butt welds in nozzles attached to vessels in a or b above	RT
	2. Welds not included in 1 above	Spot RT each 15.24m of weld. Additional RT to cover each welders work.
	3. Nonferrous vessels exceeding 9.5 mm	RT
Pump, Valve, Pipe	Pipes greater than 50A nominal size	RT, MT, or PT
	Pumps & valves greater than 50A nominal	According to the product form
Vessel	Category B (Circumferential)	
	1) a) Thickness exceeds Table ND.5211.2 for ferrous metals	RT
	b) Thickness exceeds 9.5 mm for nonferrous metals	RT
	c) Joint efficiency according to ND.3352.1(a)	RT
	d) Attachments to vessels and exceeds nominal pipe size 254 mm or thickness 28.6 mm	RT
	2. Welds not involved in 1 above	RT 152 mm long sections plus the intersections of Category A welds
Pipe, Pump and Valve	Greater than 50A nominal pipe size	RT, PT, or MT
Vessel	Category C	
	1) a) Thickness exceeds Table ND.5211.2 or ND-5211.3	RT
	b) Attachments exceed 250A NPS or 28.6 mm wall thickness	RT
	2) Welds not involved in 1 above	Spot RT to cover each welders work
Pipe, Pump, Valves	Greater than 50A nominal pipe size	RT, PT, or MT

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Vessel	Category D	
	1) Full penetration butt welds designed for joint efficiency per ND.3352.1(a)	RT
	2) In nozzles or communicating chambers attached to vessels or heads requiring full RT	RT
	3) Welds not covered by 1 and 2 above	Spot RT to cover each welders work
Pipe, Pump and Valve	Greater than 50A nominal pipe size	RT, PT, or MT
Special Welds	a) Weld metal cladding	PT
	b) Hard surfacing	PT
	i) Hard surfacing for valves with inlet connection 100A nominal pipe size or less	None
	c) Tube-tube sheet welds	PT
	d) Brazed joints	VT
Storage Tanks (Atmospheric)	a) Sidewall joints	Same as Category A or B vessel joints
	b) Roof and roof-to-sidewall	VT
	c) Bottom joints	Vacuum box testing of at least 20.6 kPaG, or PT or MT plus VT during pressure test
	d) Bottom to sidewall	Same as bottom joints
	e) Nozzle to tank side	MT or PT
	f) Nozzle to roof	VT
	g) Joints in nozzles ex. roof nozzles	MT or PT
	h) Others	Similar welds in vessels
Storage Tanks (0–103.42 kPaG)	a) Sidewall	Same as Category A or B vessel joints
	b) Roof	Same as Category A vessel joints
	c) Roof-to-sidewall	Same as above, if possible, or MT or PT
	d) Bottom to bottom-to-side	Vacuum box testing at least 20.6 kPaG, or PT or MT plus VT during pressure test
	e) Nozzle to tank	MT or PT
	f) Joints in nozzles	MT or PT
	g) Others	Same as similar vessel joints

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Components Supports (1) (2) (7)		
Class 1 Supports	Primary member, full penetration butt welds All other welds Secondary member welds	RT MT or PT VT
Class 2 and MC Supports	Primary member, full penetration butt welds Partial penetration or fillet welds throat greater than 25.4 mm All other welds Secondary member welds	RT MT or PT VT VT
Class 3 Supports	Primary member, groove or throat greater than 25.4 mm All other welds Secondary member welds	MT or PT VT VT
Special Requirements, All Classes	Welds transmitting loads in the through thickness direction in members greater than 25.4 mm	UT base metal beneath the weld
Core Support Structures (1) (2) (8)		
Core Support Structures (Provide direct support or restraint of the fuel, etc. under normal operating conditions.)	Category A, longitudinal butt welds Category B, circumferential butt welds Category C, flange to shell welds Category D, nozzle to shell welds Category E, beam end connections to other structures Repair welds under 9.5 mm or 10% deep Repair welds over 9.5 mm or 10% deep	Examination may be by any technique or certain combinations of techniques, from simple VT to MT or PT plus RT or UT. Quality factor *n* and fatigue factor *f* are dependent on the technique(s) selected, in accordance with Table NG-3352-1. MT or PT MT or PT plus RT or UT
Internal Structures (Can be any other structure within the reactor vessel.) nonmandatory	Same as above	Same as above
Temporary Attachments (Removed before operation.)	All	MT or PT

NOTES:

(1) The required confirmation that facility welding activities are in compliance with the certified commitments will include the following third-party verifications:

(a) Facility welding and the applicable ASME Code requirements

(b) Facility welding activities are performed with the applicable ASME Code requirements;

(c) Welding activities related records are prepared, evaluated and maintained in accordance withers the ASME Code Requirements.

(d) Welding processes used to weld dissimilar base metal and welding filler metal combinations for the intended applications.

(e) The facility has established procedures for qualifications of welders and welding operators in accordance with the applicable ASME Code requirements.

(f) Approved procedures are available and used for preheating and post heating of welds, and those procedures meet the applicable requirements of the ASME Code requirements.

(g) Completed welds are examined in accordance with the applicable examination method required by the ASME Code.

(2) Radiographic film will be reviewed and accepted by the licensee's nondestructive examination (NDE), Level III examiner prior to final acceptance.

(3) The NDE requirements for Class 1 components will be as stated in subarticle NB-5300 of Section III of the ASME Code.

(4) The NDE requirements for Class 2 components will be as stated in subarticle NC-5300 of Section III of the ASME Code.

(5) The NDE requirements for Class 3 components will be as stated in subarticle ND-5300 of Section III of the ASME Code.

(6) Deleted

(7) The NDE requirements for component supports will be as stated in subarticle NF-5300 of Section III of the ASME Code.

(8) The NDE requirements for Core Support structures will be as stated in subarticle NG-5300 of Section III of the ASME Code.

(9) For corner joints UT may be used instead of RT. For Type 2 full penetration corner weld joints, if RT is used, the fusion zone, and parent metal beneath the attachment surface shall be UT examined after welding.

LEGEND:

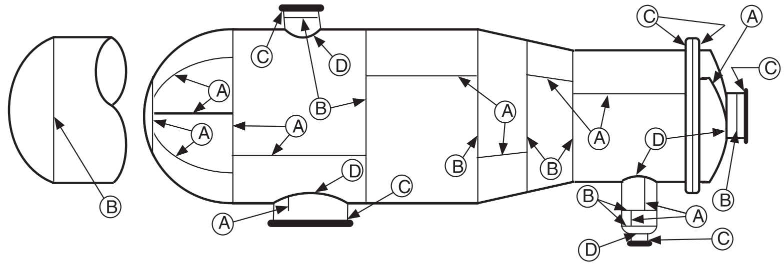
RT–Radiographic Examination

UT–Ultrasonic Examination

MT–Magnetic Particle Examination

PT–Liquid Penetrant Examination

VT–Visual Examination



Categories A, B, C, and D Welded Joint Typical Locations

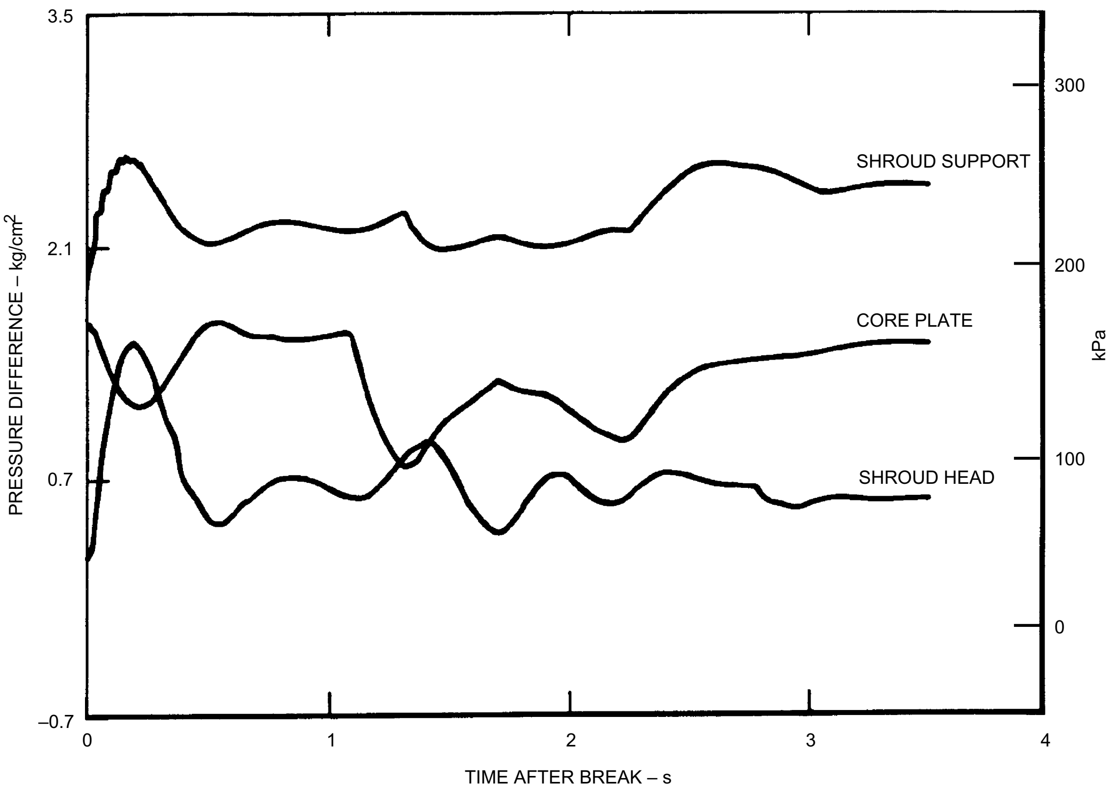


Figure 3.9-1 Transient Pressure Differentials Following a Steam Line Break

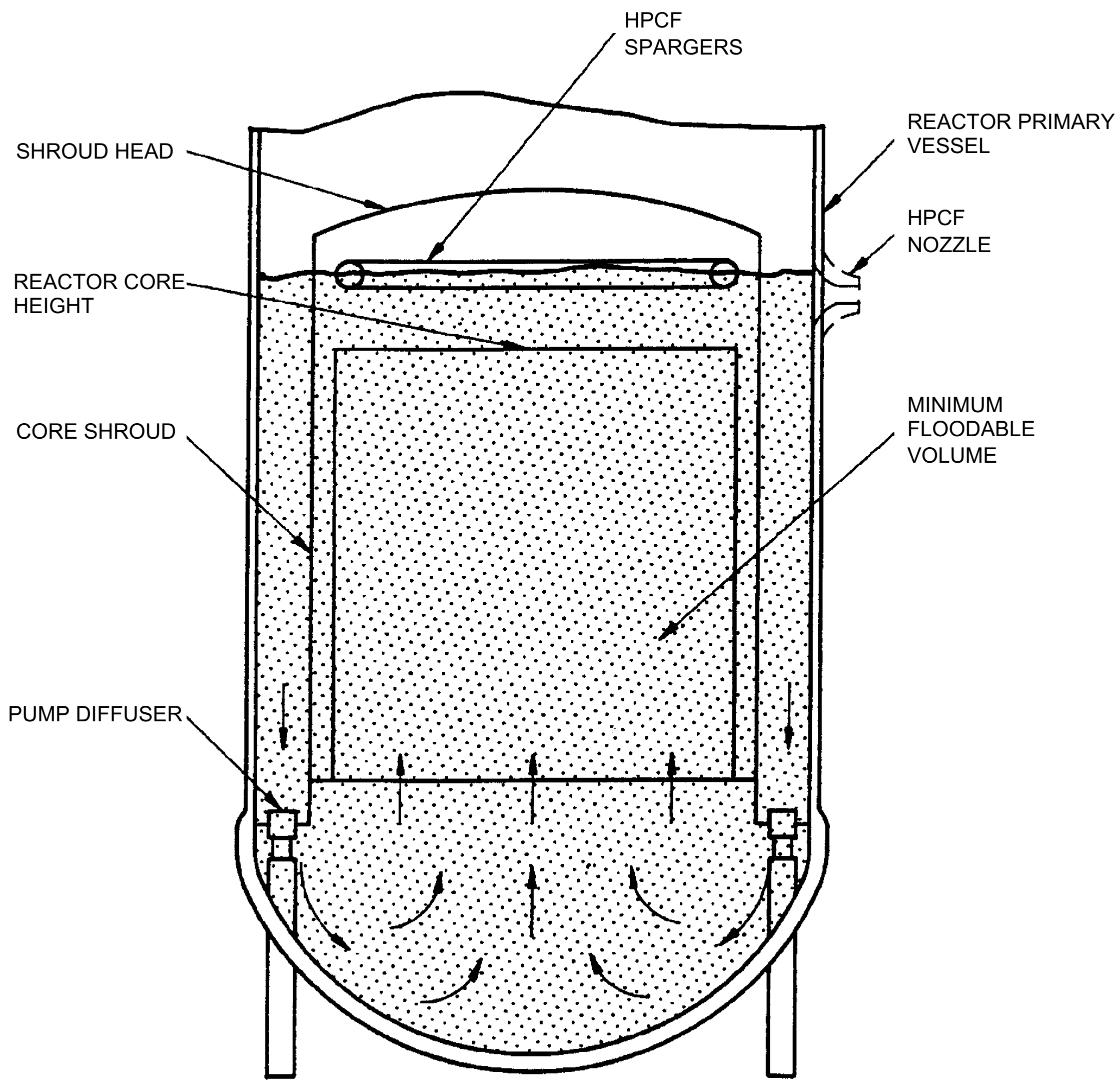


Figure 3.9-2 Reactor Internal Flow Paths and Minimum Floodable Volume

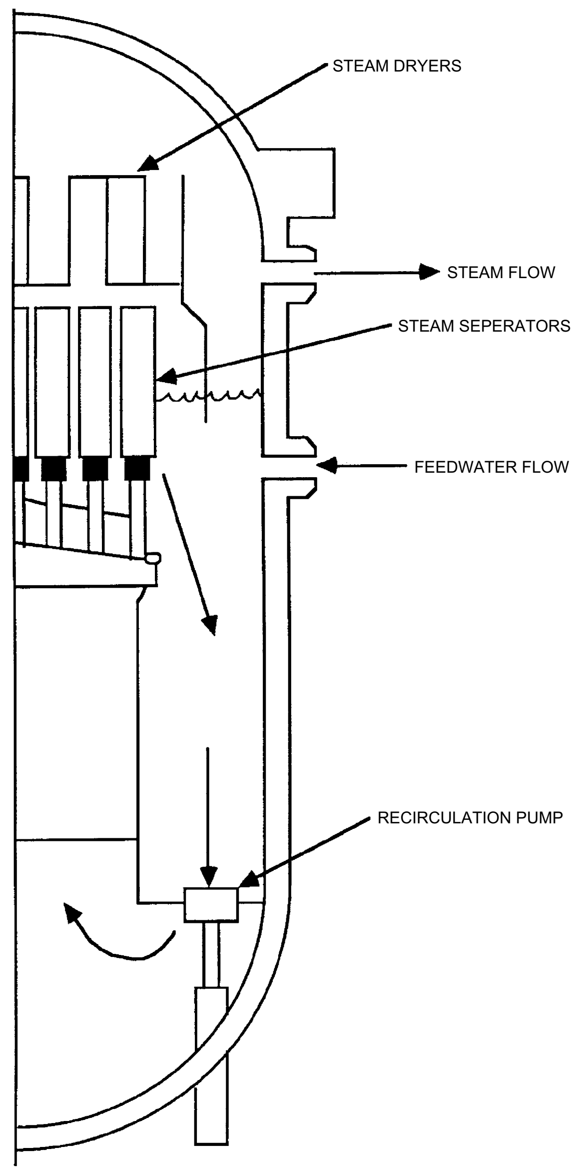


Figure 3.9-3 ABWR Recirculation Flow Path

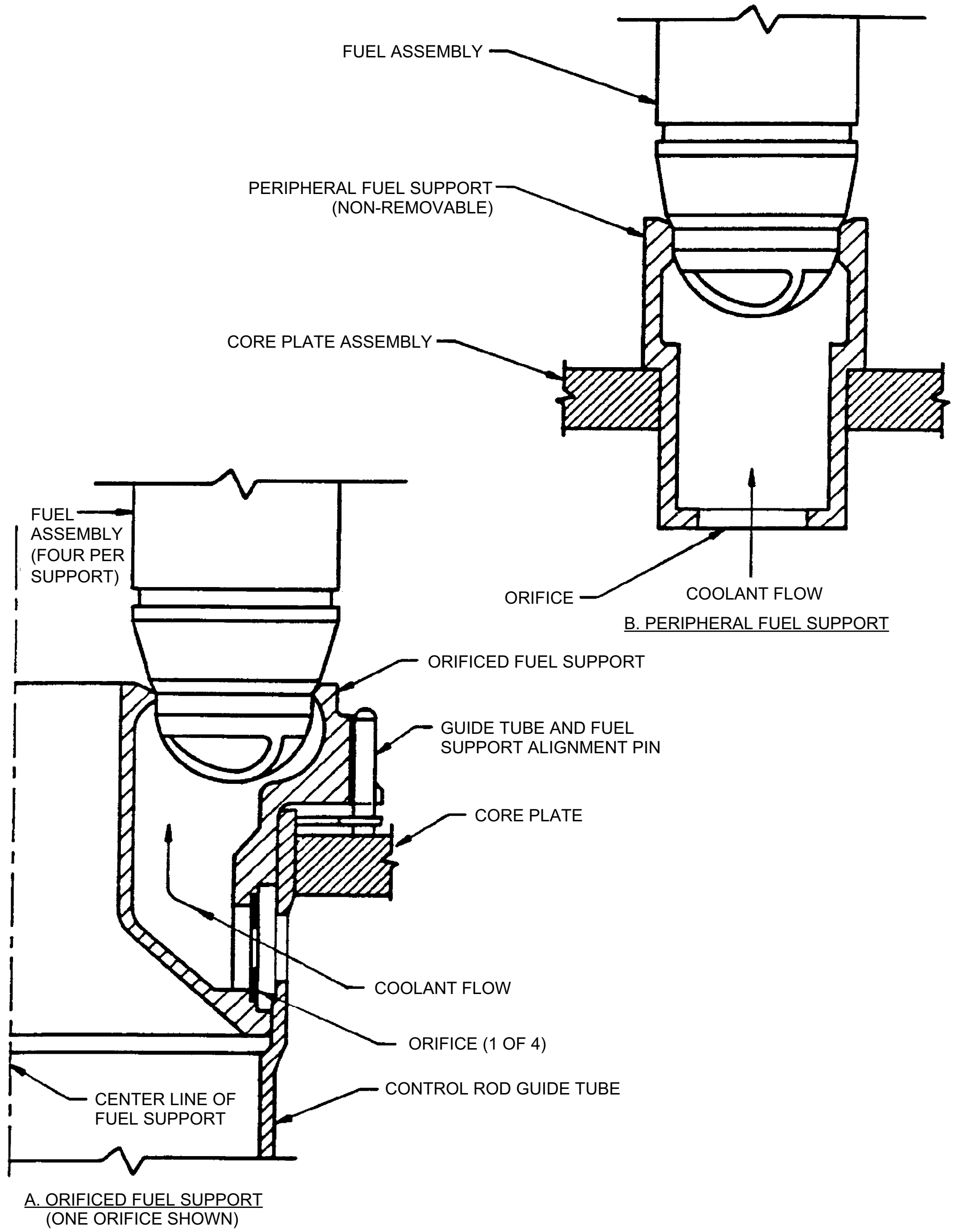


Figure 3.9-4 Fuel Support Pieces

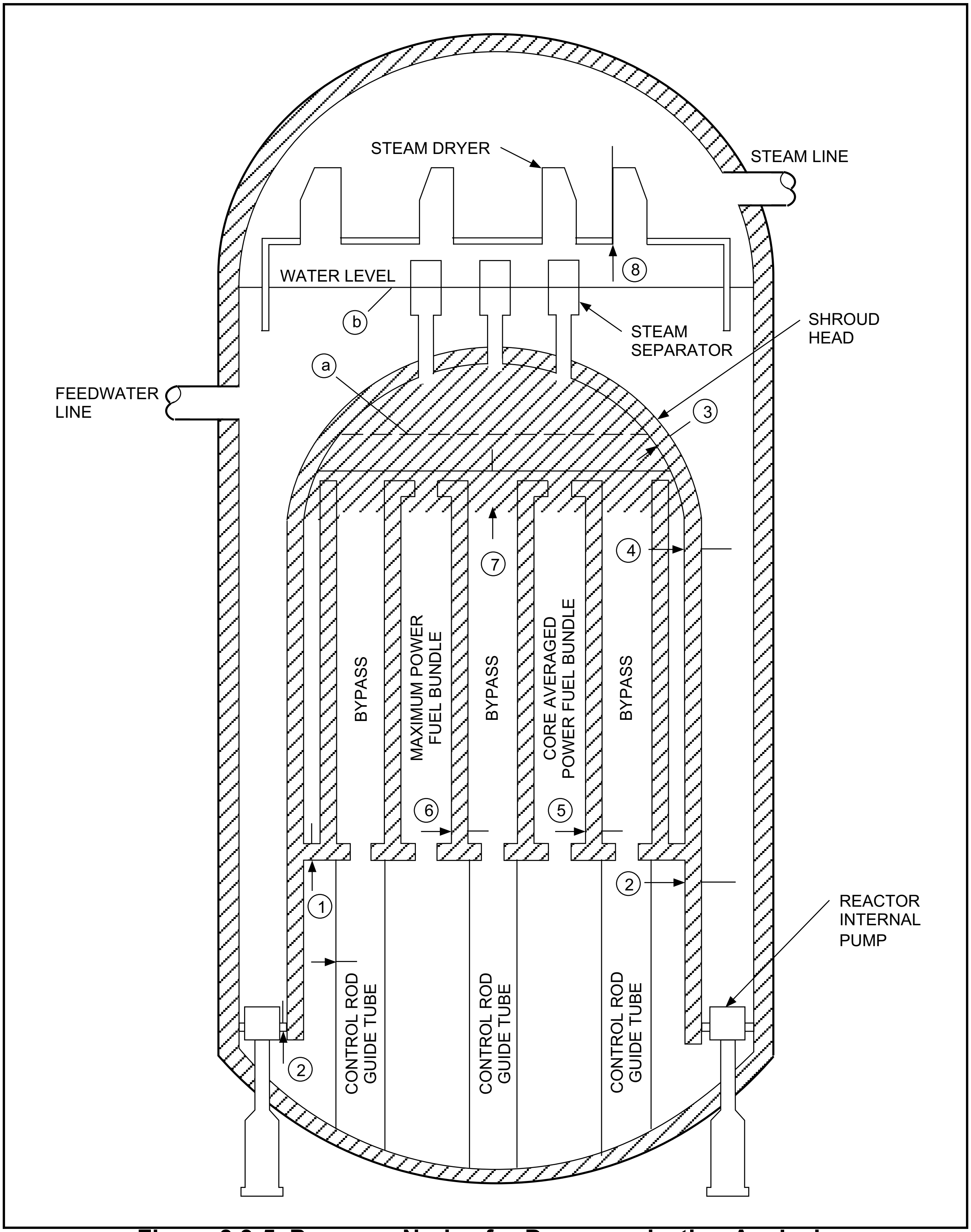


Figure 3.9-5 Pressure Nodes for Depressurization Analysis

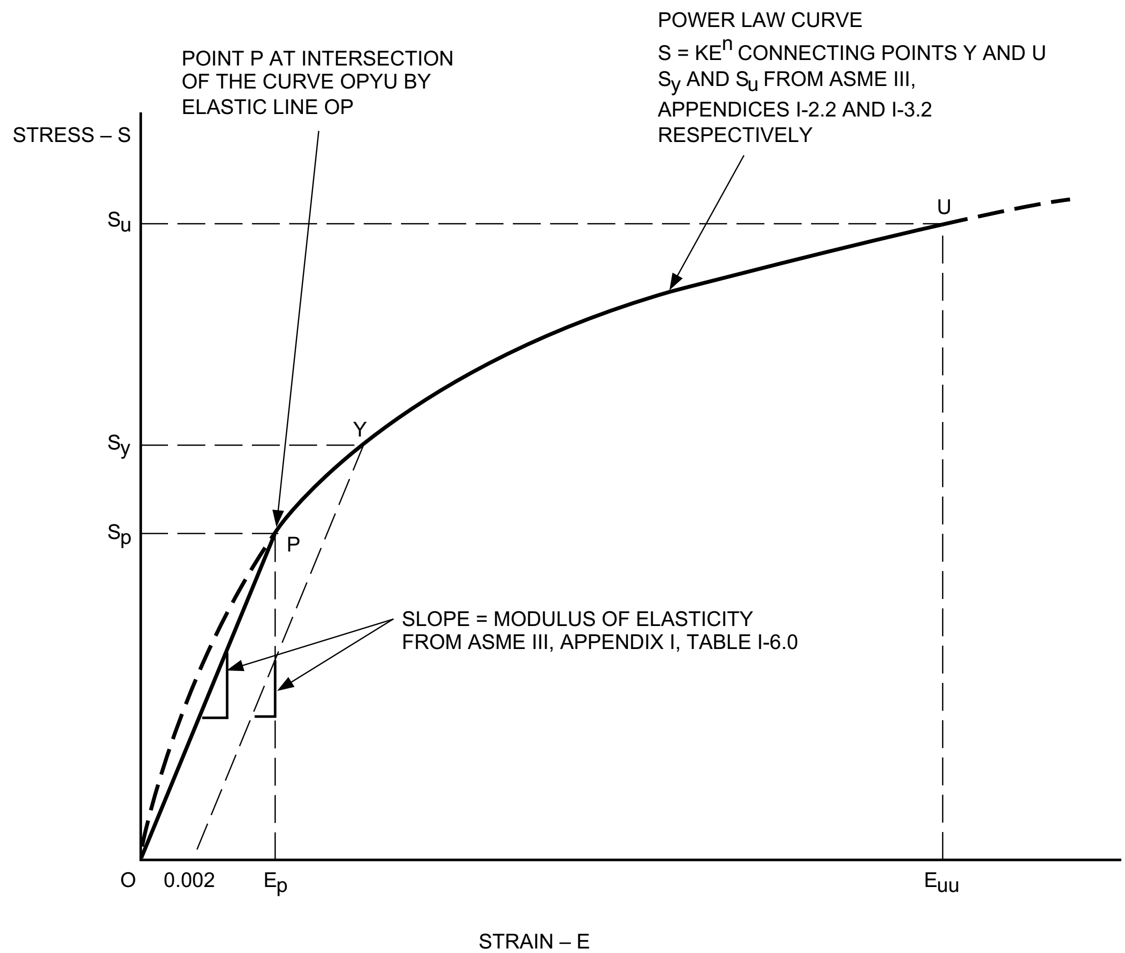


Figure 3.9-6 Stress-Strain Curve for Blowout Restraints

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

This section is supposed to address only seismic qualification of electrical components and equipment in accordance with NRC Regulatory Guide 1.70. However, recognizing that dynamic loads due to suppression pool dynamics associated with a loss-of-coolant accident (LOCA) and safety/relief valve (SRV) discharge can have a significant vibratory effect on the Reactor Building, and, hence, on the design of structures, systems, and equipment in the Reactor Building, GE has elected to address equipment qualification for both seismic and other Reactor Building vibration (RBV) dynamic loads in this section. The format utilized is consistent with Regulatory Guide 1.70; thus, reference to the safe shutdown earthquake (SSE) in this section include the combined seismic and other RBV dynamic loads. The non-seismic RBV dynamic loads are described in Table 3.9-2. The COL applicant must ensure that specific seismic and dynamic input response spectra are properly defined and enveloped in the methodology for its specific plant and implemented in its equipment qualification program.

[*Table 6 of DCD/Introduction identifies the seismic and dynamic qualification commitments, which, if changed, requires NRC Staff review and approval prior to implementation. The applicable portions of the Tier 2 sections and tables, identified on Table 6 of DCD/Introduction for this restriction, are italicized on the sections and tables themselves.*]*

The mechanical components and equipment and the electrical components that are integral to the mechanical equipment are dynamically qualified as described in Section 3.9.

Principal Seismic Category I structures, systems and components are identified in Table 3.2-1. Most of these items are safety-related as explained in Subsection 3.2.1. The safety-related functions are defined in Section 3.2, and include the functions essential to emergency reactor shutdown, containment isolation, reactor core cooling, reactor protection, containment and reactor heat removal, and emergency power supply, or otherwise are essential in preventing significant release of radioactive material to the environment.

3.10.1 Seismic Qualification Criteria (Including Other Dynamic Loads)

3.10.1.1 Selection of Qualification Method

Dynamic qualification of Seismic Category I instrumentation and electrical equipment is accomplished by test, analysis, or a combination of the two methods.

* See Section 3.5 of DCD/Introduction.

[*Qualification by analysis alone, without testing, is acceptable only if the necessary functional operability of the equipment is assured by its structural integrity alone. When complete testing is impractical, a combination of test and analysis is acceptable.*]*

In general, analysis is used to supplement test data, although simple components may lead themselves to dynamic analysis in lieu of full-scale testing. The deciding factors for choosing between tests or analysis include:

(1) Magnitude and frequency of seismic and other RBV dynamic loadings

(2) Environmental conditions (Subsection 3.11.1)associated with the dynamic loadings

(3) Nature of the safety-related function(s)

(4) Size and complexity of the equipment

(5) Dynamic characteristics of expected failure modes (structural or functional)

(6) Partial test data upon which to base the analysis

(7) [*Dynamic coupling between equipment and related systems, if any, such as connected piping and other mechanical components should be considered*]*

The selection of qualification methods to be used is largely a matter of engineering judgement; however, tests, and/or analyses of assemblies are preferable to tests or analyses on separate components (e.g., a motor and a pump, including the coupling and other appurtenances should be tested or analyzed as an assembly).

Qualification by experience is drawn from previous dynamic qualification or from other documented experience such as exposure to natural seismic disturbance. Qualification by experience is based on dynamic similarity of the equipment. [*If dynamic qualification of Seismic Category I instrumentation or electrical equipment is accomplished by experience, the COL applicant will provide the following to the NRC for review and approval: identification of the specific equipment, the details of the methodology for each piece of equipment and the corresponding experience data.*]* See Subsection 3.10.5.3 for COL license information.

3.10.1.2 Input Motion

The input motion for the qualification of equipment and supports is defined by response spectra. The required response spectra (RRS) are generated from the buildings dynamic analysis, as described in Section 3.7. They are grouped by buildings and by elevations. This RRS definition incorporates the contribution and other RBV dynamic loads as specified by the load combination Table 3.9-2. The response spectra curves for the SSE is presented in Appendix 3A.

* See Section 3.10.

When one type of equipment is located at several elevations and/or in several buildings, the governing response spectra are specified.

3.10.1.3 Dynamic Qualification Program

The dynamic qualification program is described in Section 4.4 of GE's Environmental Qualification Program, which is referenced in Subsection 3.11.2. [*The program conforms to the requirements of IEEE-323 as modified and endorsed by the Regulatory Guide 1.89, and meets the criteria contained in IEEE-344 as modified and endorsed by Regulatory Guide 1.100.*]*

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

The following subsections describe the methods and procedures incorporated in the above mentioned dynamic qualification program. Described here are the general methods and procedures to qualify by test or analysis Seismic Category I instrumentation and electrical equipment for operability during and after an SSE including other RBV dynamic loads and to ensure structural and functional integrity of the equipment after an SSE including other RBV dynamic loads.

3.10.2.1 Qualification by Testing

The testing methodology for Seismic Category I instrumentation and electrical equipment includes the hardware interface requirements and the test methods. [*The methodology for qualifying relays shall be such that testing is performed in both the open and closed positions.*]*

3.10.2.1.1 Interface Requirements

Intervening structure or components (such as interconnecting cables, bus ducts, conduits, etc.) that serve as interfaces between the equipment to be qualified and that supplied by others are not qualified as part of this program. However, the effects of interfacing are taken into consideration. When applicable, accelerations and frequency content at locations of interfaces with interconnecting cables, bus ducts, conduits, etc., are determined and documented in the test report. This information is specified in the form of interface criteria.

To minimize the effects of interfaces on the equipment, standard configurations using bottom cable entry are utilized whenever possible. Where non-rigid interfaces are located at the equipment support top, equipment qualification is based on the top entry requirements. A report, including equipment support outline drawings, is furnished specifying the equipment maximum displacement during an SSE including other RBV dynamic loads. Embedment loads and mounting requirements for the equipment supports are also specified in this manner.

* See Section 3.10.

3.10.2.1.2 Test Methods

The test method is multiaxial, random single- and/or multi-frequency excitation to envelope generic RRS levels in accordance with Sections 6.6.3 and 6.6.6 of IEEE-344. Past testing has demonstrated that Seismic Category I instrumentation and electrical equipment have critical damping ratios equal to or less than 5%. Hence, RRS at 5% or less critical damping ratio are developed as input to the equipment base.

Multiaxial testing applies input motions to both the vertical and one or both horizontal axes simultaneously. Independent random inputs are preferred and, when used, the test is performed in two steps with equipment rotated 90 degrees in the horizontal plane in the second step.

When independent random tests are not available, four tests are preformed:

(1) With the inputs in phase

(2) With one input 180 degrees out of phase

(3) With the equipment rotated 90 degrees horizontally and the inputs in phase

(4) With the same orientation as in the step (3) but with one input 180 degrees out of phase

3.10.2.1.2.1 Selection of Test Specimen

Representative samples of equipment and supports are selected for use as test specimens. Variations in the configuration of the equipment are analyzed with supporting test data. For example, these variations may include mass distributions that differ from one cabinet to another. From test or analysis, it is determined which mass distribution results in the maximum acceleration and/or frequency content, and this worst-case configuration is used as the test specimen. The test report includes a justification that this configuration envelops all other equipment configurations.

3.10.2.1.2.2 Mounting of Test Specimen

The test specimen is mounted to the test table so that inservice mounting, including interfaces, is simulated.

For interfaces that cannot be simulated on the test table, the acceleration and any resonances at such interface locations are recorded during the equipment test and documented in the test report.

3.10.2.1.3 Dynamic Testing Sequence

The test sequence includes vibration conditioning, exploratory resonance search and the SSE, including other RBV dynamic loads.

3.10.2.1.3.1 Vibration Conditioning

If required by Paragraph 4.4.2.4.5 of Reference 3.11-2 in Section 3.11, vibration aging program, vibration conditioning is performed at this point in the sequence and the vibration conditioning details are given.

3.10.2.1.3.2 Exploratory Tests

Exploratory tests are sine-sweep tests to determine resonant frequency and transmission factors at locations of Seismic Category I devices in the instrument panel. The exploratory tests are run at an acceleration level of 0.2g, which is intended to excite all modes between 1 and 60 Hz and at a sweep rate of 2 octaves per minute or less. This acceleration level is chosen to provide a usable signal-to-noise ratio for the sensing equipment to allow accurate detection of natural test frequencies of the test specimens.

These tests are run for one axis at a time in three mutually perpendicular major axes corresponding to the side-to-side, front-to-back, and vertical directions.

3.10.2.1.3.3 SSE Testing Including Other RBV Dynamic Loads

An SSE test including other appropriate RBV dynamic loads is performed on all test specimens. This test is conducted to demonstrate that the SSE (as defined in Section 3.7) combined with other RBV dynamic loads will not prevent the equipment from performing its safety-related functions. The test inputs are applied for a minimum of 15 seconds in each orientation. Operability of equipment is verified as described in Subsection 3.10.2.1.3.4 using the criteria in Subsection 3.7.3.2.

3.10.2.1.3.4 Qualification for Operability

In general, analyses are only used to supplement operability test data. However, analyses, without testing, are used as a basis for demonstration of functional capability, if the necessary functional operability of the instrumentation or equipment is assured by its structural integrity alone.

Equipment is tested in an operational condition. Most Seismic Category I instrumentation and electrical equipment have safety-related function requirements before, during, and after seismic events.

Other equipment (such as plant status display equipment) has requirements only before and after seismic events. All equipment is operated at appropriate times to demonstrate ability to perform its safety-related function.

If a malfunction is experienced during any test, the effects of the malfunction are determined and documented in the final test report.

Equipment that has been previously qualified by means of tests and analyses equivalent to those described in this section are acceptable provided proper documentation of such tests and analyses is available.

3.10.2.1.4 Final Test Report

The final test report contains a summary of test/analysis results, which is readily available for audit. See Subsection 3.10.5.1 for COL license information requirements. The report normally includes but is not limited to the following:

(1) Locations of accelerometers

(2) Resonant frequency if any and transmission ratios

(3) Calculation of equipment damping coefficient if there is resonance in the 1–60 Hz range or over the range of the test response spectra

(4) Test equipment used

(5) Approval signature and dates

(6) Description of test facility

(7) Summary of results

(8) Conclusion as to equipment seismic (including other RBV dynamic loads) qualification

(9) Justification for using single axis or single frequency tests for all items that are tested in this manner

3.10.2.2 Qualification by Analysis

The discussions presented in the following subsections apply to the qualification of equipment by analysis.

3.10.2.2.1 Analysis Methods

Dynamic analysis or an equivalent static analysis is employed to qualify the equipment. In general, the choice of the analysis is based on the expected design margin, since the static coefficient method (the easiest to perform) is far more conservative than the dynamic analysis method.

If the fundamental frequency of the equipment is above the input excitation frequency, the equipment is considered rigid. In this case, the loads on each component can be determined statically by concentrating its mass at its center of gravity and multiplying the values of the mass with the appropriate maximum floor acceleration (i.e., floor spectra acceleration at the high frequency asymptote of the RRS) at the equipment support point.

A static coefficient analysis may be also used for certain equipment in lieu of the dynamic analysis. No determination of natural frequencies is made in this case. The seismic loads are determined statically by multiplying the actual distributed weight of the equipment by a static coefficient equal to 1.5 times the peak value of the RRS at the equipment mounting location.

This method is only applicable to equipment with simple frame-type structures and can be represented by a simple model. For equipment having configuration other than simple frame-type structure, this method may be applied when justification can be provided for the static factor which is used on a case-by-case basis.

If the equipment is determined to be flexible (i.e., within the frequency range of the input spectra) and not simple enough for equivalent static analysis, a dynamic analysis method is applied. Dynamic analysis by the response spectrum method is outlined in Subsection 3.7.2.1.3.

3.10.2.2.2 Analysis for SSE Including Other RBV Dynamic Loads

An analysis is performed for the SSE (including appropriate other RBV dynamic loads) in accordance with the criteria in Subsection 3.7.3.2. The analysis must show that following such an SSE, including appropriate other RBV dynamic loads, failure of the equipment to perform its safety-related function(s) does not result.

3.10.2.2.3 Documentation of Analysis

The demonstration of qualification is documented, including the requirements of the equipment specification, the results of the qualification, and the justification that the methods used are capable of demonstrating that the equipment will not malfunction. See Subsection 3.10.5.1 for COL license information requirements.

3.10.2.3 Qualification by Combined Testing and Analysis

In some instances, it is not practical to qualify Seismic Category I instrumentation and electrical equipment solely by testing or analysis. This may be because of the size of the equipment, its complexity, or the large number of similar configurations. The following subsections address the cases in which combined analysis and testing may be warranted.

3.10.2.3.1 Low Impedance Excitation

Large equipment may be impractical to test due to limitations in vibration equipment loading capability. With the equipment mounted to simulate service mounting, a number of exciters are attached at points which will best excite the various modes of vibration of the equipment. Data is obtained from sensors for subsequent analysis of the equipment performance under seismic plus other RBV dynamic loads. The amplification of resonant motion is used to determine the appropriate modal frequency and damping for a dynamic analysis of the equipment.

This method can be used to qualify the equipment by exciting the equipment to levels at least equal to the expected response from an SSE, including other RBV dynamic loads, using analysis to justify the excitation or utilization of the test data on modal frequencies in a mathematical model to verify performance.

3.10.2.3.2 Extrapolation of Similar Equipment

As discussed in IEEE-344, the qualification of complex equipment by analysis is not recommended because of the great difficulty in developing an accurate analytical model.

In many instances, however, similar equipment has already been qualified but with changes in size or in specific qualified devices in a fixed assembly or structure. In such instances, a full test program (Subsection 3.10.2.1) is conducted on a typical piece of equipment. A single frequency test is used in addition to any multi-frequency test.

If the equipment is not rigid, the effects of the changes are analyzed. The test results, combined with the analysis, allow the model of the similar equipment to be adjusted to produce a revised stiffness matrix and to allow refinement of the analysis for the modal frequencies of the similar equipment. The result is a verified analytical model that is used to qualify the similar equipment.

3.10.2.3.3 Extrapolation of Dynamic Loading Conditions

Test results can be extrapolated for dynamic loading conditions in excess of or different from previous tests are given on a piece of equipment when the test results are in sufficient detail to allow an adequate dynamic model of the equipment to be generated. The model provides the capability of predicting failure under the increased or different dynamic load excitation.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

The following subsections describe the general methods and procedures, as incorporated in the dynamic qualification program (Subsection 3.10.1.3), for analysis and testing of supports of Seismic Category I instrumentation and electrical equipment. When possible, the supports of most of the electrical equipment (other than motor and valve-mounted equipment supports, mostly control panels and racks) in the nuclear steam supply systems (NSSS) are tested with the equipment installed. Otherwise, a dummy is employed to simulate inertial mass effect and dynamic coupling to the support.

Combined stresses of the mechanically designed component supports are maintained within the limits of ASME Code Section III, Division 1, Subsection NF, up to the interface with building structure, and the combined stresses of the structurally designed component supports defined as building structure in the project design specifications are maintained within the limits of the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

3.10.3.1 NSSS Electrical Equipment Supports (Other Than Motors and Valve-Mounted Equipment)

The seismic and other RBV dynamic load qualification tests on equipment supports are performed over the frequency range of interest.

Some of the Seismic Category I supports are qualified by analysis only. Analysis is used for passive mechanical devices and is sometimes used in combination with testing for larger assemblies containing Seismic Category I devices. For instance, a test is run to determine if there are natural frequencies in the support equipment within the critical frequency range. If the support is determined to be free of natural frequencies (in the critical frequency range), then it is assumed to be rigid and a static analysis is performed. If natural frequencies are present in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations are determined to see if Seismic Category I devices mounted in the assembly would operate without malfunctioning. In general, the testing of Category I supports is accomplished using the following procedure:

Assemblies (e.g., control panels) containing devices which have dynamic load malfunction limits established are tested by mounting the assembly on the table of a vibration machine in the manner it is to be mounted when in use and vibration testing it by running a low-level resonance search. As with the devices, the assemblies are tested in the three major orthogonal axes.

The resonance search is run in the same manner as described for devices. If resonances are present, the transmissibility between the input and the location of each device is determined by measuring the accelerations at each device location and calculating the magnification between it and the input. Once known, the transmissibilities could be used analytically to determine the response at any Seismic Category I device location for any given input. (It is assumed that the transmissibilities are linear as a function of acceleration even though they actually decrease as acceleration increases; therefore, it is a conservative assumption.)

As long as the device input accelerations are determined to be below their malfunction limits, the assembly is considered a rigid body with a transmissibility equal to one so that a device mounted on it would be limited directly by the assembly input acceleration.

Control panels and racks constitute the majority of Seismic Category I electrical assemblies. There are basically four generic panel types. One or more of each type are tested using these procedures. Figures 3.10-1 through 3.10-4 illustrate the four basic panel types and show typical accelerometer locations.

From these full acceleration level tests, it is concluded that most of the panel types have more than adequate structural strength and that a given panel design acceptability is just a function of its amplification factor and the malfunction levels of the devices mounted in it.

Subsequent panels are, therefore, tested at lower acceleration levels and the transmissibilities measured to the various devices as described. By dividing the devices' malfunction levels by the panel transmissibility between the device and the panel input, the panel dynamic qualification level could be determined. Several high level tests are run on selected generic panel designs to assure the conservativeness in using the transmissibility analysis described.

3.10.3.2 Other Seismic Category I Instrumentation and Electrical Equipment Supports

3.10.3.2.1 Supports for Battery Racks, Instrument Racks, Control Consoles, Cabinets, and Panels

Response spectra are specified for floors where Seismic Category I equipment is located. Test data, operating experience, and/or calculations shall be provided to verify that the equipment will not suffer any loss of function before, during, or after the specified dynamic disturbance. Analysis and/or testing procedures are in accordance with Subsection 3.10.2.

In essence, these supports are inseparable from their supported items and are qualified with the items. During testing, the supports are fastened to the test table with fastening devices or methods used in the actual installation, thereby qualifying the total installation.

3.10.3.2.2 Local Instrument Supports

For field-mounted Seismic Category I instruments, the following is applicable:

(1) The mounting structures for the instruments have a fundamental frequency above the excitation frequency of the RRS.

(2) The stress level in the mounting structure does not exceed the material allowable stress when the mounting structure is subjected to the maximum acceleration level for its location.

3.10.4 Operating License Review (Tests and Analyses Results)

See Subsection 3.10.5.2 for COL license information requirements.

3.10.5 COL License Information

3.10.5.1 Equipment Qualification

COL applicants will provide plant specific seismic and dynamic parameters for the equipment qualification program in accordance with Subsection 3.10.

The equipment qualification records including the reports (Subsections 3.10.2.1.4 and 3.10.2.2.3) shall be maintained in a permanent file and shall be readily available for audit.

3.10.5.2 Dynamic Qualification Report

A dynamic qualification report (DQR) shall be prepared identifying all Seismic Category I instrumentation and electrical parts and equipment therein and their supports. The DQR shall contain the following:

(1) A table or file for each system that is identified in Table 3.2-1 to be safety-related or having Seismic Category I equipment shall be included in the DQR containing the MPL item number and name, the qualification method and the input motion for all Seismic Category I equipment and the supporting structure in the system, and the corresponding qualification summary table or vendor's qualification report.

(2) The mode of safety-related operation (i.e., active, manual active or passive) of the instrumentation and equipment along with the manufacturer identification and model numbers shall also be tabulated in the DQR. The operational mode identifies the instrumentation or equipment

(a) That performs the safety-related functions automatically

(b) That is used by the operators to perform the safety-related functions manually

(c) Whose failure can prevent the satisfactory accomplishment of one or more safety-related functions.

3.10.5.3 Qualification by Experience

If qualification by experience is utilized, the COL applicant must provide the information delineated in Subsection 3.10.1.1 for NRC review and approval.

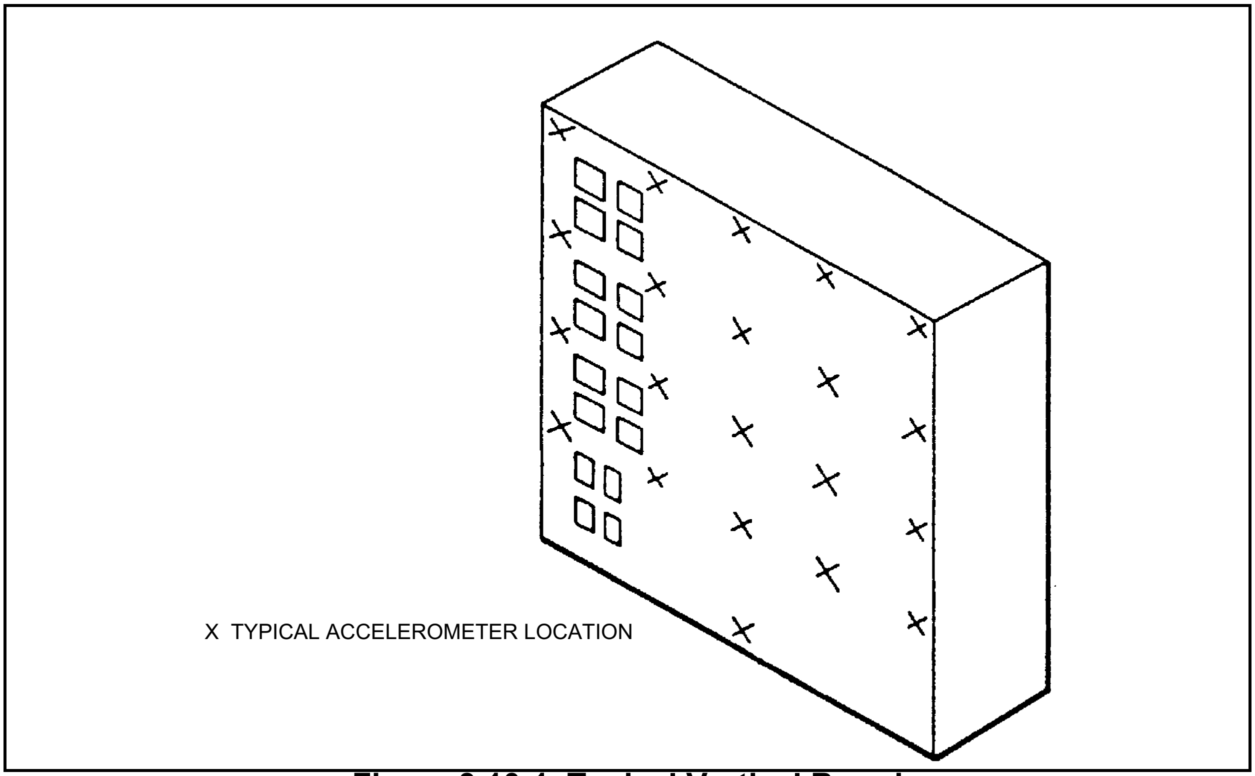


Figure 3.10-1 Typical Vertical Board

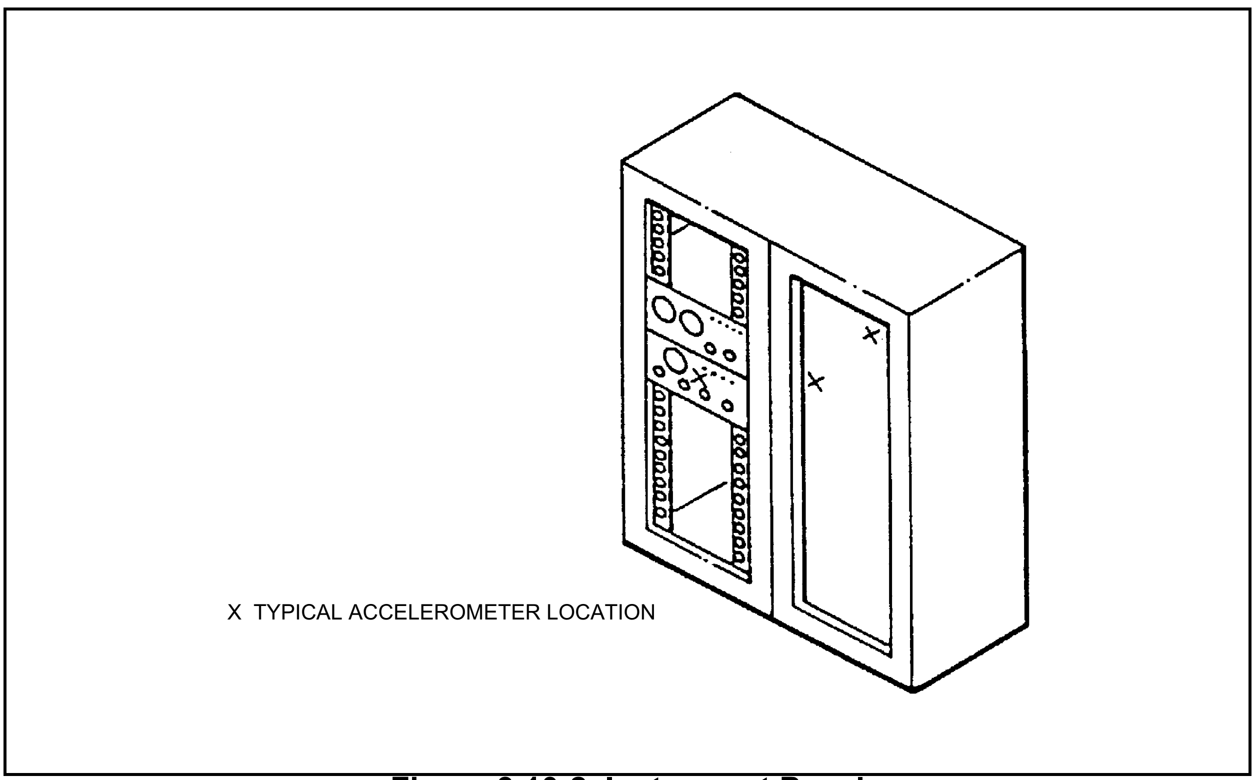


Figure 3.10-2 Instrument Panel

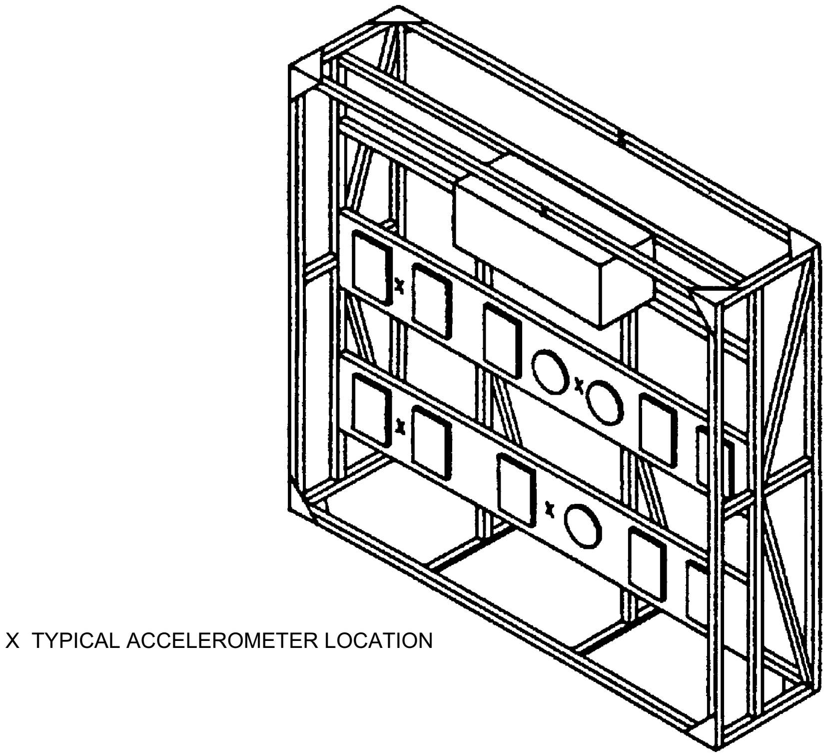


Figure 3.10-3 Typical Local Rack

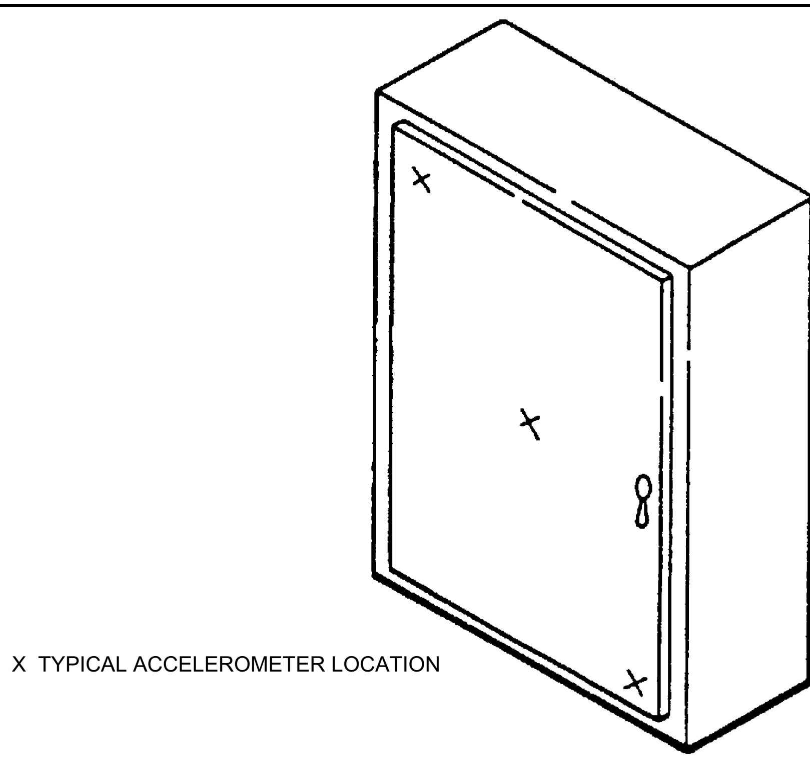


Figure 3.10-4 NEMA Type-12 Enclosure

3.11 Environmental Qualification of Safety-Related Mechanical and Electrical Equipment

This section defines the environmental conditions with respect to limiting design conditions for all the safety-related mechanical and electrical equipment, and documents the qualification methods and procedures employed to demonstrate the capability of this equipment to perform safety-related functions when exposed to the environmental conditions in their respective locations. The safety-related equipment within the scope of this section are defined in Subsection 3.11.1. Dynamic qualification is addressed in Sections 3.9 and 3.10 for Seismic Category I mechanical and electrical equipment, respectively.

Limiting design conditions include the following:

(1) Normal Operating Conditions—planned, purposeful, unrestricted reactor operating modes including startup, power range, hot standby (condenser available), shutdown, and refueling modes.

(2) Abnormal Operating Conditions—any deviation from normal conditions anticipated to occur often enough that the design should include a capability to withstand the conditions without operational impairment.

(3) Test Conditions—planned testing including pre-operational tests.

(4) Accident Conditions—a single event not reasonably expected during the course of plant operation that has been hypothesized for analysis purposes or postulated from unlikely but possible situations or that has the potential to cause a release of radioactive material (a RCPB rupture may qualify as an accident; a fuel cladding defect does not).

(5) Post-Accident Conditions—during the length of time the equipment must perform its safety-related function and must remain in a safe mode after the safety-related function is performed.

3.11.1 Equipment Identification and Environmental Conditions

Safety-related electrical equipment within the scope of this section includes all three categories of 10CFR50.49(b) (Reference 3.11-1). Safety-related mechanical equipment (e.g., pumps, MOVs, SRVs, and check valves) are as defined and identified in Section 3.2. Electrical and mechanical equipment safety classifications are further defined on the system design drawings.

Safety-related equipment must perform its proper safety function during normal, abnormal, test, design basis accident and post-accident environments as applicable. A list of all safety-related electrical and mechanical equipment that is located in a harsh environment area will be included in the Environmental Qualification Document (EQD) to be prepared as mentioned in Subsection 3.11.6.1. The COL applicant will provide a list of impacted non-safety-related control systems and the design features for preventing the potential adverse consequences identified in IE Information Notice 79-22, Qualification of Control Systems. The COL applicant will also address issues related to equipment wetting and flooding above the flood level identified in IE Information Notice 89-63, Possible Submergence of Electrical Circuits Located Above the Flood Level Because of Water Intrusion and Lack of Drainage, as required in Subsection 3.11.6.

Environmental conditions for the zones where safety-related equipment is located are calculated for normal, abnormal, test, accident and post-accident conditions and are documented in Appendix 3I, Equipment Qualification Environmental Design Criteria (EQEDC). Environmental conditions are tabulated by zones, contained in the referenced building arrangements. Typical equipment in the noted zones are shown in the referenced system P&ID and IED design drawings.

Environmental parameters include temperature, pressure, relative humidity, and neutron dose rate and integrated dose. Where applicable, these parameters are given in terms of a time-based profile.

Occurences of anticipated abnormal operating conditions are similar to test conditions and their significant environments are comparable. Equipment significant deviations (magnitude and 60 year frequency) from normal environments have minimal effects on equipment total normal or accident thermal aging. Cumulative abnormal conditions are much less than the bounding accident conditions in the Appendix 3I tables.

Margin is defined as the difference between the most severe specified service conditions of the plant and the conditions used for qualification. Margins shall be included in the qualification parameters to account for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance. The environmental conditions shown in the Appendix 3I tables do not include margins.

Some mechanical and electrical equipment may be required by the design to perform an intended safety function within minutes of the occurrence of the event but less than 10 hours into the event. Such equipment shall be shown to remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis unless a time margin of less than 1 hour can be justified. Such justification will include for each piece of equipment:

(1) Consideration of a spectrum of breaks,

(2) The potential need for the equipment later in the event or during recovery operations,

(3) Determination that failure of the equipment after performance of its safety function will not be detrimental to plant safety or mislead the operator, and

(4) Determination that the margin applied to the minimum operability time, when combined with other test margins, will account for the uncertainties associated with the use of analytical techniques in the derivation of environmental parameters, the number of units tested, production tolerances, and test equipment inaccuracies.

The environmental conditions shown in the Appendix 3I tables are upper-bound envelopes used to establish the environmental design and qualification bases of safety-related equipment. The upper bound envelopes indicate that the zone data reflect the worst case expected environment produced by a compendium of accident conditions. Estimated chemical environmental conditions are also reported in Appendix 3I.

3.11.2 Qualification Tests and Analyses

Safety-related mechanical and electrical equipment is qualified by type testing, operating experience analysis, or any combination thereof as described in IEEE-323 and permitted by 10CFR50.49(f)(Reference 3.11-1). Equipment type test is the preferred method of qualification. Equipment in a harsh environment is designed and qualified to survive the combined effects of temperature, pressure, humidity, radiation, and other conditions related to LOCA or other design-bases accident environment as a portion of their qualified and/or design life.

The qualification methodology is described in detail in the NRC approved licensing Topical Report on GE's environmental qualification program (Reference 3.11-2). This report also addresses compliance with the applicable portions of the General Design Criteria of 10CFR50, Appendix A, and the Quality Assurance Criteria of 10CFR50, Appendix B. Additionally, the report describes conformance to NUREG-0588 (Reference 3.11-3), and Regulatory Guides (i.e., RG 1.89) and IEEE Standards referenced in Section 3.11 of NUREG-0800 (Standard Review Plan).

A mild environment is that which, during or after a design basis event (DBE, as defined in Reference 3.11-2), would at no time be significantly more severe than that which exists during normal, test and abnormal events.

Safety-related mechanical equipment that is located in a harsh environment is qualified by analysis of materials data which are generally based on test and operating experience.

For equipment located in a mild environment, certificate of compliance shall be submitted certifying that the equipment has been qualified to assure its required safety-related function in its applicable environment. This equipment is qualified for dynamic loads as addressed in Sections 3.9 and 3.10. Further, a surveillance and maintenance program will be developed to ensure equipment operability during its designed life (Subsection 3.11.6).

3.11.3 Qualification Test Results

The results of qualification tests for safety-related equipment will be documented, maintained, and reported as mentioned in Subsection 3.11.6.

3.11.4 Loss of Heating, Ventilating, and Air Conditioning

To ensure that loss of heating, ventilating, and air conditioning (HVAC) system does not adversely affect the operability of safety-related controls and electrical equipment in buildings and areas served by safety-related HVAC systems, the HVAC systems serving these areas meet the single-failure criterion. Section 9.4 describes the safety-related HVAC systems, including the detailed safety evaluations. The loss of ventilation calculations are based on maximum heat loads and consider operation of all operable equipment regardless of safety classification.

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Environment

Equipment located in the containment drywell and wetwell is potentially subject to water spray modes of the RHR System. In addition, equipment in the lower portions of the containment is potentially subject to submergence. The chemical composition and resulting pH to which safety-related equipment is exposed during normal operation and design basis accident conditions are reported in Appendix 3I.

Sampling stations are provided for periodic analysis of reactor water, refueling and fuel storage pool water, and suppression pool water to assure compliance with operational limits of the plant technical specifications.

3.11.5.2 Radiation Environment

Safety-related systems and components are designed to perform their safety-related function when exposed to the normal operational radiation levels and accident radiation levels.

Electronic equipment subject to radiation exposure in excess of 10 Gy and other electrical and electrically driven mechanical equipment in excess of 100 Gy will be qualified in accordance with Reference 3.11-1. The normal operational exposure is based on the radiation source terms provided in Section 11.1 and inventories for components provided in Section 12.2. Radiation sources associated with the DBA and developed in accordance with NUREG-0588 (Reference 3.11-3) are evaluated using source terms derived from TID-14844. The DBA source defined for the maximum condition are given in each table. For example Table 3I-18 uses the LOCA analysis (Subsection 15.6.5) for evaluation of dose rates and integrated doses for six months. For Table 3I-17 various components use different DB accidents and integration times as indicated. For the LOCA case, the source term is defined as 100% noble gases and 50% halogens for airborne species (Regulatory Guide 1.3) and 50% halogens and 1% all others for water borne species (Regulatory Guide 1.7). Integrated doses associated with normal plant operation and the DBA condition for various plant compartments are described in Appendix 3I.

3.11.6 COL License Information

3.11.6.1 Environmental Qualification Document (EQD)

The EQD shall be prepared summarizing the qualification results for all safety-related equipment located in harsh environments (Subsection 3.11.3). The EQD shall include the following:

(1) The test environmental parameters and the methodology used to qualify the equipment located in mild and harsh environments shall be identified.

(2) A summary of environmental conditions and qualified conditions for the safety-related equipment located in a harsh environment zone shall be presented in the system component evaluation work (SCEW) sheets.

3.11.6.2 Environmental Qualification Records

The results of the qualification tests shall be recorded and maintained in accordance with the requirements of Reference 3.11-1 (Subsection 3.11.1).

3.11.6.3 Surveillance, Maintenance and Experience Information

The COL applicant will require equipment certificates of qualification compliance and will develop a surveillance and maintenance program in accordance with Subsection 3.11.2.

Non-safety-related control systems subjected to adverse environments will be evaluated for safety implications to safety-related protective functions, and equipment wetting and flooding above the flood level will be addressed in accordance with Subsection 3.11.1

3.11.7 References

3.11-1 Code of Federal Regulations, Title 10, Chapter I, Part 50, Paragraph 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plant.

3.11-2 [“*General Electric Environmental Qualification Program”, NEDE-24326-1-P, Proprietary Document, January 1983*.]*

3.11-3 Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, NUREG-0588.

* See Section 3.10 and Appendix 3K. This reference is same as Reference 3.9-6 (Subsection 3.9.8).

3.12 Tunnels

3.12.1 Main Steam Tunnel

3.12.1.1 Design Basis

The main steam line tunnel is a low leakage reinforced concrete structure. The structure is designed to withstand the forces of a high energy line break.

The floor and the tunnel walls that could be subjected to standing water will be designed to be watertight to the hydrostatic head of standing water plus any additional pressures that the water head may be subjected to. The portion of the tunnel over the Control Building will be subjected to very small amounts of standing water. Water released to the tunnel will flow to either the Turbine Building or the Reactor Building. Water in the Reactor Building portion of the tunnel will accumulate in the recessed portion to a height approximately equal to the bottom of the tunnel over the control building. The water in this space can be drained to the Reactor Building sumps by opening two normally closed manual drain valves located inside secondary containment.

The junction of the tunnel at each building require structural gaps to permit relative displacements of the tunnel due to seismic and thermal movement. The junctions will also be required to include features to provide shielding from the radiation sources in the tunnel. The structural gaps between the tunnel sections will be sealed to provide a confined environment during normal operation. In addition, the seals will be designed to prevent a main steam line or feedwater line break from effecting the plants ability to be brought to a safe shutdown.

3.12.1.2 Description

The Main Steam Tunnel space consists of portions of the Reactor Building, Control Building, and Turbine Building. The tunnel is basically a horizontal rectangular chase that is closed at the Reactor Building end and open at the Turbine Building. The tunnel houses the four main steam lines, the outboard Main Steam Isolation Valves (MSIVs), the two feedwater lines, outboard feedwater isolation valves, other piping, instrumentation, electrical cabling, and other associated equipment.

3.12.1.3 Safety Evaluation

The main steam tunnel is designed to withstand the effects of high energy line breaks and to vent the resulting pressure buildup to atmosphere via blow out panels in the Turbine Building.

3.12.2 Safety-Related Tunnels

3.12.2.1 Design Basis

(1) The tunnels will be designed to applicable safety requirements involving site seismic, flood, fire, and environmental conditions in order to maintain the safety function of

the safety-related divisional equipment contained inside. Specific seismic requirements are included in Subsection 3.7.3.12 and specified in SRP 3.7.3.

(2) The tunnels will be routed independently or provide separate compartments or internal substructures to assure necessary divisional separation requirements between the three (3) divisions.

(3) The tunnels will be designed to withstand the combined effects of hydrostatic head from site flooding and the dynamic effects resulting from internal piping system breaks. Provisions for relieving pressure resulting from pipe breaks will be provided as necessary including the use of external manways.

(4) The tunnels will be designed to ensure that the integrity of the piping penetrations at the interfacing buildings are maintained under design conditions.

(5) The tunnels will be designed to allow periodic inspection of the piping, cables, and piping penetrations.

(6) The tunnels will contain leak detection equipment and provisions for water removal.

(7) Entrances to the tunnels shall be provided with appropriate means to prevent unauthorized access.

(8) Tunnels used for routing fuel oil lines will be constructed in a manner that prevents fuel oil from accumulating next to safety-related structures by sloping them downward away from the building.

3.12.2.2 Description

The purpose of the safety-related tunnels is to provide protected and divisionalized pathways for piping, power cable, and instrumentation and control cable. The safety-related tunnels will be used to route piping and cabling from the Reactor and Control Buildings, to the Emergency Diesel Generator fuel oil storage tanks and the Reactor Service Water pump house.

3.12.2.3 Safety Evaluation

Divisional separation is to be maintained within the tunnel structures. A safety-related division will always be available after considering any combination of a single divisional piping break and a single active component failure.

Pipe break flooding in one division will not degrade the operation of the other two divisions. Ground water intrusion will be prevented.

Penetrations into safety-related structures will be designed to withstand pipe breaks and their effects including hydrostatic forces resulting from pipe tunnel flooding.

Flooding of the tunnels due to site flood conditions will be precluded by protecting the entrances of the tunnel from water entry.

3.12.3 Miscellaneous Non-Safety Related Tunnels

3.12.3.1 Design Basis

The equipment contained in these tunnels is non-safety related and therefore the tunnel will be non-safety related (e.g. radwaste tunnel). The design of these tunnels includes consideration of requirements for water tightness, accessibility, leak detection, and water removal. The design requirements of the tunnel must ensure that tunnel failure will not effect the ability of the plant to be shutdown. The tunnel structures shall be designed so that in the unlikely event of structural failure of a tunnel will not result in unacceptable damage to penetration seals at the interface with safety-related structures. The design must also include consideration for the potential of communicating the effects of pipe breaks from one building to another.

Penetrations at the interface with the safety-related structures will be designed to withstand the combined effects of hydrostatic head from site flooding and the dynamic effects resulting from internal piping system breaks. Provisions for relieving pressure resulting from pipe breaks will be provided as necessary. These tunnels will contain leak detection equipment and provisions for water removal.

3.12.3.2 Safety Evaluation

The use of non-safety related tunnels will not negatively impact the ability of the plant to be shutdown safely. Inter-building flooding via the non-safety-related tunnels will be precluded by penetration seals at each building/tunnel interface. Flooding of the tunnel from external sources (site flood) shall be prevented.

3.12.3.3 Description

The use of tunnels to route piping, cabling and other services from one structure to another will be determined by site characteristics and other design considerations. The ABWR presently includes a radwaste tunnel for routing radwaste piping and other services to and from the Radwaste, Control, Reactor, and Turbine Buildings.

3.13 Secondary Containment and Divisional Separation Zones – Barrier Considerations

3.13.1 Introduction

The ABWR design extends the defense-in-depth concept to its plant buildings and structures. The ABWR design utilizes five fission product barriers—fuel pellets and cladding; reactor pressure vessel and primary coolant system pressure boundary; the primary containment; the secondary containment including the divisional separation zones; the reactor building; and the below ground and elevated release plant features. In addition, the ABWR design also provides multiple division, spacial, physical, electrical and environmental arrangements. These assure total independence between redundant and diverse engineered safeguard systems and their equipment. Even the I&C elements including sensors, data transmission, logic analysis and actuators are divisionally, electrically, and physically separated. In essence, the ABWR design is a three dimensional, multiple barrier, extensively separated, protection network. All of the above features are designed to prevent, mitigate or accommodate a wide spectrum of design basis events. The same features are also capable of successfully addressing a wider spectrum of beyond design basis and/or severe accident events such that the safety risk to the plant and general public is always maintained at an extremely low level.

The design basis of each of the plant containment-type structures, systems and barriers, and their interrelationships varies with the wide spectrum of plant events encountered. The events themselves vary with their anticipated frequency of occurrences and the severity of their consequences. Several key examples are given below:

- Primary and Secondary Containments are required for all DBA pipe breaks Occurring within primary containment and reactor fuel failure or damage events since the radiological consequences associated with core uncovery and for full cladding integrity loss considerations are usually severe and unacceptable without containment. Again, all of these events (e.g., LOCAs) occur inside primary containment.
- Potential breaks outside primary containment must isolate prior to fuel damage or core uncovery. Therefore, breaks outside primary containment do not result in fuel damage.
- Neither Primary or Secondary containment are required for DBA main steam line breaks outside primary containment.
- Secondary Containment is required for DBA refueling accident events since it serves as a primary containment function.
- Secondary Containment-Divisional Separation Zone integrities are required for DBA internal fires.
- ECCS Compartment/Divisional Separation is required for DBA flood events.

- Reactor Building-Divisional Separation Zone integrities are required for DB site related external events.

Each of the cited containment structures and systems has a variety of barrier features. In the case of the Divisional Separation Zones inside the Secondary Containment, the following barriers exist:

- Structural Walls
  - Between Reactor Building and Secondary Containment
  - Between Secondary Containment and Divisional Zones
  - Between Divisional Zones
  - Between Divisional Zones and Primary Containment
  - Between Divisional Zone Individual Compartments
  - Between Divisional Zones and Non-Divisional Areas
  - Ceilings and Floors between all of the above
- Access Openings
  - Access/Egress Doors (Fire, Water-Tight, Water-Resistant, Entry)
  - Hatches (Personnel, Equipment, Inspection)
  - Removable Walls (Block, Shield, Partition)
  - Stairwells
  - Elevators
  - Major Equipment Entries
  - Relief Panels (Blowout, Vents, Vacuum)
  - Piping Electrical and HVAC Tunnel Chases
- Penetrations
  - Piping (Water, Air, Gaseous, Oil)
  - Electrical (I&C, Power)
  - HVAC (Hardened, Soft Ductwork)

– Drains (Equipment, Floor)

– Sumps (HCW, LCW)

– Tunnel Connections (Internal, External)

– Entry Tunnels

Each of structures, systems and barriers are ultimately evaluated by a spectrum of different methodologies: DBA-deterministic and PRA-probabilistic radiologically and environmentally; structural and functional; system interaction-wise; event and effects duration-wise; man-machine wise, etc. The Secondary Containment (SC) and its Divisional Separation Zones (DSZs) serves a multitude of safety and non-safety related functions. This section addresses the general critical design basis considerations of these structures and systems with emphasis on the barrier aspects.

Each of structures, systems and barriers provide hardened separation protection for a vast majority of plant events. For a minor number of low probability events, the integrity of barriers are designed or permitted to be compromised or softened. The resultant integrity loss consequences from these softening or compromising allowances are shown to be acceptable. The accommodation of the resultant effects are an integral part of the design basis considerations (e.g., outside containment break environmental effects in any affected DSZs are tolerable since the equipment in those zones are qualified and specified for those adverse conditions).

3.13.2 Secondary Containment and Divisional Separation Barriers – General Design Basis

The Secondary Containment (SC) and the Division Separation Quadrants or Zones and their individual and collective compartments serve a number of safety and non safety-related functions. This section addresses the general design basis considerations of the Reactor Building, Secondary Containment and Divisional Separation Zones, their structures, systems and barriers. The barriers include – walls, floors, ceilings, doors, hatches, penetrations, HVAC duct work.

3.13.2.1 Reactor Building (RB)

The general design basis considerations for the Reactor Building include:

- The Reactor Building is classified as a safety -related structure.

- The Reactor Building protects the equipment required safe and orderly shutdown equipment from adverse site-related environmental events (e.g., seismic, flood, storm, wind, snow, etc.).

- The Reactor Building encompasses the Secondary Containment and its ECCS Divisional Separation Zones.
- The Reactor Building also houses and provides spacial, physical and electrical separation to other Divisional Separation Equipment Zones or Compartments (e.g., Emergency DG Rooms, Emergency Electrical Equipment Rooms)
- The Reactor Building provides environmental controls to safety related equipment during normal operation and plant transients.
- The Reactor Building is devoid of HELB sources. It also contains only a limited number of fire, flood or radiological sources.
- The Reactor Building and the Secondary Containment share structural and barrier walls, and penetrations.
- Reactor Building is a relatively friendly environs although it provides controlled access to important safety equipment.
- Reactor Building’s radiological barrier capabilities are not required for DBA events.
- Reactor Building’s structural integrity is assured for DBA events.

3.13.2.2 Secondary Containment (SC)

The general design basis considerations for the Secondary Containment include:

- The Secondary Containment provides an additional (secondary) radiological barrier to the Primary Containment. It provides a controlled collection, treatment and elevated release pathway for design basis LOCAs caused leakage from the primary containment. It also provides an environmentally controllable atmosphere for vital equipment required to safely shut the plant down under these conditions.
- The secondary containment provides primary containment during refueling or shutdown operations when postulated refueling pool or open primary coolant system accidents are assumed to occur.
- The Secondary Containment also provides a primary containment function for steam or liquid leaks from reactor coolant pathways outside the primary containment during normal or transient operations.
- The Reactor Building encloses the Secondary Containment and the lower portions of the secondary containment are situated below site ground level.
- Under design basis LOCA inside primary containment conditions, the Secondary Containment is subjected to isolation and standby gas treatment operation. Normal HVAC

is terminated. Breaks inside primary containment are assumed to result in core uncoveries and fission product release although the ABWR design does not show this result. Divisional Separation Zone compartments will be relatively unaffected by the break effects.

- Under design bases LOCA outside containment breaks in the MS tunnel, the Secondary Containment may be subjected to isolation. The post event use of the Standby Gas Treatment System and need for secondary containment integrity is not required although they may be available. These breaks do not result in core uncovery or significant fission product releases. Therefore, Secondary Containment is radiological, controls are not needed. Primary containment is also not needed.

3.13.2.3 Divisional Separation Zones (DSZs)

The general design basis considerations for the Divisional Separation Zones include:

- The three (3) special divisional separation zones or compartments are provided to independently house one of the three (3) ECCS/ESF divisions. A fourth but unique zone is set aside for non-safety-related equipment. The reactor, suppression pool and spent fuel cleanup systems are housed in this fourth quadrant or zone.

- The Divisional Separation Zone compartments also protect each division’s equipment from any potential adverse effects of design basis breaks inside primary containment. Special individual DSZ HVAC systems provide heat removal service to the operating ECCS equipment and the surrounding rooms.

- The Divisional Separation Zone compartments provide limited protection from breaks outside the primary containments but, inside secondary containment (e.g., CUW System breaks and RCIC System breaks). The DSZ provide complete protection for breaks in the MS tunnel. The effects of this event do not adversely affect the Divisional Separation Zone rooms or equipment.

- The Divisional Separation Zone compartments can maintain their integrity for minor leaks within the compartment’s barrier. They can accommodate larger leaks without compromising other DSZ compartments or equipments.

- Breaks outside primary containment do not require secondary containment or divisional separation compartment integrity. These breaks do, however, require successful operation of at least the equivalent of some parts of the one or more divisions of ECCS. Equipment in divisional compartments are designed for intra and inter DSZ compartment breaks.

- Not all of the break, fire, flood and harsh environs protection features in the Divisional Separation Zone compartments are required to be maintained at all times and under all conditions. Fire barriers do not preclude break effect impositions. Flood door barriers integrities are not expected nor required during outside break conditions.

- Divisional compartments are entered from common corridors to enhance inspection and maintenance capabilities. These corridors are defined as divisional or non-divisional zones depending of fire, flood or break aspects.
- Divisional separation throughout the secondary containment is not necessarily required for all events (fire, flood, breaks or adverse environs).
- Under both design basis fire conditions, the Divisional Separation Zone barriers maintain their design integrity. It's only under outside break conditions that the barriers are challenged and allowed to be breached. Under flooding in one DSZ compartment, excess water is permitted into the non-divisional corridor. Entry into other DSZ compartments is precluded.
- Breaks inside the Divisional Separation compartments are vented outside the compartment and within the secondary containment then out of the secondary containment to the site environs in a relatively controlled manner.
- For breaks in the Secondary Containment or in Divisional Separation Zone compartments, the break effects are by design quickly terminated (valve closures), vented to the MS tunnel (through blowout panels) or to other secondary containment volumes. These immediate break effects are limited to the affected area. Ultimately, residual or possible carryover effects to other compartments or zones is expected and designed into the shutdown equipment qualification specifications.
- Only two areas have high energy lines (CUW and RCIC).
- Based on evaluation of the outside containment break risk (frequency of occurrence X severity of consequences), these events represent less than 1% of the total plant risk.
- During normal operation or during transients, the Secondary Containment and Divisional Separation Zone barriers are not subject to abnormal operating conditions, their integrity is maintained, and their status is monitored.

3.13.3 General ABWR Containment Structures, Systems and Barrier Descriptions

General structure, system and barrier descriptions are located throughout the SSAR. References are given to specific locations below:

- Overall Plant Design and Equipment Layout
  - Refer to Figures 1.2-2 thru 1.2-31
- Reactor Building
  - Refer to Section 3.8, Figures 1.2-4 thru 1.2-12

- Secondary Containment
  - Refer to Section 6.2, Figures 1.2-4 thru 1.2-12
- Divisional Separation Zones
  - Refer to Figures 1.2-2 thru 1.2-12
- Design Basis Accident Inside Containment
  - Refer to Section 6.3 and Chapter 15
- Design Basis Breaks Outside Containment Inside Secondary Containment
  - Refer to Section 6.3 and Chapter 15
- Reactor Building/Secondary Containment/Divisional Separation Zone – HVAC Systems
  - Refer to Section 9.4, Figures 9.4-3 thru 9.4-5
- Secondary Containment Penetrations
  - Refer to Section 6.2, Table 6.2-9
- RB/SC Fire Hazard Analysis
  - Refer to Appendix 9A, Table 9A.6-2
- RB/SC Flooding Analysis (Internal and External)
  - Refer to Section 3.4, Table 3.4-1, Section 2.0, Table 2.0-1
- RB/SC/DSZ Safe Shutdown Equipment – Qualifications
  - Refer to Appendix 3I, Tables 3.6-1 and 3.6-2
- Engineered Safety Features
  - Refer to Sections 6.2, 6.3 and 6.4
- Postulated Pipe Break Aspects
  - Refer to Section 3.6
- PRA - Plant and Public Risk Analysis
  - Refer to Sections 15.5.6, 19.8 and 19.9, and Appendix 19E

- Technical Specifications- Containment Structures and Systems
  - Refer to Chapter 16

3.13.4 General Safety Evaluation

3.13.4.1 Overall Perspective

The Reactor Building, the Secondary Containment and the Divisional Separation Zones have been evaluated for the four major (faulted conditions) design basis accident events—a) the reactor core/recirculation internal pump trip-thermohydraulic/ thermodynamic anomaly, b) the refueling bundle drop accident, c) large coolant piping breaks inside primary containment, and d) large coolant piping breaks outside of primary containment. All four events are examined in Chapter 15. All of the events treated with deterministic safety requirements (e.g., single failure criteria, expected isolation valve closures, etc.)

The same subject structures, systems and barriers are evaluated for other postulated plant events (e.g., internal fires, internal floods, external floods). These evaluations are presented in detail in Sections 3.4, Appendix 9A and Chapter 19. In other words, all of the above events were treated with both deterministic and probabilistic evaluation methodologies.

The same subject structures, systems and barriers are also evaluated for less probable and more severe consequence events. These include unrecoverable breaks inside containment, breaks outside containment without immediate isolation, SBO events, loss of all MUX, ATWS, Loss of Immediate Core Cooling, etc. All of these events were treated with probabilistic analysis methodologies in Section 19.

In essence, the ABWR is designed and evaluated to the following critical key events or conditions:

(1) Reactor Building is designed and evaluated to withstand site related events (seismic, wind, snow, storm and flood conditions)

(2) Primary Containment is designed and evaluated to maintain its integrity for DBA-LOCA inside containment events.

(3) Secondary Containment is designed and evaluated to maintain its integrity for DBA-LOCA and core anomaly events inside primary containment.

Divisional Separation Zones are designed and evaluated to maintain their separation criteria and provide adequate safe shutdown capabilities for:

(i) DBA-LOCA breaks inside containment; DBA-core anomaly events and DBA refueling events. All potential core damage resultant events;

(ii) divisional and non-divisional fires, floods, and harsh environs effects. These are non-radiological effects events;

(iii) high energy line breaks outside containment with timely isolation valve closures. These are non-fuel damage events; and

(iv) minor or mild ECCS system leaks in DSZ components events. These are non-radiological concern events.

The barriers for the Divisional Separation Zones in general have a number of different and often conflicting safety and non-safety functions. They are therefore designed to a number of different criteria. The Divisional Separation Zone walls are designed and expected to maintain their structural integrity during all DBA events. The access doors are selectively designed for several uses. Most are expected under normal conditions to be closed. Some are designed to be water tight. Many are fire resistant. Most will remain in place but not necessarily closed under outside containment break conditions. Divisional Separation Zone internal penetrations are fire resistant and flood protected. Under outside break conditions their integrity is not assured nor assumed. The equipment in the DSZ compartments are qualified for environmental conditions that do not depend on most barrier functions.

3.13.4.2 Fire Events

Divisional Separation Zones separation criteria are maintained during design basis fire events. Internal fire in one affected zone will not propagate to other divisions. Smoke is removed from the affected zone. Other zones are pressurized and also vented. Fire suppression techniques are confined to the affected zone. Refer to Fire FIVE Analysis in Appendix 9A. Divisional barriers doors, penetrations, walls, etc. are designed to preclude any external fire effects. Divisional fire impact on other zones is negligible. Fire threat has been minimized by fire source reduction, fire retardant materials and timely fire detection and adequate fire suppression capabilities.

3.13.4.3 Flood Events

Divisional Separation Zones separation criteria are maintained during design basis internal or external flooding events. Internal flooding in one affected zone will not propagate to other divisions. Divisional flooding will be initially directed to basement levels (–8,200 mm). Divisional sumps will collect and remove residual waters. For a Special Event--Suppression Pool drain-out to a specific LPCS room (Divisional Separation Zone), the affected divisional zone will direct some controlled flooding to the non-divisional corridor at the –8,200 mm level to a level of 2.5 meters high. This subsequent flooding to the non-divisional corridor will be confined to the corridor only. The flooding of the other divisional compartments will be precluded by the water tight doors of the other divisions at that level. This is a unique event due to the volume of the Suppression Pool. Most other potential flood sources are small and well controlled within the affected compartment.

3.13.4.4 Pipe Break Events

- Breaks inside Primary Containment will not affect the Divisional Separation Zones. The RB/SC HVAC will be, of course, isolated and the SGTS will be operated. External Secondary Containment barriers (doors) are expected to be closed. Secondary Containment is required (due to potential radiological effects) and automatically assured by isolation valve closures.

- Breaks inside the Main Steam Tunnel will not effect the Divisional Separation Zones. However, Secondary Containment is not required (due to low potential radiological effects) but probably will be available. The event does not result in core damage or core uncovery. Radiological releases are momentary (5.5 seconds). Residual radiation is extremely low. Break effects (release of steam and water) will be confined to MS tunnel, Turbine Building and outside site environs.

- Breaks outside Primary Containment but inside the Secondary Containment (e.g., CUW and/or RCIC breaks) are treated rather uniquely. Since the events do not result in core damage nor core uncovery throughout these events (similar for Main Steam Line Break inside the Main Steam Tunnel), there is no need for Secondary Containment during the blowdown or recovery phase. Although the subject valve closures are longer, coolant release is less and release pathways to the environs are more torturous. Environmental effects on the Secondary Containment and enclosed Divisional Separation Zones are taken into account. Equipment qualifications are defined by the worst break, worst-pressurization analysis. Little credit is assumed for non-affected DSZ areas. Existing barriers are conservatively treated. CUW break pressurization analysis (Chapter 6.2.3) limits the affected Secondary Containment volumes used in the analysis in order to maximize the blowdown effects. It then, however, applies the peak effects results to all Divisional Separation Zone areas. (Refer to Appendix 3I). Realistic analysis shows a significant reduction in break pressurization effects when using reasonable blowdown volumes (complete non divisional corridor plus upper levels of secondary containment). The further use of other available volumes (divisional zones themselves) would result in even lower break effects.

- The design and justification of softened versus hardened Divisional Separation Zones and barriers for extremely low probability breaks outside containment has been thoroughly examined and evaluated in a later paragraph.

- The above evaluation assumes that most of the Divisional Separation Zone barriers—except structural walls—are neglected or ignored in the pressurization and effects analysis. This is a very, very conservative assumption. Some access doors and penetration pathways are all assumed to be open. In realistic situations some will remain closed. Where conservative analysis might require their closure for worst case pressurization analysis, they are assumed closed.

3.13.4.5 Harsh Environs

Divisional Separation Zones are maintained during harsh environs events. Events other than the above cited that events can occur individually or collectively (e.g., SBO—no HVAC) have been evaluated for abnormal or compromising effects. Required Divisional Separation Zone requirements were examined (e.g., restoration of one division HVAC for safe shutdown requirements without the availability of other the divisional HVACs). Most of these have been evaluated throughout Tier 2. For most such events, divisional barriers do not play a significant role in these localized environmental events. That is, these events are not extremely important relative to divisional separation functions.

Loss of RB/SC HVAC due to an inadvertent isolation is accommodated by the operation of the individual Divisional Separation Zone equipment compartment emergency HVAC coolers. This event can affect all the divisional zones at once but divisionalized cooling mitigates the event initial effects. Likewise, loss of an Emergency HVAC system will only affect one divisional zone area.

Divisional barriers appear to be less important for operationally harsh environs threats.

3.13.4.6 Divisional Separation Aspects Throughout the Plant

Divisional Separation Areas exist outside the Reactor Building–Secondary Containment Structures (e.g., in Control Building, RSWS Pump House, Pipe Chases and Tunnels, etc.). Both structural and operational feature barriers exist in these divisional separation areas. Separation criteria or requirements have been established in the SSAR sections addressing these divisional separation areas.

3.13.4.7 Divisional Separation Assurance

Most divisional separated systems are shown on separate drawings; powered by divisional sources; housed in divisional areas; and supported by divisional auxiliary systems. Most divisional separation structures are shown in identified areas on plant layout drawings. Most divisional separation barriers requirements are analyzed and justified in separate event evaluations. The documentation locations of these analysis are spread throughout Tier 2.

3.13.4.8 Soft vs. Hardened Divisional Separation Zones – Outside Breaks

The Secondary Containment, the Divisional Separation Zones, and both of their barriers provide rather robust and hardened inter- and intra-protection for most plant design basis events. They even provide under some extremely low probability and high consequence events a sustained level of total independence.

Most of these events are dictated by the need for Secondary Containment due to severe radiological considerations caused by inside containment breaks. Others involved the need to assure immediate and sustained safe and orderly shutdown capabilities for plant internal events within the Secondary Containment. Fire or flood events require complete isolation of the event effects. Environmental rather than radiological effects governed these mitigation requirements. Barrier performance is critical to the event impact and outcome of all of the above. The timely response of the mitigation engineered safety features also determined the significance of the event effects. Primary Containment radiological barrier isolation valves closed per design, fire doors were closed or closed, flood leak detection actuated alarms and isolations, etc. To most extent the containment features or barriers are inherent, hardened, mostly passive, uni-directional and predisposited.

However, some events are better serviced and mitigated by circumventing conventional barriers or utilizing the barriers for another function. A rapid pressurization is better served by a rapid venting through barriers rather than bottling within the barriers. This is the softened approach to barriers. This is the technique used for breaks outside containment in RB/SC/DSZs. The barrier shortcomings are utilized and engineered for other purposes.

3.13.5 Hardened–Softened Barrier Concept Approach– Special Critique

3.13.5.1 General

The Secondary Containment and Divisional Separation Zone Barriers are subject to wide spectrum of events, accident effects, and performance objectives. Some of these often require diametrically opposed response functions. (e.g., in case of Secondary Containment: maintain integrity or provide a ventable pathway). Each function cited here depends on location of the specific pipe break. That is, whether its inside or outside containment. Whether the outside containment break isolatable or unisolatable. Whether steam or water release is finite or infinite. The current design contains provisions to address all of the barriers functions for all of the design basis events and to also address less frequent and higher consequence events as well. The net result is that the ABWR design has both hardened divisional separation objectives and softened objectives. The hardened objectives demand the robust maintenance of integrity of barriers or boundaries between Divisional Separation Zones in the Secondary Containment for most events. The softened objectives allow more relaxed communication between the zones for a negligible number of unique events. Fire analysis demands strict separation and time maintenance of barriers. Breaks inside CUW areas allow depressurization throughout the Secondary Containment, Non-Divisional Corridor, Stairwells, Vertical HVAC and pipe chases, etc. to the site environs.

The inside containment break demands Secondary Containment full and complete isolation and integrity. The inside flood event in a DSZ compartment allows flooding outside the affected compartment into a non-divisional corridor but no to other DSZ rooms.

3.13.5.2 Specific Critique

The Secondary Containment, the Divisional Separation Zones, and their respective barriers could be viewed as being somewhat softer in protection for pipe breaks occurring within themselves. The barriers do not appear to be designed to sustain the same separation capabilities exhibited above for other areas and events.

There are a number of reasons for considering breaks outside containment, harsh environment occurrences and barrier design basis differently. The reasons include:

- Secondary Containment does not have specific safety function for outside containment breaks.
- Breaks outside containment are less frequent; they result in less consequences; and they are more readily preventable by frequent periodic inspection, increased monitoring and more sensitive leak before break detection.
- Breaks outside containment are more likely to be isolatable and terminated by automatic, timely and responsive break detection and isolation valve closure actions.
- Breaks of the type designed for HELB events (e.g., CUW and RCIC) do not result in core damage, core uncovery or appreciable radiological or environmental effects.
- These breaks result in immediate but short term environmental effects. Their effects are not curtailable by rapid valve closure, early break detections, etc. or even reasonable barrier considerations.
- The most effective and efficient means to accommodate such sudden and momentary energy releases is to provide a large blowdown volume and a large ventable pathway for the released effluents to the outside environs.
- Safe shutdown event mitigation equipment can be and is sheltered out of the direct effects of the break blowdown. Residual effects of the blowdown are included in the equipment qualifications. It is essentially engineering the blowdown pathway.
- Many of the current DSZ barriers, have conflicting missions when used for other events (e.g., fire and flood door closures are rigid barrier features). For pressurization events, door opening are very helpful. They provide additional blowdown pathways.
- The failure modes and effects of most barriers tend to assist the depressurization objective rather than resist it. Door openings are more predictable than door closures.
- Sensitivity of most barrier performances have minimal effects on the depressurization/ event outcome. Blowout panels go over a wide range of pressures. Ventilation dampers closure characteristics are very hard to protect and predict.

- The risk to plant and public is a very small percentage of the total risk for the CUW and RCIC breaks. Refer to Section 15.6.6, 19.8 and 19.9, and Appendix 19E for specific information relative to these breaks and their treatment both deterministically and probabilistically.

3.13.5.3 Conclusions

The Secondary Containment—Divisional Separation Zone Barriers serve both hardened and softened objectives very well. They rigidly resist integrity loss for most plant internal and external influenced events and flexibly relax and allow depressurization pathways for a small number of internally generated events. In essence, they inherently can serve two masters at once successful.

A series of comprehensive evaluations of means to further harden the SC and DSZ barriers was carried out. Blast dampers for HVAC ducting were considered. Increased venting via blowout panels to the MSL tunnel was examined. Faster valve closures on CUW, separate and hardened HVAC subsystems to each DSZ, containment-like DSZ penetrations, etc. were evaluated. However, risk evaluations failed to indicate any appreciable risk reduction by incorporating totally hardened barriers (such as the above).

3.13.6 Specific Barrier Design Basis and Safety Evaluation

The following subsection singles out important individual barriers and provide a brief design basis and safety evaluation of them.

3.13.6.1 Divisional Structural Walls

All the structural load bearing, etc. walls are designed to building code structural requirements. Structural integrity will be assured during all DBA events. These are discussed in Subsection 3.8.

The Reactor Building exterior walls and the divisional walls used for flood protection on the –8,200 mm elevation of the Reactor Building will be designed to withstand the differential pressure resulting from a HELB that is vented only into the corridor spaces within the division on that elevation. Credit could be taken for all the non-divisional corridor volume at –8,200 mm. The Secondary Containment and divisional walls on elevation –1,700 mm and above in the Reactor Building will be designed to withstand the resulting differential pressure from HELB that is assumed to expand into the volumes of these elevations.

Appendix 3H.4 provides the Secondary Containment and Divisional Separation Zone wall design thickness and capabilities. Lower level walls are shown to be capable of maintaining their structural integrity for the pressurization analysis pressures cited.

Divisional structural walls can maintain their structural integrity for internal design loads for fire and flood conditions cited in Section 3.4 and Appendix 9A.

3.13.6.2 Divisional Access Doors

Lower corridor divisional compartment water tight doors are expected to maintain their leak tight closed position during design basis internal flood events.

Fire doors at all levels are expected to maintain their integrity for 3 hours during internal design basis fire events.

Lower corridor doors are expected to open on breaks outside primary containment but inside secondary containment pressurization events.

Non-affected lower divisional doors are expected to stay closed during similar events.

Upper divisional level doors are expected to be less affected by break outside containment venting pressures. They may or may not open depending on vent pressure pathways.

Secondary Containment external access doors are expected to maintain their closed position during fire, flood and break events.

Blowout Panels will not become missiles but will be retained in place.

Elevator Shaft will not be affected pressurization transient.

Equipment Hatches will leak but will be retained in place.

Vertical HVAC and Piping Chimneys are expected to be available as a vent pathway.

3.13.6.3 Divisional Penetrations

Lower corridor divisional penetrations (water, power, I&C) are expected to maintain their integrity under all internal design basis fire and flood conditions.

Lower corridor to divisional compartment penetrations are expected to leak under breaks outside containment pressurization events. However, divisional flooding will not occur.

Division compartment HVAC are expected to maintain their integrity under inside DBA events and internal fire and flood event conditions but not function after an outside break.

Divisional compartment HVAC penetrations are expected to leak or open upon outside break pressurization events.

3.13.6.4 Divisional Safe Shutdown Equipment

Divisional safe shutdown equipment is designed and will be procured for the very conservative normal and accident conditions cited in Appendix 3I. Reasonable and/or realistic analysis show that these requirements are indeed conservative and far beyond anticipated barrier performance and equipment qualification needs. Subsection 6.2.3 analysis shows that for the most limiting events—CUW or RCIC system breaks outside containment—the required safe shutdown equipment in the DSZs are maintained within their environmental qualification limits by wide margins.

3.13.6.5 Pressurization Analysis

The subject analysis conservatively bounds break outside containment event effects. Refer to Subsection 6.2.3. The basis and results of the CUW or RCIC break evaluations are described there. The resulting environmental effects are the most limiting conditions for Secondary Containment and Divisional Separation Zone safe shutdown equipment. The event conditions are momentary as shown by the time at temperature analysis. The conditions cited are used as their equipment qualification requirements. Refer to Appendix 3I.

3.13.6.6 Post Event Recovery

The cited DSZ barriers do not play a critical role in the post event recovery actions. Use of internally located equipment that assure safe shutdown and long-term recovery operations depend only on isolation valve closures within a reasonable length of time (0-1 hour). Externally located equipment that can assure safe shutdown (e.g. Feedwater) depends only on the supply of make-up water available and not on RB/SC/DSZ environmental conditions.

3.13.6.7 Divisional Equipment Reliability

The engineered safety feature equipment required for short term post event safe and orderly shutdown and for long term recovery have been evaluated for the break-environmental effects. Considerations are given both to the effects on the equipment operation and its reliability. Breaks outside containment will subject equipment to brief, momentary environmental conditions below the equipment qualification levels. Breaks inside containment will not subject the equipment to any abnormal conditions. Timely automatic or manual post recovery actions will significantly reduce these momentary conditions to levels significantly below their qualification or capability levels. The reliability of the equipment to these momentary conditions was taken into account. Decreases in equipment reliability is not expected.

3.13.7 Protection of Environmentally Sensitive Equipment

3.13.7.1 Overview

Special attention has been given to environmentally sensitive equipment, especially to equipment located with the Reactor Building, the Secondary Containment or the Divisional Separation Zones which are required or utilized in safe shutdown operations. Certain digital, solid-state electronic I&C equipment falls into this category, (e.g. safety-related remote multiplex units (RMUs)).

Special protection precautions are incorporated in the ABWR design to address environmental sensitivities. These include:

(1) Locating the subject equipment in environmentally suitable and acceptable areas where mild environmental conditions already exists.

(2) Protecting the equipment from expected or potential plant abnormal plant events and their effects or locating the equipment at alternate locations.

(3) Providing alternative, diverse backup equipment which is less environmentally sensitive.

(4) Separating the equipment such that a single severe or challenging plant event or its effects will not be experienced by redundant components or functions.

(5) Providing reliable environmental controls to maintain acceptable conditions at all times even during abnormal events.

(6) Including environmental condition margins in the design and procurement of equipment.

(7) Being able to identify and discriminate between I&C outputs from equipment subjected to abnormal conditions and being able to operate or function around them.

(8) Monitoring and reacting to abnormal environmental conditions with timely remedial actions.

Normal and Accident Equipment Environmental Conditions are cited in Appendix 3I. The Appendix 3I tables contained in this appendix identify environmental conditions at various plant building and equipment locations for a wide spectrum of plant design bases conditions. Pipe breaks both inside and outside primary containment are the dominant contributors to the abnormal plant conditions cited in the tables. Plant impact aspects from design basis fire or flood, or harsh environmental conditions have less of an impact. Due to divisional separation requirements enough equipment is isolated from the single plant fire and flood sources to assure safe shutdown. That is, the event only affects a limited amount of equipment and plant area. Beyond design basis events plant effects like ATWS and SBO are enveloped by the above cited tables since the environmental conditions and effects of these events are less pronounced or momentary. Specific design requirements for these event demand inherent coping capabilities. A number of engineered safety features also mitigate the initial hostile conditions. Ultimately, other equipment is available to restore normal conditions, (e.g. CTG operation and restoration of HVAC). The subject Appendix 3I tables apply to safety-related equipment and their environmental qualifications. However, other equipment unaffected by these environmental conditions may also provide mitigation service (e.g., Turbine Building Feedwater).

3.13.7.2 Reactor Building Housed Equipment

The Reactor Building houses environmentally sensitive equipment in isolated and protected clean zones. These are areas which are not subject to design basis accident pipe break (inside or outside containment) effects. These clean rooms, areas or zones have their own independent and redundant component environmental control HVAC systems. The clean zones house a number of safety-related systems or related components (e.g. emergency electrical equipment rooms, the remote shutdown panel rooms, diesel generator rooms, etc.). The clean zones for redundant safety equipment are in themselves separated by divisional requirements related to fire, flood, and break aspects. Environmentally sensitive I&C equipment is housed in the Emergency Electrical Equipment (EEE) rooms. Not all equipment in clean zones are environmentally sensitive, In fact, only a small portion of the equipment are environmentally sensitive to changes in normal environmental conditions.

Safety-related RMUs and other MUX equipment are housed in EEE rooms. Severe plant event effects do not effect their safety functions. They are inherently unaffected by their own heat sources. They are also capable of prolonged loss of HVAC services due to their environmental locations and their low self heatup characteristics. Since there are three I&C divisions, environmental effects in one will not negate any demanded safety functions from the other locations.

3.13.7.3 Secondary Containment Housed Equipment

The Secondary Containment houses both safety-related and non-safety-related equipment. Little environmentally sensitive equipment is located inside the Secondary Containment. Although all equipment is ultimately affected by beyond normal operation condition, the threshold EQ for most equipment is high and maximum event effect results are low to it. A limited number of potential pipe breaks inside the Secondary Containment require that housed safe shutdown equipment be designed and qualified for significantly elevated (above normal) environmental conditions. Even though these conditions are only momentary (a few seconds to a minute), equipment is qualified for them. The equipment is generally capable of operating for longer times at abnormal effect conditions than required by the design basis event effects.

No safety-related environmentally sensitive I&C equipment resides inside Secondary Containment (e.g. RMUs). Some non-safety related operational MUX equipment (e.g. RMUs) are housed in the Secondary Containment. Their failure or mal-operation due to abnormal secondary containment conditions will not negate safety-related equipment abnormal event functions. The safety-related equipment in the RB/EEE rooms and the qualified safe shutdown equipment in the secondary containment will accomplish their safety function regardless of any non-safety system failures due to environmental conditions.

3.13.7.4 Divisional Separation Zones Housed Equipment

For most plant events and their effects the divisional separation zones generally afford another level of environmental protection and control. Each division has its own emergency HVAC system. For fire, flood and breaks inside containment, the divisional separation barriers assure complete independence, electrical, physical, environmental, etc. A small number of outside containment breaks limit the barriers effectiveness in regards to environmental effects. The equipment qualification requirements are designed to take these low probability events into account.

Less pronounced abnormal environmental conditions (e.g. divisional pipe leaks, fires, floods, HVAC loss, etc.) are readily isolated to the affected divisional zone and not allowed to propagate to the other divisional zones. Even postulated beyond design basis long term environmental effects (total loss of HVAC, extended SBOs, unisolated breaks, etc.) are accommodated. They are accommodated in the short term by the current conservative equipment environmental qualifications and alternative heat removal capabilities and in the long term by power recoveries, valve closures and break isolations, HVAC restoration and alternate heat removal systems.

3.13.7.5 Control Building Housed Equipment

The same protection afforded the above equipment is provided in the Control Building. Control Building environmental effects are induced and self-correcting. (e.g., divisional separation, independent emergency HVAC systems, isolation capabilities.) No high energy related events can occur in the Control Building.

3.13.8 Summary Conclusions

The following overall summary conclusions are offered:

- The ABWR Design Containment structures, systems and barriers provide adequate protection to the plant and public for a wide spectrum of events—Design Basis Accidents, Special Events and Severe Accidents.
- The individual containment structures, systems and components including barriers comply with a wide spectrum of design basis and performance requirements.
- Plant Containment Structures will maintain their structural integrity for all postulated design basis events, (e.g. fire, floods, breaks, site-related events)
- The Secondary Containment and the Divisional Separation Zones will maintain their design basis barriers for all radiologically significant events—DBA breaks inside containment, core/fuel integrity anomalies and refueling accidents.
- Breaks within the Secondary Containment can be accommodated and safe shutdown achieved for a variety of event scenarios. The Divisional Separation Zone equipment will continue to operate after these breaks and will reliably assure safe and order shutdown.

- The Secondary Containment and Divisional Separation Zone envelope integrities are not required for breaks outside Primary Containment especially those within the Secondary Containment or the zones.
- The primary function of the Secondary Containment–Divisional Separation Zones is to assure physical, electrical, divisional and environmental separation during normal plant operations, plant transients and DBA events requiring secondary containment.
- Divisional Separation is most affected during plant design basis fire, flood, breaks inside containment and harsh environs, plant-related events and by site-related threats.
- The Hardened–Softened Barrier Concept utilized in this design is the most practical and effective means of addressing the wide spectrum of events cited.
- The ABWR Design containment systems meet all regulatory requirements and regulations.
- Environmentally sensitive equipment is afforded a significant amount of protection by the ABWR Reactor Building-Secondary Containment-Divisional Separation-Control Building physical configuration arrangements; by engineered safety features -- emergency HVAC system, alternative power supplies and heat removal techniques, by alternate hardwired I&C networks, and by divisional separation requirements; and by conservative fire, flood, break and harsh environs event evaluations, system interaction analysis and conservative EQ considerations.

3A Seismic Soil Structure Interaction Analysis

3A.1 Introduction

This appendix presents soil-structure interaction (SSI) analysis performed for the generic site conditions adopted for establishing seismic design loads and seismic adequacy of the Reactor Building (R/B) and Control Building (C/B) complex of the ABWR standard plant for a 0.3g safe shutdown earthquake (SSE) excitation. The free-field design spectra of SSE are described in Subsection 3.7.1 and are defined per Regulatory Guide 1.60. The SSI analysis results in the form of site-enveloped seismic responses at key locations in the R/B and C/B complex are presented herein.

In order to ensure the seismic adequacy of the R/B and C/B structures and the associated reactor and other equipment, an extensive seismic analysis is required. For a standard plant design, the analysis must be performed over a range of site parameters. The site parameters considered and their ranges together form the generic site conditions. The generic site conditions are selected to provide an adequate seismic design margin for the standard plant located at any site with site parameters within the range of parameters considered in this study. For sites to be located with these facilities, site-specific geotechnical data will be developed and submitted to the NRC demonstrating compatibility with the design analyses assumptions (see Subsection 2.3.1.2).

This appendix details the basis for selecting the site conditions and analysis cases, and the method of the seismic soil-structure interaction analysis. A description of the input motion and damping values, the structural model, and the soil model are included. The parametric study SSI results as well as enveloping seismic responses are also presented.

Seismic adequacy of the R/B and C/B are demonstrated in Section 3.8 using the seismic design loads presented in this appendix, which are obtained as a result of application of the SSI on the methodology described in this appendix.

To demonstrate the seismic adequacy of the standard ABWR R/B design, a total of 22 SSI cases are analyzed for generic site conditions using the finite element method for the SSE condition. The enveloped results reported in this appendix form the design SSE loads.

3A.2 ABWR Standard Plant Site Plan

The typical site plan of the ABWR standard plant is shown in Figure 3A-1. The plan orientations are identified by 0°–180° and 90°–270° directions. The cross section along the 0°–180° is shown in Figure 3A-2. The R/B is nearly square in plan with dimensions of 56.6m x 59.6m. This building is deeply embedded with embedment depth of 25.7m. Adjacent to the R/B along the 0° direction is the C/B. This building is rectangular in plan having plan dimensions of 24m x 56m and the embedment depth of 23.2m. Adjacent to the C/B in the 0° direction farther away from the R/B is the turbine building (T/B) which is 106m long, 58.5m wide. All buildings are supported on separated basemats. The separation distance between the R/B and C/B is 2m. The separation distance between the C/B and T/B is 6m.

The primary objective of the analysis presented in this appendix is to obtain seismic responses for the R/B and C/B complex. Other buildings are considered in the seismic analysis when they are expected to affect the seismic responses of the R/B and C/B.

In the modeling of the buildings, the 0°–180° and 90°–270° directions are designated as X- and Y-axes, respectively. The Z-axis follows the elevations shown in Figure 3A-2. As evident from the site plan, the consideration of structure-to-structure interaction effect in the SSI analysis for the R/B and C/B is required only for X-direction. For the analysis in the Y-direction, the R/B and C/B are considered individually since the structure-to-structure interaction effect in this direction is expected to be less insignificant.

3A.3 Generic Site Conditions

This section describes the generic site conditions and their design parameters based on a range of soil properties used in the soil-structure interaction analysis described in this appendix.

The site conditions are varied to cover a range of expected site conditions in relatively high seismic areas where a nuclear power plant may be constructed.

From the SSI analysis point of view, site conditions can be characterized in terms of: (1) soil deposit depth above bedrock, (2) soil profile and properties, and (3) ground water level. Parametric variations in each of these three areas for establishing generic site design envelopes are presented as follows.

3A.3.1 Soil Deposit Depth

To encompass most of the potential site conditions, a broad range of soil deposit depth is considered. The minimum depth is the embedment depth for which the R/B is supported directly on rock. For this case, the soil depth is 25.7m which is the R/B embedment depth.

The other soil depths are determined using the GESSAR (References 3A-1 and 3A-2) as the guide. By maintaining the ratio of soil depth below the basemat to the basemat width to be approximately the same as for the GESSAR design, the shallow depth is chosen to be 45.7m.

The deepest soil deposit considered in GESSAR is 91.5m. This depth is selected on the basis of a survey of sixty-two U.S. nuclear power plant sites, which indicates that the rock level in the majority of sites is located at a depth less than 61m from the ground surface. A 91.5m depth is, thus, a reasonable upper bound and, therefore, is selected to be the deep soil deposit case for this appendix. Between the shallow and deep soil cases, an intermediate depth is chosen to be at 61m.

In summary, the variations of soil deposit thickness are accounted for by considering the following four representative soil deposit depths.

Minimum (embedment depth)	25.7m
Shallow soil deposit	45.7m
Intermediate soil deposit	61.0m
Deep soil deposit	91.5m

3A.3.2 Soil Profile and Properties

The range of soil profiles considered in this appendix is based on the velocity profiles used in GESSAR (References 3A-1 and 3A-2). A total of six velocity profiles are selected and shown in Figure 3A-3. These velocity profiles are designated with the abbreviations: UB, VP3, VP4, VP5, VP7 and an upper bound case with rigid soil properties (R cases).

The profile UB represents a soil profile of which the shear wave velocity at a depth z below the ground surface is obtained using the modulus parameter K_{2max} and the following equations.

$$G_{max}(z) = 218.8K_{2max}\sqrt{\sigma_m(z)} \tag{3A-1}$$

$$V_s = \sqrt{G_{max}/\rho} \tag{3A-2}$$

where

G_{max}	=	maximum shear modulus, kPa
σ_m	=	effective mean pressure (kPa) at depth (z); it is assumed equal to 0.7 times the effective overburden pressure, which corresponds to the use of an at rest coefficient of lateral pressure equal to 0.55.
K_{2max}	=	modulus parameter
ρ	=	mass density
V_s	=	shear wave velocity

This soil profile is assumed to consist of seven horizontal layers. The K_{2max} value, total unit weight, and Poisson's ratio for each layer are as shown in Table 3A-1.

Note that for submerged layers the Poisson's ratio is to be adjusted such that the minimum P-wave velocity of water is retained. The values of average shear wave velocity, shown in Figure

3A-3 for UB, are computed using Equations 3A-1 and 3A-2 at the mid-depth of each layer for the ground water level at a depth of 0.61m below grade.

The profiles UB, VP3 through VP5 are selected based on three generalized soil zones shown in Figure 3A-4: a soil zone (sands, silts, clays, and gravely soils), a transition zone, and a soft rock and well-cemented soil zone. Velocity profile UB represents an average profile of the soil zone; VP3 and VP4 bound the transition zone, and VP5 represents an average profile for well-cemented soil zone. Those velocity profiles are smooth curves representative of the average variation of shear modulus with depth that can be expected within each of the soil zones.

The profile VP7 represents a hard rock site with a uniform shear wave velocity of 1524 m/s. The profile for R cases represents an upper bound profile with rigid soil properties (uniform shear wave velocity of 6096 m/s).

The lower bound shear wave velocity for the top 9m of soil among all profiles considered is 303 m/s. The upper bound velocity is 6096 m/s. This constitutes a wide range of potential site conditions that are suitable for nuclear power plants.

The general soil layer properties (layer thickness, total unit weight, and Poisson's ratio) defined for the UB profile are also adopted for all other profiles except for VP7 and R. The profiles VP7 and R are considered to be uniform elastic half-space with a constant Poisson's ration of 0.3. The unit weight density for VP7 profile is the same as UB profile. The unit weight density for R profiles is 2.2 t/m^3. The base rock in all soil profiles is modeled using density 2.2 t/m^3, shear wave velocity of 1594 m/s. Poisson's ratio of 0.3, and material damping of 1%. The average shear wave velocities in layers for all soil profiles are tabulated in Table 3A-2.

The shear modulus and material damping of soil are strain dependent. Figure 3A-5 shows the variation of shear modulus and damping ratio with shear strain for various soil profiles considered. The soil curves shown correspond to average curves of Reference 3A-6. On the basis of the recommendations made in Reference 3A-3 the soil material damping of a hysteretic nature is limited to a maximum of 15% of critical. In addition to use of average soil curves, a parametric study was performed in which the upper bound shear modulus soil degradation curve of Reference 3A-6 is used. This curve is the same as the shear modulus degradation curve reported in Reference 3A-7 for sands. The results of this case are presented in Section 3A.9.5. Variation of shear modulus and damping for rock profiles (VP5 and VP7) are shown in Figure 3A-6. The shear modulus reduction factors and damping ratios at various strain levels are shown in Tables 3A-3 and 3A-4. For VP7 profile, the free-field site response analysis results has shown that the strain-compatible shear modulus and material damping are essentially unchanged from their initial values. The SSI analysis for R profile is performed using uniform velocity of 6098 m/s.

3A.3.3 Ground Water Table

The effect of ground water on the soil properties is considered using the following procedure:

(1) Perform one dimensional convolution or deconvolution analysis for the horizontal excitation component to obtain strain compatible shear modulus, G, corresponding to the induced strain level. The corresponding shear wave velocity, Vs, is then computed as

$$V_s = \sqrt{G/\rho} \qquad (3A\text{-}3)$$

(2) Compute the corresponding compression wave velocity, Vp, using the following equation.

$$V_p = V_s\sqrt{\frac{2(1-\mu)}{1-2\mu}} \qquad (3A\text{-}4)$$

where μ is a Poisson's ratio. The lower bound of V_p for the submerged soil below water table is the compression wave velocity of water taken to be 1463 m/s. When the computed V_p of soil is smaller than 1463 m/s, adjustment of soil Poisson's ratio is required to increase the P-wave velocity.

In order to evaluate the effects of water table location variation (assuming no soil failure) on structural response, three water table locations are considered, namely, the high, intermediate, and low water tables. The high water table is the base case which is located at 0.61m below grade. The low water table is taken to be 25.7m which is at the base of the R/B foundation basemat. The intermediate water table is assumed at 12.2m below grade which is at about the midheight of the R/B embedment. Since the ground water may have more pronounced effects on soft sites, the UB profile with 45.7m depth of soil deposit is investigated for all three water tables defined above. The water table level for other site conditions is based on the basic high water table case which is at 0.61m below grade.

3A.3.4 Summary of Site Conditions

The above discussions cover the range of site parameters in terms of soil deposit depth, soil profile and properties, and water table location. Based on the GESSAR experience (References 3A-1 and 3A-2) that the shallower soil depth in general resulted in higher structural response, all velocity profiles, except the VP7 and R profiles which are uniform rock profiles with zero soil depth, are considered for the 45.7m shallow soil case, and only limited soil velocity profiles need to be considered for other depths. For the deeper deposits of 61m and 91.5m in depth, it is sufficient to consider only the lower bound UB and VP3 profiles. Consequently, the effects of variation in soil properties with depth is accounted for more representatively. The minimum soil depth of 25.7m is considered for the UB and VP3 profiles which adequately cover the range of soil profiles considered. As mentioned before, the water table variation is taken into account by the UB profile for the shallow soil deposit case. The site conditions in terms of the combination of shear wave velocity profile, soil depth and ground water table locations are schematically summarized in Figure 3A-3.

3A.4 Input Motion and Damping Values

3A.4.1 Input Motion

The time-history method is used in performing the seismic soil-structure interaction analysis. Earthquake (input or control) motion in the form of synthetic acceleration time histories are generated as described in Subsection 3.7.1.2 for all three components designated as H_1, H_2, and V. The H_1 and H_2 are the two horizontal components mutually perpendicular to each other. In the SSI analyses, H_1 and H_2 components are used in the horizontal X-(0°) and Y-(90°) directions, respectively. The V component is used in the vertical Z-direction.

The input motion (or control motion) is defined at the finished grade in the free-field. The motion is assumed to be generated by vertically propagating plane seismic shear waves for the horizontal components and compression waves for the vertical component.

3A.4.2 Damping Values

The structural components damping values used in the seismic analysis are in accordance with those specified in Regulatory Guide 1.61. These values for the SSE are summarized in Table 3.7-1.

The soil material damping values used for the SSI analysis are based on the strain-compatible soil damping values resulting from the free-field site response analysis as described in Subsection 3A.6. The strain-dependent soil damping versus strain is described in Subsection 3A.3.2.

3A.5 Soil-Structure Interaction Analysis Method

3A.5.1 Introduction

Nuclear Island structures are massive structures typically embedded to a considerable depth in a soil deposit. An important aspect in the seismic design of these structures is the evaluation of the dynamic interaction between the structures and the soil. Such interaction effects may significantly affect the response of the structures as well as the equipment systems.

The approach used to obtain seismic design envelopes in this appendix is based on the finite-element method using substructuring technique. The newly developed, two and three-dimensional (2-D and 3-D), linear finite element computer program SASSI (Reference 3A-4) is used. The program uses finite elements with complex moduli for modeling the structure and foundation properties and is based on the flexible volume method of substructuring and the frequency domain complex response method of analysis. Detailed descriptions of the complex response method and SSI methodology are presented in Subsections 3A.5.2 and 3A.5.3.

In performing the SSI analysis using the finite element method, the detailed structural models described in Subsection 3.7.2 are coupled with the soil model. Structural responses in terms of accelerations, forces, moments, and stresses are computed directly. Floor response spectra are obtained from the calculated response acceleration time histories (see Subsection 3.7.2.5). This effectively eliminates the need for a second step structural response analysis in which the fixed-based structural model is subjected to the base motions resulting from the first step SSI analysis. The direct solution also has an added advantage that the structural response to all components of base motion including rocking motion components for an embedded foundation is automatically accounted for in the solution.

The SSI analyses for the three directional earthquake components are performed separately. The maximum co-directional responses to each of the three earthquake components are combined using the SRSS method to obtain the combined maximum structural response in each selected degree of freedom of interest.

3A.5.2 Complex Response Method

For the dynamic analysis of a linear damped system, the equation of motion in the time domain at time t may be written in the following matrix form:

$$M\ddot{U} + D\dot{U} + KU = P \qquad (3A\text{-}5)$$

where $\ddot{U}$, $\dot{U}$ and U are respectively the acceleration, velocity and displacement vector at time t; K and M are the assembled total stiffness and mass matrices; D is the damping matrix and P is the dynamic load vector acting on the system at time t. For an SSI system for which the soil material damping is characterized by the constant hysteresis damping model and the foundation impedances are frequency dependent, the equation of motion is most conveniently formed and solved in the frequency domain using the complex response method. The application of the complex response method involves two solution steps: (1) steady-state response analysis to obtain the frequency response functions, and (2) Fourier analysis of the input and convolution to determine the transient response.

For the steady-state harmonic excitation at the circular frequency ω, the exciting force vector is written in the form:

$$P(t) = P^* \cdot e^{i\omega t} \qquad (3A\text{-}6)$$

where P^* is a complex constant vector representing the applied force amplitude and the phase angle.

The steady-state displacement response is also harmonic with the frequency ω and can accordingly be written as

$$U(t) = U^* \cdot e^{i\omega t} \qquad (3A\text{-}7)$$

Substitution of Equations 3A-6 and 3A-7 into Equation 3A-5 yields the matrix equation

$$C^*U^* = P^* \tag{3A-8}$$

where C* is the frequency dependent complex dynamic stiffness matrix in the form

$$C^* = K + i\omega D - \omega^2 M \tag{3A-9}$$

Equation 3A-8 is a system of linear equations with the complex coefficient matrix which can be solved for the complex response vector U^* for each frequency. The solution for U^* for a unit applied force, i.e., $P^* = 1$, is the transfer function.

For a system with material damping, the first two terms of Equation 3A-9 can be combined into a complex stiffness matrix K^*

$$K^* = K + i\omega D \tag{3A-10}$$

For a system with constant hysteresis material damping, K^* can be formed directly using complex shear moduli E_s^*, and complex constrained moduli E_p^*, defined by the following equations:

$$E_s^* = E_s\left(1-2\beta_s^2 + 2i\beta_s\sqrt{1-\beta_s^2}\right) \tag{3A-11}$$

$$E_p^* = E_p\left(1-2\beta_p^2 + 2i\beta_p\sqrt{1-\beta_p^2}\right) \tag{3A-12}$$

Where E_s and E_p are the real numbers corresponding to the shear and constrained moduli of the material, respectively, and β_s and β_p are the critical damping ratios associated with S-waves and P-waves, respectively. For practical applications, β_s and β_p are usually taken to be equal, i.e., $\beta_s = \beta_p = \beta$, in which case the Poisson's ratio is a real number.

Using the transfer function solution obtained from the complex response method, the transient excitations such as earthquake motions or applied dynamic loads, can be analyzed by decomposing the input excitation into Fourier components with period N•Δt, where Δt is the time interval of digitization of excitation and N is the number of digitized points. Using the discrete representation, the discretized load function P (t) is defined by the series.

$$P_K = P(K \cdot \Delta t),\ K = 1, 2, ...,N \tag{3A-13}$$

which can be expanded into a discrete Fourier series in the form

$$P(t) = \mathrm{Re}\sum_{s=1}^{N/2} P_s^* e^{i\omega_s t} \tag{3A-14}$$

where ω_s are the discrete circular frequencies defined by:

$$\omega_s = \frac{2\pi s}{N \cdot \Delta t} \tag{3A-15}$$

and P^* is complex amplitude defined by

$$P_s^* = \frac{1}{N}\sum_{K=1}^{N-1} P_k \exp(-i\omega_s k\Delta t);\ s = 0, s = \frac{N}{2} \tag{3A-16}$$

$$P_s^* = \frac{2}{N}\sum_{K=1}^{N-1} P_k \exp(-i\omega_s k\Delta t);\ 1 \le s \text{ and} \le \frac{N}{2} \tag{3A-17}$$

The numerical operations in Equations 3A-16 and 3A-17 are performed using the efficient Fast Fourier Transform (FFT) algorithm. This technique requires that N be a power of 2 which can always be achieved by adding trailing zeros to the discrete excitation time series. For damped systems these trailing zeros also serve as a quiet zone which allows the transient response motions to die out at the end of the duration to avoid cyclic overlapping error in the discrete Fourier transform procedure.

Once the P_s values are obtained, the Fourier transform of the transient response can be obtained by convolution of the transfer function and the Fourier coefficient, P_s, giving the complex frequency response function U_s for each discrete frequency ω_s.

Finally the transient response in the time domain U(t) can be obtained by the Fourier synthesis, i.e.,

$$U(t) = \mathrm{Re}\sum_{s=0}^{N/2} U_s^* e^{i\omega_s t} \tag{3A-18}$$

Numerically, this operation is performed using the inverse Fast Fourier Transform algorithm. In principle, the system of complex linear equations defined by Equation 3A-8 has to be solved for each of the 1 + N/2 frequencies shown in Equation 3A-15. For the excitation defined by a real time function, the highest frequency corresponds to $1/(2\Delta t)$. This is called the Nyquist (or folding) frequency. The choice of Δt usually results in the Nyquist frequency which is much higher than the highest frequency of interest. The series in Equation 3A-15 can thus be truncated at some lower frequency number, s_{cut}. The corresponding cutoff frequency is

$$f_{cut} = \frac{s_{cut}}{N \cdot \Delta t} \quad s = 0,...,s_{cut} < \frac{N}{2} \tag{3A-19}$$

and only s_{cut} solution to Equation 3A-8 are required. For seismic analysis, s_{cut} is still a large number (typically of the order of 500). In practice Equation 3A-8 is solved for only a limited number of frequencies in the frequency range of interest. The solutions for the remaining frequencies are obtained by interpolation in the frequency domain.

3A.5.3 Methodology and Analysis Procedure

The flexible volume substructuring method used in SASSI considers the structure(s) and the foundation(s) to be partitioned as shown in Figure 3A-7. In this partitioning, the structure (Figure 3A-7) consists of the superstructure plus the basement minus the excavated soil. The foundation (Figure 3A-7), consists of the original site, i.e., the site without soil excavation for the structure basement. Interaction between the structure and the foundation occurs at all basement nodes. For seismic analysis, the equation of motion for substructure C in Figure 3A-7 can be written in the frequency domain as follows:

$$\begin{bmatrix} C^*_{ss} & C^*_{si} \\ C^*_{is} C^*_{ii} - C^*_f + X^*_f \end{bmatrix} \begin{Bmatrix} U^*_s \\ U^*_f \end{Bmatrix} = \begin{Bmatrix} O \\ X^*_f \bullet U'^*_f \end{Bmatrix} \tag{3A-20}$$

where the submatrices C^* are the complex dynamic stiffness matrices and defined by the following relation:

$$C^* = K^* - \omega^2 M \tag{3A-21}$$

where K^* is the complex stiffness matrix and M is the mass matrix, and ω is the excitation circular frequency. The subscripts used in Equation 3A-20 are defined in Figure 3A-7. The matrix X_f^* is the impedance computed for all the interacting nodes, U'^*_f is the free–field motion and U_s^* and U_f^* are the complex total displacements of the superstructure and foundation nodes, respectively.

The complex modulus method of material representation described in the previous subsection is used to compute the complex stiffness of the elements allowing for variation of material damping in each element.

Based on the above equation of motion, the solution to the harmonic soil structure interaction problem can be achieved in the following five steps.

(1) Solve the site response problem. This step involves determining the free-field displacement amplitude U'^*_f within the excavated volume. The site is modeled by horizontal soil layers resting on rigid base or elastic halfspace. The halfspace at the

bottom boundary is modeled using the variable depth method in which an extra layer of soil with a total depth of 1.5λ (where λ is the shear wave length in the halfspace) is added to the soil profile. Thus, the added soil layer depth varies with frequency. Furthermore, the added soil column is subdivided into n sublayers. A value of n = 10 is adequate and is used in the analyses.

In addition to variable depth soil layer, the halfspace simulation also uses viscous dashpots in the horizontal and vertical directions at the base of the added soil layer.

SASSI is a linear analysis program. The iteration on strain-dependent soil properties cannot be directly incorporated in the analysis. It is therefore, necessary first to perform the one-dimensional free-field soil column deconvolution analysis to obtain the equivalent linear soil properties compatible with seismic strains induced in the free-field. These properties are then used for the SASSI site response analysis. The free-field site response analysis is described in Subsection 3A.6.

(2) Solve the impedance problem. This step involves determining the matrix X_f^* which can be obtained efficiently from a series of 2-D plane strain or 3-D axisymmetric solution of sub-structure (b) in Figure 3A-7. The halfspace simulation, if requested in Step 1, will be automatically included in this step of analysis as well.

(3) Form the load vector. The seismic excitation load vector in Equation 3A-20 can be computed from the product of solutions in Steps 1 and 2.

(4) Form the complex dynamic stiffness matrix. This step involves forming the coefficient matrix C^* in the left-hand side of Equation 3A-20. The superstructure is modeled using finite elements from the finite element library of SASSI. The foundation is modeled using two sets of elements. The first set is used to represent excavated soil in the basement volume. Usually brick elements are used in 3-D analysis and four-node plane strain elements are used in 2-D analysis. The second set of elements are used to model the embedded portion of the superstructure such as the basement or side walls of the structural foundation below the grade. Similarly, brick and plane strain elements and/or plate elements are used for this purpose.

In the finite element discretization of the basement volume, the element side dimensions are selected to be smaller than or equal to 0.2 λ where λ is the shortest shear wave length of interest in the soil layer. Thus the maximum side dimension of the elements in the foundation model are governed by the shear wave velocity of the soil layer and highest frequency of interest in the analysis.

In selecting the highest frequency of analysis, the frequency content of input motion and fixed-base modal frequencies and modal mass participation factor of the superstructure are considered. Generally, it is adequate to solve the SSI problem for frequencies up to the fixed-base modal frequency whose fixed-base cumulative modal mass amounts to 90% of the total mass.

Furthermore, if the nodal points for which responses are to be computed are located on the subcomponents of the superstructure model, then the local modal frequencies corresponding to these subcomponents should be considered in selecting the highest frequency of analysis.

(5) Solve the structural problem. This step involves solving the linear equation of Equation 3A-20 for each frequency.

For seismic excitation, the above steps are performed at several selected frequencies. These frequencies are usually selected to be close to major fixed-base modal frequencies of the structure and those of its subcomponents that are of interest. An effective interpolation scheme is used to compute the response at all frequencies required by the Fourier Transform Techniques to compute the response in time domain.

3A.6 Free-Field Site Responses Analysis

The behavior of soil is nonlinear under seismic excitation. The soil nonlinearity is caused by primary and secondary nonlinear effects. The primary nonlinearity is associated with the state of deformations induced by the free-field ground motion. The secondary nonlinearity is attributed to the SSI effects. This secondary effect on structural response is usually not significant and is neglected in the appendix.

The soil nonlinearity is approximately accounted for by an iterative process to obtain equivalent linear soil properties of shear modulus and material damping corresponding to effective strains developed in the free-field. The strain-compatible properties are then assigned to be the linear soil properties in the SSI analysis. These properties also determine the maximum element size of the foundation model using the finite-element approach.

The free-field site response analysis is performed using computer program SHAKE (Reference 3A-5) which employs the principle of one-dimensional propagation of waves in the vertical direction for a system of homogeneous, visco-elastic soil layers. An equivalent linear method is used to compute strain-compatible shear modulus and material damping for each soil layer. All sites except for the upper bound R profile defined in Table 3A-5 are analyzed using the H_1 and H_2 horizontal components of input motion at the ground surface. The free-field site response analysis is not performed for the R profile, since the next hardest profile VP7 already shows that the shear modulus and material damping corresponding to the induced strain level are essentially the same as their initial values at low strains. Therefore, the SSI analysis for the R profile is performed using the initial modulus and a conservative material damping of 1%.

In all analysis cases, the input motion was prescribed at the grade level.

Table 3A-6 summarizes the results of SSE free-field response analyses in terms of the strain-compatible shear wave velocity, averaged for all soil layers of the soil deposit, and the natural frequency of the soil column. These frequencies vary from 1.18 CPS to >33 CPS reflecting the wide range of dynamic soil properties and soil deposit configurations considered in this appendix.

For the vertical earthquake component, the P-wave velocity corresponding to the calculated strain-compatible shear modulus (or shear wave velocity) for each layer is determined following the steps described in Subsection 3A.3.3. The soil damping associated with the P-wave is considered to be the same as the final iterated damping associated with the shear wave.

3A.7 Soil-Structure Interaction Analysis Cases

To establish design envelopes of seismic responses of the R/B and C/B, SSI analyses of these buildings are performed for a total of 22 cases using the properties of the 14 generic sites defined in Subsection 3A.6. SSI cases are summarized in Table 3A-7. The analyses cases can be categorized into two groups. In group 1, 3-D SSI analysis of the R/B and C/B are performed individually without consideration to structure-to-structure interaction effects (cases 1 through 16 in Table 3A-7). In group 2, 2-D SSI analysis of the R/B and C/B are performed considering individual building as well as multiple buildings including the turbine building to evaluate the structure-to-structure interaction effects (cases 17 through 22 in Table 3A-7).

Group 1 includes cases which consider variation of shear wave velocity, depth to base rock, depth to water table, separation of the basement walls from side soil medium as well as concrete cracked condition. Variation of shear wave velocity includes cases from the upper bound R cases to softest UB profiles in the 0°–180° (X) and vertical (Z) direction. Only selected profiles to cover the range of shear wave velocities are used in the 90°–270° (Y) direction. Separation of side wall is allowed for one story for both the R/B and C/B. For R/B the separation considered is 7.5m and for C/B is 4.4m. For cracked condition, properties of the building walls are reduced by 50% and the RCCV by 30%. Variation of depth to base rock is considered for two soil profiles UB and VP3 where the effect is expected to be more significant due to contrast of velocity between the rock and the respective soil velocity. Variation of depth to water table was considered for the UB profile only where larger variation in P-wave velocity is expected due to submerged conditions.

In group 2 of SSI analysis cases, the effect of structure-to-structure interaction is evaluated. These cases are analyzed using 2-D option of SASSI in the 0°–180° (X) direction. In order to provide a one to one comparison, both the R/B and C/B are analyzed individually and together with T/B(R/B + C/B + T/B cases in Table 3A-7) using UB, VP3, and VP5 velocity profiles. The effects of structure-to-structure interaction are evaluated using the 2-D results. These results are compared with the respective 3-D results of each building.

The enveloping results are based on enveloping the responses of the governing SSI cases to cover a wide range of site conditions and variation in ground motion, soil foundation separation and change of structural properties due to concrete cracked condition.

3A.8 Analysis Models

3A.8.1 Reactor Building Structural Models

The structural model of the R/B complex including the reactor pressure vessel and internal components as described in Subsection 3.7.2.1.5.1.1 is used in SSI analysis. The model figures are shown as Figures 3A-8 and 3A-9.

In the SASSI analyses of the R/B, the properties of the outer walls represented by the outer stick below the grade in Figure 3A-8 are adjusted to subtract the properties of the four embedded outer walls. These walls, as described in the following subsection, are modeled as part of the finite element foundation model.

The SSI model of the R/B is a quarter model taking advantage of the two planes of symmetry. The model properties and boundary conditions are adjusted to correspond to the quarter model. In order to include the coupling effects, rigid arms are added to the stick models at each floor elevation to obtain the response of extreme floor locations. The rigid arms are shown in Figures 3A-10 and 3A-11 in the XZ and YZ planes, respectively.

In the 2-D SSI analyses the structural model properties (stiffness and mass) are converted into values corresponding to per unit foundation depth (dimension in the third direction) to maintain compatibility with the 2-D foundation model.

3A.8.2 Reactor Building Foundation Models

3A.8.2.1 2-D Foundation Model

The 2-D model consists of the R/B foundation model for X–direction of analysis. The model developed for X-direction is also applicable for the vertical analysis. The same R/B model in conjunction with the 2-D control and turbine buildings models is used for structure-to-structure interaction cases.

The foundation model of the R/B below grade is shown in Figure 3A-12. This model consists of 60 four-node plane strain elements with the properties of concrete to represent the base slab. The side walls are modeled by beam elements. The base slab and side wall elements are massless since the mass of all structural components is already included in the structural model. The stick models are connected to the base slab by rigid beams at their respective footprints (see Figure 3A-12)

The finite element mesh in the XZ plane representing the excavated soil elements for the R/B foundation is shown in Figure 3A-13. This model consists of 384 four-node plane strain elements. The strain-compatible equivalent linear soil properties obtained from the free-field analysis are used for the soil finite elements.

To properly transfer the rotation of the stick model to the base slab (and vice versa), a set of rigid beams are placed at the top of the slab connecting each stick to its respective footprint (see Figure 3A-12). The stick representing the outer walls of the R/B is connected to the side walls by a set of rigid beams in horizontal directions to reflect the direct contact condition of the outside wall with the soil.

3A.8.2.2 3-D Foundation Model

By taking advantage of two planes of symmetry, the quarter model of R/B foundation is constructed as shown in Figure 3A-14. The excavated soil volume is shown in Figure 3A-15. The nodal points of the finite element mesh at different elevations are shown in Figures 3A-16 through 3A-24. A total of 504 brick elements are used for modeling the excavated soil in the embedded foundation volume. The base slab is modeled with 81 plate elements with rigid properties due to large thickness of the slab (see Figure 3A-16). The exterior walls of the building are also modeled with plate elements with concrete properties as shown in Figures 3A-25 and 3A-26.

The outer stick of the R/B model (Figure 3A-8) is connected to the base slab and the side walls with rigid beams as shown in Figure 3A-14. Since the sectional properties of the R/B stick model below grade are mainly due to the properties of the four side walls modeled explicitly by the plate elements, the stick model below grade, with adjusted properties, is connected to the side walls in both directions with rigid beams at each respective floor elevation (see Figure 3A-14). The foundation model shown is used in conjunction with the UB profile. For stiffer sites (VP and HR) foundation models with coarser element size is used to reduce the computer cost.

3A.8.3 Control Building Structural Model

The C/B model as described in Subsection 3.7.2.1.5.1.2 is used in SSI analysis. The model is shown in Figure 3A-27. The model has 8 lumped masses connected by massless beam elements. The shear rigidity in the direction of excitation is provided by the parallel walls. The bending rigidity includes the cross wall contribution. In the vertical direction a single mass point is used for each slab to represent the vertical slab frequency. The stick model with rigid arms is shown in Figure 3A-28. The rigid arms located at each floor elevation are used to obtain the response of extreme floor locations to include the coupling effect. Similarly, in the SASSI model of the C/B, the properties of the outer walls below the grade are subtracted from the stick model properties at respective locations. The walls are modeled as part of the finite element foundation model. The SSI model is also a quarter of model, taking advantage of two planes of symmetry in the building.

For the 2-D SSI model, the structural model properties are divided by the depth of the model to maintain compatibility with the 2-D foundation model.

3A.8.4 Control Building Foundation Models

3A.8.4.1 2-D Foundation Model

A 2-D model of the C/B foundation for X-direction of analysis was developed. The model for the X direction is also applicable for the vertical analysis.

The foundation model of the C/B in the XZ plane below grade is shown in Figure 3A-29. This model consists of 14 four-node plane strain elements with the properties of concrete to represent the base slab. The cross walls of the C/B are modeled by beam elements. The base slab and side wall elements are massless since the mass of all structural components is already included in the structural model. The structural model is connected to the base slab by a rigid beam connecting Node 2126 through Node 2130 at the top of the slab and another rigid beam connecting node 2106 through 2110 at the bottom of the slab (see Figure 3A-29). The rigid beams are not extended to the edge of the basemat in order to include the effect of basemat flexibility in the analysis.

The finite element mesh in the XZ plane representing the excavated soil elements for the C/B foundation is shown in Figure 3A-30. This model consists of 154 four-node plane strain elements. The strain-compatible equivalent linear properties obtained from the free-field analysis are used for the soil finite elements (see Appendix 3A.6 for additional information on free field analyses).

To properly transfer the rotation of the stick model to the base slab (and vice versa), a set of rigid beams are placed at the bottom of the slab connecting Nodes 2106 through Nodes 2110. The stick representing the walls of the C/B are connected to the cross walls by a set of rigid beams in the horizontal directions to reflect the direct contact condition of the outside wall with the soil.

3A.8.4.2 3-D Foundation Model

By taking advantage of two planes of symmetry, the quarter model of the C/B foundation is constructed as shown in Figure 3A-31. The excavated soil model is shown in Figure 3A-32. A total of 120 brick elements are used for modeling the excavated soil in the embedded foundation volume. The nodal points of the foundation model are shown in Figures 3A-33 through 3A-43. The base slab is modeled with 60 plate elements (see Figure 3A-33). The exterior walls of the building below grade are modeled with plate elements as shown in Figures 3A-44 and 3A-45.

The foundation model and the connection of the building stick model to the foundation are shown in Figure 3A-31. As shown in the figure, the base of the stick model is connected to the base slab by a set of rigid beams to maintain rotation compatibility. Rigid beams are placed in the walls at respective floor elevations (see Figure 3A-31) but the stick model is connected to the wall parallel to the direction of shaking. For this reason the rigid beams on the other side wall are not fully extended in the full depth of the wall so that mainly the stiffness of the wall parallel to the shaking direction is considered compatible with the assumption used in development of stick model. For shaking in the other direction, stick model is similarly connected to the side wall which is parallel to the direction of shaking. The model shown is used in conjunction with the UB soil velocity profile. For stiffer profiles coarser models are used to reduce computer cost.

3A.8.5 Turbine Building Structural and Foundation Models

The turbine building model considered is shown in Figure 3A-46. It consists of two concentric lumped-mass beam sticks representing the building structures and the turbine generator pedestal. This simple representation is sufficient since the turbine building response is of no concern in this appendix and it is only used to evaluate its effect on the R/B and C/B. This effect can be adequately accounted for by maintaining the turbine building foundation configuration and the building total mass as well as the building fundamental frequency in the simplified model representation. Turbine building model is used in the 2-D SSI analysis in conjunction with the 2-D R/B and C/B models.

The foundation is assumed to be rigid for the purpose of evaluating the structure-to-structure interaction effect on the R/B and C/B. Figure 3A-46 also shows the 2-D foundation model.

3A.9 Analysis Results

In this section, typical SSI results are presented to show the effect of different parameters in terms of soil and structural properties on seismic responses of the plant structures and components. These results are grouped selectively in order to provide a reasonable basis for comparison of the results. In this section, only limited SSI responses in terms of acceleration response spectra and seismic forces at the key locations in the plant are presented. The complete seismic responses for all cases are presented in Section 3A.10.

For comparison study, the acceleration response spectra at 2% damping are shown for the following locations.

(1) Reactor Building

Location	Node No.	Respective Figure
RPV/MS Nozzle	33	3A-9
RCCV Top	89	3A-8
R/B Top	95	3A-8
Basemat	210	3A-14
	(UB model)	
	208	
	(all other models)	

(2) Control Building

C/B Top	108/181/183	3A-28
Basemat	102	3A-31

The seismic forces are presented at the following locations.

(1) Reactor Building

Location	Element No.	Respective Figure
Shroud Support	28	3A-9
RPV Skirt	69	3A-9
RSW Base	78	3A-8
Pedestal Base	86	3A-8
RCCV–Grade Level	89	3A-8
R/B–Grade Level	99	3A-8

(2) Control Building

C/B Grade Level	6	3A-27

For comparison study, the responses in each direction (X or Y or Z) due to shaking in the respective direction are presented. In Section 3A.10, however, the coupling effect due to shaking in all 3 directions are considered. In addition the results of the upper bound soil cases (rigid soil) are shown as a reference solution with the results of all selected groups.

3A.9.1 Effect of Soil Stiffness

In order to study the effect of soil stiffness on the SSI responses, both the R/B and C/B were analyzed in the horizontal X- and vertical Z-directions for the upper bound (R2) case, and soil with velocity profiles VP7, VP5, VP4, VP3, and UB1. The results due to horizontal X- shaking are shown in Figures 3A-47 through 3A-51 for the R/B. Figure 3A-47 shows the vertical response at the edge of the basemat due to horizontal shaking. The rocking frequency of the basemat increases as the soil stiffness increases. The results at other locations in the R/B show the successive shift in the peak frequency of the response due to the change in soil stiffness.

The results due to X-shaking for the C/B are shown in Figures 3A-52 through 3A-54. Similarly, a successive shift in the peak frequency of the response is observed.

The results in the vertical direction due to Z-shaking are shown in Figures 3A-55 through 3A-58 for the R/B and in Figures 3A-59 through 3A-60 for the C/B. As shown in these figures, the results are generally governed by the upper bound (R2) case results.

The results in terms of seismic forces are compared in Table 3A-8. As shown in this table, the results of the upper bound case generally governs the seismic responses. The results of all soil cases shown are used to obtain the enveloping results (Subsection 3A.10).

3A.9.2 Effect of Depth to Base Rock

In order to evaluate the effect of depth to base rock, both the R/B and C/B were analyzed in the X-direction using soil shear wave velocity profiles UB1 and VP3 for soil column depths of 25.7m, 45.7m, 61.0m, and 91.5m. The results for the R/B are compared with the upper bound case results in Figures 3A-61 through 3A-64 for the UB1 profile and in Figures 3A-65 through 3A-68 for the VP3 profile. As shown in these figures, the effect of the depth to base rock on the floor acceleration responses is insignificant except for the shallow soil site with 25.7m depth. In this case, the bottom of the basemat is resting on the base rock. Consequently, the responses for this case are larger than other soil cases considered. The results for the C/B are compared with the upper depth bound results in Figures 3A-69 through 3A-72. Similarly, except for the shallow soil case of 25.7m, the results of all other cases are relatively close. The results in terms of seismic forces are compared in Tables 3A-9 and 3A-10 for the UB and VP3 profiles, respectively. As shown in this table, except for the shallow soil case of 25.7m, the results are relatively close for other depth to base rock cases.

The results of all cases for depth to base rock study are used to obtain the enveloping responses (Subsection 3A.10)

3A.9.3 Effect of Depth to Ground Water Table

In order to evaluate the effect of depth to water table on the seismic responses, both the R/B and C/B were analyzed using the soil profile UB1D150 in the X- and Z-directions. The depth to water tables of 0.61m, 12.2m, and 25.7m were considered. The results in terms of acceleration response spectra due to horizontal X-shaking are shown in Figures 3A-73 through 3A-76 for the R/B and in Figures 3A-77 through 3A-78 for the C/B. As shown in these figures, the effect of depth to water table on horizontal responses is relatively small. The results due to vertical Z-shaking are shown in Figures 3A-79 through 3A-82 for the R/B and in Figures 3A-83 through 3A-84 for the C/B. As shown in these figures the effect of depth to water table is relatively more significant on the vertical responses. The results in terms of seismic forces are shown in Table 3A-11. The results in terms of axial loads for the shallow water table case are larger than other depths to water table cases considered. Seismic forces due to horizontal shaking are not affected significantly.

The results of all cases considered for the depth to water table are used to obtain the enveloping results (Subsection 3A.10)

3A.9.4 Effect of Concrete Cracking

In order to evaluate the effect of concrete cracking on the SSI responses, the upper bound case (R1) was analyzed using concrete cracked properties. These properties were obtained by reducing the stiffness of the R/B and C/B walls by 50%. Properties of the RCCV were reduced by 30%. The results of uncracked case due to horizontal X-shaking (R1UX) are compared with the results of the cracked case (R1CX) in Figures 3A-85 through 3A-88 for the R/B and in Figures 3A-89 and 3A-90 for the C/B. As expected, concrete cracking reduces the peak frequency of the response and amplifies the response at most locations. The effect on the basemat response is insignificant. The effect on the seismic forces is shown in Table 3A-12. Similarly, concrete cracking also affects the seismic forces in the plant structure.

Due to significance of concrete cracking on the seismic responses, the result of both cracked and uncracked cases are used with the results of other cases to obtain the enveloping results (Subsection 3A.10).

3A.9.5 Effects of Change in Soil Degradation Curve

For all SSI soil cases, strain-compatible soil properties were obtained using free-field SHAKE analysis results as discussed in Subsection 3A.6. For the free-field analysis, the strain-dependent soil properties were obtained from the generic Seed and Idriss soil curves (Reference 3A-6) using the mean curves. Recent data indicate that the amount of soil degradation due to shaking can be less than the mean values of the generic curves and is closer to the upper bound of the 1970 soil curves (Reference 3A-6). The soil shear modulus degradation curves (mean and upper bound) are shown in Figure 3A-91 The upper bound of the 1970 soil shear modulus curve is the same as the curve reported for sand in Reference 3A-7.

In order to evaluate the effect of change in soil degradation curves on SSI responses, the soil case with velocity profile VP3 and soil column depth of 45.7m was re-analyzed using the upper bound soil degradation curve. The results of the analysis in the horizontal X-direction for this case (SSI case VP3D1AX) are compared with the respective SSI case results using the mean curve (SSI case VP3D1X) and the very rigid soil case results (R1UX) for both the R/B and C/B in Figures 3A-92 through 3A-97. As shown in these results, the effect of change in soil degradation curve on the spectral values is insignificant. The effect on the seismic forces is shown in Table 3A-13. The effect on seismic forces is also insignificant.

Based on these results, variation of soil degradation for other soil cases was not considered warranted. The results based on mean soil degradation curve were used to obtain the enveloping responses.

3A.9.6 Effect of Side Soil-Wall Separation

In order to evaluate the effect of separation between the side walls and the side soil, the upper bound case was re-analyzed considering one story height separation for each building. For the R/B, R1 cases (R1UX and R1UZ) consider separation up to Elevation 4.8m (Figure 3A-8). For the C/B, R1 cases (R1UX and R1UZ) consider separation up to Elevation 3.5m (Figure 3A-27).

The spectral results for the R/B due to horizontal X-shaking are compared with the results of R2 cases (no separation) in Figures 3A-98 through 3A-101. The results due to vertical Z-shaking are shown in Figures 3A-102 through 3A-105. As expected, separation of side walls reduces the peak frequency of the response. The reduction is less significant for vertical responses. The effect on the basement response is insignificant.

The effect on the C/B responses are shown in Figures 3A-106 through 3A-109. Similarly, the separation effect is more pronounced in the horizontal responses.

The results in terms of seismic forces are shown in Table 3A-14. As shown in the table, the effect on the axial forces is relatively insignificant. However, seismic forces due to horizontal shaking are generally larger for the case with no separation. Due to considerable effect of side soil separation on the seismic responses, the results of both cases (R1 and R2) are used with the results of other SSI cases to obtain the enveloping results (Subsection 3A.10).

3A.9.7 Effect of Adjacent Buildings

In order to evaluate the effect of structure-to-structure interaction on the seismic responses, both the R/B and C/B were analyzed in the horizontal X-direction for soil profiles UB1D150, VP3D150, and VP5D150. Each building was analyzed individually using both the 2-D and 3-D models. The analysis was repeated using the multiple 2-D models of the reactor, control, and turbine buildings in one SSI model to consider structure-to-structure interaction effects. The results for the R/B are compared with the upper bound case results in Figures 3A-110 through 3A-121 for the soil profiles UB1D150, VP3D150, and VP5D150, respectively. As shown in these figures, the effect of structure-to-structure interaction on the R/B response is relatively insignificant and in general, tends to reduce the overall response of the building. The result of structure-to-structure interaction cases are either enveloped by the respective 3-D case results or the results of the upper bound case.

The results for the C/B are compared in Figures 3A-122 through 3A-127 for soil profiles UB1D150, VP3D150, and VP5D150, respectively. As shown in these results, the effect of structure-to-structure interaction is more pronounced in the response of the C/B. The SSI frequency of the R/B is present in the C/B response through a dip in the response spectrum. Similarly, the results of structure-to-structure cases generally are either enveloped by the respective 3-D case results or the results of the upper bound case. The results in terms of seismic forces are compared in Tables 3A-15 through 3A-17 for UB1, VP3, and VP5 soil velocity profiles. As shown in these tables, the results are generally governed by the upper bound cases.

The results in terms of seismic soil pressure were also computed. As expected, seismic soil pressure in between the R/B and C/B increased due to structure-to-structure interaction effect. The seismic soil pressure results obtained from analysis of individual buildings as well as multiple buildings were enveloped and used in the design of respective walls of each building. The enveloping seismic soil pressure results are shown in Table 3A-18.

Based on these results, except for seismic soil pressure, consideration to structure-to-structure interaction effect is not warranted for the purpose of obtaining enveloping seismic forces and floor acceleration response spectra. Consequently, the enveloping results presented in Section 3A.10 are based on the analysis of each individual building.

3A.9.8 Summary

On the basis of the analysis results for a wide range of site parameters and various analysis conditions presented in the preceding subsections, the following conclusions can be made.

(1) Stiffer sites result in higher structural responses, primarily due to less ground motion attenuation with depth in the free-field for stiffer sites. The R profile can be considered to be the upper bound profile.

(2) The effects of soil depth variation and ground water table variation on structural response for the same soil profile are relatively insignificant as compared to the effects of variation in shear wave velocities (i.e., soil stiffness) considered for a range of soil/rock profiles.

(3) The effect of structure-to-structure interaction on the R/B and C/B response is bounded by the response of the individual building considering all soil cases except for the seismic soil pressure between the R/B and C/B.

(4) The effects of concrete cracking and side soil separation on the floor acceleration response spectra are relatively significant.

3A.10 Site Enveloping Seismic Response

3A.10.1 Enveloping Maximum Structural Loads

The site-envelope seismic loads are established from the envelopes of all SASSI analysis results from 3-D SSI cases summarized in Table 3A-7 (Cases 1 through 16). The site-envelope seismic loads obtained are applicable for the design of the R/B and C/B structures and associated reactor and other equipment housed in the ABWR standard plant.

The enveloping maximum shear and moment distributions along the R/B walls, RCCV, RSW, pedestal and key RPV internal components are shown in Table 3A-19a through 3A-19d. The shear and moment are the envelope of all SSI cases additionally enveloped over the enveloped responses due to X-(0°–180°) and Y-(90°–270°) shaking. The torsional moments for building structures are obtained using the enveloping shear force at each elevation multiplied by 5% eccentricity of the respective maximum floor dimension.

Similarly, the results for the C/B were obtained. These results are shown in Table 3A-20.

It should be noted that for design purposes, no credit has been taken for the reduction of horizontal forces at the R/B and C/B elements below the ground surface as predicted from analysis. The largest shear force above the ground surface is maintained as minimum for these elements.

The vertical loads are expressed in terms of enveloping absolute acceleration. The enveloping maximum acceleration values for the R/B and C/B are shown in Tables 3A-21a through 3A-21c and 3A-22. These acceleration values do not include the coupling effect and are only applicable for structural analysis in combination with the seismic loads due to horizontal shaking. For equipment design, however, SRSS was used to include the coupling vertical response due to horizontal shaking. These results are presented in Section 3A.10.3.

3A.10.2 Enveloping Floor Response Spectra

The site-envelope floor response spectra due to the 0.3g SSE are obtained according to the following steps.

(1) The calculated 2, 3, 5, and 10% damping response spectra of all SASSI 3-D cases are enveloped at all required locations in each of the three directions. In the vertical direction, where applicable, SRSS was used to include the coupling vertical response due to horizontal shaking.

(2) The envelope spectra in the two horizontal directions at each location obtained in step 1 are subsequently enveloped to form the bounding horizontal spectra.

(3) The envelope spectra were subsequently peak broadened by ±15%.

The site-envelope peak broadened SSE floor response spectra at critical damping ratios 2, 3, 5, and 10% for the R/B are shown in Figures 3A-128 through 3A-165 for the horizontal direction and in Figures 3A-166 through 3A-209 for the vertical direction including floor oscillator responses. The vertical responses for the reactor building walls also include the responses of the finite element model (Section 3H.1) for the upper bound R1 case. The results for the C/B are shown in Figures 3A-210 through 3A-228. For seismic design of equipment and piping, the alternative seismic input can be individual floor response spectra of each site condition considered in generating the site-envelope spectra.

Furthermore, vertical response spectra for floor oscillators should be used for the components supported by the floor slabs. However, for the components in the vicinity of the walls or on the walls, the building vertical response spectra at respective elevation can be used.

3A.10.3 Enveloping Maximum Absolute Accelerations

The site-envelope absolute acceleration responses (enveloping maximum zero period accelerations) are shown in Tables 3A-23a through 3A-23d and 3A-24 for the R/B and C/B. The vertical responses include the coupling vertical response due to horizontal shaking where applicable.

3A.10.4 Enveloping Maximum Relative Displacements

The site-envelope maximum relative displacements with respect to input motion at grade level in the free-field are shown in Tables 3A-25a through 3A-25d and 3A-26 for the R/B and C/B. These results may be used for design of components supported on the surrounding soil medium and connected to the respective building.

The site-envelope maximum relative displacements of the nodal points with respect to the base of each respective stick model are shown in Tables 3A-27a through 3A-27d and 3A-28. These results may be used for design of components located within the respective building.

3A.10.5 Summary

The site-envelope maximum seismic responses presented in Subsection 3A.10 envelop the maximum seismic SSE responses of the ABWR plant structures and components for a wide range of subsurface properties and conditions as well as the effect of concrete cracking and side soil-wall separation. These responses are used to design the ABWR plant structures and components.

3A.11 References

3A-1 Appendix 3A, General Electric Company BWR/6-238 Standard Safety Analysis Report (GESSAR), Docket No. STN 50-447, July 30, 1973.

3A-2 Appendix 3A, General Electric Company GESSAR II BWR/6 Nuclear Island Design (22A7007), March 1980.

3A-3 NUREG/CR-1161, Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria, May 1980.

3A-4 Lysmer, J., Tabatabaie-Raissi, M., Tajirian, F., Vahdani, S., and Ostadan, F., SASSI--A System for Analysis of Soil-Structure Interaction, Report No. UC/B/GT/81-02, Geotechnical Engineering, University of California, Berkeley, CA, April, 1981; also Ostadan, F., Computer Program SASSI, CE (944), Theoretical, User’s and Validation Manuals (1991), Bechtel Corporation, San Francisco, California

3A-5 Schnabel, P.B., Lysmer, J., and Seed, H.B., SHAKE--A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites, Report No. RC 72-12, Earthquake Engineering Research Center, University of California, Berkeley, CA, 1972.

3A-6 Seed, H.B. and Idriss, I.M–Soil Moduli and Damping Factors for Dynamic Response Analysis, Report No. RC 70-10, Earthquake Engineering Research Center, University of California, Berkeley, CA, 1970.

3A-7 Idriss, I.M–Response of Soft Soil Sites During Earthquakes, H. Bolton Seed Memorial Symposium Proceedings, Volume 2, Bi Tech Publishers, May 1990.

Table 3A-1 Soil Properties for UB Profile

Layer Depth m (Nominal)	K_{2max}	Total Unit Weight t/m^3	Poisson's[1] ratio
0–9	140	1.92	0.38
9–15	140	2.00	0.35
15–22.5	140	2.00	0.35
22.5–30	140	2.00	0.35
30–45	160	2.09	0.32
45–60	160	2.09	0.32
60–90	240	2.17	0.30

1 Total unit weight and Poisson's ratio for each layer are also used for other soil profiles. For rock profiles the total unit weight is 2.2 t/m^3 and Poisson's ratio is 0.3.

Table 3A-2 Average Shear Wave Velocities in Layers

	Shear Wave Velocity Profiles					
Layer Depth (m) (nominal)	UB (m/s)	VP3 (m/s)	VP4 (m/s)	VP5 (m/s)	VP7 (m/s)	R (m/s)
0–9	303	549	732	887	1524	6096
9–15	374	576	738	893	1524	6096
15–22.5	418	585	756	912	1524	6096
22.5–30	455	610	762	915	1524	6096
30–45	524	622	768	918	1524	6096
45–60	573	643	793	930	1524	6096
60–90	758	674	817	945	1524	6096

Table 3A-3 Strain-Dependent Shear Modulus

Effective Shear Strain γ_{eff}(%)	$LOG_{10}\gamma_{eff}$	Modulus Reduction Factor[1] UB, VP3 & VP4	VP5 & VP7
1.00E–4	–4.0	1.00	1.00
3.16E–4	–3.5	0.98	1.00
1.00E–3	–3.0	0.96	0.99
3.16E–3	–2.5	0.88	0.95
1.00E–2	–2.0	0.74	0.90
3.16E–2	–1.5	0.52	0.81
1.00E–1	–1.0	0.29	0.72
0.316	–0.5	0.15	0.40
1.00	0.0	0.06	0.40
10.00	1.0	0.06	0.40

1 This is factor has to be applied to the shear modulus at low shear strain amplitudes defined here as 1E–4% to obtain the modulus at higher strain levels.

Table 3A-4 Strain-Dependent Soil Damping

Effective Shear Strain γ_{eff}(%)	$LOG_{10}\gamma_{eff}$	% Of Critical Damping UB, VP3, VP4	VP5, VP7
1.00E–4	–4.0	0.60	0.60
3.16E–4	–3.5	0.8	0.70
1.00E–3	–3.0	1.7	0.8
3.16E–3	–2.5	3.1	1.1
1.00E–2	–2.0	5.6	1.5
3.16E–2	–1.5	9.6	2.1
1.00E–1	–1.0	15.00	3.0
0.316	–0.5	15.00	–
1.00	0.0	15.00	–
3.16	0.5	15.00	–
10.00	1.0	15.00	–

Table 3A-5 Case IDs for Site Conditions Considered

Soil Profile Depth m	Depth to Water Table m	Depth to Base Rock m	Shear Wave Velocity Profiles					
			UB	VP3	VP4	VP5	VP7[1]	R^2 Upper Bound
25.7	.61	25.7	UB1D85a	VP3D85a			–	–
45.7	.61	45.7	UB1D150	VP3D150	VP4D150	VP5D150		–
	12.2	45.7	UB2D150					
	25.7	45.7	UB3D150					
61	.61	61	UB1D200	VP3D200	–	–	–	–
91.5	.61	91.5	UB1D300	VP3D300	–	–	–	–

1 These are uniform sites.

2 These are uniform sites. One-story height separation (R1 Cases) and no separation of structures (R2 Cases) with side soil are considered for this site.

Table 3A-6 SSE Free-Field Site Response Results for all Soil Profiles (Average Properties)

Soil Profile ID	Average Shear Wave Velocity m/s	Soil Column Frequency (hertz)
UB1D85a	283	2.73
VP3D85a	499	4.82
UB1D150	320	1.75
UB2D150	320	1.75
UB3D150	320	1.75
VP3D150	487	2.67
VP4D150	659	3.60
VP5D150	877	4.80
UB1D200	345	1.41
VP3D200	483	1.98
UB1D300	431	1.18
VP3D300	486	1.33
VP7D300	1486	4.06
R Cases	6097	> 33

Table 3A-7 Summary of SSI Cases Considered (Reactor and Control Buildings)

Case No.	Soil Case ID	Soil Profile	Depth to Base Rock m	Depth to Water Table m	Uncracked/ Cracked	Input Motion	Dimension of Analysis	SSI Case ID (X)	SSI Case ID (Y)	SSI Case ID (Z)	Description
1	R1U	Upper Bound	NA	NA	UNCRKD	RG	3-D	R1UX	R1UY	R1UZ	Hard Rock, with separation
2	R2U	Upper Bound	NA	NA	UNCRKD	RG	3-D	R2UX	R2UY	R2UZ	Hard Rock, no separation
3	R1C	Upper Bound	NA	NA	CRKD	RG	3-D	R1CX	R1CY	NA	Cracked Case
4	VP7D150	VP7	45.7	NA	UNCRKD	RG	3-D	VP7D1X	VP7D1Y	VP7D1Z	SSI Cases
5	VP5D150	VP5	45.7	0.61	UNCRKD	RG	3-D	VP5DX	VP5DY	VP5DZ	SSI Cases
6	VP4D150	VP4	45.7	0.61	UNCRKD	RG	3-D	VP4D1X	—	—	SSI Cases
7	VP3D150	VP3	45.7	0.61	UNCRKD	RG	3-D	VP3D1X	VP3D1Y	VP3D1Z	SSI Cases
8	UB1D150	UB	45.7	0.61	UNCRKD	RG	3-D	UB1D1X	UB1D1Y	UB1D1Z	SSI Cases
9	VP3D85	VP3	25.7	0.61	UNCRKD	RG	3-D	VP3D3X	—	—	Depth to base rock study
10	VP3D200	VP3	61.0	0.61	UNCRKD	RG	3-D	VP3D4X	—	—	Depth to base rock study
11	VP3D300	VP3	91.5	0.61	UNCRKD	RG	3-D	VP3D5X	—	—	Depth to base rock study
12	UB1D85	UB	25.7	0.61	UNCRKD	RG	3-D	UB1D3X	—	—	Depth to base rock study
13	UB1D200	UB	61.0	0.61	UNCRKD	RG	3D	UB1D4X	—	—	Depth to base rock study
14	UB1D300	UB	91.5	0.61	UNCRKD	RG	3-D	UB1D5X	—	—	Depth to base rock study
15	UB2D150	UB	45.7	12.2	UNCRKD	RG	3-D	UB2D1X	—	UB2D1Z	Water table location study

Table 3A-7 Summary of SSI Cases Considered (Reactor and Control Buildings) (Continued)

Case No.	Soil Case ID	Soil Profile	Depth to Base Rock m	Depth to Water Table m	Uncracked/ Cracked	Input Motion	Dimension of Analysis	SSI Case ID (X)	SSI Case ID (Y)	SSI Case ID (Z)	Description
16	UB3D150	UB	45.7	25.7	UNCRKD	RG	3-D	UB3D1X	—	UB3D1Z	Water table location study
17	UB1D150	UB	45.7	0.61	UNCRKD	RG	2-D	UB1D1X2	—		One Building: R/B, C/B
18	VP3D150	VP3	45.7	0.61	UNCRKD	RG	2-D	VP3D1X2	—		One Building: R/B, C/B
19	VP5D150	VP5	45.7	0.61	UNCRKD	RG	2-D	VP5D1X2	—		One Building: R/B, C/B
20	UB1D150	UB	45.7	0.61	UNCRKD	RG	2-D	RCTUB1X	—		R/B + C/B + T/B
21	VP3D150	VP3	45.7	0.61	UNCRKD	RG	2-D	RCTVP3X	—		R/B + C/B + T/B
22	VP5D150	VP5	45.7	0.61	UNCRKD	RG	2-D	RCTVP5X	—		R/B + C/B + T/B

Table 3A-8 Effect Of Soil Stiffness on Maximum Forces

			Soil Stiffness					
			Stiff				Soft	
Beam Element	Location	Response Type	R2UX/ R2UZ	VP7D5X/ VP7D5Z	VP5DX/ VP5DZ	VP4D1X	VP3D1X/ VP3D1Z	UB1D1X / UB1D1Z
a. Reactor Building								
28	Shroud Support	Shear	237	258	244	211	178	98
		Moment	14.14	15.49	15.03	12.82	10.66	6.35
		Axial	151	96	111		96	82
69	RPV Skirt	Shear	746	663	719	603	540	438
		Moment	38.36	42.11	40.32	34.94	32.11	20.92
		Axial	976	634	725		645	543
78	RSW Base	Shear	630	646	619	545	460	359
		Moment	32.10	32.94	31.27	28.01	23.37	17.63
		Axial	413	334	319		314	286
86	Pedestal Base	Shear	725	1,197	1,828	1,867	1,990	1,704
		Moment	60.20	198.40	319.22	333.73	347.27	319.12
		Axial	3,492	3,196	2,869		2,497	2,513
89	RCCV at Grade	Shear	14,620	14,330	13,120	12,120	10,130	7,269
		Moment	1,692.69	1,614.23	1,159.19	1,002.28	923.13	575.08
		Axial	22,490	20,000	15,140		12,090	12,100
99	R/B at Grade	Shear	27,630	26,650	27,730	24,760	20,790	11,430
		Moment	6,135.26	6,796.25	5,496.82	5,043.74	4,677.94	2,444.89
		Axial	17,050	20,070	15,040		12,610	12,310
b. Control Building								
6	C/B at Grade	Shear	4,353	4,957	4,716	4,462	3,497	2,564
		Moment	321.57	385.81	369.33	356.78	282.73	217.81
		Axial	3,360	3,616	3,429		3,180	2,998

Units: Shear & Axial Forces in Metric Ton (t); Moment in Meganewton-meters(MN-m)

Table 3A-9 Effect Of Depth to Base Rock, UB Case

				Depth to Base Rock			
				(25.7m)	(45.7m)	(61.0m)	(91.5m)
Beam Element	Location	Response Type	R1UX	UB1D3X	UB1D1X	UB1D4X	UB1D5X
a. Reactor Building							
28	Shroud Support	Shear	287	191	98	114	111
		Moment	18.83	11.07	6.35	7.09	7.06
69	RPV Skirt	Shear	798	647	438	449	427
		Moment	36.10	32.56	16.40	16.97	16.16
78	RSW Base	Shear	747	602	359	367	374
		Moment	38.88	30.24	17.63	17.78	17.84
86	Pedestal Base	Shear	2,218	2,481	1,704	1,830	1,668
		Moment	350.01	508.49	319.12	315.59	299.31
89	RCCV at Grade	Shear	17,010	10,070	7,269	10,940	8,104
		Moment	1,823.12	915.78	575.08	660.31	605.09
99	R/B at Grade	Shear	34,600	16,000	11,430	12,620	13,200
		Moment	8,073.12	4,055.19	2,444.89	2,719.48	2,535.10
b. Control Building							
6	C/B at Grade	Shear	5,624	2,425	2,564	2,320	2,294
		Moment	438.86	208.50	217.81	200.26	193.20

Units: Shear & Axial Forces in Metric Ton (t); Moment in Meganewton-meters (MN-m)

Table 3A-10 Effect Of Depth to Base Rock, VP3 Case

				Depth to Base Rock			
				(25.7m)	(45.7m)	(61.0m)	(91.5m)
Beam Element	Location	Response Type	R1UX	VP3D3X	VP3D1X	VP3D4X	VP3D5X
a. Reactor Building							
28	Shroud Support	Shear	287	178	178	166	164
		Moment	18.83	11.04	10.66	9.93	9.70
69	RPV Skirt	Shear	798	585	540	517	531
		Moment	43.85	35.68	32.11	28.55	29.01
78	RSW Base	Shear	747	515	460	442	427
		Moment	38.88	25.90	23.37	21.13	21.36
86	Pedestal Base	Shear	2,218	2,256	1,990	1,883	1,852
		Moment	350.01	425.04	347.27	326.77	236.57
89	RCCV at Grade	Shear	17,010	10,490	10,130	9,442	9,222
		Moment	1,823.12	887.63	923.13	759.65	727.48
99	R/B at Grade	Shear	34,600	20,270	20,790	17,720	18,020
		Moment	8,073.12	3,958.11	4,677.94	3,808.06	3,777.66
b. Control Building							
6	C/B at Grade	Shear	5,624	4,279	3,497	3,542	3,893
		Moment	438.86	310.39	282.73	287.35	31157

Units: Shear & Axial Forces in Metric Ton (t); Moment in Meganewton-Meters (MN-m)

Table 3A-11 Effect Of Depth to Water Table Location

Beam Element	Location	Response Type	R1UX/ R1UZ	GWT Below Grade (0.61m) UB1D1X/ UB1D1Z	(12.2m) UB2D1X/ UB2D1Z	(25.7m) UB3D1X/ UB3D1Z
a. Reactor Building						
28	Shroud Support	Shear	287	98	109	106
		Moment	18.83	6.35	6.31	7.35
		Axial	150	82	74	59
69	RPV Skirt	Shear	798	438	419	462
		Moment	43.85	20.92	20.30	21.78
		Axial	974	543	493	396
78	RSW Base	Shear	747	359	309	351
		Moment	38.88	17.63	14.84	16.87
		Axial	401	286	260	209
86	Pedestal Base	Shear	2,218	1,704	2,103	2,269
		Moment	350.01	319.12	392.28	414.15
		Axial	3,449	2,513	2,299	1,931
89	RCCV at Grade	Shear	17,010	7,269	7,973	7,326
		Moment	1823.12	575.08	402.80	604.31
		Axial	22,740	12,100	10,600	10,080
99	R/B at Grade	Shear	34,600	11,430	13,640	12,980
		Moment	8,073.12	2,444.89	3,136.28	3,034.29
		Axial	18,190	12,310	10,960	9,634
b. Control Building						
6	C/B at Grade	Shear	5,624	2,564	3,008	2,849
		Moment	430.86	217.81	253.12	240.66
		Axial	3,612	2,998	2,741	2,354

Units: Shear & Axial Forces in Metric Ton (t); Moment in Meganewton-meters (MN-m)

Table 3A-12 Effect of Concrete Cracking

Beam Element	Location	Response Type	Cracking Assumptions: Uncracked R1UX	Cracking Assumptions: Cracked R1CX
a. Reactor Building				
28	Shroud Support	Shear	287	414
		Moment	18.83	29.16
69	RPV Skirt	Shear	798	1,146
		Moment	43.85	69.16
78	RSW Base	Shear	747	1,043
		Moment	38.88	53.06
86	Pedestal Base	Shear	2,218	2,519
		Moment	350.01	483.19
89	RCCV at Grade	Shear	17,010	24,820
		Moment	1,823.12	1,504.39
99	R/B at Grade	Shear	34,600	36,790
		Moment	8,073.12	9,158.76
b. Control Building				
6	C/B at Grade	Shear	5,624	7,400
		Moment	438.86	593.52

Units: Shear & Axial Forces in Metric Ton (t); Moment in Meganewton-Meters (MN-m)

Table 3A-13 Effect of Change in Soil Degradation Curves

			Soil Degradation Curves		
				Average	Upper Bound
Beam Element	Location	Response Type	R1UX	VP3D1X	VP3D1AX
a. Reactor Building					
28	Shroud Support	Shear	287	178	252
		Moment	18.83	10.66	11.46
69	RPV Skirt	Shear	798	540	576
		Moment	43.85	32.11	34.74
78	RSW Base	Shear	747	460	508
		Moment	38.88	23.37	25.74
86	Pedestal Base	Shear	2,218	1,990	1,917
		Moment	350.01	347.27	350.89
89	RCCV at Grade	Shear	17,010	10,130	10,770
		Moment	1,823.12	923.13	951.48
99	R/B at Grade	Shear	34,600	20,790	22,090
		Moment	8,073.12	4,677.94	4,842.70
b. Control Building					
6	C/B at Grade	Shear	5,624	3,497	3,764
		Moment	438.86	282.74	308.14

Units: Shear & Axial Forces in Metric Ton (t); Moment in Metanewton-Meters (MN-m)

Table 3A-14 Effect Of Separation Between the Side Soil and Foundation Walls

Beam Element	Location	Response Type	Separating Assumptions: With Separation R1UX/R1UZ	Separating Assumptions: No Separation R2UX/R2UZ
a. Reactor Building				
28	Shroud Support	Shear	287	237
		Moment	18.83	14.14
		Axial	150	151
69	RPV Skirt	Shear	798	746
		Moment	43.85	38.36
		Axial	974	976
78	RSW Base	Shear	747	630
		Moment	38.88	32.10
		Axial	401	413
86	Pedestal Base	Shear	2,218	725
		Moment	350.01	60.20
		Axial	3,449	3,492
89	RCCV at Grade	Shear	17,010	14,620
		Moment	1,823.12	1,692.69
		Axial	22,740	22,490
99	R/B at Grade	Shear	34,600	27,630
		Moment	8,073.12	6,135.26
		Axial	18,190	17,050
b. Control Building				
6	C/B at Grade	Shear	5,624	4,353
		Moment	438.86	321.57
		Axial	3,612	3,360

Units: Shear & Axial Forces in Metric Ton (t); Moment in Meganewton-Meter (MN-m)

Table 3A-15 Effect Of Adjacent Buildings, UB Case

			Model in SASSI Analysis			
			3-D R/B	3-D R/B	2-D R/B	2-D R/B+C/B+T/B
Beam Element	Location	Response Type	ENV. of R1UX/R2UX	UB1D1X	UB1D1X2	RCTU1X
a. Reactor Building						
28	Shroud Support	Shear	287	98	115	106
		Moment	18.83	6.35	6.96	6.78
69	RPV Skirt	Shear	798	438	516	419
		Moment	43.85	20.92	25.98	21.52
78	RSW Base	Shear	747	359	377	325
		Moment	38.88	17.63	18.37	15.64
86	Pedestal Base	Shear	2,218	1,704	2,154	2,126
		Moment	350.01	319.12	395.93	394.83
89	RCCV at Grade	Shear	17,010	7,269	7,370	5,970
		Moment	1,823.12	575.08	626.28	594.40
99	R/B at Grade	Shear	34,600	11,430	15,380	13,700
		Moment	8,073.12	2,444.89	3,125.49	3,272.60
b. Control Building						
6	C/B at Grade	Shear	5,624	2,564	3,160	2,787
		Moment	438.86	217.81	264.69	266.69

Units: Shear & Axial Forces in Metric Ton (t); Moment in Meganewton-Meters (MN-m)

Table 3A-16 Effect Of Adjacent Buildings, VP3 Case

			Model in SASSI Analysis			
			3-D R/B	3-D R/B	2-D R/B	2-D R/B+C/B+T/B
Beam Element	Location	Response Type	ENV. of R1UX/R2UX	VP3D1X	VP3D1X2	RCTV3X
a. Reactor Building						
28	Shroud Support	Shear	287	178	181	153
		Moment	18.83	10.66	11.19	11.06
69	RPV Skirt	Shear	798	540	525	455
		Moment	43.85	32.11	27.84	24.77
78	RSW Base	Shear	747	460	420	382
		Moment	38.88	23.37	20.67	18.42
86	Pedestal Base	Shear	2,218	1,990	2,099	2,556
		Moment	350.01	347.27	352.66	326.87
89	RCCV at Grade	Shear	17,010	10,130	7,264	8,244
		Moment	1,823.12	923.13	662.46	688.26
99	R/B at Grade	Shear	34,600	20,790	14,680	18,690
		Moment	8,073.12	4,677.94	3,707.05	3,555.04
b. Control Building						
6	C/B at Grade	Shear	5,624	3,497	4,208	3,030
		Moment	438.86	282.74	363.25	257.40

Units: Shear & Axial Forces in MetricTon (t); Moment in Meganewton-Meters (MN-m)

Table 3A-17 Effect Of Adjacent Buildings, VP5 Case

Beam Element	Location	Response Type	Model in SASSI Analysis 3-D R/B ENV. of R1UX/R2UX	3-D R/B VP5DX	2-D R/B VP5D1X2	2-D R/B+C/B+T/B RCTV5X
a. Reactor Building						
28	Shroud Support	Shear	287	244	330	335
		Moment	18.83	15.03	26.88	29.03
69	RPV Skirt	Shear	798	719	740	684
		Moment	43.8	40.32	39.41	36.12
78	RSW Base	Shear	747	619	579	567
		Moment	38.88	31.27	29.61	27.99
86	Pedestal Base	Shear	2,218	1,828	2,280	1,802
		Moment	350.01	319.22	383.06	343.54
89	RCCV at Grade	Shear	17,010	13,120	13,220	10,550
		Moment	1,823.12	1,159.19	1,014.04	1,001.29
99	R/B at Grade	Shear	34,600	27,730	28,690	22,070
		Moment	8,073.12	5,496.82	5,671.39	4,717.17
b. Control Building						
6	C/B at Grade	Shear	5,624	4,716	4,940	3,895
		Moment	438.86	369.33	393.75	289.53

Units: Shear & Axial Forces in MetricTon (t); Moment in Meganewton-meters (MN-m)

Table 3A-18 Effect of Adjacent Buildings Enveloping Seismic Soil Pressures

Elevation (m)	R/B (kPa)	C/B (kPa)
12.0 to 9.9	921.86	921.86
9.9 to 7.9	431.51	431.51
7.9 to 6.3	215.75	254.98
6.3 to 4.8	156.91	137.30
4.8 to 3.5	147.11	215.75
3.5 to 0.90	117.68	137.30
0.90 to –1.7	166.72	166.72
–1.7 to –3.8	235.37	235.37
–3.8 to –5.9	156.91	333.44
–5.9 to –8.2	392.28	392.28

Table 3A-19a ABWR Reactor Building Walls and Floors Summary of Enveloping Seismic Loads

Element No.	Node	Elev TMSL(m)	Max. Shear (t)	Max. Moment (MN-m)	Max. Torsion (MN-m)
93	95	49.70	8,700	519.77	254.98
	96	38.20	8,700	1,471.05	
94	96	38.20	19,000	2,059.47	539.39
	98	31.70	19,000	2,942.10	
96	98	31.70	25,000	3,628.59	715.91
	100	23.50	25,000	5,295.78	
98	100	23.50	35,000	5,688.06	1,078.77
	102	18.10	35,000	7,159.11	
99	102	18.10	40,000	7,257.18	1,176.84
	103	12.30	40,000	9316.65	
100	103	12.30	53,000	9,414.72	1,569.12
	104	4.80	53,000	13,729.80	1,569.12
101	104	4.80	53,000	13,729.80	1,569.12
	105	–1.70	53,000	16,671.90	
102	105	–1.70	53,000	16,671.90	1,569.12
	88	–8.20	53,000	20,594.70	

NOTE: (1) Seismic loads are the envelopes of horizontal X and Y direction responses

Table 3A-19b ABWR Reactor Building RCCV Summary of Enveloping Seismic Loads

Element No.	Node	Elev TMSL(m)	Max. Shear (t)	Max. Moment (Mn-m)	Max. Torsion (Mn-m)
87	89	31.70	9,500	951.28	74.53
	90	23.50	9,500	1,471.05	
88	90	23.50	23,000	1,765.26	362.86
	91	18.10	23,000	2,353.68	
89	91	18.10	25,000	3,138.24	402.09
	92	12.30	25,000	3,824.73	
90	92	12.30	25,000	4,609.29	402.09
	93	4.80	25,000	5,688.06	
91	93	4.80	25,000	5,982.27	402.09
	94	–1.70	25,000	7,257.18	
92	94	–1.70	26,000	7,257.18	421.70
	88	–8.20	26,000	8,826.30	

NOTE: (1) Seismic loads are the envelopes of horizontal X and Y direction responses

Table 3A-19c ABWR Reactor Building RSW/Pedestal Summary of Enveloping Seismic Loads

Element No.	Node	Elev TMSL(m)	Max. Shear (t)	Max. Moment (MN-m)	Max. Torsion (MN-m)
74	70	21.20	80	0	0.39
	78	18.44	80	2.06	
75	78	18.44	630	2.06	3.33
	79	17.02	630	10.79	
76	79	17.02	700	10.79	3.73
	80	15.60	700	20.59	
77	80	15.60	980	20.59	5.10
	81	13.95	980	36.29	
78	81	13.95	1,100	36.29	5.49
	82	12.30	1,100	53.94	
79	82	12.30	2,400	53,94	16.67
	71	8.20	2,400	107.88	
80	71	8.20	2,700	137.30	18.63
	83	7.00	2,700	156.91	
81	83	7.00	2,700	156.91	18.63
	72	4.50	2,700	215,75	
103	72	4.50	2,700	215.75	18.63
	84	3.50	2,700	245,18	
82	84	3.50	2,800	245.18	19.61
	73	1.70	2,800	294.21	

Table 3A-19c ABWR Reactor Building RSW/Pedestal Summary of Enveloping Seismic Loads (Continued)

Element No.	Node	Elev TMSL(m)	Max. Shear (t)	Max. Moment (MN-m)	Max. Torsion (MN-m)
83	73	1.70	29.42	294.21	20.59
	85	–0.18	29.42	343.25	
84	85	–0.18	29.42	343.25	20.59
	86	–2.10	29.42	402.09	
85	86	–2.10	30.40	402.09	21.58
	87	–4.70	30.40	470.74	
86	87	–4.70	31.38	470.74	21.58
	88	–8.20	31.38	588.42	

NOTE: (1) Seismic loads are the envelopes of horizontal X and Y direction responses

Table 3A-19d ABWR Reactor Building Key RPV/Internal Components Summary of Enveloping Seismic Loads

Element No.	Node	Elev TMSL(m)	Max. Shear (t)	Max. Moment (MN-m)	Max. Torsion (MN-m)
28	56	7.39	420	27.46	1.57
	27	6.75	420	29.42	1.57
69	46	9.29	1,200	58.842	9.61
	71	8.20	1,200	69.63	9.61

NOTE: (1) Seismic loads are the envelopes of horizontal X and Y direction responses

Table 3A-20 ABWR Control Building Summary of Enveloping Seismic Loads

Element No.	Node	Elev TMSL(m)	Max. Shear (t)	Max. Moment (MN-m)	Max. Torsion (MN-m)
7	108	22.2	4,000	50.01	117.68
	107	17.15	4,000	254.98	117.68
6	107	17.15	8,200	333.44	225.56
	106	12.3	8,200	686.49	225.56
5	106	12.3	11,000	735.53	294.21
	105	7.9	11,000	1,176.84	294.21
4	105	7.9	11,000	1,176.84	294.21
	104	3.5	11,000	1,667.19	294.21
3	104	3.5	11,000	1,667.19	294.21
	103	–2.15	11,000	2,255.61	294.21
2	103	–2.15	11,000	2,255.61	294.21
	102	–8.2	11,000	2,942.10	294.21

NOTE: (1) Seismic loads are the envelopes of horizontal x and y direction responses.

Table 3A-21a ABWR Reactor Building Walls and Floors Summary of Enveloping Maximum Vertical Accelerations

Node	Elev TMSL(m)	Location	Max. Vertical Acceleration (g)
95	49.70	R/B	0.62
96	38.20	R/B	0.51
98	31.70	R/B	0.47
100	23.50	R/B	0.43
102	18.10	R/B	0.38
103	12.30	R/B	0.34
104	4.80	R/B	0.31
105	–1.70	R/B	0.31
88	–8.20	R/B	0.30
107	31.70	R/B FLOOR	1.20
108	23.50	R/B FLOOR	1.55
109	18.10	R/B FLOOR	1.84
110	12.30	R/B FLOOR	1.02
111	4.80	R/B FLOOR	0.54
112	–1.70	R/B FLOOR	0.46

NOTE: Vertical values do not include the coupling effect due to horizontal shaking

Table 3A-21b ABWR Reactor Building RCCV Summary of Enveloping Maximum Vertical Accelerations

Node	Elev TMSL(m)	Location	Max. Vertical Acceleration (g)
89	31.70	RCCV	0.71
90	23.50	RCCV	0.67
91	18.10	RCCV	0.64
92	12.30	RCCV	0.57
93	4.80	RCCV	0.43
94	–1.70	RCCV	0.34
88	–8.20	RCCV	0.30

NOTE: Vertical values do not include the coupling effect due to horizontal shaking

Table 3A-21c ABWR Reactor Building RSW/PED Summary of Enveloping Maximum Vertical Accelerations

Node	Elev TMSL(m)	Location	Max. Vertical Acceleration (g)
70	21.20	RSW/PED	0.41
78	18.44	RSW/PED	0.41
79	17.02	RSW/PED	0.41
80	15.60	RSW/PED	0.40
81	13.95	RSW/PED	0.39
82	12.30	RSW/PED	0.38
71	8.20	RSW/PED	0.37
83	7.00	RSW/PED	0.36
72	4.50	RSW/PED	0.34
84	3.50	RSW/PED	0.34
73	1.70	RSW/PED	0.33
85	–0.18	RSW/PED	0.32
86	–2.10	RSW/PED	0.31
87	–4.70	RSW/PED	0.31
88	–8.20	RSW/PED	0.30

NOTE: Vertical values do not include the coupling effect due to horizontal shaking

Table 3A-22 ABWR Control Building Summary of Enveloping Maximum Vertical Accelerations

Node	Elev TMSL(m)	Location	Max. Vertical Acceleration (g)
108	22.2	C/B STICK	0.37
107	17.15	C/B STICK	0.35
106	12.3	C/B STICK	0.32
105	7.9	C/B STICK	0.32
104	3.5	C/B STICK	0.31
103	–2.15	C/B STICK	0.31
102	–8.2	C/B STICK	0.30
113	17.15	C/B FLOOR	0.62
112	12.3	C/B FLOOR	0.58
111	7.9	C/B FLOOR	0.55
110	3.5	C/B FLOOR	0.51
109	–2.15	C/B FLOOR	0.49

NOTE: Vertical values do not include the coupling effect due to horizontal shaking.

Table 3A-23a ABWR Reactor Building Walls and Floors Summary of Enveloping Maximum Accelerations

Node No.	Elev TMSL(m)	Location	Max. Horizontal Acceleration (g)	Max. Vertical Acceleration (g)
95	49.70	R/B	1.44	1.03
96	38.20	R/B	1.10	0.83
98	31.70	R/B	0.93	0.80
100	23.50	R/B	0.75	0.63
102	18.10	R/B	0.59	0.50
103	12.30	R/B	0.47	0.39
104	4.80	R/B	0.32	0.33
105	–1.70	R/B	0.31	0.32
88	–8.20	R/B	0.31	0.33
107	31.70	R/B FLOOR	—	1.20
108	23.50	R/B FLOOR	—	1.55
109	18.10	R/B FLOOR	—	1.84
110	12.30	R/B FLOOR	—	1.02
111	4.80	R/B FLOOR	—	0.54
112	–1.70	R/B FLOOR	—	0.46

NOTE: (1) Horizontal accelerations are the envelopes of X and Y direction responses
(2) The results represent the maximum responses at the respective floor elevation

Table 3A-23b ABWR Reactor Building RCCV Summary of Enveloping Maximum Accelerations

Node No.	Elev TMSL(m)	Location	Max. Horizontal Acceleration (g)	Max. Vertical Acceleration (g)
89	31.70	RCCV	0.96	0.82
90	23.50	RCCV	0.82	0.95
91	18.10	RCCV	0.65	0.89
92	12.30	RCCV	0.55	0.77
93	4.80	RCCV	0.48	0.56
94	–1.70	RCCV	0.36	0.38
88	–8.20	RCCV	0.31	0.33

NOTE: (1) Horizontal accelerations are the envelopes of X and Y direction responses

(2) The results represent the maximum responses at the respective floor elevation

Table 3A-23c ABWR Reactor Building RSW/PED Summary of Enveloping Maximum Accelerations

Node No.	Elev TMSL(m)	Location	Max. Horizontal Acceleration (g)	Max. Vertical Acceleration (g)
70	21.20	RSW/PED	1.02	0.55
78	18.44	RSW/PED	0.92	0.55
79	17.02	RSW/PED	0.83	0.55
80	15.60	RSW/PED	0.73	0.52
81	13.95	RSW/PED	0.63	0.52
82	12.30	RSW/PED	0.58	0.51
71	8.20	RSW/PED	0.55	0.49
83	7.00	RSW/PED	0.53	0.49
72	4.50	RSW/PED	0.50	0.43
84	3.50	RSW/PED	0.50	0.41
73	1.70	RSW/PED	0.47	0.38
85	–0.18	RSW/PED	0.45	0.33
86	–2.10	RSW/PED	0.41	0.33
87	–4.70	RSW/PED	0.35	0.31
88	–8.20	RSW/PED	0.31	0.33

NOTE: (1) Horizontal accelerations are the envelopes of X and Y direction responses
(2) The results represent the maximum responses at the respective floor elevation

Table 3A-23d ABWR Reactor Building RPV/Internals Summary of Enveloping Maximum Accelerations

Node No.	Elev TMSL(m)	Location	Max. Horizontal Acceleration (g)	Max. Vertical Acceleration (g)
17	16.48	RPV	1.16	0.55
18	15.68	RPV	1.04	0.55
25	9.65	RPV	0.57	0.51
27	6.75	RPV	0.61	0.47
28	26.06	RPV	1.72	0.44
33	20.49	RPV	1.19	0.64
36	17.18	RPV	0.98	0.62
38	15.68	RPV	0.89	0.61
46	9.29	RPV	0.56	0.51
50	5.95	RPV	0.67	0.46
51	5.49	RPV	0.64	0.48
52	4.82	RPV	0.65	0.54
60	1.65	RPV	1.10	0.54
66	1.65	RPV	1.15	0.48

NOTE: (1) Horizontal accelerations are the envelopes of X and Y direction responses

Table 3A-24 ABWR Control Building Summary of Enveloping Maximum Accelerations

Node No.	Elev TMSL(m)	Location	Max. Acceleration (g) Horizontal	Vertical
108	22.2	C/B STICK	1.02	0.48
107	17.15	C/B STICK	0.72	0.44
106	12.3	C/B STICK	0.52	0.36
105	7.9	C/B STICK	0.34	0.33
104	3.5	C/B STICK	0.32	0.31
103	–2.15	C/B STICK	0.31	0.31
102	–8.2	C/B STICK	0.31	0.30
113	17.15	C/B FLOOR		0.62
112	12.3	C/B FLOOR		0.58
111	7.9	C/B FLOOR		0.55
110	3.5	C/B FLOOR		0.51
109	–2.15	C/B FLOOR		0.49

NOTE: (1) Horizontal accelerations are the envelopes of x and y direction responses.
(2) The results represent the maximum responses at the respective floor elevation.

Table 3A-25a ABWR Reactor Building Walls and Floors Summary of Enveloping Maximum Relative Displacements with Respect to Input Motion

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Free-Field Grade Level				
95	49.70	R/B	28.5	7.4
96	38.20	R/B	22.3	6.7
98	31.70	R/B	18.7	6.8
100	23.50	R/B	13.7	5.8
102	18.10	R/B	10.1	5.6
103	12.30	R/B	7.5	5.4
104	4.80	R/B	8.2	5.0
105	–1.70	R/B	8.8	4.7
88	–8.20	R/B	12.0	4.3

NOTE: (1) Horizontal displacements are the envelopes of X and Y direction responses
(2) The results represent the maximum responses at the respective floor elevation

Table 3A-25b ABWR Reactor Building RCCV Summary of Enveloping Maximum Relative Displacements with Respect to Input Motion

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Free-Field Grade Level				
89	31.70	RCCV	18.7	4.1
90	23.50	RCCV	13.7	6.6
91	18.10	RCCV	10.1	6.4
92	12.30	RCCV	7.5	5.6
93	4.80	RCCV	8.2	3.9
94	–1.70	RCCV	8.8	3.3
88	–8.20	RCCV	12.0	4.3

NOTE: (1) Horizontal displacements are the envelopes of X and Y direction responses
(2) The results represent the maximum responses at the respective floor elevation

Table 3A-25c ABWR Reactor Building RSW/PED Summary of Enveloping Maximum Relative Displacements with Respect to Input Motion

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Free-Field Grade Level				
70	21.20	RSW/PED	11.1	2.2
78	18.44	RSW/PED	9.5	2.2
79	17.02	RSW/PED	8.9	2.2
80	15.60	RSW/PED	8.1	2.2
81	13.95	RSW/PED	7.6	2.2
82	12.30	RSW/PED	7.6	2.7
71	8.20	RSW/PED	8.0	2.6
83	7.00	RSW/PED	8.1	2.6
72	4.50	RSW/PED	8.3	2.5
84	3.50	RSW/PED	8.4	2.4
73	1.70	RSW/PED	8.6	2.4
85	–0.18	RSW/PED	8.9	2.4
86	–2.10	RSW/PED	9.8	2.2
87	–4.70	RSW/PED	10.8	2.2
88	–8.20	RSW/PED	12.0	4.3

NOTE: (1) Horizontal displacements are the envelopes of X and Y direction responses
(2) The results represent the maximum responses at the respective floor elevation

Table 3A-25d ABWR Reactor Building RPV/Internals Summary of Enveloping Maximum Relative Displacements with Respect to Input Motion

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Free-Field Grade Level				
17	16.48	RPV	10.8	1.6
18	15.68	RPV	10.3	1.6
25	9.65	RPV	7.7	1.6
27	6.75	RPV	8.0	1.6
28	26.06	RPV	14.9	1.6
33	20.49	RPV	11.0	3.2
36	17.18	RPV	9.0	2.8
38	15.68	RPV	8.5	2.8
46	9.29	RPV	7.9	2.8
50	5.95	RPV	8.1	1.6
51	5.49	RPV	8.2	1.6
52	4.82	RPV	8.2	1.6
60	1.65	RPV	8.4	1.6
66	1.65	RPV	10.6	1.6

NOTE: (1) Horizontal accelerations are the envelopes of X and Y direction responses

Table 3A-26 ABWR Control Building Summary of Enveloping Maximum Relative Displacements with Respect to Input Motion

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Free-Field Grade Level				
108	22.2	C/B STICK	8.9	3.3
107	17.15	C/B STICK	9.0	3.3
106	12.3	C/B STICK	9.0	3.2
105	7.9	C/B STICK	9.7	3.1
104	3.5	C/B STICK	10.6	3.2
103	–2.15	C/B STICK	9.9	3.1
102	–8.2	C/B STICK	9.1	2.7
113	17.15	C/B FLOOR		1.5
112	12.3	C/B FLOOR		1.5
111	7.9	C/B FLOOR		1.5
110	3.5	C/B FLOOR		1.6
109	–2.15	C/B FLOOR		1.6

NOTE: (1) Horizontal accelerations are the envelopes of x and y direction responses.
(2) The results represent the maximum responses at the respective floor elevation.

Table 3A-27a ABWR Reactor Building Walls and Floors Summary of Enveloping Maximum Relative Displacements with Respect to Basemat

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Node 88				
95	49.70	R/B	24.1	7.4
96	38.20	R/B	19.0	6.7
98	31.70	R/B	15.6	6.8
100	23.50	R/B	11.4	5.8
102	18.10	R/B	8.2	5.6
103	12.30	R/B	5.7	5.3
104	4.80	R/B	3.8	4.9
105	–1.70	R/B	1.7	4.4
88	–8.20	R/B	0.0	4.0

NOTE: (1) Horizontal displacements are the envelopes of X and Y direction responses
(2) The results represent the maximum responses at the respective floor elevation

Table 3A-27b ABWR Reactor Building RCCV Summary of Enveloping Maximum Relative Displacements with Respect to Basemat

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Node 88				
89	31.70	RCCV	15.5	4.1
90	23.50	RCCV	11.9	6.6
91	18.10	RCCV	8.9	6.4
92	12.30	RCCV	5.8	5.6
93	4.80	RCCV	3.8	3.9
94	–1.70	RCCV	1.7	2.9
88	–8.20	RCCV	0.0	4.0

NOTE: (1) Horizontal displacements are the envelopes of X and Y direction responses
(2) The results represent the maximum responses at the respective floor elevation

Table 3A-27c ABWR Reactor Building RSW/PED Summary of Enveloping Maximum Relative Displacements with Respect to Basemat

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Node 88				
70	21.20	RSW/PED	8.8	2.2
78	18.44	RSW/PED	8.0	2.2
79	17.02	RSW/PED	7.4	2.2
80	15.60	RSW/PED	6.9	2.2
81	13.95	RSW/PED	6.3	2.7
82	12.30	RSW/PED	5.8	2.7
71	8.20	RSW/PED	4.6	2.6
83	7.00	RSW/PED	4.2	2.6
72	4.50	RSW/PED	3.4	2.4
84	3.50	RSW/PED	3.1	2.3
73	1.70	RSW/PED	2.5	2.0
85	–0.18	RSW/PED	2.7	2.0
86	–2.10	RSW/PED	1.6	1.4
87	–4.70	RSW/PED	1.3	1.4
88	–8.20	RSW/PED	0.0	4.0

NOTE: (1) Horizontal displacements are the envelopes of X and Y direction responses
(2) The results represent the maximum responses at the respective floor elevation

Table 3A-27d ABWR Reactor Building RPV/Internals Summary of Enveloping Maximum Relative Displacements with Respect to Basemat

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Node 88				
17	16.48	RPV	9.0	0.4
18	15.68	RPV	8.6	0.4
25	9.65	RPV	5.6	0.4
27	6.75	RPV	4.3	0.4
28	26.06	RPV	11.0	0.3
33	20.49	RPV	8.9	1.9
36	17.18	RPV	7.6	1.8
38	15.68	RPV	7.1	1.8
46	9.29	RPV	4.9	1.6
50	5.95	RPV	4.1	0.3
51	5.49	RPV	4.0	0.4
52	4.82	RPV	3.8	0.4
60	1.65	RPV	3.1	0.4
66	1.65	RPV	5.0	0.4

NOTE: (1) Horizontal displacements are the envelopes of X and Y direction responses

Table 3A-28 ABWR Control Building Summary of Enveloping Maximum Relative Displacements with Respect to Basemat

Node No.	Elev TMSL(m)	Location	Max. Rel. Displacement (mm) Horizontal	Vertical
Reference Motion: Displacement at Node 102				
108	22.2	C/B STICK	6.6	3.0
107	17.15	C/B STICK	6.1	3.0
106	12.3	C/B STICK	6.4	2.9
105	7.9	C/B STICK	6.4	2.9
104	3.5	C/B STICK	7.6	2.8
103	-2.15	C/B STICK	2.6	2.6
102	-8.2	C/B STICK	0.0	0.0
113	17.15	C/B FLOOR		0.9
112	12.3	C/B FLOOR		0.9
111	7.9	C/B FLOOR		0.8
110	3.5	C/B FLOOR		0.7
109	-2.15	C/B FLOOR		0.5

NOTE: (1) Horizontal displacements are the envelopes of x and y direction responses.
(2) The results represent the maximum responses at the respective floor elevation.

Figure 3A-1 (Refer to Figure 1.2-1)

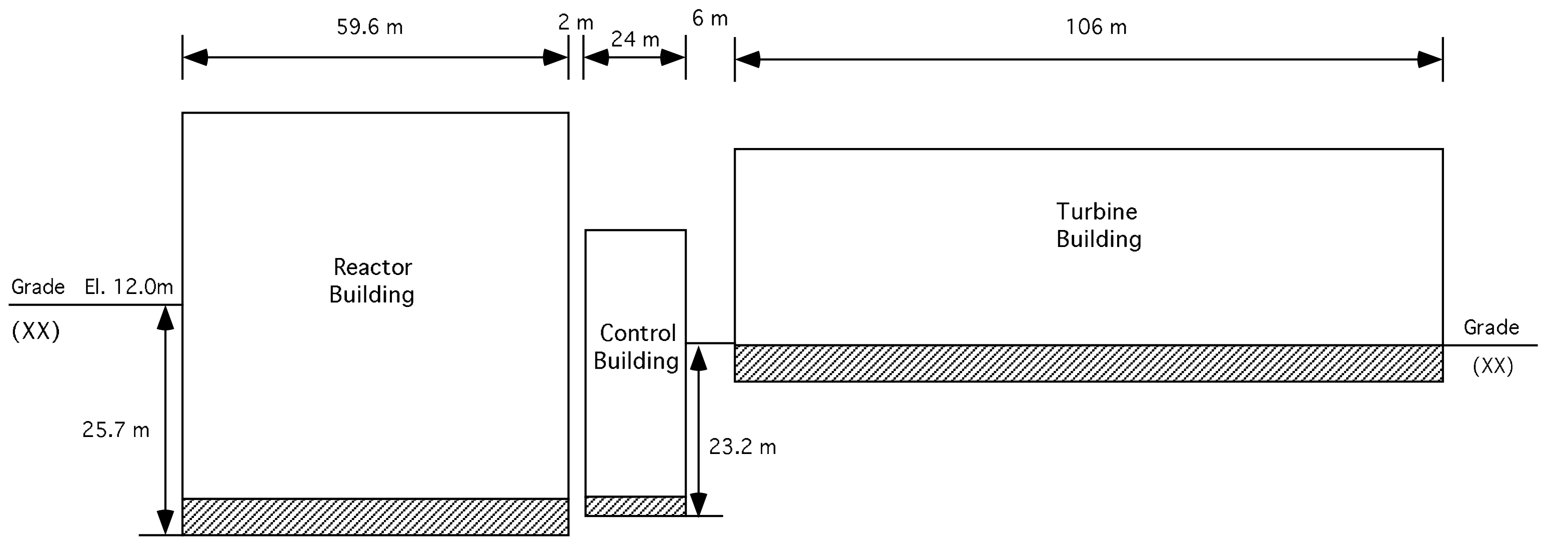


Figure 3A-2 0°–180° Section View

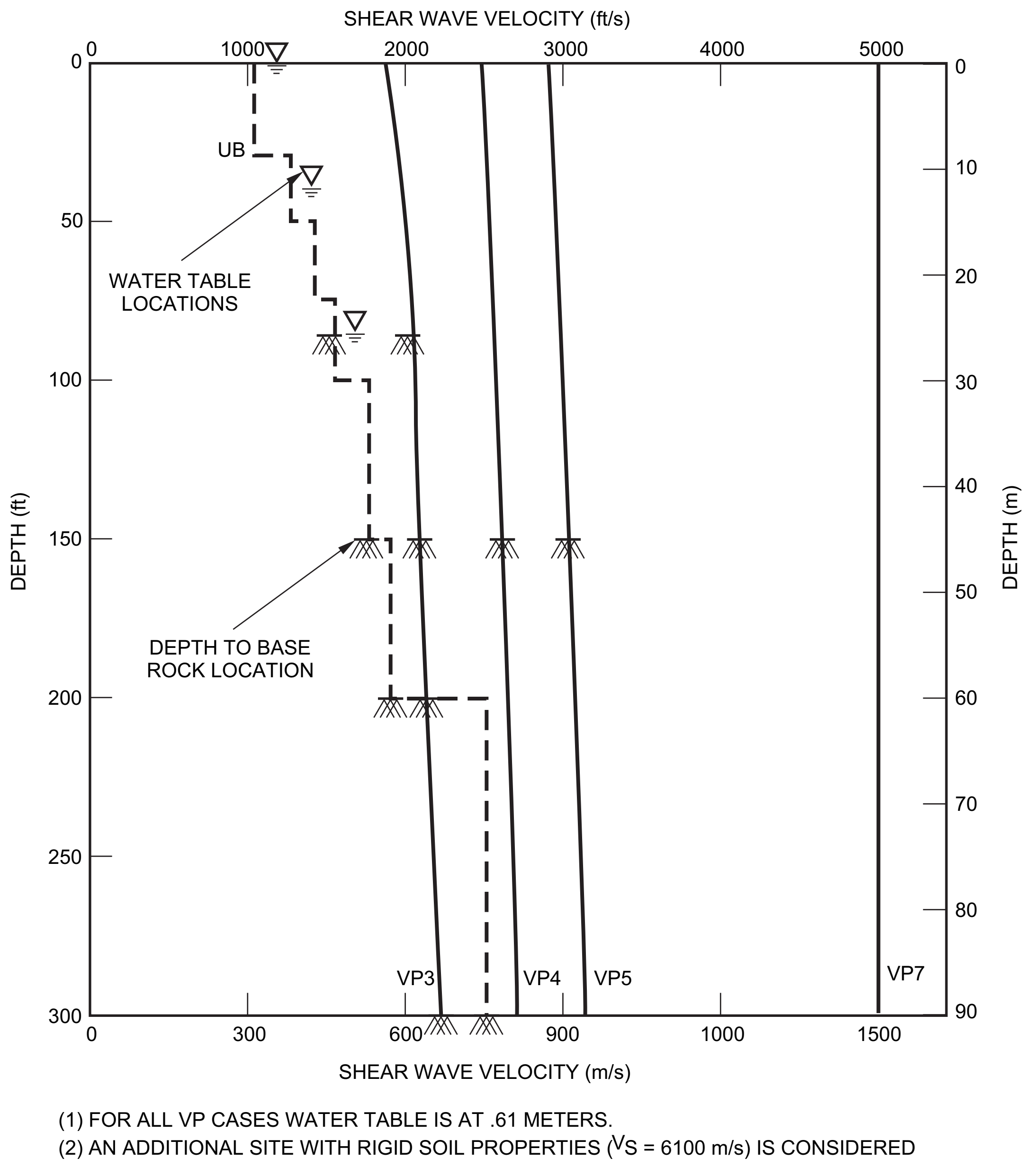


Figure 3A-3 Shear Wave Velocity Profiles Considered for SSI Analyses

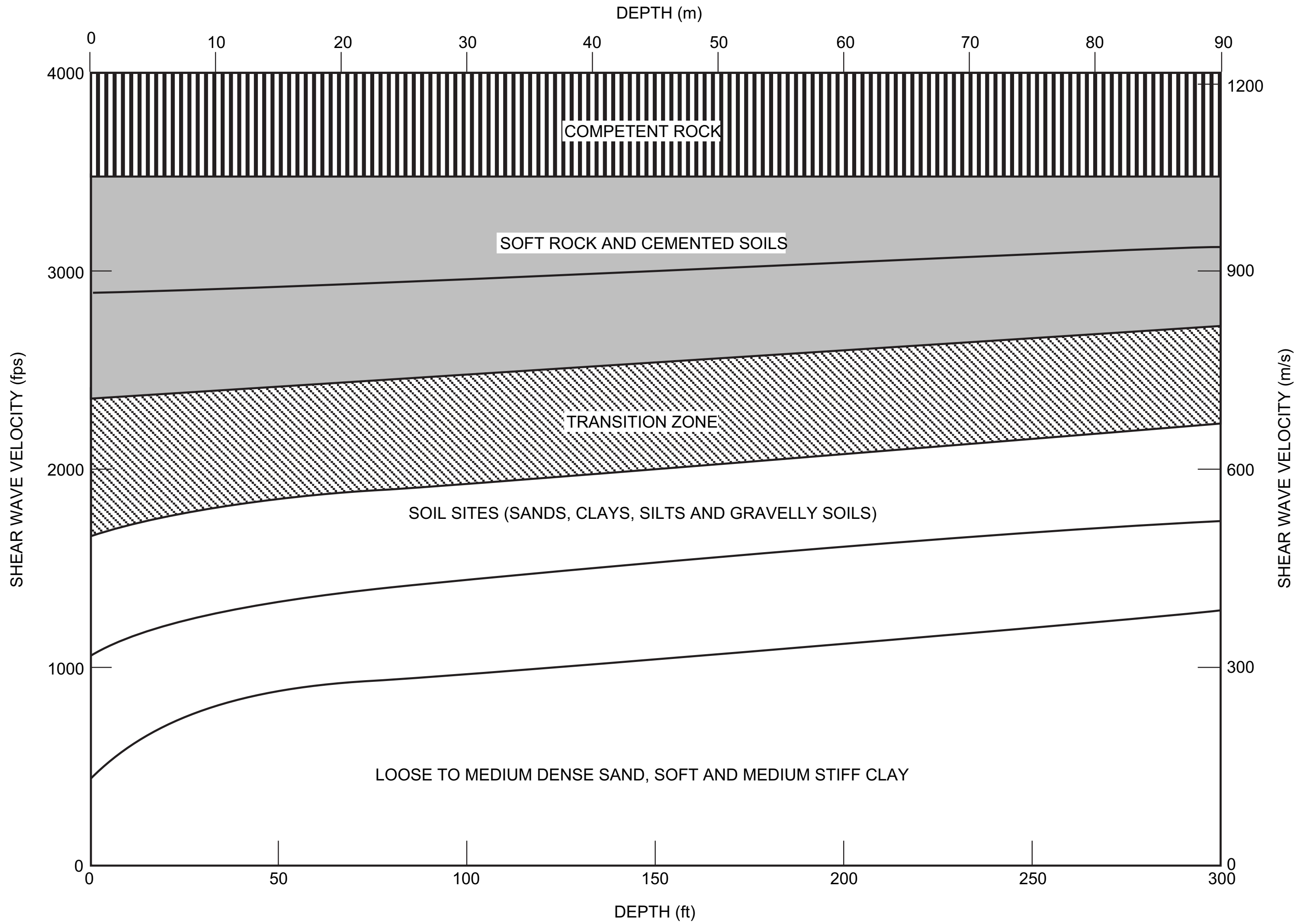


Figure 3A-4 Range of Shear Wave Velocities for Nuclear Power Plant Sites in High Seismic Areas

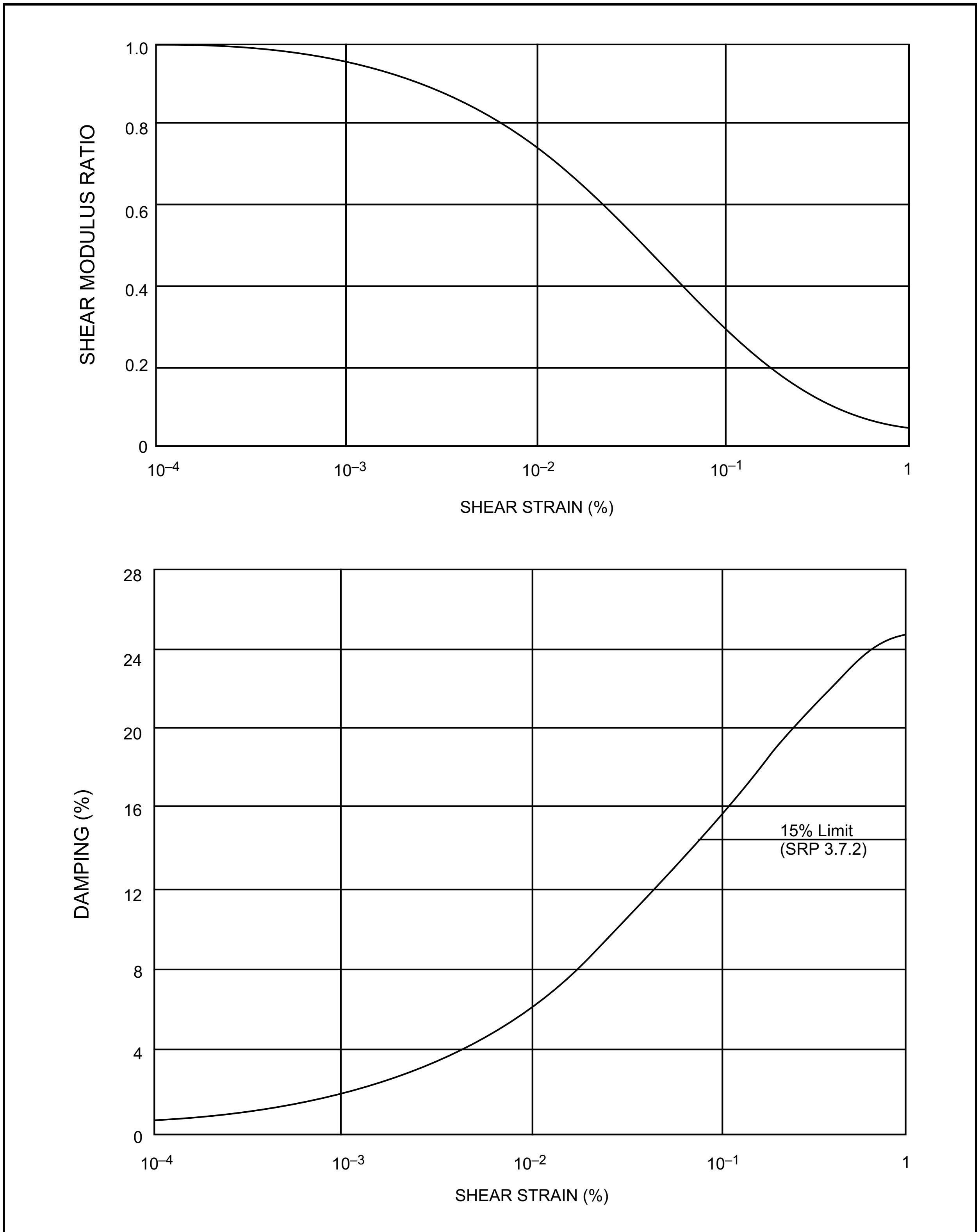


Figure 3A-5 Strain Dependent Soil Properties

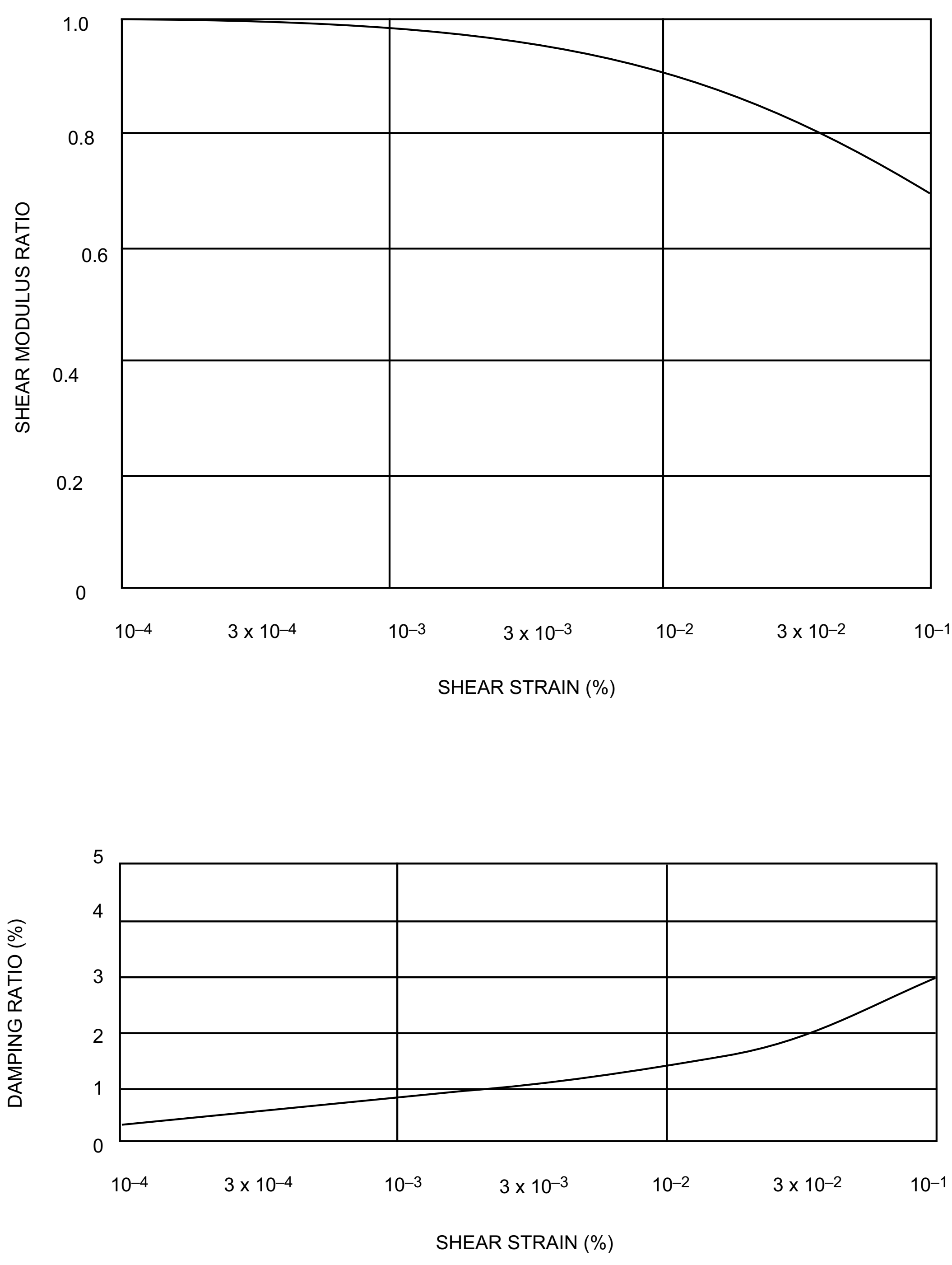


Figure 3A-6 Strain Dependent Rock Properties

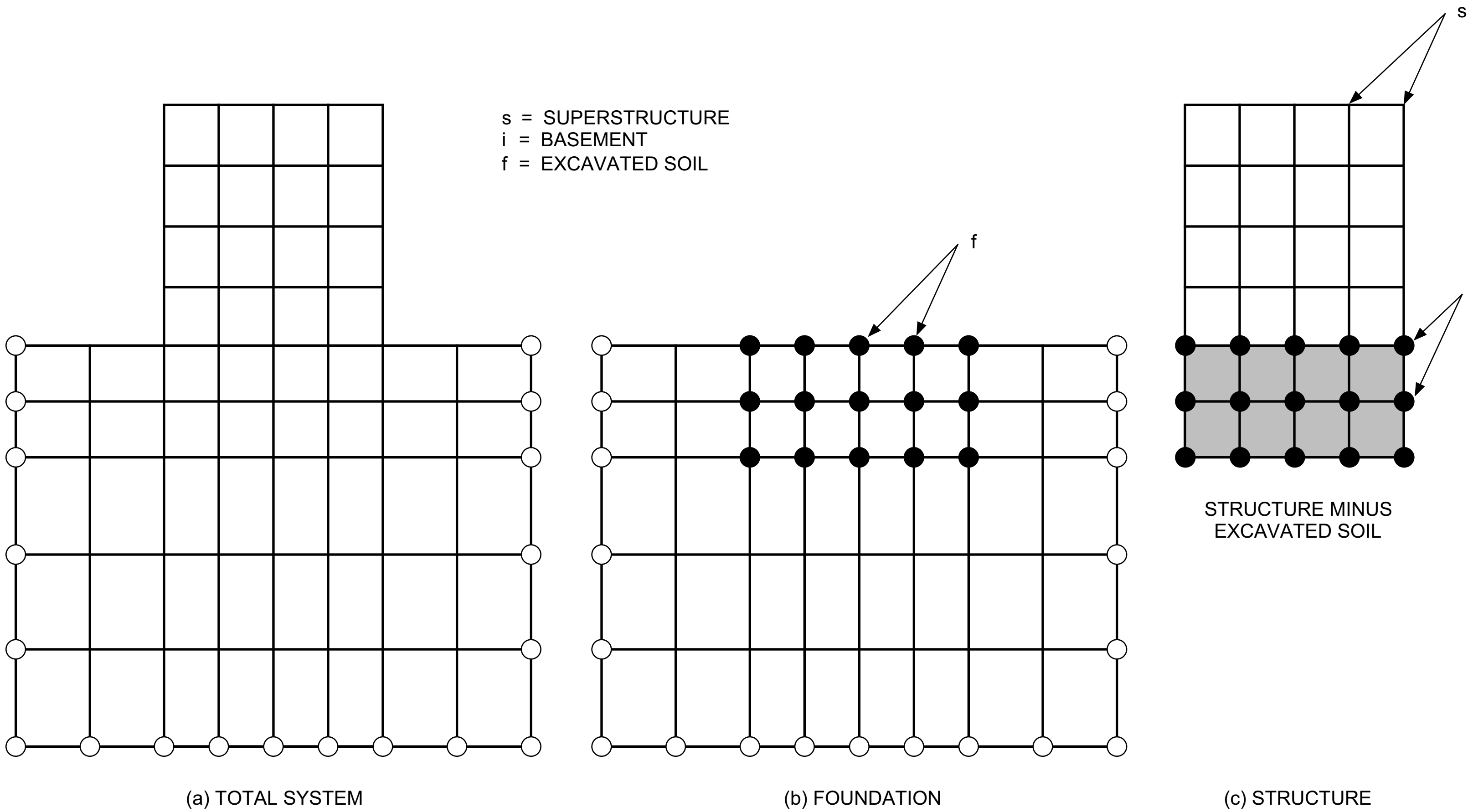


Figure 3A-7 Substructuring of Interaction Model

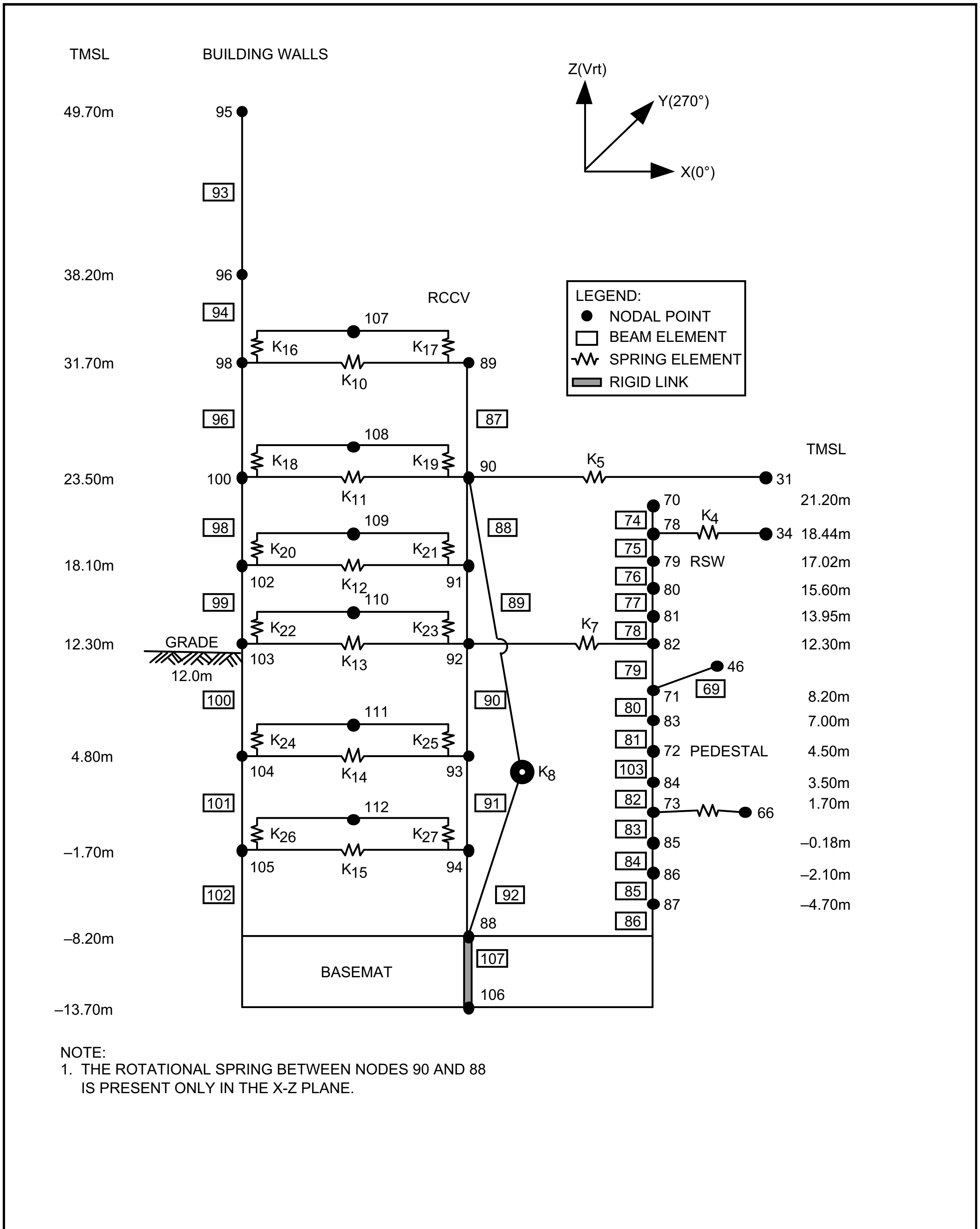


Figure 3A-8 Reactor Building Stick Model

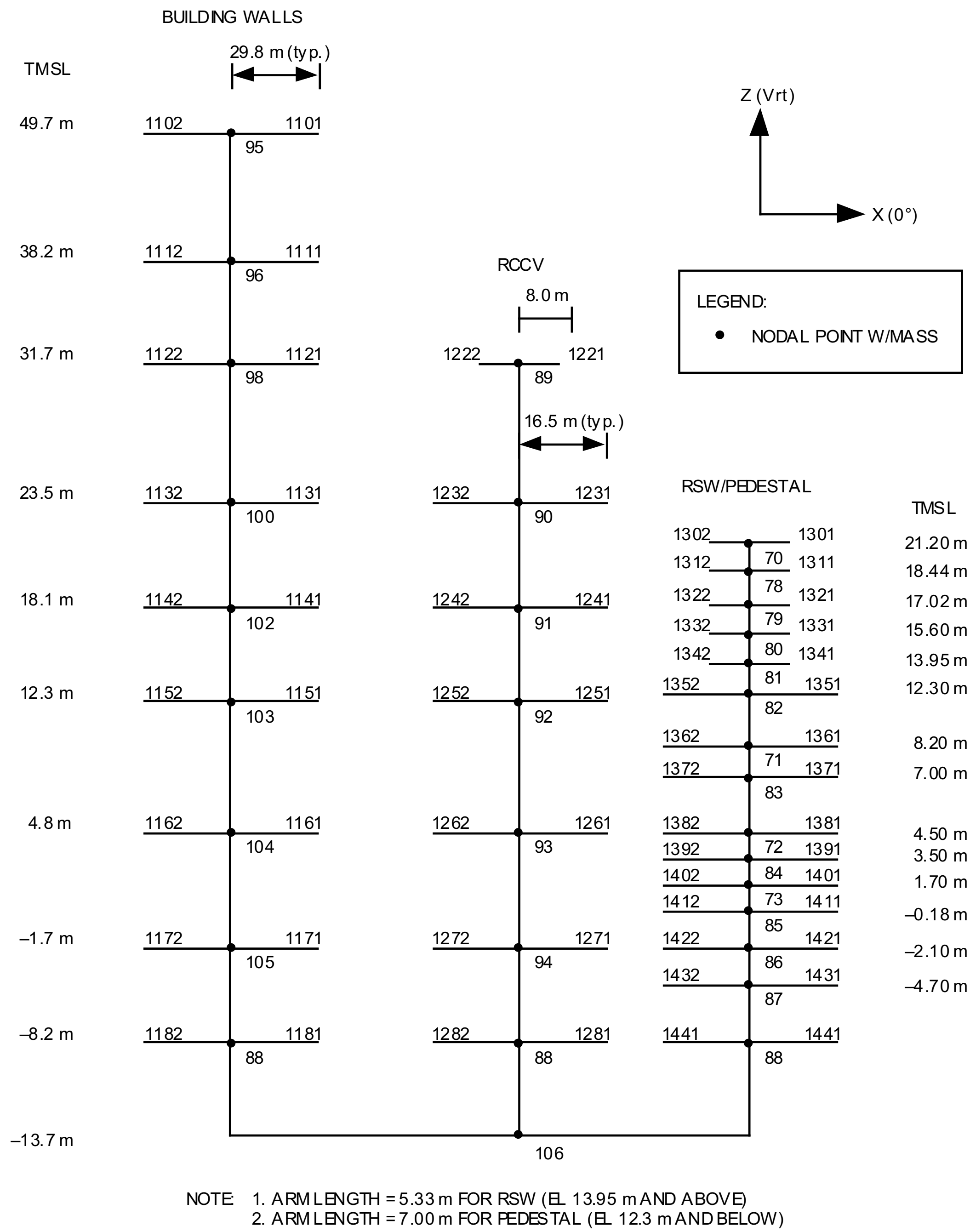


Figure 3A-9 RPV Stick Model

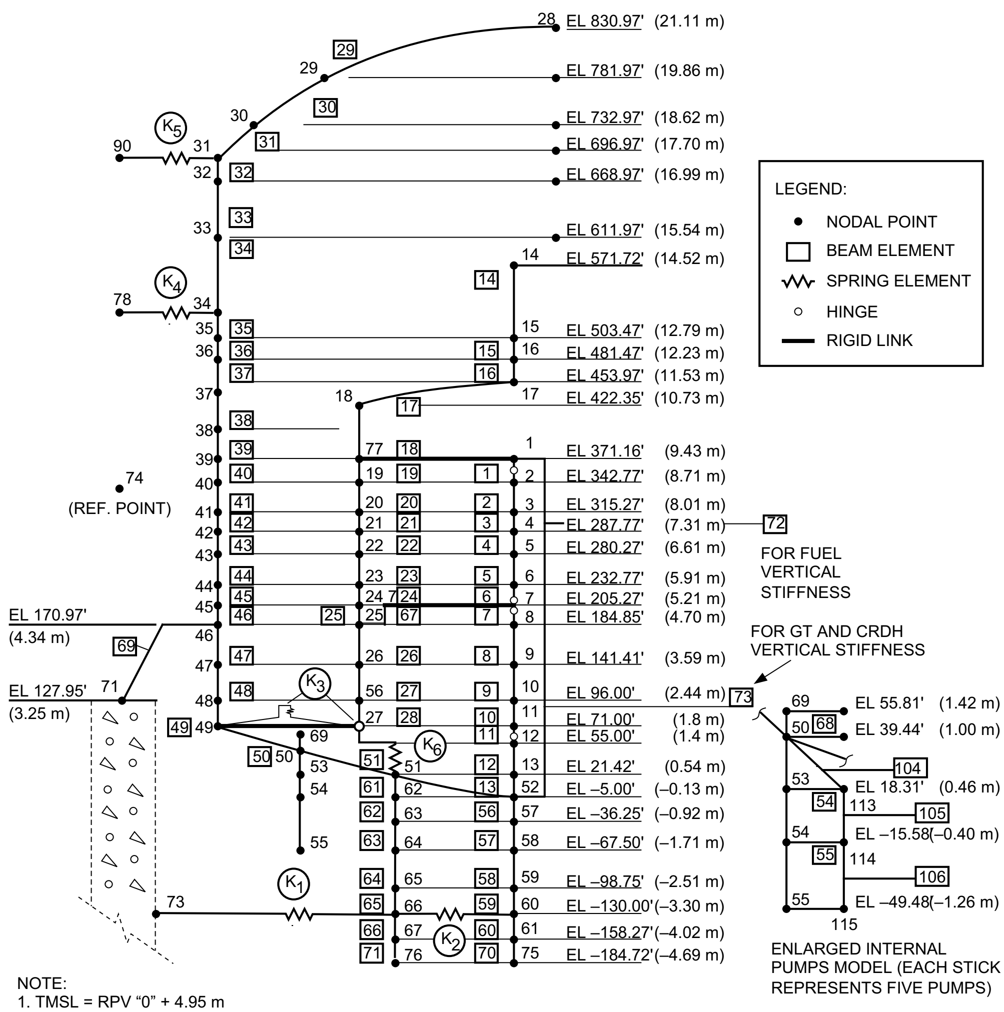


Figure 3A-10 Reactor Building Stick Model with Rigid Arms for X & Z Shaking

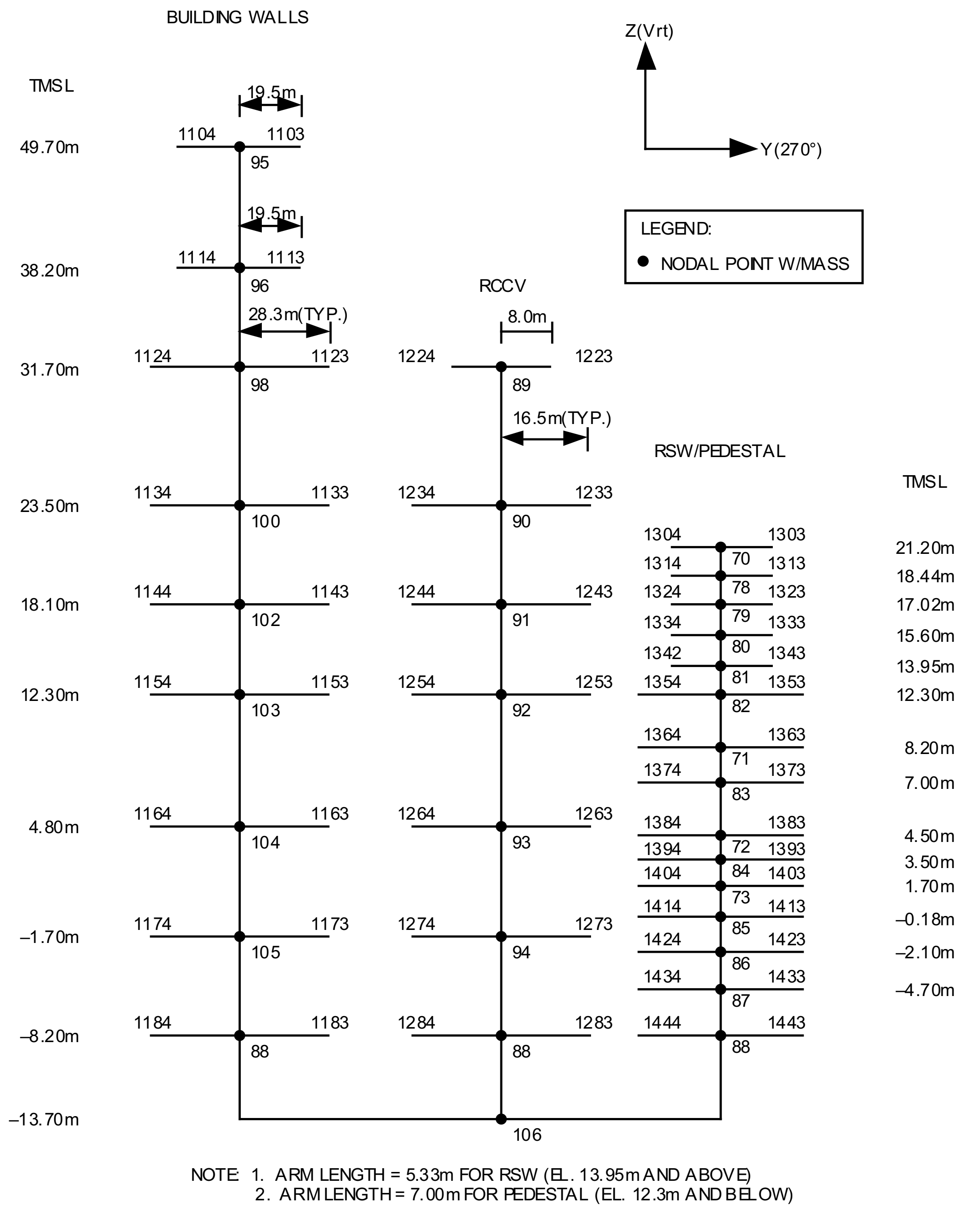
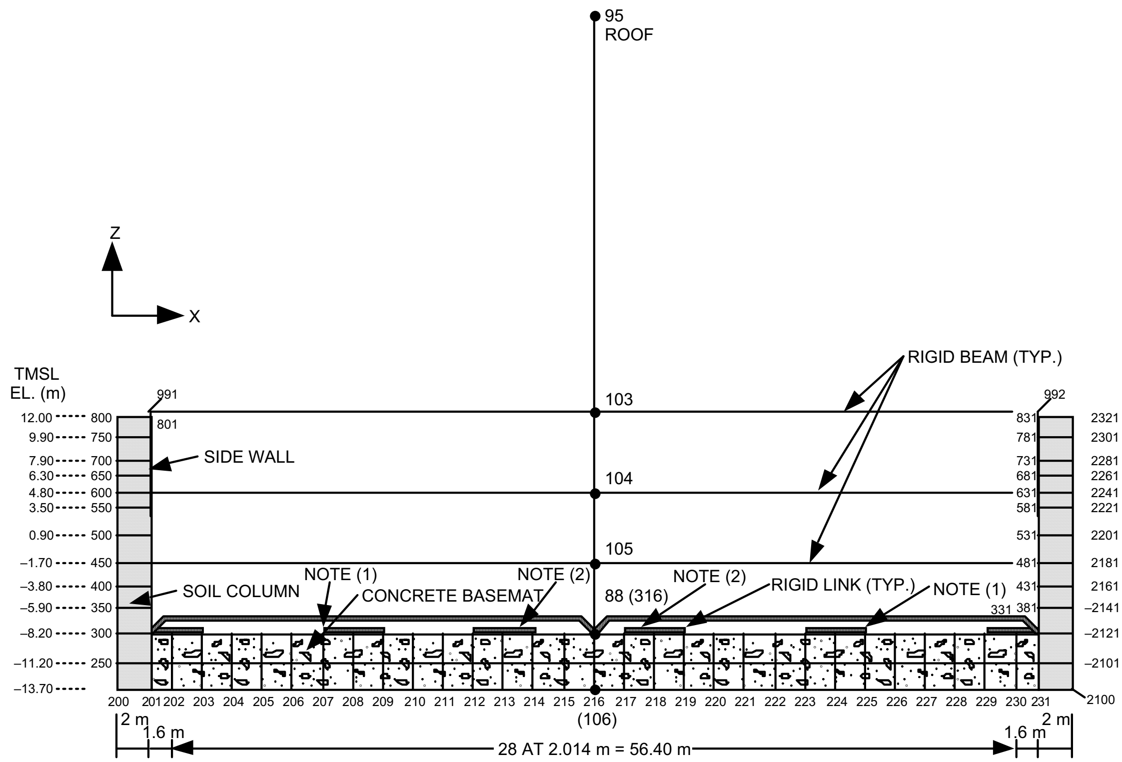


Figure 3A-11 Reactor Building Stick Model with Rigid Arms for Y Shaking



NOTES:
(1) FOOTPRINT CONNECTING TO RCCV STICK
(2) FOOTPRINT CONNECTING TO RSW/PEDESTAL STICK

Figure 3A-12 Reactor Building Foundation 2-D Model XZ Direction

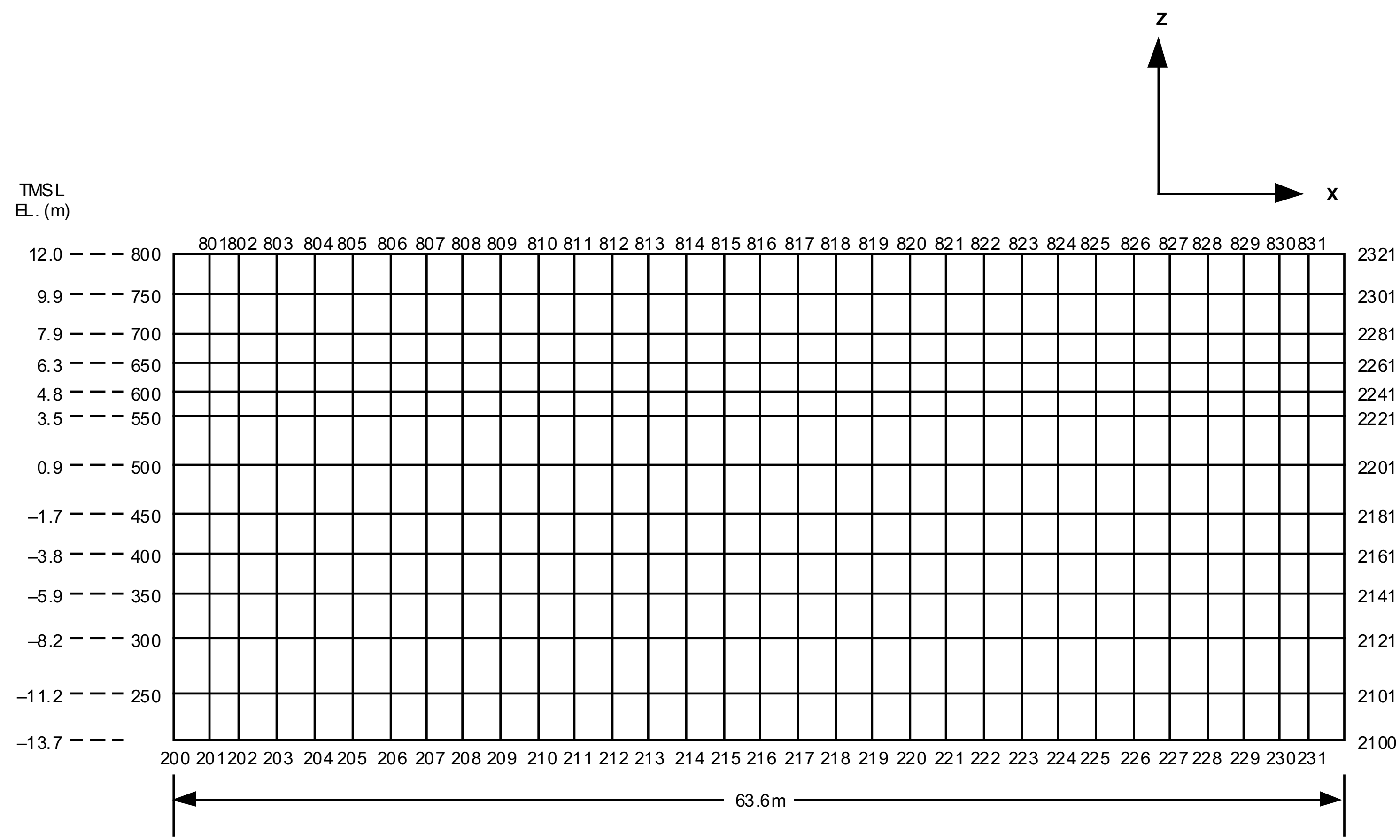


Figure 3A-13 The Excavated Soil Elements of Reactor Building 2-D Model

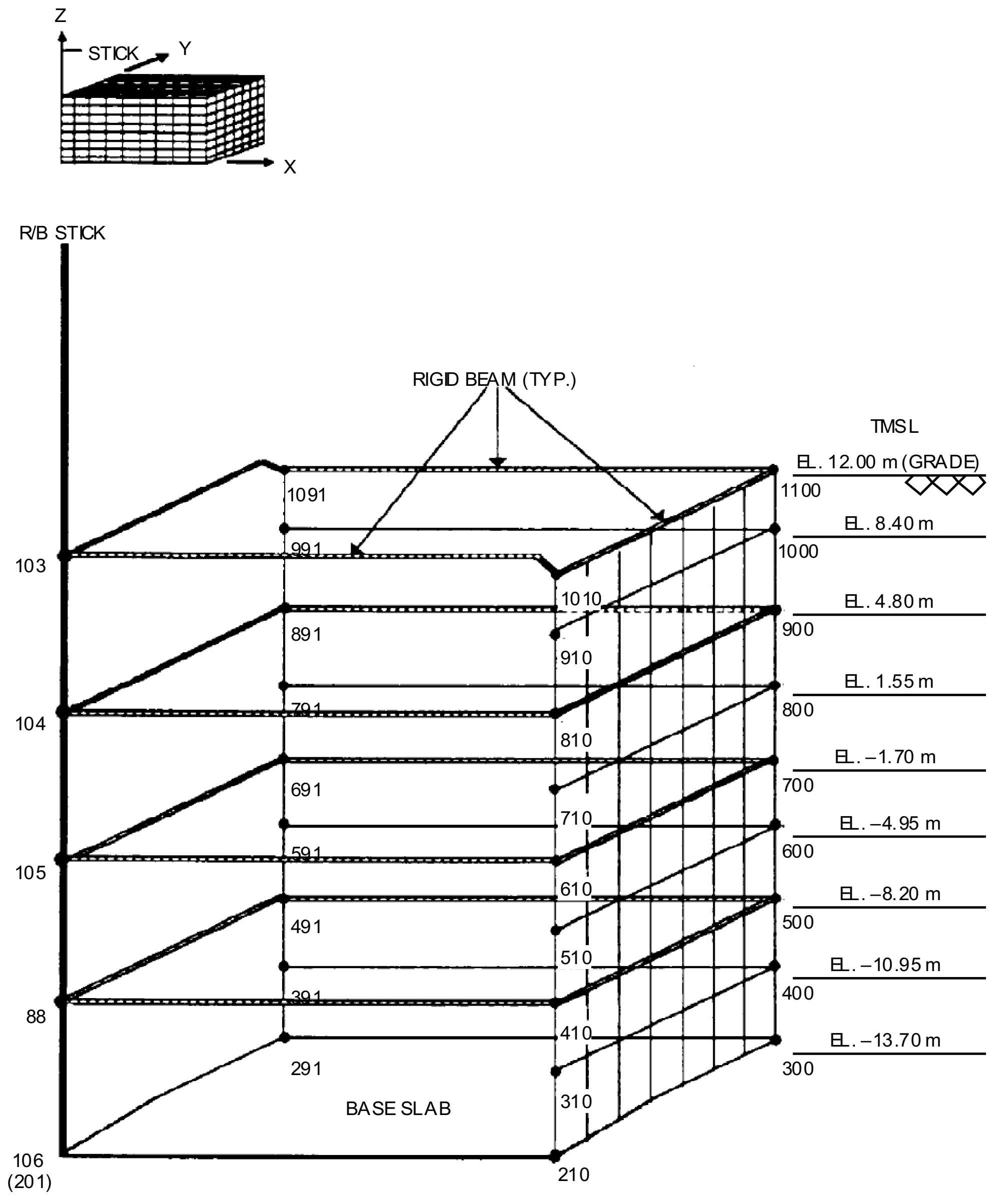


Figure 3A-14 Connection of the Main Stick to the Side Walls (For Reactor Building UB Cases)

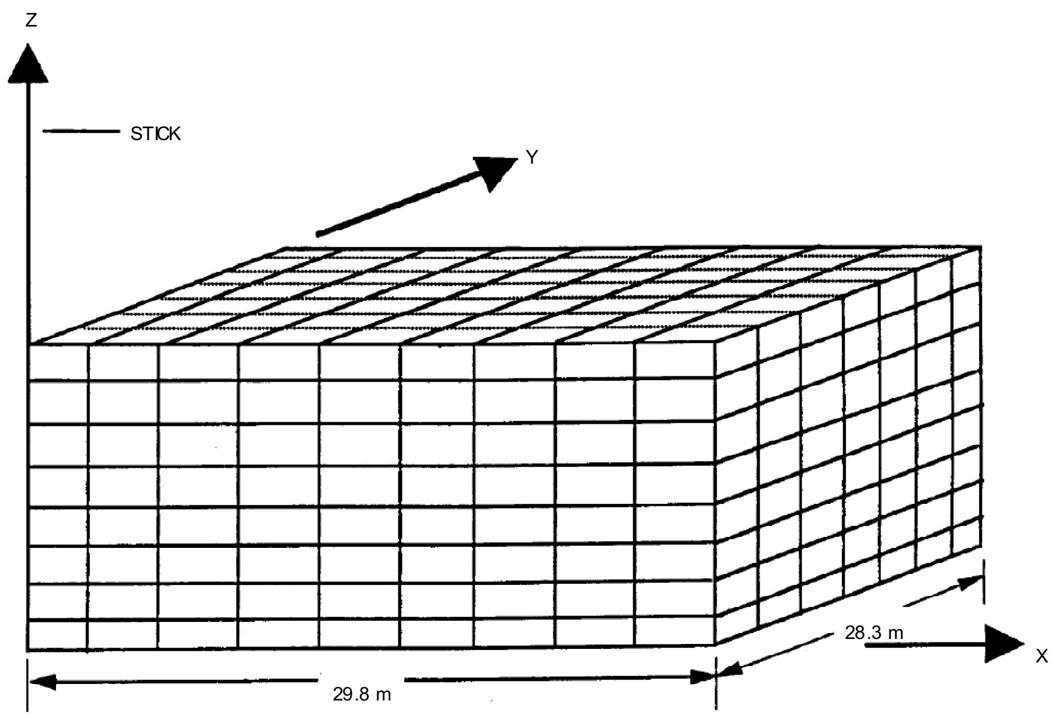


Figure 3A-15 R/B Excavated Soil Model (UB Soil Profiles)

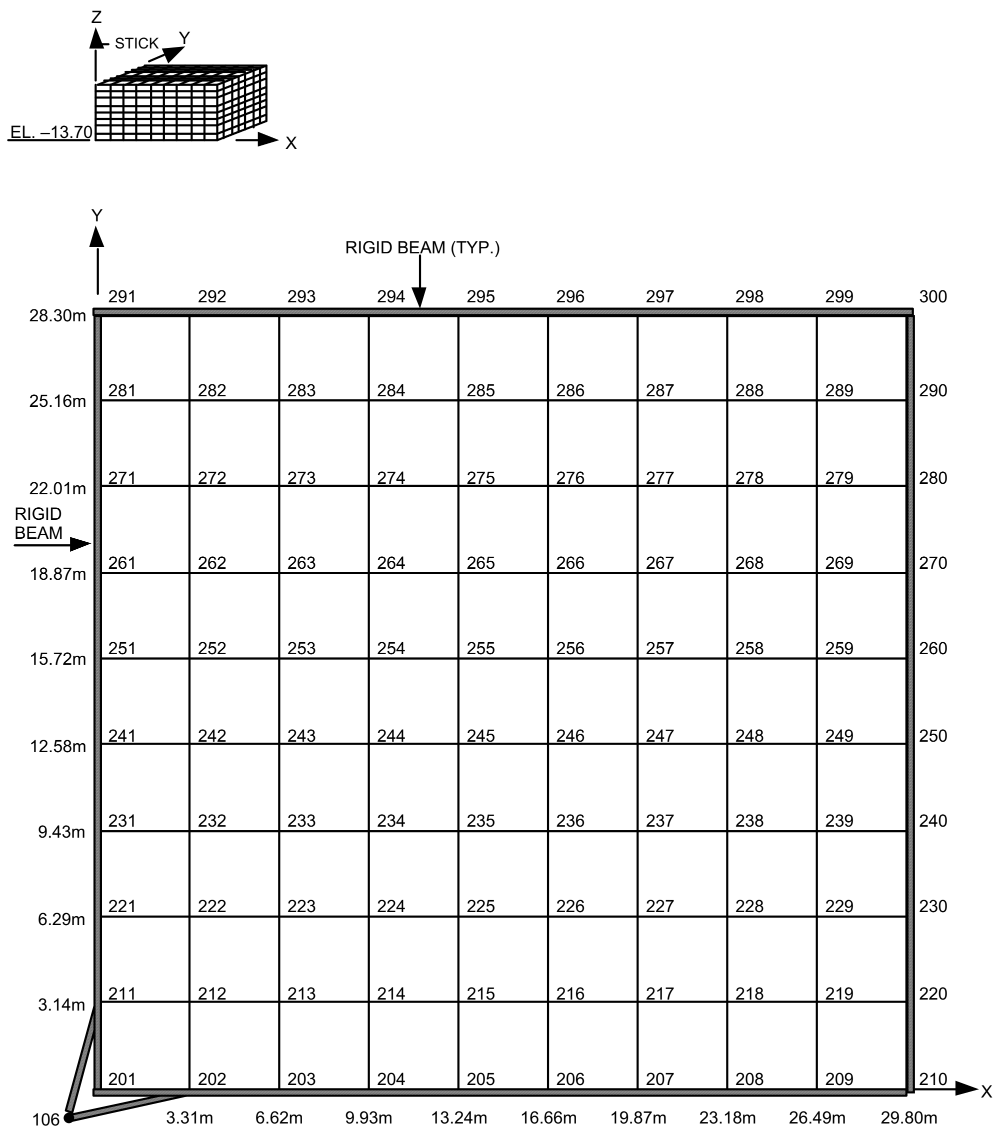


Figure 3A-16 UB Case: Nodal Points at Elevation –13.70m

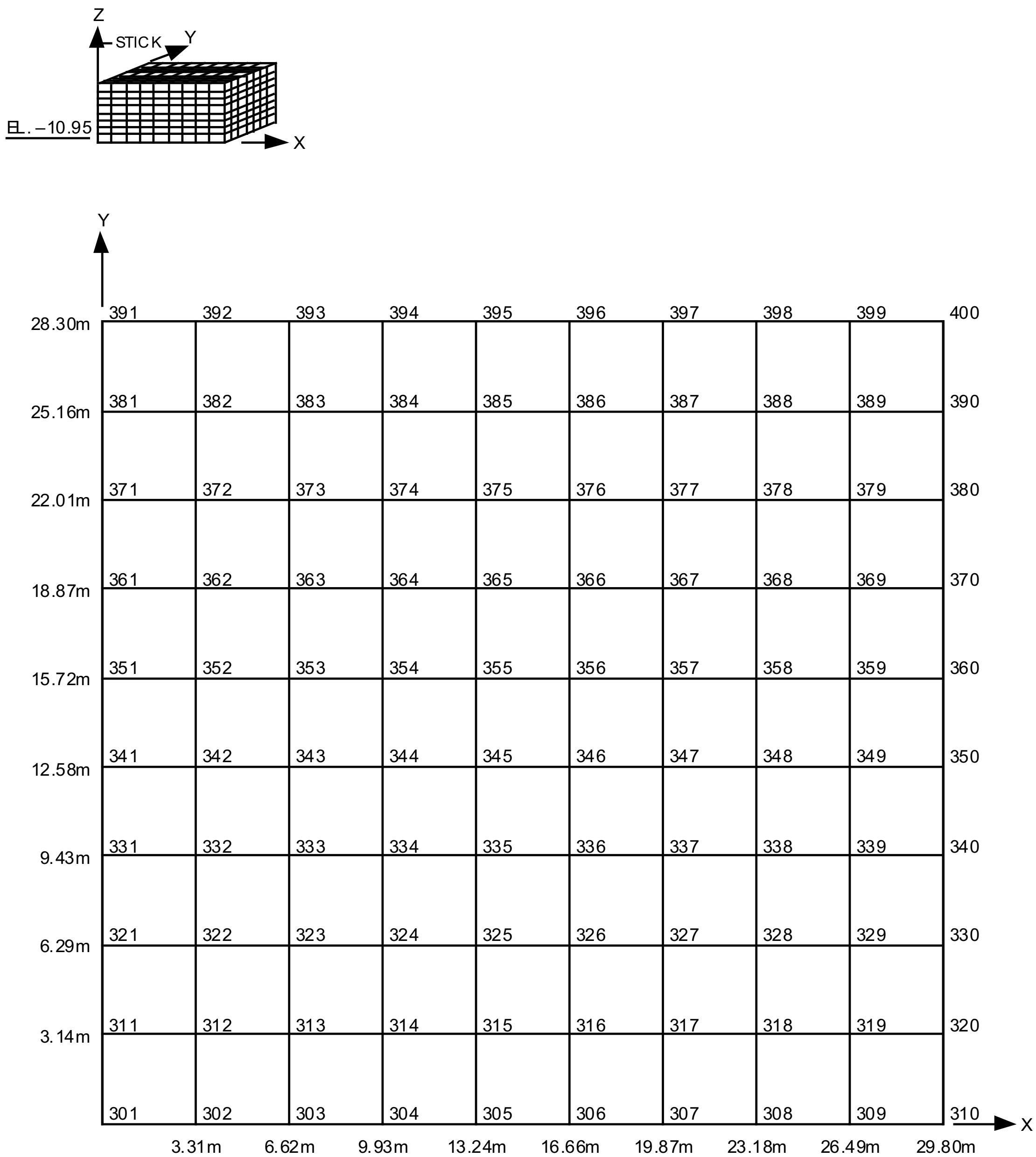


Figure 3A-17 UB Case: Nodal Points at Elevation –10.95m

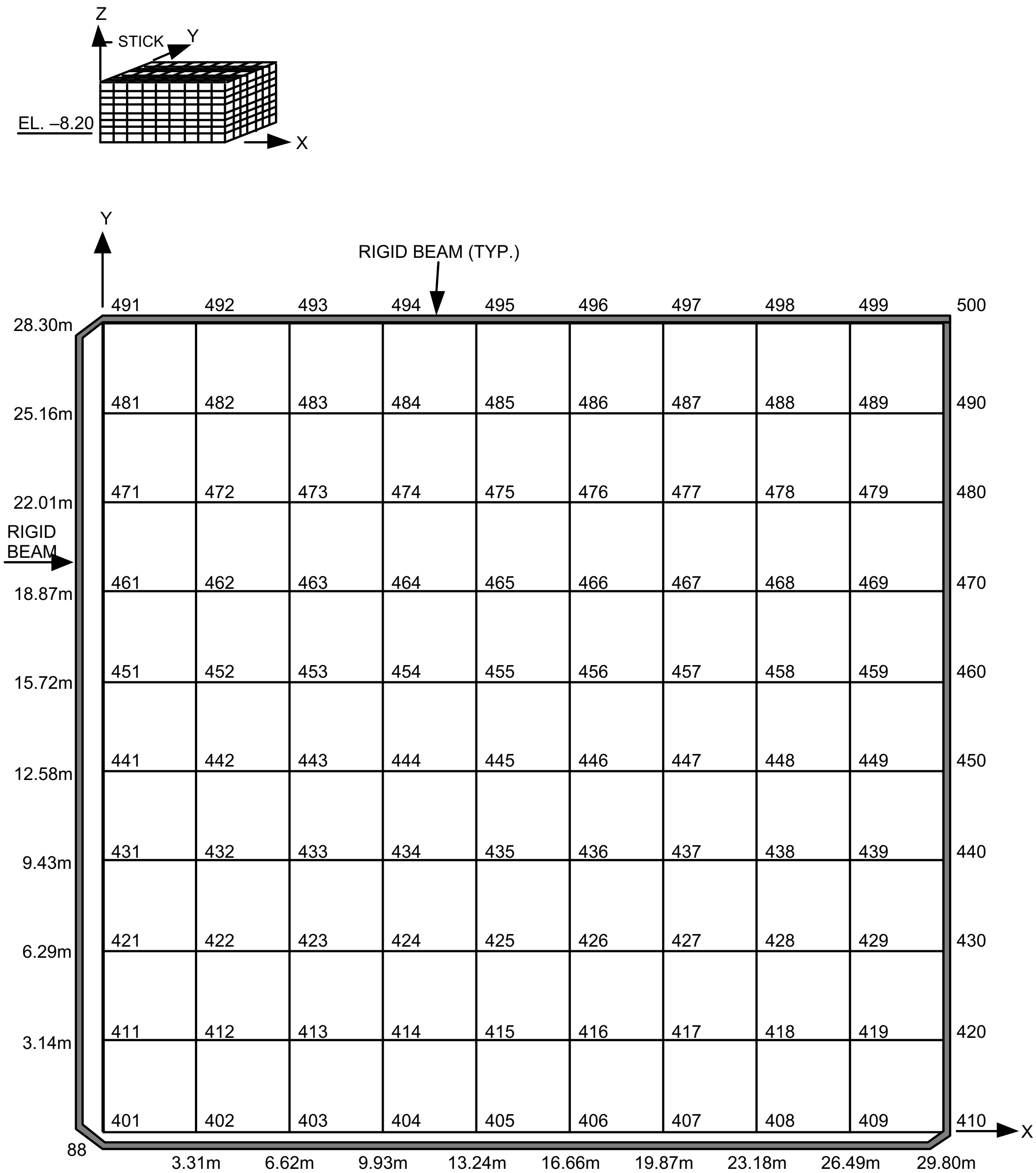


Figure 3A-18 UB Case: Nodal Points at Elevation –8.20m

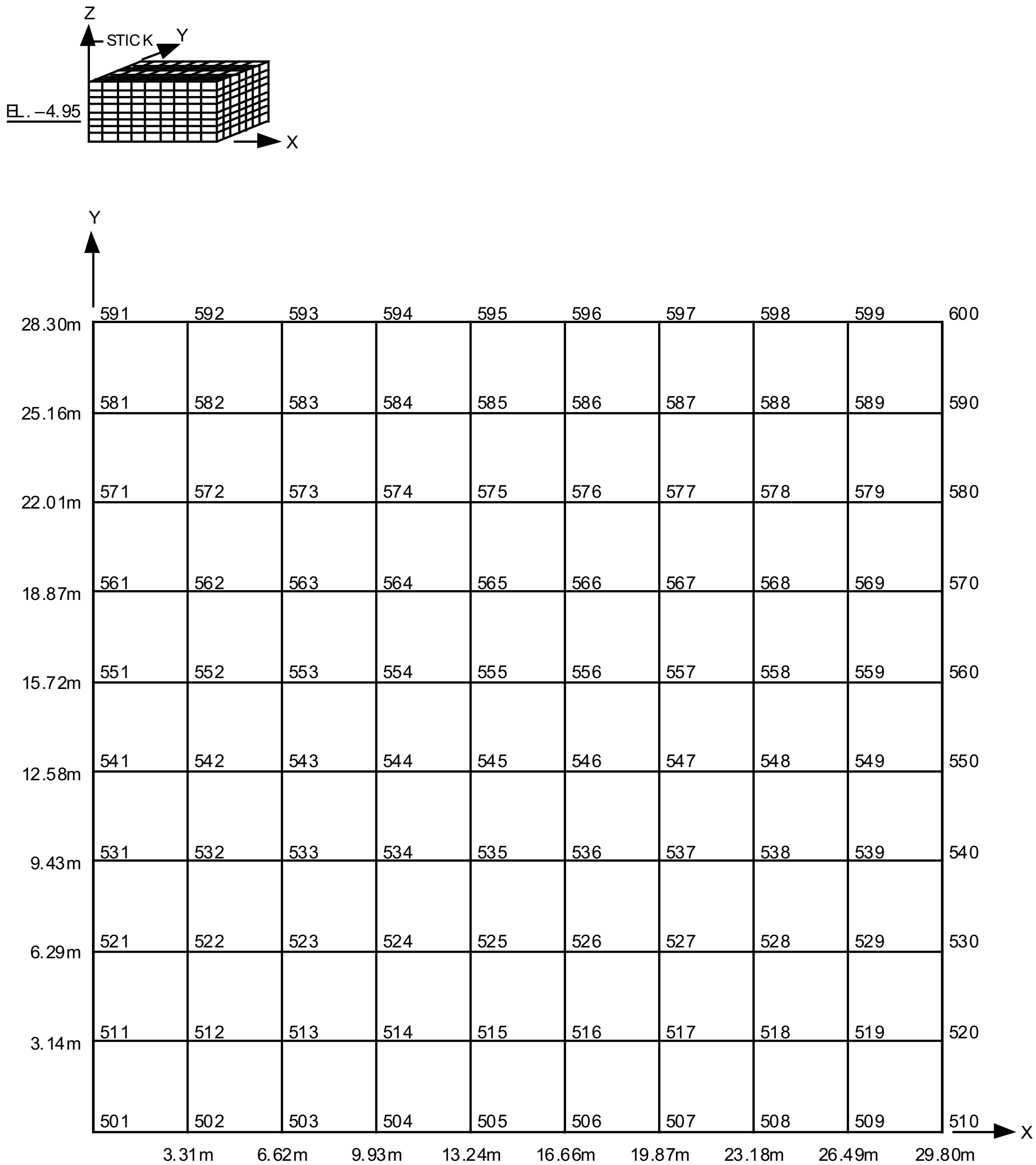


Figure 3A-19 UB Case: Nodal Points at Elevation –4.95m

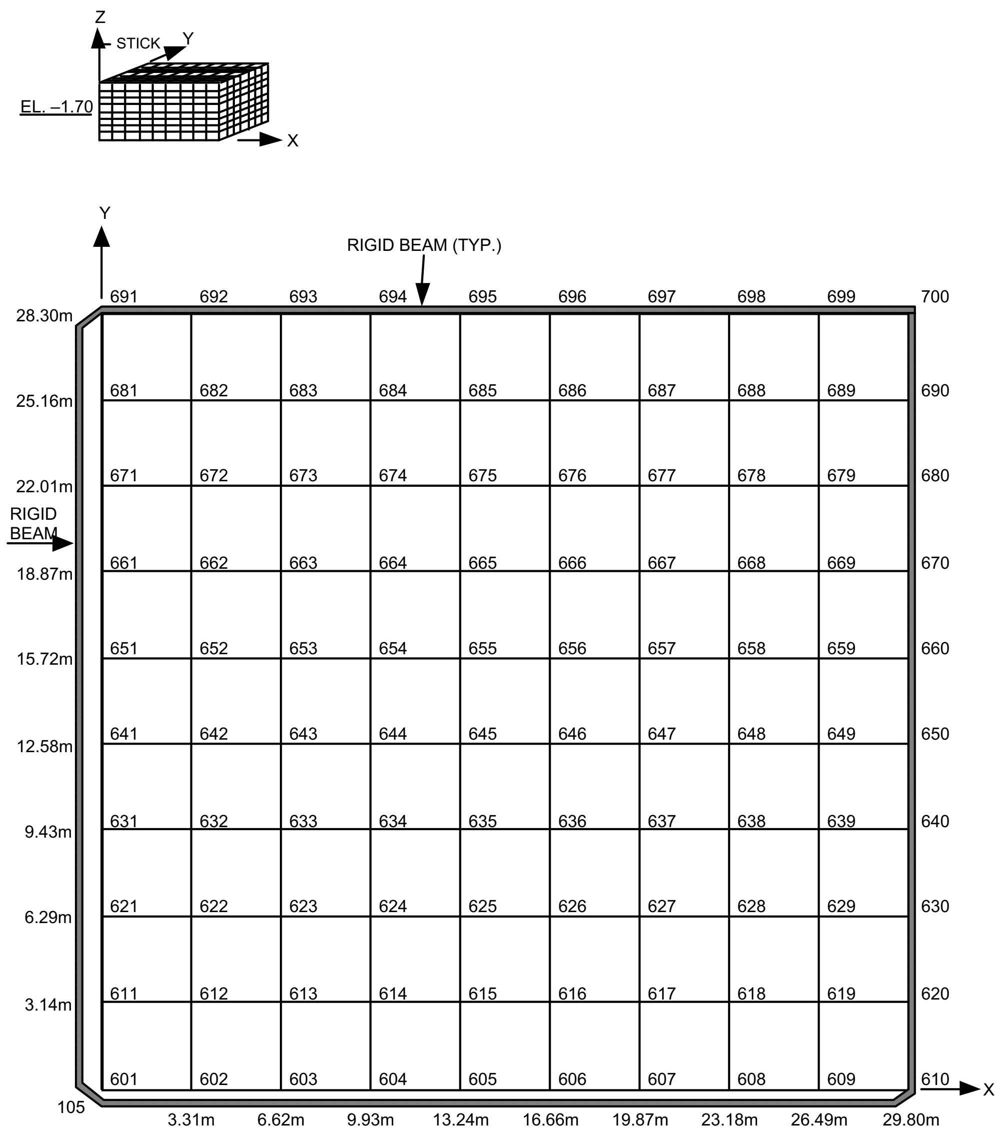


Figure 3A-20 UB Case: Nodal Points at Elevation –1.70m

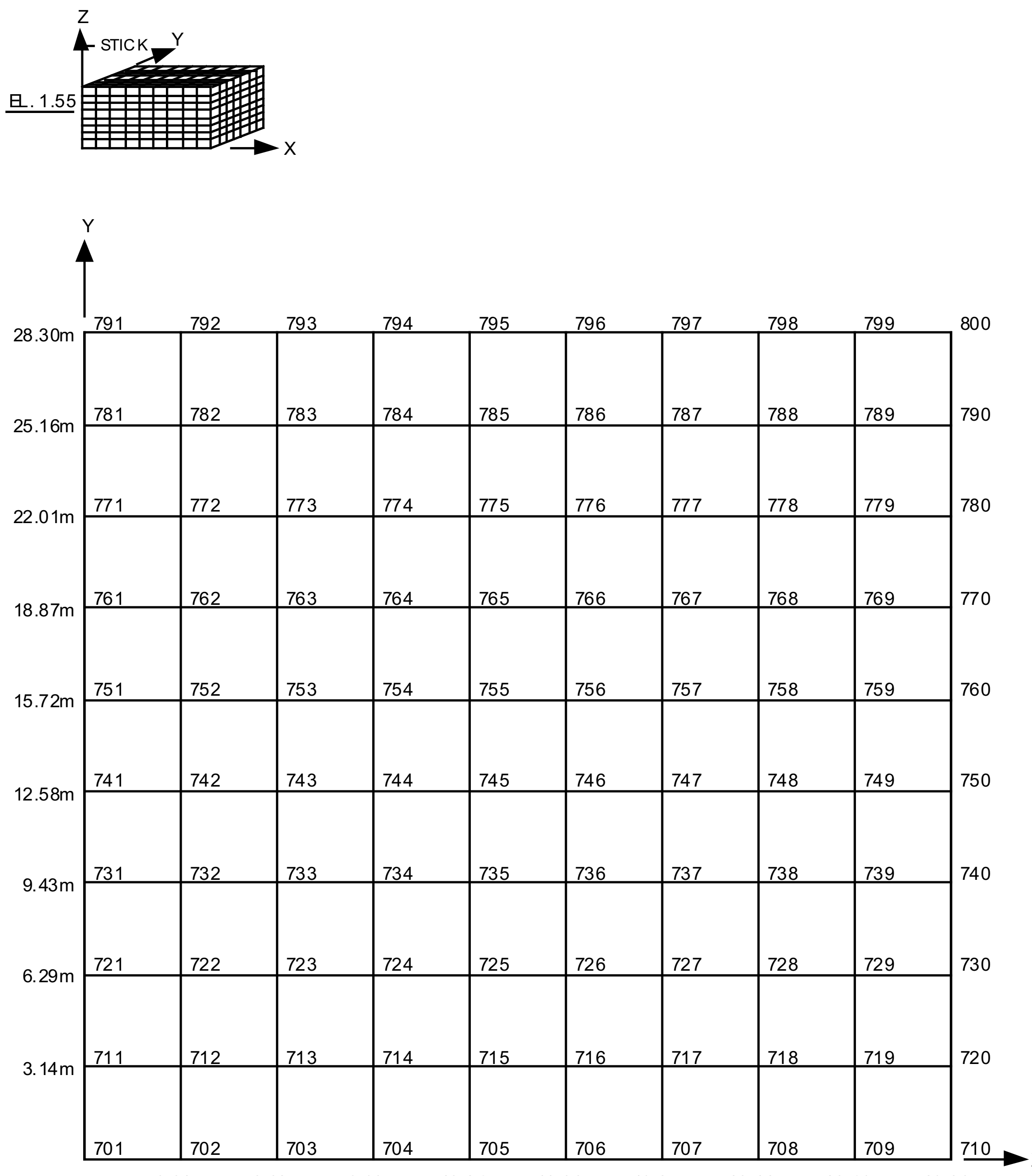


Figure 3A-21 UB Case: Nodal Points at Elevation 1.55m

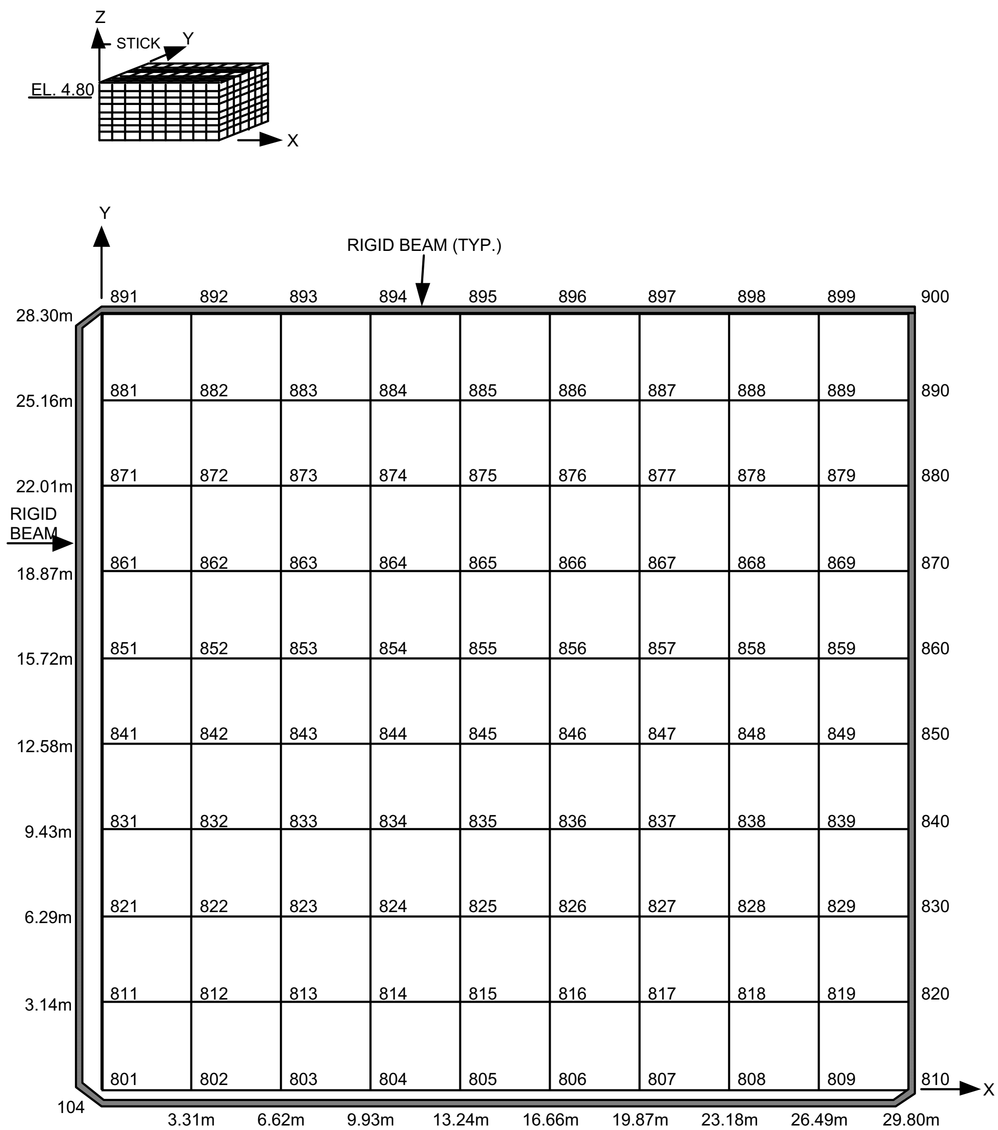


Figure 3A-22 UB Case: Nodal Points at Elevation 4.80m

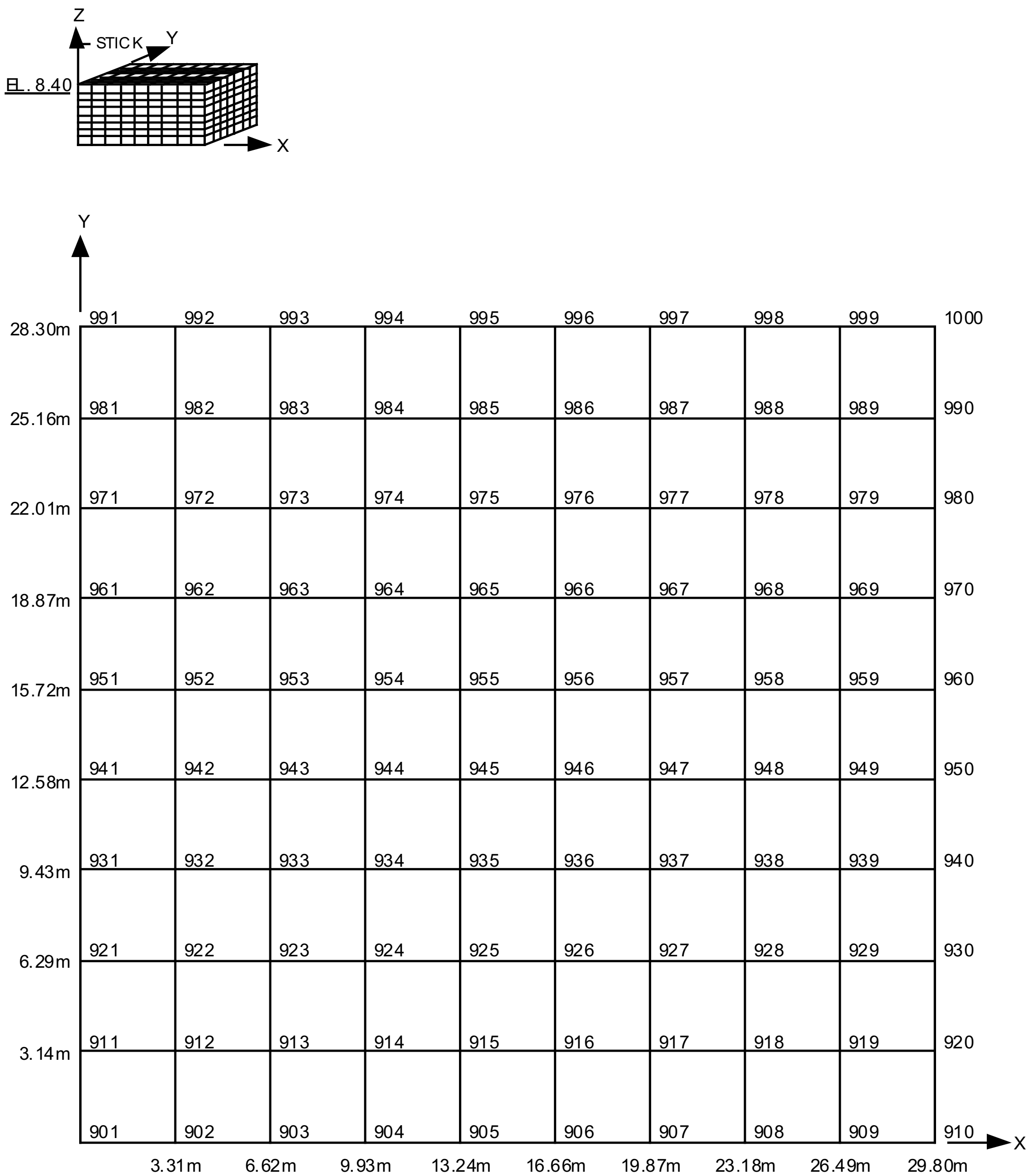


Figure 3A-23 UB Case: Nodal Points at Elevation 8.40m

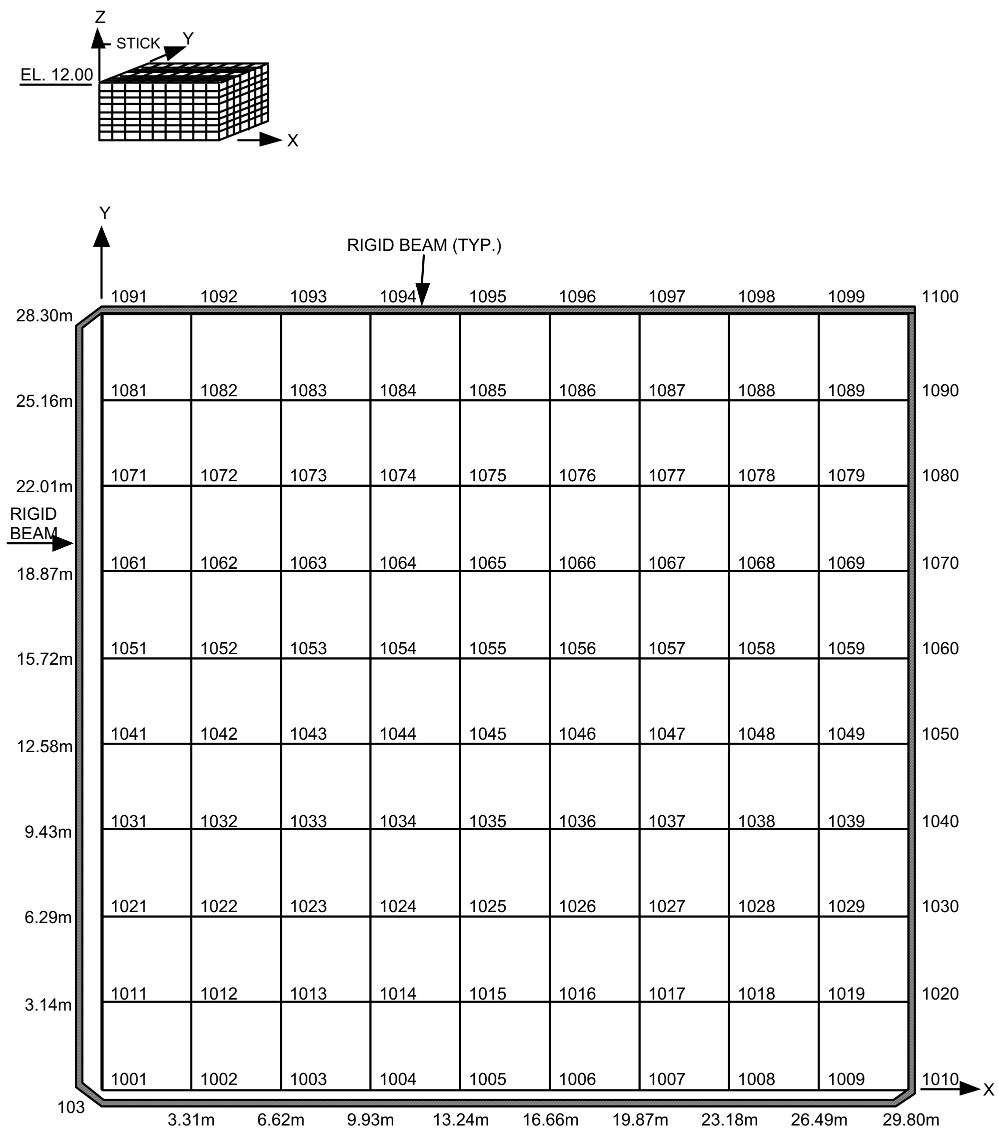


Figure 3A-24 UB Case: Nodal Points at Elevation 12.00m

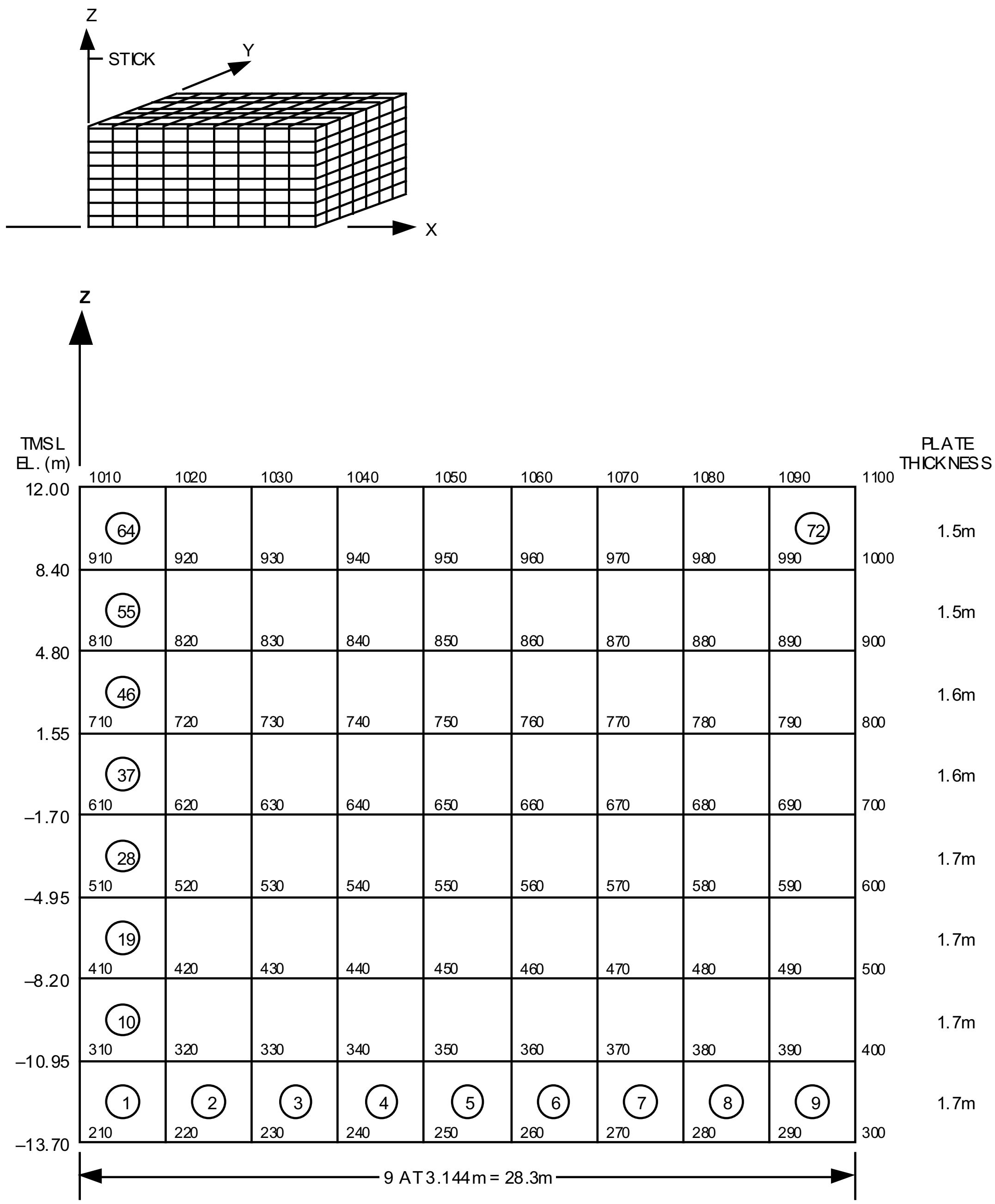


Figure 3A-25 Plate Elements of the Side Wall (X = 29.80m)

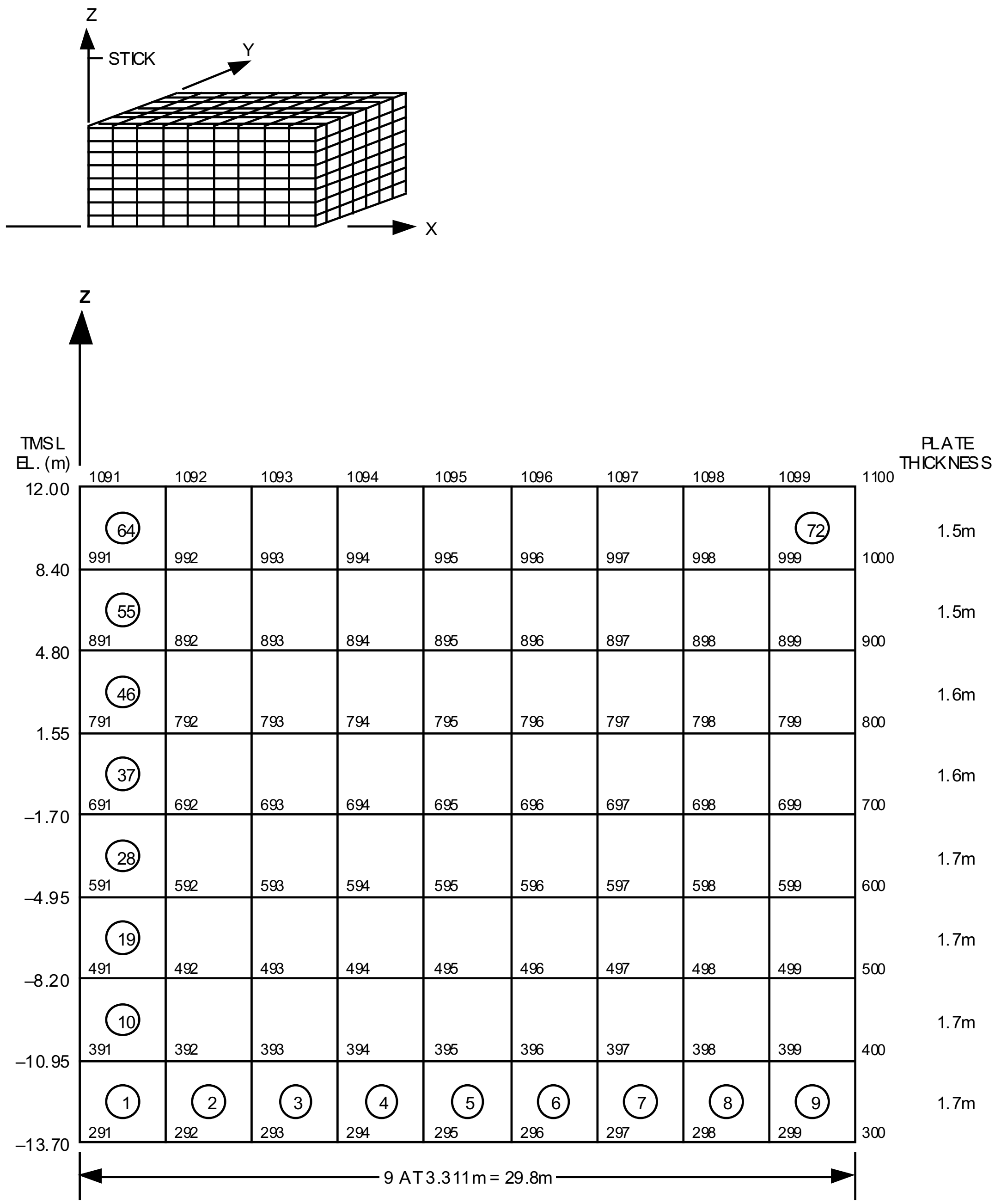


Figure 3A-26 Plate Elements of the Side Wall (Y = 28.30m)

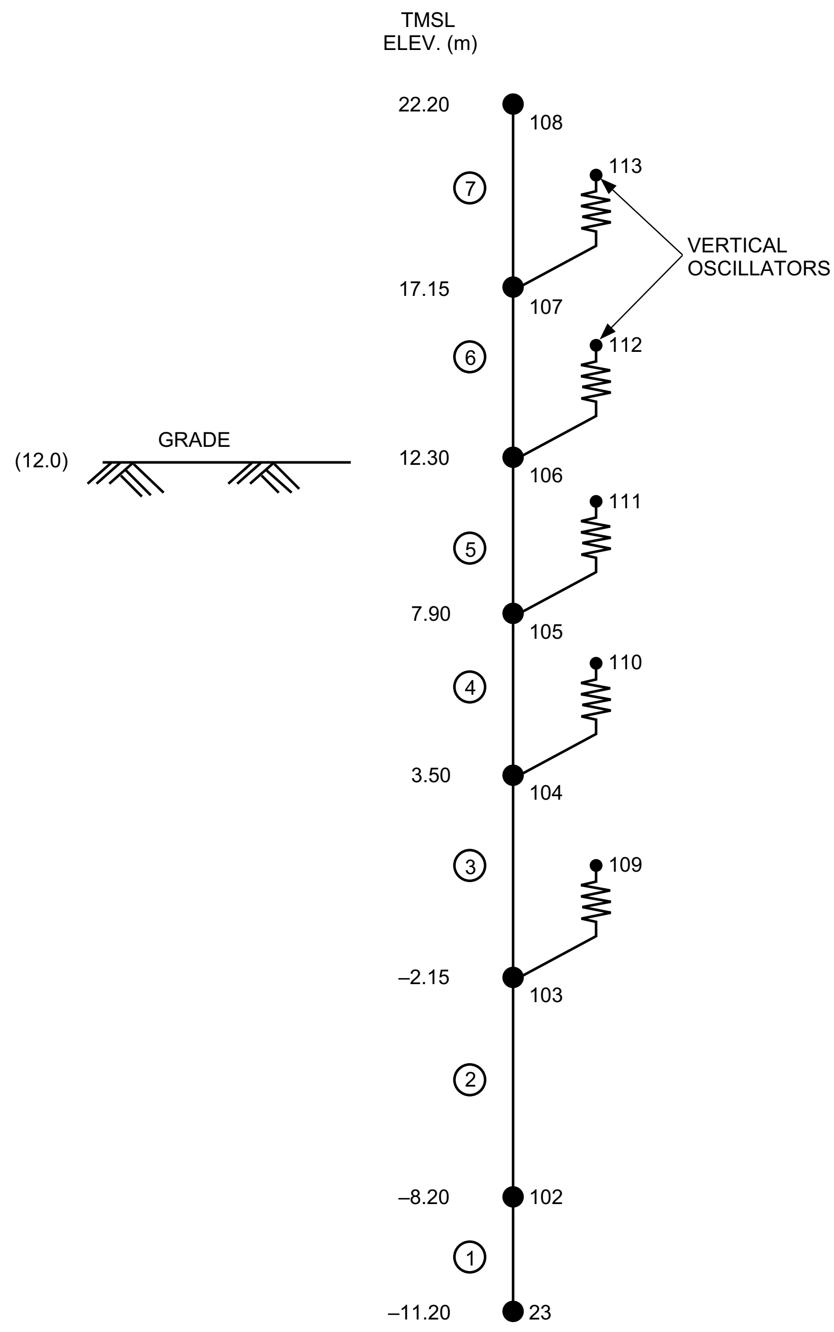


Figure 3A-27 Stick Model for the Control Building

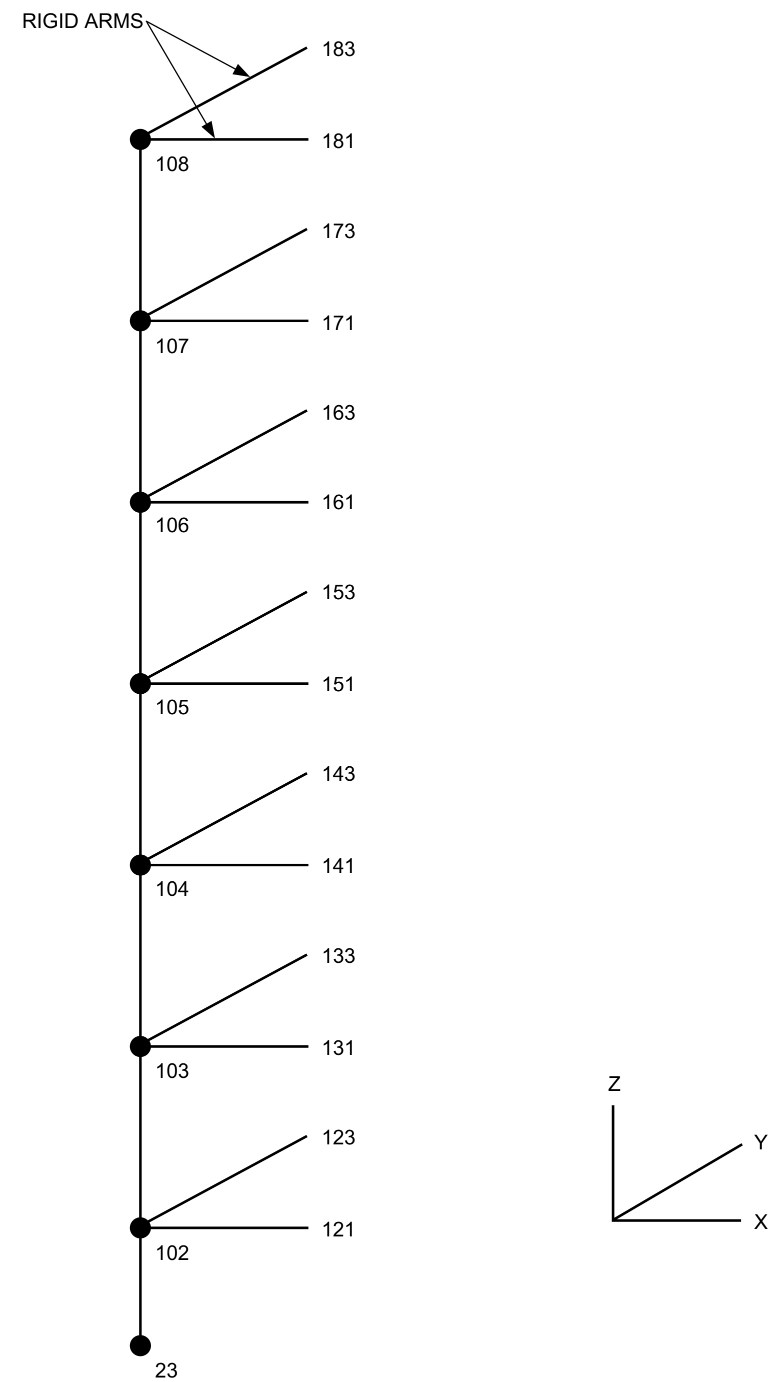


Figure 3A-28 1/4 Stick Model Showing Rigid Arms

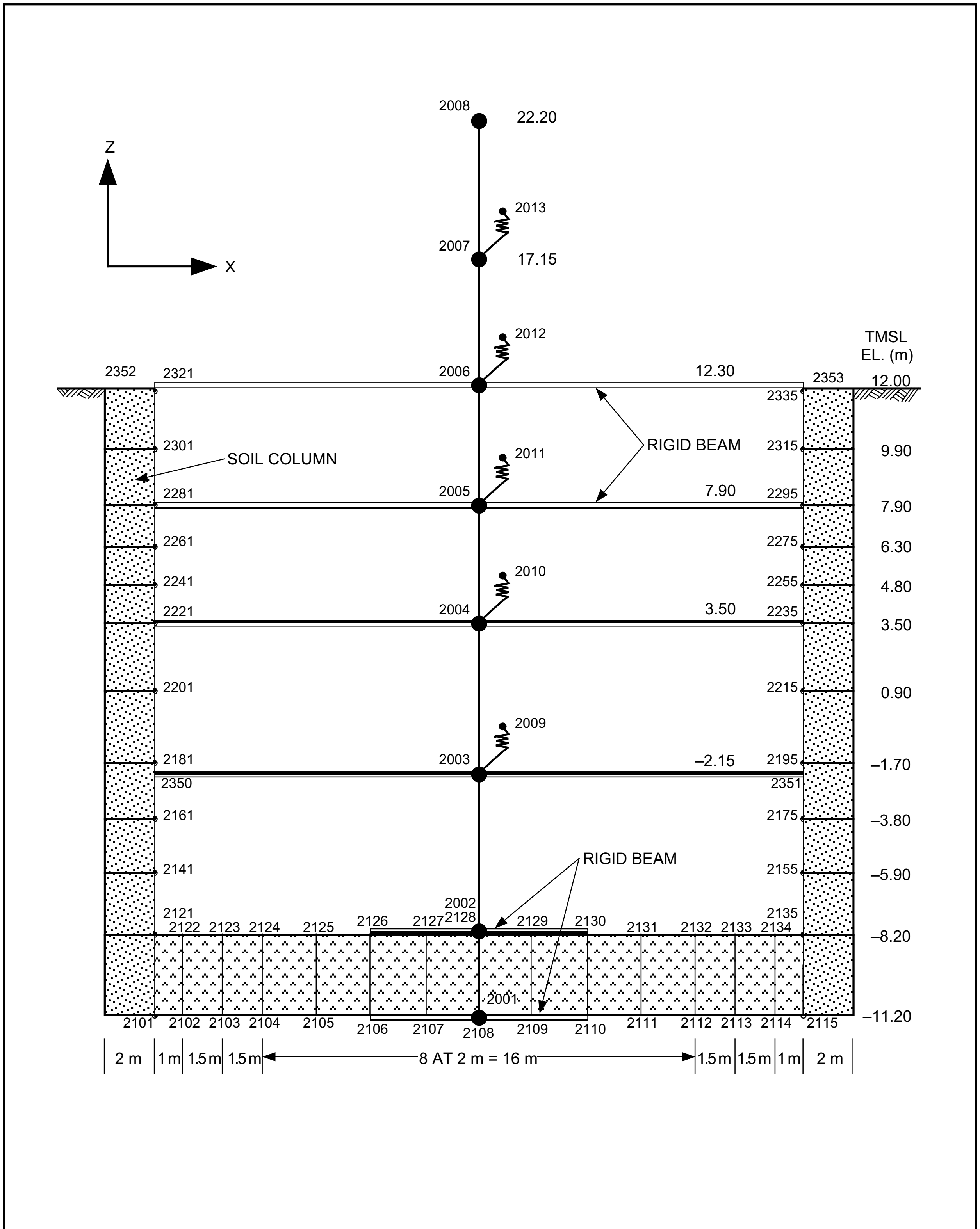


Figure 3A-29 Control Building Foundation 2-D Model XZ Direction

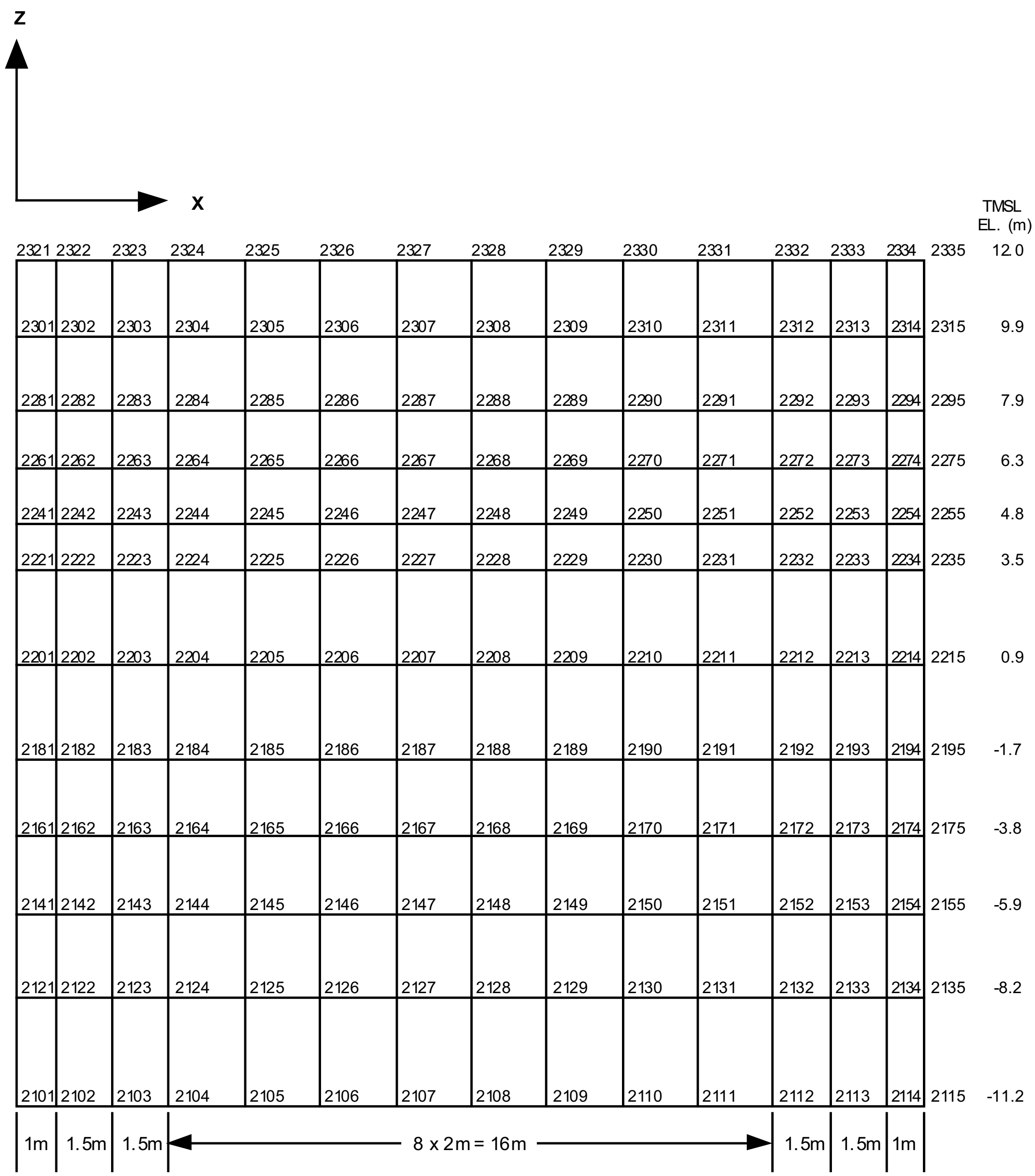


Figure 3A-30 Control Building Excavated Soil 2-D Model XZ Direction

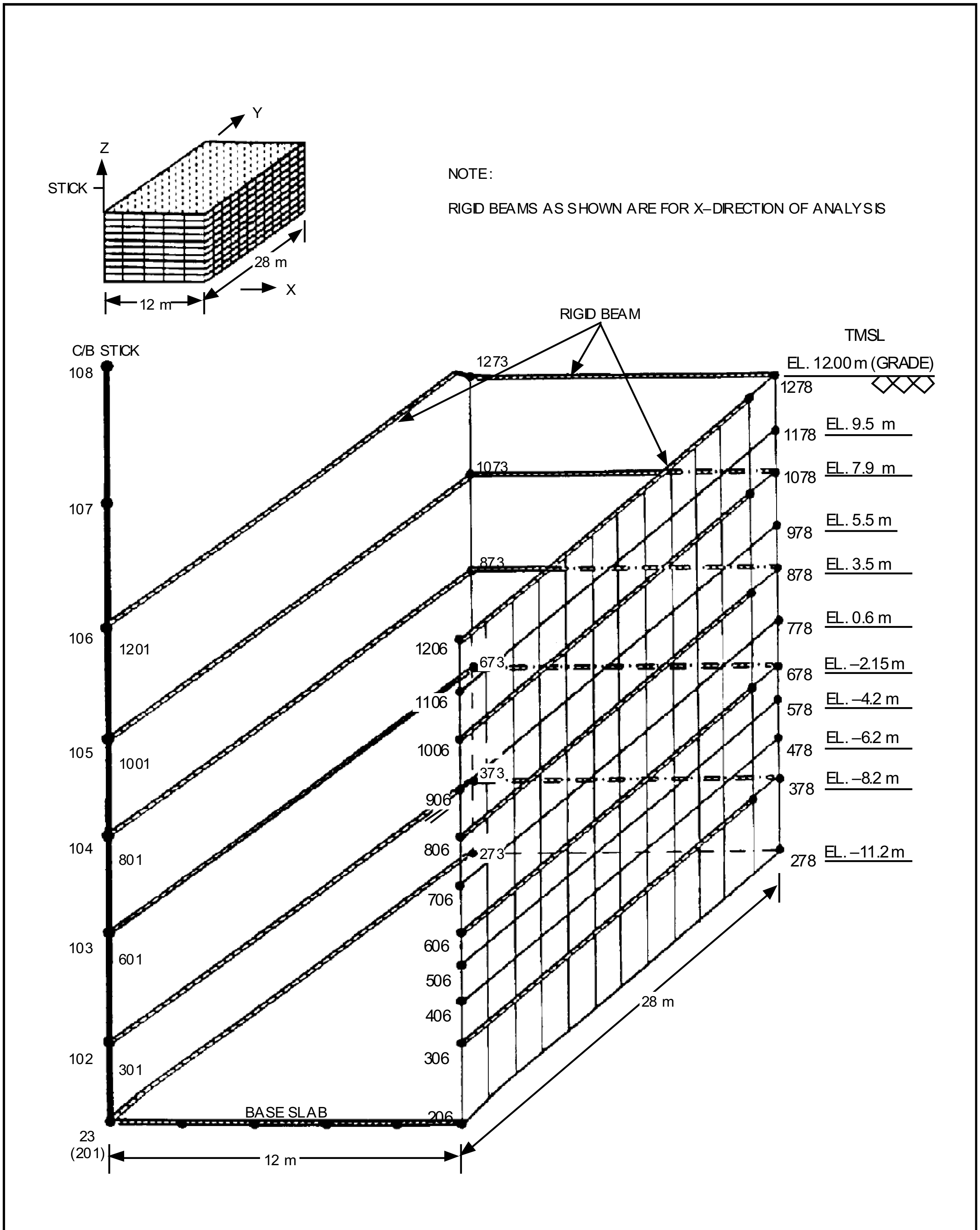


Figure 3A-31 Connection of the Main Stick to the Side Wall (Control Building for UB Soil Profiles)

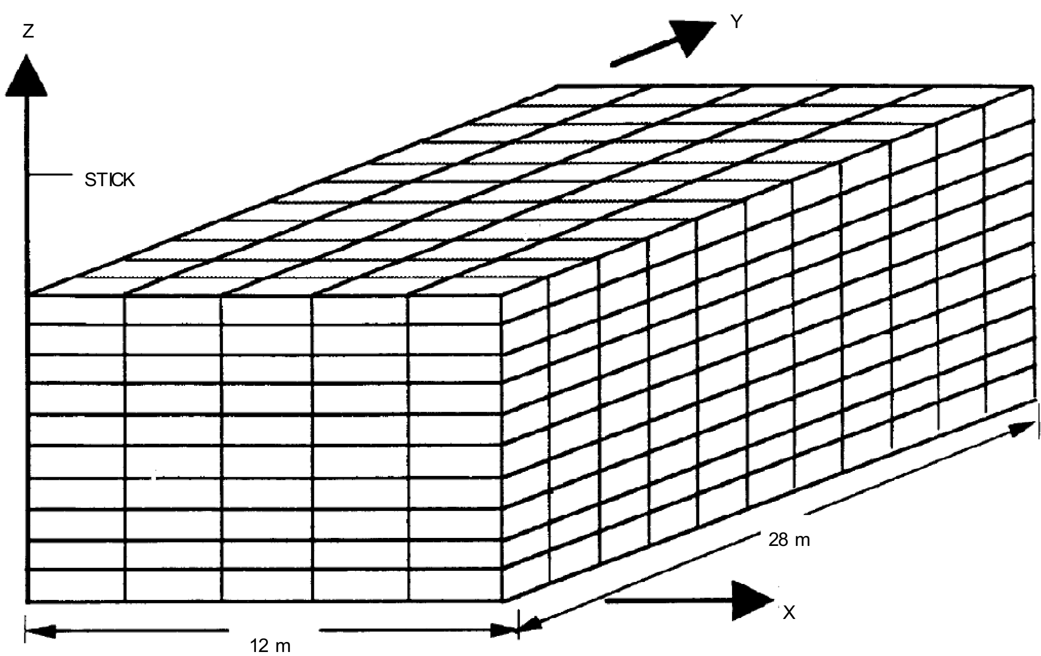


Figure 3A-32 C/B Excavated Soil Model (UB Soil Profiles)

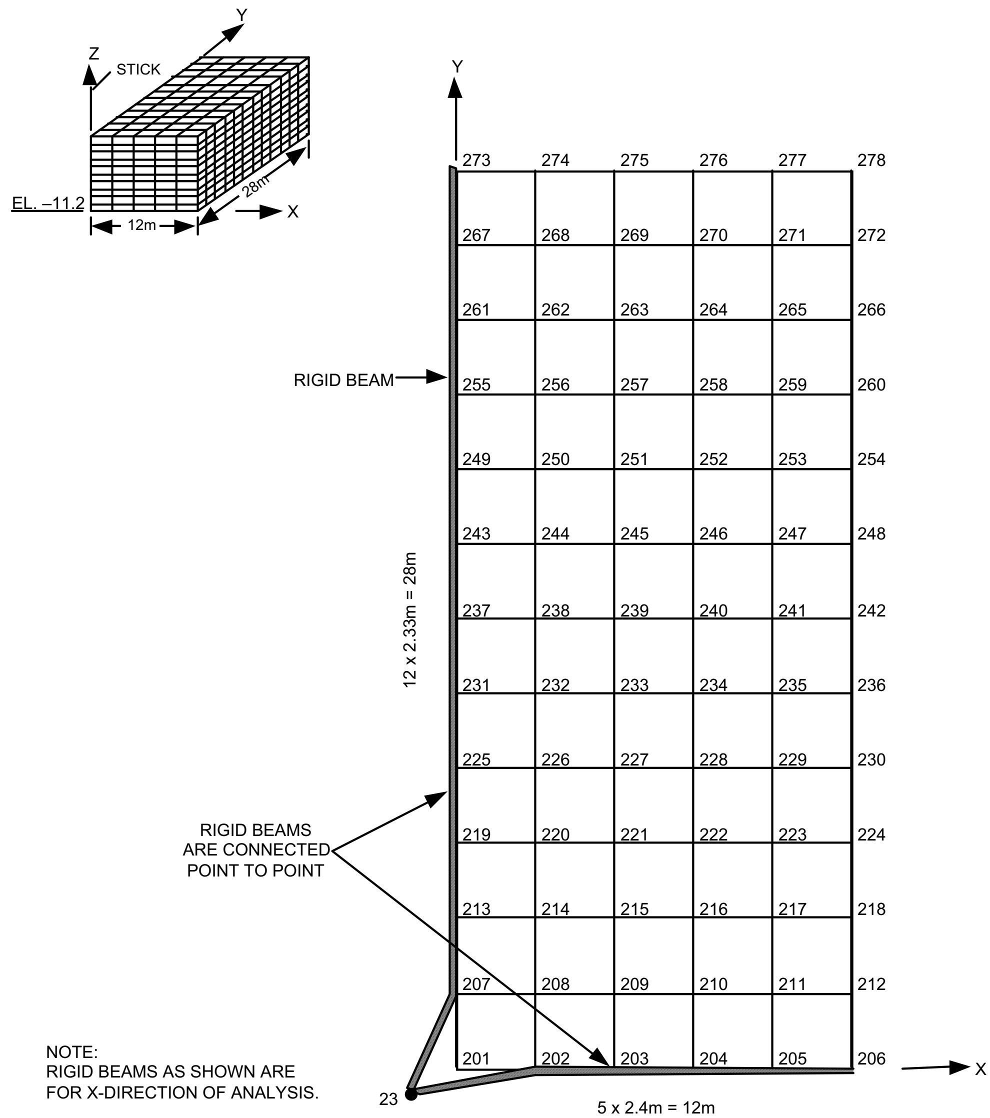


Figure 3A-33 UB Case: Nodal Points at Elevation –11.20m

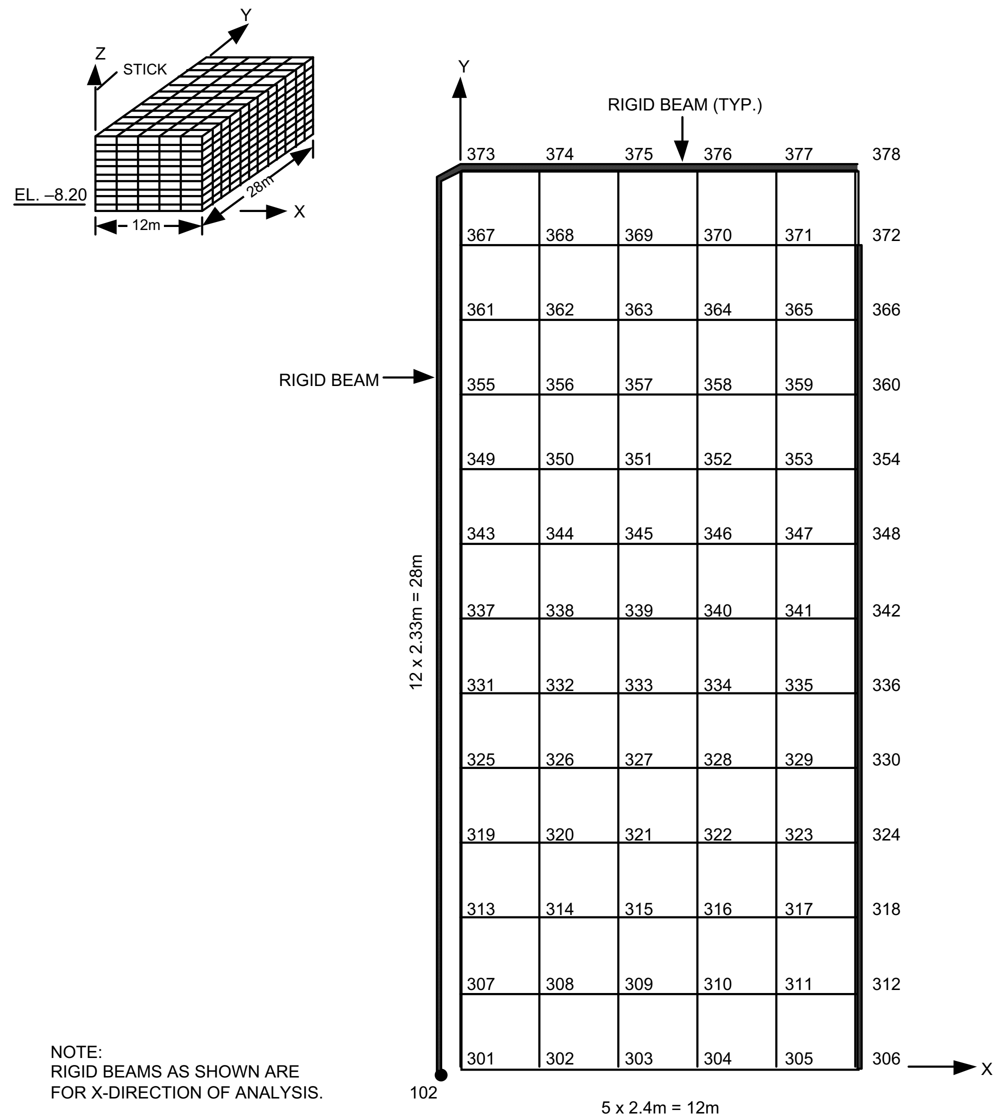


Figure 3A-34 UB Case: Nodal Points at Elevation –8.20m

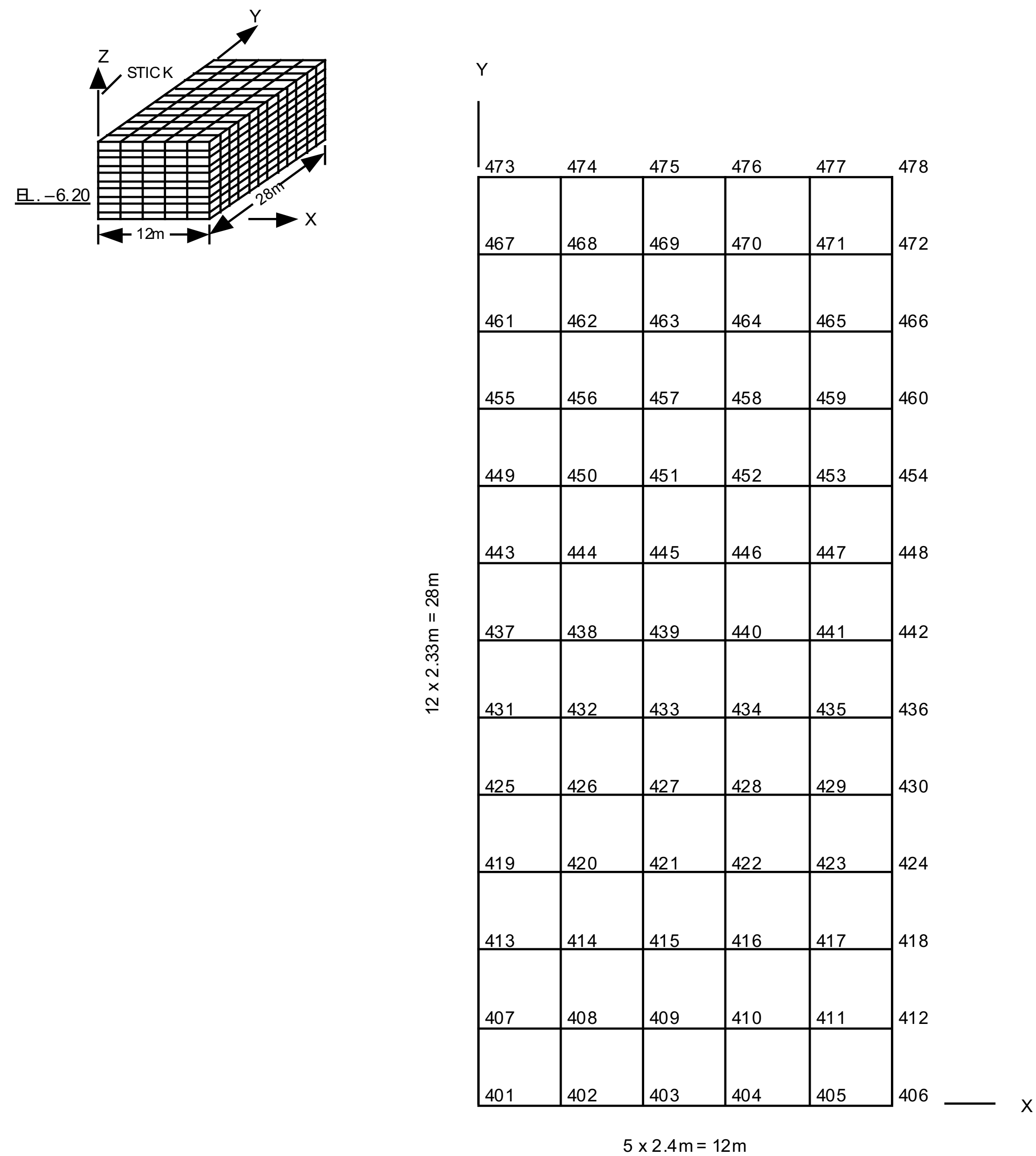


Figure 3A-35 UB Case: Nodal Points at Elevation –6.20m

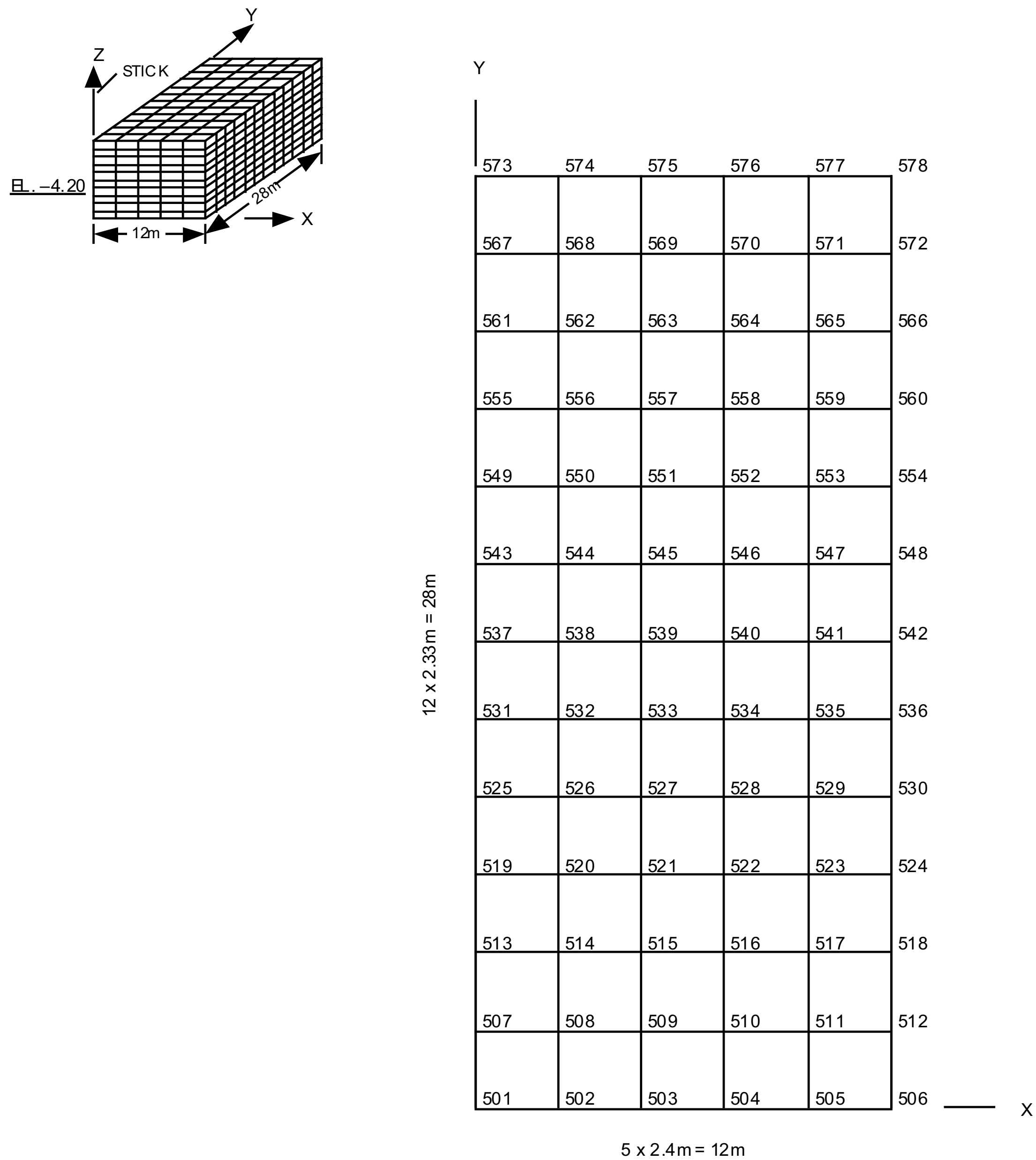


Figure 3A-36 UB Case: Nodal Points at Elevation –4.20m

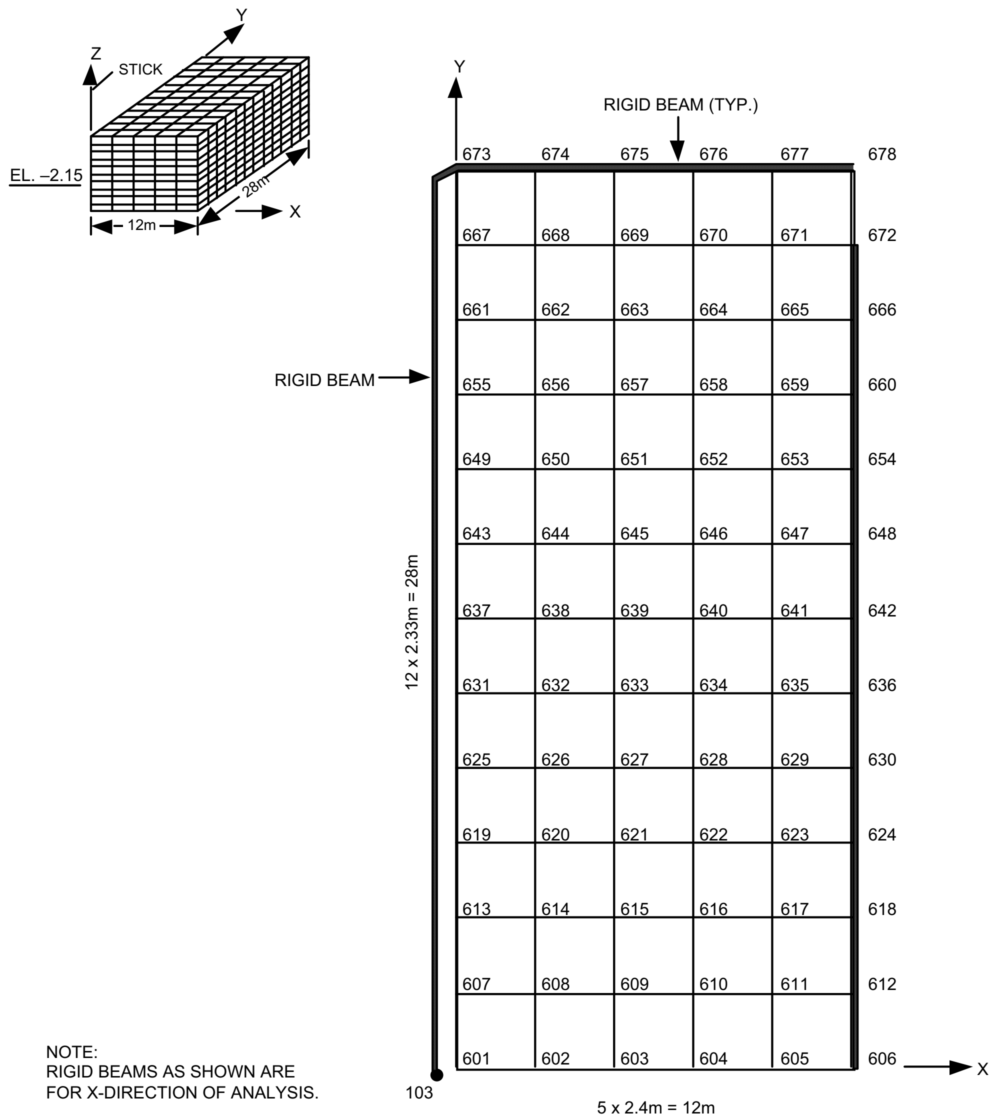


Figure 3A-37 UB Case: Nodal Points at Elevation –2.15m

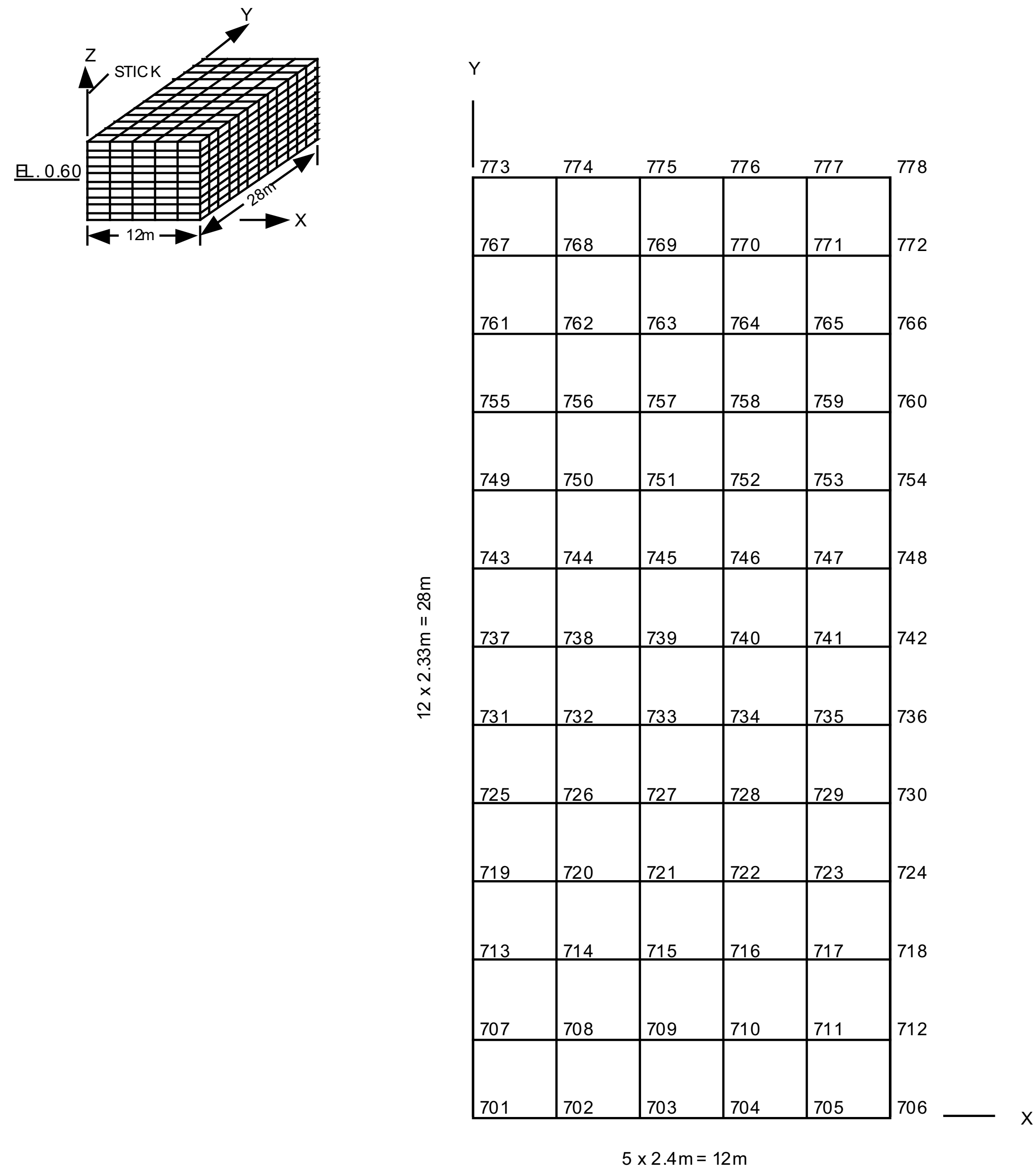


Figure 3A-38 UB Case: Nodal Points at Elevation 0.60m

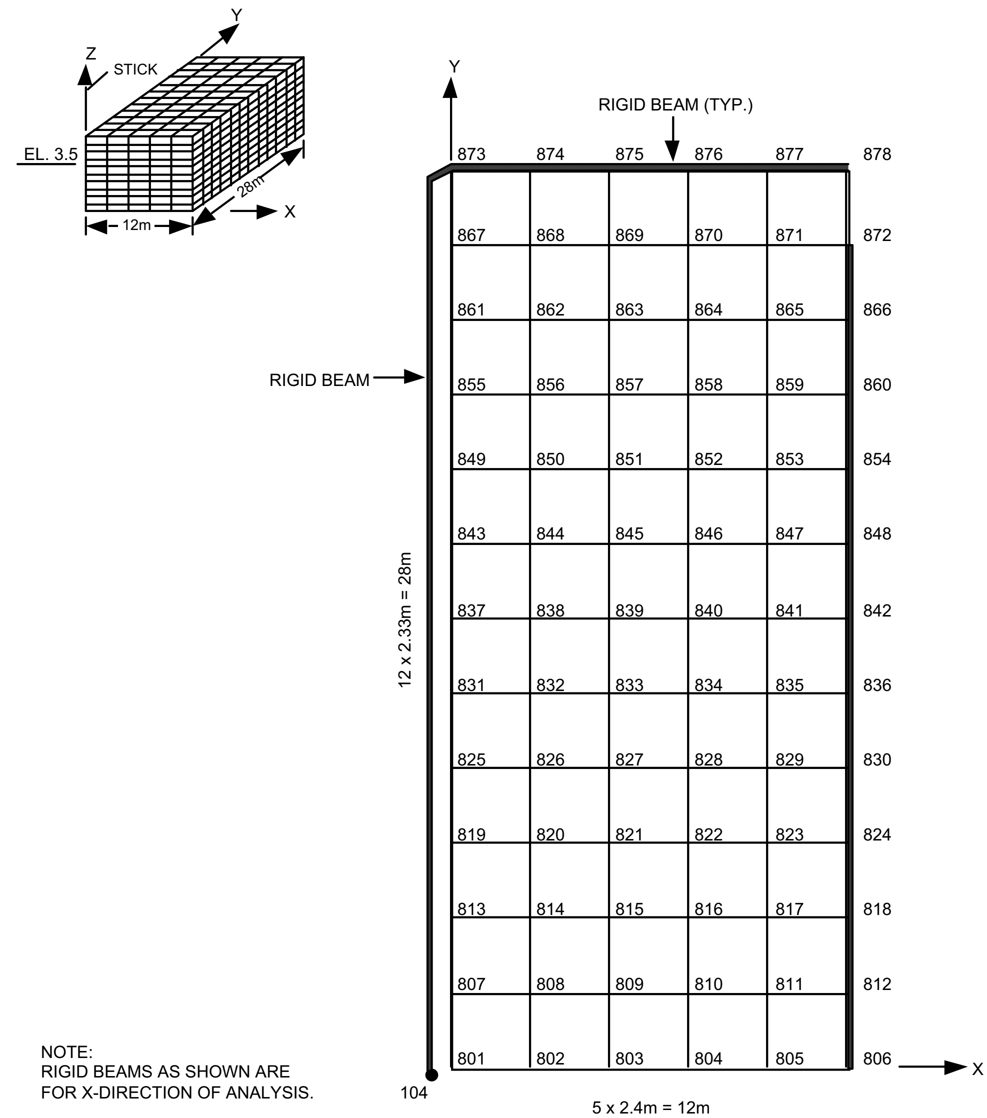


Figure 3A-39 UB Case: Nodal Points at Elevation 3.50m

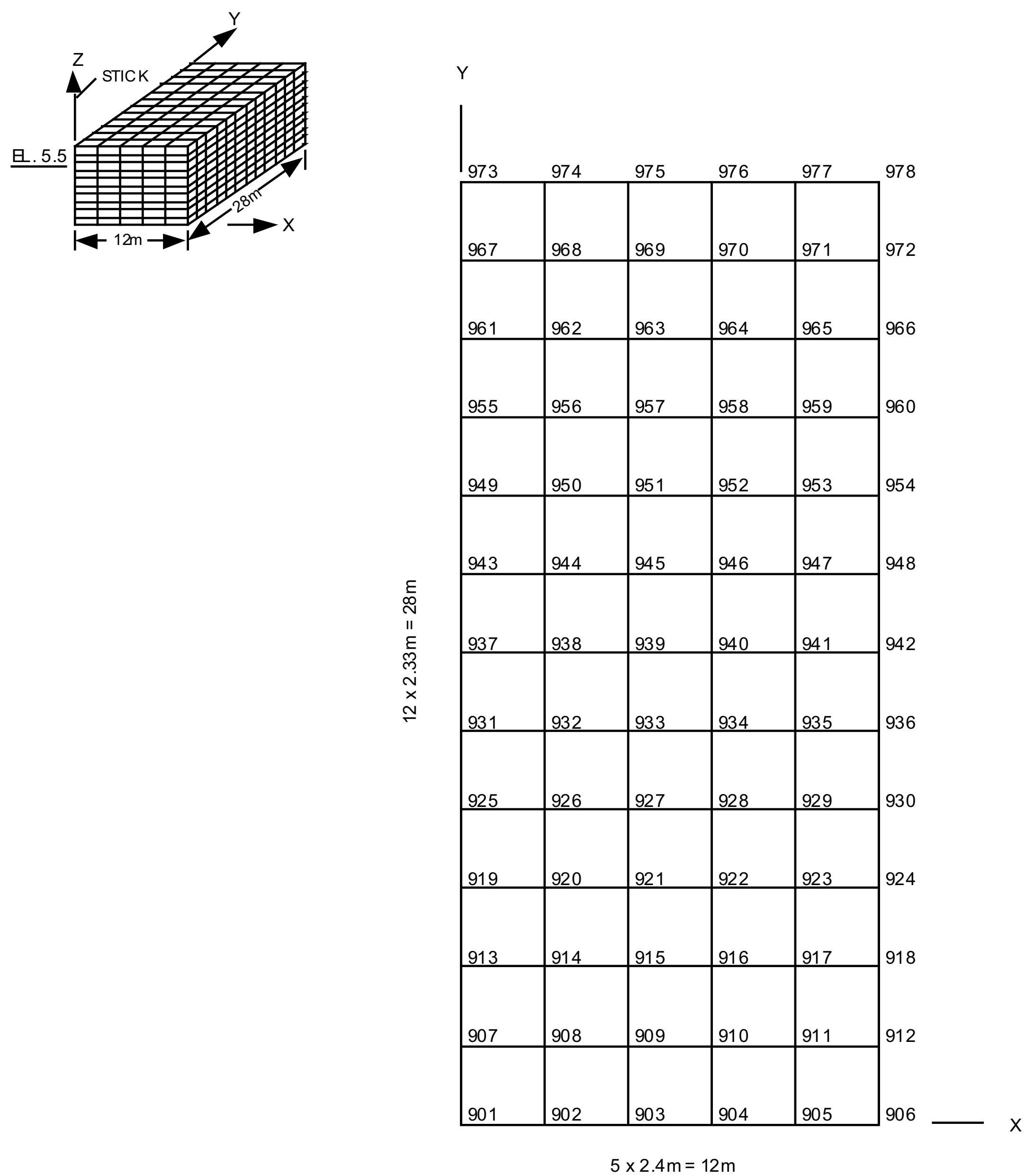


Figure 3A-40 UB Case: Nodal Points at Elevation 5.50m

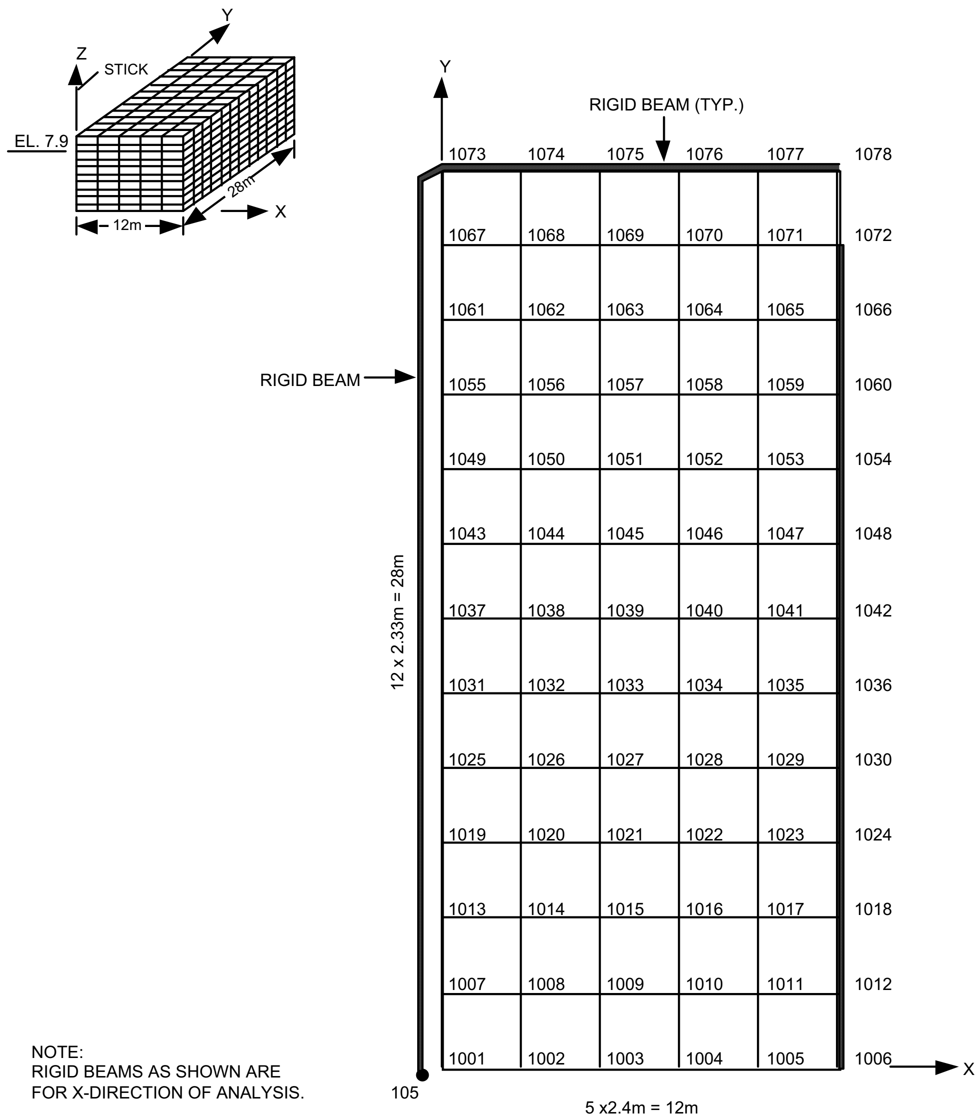


Figure 3A-41 UB Case: Nodal Points at Elevation 7.90m

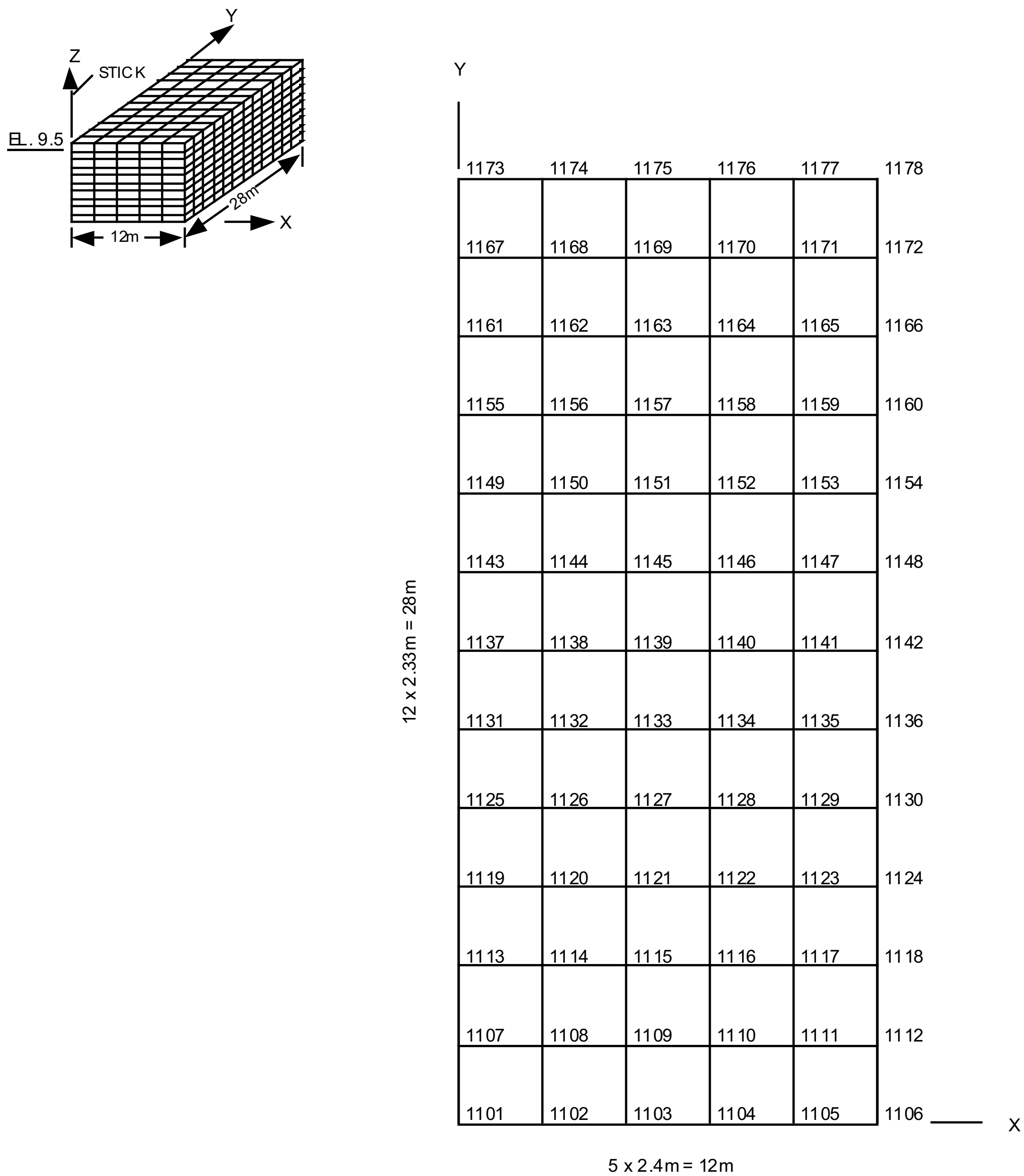


Figure 3A-42 UB Case: Nodal Points at Elevation 9.50m

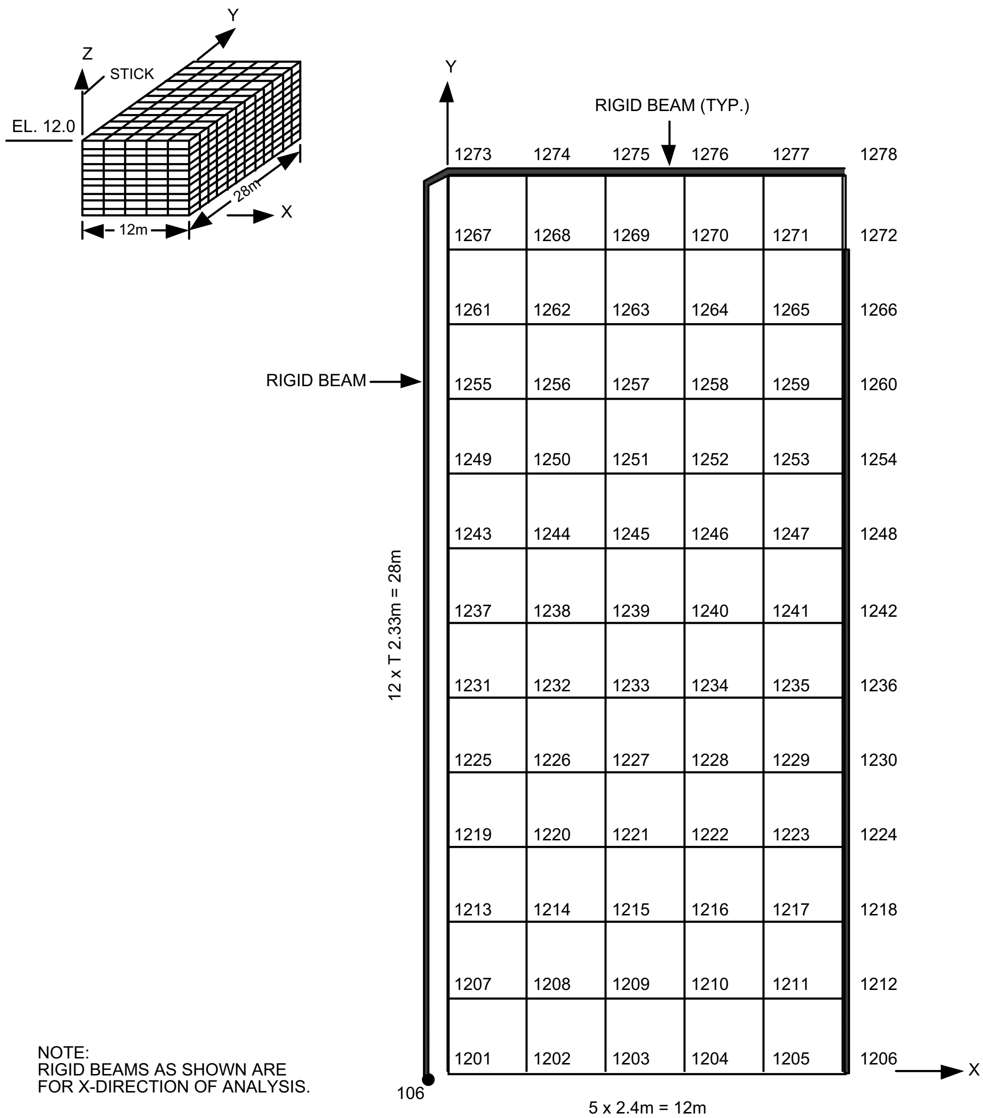


Figure 3A-43 UB Case: Nodal Points at Elevation 12.00m

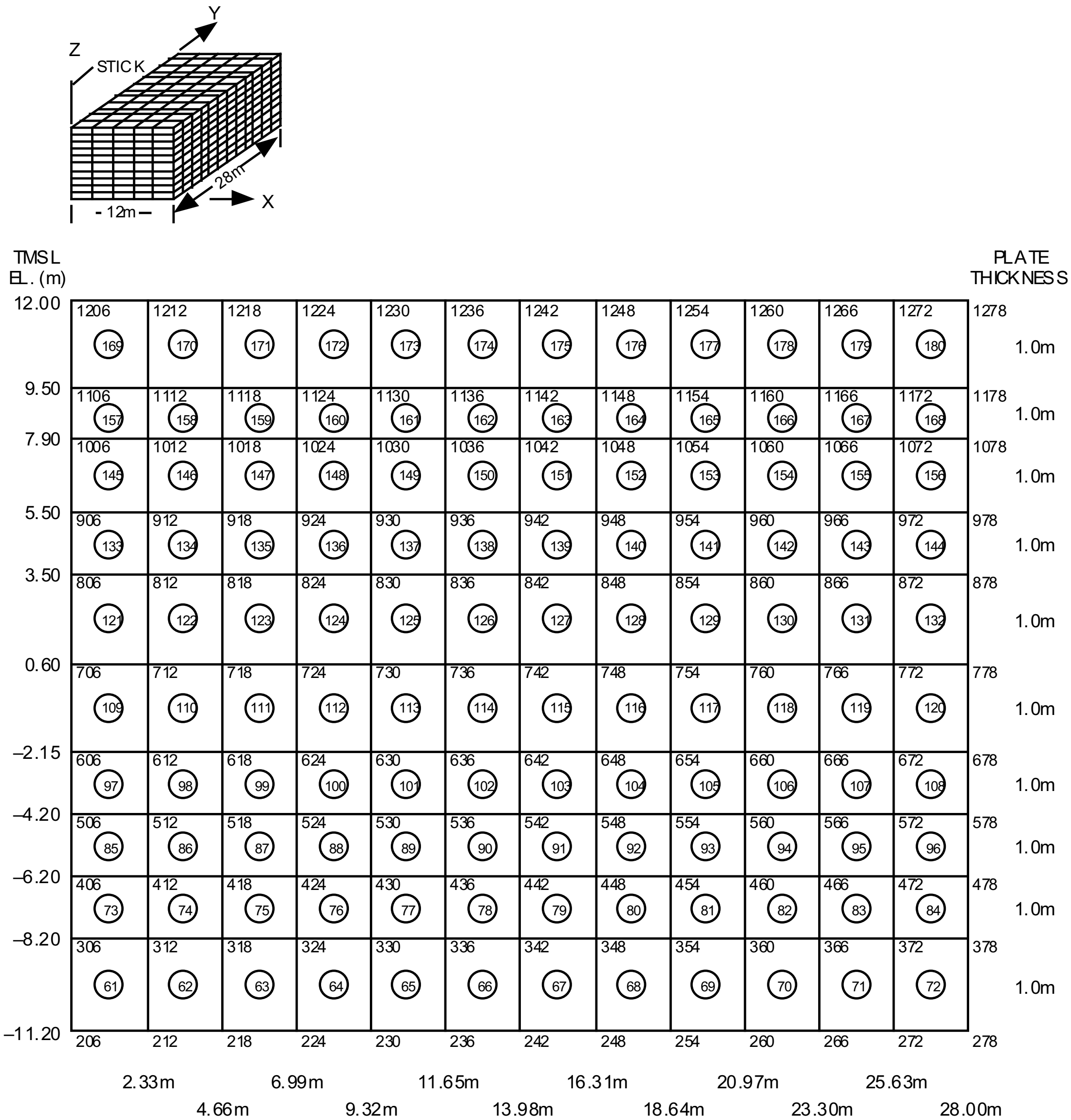


Figure 3A-44 Plate Elements of the Side Wall (X = 12.00m)

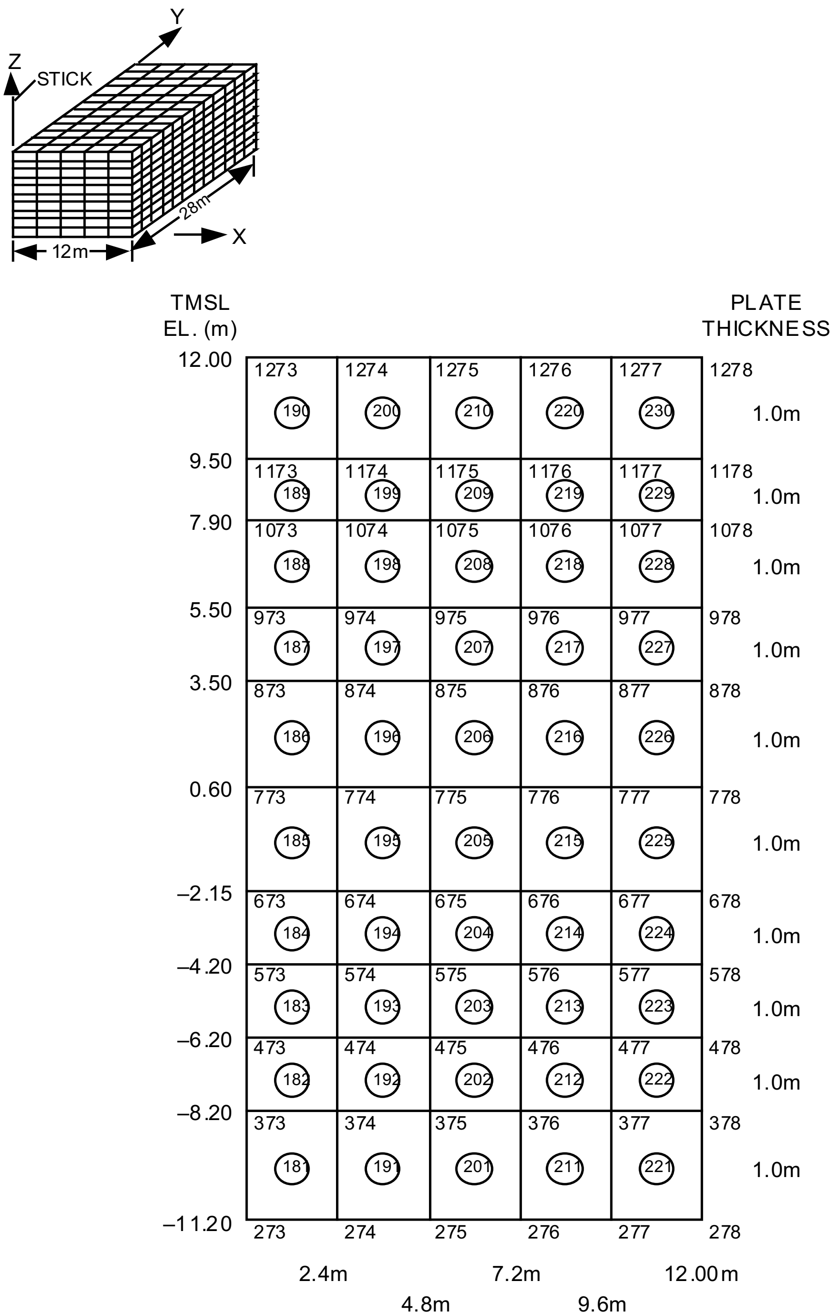


Figure 3A-45 Plate Elements of the Side Wall (Y = 28.00m)

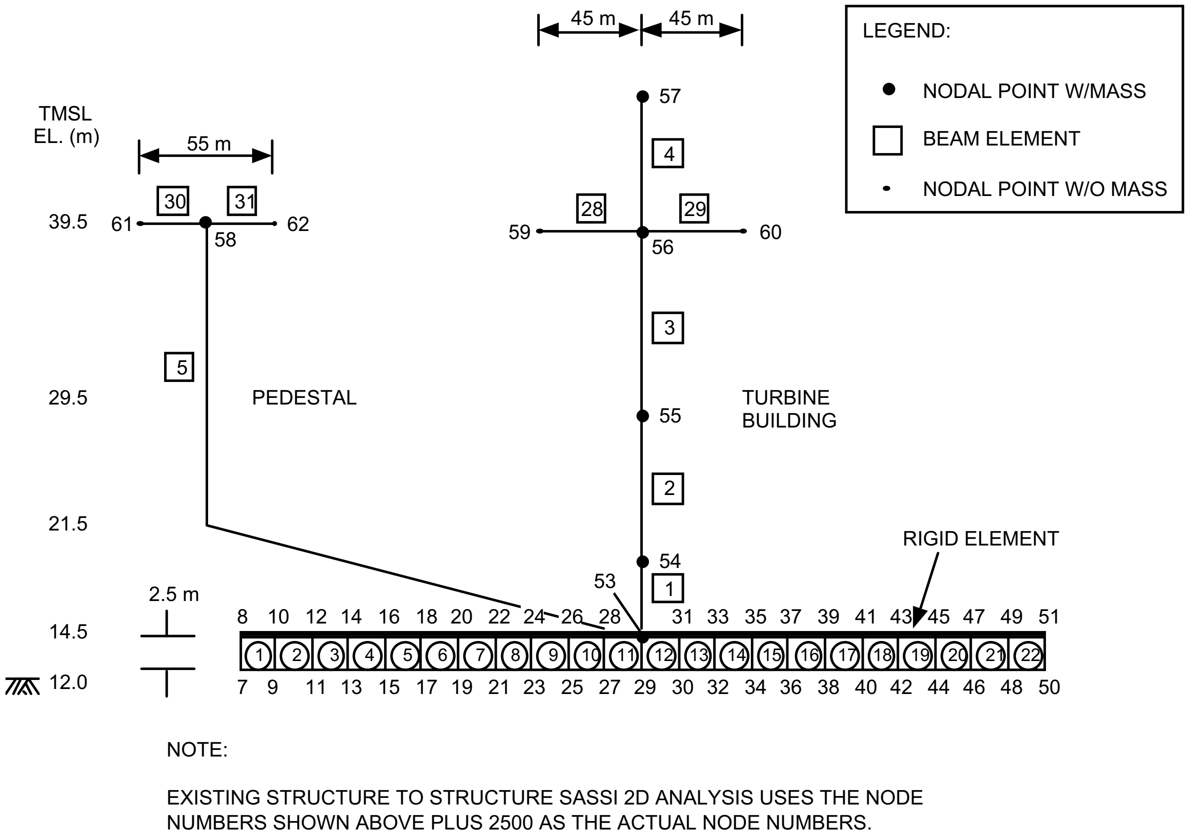


Figure 3A-46 Turbine Building Model

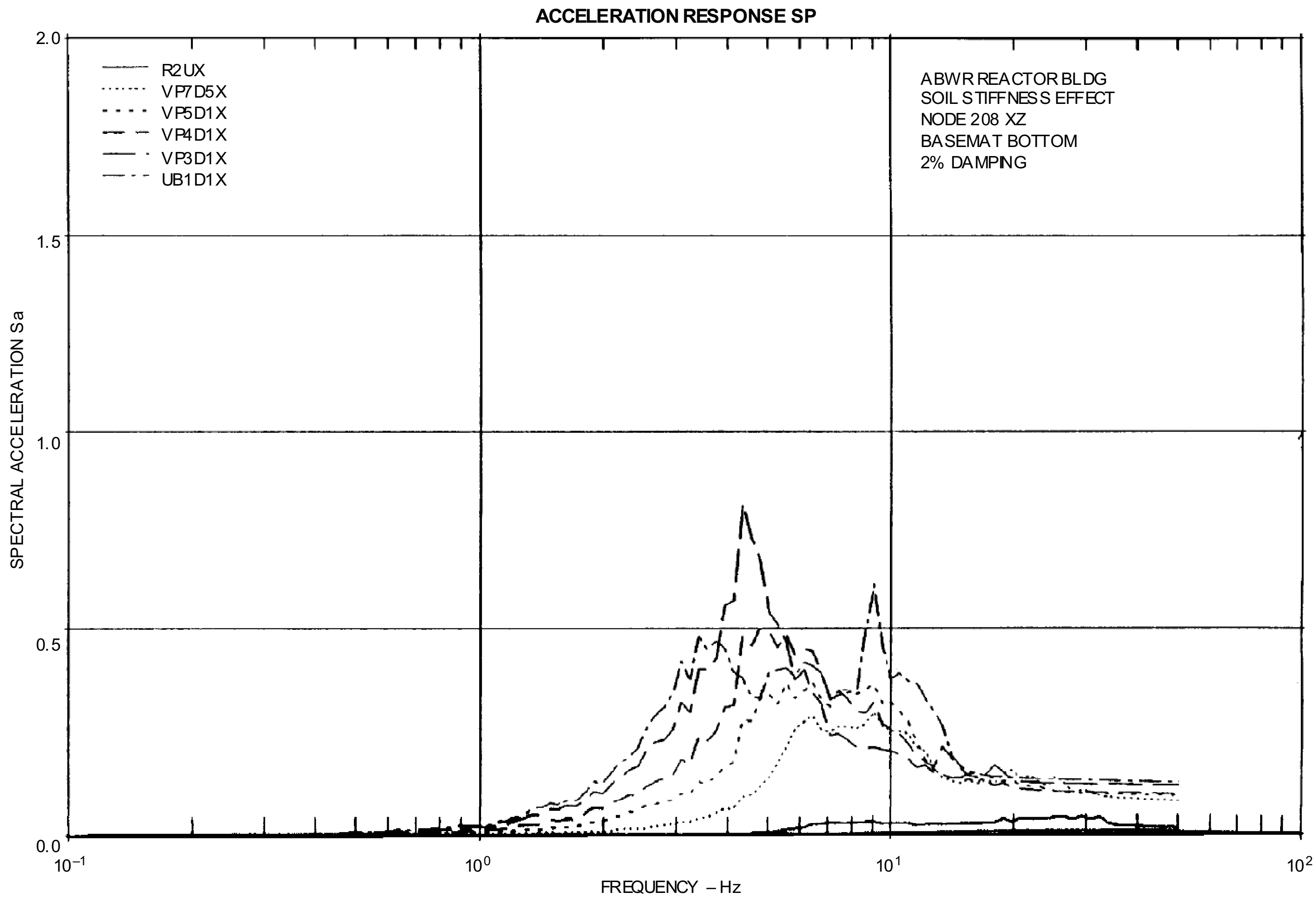


Figure 3A-47 ABWR Reactor Bldg. Soil Stiffness Effect, Node 208 XZ Basemat Bottom, 2% Damping

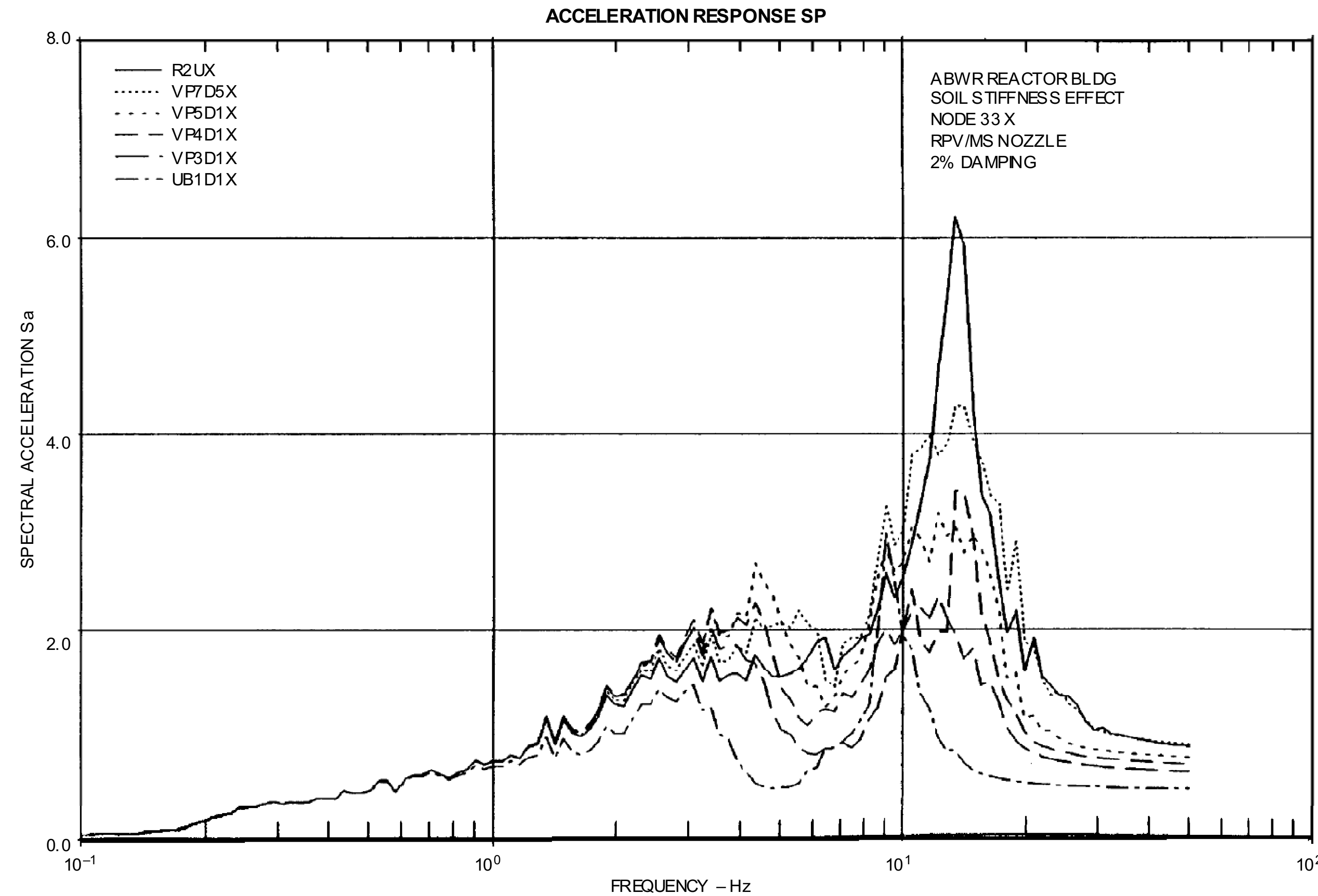


Figure 3A-48 ABWR Reactor Bldg. Soil Stiffness Effect, Node 33 X RPV/MS Nozzle, 2% Damping

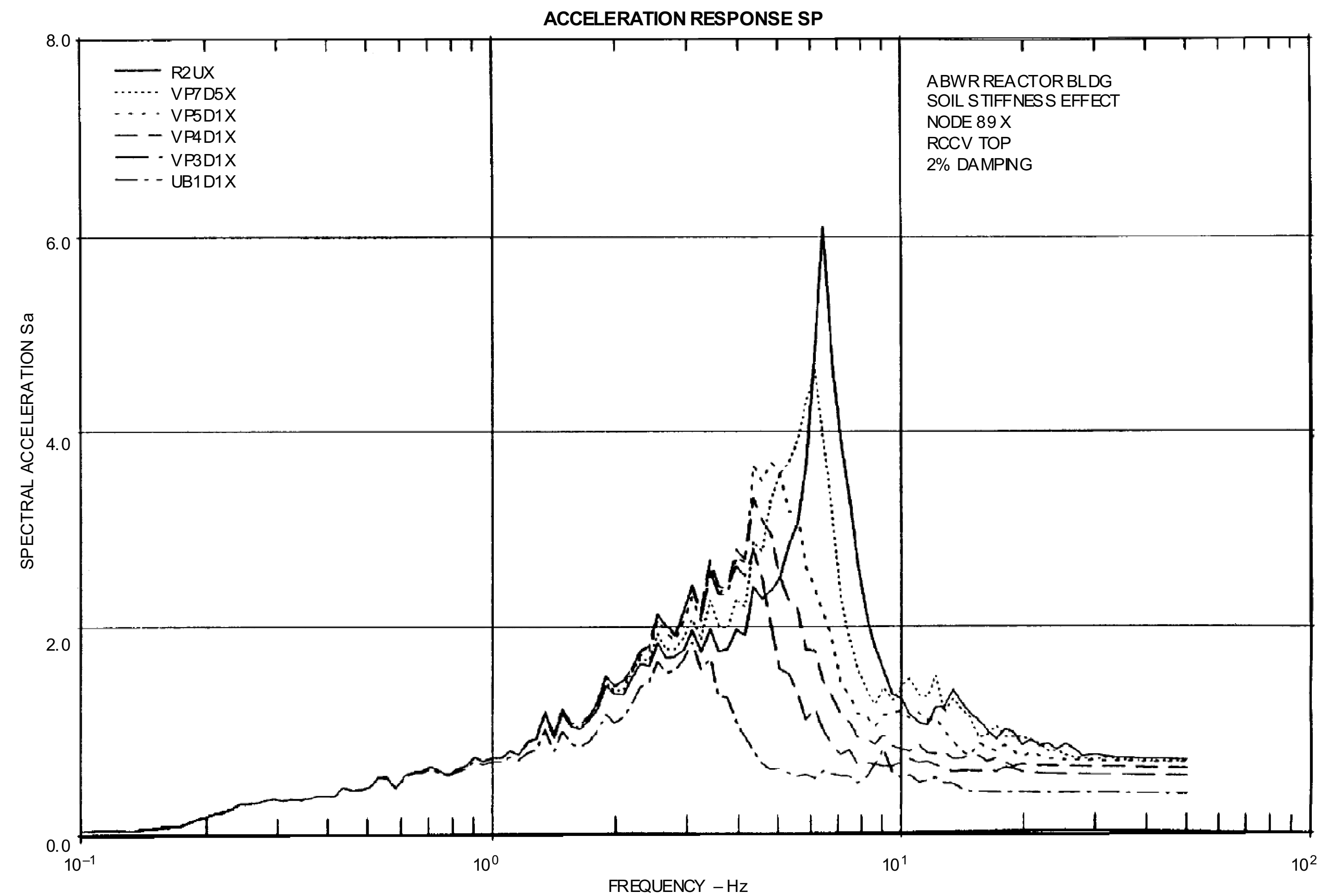


Figure 3A-49 ABWR Reactor Bldg. Soil Stiffness Effect, Node 89 X RCCV Top, 2% Damping

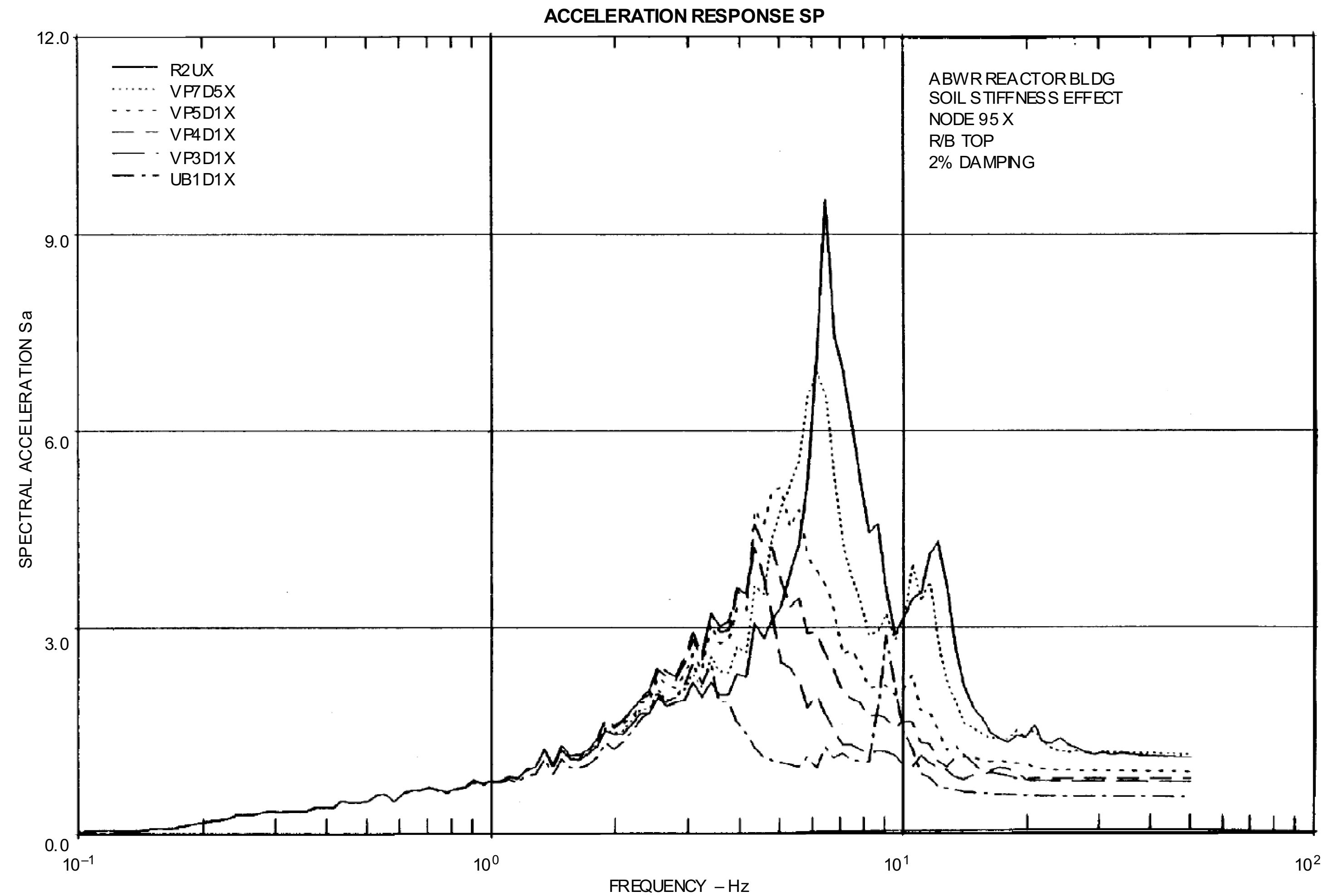


Figure 3A-50 ABWR Reactor Bldg. Soil Stiffness Effect, Node 95 X R/B Top, 2% Damping

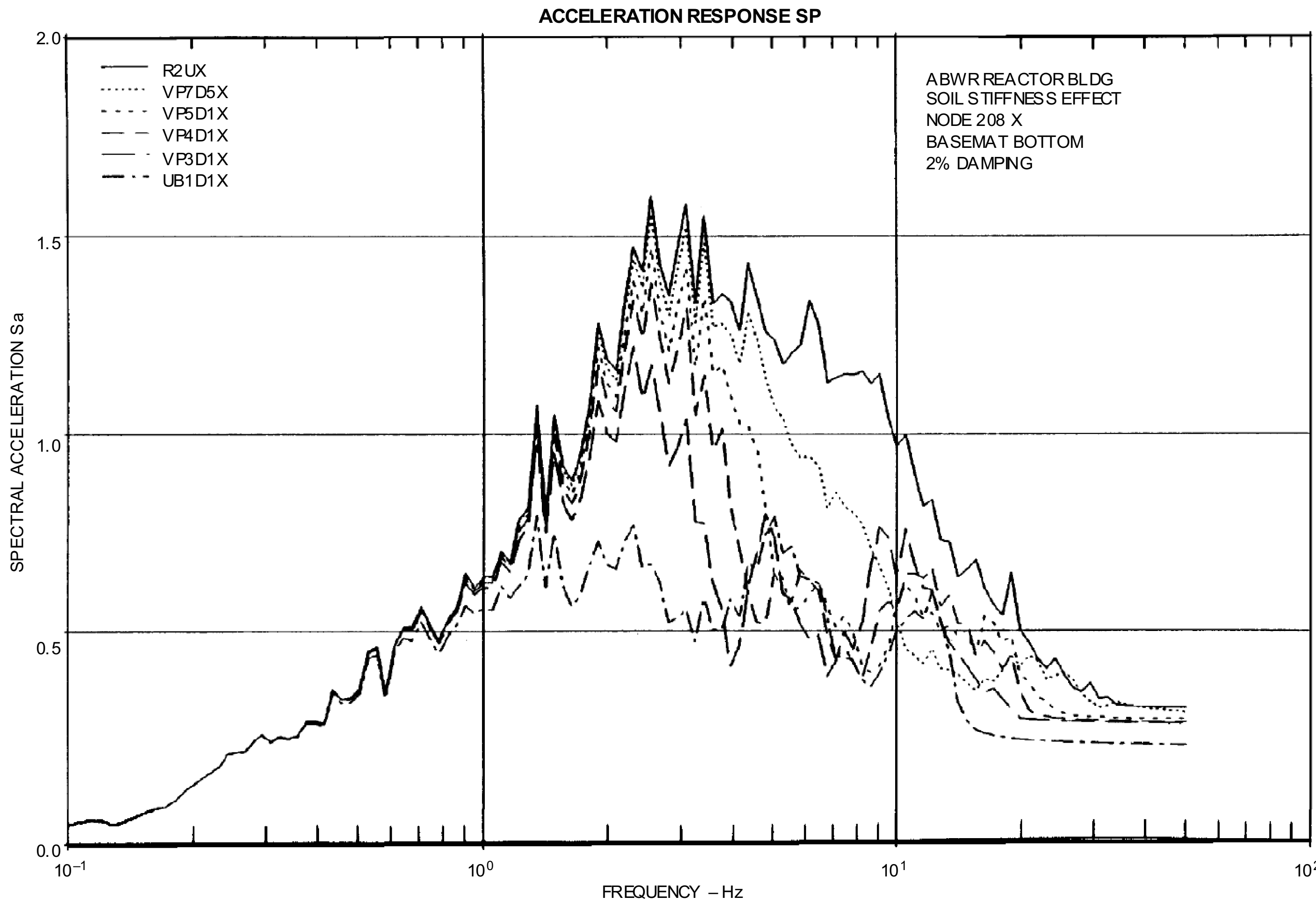


Figure 3A-51 ABWR Reactor Bldg. Soil Stiffness Effect, Node 208 X Basemat Bottom, 2% Damping

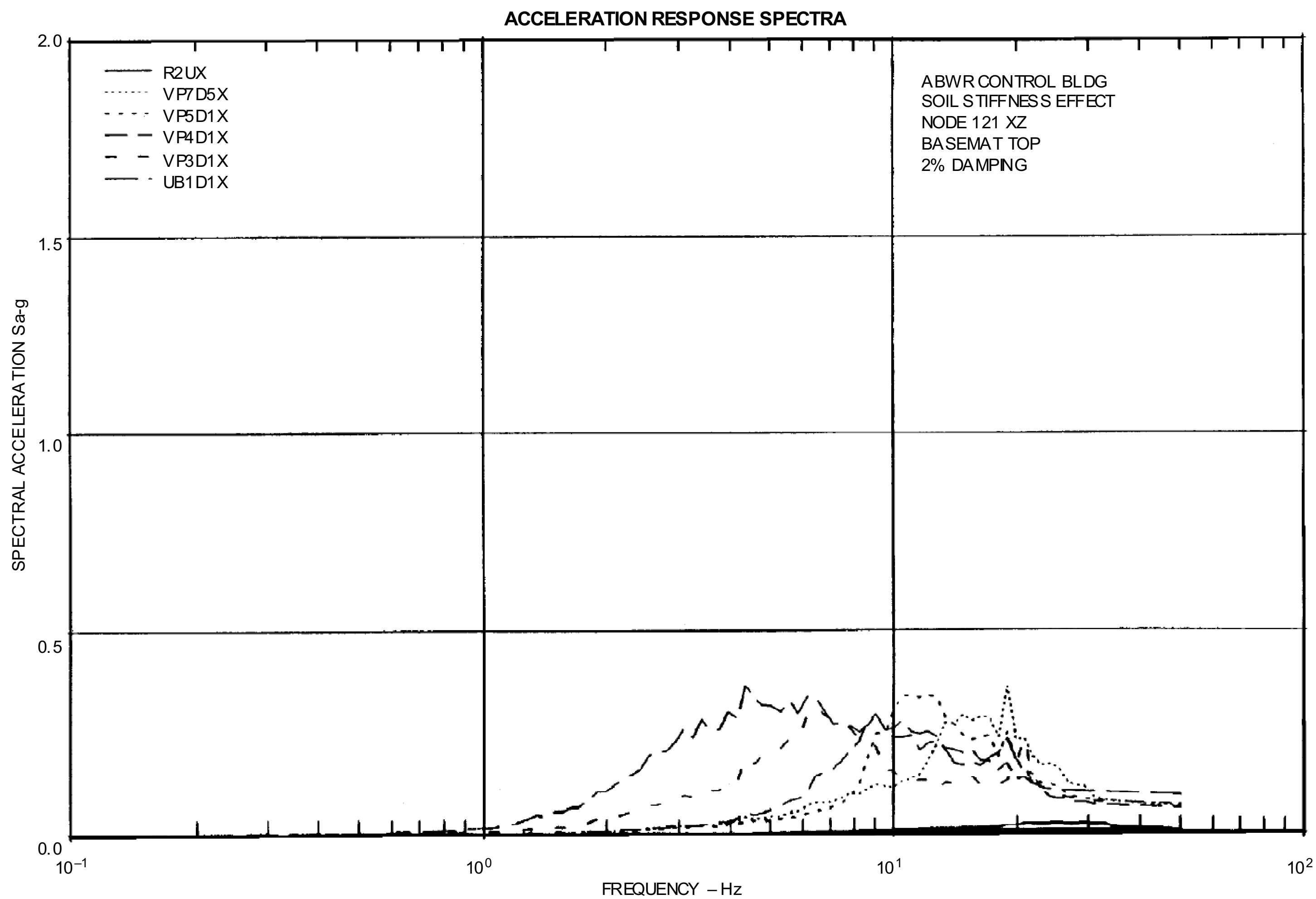


Figure 3A-52 ABWR Control Bldg. Soil Stiffness Effect, Node 121 XZ Basemat Top, 2% Damping

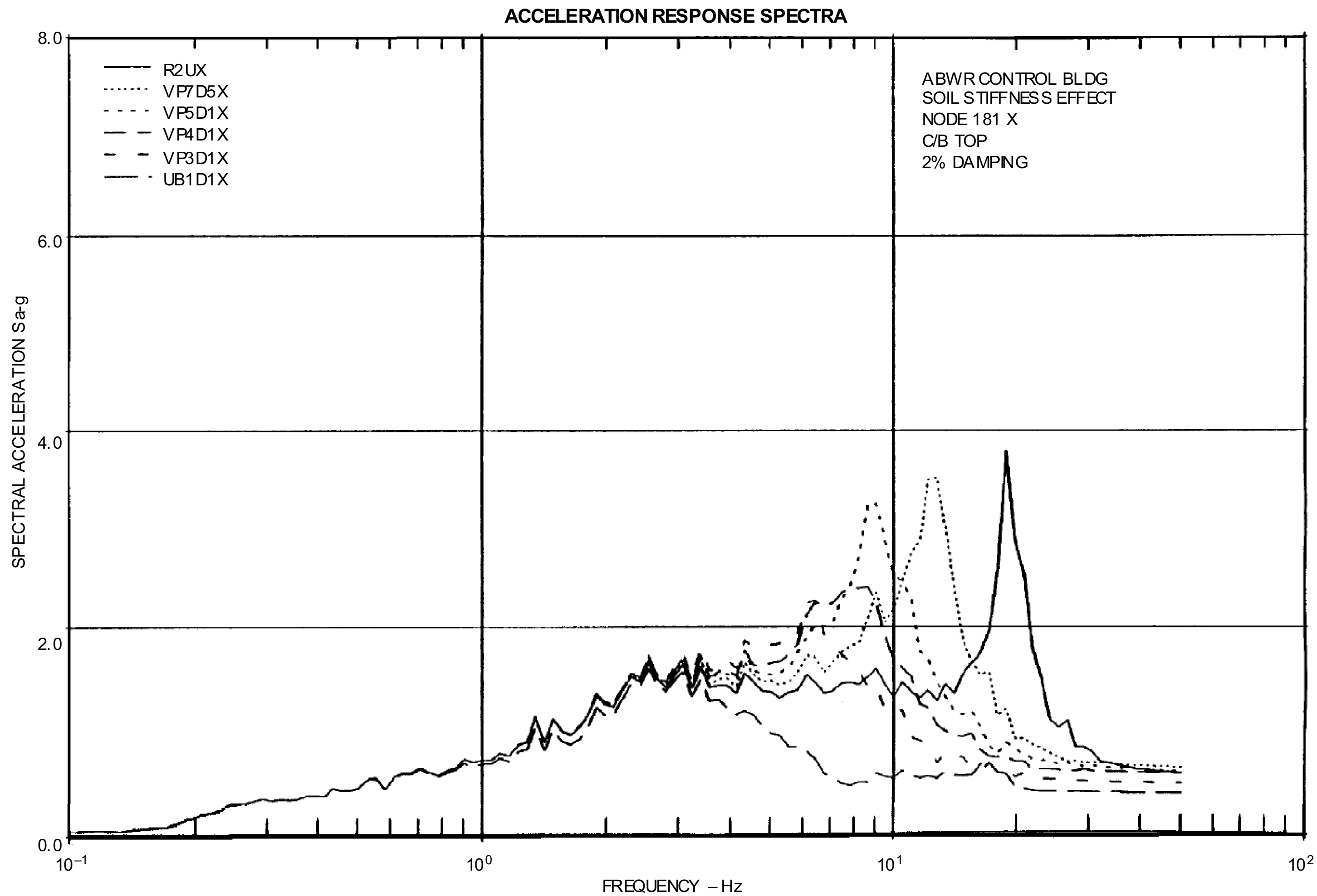


Figure 3A-53 ABWR Control Bldg. Soil Stiffness Effect, Node 181 X C/B Top, 2% Damping

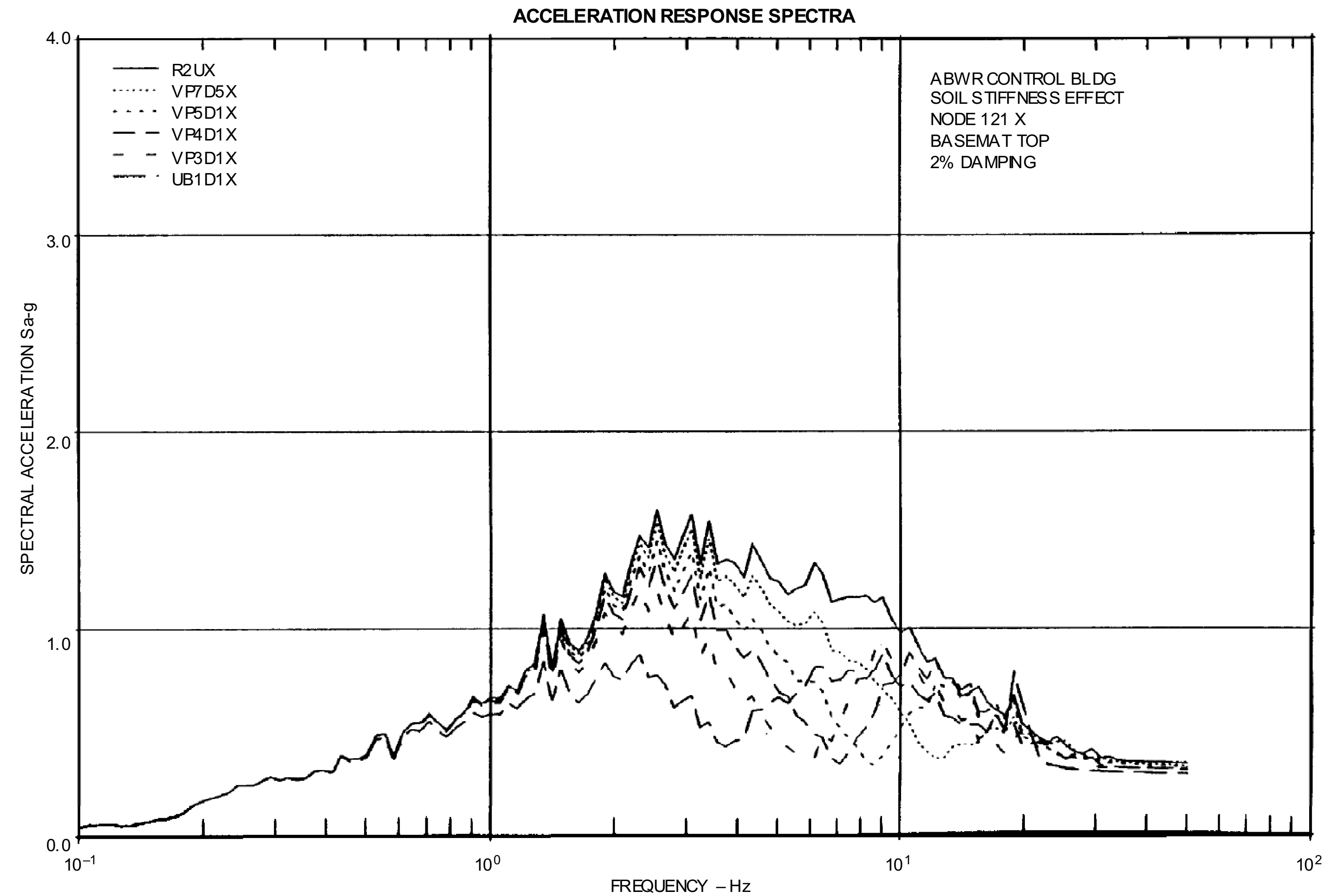


Figure 3A-54 ABWR Control Bldg. Soil Stiffness Effect, Node 121 X Basemat Top, 2% Damping

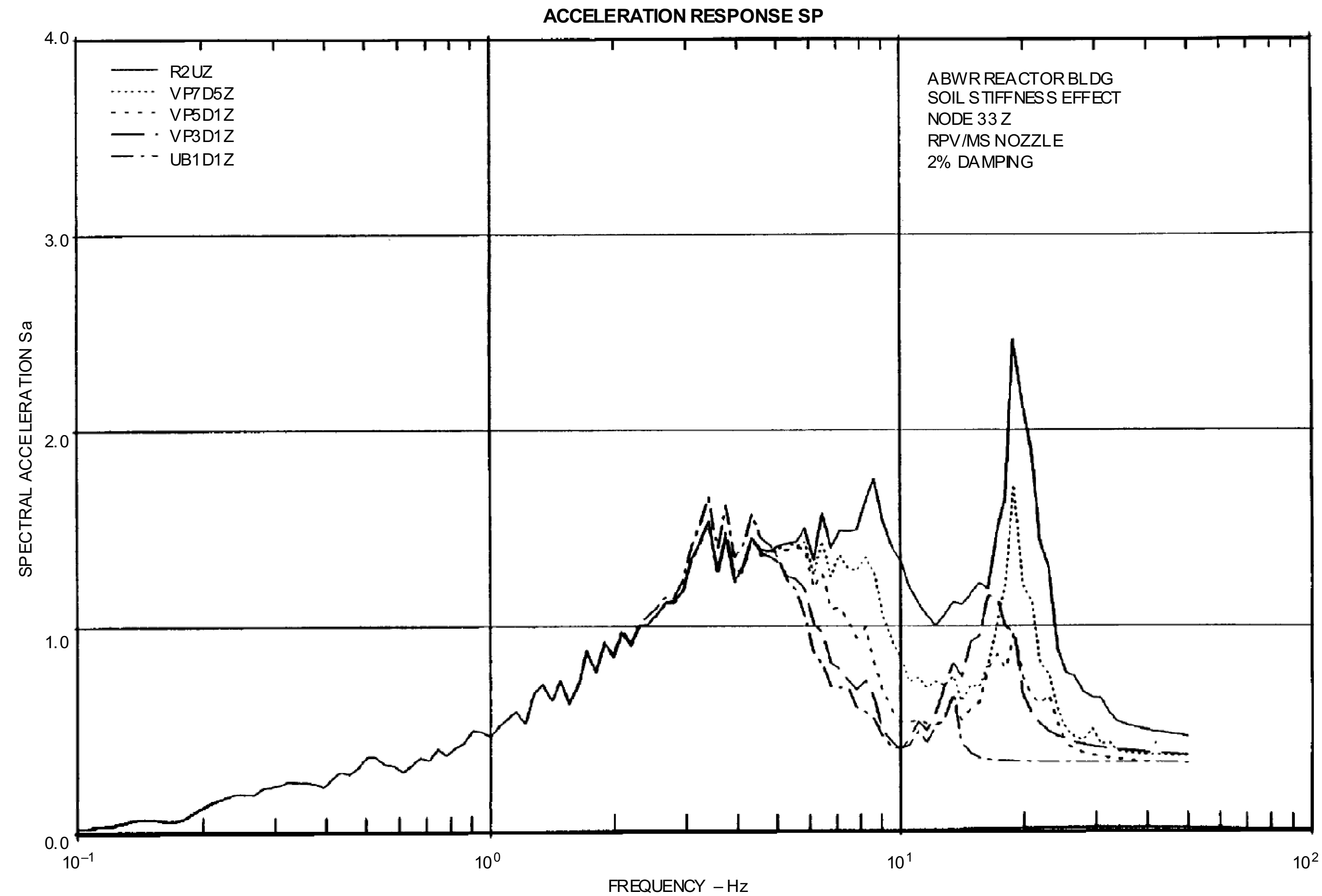


Figure 3A-55 ABWR Reactor Bldg. Soil Stiffness Effect, Node 33 Z RPV/MS Nozzle, 2% Damping

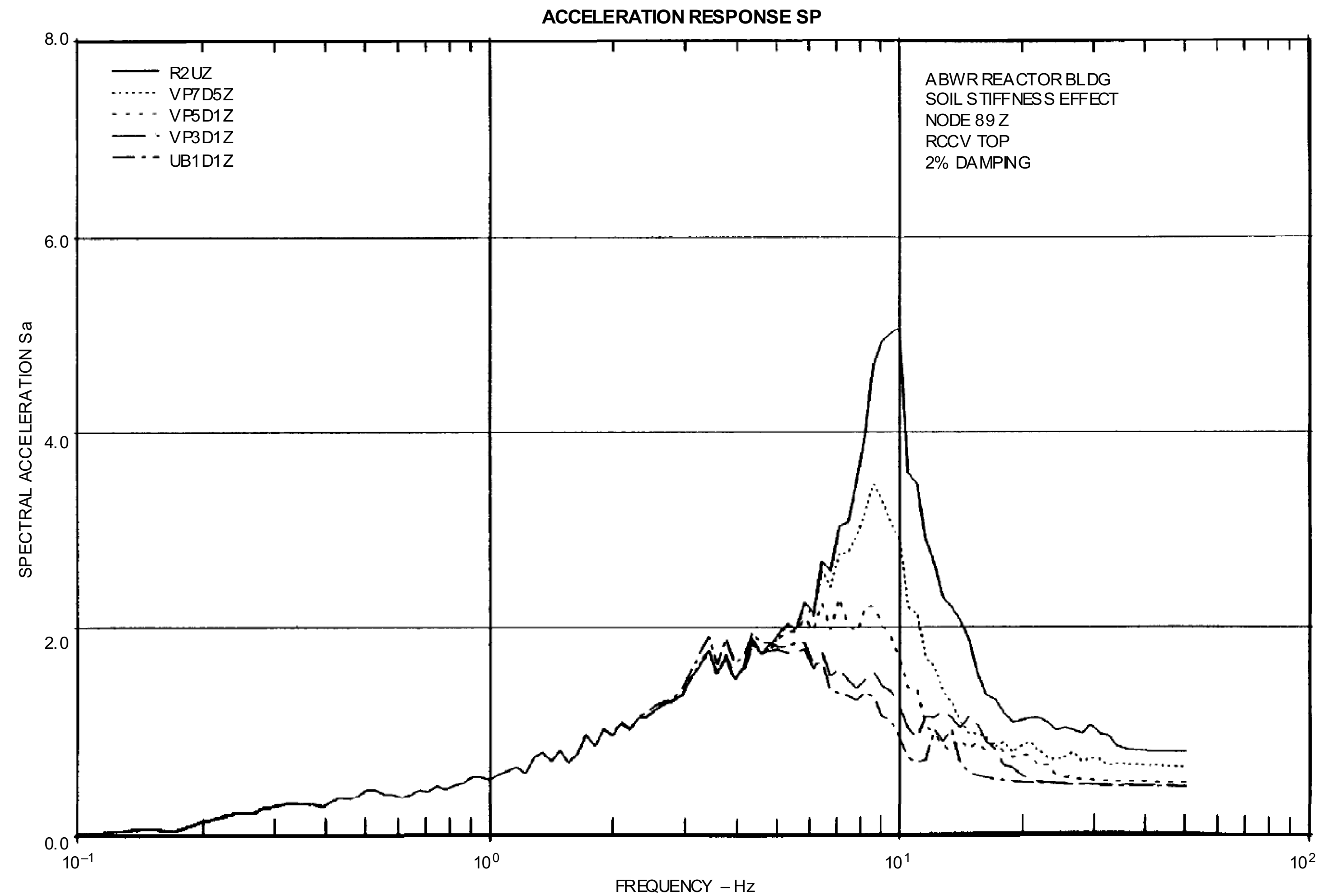


Figure 3A-56 ABWR Reactor Bldg. Soil Stiffness Effect, Node 89 Z RCCV Top, 2% Damping

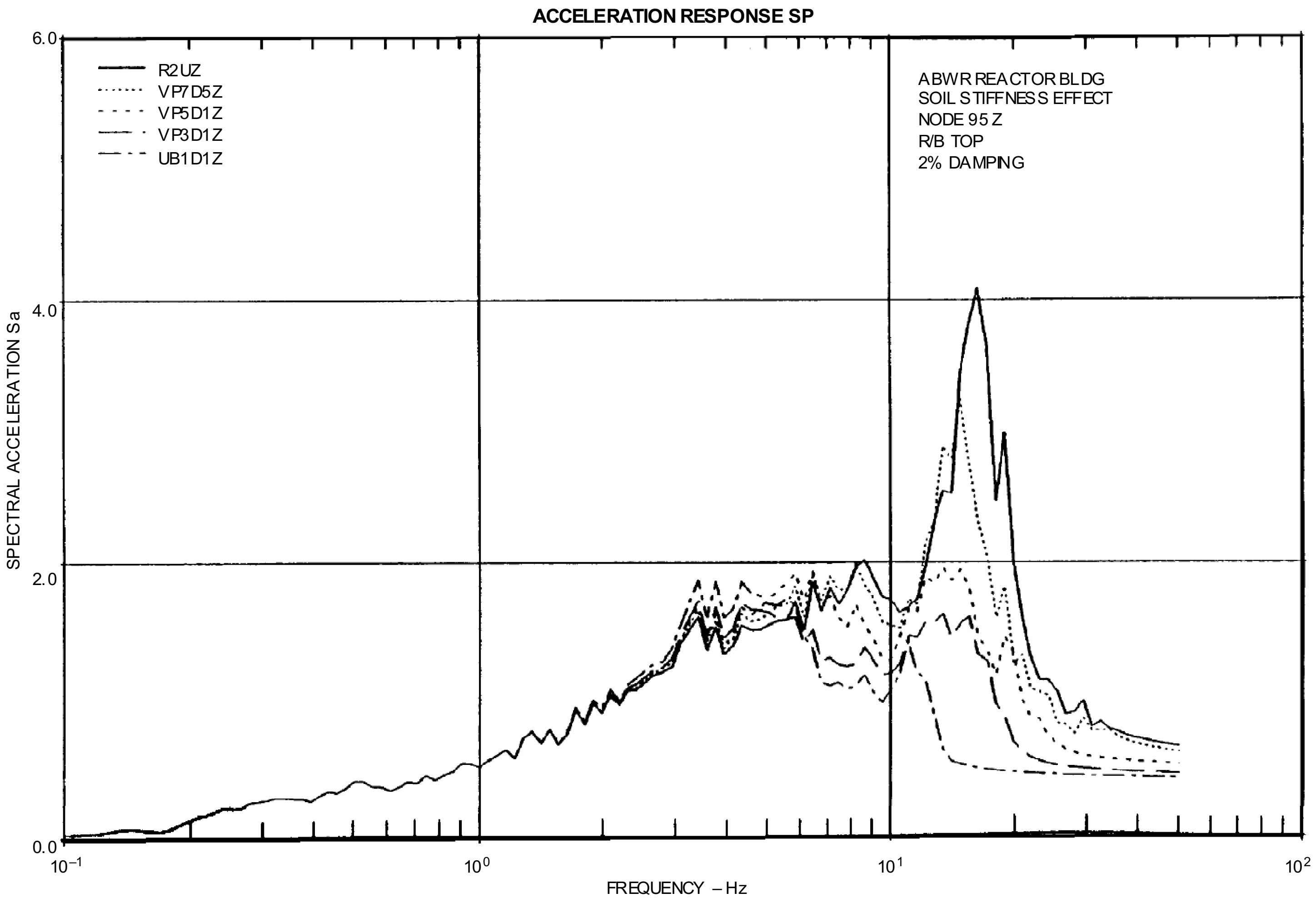


Figure 3A-57 ABWR Reactor Bldg. Soil Stiffness Effect, Node 95 Z R/B Top, 2% Damping

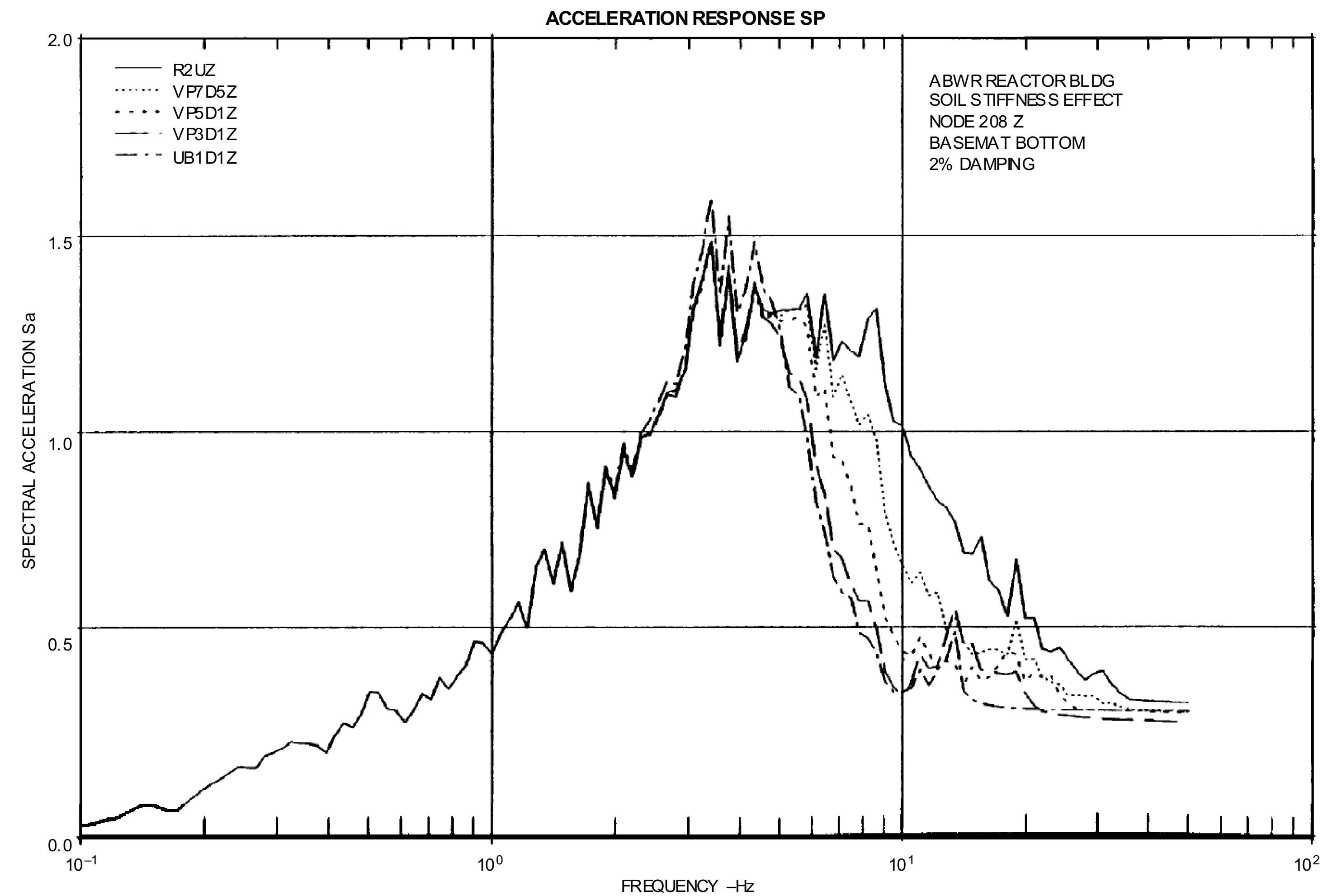


Figure 3A-58 ABWR Reactor Bldg. Soil Stiffness Effect, Node 208 Z Basemat Bottom, 2% Damping

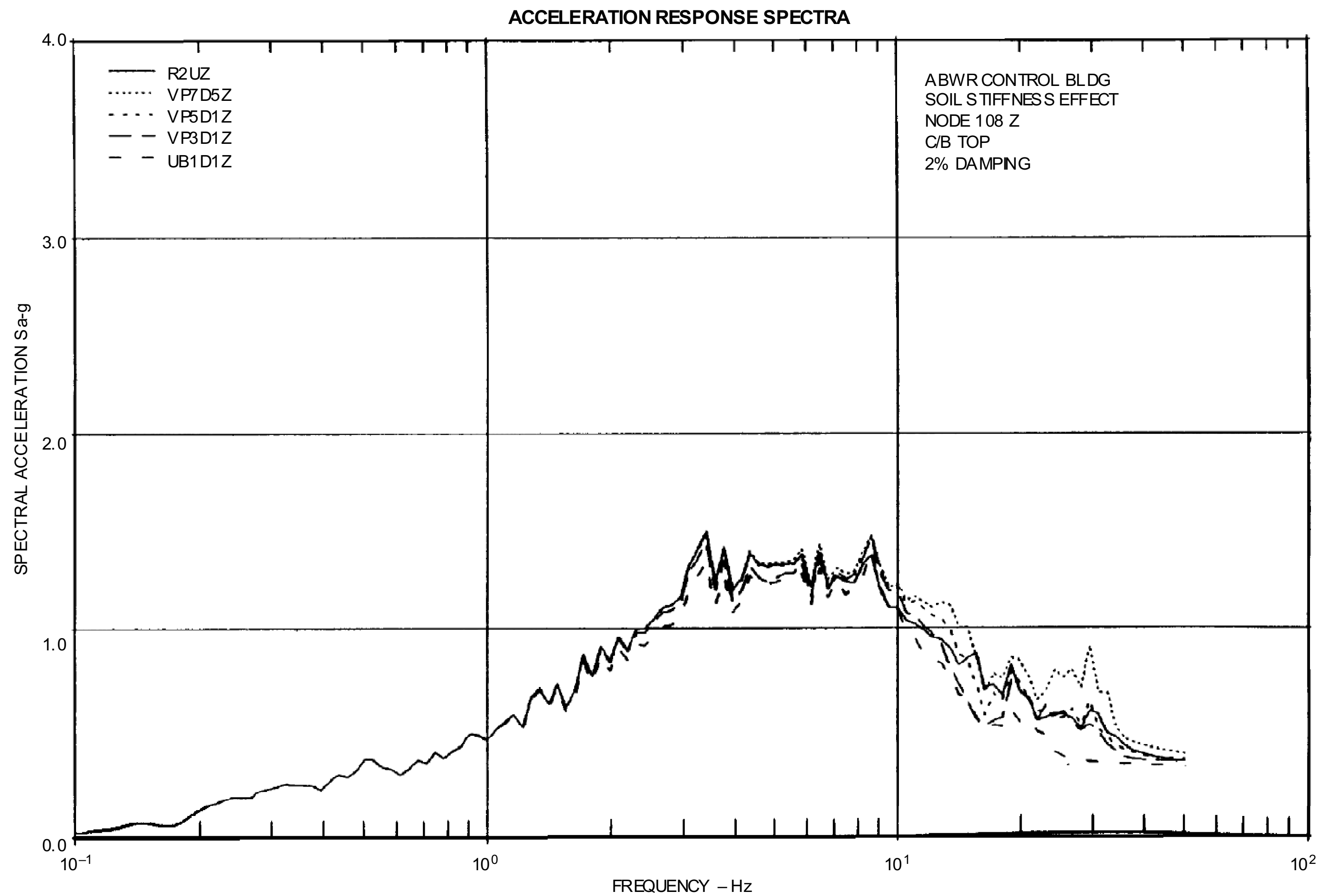


Figure 3A-59 ABWR Control Bldg. Soil Stiffness Effect, Node 108 Z C/B Top, 2% Damping

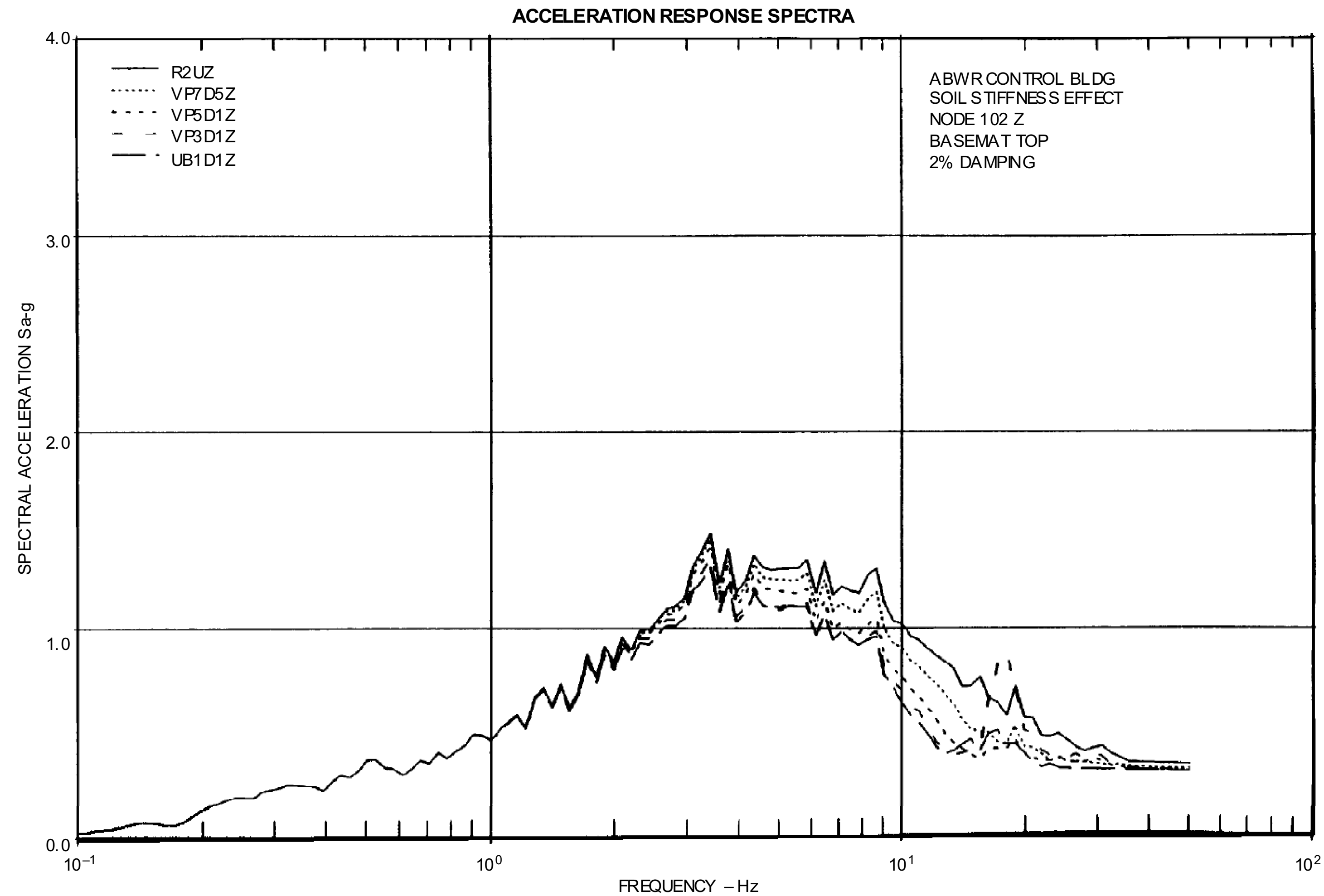


Figure 3A-60 ABWR Control Bldg. Soil Stiffness Effect, Node 102 Z Basemat Top, 2% Damping

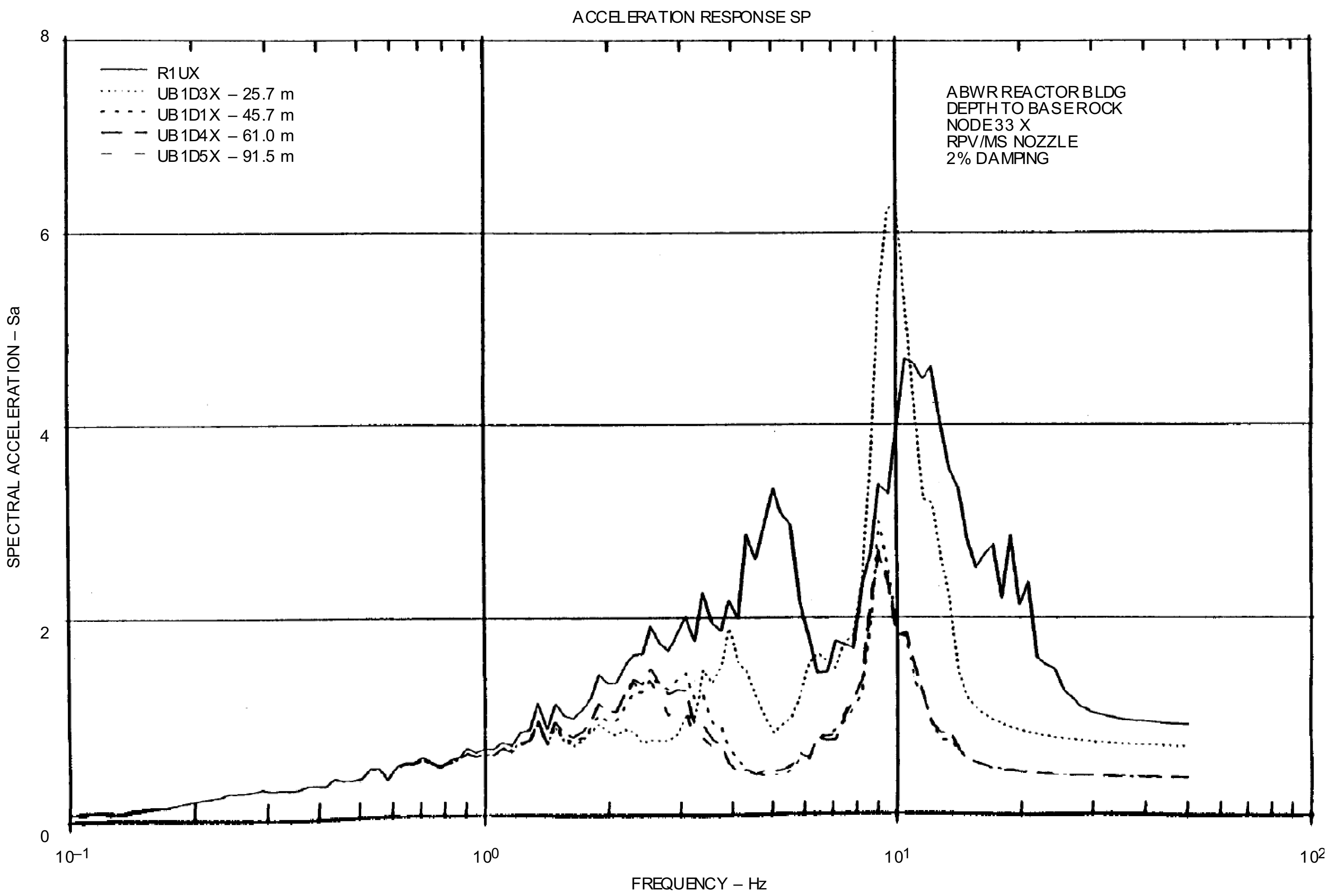


Figure 3A-61 ABWR Reactor Bldg. Depth to Base Rock, Node 33 RPV/MS Nozzle, 2% Damping, UB

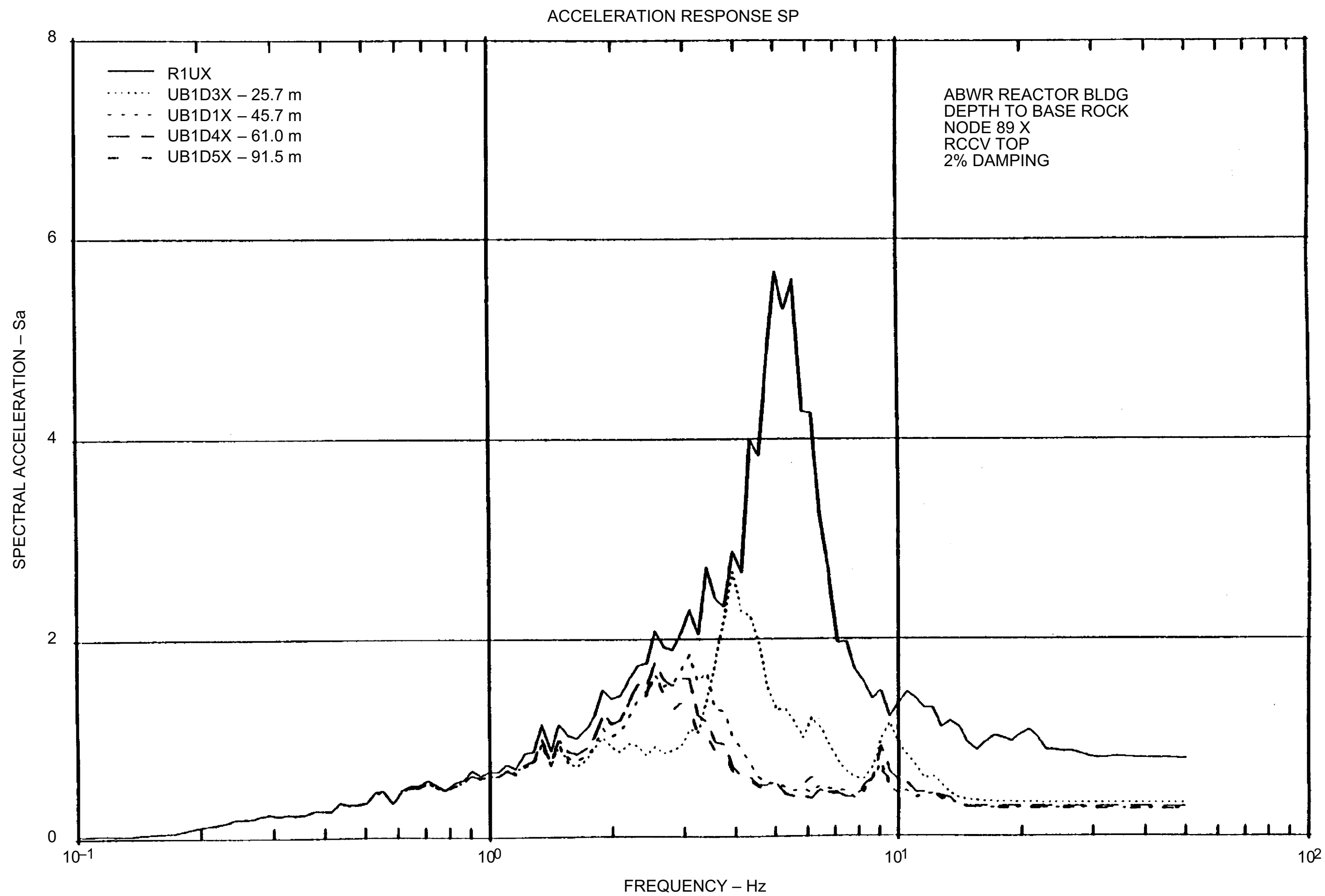


Figure 3A-62 ABWR Reactor Bldg. Depth to Base Rock, Node 89 X RCCV Top, 2% Damping, UB

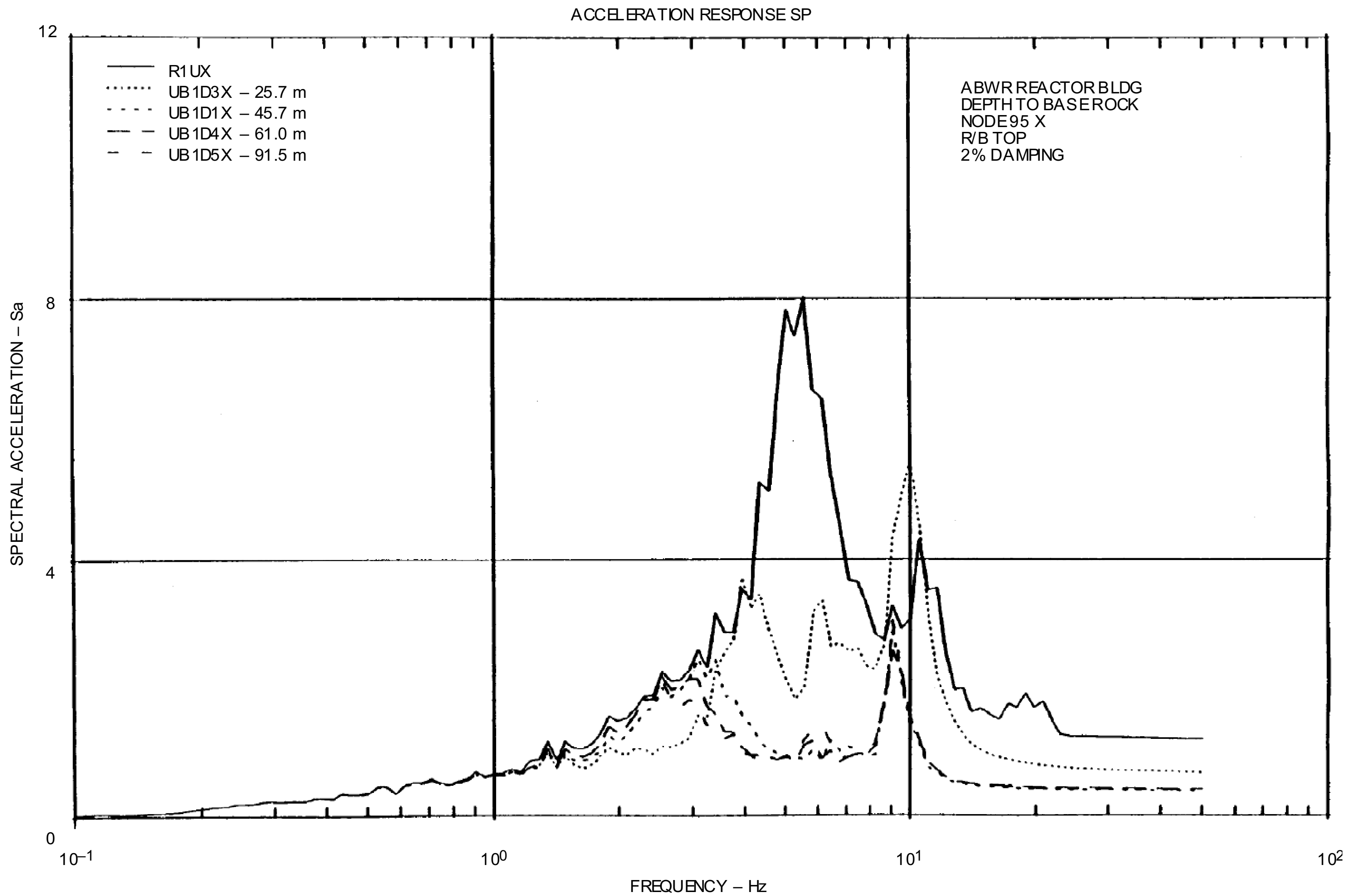


Figure 3A-63 ABWR Reactor Bldg. Depth to Base Rock, Node 95 X R/B Top, 2% Damping, UB

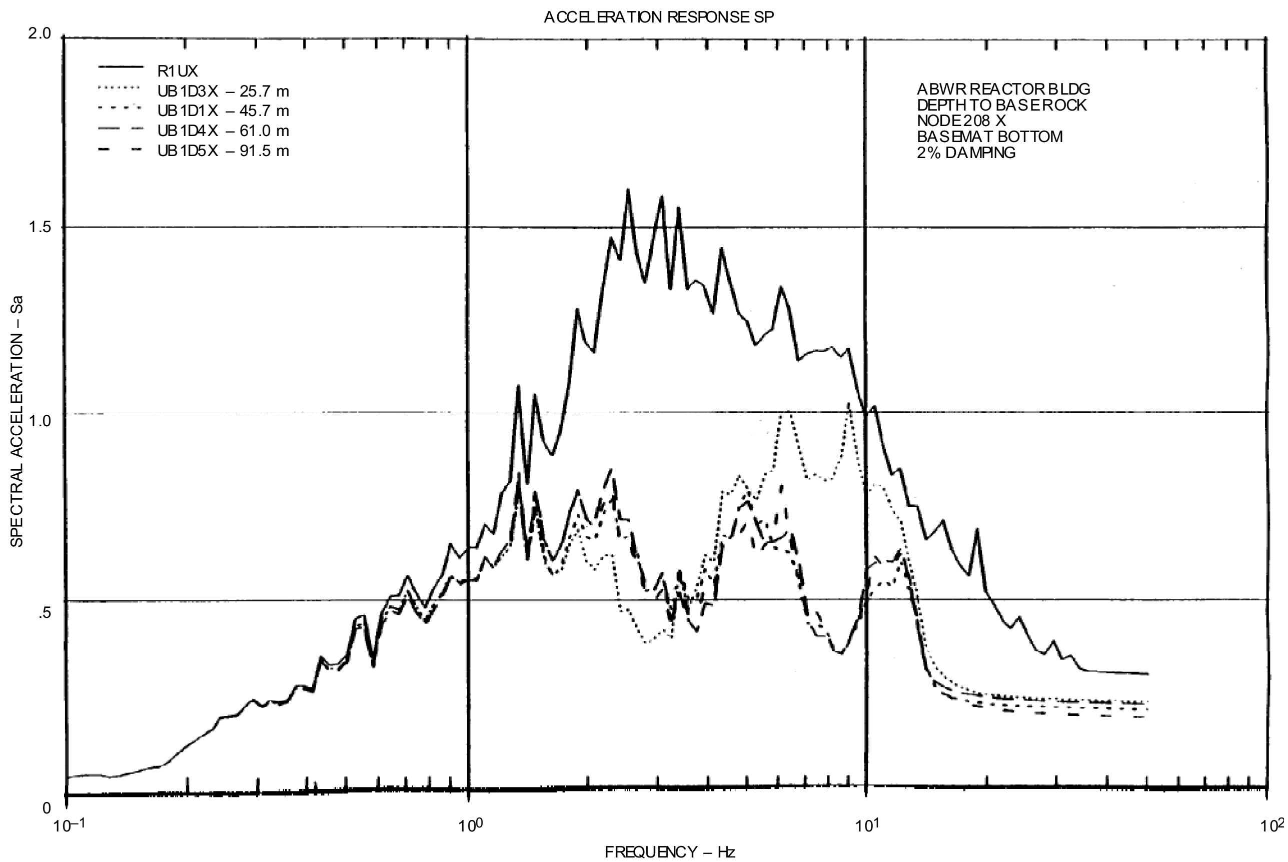


Figure 3A-64 ABWR Reactor Bldg. Depth to Base Rock, Node 208 X Basemat Bottom, 2% Damping, UB

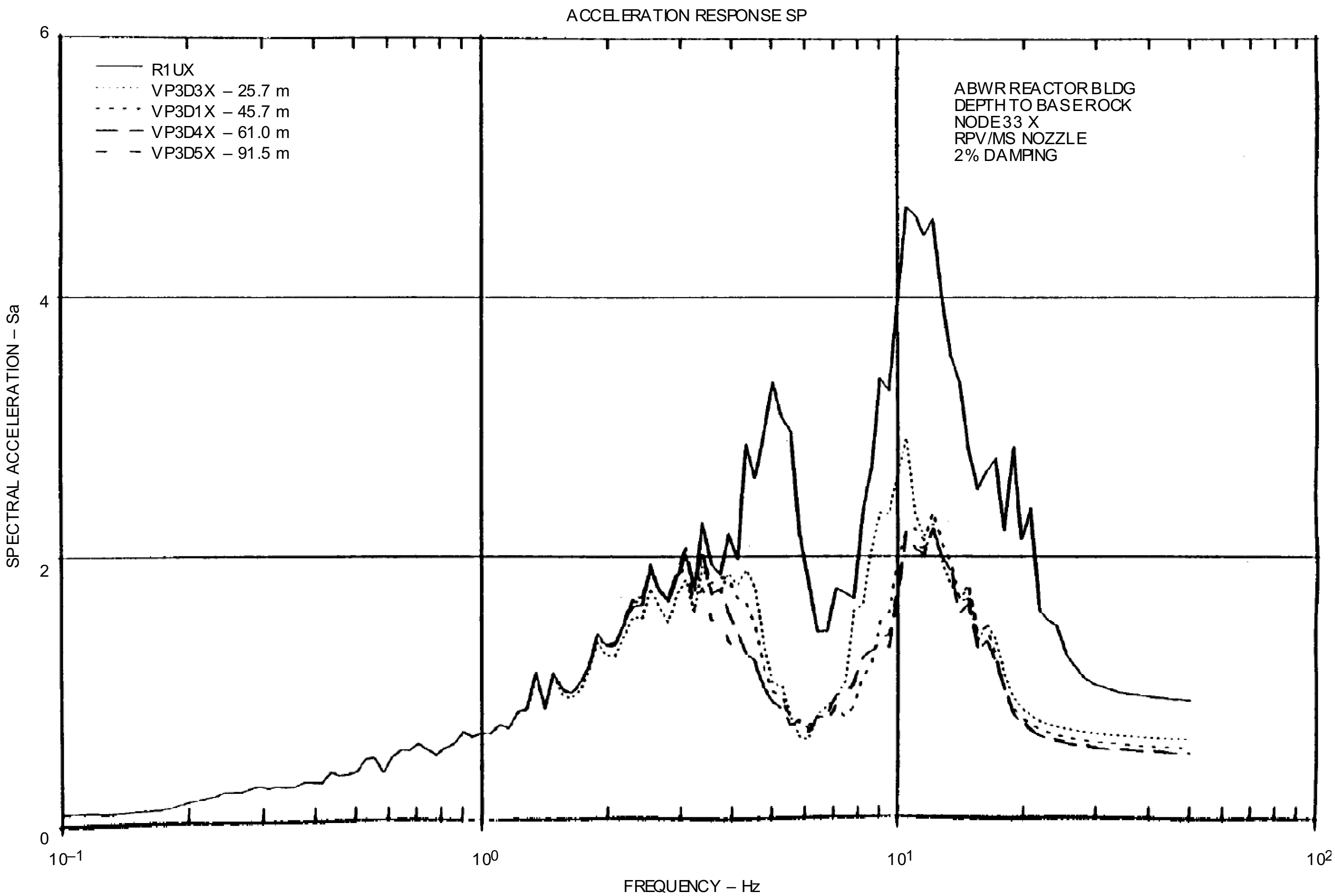


Figure 3A-65 ABWR Reactor Bldg. Depth to Base Rock, Node 33 X RPV/MS Nozzle, 2% Damping, VP3

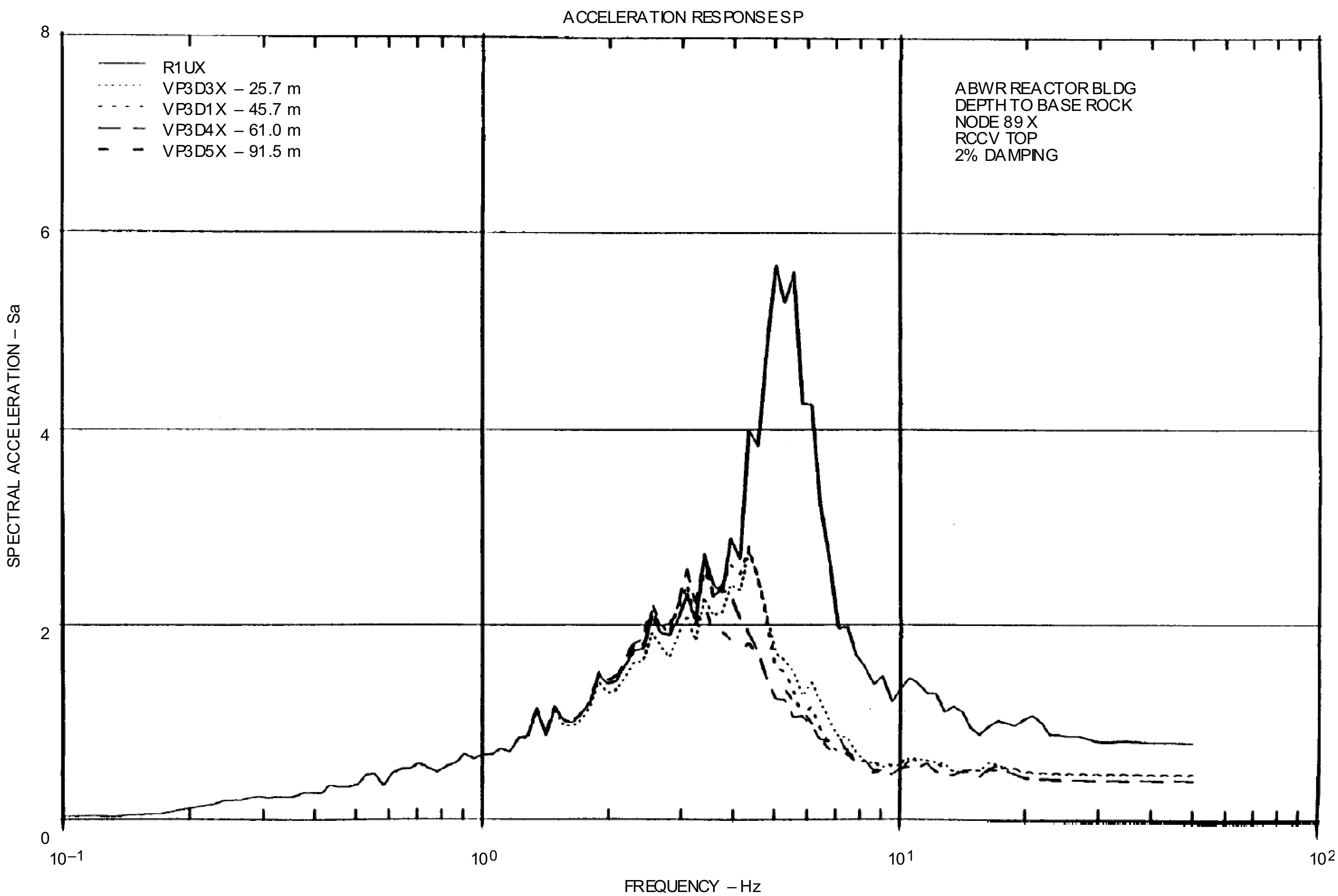


Figure 3A-66 ABWR Reactor Bldg. Depth to Base Rock, Node 89 X RCCV Top, 2% Damping, VP3

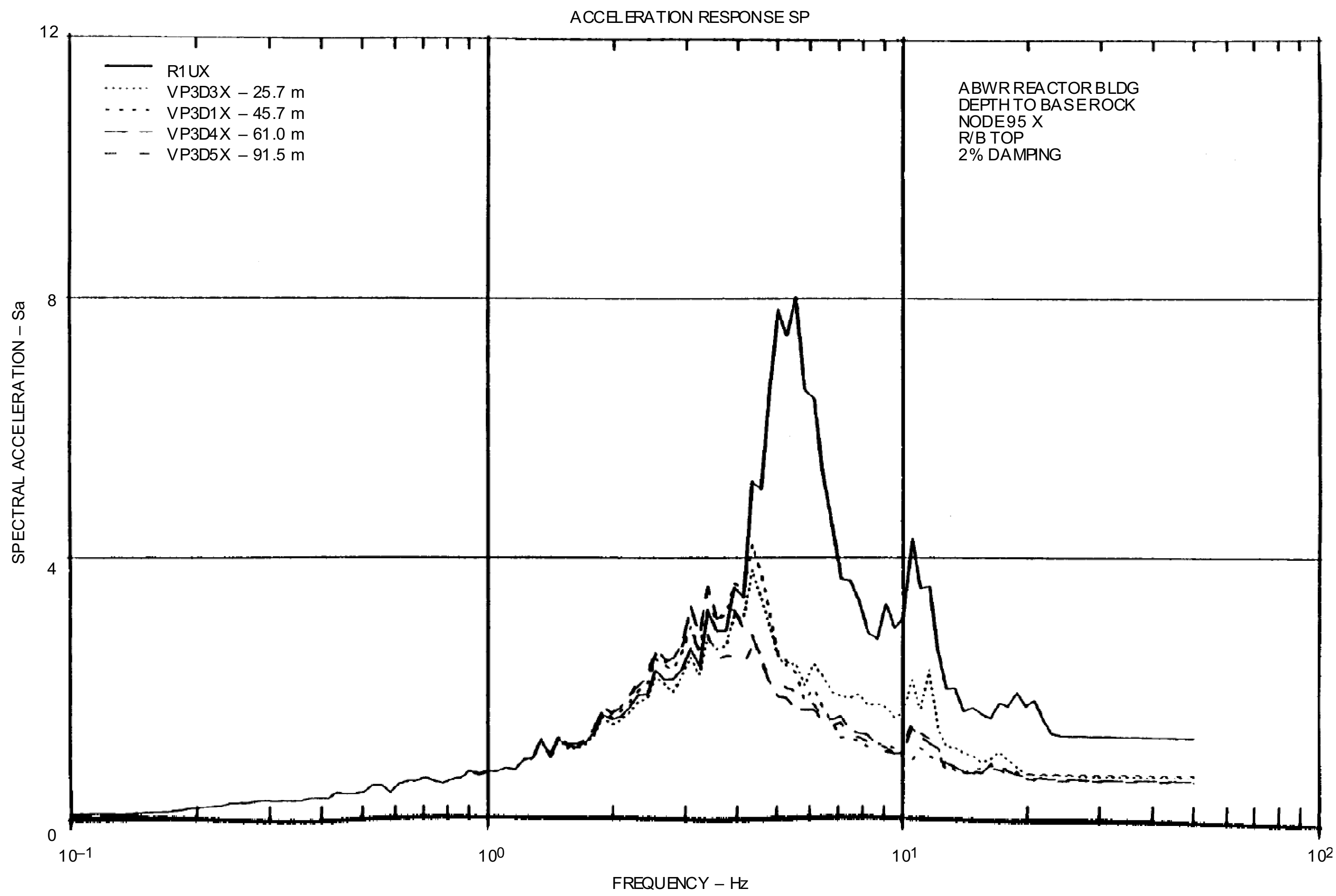


Figure 3A-67 ABWR Reactor Bldg. Depth to Base Rock, Node 95 X R/B Top, 2% Damping, VP3

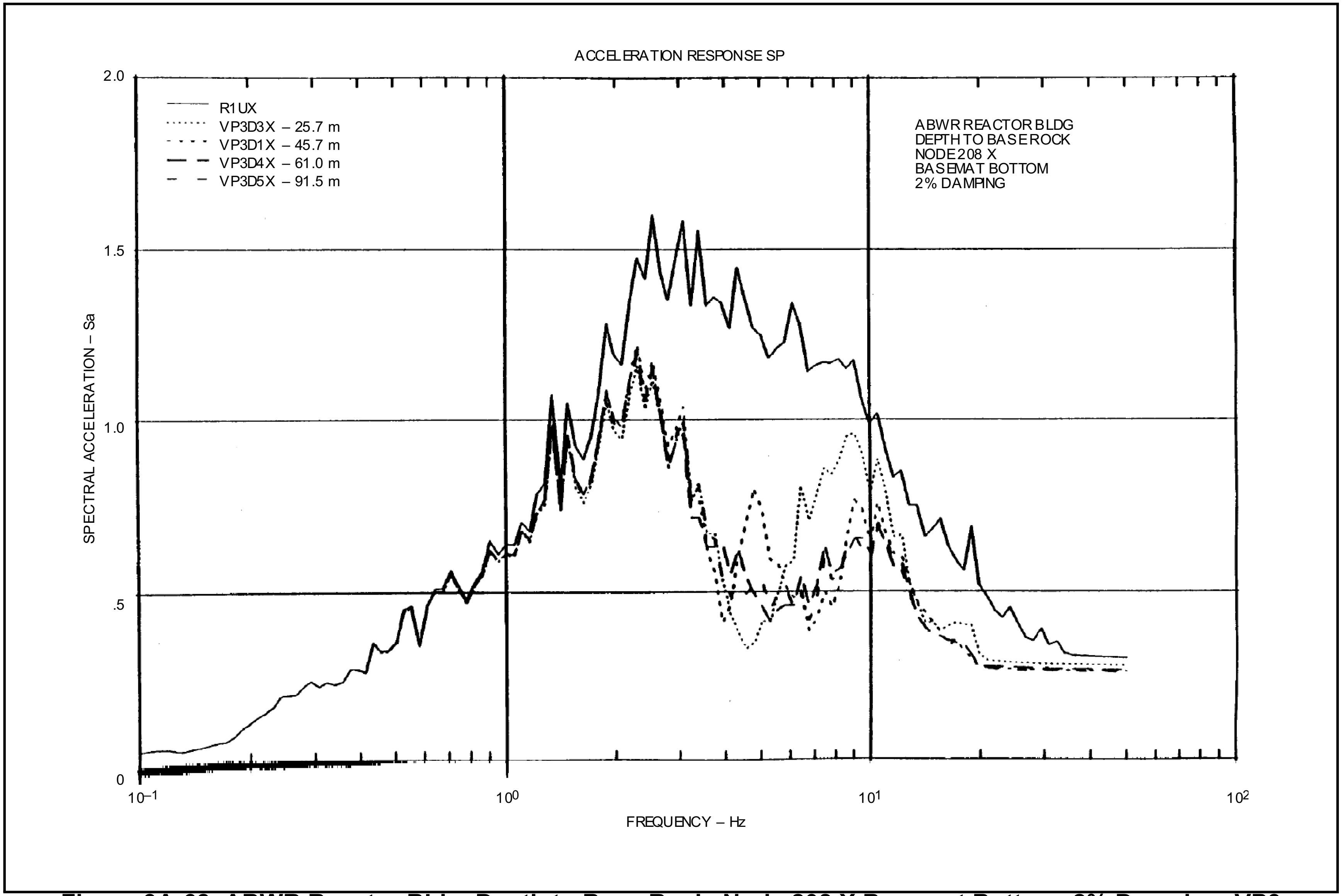


Figure 3A-68 ABWR Reactor Bldg. Depth to Base Rock, Node 208 X Basemat Bottom, 2% Damping, VP3

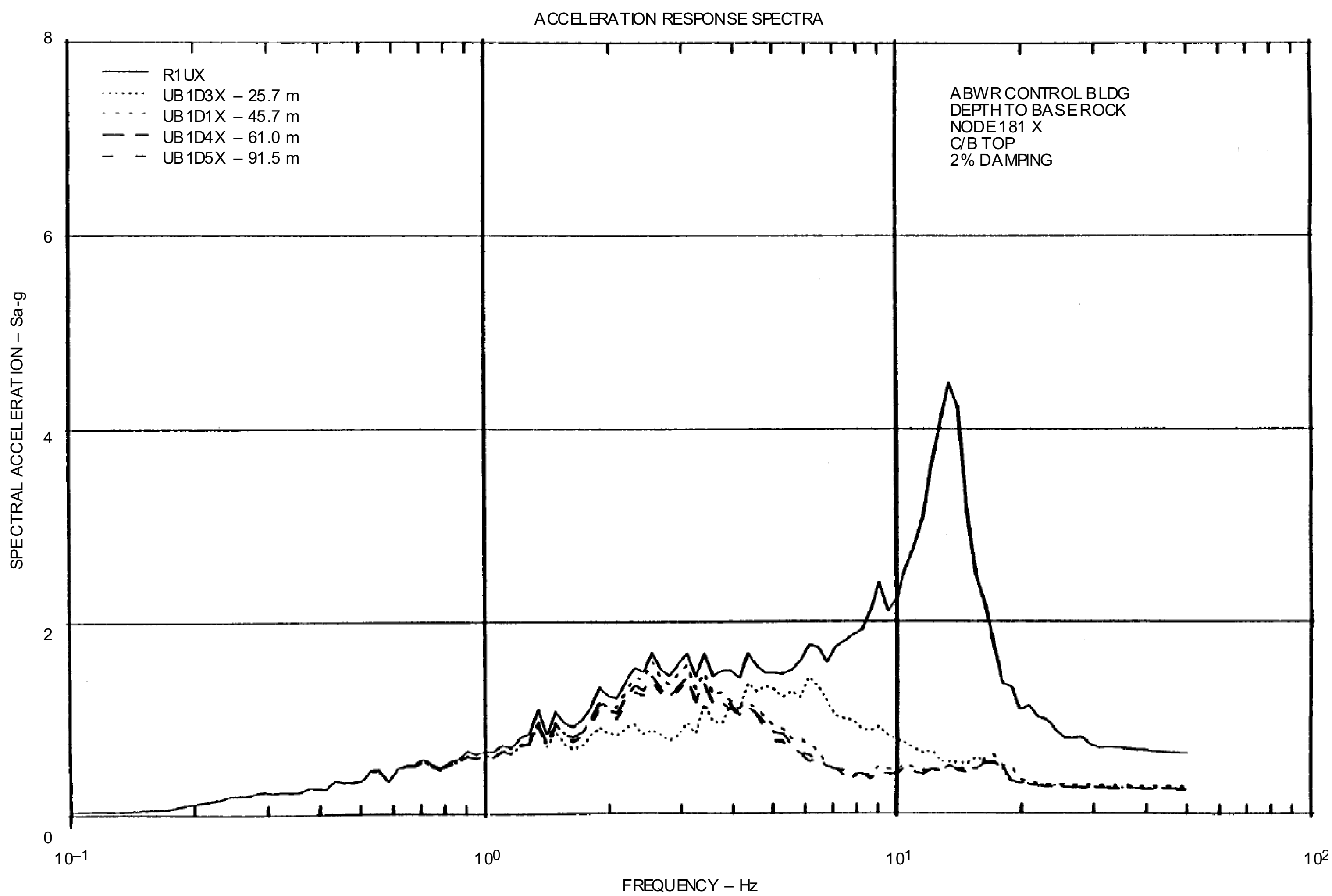


Figure 3A-69 ABWR Control Bldg. Depth to Base Rock, Node 181 X C/B Top, 2% Damping, UB

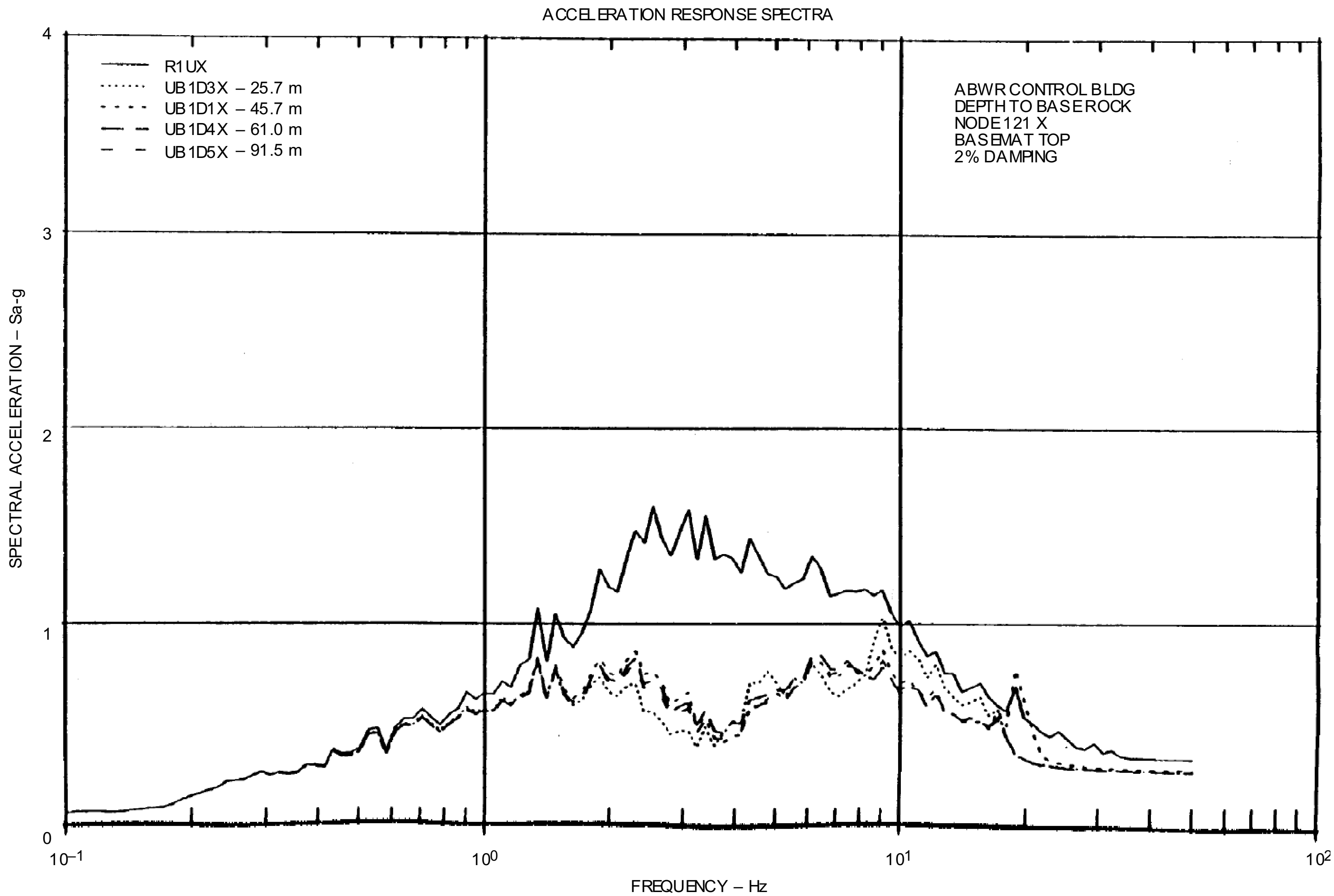


Figure 3A-70 ABWR Control Bldg. Depth to Base Rock, Node 121 X Basemat Top, 2% Damping, UB

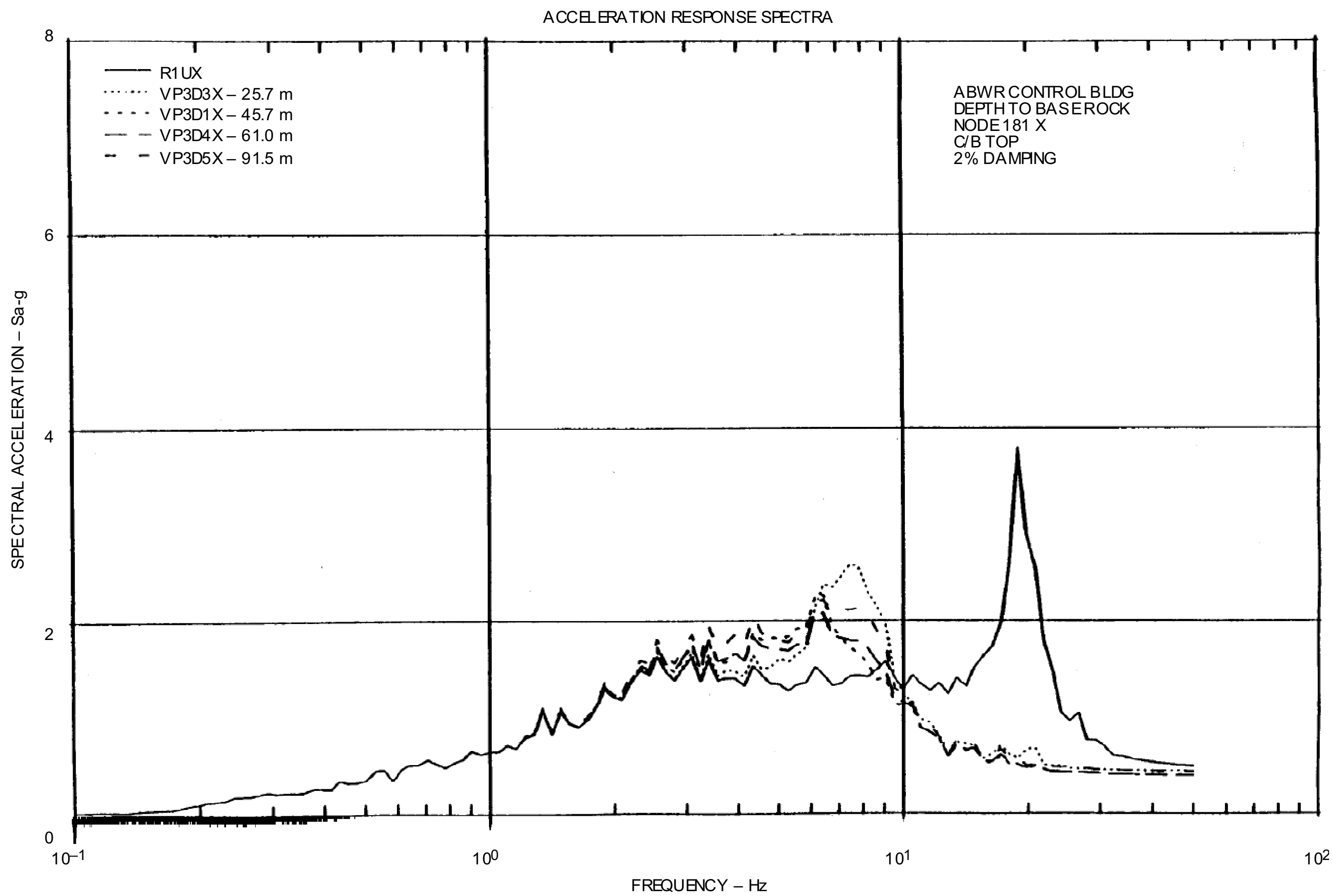


Figure 3A-71 ABWR Control Bldg. Depth to Base Rock, Node 181 X C/B Top, 2% Damping, VP3

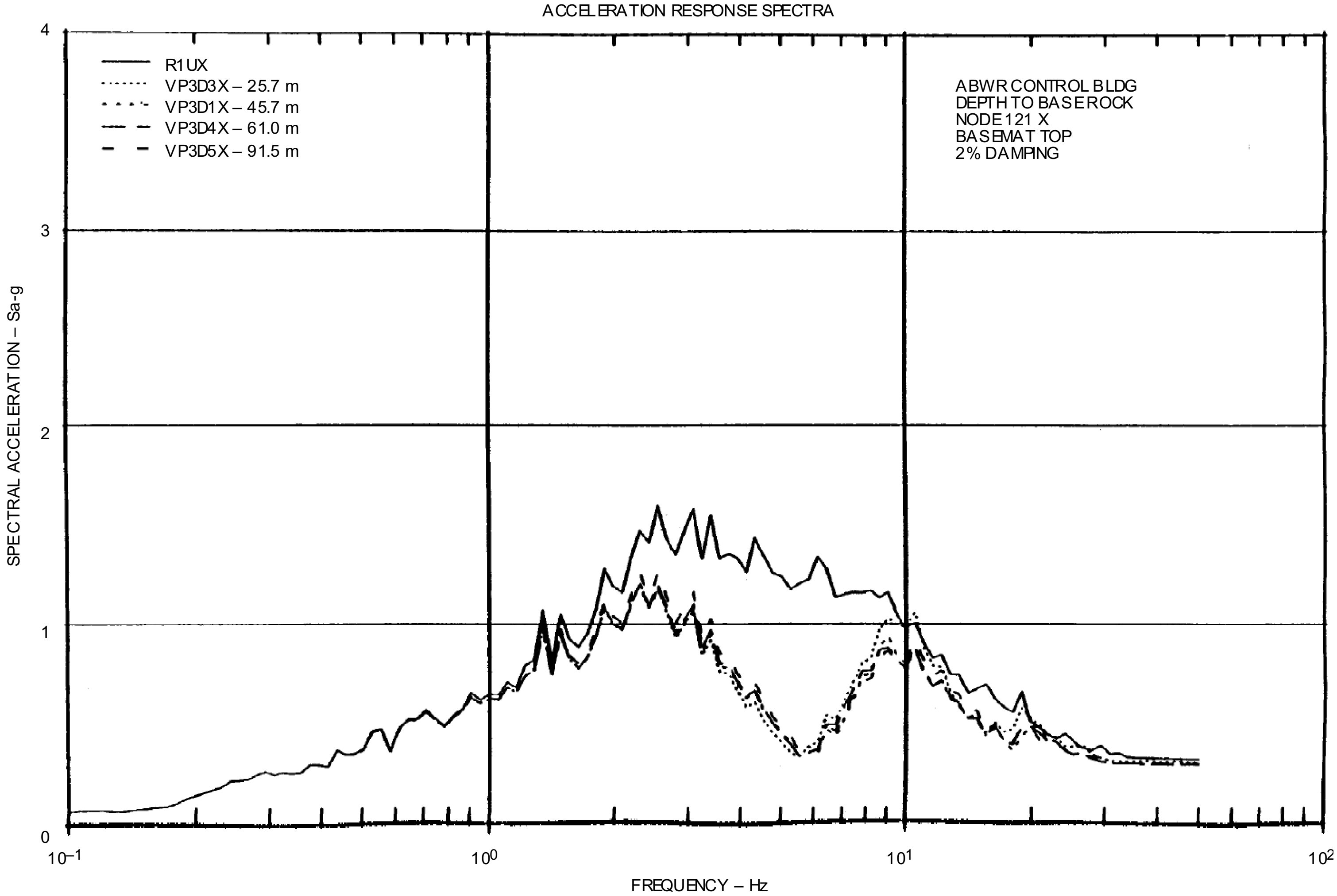


Figure 3A-72 ABWR Control Bldg. Depth to Base Rock, Node 121 X Basemat Top, 2% Damping, VP3

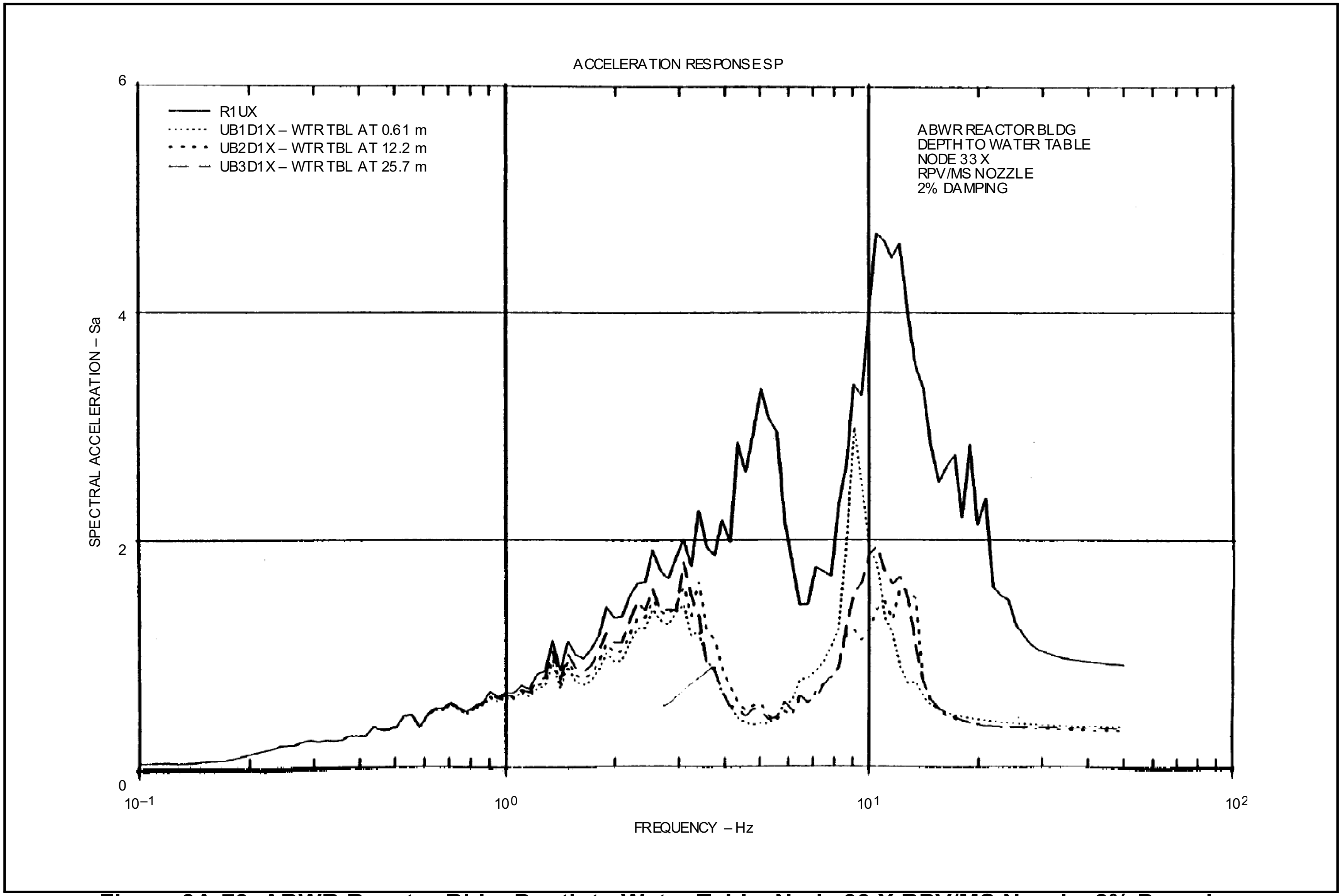


Figure 3A-73 ABWR Reactor Bldg. Depth to Water Table, Node 33 X RPV/MS Nozzle, 2% Damping

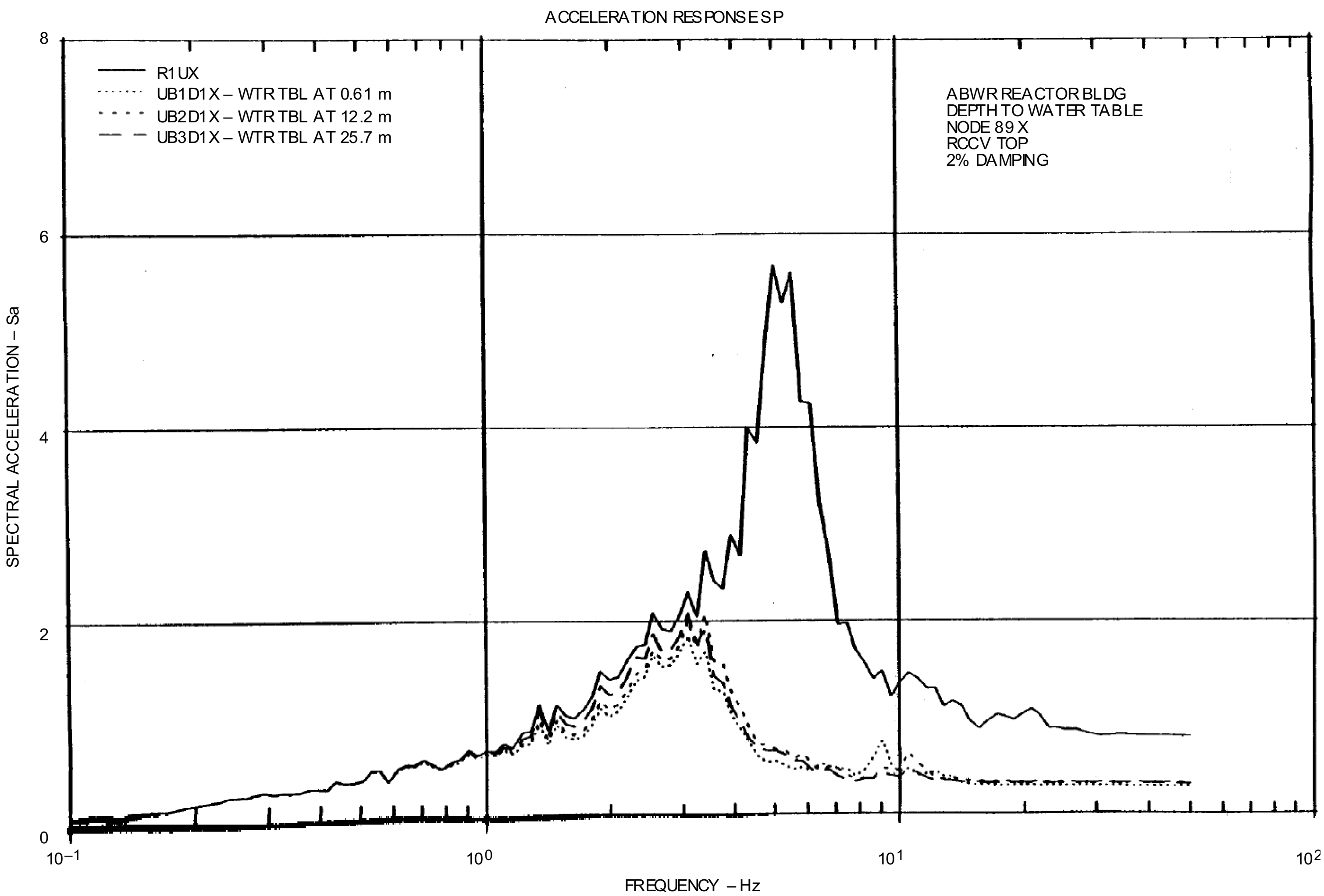


Figure 3A-74 ABWR Reactor Bldg. Depth to Water Table, Node 89 X RCCV Top, 2% Damping

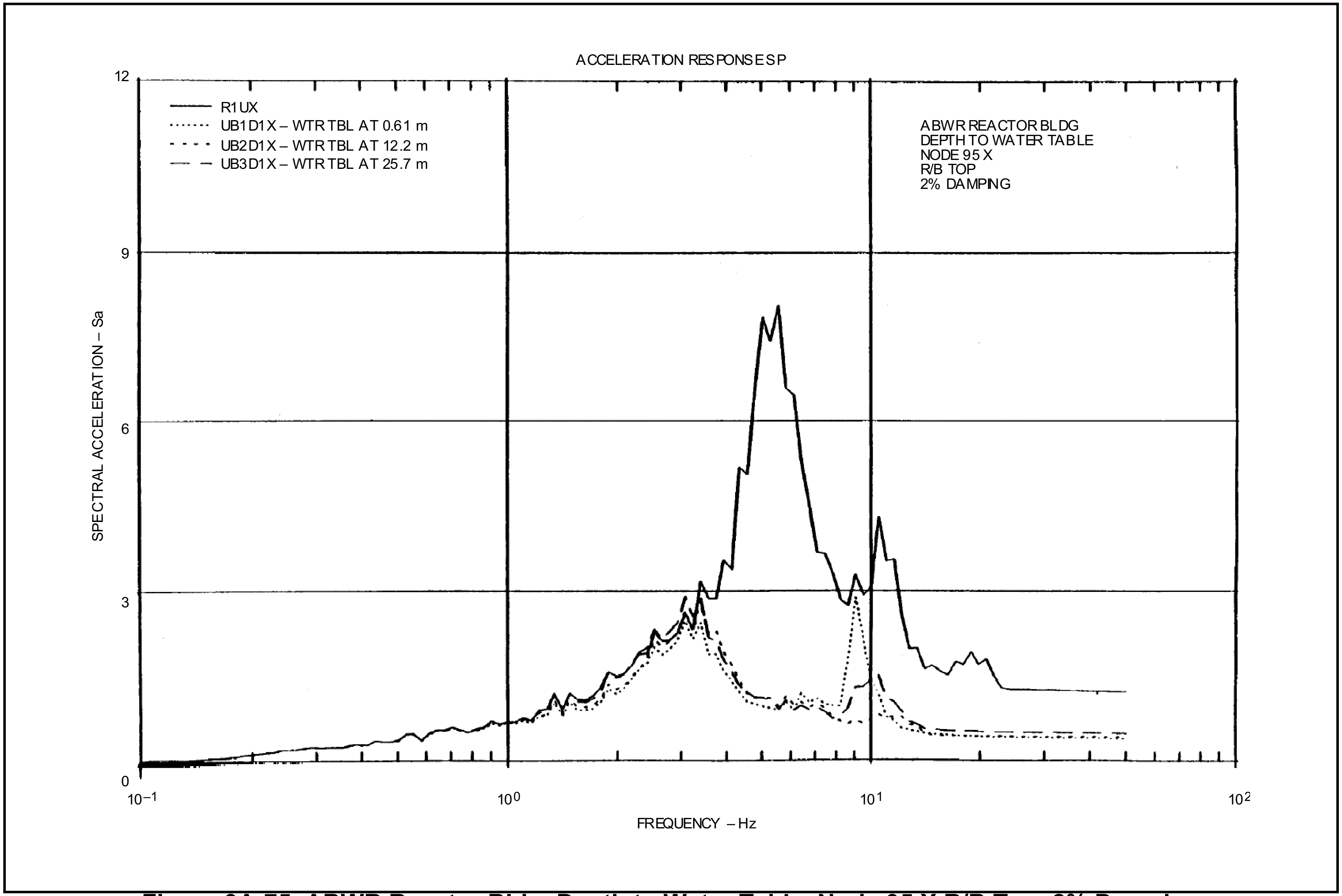


Figure 3A-75 ABWR Reactor Bldg. Depth to Water Table, Node 95 X R/B Top, 2% Damping

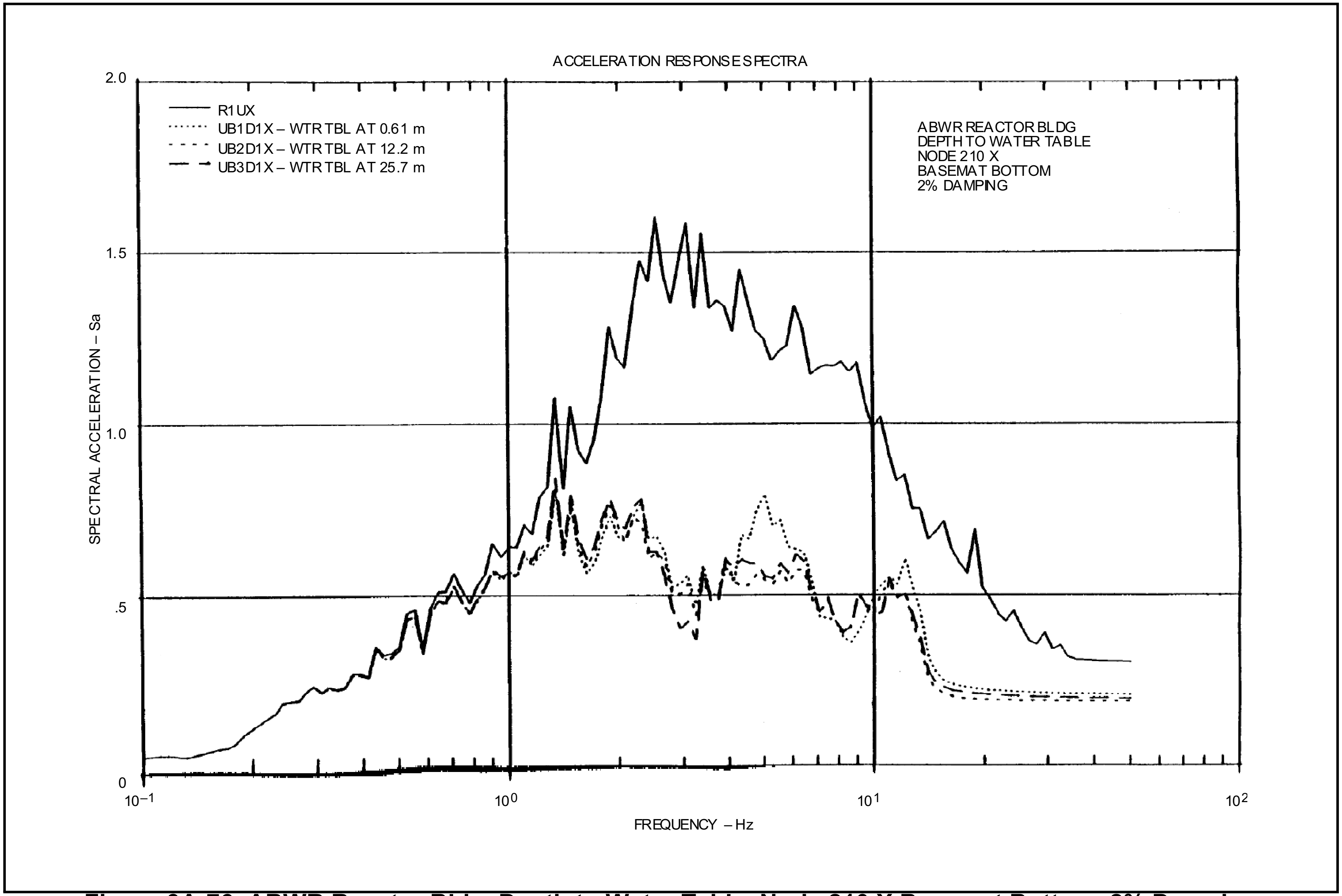


Figure 3A-76 ABWR Reactor Bldg. Depth to Water Table, Node 210 X Basemat Bottom, 2% Damping

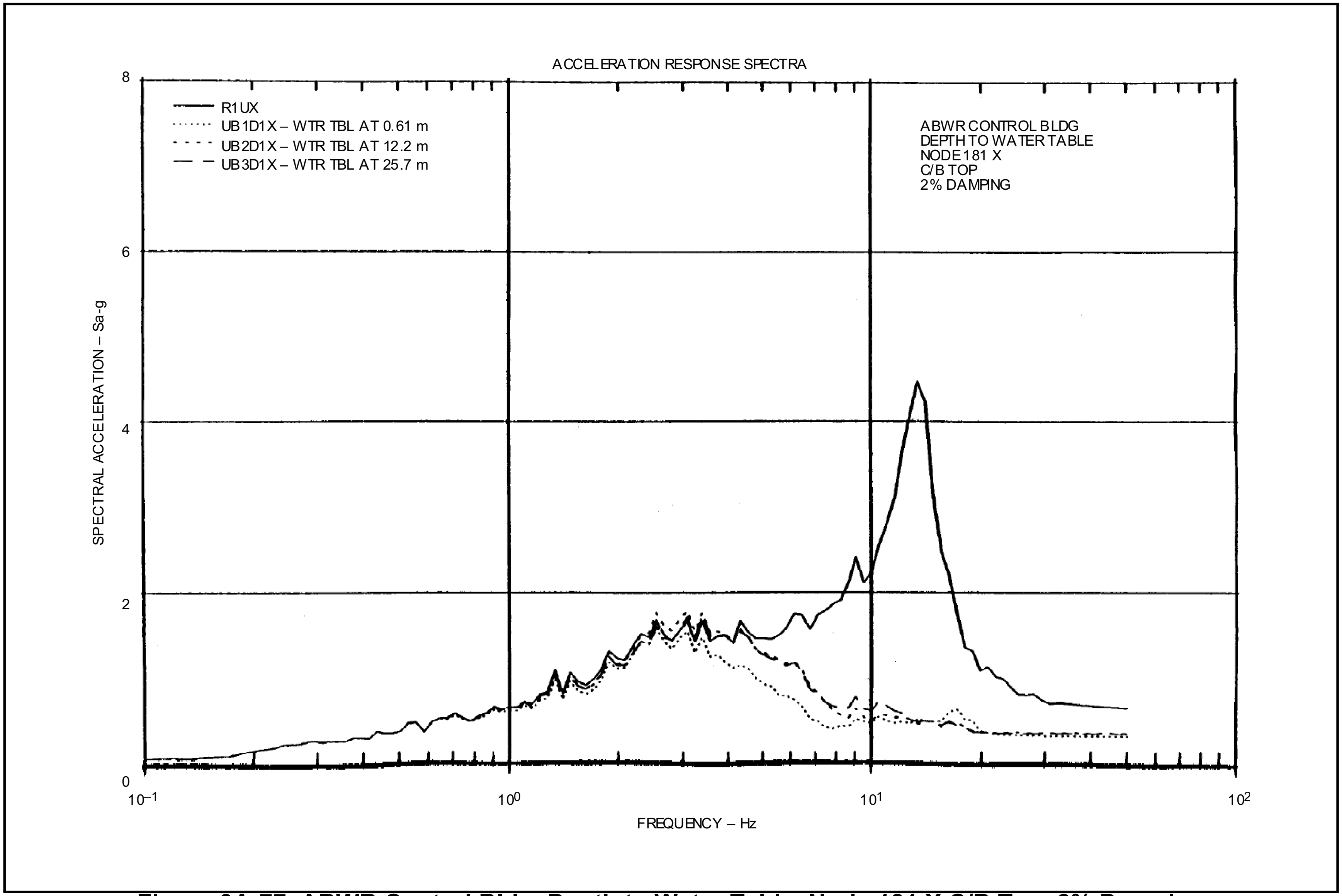


Figure 3A-77 ABWR Control Bldg. Depth to Water Table, Node 181 X C/B Top, 2% Damping

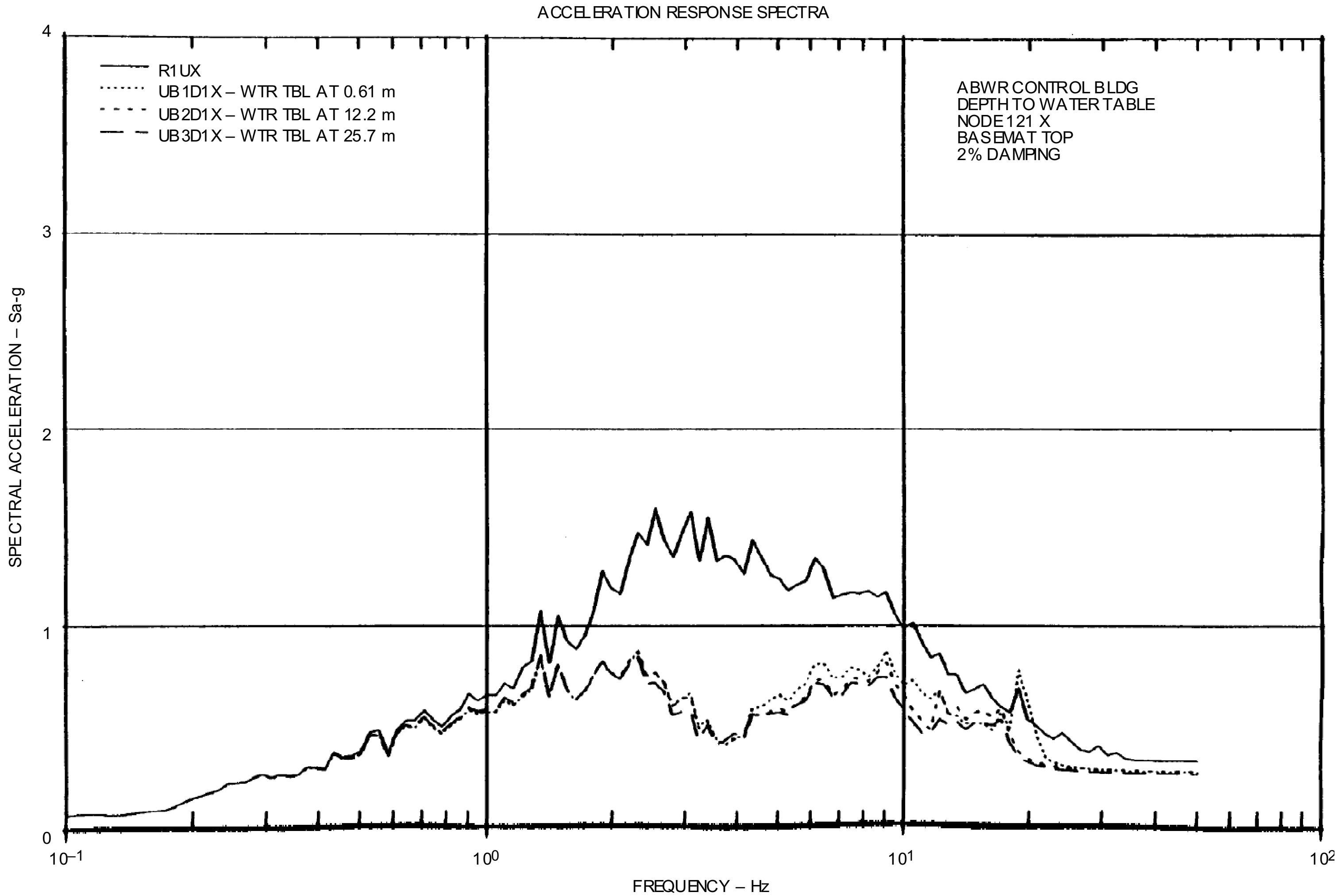


Figure 3A-78 ABWR Control Bldg. Depth to Water Table, Node 121 X Basemat Top, 2% Damping

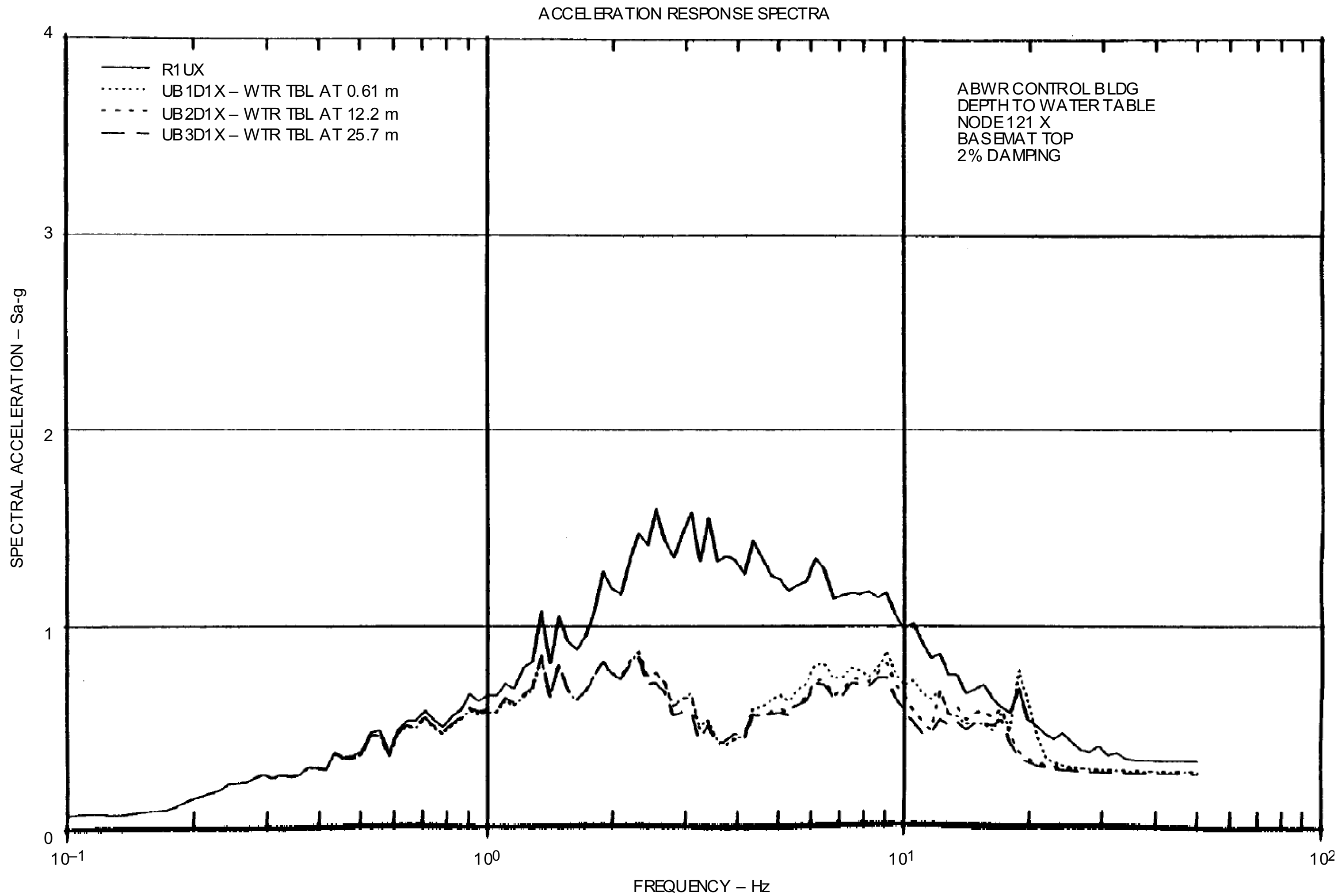


Figure 3A-79 ABWR Reactor Bldg. Depth to Water Table, Node 33 Z RPV/MS Nozzle, 2% Damping

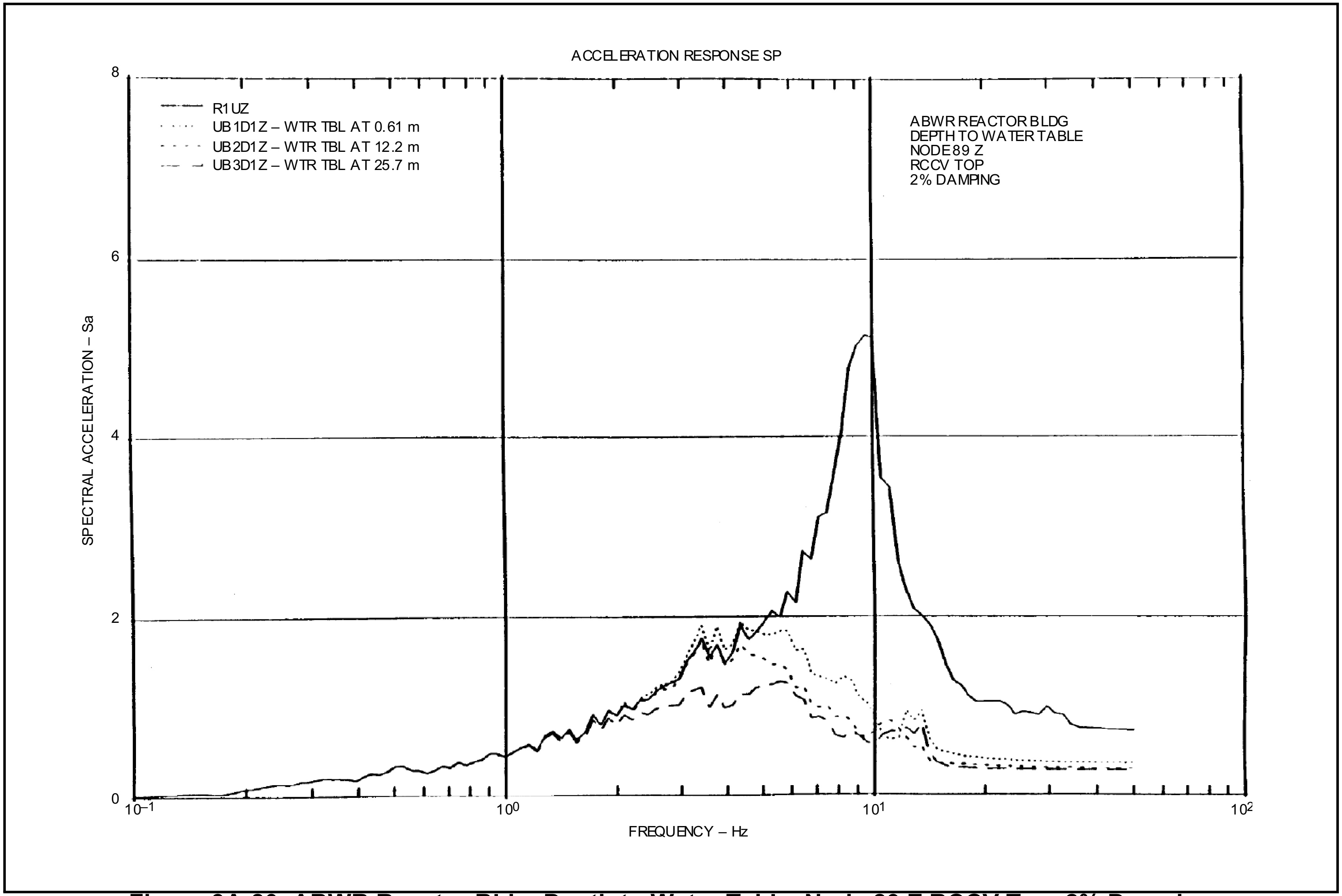


Figure 3A-80 ABWR Reactor Bldg. Depth to Water Table, Node 89 Z RCCV Top, 2% Damping

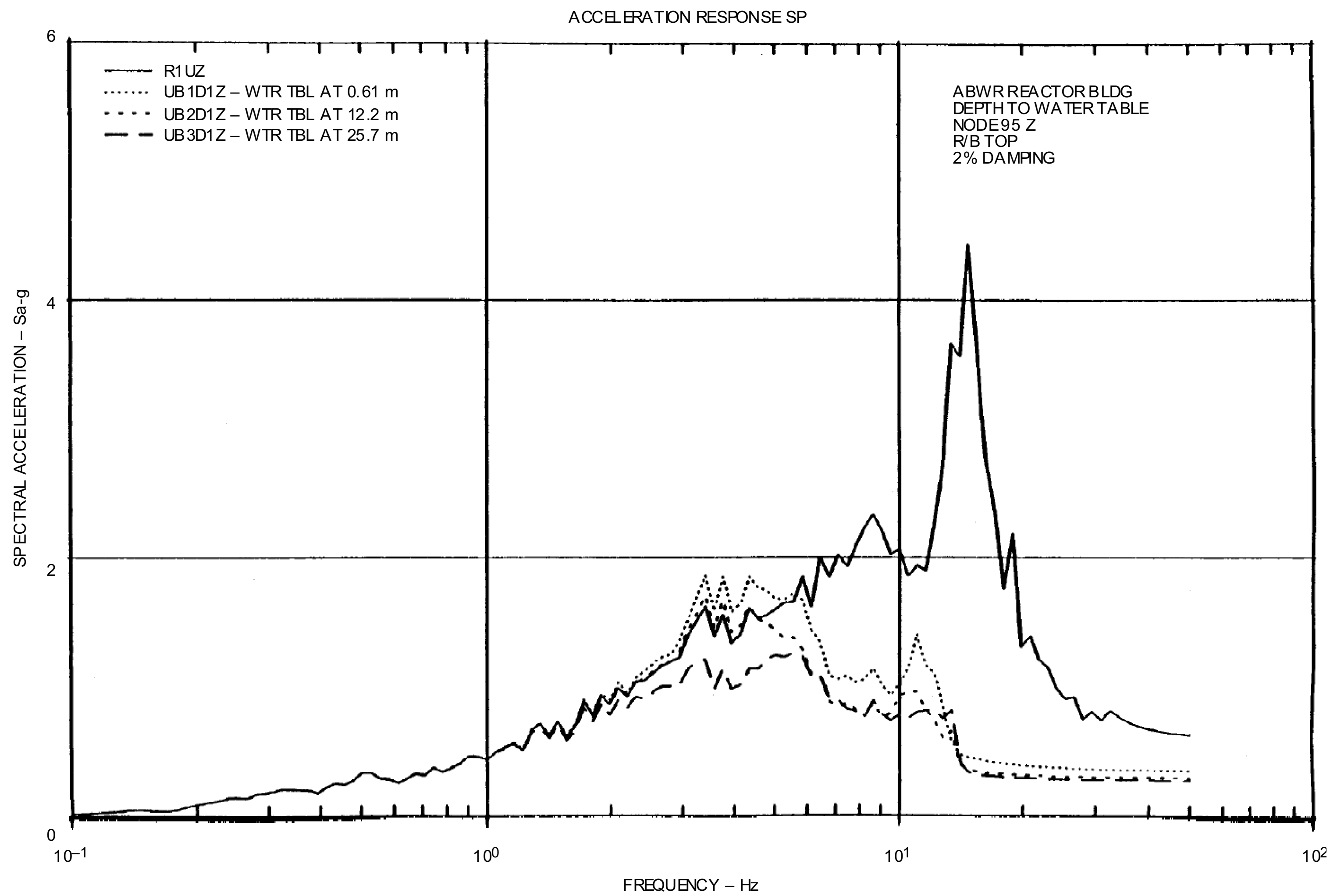


Figure 3A-81 ABWR Reactor Bldg. Depth to Water Table, Node 95 Z R/B Top, 2% Damping

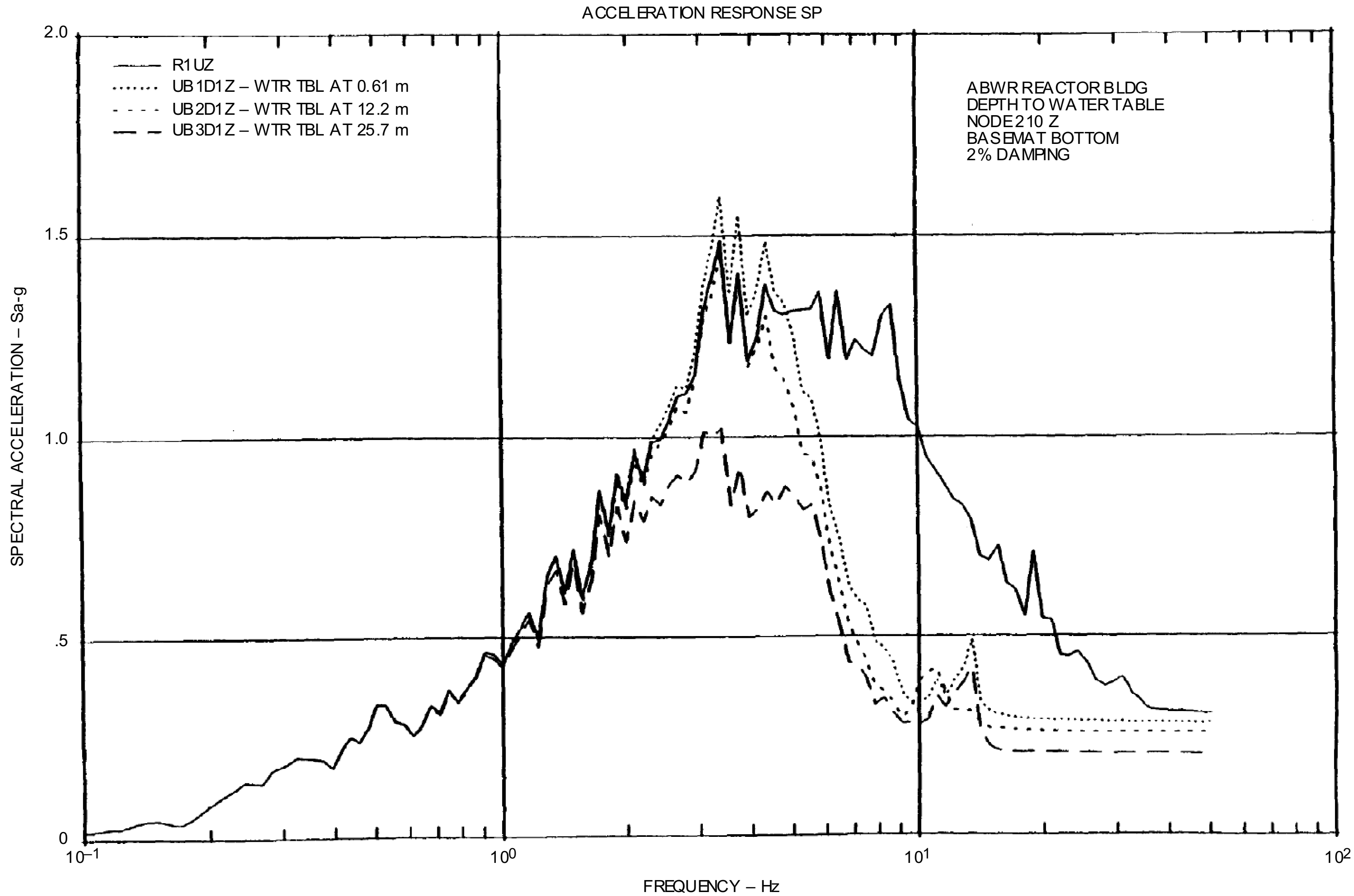


Figure 3A-82 ABWR Reactor Bldg. Depth to Water Table, Node 210 Z Basemat Bottom, 2% Damping

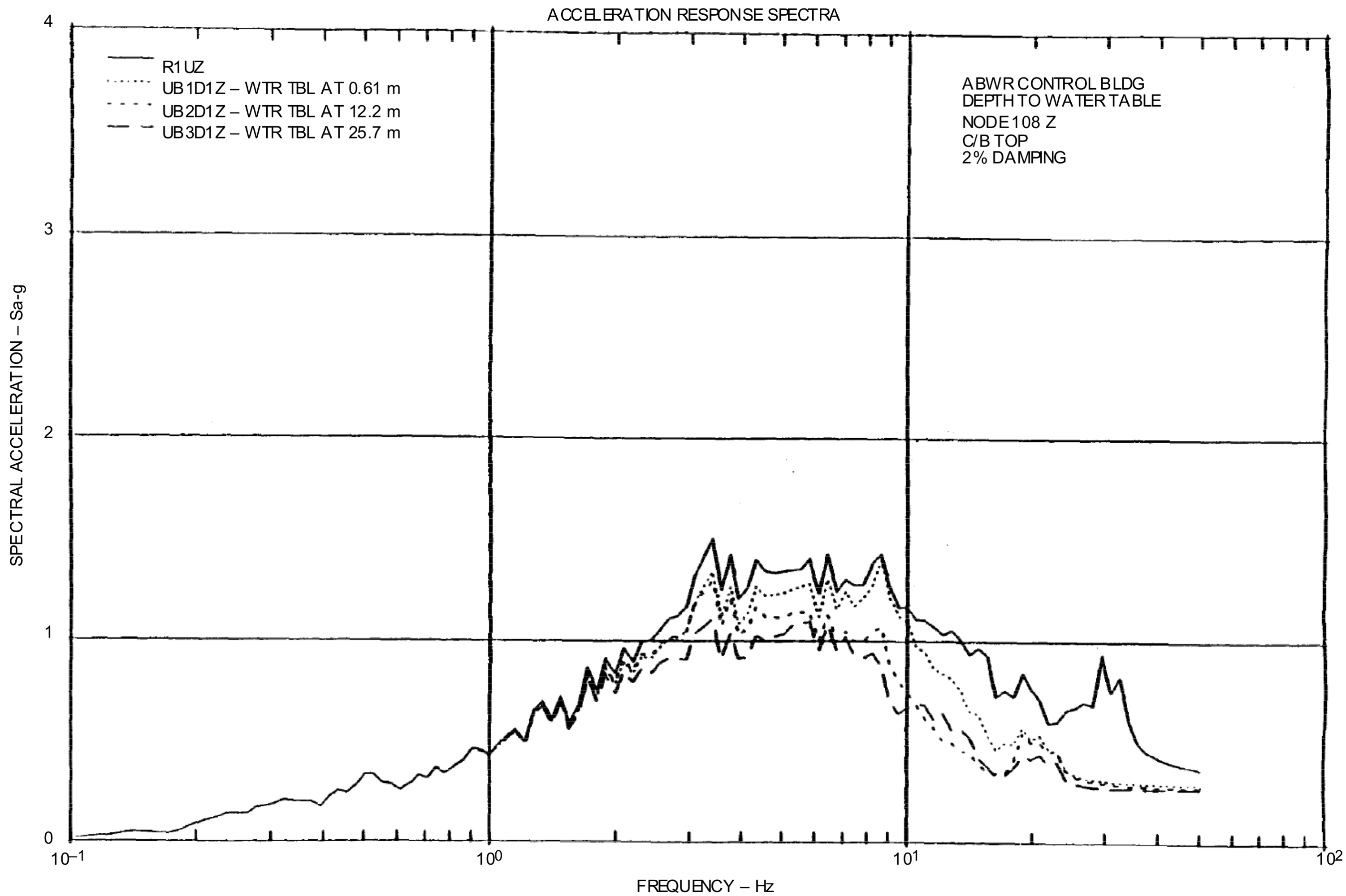


Figure 3A-83 ABWR Control Bldg. Depth to Water Table, Node 108 Z C/B Top, 2% Damping

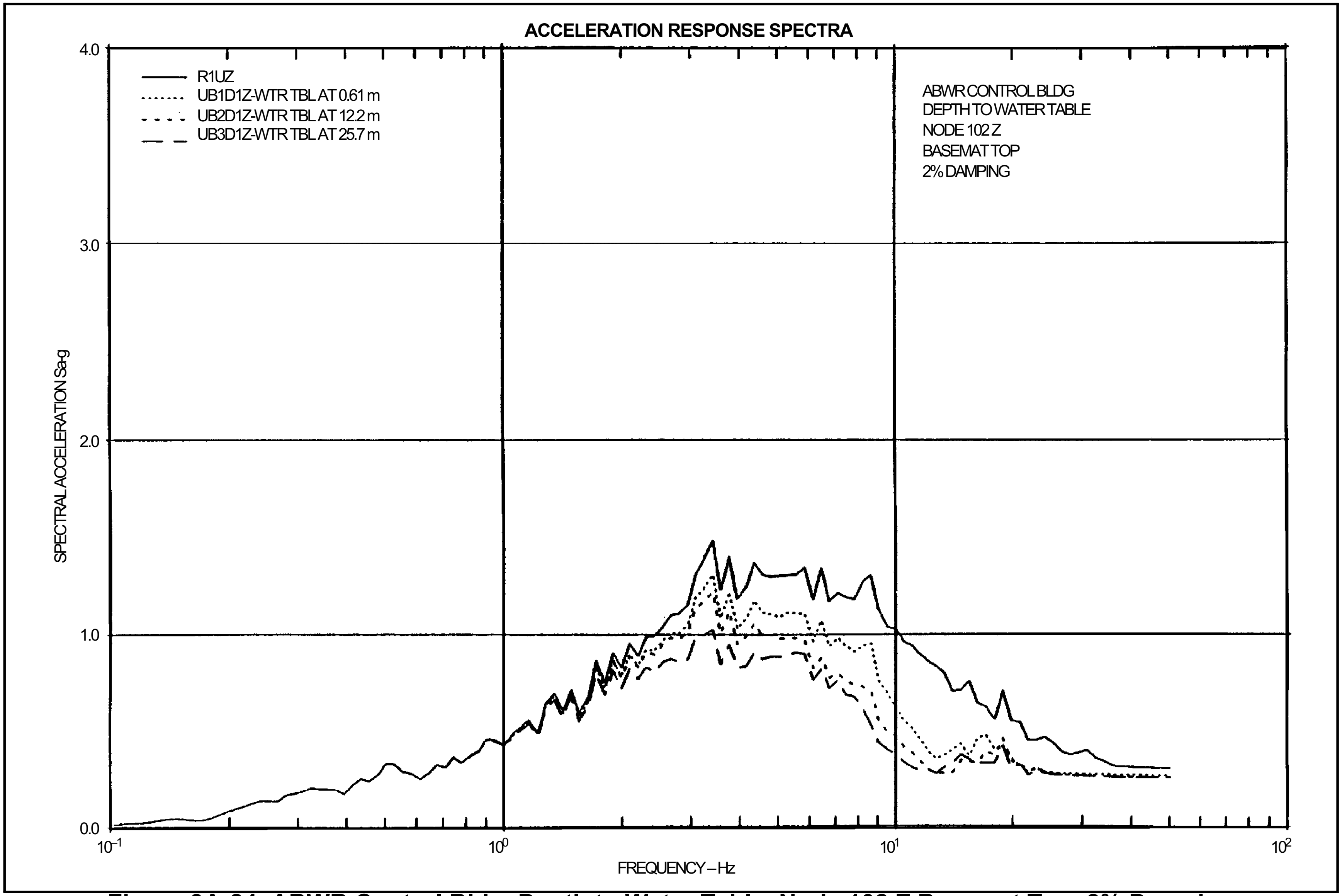


Figure 3A-84 ABWR Control Bldg. Depth to Water Table, Node 102 Z Basemat Top, 2% Damping

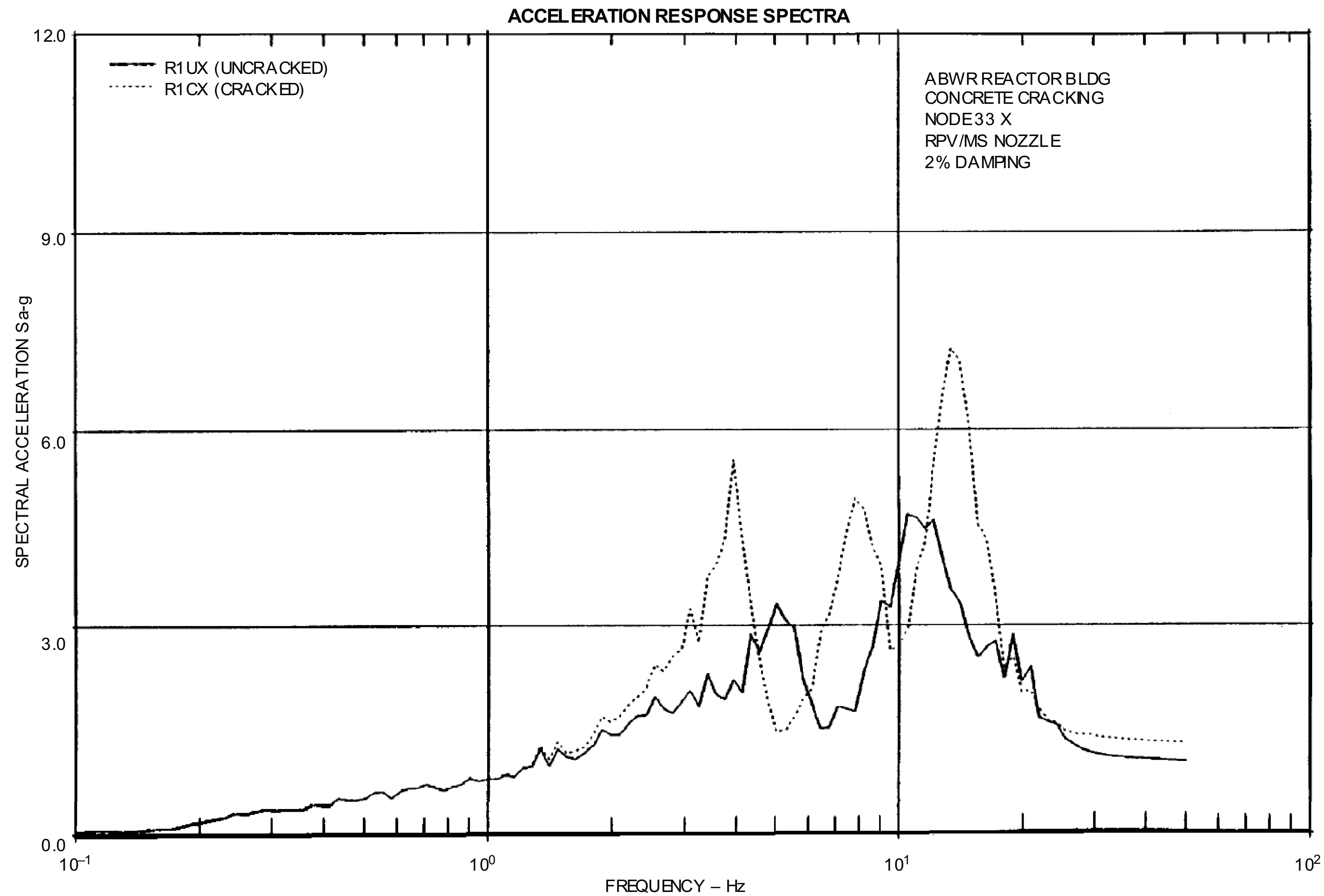


Figure 3A-85 ABWR Reactor Bldg. Concrete Cracking, Node 33 X RPV/MS Nozzle, 2% Damping

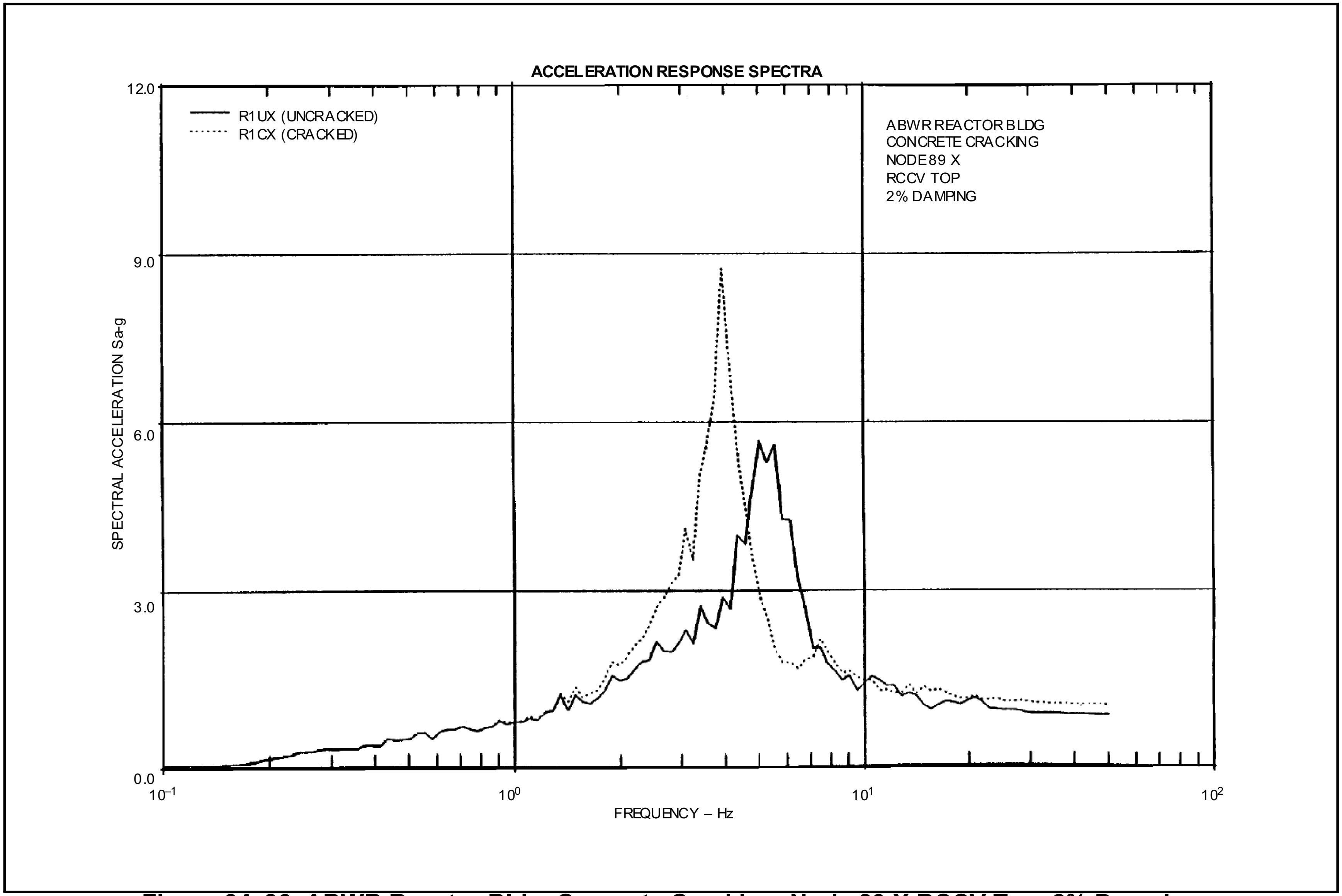


Figure 3A-86 ABWR Reactor Bldg. Concrete Cracking, Node 89 X RCCV Top, 2% Damping

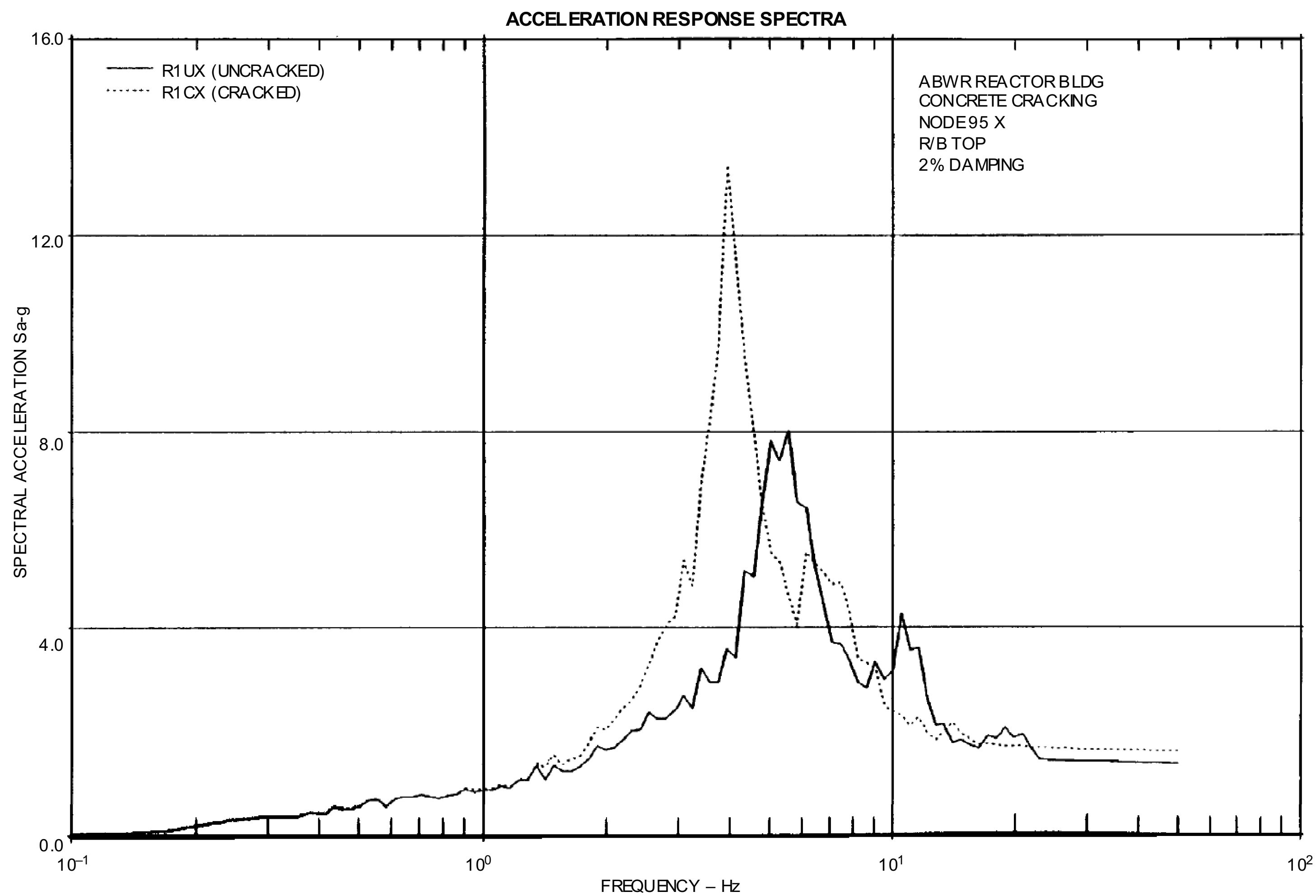


Figure 3A-87 ABWR Reactor Bldg. Concrete Cracking, Node 95 X R/B Top, 2% Damping

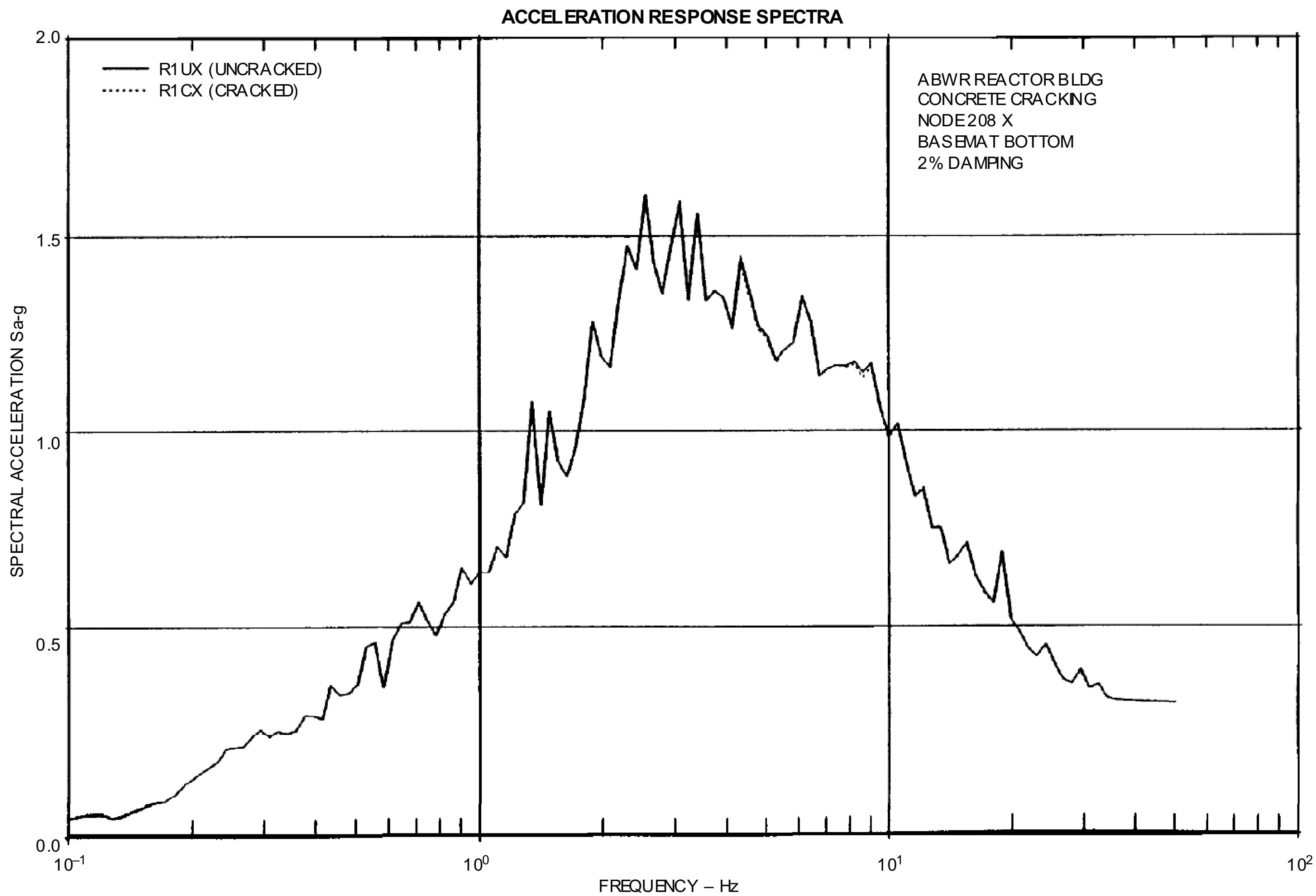


Figure 3A-88 ABWR Reactor Bldg. Concrete Cracking, Node 208 X Basemat Bottom, 2% Damping

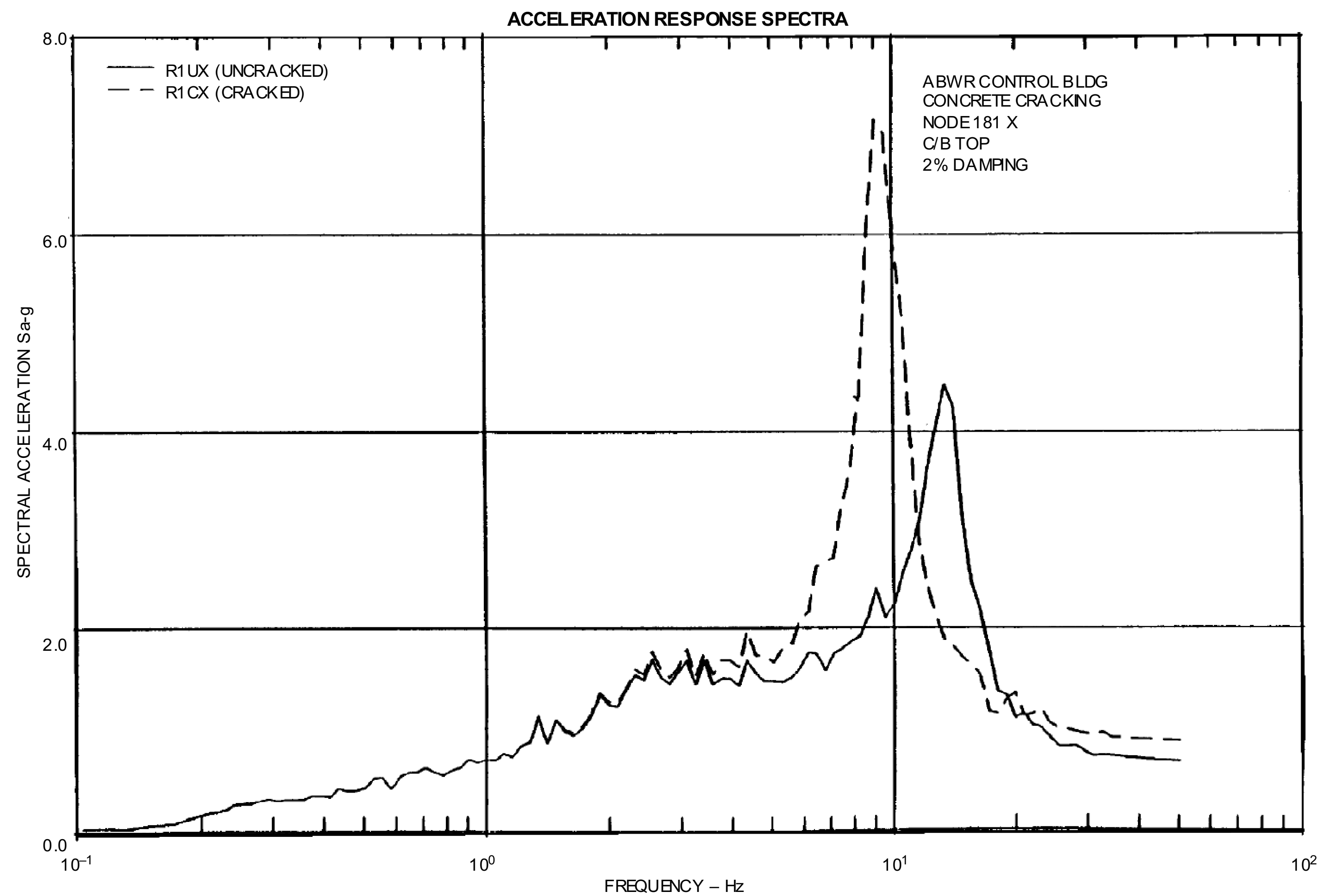


Figure 3A-89 ABWR Control Bldg. Concrete Cracking, Node 181 X C/B Top, 2% Damping

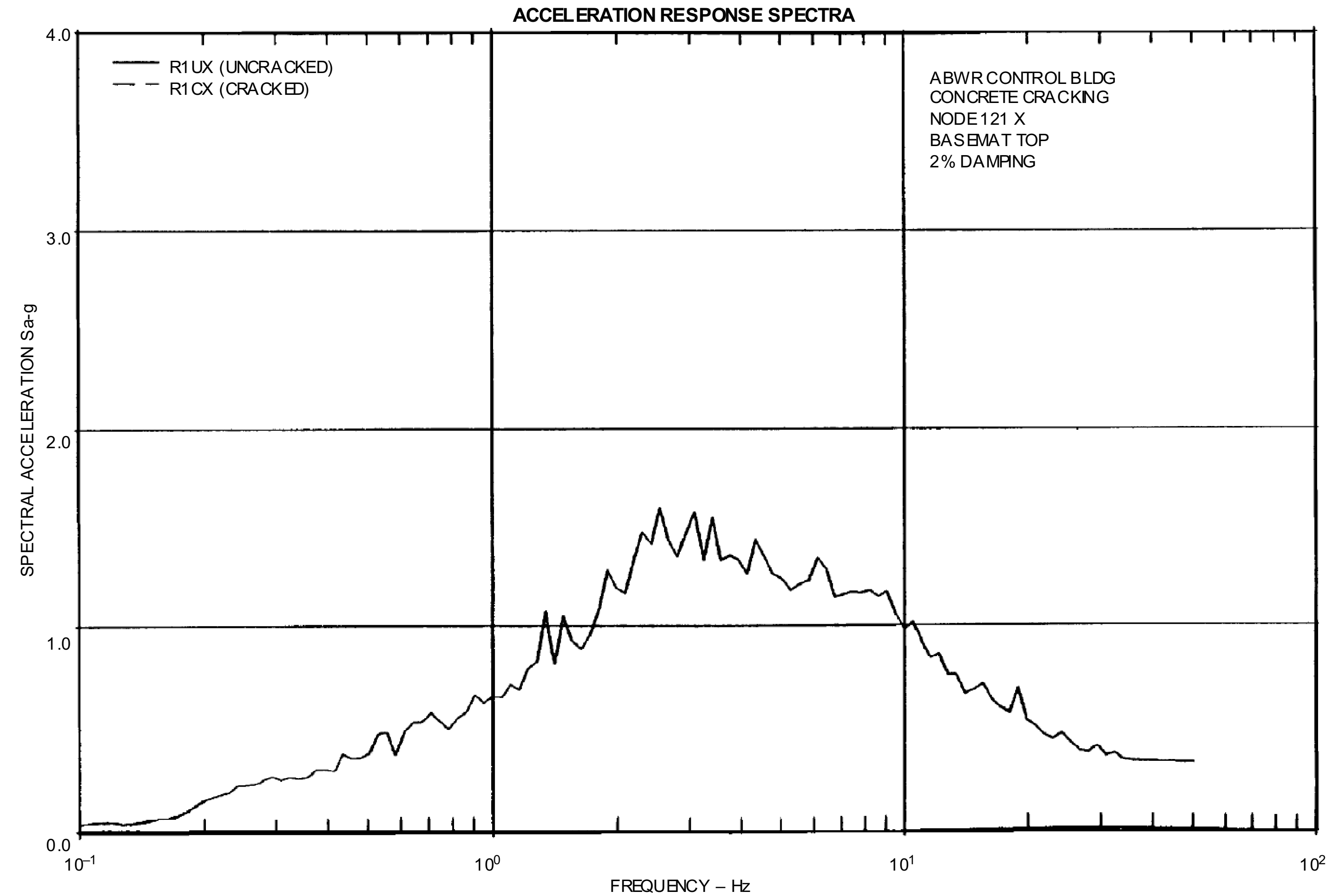


Figure 3A-90 ABWR Control Bldg. Concrete Cracking, Node 121 X Basemat Top, 2% Damping

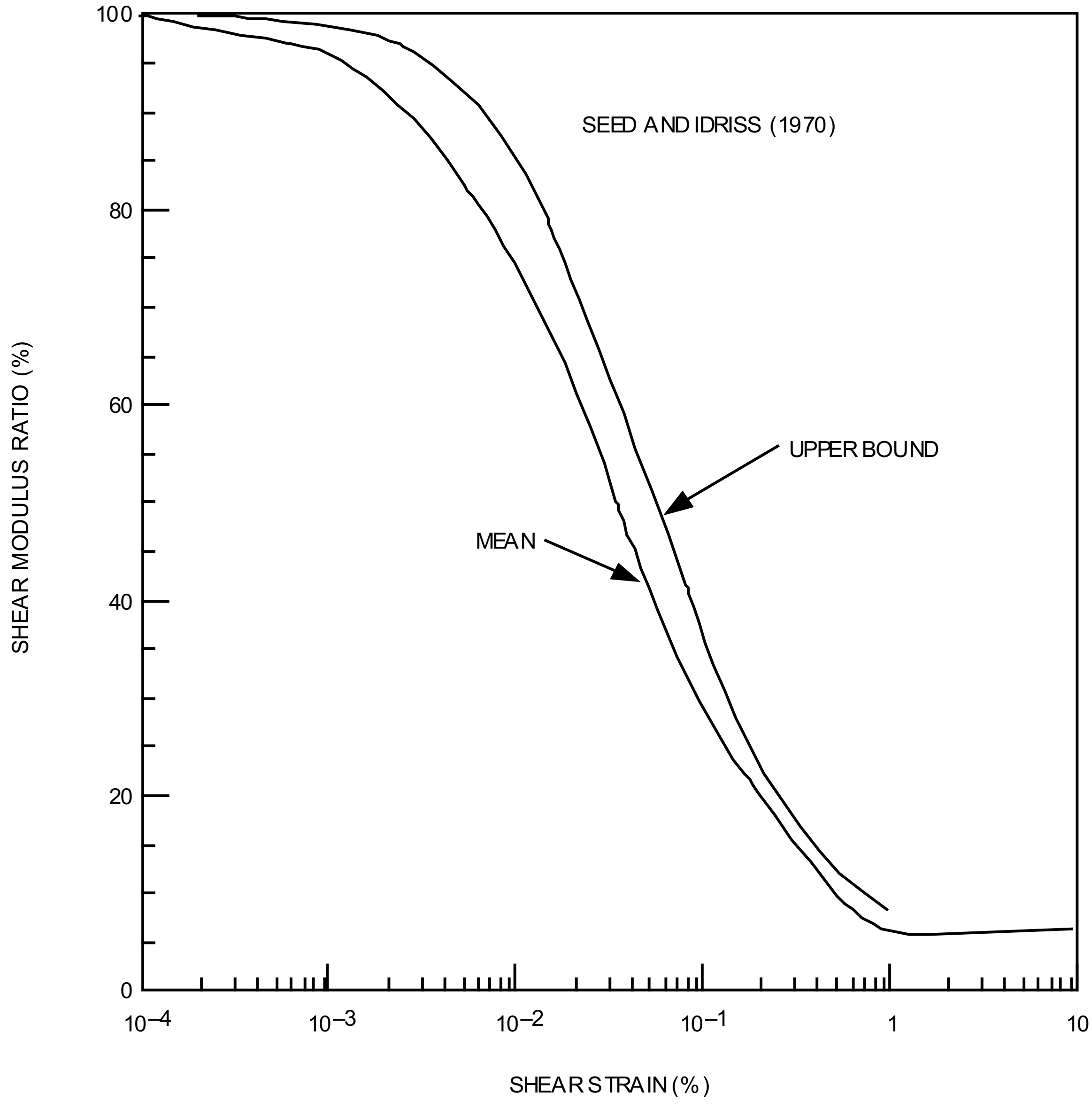


Figure 3A-91 Shear Modulus vs Shear Strain

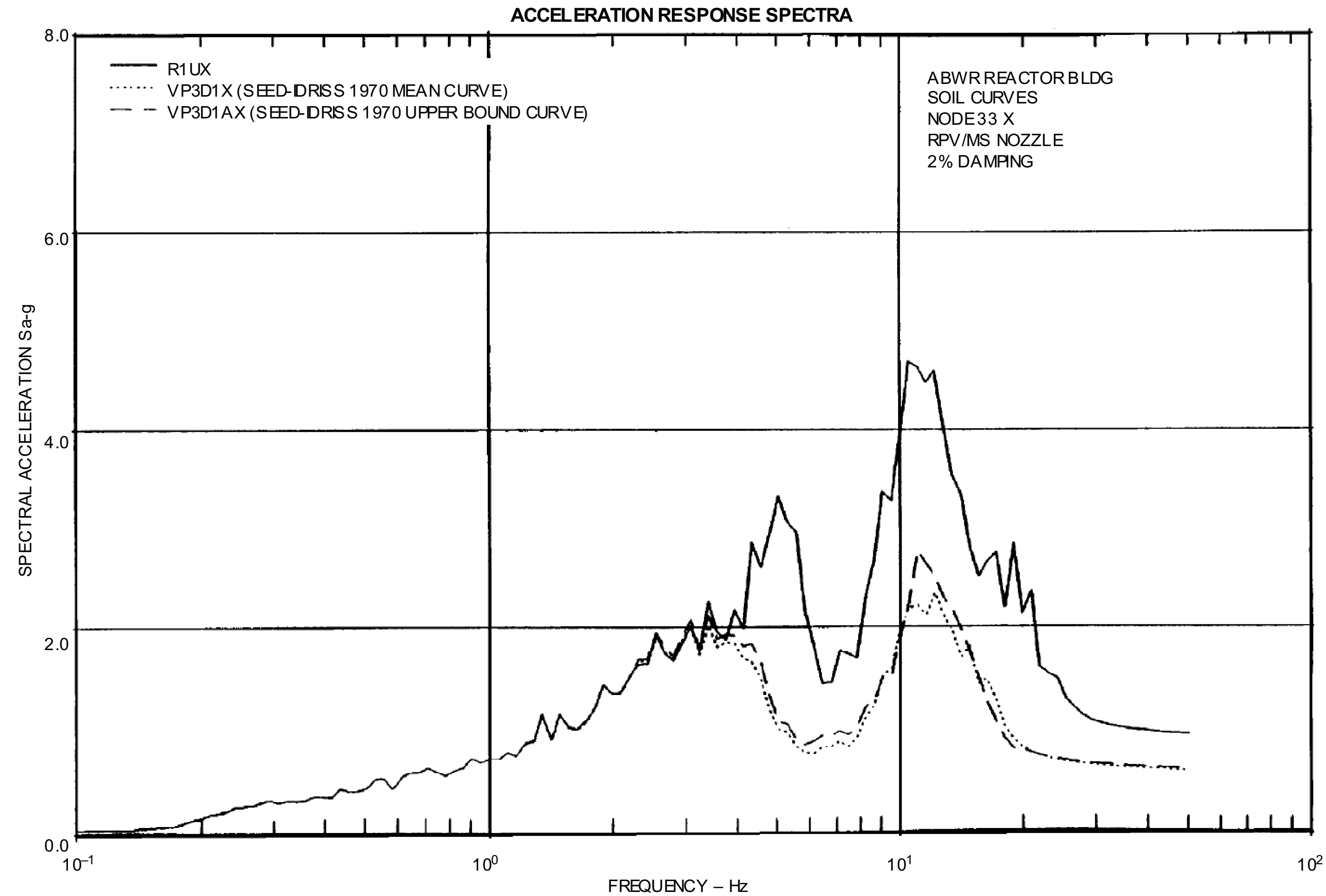


Figure 3A-92 ABWR Reactor Bldg. Soil Curves, Node 33 X RPV/MS Nozzle, 2% Damping

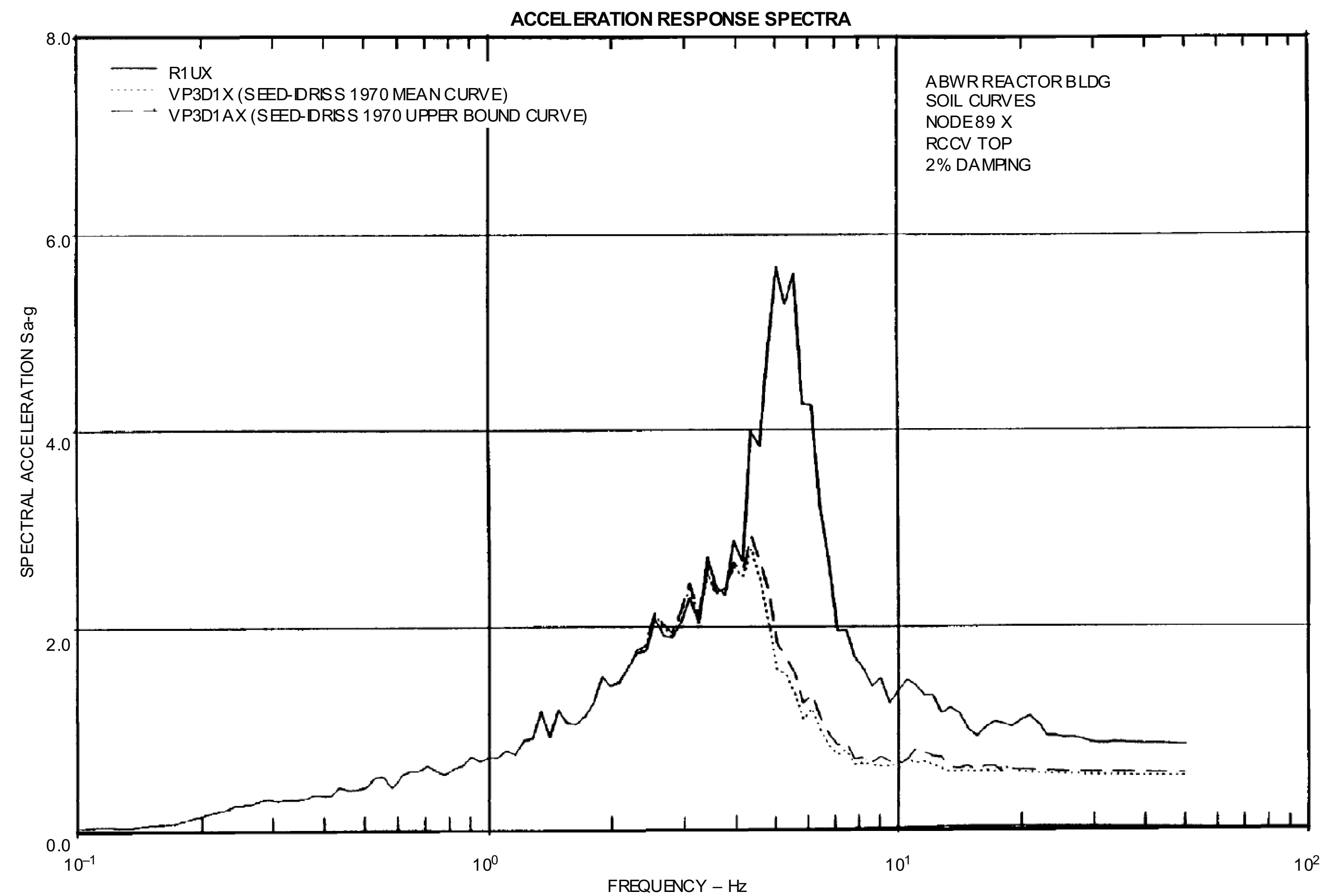


Figure 3A-93 ABWR Reactor Bldg. Soil Curves, Node 89 X RCCV Top, 2% Damping

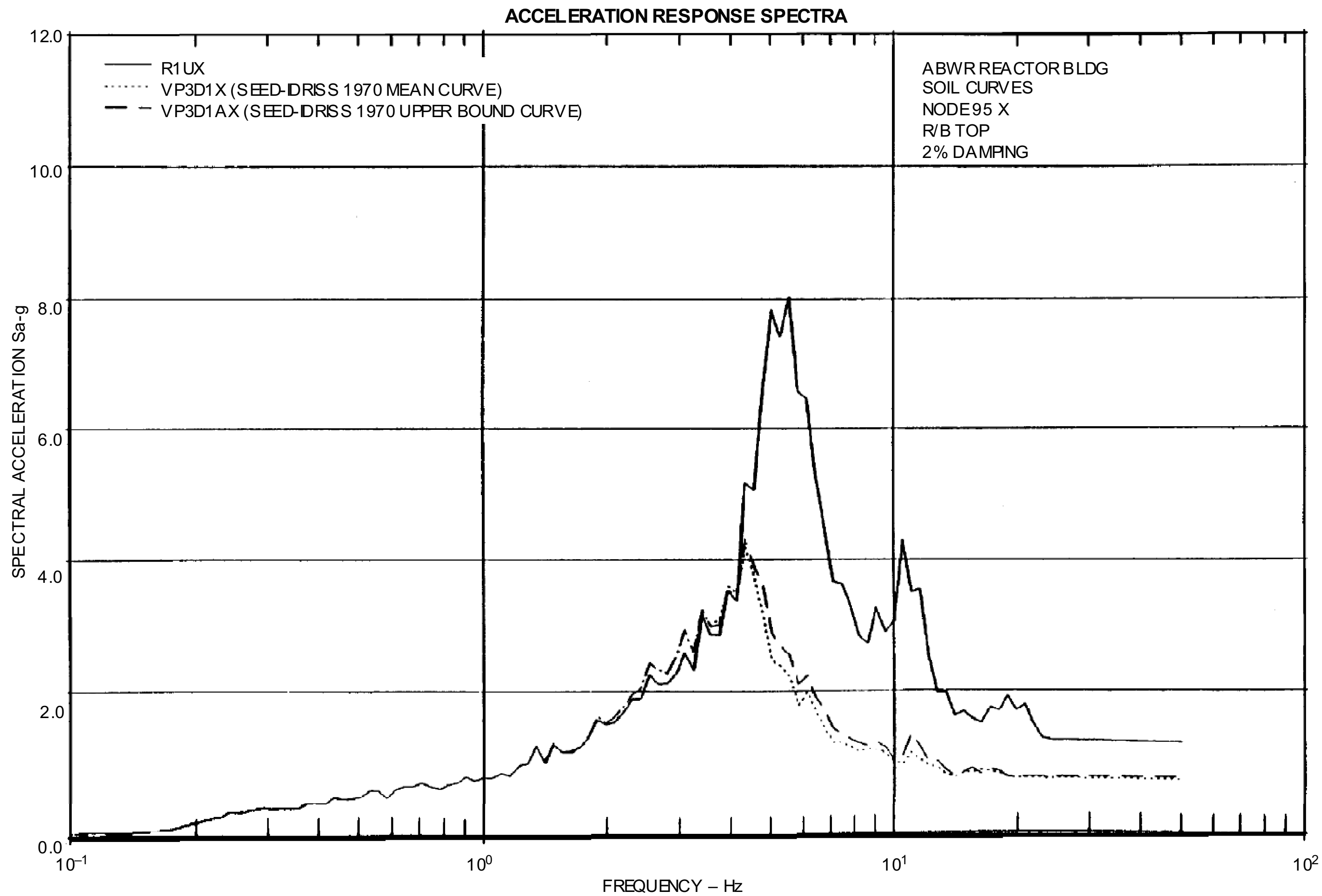


Figure 3A-94 ABWR Reactor Bldg. Soil Curves, Node 95 X R/B Top, 2% Damping

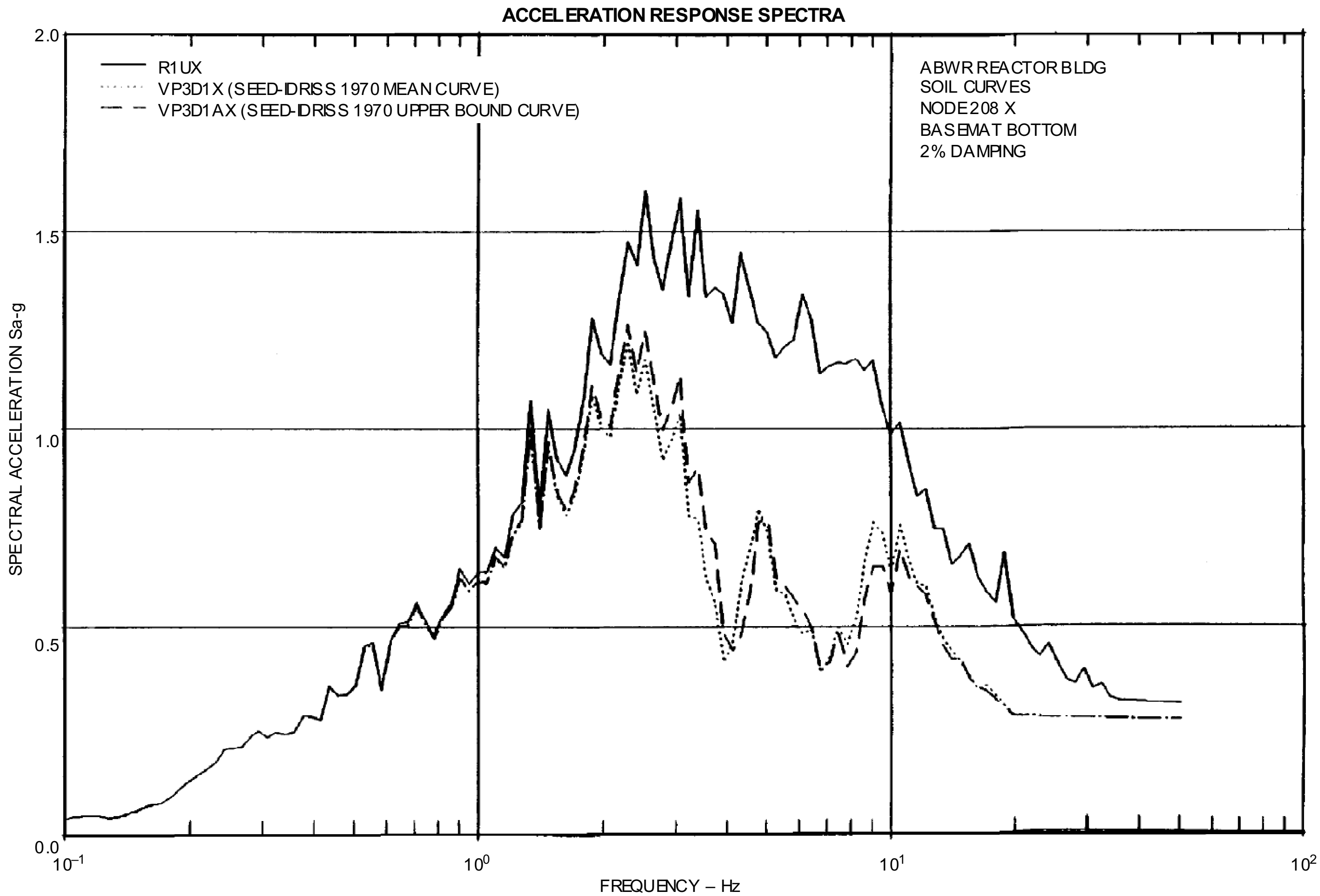


Figure 3A-95 ABWR Reactor Bldg. Soil Curves, Node 208 X Basemat Bottom, 2% Damping

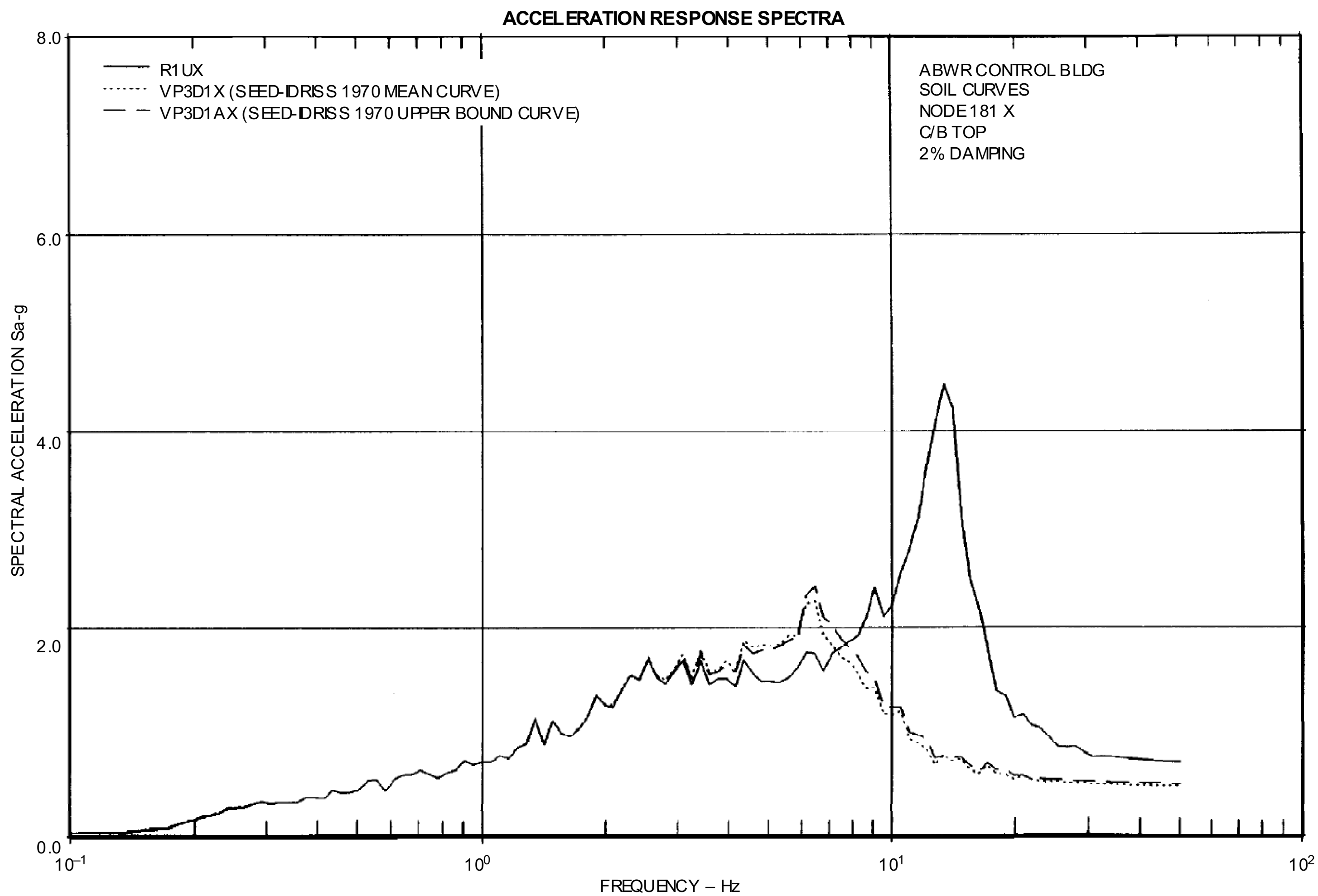


Figure 3A-96 ABWR Control Bldg. Soil Curves, Node 181 X C/B Top, 2% Damping

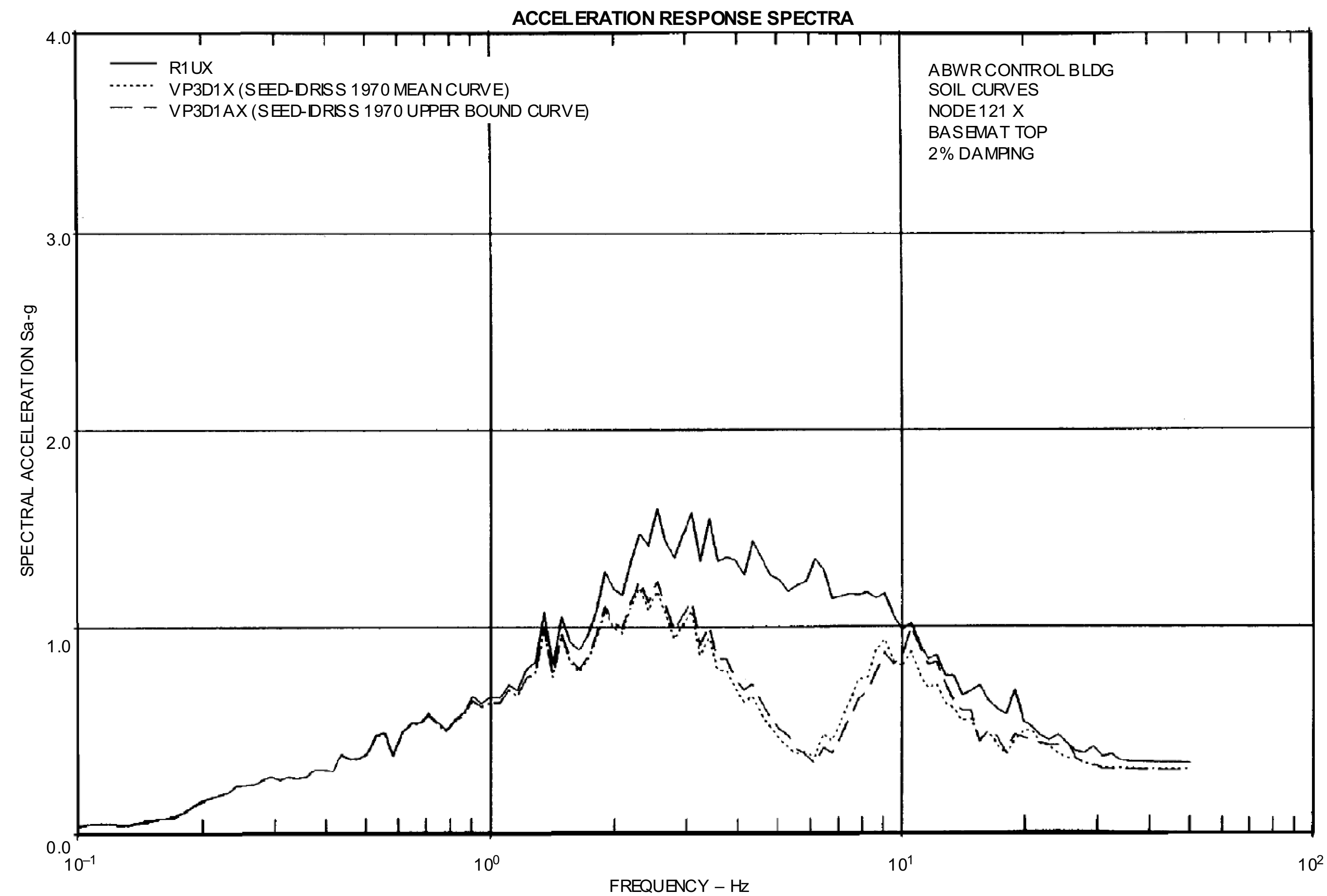


Figure 3A-97 ABWR Control Bldg. Soil Curves, Node 121 X Basemat Top, 2% Damping

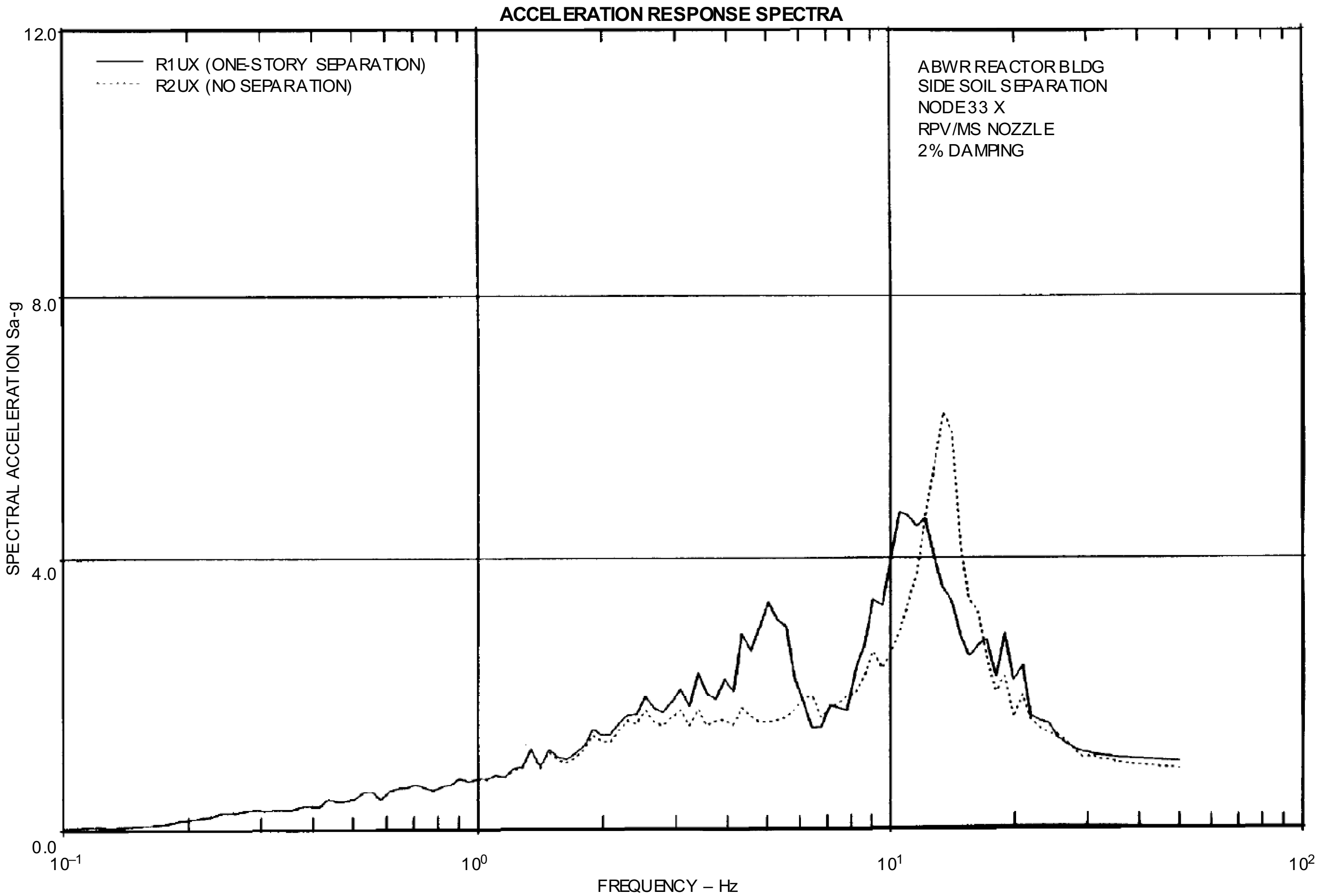


Figure 3A-98 ABWR Reactor Bldg. Side Soil Separation, Node 33 X RPV/MS Nozzle, 2% Damping

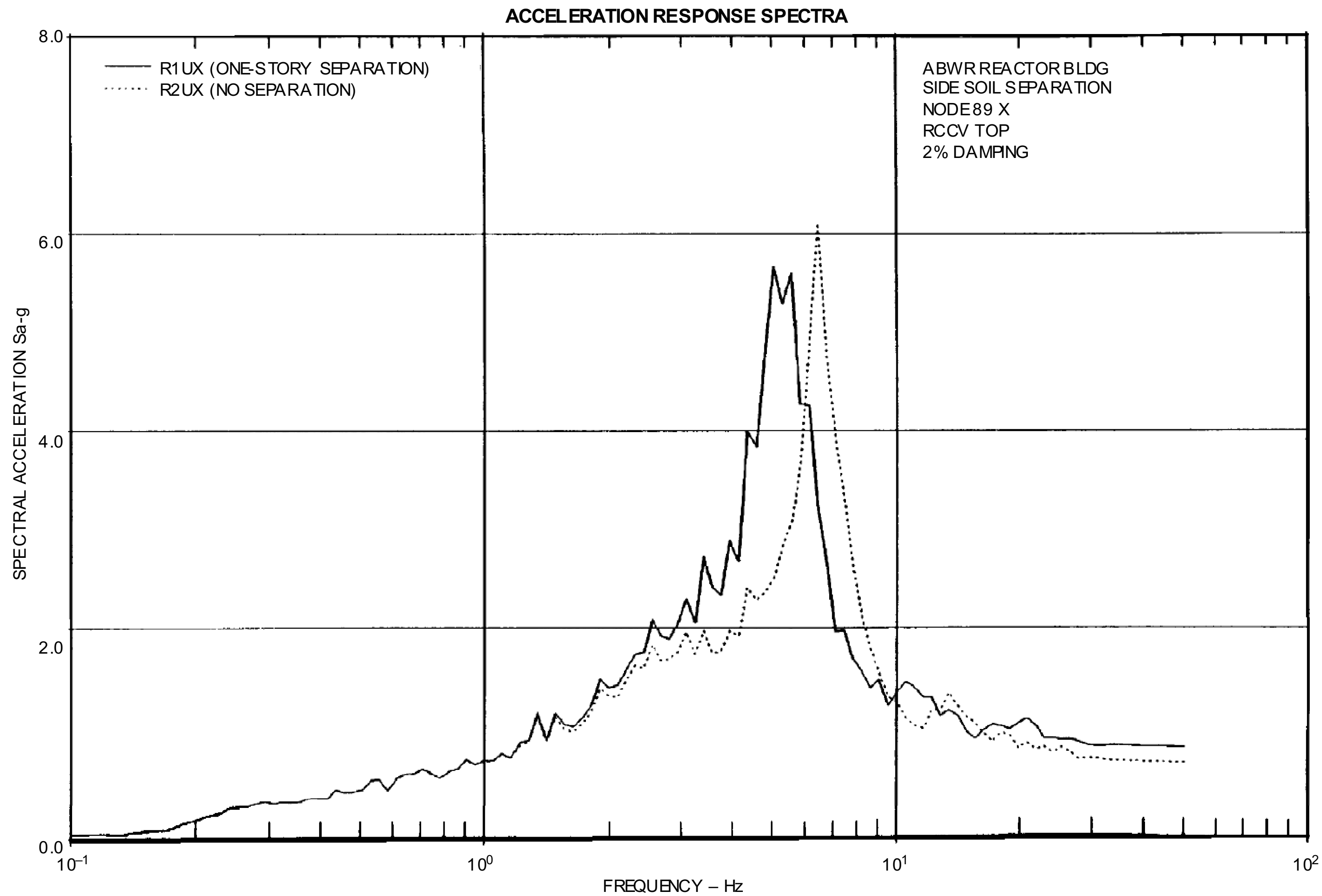


Figure 3A-99 ABWR Reactor Bldg. Side Soil Separation, Node 89 X RCCV Top, 2% Damping

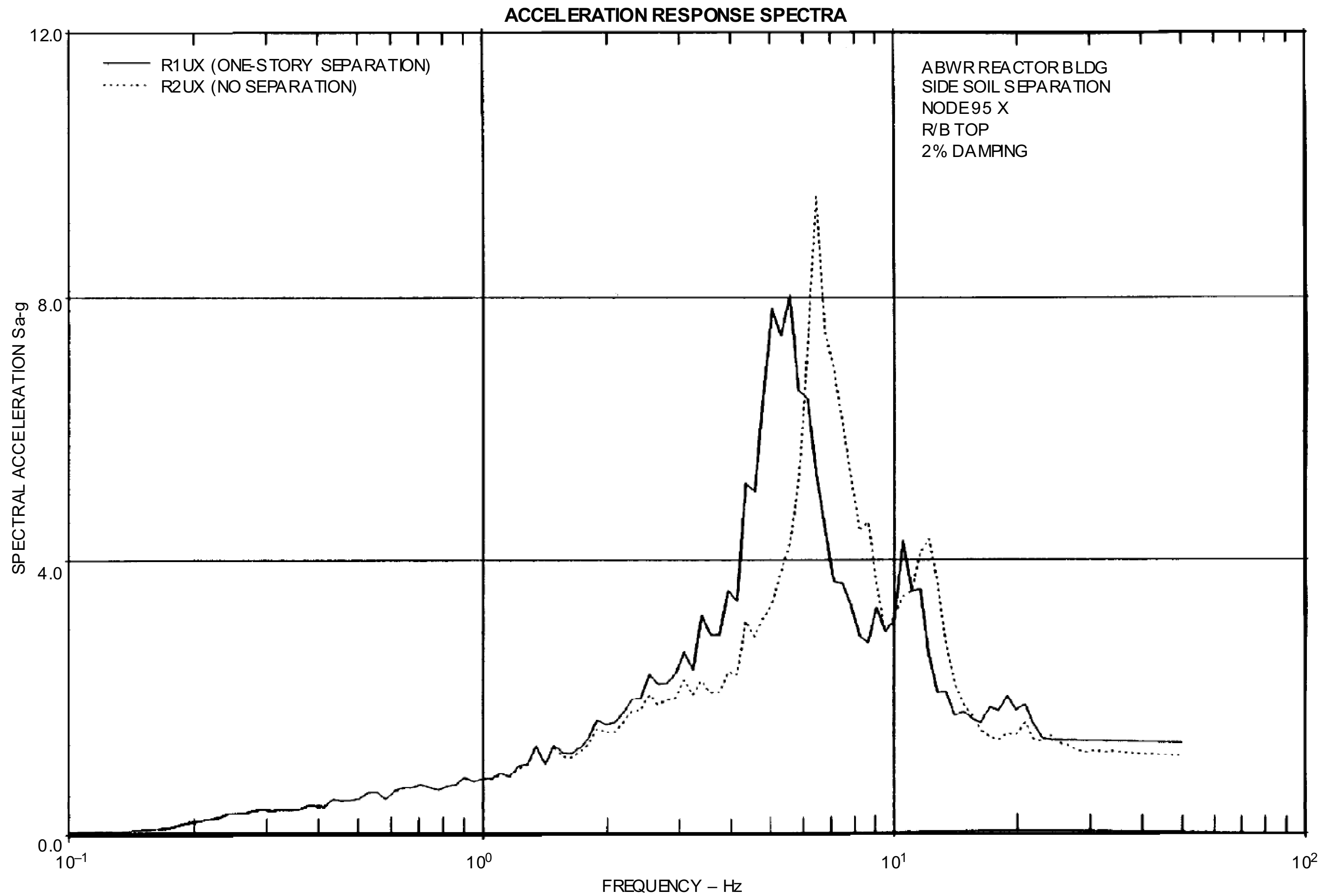


Figure 3A-100 ABWR Reactor Bldg. Side Soil Separation, Node 95 X R/B Top, 2% Damping

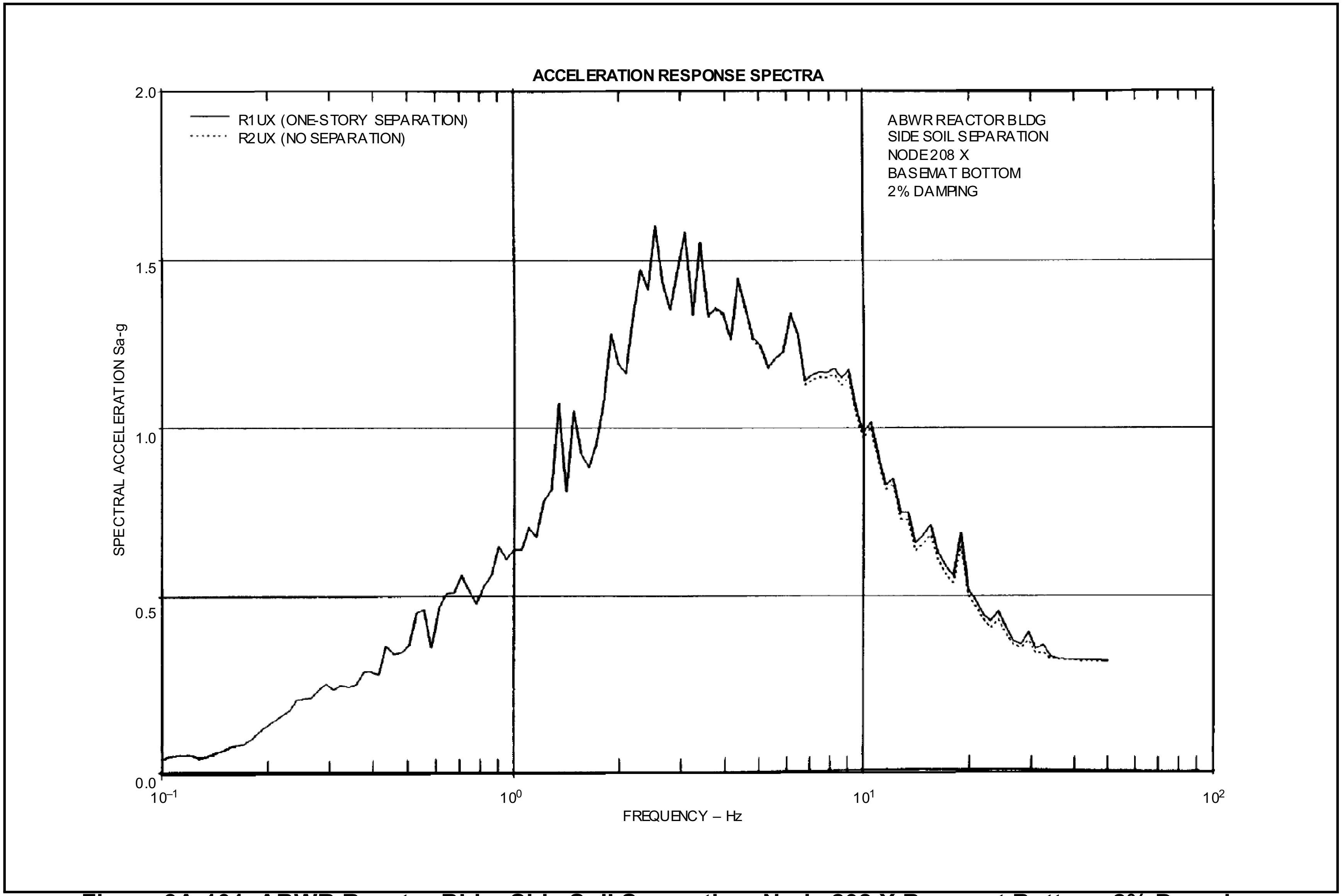


Figure 3A-101 ABWR Reactor Bldg. Side Soil Separation, Node 208 X Basemat Bottom, 2% Damping

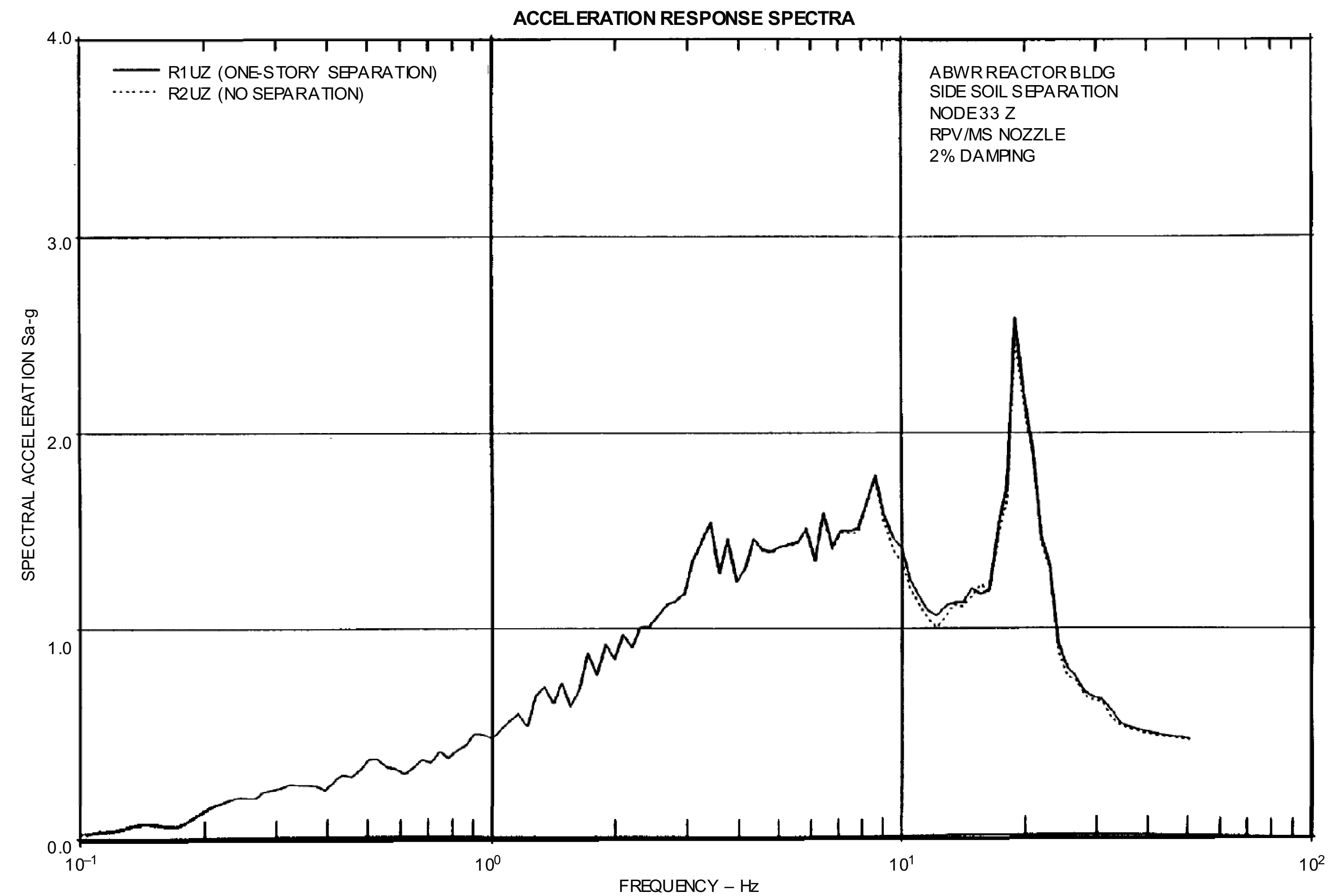


Figure 3A-102 ABWR Reactor Bldg. Side Soil Separation, Node 33 Z RPV/MS Nozzle, 2% Damping

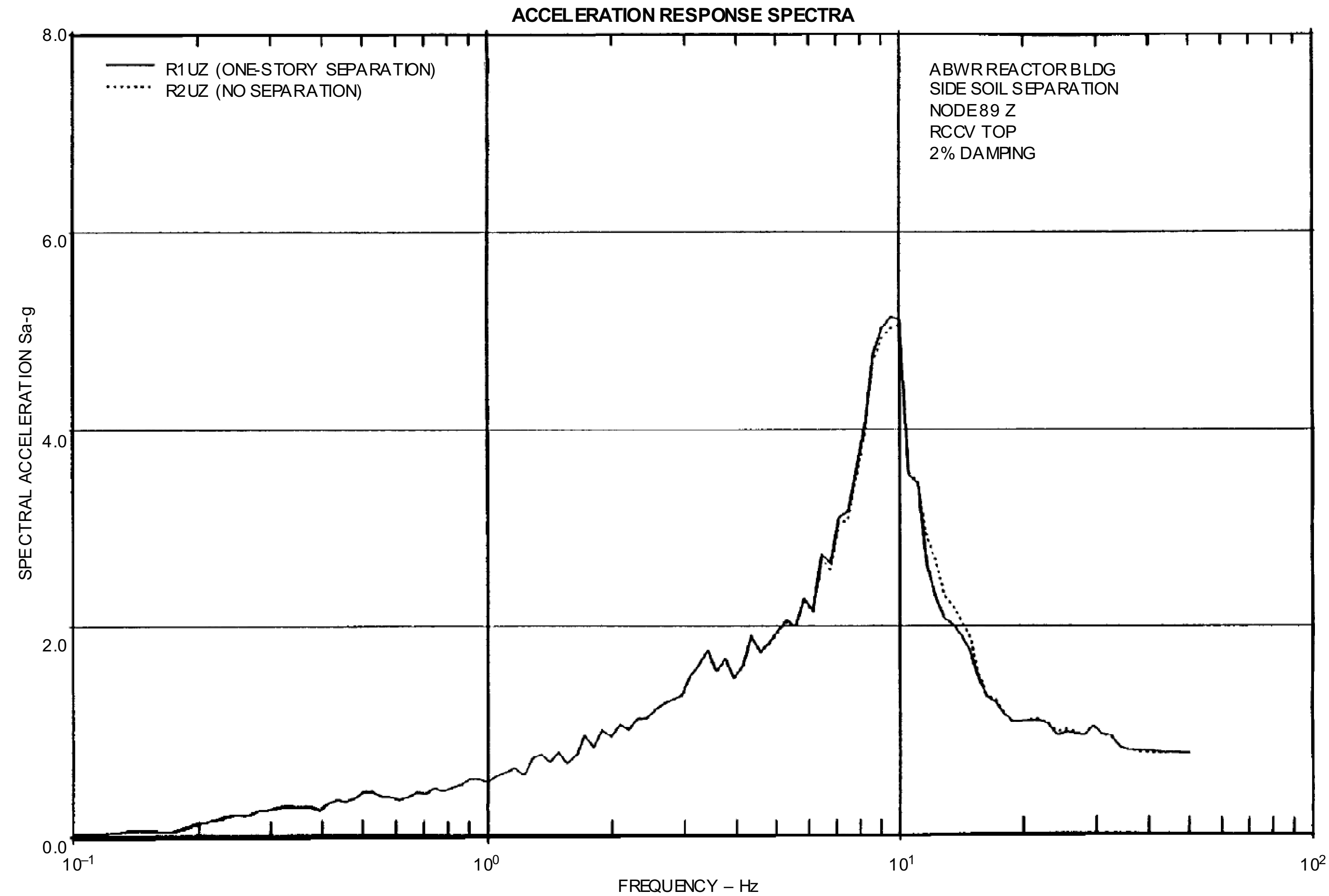


Figure 3A-103 ABWR Reactor Bldg. Side Soil Separation, Node 89 Z RCCV Top, 2% Damping

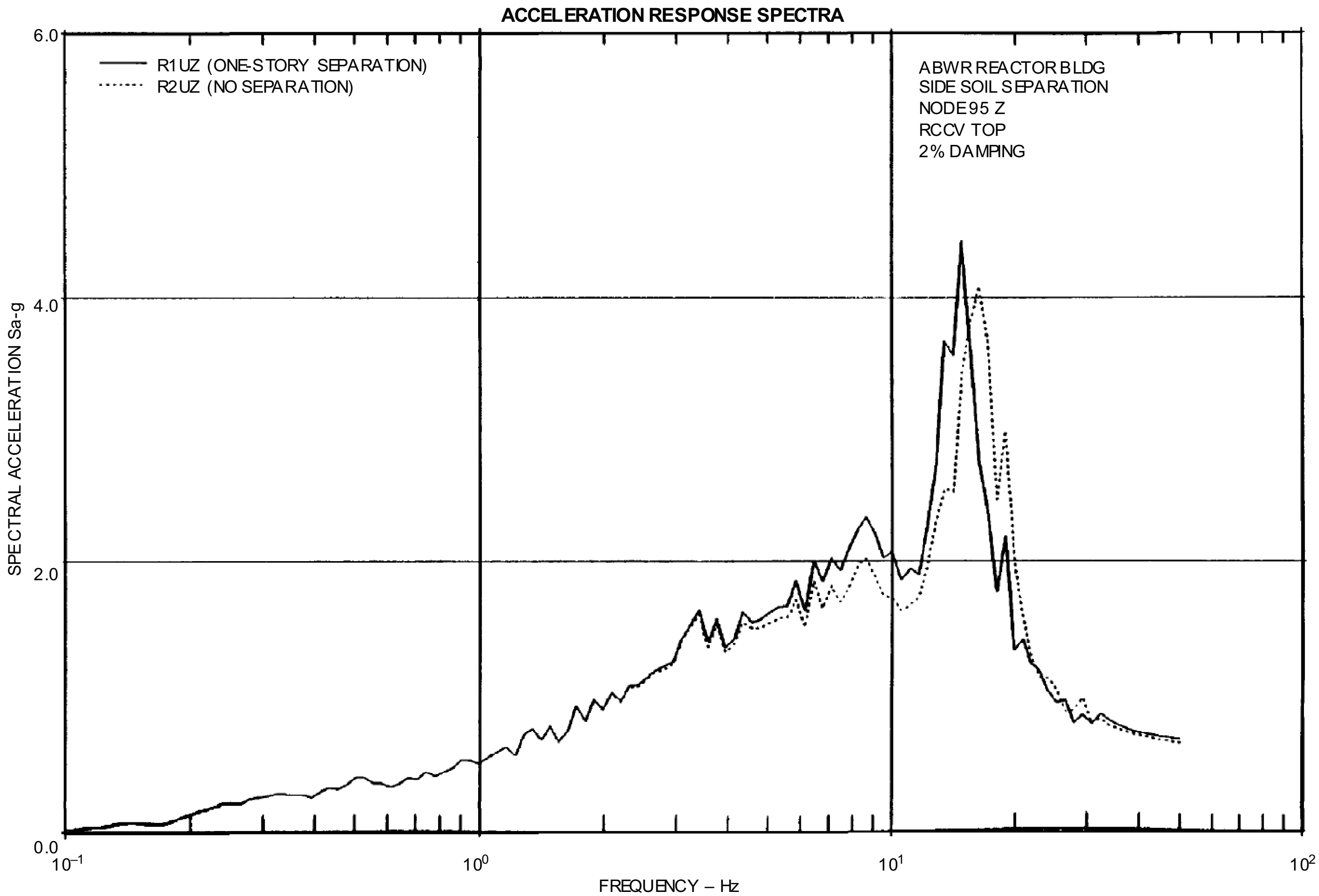


Figure 3A-104 ABWR Reactor Bldg. Side Soil Separation, Node 95 Z R/B Top, 2% Damping

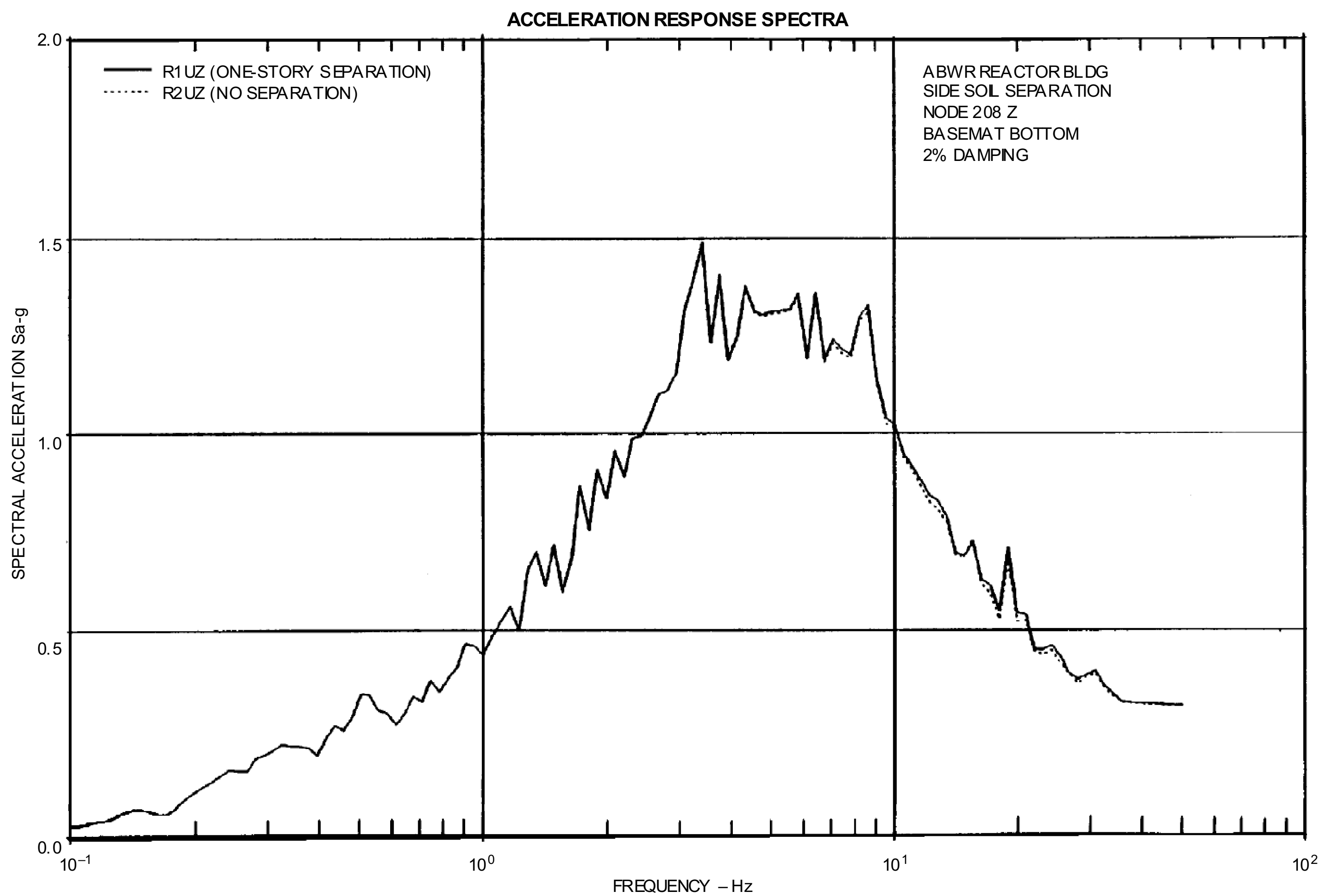


Figure 3A-105 ABWR Reactor Bldg. Side Soil Separation, Node 208 Z Basemat Bottom, 2% Damping

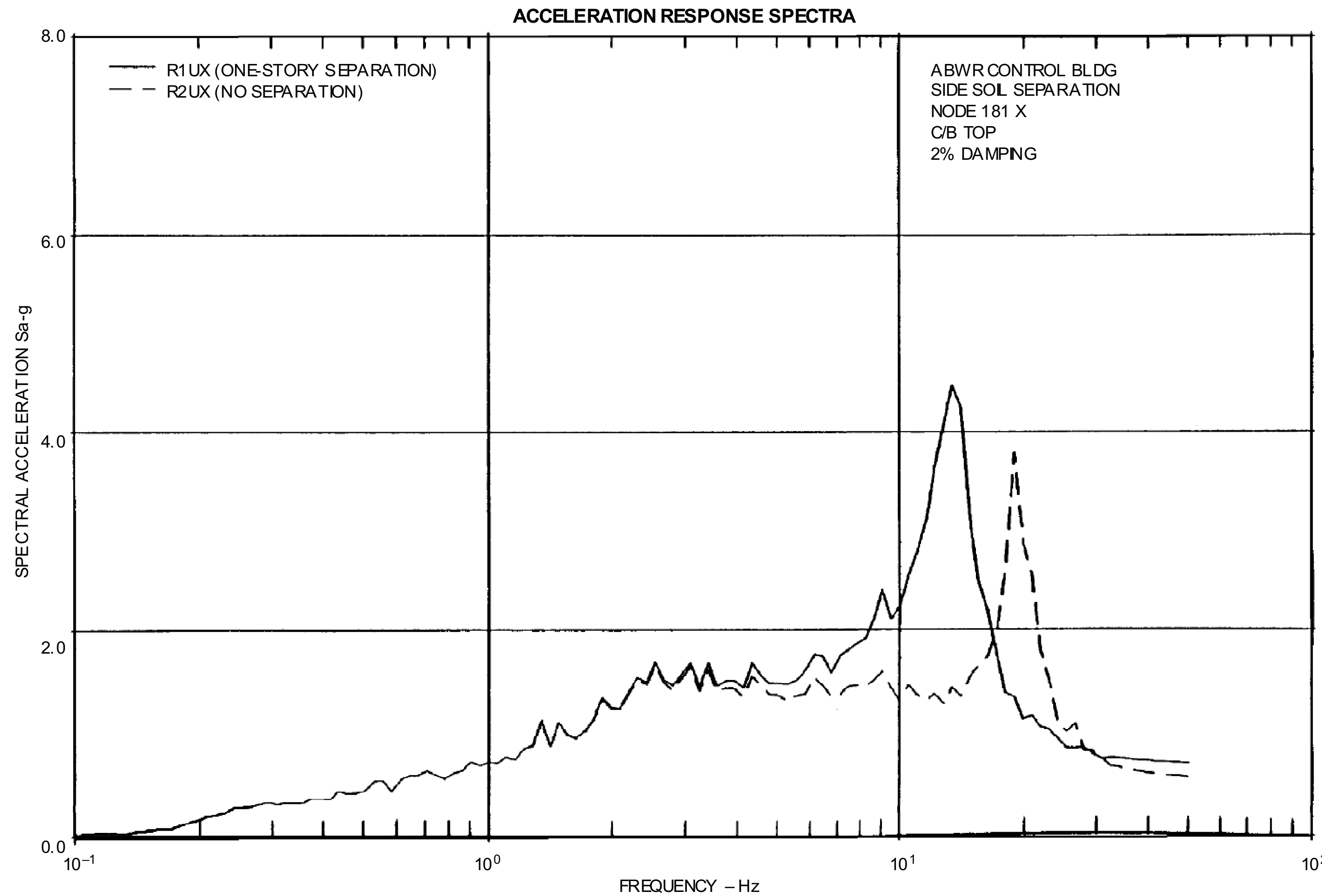


Figure 3A-106 ABWR Control Bldg. Side Soil Separation, Node 181 X C/B Top, 2% Damping

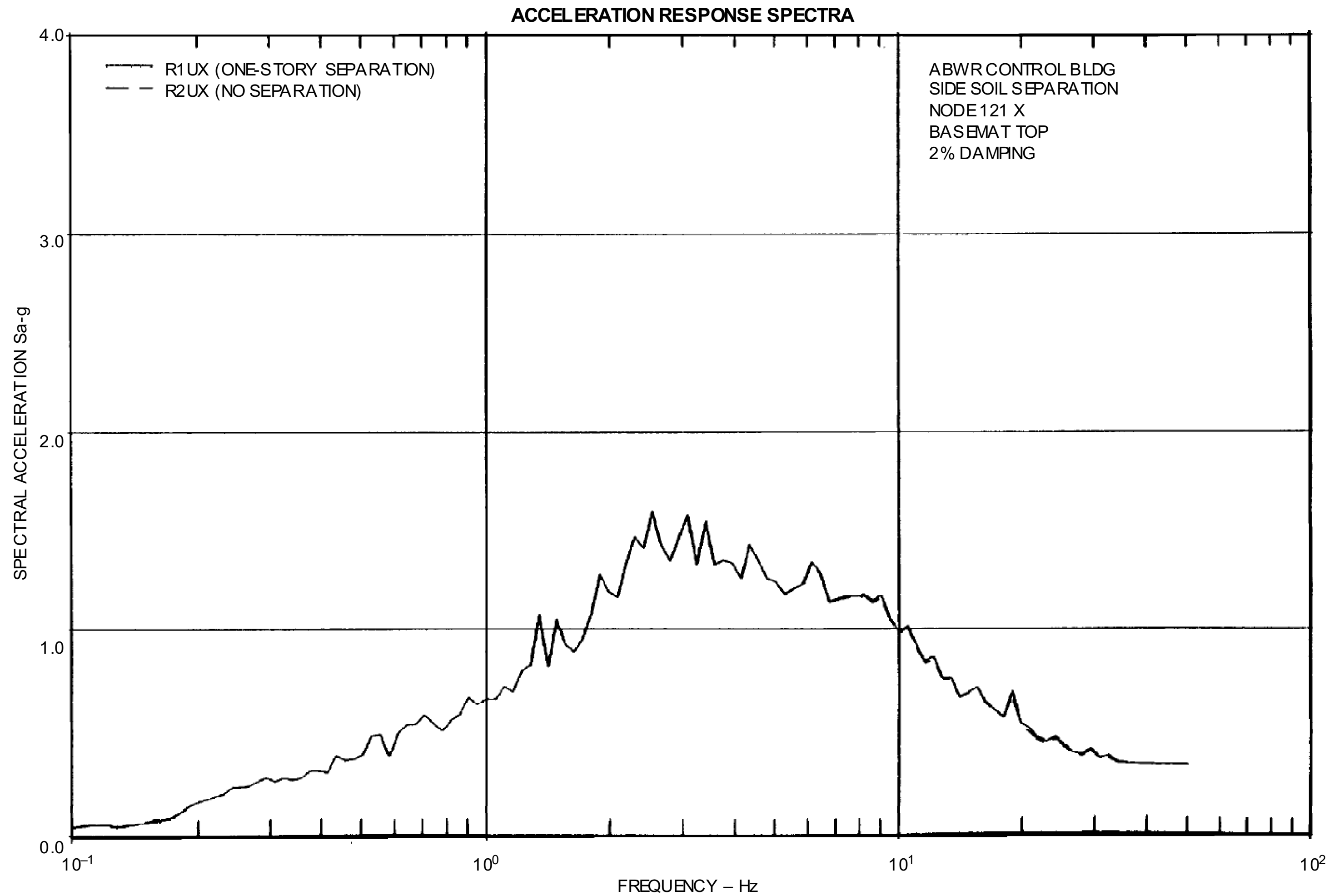


Figure 3A-107 ABWR Control Bldg. Side Soil Separation, Node 121 X Basemat Top, 2% Damping

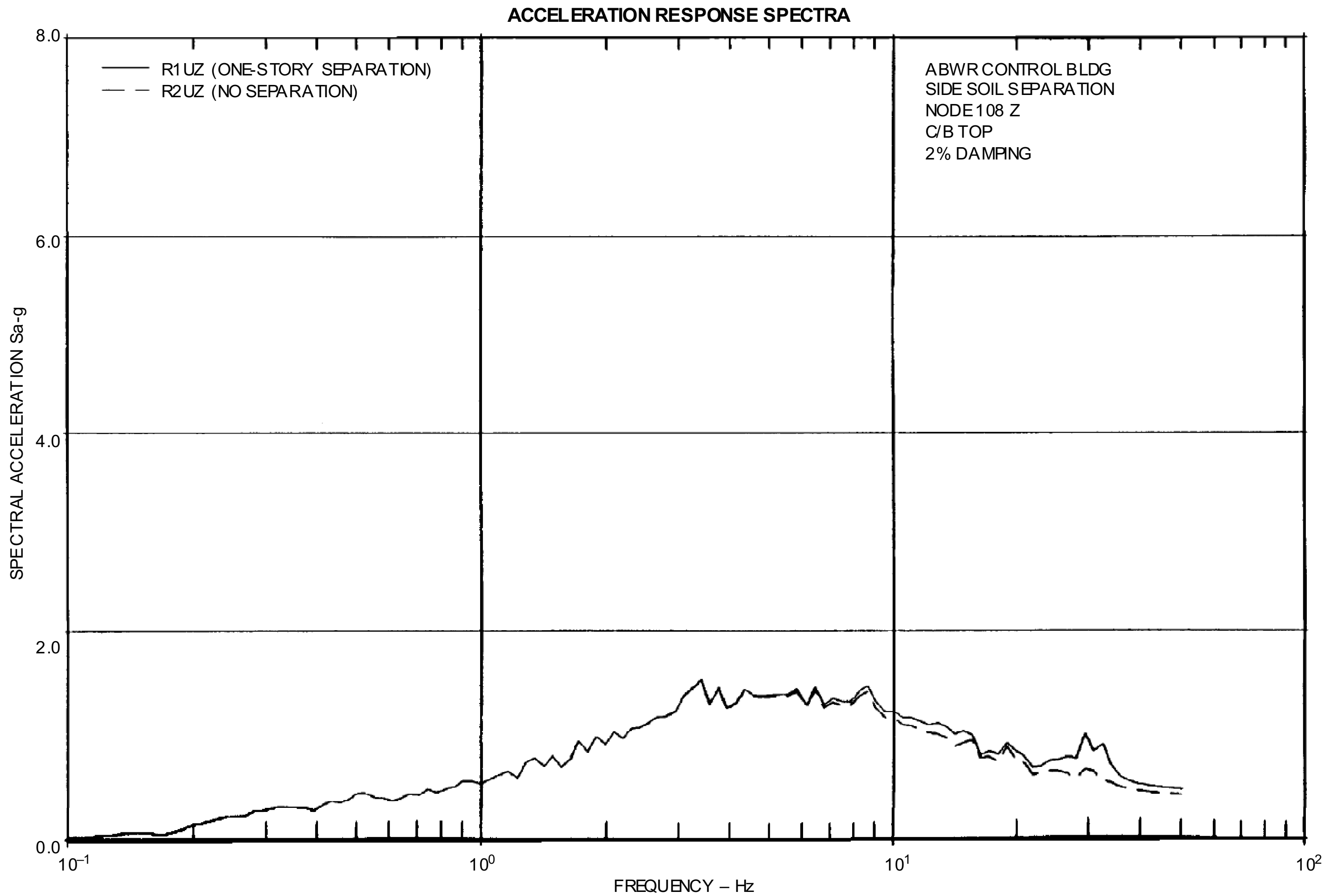


Figure 3A-108 ABWR Control Bldg. Side Soil Separation, Node 108 Z C/B Top, 2% Damping

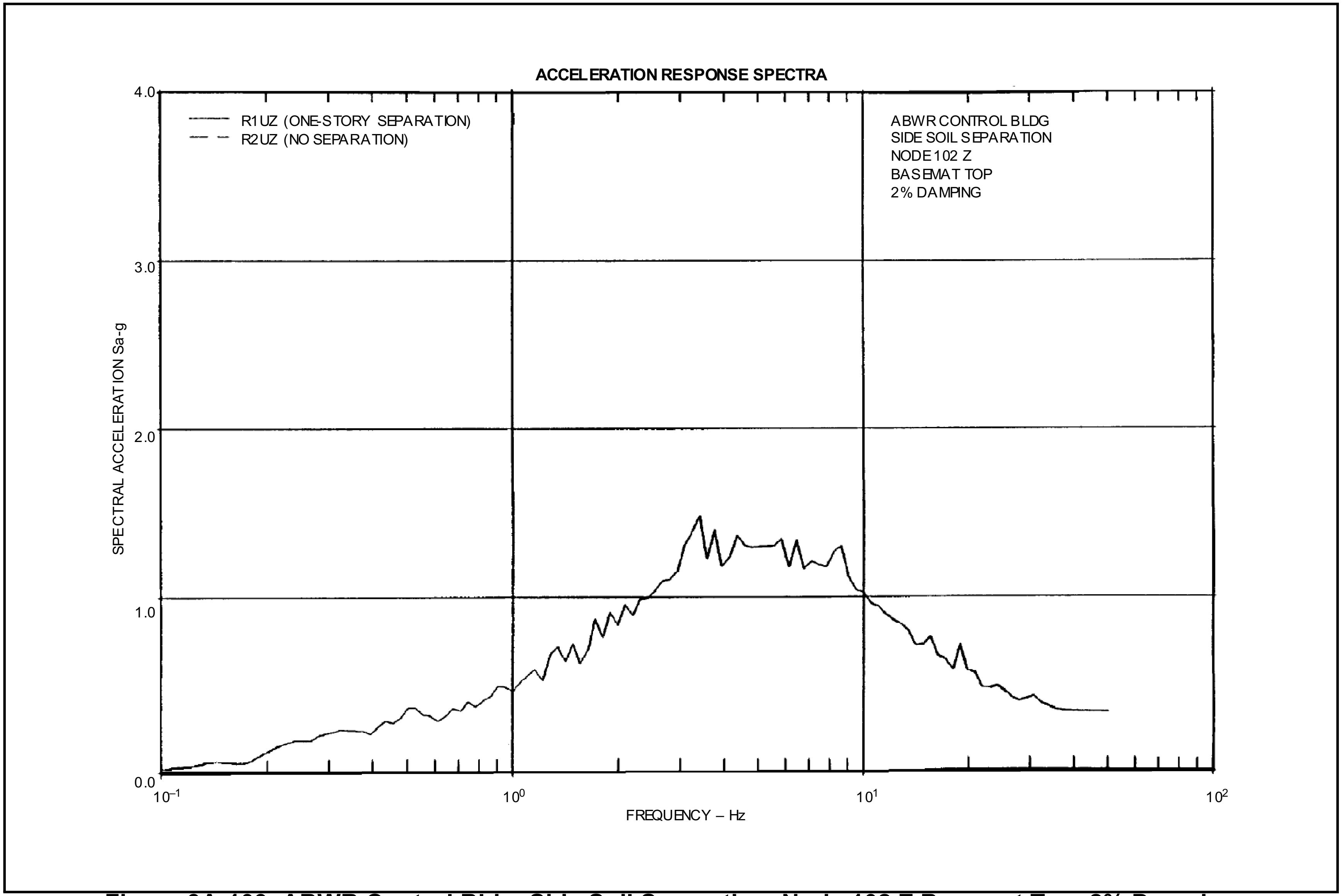


Figure 3A-109 ABWR Control Bldg. Side Soil Separation, Node 102 Z Basemat Top, 2% Damping

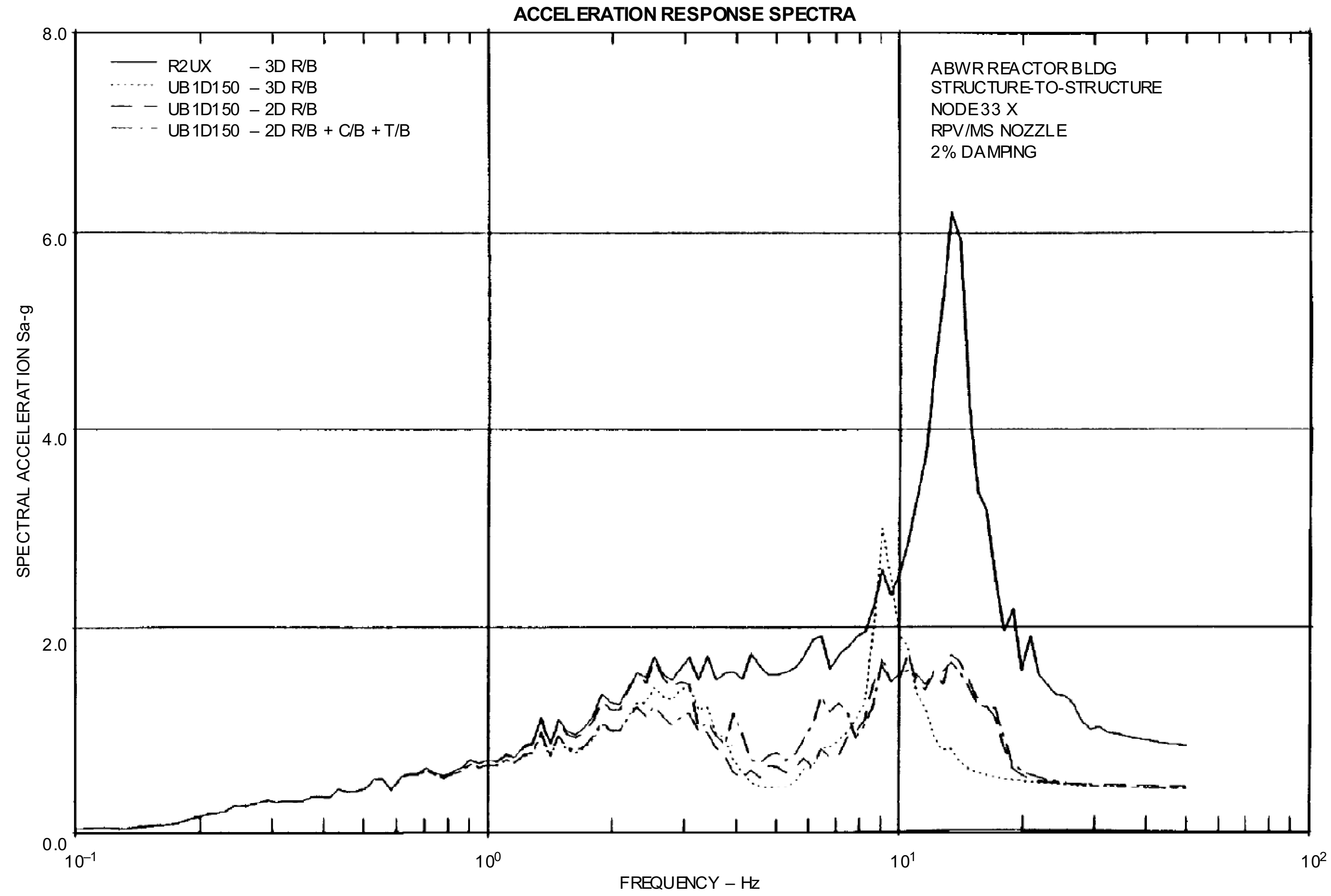


Figure 3A-110 ABWR Reactor Bldg. Structure-to-Structure, Node 33 X RPV/MS Nozzle, 2% Damping, UB1D150

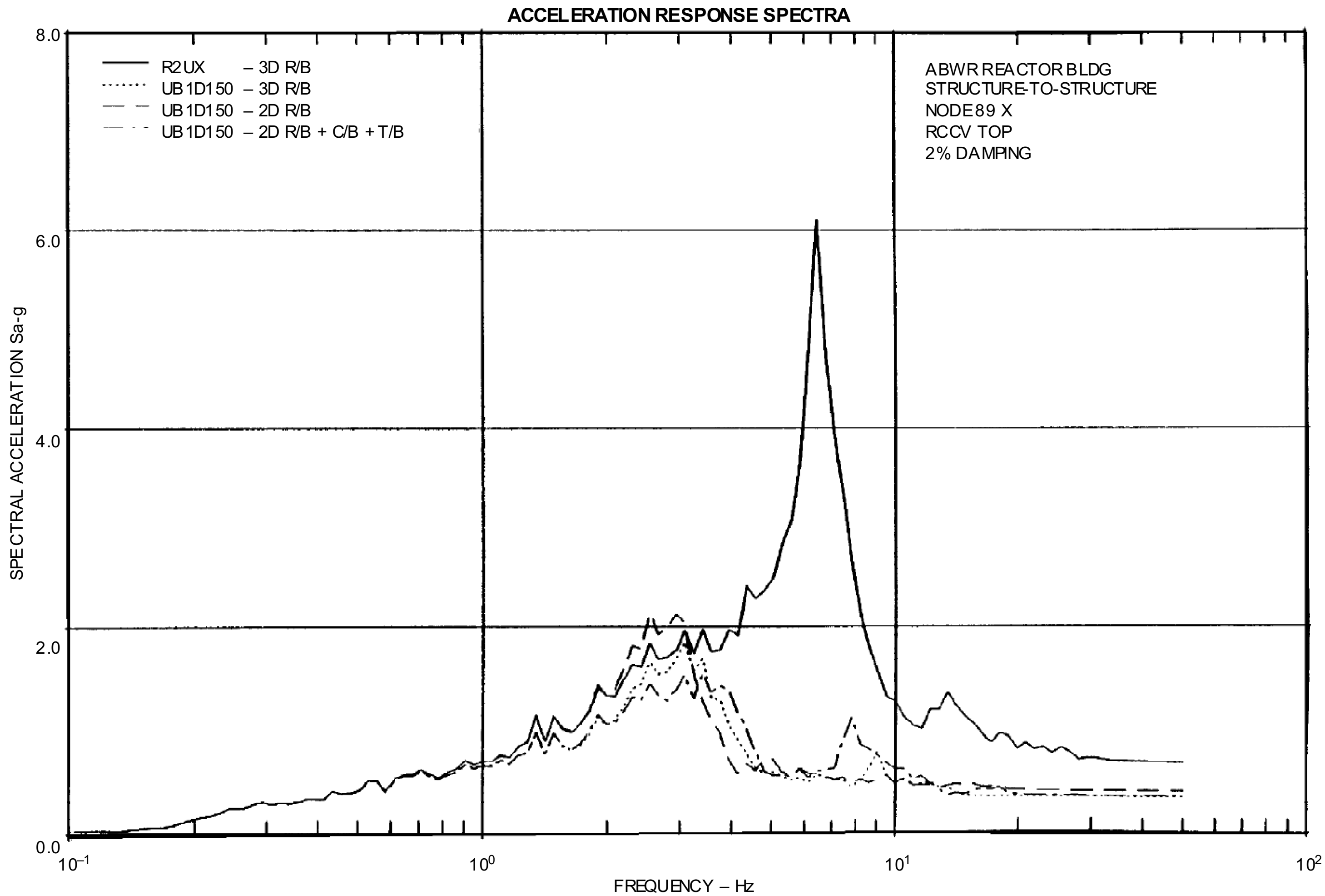


Figure 3A-111 ABWR Reactor Building Structure-to-Structure, Node 89 X, RCCV Top, 2% Damping, UB1D150

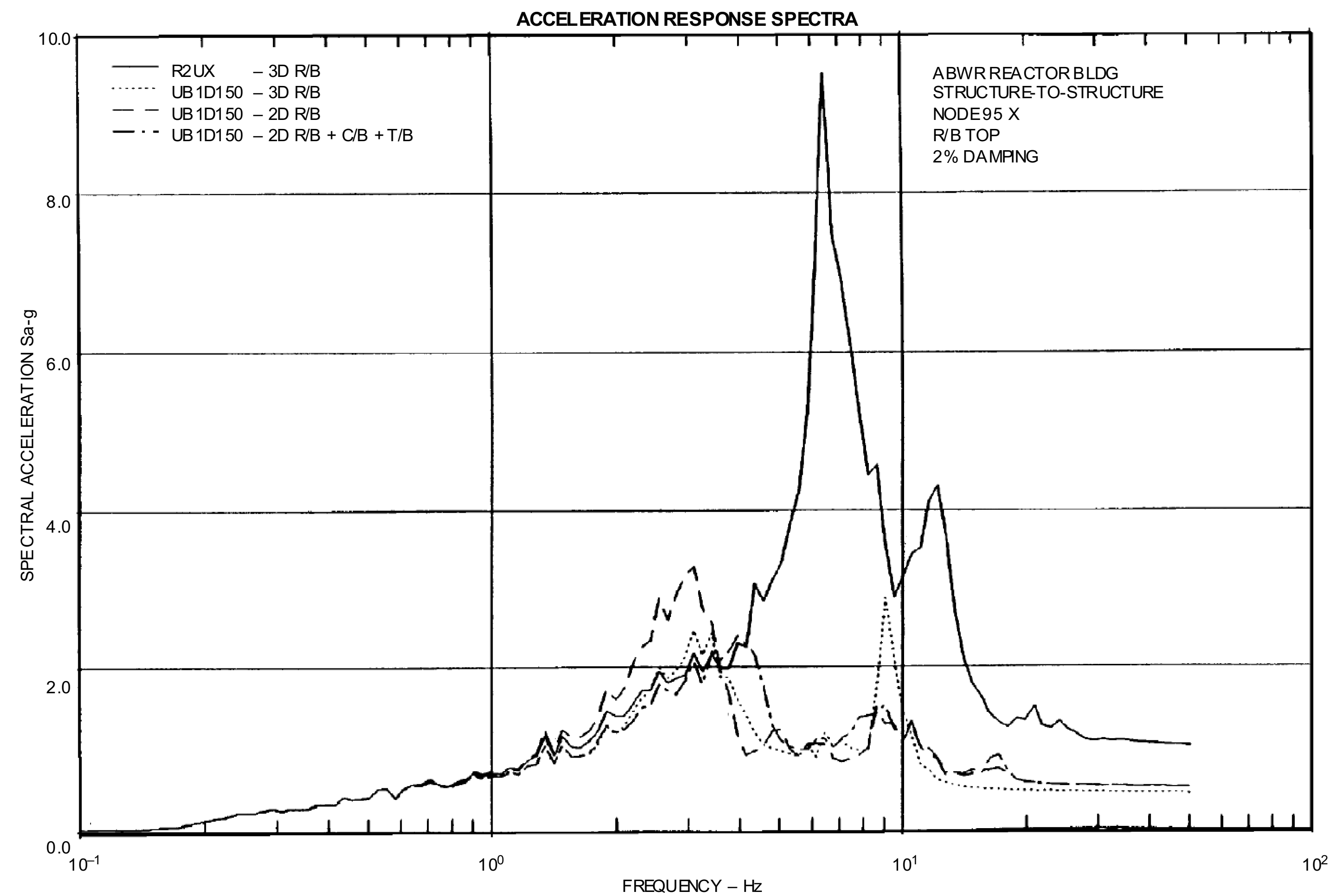


Figure 3A-112 ABWR Reactor Building Structure-to-Structure, Node 95 X, R/B Top, 2% Damping, UB1D150

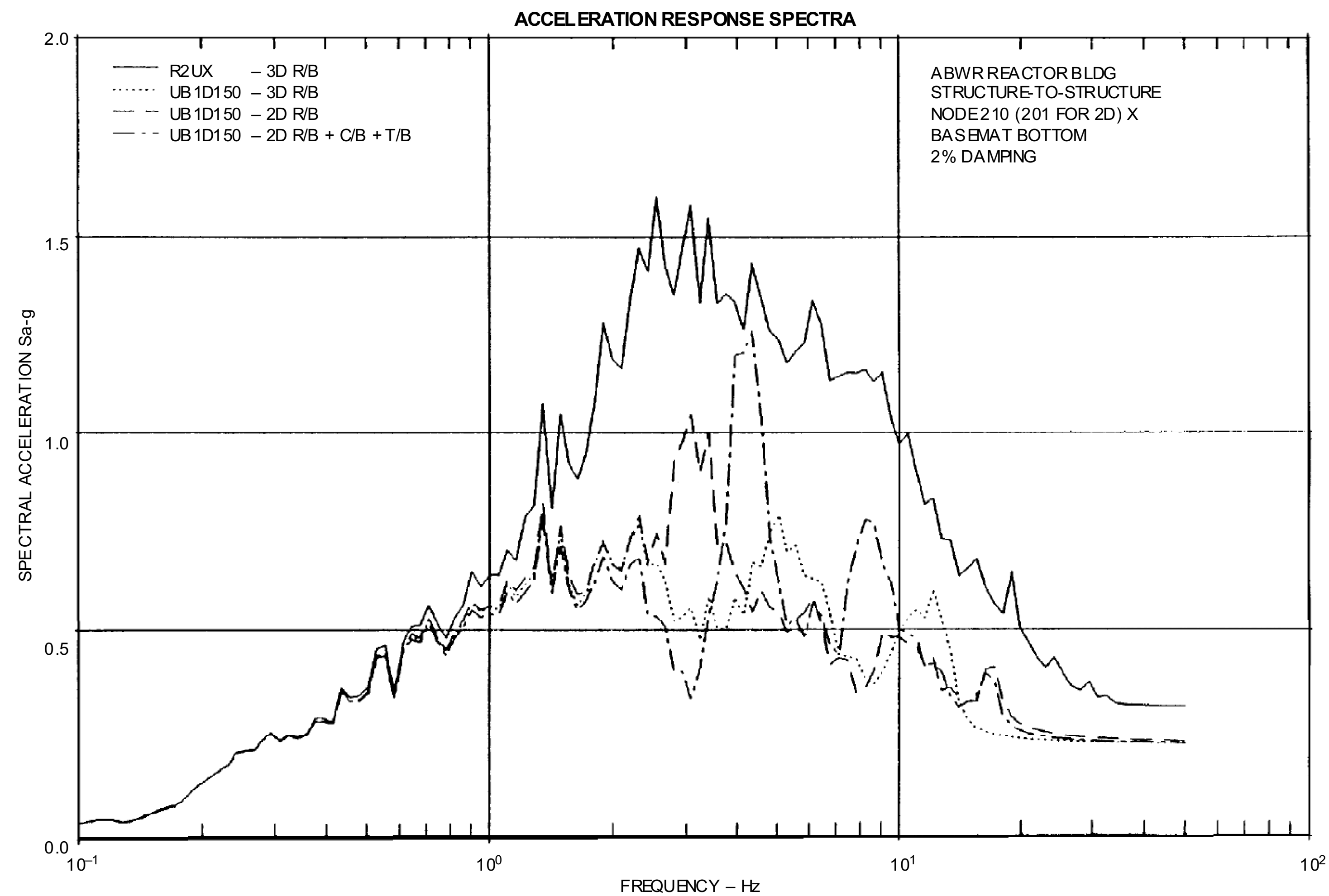


Figure 3A-113
ABWR Reactor Building Structure-to-Structure, Node 210 (201 for 2D) X, Basemat Bottom, 2% Damping, UB1D150

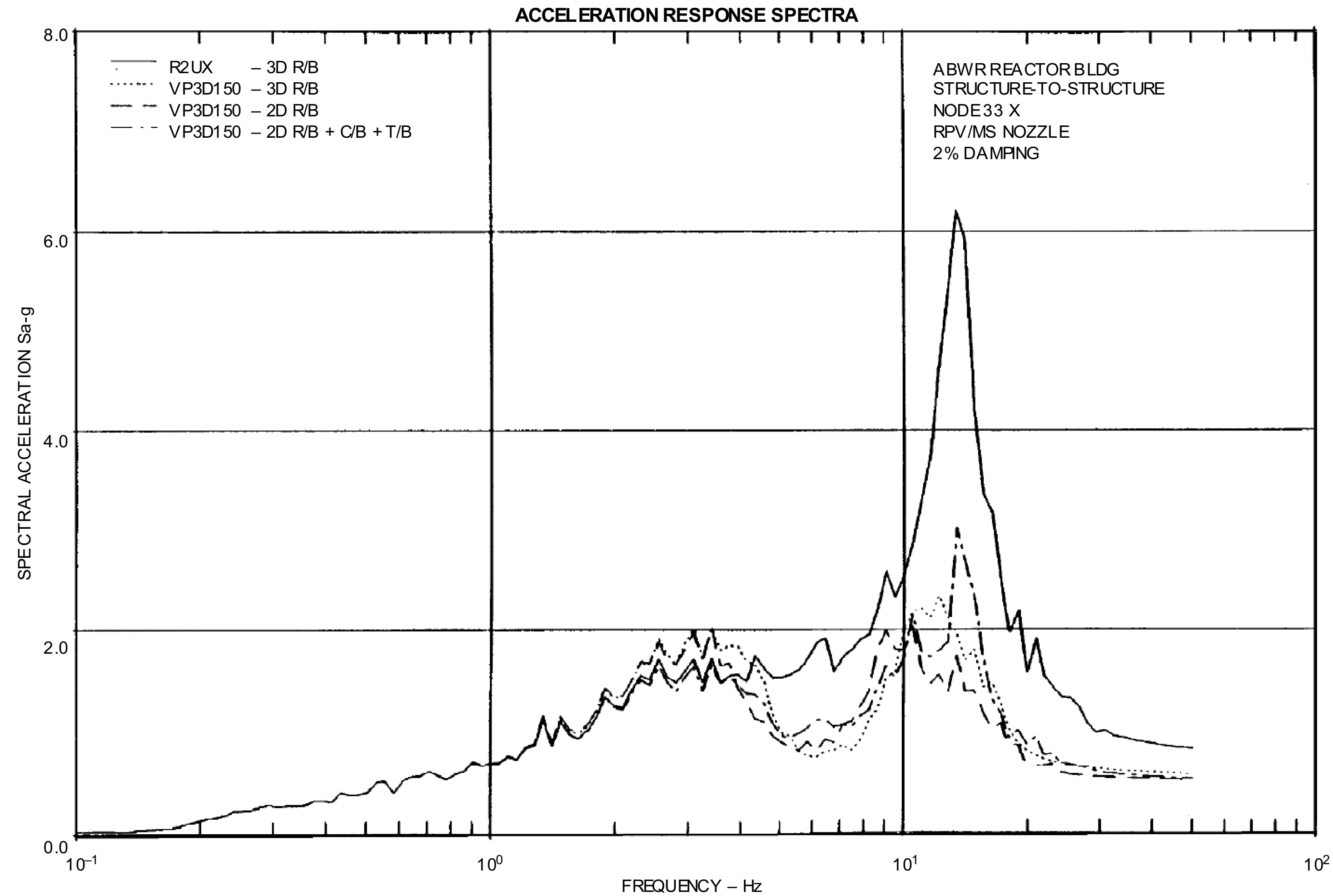


Figure 3A-114 ABWR Reactor Building Structure-to-Structure, Node 33 X, RPV/MS Nozzle, 2% Damping, VP3D150

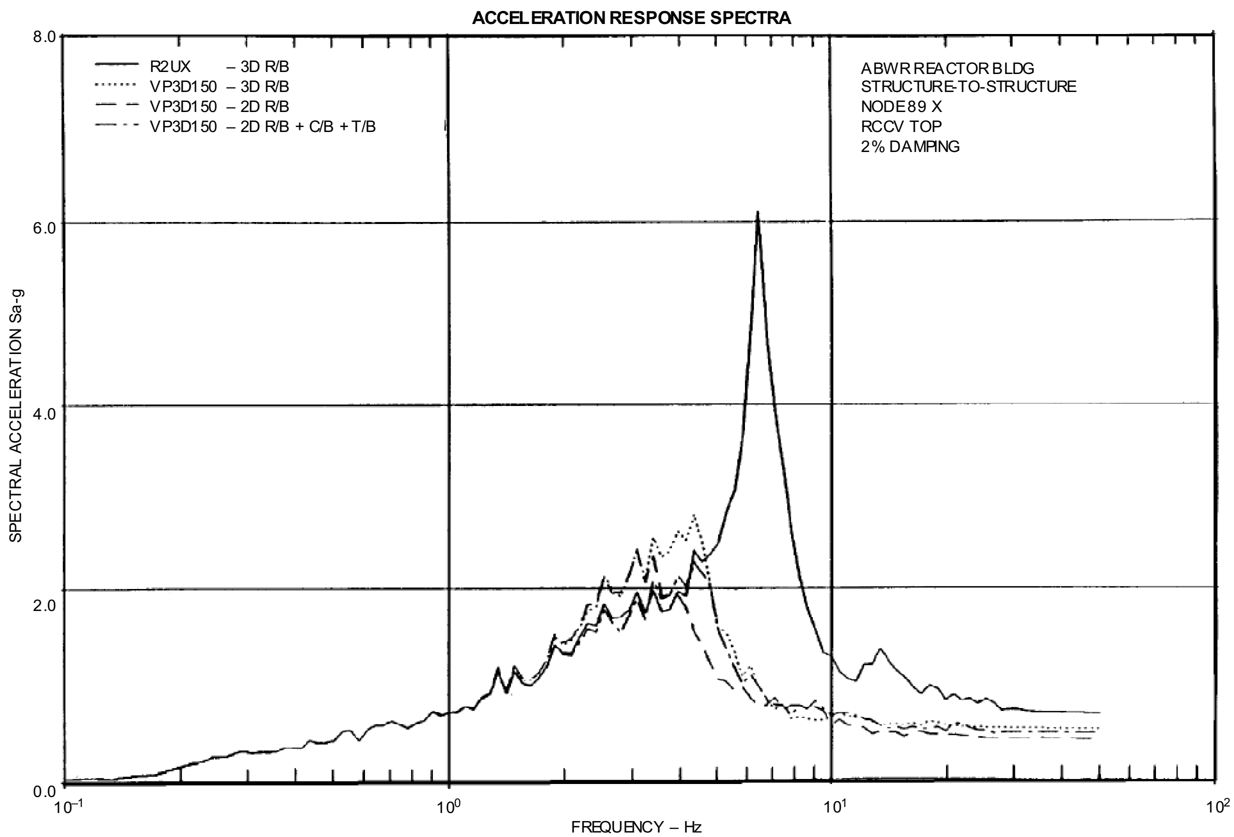


Figure 3A-115 ABWR Reactor Bldg. Structure-to-Structure, Node 89 X RCCV Top, 2% Damping, VP3D150

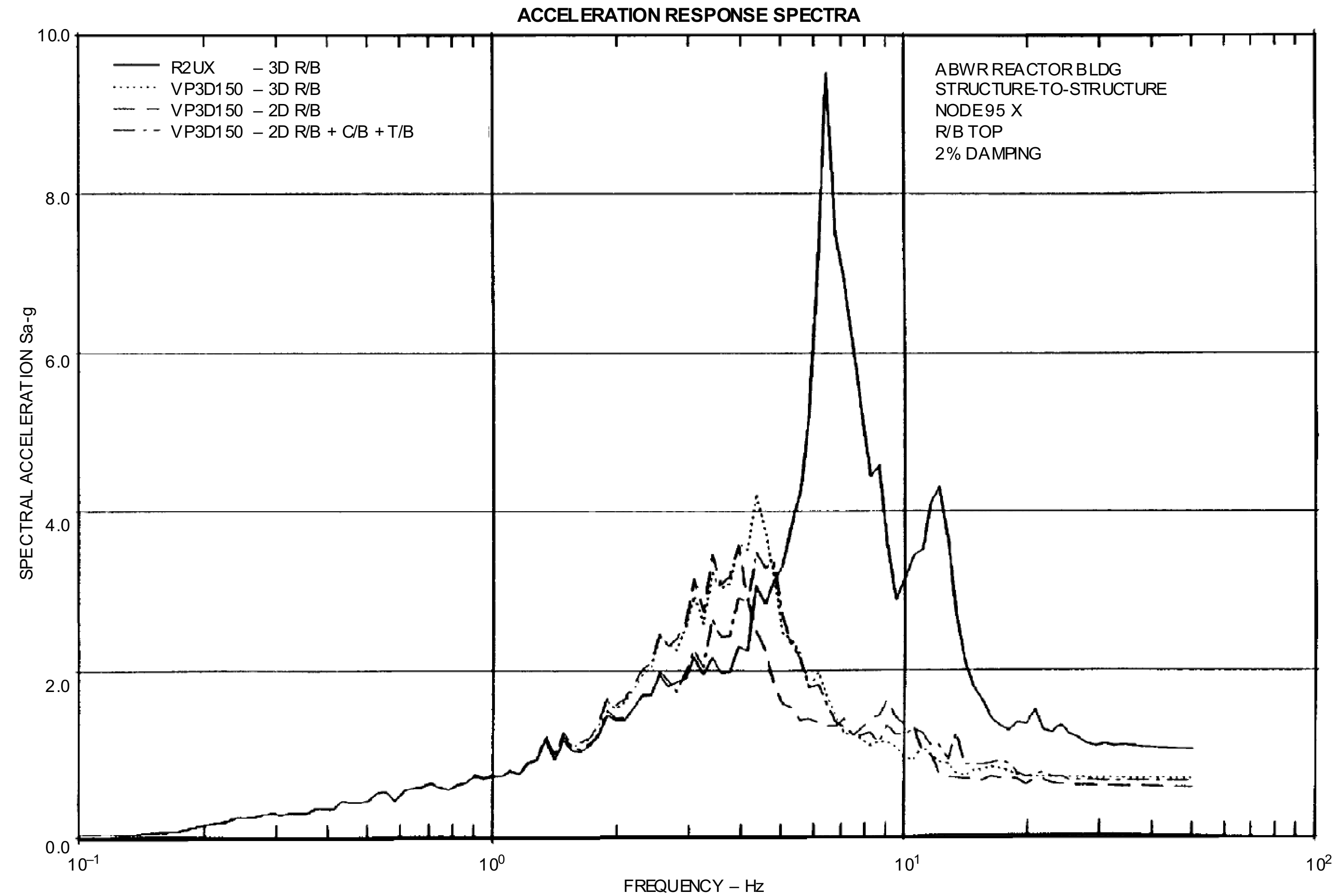


Figure 3A-116 ABWR Reactor Bldg. Structure-to-Structure, Node 95 X R/B Top, 2% Damping, VP3D150

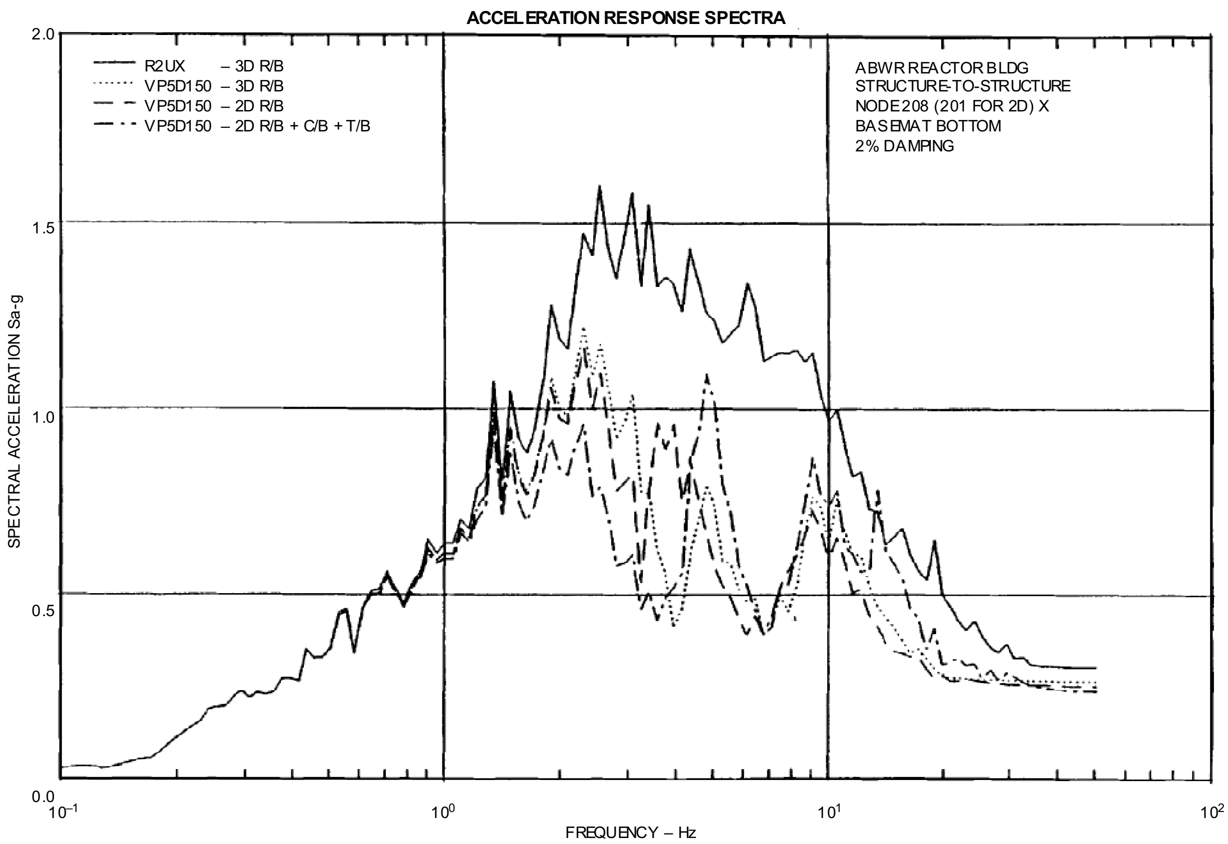


Figure 3A-117 ABWR Reactor Bldg. Structure-to-Structure, Node 208 (201 for 2 D) X Basemat Bottom, 2% Damping, VP5D150

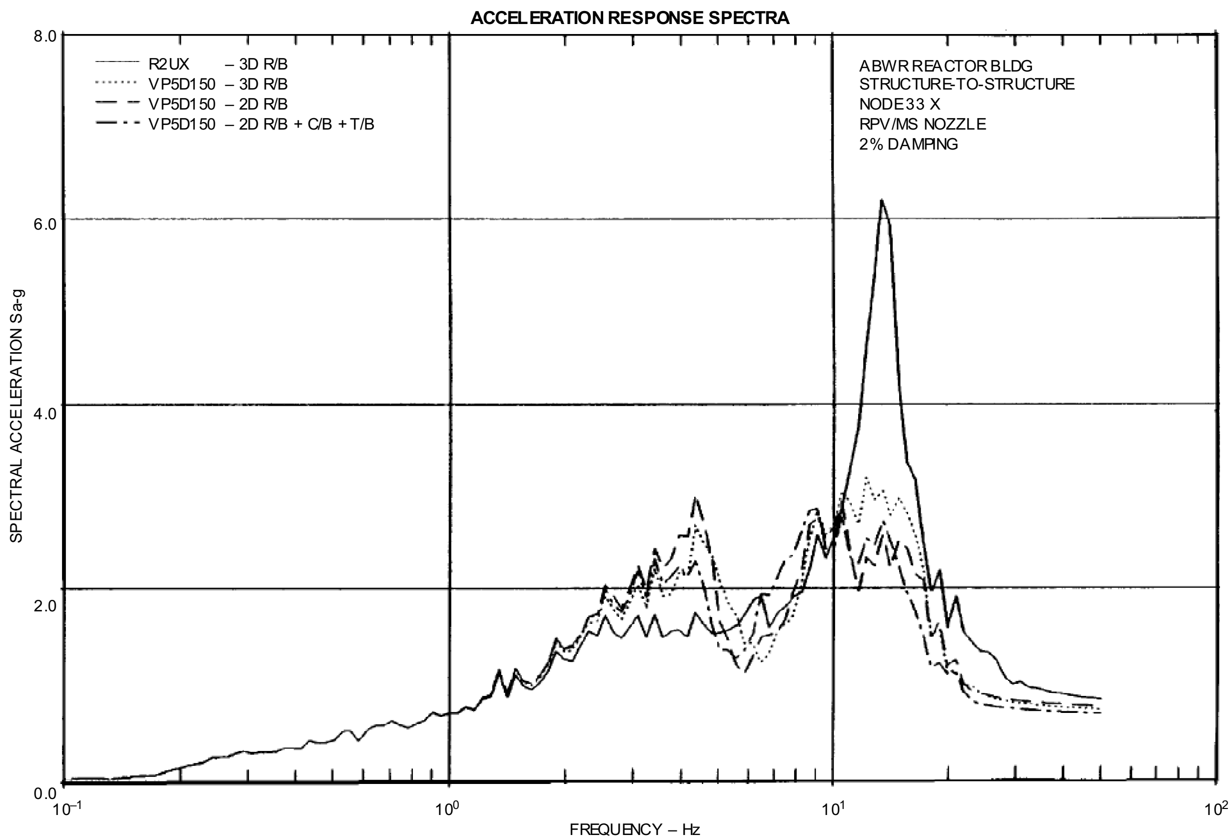


Figure 3A-118 ABWR Reactor Bldg. Structure-to-Structure, Node 33X RPV/MS Nozzle, 2% Damping, VP5D150

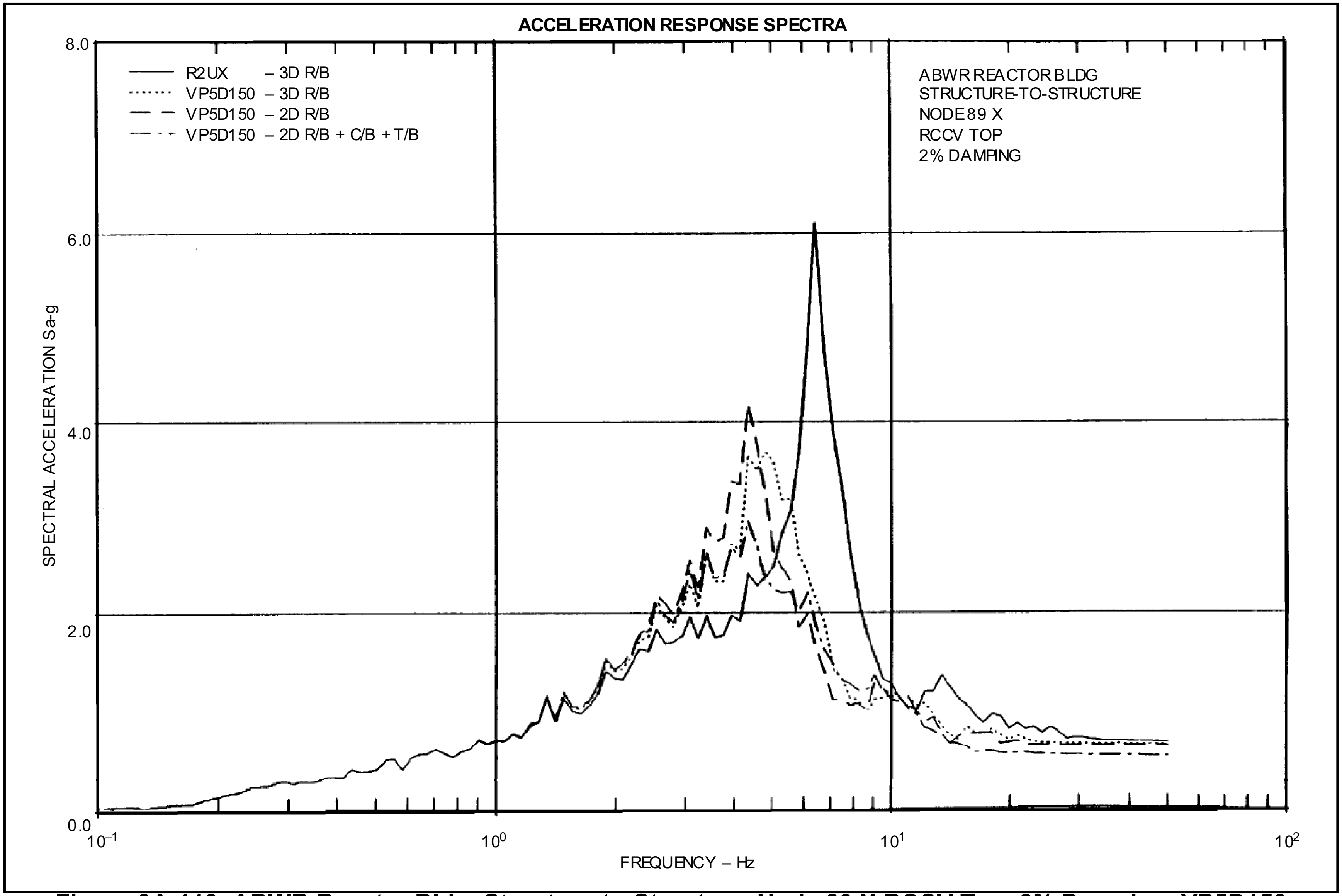


Figure 3A-119 ABWR Reactor Bldg. Structure-to-Structure, Node 89 X RCCV Top, 2% Damping, VP5D150

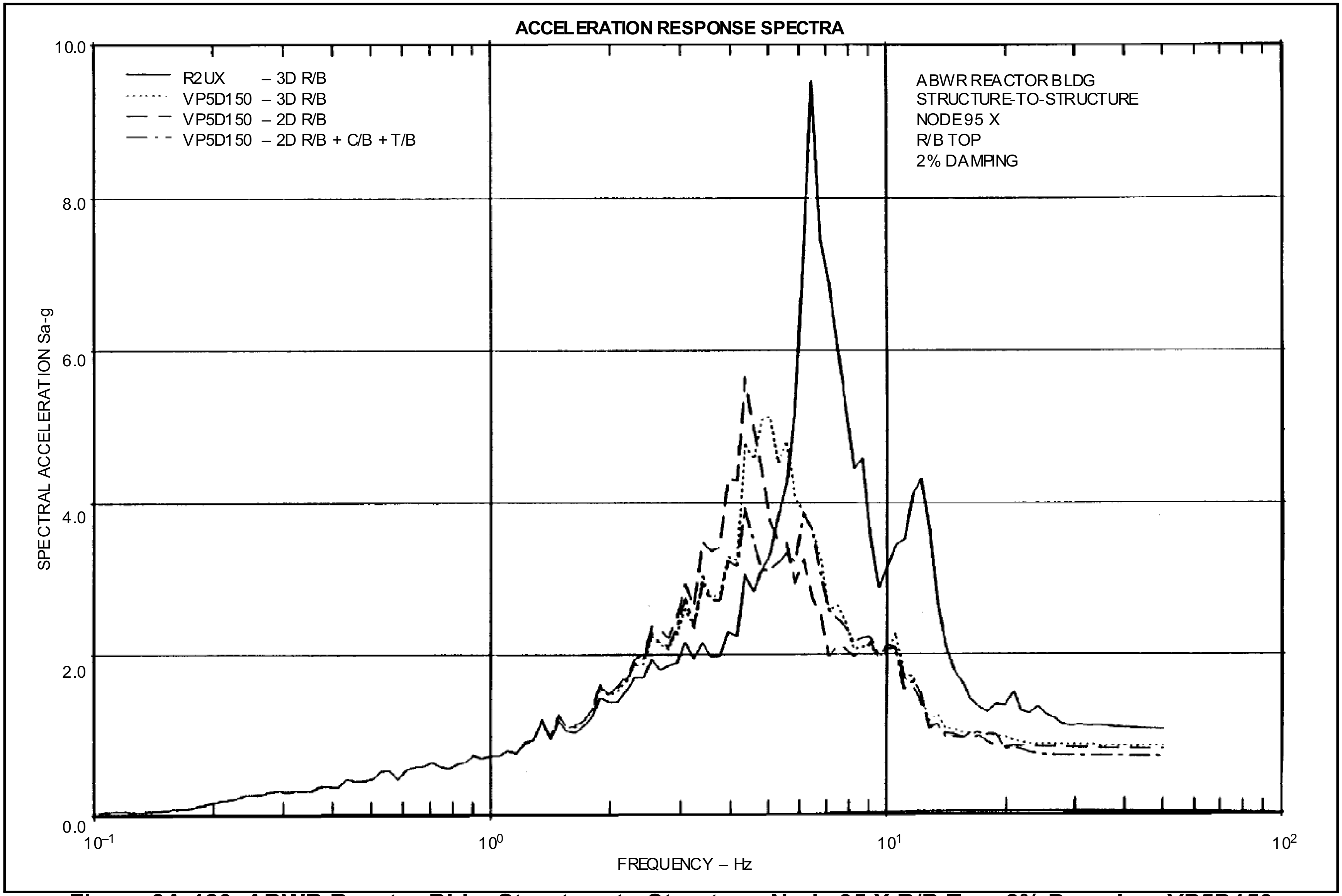


Figure 3A-120 ABWR Reactor Bldg. Structure-to-Structure, Node 95 X R/B Top, 2% Damping, VP5D150

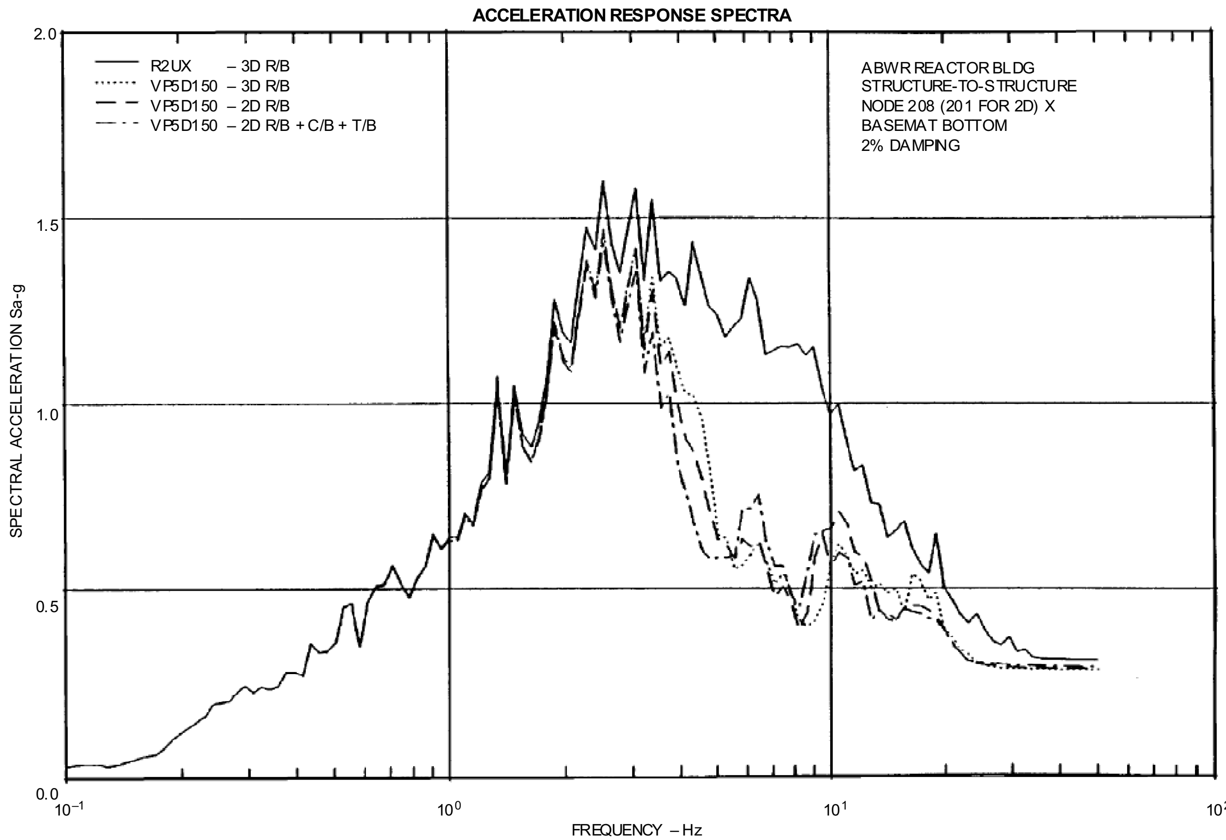


Figure 3A-121 ABWR Reactor Bldg. Structure-to-Structure, Node 208 (201 for 2 D) X, Basemat Bottom, 2% Damping, VP3D150

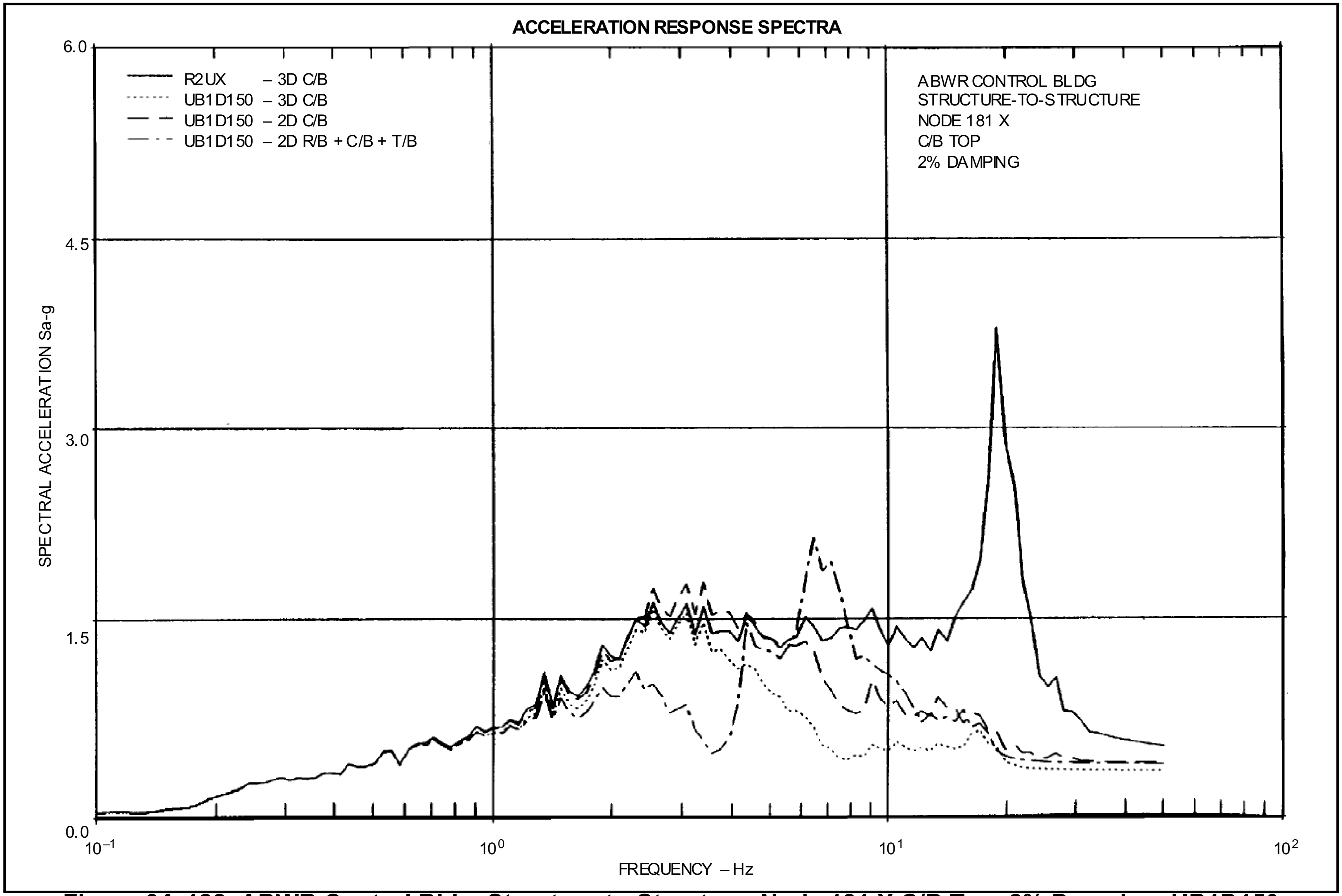


Figure 3A-122 ABWR Control Bldg. Structure-to-Structure, Node 181 X C/B Top, 2% Damping, UB1D150

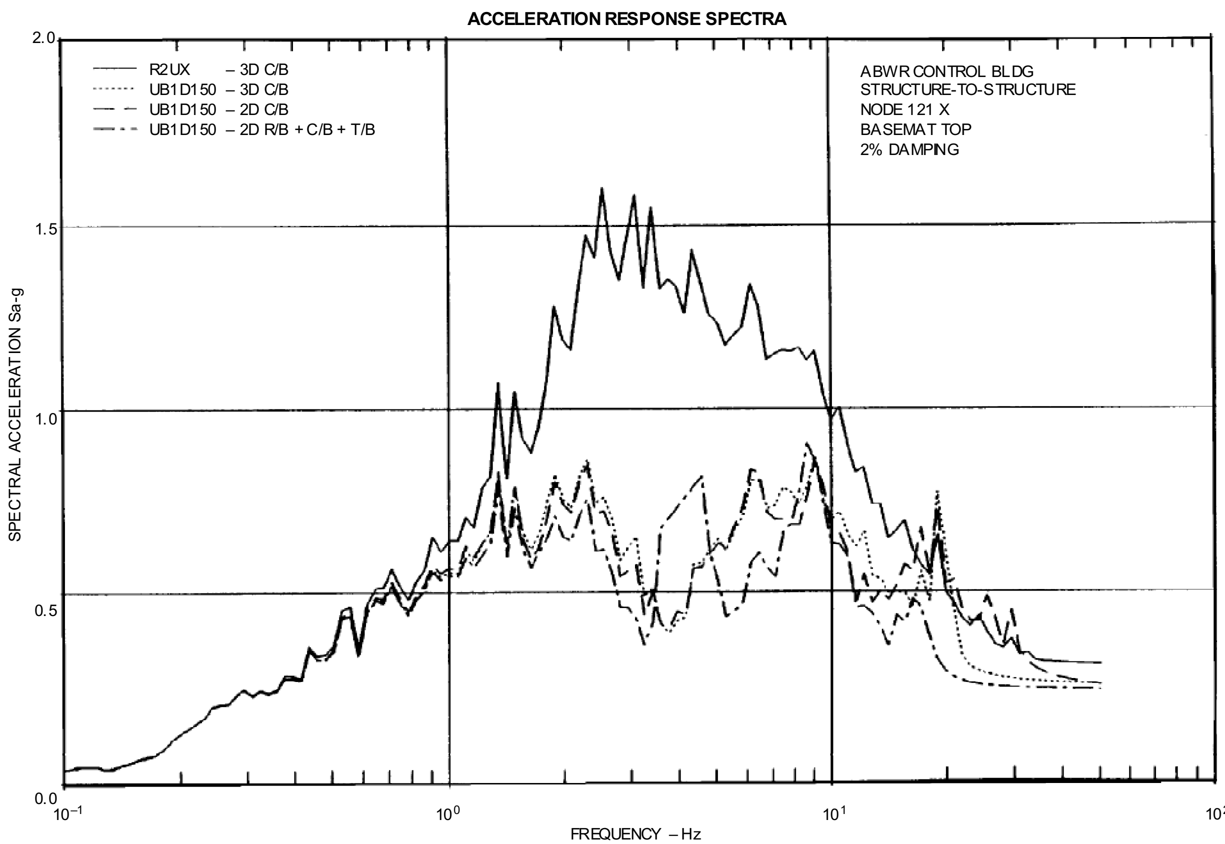


Figure 3A-123 ABWR Control Bldg. Structure-to-Structure, Node 121 X Basemat Top, 2% Damping, UB1D150

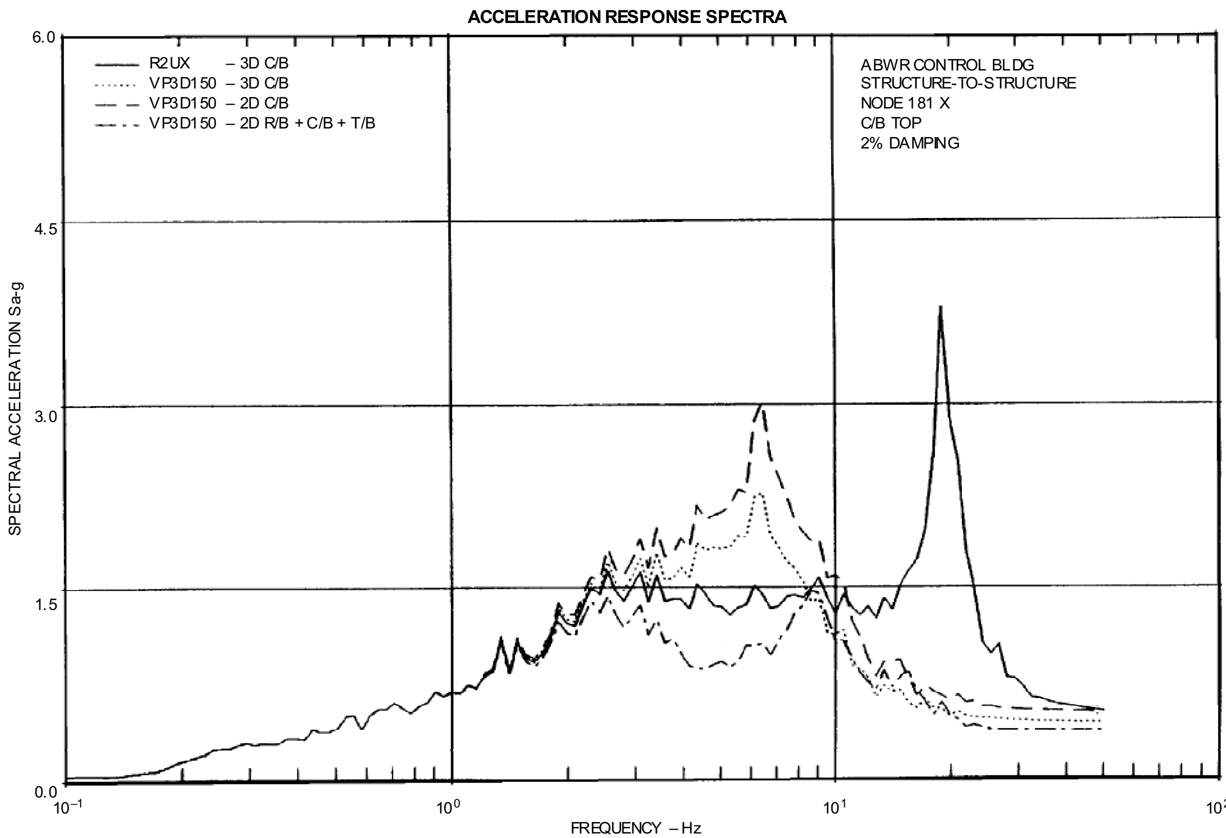


Figure 3A-124 ABWR Control Bldg. Structure-to-Structure, Node 181 X C/B Top, 2% Damping, VP3D150

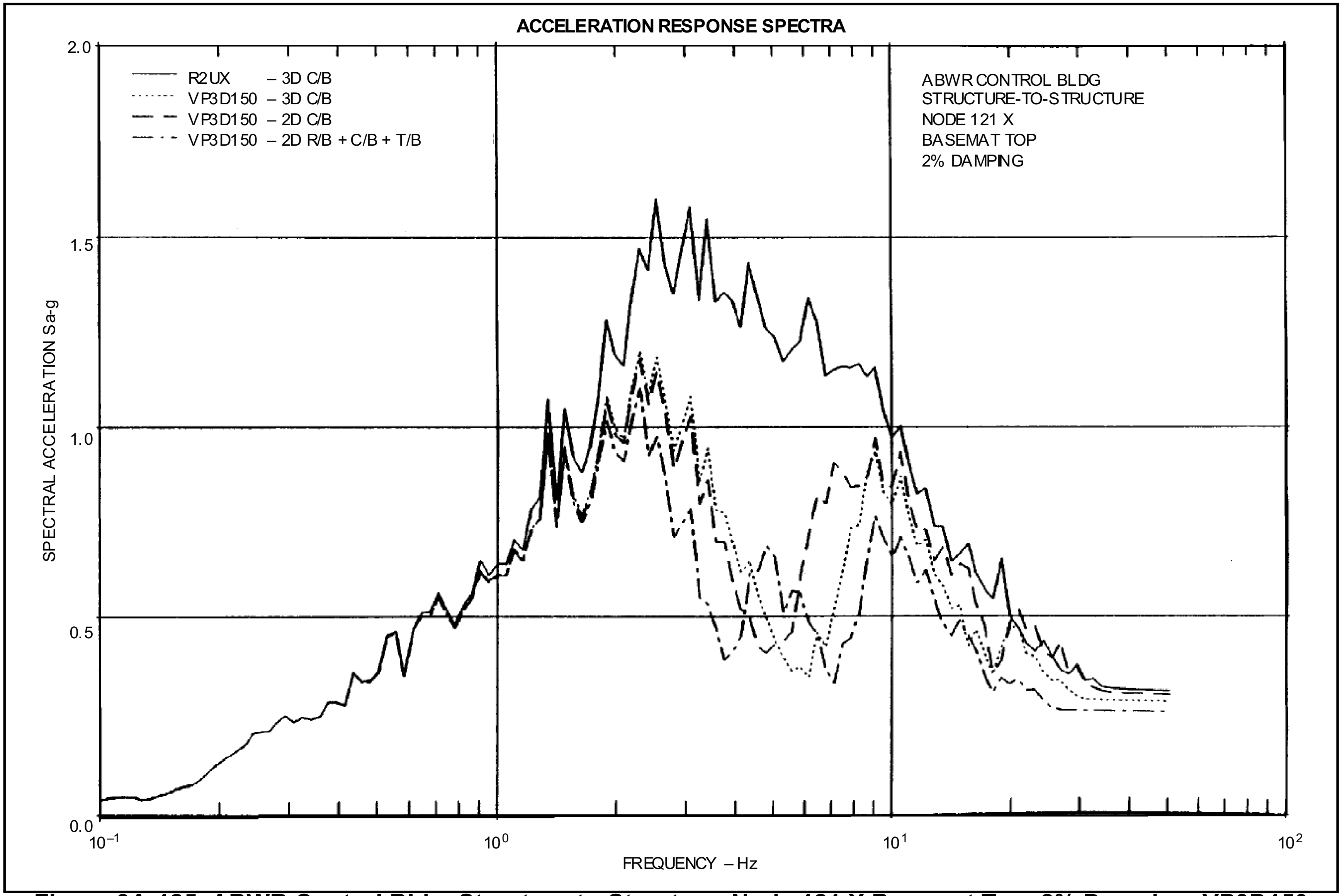


Figure 3A-125 ABWR Control Bldg. Structure-to-Structure, Node 121 X Basemat Top, 2% Damping, VP3D150

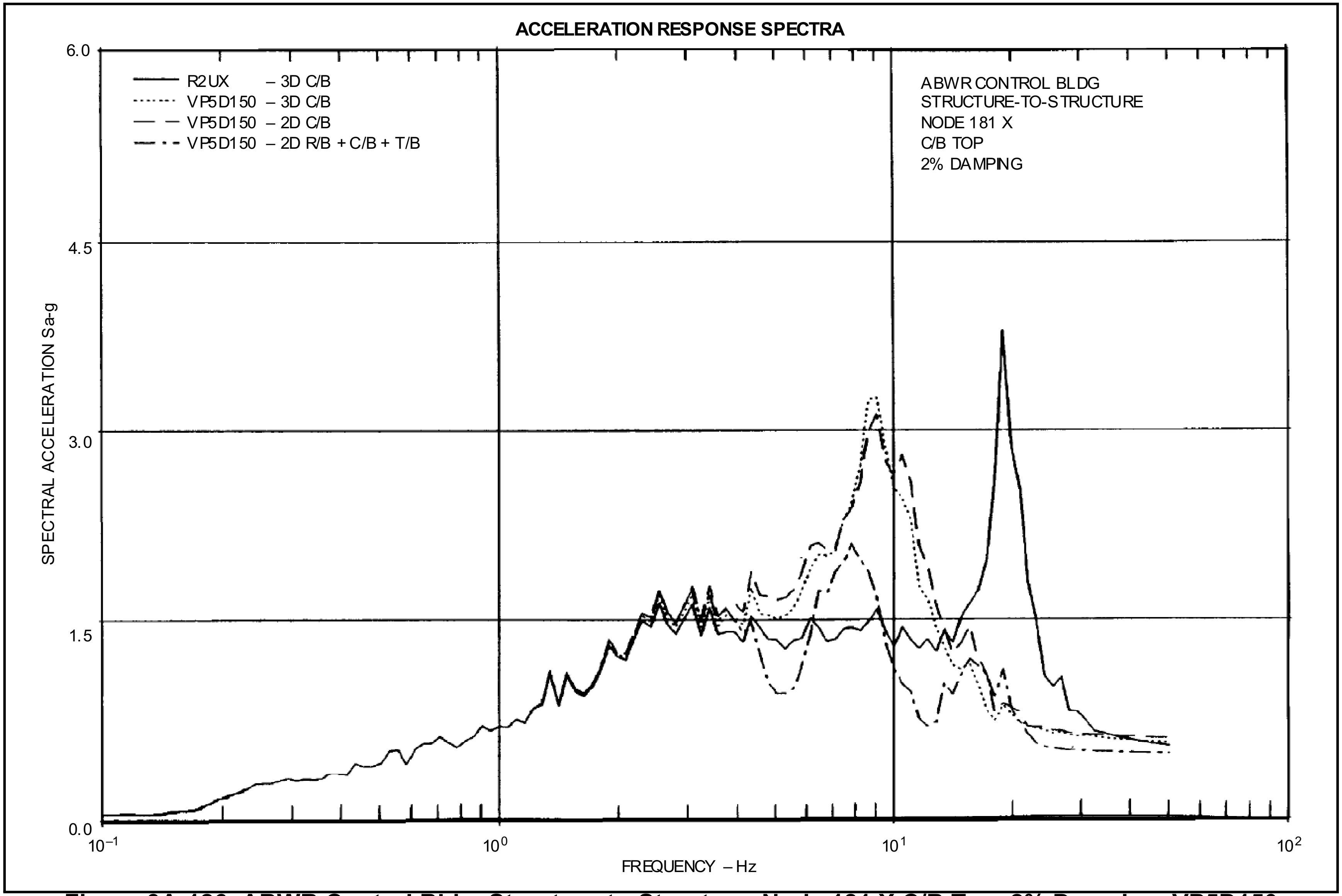


Figure 3A-126 ABWR Control Bldg. Structure-to-Structure, Node 181 X C/B Top, 2% Damping, VP5D150

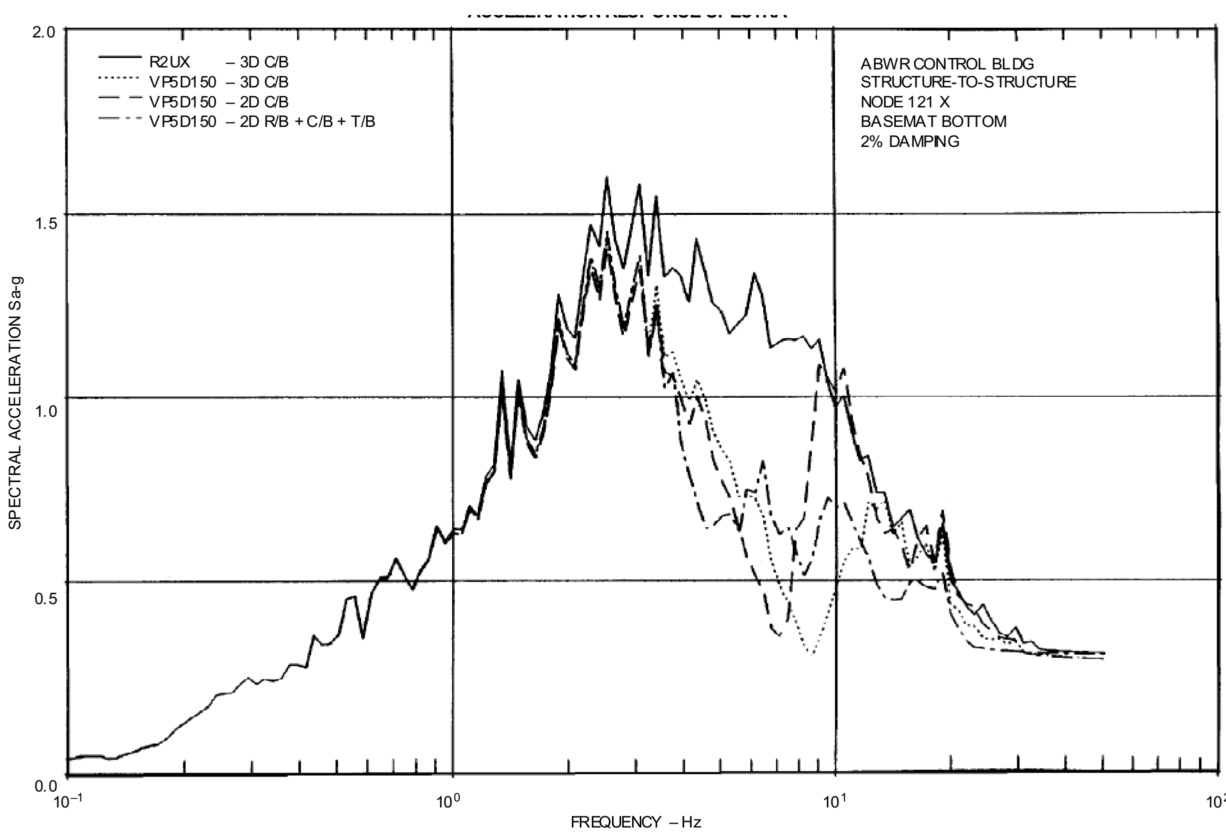


Figure 3A-127 ABWR Control Bldg. Structure-to-Structure, Node 121 X Basemat Bottom, 2% Damping, VP5D150

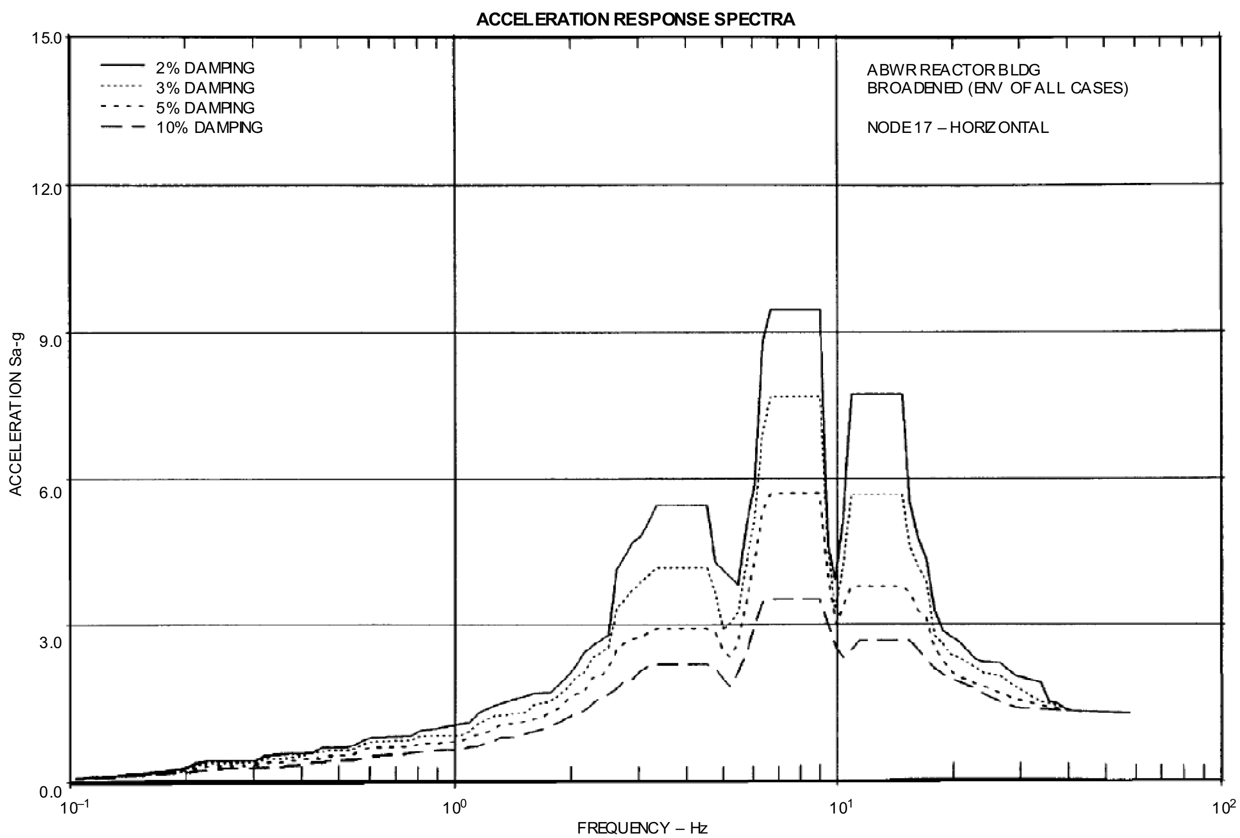


Figure 3A-128 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 17–Horizontal

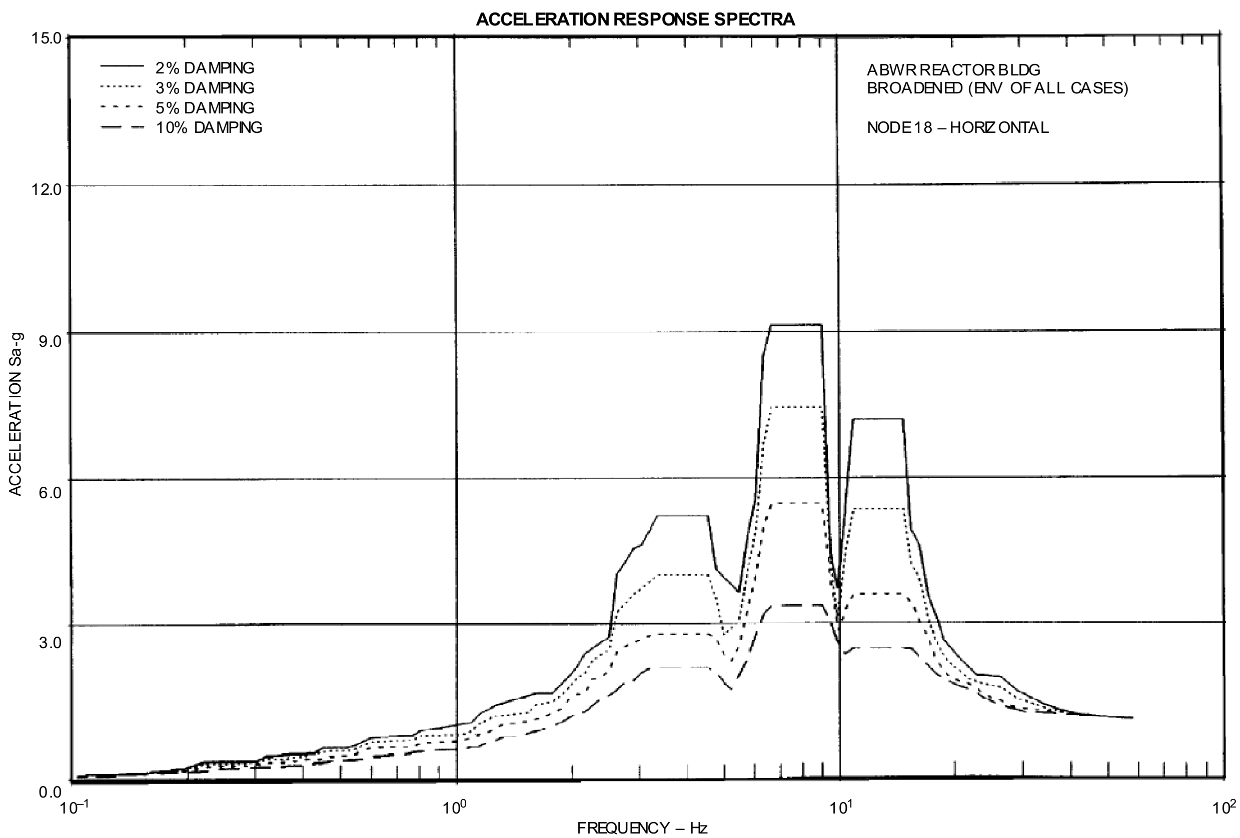


Figure 3A-129 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 18–Horizontal

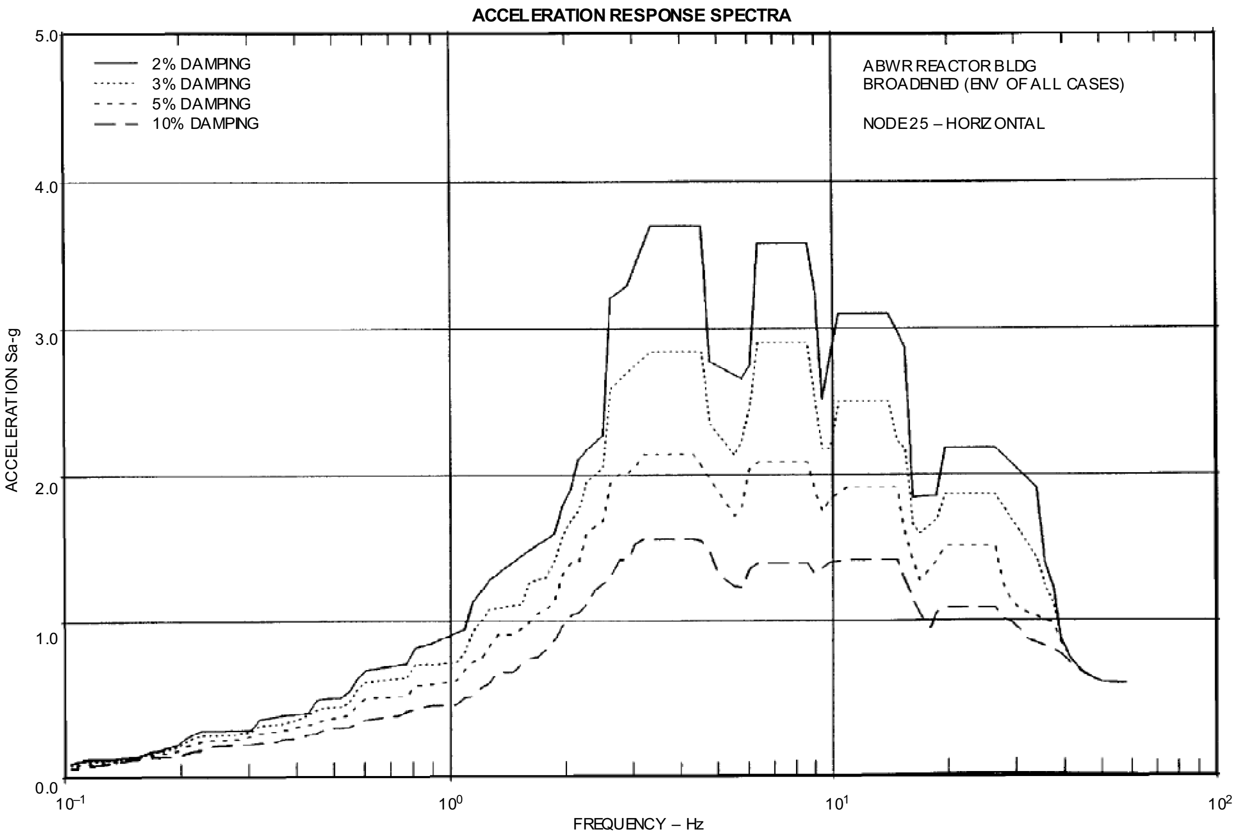


Figure 3A-130 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 25–Horizontal

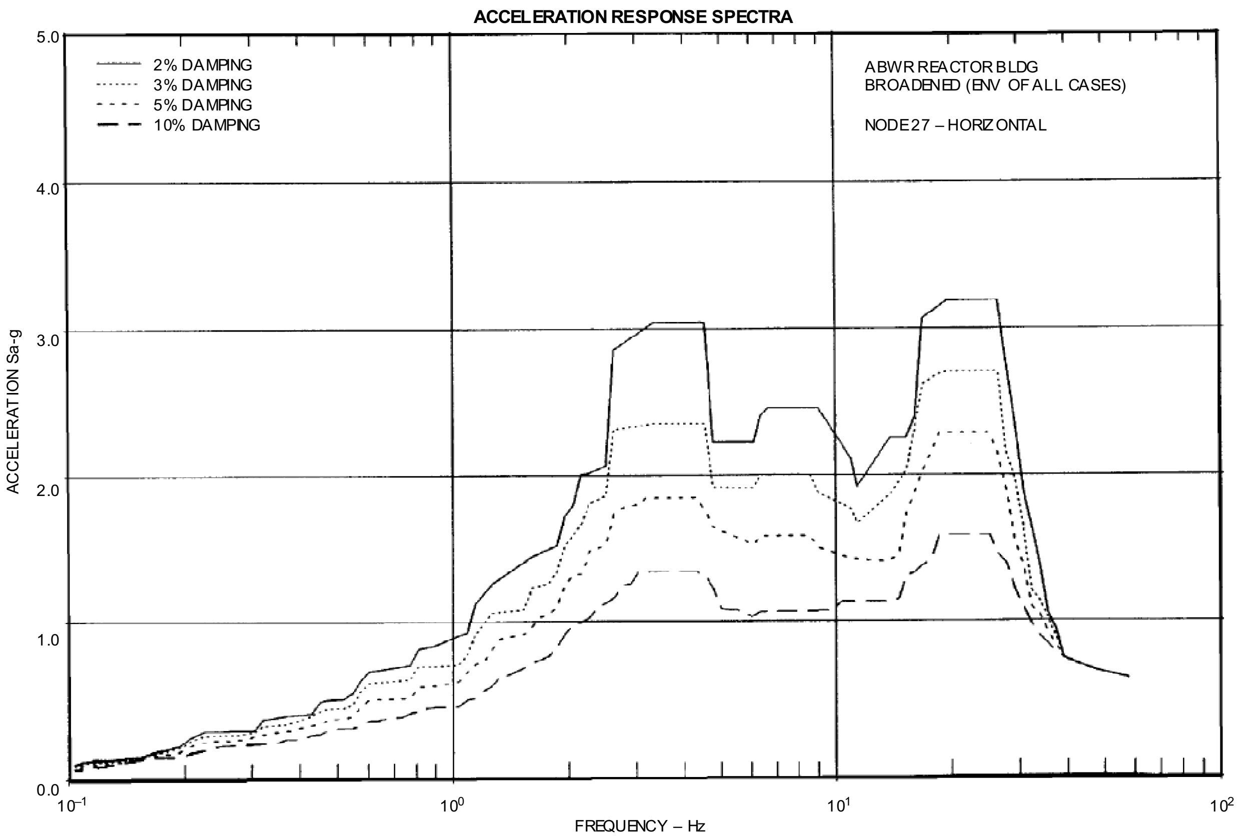


Figure 3A-131 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 27–Horizontal

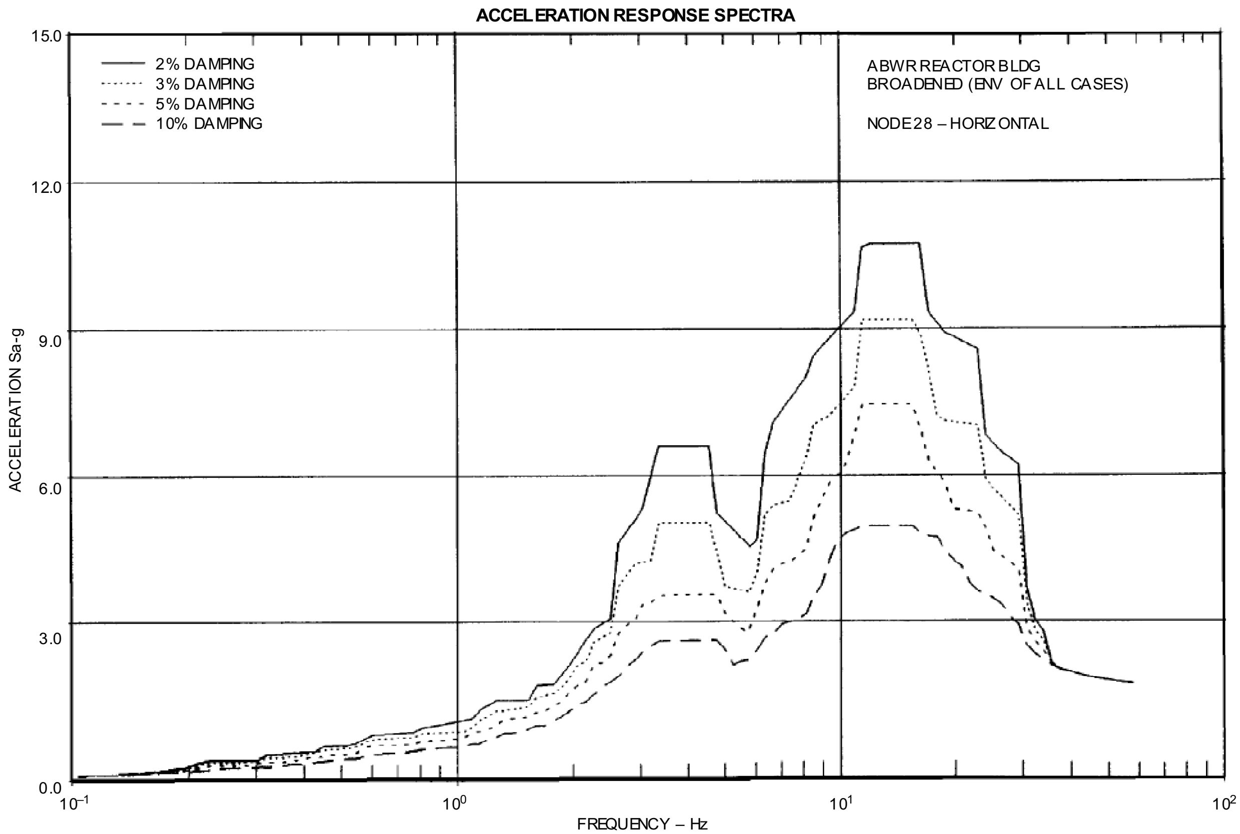


Figure 3A-132 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 28–Horizontal

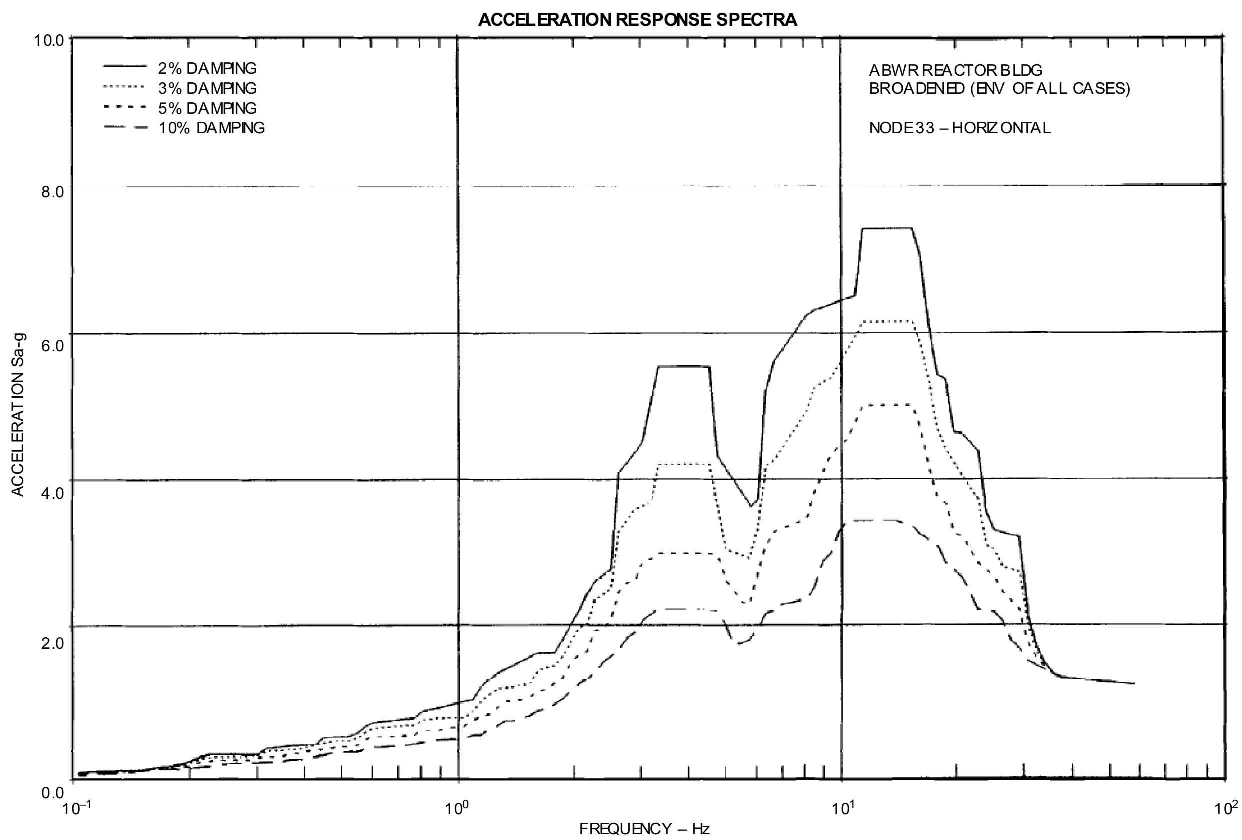


Figure 3A-133 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 33–Horizontal

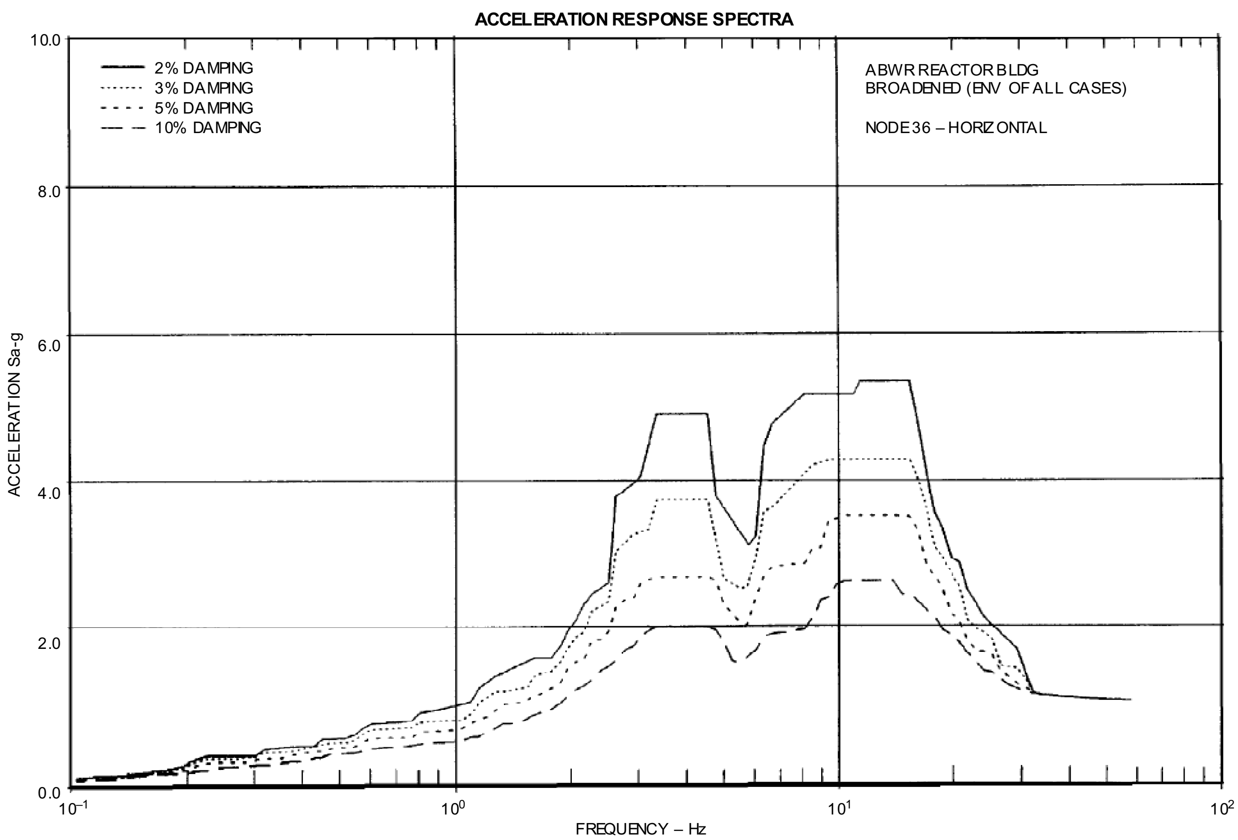


Figure 3A-134 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 36–Horizontal

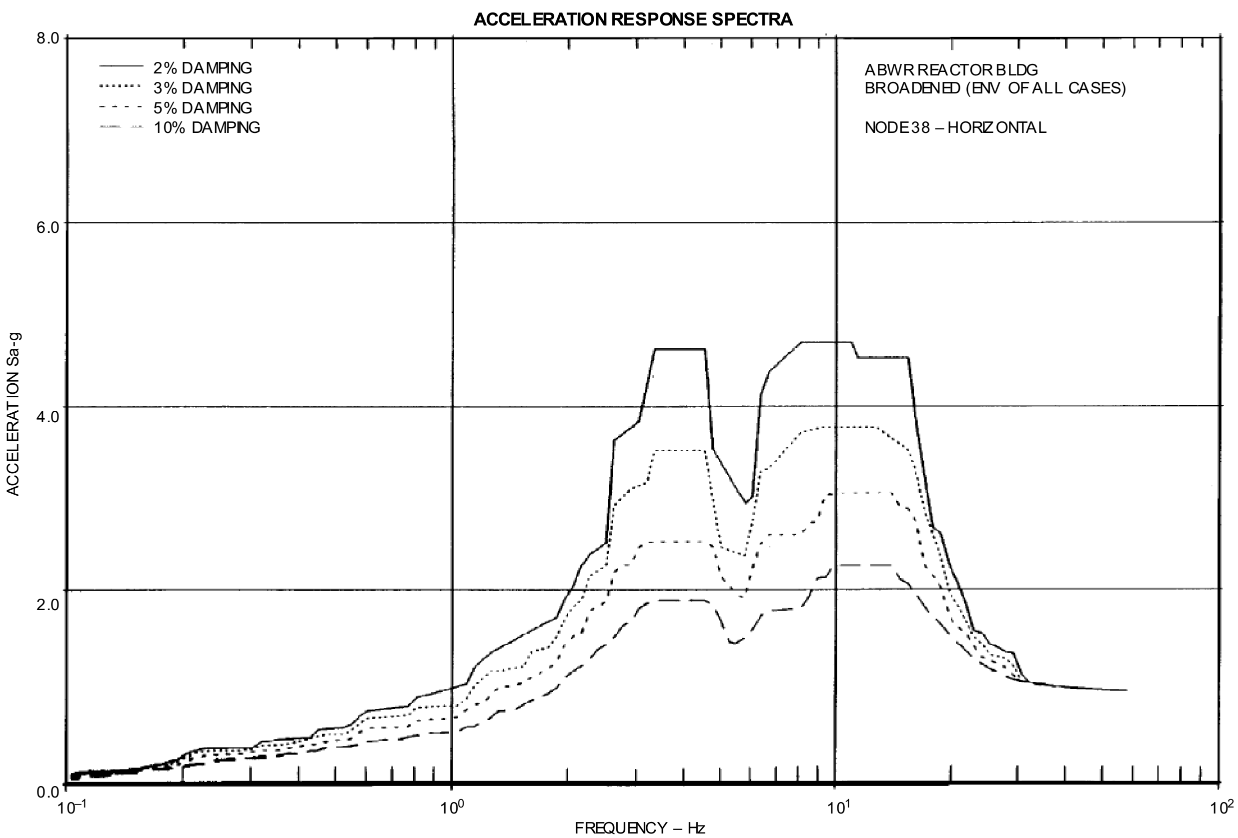


Figure 3A-135 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 38–Horizontal

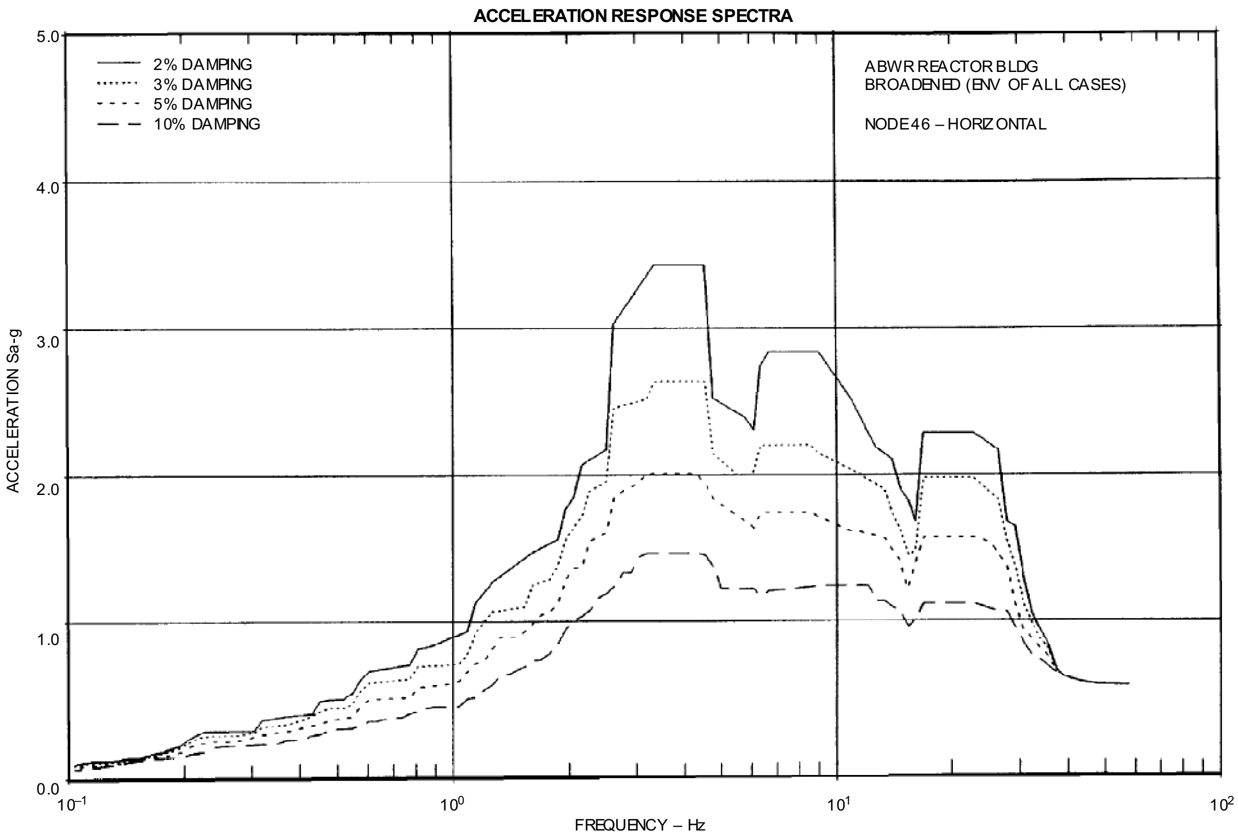


Figure 3A-136 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 46–Horizontal

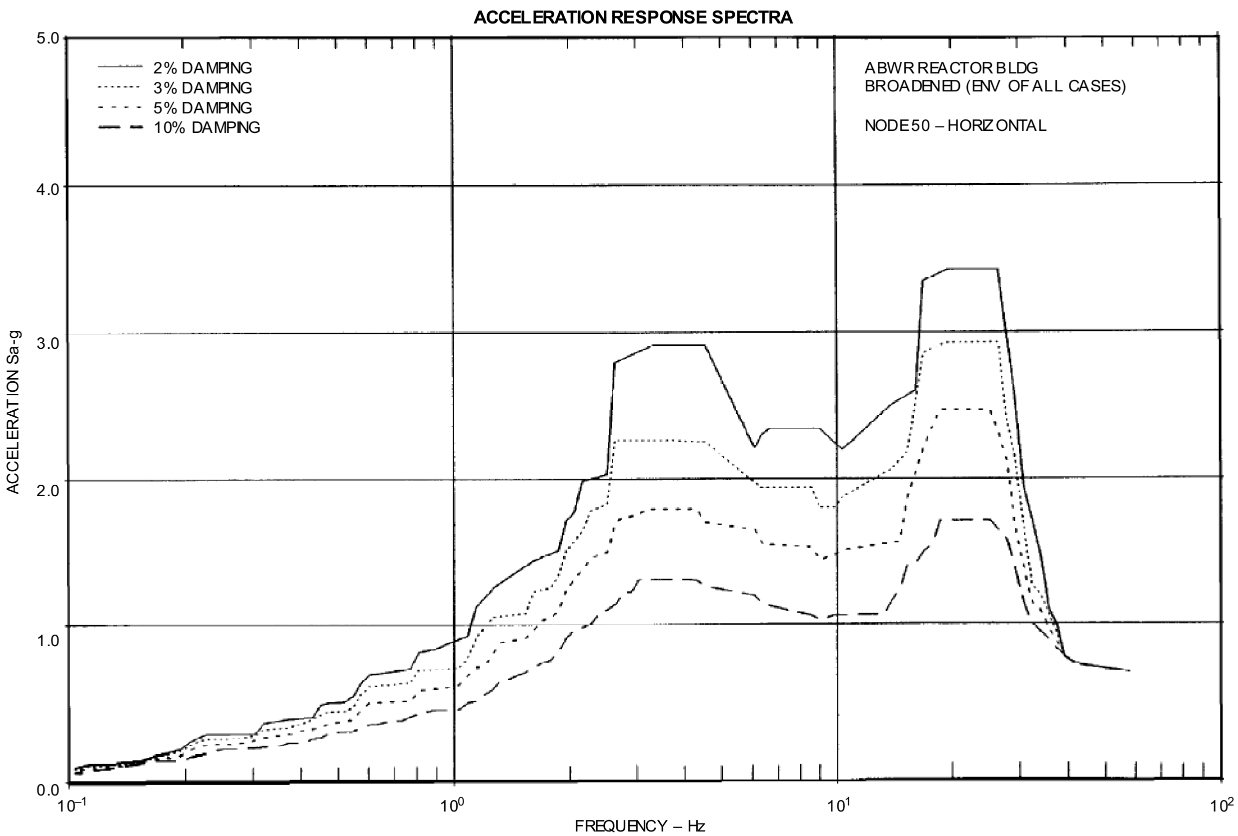
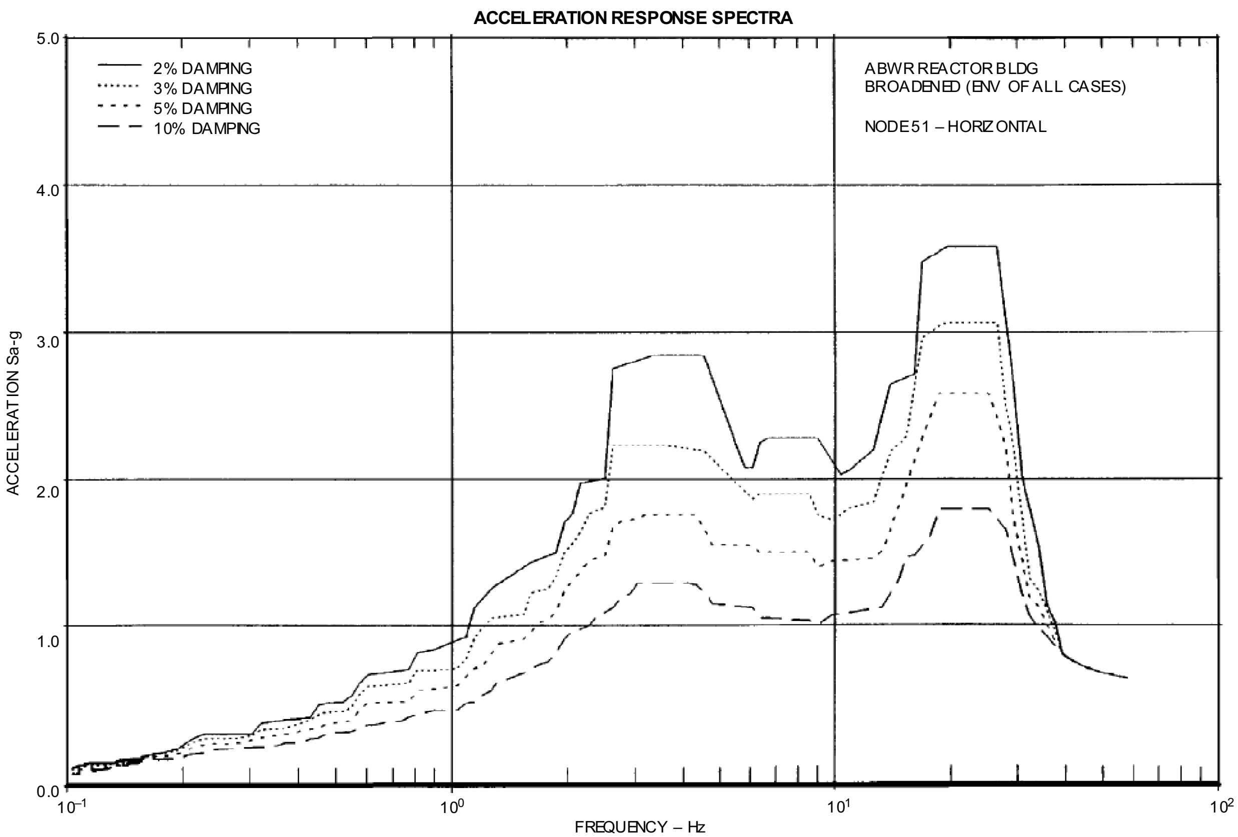


Figure 3A-137 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 50–Horizontal



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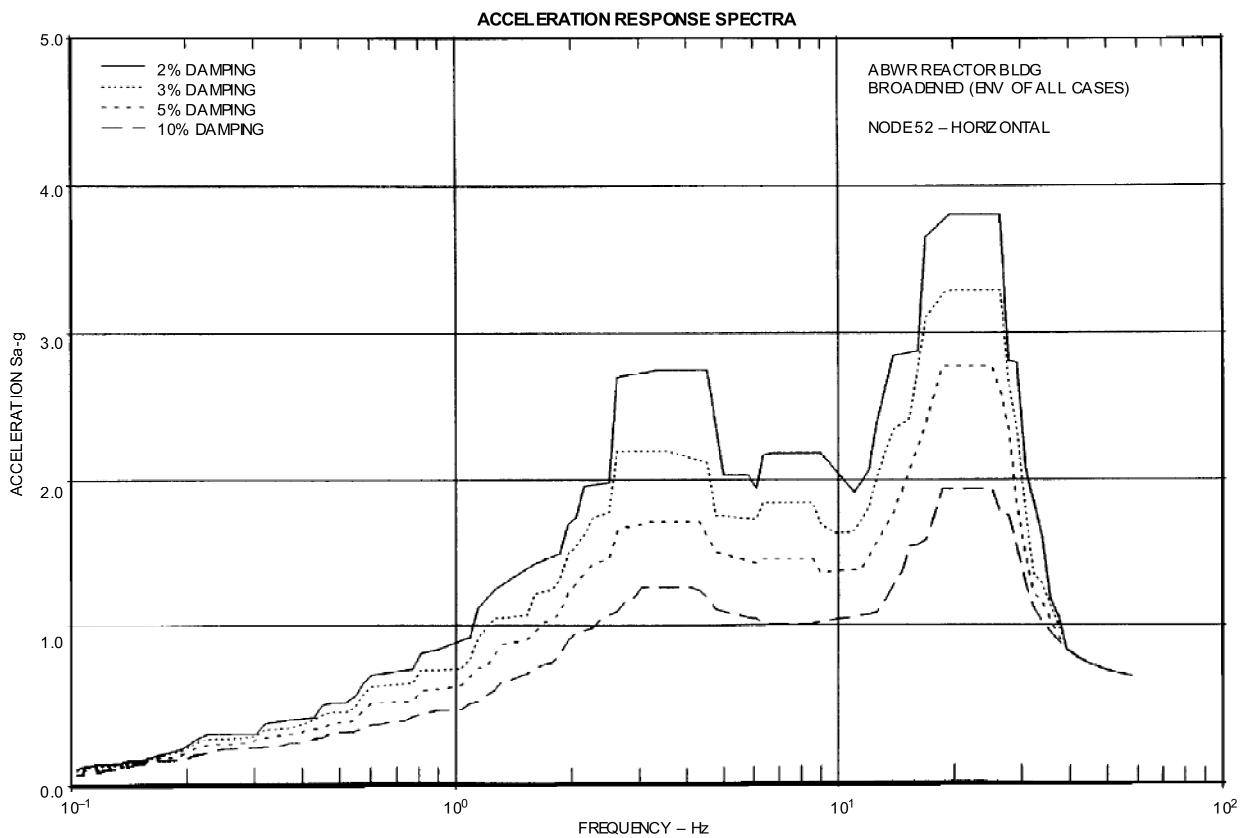

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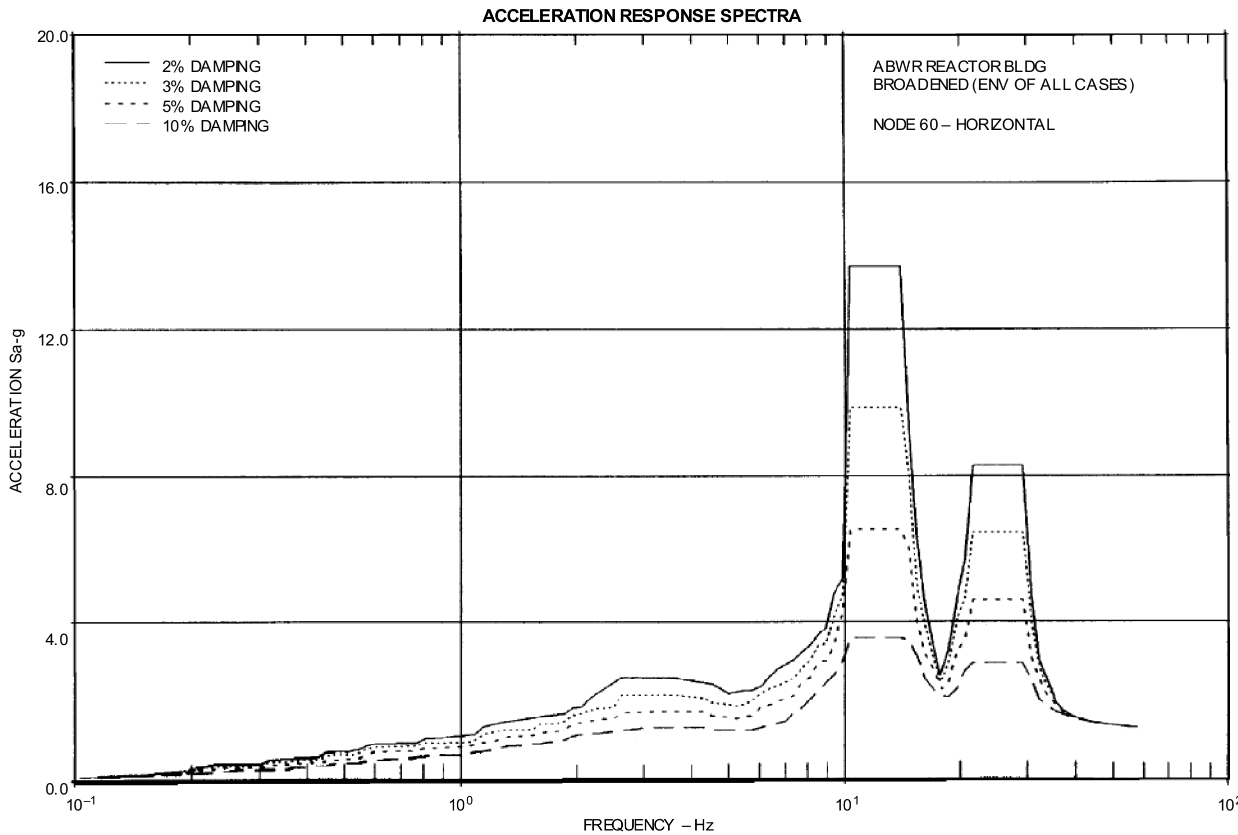

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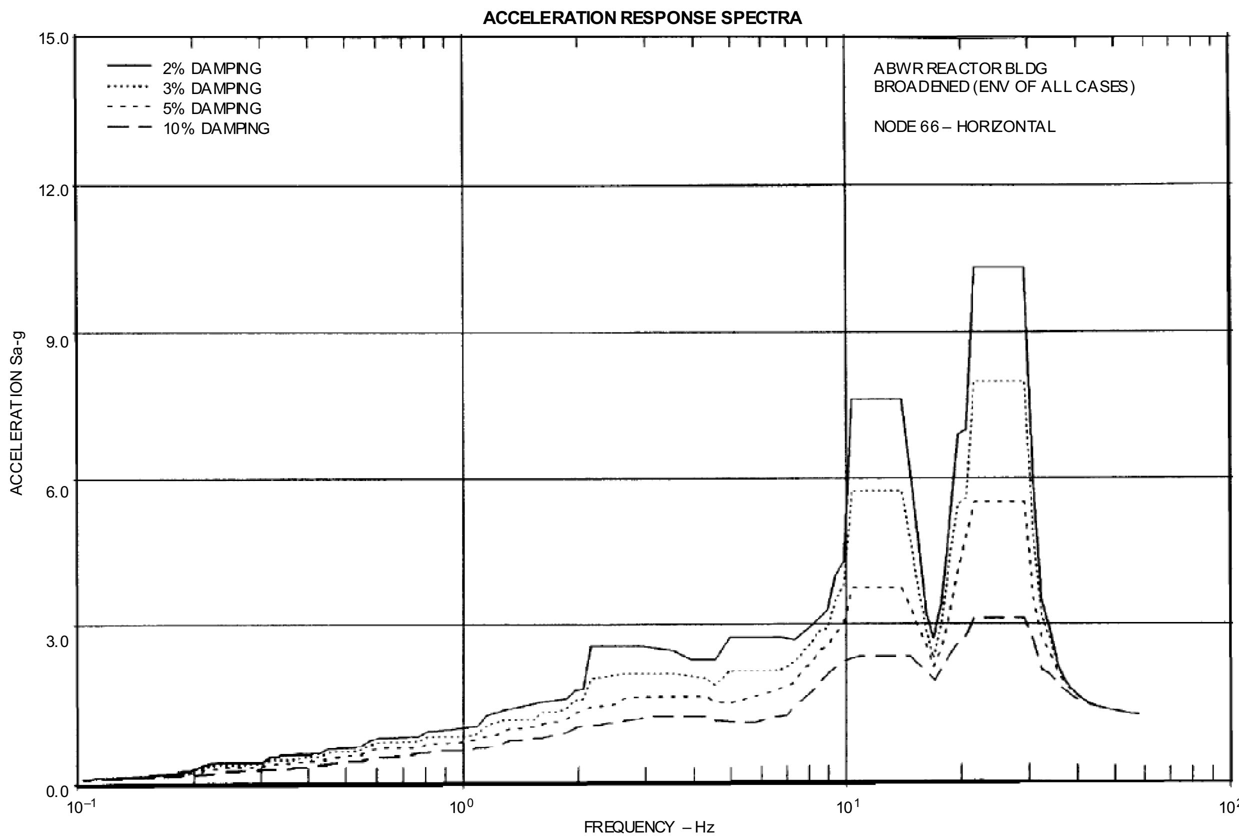

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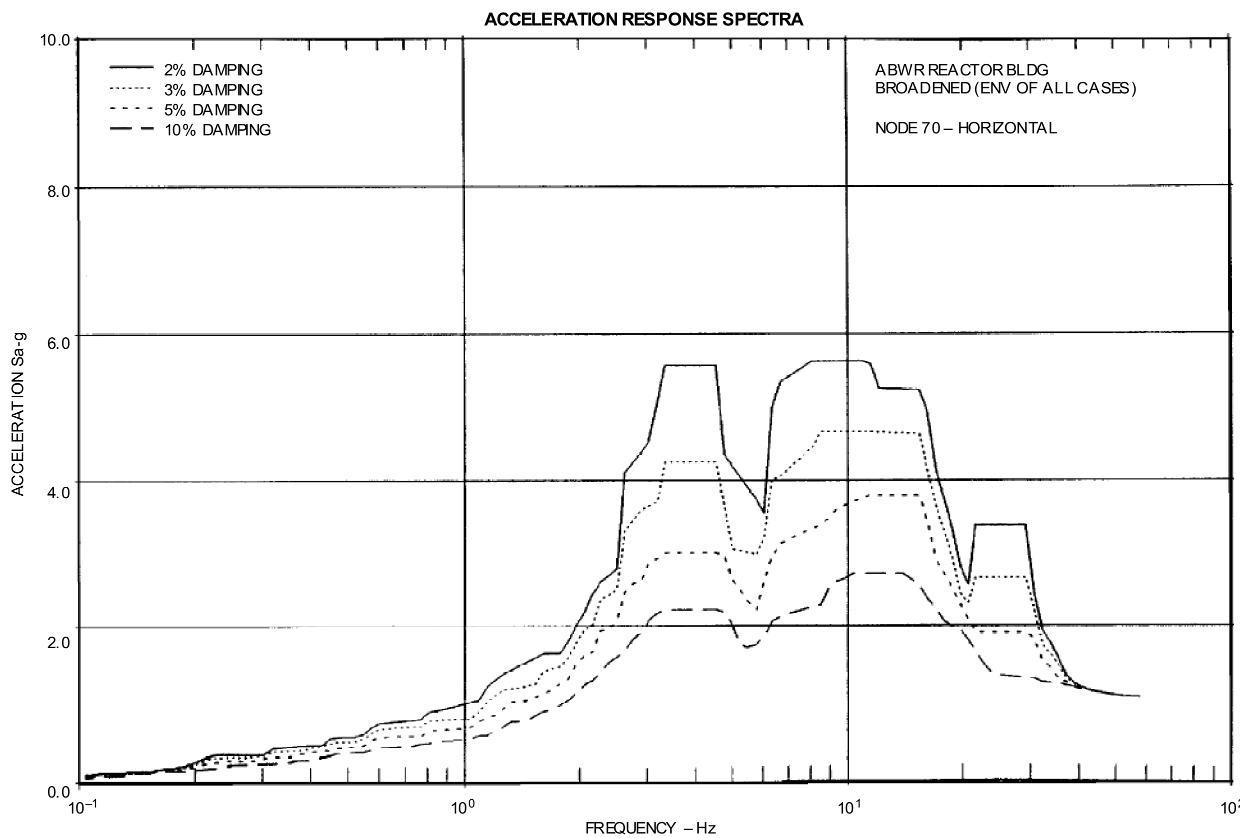
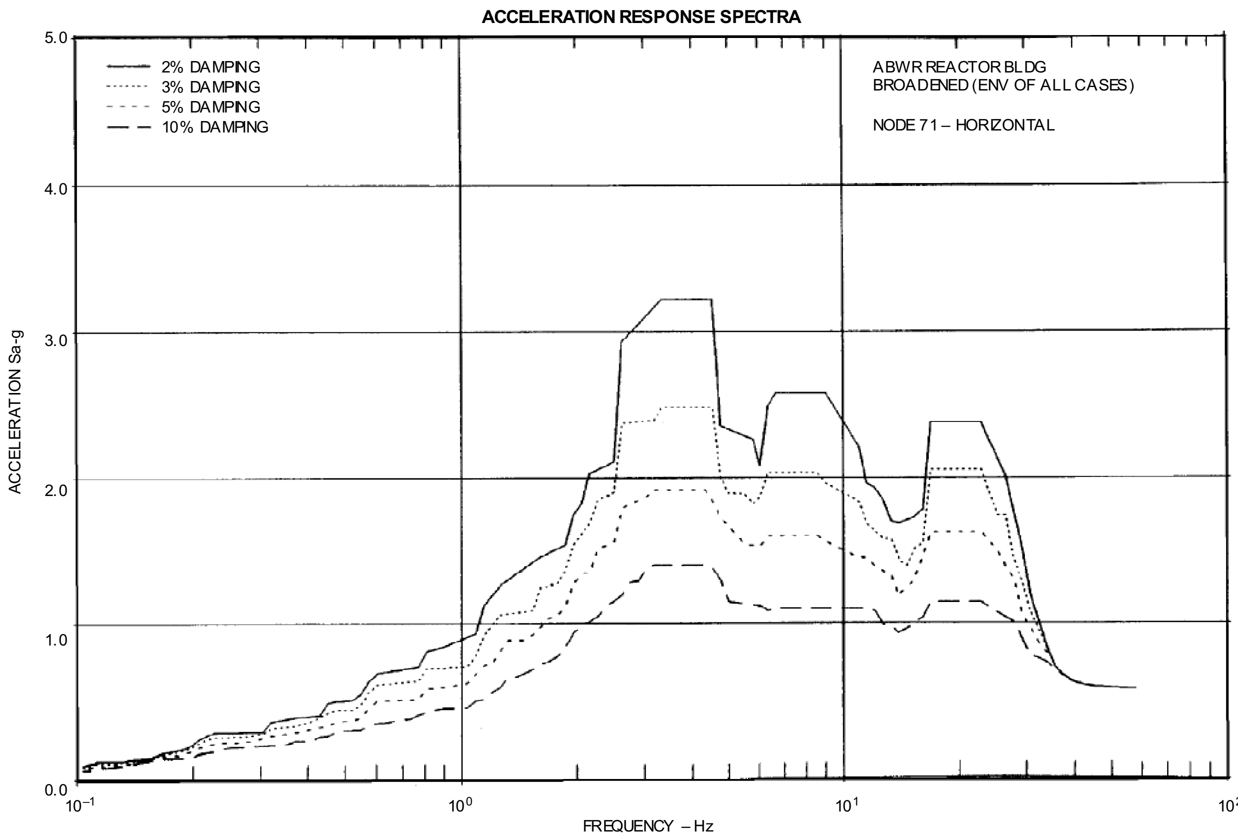


Figure 3A-142 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 70–Horizontal



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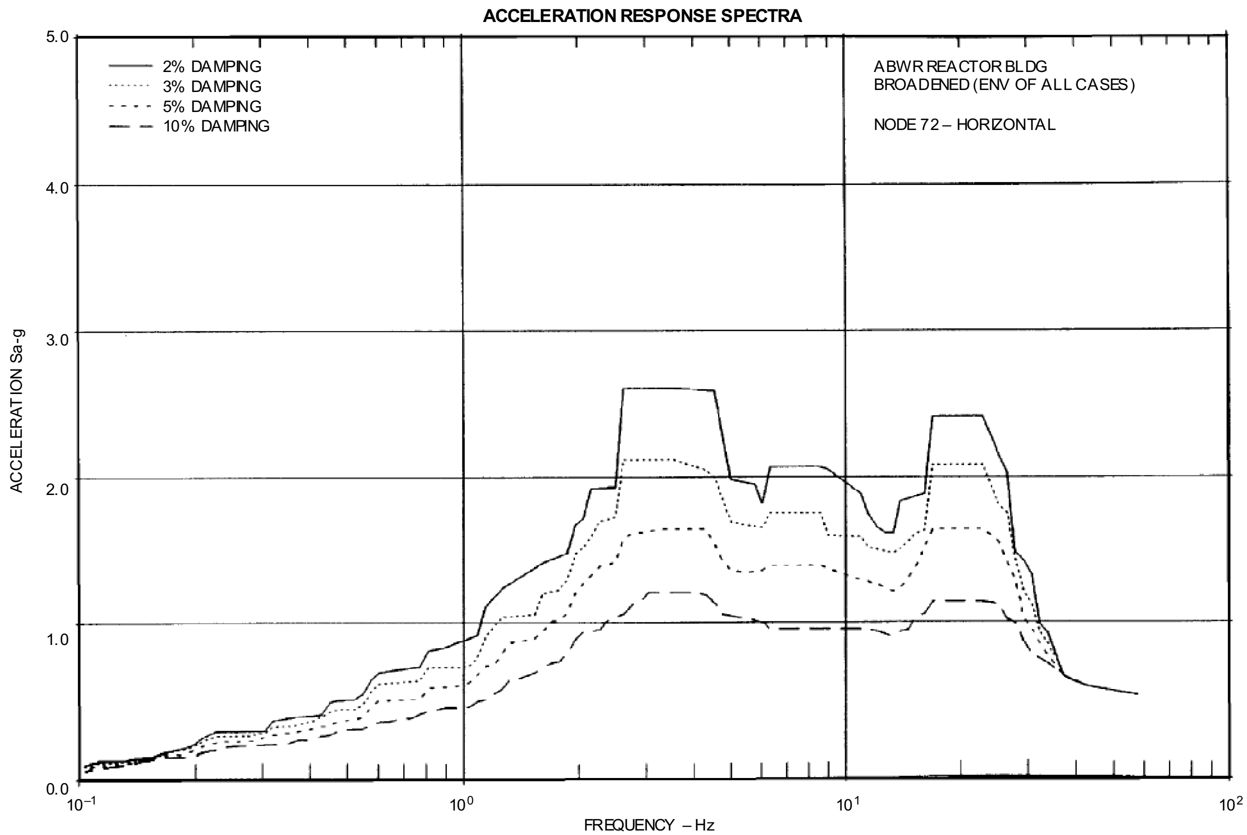

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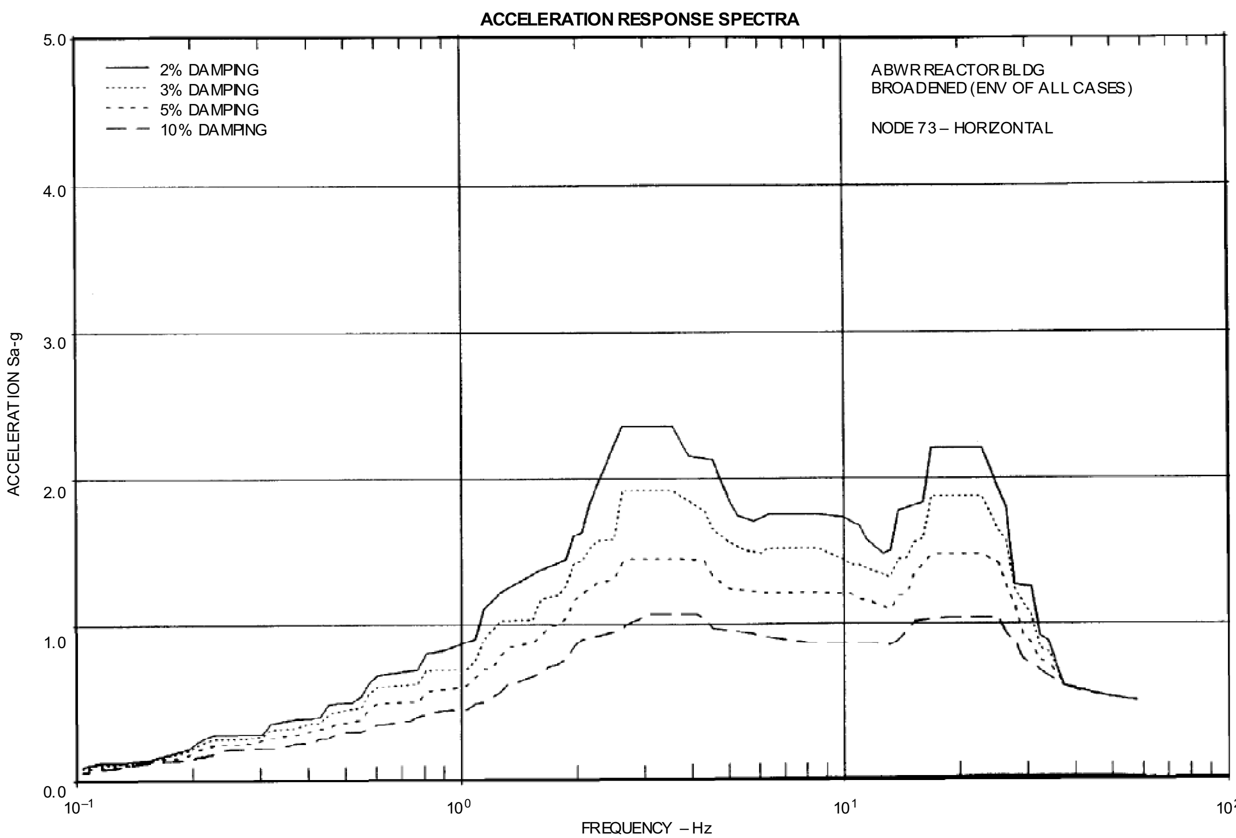

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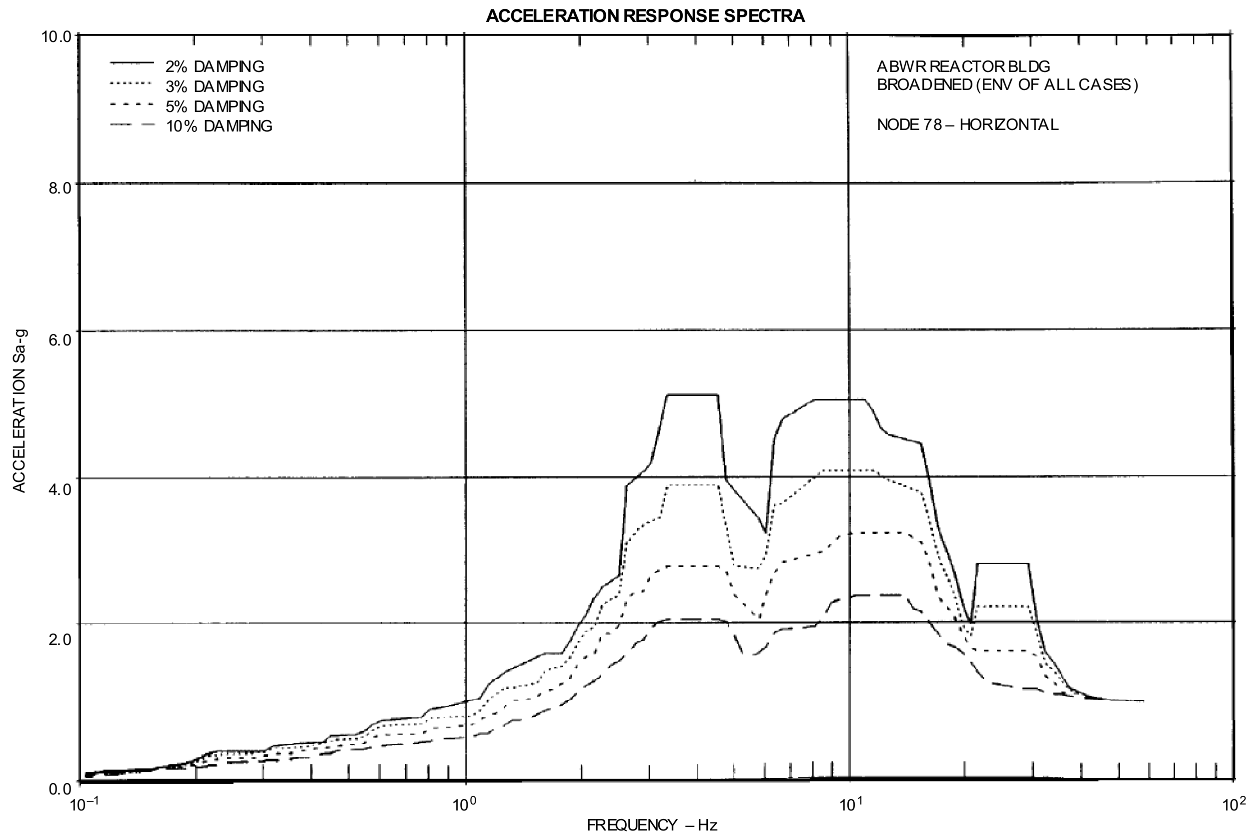

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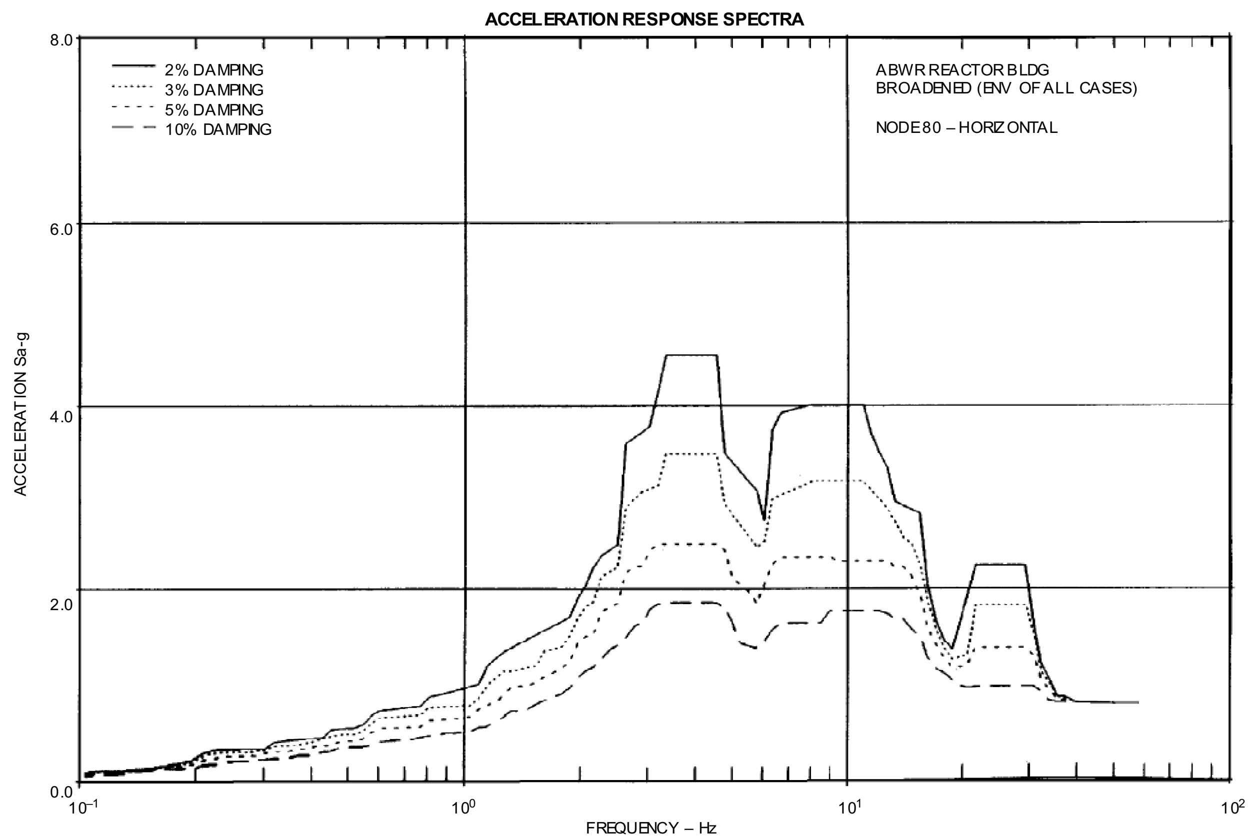

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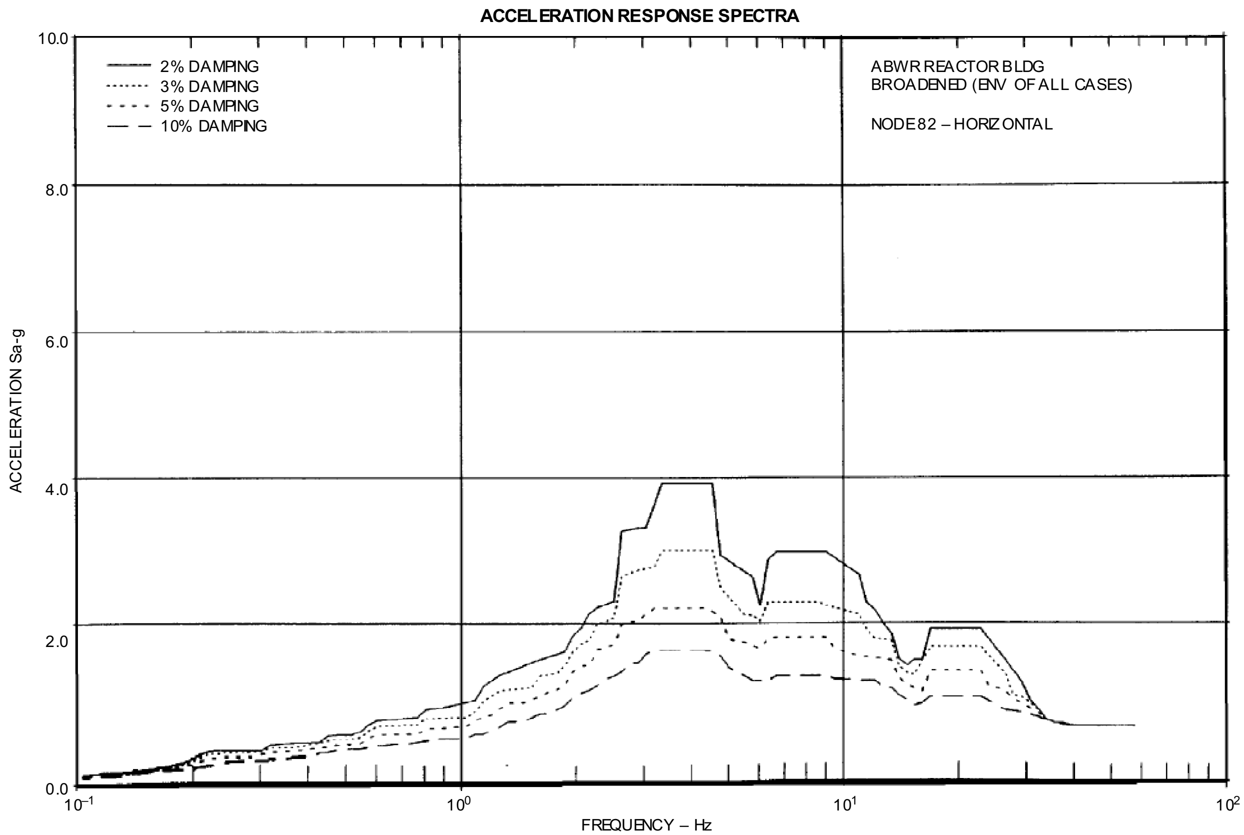

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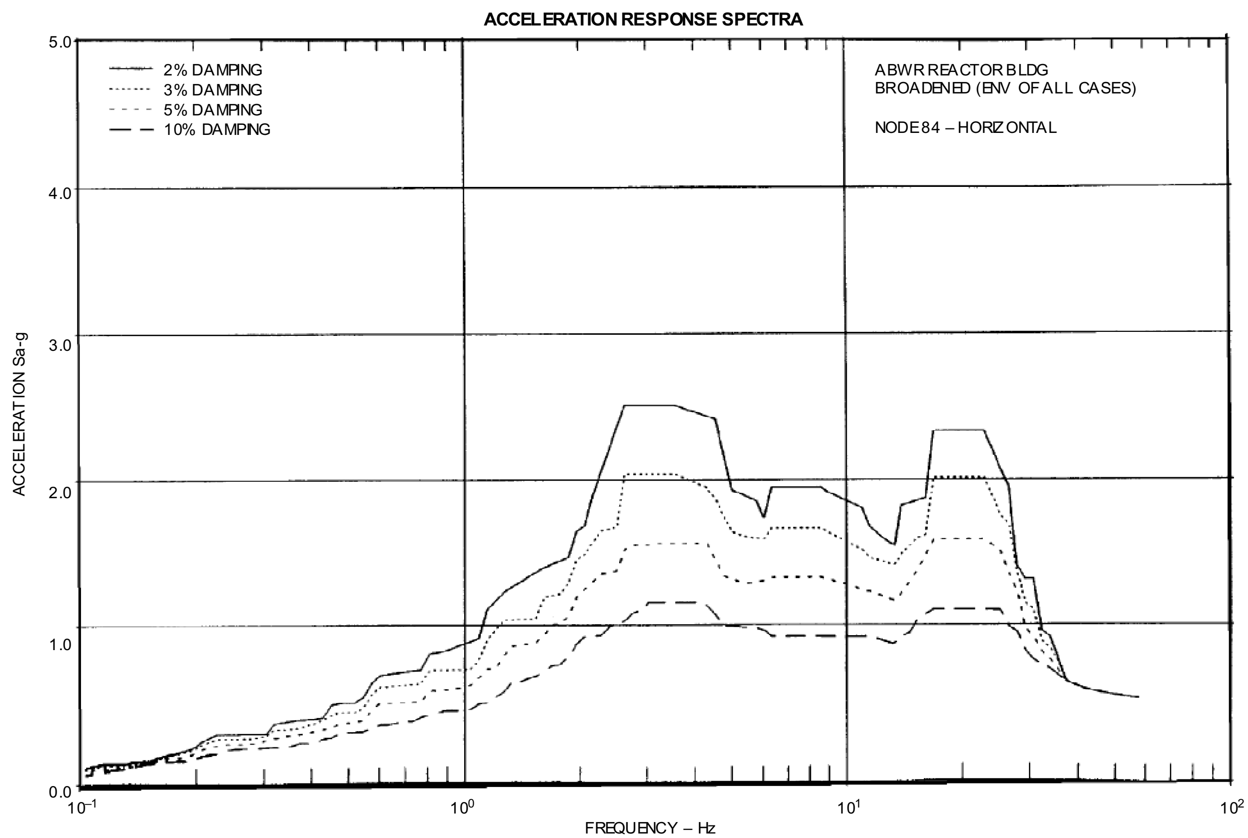

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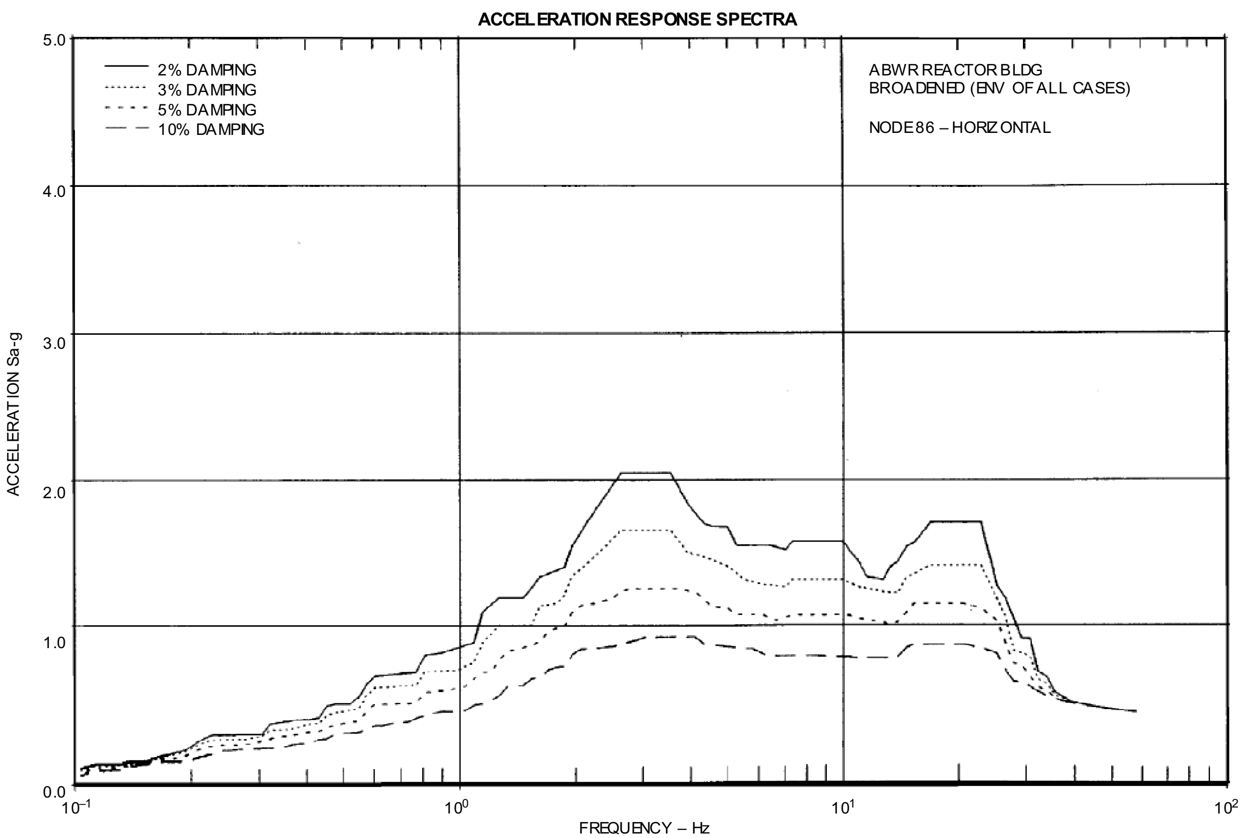

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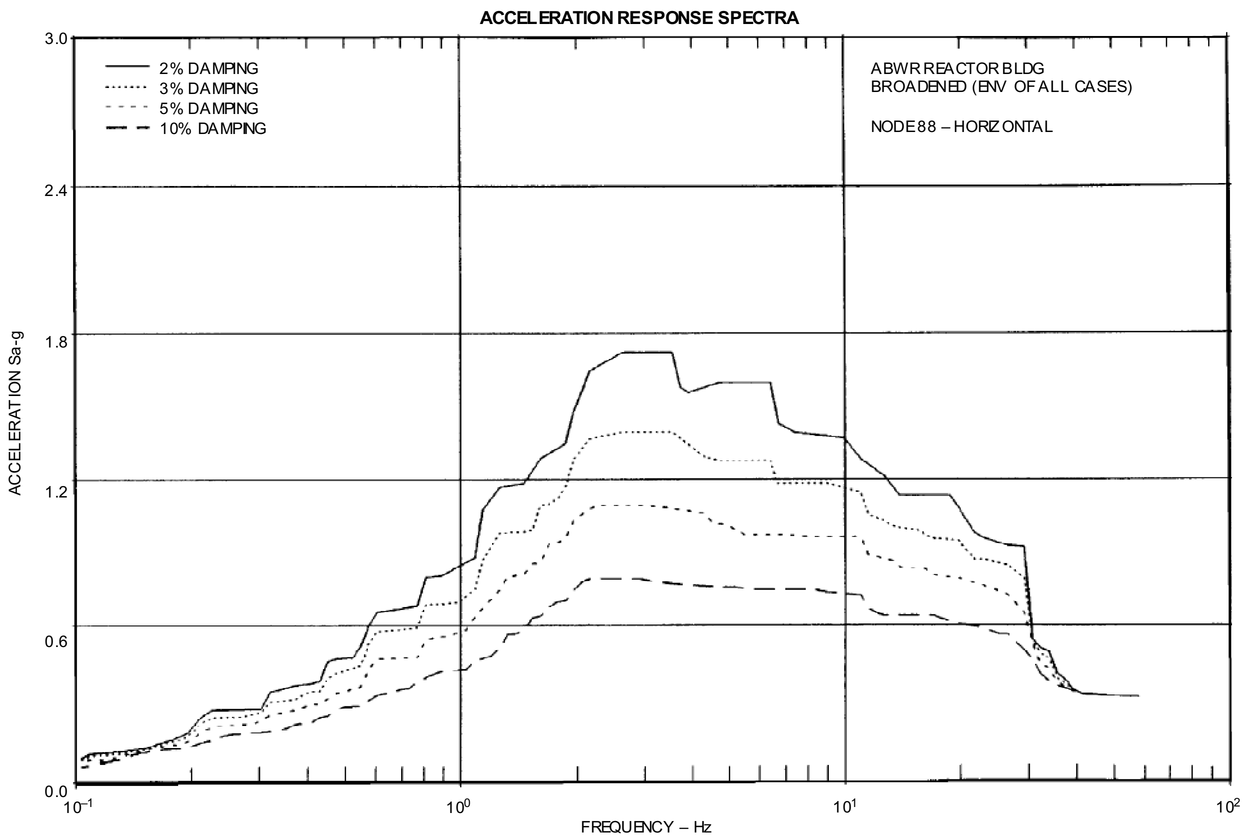


Figure 3A-151 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 88–Horizontal

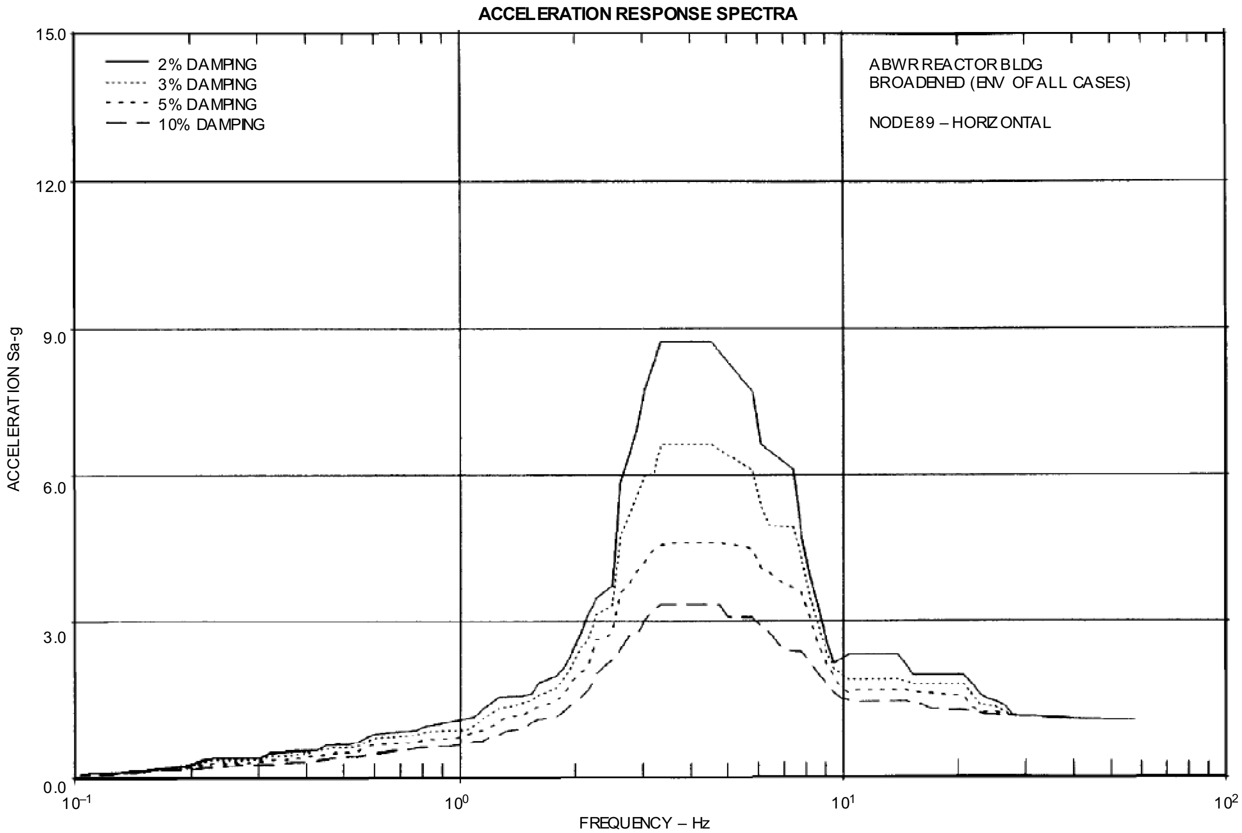
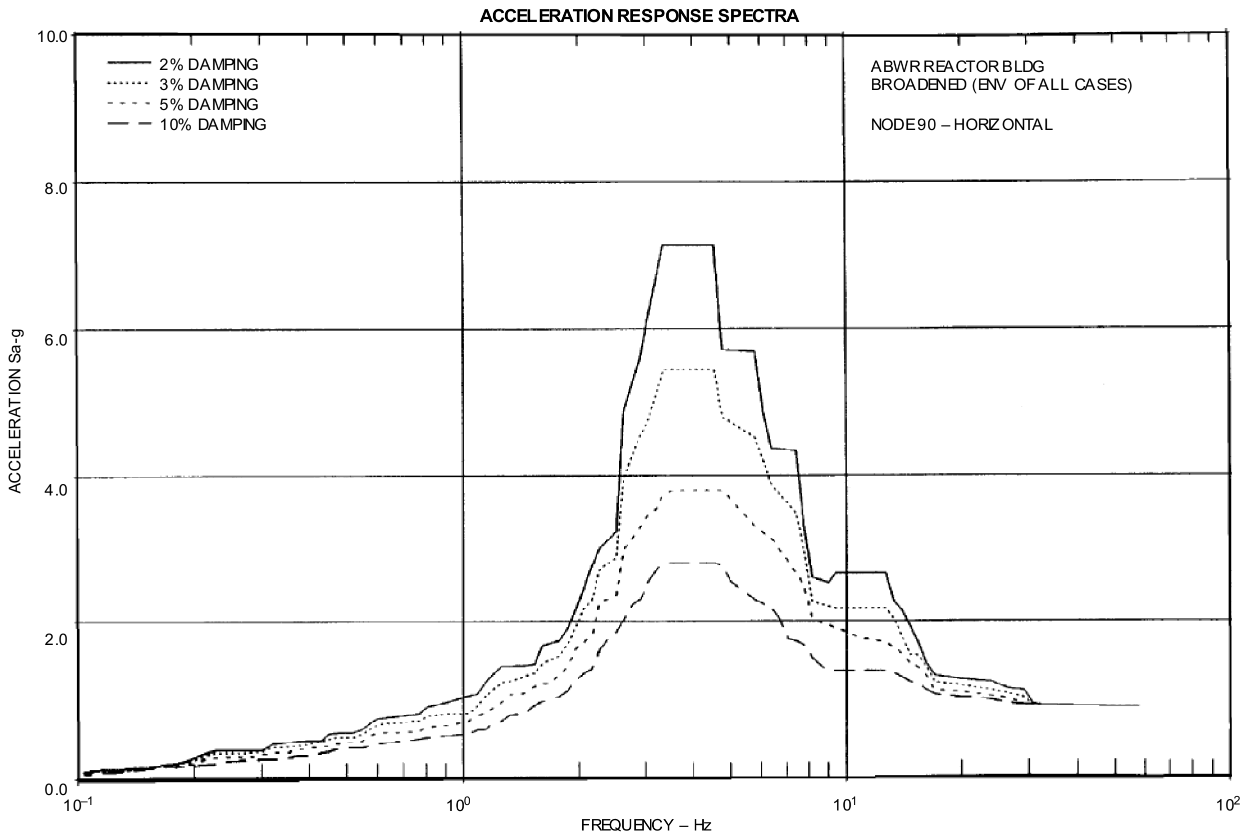


Figure 3A-152 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 89–Horizontal



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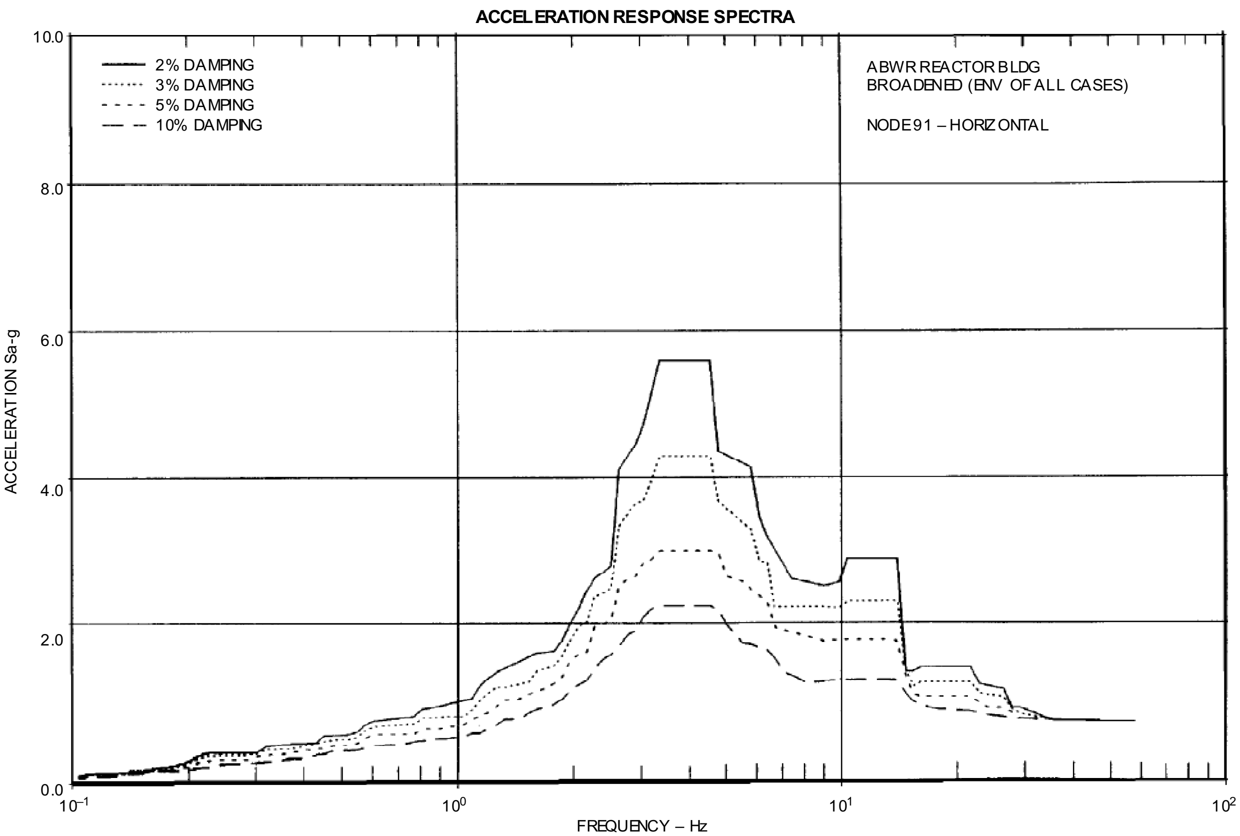


Figure 3A-154 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 91–Horizontal

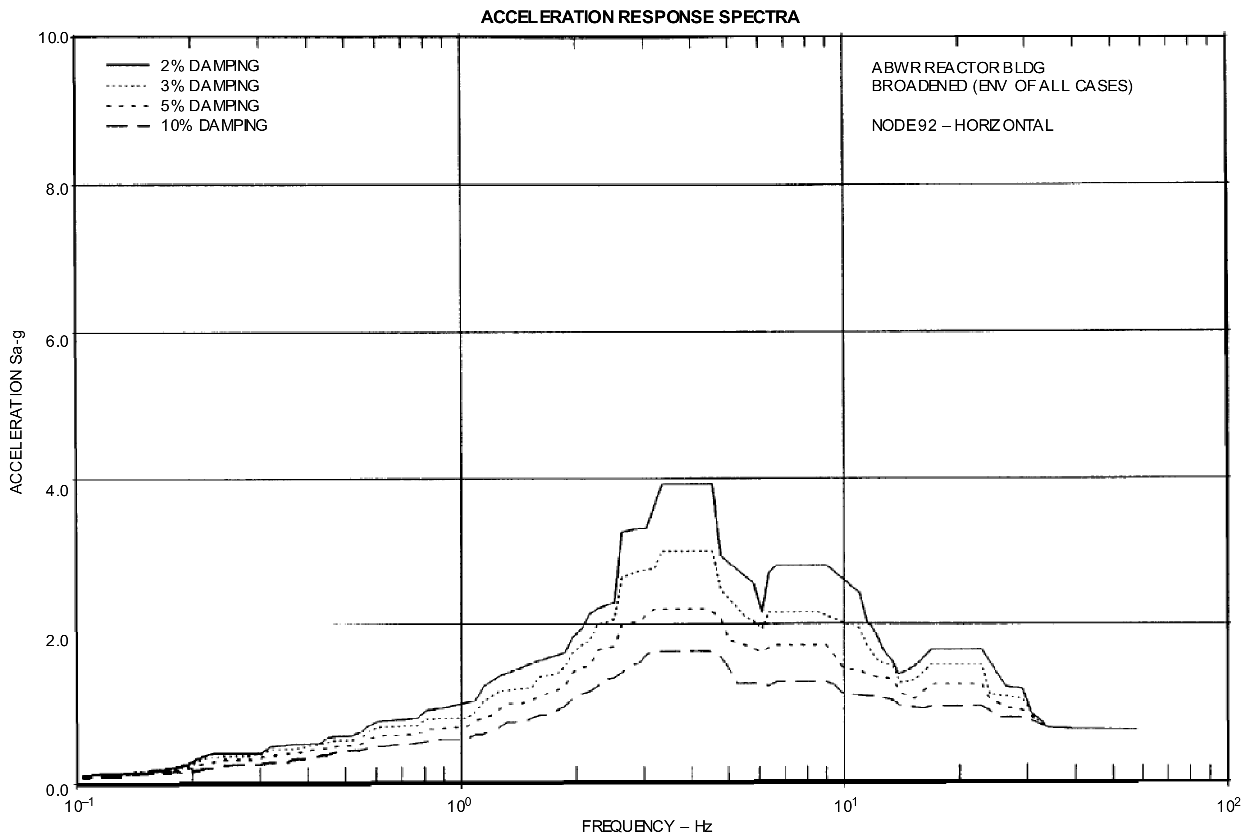


Figure 3A-155 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 92–Horizontal

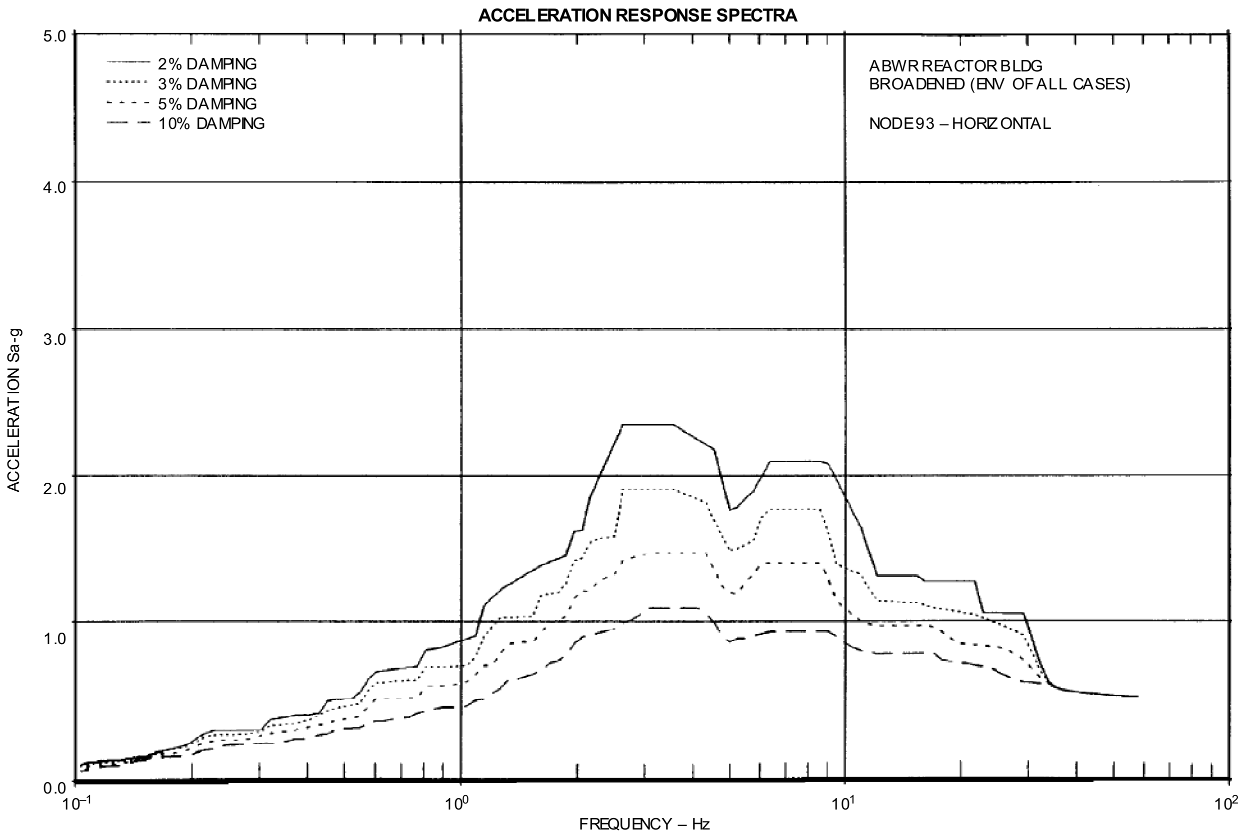
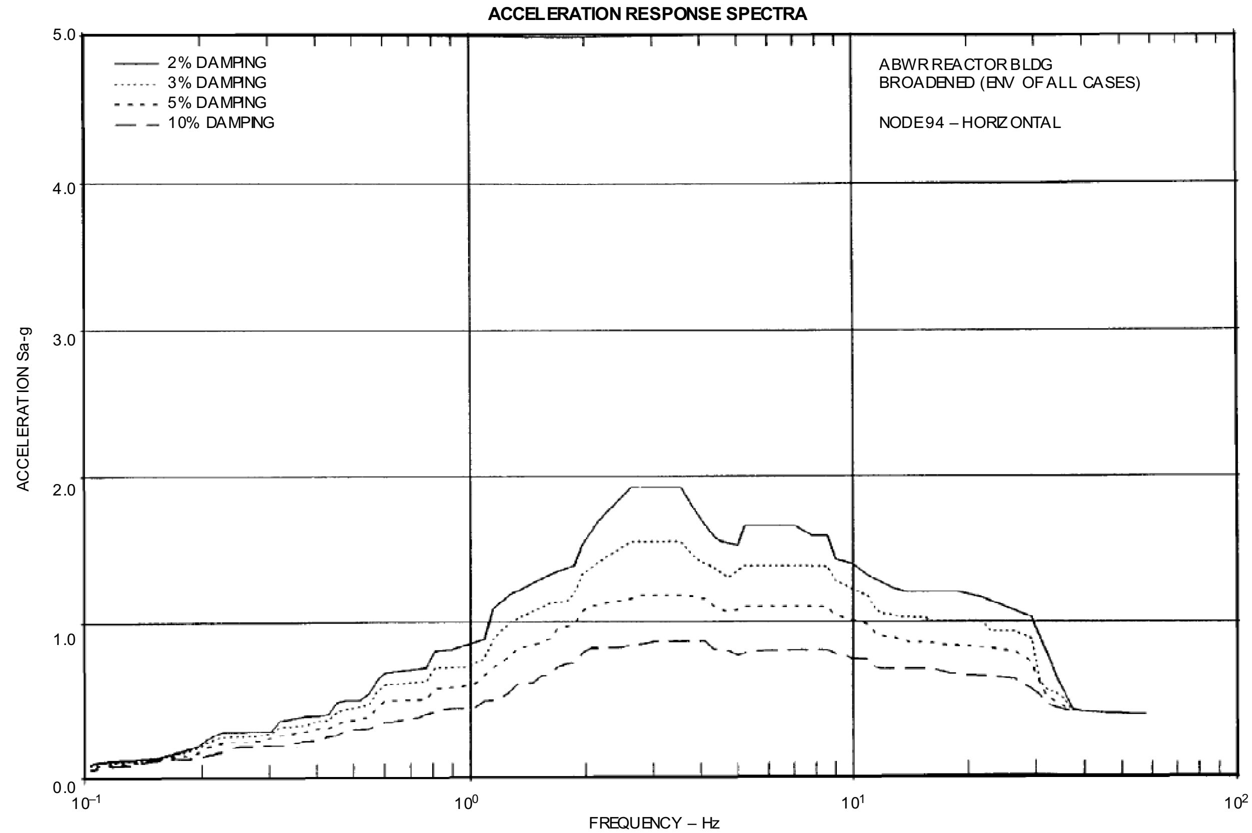


Figure 3A-156 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 93–Horizontal



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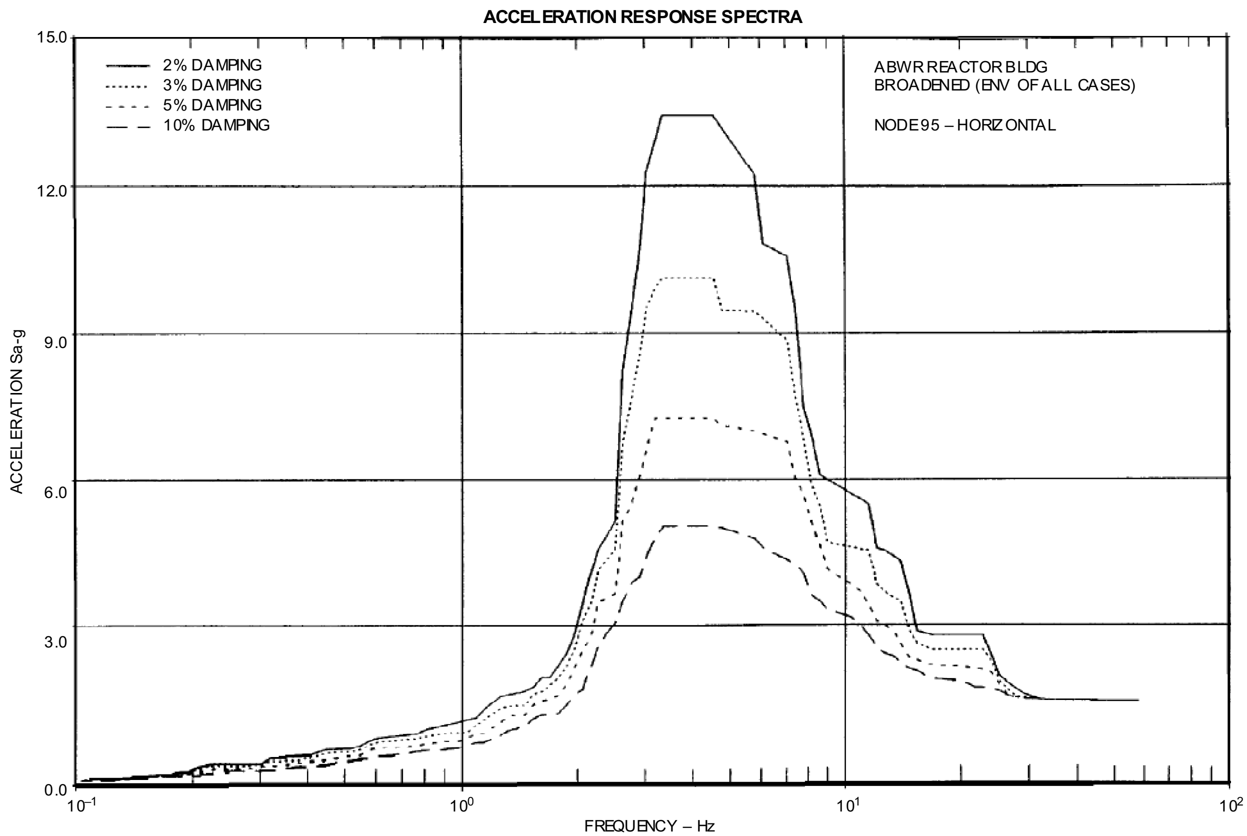

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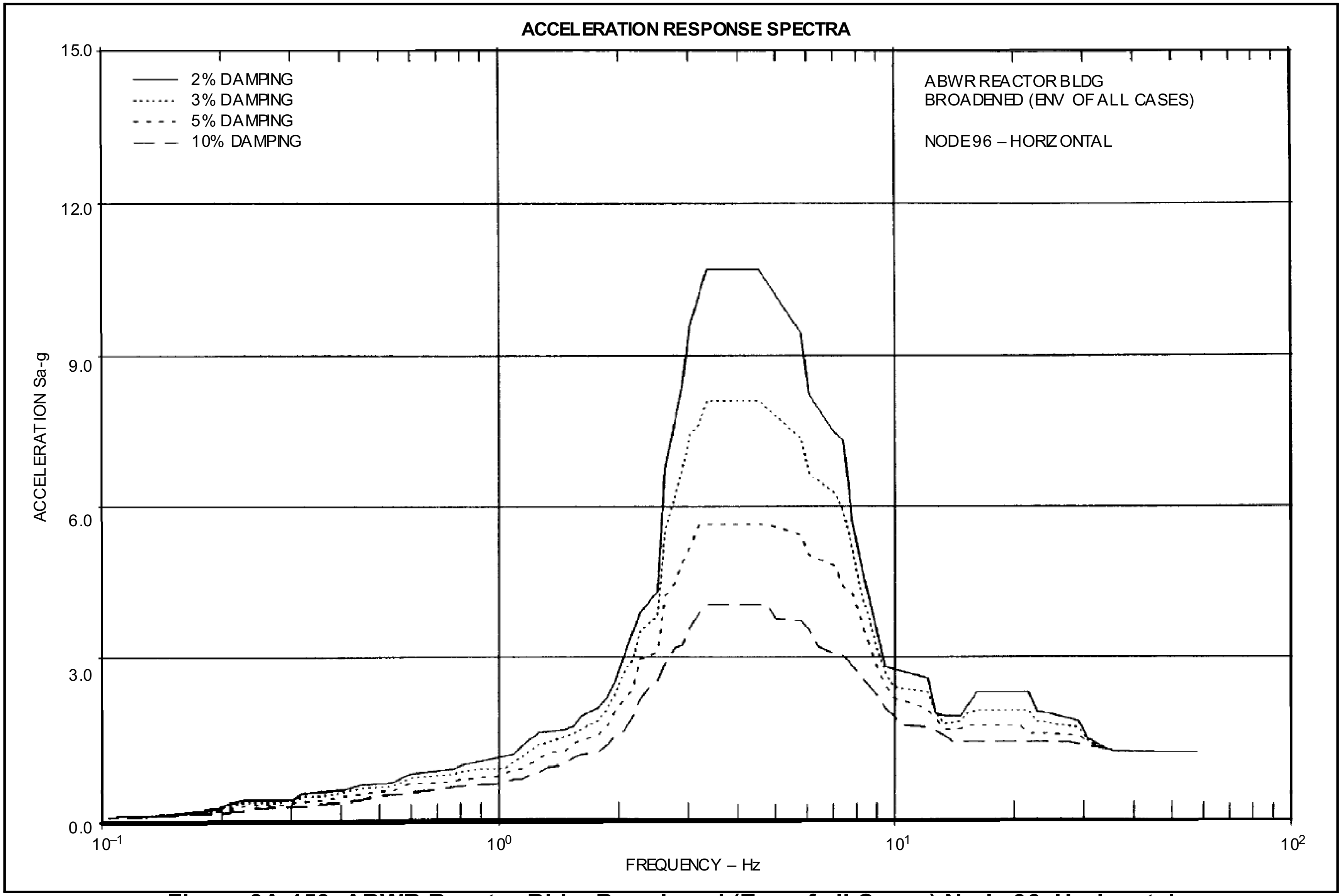

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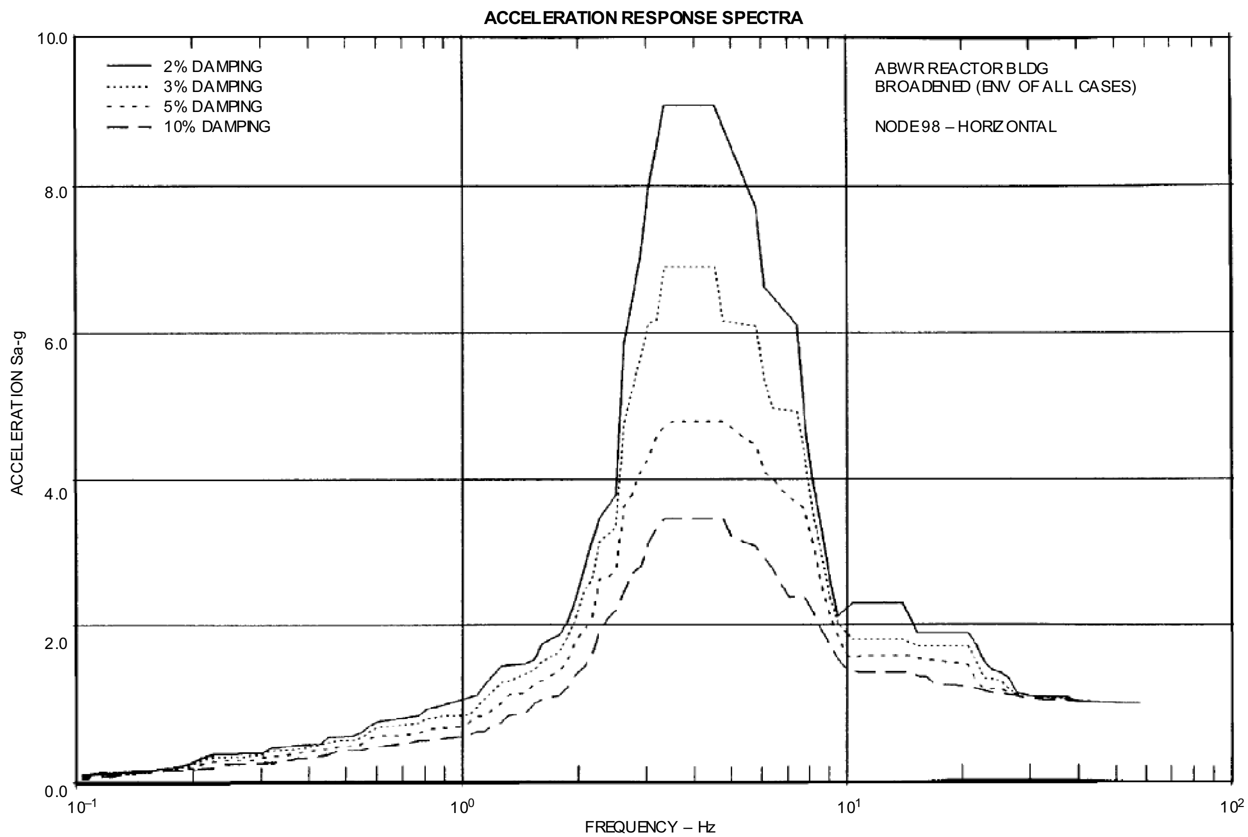

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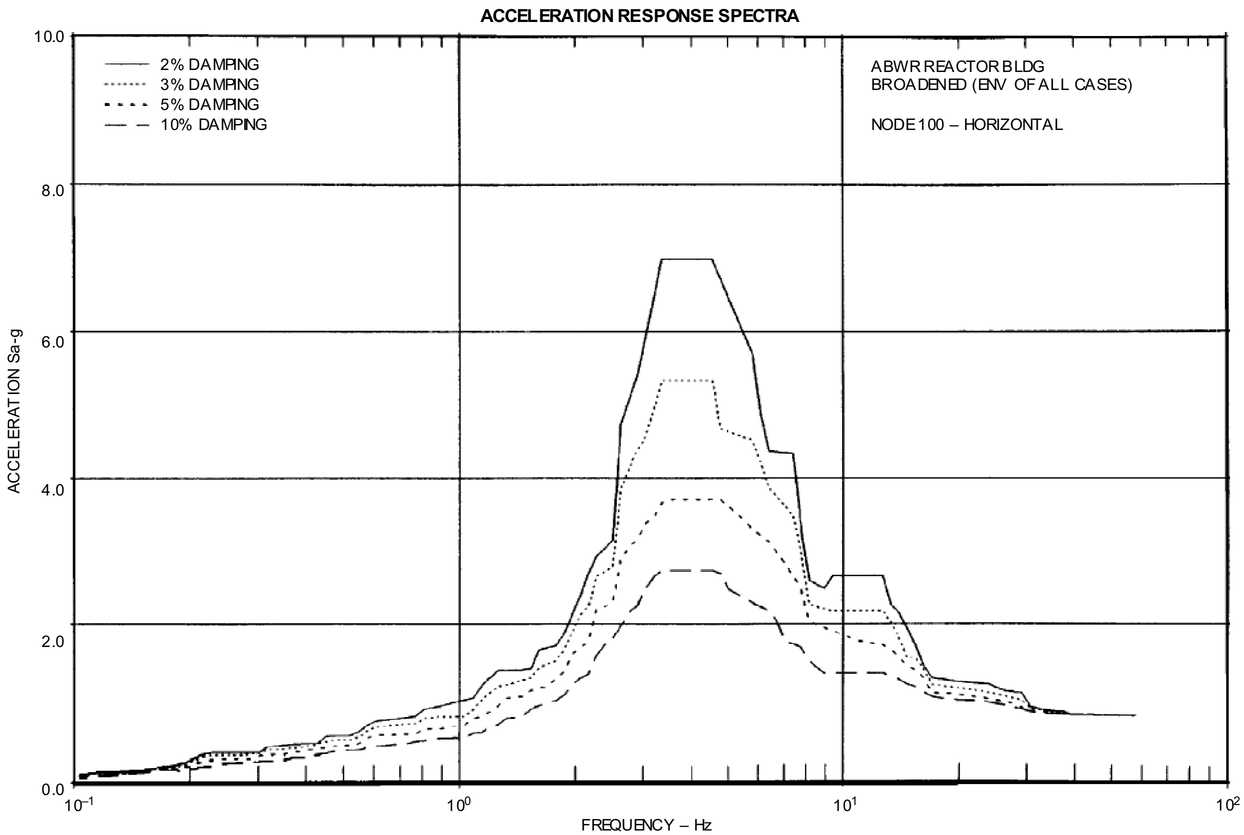


Figure 3A-161 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 100–Horizontal

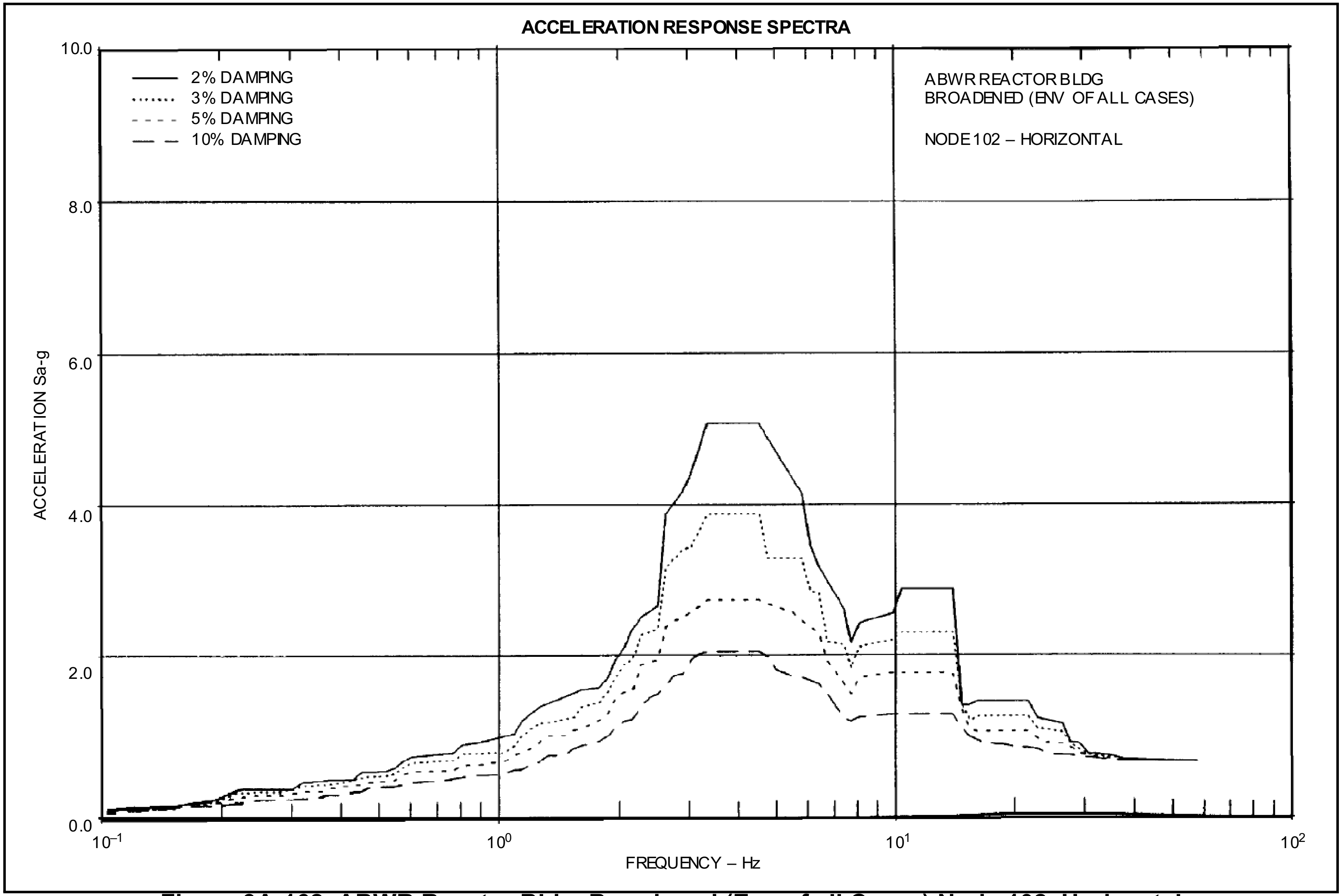


Figure 3A-162 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 102–Horizontal

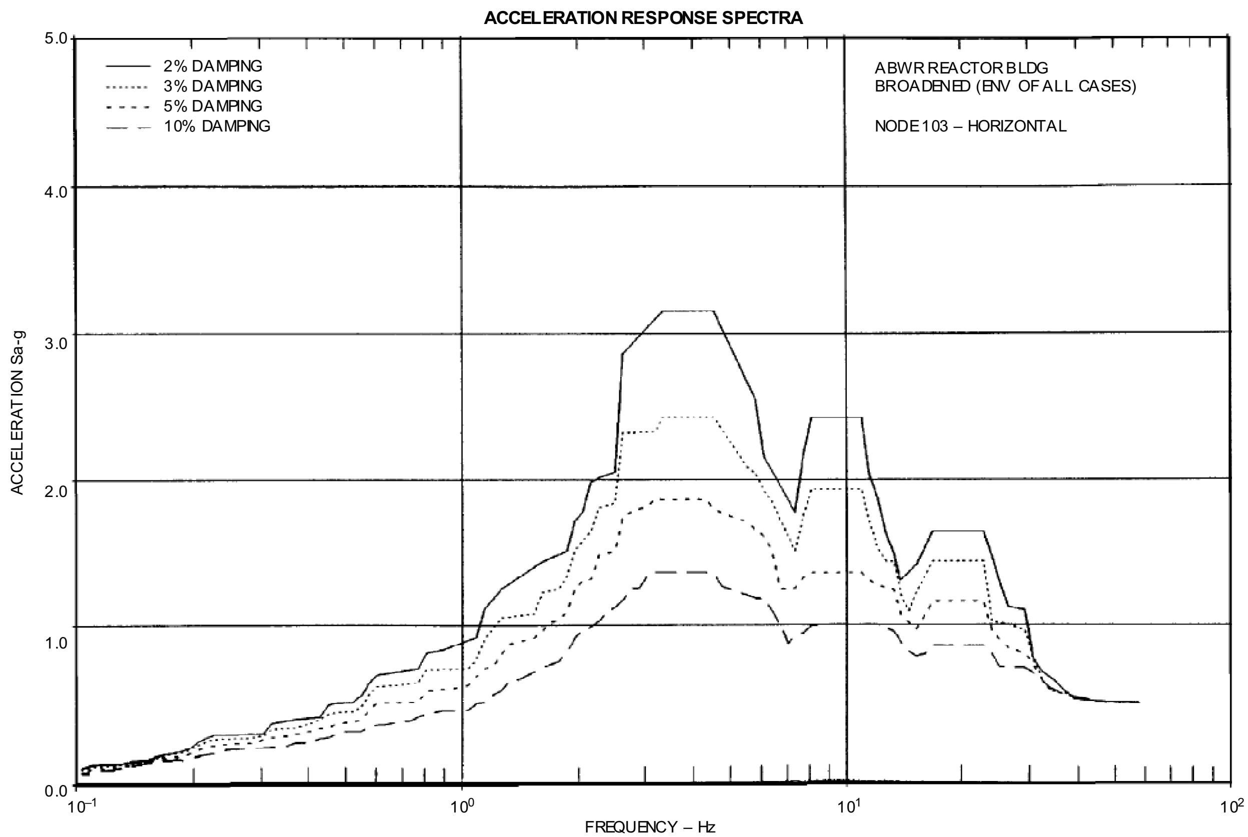


Figure 3A-163 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 103–Horizontal

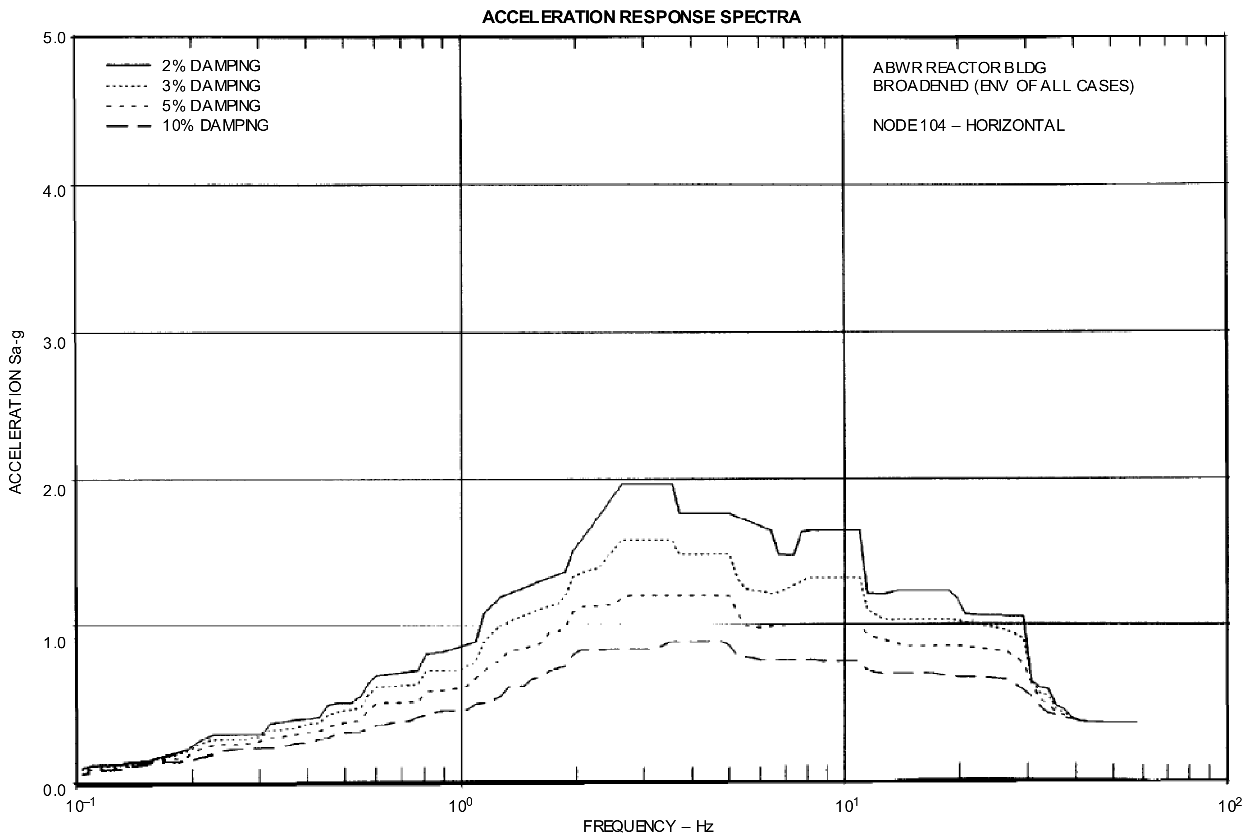


Figure 3A-164 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 104–Horizontal

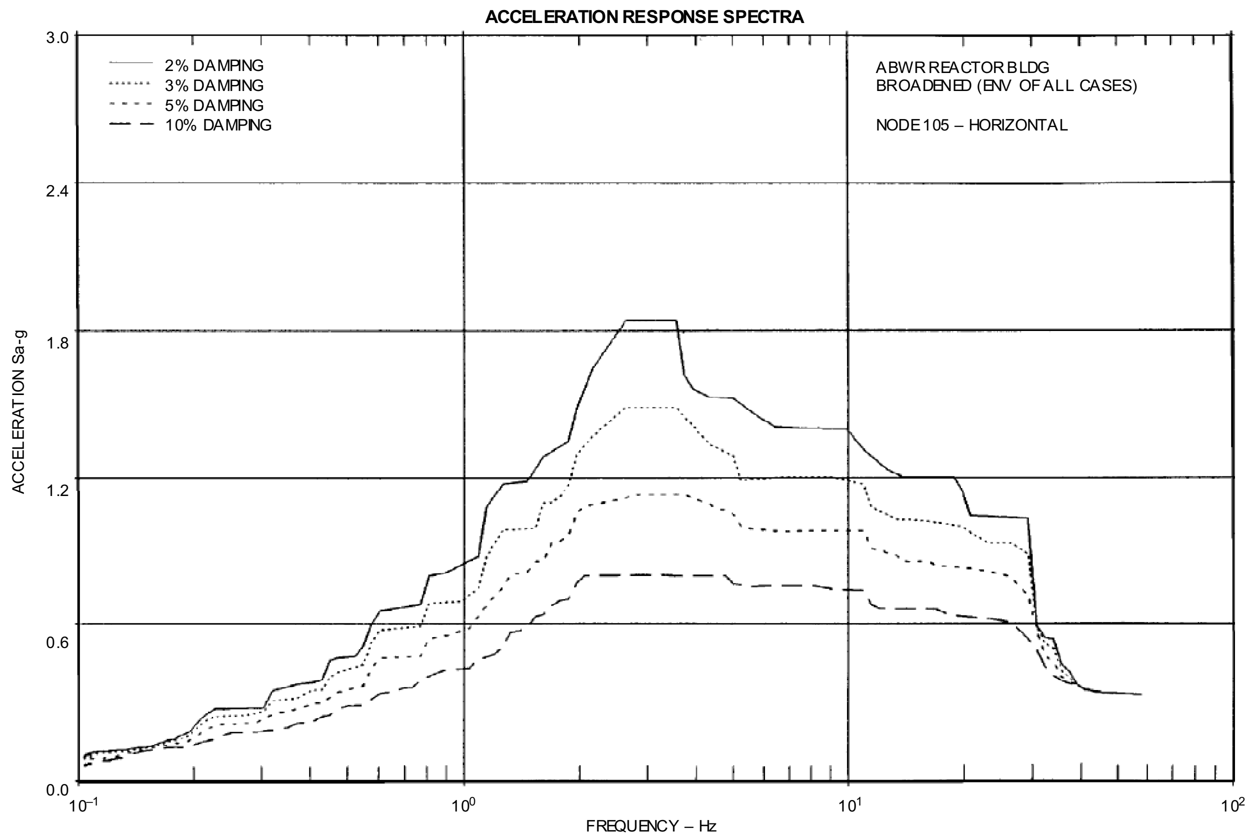


Figure 3A-165 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 105–Horizontal

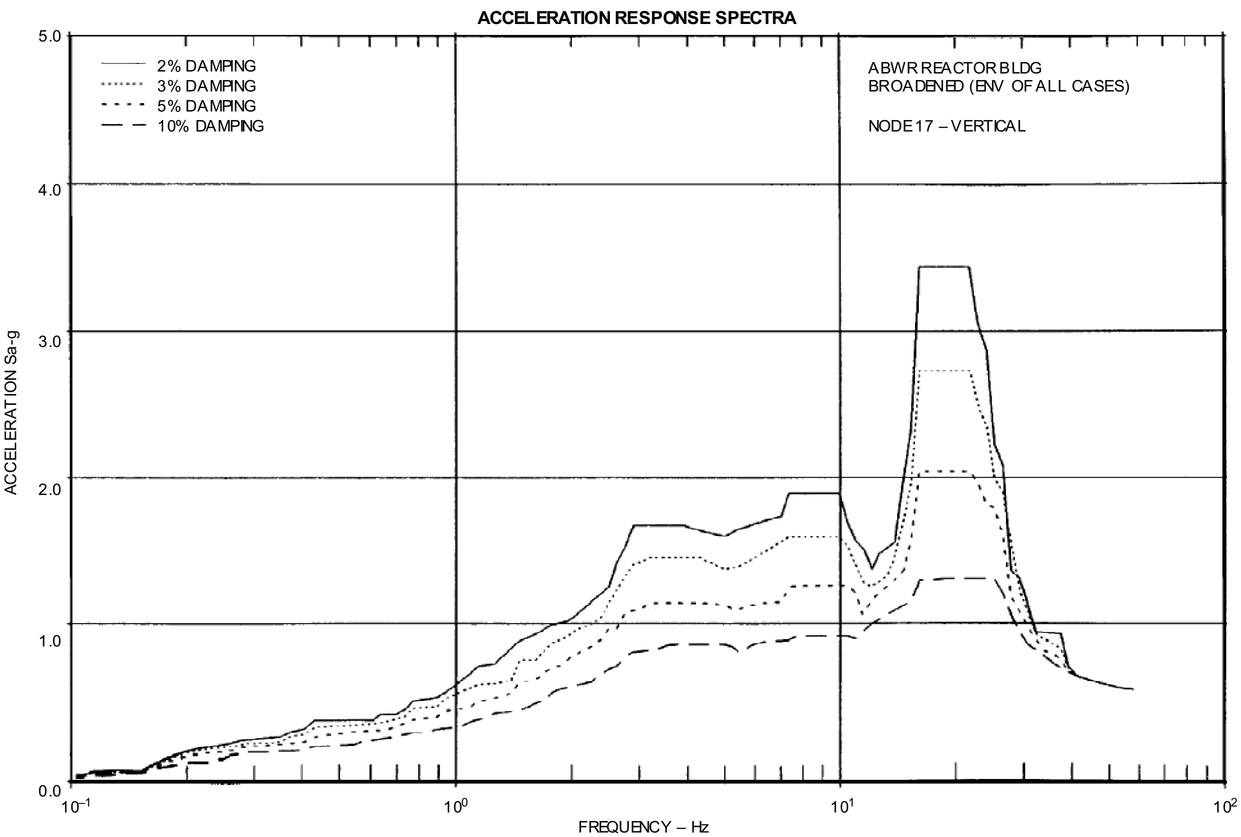


Figure 3A-166 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 17–Vertical

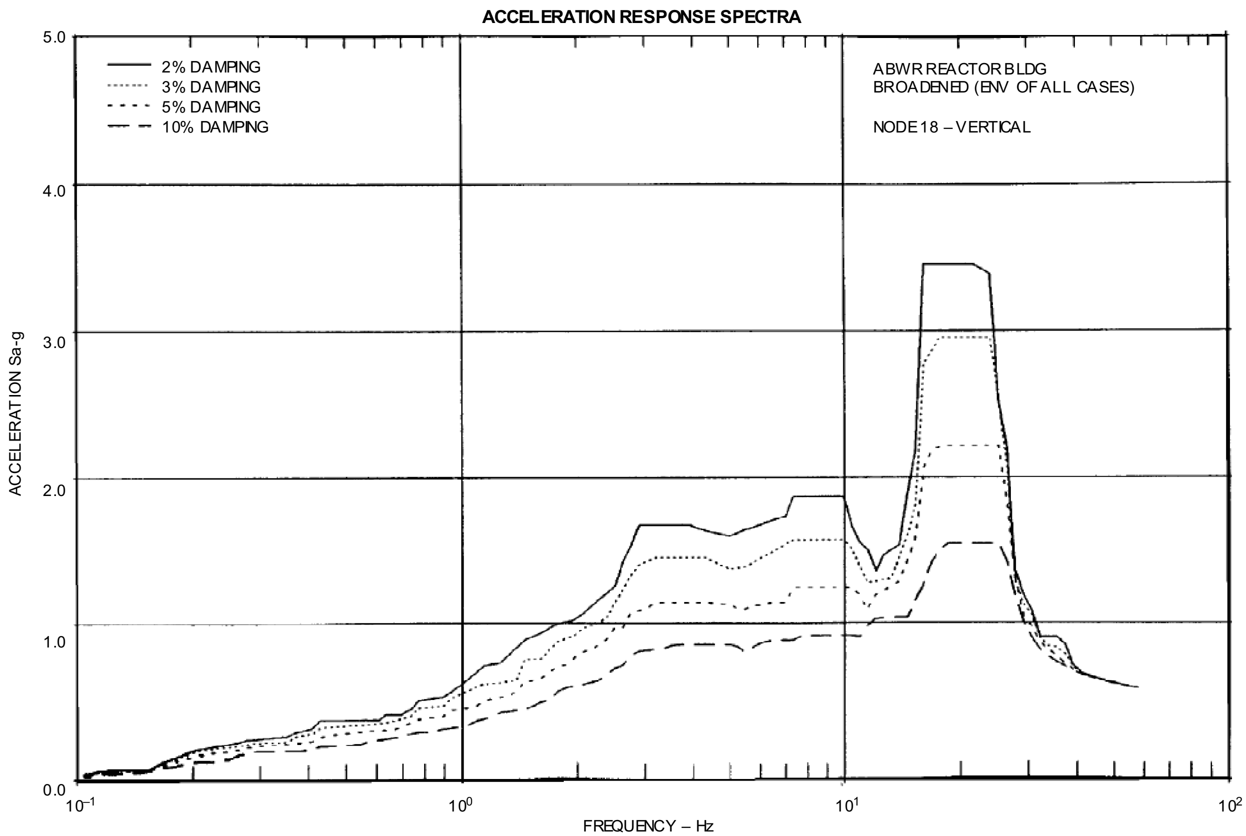


Figure 3A-167 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 18–Vertical

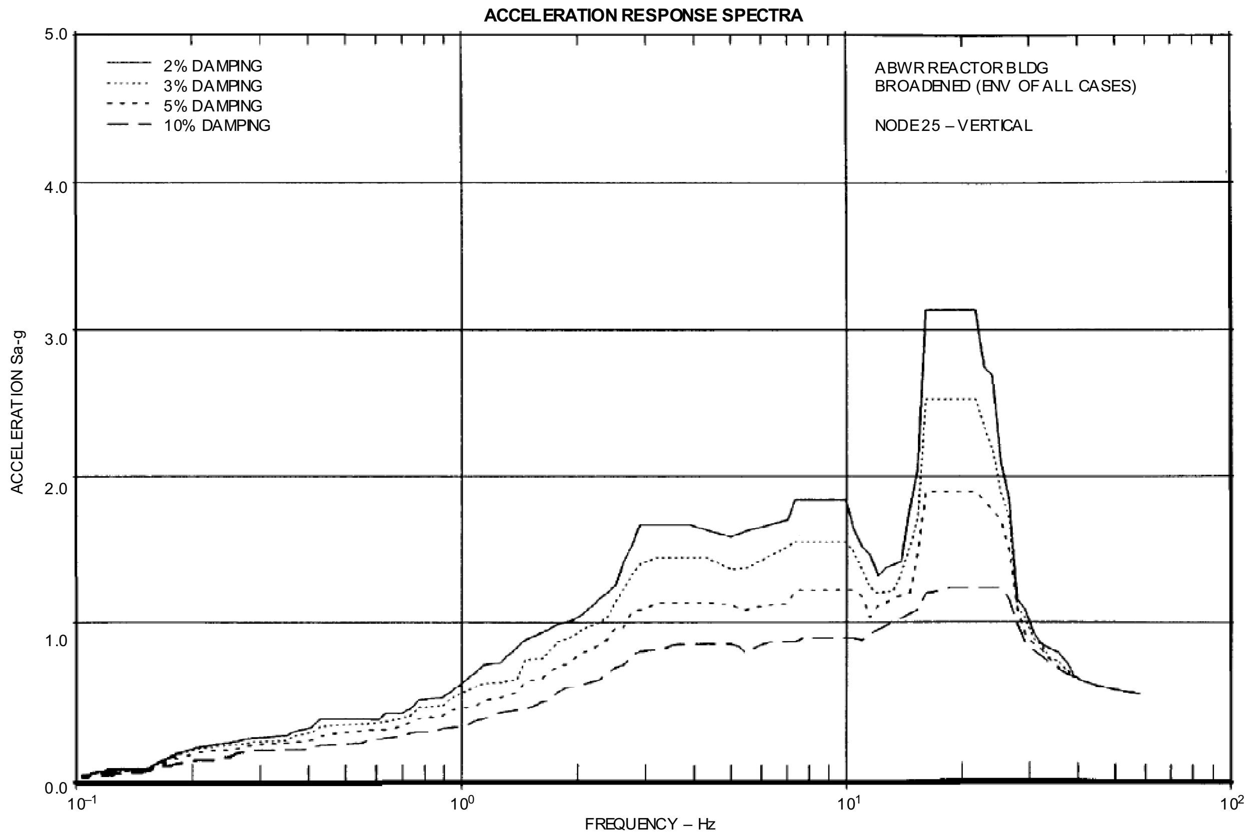


Figure 3A-168 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 25–Vertical

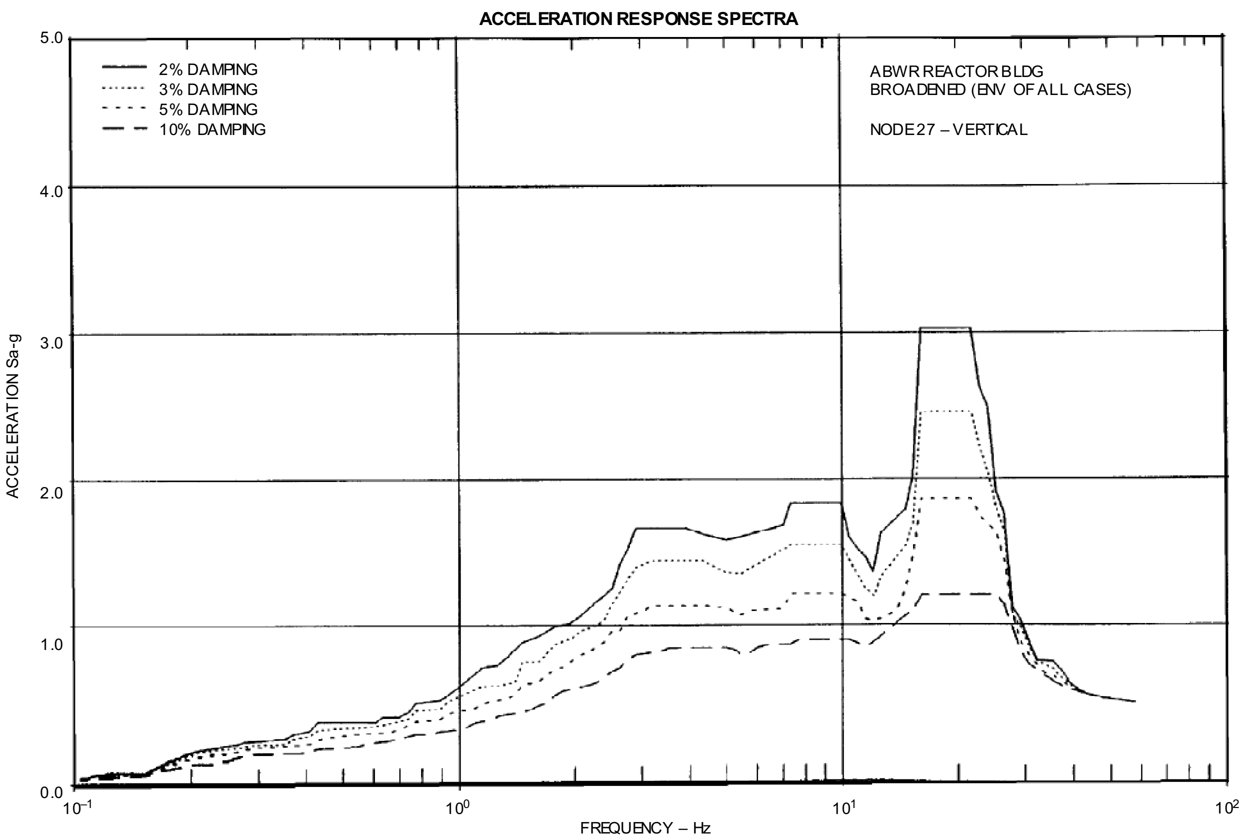


Figure 3A-169 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 27–Vertical

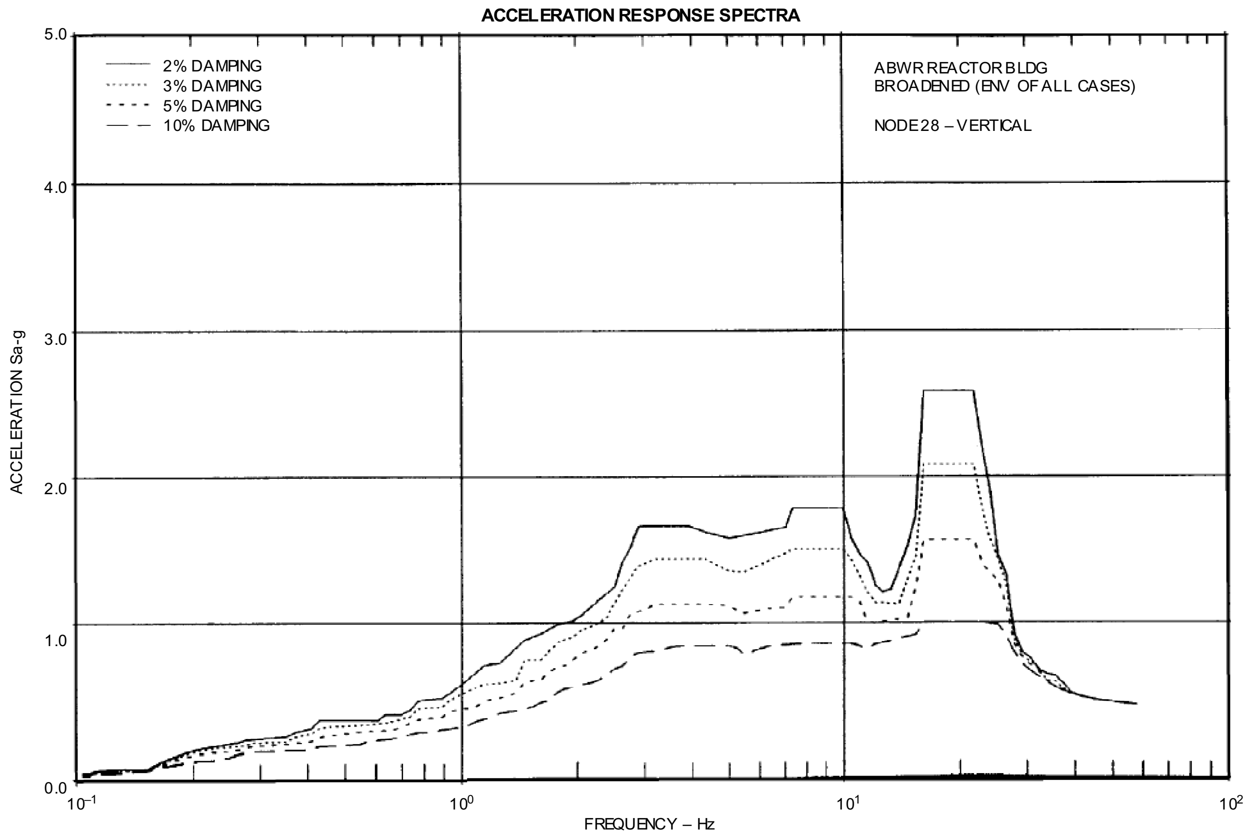


Figure 3A-170 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 28–Vertical

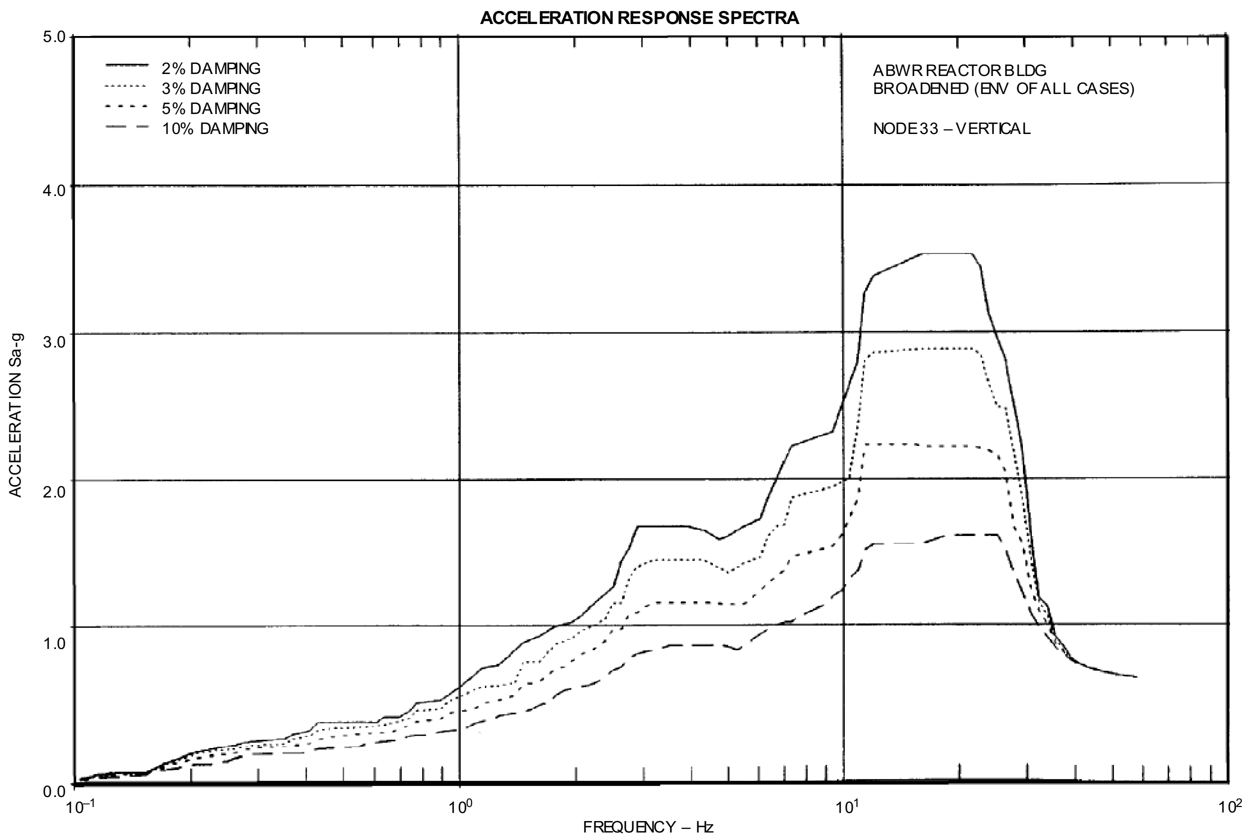


Figure 3A-171 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 33–Vertical

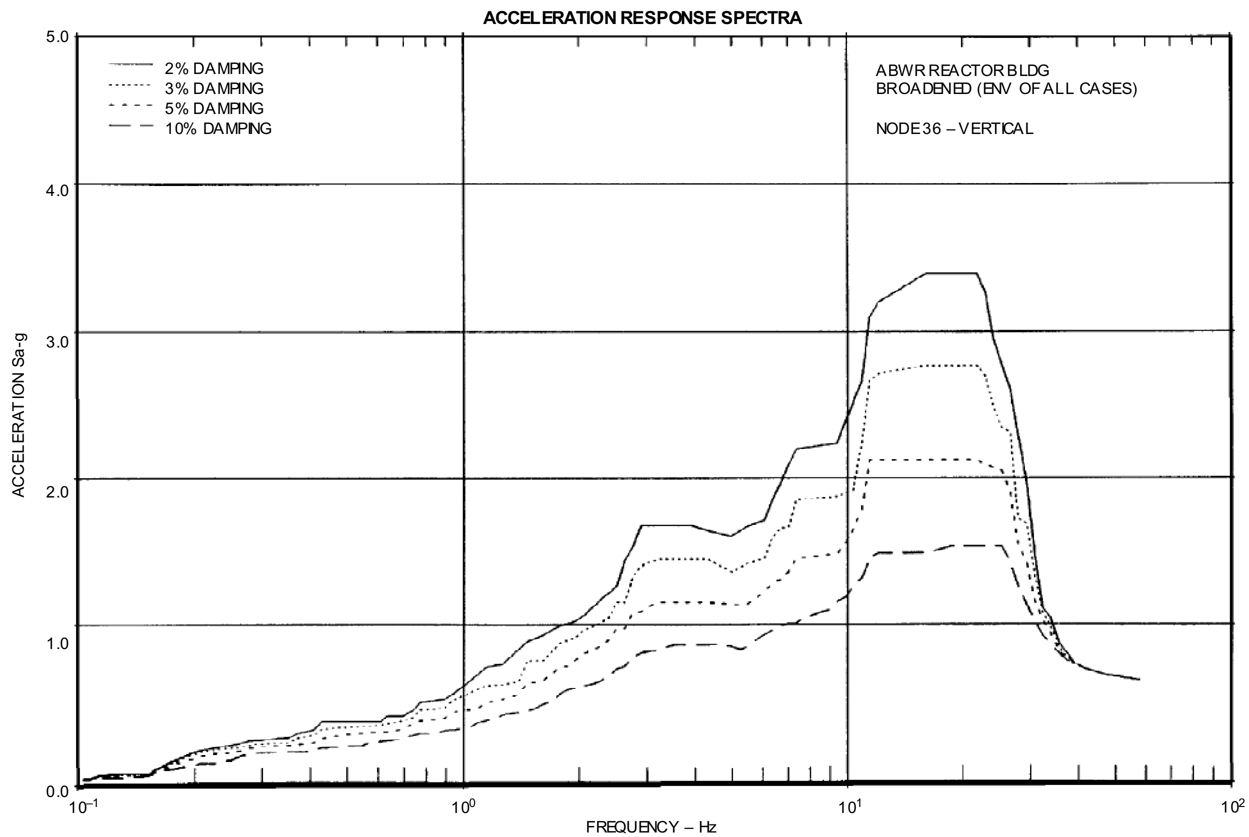


Figure 3A-172 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 36–Vertical

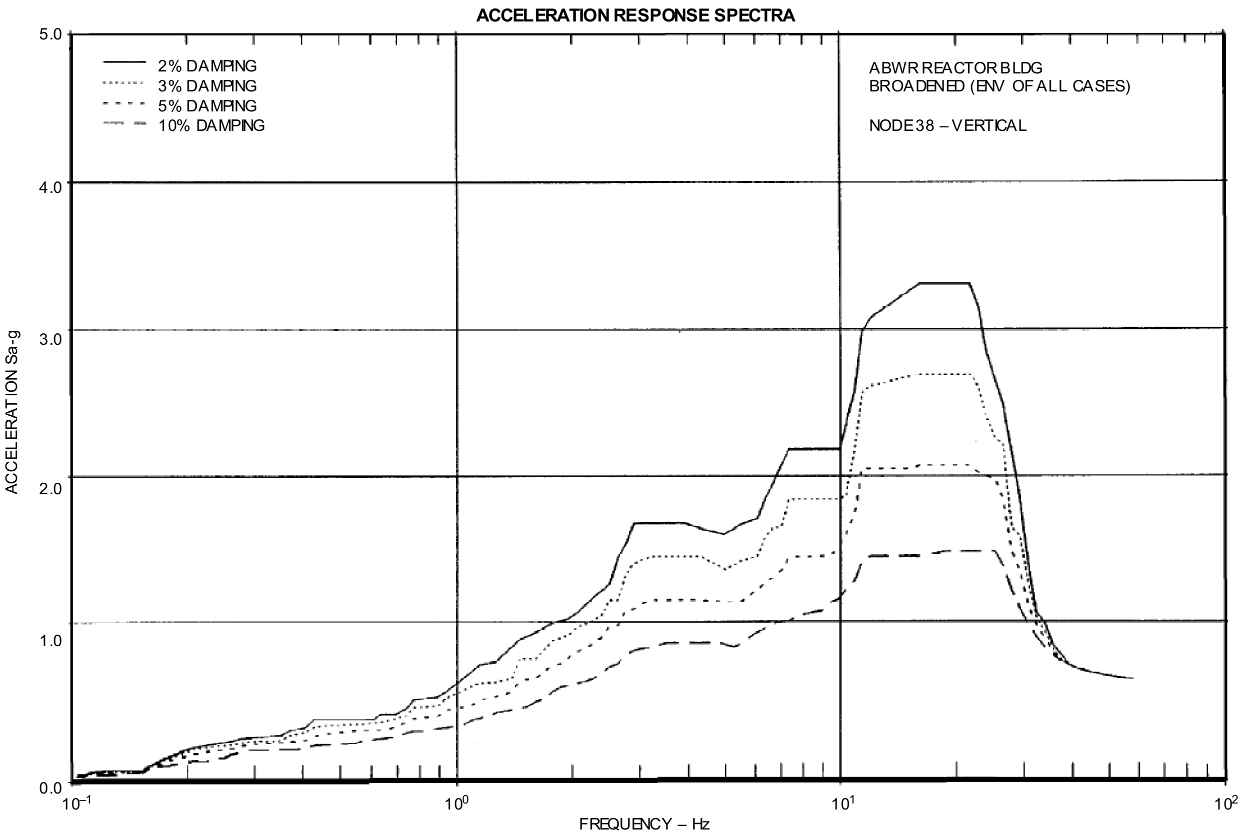


Figure 3A-173 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 38–Vertical

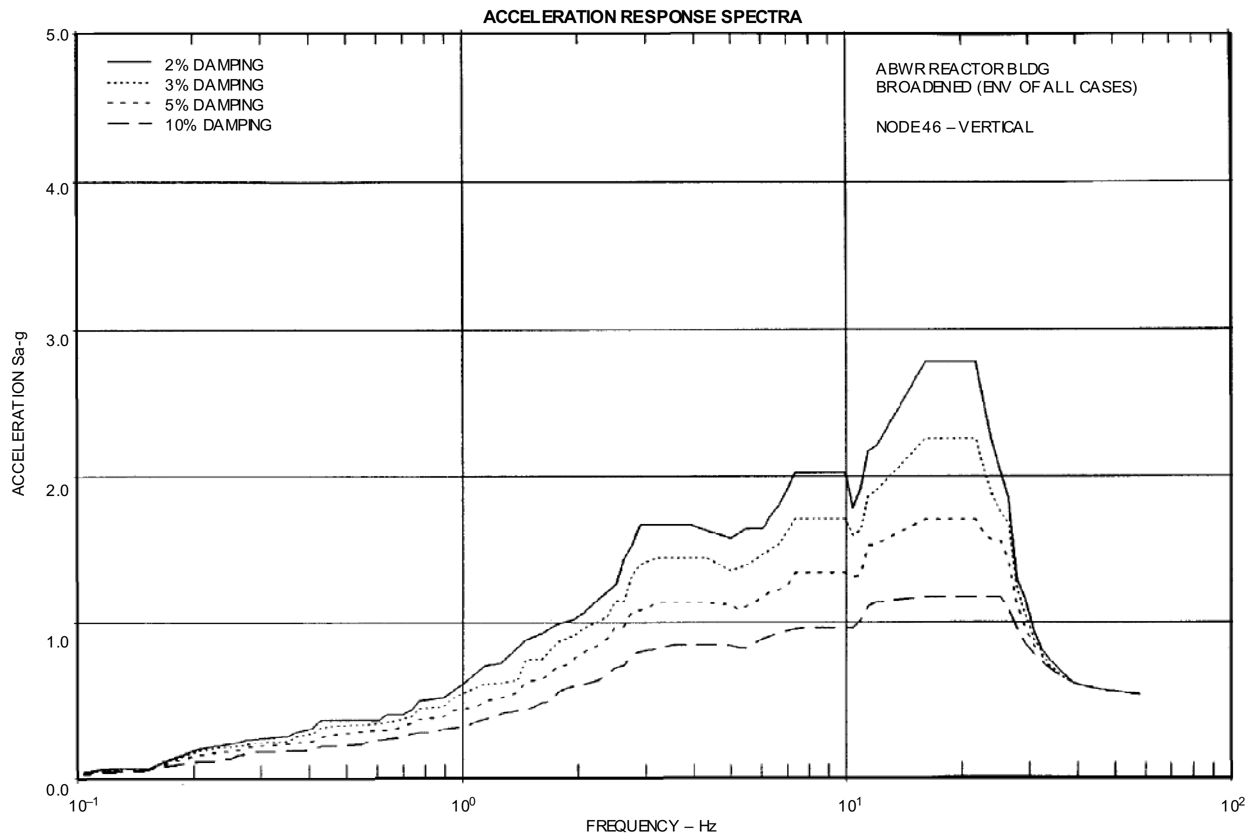


Figure 3A-174 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 46–Vertical

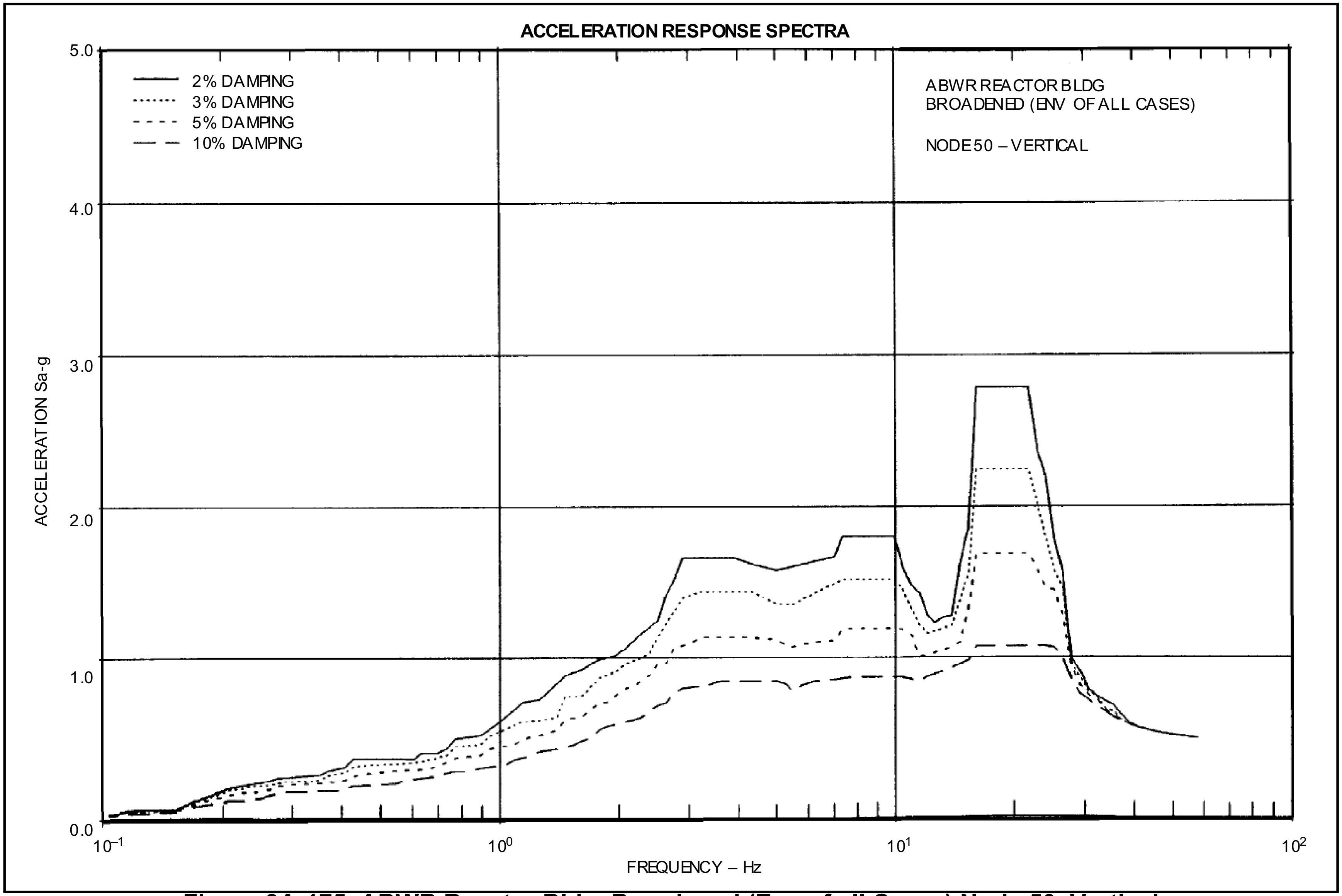


Figure 3A-175 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 50–Vertical

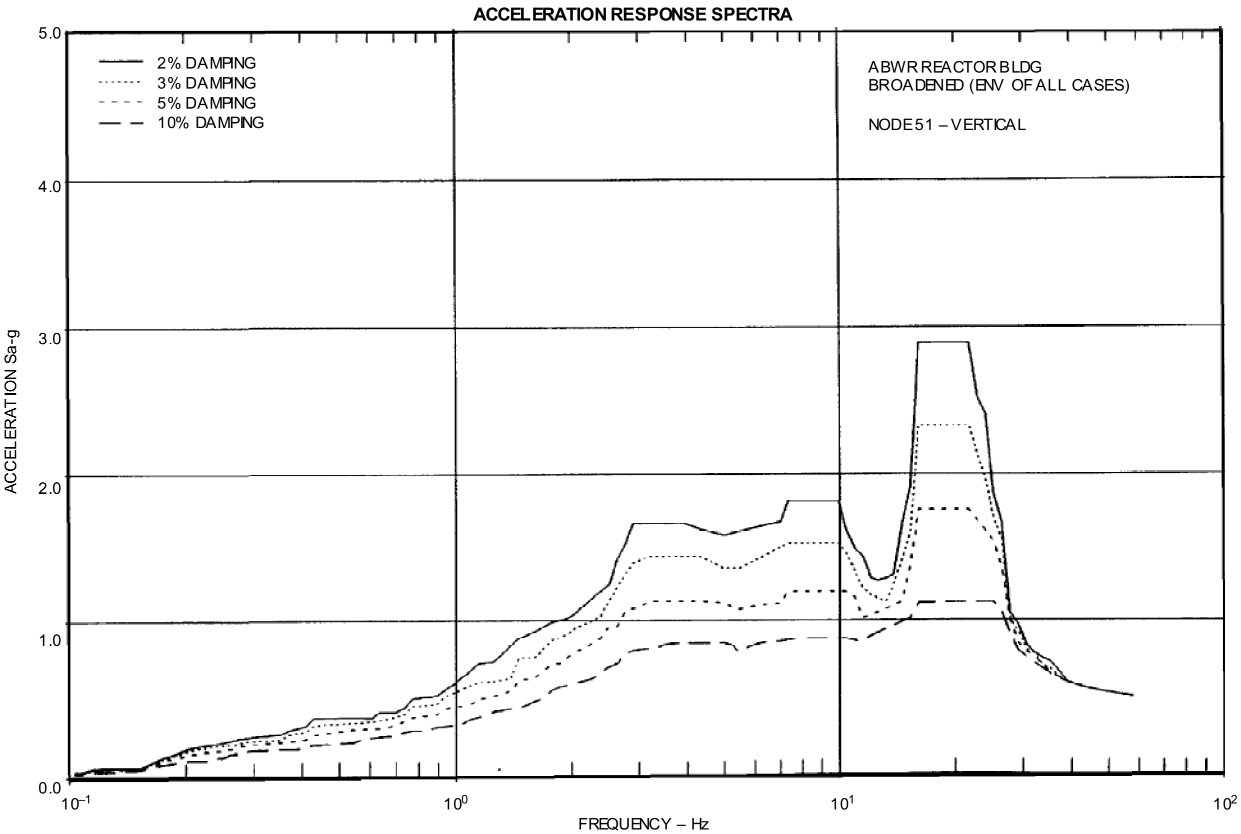


Figure 3A-176 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 51–Vertical

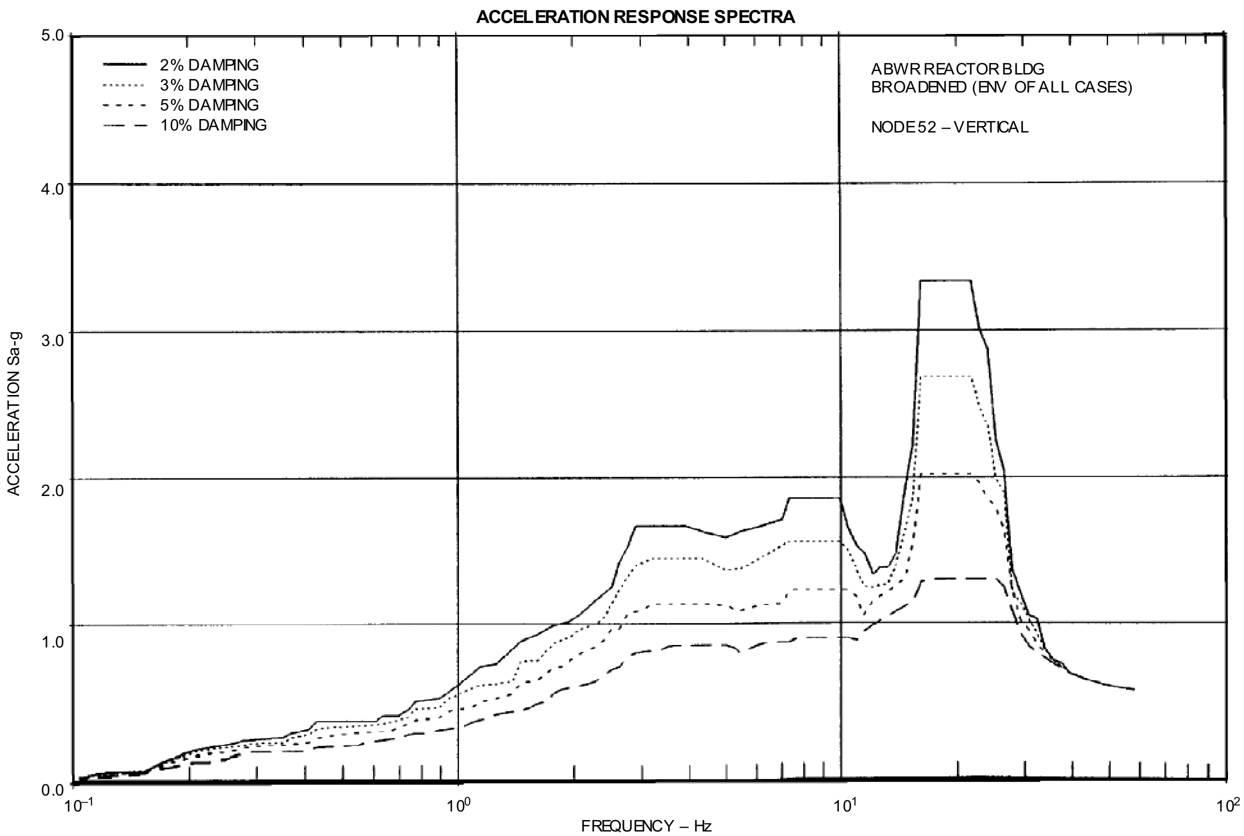


Figure 3A-177 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 52–Vertical

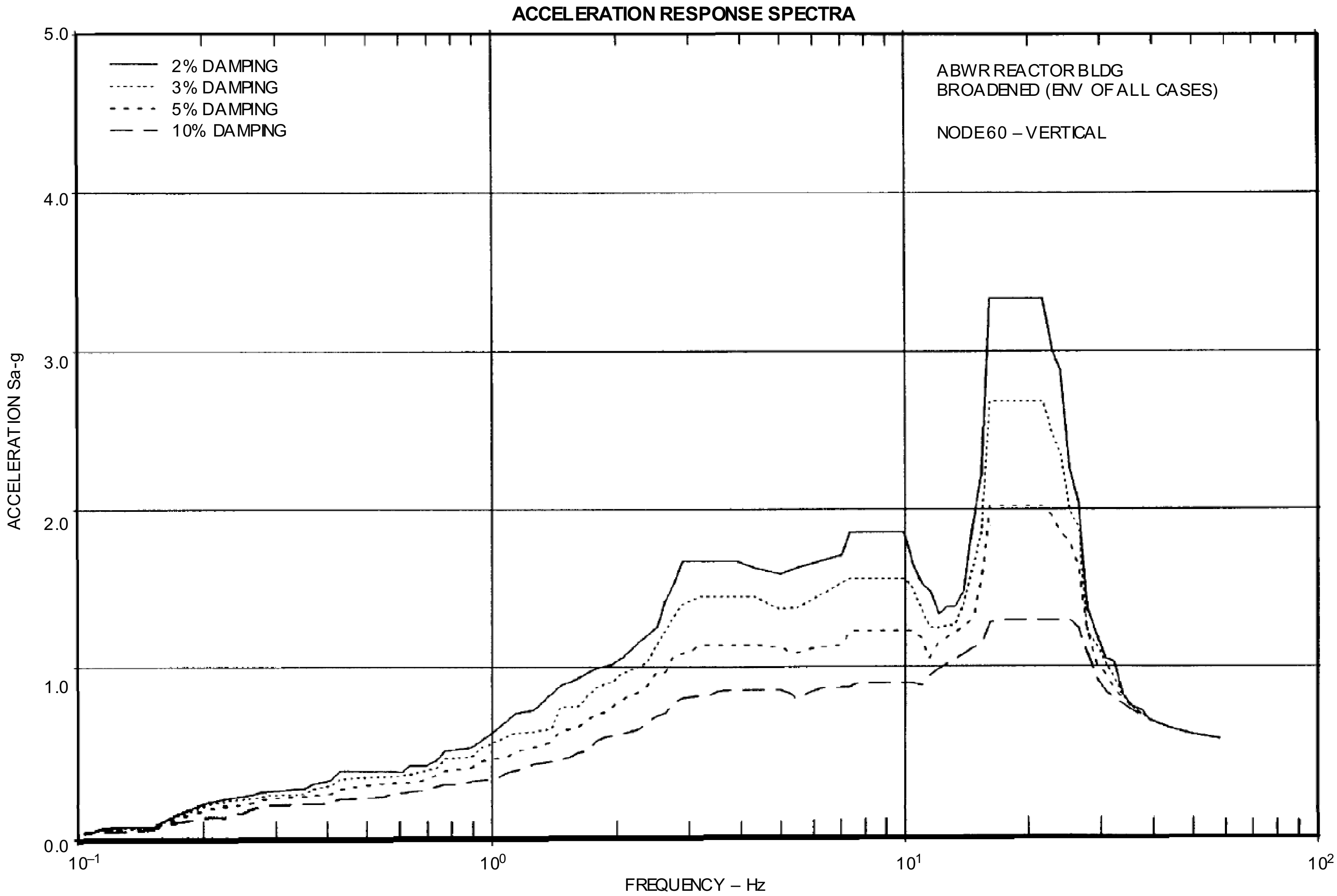


Figure 3A-178 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 60–Vertical

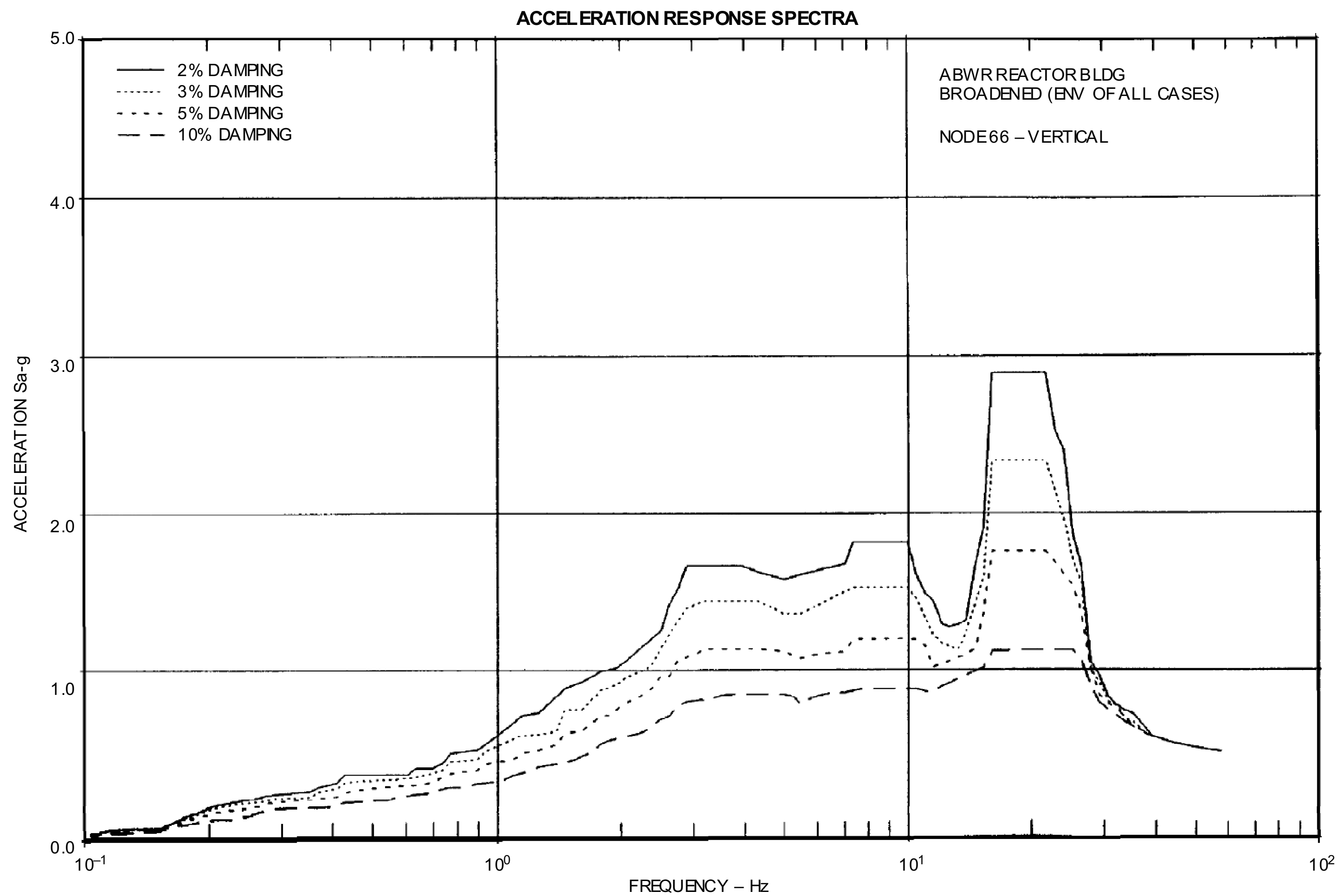


Figure 3A-179 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 66–Vertical

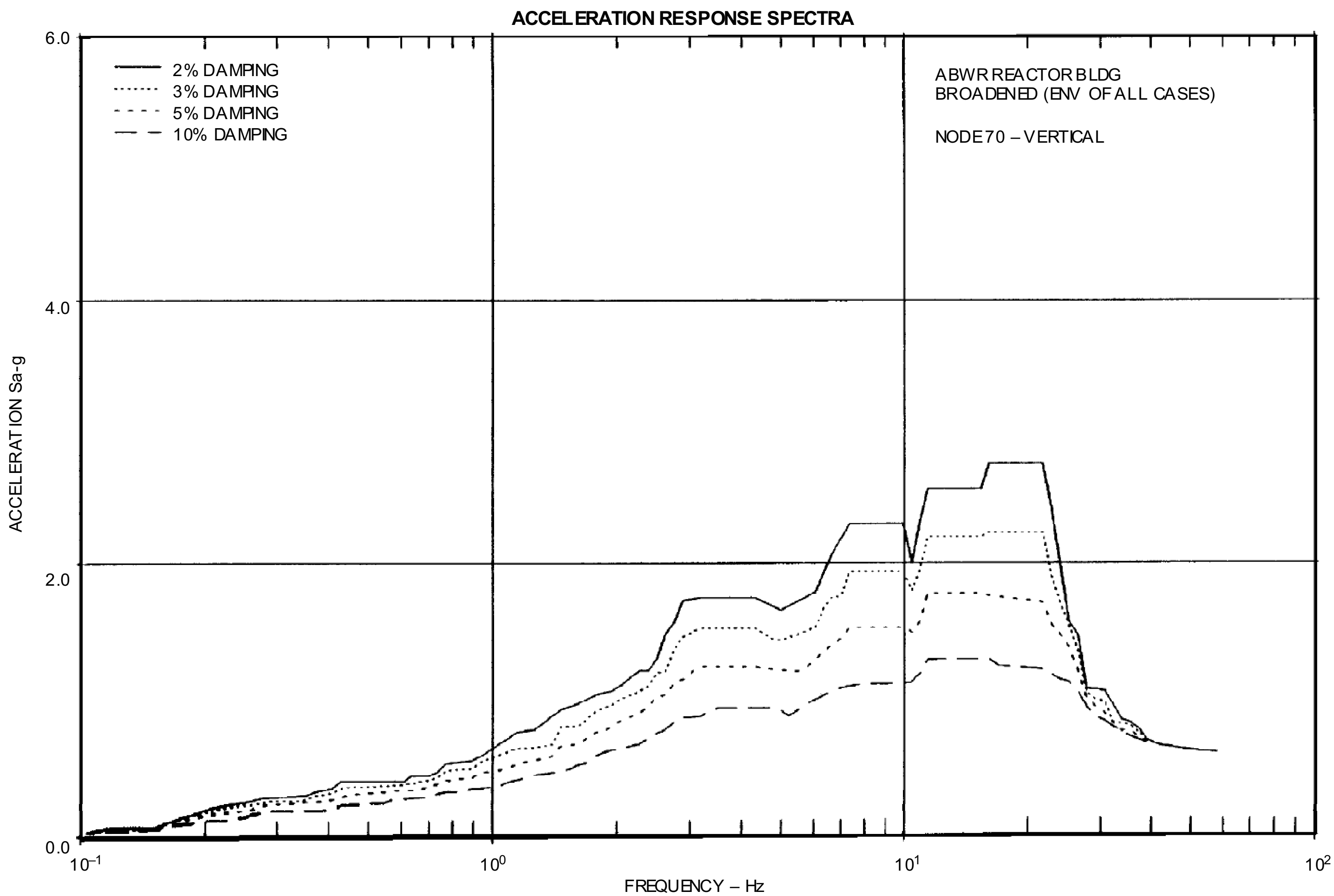


Figure 3A-180 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 70–Vertical

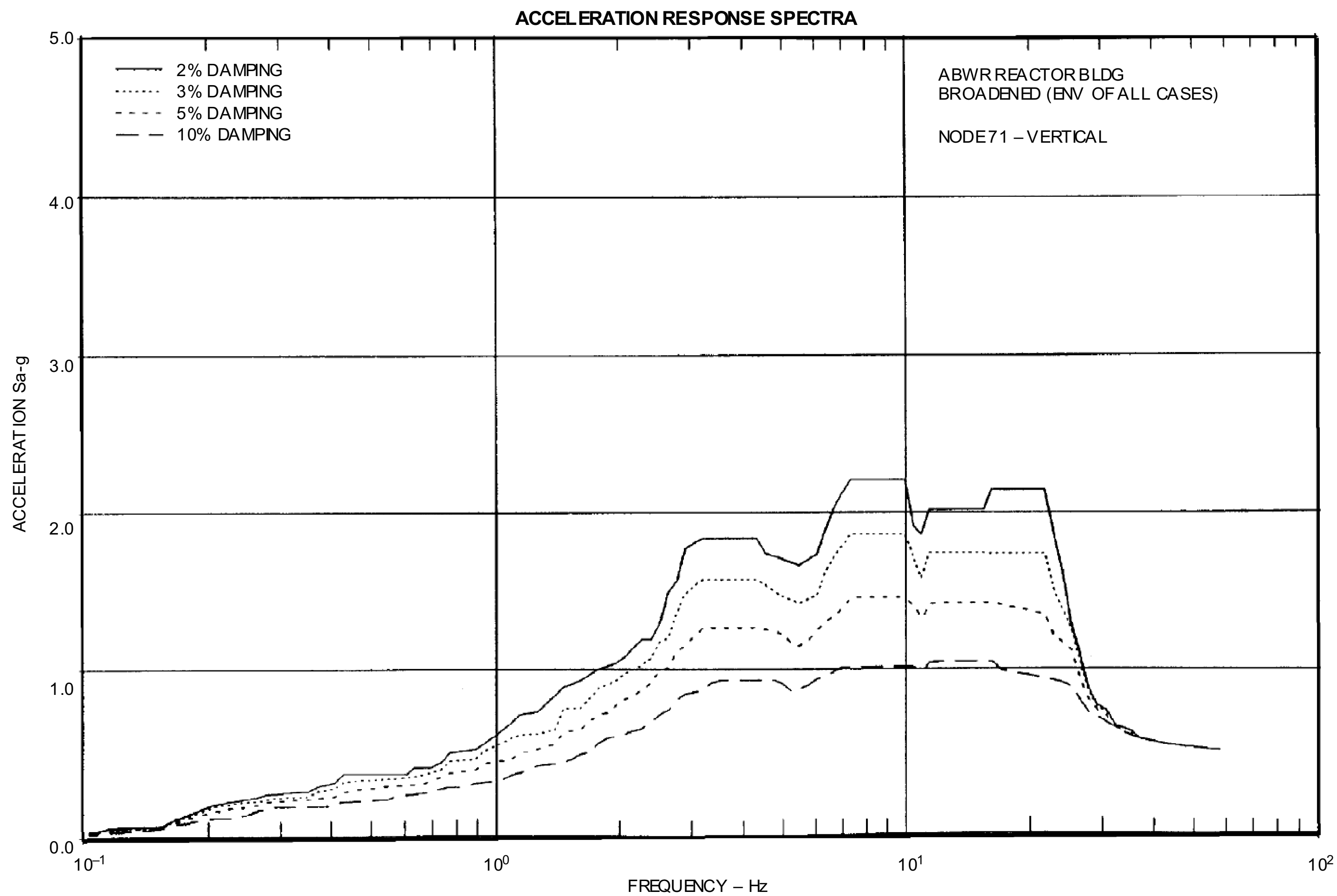


Figure 3A-181 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 71–Vertical

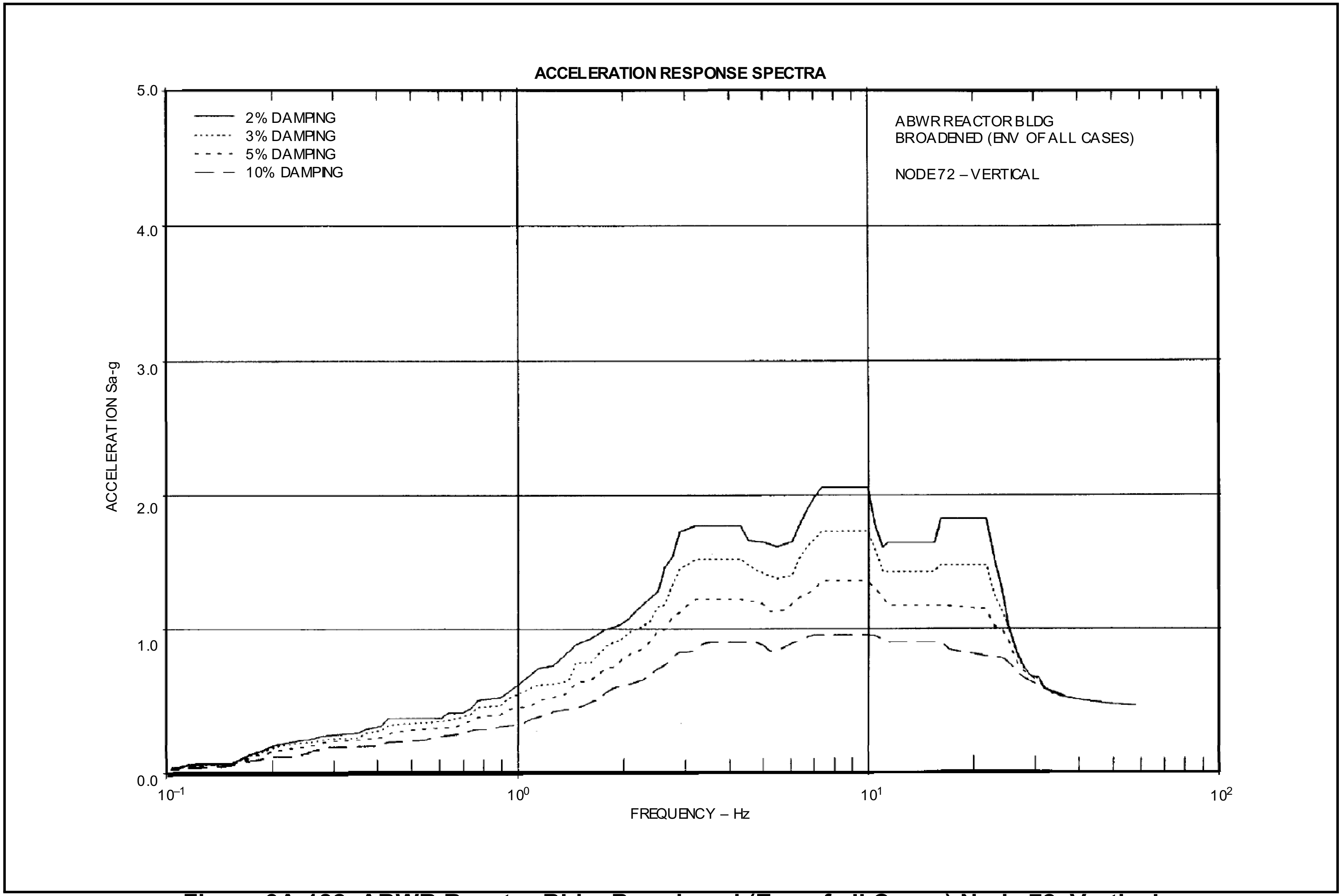


Figure 3A-182 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 72–Vertical

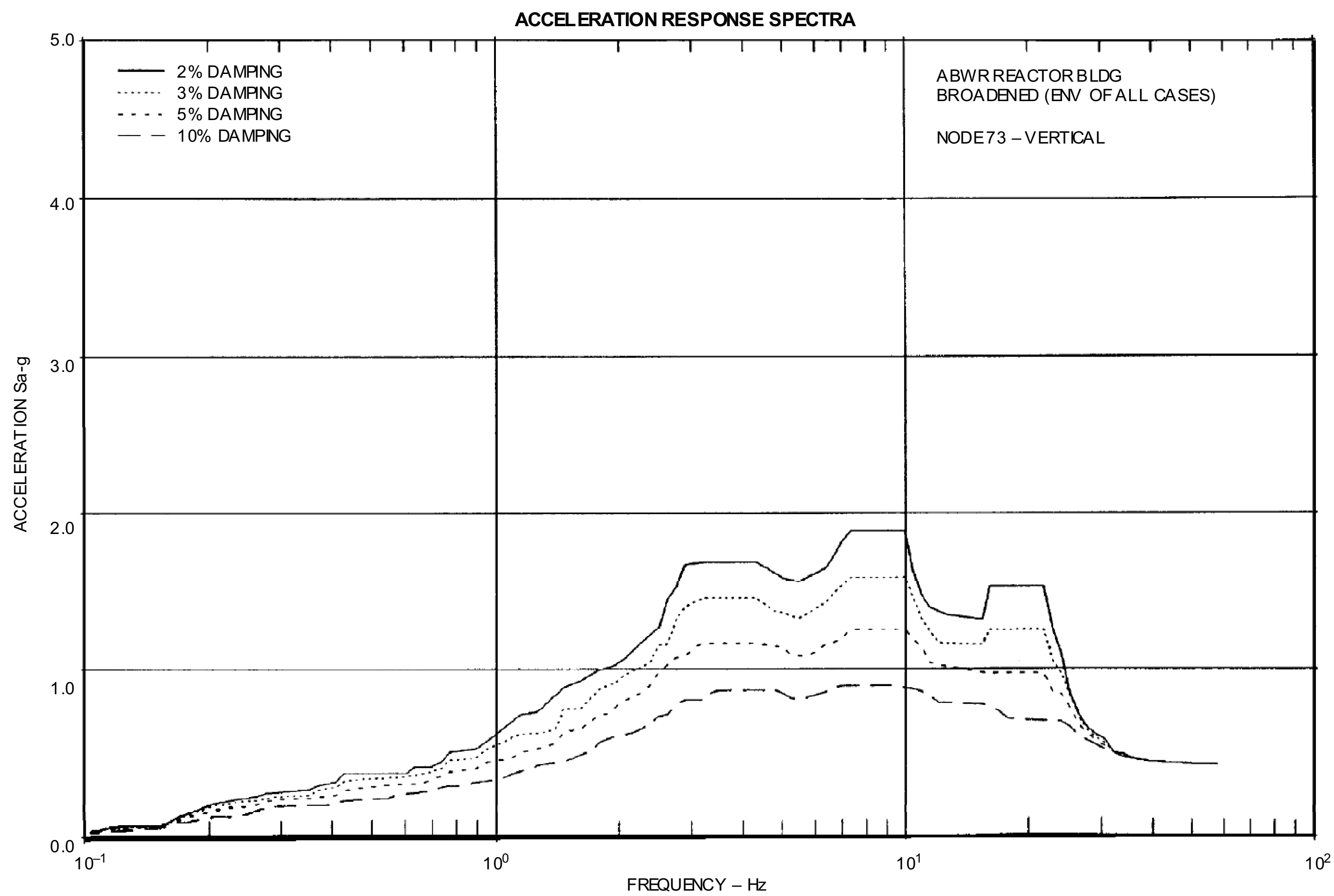


Figure 3A-183 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 73–Vertical

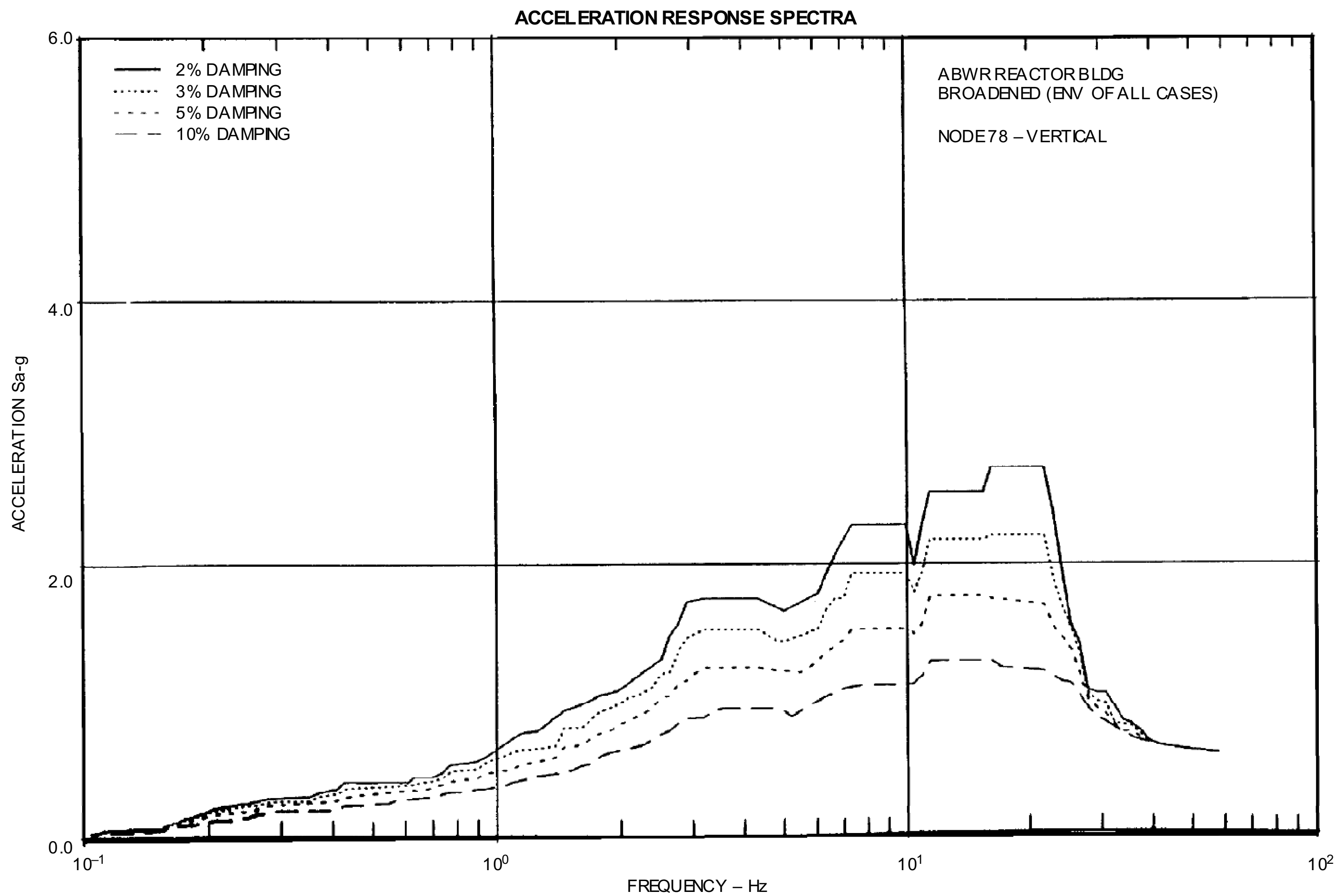


Figure 3A-184 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 78–Vertical

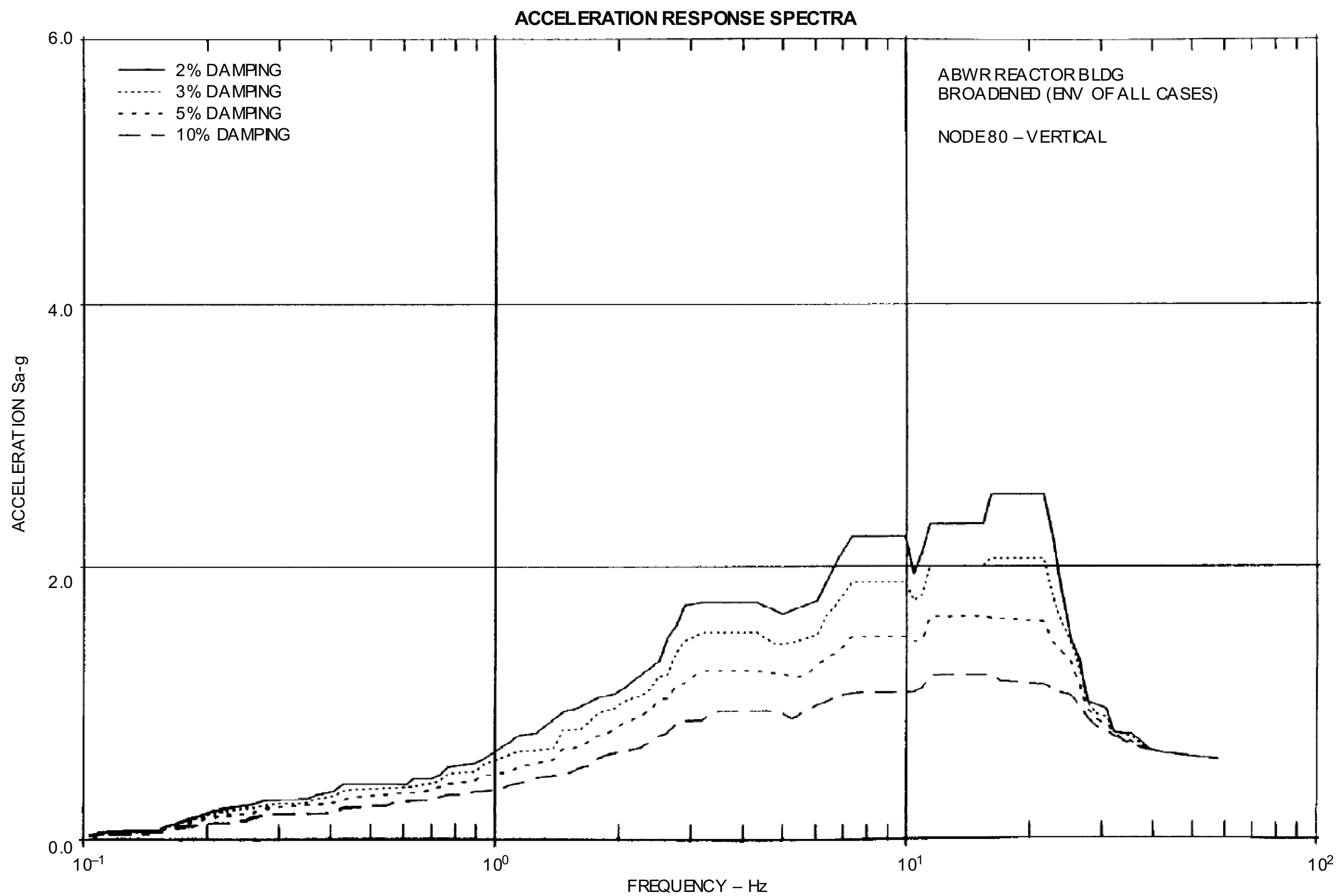


Figure 3A-185 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 80–Vertical

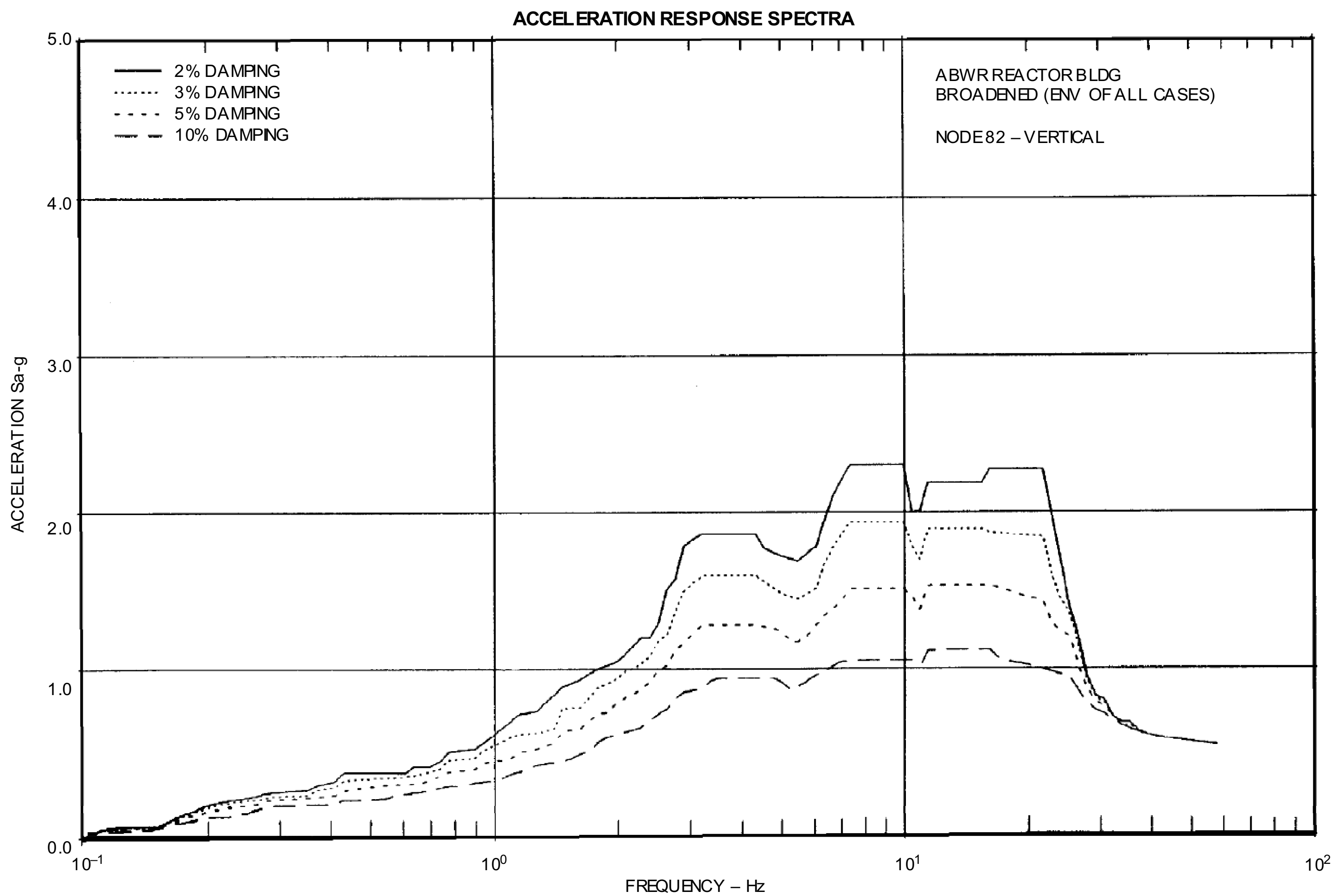


Figure 3A-186 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 82–Vertical

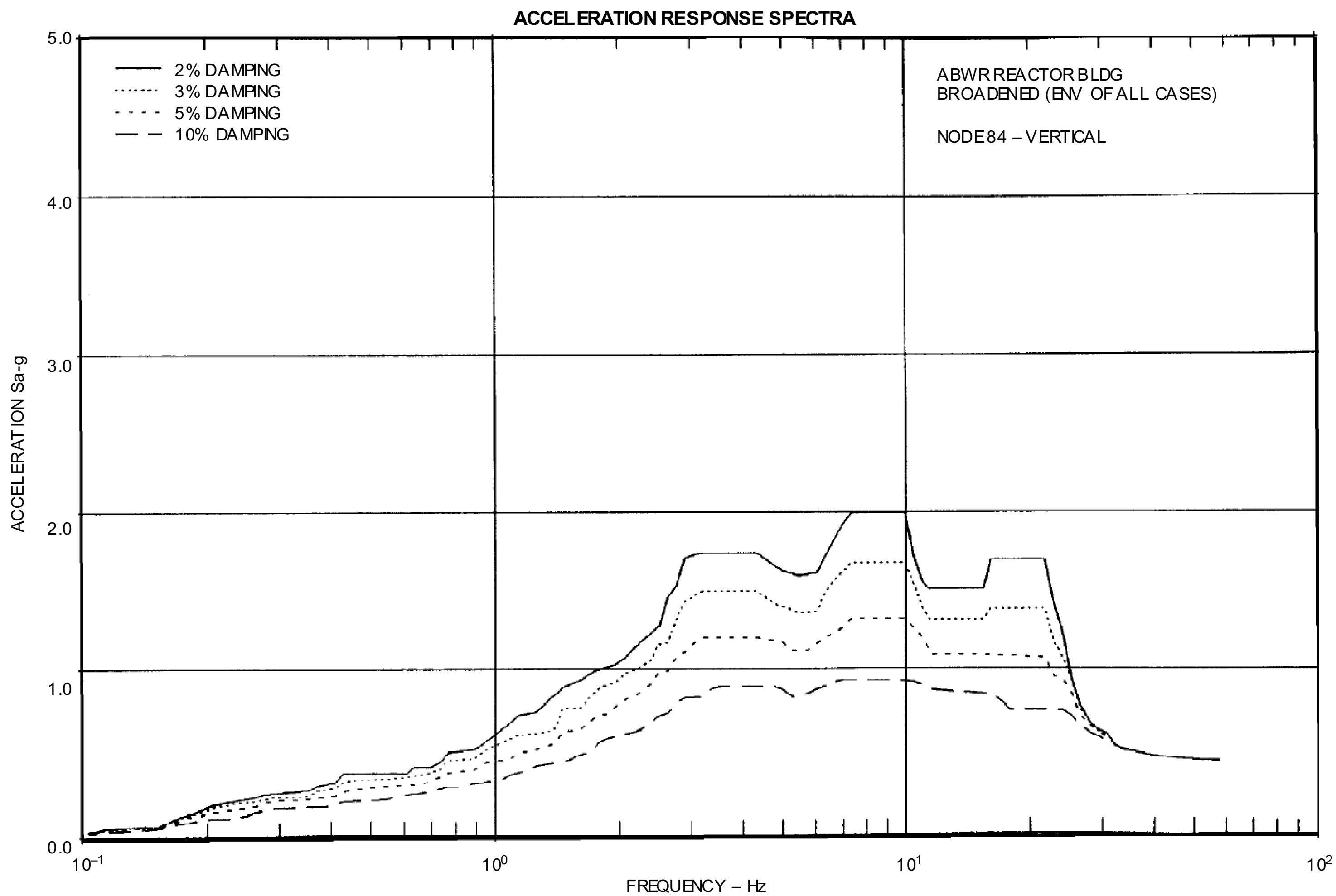


Figure 3A-187 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 84–Vertical

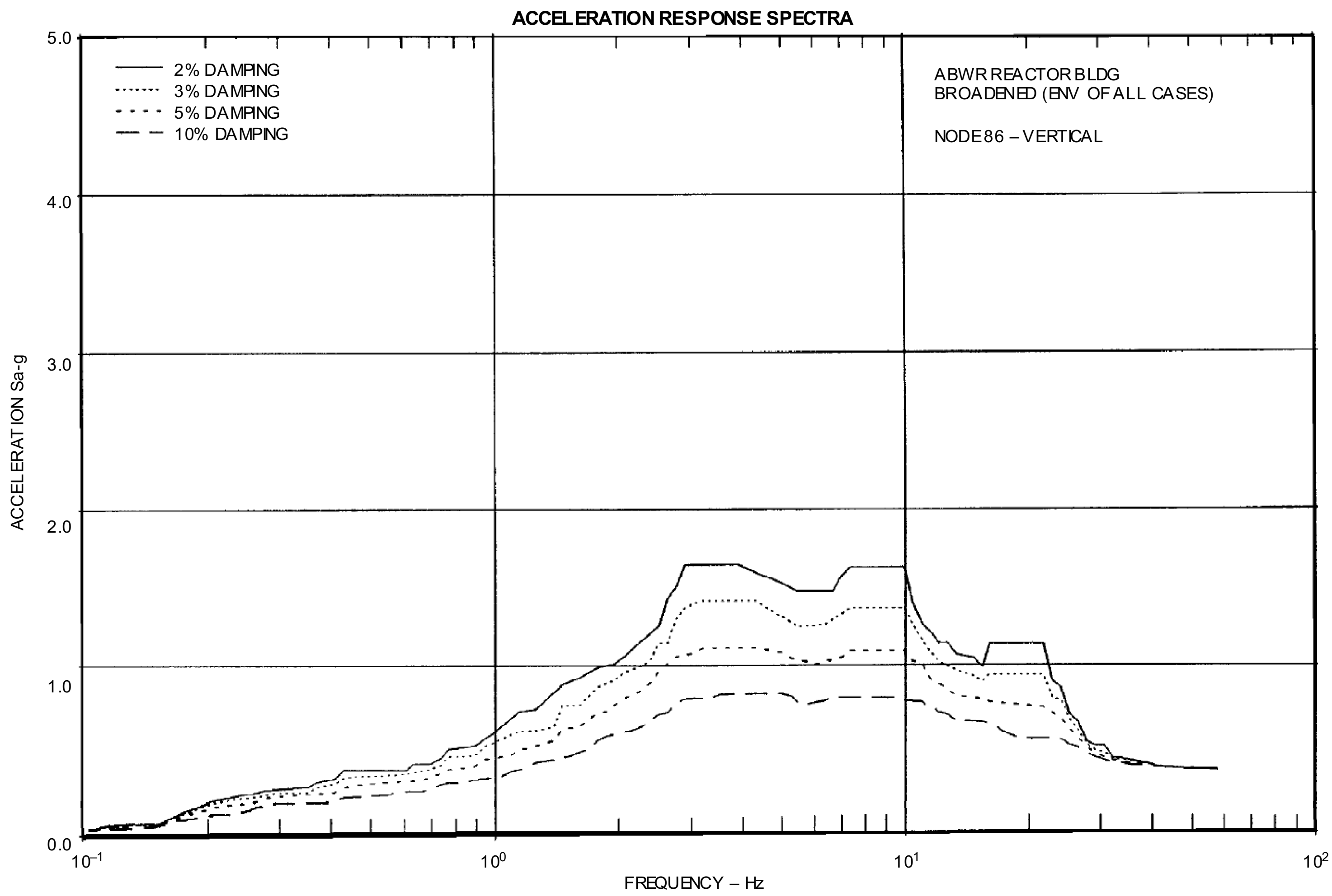


Figure 3A-188 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 86–Vertical

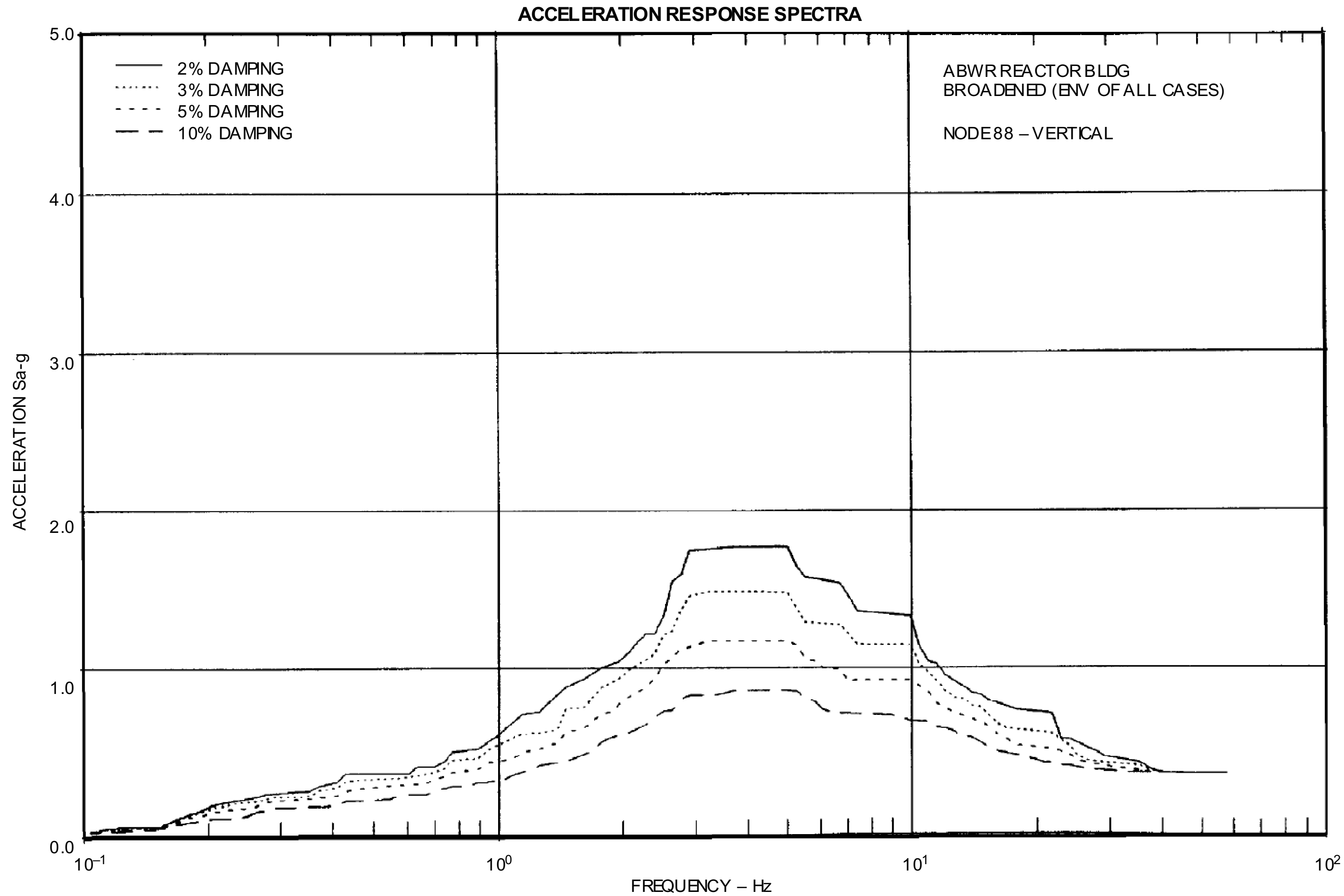


Figure 3A-189 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 88–Vertical

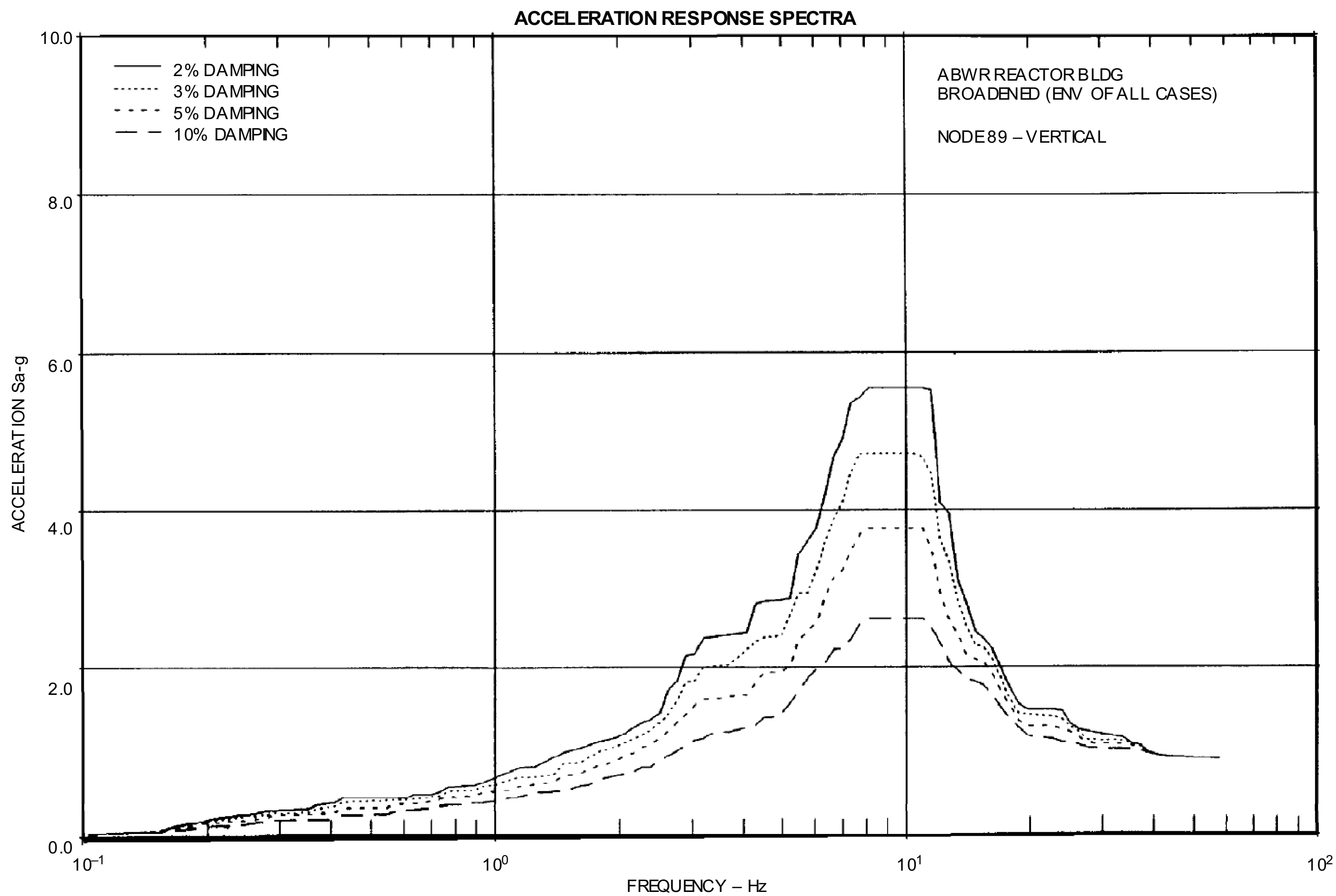


Figure 3A-190 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 89–Vertical

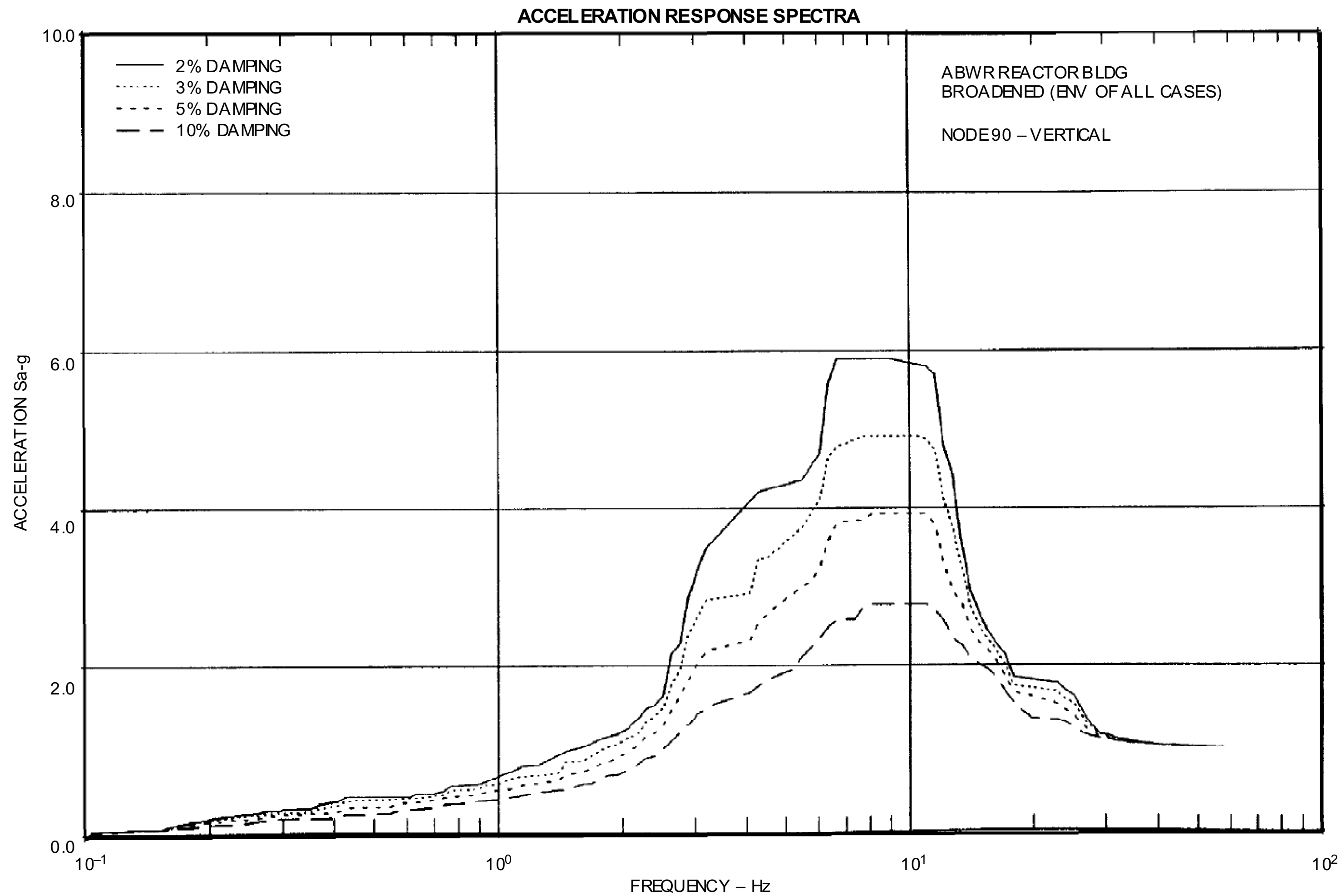


Figure 3A-191 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 90–Vertical

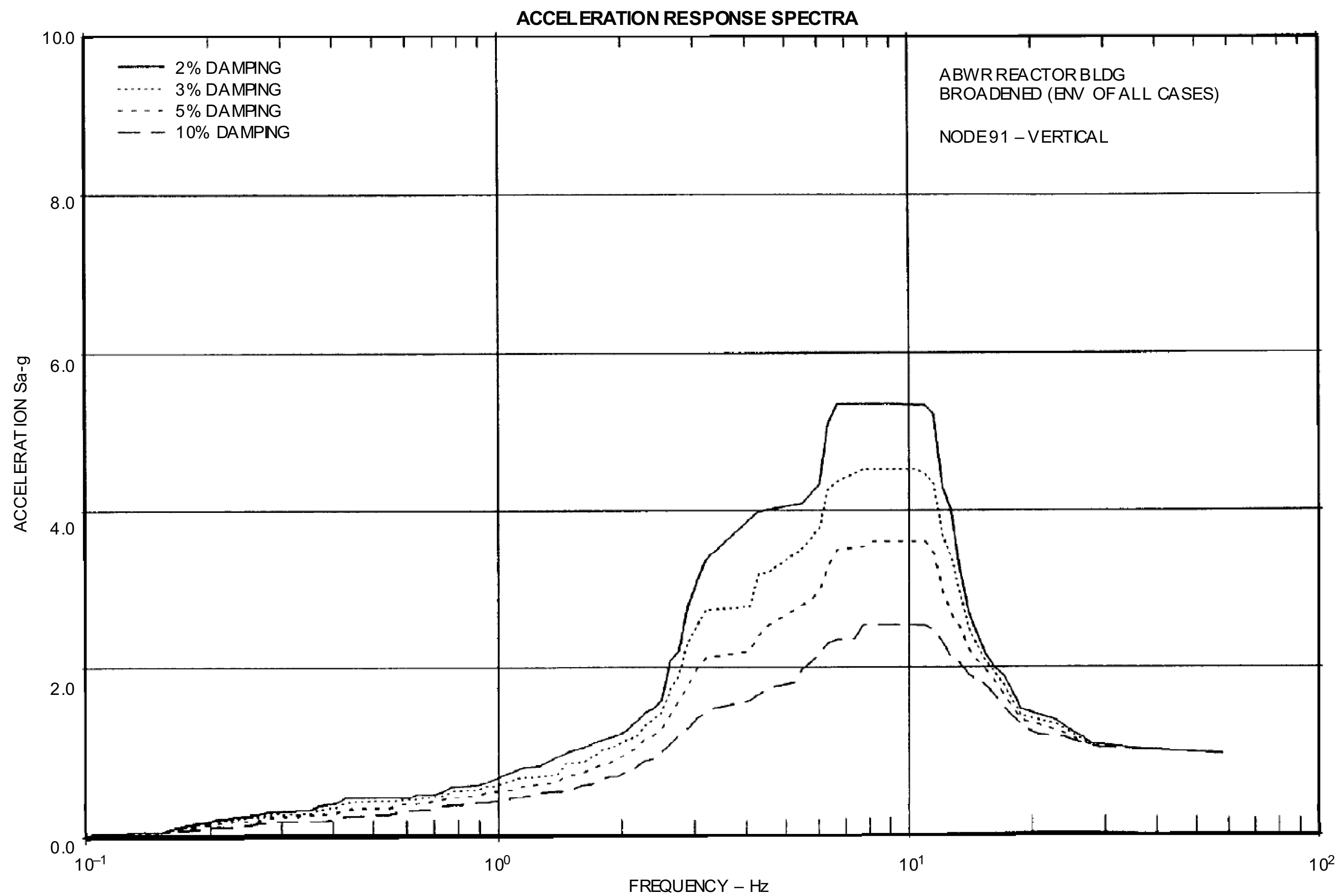
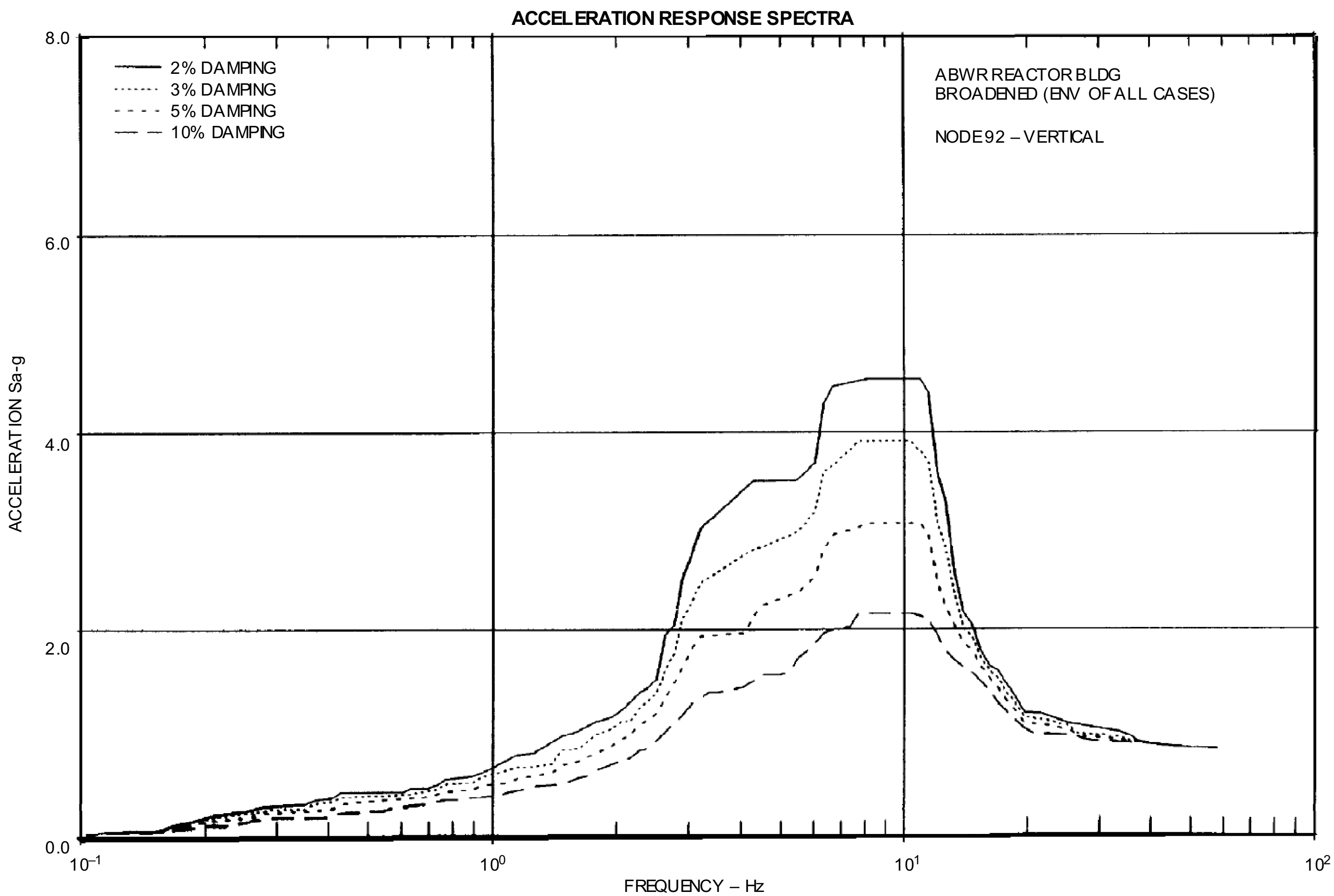


Figure 3A-192 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 91–Vertical



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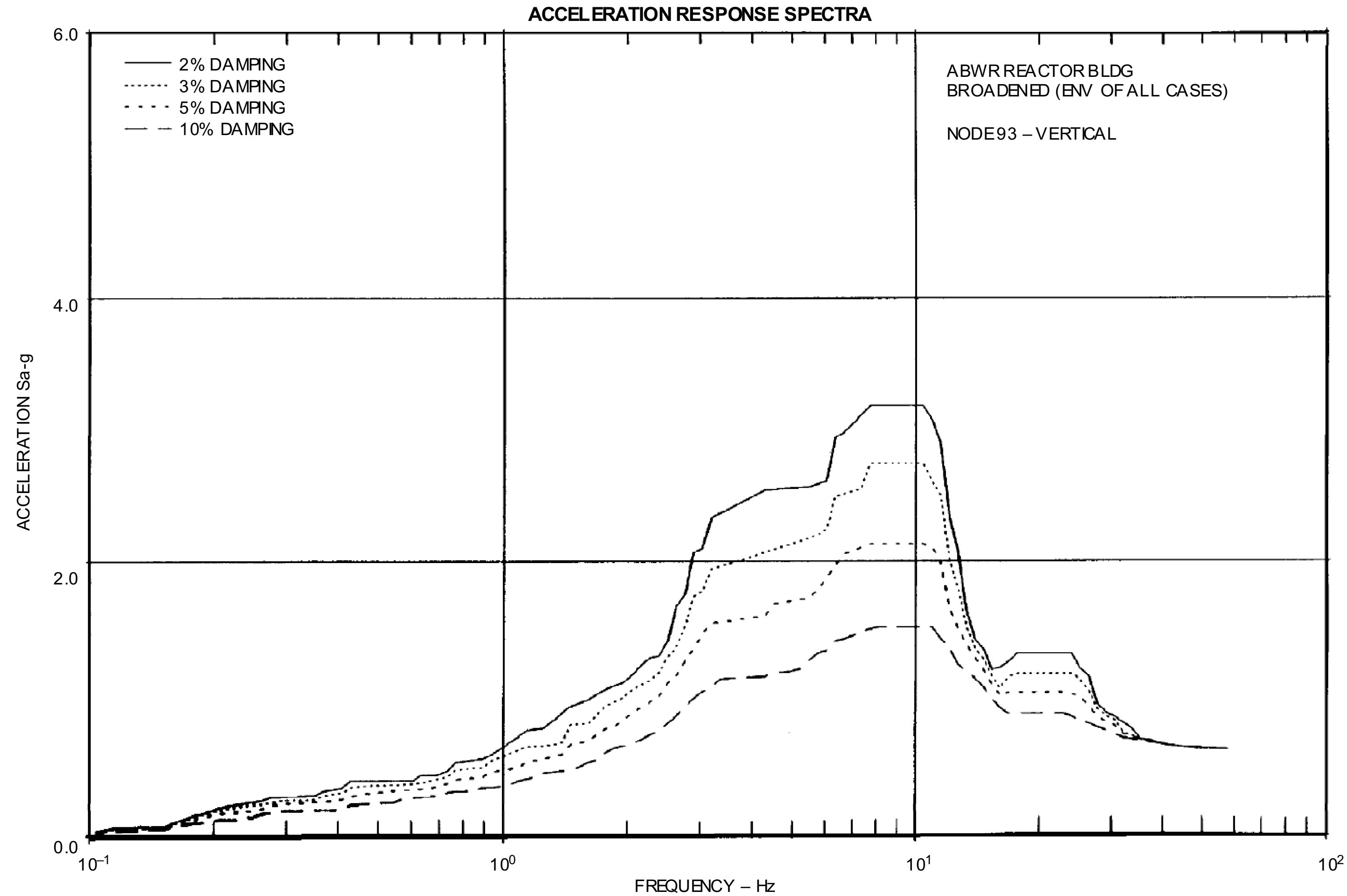

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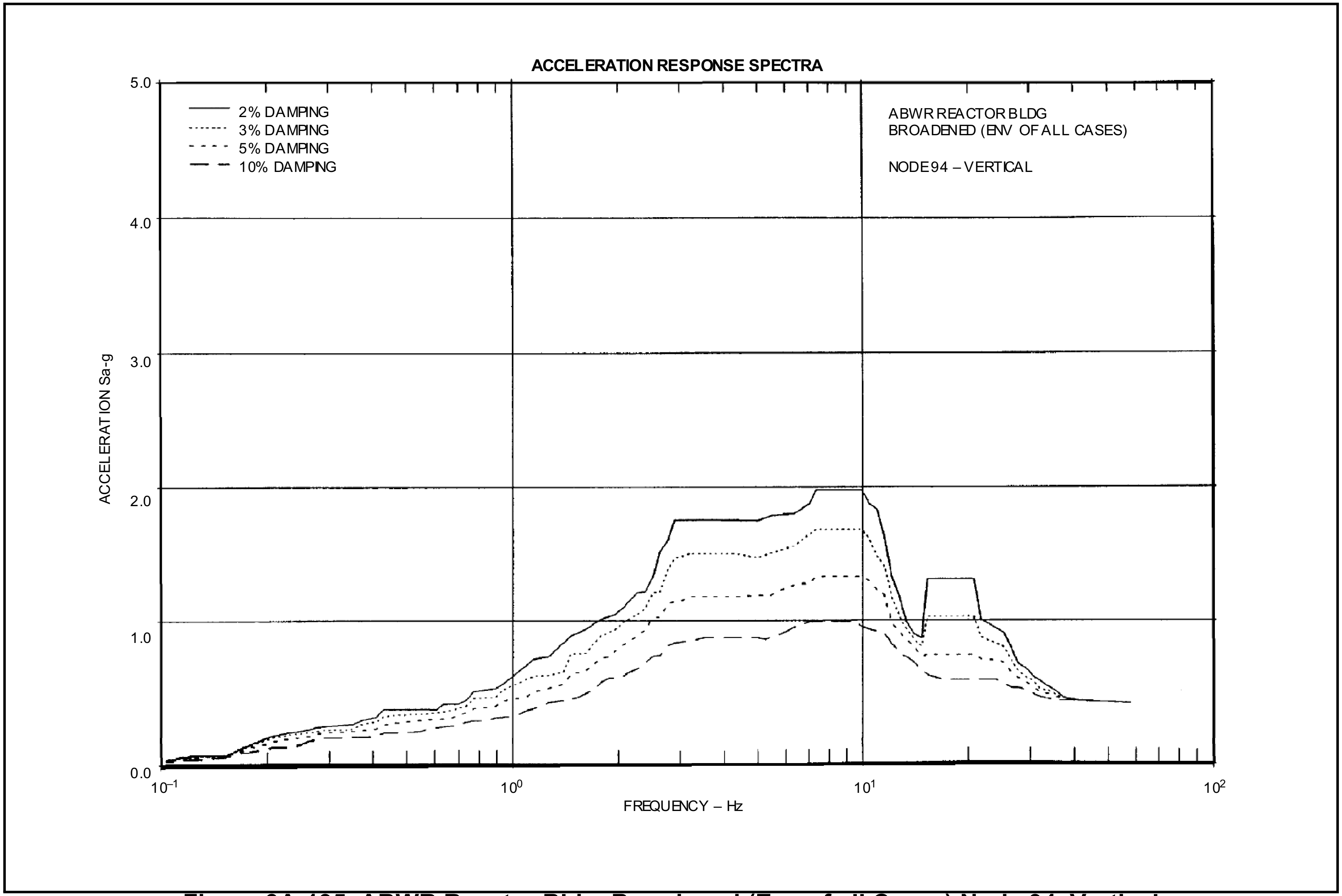
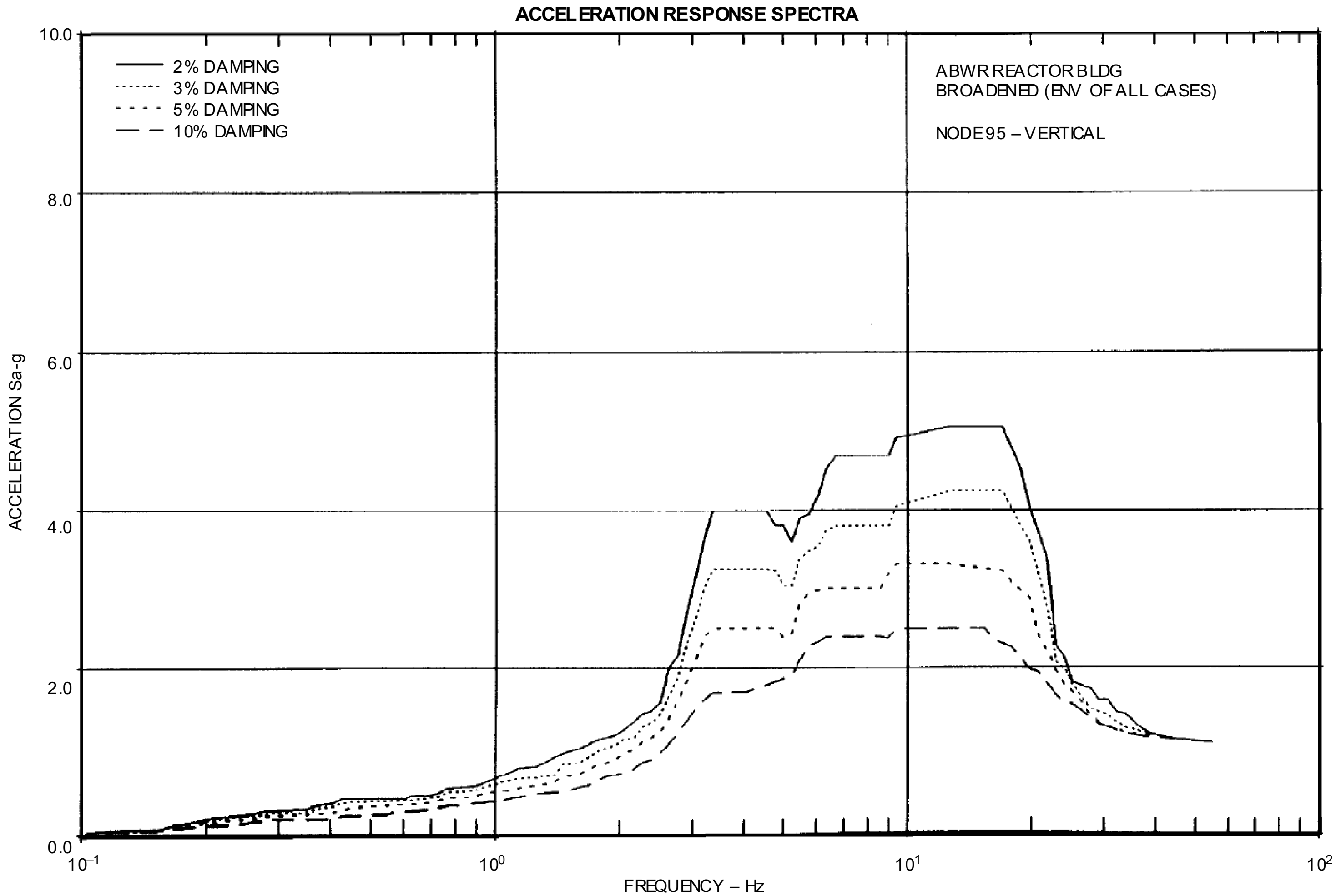


Figure 3A-195 ABWR Reactor Bldg. Broadened (Env of all Cases) Node 94–Vertical



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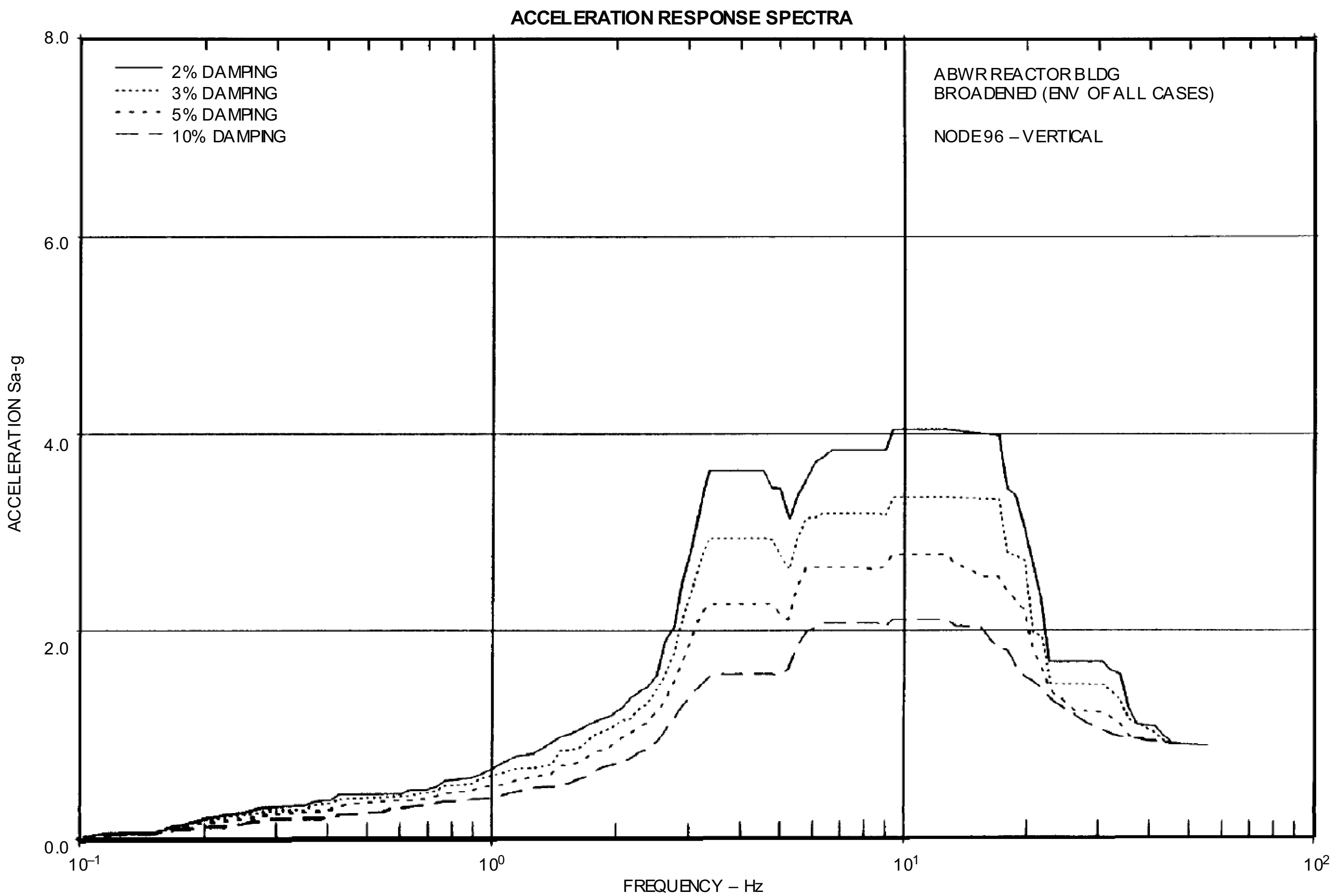

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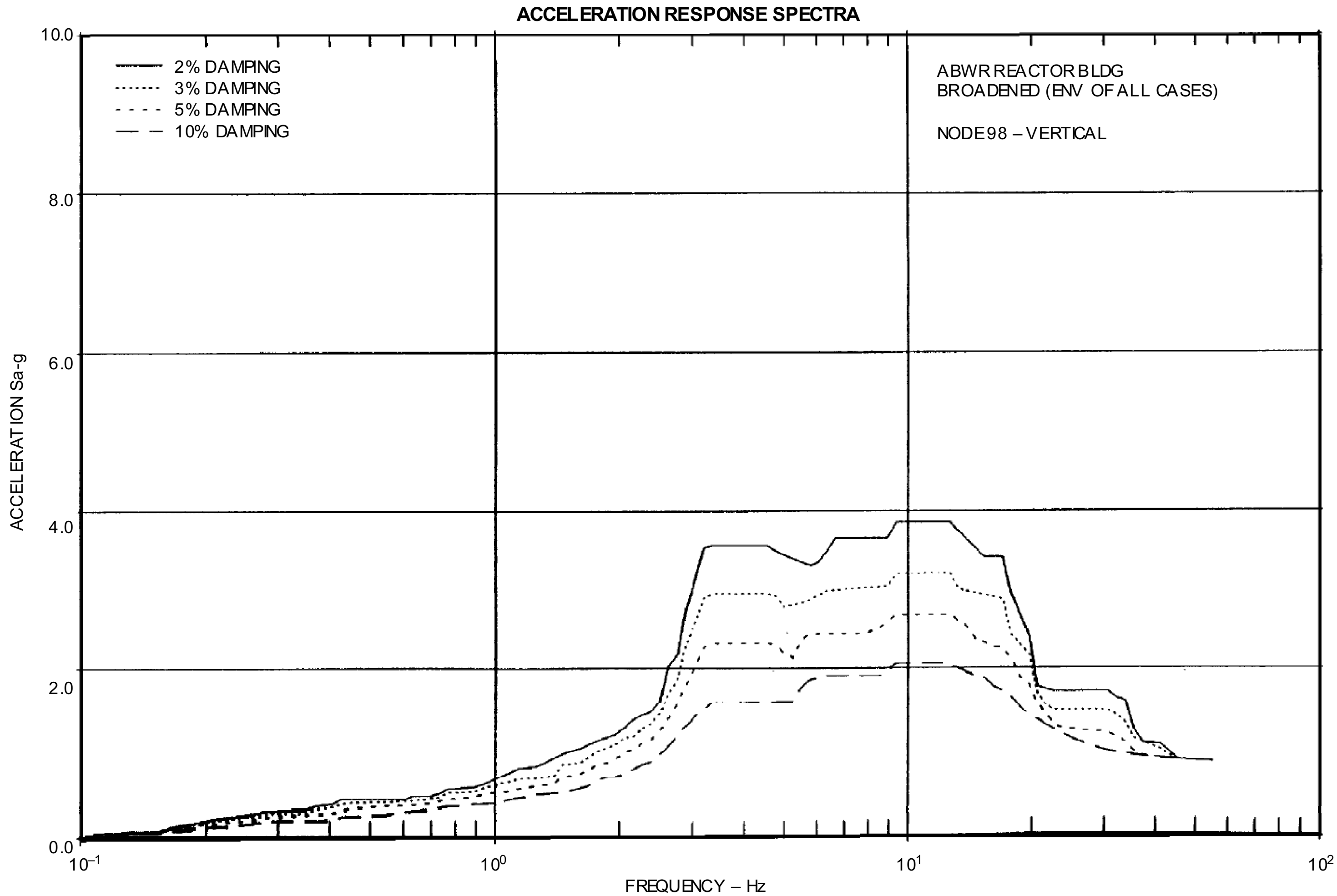

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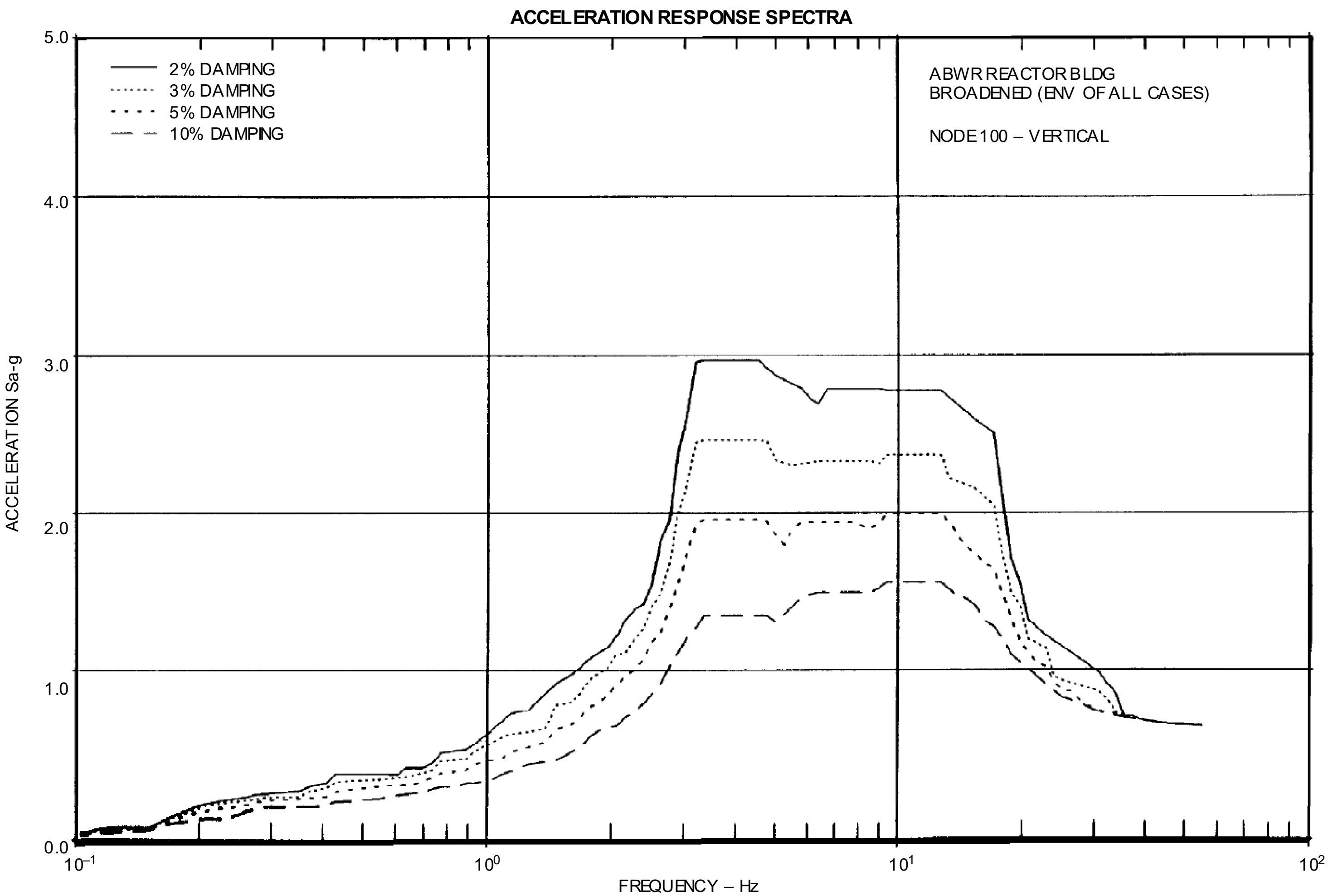

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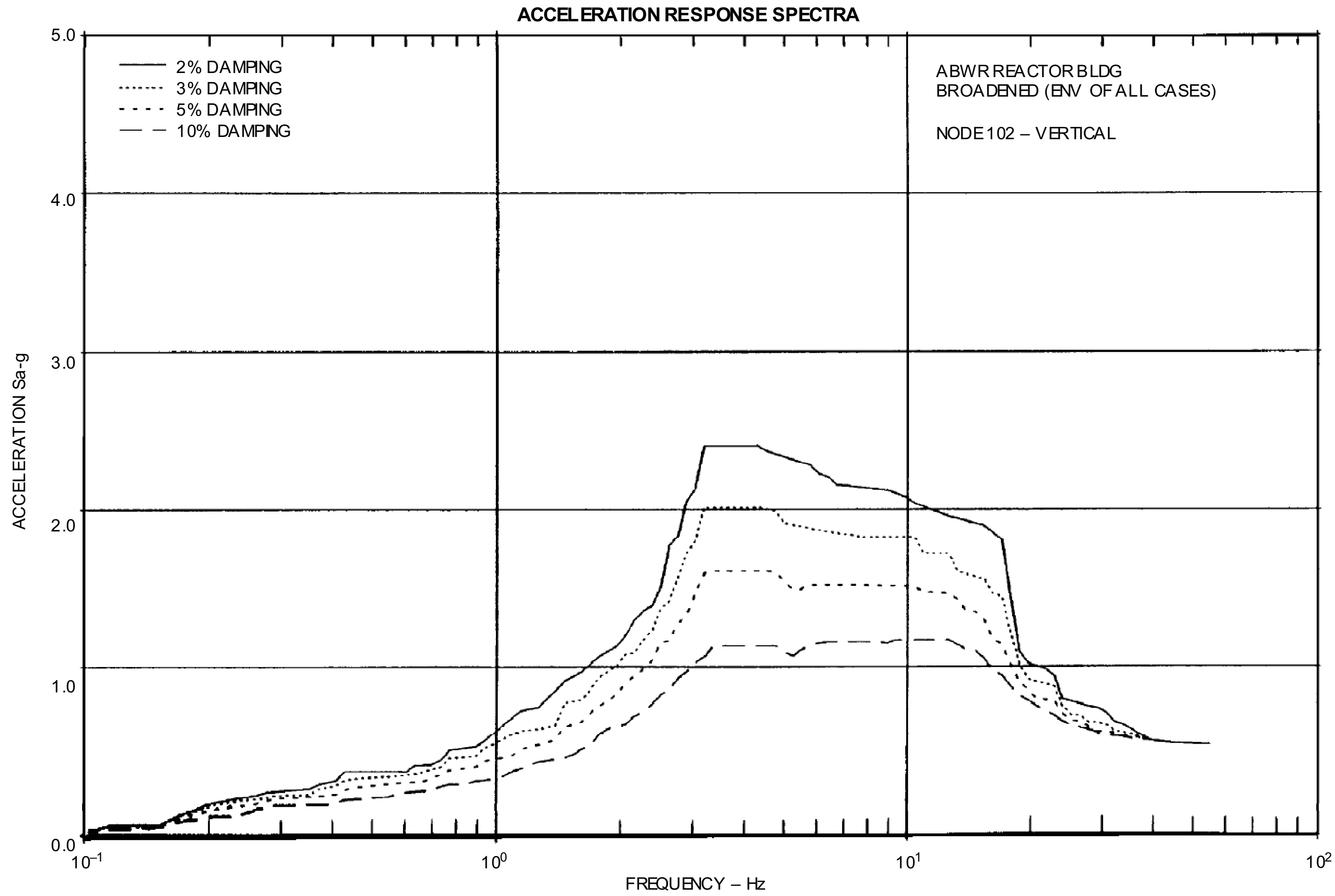

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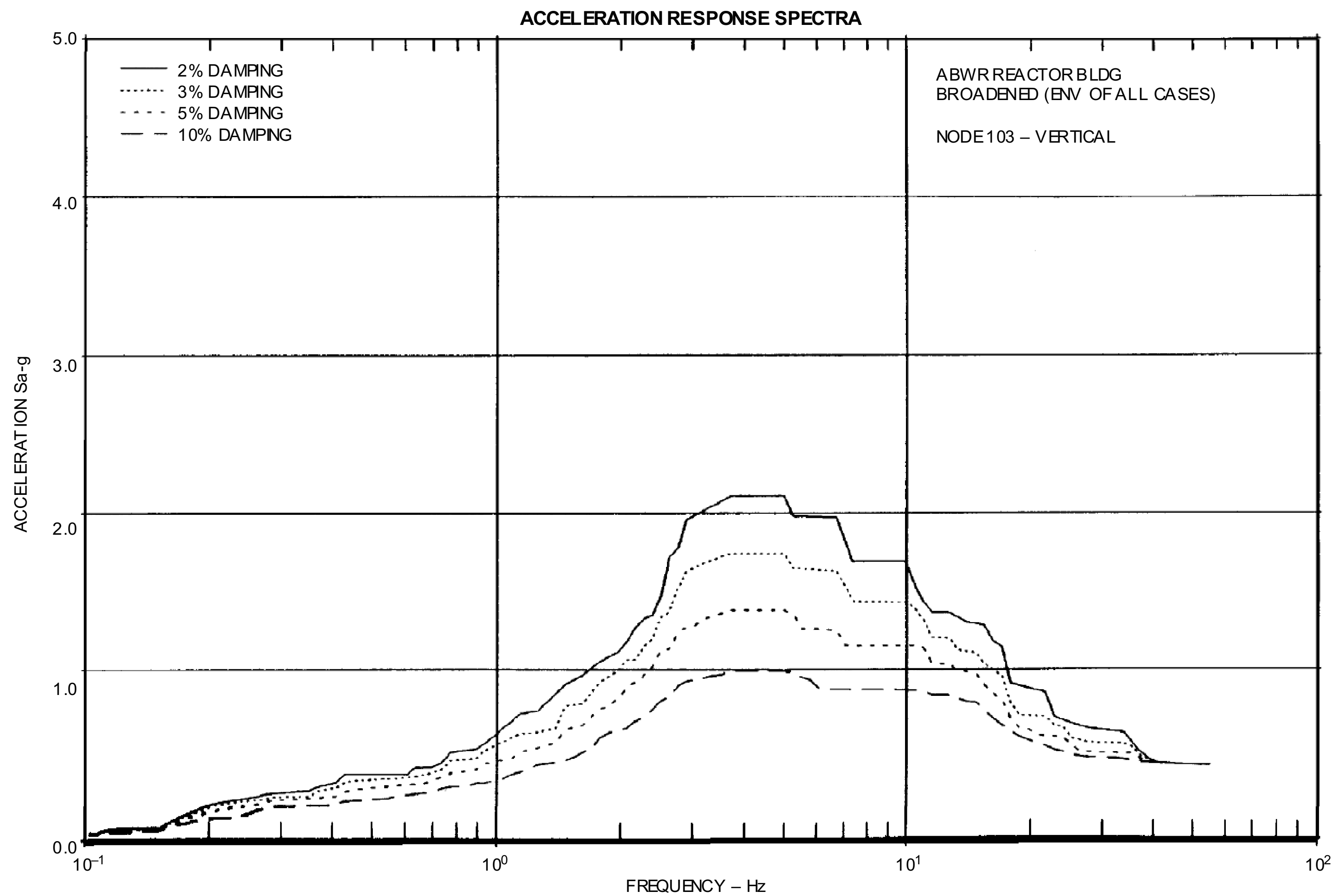

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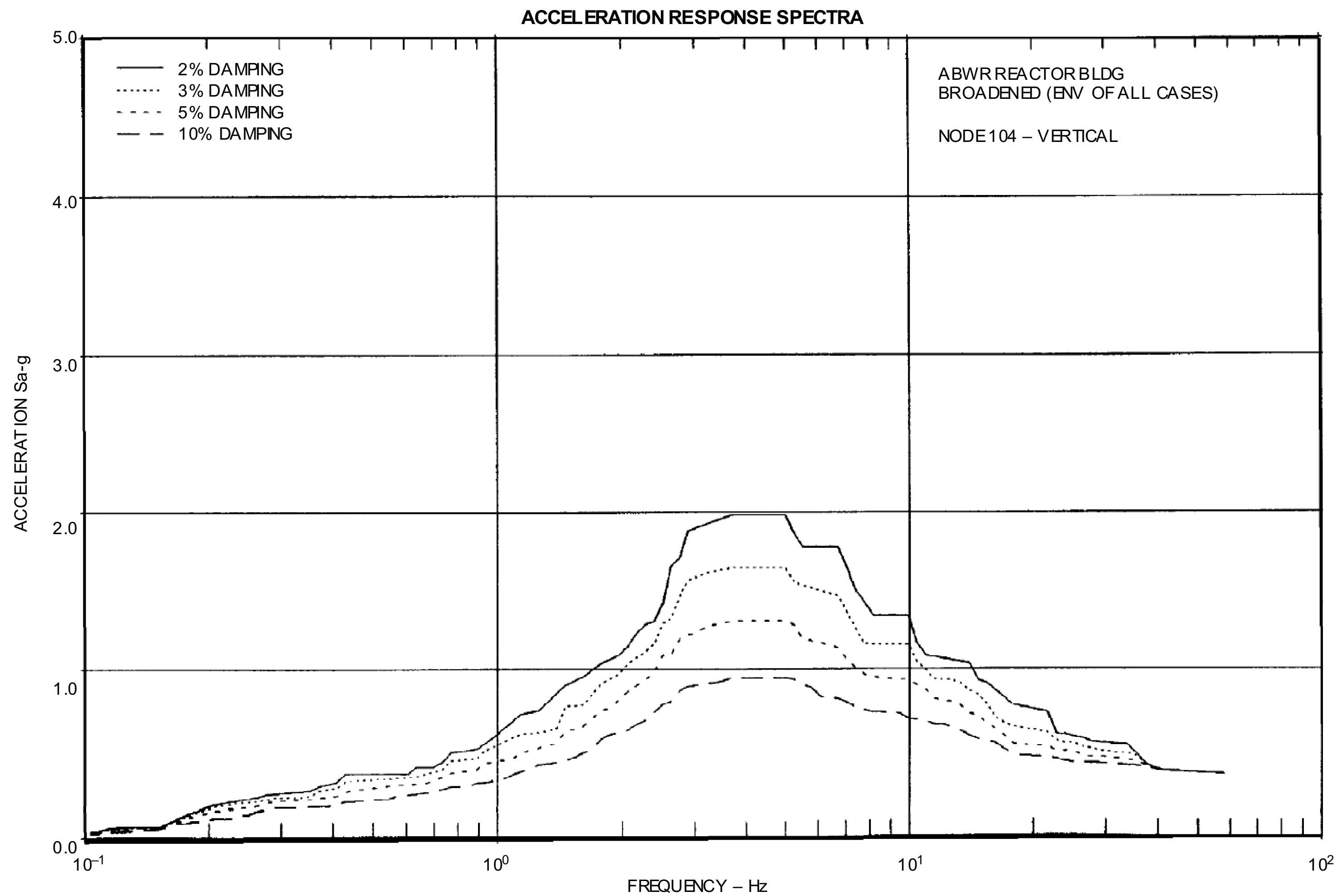

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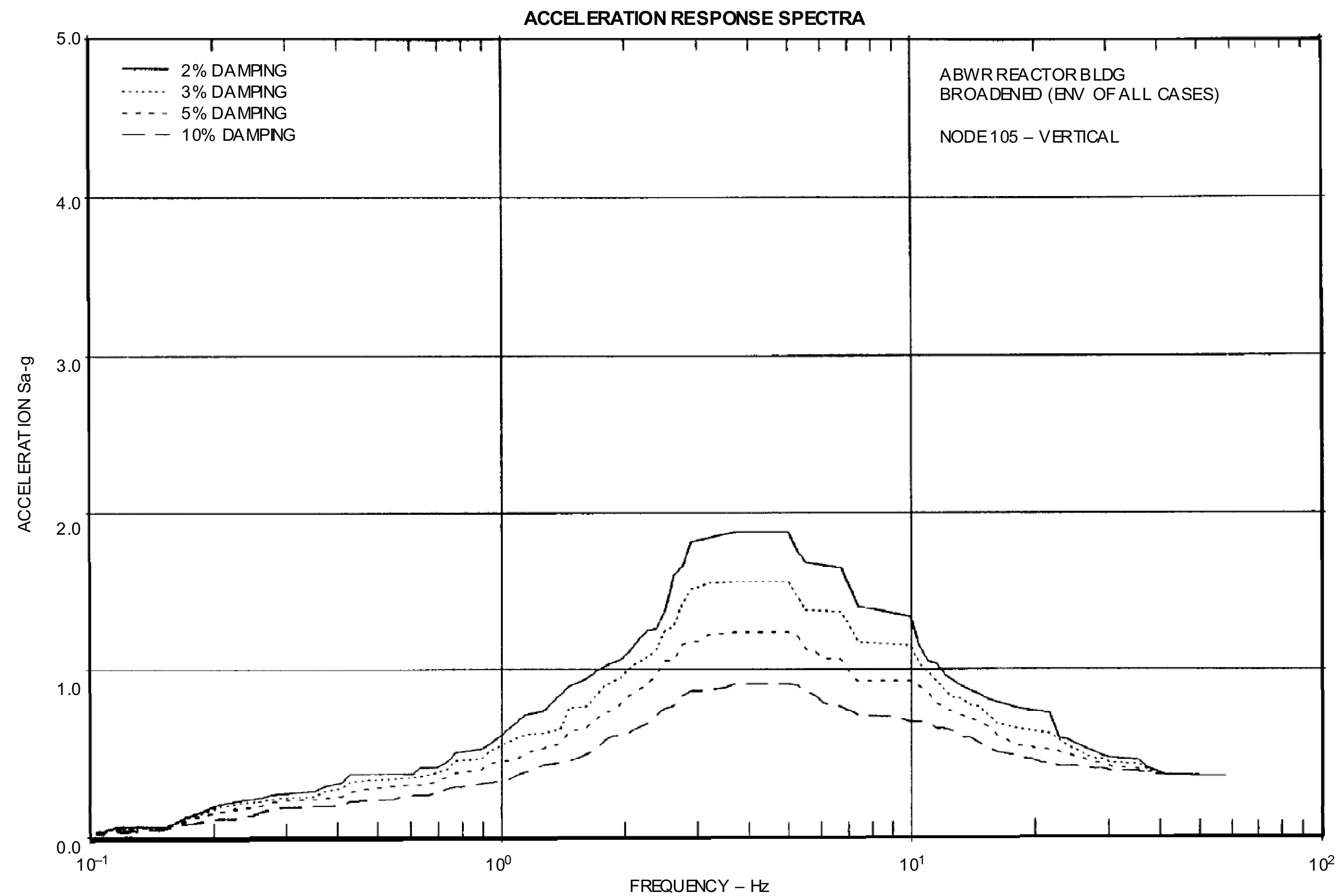

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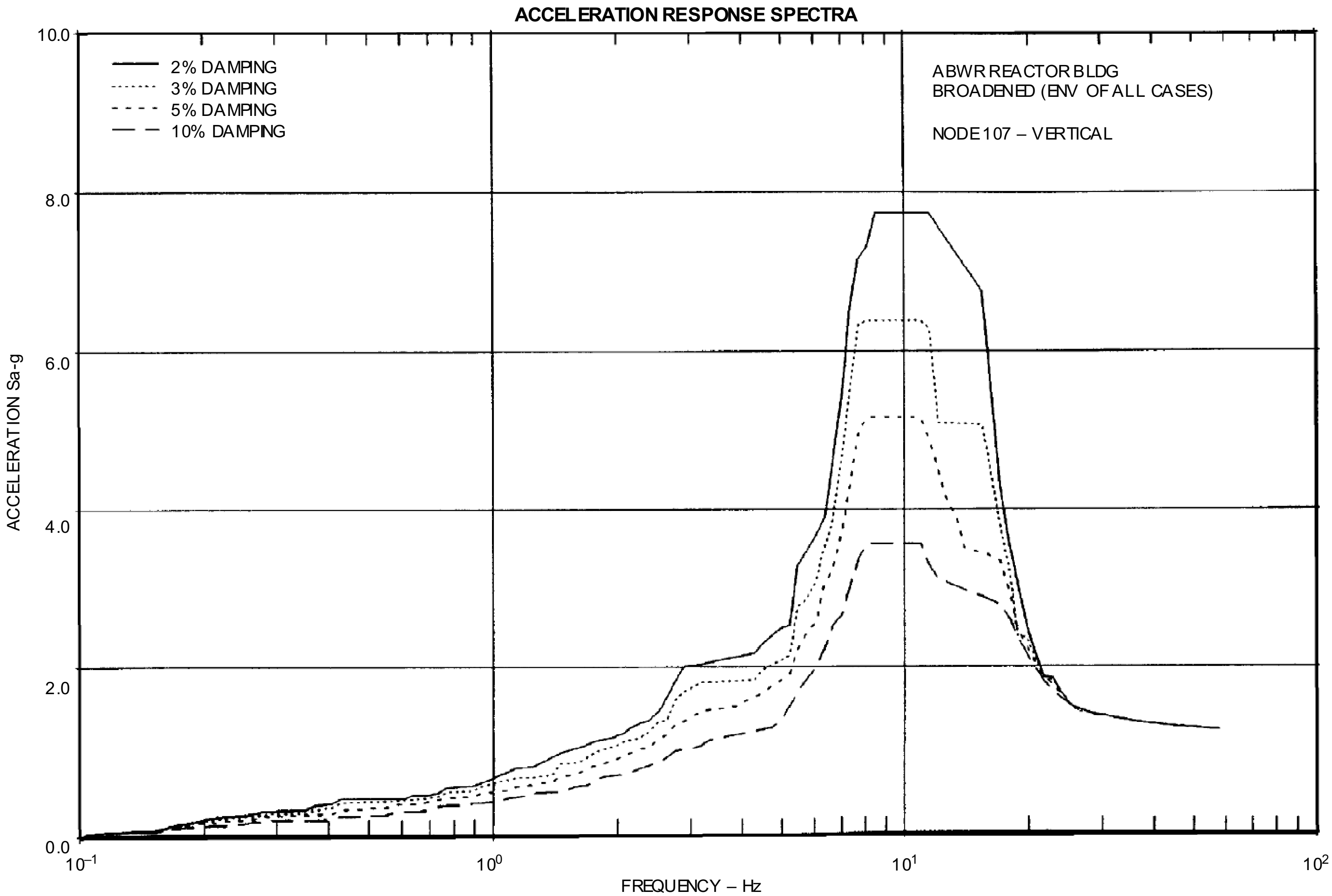

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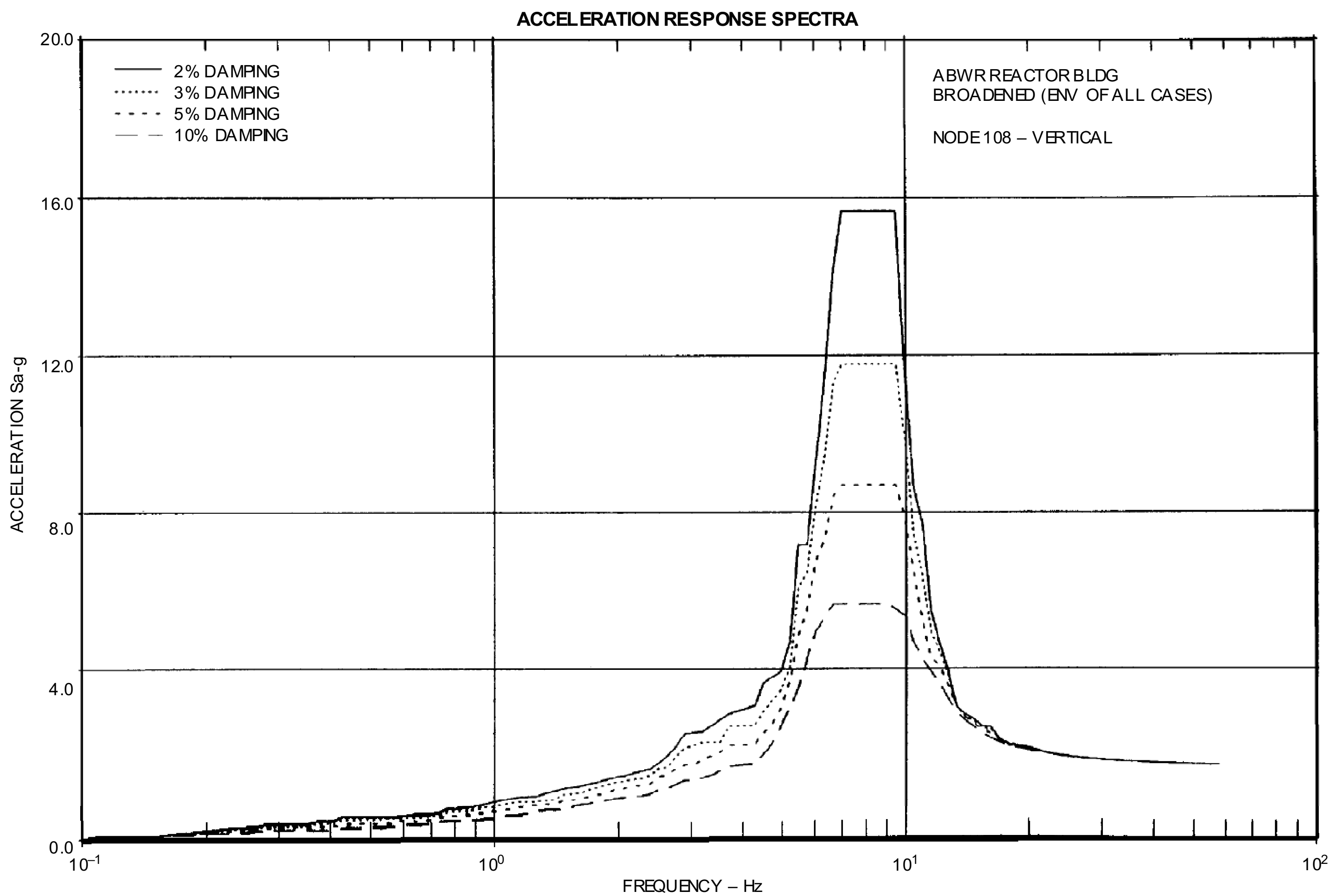

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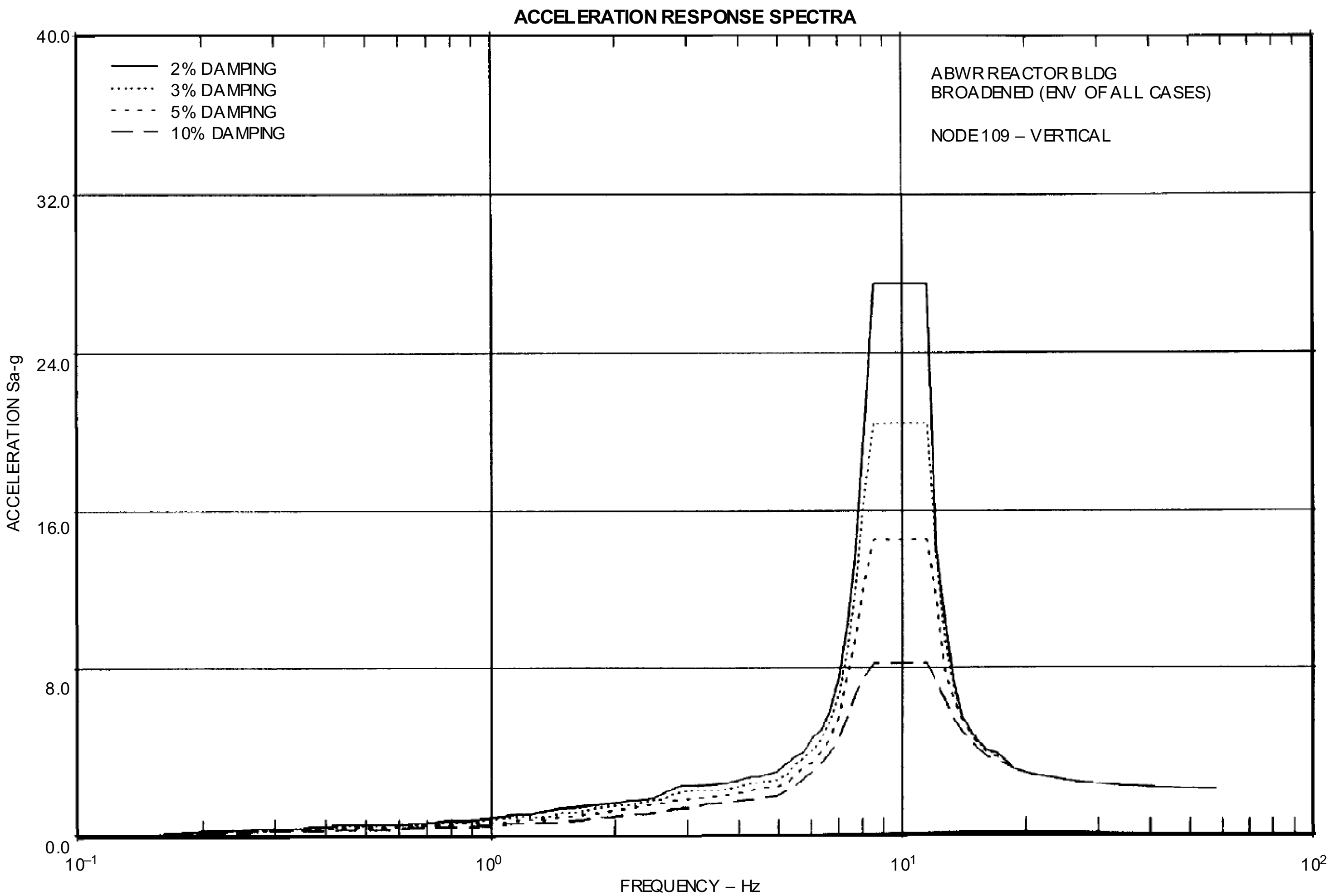

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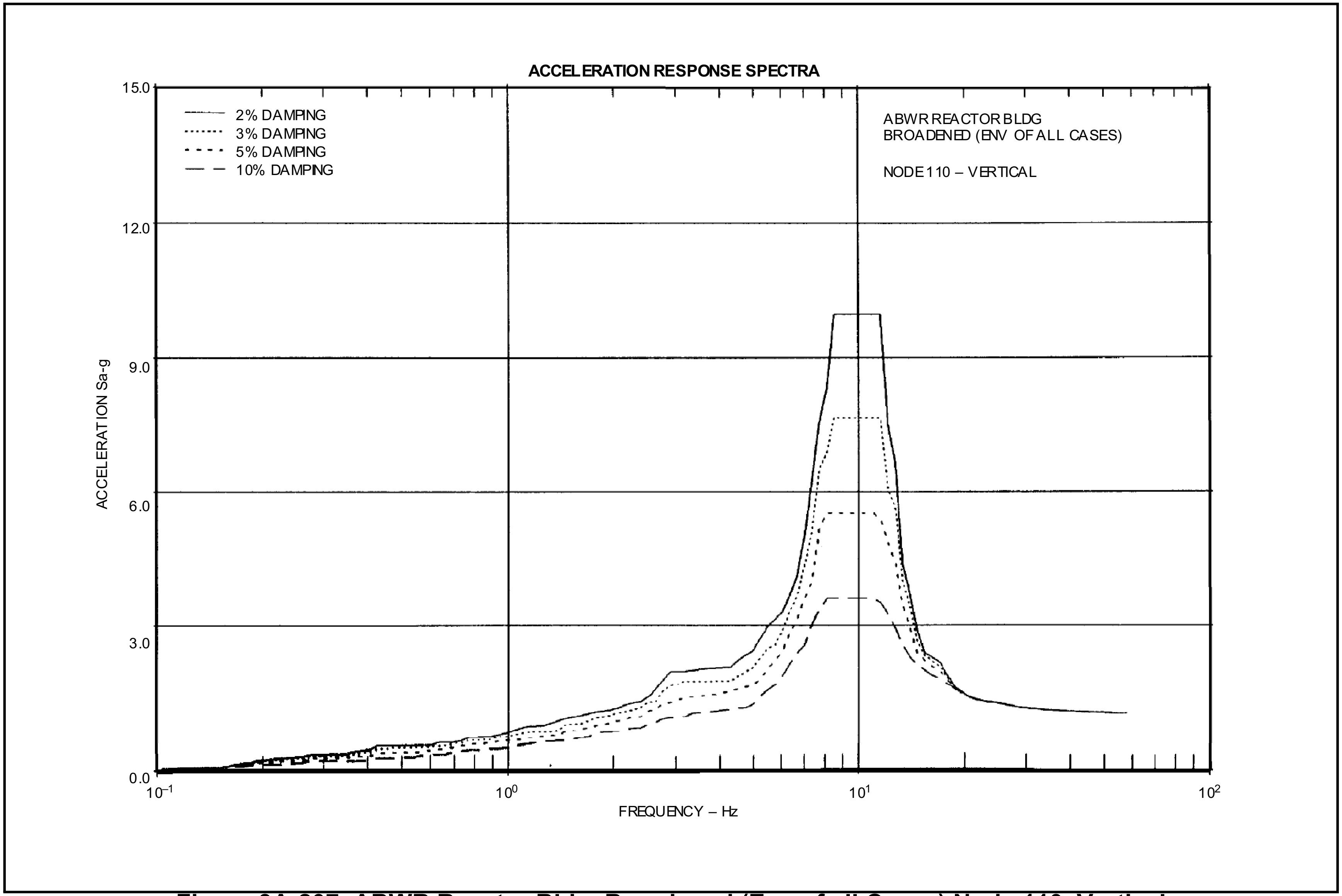

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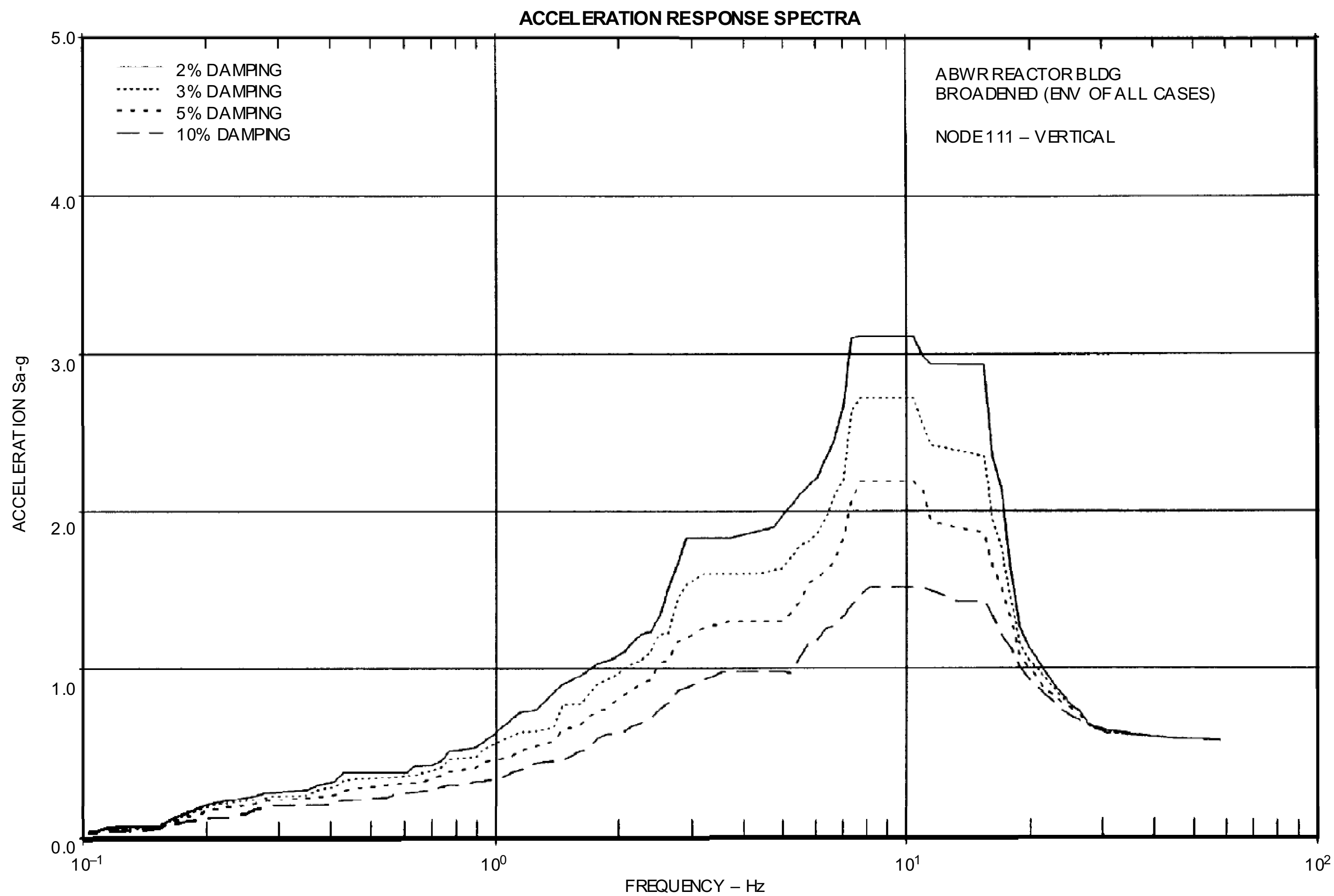

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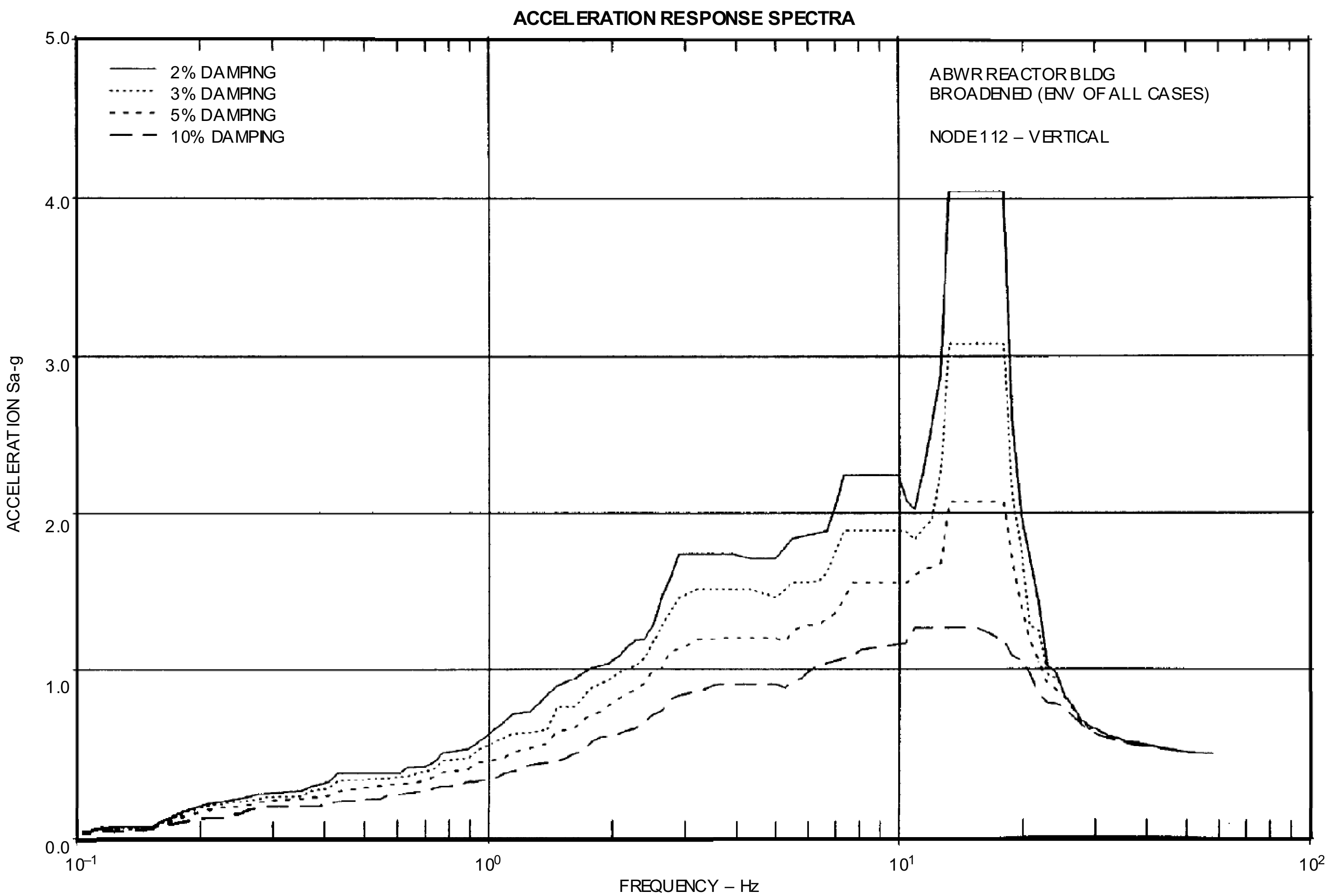

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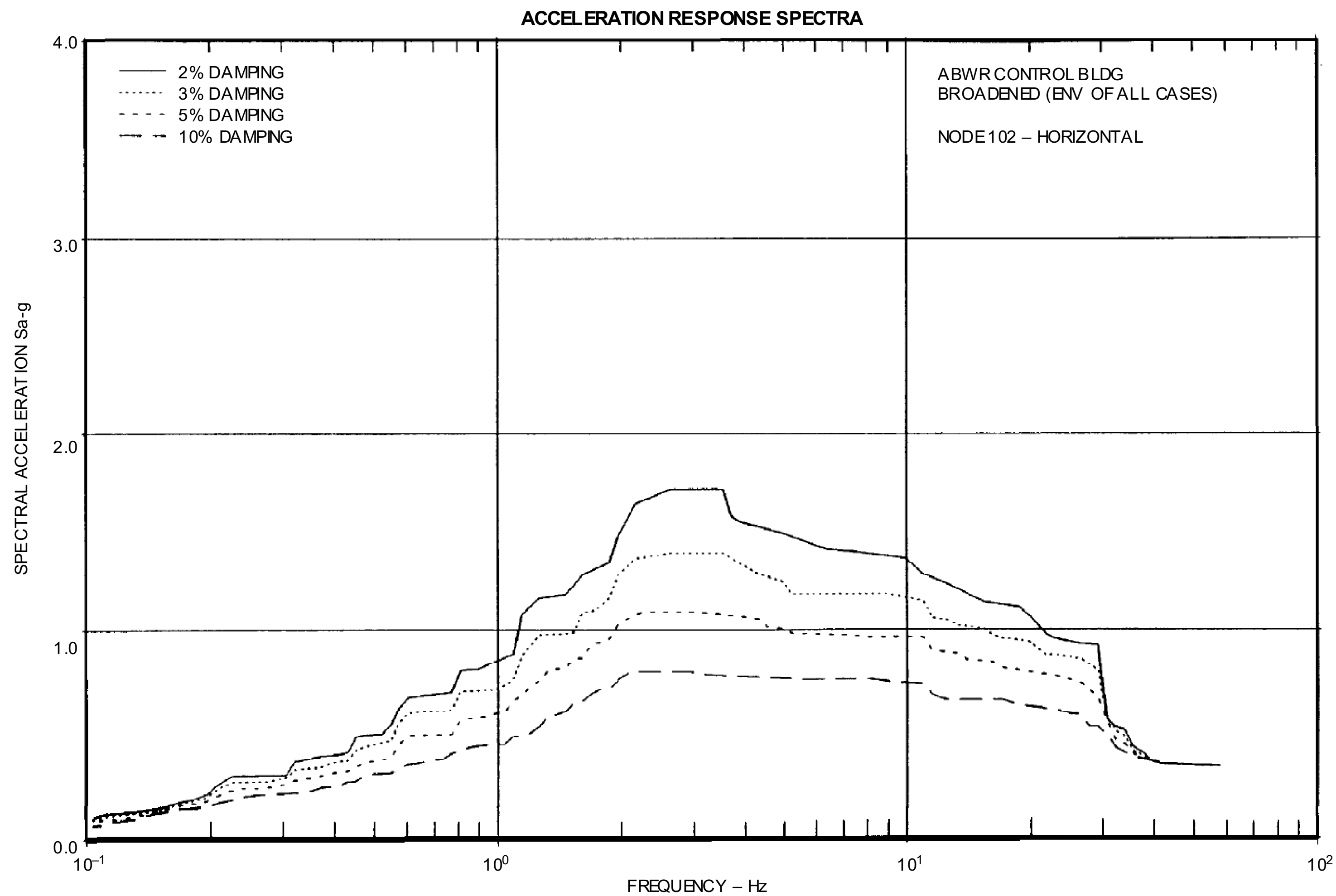

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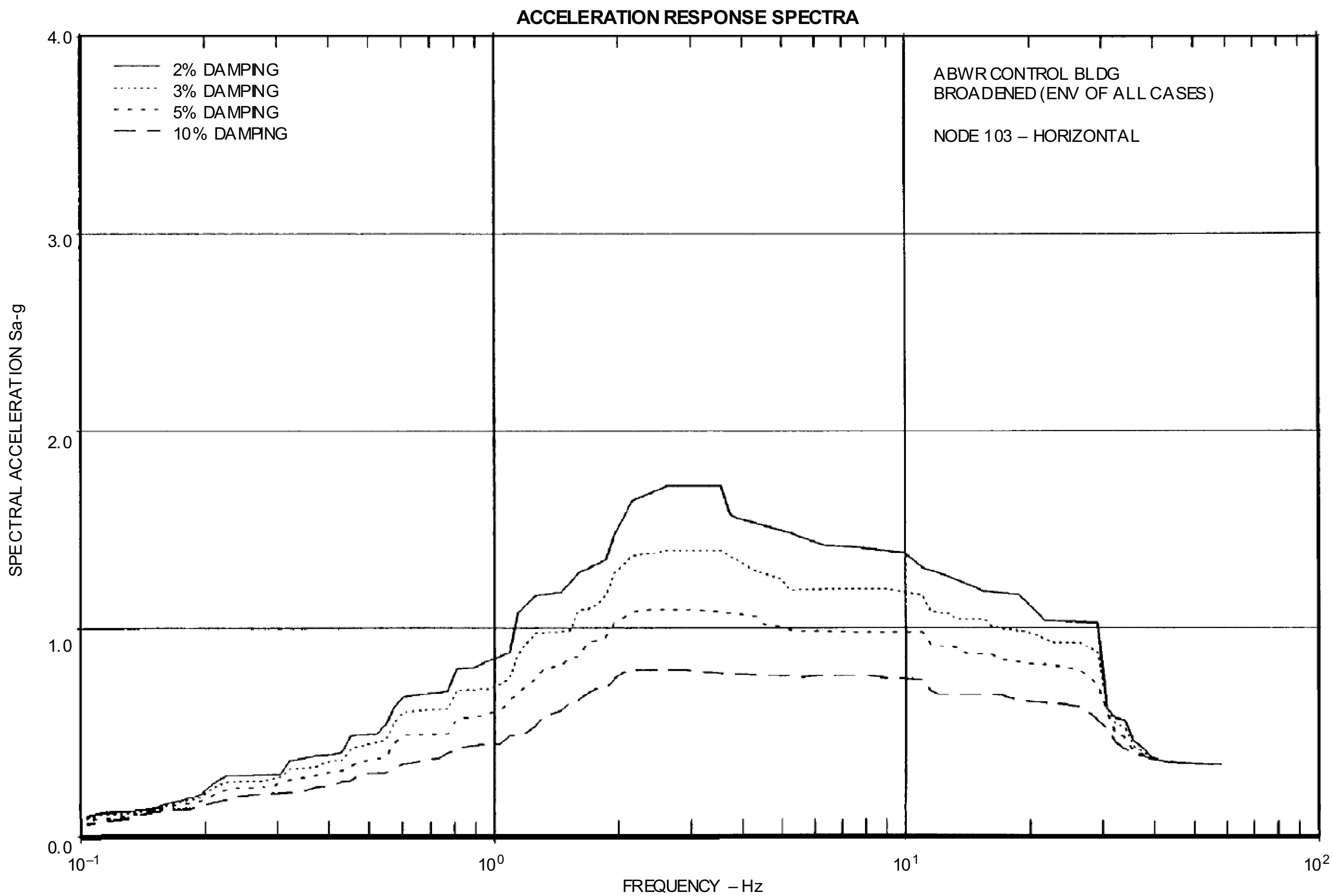

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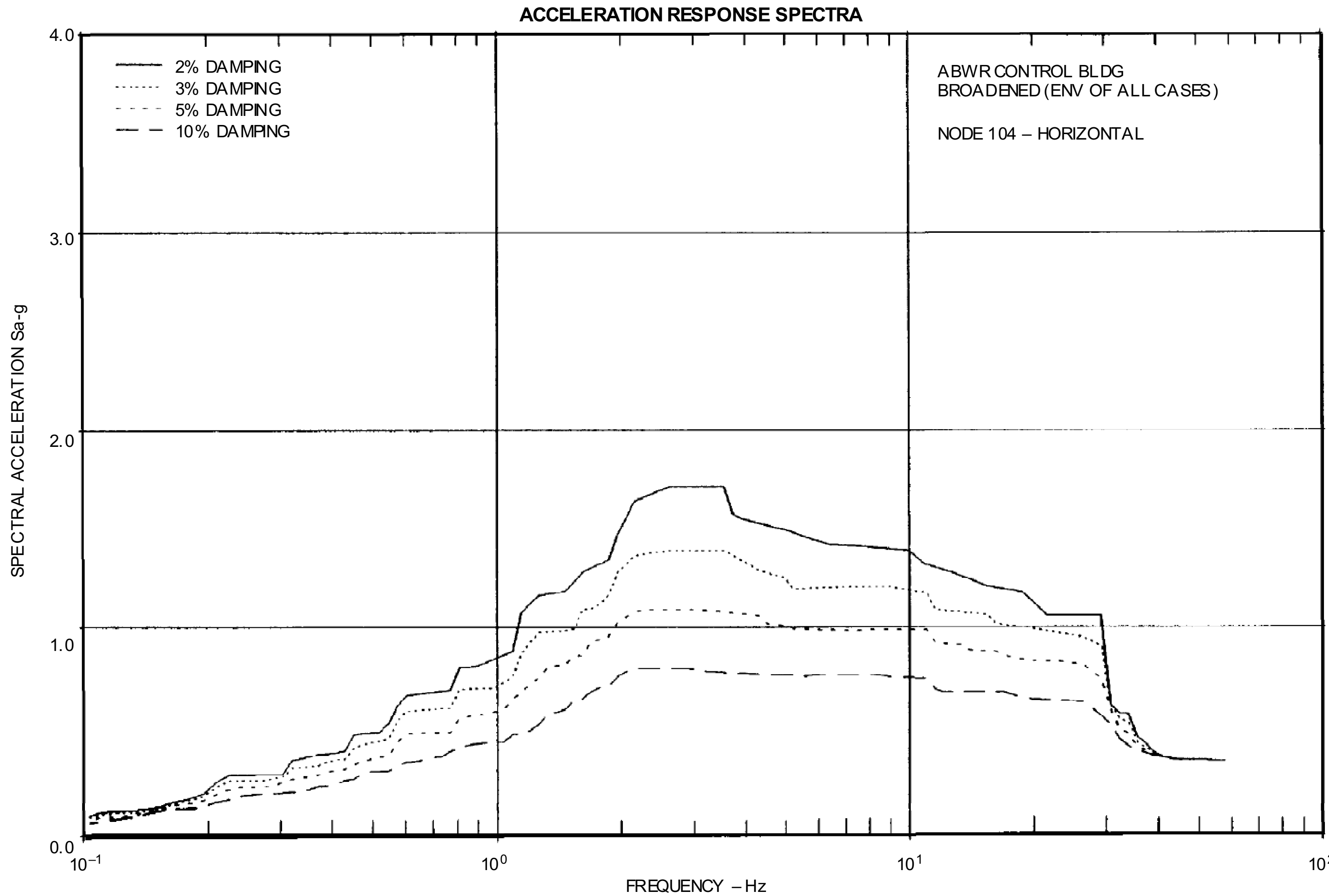

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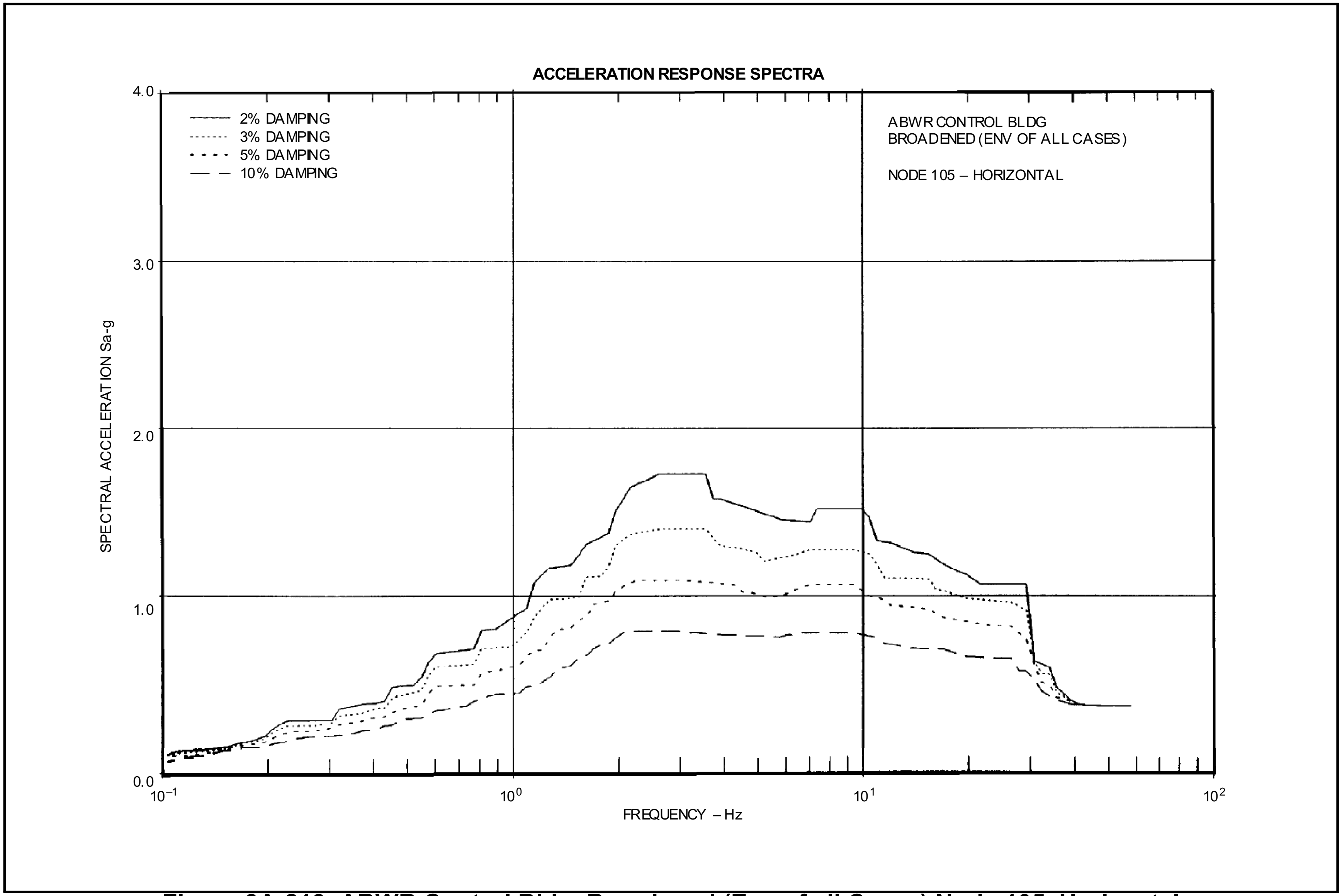

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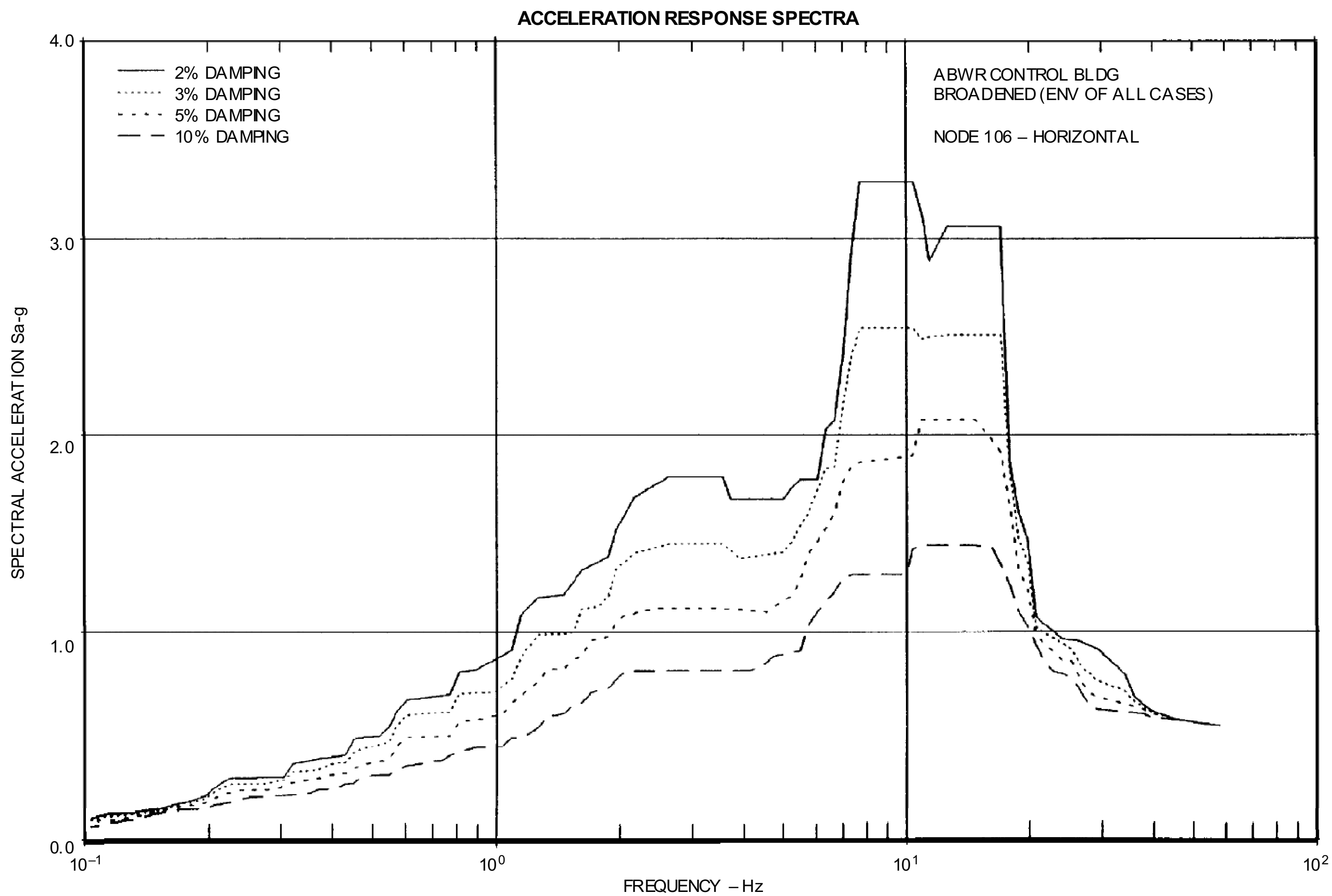

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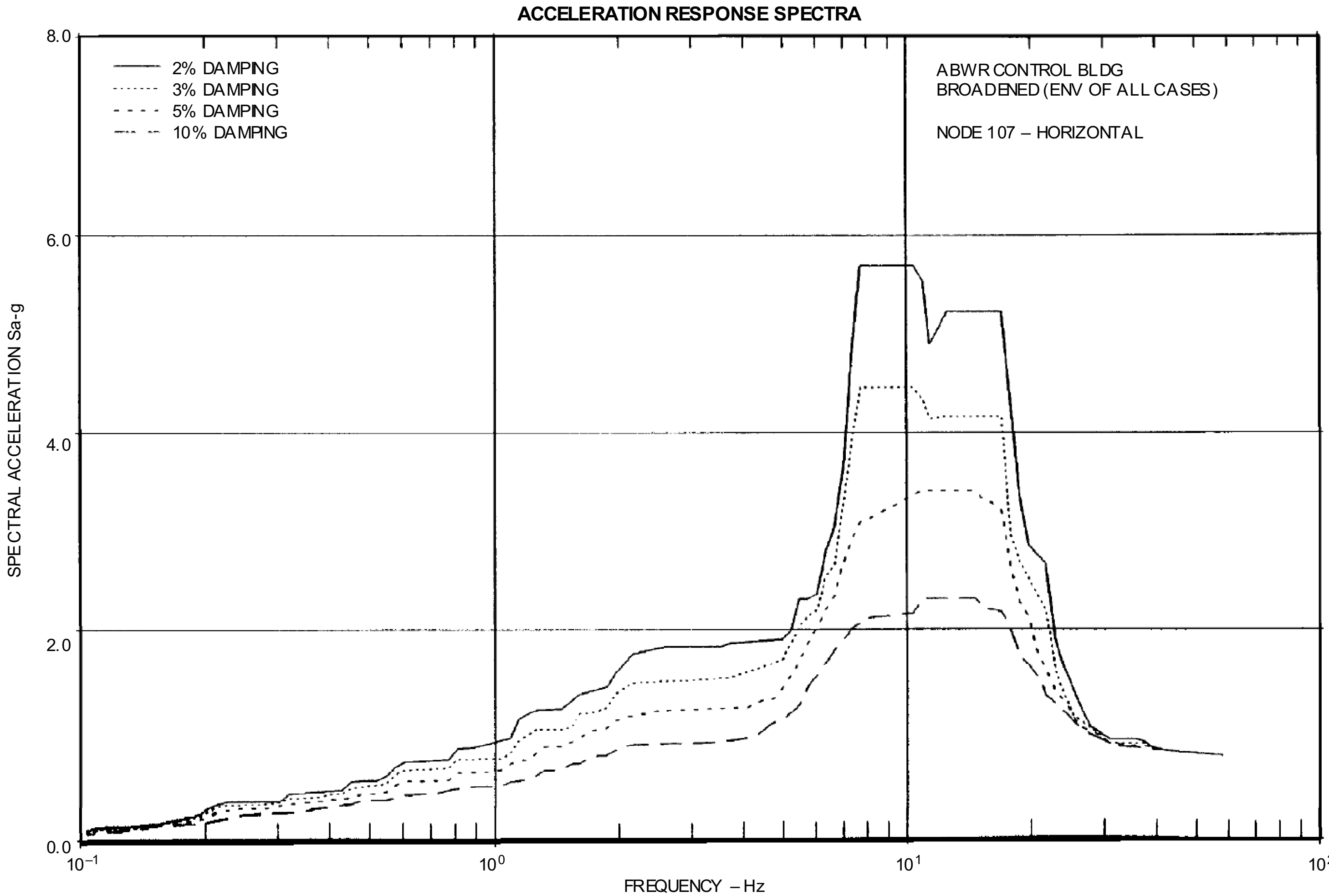

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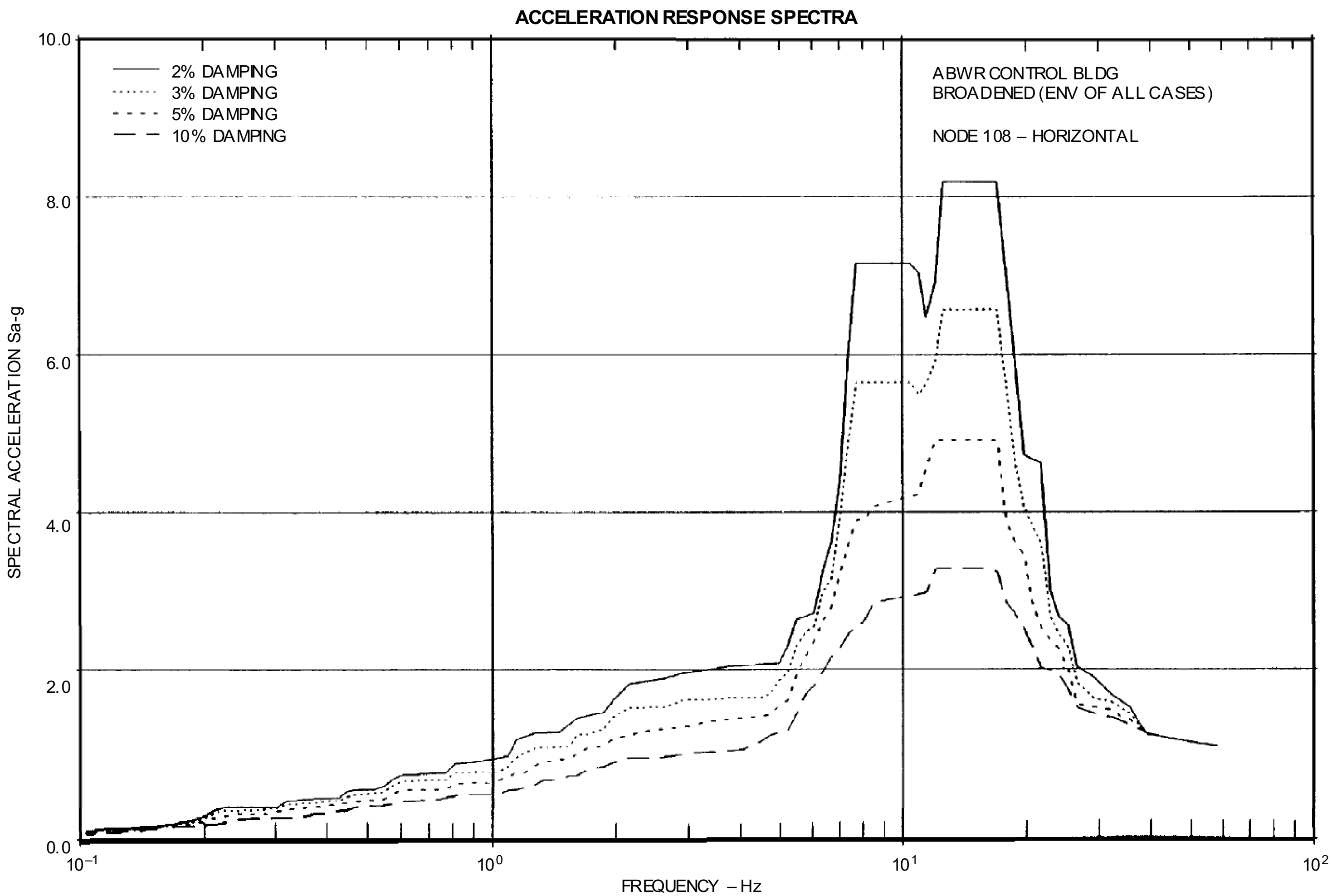

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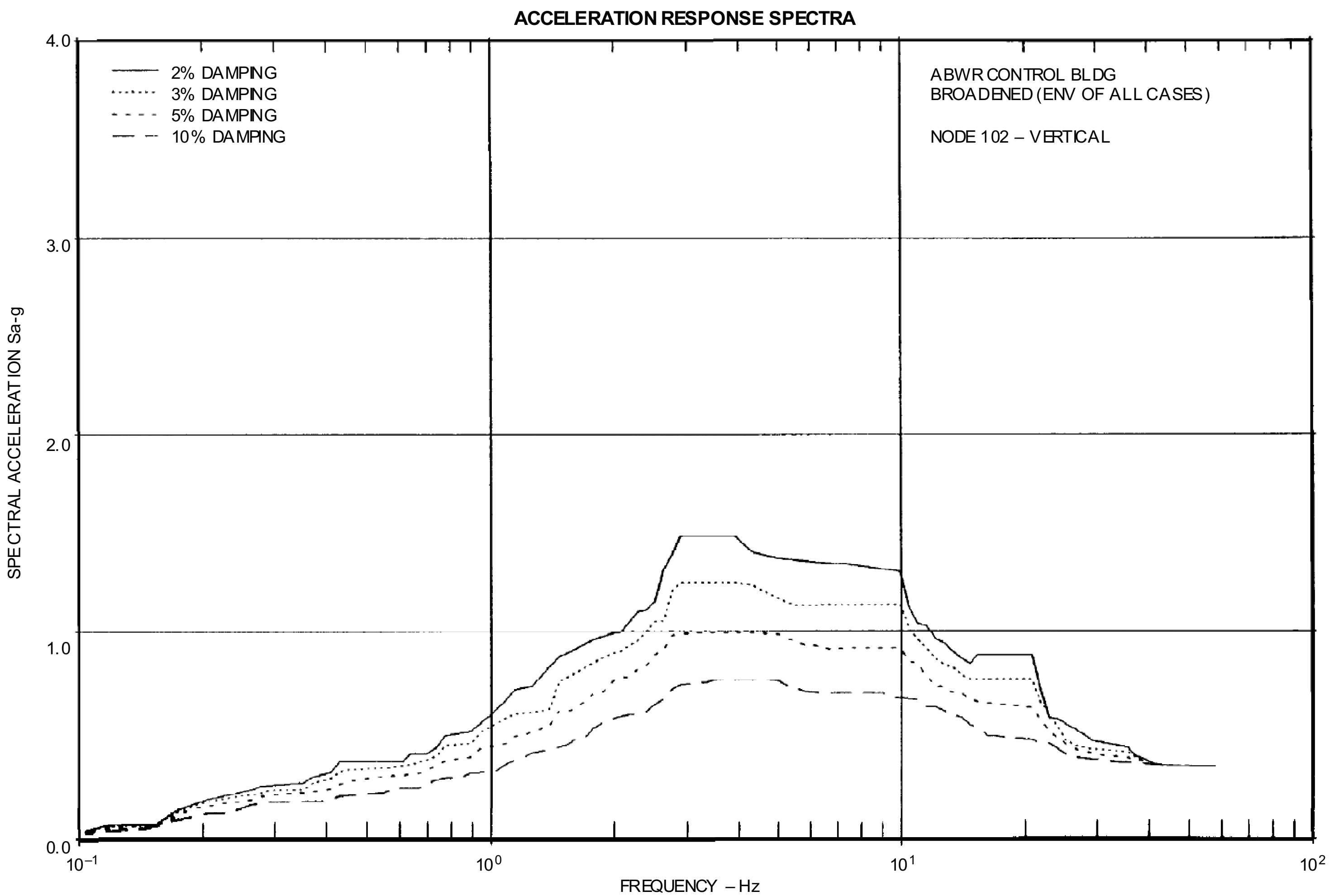

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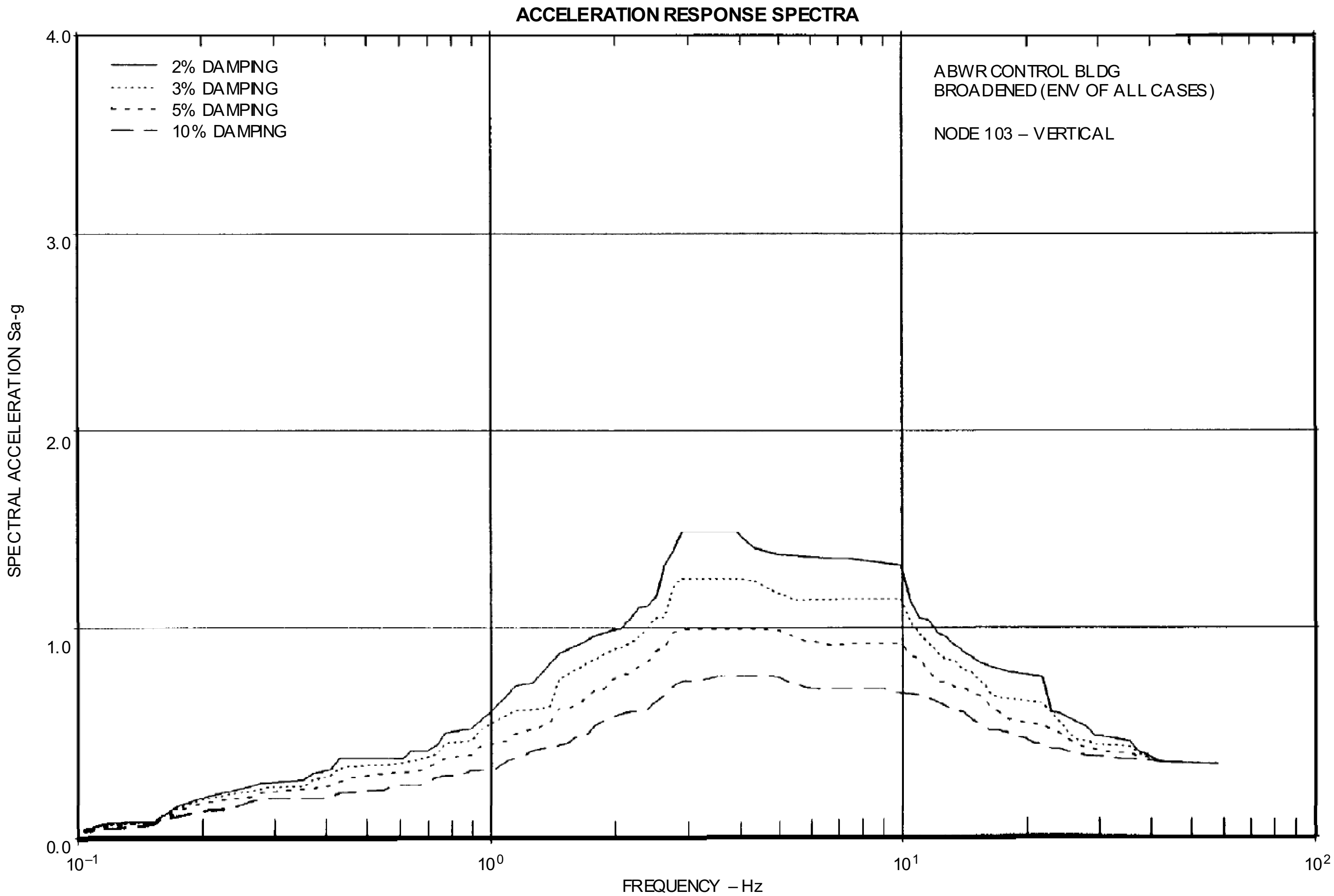

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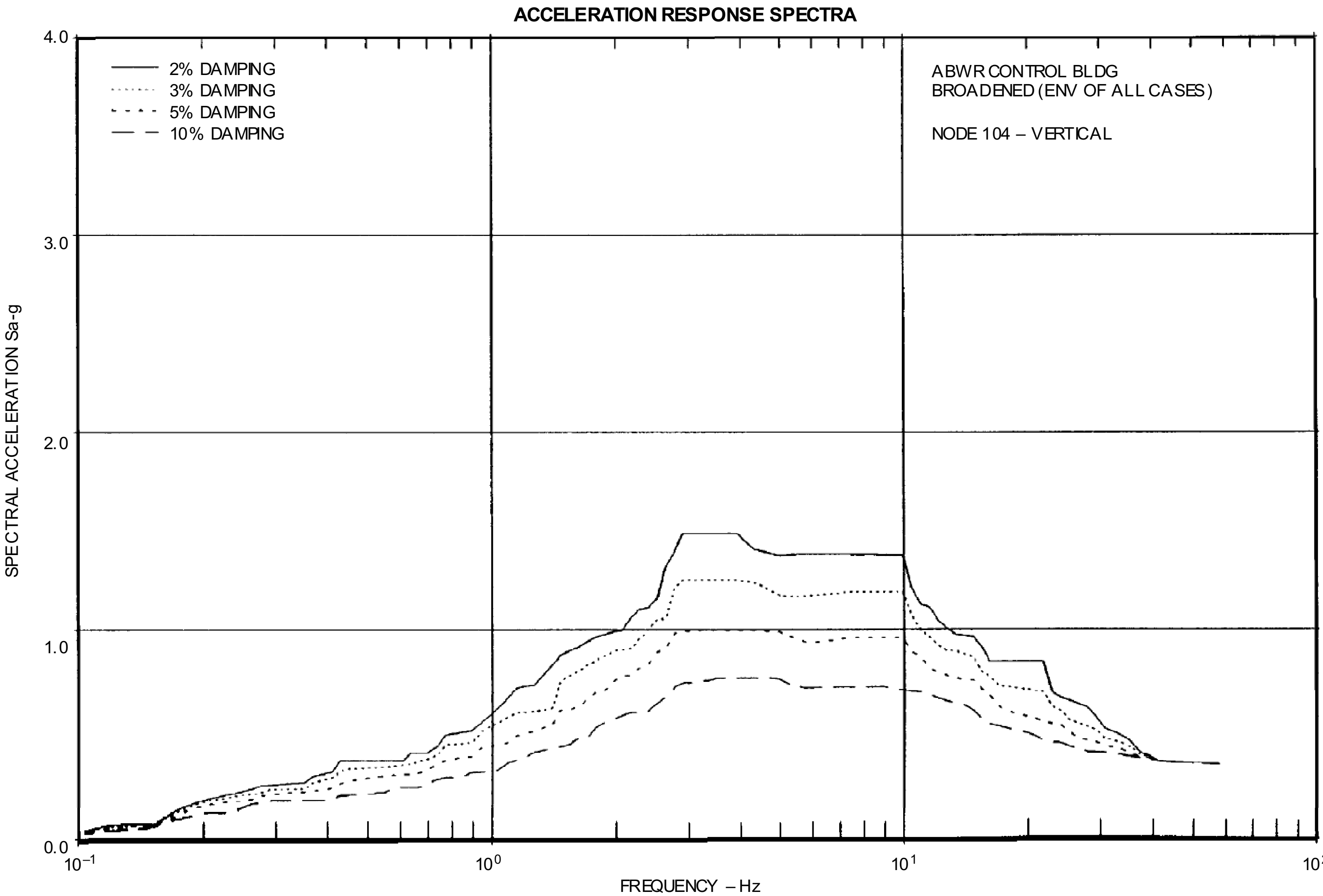

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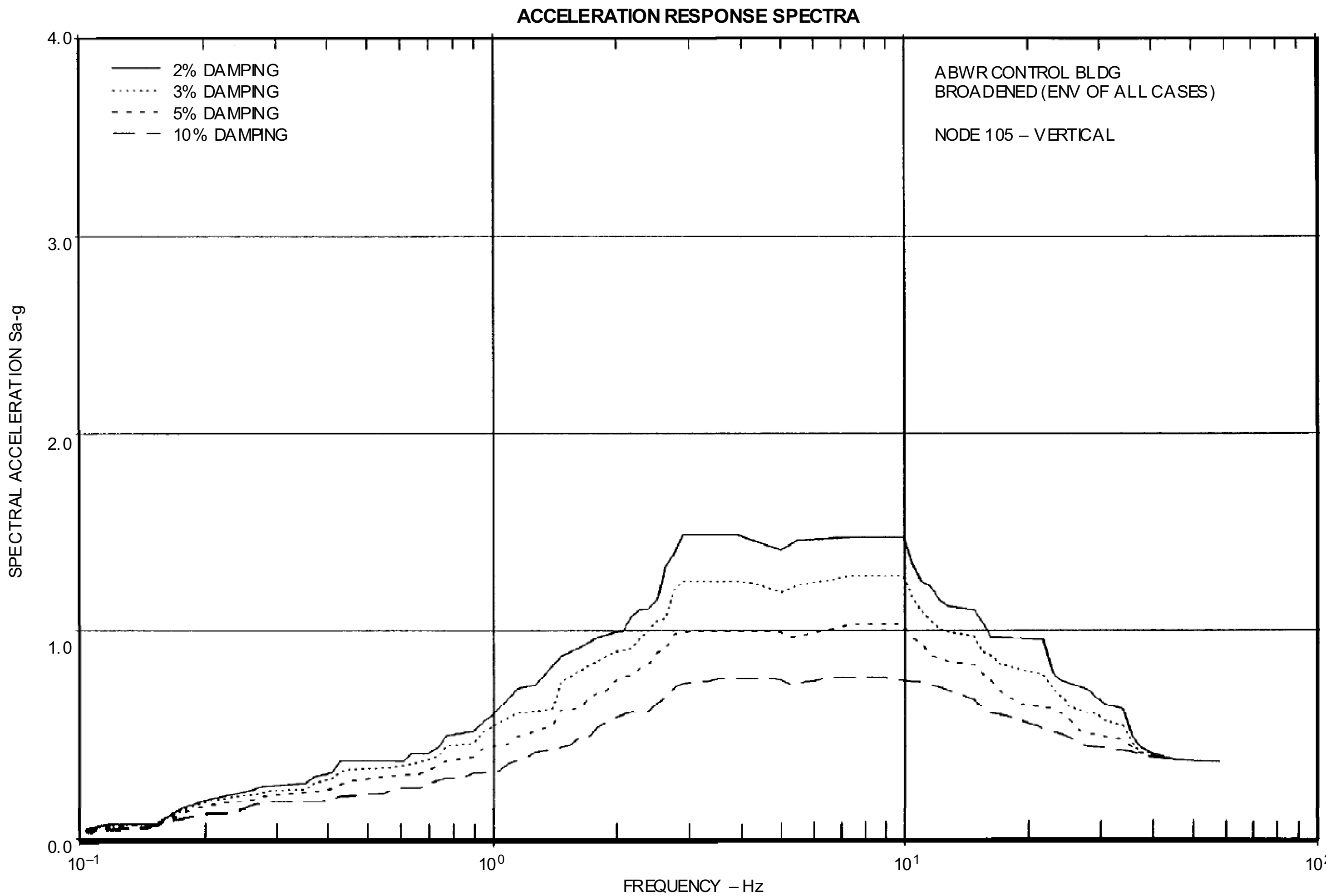

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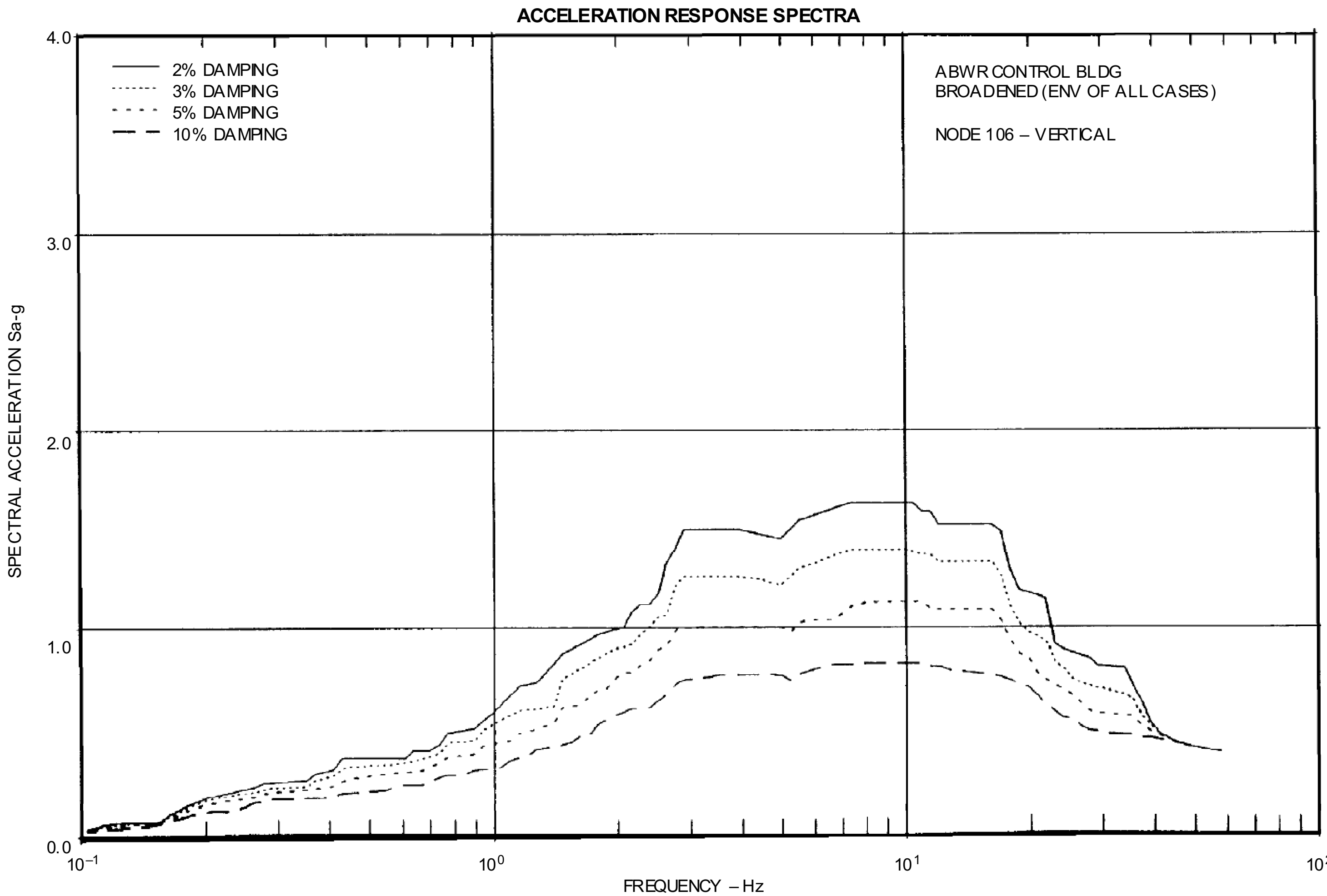

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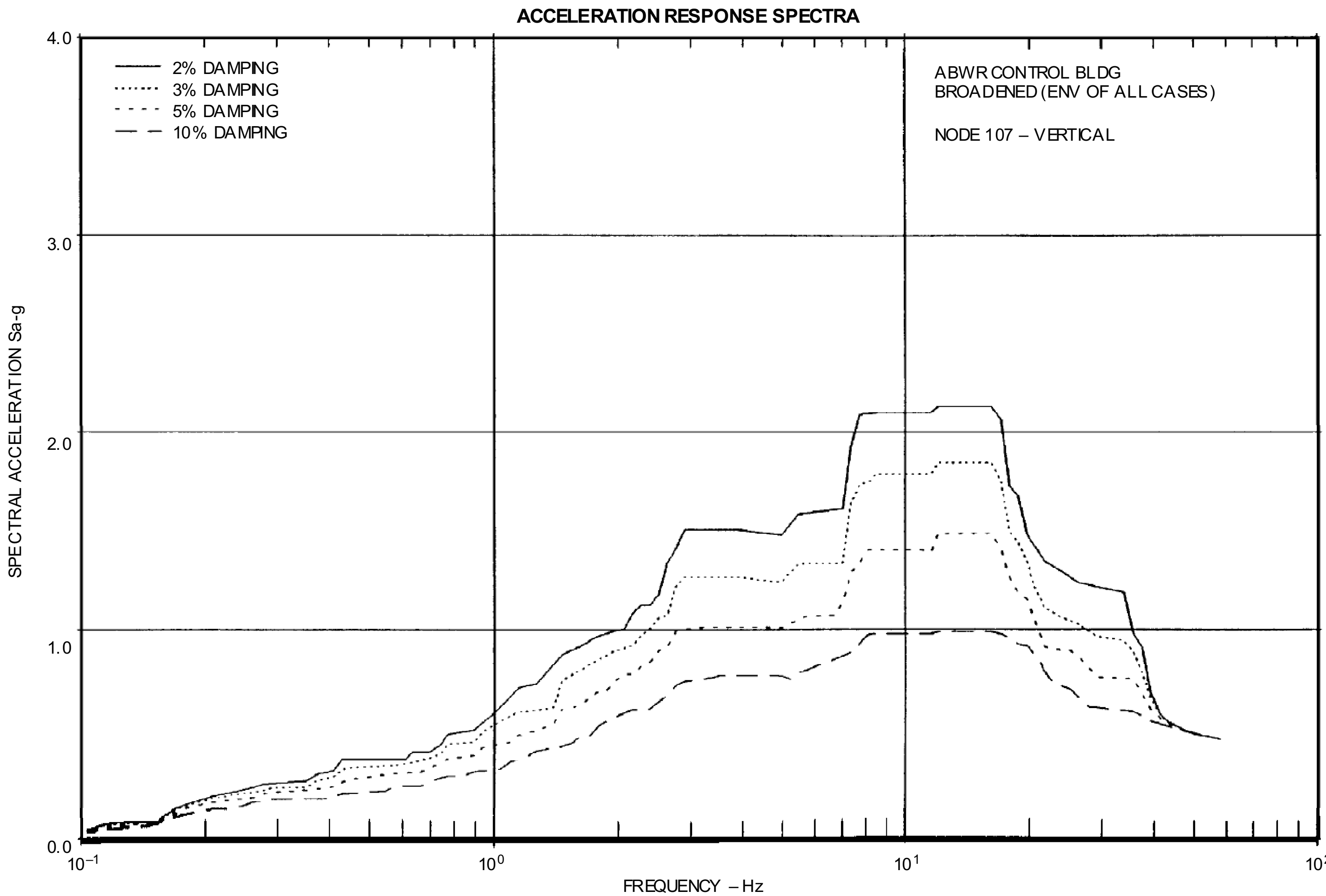

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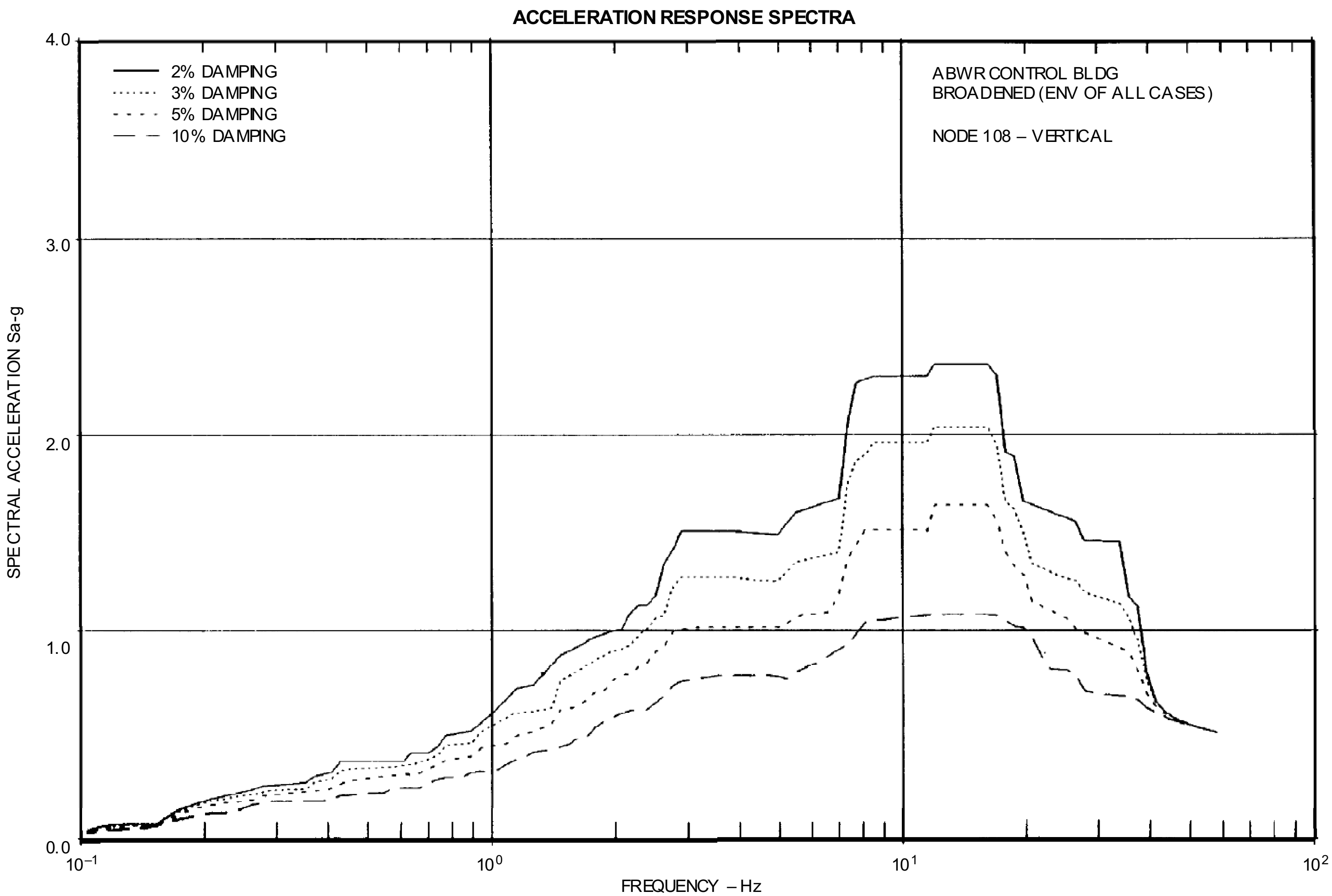

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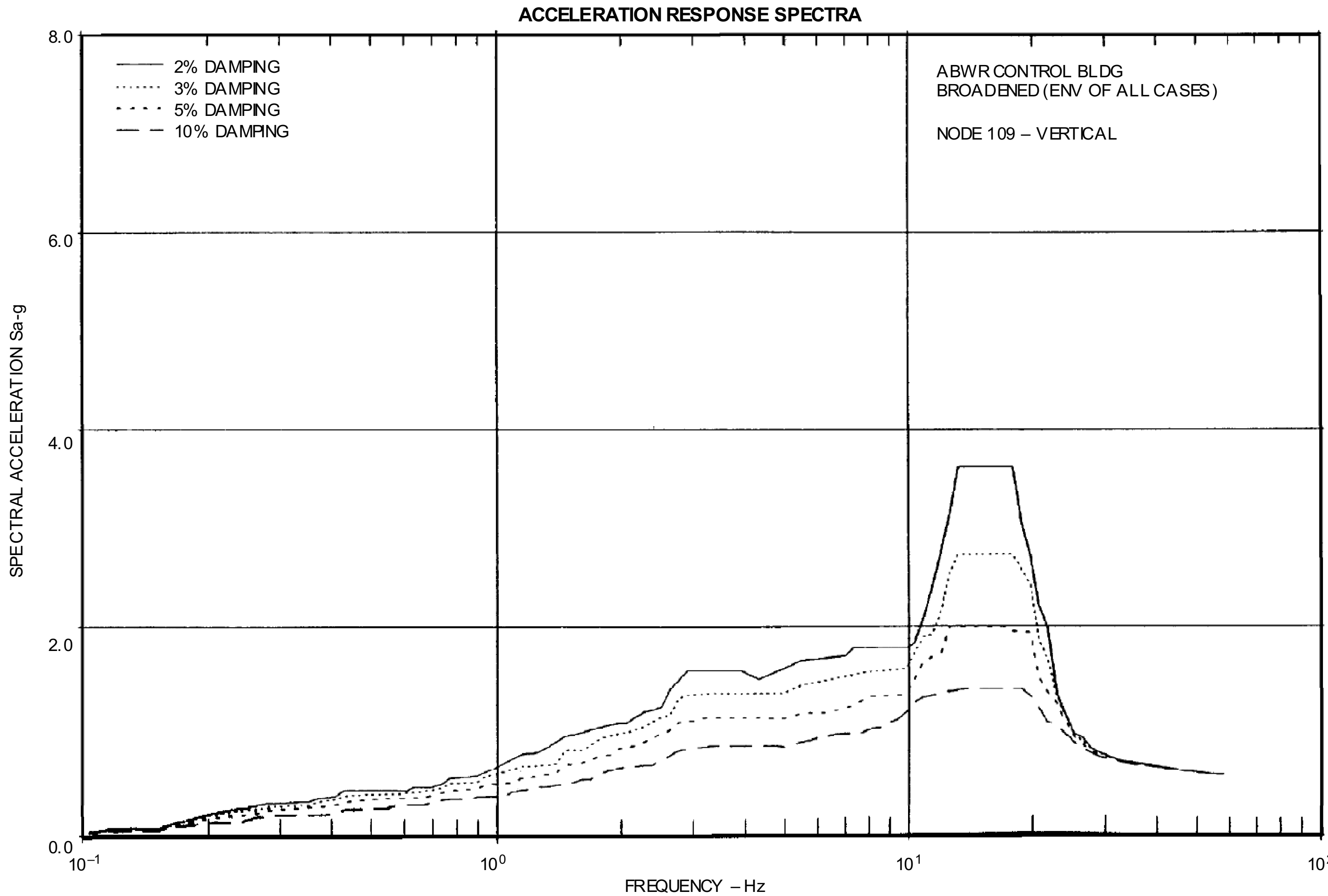

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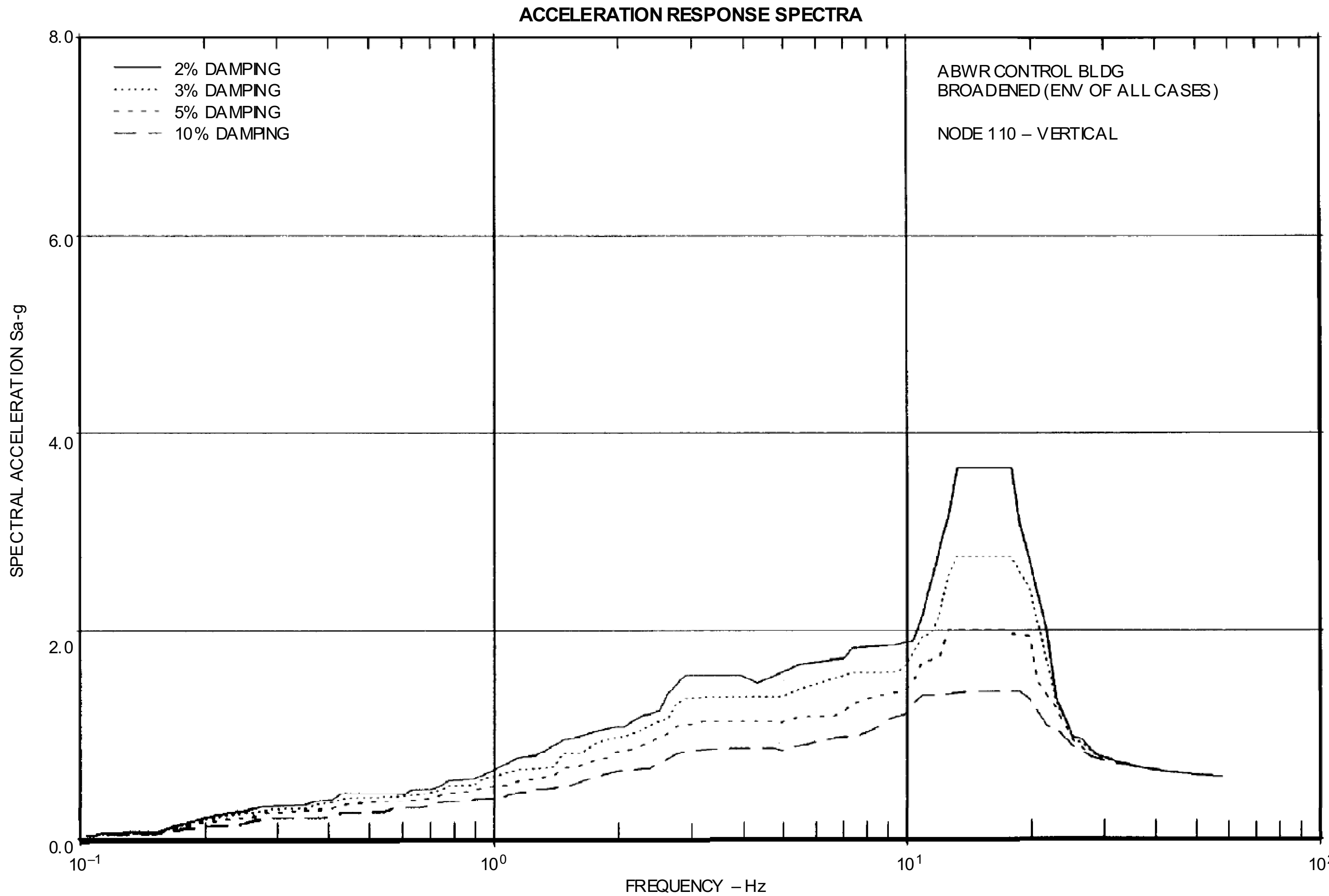

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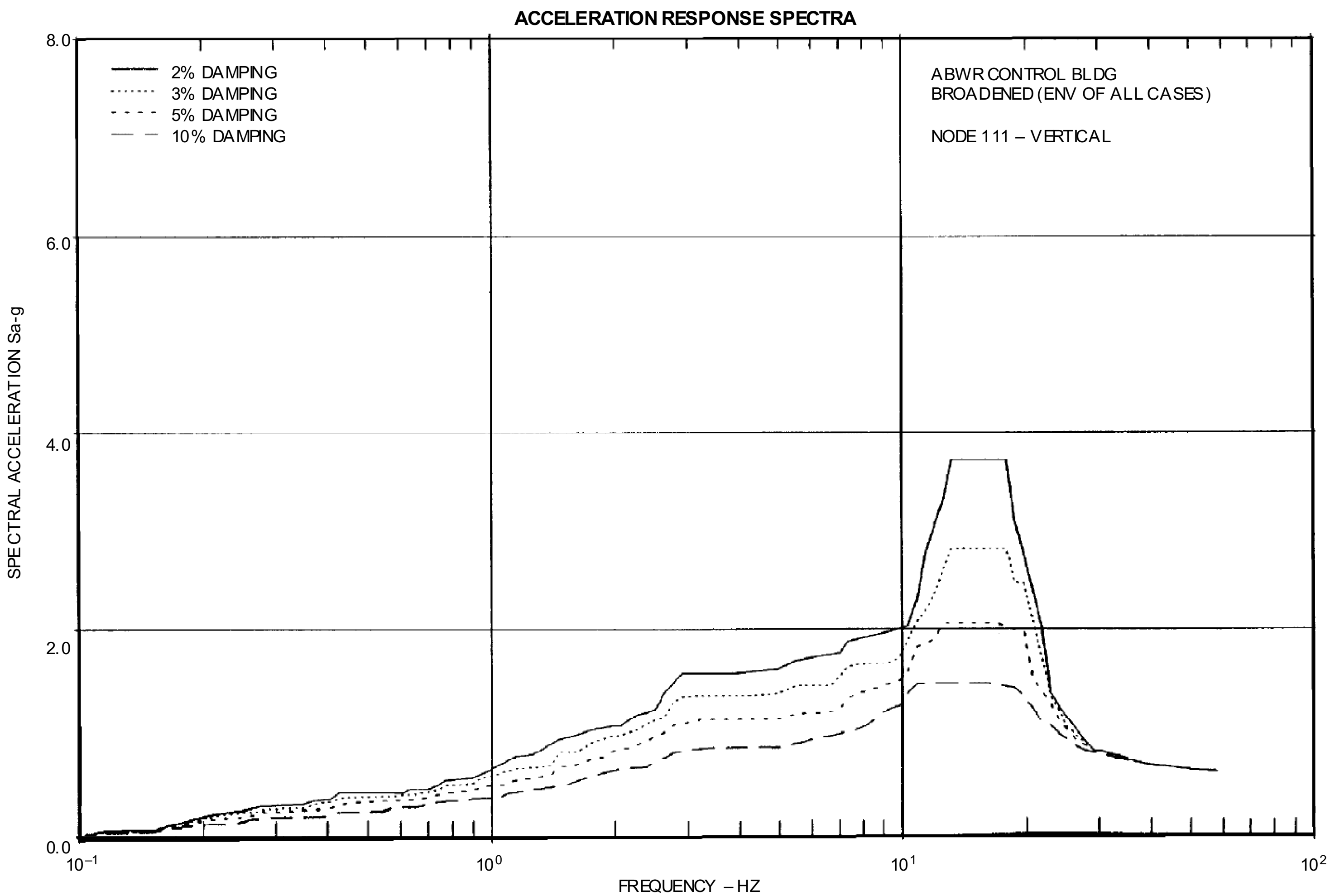

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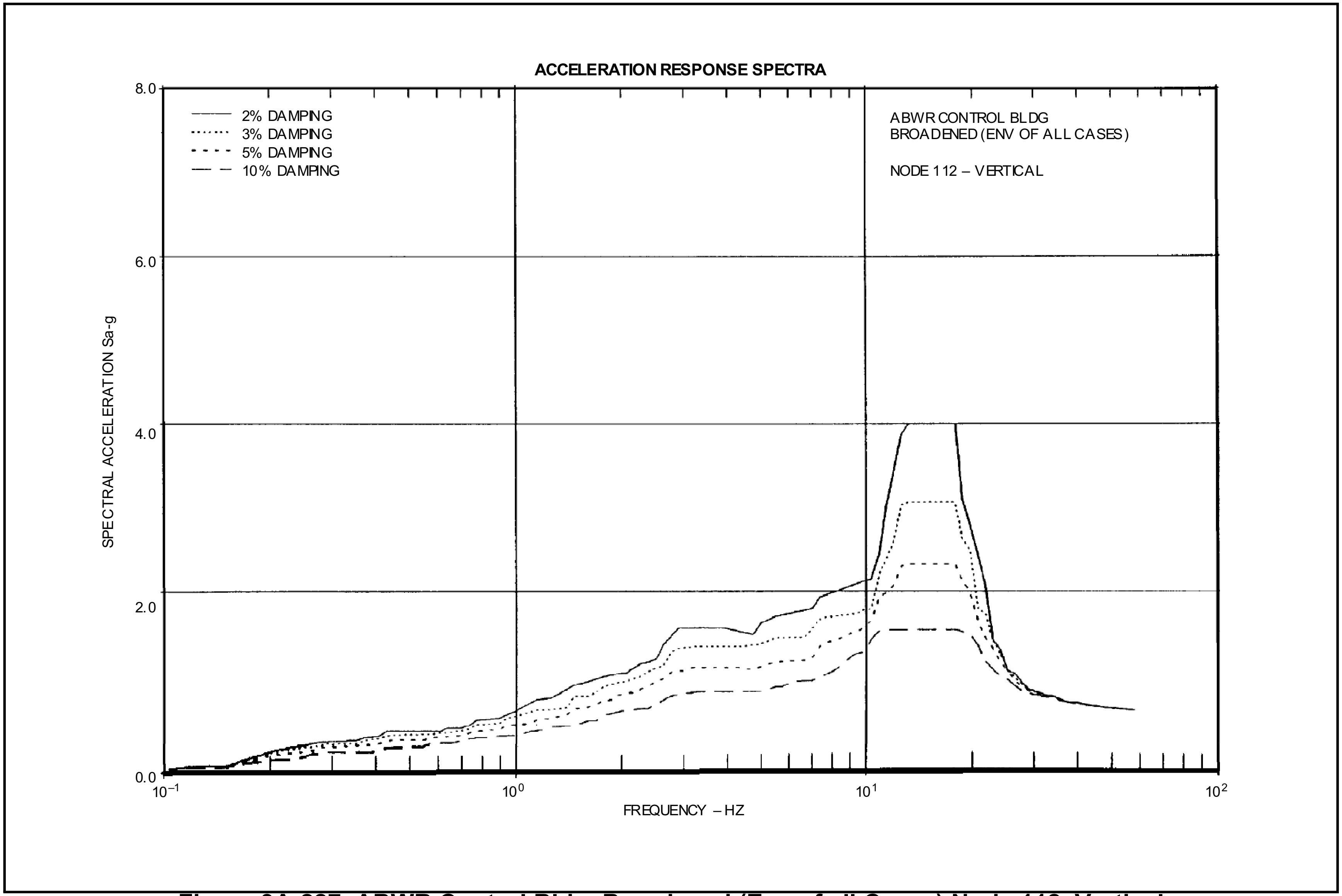

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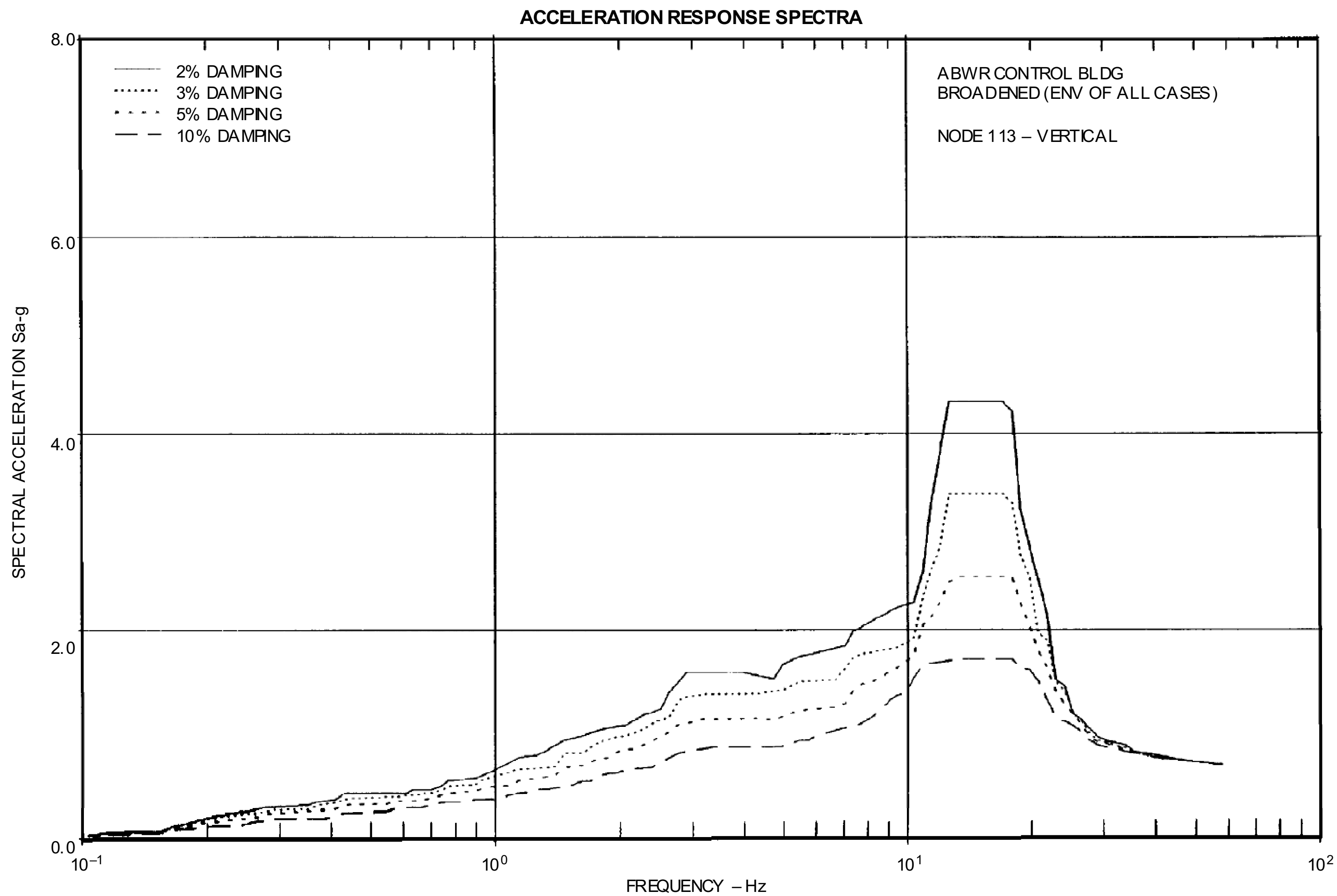


Figure 3A-228 ABWR Control Bldg. Broadened (Env of all Cases) Node 113–Vertical

3B Containment Hydrodynamic Loads

3B.1 Introduction

3B.1.1 Purpose

This appendix provides a description and load definition methodology for hydro-dynamic loading conditions inside the primary containment in an Advanced Boiling Water Reactor (ABWR) during safety/relief valve (SRV) actuation and a loss-of-coolant accident (LOCA) events. Overall, the load definition methodology used for the ABWR containment design is similar to that used for prior BWR containment designs. Wherever the ABWR unique design features warranted additional information for defining ABWR design loads, ABWR unique analyses and tests were conducted to provide an adequate database for defining the pertinent hydrodynamic loads.

3B.1.2 ABWR Containment Design Features

Subsection 6.2.1.1.2 provides a description of the design features of the ABWR containment system. The basic ABWR containment features are shown in Figure 3B-1. The ABWR design utilizes a horizontal vent system that is similar to the prior Mark III design, but includes some design features which are unique to the ABWR. These unique features include pressurization of the wetwell gas space, the presence of a lower drywell (L/D), the smaller number of horizontal vents (30 in ABWR vs. 120 in Mark III), extension of horizontal vents into the pool, vent submergence, and suppression pool width.

3B.2 Review of Phenomena

This section describes the sequence of events occurring during safety/relief valve (SRV) actuations and a postulated LOCA, and describes the potential load producing conditions. A spectrum of break sizes is considered for the LOCA event.

3B.2.1 Safety/Relief Valve Actuation

SRVs are utilized in a BWR pressure suppression system to provide pressure relief during certain reactor transients. Steam discharge through the SRVs is routed through discharge lines into the pressure suppression pool, where it is condensed. Each discharge pipe is fitted at the end with a device called a quencher to promote heat transfer during SRV actuation between the high temperature compressed air and steam mixture and the cooler water in the suppression pool. This enhances heat transfer results in a low amplitude oscillating pressure in the pool and eliminates concern over operation at high suppression pool temperature. For ABWR plants, the discharge device is an X-quencher which has been used for prior plants.

SRV actuation may occur (1) in response to a reactor system transient pressure increase (pressure actuation), (2) as the result of planned operator action (manual operation), or (3) as the result of a failure or error affecting one SRV (inadvertant opening). Inadvertant operation involves a single SRV, as does manual operation. Pressure actuation involves from one to all

SRVs opening sequentially during the vessel pressure rise. The opening sequence depends on the SRV pressure setpoints.

The discharge piping of an SRV contains ambient air and a column of water whose volume determined by (1) the submergence and inclination of the SRV discharge line in the suppression pool, and (2) the difference in the drywell and wetwell airspace pressures. Upon SRV actuation, pressure builds up inside the piping as SRV blowdown steam compresses the air and forces the water column through the quencher into the suppression pool. The expulsion of water from the SRV line into the suppression pool is called the “water-clearing phase” of the SRV discharge. The loads associated with the water-clearing phase are: (1) transient SRV pipe pressure and thermal loads; (2) pipe reaction forces from transient pressure waves and fluid motion in the pipe; (3) drag loads on structures located in the path of the submerged water jet, and (4) pool boundary loads.

Air follows the water column into the pool in the form of high-pressure bubbles. Once in the pool, the bubbles expand because the ambient pool pressure is lower than the bubble pressure. The subsequent interaction of the air bubbles and the pool water manifests itself as an oscillatory pressure field which persists with decaying amplitude until the air rises to the pool surface. The frequency of the pressure oscillation is influenced by (1) the initial mass of air in the SRV line, (2) the submergence of the SRV discharge line in the pool, (3) the suppression pool temperature, (4) the pool geometry, and (5) the wetwell airspace pressure. Loads associated with the air bubble dynamic phenomena are transient drag loads on submerged structures caused by the velocity field (standard drag) and the acceleration field (inertial drag), and oscillating pressure loads on the pool boundary.

Following the air-clearing phase, steady steam discharge flow is established and continues until the SRV is closed or the reactor is depressurized. The steam enters the pool from the quencher as submerged jets and is completely condensed outside the quencher. Loads associated with the steady steam flow phase of the SRV discharge include (1) pipe reaction forces caused by steady steam flow through pipe bends, (2) thrust forces on the quencher, (3) thermal loads on structures contacted by the steam, and (4) pool boundary loads caused by the oscillation of the condensing steam jets at the quencher.

Following SRV closure, the steam in the discharge line will condense and the resulting vacuum will draw water back into the line. Vacuum breakers on each SRV discharge line are provided to admit drywell air to the discharge line to limit the water reflood rise within the discharge line and allow the water to return to near normal level. This is to prevent large loads during subsequent actuations due to large water leg. Conditions at this time are effectively the same as those prior to actuation, and reopening of the SRV will result in a repeat of the previously described sequence of events. Subsequent actuation of the SRV can influence the magnitude of loads associated with the SRV because the SRV discharge line heated by the previous actuation.

For multiple SRV discharge conditions, the basic discharge line clearing phenomena are the same as for single SRV discharges. The loads on structures in the suppression pool, including the pool boundary, will be the result of combined effects of the SRV discharge at a number of locations in the suppression pool.

3B.2.2 Loss-of-Coolant Accidents

A spectrum of postulated loss-of-coolant-accidents (LOCAs) is considered to assess the design adequacy of the ABWR containment structure. Each accident condition is described in this section.

3B.2.2.1 Large Break Accident

Two types of large break accidents for the ABWR control the design: the feedwater line break and the main steamline break. In these transients, the upper drywell pressure increases as a result of the mass and energy release from the break, and a steam-air mixture is forced through the drywell connecting vents into the lower drywell. The pressure differential before the pressure is equalized causes a loading on the vessel skirt. Additionally, the flow through the drywell connecting vents causes a drag loading on all structures in those vents.

Concurrent with the air-steam mixture being forced into the lower drywell, the water initially contained in the vent system is accelerated out of the vents. During this vent-clearing process, the water exiting the vents will form submerged jets in the suppression pool which can produce loads on structures near the vent exits and on the suppression pool basemat.

Immediately following the water clearing, bubbles containing air and steam form at the vent exits. As the flow of air and steam from the drywell becomes established in the vent system, the initial bubbles at the vent exits expand with the bubble pressure nearly equal to the drywell pressure. The steam fraction of the flow into the pool will be condensed, but continued injection of drywell air and the resultant expansion of the air bubbles will produce a rapid rise of the suppression pool surface (termed as pool swell). The expanding bubble will cause loads on submerged structures and the suppression pool boundaries.

During the early stages of pool swell, a slug of water is accelerated upward by the expanding air bubbles. Structures and equipment close to the pool surface will experience impact loads as the rising pool surface hits the bottom surface of the structures. Along with these inertial loads, dissipative drag loads will develop as water flows past structures and equipment at elevations above the vent exit and below the maximum pool swell height. These rising and expanding bubbles eventually break through the water ligament and communicates with the wetwell airspace. Breakthrough occurs when the instabilities formed in the rising ligament causes the surface to become unstable and shatter. Froth will continue upward until decelerated to zero velocity by gravity. Because of increasing wetwell airspace pressure, froth would not reach the diaphragm floor, so a pool swell uplift load on the diaphragm floor will not occur.

Following the pool swell transient, a period of high steam flow rate through the vent system will commence. This is followed by a decreasing flow rate as the reactor vessel blowdown progresses. Prior test data have indicated that the steam will be entirely condensed in the vent exit region.

The steam condensation process is influenced by the vent steam mass flow rate, the subcooling at the vent exit, and the vent flow air content fraction. At medium vent flow rates, the water-to-steam condensation interface will oscillate, causing pressure oscillations in the pool. This phenomenon, referred to as “condensation oscillation”, produces oscillatory and steady loadings potentially severe enough to establish some of the containment design parameters. As the vessel blowdown continues, the vent flow rate will decrease and vent flow air content fraction will become negligibly small. At lower vent flow rates (below a threshold level), the steam bubbles at the vent exit alternately grow, and then nearly instantaneously collapse in a condensation process referred to as “chugging”. This chugging process produces transient dynamic loading on the vents and suppression pool boundary which must be considered in design evaluation of the containment system.

Shortly after a large break accident, the Emergency Core Cooling System (ECCS) pumps will automatically start pumping water from condensate storage pool or suppression pool into the reactor pressure vessel (RPV) to flood the reactor core. Eventually water will cascade into the upper drywell from the pipe break. The time at which this occurs depends on the break size, type, and location. Because the drywell would be full of steam at the time of the vessel flooding, the sudden introduction of water into the drywell will cause condensation of the steam in the drywell and thus depressurization of the drywell. When the drywell pressure falls below the wetwell pressure, the wetwell-to-drywell vacuum breaker (WDVB) system will open vacuum breakers admitting air (noncondensables) from the wetwell airspace to the drywell. This depressurization of the drywell could cause upward loads on the diaphragm floor, whose magnitude would depend on the WDVB system characteristics, and the drywell and wetwell pressure histories. Eventually, a sufficient amount of air will return to equalize the drywell and wetwell pressures.

Following the vessel flooding and drywell-to-wetwell pressure equalization, suppression pool water will continuously recirculate through the reactor vessel via the ECCS pumps. The energy associated with the core decay heat will gradually raise the temperature of the suppression pool. Heat energy is subsequently removed from the suppression pool by the residual heat removal (RHR) heat exchangers. The capacity of the RHR heat exchangers is such that the maximum suppression pool temperature, reached after several hours, remains below the allowable limit. The increase in pool temperature and the corresponding increase in wetwell pressure are considered in the design of the ABWR containment systems.

3B.2.2.2 Intermediate Break Accident (IBA)

The IBA is defined as a break sized such that rapid depressurization of the RPV does not occur due to break flow. However, the reactor inventory loss is sufficiently rapid to cause a reduction in the reactor water level which may have potential for core uncovery. Since the ABWR design provides three high-pressure ECCSs, the vessel will be flooded without having to depressurize the reactor vessel.

The IBA will increase drywell pressure and temperature at a moderate rate, compared to that due to a large break accident. Water initially contained in the vent system will be accelerated from the vents. During the vent-clearing process, the water exiting the vents will form water jets in the suppression pool, which will cause loads on structures and equipment near the vent exits. The submerged structure loads from an IBA are less severe than those from a design basis accident (DBA). Structures and equipment designed for the DBA water jet loads can readily accommodate the less severe IBA water jet loads.

Immediately following vent water clearing, air and steam bubbles will form at the vent exits. The drywell pressurization rate for an IBA is less than due to an DBA. Consequently, the bubble pressure in the suppression pool is less severe and the moderate rate of drywell pressurization does not result in significant pool swell. The resulting IBA loads on pool boundaries, submerged structures, and equipment are bounded by the corresponding loads from a DBA.

A high drywell pressure signal scrams the reactor during the IBA. The sequence of events following the scram can lead to closure of the main steamline isolation valves (MSIVs) due to low reactor water level. The closure of the MSIVs can result in an increase in RPV pressure, which is relieved by opening of the SRVs. SRV discharge may continue intermittently to regulate reactor pressure and remove decay heat. Consequently, the suppression pool boundary may be subjected to a pressure loading resulting from the SRV discharge during IBA.

For intermediate size breaks, the steam flow rate through the vents may be insufficient to cause condensation oscillation (CO) loads as severe as those during a DBA. Following air carryover, however, chugging loads will be experienced until the reactor vessel blowdown is reduced to a flow rate where chugging becomes inginificant. The subsequent long-term pool temperature transient is essentially the same as that described for the DBA.

3B.2.2.3 Small Break Accident (SBA)

The SBA is defined as an event in which the fluid loss from the RPV is insufficient to either depressurize the reactor or result in a decrease of reactor water level. Following the break, the drywell pressure will slowly increase until the high drywell pressure scram setting is reached. The reactor will scram, but the MSIVs do not close immediately.

The drywell pressure will continue to increase at a rate dependent on the size of the postulated break. The pressure increase will depress the water level in the vent system until the water is expelled out and air and steam mixture enters the suppression pool. The air flow rate will be such that the air will bubble through the pool without causing any appreciable pool swell. The steam will be condensed and the drywell air will pass through the pool into the wetwell airspace. The wetwell airspace will gradually pressurize at a rate dependent upon the air carryover rate, which, in turn, depends upon the break size. Eventually, the steam and air flow through the vents will transfer essentially all the drywell air to the wetwell airspace. Following the air transfer, wetwell pressurization will increase at a rate dependent on the rate of increase of the suppression pool temperature.

The vent steam mass flux for an SBA is expected to be insufficient to produce steady condensation oscillation type loading conditions. However, there may be sufficient steam flow rate to cause chugging type of loading conditions. As the RPV depressurizes and cools down, the vent steam mass flux will decrease so that the vents will not remain cleared. Steam condensation will occur at the water interface inside the vents and on the walls of the vent system.

As a result of a postulated MSIV closure, the SRVs will initially discharge to control reactor vessel pressure in response to the isolation transient. Following the initial SRV discharge, SRV cycling will occur at the SRV setpoint pressure. When the temperature of the suppression pool reaches the Technical Specification limit of 54°C during normal operation, the operator will take action to begin a controlled depressurization of the reactor vessel, using manual operation of the SRVs if the MSIVs are closed, or using the main condenser if the MSIVs are open. The rate of depressurization, and thus the total duration of the SBA event, is dependent on operator action. A conservative value for analysis is taken as 56°C/h.

3B.3 Safety/Relief Valve Discharge Loads

During the actuation of a safety/relief valve (SRV), the air initially contained inside the SRV discharge line is compressed and subsequently expelled into the suppression pool by the SRV blowdown steam entering the SRV discharge line. The air exits through holes drilled into an X-quencher device which is attached to the SRV discharge line. The X-quencher discharge device is utilized in the ABWR design to promote effective heat transfer and stable condensation of discharged steam in the suppression pool.

3B.3.1 Quencher Description and Arrangement

The X-quencher discharge device used in the ABWR design is the same as that used in Mark III and Mark II designs. Reference 3B-1 contains a detailed description of design configuration features of the X-quencher discharge device. This discharge device (Figure 3B-2), is a diffuser device comprised of a short conical extension of the vertical terminus of the SRV discharge line and a capped cylindrical central section or plenum, from which four perforated, capped arms extend. The X-quencher, with four arms and many small holes in each arm, directs the air and steam over a broad area. Experimental data (References 3B-2 and 3B-3) have demonstrated the X-quencher characteristics of low air-clearing pressure loadings and no instability in the steam condensation process.

Figure 3B-3 shows the quencher azimuthal locations in the suppression pool. This arrangement distributes low, intermediate and high pressure-switch set valves uniformly around the pool to preclude concurrent adjacent valves operation.

3B.3.2 Quencher Discharge Loads

3B.3.2.1 Load Definition Methodology

After the air exits into the suppression pool, during the actuation of SRV, the air bubbles coalesce and oscillate as Rayleigh bubbles while rising to the pool free surface. The oscillating air bubbles produce hydrodynamic loads on the pool boundary and drag loads on structures submerged in the pool. After the air has been expelled, steam exits and condenses in the pool. The condensing steam produces negligible (pressure) amplitude loads on the pool boundary, as observed from X-quencher discharge testing.

The methodology for defining the quencher discharge loads (due to initial and subsequent SRV actuations) on the pool boundary for the ABWR containment will be consistent with previously approved methodology for Mark II and Mark III containment designs (NUREG-0802). Reference 3B-4 provides a detailed description of the calculational methodology. This methodology is based directly on empirical correlations obtained from mini-scale, small-scale, and large-scale (including inplant tests) tests conducted to develop a load definition methodology for X-quencher discharge loads during the SRV actuation events.

The X-quencher test data were statistically correlated to calculate the magnitude of quencher air clearing pressure loads on pool boundary as a function of several key parameters. The correlation was developed for use in both Mark II and Mark III containment systems using X-quencher discharge devices for the SRV discharge lines. Detailed descriptions of (1) the database, (2) a quantitative assessment of the test data in terms of the physical phenomena, (3) the procedure for identification and justification of key parameters used in the statistical correlations, (4) the statistical analysis of the data, and the resulting correlation equations, are provided in Section A12 of Reference 3B-1.

In summary, the calculation methodology consists of (1) a statistically derived correlation for predicting the magnitude of the peak positive bubble pressure and a relationship for calculating maximum negative pressure from the maximum positive pressure, (2) an idealized oscillatory pressure history representing subsequent interaction of the quencher air bubble with the suppression pool fluid, (3) a relationship for determining the pressure field in the pool as a function of distance from the quencher, and (4) a technique for determining the total air bubble pool boundary load for subsequent actuation from the first actuation loads, and when more than one quencher bubble exists in the pool (multiple valve actuation conditions).

3B.3.2.2 ABWR Design Quencher Discharge Loads

Quencher discharge pool boundary loads for design evaluation of the ABWR design will be defined after finalization of the ABWR SRV discharge line arrangement layout. After the SRV discharge line arrangement layout is finalized, the quencher pool boundary design loads will be computed using the methodology described above. The pool boundary pressure $P(r)$ will be calculated from bubble pressure, P_b, using the following relationship:

$$P(r) = \frac{2P_b x r_o}{r} \quad :\text{for } r > 2r_o \qquad (3B\text{-}1)$$

$$P(r) = P_b \quad :\text{for } r \le 2r_o \qquad (3B\text{-}2)$$

where

r_o	=	Quencher radius
r	=	Line-of-sight distance from quencher center point to the evaluation point (Figure 3B-4)

Air bubble pressure loads from a particular quencher location are considered to act only on boundaries which can be viewed from the quencher bubble with direct line of sight, as illustrated in Figure 3B-5. Figure 3B-6 shows the ideal quencher bubble pressure time history which is normalized with the maximum pressure value. This pressure time history profile will be used in determining pressure amplitude variation with time and the number of pressure cycles. It should be noted that the bubble pressure decay to $1/3P_{max}$ occurs in five cycles for any frequency between 5 and 12 Hz. The justification for this application is from examination of full-scale plant data where most traces were observed to decay to a small fraction of their peak value in two or three cycles.

The design loads will consider and include the following SRV actuation cases:

(1) Single valve discharge for first and subsequent actuations.

(2) Multiple valves discharge.

3B.3.2.2.1 Single Valve Discharge

The most frequent SRV discharge case during the plant lifetime is the single valve discharge of the low setpoint valve.This load case of single valve discharge deals with events such as inadvertent opening of an SRV and actuation of an SRV following small- or intermediate-line breaks in the primary system. A subsequent single SRV actuation may also result to provide pressure relief following a multiple-valve actuation, as described in Subsection 3B.3.2.2.2 below. This actuation, however, involves opening, closing, and reopening of an SRV. Therefore, pressure loading resulting from both first and subsequent SRV actuation will be considered. The SRV line resulting in the most severe pressure loading will be selected for design assessment.

Air bubble pressure loads from a particular quencher (SRV) are considered to act only on boundaries which can be viewed from the quencher bubble with direct line of sight as illustrated in Figure 3B-5, and no load acts on the shaded regions. Load attenuation in the vertical direction will be assumed as shown in Figure 3B-7.

3B.3.2.2.2 Multiple Valves Discharge

This case will cover the events in which all SRVs actuate and the resulting load on the pool boundary will be most severe.Events that are expected to actuate more than one SRV include generator load rejection, loss of main condenser vacuum, turbine trip, closure of all main steam isolation valves, and some less severe transients such as pressure regulator failure and loss of auxiliary power. Some of these anticipated transients may result in actuation of all SRVs. However, variation in time of actuation, valve opening time, and differences in individual discharge line lengths (which influence the time to complete line clearing) will introduce differences in phasing of the oscillating air bubbles in the suppression pool. Air bubbles oscillating out of phase will result in mitigating the pool boundary loads.

The pressure loading for multiple valve discharge will correspond to that resulting from simultaneous first actuation of all SRVs. In view that each SRV has a large value (of about 490 kPa) of blowdown pressure setpoint range, simultaneous subsequent actuation of all SRVs is not expected. Lowest relief set pressure a SRV may cycle and have subsequent actuation to provide pressure relief after multiple valve first actuation. In determining combined pressure loading on pool boundary due to multiple valve actuation, pressure loading due to individual SRV will be assumed equal to the largest of pressure loading calculated for individual valves. Pressure loading due to an individual valve is primarily determined by its relief pressure setpoint and discharge line air volume.The combined pressure loading from multiple valves at an evaluation point will be obtained by SRSS (Square Root of the Sum of Squares) of the individual loads from single valves.

As a bounding and conservative approach for structural evaluation, the multiple valves discharge case will consider and include most severe symmetric and asymmetric load cases. The most severe symmetric load case will assume oscillating air bubbles (from all valves) in phase, and the most severe asymmetric load case will assume one half of oscillating air bubbles 180° out of phase with the other half of oscillating air bubbles. These two load cases will bound all combinations of multiple valve actuation cases.

3B.3.3 Quencher Condensation Performance

After air discharge through the SRV line is completed, steady steam flow from the quencher will be established. Discharged steam condenses in the immediate vicinity of the discharge device. Thermal loads associated with steam jet contact can generally be avoided by appropriate orientation of the discharge device in the suppression pool.

Operating practice of early BWRs, in anticipation that extended SRV steady steam blowdown will heat the pool to a level where the condensation process may become unstable, a temperature limit for BWR suppression pools was established. This pool temperature limit, specified in NUREG-0783, was established because of concern that unstable steam condensation at high pool temperature could result in high loads on containment structure. Although quencher discharge devices (like the X-quencher) were found to produce smooth steam condensation process, at the time the pool temperature limit was established there were insufficient data available to confirm that quenchers were effective in mitigating loads sue to unstable steam condensation process. NUREG-0783 currently specifies acceptance criteria related to the suppression pool temperature limits for steady state steam condensation condition for the quencher discharge devices.

Recent studies, subsequent to the issuance of NUREG-0783, conclude that steady steam flow through quencher devices (like the X-quencher) is expected to be a stable and smooth condensation process over the full range of pool temperature up to saturation. It is also concluded that the condensation loads for stem discharge less than the loads from equivalent straight pipes. These recent studies are described and discussed in Reference 3B-5.

Subsequent to the studies reported in Reference 3B-5, there were additional test data from quencher discharge tests at high pool temperatures. These tests, reported in Reference 3B-15, showed a long, steady, turbulent, forced plume which consisted of a random two-phase mixture of entrained water and steam bubbles. This additional data, which showed formation of a long continuous steam plume at high pool temperatures, raised an additional concern. It was postulated that large continuous steam plumes may give rise to large bubbles that drift into a cooler region of the pool and suddenly collapse which could transmit significant loads to the pool boundary.

This additional concern was evaluated in a recent study, and i was determined that the continuous plume was not a transient flow shedding large coherent bubbles which might drift away and collapse in a cooler region of the pool. This recent study, described in deal in Reference 3B-16, concludes that the condensation process with SRV dischargers through quenchers (like the X-quencher) into the suppression pool would result in low amplitude loads for all suppression pool temperature.

In view of findings and conclusions from these recent studies discussed in above, it is concluded that suppression pool temperature limits (specified in NUREG-0783) for SRV discharge with quenchers are no longer necessary. Therefore, given that the ABWR design utilizes X-quencher discharge devices, the pool temperature limit specified in NUREG-0783 were not considered. However, ABWR design retains the restrictions on the allowable operating temperature envelope of the pool, similar to those in place for operating BWRs.

Further, the studies in Reference 3B-5 conclude that steam condensation loads with X-quenchers over the full range of pool temperature up to saturation are low compared to loads due to SRV discharge line air clearing and LOCA events. Therefore, considering that ABWR design considers SRV air Clearing and LOCA steam condensation loads for containment design evaluation, dynamic loads during the quencher steam condensation process will not be defined and considered for containment design evaluation.

3B.4 Loss-of-Coolant Accident Loads

In this section, methodologies for calculating the dynamic loading conditions associated with the various LOCA phenomena are presented.

3B.4.1 Pressure and Temperature Transients

A LOCA causes a pressure and temperature transient in the drywell and wetwell due to mass and energy released to the drywell. The severity of this transient loading condition depends upon the type and size of LOCA. Section 6.2 provides pressure and temperature transient data in the drywell and wetwell for the most severe LOCA case [design basis accident(DBA)]. This transient data establish the structural loading conditions in the containment.

3B.4.2 Vent Clearing and Pool Swell Loads Methodology

Following a postulated DBA, the drywell pressurizes and the water in the vents is expelled out into the suppression pool. The water forms jets in the suppression pool which induce loads on structures near the vent exits. After the water is cleared from the vents, the air in the vents and the drywell flows into the suppression pool. Air bubbles form at the vent exits, expand under the pool surface, and produce pressure loading on the suppression pool boundary. The expansion of the air bubble forces the slug of water above the bubbles to accelerate upward (pool swell), which causes both impact and drag loads on structures within the swell zone. Upon reaching the maximum swell height, the air bubbles that drive the water slug penetrates through the surface, resulting in froth formation. This froth will impact structures located above the maximum bulk swell height. The froth created after breakthrough experiences gravity-induced phase separation and will fallback toward the pool bottom. During this fallback, structures will be subjected to fallback drag loads.

Consistent with the load definition methodology for Mark I, II and III containments, sonic and compression wave loadings, occurring immediately following the postulated instantaneous rupture of a large primary system pipe, will not be considered and defined for design evaluation of the containment structure. It was concluded that these waves would result in a negligible structural response.

3B.4.2.1 Pool Boundary Loads

Following a postulated LOCA and after the water is cleared from the vents, air/steam mixture from the drywell flows into the suppression pool creating a large bubble at vent exit as it exits into the pool. The bubble at vent exit expands to suppression pool hydrostatic pressure, as the air/steam mixture flow continues from the pressurized drywell. Water ligament above the expanding bubble is accelerated upward by the difference between the bubble pressure and the air space pressure above the pool. This acceleration of water ligament gives rise to pool swell phenomena, which, typically, lasts for a couple of seconds.

During this pool swell phase, wetwell region is subjected to the hydrodynamic loading conditions, and they are:

- Loads on suppression pool boundary and drag loads on structures initially submerged in the pool, due to the pressurized and expanding bubble at vent exit
- Loads on wetwell airspace boundary (including the diaphragm floor), due to rising pool which compresses the wetwell air space
- Impact and drag loads on structures located above the initial pool surface, due to the rising pool surface

From a structural design standpoint, the most important aspects of the pool swell phenomena are peak pool swell height and peak pool swell velocity. The former determines a region of impact/drag loading condition, whereas the latter determines severity of the loading condition.

ABWR Pool Swell Loads

ABWR pool swell response calculations to quantify pool swell loads were based on a simplified, one dimensional analytical model, same as that reviewed and accepted by the staff (NEDE-21544P/NUREG-0808) for application to Mark II plants. This analytical model was qualified against Mark II full-scale test data. The ABWR design utilizes a confined wetwell airspace similar to that in Mark II design, but its vent system design is quite different than that in Mark II design. The ABWR vent system design utilizes horizontal vents similar to that in Mark III design. Therefore, recognizing this difference in vent system design, additional studies comparing model against Mark III horizontal vent test data were performed to assure adequacy of the model for application to ABWR.

Model Vs. Mark III Horizontal Vent Test Data

Model input/ assumptions used in predicting Mark III test data for model comparison were the same as prescribed in NEDE-21544P. Mark III horizontal vent system features were modeled in the following manner:

- Pool swell water slug was approximated by a consistent thickness equal to top vent submergence
- Drywell pressure transient and vent clearing times input based on test data
- Vent flow area increased in order with the clearing of middle and bottom vents

Test data used for model comparison were taken from full-scale and sub-scale tests, and they were representative of ABWR submergence to pool width ratio. The test data used in model comparison are listed in Table 3B-8.

Comparison results, summarized in Table 3B-9 and sample results shown graphically in Figures 3B-9 and 3B-10, demonstrate that the model over predicts the horizontal vent test data.

These comparison results demonstrate and assure adequacy of the model for calculating ABWR pool swell response.

Pool Swell Loads

Pool swell response calculations were done using the analytical model described above. Reference 3B-14 provides a detailed description of the model. The modeling scheme for calculations was consistent with that used for model vs. test data comparison. For an added conservatism in model predictions, water slug surface area occupied by the air bubble was taken as 80% of the total pool surface area in pool swell response calculations.

In modeling and simulating the pool swell phenomenon, the following assumptions were made:

(1) Noncondensable gases are assumed to behave as an ideal gas.

(2) After the vent clearing, only noncondensable gases flow through the vent system.

(3) The flow rate of noncondensable gases through the vent system is calculated assuming one-dimensional flow under adiabatic conditions and considering the pipe friction effects.

(4) The noncondensable gases contained initially in the drywell are compressed isentropically.

(5) The temperature of bubbles (noncondensable gas) in the pool is taken to be the same as that of the noncondensable gases in the drywell (from (4)).

(6) After the vent clearing, pool water of constant thickness above the top horizontal vent outlet is accelerated upward.

(7) Friction between pool water and the pool boundary and fluid viscosity are neglected.

(8) Noncondensable gases present in the wetwell airspace are assumed to undergo a polytropic compression process during the pool swell phase.

(9) For conservative estimates, a polytropic index of 1.2 will be used for computing the pool swell height and pool swell velocity, and an index of 1.4 for computing pressurization of the wetwell airspace.

(10) For added conservatism, pool swell velocity obtained in (9) above will be multiplied uniformally by a factor of 1.1 in defining impact/drag loads.

The calculated pool swell loading conditions for the ABWR containment system obtained from the analytical model (identified above) and the above modeling assumptions are shown in Table 3B-1.

Pool boundary pressure distribution is shown in Figure 3B-11.

Figure 3B-13 shows types of loading regions for structures located above the initial pool surface during the pool swell phase of LOCA. Key structures that will be subjected to these loads are: SRV discharge piping; catwalk structure; wetwell-to-drywell vacuum breakers.

Structures located between 0 and 7.6m above the initial surface will be subjected to impact load by an intact water ligament, where the 7.6m value corresponds to the calculated maximum pool swell height. The load calculation methodology will be based on that approved for Mark II and Mark III containments (NUREG-0487 and NUREG--0978).

Structures located at elevations between 7.6m and 10.9m will be subjected to froth impact loading. This is based on the assumption that bubble breakthrough (i.e., where the air bubbles penetrate the rising pool surface) occurs at 7m height, and the resulting froth swells to a height of 3.3m. This froth swell height is the same as that defined for Mark III containment design, and this is considered to be a conservative value for the ABWR containment design. Because of substantially smaller wetwell gas space volume (about 1/5th of the Mark III design), the ABWR containment is expected to experience a froth swell height substantially lower than that in Mark III design. The wetwell gas space is compressed by the rising liquid slug during pool swell, and the resulting increase in the wetwell gas space pressure will decelerate the liquid slug before the bubble break-through process begins. The load calculation methodology will be based on that approved for the Mark III containment (NUREG-0978).

As shown in Figure 3B-13 the gas space above the 10.9m elevation will be exposed to spray condition, which is expected to induce no significant loads on structures in that region.

As drywell air flow through the horizontal vent system decreases and the air/water suppression pool mixture experiences gravity-induced phase separation, pool upward movement stops and the “fallback” process starts. During this process, structures between the bottom vent and the 10.9m elevation can experience loads as the mixture of air and water fall past the structure. The load calculation methodology for the defining such loads will be based on that approved for Mark III containment (NUREG-0978).

3B.4.2.2 Loads on Access Tunnel

The ABWR design provides two access tunnels through the suppression chamber for access from the reactor building to the lower drywell. These tunnels provide access for personnel and equipment, and the bottom of these two tunnels is partially submerged when suppression pool water is at its nominal level position. During pool swell, the access tunnel will be subjected to drag load only. Because of their initial partial submergence, the tunnels are not expected to experience any impact load due to pool swell.

The drag load imposed on the access tunnel due to pool swell will be calculated by the following equation:

$$P_d = 1/2 C_d \zeta \frac{V_{Max}^2}{g_c} \tag{3B-3}$$

where

P_d = Drag pressure load

C_d = Drag coefficient

V_{Max} = Maximum pool swell velocity

ζ = Density of water

g_c = Conversion constant

The drag coefficient, C_d, will be determined from Figure 3B-14. The maximum velocity, V_{Max} in the above equation will be 1.1 times the maximum vertical velocity calculated from the pool swell analytical model.

The pressure loading due to air bubble pressure will be calculated and added to the drag load. This pressure loading will consider and account for the wetwell airspace pressure as shown below:

$$P_b = P_B - P_{w/w} \tag{3B-4}$$

where

P_b = Net pressure loading due to air bubble pressure

P_B = Air bubble pressure

$P_{w/w}$ = Wetwell airspace pressure

In addition to the drag and air bubble pressure loading, the access tunnel will also be subjected to buoyancy loading, as shown by the following equation:

$$P_{by} = \frac{\zeta V_T}{A} \tag{3B-5}$$

where

P_{by} = Pressure loading due to buoyancy

ζ = Density of water

V_T = Displaced volume of access tunnel

A = Projected area of access tunnel

3B.4.2.3 Impact and Drag Loads

As the pool level rises during pool swell, structures or components located above the initial pool surface (but lower than its maximum elevation) will be subjected to water impact and drag loads. The following equations will be used to compute the applicable impact and drag loads on affected structures.

Impact Load

The impact loading on structures between initial pool surface and the maximum swell height due to pool swell will be calculated by the following equation:

$$P(t) = P_{Max}\frac{(1 - \mathrm{Cos}(2\pi t/T))}{2} \quad \text{(3B-6)}$$

where

$P(t)$ = Pressure acting on the projected area of the structure

P_{Max} = The temporal maximum of pressure acting on projected area of the structure

t = Time

T = Duration of impact

For both cylindrical and flat structures, the maximum pressure P_{Max} and pulse duration T will be determined as follows:

(a) The impulse will be calculated using the equation given below:

$$I_P = \left(\frac{M_H}{A}\right)\frac{V}{gc} \quad \text{(3B-7)}$$

where

I_P = Impulse per unit area

$\left(\frac{M_H}{A}\right)$ = Hydrodynamic mass per unit area

V = Impact velocity

gc = Conversion Constant

(b) The hydrodynamic mass per unit area for impact loading will be obtained from the appropriate correlation for a cylindrical or flat target from Figure 6-8 of Reference 3B-6.

(c) The pulse duration T will be obtained from the following equation:

Cylindrical Target:

$$T = (0.0463 \times D)/V \tag{3B-8}$$

Flat Target:

$$T = (0.011 \times W)/V \quad \text{for} \quad V \geq 2.13\,\text{m/s}$$

$$= (0.0016 \times W) \quad \text{for} \quad V < 2.13\,\text{m/s}$$

where

T = Pulse duration, s

D = Diameter of cylindrical pipe, m

W = Width of the flat structure, m

V = Impact velocity, m/s

(d) The value of P_{Max} will be obtained using the following equation:

$$P_{Max} = (2I_P)/T \tag{3B-9}$$

For both cylindrical and flat structures, a margin of 35% will be added to the P_{Max} values (as specified above) to obtain conservative design loads.

Drag Load

Following the impact loading, the structure above the initial pool surface (but below the maximum swell height) will be subjected to the standard drag loading. This drag loading will be calculated using the methodology described in Subsection 3B.4.2.2.

$$P_d = 1/2C_d\zeta(V^2/g_c) + V_A\zeta(\dot{V}/g_c) \qquad (3B\text{-}10)$$

where

P_d	=	Drag pressure
C_d	=	Standard drag coefficient
V	=	Pool swell velocity
ζ	=	Density of water
g_c	=	Conversion constant
V_A	=	Acceleration drag volume
$\dot{V}$	=	Pool acceleration

The standard drag coefficient, C_d, in the above equation and acceleration drag volume, V_A, will be used consistent with those defined and used in Reference 3B-1. The time history of pool swell velocity will be calculated from the analytical model described in Subsection 3B.4.2. The pool swell velocity calculated from the analytical model will be multiplied by a factor of 1.1 (for conservatism) for application in calculation of the drag load. Time history of pool swell surface level will be determined from the same analytical model used for pool swell velocity calculation. The pool swell acceleration, $\dot{V}$, will be computed from the pool swell velocity data.

3B.4.2.4 Loads on Diaphragm Floor

Rapid pressurization of the wetwell airspace during the pool swell transient has a potential for upward differential pressure loading on the diaphragm floor. Results from the pool swell analytical model, however, showed that wetwell airspace pressure did not exceed the drywell pressure during the pool swell transient. Hence, it is concluded that the diaphragm floor will not be subjected to an upward differential pressure loading. The diaphragm floor will be subjected to only downward differential pressure loading, during the pool swell phase.

3B.4.3 LOCA Steam Condensation Loads

3B.4.3.1 ABWR Horizontal Vent Test Program

LOCA loads with the horizontal vent system design have been well characterized during the Mark III Confirmatory Test Program. More than 200 tests have been performed to determine horizontal vent system performance and associated LOCA loads. However, all of these tests have utilized the low containment pressure characteristics of the Mark III containment system (about 34.32k Pa G). Because of some thermodynamic and geometrical differences between the ABWR and Mark III designs, it was anticipated that condensation oscillation (CO) and chugging (CH) loads might differ from prior (Mark III) testing in horizontal-vent facilities. These included (1) increased ABWR wetwell airspace pressure, and hence subcooling, (2) the presence of a lower drywell (L/D), (3) the smaller number of vents (30 in ABWR vs. 120 in Mark III), (4) extension of the vents in the pool, (5) vent submergence, and (6) suppression pool width.

Considering the existence of the above thermodynamic and geometrical differences, a test program was conducted to confirm the CO and CH loads which would occur in the event of a LOCA in an ABWR plant. The test program, test data, and interpretation of test data are documented in Reference 3B-7.

The test program consisted of 24 simulated blowdowns (Table 3B-2) in test facilities representing the horizontal-vent ABWR design. The tests were divided into two parts utilizing sub-scale (SS) and partial full-scale (FS*) test facilities shown in Figures 3B-14 and 3B-15, respectively. Figures 3B-17 and 3B-18 show test sensors common to FS* and SS tests, and unique to FS* and SS tests, respectively.

The SS facility had all linear dimensions reduced by a factor of 2.5 from prototypical ABWR dimensions. Thirteen SS tests were performed primarily for the purpose of obtaining CO data. A full-scale vertical and horizontal-vent configuration was installed for the FS* tests. The upper drywell (U/D) was enlarged but not to prototypical dimensions. Eleven FS* tests were performed primarily for the purpose of obtaining CH data. The test matrix for the 24 blowdowns (Table 3B-2) included variations in pool temperature, break size, wetwell backpressure, and type of break (steam or liquid). The test facilities were equipped adequately with the data sensors to obtain necessary data for understanding the phenomena and establishing a database for defining CO and CH loads for the ABWR containment. In addition to the geometrical considerations, the facility was designed to minimize the potential for fluid-structure interaction (FSI). Measurements were taken at seven locations on the wetted suppression pool boundary to record dynamic pressure oscillations. Structural instrumentation (strain gauges and accelerometers on the basemat, pedestal, and containment walls) was used to confirm that FSI effects were minimal. Pressure transducers in the vertical and horizontal vents recorded dynamic loads on the vent system.

Loads due to condensation oscillation and chugging are described and discussed in Subsections 3B.4.3.2 and 3B.4.3.3, respectively.

3B.4.3.2 Condensation Oscillation (CO) Loads

The condensation oscillation (CO) period of a postulated LOCA follows the pool swell transient. During the CO period, both the vent steam mass flux and vent air content are decreasing. The steam-water interface at the vent exit oscillates as the steam is condensed. The vent steam mass flux is sufficient to prevent water flow into the vent. The steam condensation process at the vent exit induces pressure loads on the containment system, including the suppression pool boundaries and structures submerged in the suppression pool.

3B.4.3.2.1 Description of CO Database

A detailed description, evaluation, and discussion of CO data are given in Reference 3B-7.

The test program consisted of a total of 13 simulated blowdowns in sub-scaled test facility representing a one-cell (360°) sector of the ABWR horizontal vent design, which included a signal vertical/horizontal vent module. The subscaled (SS) test facility was geometrically (all liner dimensions scaled by a factor of 2.5) similar to the prototypical ABWR design, and the single vertical/horizontal vent module included all three horizontal vents, as shown in Figure 3B-15. In these tests, full-scale thermodynamic conditions were employed. This approach is based on the belief that condensation phenomena at the vent exit are mainly governed by the thermodynamic properties of the liquid and vapor phases. In accordance with this scaling procedure, measured pressure amplitudes are equal to full-scale values at geometrically similar locations, whereas measured frequencies are 2.5 times higher than the corresponding full-scale frequencies.The technical basis for using this scaling approach was based on extensive review and evaluation of the available literature on CO scaling and scaled tests performed for Mark II and Mark III containments, as well as general consensus of technical experts in this field. The CO scaling studies, which have been performed independently by various technical experts, show that for tests in a geometrically scaled facility with full-scale thermodynamic conditions, the measured pressure amplitudes are the same as full-scale values at geometrically similar locations, and measured pressure frequencies are the scale factor times higher than the corresponding full-scale frequencies.

Therefore, CO frequencies for the full-scale ABWR design are obtained by scaling the frequencies measured in SS tests by a factor of 2.5. A similar technique is applicable to scaling adjustment in frequency for obtaining full-scale values. Thus, this scaling procedure made it possible to use the measured SS data (pressure time history) directly for load definition purpose after the time scale is compressed by a factor of 2.5.

Out of the 13 SS tests, the tests recommended for definition of the CO load are SST-1, 2, 3, 9, 11, and 12. These six tests, summarized in Table 3B-2, were run at prototypical conditions. Of the remaining tests, SST-4, 5, 6, 7, 8, and 14 were run with a prepurged vent system, and SST-10 was run with the L/D blocked off. These tests are valuable for understanding CO phenomena and the effects of system variables, but they are not considered to be an appropriate basis for the CO load definition.

3B.4.3.2.2 Evaluation of CO Database

Each of the CO load definition tests shows significant frequency peaks at 5 and 9 Hz. Figure 3B-19 illustrates this behavior at the basemat (019P) sensor location. The 9 Hz frequency corresponds to a full-scale CO driver frequency near 4 Hz, which is representative of diameter at vent exit. The lower frequency is associated with the vent acoustic frequency which is representative of drywell-to-wetwell connecting vent.

Further examination of the data shows that, in general, the largest amplitude loads occurred at transducer location 019P (on basemat, near pedestal wall). It was observed that the highest amplitude CO loads occurred during the first 30 seconds of tests SST-1 and 2 (large liquid breaks at 49°C pool temperature). Examination of the Power Spectral Density (PSD) data showed that the envelope PSD of the pressure at 019P from a 12-second segment in SST-1 and an 18-second segment in SST-2 matched the envelope PSD of the 019P pressure from the six-test database. This is shown in Figure 3B-20 where the two envelope PSDs at 019P are compared. The PSD from the SST-1 and 2 time segments is indistinguishable from the six-test envelope except for a small frequency interval near the origin.

3B.4.3.2.3 CO Load Definition

The load definition methodology for the design CO load for the ABWR containment uses a source load approach which is described in the following subsections. In development of this methodology, all pertinent data from Mark II and Mark III tests were reviewed and considered.

3B.4.3.2.3.1 Source Load Approach

For prior BWR containments, the CO loading is defined and applied as a rigid-wall load at the fluid-structure interface. This application of the CO loading to a coupled containment model can produce relatively large structural responses, since the pressure oscillation frequencies are close to the natural frequencies of the fluid-structure system.

An alternate formulation of the CO load, termed as “Source Load Approach”, is to develop a source load. The source load is a series of pulses which simulates the oscillation of the steam/water interface at the horizontal vent exits. This approach has recently been used successfully for BWR containments. In this approach the CO source load would be applied to a coupled fluid-structure model of the ABWR containment as an excitation of the steam/water interface at the exits of the horizontal vents. It is the oscillatory motion of the steam/water interface which produces the characteristic oscillatory pressure loading on the wall. With a source load, it will be possible to account for the spatial distribution of the load and the variation of pool and vent fluid properties in a natural way. This approach avoids the problem of artificial resonant amplification at the system frequencies.

Figure 3B-21 describes the CO source load methodology. In order to develop a technically justified source loading function, the methodology includes the following elements:

(1) A comprehensive test database

(2) A coupled steam-water-structure interaction model of the test facility from which the data were obtained

(3) A procedure to develop a “test source” loading configuration

(4) A criteria to evaluate the test source loading configuration and test facility model

(5) A procedure to scale up the test source to a full-scale design source for the ABWR containment system

(6) A full-scale coupled steam-water-structure interaction model of the ABWR containment system

(7) A criteria to evaluate the design source loading condtion for the ABWR containment system

(8) Calculation of CO design (wall pressure) from the ABWR analysis using the design source

Criteria for CO Source Load

An acceptance criterion is specified in order to provide a basis for judging the acceptability of the source loading function with respect to prediction of wall pressure loadings and their frequency contents. The criteria include the following elements:

(1) Wall pressure histories for the SS test facility produced by the test source match with the pressures measured in the SS test facility.

(2) Frequency content of the predicted pressure histories, as defined by a power spectral density (PSD) and by an amplitude response spectrum (ARS), matches with the data obtained from the SS test facility.

(3) Spatial distribution of the root mean square (RMS) of the predicted loading matches with the data measured from the SS test facility.

(4) Wall pressures predicted by the design source for the full-scale (ABWR) facility match with the pressures measured in the SS test facility at geometrically similar locations. Note, this is required by the CO scaling laws (References 3B-8 and 3B-9).

Figure 3B-22 shows pressure time history representative of ABWR CO load determined using the source load approach described above. Figure 3B-23 shows pressure time history representative of Mark III CO loads. This pressure time history is based on Mark III CO correlation described and discussed in GESSAR. In comparison, ABWR pressure amplitudes are higher than for Mark III design.

The higher amplitudes in ABWR can be attributed to a deeper submergence in ABWR (3.5m vs. 2.3m), and the fact that in ABWR tests all three horizontal vents remained open during the maximum CO period whereas in Mark III tests the bottom and middle vents were closed at the onset of CO conditions. There may be some partial contribution from increasing wetwell overpressure during CO period in ABWR tests.

Load Application Methodology

For design evaluation of containment structure, the pool boundary pressure loads obtained from analysis of single-vent (36°) model of the prototypical ABWR design were specified and applied over the full (360°) model of the ABWR configuration. This CO loading specification implies all vertical vents are in phase (i.e., no credit for phasing among vents), which is considered to be a conservative load definition approach.

3B.4.3.3 Chugging Loads

Chugging, a hydrodynamic phenomenon associated with a LOCA, follows the CO period and occurs during periods of low vent steam mass flux and, typically, produces a sharp pressure pulse followed by a damped oscillation. During chugging, rapid steam condensation causes the pool water to re-enter the vents. This is followed by a quiescent period until the steam-water interface is forced out into the pool. Thus, chugging, an intermittent event, is the result of unsteady condensation occurring in the last stages of the blowdown. As stated earlier, specific tests were conducted to obtain chugging data for defining the chugging loads for the ABWR containment system.

3B.4.3.3.1 Description of Chugging Data

As shown in Table 3B-2 under the FS* part of the Horizontal Vent Test (HVT) matrix, 11 tests were performed primarily for the purpose of establishing a database for definition of the CH load for ABWR design evaluation. The HVT facility for the FS* test series was run with a full-scale vertical vent and horizontal vent system and an enlarged U/D. The tests were run at prototypical mass flux and pool temperature and with the vent system purged of air. It is known from previous blowdown testing and observations that presence of air in the vent reduces CH loads, so running chugging tests at prepurged conditions is conservative.

3B.4.3.3.2 Evaluation of Chugging Data

A detailed description and discussion of chugging data are contained in Reference 3B-7.

A typical large chug from the full-scale database is shown in Figure 3B-21 and its PSD in Figure 3B-25. It is characterized by a small underpressure, followed by a positive pressure pulse, and a decaying ringout. These phenomena are associated with the initial contraction of the steam bubble, the rapid deceleration of pool water converging on the vent exit, and the excitation of an acoustic standing wave in the pool. Chugging data from Mark II and Mark III testing also exhibited similar characteristic features.

Mean and standard deviation values for peak overpressure (POP) and pressure root-mean-square (PRMS) are shown in Table 3B-3. This amplitude data clearly show that the most severe chugging occurs for the steam breaks with an initial pool temperature of 21°C. Both POP and PRMS decrease significantly as the pool temperature is raised to 49°C. The variation is much less significant between 49 and 68°C. In general, the data support the understanding (observed from prior tests) that chugging has some dependence on system parameters, such as mass flux and pool temperature, along with a substantial degree of randomness.

Tables 3B-4 to 3B-6 show the average periods by 10-second segment for the steam breaks at 21, 49, and 68°C, respectively. Table 3B-7 presents the weighted average of the chug periods within each 10-second segment for all the tests at a given temperature.

3B.4.3.3.3 Chugging Load Definition

Figure 3B-26 shows various elements of the source load methodology for defining the chugging load on the pool boundary. The database consisted of 11 tests conducted in the HVT facility with the full-scale vent system. From this database, key chugs were selected which serve as criteria for the development of the source load. The key-chug approach was used successfully for the definition of the Mark II chugging load (Reference 3B-10).

Key chug selection was determined by requiring that the PSD envelope of the selected key chugs matches the PSD envelope of the FS* chugging database. The criterion for a technically justifiable chug design source is that the design source load, when applied to an analytical model of the HVT facility, produces a wall pressure which matches the selected data and a PSD envelope which envelopes the PSD envelope of the selected data.

3B.4.3.3.3.1 Pool Boundary Loads

Eight different chugging design sources, represented by a single pulse acting at the exit of top vent in a full-scale model, were defined. The design sources were determined by imposing a requirement that the PSD envelope generated by these design sources bounds the PSD envelope from the selected chugging data. A comparison of PSD results from analysis and test data (for sensor location 019P) is shown in Figure 3B-24. Figure 3B-28 shows spatial distribution of maximum pool boundary pressure. For design evaluation of affected structures, a total of eight pressure time histories, corresponding to eight design sources, will be computed and specified. A typical pressure time history at the bottom pool boundary is shown in Figure 3B-29 which will form the basis for the spatial distribution shown in Figure 3B-28.

Load Application Methodology

The pool boundary pressure loads obtained from analysis of a single-vent (36°) sector model of the prototypical ABWR design were specified for application over the full (360°) model of the prototypical ABWR facility. To bound symmetric and asymmetric loading conditions, two load cases were defined.

Case 1: All vents chugging in phase.

Case 2: Vents in one half chugging 180° out of phase with the other half vents

3B.4.3.3.3.2 Loads on Access Tunnel

The bottom of the access tunnel is treated as a wetted boundary at the top of the top pool surface. The axial and circumferential pressure loadings on the submerged portion of the access tunnel are as shown in Figures 3B-28 and 3B-30, respectively. The pressure attenuation in the axial direction is assumed to vary linearly from the pedestal to the opposite containment wall, and the circumferential attenuation on the tunnel perimeter is assumed to vary with the submergence.

3B.4.3.3.3.3 Loads on Horizontal Vent

The HVT FS facility was instrumented with two load cells on the top horizontal vent to measure the vertical force and bending moment experienced by the vent during chugging. Steam bubble collapsing inside the vent has a potential to induce significant loading on the horizontal vent. With the ABWR vent system design, in which the horizontal vents project into the pool, it is anticipated that these may be significant for structure evaluation.

Typical test results from the HVT program are exhibited in Figures 3B-31 and 3B-32. The maximum upward load and maximum moment values were not observed to occur simultaneously on the horizontal vent.

For structure evaluation of the horizontal vent pipe and pedestal, an upward load, based on the HVT test data, is conservatively defined as shown in Figure 3B-24.

For building structure response analysis for the evaluation of RPV and its internals, the horizontal vent upward load is specified as shown in Figure 3B-28. To bound symmetrical and asymmetrical loading conditions, the following two load cases will be considered and analyzed.

(1) Upward load on the pedestal wall simultaneously at all top 10 horizontal vents

(2) Upward load on the pedestal wall simultaneously at top five vents in one-half side of pedestal

3B.4.4 RCIC Turbine Exhaust Steam Condensation

The Reactor Core Isolation Cooling (RCIC) system, which forms part of the Emergency Core Cooling Systems (ECCS), will maintain sufficient reactor water inventory in the event that the reactor vessel is isolated and the feedwater supply unavailable. The RCIC system injects water into a feedwater line, using a pump driven by a steam turbine. The steam turbine is driven with a portion of the decay heat steam from the reactor vessel and the turbine exhaust steam is piped into the suppression pool where it is condensed. The RCIC system is designed to perform its intended function without AC power for at least 2 hours with a capability up to 8 hours.

In view that the turbine steam discharges and condenses in the suppression pool and the expected long duration of RCIC operation, there exists a potential for steam condensation loading on the pool boundary. Significance of such potential loading on the pool boundary (steel liner, in specific) was examined, and it was determined that this loading condition will be well bounded by the LOCA steam condensation design loads.

3B.4.4.1 Exhaust Steam Condensation Loading

The RCIC system is a safety system, consisting of a steam turbine, pump, piping, accessories, and necessary instrumentation. The steam turbine exhaust steam piping is ASME Code Class 2 piping, as noted in the RCIC P&ID in Tier 2 Figure 5.4-8. To minimize exhaust steam line vibration and noise levels, the discharge end of the turbine exhaust line will be equipped with a condensing sparger. The sparger design configuration will be similar to that currently used successfully for the operating BWRs.

The condensing sparger is expected to produce a very smooth steam condensation operation resulting in low pressure fluctuations in the poll, which would imply low pressure on the pool boundary. During RCIC operation, steam mass flux in the neighborhood of 470.72 Pa·s are expected, which should assure smooth steam condensation process. During the extended RCIC operation, condensing exhaust steam will bring the poll to high temperature. At high poll temperatures, long plumes consisting of a random two-phase mixture of entrained water and steam bubbles are expected to exist. As reported in Reference 3B-16, the condensation of the steam within such a mixture will not give rise to large bubbles that drift in to a cooler region of the pool and suddenly collapse which could transmit significant loads to the pool boundary.

Therefore, in view of above, steam discharge through the condensing sparger is expected to be a smooth condensation process which would result in low pressure fluctuation loading on the pool boundary. This expected asymmetric loading condition, which is expected to be a low pressure fluctuation loading, should be bounded by the LOCA steam condensation (CO and Chugging) loads defined for the ABWR design. Further, the ABWR design load definition specifies a bounding asymmetric load case which assumes vents in one half chugging 180° out of phase with the other half vents. This is a conservative representation of asymmetric loading.

In summary, it is concluded that steam condensation loads associated with the RCIC turbine exhaust steam discharge (via condensing sparger) to the pool will produce low pressure fluctuation loads on the pool boundary. Such loads should be well bounded by the LOCA steam condensation loads. The turbine exhaust piping, being designated as ASME Class 2 piping, shall be designed to retain its pressure integrity and functional capability.

3B.5 Submerged Structure Loads

Structures submerged in the suppression pool can be subjected to flow-induced hydrodynamic loads due to LOCA and SRV actuations.

During a LOCA, steam/water mixture rapidly escapes from the break, and the drywell is rapidly pressurized. The water initially in the vent system is expelled out into the suppression pool. A highly localized induced flow field is created in the pool and a dynamic loading is induced on submerged structures. After the water is expelled from the vent system, the air initially in the drywell is forced out through the horizontal vents into the suppression pool. The air exiting from the vents forms expanding bubbles which create moderate dynamic loads on structures submerged in the pool. The air bubbles cause the pool water surface to rise until they break through the pool water surface. The pool surface water slug decelerates and falls back to the original pool level. Steam/water mixture from the break soon fills the drywell space and is channeled to the pool via the vent system. Steam condensation starts and the vibratory nature of pool water motion causes an oscillatory load on submerged structures.

The CO loading continues until the pressure in the drywell decreases. This is followed by a somewhat regular but less frequent vibration called chugging (CH). During the CH period, a high frequency spike is propagated, which causes an acoustic loading on submerged structures.

During SRV actuations, the dynamic process of the steam blowdown is quite similar to LOCA steam blowdown but the induced load is mitigated by the X-quencher device attached at the end of each discharge device. Two types of loads are important. One is due to the water jet formed at the confluence of the X-Quencher arm discharges and another is due to the four air bubbles formed between the arms of the X-Quencher. These air bubbles are smaller in size than the LOCA air bubbles, reside longer in the pool, and oscillate as they rise to the free surface of the pool.

Key submerged structures that will be subjected to significant loads due to LOCA (pool swell, CO, CM) and SRV actuation events are:

- Submerged portion of SRV discharge lines
- SRV discharge line X-quencher discharge device and its support structure
- Personnel and equipment access tunnels (partially submerged)
- ECCS suction lines and strainers.

3B.5.1 Pool Swell Submerged Structure Loads

During the initial phase of the DBA, the drywell airspace is pressurized and the water in the vents is expelled to the pool and induces a flow field throughout the suppression pool. This induced flow field is not limited to direct jet contact and creates a dynamic load on structures submerged in the pool.

However, since none of the submerged structures in the ABWR containment is in the direct path of these jets, the dynamic load on these structures is less than the load induced by the LOCA air bubble that forms after the water is expelled out. Since the air bubble induced dynamic load is bounding, this load is conservatively used in place of water jet load.

After the vents are cleared of initially contained water, pressurized drywell air is purged into the suppression pool, and a single bubble is formed around each vent exit. It is during the bubble growth period that unsteady fluid motion is created within the suppression pool. During this period, all submerged structures below the pool surface will be exposed to transient hydrodynamic loads.

The load definition methodology for defining the LOCA bubble-induced loads on submerged structures will be consistent with the methodology used for prior plants, as described in Reference 3B-11.

3B.5.2 Condensation Oscillation Submerged Structure Loads

During a LOCA, after the vent is cleared of water and the drywell air has been carried over into the wetwell, steam condensation begins. This condensation oscillation phase induces bulk water motion and, therefore, creates drag loads on structures submerged in the pool.

The load definition methodology for defining the LOCA steam condensation oscillation loads on submerged structures will be consistent with the methodology used for prior plants. The methodology is described in Reference 3B-12.

3B.5.3 Chugging Submerged Structure Loads

Chugging occurs after drywell air has been purged and carried over into the wetwell, and the vent steam mass flux falls below a critical value. Chugging then induces acoustic pressure loads on structures submerged in the pool.

The load definition methodology for defining the LOCA chugging loads on submerged structures will be consistent with methodology used for prior plants. The methodology is described in Reference 3B-12.

3B.5.4 SRV Submerged Structure Loads

Following the actuation of a SRV, water contained initially in the discharge line is rapidly discharged through the X-Quencher discharge device attached at the end of the SRV discharge line. A highly localized water jet is formed around the X-Quencher arms. The hydrodynamic load induced outside a sphere circumscribed around the quencher arms by the quencher water jet is not significant. There are no submerged structures located within the sphere mentioned above in the ABWR arrangement. The induced load for submerged structures located outside the circumscribed sphere by the quencher arm is negligible and is ignored.

After the water discharge, the air initially contained in the discharge line is forced into the suppression pool under high pressure. The air bubbles formed interact with the surrounding water and produce oscillating pressure and velocity fields in the suppression pool. This pool disturbance gives rise to hydrodynamic loads on submerged structures in the pool.

The load definition methodology for defining the SRV air bubble loads on submerged structures will be consistent with that used for prior plants. The methodology is described in References 3B-11 and 3B-13.

3B.6 Loads Combination

Under certain plant conditions, the containment structures can be subjected simultaneously to hydrodynamic loads due to LOCA and SRV actuations. Event-time relationships showing load combination histories for design assessment of the ABWR containment system will be, in general, consistent with the approach used for prior plants. If found necessary, any ABWR-unique features will be considered and addressed appropriately.

3B.7 References

3B-1 General Electric Company, “Containment Loads Report (CLR), Mark III, Containment”, 22A4365AB, Rev. 4, Class III, January 1980.

3B-2 General Electric Company, “Caorso SRV Discharge Tests Phase I Test Report”, NEDE-25100-P, May 1979.

3B-3 General Electric Company, “Caorso SRV Discharge Tests Phase II ATR”, NEDE - 25118, August 1979.

3B-4 GESSAR II, Appendix 3B, Attachment A, 22A7007, 1984.

3B-5 General Electric Company, “Elimination of Limit on BWR Suppression Pool Temperature For SRV Discharge With Quenchers”, NEDO - 30832, Class I, December 1984.

3B-6 McIntyre, T. R. et al., “Mark III Confirmatory Test Program One-Third Scale Pool Swell Impact Tests - Test Series 5805”, General Electric Company, NEDE - 13426P, Class III, August 1975.

3B-7 General Electric Company, “Horizontal Vent Confirmatory Test, Part I”, NEDC - 31393, Class III, March 1987.

3B-8 Sonin A. A., “Scaling Laws In Small-Scale Modeling of Steam Relief Into Water Pool”, ASME Winter Meeting, Chicago, November 1980.

3B-9 Dodge, F. T., "Scaling Study of the GE PSTF Mark III Long Range Program, Task 2.2.1, SwRI", General Electric Company Report NEDE - 25273, March 1980.

3B-10 Mark II Containment Program, "Generic Chugging Load Definition Report", NEDE-24302-P, Class III April 1981.

3B-11 F. J. Moody, "Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and Safety Relief Valve Ramshead Air Discharges", NEDE-21471; revised by L. C. Chow and L. E. Lasher, September 1977.

3B-12 L. E. Lasher, "Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by Steam Condensation and Chugging", NEDO-25153, July 1978.

3B-13 T. H. Chuang, L. C. Chow, and L. E. Lasher, "Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and Safety Relief Valve Ramshead Air Discharges", NEDO - 21471, Supplement 1, June 1978.

3B-14 Ernst, "Mark II Pressure Suppression Containment Systems: An Analytical Model of the Pool Swell Phenomenon", NEDE - 21544-P, General Electric Company, December 1976.

3B-15 J-H, Chun and A.A. Sonin, "Small-scale Simulation of Vapor Discharge into Subcooled Liquid Pools", Nuclear Engineering and Design **85**(1985) pp 353-362.

3B-16 Letter, January 20, 1994, General Electric (Jack Fox) to the Staff (Chet Poslusny), "Containment Emergency Procedure Guidelines Issue on Heat Capacity Temperature Limit (HCTL)", Docket No. 52-001.

Table 3B-1 Pool Swell Calculated Values

Description	Value
1. Air bubble pressure (maximum)	148.7 kPaG
2. Pool swell velocity (maximum)	7.5m/s
3. Wetwell airspace pressure (maximum)	98.7 kPaG
4. Pool swell height (maximum)	7.6m

Table 3B-2 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Table 3B-2 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)] (Continued)

Table 3B-3 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Table 3B-4 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Table 3B-5 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Table 3B-6 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Table 3B-7 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Table 3B-8 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Table 3B-9 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

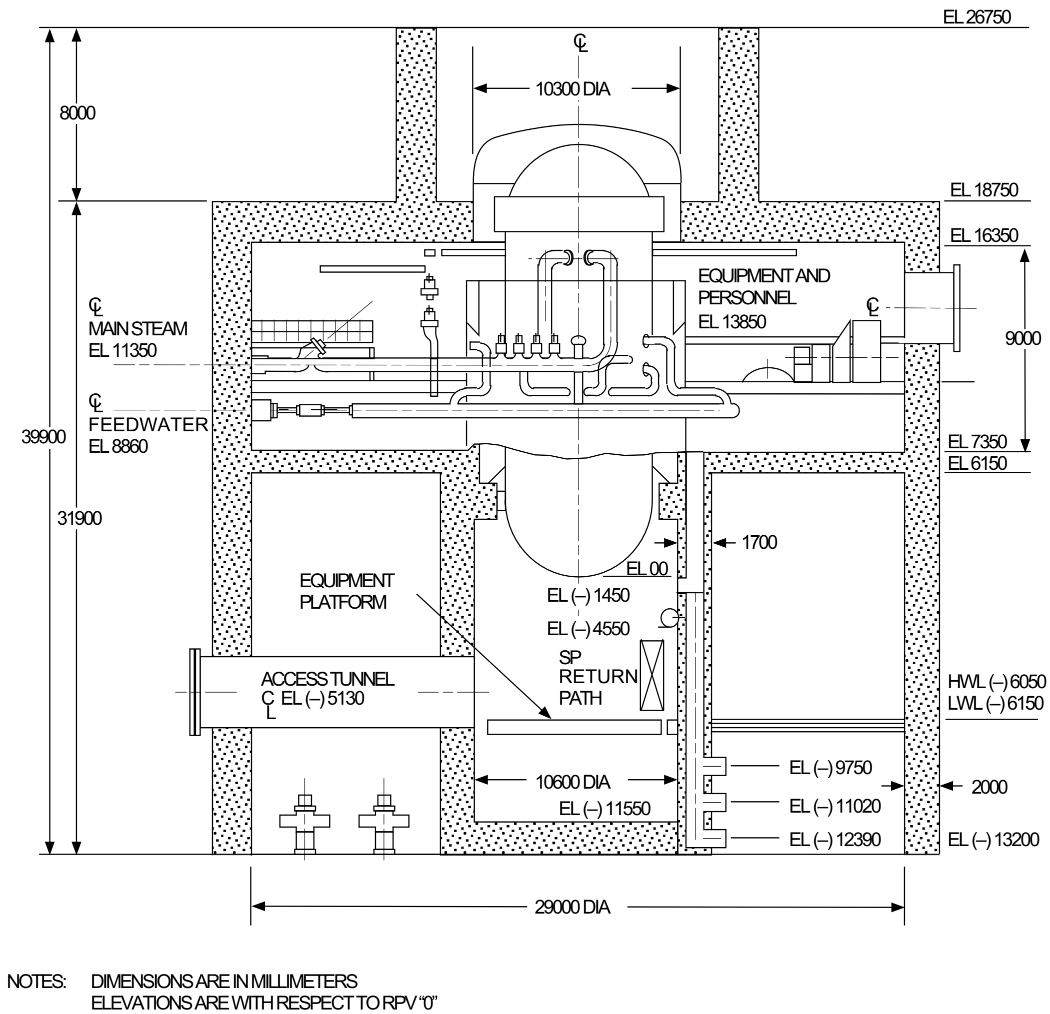
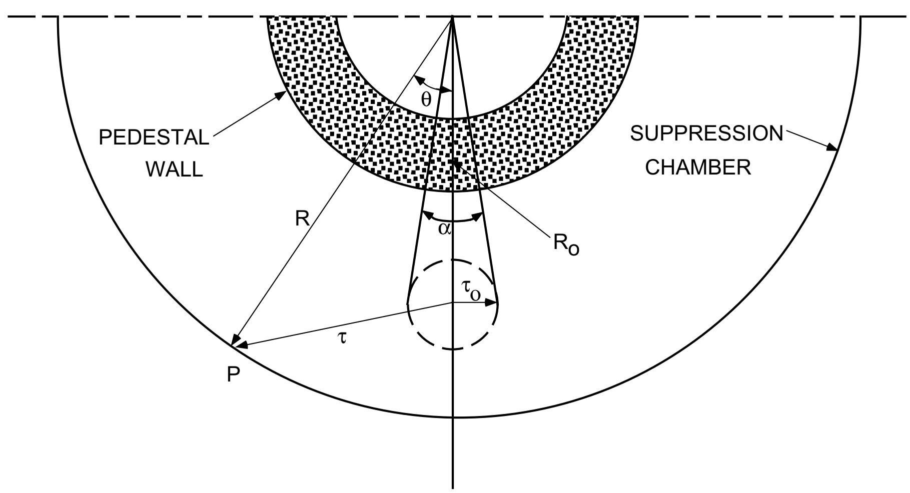


Figure 3B-1 ABWR Primary Containment Configuration

Figure 3B-2 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-3 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]



$\tau = R - R_o$: for $\theta < \alpha/2$

$\tau = \sqrt{R^2 + R_o^2 - 2RR_o \cos\theta}$: for $\theta \geq \alpha/2$

Figure 3B-4 Dimensions for P(r) Calculation

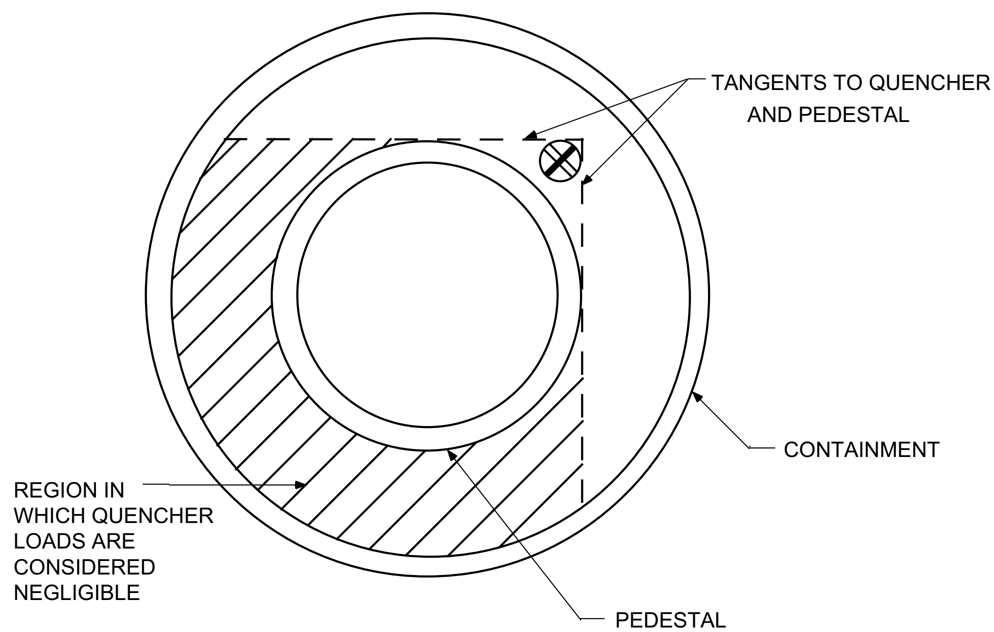


Figure 3B-5 Circumferential Distribution

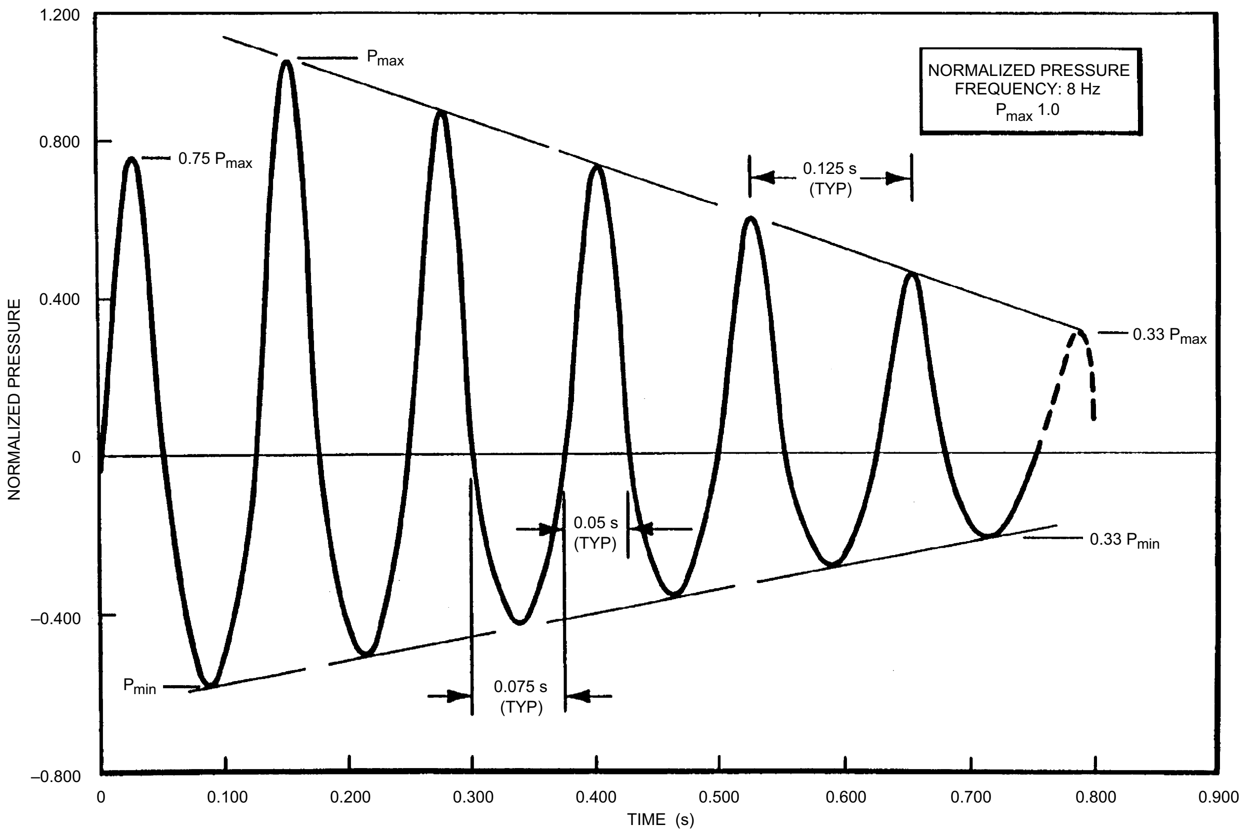


Figure 3B-6 Quencher Bubble Pressure Time History

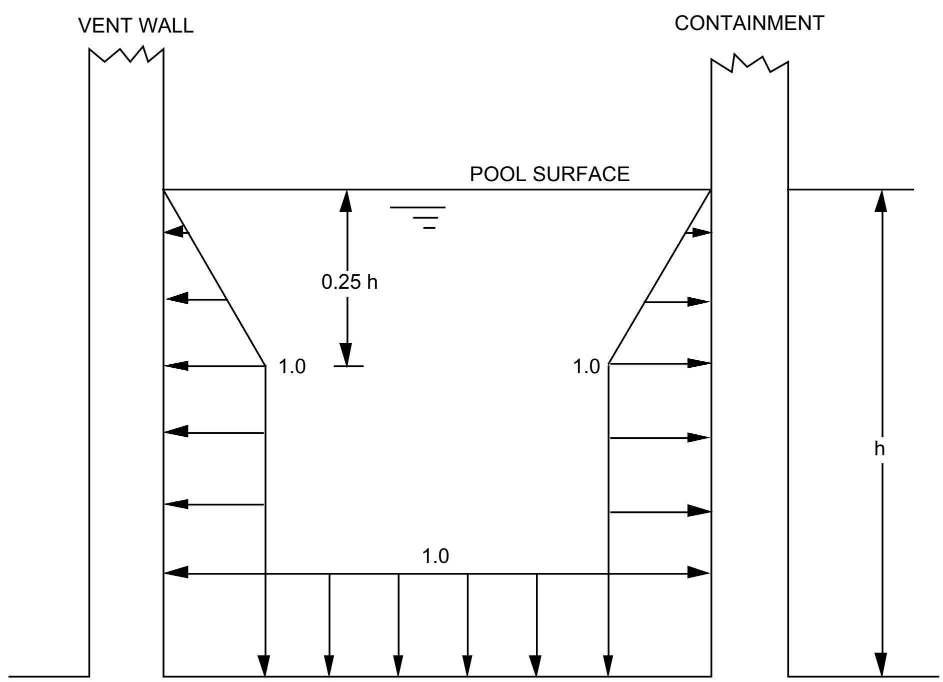


Figure 3B-7 Spatial Load Distribution for SRV Loads

Figure 3B-8 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-9 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-10 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

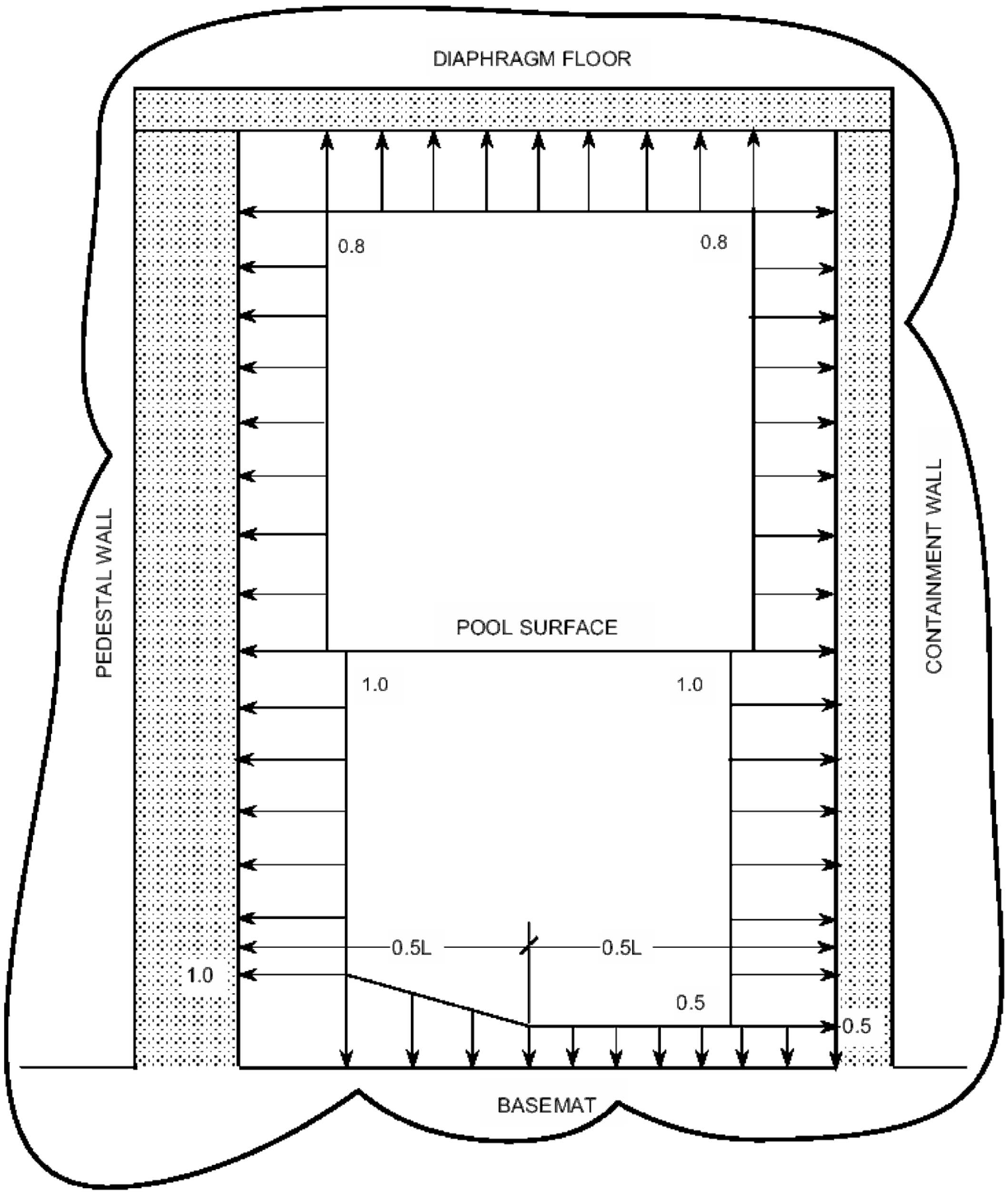


Figure 3B-11 Pool Boundary Pressure During Pool Swell, Normalized to Bubble Pressure

Figure 3B-12 Deleted

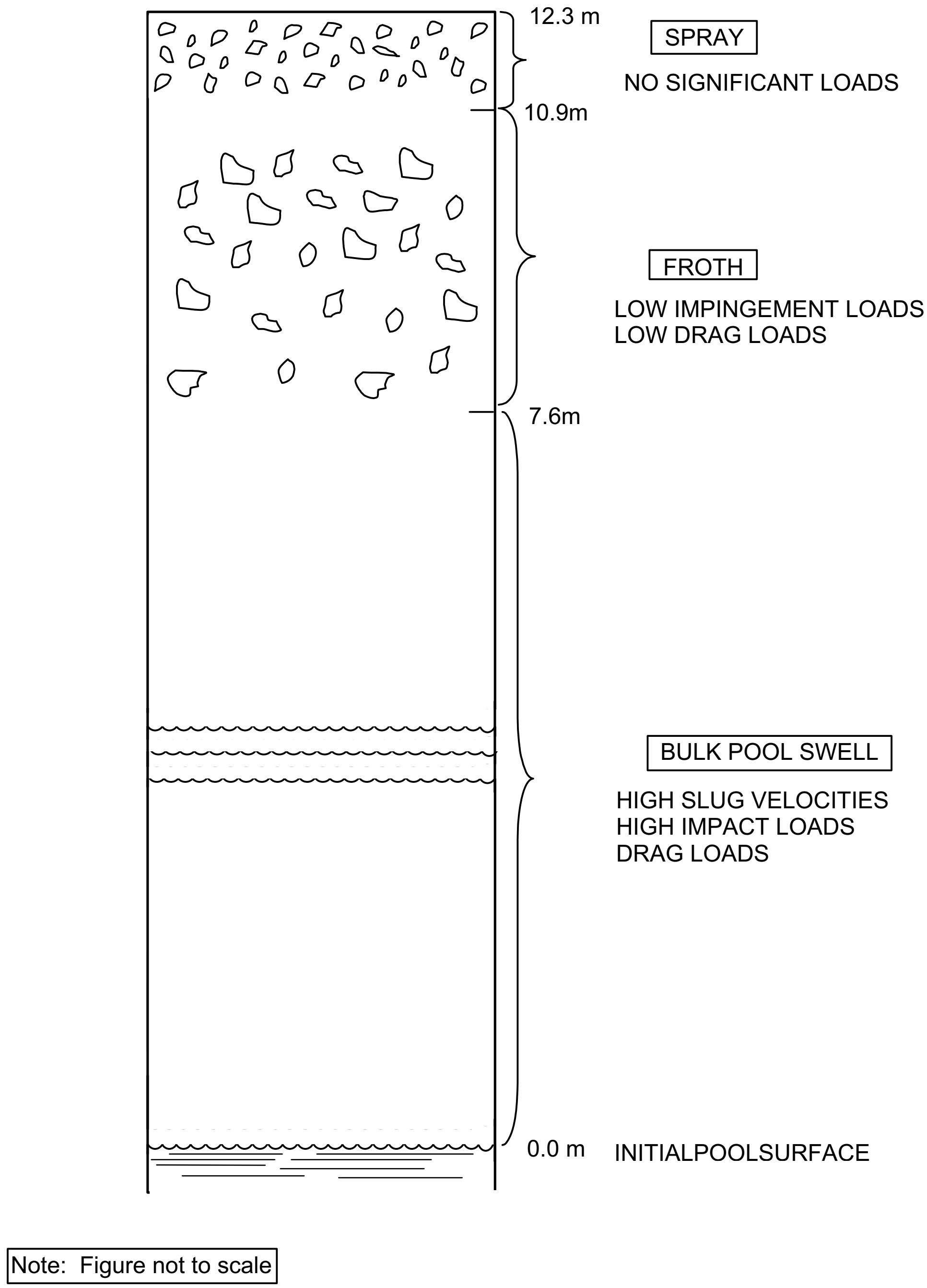


Figure 3B-13 Schematic of the Pool Swell Phenomenon

Figure 3B-14 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-15 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-16 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-17 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-18 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-19 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-20 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

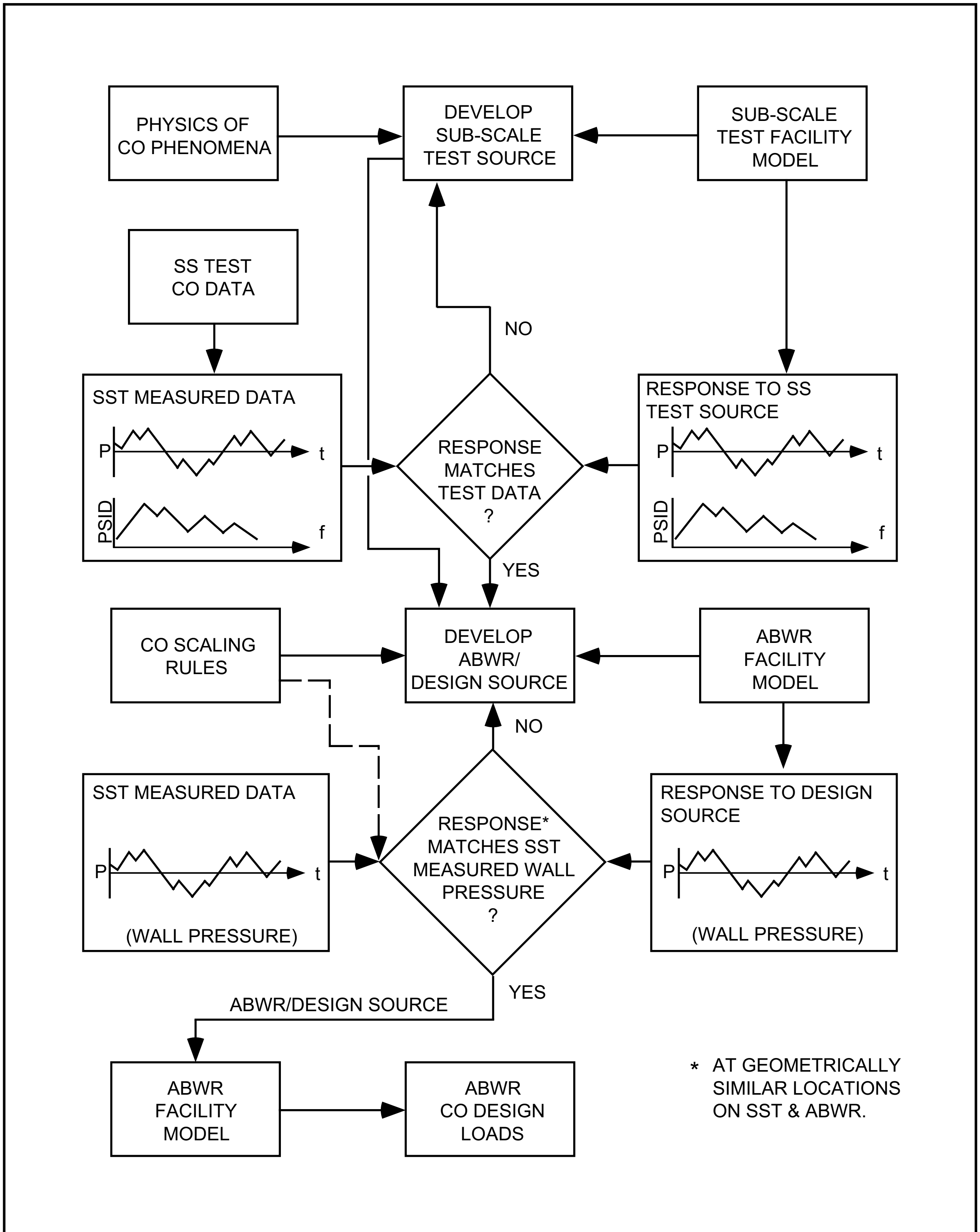


Figure 3B-21 ABWR CO Source Load Methodology

Figure 3B-22 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-23 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-24 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-25 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

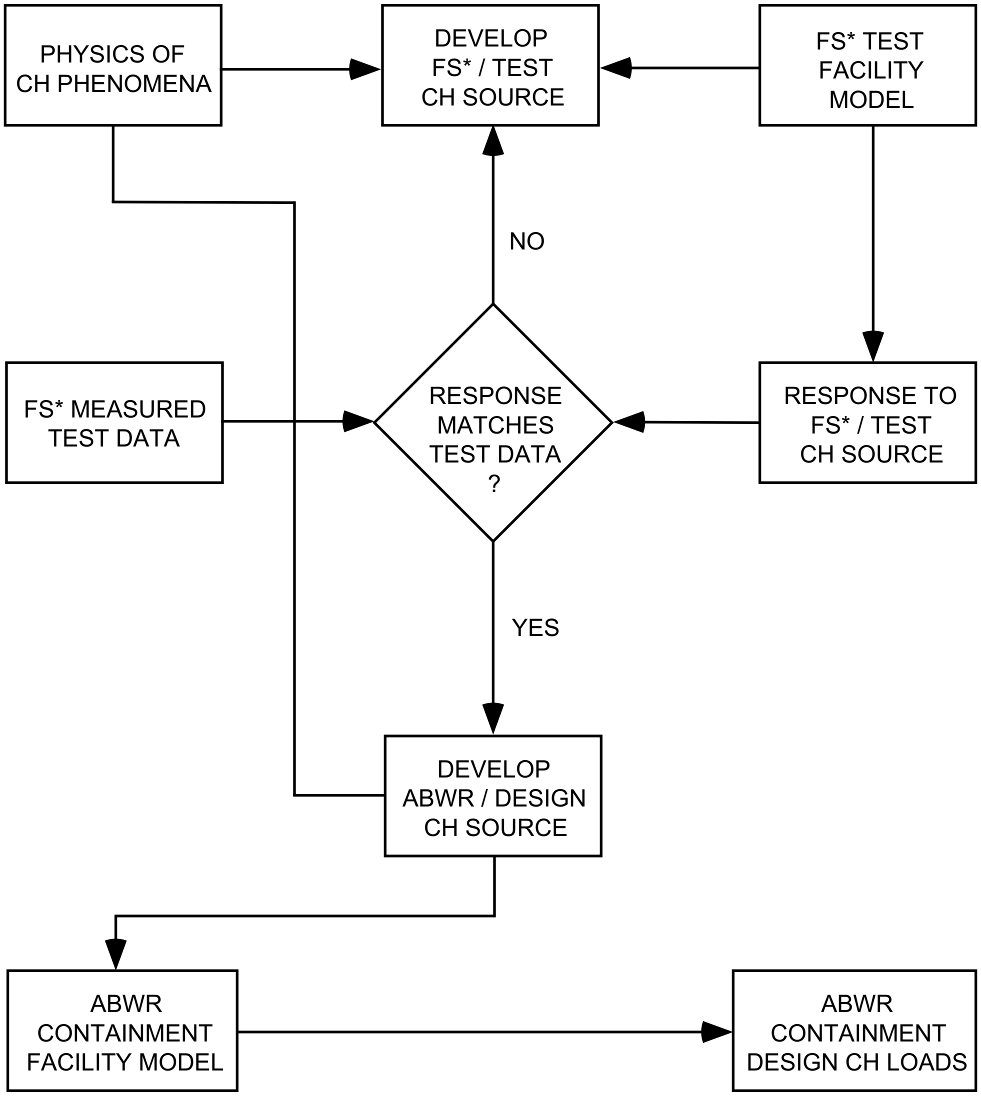


Figure 3B-26 ABWR Chug Source Load Methodology

Figure 3B-27 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

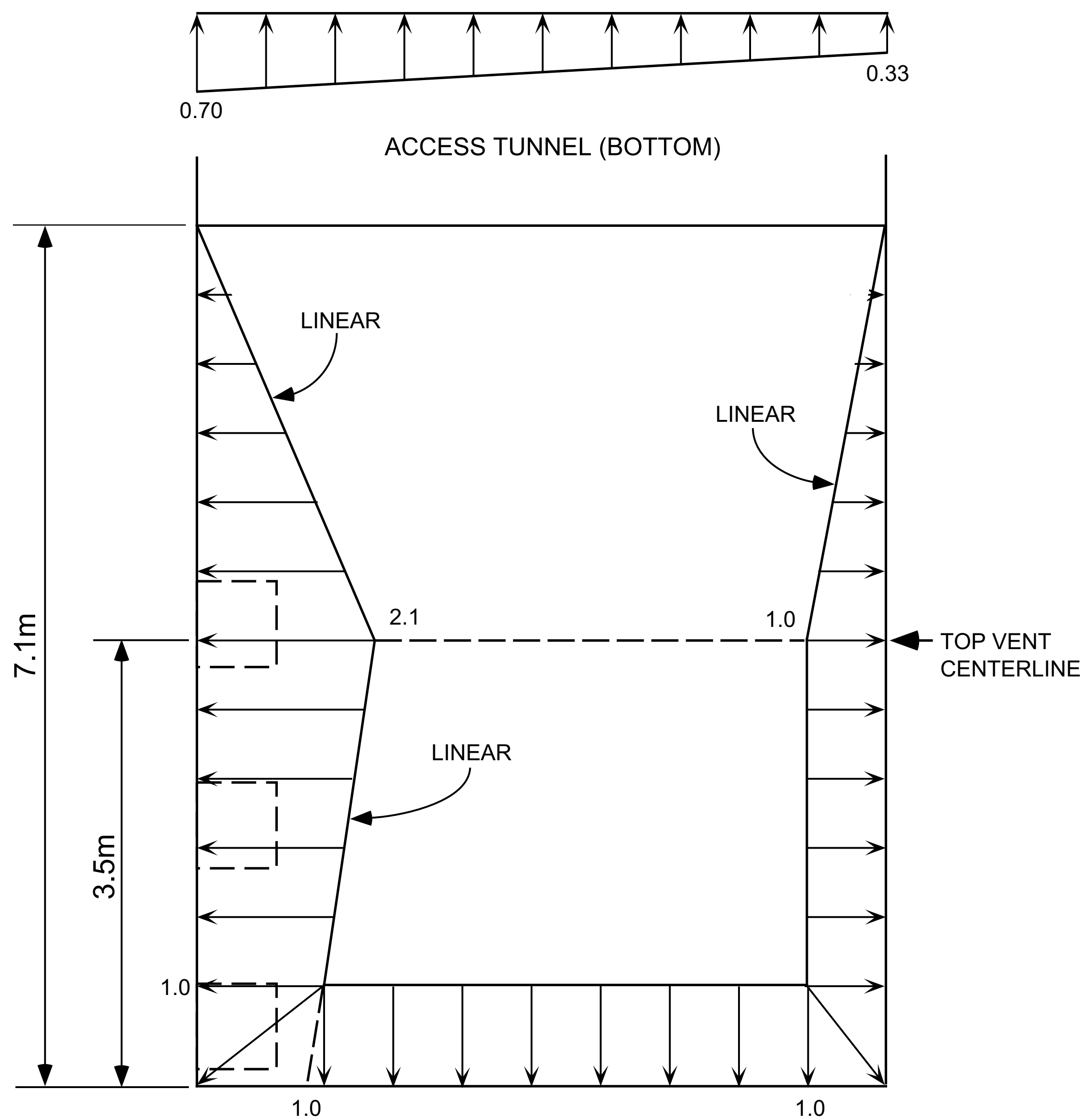


Figure 3B-28 Spatial Load Distribution for CH

Figure 3B-29 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

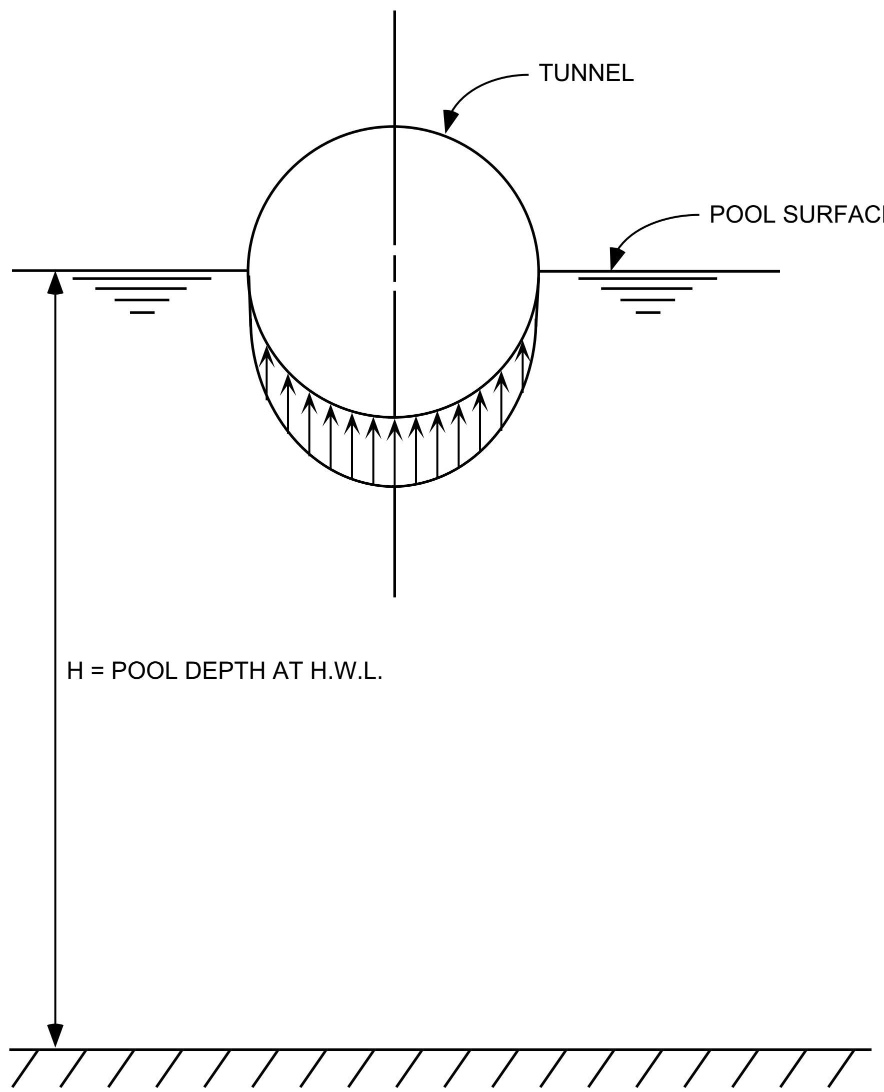


Figure 3B-30 Circumferential Pressure Distribution on Access Tunnel Due to CH

Figure 3B-31 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-32 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-33 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

Figure 3B-34 [Proprietary information not included in DCD (Refer to SSAR Appendix 3B, Amendment 34)]

3C Computer Programs Used in the Design and Analysis of Seismic Category I Structures

3C.1 Introduction

The following Seismic Category I structures and their foundations of the Nuclear Island are analyzed and/or designed using the computer programs described in this appendix:

(1) Reactor Building

(2) Concrete Containment Structure

(3) Control Building

3C.2 Static and Dynamic Structural Analysis Systems (STARDYNE)

3C.2.1 Description

STARDYNE is a large-scale, finite-element program that has a broad range of analysis types and many different structural elements. The STARDYNE system of structural analysis programs is segmented into individual programs. A variety of static or dynamic analyses may be performed by using one or more of the individual programs in a coordinated series of computer runs.

3C.2.2 Validation

STARDYNE is written and maintained by the System Development Corporation of Santa Monica, California, and is available on the Control Data Corporation (CDC) system. Program validation documentation is available at CDC in San Francisco, California.

3C.2.3 Extent of Application

This program is used for the static analysis of the Reactor Building and containment structure.

3C.3 Concrete Element Cracking Analysis Program (CECAP)

3C.3.1 Description

CECAP computes stresses in a concrete element under thermal and/or nonthermal (real) loads, considering effects of concrete cracking. The element represents a section of a concrete shell or slab, and may include two layers of reinforcing, transverse reinforcing, prestressing tendons, and a liner plate.

The program outputs stresses and strains along the element in the concrete, reinforcement, and liner plate, and resultant forces and moments for the composite concrete element.

CECAP assumes linear stress-strain relationships for steel and for concrete in compression. Concrete is assumed to have no tensile strength. The solution is an iterative process, whereby tensile stresses found initially in concrete are relieved (by cracking) and redistributed in the element. The equilibrium of nonthermal loads is preserved. For thermal effects, the element is assumed free to expand inplane, but is fixed against rotation. The capability for expansion and cracking generally results in a reduction in thermal forces and moments from the initial condition.

3C.3.2 Validation

CECAP is written and maintained by Bechtel Power Corporation (BPC) of San Francisco, California. Program validation documentation is available at BPC in San Francisco.

3C.3.3 Extent of Application

This program is used for the analysis of the Reactor Building and containment structure.

3C.4 Finite Element Program for Cracking Analysis (FINEL)

3C.4.1 Description

FINEL is a proprietary computer program of Bechtel Power Corporation, San Francisco, California. The FINEL program performs a static analysis of stresses and strains in plane and axisymmetric structures by the finite element method. The program performs the non-linear static analysis utilizing a stepwise linear iteration solution technique. Within each solution cycle, status of all elements is determined and their stiffness adjusted by the program prior to the next iteration cycle. The Von Mises yield criterion is used to determine the status of all ductile materials and brittle materials which are in compression. A ductile material is assumed to yield in all directions when the yield criterion is exceeded. A brittle material is assumed to be cracked in the direction in which the maximum principal stress exceeds the specified tensile stress. The modulus of elasticity for each material is adjusted for the next solution cycle to conform to the secant modulus corresponding to the calculated strain in the element following the bilinear stress strain relationship specified. The numerical algorithm assumes that the state of stress which exists when converged solution is achieved is independent of the stress history of the loading.

3C.4.2 Validation

The FINEL code has been extensively used in the past to design reinforced concrete containment structures and to predict the strains and deflections during their structural integrity tests (SIT). The correlation between the predicted and observed deflections and strains during such tests has been very satisfactory. FINEL code predictions have been on the conservative side. The FINEL code was validated by comparing results based on FINEL with results based on classical solutions and/or experimental results existing in literature. A further confirmation of its validity is provided by comparison of predicted strains and deformations based on FINEL with experimental results obtained during SIT on various Bechtel designed containments.

3C.4.3 Extent of Application

This program is used for the static load analysis of the reactor building and containment to determine stresses and strains in the various structural elements and resultant forces and moments at selected sections, and is used to evaluate the ultimate strength of the containment.

3D Computer Programs Used in the Design of Components, Equipment and Structures

3D.1 Introduction

As discussed in Subsection 3.9.1.2, this appendix describes the major computer programs used in the analysis of the safety-related components, equipment and structures. The quality of the programs and the computed results is controlled. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems.

Updates to Appendix 3D will be provided to indicate any additional programs used by GE and especially by vendors of components and equipment, or the later version of the described programs, and the method of their verification.

3D.2 Fine Motion Control Rod Drive

3D.2.1 Fine Motion Control Rod Drive—FMCRD01

The FMCRD01 program is used to obtain scram performance data for various inputs to the fine motion control rod drive (FMCRD) stress analysis for both code and non-code parts. The use of this program is addressed in Subsection 3.9.1.3.2. Experimental data on pressure drops, friction factors, effects of fuel channel detection, etc., are used in the setting up and perfecting of this code. Internal drive pressures and temperatures used in the stress analysis are also determined during actual testing of the prototype FMCRD.

3D.2.2 Structural Analysis Programs

Structural analysis programs, such as NASTRO4V and ANSYS, that are mentioned in Subsections 3D.3 and 3D.5 are used in the analysis of the FMCRD.

3D.3 Reactor Pressure Vessel and Internals

The following computer programs are used in the analysis of the reactor pressure vessel, core support structures, and other safety class reactor internals: NASTR04V, SAP4G07, HEATER, FATIGUE, ANSYS, CLAPS, ASSIST, SEISM03 AND SASSI. These programs are described in Subsection 4.1.4.

3D.4 Piping

3D.4.1 Piping Analysis Program—PISYS

PISYS is a computer code for analyzing piping systems subjected to both static and dynamic piping loads. Stiffness matrices representing standard piping components are assembled by the program to form a finite element model of a piping system. The piping elements are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment, and loading of the piping system becomes possible. PISYS is based on the linear elastic analysis in which the resultant deformations, forces, moments and accelerations at each joint are proportional to the loading and the superposition of loading is valid.

PISYS has a full range of static dynamic load analysis options. Static analysis includes dead weight, uniformly-distributed weight, thermal expansion, externally-applied forces, moments, imposed displacements and differential support movement (pseudo-static load case). Dynamic analysis includes mode shape extraction, response spectrum analysis, and time-history analysis by modal combination or direct integration. In the response spectrum analysis [i.e., uniform support motion response spectrum analysis (USMA) or independent support motion response spectrum analysis (ISMA)], the user may request modal response combination in accordance with NRC Regulatory Guide 1.92. In the ground motion (uniform motion) or independent support time-history analysis, the normal mode solution procedure is selected. In analysis involving time-varying nodal loads, the step-by-step direct integration method is used.

The PISYS program has been benchmarked against Nuclear Regulatory Commission piping models. The results are documented in a report to the Commission, “PISYS Analysis of NRC Benchmark Problems”, NEDO-24210, August 1979, for mode shapes and USMA options. The ISMA option has been validated against NUREG/CR-1677, “Piping Benchmark Problems Dynamic Analysis Independent Support Motion Response Spectrum Method,” published in August 1985.

3D.4.2 Component Analysis—ANSI7

ANSI7 is a computer code for calculating stresses and cumulative usage factors for Class 1, 2 and 3 piping components in accordance with articles NB, NC and ND-3650 of ASME Code Section III. ANSI7 is also used to combine loads and calculate combined service level A, B, C and D loads on piping supports and pipe-mounted equipment.

3D.4.3 Area Reinforcement—NOZAR

The Nozzle Area Reinforcement (NOZAR) computer program performs an analysis of the required reinforcement area for openings. The calculations performed by NOZAR are in accordance with the rules of ASME Code Section III, 1974 edition.

3D.4.4 Dynamic Forcing Functions

3D.4.4.1 Relief Valve Discharge Pipe Forces Computer Program—RVFOR

The relief valve discharge pipe connects the pressure-relief valve to the suppression pool. When the valve is opened, the transient fluid flow causes time-dependent forces to develop on the pipe wall. This computer program computes the transient fluid mechanics and the resultant pipe forces using the method of characteristics.

3D.4.4.2 Turbine Stop Valve Closure—TSFOR

The TSFOR program computes the time-history forcing function in the main steam piping due to turbine stop valve closure. The program utilizes the method of characteristics to compute fluid momentum and pressure loads at each change in pipe section or direction.

3D.4.5 Response Spectra Generation

3D.4.5.1 ERSIN Computer Program

ERSIN is a computer code used to generate response spectra for pipe-mounted and floor-mounted equipment. ERSIN provides direct generation of local or global acceleration response spectra.

3D.4.5.2 RINEX Computer Program

RINEX is a computer code used to interpolate and extrapolate amplified response spectra used in the response spectrum method of dynamic analysis. RINEX is also used to generate response spectra with nonconstant model damping. The nonconstant model damping analysis option can calculate spectral acceleration at the discrete eigenvalues of a dynamic system using either the strain energy weighted modal damping or the ASME Code Class N-411-1 damping values.

3D.4.6 Piping Dynamic Analysis Program—PDA

The pipe whip dynamic analysis is performed using the PDA computer program, as described in Appendix 3L. PDA is a computer program used to determine the response of a pipe subjected to the thrust force occurring after a pipe break. It also is used to determine the pipe whip restraint design and capacity.

The program treats the situation in terms of generic pipe break configuration, which involves a straight, uniform pipe fixed (or pinned) 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 relations 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 (or pinned) 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 moment-angular rotation relations, nonlinear equations of motion are formulated using energy considerations, and the equations are numerically integrated in small time steps to yield the time-history of the pipe motion.

3D.4.7 Not Used

3D.4.8 Thermal Transient Program—LION

The LION program is used to compute radial and axial thermal gradients in piping. The program calculates a time-history of ΔT_1, ΔT_2, Ta, and Tb (defined in ASME Code Section III, Subsection NB) for uniform and tapered pipe wall thickness.

3D.4.9 Not Used

3D.4.10 Engineering Analysis System—ANSYS

The ANSYS computer program is a large-scale general-purpose program for the solution of several classes of engineering analysis problems. Analysis capabilities include static and dynamic, plastic, creep and swelling, small and large deflections, and other applications.

This program is used to perform non-linear analysis of piping systems for time varying displacements and forces due to postulated pipe breaks.

3D.5 Pumps and Motors

Following are the computer programs used in the dynamic analysis to assure the structural and functional integrity of the pump and motor assemblies, such as those used in the ABWR ECCS.

3D.5.1 Structural Analysis Program—SAP4G07

SAP4G07 is used to analyze the structural and functional integrity of the pump/motor systems. This program is also identified in Subsections 4.1.4.1.2, 3D.3 and 3D.6. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite-element displacement method is used to solve the displacement and stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plane strain-plane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time-history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacement of each nodal point as well as stresses at the surface of each element.

3D.6 Heat Exchangers

The following computer programs are used in dynamic and static analyses to determine the structural and functional integrity of the heat exchangers, such as those used in the ABWR RHR System.

3D.6.1 Structural Analysis Program—SAP4G07

The structural integrity of the heat exchanger is evaluated using SAP4G07. This program is described in Subsection 3D.5.1.

3D.7 Soil-Structure Interaction

3D.7.1 A System for Analysis of Soil-Structure Interaction—SASSI

This program consists of a number of interrelated computer program modules which can be used to solve a wide range of dynamic soil-structure interaction (SSI) problems in two or three dimensions. This program is used to obtain enveloped seismic design loads based on the finite-element method using substructuring technique, as described in Section 3A.5 of Appendix 3A of this document. A description of this program is included in Subsection 4.1.4.1.9.

The SASSI computer program was developed at the University of California, Berkeley, under the technical direction of Prof. John Lysmer. The Bechtel version of the program was obtained from the University of California, Berkeley. During the course of installation, testing, and validation of the Bechtel version of the program on the IBM System, some modifications and enhancements were made to the program to improve the performance. These include correcting the motion phases in Rayleigh wave calculation, replacing the plate element, modifying the spring element to include damping capability, and providing the option for local end release condition in beam element. The program was verified against benchmark results reported by various investigators in the technical literature.

3D.7.2 Not Used

3D.7.3 Free-Field Response Analysis—SHAKE

This program is used to perform the free-field site response analysis required in the seismic SSI analysis (Subsection 3A.6).

SHAKE is a computer program developed at the University of California, Berkeley, by Schnable, Lysmer and Seed (Reference 3A-5 of Subsection 3A.11). The program uses the principle of one-dimensional propagation of shear waves in the vertical direction for a system of horizontal, visco-elastic soil layers to compute soil responses in the free-field. The nonlinearities in soil shear modulus and damping are accounted for by the use of equivalent linear soil properties using an iterative procedure to obtain values for modulus and damping compatible with the effective shear strains in each layer. The final iterated, strain-compatible properties are used as equivalent linear soil properties in seismic SSI analysis.

3E Guidelines for LBB Application

3E.1 Introduction

As discussed in Subsection 3.6.3, this appendix provides detailed guidelines for the COL applicant's use in applying for NRC's approval of leak-before-break (LBB) for specific piping systems. Also included in this appendix are the fracture mechanics properties of ABWR piping materials and analysis methods, including the leak rate calculation methods. Table 3E-1 gives a list of piping systems inside and outside the containment that are preliminary candidates for LBB application. As noted on Table 3E-1, most candidate piping systems are carbon steel piping. Therefore, this appendix deals extensively with the evaluation of carbon steel piping.

Piping qualified by LBB would be excluded from the non-mechanistic postulation requirements of double-ended guillotine break (DEGB) specified in Subsection 3.6.3. The LBB qualification means that the throughwall flaw lengths that are detectable by leakage monitoring systems (Subsection 5.2.5) are significantly smaller than the flaw lengths that could lead to pipe rupture or instability.

Section 3E.2 addresses the fracture mechanics properties aspects required for evaluation in accordance with Subsection 3.6.3. Section 3E.3 describes the fracture mechanics techniques and methods for the determination of critical flaw lengths and evaluation of flaw stability. Explained in Section 3E.4 is the determination of flaw lengths for detectable leakages with margin. A brief discussion on the leak detection capabilities is presented in Section 3E.5. Finally, Section 3E.6 provides general guidelines for the preparation of LBB justification reports by providing two examples.

Material selection and the deterministic LBB evaluation procedure are discussed in this section.

3E.1.1 Material Selection Guidelines

The LBB approach is applicable to piping systems for which the materials meet the following criteria: (1) low probability of failure from the effects of corrosion (e.g., intergrannular stress corrosion cracking), and (2) adequate margin before susceptibility to cleavage type fracture over the full range of consequences.

The ABWR plant design specifies use of austenitic stainless steel piping made of material (e.g., nuclear grade or low carbon type) that is recognized as resistant to IGSCC. The carbon steel or ferritic steels specified for the reactor pressure boundary are described in Subsection 3E.2.2. These steels are assured to have adequate toughness to preclude a fracture at operating temperatures. A COL applicant is expected to supply a detailed justification in the LBB evaluation report considering system temperature, fluid velocity and environmental conditions.

3E.1.2 Deterministic Evaluation Procedure

The following deterministic analyses and evaluations are performed as an NRC-approved method to justify applicability of the LBB concept.

(1) Use the fracture mechanics and the leak rate computational methods that are accepted by the NRC staff, or are demonstrated accurate with respect to other acceptable computational procedures or with experimental data.

(2) Identify the types of materials and materials specifications used for base metal, weldments and safe ends, and provide the materials properties including toughness and tensile data, long-term effects such as thermal aging, and other limitations.

(3) Specify the type and magnitude of the loads applied (forces, bending and torsional moments), their source(s) and method of combination. For each pipe size in the functional system, identify the location(s) which have the least favorable combination of stress and material properties for base metal, weldments and safe ends.

(4) Postulate a throughwall flaw at the location(s) specified in (3) above. The size of the flaw should be large enough so that the leakage is assured detection with sufficient margin using the installed leak detection capability when pipes are subjected to normal operating loads. If auxiliary leak detection systems are relied on, they should be described. For the estimation of leakage, the normal operating loads (i.e., deadweight, thermal expansion, and pressure) are to be combined based on the algebraic sum of individual values.

Using fracture mechanics stability analysis or limit load analysis based on (11) below, and normal plus SSE loads, determine the critical crack size for the postulated throughwall crack. Determine crack size margin by comparing the selected leakage size crack to the critical crack size. Demonstrate that there is a margin of 2 between the leakage and critical crack sizes. The same load combination method selected in (5) below is used to determine the critical crack size.

(5) Determine margin in terms of applied loads by a crack stability analysis. Demonstrate that the leakage size cracks will not experience unstable crack growth if 1.4 times the normal plus SSE loads are applied. Demonstrate that crack growth is stable and the final crack is limited such that a double-ended pipe break will not occur. The dead-weight, thermal expansion, pressure, SSE (inertial), and seismic anchor motion (SAM) loads are combined based on the same method used for the primary stress evaluation by the ASME Code. The SSE (inertial) and SAM loads are combined by square-root-of-the-sum-of-the-squares (SRSS) method.

(6) The piping material toughness (J-R curves) and tensile (stress-strain curves) properties are determined at temperatures near the upper range of normal plant operation.

(7) The specimen used to generate J-R curves is assured large enough to provide crack extensions up to an amount consistent with J/T condition determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques are used as described in NUREG-1601, Volume 3, or in NUREG/CR-4575. Other techniques can be used if adequately justified.

(8) The stress-strain curves are obtained over the range from the preoperational limit to maximum load.

(9) Preferably, the materials tests should be conducted using archival materials for the pipe being evaluated. If archival material is not available, plant specific or industry wide generic material databases are assembled and used to define the required material tensile and toughness properties. Test material includes base and weld metals.

(10) To provide an acceptable level of reliability, generic databases are reasonable lower bounds for compatible sets of material tensile and toughness properties associated with materials at the plant. To assure that the plant-specific generic database is adequate, a determination is made to demonstrate that the generic database represents the range of plant materials to be evaluated. This determination is based on a comparison of the plant material properties identified in (2) above with those of the materials used to develop the generic database. The number of material heats and weld procedures tested is adequate to cover the strength and toughness range of the actual plant materials. Reasonable lower bound tensile and toughness properties from the plant-specific generic database are to be used for the stability analysis of individual materials, unless otherwise justified.

Industry generic databases are reviewed to provide a reasonable lower bound for the population of material tensile and toughness properties associated with any individual specification (e.g., A106, Grade B), material type (e.g., austenitic steel) or welding procedures.

The number of material heats and weld procedures tested should be adequate to cover the range of the strength and tensile properties expected for specific material specifications or types. Reasonable lower bound tensile and toughness properties from the industry generic database are used for the stability analysis of individual materials.

If the data are being developed from an archival heat of material, three stress-strain curves and three J-resistance curves from that one heat of material are sufficient. The tests should be conducted at temperatures near the upper range of normal plant operation. Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. The lower toughness should be used in the fracture mechanics evaluation. One J-R curve and one stress-strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

(11) There are certain limitations that currently preclude generic use of limit load analyses to evaluate LBB conditions deterministically. However, a modified limit-load analysis can be used for austenitic stainless steel piping to demonstrate acceptable margins as described in Subsection 3E.3.3.

3E.2 Material Fracture Toughness Characterization

This subsection describes the fracture toughness properties and flow stress evaluation for the ferritic and austenitic steel materials used in ABWR plant piping, as required for evaluation according to Subsection 3E.1.2.

3E.2.1 Fracture Toughness Characterization

When the elastic-plastic fracture mechanics (EPFM) methodology or the J-T methodology is used to evaluate the LBB conditions with postulated throughwall flaws, the material toughness property is characterized in the form of J-integral resistance curve (or J-R curve) (References 3E-1, 3E-2, 3E-3). The J-R curve, schematically shown in Figure 3E-1, represents the material's resistance to crack extension. The onset of crack extension is assumed to occur at a critical value of J. Where the plane strain conditions are satisfied, initiation J is denoted by J_{IC}. Plane strain crack conditions, achieved in test specimen by side grooving, generally provide a lower bound behavior for material resistance to stable crack growth.

Once the crack begins to extend, the increase of J with crack growth is measured in terms of slope or the nondimensional tearing modulus, T, expressed as:

$$T = \frac{E}{\sigma_f^2} \cdot \frac{dJ}{da} \qquad (3E\text{-}1)$$

The flow stress, σ_f, is a function of the yield and ultimate strength, and E is the elastic modulus. Generally, σ_f is assumed as the average of the yield and ultimate strength. The slope dJ/da of the material J-R curve is a function of crack extension a. Generally, dJ/da decreases with crack extension, thereby giving a convex upward appearance to the material J-R curve in Figure 3E-1.

To evaluate the stability of crack growth, it is convenient to represent the material J-R curve in the J-T space as shown in Figure 3E-2. The resulting curve is labeled as J-T material. Crack instability is predicted at the intersection point of the J/T material and J/T applied curves.

The crack growth invariably involves some elastic unloading and distinctly nonproportional plastic deformation near the crack tip. J-integral is based on the deformation theory of plasticity (References 3E-4, 3E-5) which inadequately models both of these aspects of plastic behavior. In order to use J-integral to characterize crack growth (i.e., to assure J-controlled crack growth), the following sufficiency condition in terms of a nondimensional parameter proposed by Hutchinson and Paris (Reference 3E-6), is used:

$$\omega = \frac{b}{j} \cdot \frac{dJ}{da} >> 1 \qquad (3E\text{-}2)$$

where b is the remaining ligament. Reference 3E-7 suggests that $\omega > 10$ would satisfy the J-controlled growth requirements. However, if the requirements of this criteria are strictly followed, the amount of crack growth allowed would be very small in most test specimen geometries. Use of such a material J-R curve in J/T evaluation would result in grossly underpredicting the instability loads for large diameter pipes where considerable stable crack growth is expected to occur before reaching the instability point. To overcome this difficulty, Ernst (Reference 3E-8) proposed a modified J-integral, J_{mod}, which was shown to be effective even when limits on ω were grossly violated. The Ernst correction essentially factors-in the effect of crack extension in the calculated value of J. This correction can be determined experimentally by measuring the usual parameters: load, displacement and crack length.

The definition of J_{mod} is:

$$J_{mod} = J - \int_{a_0}^{a} \left| \frac{\partial(J - G)}{\partial_a} \right|_{\delta_{pl}} da \qquad (3E\text{-}3)$$

where:

J	Is based on deformation theory of plasticity
G	Linear elastic Griffith energy release rate or elastic J, J_{el}
δ_{pl}	Nonlinear part of the load-point displacement, (or simply the total minus the elastic displacement)
a_0, a	Initial and current crack lengths, respectively

For the particular case of the compact tension specimen geometry, the preceding equation and the corresponding rate take the form:

$$J_{mod} = J + \int_{a_0}^{a} \gamma \cdot \frac{J_{pl}}{b} \cdot da \qquad (3E\text{-}4)$$

where J_{pl} is the nonlinear part of the deformation theory J, and b is the remaining ligament and is

$$= (1 + 0.76b/W) \qquad (3E\text{-}5)$$

Consequently, the modified material tearing modulus T_{mod} can be defined as:

$$T_{mod} = T_{mat} + \frac{E}{\sigma_f^2}\left(\frac{\gamma}{b} \cdot J_{pl}\right) \qquad (3E\text{-}6)$$

Since in most of the test J-R curves the $\omega > 10$ limit was violated, all of the material J-T data were recalculated in the J_{mod}, T_{mod} format. The J_{mod}, T_{mod} calculations were performed up to crack extension of $a = 10\%$ of the original ligament in the test specimen. The J-T curves were then extrapolated to larger J values using the method recommended in NUREG 1061, Vol. 3 (Reference 3E-9).

The J_{mod}-T_{mod} approach is used in this appendix for illustrative purposes. It should be adopted if justified based on its acceptability by the technical literature. A J_D- approach is another more justifiable approach.

3E.2.2 Carbon Steels and Associated Welds

The carbon steels used in the ABWR reactor coolant pressure boundary (RCPB) piping are: SA 106 Grade B, SA 333 Grade 6 and SA 672, Grade C70. The first specification covers seamless pipe and the second one pertains to both seamless and seam-welded pipe. The last one pertains to seam-welded pipe for which plate stock is specified as SA 516, Grade 70. The corresponding material specifications used for carbon steel flanges, fittings and forgings are equivalent to the piping specifications.

While the chemical composition requirements for a pipe per SA 106 Grade B and SA 333 Grade 6 are identical, the latter is subjected to two additional requirements: (1) a normalizing heat treatment which refines the grain structure and, (2) a Charpy test at –46.6°C with a specified minimum absorbed energy of 85.5 N·m. The electrodes and filler metal requirements for welding carbon steel to carbon or low alloy steel are as specified in Table 3E-2.

A comprehensive test program was undertaken at GE to characterize the carbon steel base and weld material toughness properties. The next section describes the scope and results of this program. The purpose of the test program was to generate the necessary data for application in Section 3E.6 and to illustrate a general procedure of conducting the tests per requirements of

Item (10) in Subsection 3E.1.2. The extent of the test program for the NRC's approval of an application will depend upon the identified requirements.

3E.2.2.1 Fracture Toughness Test Program

The test program consisted of generating true-stress/true-strain curves, J-Resistance curves and the Charpy V-notch tests. Two materials were selected: (1) SA333 Grade 6, 400A diameter, Schedule 80 pipe and (2) SA516, Grade 70, 2.54 cm thick plate. Table 3E-3 shows the chemical composition and mechanical property test information provided by the material supplier. The materials were purchased to the same specifications as those to be used in the ABWR applications.

To produce a circumferential butt weld, the pipe was cut in two pieces along a circumferential plane and welded back using the shielded metal arc process. The weld prep was of single V design with a backing ring. The preheat temperature was 93.3°C.

The plate material was cut along the longitudinal axis and welded back using the SAW process. The weld prep was of a single V type with one side as vertical and the other side at 45 degrees. A backing plate was used during the welding with a clearance of 0.64 cm at the bottom of the V. The interpass temperature was maintained at less than 260°C.

Both the plate and the pipe welds were X-rayed according to Code (Reference 3E-11) requirements and were found to be satisfactory.

It is well-known that carbon steel base materials show considerable anisotropy in fracture toughness properties. The toughness depends on the orientation and direction of propagation of the crack in relation to the principal direction of mechanical working or gain flow. Thus, the selection of proper orientation of charpy and J-R curve test specimen is important. Figure 3E-2 shows the orientation code for rolled plate and pipe specimen as given in ASTM Standard E399 (Reference 3E-12). Since a throughwall circumferential crack configuration is of most interest from the DEGB point of view, the L-T specimen in a plate and the L-C specimen in a pipe provide the appropriate toughness properties for that case. On the other hand, T-L and C-L specimen are appropriate for the axial flaw case.

Charpy test data are reviewed first, since they provide a qualitative measure of the fracture toughness.

3E.2.2.1.1 Charpy Tests

The absorbed energy or its complement, the lateral expansion measured during a Charpy V-notch test, provides a qualitative measure of the material toughness. For example, in the case of austenitic stainless steel flux weldments, the observed lower Charpy energy relative to the base metal was consistent with the similar trend observed in the J-Resistance curves. The Charpy tests in this program were used as preliminary indicators of relative toughness of welds, HAZs and the base metal.

The carbon steel base materials exhibit considerable anisotropy in the Charpy energy as illustrated by Figure 3E-3 from Reference 3E-13. This anisotropy is associated with development of grain flow due to mechanical working. The Charpy orientation C in Figure 3E-3 (orientations LC and LT in Figure 3E-2) is the appropriate one for evaluating the fracture resistance to the extension of a throughwall circumferential flaw. The upper shelf Charpy energy associated with axial flaw extension (orientation A in Figure 3E-3) is considerably lower than that for the circumferential crack extension.

A similar trend in the base metal charpy energies was also noted in this test program. Figures 3E-4 and 3E-5 show the pipe and plate material Charpy energies for the two orientations as a function of temperature. The tests were conducted at six temperatures ranging from room temperature to 288°C. From the trend of the Charpy energies as a function of temperature in Figures 3E-4 and 3E-5, it is clear that even at room temperature the upper shelf conditions have been reached for both the materials.

No such anisotropy is expected in the weld metal, since it does not undergo any mechanical working after its deposition. This conclusion is also supported by the available data in the technical literature. The weld metal Charpy specimens in this test program were oriented the same way as the LC or LT orientations in Figure 3E-2. The HAZ Charpy specimens were also oriented similarly.

Figure 3E-6 shows a comparison of the Charpy energies from the SA 333, Grade 6 base metal, the weld metal and the HAZ. In most cases, two specimens were used. Considerable scatter in the weld and HAZ Charpy energy values is seen. Nevertheless, the average energies for the weld metal and the HAZ seem to fall at or above the average base metal values. This indicates that, unlike the stainless steel flux weldments, the fracture toughness of carbon steel weld and HAZ, as measured by the Charpy tests, is at least equal to the carbon steel base metal.

The preceding results and the results of the stress-strain tests discussed in the next section or other similar data are used as a basis to choose between the base and the weld metal properties for use in the J-T methodology evaluation.

3E.2.2.1.2 Stress-Strain Tests

The stress-strain tests were performed at three temperatures: room temperature, 177°C and 288°C. Base and weld metal from both the pipe and the plate were tested. The weld specimens were in the as-welded condition. The standard test data obtained from these tests are summarized in Table 3E-4.

An examination of Table 3E-4 shows that the measured yield strength of the weld metal, as expected, is considerably higher than that of the base metal. For example, the 288°C yield strength of the weld metal in Table 3E-4 ranges from 365.4 MPa to 407.8 M Pa, whereas the base metal yield strength is only 234.5 MPa. The impact of this observation in the selection of appropriate material (J/T) curve is discussed in later sections.

Figures 3E-7 through 3E-10 show the plots of the 288°C and 177°C stress-strain curves for both the pipe and the plate used in the test. As expected, the weld metal stress-strain curve in every case is higher than the corresponding base metal curve. The Ramberg-Osgood format characterization of these stress-strain curves is given in Section 3E.3.2.

3E.2.2.1.3 J-R Curve Tests

The test temperatures selected for the J-R curve tests were: room temperature, 177°C and 288°C. Both the weld and the base metal were included. Due to the curvature, only the 1T plan compact tension (CT) specimens were obtained from the 0.41m diameter test pipe. Both 1T and 2T plan test specimens were prepared from the test plate. All of the CT specimens were side-grooved to produce plane strain conditions.

Table 3E-5 shows some details of the J-R curve tests performed in this test program. The J-R curve in the LC orientation of the pipe base metal and in the LT orientation of the plate base metal represent the material's resistance to crack extension in the circumferential direction. Thus, the test results of these orientations were used in the LBB evaluations. The orientation effects are not present in the weld metal. As an example of the J-R curve obtained in the test program, Figure 3E-11 shows the plot of J-R curve obtained from specimen OWLC-A.

3E.2.2.2 Material (J/T) Curve Selection

The normal operating temperatures for most of the carbon steel piping in the reactor coolant pressure boundary in the ABWR generally fall into two categories: 288°C and 216°C. The latter temperature corresponds to the operating temperature of the feedwater piping system. The selections of the appropriate material (J/T) curves for these two categories are discussed next.

3E.2.2.2.1 Material J/T Curve for 288°C

A review shows that five tests were conducted at 288°C. Two tests were on the weld metal, two were on the base metal and one was on the heat-affected zone (HAZ). Figure 3E-12 shows the plot of material J_{mod}, T_{mod} values calculated from the J-Δ a values obtained from the 288°C tests. The value of flow stress, σ_f, used in the tearing modulus calculation (Equation 3E-1) was 358.5 MPa based on data shown in Table 3E-4. To convert the deformation J and dJ/da values obtained from the J-R into J_{mod}, T_{mod}, Equations 3E-4 and 3E-6 were used. Only the data from the pipe weld (Specimen ID OWLC-A) and the plate base metal (Specimen ID BMLI-12) are shown in Figure 3E-12. A few unreliable data points were obtained in the pipe base metal (Specimen ID OBLC-2) J-R curve test due to a malfunction in the instrumentation. Therefore, the data from this test were not included in the evaluation. The J-R curves from the other two 288°C tests were evaluated as described in the next paragraph. For comparison purposes, Figure 3E-12 also shows the SA106 carbon steel J-T data obtained from the J-R curve reported by Gudas (Reference 3E-14). The curve also includes extrapolation to higher J values based on the method recommended in NUREG-1061, Vol. 3 (Reference 3E-9).

The J_{mod}-T_{mod} data for the plate weld metal and the plate HAZ were evaluated. A comparison shows that these data fall slightly below those for the plate base metal shown in Figure 3E-12. On the other hand, as noted in Subsection 3E.2.2.1.2, the yield strength of the weld metal and the HAZ is considerably higher than that of the base metal. The material stress-strain and J-T curves are the two key inputs in determining the instability load and flaw values by the (J/T) methodology. Calculations performed for representative throughwall flaw sizes showed that the higher yield strength of the weld metal more than compensates for the slightly lower J-R curve and, consequently, the instability load and flaw predictions based on base metal properties are smaller (i.e., conservative). Accordingly, it was concluded that the material (J-T) curve shown in Figure 3E-12 is the appropriate one to use in the LBB evaluations for carbon steel piping at 288°C.

3E.2.2.2.2 Material J/T Curve for 216°C

Since the test temperature of 177°C can be considered reasonably close to the 216°C, the test J-R curves for 177°C were used in this case. A review of the test matrix in Table 3E-5 shows that three tests were conducted at 177°C. The J_{mod}, T_{mod} data for all three tests were reviewed. The flow stress value used in the tearing modulus calculation was 372.4 MPa based on Table 3E-4. Also reviewed were the data on SA106 carbon steel at 149°C reported by Gudas (Reference 3E-14).

Consistent with the trend of the 288°C data, the 177°C weld metal (J-T) data fell below the plate and pipe base metal data. This probably reflects the slightly lower toughness of the SAW weld in the plate. The (J/T) data for the pipe base metal fell between the plate base metal and the plate weld metal. Based on the considerations similar to those presented in the previous section, the pipe base metal J-T data, although they may lie above the weld J-T data, were used for selecting the appropriate (J-T) curve. Accordingly, the curve shown in Figure 3E-13 was developed for using the (J-T) methodology in evaluations at 216°C.

3E.2.3 Stainless Steels and Associated Welds

The stainless steels used in the ABWR RCPB piping are either Nuclear Grade or low carbon Type 304 or 316. These materials and the associated welds are highly ductile and, therefore, undergo considerable plastic deformation before failure can occur. Toughness properties of Type 304 and 316 stainless steels have been extensively reported in the open technical literature and, thus, are not discussed in detail in this section. Due to high ductility and toughness, modified limit load methods can be used to determine critical crack lengths and instability loads (Subsection 3E.3.3).

3E.3 Fracture Mechanics Methods

This subsection deals with the fracture mechanics techniques and methods for the determination of critical flaw lengths and instability loads for materials used in ABWR. These techniques and methods comply with Criteria (5) through (11) described in Subsection 3E.1.2.

3E.3.1 Elastic-Plastic Fracture Mechanics or (J/T) Methodology

Failure in ductile materials such as highly tough ferritic materials is characterized by considerable plastic deformation and significant amount of stable crack growth. The EPFM approach outlined in this subsection considers these aspects. Two key concepts in this approach are: (1) J-integral (References 3E-15, 3E-16) which characterizes the intensity of the plastic stress-strain field surrounding the crack tip and (2) the tearing instability theory (References 3E-17, 3E-18), which examines the stability of ductile crack growth. A key advantage of this approach is that the material fracture toughness characteristic is explicitly factored into the evaluation.

3E.3.1.1 Basic (J/T) Methodology

Figure 3E-14 schematically illustrates the J/T methodology for stability evaluation. The material (J/T) curve in Figure 3E-14 represents the material's resistance to ductile crack extension. Any value of J falling on the material R-curve is denoted as J_{mat} and is a function solely of the increase in crack length ýa. Also defined in Figure 3E-14 is the 'applied' J, which, for given stress-strain properties and overall component geometry, is a function of the applied load P and the current crack length, a, Hutchinson and Paris (Reference 3E-18) also define the following two nondimensional parameters:

$$\begin{aligned} T_{applied} &= \frac{E}{\sigma_f^2} \cdot \frac{\partial J_{applied}}{\partial a} \\ T_{mat} &= \frac{E}{\sigma_f^2} \cdot \frac{dJ_{mat}}{da} \end{aligned} \qquad (3E\text{-}7)$$

where E is Young's modulus and σ_f is an appropriate flow stress.

Intersection point of the material and applied (J/T) curves denotes the instability point. This is mathematically stated as follows:

$$J_{applied}(a, P) = J_{mat}(a) \qquad (3E\text{-}8)$$

$$\begin{aligned} T_{applied} &< T_{mat}(\text{stable}) \\ T_{applied} &> T_{mat}(\text{unstable}) \end{aligned} \qquad (3E\text{-}9)$$

The load at instability is determined from the J versus load plot also shown schematically in Figure 3E-14. Thus, the three key curves in the tearing stability evaluation are: $J_{applied}$ versus $T_{applied}$, J_{mat} versus T_{mat} and $J_{applied}$ versus load. The determination of appropriate J_{mat} versus T_{mat} or the material (J/T) curve has been already discussed in Subsection 3E.2.1. The $J_{applied} – T_{applied}$ or the (J/T) applied curve can be easily generated through perturbation in the crack length once the $J_{applied}$ versus load information is available for different crack lengths. Therefore, only the methodology for the generation of $J_{applied}$ versus load information is discussed in detail.

3E.3.1.2 J Estimation Scheme Procedure

The $J_{applied}$ or J as a function of load was calculated using the GE/EPRI estimation scheme procedure (References 3E-19, 3E-20). The J in this scheme is obtained as sum of the elastic and fully plastic contributions:

$$J = J_e + J_p \tag{3E-10}$$

The material true stress-strain curve in the estimation scheme is assumed to be in the Ramberg-Osgood format:

$$\left(\frac{\varepsilon}{\varepsilon_0}\right) = \left(\frac{\sigma}{\sigma'_0}\right) + \alpha\left(\frac{\sigma}{\sigma_0}\right)^n \tag{3E-11}$$

where, σ_0 is the material yield stress, $\varepsilon_0 = \sigma_0/E$, and α and n are obtained by fitting the preceding equation to the material true stress-strain curve.

The estimation scheme formulas to evaluate the J-integral for a pipe with a throughwall circumferential flaw subjected to pure tension or pure bending are as follows:

Tension

$$J = f_1\left(a_e, \frac{R}{t}\right)\frac{P^2}{E} + \alpha\sigma_0\varepsilon_0 c\left(\frac{a}{b}\right)h_1\left(\frac{a}{b}, n, \frac{R}{t}\right)\left[\frac{P}{P_0}\right]^{n+1} \tag{3E-12}$$

where:

$$f_1\left(\frac{a}{b}, n, \frac{R}{t}\right) = \frac{aF^2\left(\frac{a}{b}, n, \frac{R}{t}\right)}{4\pi R^2 t^2}$$

$$P_0 = 2\sigma_0 Rt\left[\pi - \gamma - 2\,\mathrm{asin}\left(\frac{1}{2}\sin\gamma\right)\right]$$

Bending

$$J = f_1\left(a_e, \frac{R}{t}\right)\frac{M^2}{E} + \alpha\sigma_0\varepsilon_0 c\left(\frac{a}{b}\right)h_1\left(\frac{a}{b}, n, \frac{R}{t}\right)\left[\frac{M}{M_0}\right]^{n+1} \tag{3E-13}$$

where:

$$f_1\left(\frac{a}{b}, n, \frac{R}{t}\right) = \pi a\left(\frac{R}{I}\right)^2 F^2\left(\frac{a}{b}, n, \frac{R}{t}\right)$$

$$M_0 = M_0\left[\cos\left(\frac{\gamma}{2}\right) - \frac{1}{2}\sin(\gamma)\right]$$

The nondimensional functions F and h are given in Reference 3E-20.

While the calculation of J for given α, n, α_0 and load type is reasonably straight-forward, one issue that needs to be addressed is the tearing instability evaluation when the loading includes both the membrane and the bending stresses. The estimation scheme is capable of evaluating only one type of stress at a time.

This aspect is addressed next.

3E.3.1.3 Tearing Instability Evaluation Considering Both the Membrane and Bending Stresses

Based on the estimation scheme formulas and the tearing instability methodology just outlined, the instability bending and tension stresses can be calculated for various throughwall circumferential flaw lengths. Figure 3E-15 shows a schematic plot of the instability stresses as a function of flaw length. For the same stress level, the allowable flaw length for the bending is expected to be larger than the tension case.

When the applied stress is a combination of the tension and bending, a linear interaction rule is used to determine the instability stress or conversely the critical flaw length. The application of linear interaction rule is certainly conservative when the instability load is close to the limit load. The applicability of this proposed rule should be justified by providing a comparision of the predictions by the proposed approach (or an alternate approach) with those available for cases where the combination is treated together.

The interaction formulas are as follows (Figure 3E-15):

Critical Flaw Length

$$a_c = \frac{(\sigma_t)}{\sigma_t + \sigma_b} a_{c,\,t} + \frac{(\sigma_b)}{\sigma_t + \sigma_b} a_{c,\,b} \tag{3E-14}$$

where:

σ_t = Applied membrane stress

σ_b = Applied bending stress

$a_{c,t}$ = Critical flaw length for a tension stress of $(\sigma_t + \sigma_b)$

$a_{c,b}$ = Critical flaw length for a bending stress of $(\sigma_t + \sigma_b)$

Instability Bending Stress

$$S_b = \left(1 - \frac{\sigma_t}{\sigma'_t}\right)\sigma'_b \tag{3E-15}$$

where:

S_b = Instability bending stress for flaw length, a, in the presence of membrane stress, σ_t

σ_t = Applied membrane stress

σ'_t = Instability tension stress for flaw length, a

σ'_b = Instability bending stress for flaw length, a

Once the instability bending stress, S_b, in the presence of membrane stress, σ_t, is determined, the instability load margin corresponding to the detectable leak-size crack (as required by LBB criterion in Section 3.6.3) can be calculated as follows:

Instability Load Margin

$$\frac{\sigma_t + S_b}{\sigma_t + \sigma_b} \tag{3E-16}$$

It is assumed in the preceding equation that the uncertainty in the calculated applied stress is essentially associated with the stress due to applied bending loads and that the membrane stress, which is generally due to the pressure loading, is known with greater certainty. This method of calculating the margin against loads is also consistent with the definition of load margin employed in Paragraph IWB-3640 of Section XI (Reference 3E-21).

3E.3.2 Application of (J/T) Methodology to Carbon Steel Piping

From Figure 3E-3, it is evident that carbon steels exhibit transition temperature behavior marked by three distinct stages: lower shelf, transition and upper shelf. The carbon steels generally exhibit ductile failure mode at or above upper shelf temperatures. This would suggest that a net-section collapse approach may be feasible for the evaluation of postulated flaws in carbon steel piping. Such a suggestion was also made in a review report prepared by the Naval Research Lab (Reference 3E-22). Low temperature (i.e. less than 51.7°C) pipe tests conducted by GE (Reference 3E-23) and by Vassilaros (Reference 3E-24) which involved circumferentially cracked pipes subjected to bending and/or pressure loading, also indicate that a limit load approach is feasible. However, test data at high temperatures specially involving large diameter pipes are currently not available. Therefore, a (J/T) based approach is used in the evaluation.

3E.3.2.1 Determination of Ramberg-Osgood Parameters for 288°C Evaluation

Figure 3E-7 shows the true stress-true strain curves for the carbon steels at 288°C. The same data is plotted here in Figure 3E-16 in the Ramberg-Osgood format. It is seen that, unlike the stainless steel case, each set for stress-strain data (i.e. data derived from one stress-strain curve) follow approximately a single slope line. Based on the visual observation, a line representing $\sigma = 2, n = 5$ in Figure 3E-16 was drawn as representing a reasonable upper bound to the data shown.

The third parameter in the Ramberg-Osgood format stress-stain curve is σ_0, the yield stress. Based on the several internal GE data on carbon steels such as SA 333 Grade 6, and SA 106 Grade B, a reasonable value of 288°C yield strength was judged as 238.6 MPa. To summarize, the following values are used in this appendix for the (J/T) methodology evaluation of carbon steels as 288°C:

α = 2.0

n = 5.0

σ_0 = 238.6 MPa

E = 1.79×10^5 MPa

3E.3.2.2 Determination of Ramberg-Osgood Parameters for 216°C Evaluation

Figure 3E-17 shows the Ramberg-Osgood (R-O) format plot of the 177°C true stress-stain data on the carbon steel base metal. Also shown in Figure 3E-17 are the CE data a SA 106 Grade B at 204°C. Since the difference between the ASME Code Specified minimum yield strength at 177°C and 216°C is small, the 177°C stress-strain data were considered applicable in the determination of R-O parameters for evaluation at 216°C.

A review of Figure 3E-17 indicates that the majority of the data associated with any one test can be approximated by one straight line.

It is seen that some of the data points associated with the yield point behavior fall along the y-axis. However, these data points at low stain level were not considered significant and, therefore, were not included in the R-O fit.

The 177°C yield stress for the base material is given in Table 3E-4 as 261.2 MPa. Since the difference between the ASME Code specified minimum yield strengths of pipe and plate carbon steels at 216°C and 177°C is roughly 6.18 MPa, the σ_0 value for use at 216°C is chosen as (261.2-6.18 MPa) or 255.0 MPa. In summary, the following values of R-O parameters are used for evaluation of 216°C:

σ_0 = 255.0 MPa

a = 5.0

n = 4.0

3E.3.3 Modified Limit Load Methodology for Austenitic Stainless Steel Piping

Reference 3E-30 describes a modified limit load methodology that may be used to calculate the critical flaw lengths and instability loads for austenitic stainless steel piping and associated welds. If appropriate, this or an equivalent methodology may be used in lieu of the (J/T) methodology described in Subsection 3E.3.1.

3E.3.4 Bimetallic Welds

For joining austenitic steel to ferritic steel, the Ni-Cr-Fe Alloys 82 or 182 are generally used for weld metals. The procedures recommended in Subsection 3E.3.3 for the austenitic welds are applicable to these weld metals. This is justified based on the common procedures adopted for flaw acceptance in ASME Code Section XI, Article IWB-3600 and Appendix C, for both types of the welds. If other types of bimetallic weld metals are used, proper procedures should be used with generally acceptable justification.

3E.4 Leak Rate Calculation Methods

Leak rates of high pressure fluids through cracks in pipes are a complex function of crack geometry, crack surface roughness, applied stresses, and inlet fluid thermodynamic state. Analytical predictions of leak rates essentially consist of two separate tasks: (1) calculation of the crack opening area, and (2) estimation of the fluid flow rate per unit area. The first task requires the fracture mechanics evaluations based on the piping system stress state. The second task involves the fluid mechanics considerations in addition to the crack geometry and its surface roughness information. Each of these tasks is now discussed separately considering the type of fluid state in ABWR piping.

3E.4.1 Leak Rate Estimation for Pipes Carrying Water

EPRI-developed computer code PICEP (Reference 3E-31) may be used in the leak rate calculations. The basis for this code and comparison of its leak rate predictions with the experimental data is described in References 3E-32 and 3E-33. This code has been used in the successful application of LBB to the primary piping system of a PWR. The basis for flow rate and crack opening area calculations in PICEP is briefly described first. A comparison with experimental data is shown next.

Other methods (e.g., Reference 3E-34) may be used for leak rate estimation at the discretion of the applicant.

3E.4.1.1 Description of Basis for Flow Rate Calculation

The thermodynamic model implemented in PICEP computer program assumes the leakage flow through pipe cracks to be isenthalpic and homogeneous, but it accounts for non-equilibrium "flashing" transfer process between the liquid and vapor phases.

Fluid friction due to surface roughness of the walls and curved flow paths has been incorporated in the model. Flows through both parallel and convergent cracks can be treated. Due to the complicated geometry within the flow path, the model uses some approximations and empirical factors which were confirmed by comparison against test data.

For given stagnation conditions and crack geometries, the leak rate and exit pressure are calculated using an iterative search for the exit pressure starting from the saturation pressure corresponding to the upstream temperature and allowing for friction, gravitational, acceleration and area change pressure drops. The initial flow calculation is performed when the critical pressure is lowered to the backpressure without finding a solution for the critical mass flux.

A conservative methodology was developed to handle the phase transformation into a two-phase mixture or superheated steam through a crack. To make the model continuous, a correction factor was applied to adjust the mass flow rate of a saturated mixture to be equal to that of a slightly subcooled liquid. Similarly, a correction factor was developed to ensure continuity as the steam became superheated. The superheated model was developed by applying thermodynamic principles to an isentropic expansion of the single phase steam.

The code can calculate flow rates through fatigue or IGSCC cracks and has been verified against data from both types. The crack surface roughness and the number of bends account for the difference in geometry of the two types of cracks. The guideline for predicting leak rates through IGSCCs when using this model was based on obtaining the number of turns that give the best agreement for Battelle Phase II test data of Collier et al. (Reference 3E-35). For fatigue cracks, it is assumed that the crack path has no bends.

3E.4.1.2 Basis for Crack Opening Area Calculation

The crack opening area in the PICEP code is calculated using the estimation scheme formulas. The plastic contribution to the displacent is computed by summing the contributions of bending and tension alone, a procedure that underestimates the displacent from combined tension and bending. However, the plastic contribution is expected to be insignificant because the applied stresses at normal operation are generally such that they do not produce significant plasticity at the cracked location.

3E.4.1.3 Comparison Verification with Experimental Data

Figure 3E-18 from Reference 3E-33 shows a comparison PICEP prediction with measured leak rate data. It is seen that PICEP predictions are virtually always conservative (i.e., the leak flow rate is underpredicted).

3E.4.2 Flow Rate Estimation for Saturated Steam

3E.4.2.1 Evaluation Method

The calculations for this case were based on the maximum two-phase flow model developed by Moody (Reference 3E-36). However, in an LBB report, a justification should be provided by comparing the predictions of this method with the available experimental data, or a generally accepted method, if available, should be used.

The Moody nodal predicts the flow rate of steam-water mixtures in vessel blowdown from pipes (see Figure 3E-19). A key parameter that characterized the flow passage in the Moody analysis is fL/D_h, where f is the coefficient of friction, L, the length of the flow passage and D_h, the hydraulic diameter. The hydraulic diameter for the case of flow through a crack is 2δ, where δ is the crack opening displacement and the length of the flow passage is t, the thickness of the pipe. Thus, the parameter fL/D_h in the Moody analysis was interpreted as $ft/2\delta$ for the purpose of this evaluation.

Figure 3E-20 shows the predicted mass flow rates by Moody for fL/D_h of 0 and 1. Similar plots are given in Reference 3E-36 for additional fL/D_h values of 2 through 100. Since the steam in the ABWR main steamlines would be essentially saturated, the mass flow rate corresponding to the upper saturation envelope line is the appropriate one to use. Table 3E-6 shows the mass flow rates for a range of fL/D_h values for a stagnation pressure of 6.89 MPa, which is roughly equal to the pressure in an ABWR piping system carrying steam.

A major uncertainty in calculating the leakage rate is the value of f. This is discussed next.

3E.4.2.2 Selection of Appropriate Friction Factor

Typical relationships between Reynolds' Number and relative roughness ε / D_h, the ratio of effective surface protrusion height to hydraulic diameter, were relied upon in this case. Figure 3E-21, from Reference 3E-37, graphically shows such a relationship for pipes. The ε / D_h ratio for pipes generally ranges from 0 to 0.50. However, for a fatigue crack consisting of rough fracture surfaces represented by a few mils, the roughness height ε at some location may be almost as much as δ. In such cases, ε / D_h would seem to approach one-half. There are no data or any analytical model for such cases, but a crude estimate based on the extrapolation of the results in Figure 3E-21 would indicate that f may be of the order of 0.1 to 0.2. For this evaluation an average value of 0.15 was used with the modification as discussed next.

For blowdown of saturated vapor, with no liquid present, Moody states that the friction factor should be modified according to:

$$f_g = f_{GSP}\left[\frac{\upsilon_f}{\upsilon_g}\right]^{1/3} \quad (3E\text{-}17)$$

where:

f_g	=	Modified friction factor
f_{GSP}	=	Factor for single phase
$\frac{\upsilon_f}{\upsilon_g}$	=	Liquid/vapor specific volume ratio evaluated at an average static pressure in the flow path

This correction is necessary because the absence of a liquid film on the walls of the flow channel at high quality makes the two-phase flow model invalid as it stands. The average static pressure in the flow path is going to be something in excess of 3.45 MPa if the initial pressure is 6.89 MPa; this depends on the amount of flow choking and can be determined from Reference 3E-36. However, a fair estimate of $(\upsilon_f/\upsilon_g)^{1/3}$ is 0.3, so the friction factor for saturated steam blowdown may be taken as 0.3 of that for mixed flow.

Based on this discussion, a coefficient of friction of $0.15 \times 0.3 = 0.045$ was used in the flow rate estimation. Currently experimental data are unavailable to validate this assumed value of coefficient of friction.

3E.4.2.3 Crack Opening Area Formulation

The crack opening areas were calculated using LEFM procedures with the customary plastic zone correction. The loadings included in the crack opening area calculations were: pressure, weight and thermal expansion.

The mathematical expressions given by Paris and Tada (Reference 3E-38) are used in this case. The crack opening areas for pressure (A_p) and bending stresses (A_b) were separately calculated and then added together to obtain the total area, A_c.

For simplicity, the calculated membrane stresses from weight and thermal expansion loads were combined with the axial membrane stress, σ_p, due to the pressure.

The formulas are summarized below:

$$A_p = \frac{\sigma_p}{E}(2\pi Rt)G_p(\lambda) \quad (3E\text{-}18)$$

where:

σ_p = Axial membrane stress due to pressure, weight and thermal expansion loads

E = Young's modulus

R = Pipe radius

t = Pipe thickness

λ = Shell parameter = $\frac{a}{\sqrt{Rt}}$

a = Half crack length

$$G_p(\lambda) = \lambda^2 + 0.16\lambda^4 (0 \le \lambda \le 1)$$
$$= 0.02 + 0.81\lambda^2 + 0.30\lambda^3 + 0.03\lambda^4 (1 \le \lambda \le 5) \qquad \text{(3E-19)}$$

$$A_b = \frac{\sigma_b}{E} \cdot \pi \cdot R^2 \cdot \frac{(3 + \cos\theta)}{4} I_t(\theta) \qquad \text{(3E-20)}$$

where:

σ_b = Bending stress due to weight and thermal expansion loads

θ = Half crack angle

$$I_t(\theta) = 2\theta^2 \left[1 + \left(\frac{\theta}{\pi}\right)^{3/2} \left\{ 8.6 - 13.3\left(\frac{\theta}{\pi}\right) + 24\left(\frac{\theta}{\pi}\right)^2 \right\} + \left(\frac{\theta}{\pi}\right)^3 \left\{ 22.5 - 75\left(\frac{\theta}{\pi}\right) + 205.7\left(\frac{\theta}{\pi}\right)^2 - 247.5\left(\frac{\theta}{\pi}\right)^3 + 242\left(\frac{\theta}{\pi}\right)^4 \right\} \right] (0 < \theta < 100°) \qquad \text{(3E-21)}$$

The plastic zone correction was incorporated by replacing a and θ in these formulas by a_e and θ_e which are given by:

$$\theta_e = \theta + \frac{K_{total}^2}{2\pi R \sigma_Y{}^2} \qquad \text{(3E-22)}$$

$$a_e = \theta_e \cdot R$$

The yield stress, σ_y, was conservatively assumed as the average of the code specified yield and ultimate strength. The stress intensity factor, K_{total}, includes contribution due to both the membrane and bending stress and is determined as follows:

$$K_{total} = K_m + K_b \tag{3E-23}$$

where:

$$K_m = \sigma_p \sqrt{a} \cdot F_p(\lambda)$$

$$F_p(\lambda) = (1 + 0.3225\lambda^2)^{1/2} (0 \le \lambda \le 1)$$

$$= 0.9 + 0.25\lambda \qquad (1 \le \lambda \le 5)$$

$$K_b = \sigma_b \cdot \sqrt{\pi a} \cdot F_b(\theta)$$

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$$F_b(\theta) = 1 + 6.8\left(\frac{\theta}{\pi}\right)^{3/2} - 13.6\left(\frac{\theta}{\pi}\right)^{5/2} + 20\left(\frac{\theta}{\pi}\right)^{7/2} (0 \le \theta \le 100°)$$

The steam mass flow rate, M, shown in Table 3E-6 is a function of parameter, $m/2\delta$. Once the mass flow rate is determined corresponding to the calculated value of this parameter, the leak rate in liter/min can then be calculated.

3E.5 Leak Detection Capabilities

A complete description of various leak detection systems is provided in Subsection 5.2.5. The leak detection system gives separate considerations to leakage within the drywell and leakage external to the drywell. The limits for reactor coolant leakage are described in Subsection 5.2.5.4.

The total leakage in the drywell consists of the identified leakage and the unidentified leakage. The identified leakage is that from pumps, valve stem packings, reactor vessel head seal and other seals, which all discharge to the equipment drain sump. The Technical Specification limit on the identified leak rate is expected to be 95 L/min.

The unidentified leak rate in the drywell is the portion of the total leakage received in the drywell sumps that is not identified as previously described. The licensing (Technical Specification) limit on unidentified leak rate is 3.785 L/min. To cover uncertainties in leak detection capability, although it meets Regulatory Guide 1.45 requirements, a margin factor of

10 is required per Reference 3E-30 of Subsection 3E.3.4 to determine a reference leak rate. A reduced margin factor may be used if accounts can be made of the effects of sources of uncertainties such as plugging of the leakage crack with particulate material over time, leakage prediction, measurement techniques, personnel and frequency of monitoring. For the piping in drywell, a reference leak rate of 37.85 L/min may be used, unless a smaller rate can be justified.

The sensitivity and reliability of leakage detection systems used outside the drywell must be demonstrated to be equivalent to Regulatory Guide 1.45 systems. Methods that have been shown to be acceptable include local leak detection (e.g., example, visual observation or instrumentation). Outside the drywell, the leakage rate detection and the margin factor depend upon the design of the leakage detection systems.

3E.6 Guidelines for Preparation of an LBB Report

Some of the key elements of an LBB evaluation report for a high energy piping system are: (1) system description, (2) evaluation of susceptibility to waterhammer and thermal fatigue, (3) material specification, (4) piping geometry, (5) stresses and (6) the LBB margin in evaluation results. Two examples are presented in the following subsections to provide guidelines and illustrations for preparing an LBB evaluation report.

3E.6.1 Main Steam Piping

3E.6.1.1 System Description

The four 700A main steam lines (MS) carry steam from the reactor to the turbine and auxiliary systems. The RCPB portion of each line being evaluated in this section connects to a flow restrictor, which is a part of the RPV nozzle and is designed to limit the rate of escaping steam from the postulated break in the downstream steamline. The restrictor is also used for flow measurements during plant operation. The SRVs discharge into the pressure suppression pool through SRV discharge piping. The SRV safety function includes protection against overpressure of the reactor primary system. The main steam line A has a branch connection to supply steam to the Reactor Core Isolation Cooling (RCIC) System turbine.

This section addresses the MS piping system in the reactor building which is designed and constructed to the requirements of ASME Code Section III, Class 1 piping (within outermost isolation valve) and Class 2 piping. It is classified as Seismic Category I and inspected according to ASME Code Section XI.

3E.6.1.2 Susceptibility to Water-Hammer

Significant pressure pulsation of the waterhammer effect in the pipe may occur as a result of opening of SRVs or closing of the turbine stop valve. A brief description of these phenomena follows. These two transients are considered in the main steam piping system design and fatigue analysis. These events are more severe than the opening or closing of a MSIV or water carryover through main steam and SRV piping. Moreover, the probability of water carryover during core flooding in case of an accident is low.

Safety/Relief Valve Lift Transient Description

The SRV produces momentary unbalanced forces acting on the discharge piping system for the period from the opening of the SRV until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug at the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the SRVs cause the discharge piping to vibrate. This, in turn, produces time dependent forces that act on the main steam piping segments.

There are a number of events/transients/postulated accidents that result in SRV lift:

(1) Automatic opening signal when main steam system pressure exceeds the setpoint for a given valve (there are different set points for different valves in a given plant).

(2) Automatic opening signal for all valves assigned to the Automatic Depressurization System function on receipt of proper actuation signal.

(3) Manual opening signal to valve selected by plant operator.

The SRVs close when the MS System pressure reaches the relief mode reseat pressure or when the plant operator manually releases the opening signals.

It is assumed (for conservatism) that all SRVs are activated at the same time, which produces simultaneous forces on the main steam piping system.

Turbine Stop Valve Closure Transient Description

Prior to turbine stop valve closure, saturated steam flows through each main steamline at nuclear boiler rated (NBR) pressure and mass flow rate. Upon signal, the turbine stop valves close rapidly and the steam flow stops at the upstream side of these valves. A pressure wave is created and travels at sonic velocity toward the reactor vessel through each main streamline. The flow of steam into each main steamline from the reactor vessel continues until the fluid compression wave reaches the reactor vessel nozzle. Repeated reflection of the pressure wave at the reactor vessel and stop valve ends of the main steamlines produces time-varying pressures and velocities at each point along the main steamlines. The combination of fluid momentum changes, shear forces, and pressure differences cause forcing functions which vary with position and time to act on the main steam piping system. The fluid transient loads due to turbine stop valve closure is considered as design load for upset condition.

Basic Fluid Transient Concept

Despite the fact that the SRV discharge and the turbine stop valve closure are flow-starting and flow-stopping processes, respectively, the concepts of mass, momentum, and energy conservation and the differential equations which represent these concepts are similar for both problems. The particular solution for either of the problems is obtained by incorporating the appropriate initial conditions and boundary conditions into the basic equations. Thus, relief valve discharge and turbine stop valve closure are seen to be specific solutions of the more general problem of compressible, non-steady fluid flow in a pipe.

The basic fluid dynamic equations which are applicable to both relief valve discharge and turbine stop valve closure are used with the particular fluid boundary conditions of these occurrences. Step-wise solution of these equations generates a time-history of fluid properties at numerous locations along the pipe. Simultaneously, reaction loads on the pipe are determined at each location corresponding to the position of an elbow.

The computer programs RVFOR and TSFOR described in Appendix 3D are used to calculate the fluid transient forces on the piping system due to SRV discharge and turbine stop valve closure. Both of the programs use method of characteristics to calculate the fluid transients.

The results from the RVFOR program have been verified with various inplant test measurements such as from the Monticello tests and Caorso tests and the test sponsored by the BWR Owner's Group for NUREG-0737 at Wyle test facilities, Huntsville, Alabama. Various data from the strain gages on the pipes and the load cells on the supports were compared with the analytical data and found to be in good correlation.

Evaluation of the ensuing effects is considered as a normal design process for the main steam piping system. The peak pressure pulses are within the design capability of a typical piping design and the piping stresses and support loads remain within the ASME Code allowables.

It is concluded that, during these waterhammer type events, the peak pressures and segment loads would not cause overstressed conditions for the main steam piping system.

3E.6.1.3 Thermal Fatigue

No thermal stratification and thermal fatigue are expected in the main steam piping, since there is no large source of cold water in these lines. A small amount of water may collect in the near horizontal leg of the main steamline due to steam condensation. However, a slope of 3.18 mm per 30.5 cm of main steam piping is provided in each main steamline. Water drain lines are provided at the end of slope to drain out the condensate. Thus, in this case no significant thermal cycling effects on the main steam piping are expected.

3E.6.1.4 Piping, Fittings and Safe End Materials

The material specified for the 700A main steam pipe is SA672 Grade C70. The corresponding specification for the piping fittings and forgings are given as SA420, WPL6 and SA350, LF2, respectively. The material for the safe end forging welded between the main steam piping and the steam nozzle is SA508 Class 3.

3E.6.1.5 LBB Margin Evaluation

The Code stress analysis of the piping is reviewed to obtain representative stress magnitudes. Table 3E-7 shows, for example purposes, the stress magnitudes due to pressure, weight, thermal expansion and SSE loads.

The leak rate calculations are performed assuming saturated steam conditions at 7.53 MPa. The leak rate model for saturated steam developed in Section 3E.4.2 is to be used in this evaluation. Pressure, weight and thermal expansion stresses are included in calculating the crack opening area. A plot of leak rate as a function of crack size is developed as is shown in Figure 3E-22. The leakage flaw length corresponding to the reference leak rate (Section 3E.5) is determined from this figure.

The calculations for the critical flaw size and instability load corresponding to leakage-size crack are performed using the J-T methodology. Specifically, the 288°C J-R curve shown in Figure 3E-12 and the Ramberg-Osgood parameters given in Subsection 3E.3.2.1 are used. A plot of instability tension and bending stresses as a function of crack length is developed. Table 3E-8 shows the example presentation of calculated critical crack size and the margin along with the instability load margin for the leakage size crack. It is noted that the critical crack size margin is greater than 2 and the instability load margin also exceeds $\sqrt{2}$.

3E.6.1.6 Conclusion

For all example main steamlines, based upon the reference leakage rates and assumed stress magnitudes, leakage flaw lengths are calculated and compared against the critical flaw length. The margin is shown to be greater than 2 for the leakage rates. Also, the leak-size crack stability evaluation is shown to have a margin of at least $\sqrt{2}$.

It is also shown that the conditions required for applicability of LBB (Subsection 3.6.3.2), such as high resistance to failure from effects of IGSCC, water hammer and thermal fatigue, are satisfied. Therefore, all four of the main steam lines qualify for LBB behavior.

3E.6.2 Feedwater Piping Example

3E.6.2.1 System Description

The function of the Feedwater (FW) System is to conduct water to the reactor vessel over the full range of the reactor power operation. The feedwater piping consists of two 550A diameter lines from the high-pressure feedwater heaters, connecting to the reactor vessel through three 300A risers on each line. Each line has one check valve inside the containment drywell and one positive closing check valve outside containment. During the shutdown cooling mode, reactor water pumped through the RHR heat exchanger in one loop is returned to the vessel by way of one feedwater line.

This section addresses the feedwater piping in the Reactor Building, extending from the vessel out to the outboard isolation valve (ASME Class 1) and further through the shutoff valve to and including the seismic interface restraint (ASME Class 2). This section of the feedwater piping is classified as Seismic Category I.

3E.6.2.2 Susceptibility to Waterhammer

There is no record of feedwater piping failure due to waterhammer. Although there are several check valves in the FW System, operating procedure and the control systems have been designed to limit the magnitude of water hammer load to the extent that a formal design is not required.

3E.6.2.3 Thermal Fatigue

Thermal fatigue is not a concern in ABWR feedwater piping. The ASME Code evaluation includes operating temperature transients, cold and hot water mixing and thermal stratification.

3E.6.2.4 Piping. Fittings and Safe End Material

The material for piping is either SA333, Grade 6 or SA-672, Grade C70.

3E.6.2.5 Piping Sizes, Geometries and Stresses

Table 3E-9 shows the normal operating temperatures, pressures and thickness for representative pipe sizes in the example FW System. The nominal thickness for both pipe sizes correspond to schedule 80. Table 3E-10 shows, for example purposes, the stress magnitudes for each pipe size due to pressure, weight, thermal expansion and SSE loads. Only the pressure weight and thermal expansion stresses are used in the leak rate evaluation, where a sum of all stresses is used in the instability load and critical flaw evaluation.

3E.6.2.6 LBB Margin Evaluation

The incoming water of the FW System is in a subcooled state. Accordingly, the leakage flaw length calculations are based on the procedure outlined in Section 3E.4.1. The saturation pressure, P_{sat}, for each pipe size is calculated from the normal operation temperatures given in Table 3E-9. The leak rates are calculated as a function of crack length. The leakage flaw lengths corresponding to the reference leak rate (Section 3E.5) are then determined.

The calculations for the critical flaw size and the instability load corresponding to leakage size cracks is performed using the J-T methodology. Specifically, the J-T curve shown in Figure 3E-13 and the Ramberg-Osgood parameters given in Subsection 3E.3.2.2 are used. Table 3E-11 shows the example presentation of calculated critical crack sizes, and the margins along with the instability load margins for the leakage size cracks. Results are shown for both the 550A and 300A lines. It is noted that the critical crack size margin is greater than 2 and the instability load margin also exceeds $\sqrt{2}$.

3E.6.2.7 Conclusion

For the example feedwater piping, based upon the reference leakage rate and assumed stress magnitudes, leakage flaw lengths are calculated for 550A and 300A lines. Comparison with critical crack lengths shows margin to be greater than 2. Also, the leak-size crack stability evaluation shows a margin of at least $\sqrt{2}$.

It is also demonstrated that the feedwater line meets other LBB criteria of Subsection 3.6.3.2, including immunity to failure from effects of IGSCC, waterhammer and thermal fatigue. Therefore, the feedwater lines qualify for LBB behavior.

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Table 3E-1 Leak Before Break Candidate Piping System

System	Location	Description	Diameter (mm)
Main Steam (4 lines)	PC	RPV to RCCV	700A
Feedwater (2 lines/ 6 risers)	PC	RPV to RCCV	550A/300A
RCIC Steam	PC	MS line to RCCV	150A
HPCF	PC	RPV to first check valve	200A
RHR/LPFL	PC	RPV to first check valve	250A
RHR/Suction	PC	RPV to first closed gate valve	350A
CUW	PC	RHR suction to RCCV	200A
Main Steam (4 lines)	Steam Tunnel	RCCV to turbine building	700A
Feedwater (2 lines)	Steam Tunnel	RCCV to turbine building	550A
RHR Div. A Suction	Steam Tunnel	FW line A to check valve	250A
RCIC Steam	SC	RCCV to turbine shutoff valve	150A
RCIC Supply	SC	FW line to first check valve	200A
CUW Suction	SC	RCCV to heat exchanger discharge	200A
CUW Discharge	SC	Heat exchanger discharge to FW suction	200A/150A

Legend:

PC: Primary Containment

SC: Secondary Containment

FW: Feedwater

MS: Main Steam

Note:

All piping in primary and secondary containment (including steam tunnel) are carbon steel piping, except the in-containment CUW piping which is stainless steel.

Table 3E-2 Electrodes and Filler Metal Requirements for Carbon Steel Welds

Base Material	P-No.	Process	Electrode Specification	or Filler Metal Classification
Carbon Steel to	P-1 to	SMAW	SFA 5.1	E7018
Carbon Steel or	P-1, P-3			
Low Alloy Steel	P-4 or P-5	GTAW PAW	SFA 5.18	E70S-2, E70S-3
		GMAW	SFA 5.18	E70S-2, E70S-3, E70S-6
			SFA 5.20	E70T-1
		SAW	SFA 5.17	F72EM12K, F72EL12

Table 3E-3 Supplier Provided Chemical Composition and Mechanical Properties Information

		Chemical Composition					Mech Property		
Material	**Product Form**	**C**	**Mn**	**P**	**S**	**Si**	**Sy (MPa)**	**Su (MPa)**	**Elongation (%)**
SA 333 Grade 6 Heat #52339	400A Sch. 80 Pipe	0.12	1.18	0.01	0.026	0.27	303.4	4,746	42.0
SA 516 Grade 70 Heat #E18767	2.54 cm Plate	0.18	0.98	0.017	0.0022	0.25	320.5	4,957	31.0

Note:

1. Pipe was normalized at 871°C. Held for 2 hours and air cooled.
2. Plate was normalized at 927°C for one hour and still air cooled.

Table 3E-4 Standard Tension Test Data at Temperature

Spec. No.	Material Temperature	Test (Temp.)	0.2% YS (MPa)	UTS (%)	Elong. (%)	RA
OW1	Pipe Weld	RT	455.7.	81.6	32	77.2
OW2	Pipe Weld	288°C	406.8.	93.9	24	56.7
ITWL2	Plate Weld	288°C	365.4	91.4	34	51.3
IBL1	Plate Base	RT	309.6.	73.7	38	51.3
IBL2	Plate Base	177°C	261.3.	64.2	34	68.9
IBL3	Plate Base	288°C	235.2.	69.9	29	59.4
OB1	Pipe Base	RT	300.9.	68.6	41	67.8
OB2	Pipe Base	177°C	299.0	74.9	21	55.4
OB3	Pipe Base	288°C	238.6	78.2	31	55.4

Table 3E-5 Summary of Carbon Steel J-R Curve Tests

No.	Specimen ID	Size	Description	Temp.
(1)	OWLC-A	1T	Pipe Weld	288°C
(2)	OBCL-1	1T	Pipe Base C-L Orientation	RT
(3)	OBLC2	1T	Pipe Base L-C Orientation	288°C
(4)	OBLC3-B	1T	Pipe Base L-C Orientation	177°C
(5)	BML-4	1T	Plate Base Metal, L-T Orientation	RT
(6)	BML4-14	2T	Plate Base Metal, L-T Orientation	RT
(7)	BML2-6	2T	Plate Base Metal, L-T Orientation	177°C
(8)	BML1-12	2T	Plate Base Metal, L-T Orientation	288°C
(9)	WM3-9	2T	Plate Weld Metal	RT
(10)	XWM1-11	2T	Plate Weld Metal	177°C
(11)	WM2-5	2T	Plate Weld Metal	288°C
(12)	HAZ	(Non-standard) Width = 7.09 cm	Heat-Affected Zone, Plate	RT
(13)	OWLC-7	1T	Pipe Weld	RT

Notes:

1. Pipe base metal, SA333 Grade 6
2. Plate base metal, SA516 Grade 70
3. Pipe weld made by shielded metal arc welding.
4. Plate weld made by submerged arc welding.

Table 3E-6 Mass Flow Rate for Several fl/D_h Values

fl/D_h	Mass Flow Rate, $kg/s{\cdot}m^2$
0	18,540
1	10,740
2	7,810
3	5,520
4	4,490
5	3,904
10	2,830
20	1,950
50	1,270
100	903

Table 3E-7 Stresses in the Main Steam Lines (Assumed for Example)

Nominal Pipe Size	Pipe O.D. (mm)	Nominal Thickness (mm)	Long. Pressure Stress (MPa)	Weight + Thermal Expansion Stress (MPa^2)	SSE Stress (MPa)
700A	711.2	36.1	35.7	20.7	34.5

Table 3E-8 Critical Crack Length and Instability Load Margin Evaluations for Main Steam Lines (Example)

					Margins on	
Pipe Size	Reference Leak Rate (l/s)	Reference Leakage Crack Length (cm)	Critical Crack Length (cm)	Instability* Bending Stress, S_b (MPa)	Critical Crack	Load† at Leakage Crack
700A	0.631‡	34.2	78.0	166.9	2.3	2.2

* Based on Equation 3E-15

† Based on Equation 3E-16.

‡ See Section 3E.5.

Table 3E-9 Data for Feedwater System Piping (Example)

Nominal Pipe Size	Pipe O.D. (mm)	Nominal Thickness (mm)	Nominal Temperature (°C)	Operating Pressure (MPa)
300A	323.9	21.4	216	7.58
550A	558.8	34.9	216	7.58

Table 3E-10 Stresses in Feedwater Lines (Assumed for Example)

Nominal Pipe Size	Longitudinal Pressure Stress (MPa)	Weight + Thermal Expansion Stress (MPa)	Safe Shut-down Earthquake (SSE) Stress (MPa)
300A	35.2	27.5	34.5
550A	37.3	27.5	34.5

Table 3E-11 Critical Crack Length and Instability Load Margin Evaluations for Feedwater Lines (Example)

Pipe Size	Reference Leak Rate (l/s)	Reference Leakage Crack Length (cm)	Critical Crack Length (cm)	Instability* Bending Stress, S_b (MPa)	Margins on Critical Crack	Margins on Load† at Leakage Crack
300A	0.631‡	14.7	33.3	165.7	2.3	2.1
550A	0.631‡	17.0	51.8	176.5	3.1	2.2

* Based on Equation 3E-15

† Based on Equation 3E-16.

‡ See Section 3E.5.

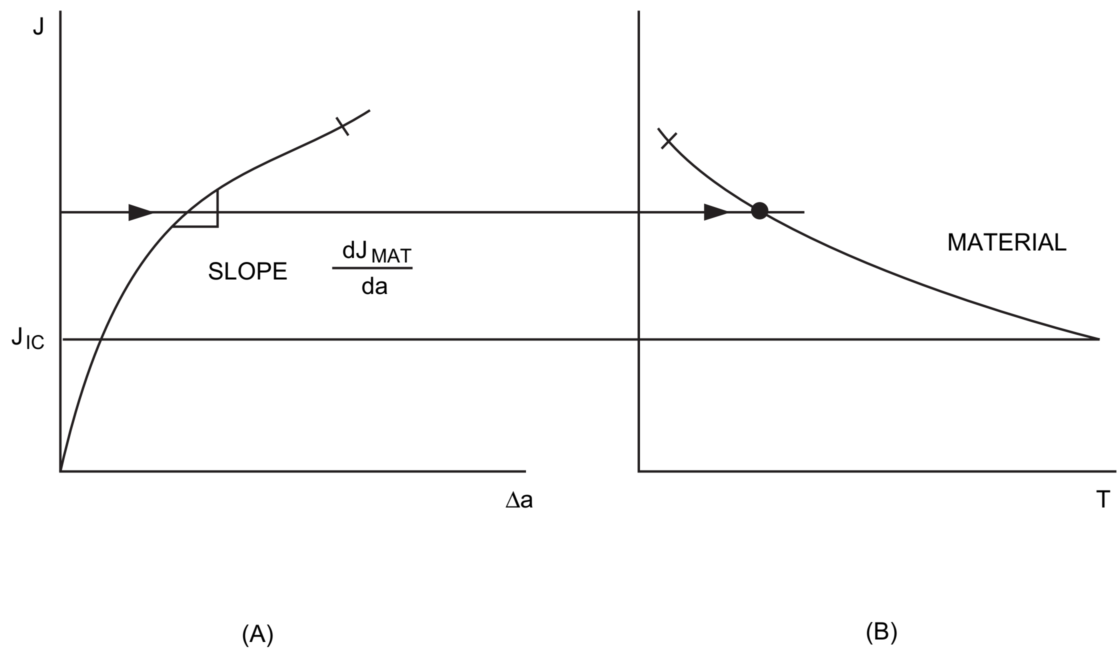

Figure 3E-1 Schematic Representation of Material J-Integral R and J-T Curves

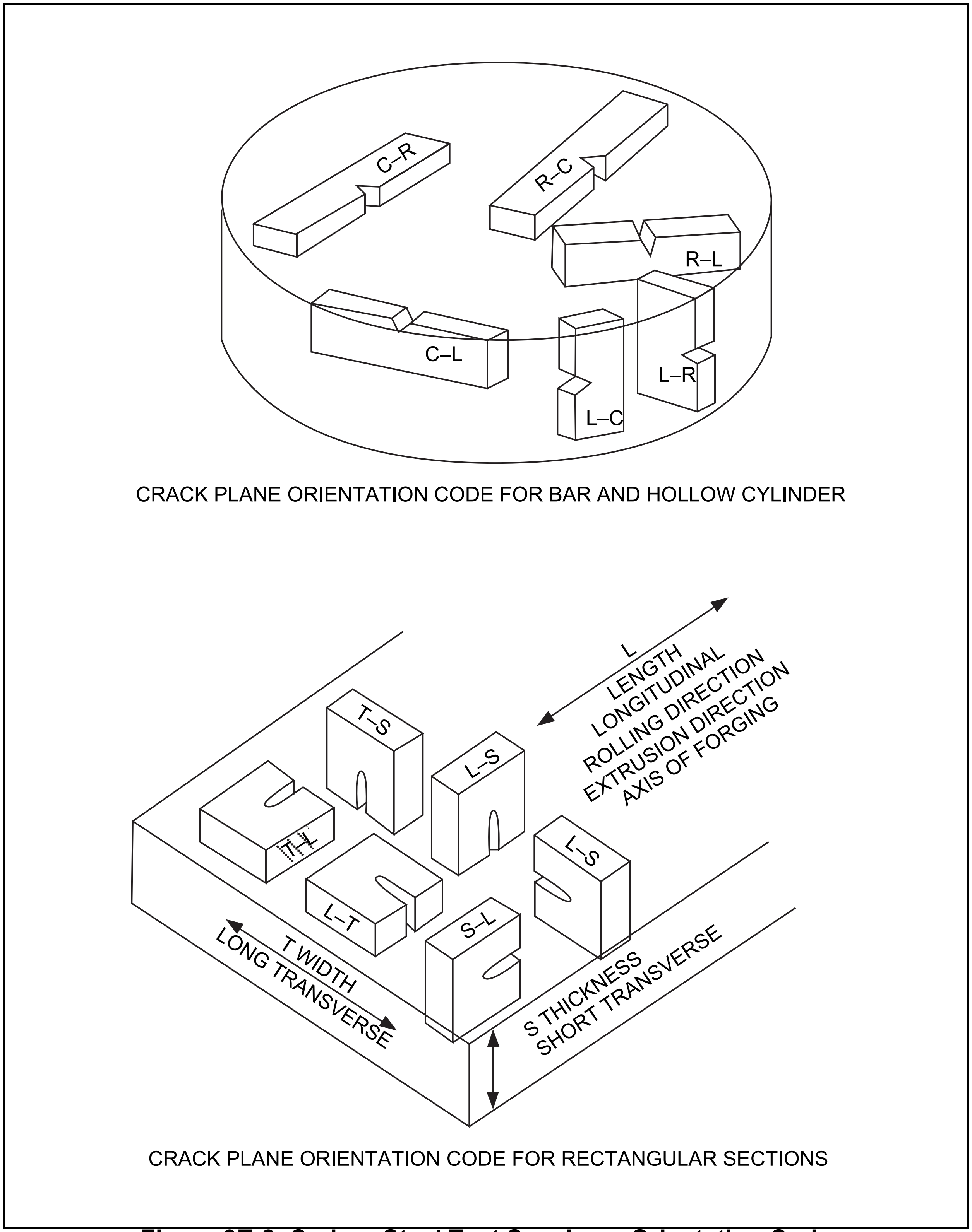

Figure 3E-2 Carbon Steel Test Specimen Orientation Code

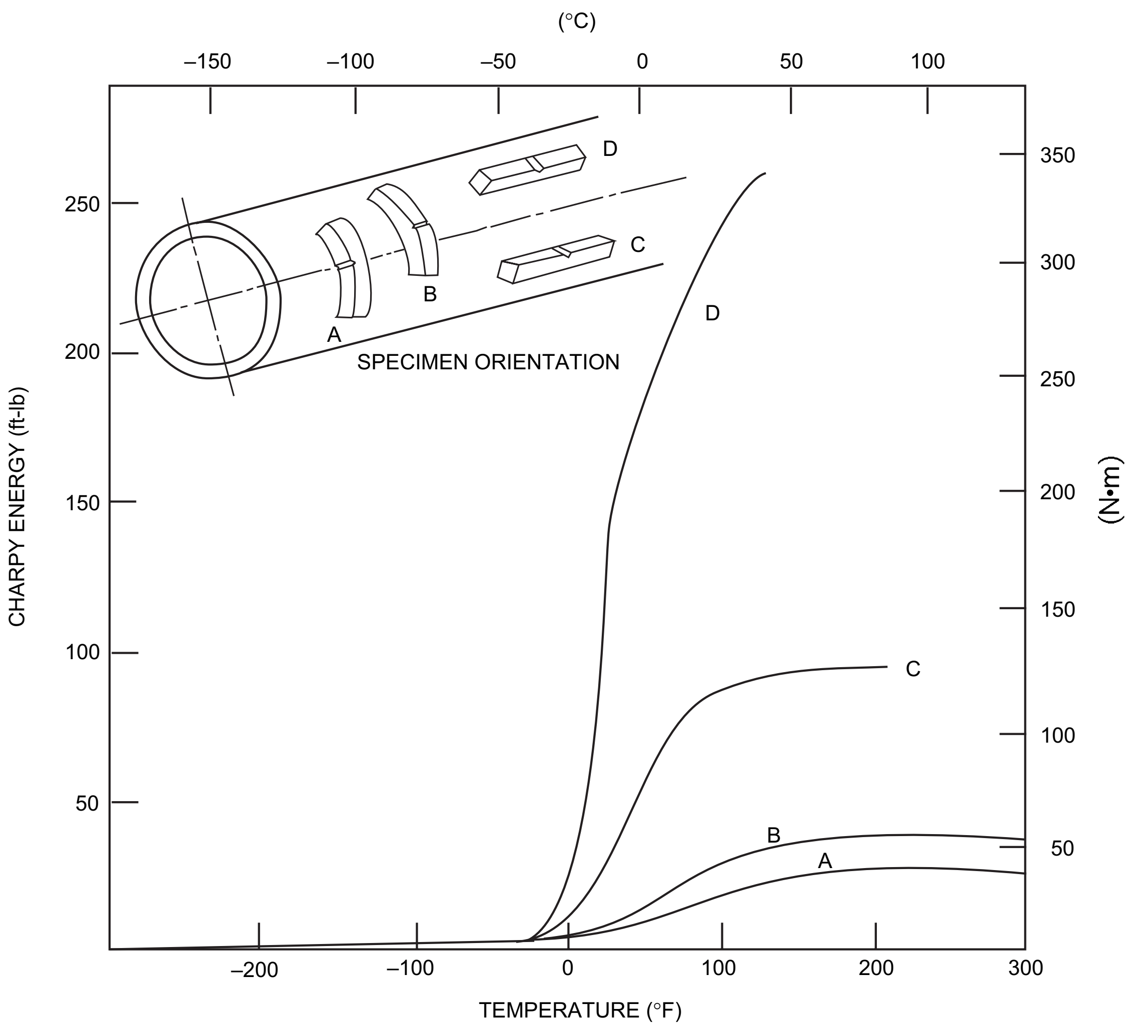

Figure 3E-3 Toughness Anisotropy of ASTM 106 Pipe (152 mm Sch. 80)

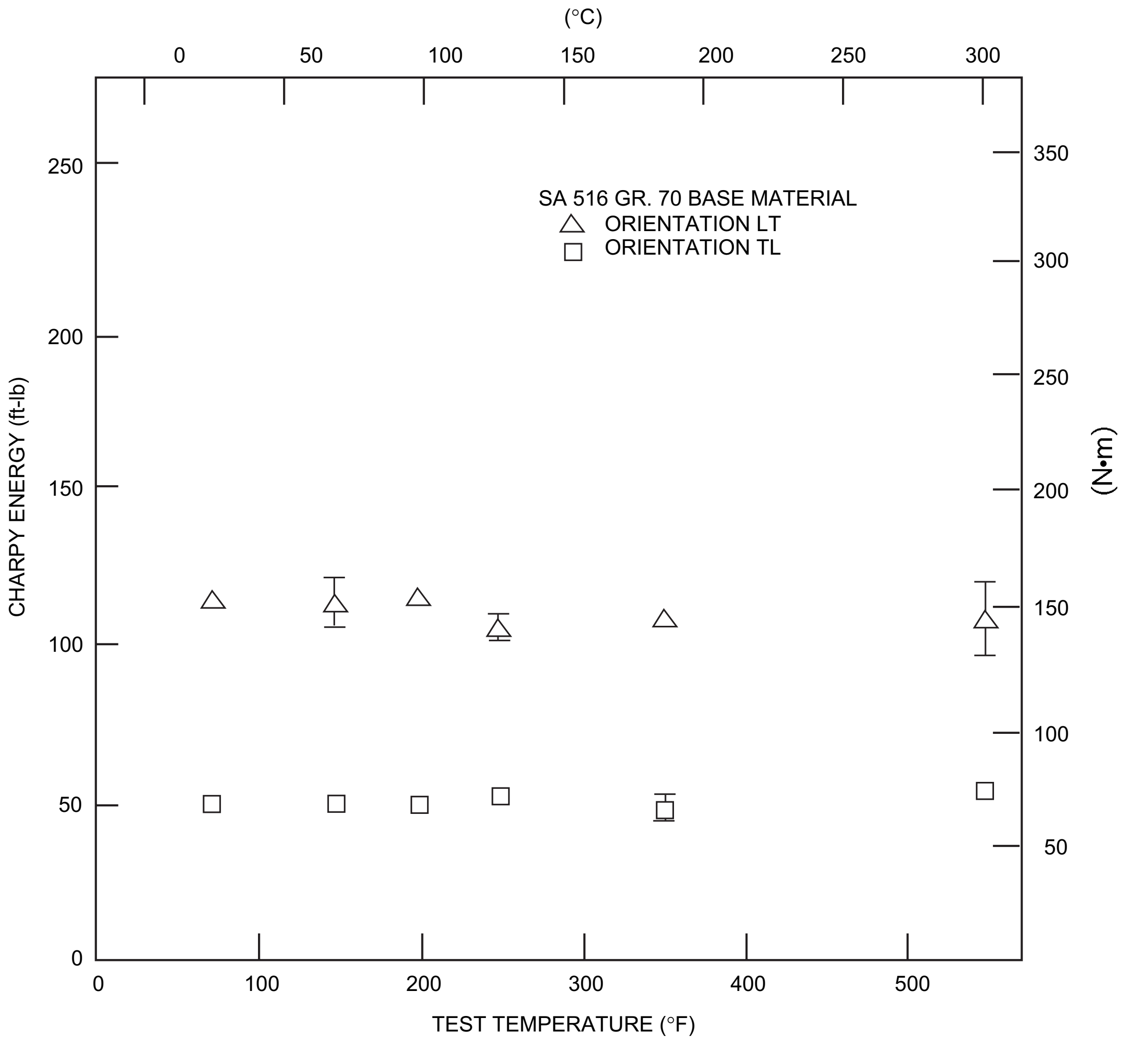

Figure 3E-4 Charpy Energies for Pipe Test Material as a Function of Orientation and Temperature

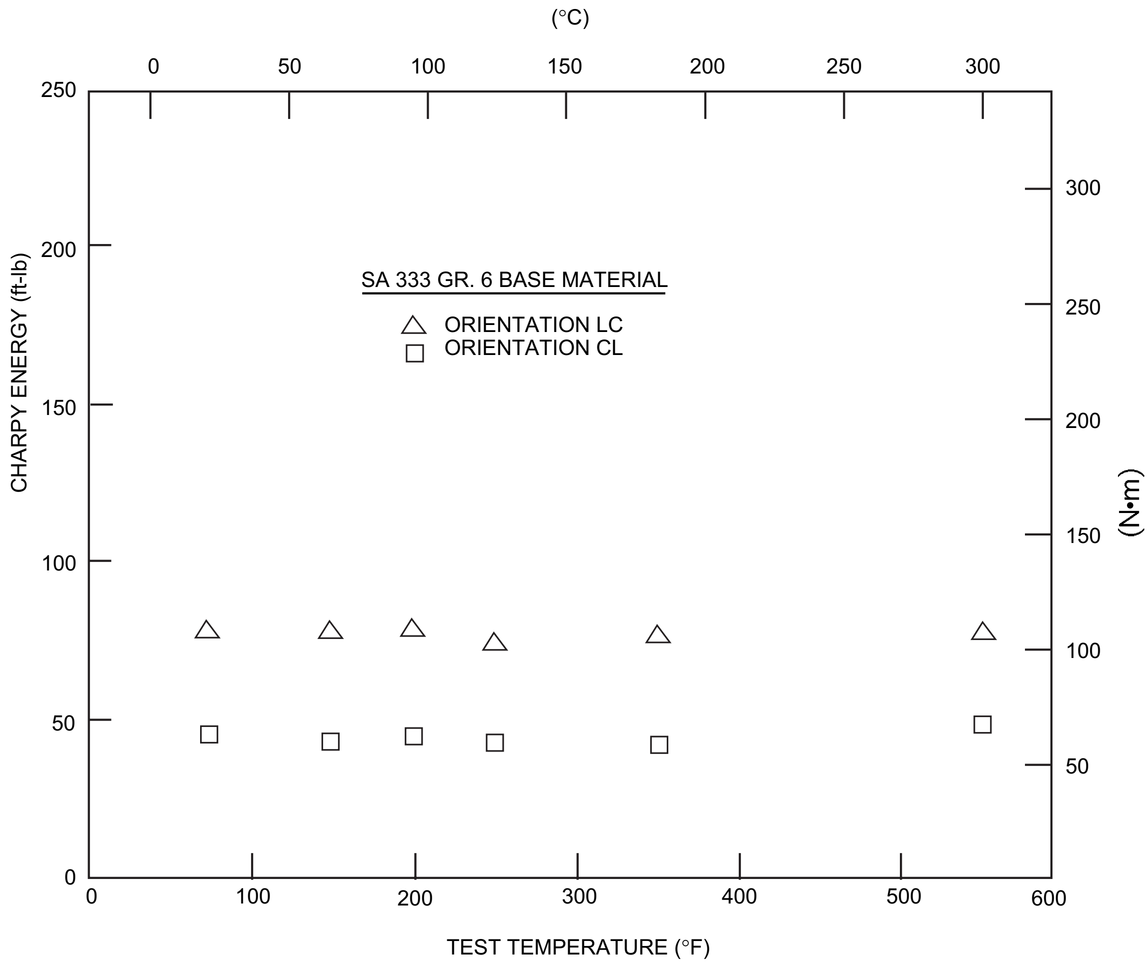

Figure 3E-5 Charpy Energies for Plate Test Material as a Function of Orientation and Temperature

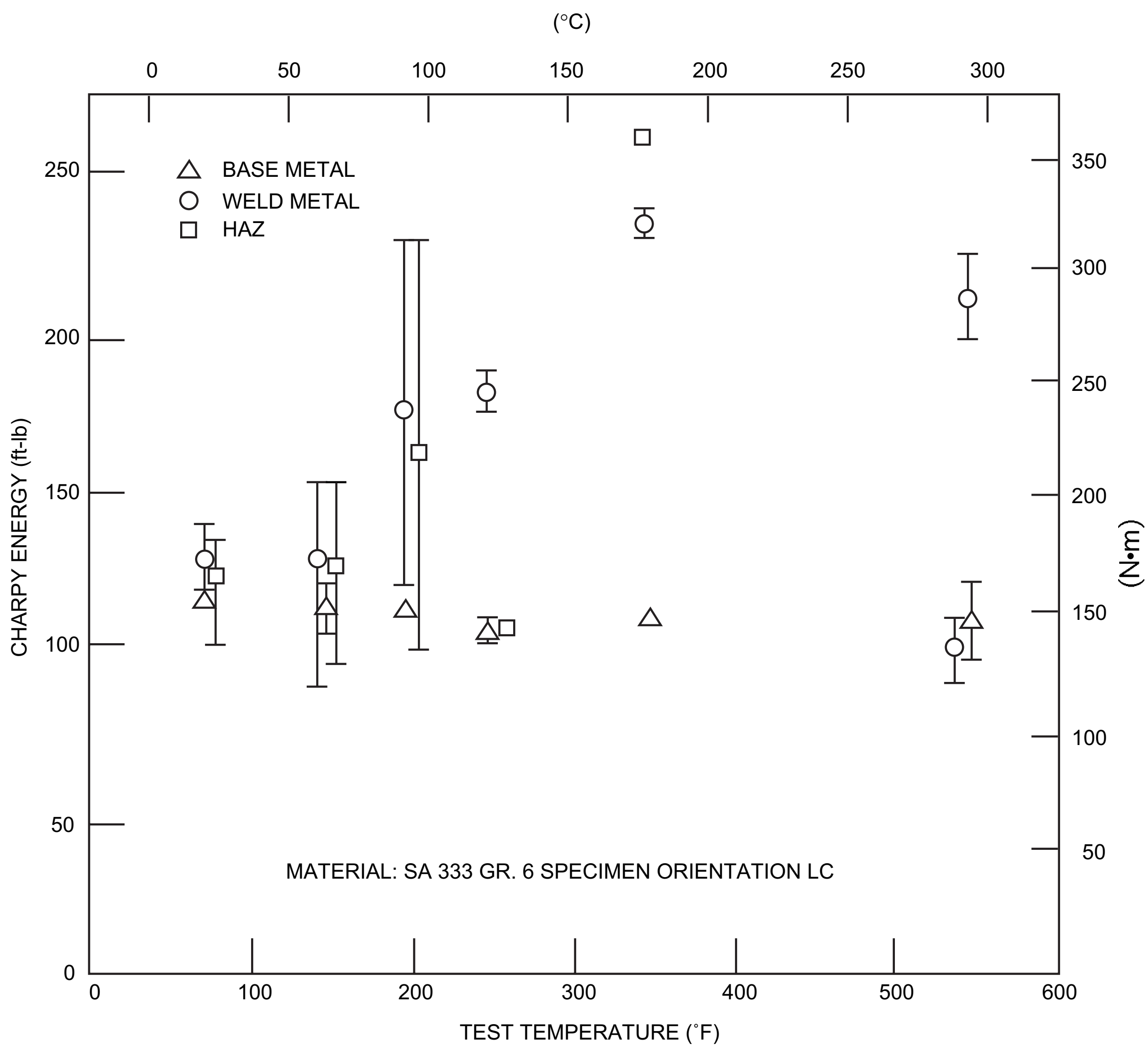

Figure 3E-6
Comparison of Base Metal, Weld and HAZ Charpy Energies for SA 333 Grade 6

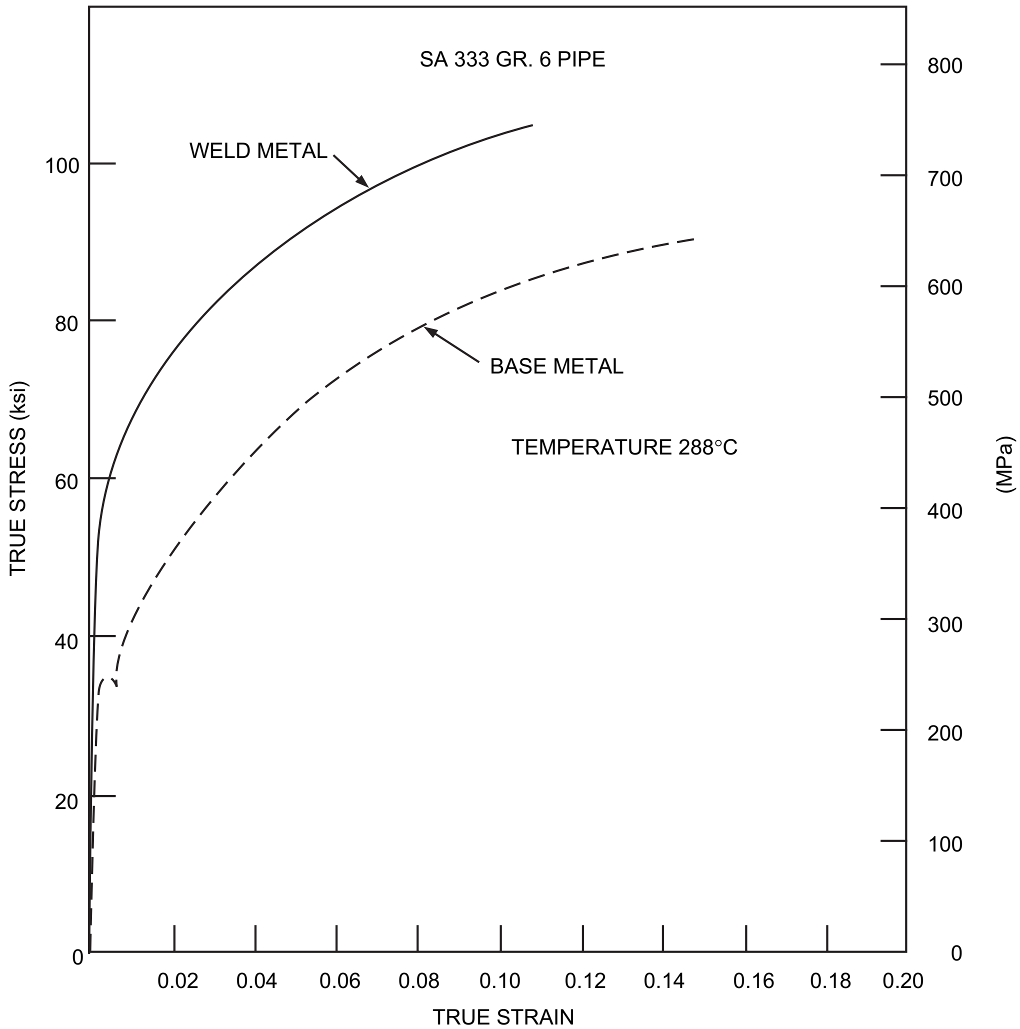

Figure 3E-7
Plot of 288°C True Stress-True Strain Curves for SA 333 Grade 6 Carbon Steel

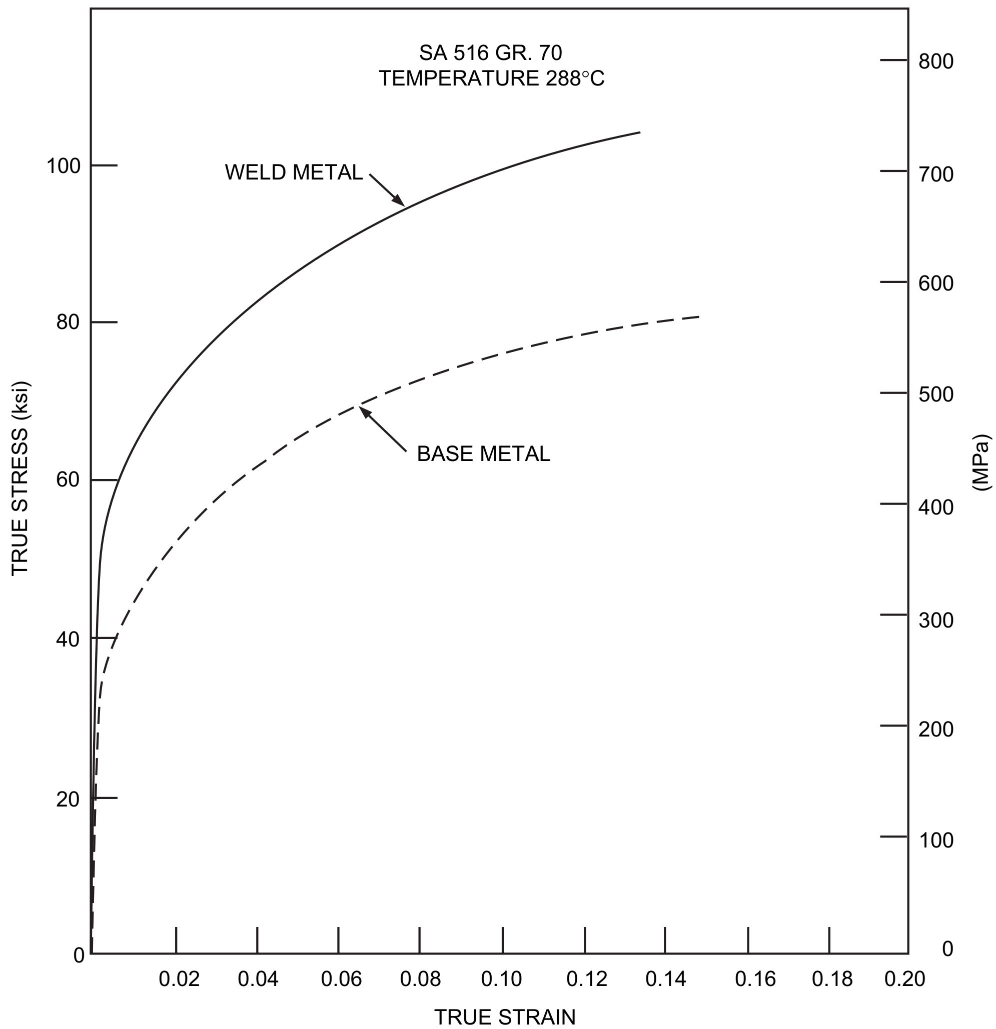

Figure 3E-8
Plot of 288°C True Stress-True Strain Curves for SA 516 Grade 70 Carbon Steel

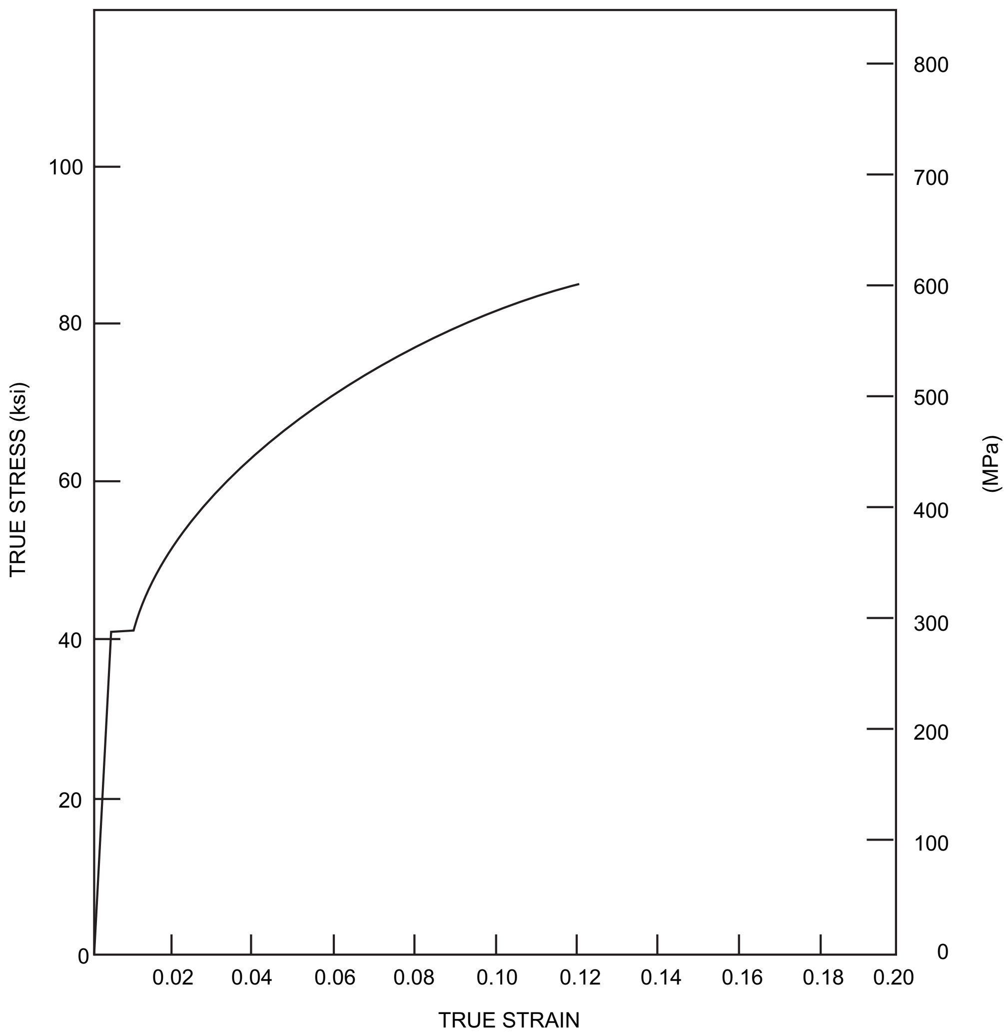

Figure 3E-9
Plot of 117°C True Stress-True Strain Curves for SA 333 Grade 6 Carbon Steel

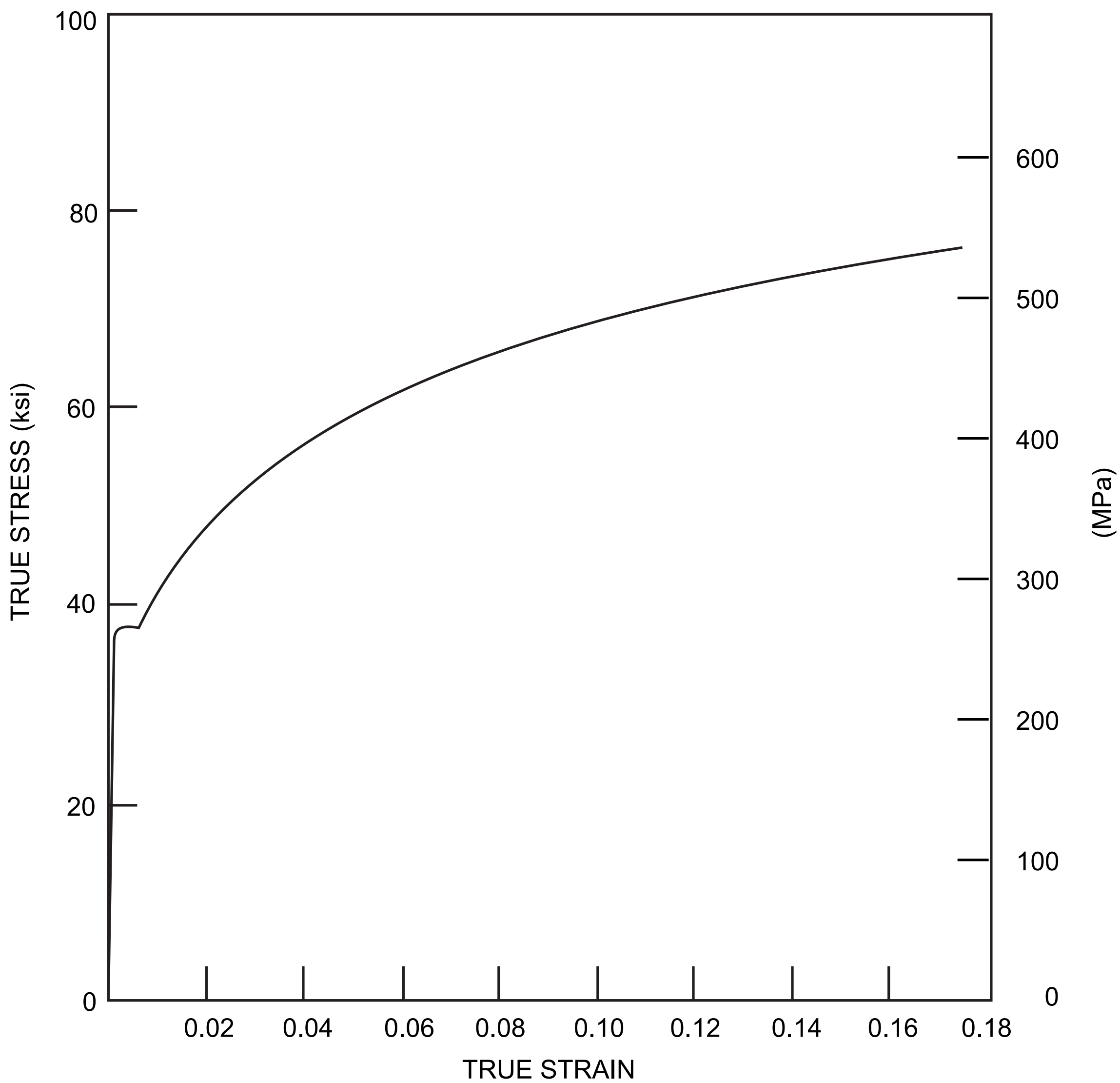

Figure 3E-10
Plot of 177°C True Stress-True Strain Curves for SA 516 Grade 70 Carbon Steel

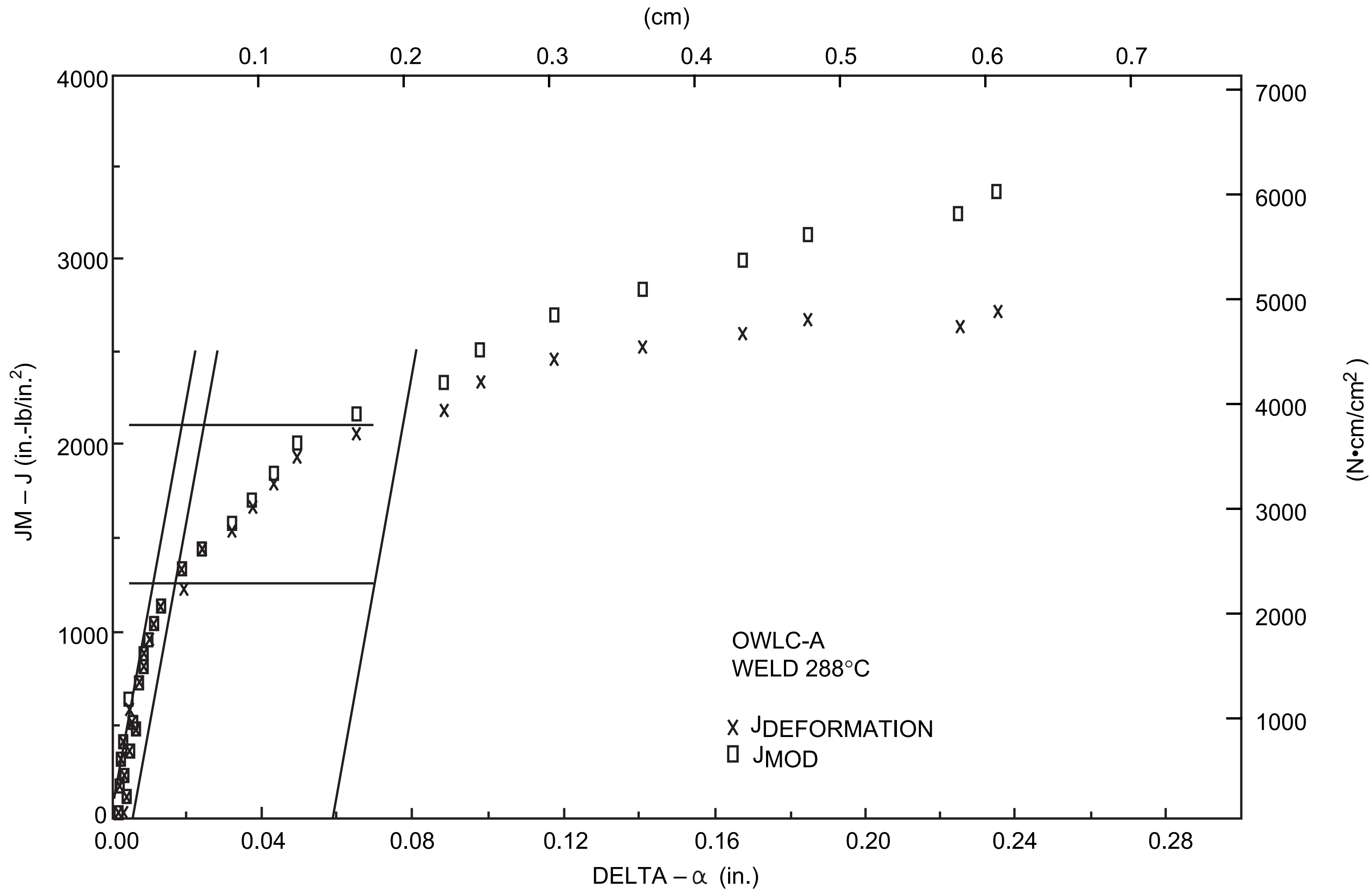

Figure 3E-11 Plot of 288°C Test J-R Curve for Pipe Weld

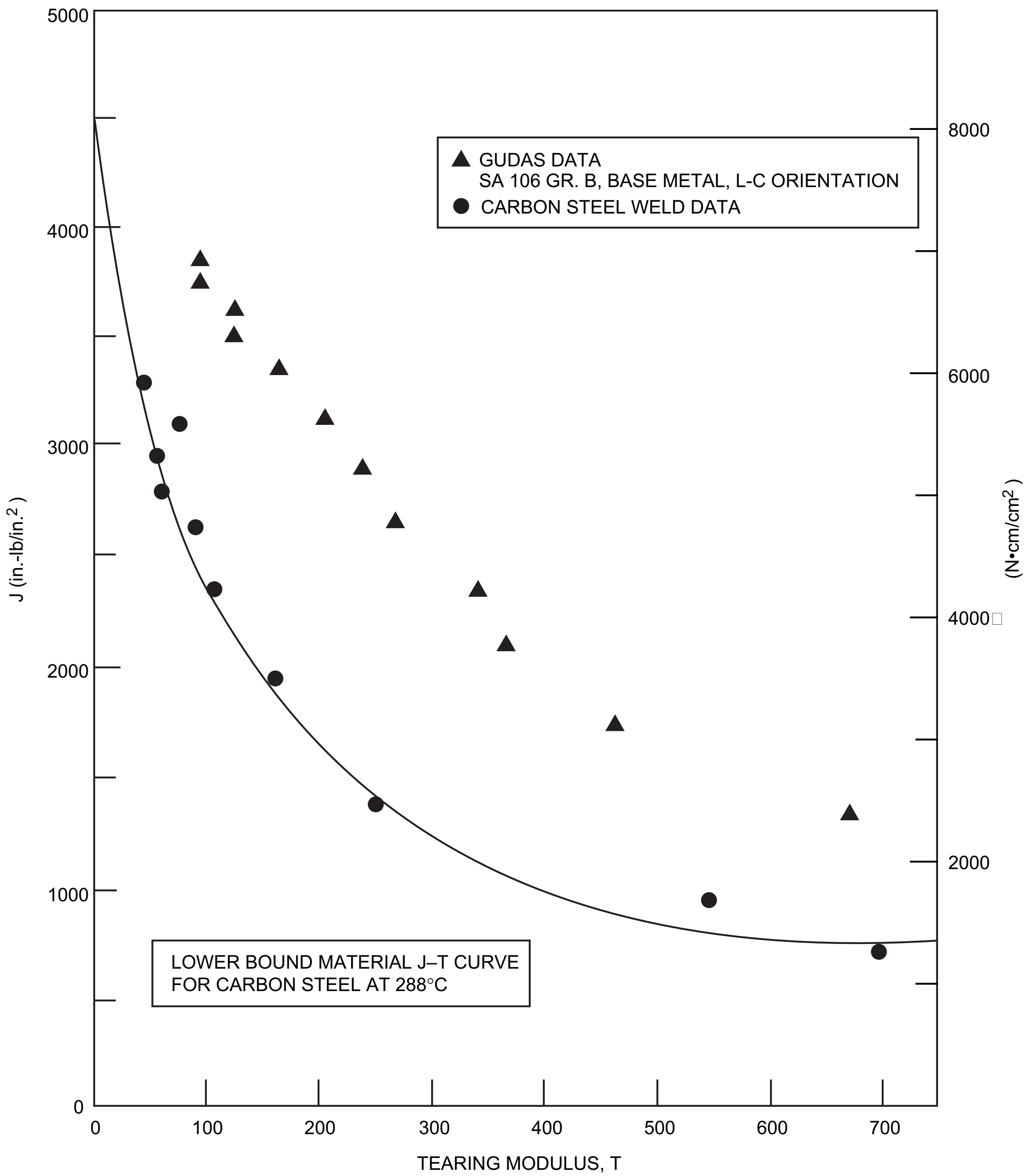

Figure 3E-12 Plot of 288°C J_{mod}, T_{mod} Data from Test J-R Curve

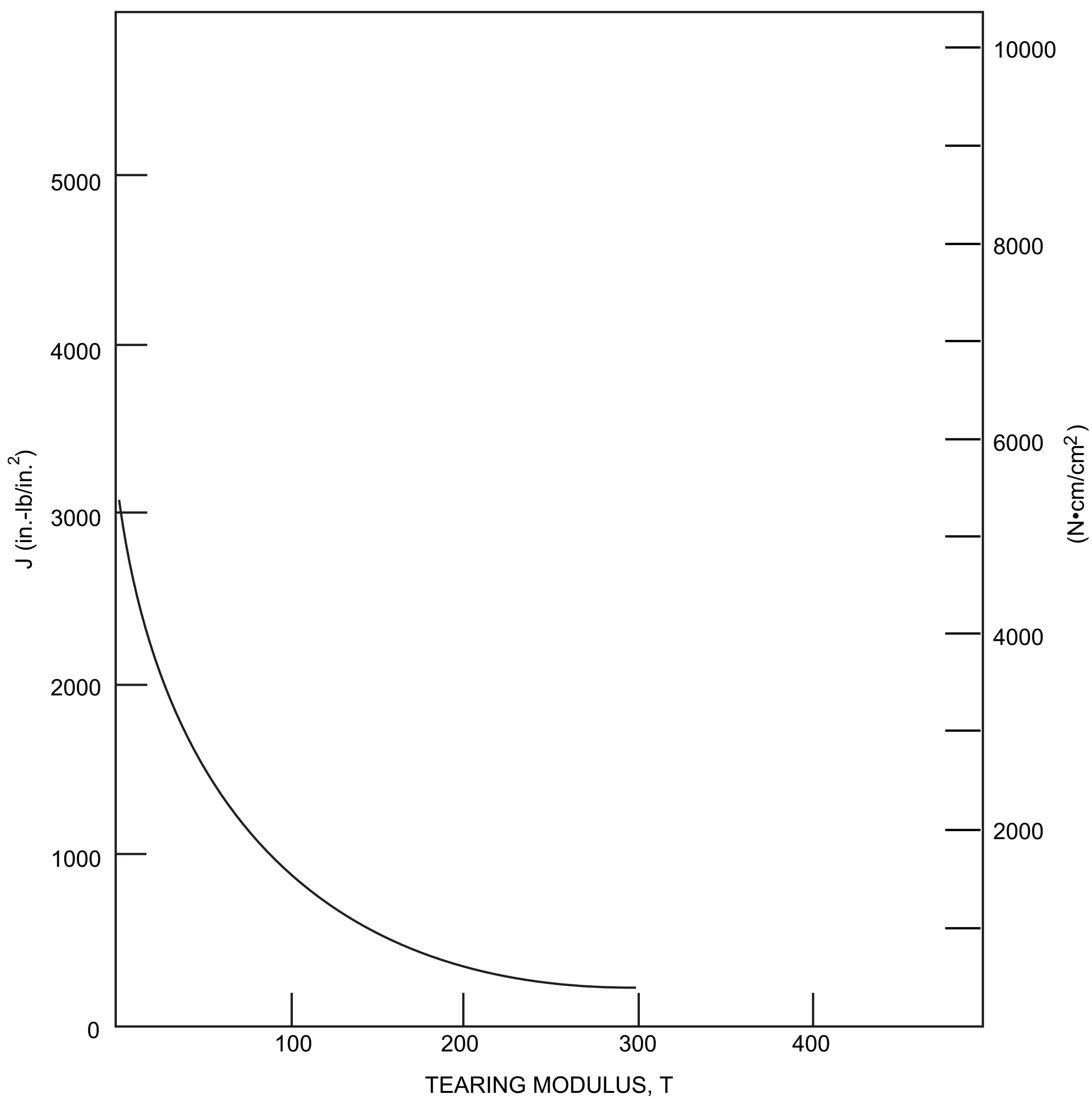

Figure 3E-13 Carbon Steel J-T Curve for 216°C

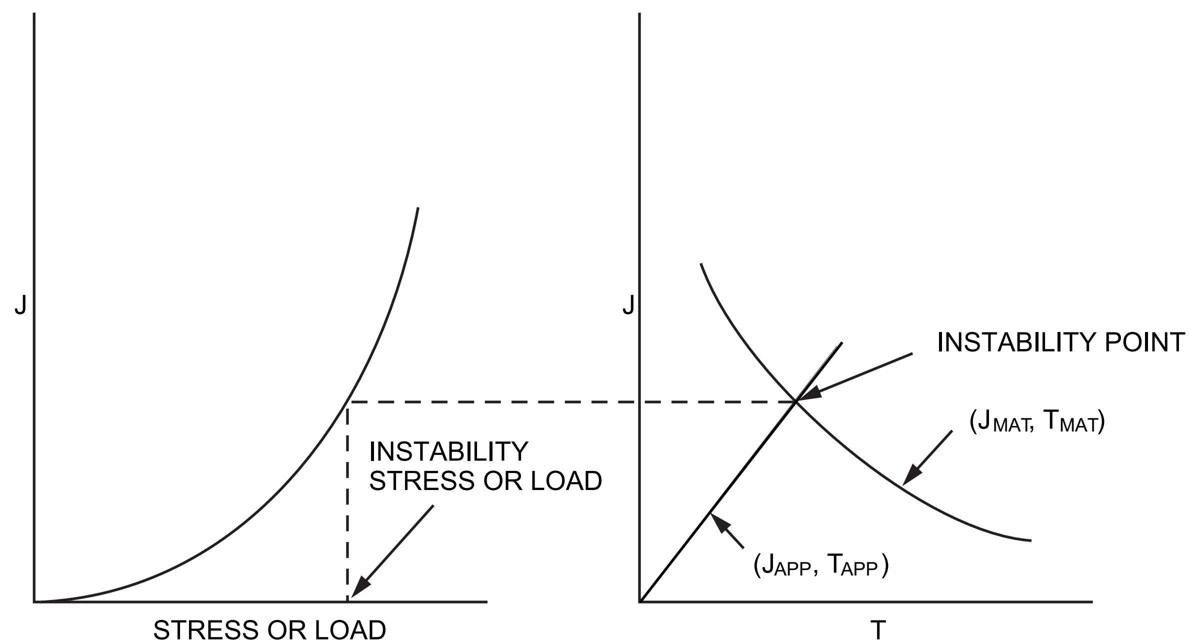

Figure 3E-14 Schematic Illustration of Tearing Stability Evaluation

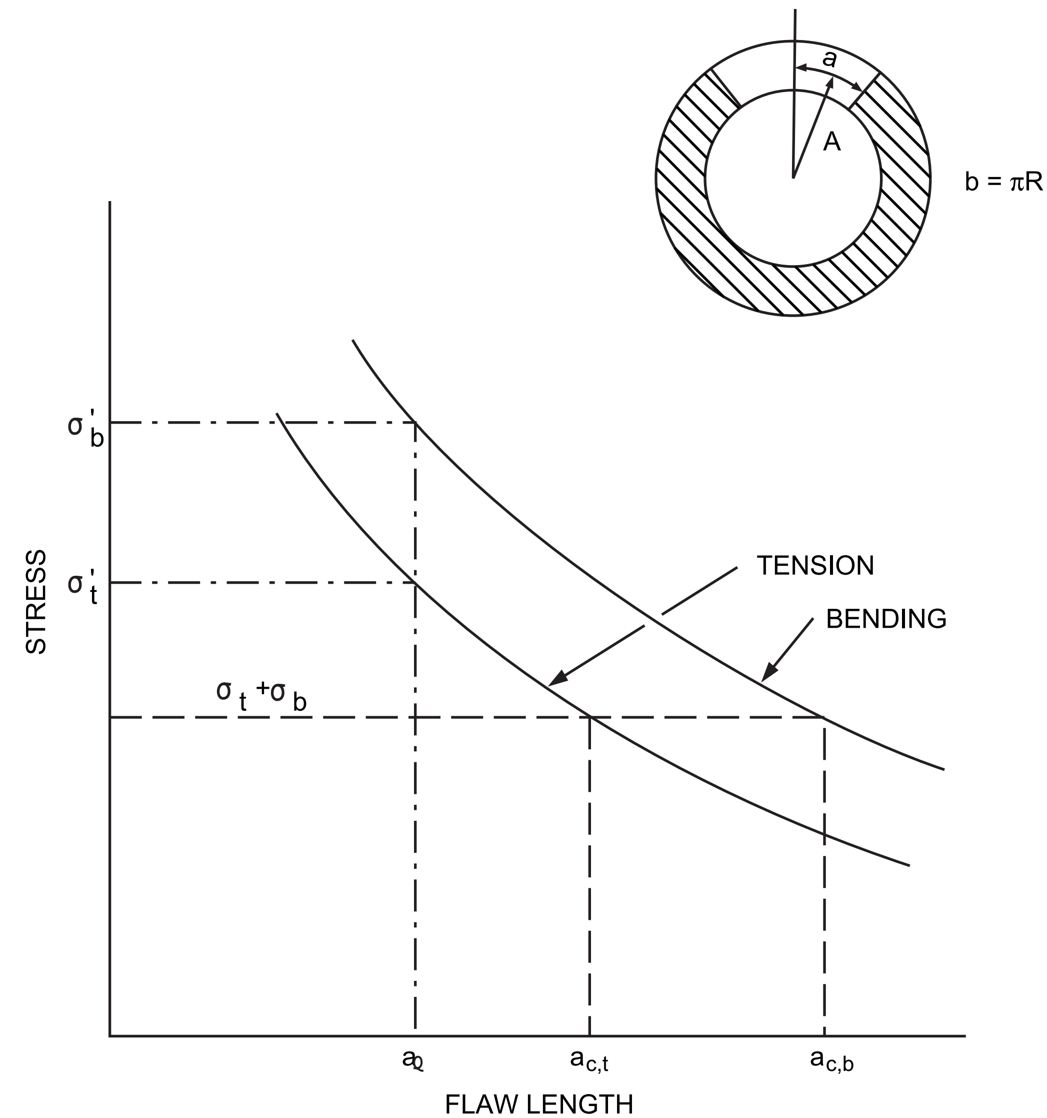

Figure 3E-15 A Schematic Representation of Instability Tension and Bending Stresses as a Function of Flaw Strength

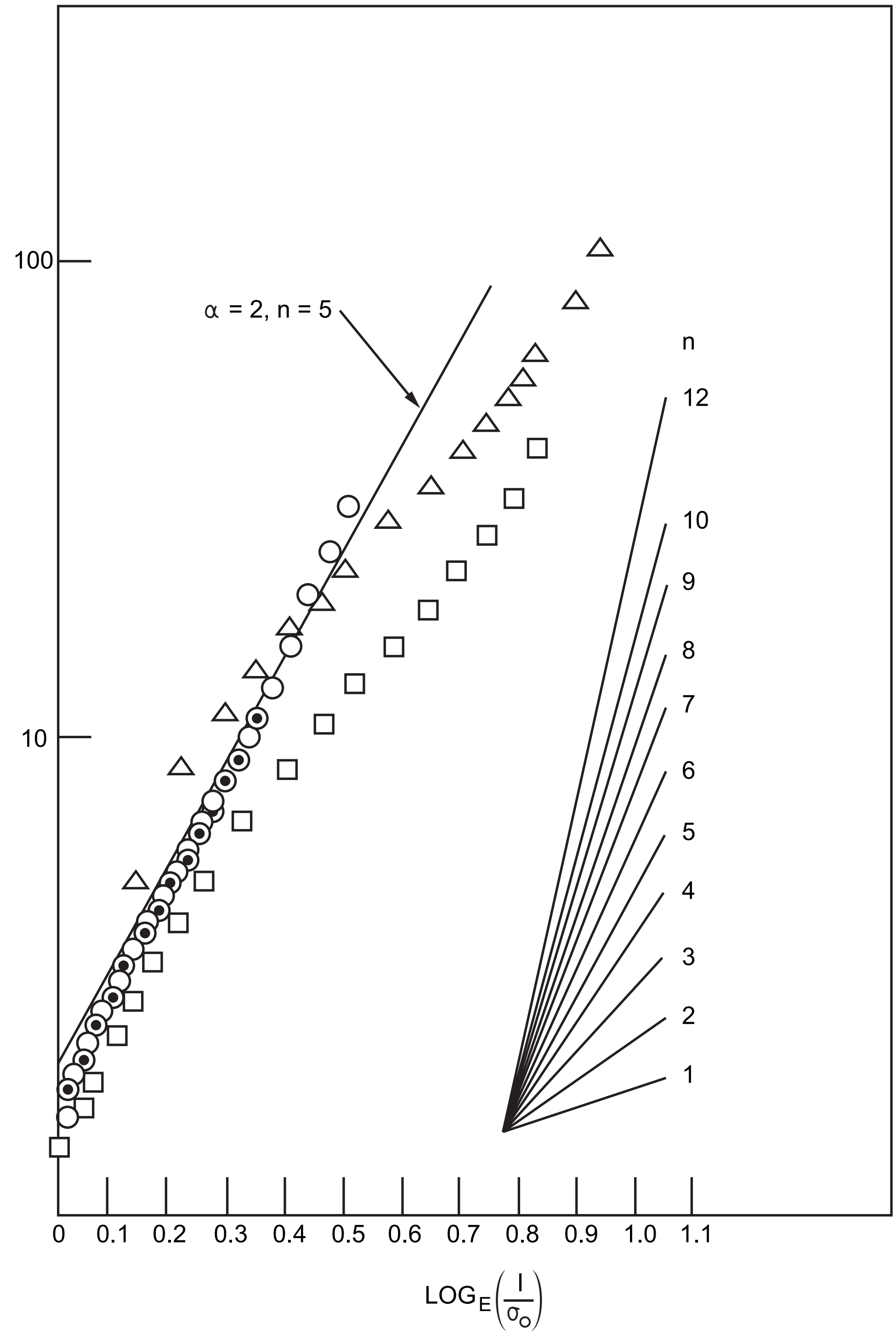

Figure 3E-16 SA 333 Grade 6 Stress-Strain Data at 288°C in the Ramberg-Osgood Format

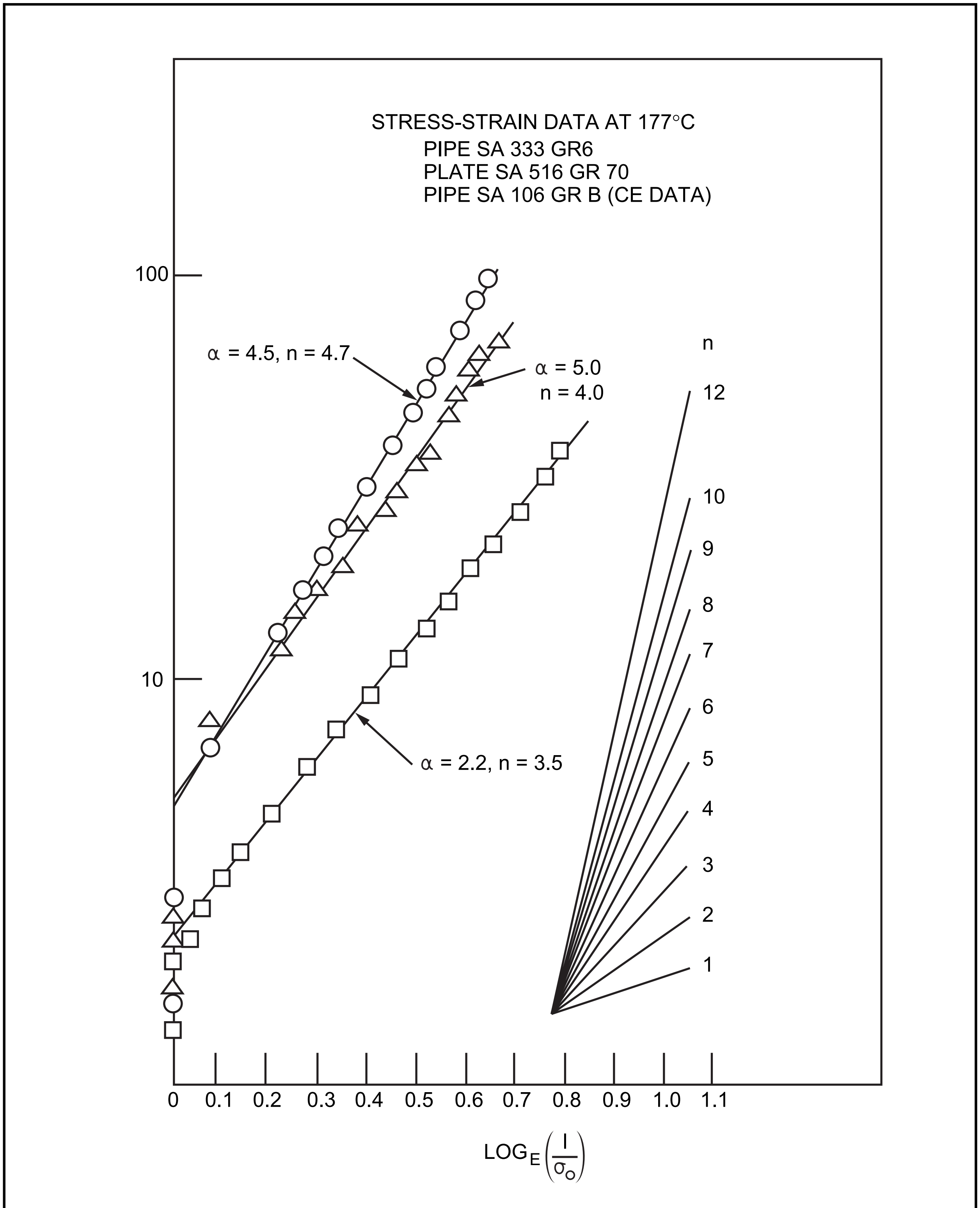

Figure 3E-17
Carbon Steel Stress-Strain Data at 177°C in the Ramberg-Osgood Format

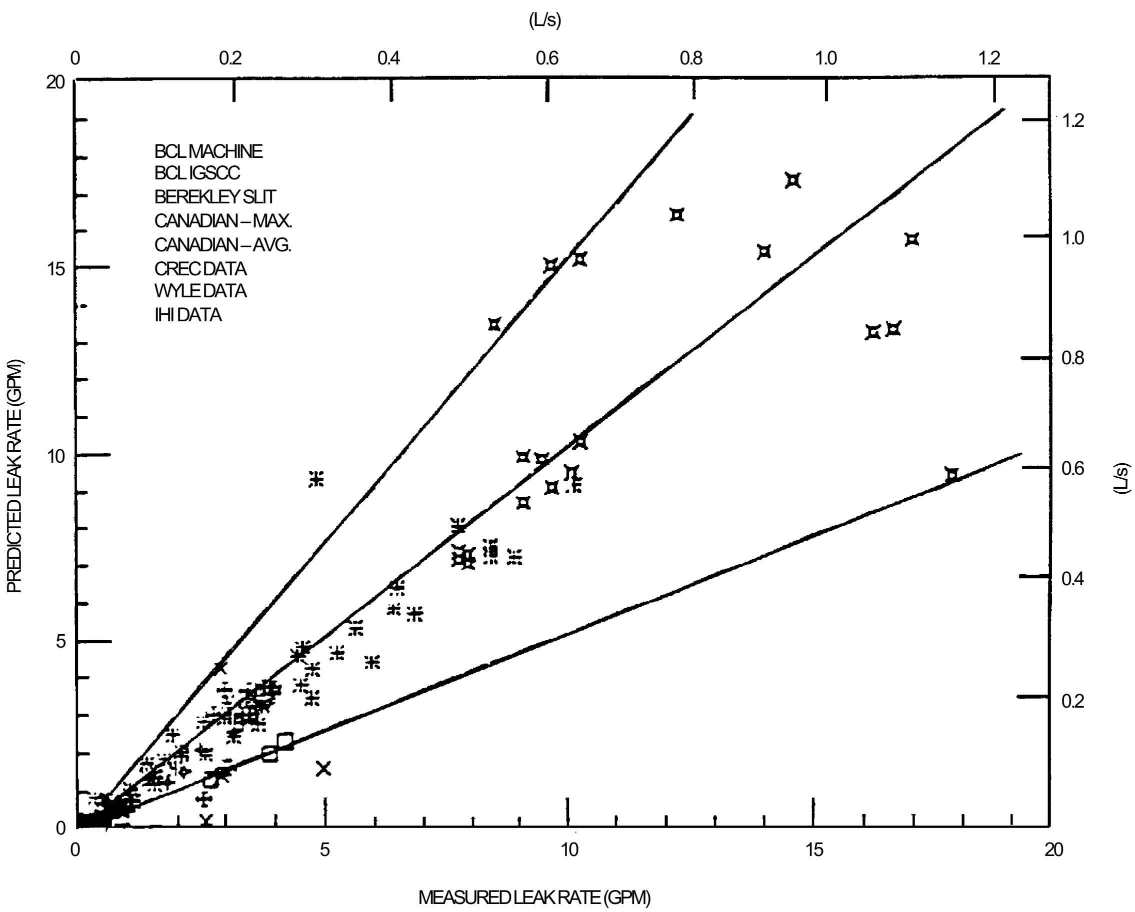

Figure 3E-18 Comparison of PICEP Predictions with Measured Leak Rates

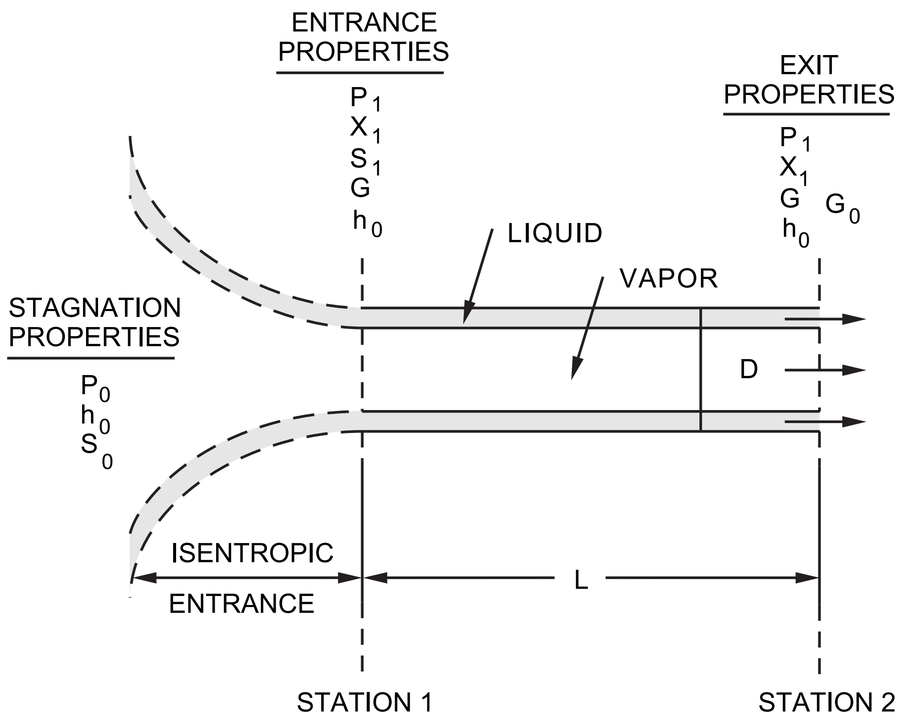

Figure 3E-19 Pipe Flow Model

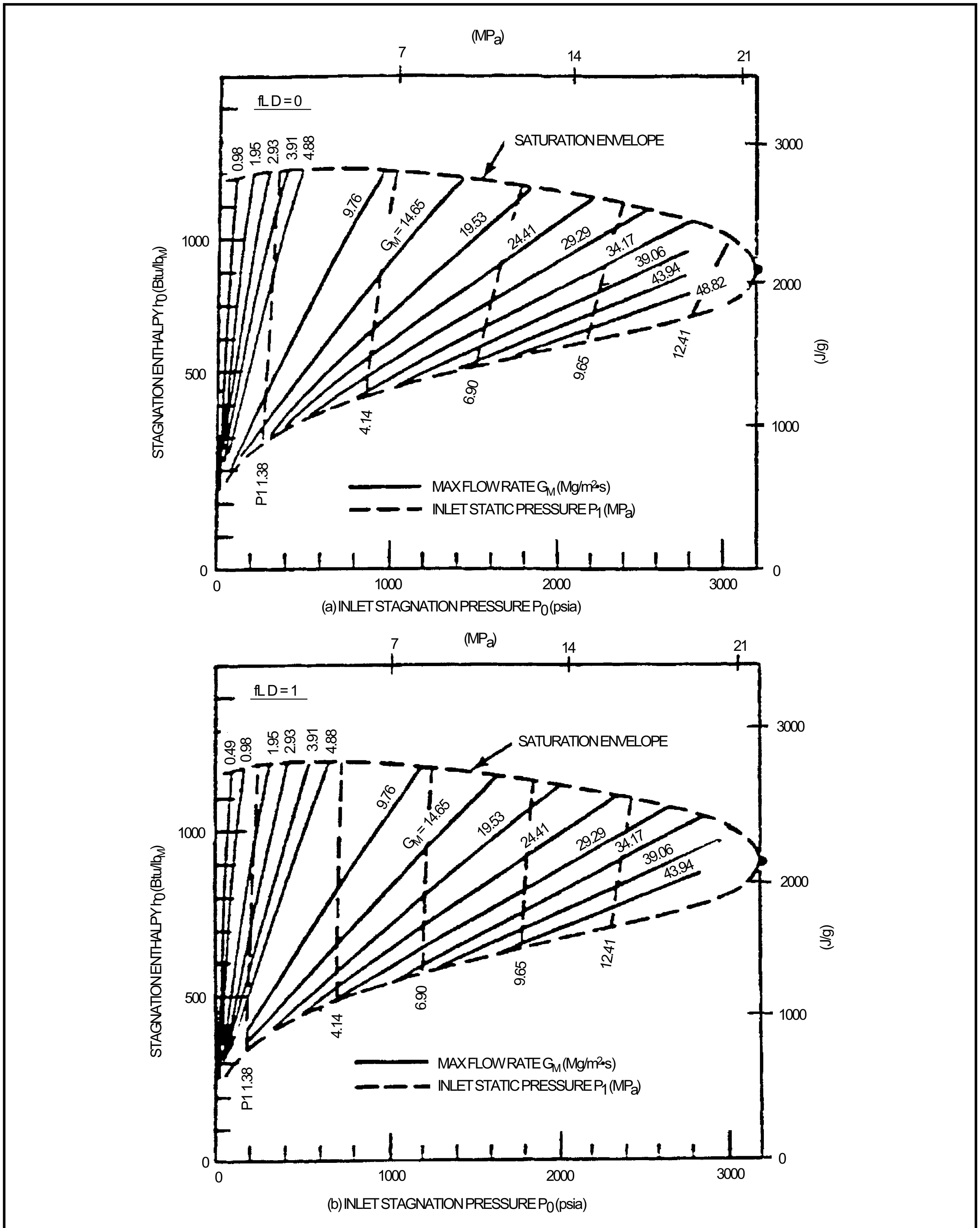

Figure 3E-20 Mass Flow Rates for Steam/Water Mixtures

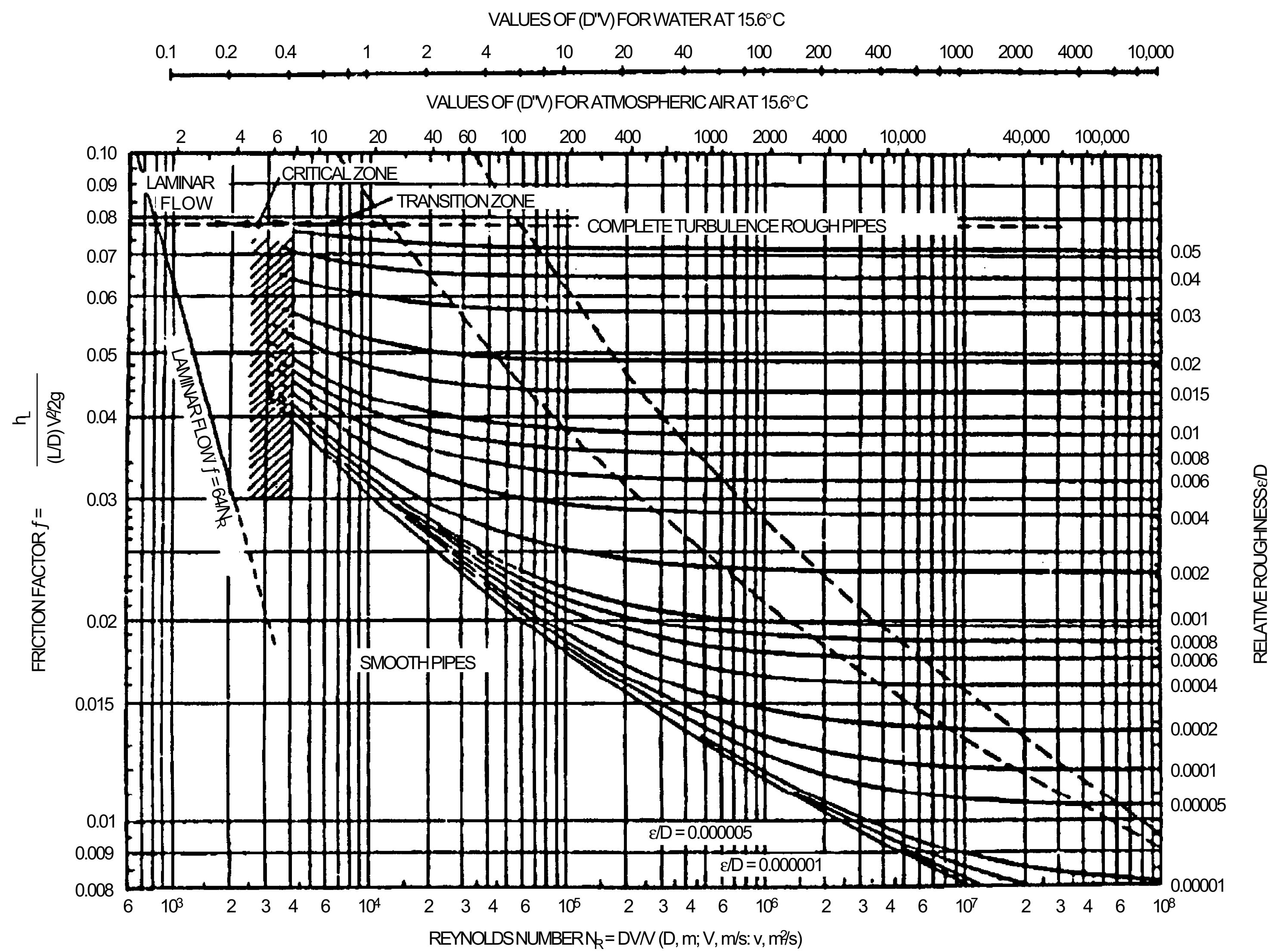

Figure 3E-21 Friction Factors for Pipes

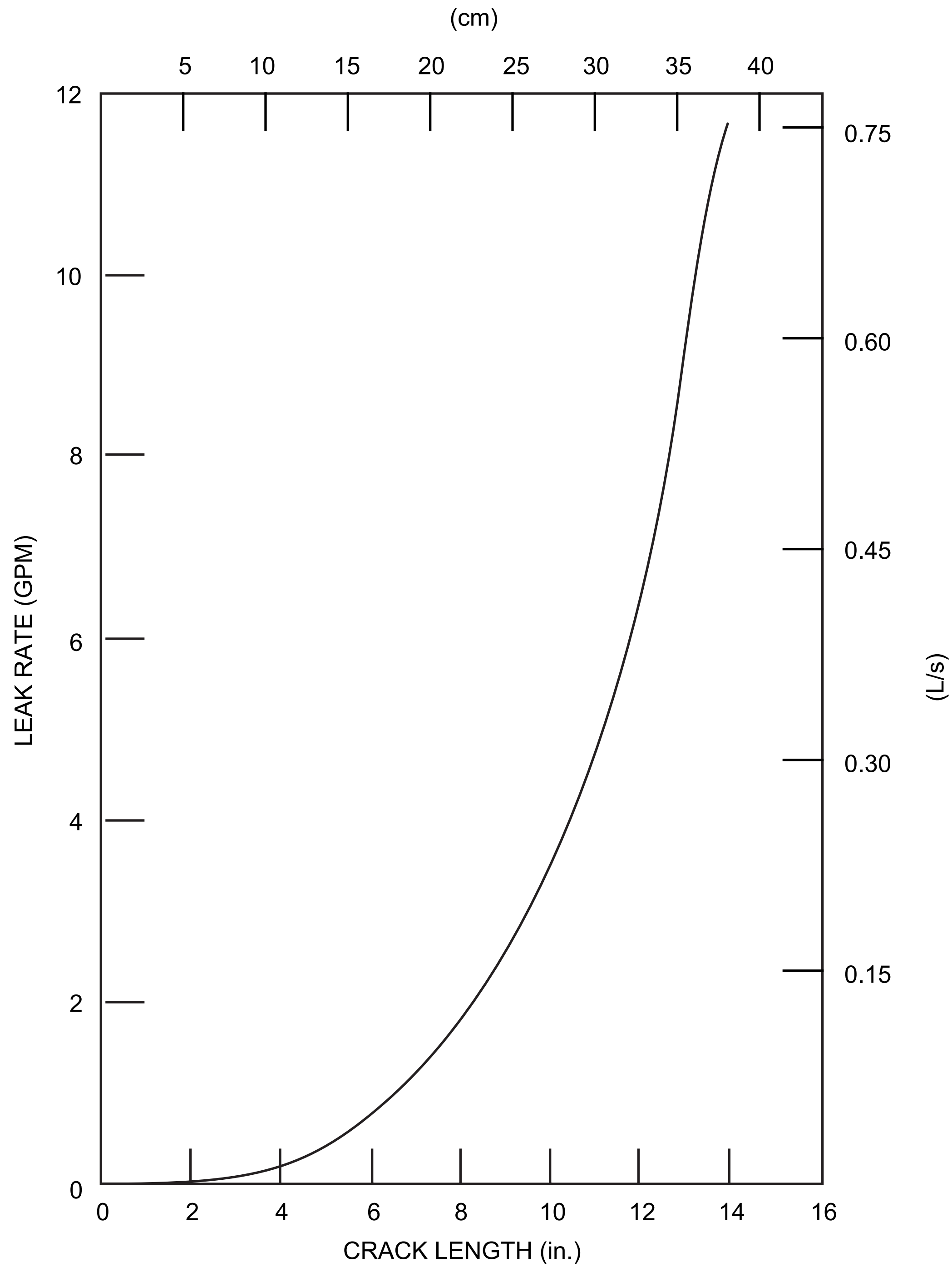

Figure 3E-22
Leak Rate as a Function of Crack Length in Main Steam Pipe (Example)

3F Not Used

3G Response of Structures to Containment Loads

3G.1 Scope

This appendix specifies the design for safety-related structures, systems, and components as applicable due to dynamic excitations originating in the primary containment in the event of operational transients and LOCA. The input containment loads are described in Appendix 3B. The containment loads considered for structural dynamic response analysis are Condensation Oscillation (CO), Pool Chugging (CH), Horizontal Vent Chugging (HV), Safety/Relief Valve discharge (SRV), and Annulus Pressurization (AP).

3G.2 Dynamic Response

3G.2.1 Classification Of Analytical Procedure

Analytical procedure of hydrodynamic loads is classified into following three groups:

- Pipe nozzle break loads
- Symmetric loads
- Asymmetric loads

3G.2.2 Analysis Models

(1) Analysis Model

The structural models used in the analyses represent a synthesis of the reactor building model and the RPV & Internals model. The beam model used in the pipe break load analysis is illustrated in Figure 3G-1. The analysis model of the building structure is illustrated in Figure 3G-2 which is coupled with the RPV model shown in Figure 3G-3 for symmetric load cases and with the RPV model shown in Figure 3G-1 for asymmetric load cases.

(2) Structural Damping

Regulatory Guide 1.61 damping values were used for SRV and LOCA loads.

3G.2.3 Load Application

(1) Pipe Break Nozzle Load

The AP pressures were converted to horizontal forces according to the following formula.

For RSW side:

$$Fj(t) = 2\sum_{i=1}^{8} Pij(t)\int_{\theta=ai}^{\theta=bi} R\cos(\theta)d\theta \qquad (3G\text{-}1)$$

For RPV side:

$$Fj(t) = -2\sum_{i=1}^{8} Pij(t)\int_{\theta=ai}^{\theta=bi} r\cos(\theta)d\theta \qquad (3G\text{-}2)$$

Fj(t)	=	Force per unit height each level
Pij(t)	=	Pressure each level and angle
i	=	Cell No.
j	=	Level No.
R	=	RSW Inner Radius
r	=	RPV Outer Radius
l	=	Angle (180°)

Jet reaction, jet impingement, and pipe whip reaction forces were considered as steady state loads whose rise time for initial pulse was set as one millisecond.

(2) SRV Load

Symmetric SRV (all) response analysis is covered by n=0 harmonic. Asymmetric case of SRV (all) actuation is covered by n=1 harmonic that corresponds to overturning moment. SRV (1) is also in this category. The SRV air bubble frequencies are expected to be within a range of 5 to 12 Hz. Ways of selecting minimum number of bubble frequencies for dynamic analysis are as follows.

Frequency range of SRV Loads: $f_1 \le f \le f_2$ (f_1 = 5 Hz, f_2 = 12 Hz)

For vertical structural frequencies (n=0):

(a) If $(f_s)_v > f_2$ then use f_2

(b) If $f_1 < (f_s)_v < f_2$ then use $(f_s)_v$

For horizontal structural frequencies (n=1):

(a) If $(f_s)_h < f_1$ then use f_1

(b) If $f_1 < (f_s)_h < f_2$ then use $(f_s)_h$

(c) If $f_2 < (f_s)_h$ then use f_2

In symmetric load case, 12 Hz was adopted as bubble frequency, because the vertical frequencies of the structure were higher than 12 Hz. In asymmetric load case, 3 horizontal beam frequencies of the structure within the above range 7.73 Hz, 9.59 Hz, and 11.58 Hz were adopted as bubble frequencies.

(3) HV Load

Both symmetric and non-symmetric upward loads on pedestal due to chugging in the top horizontal vents was considered.

(4) Chugging, Condensation Oscillation Loads

According to the study of the natural frequencies of the structure and the frequencies of the input motion, 4 critical pressure time histories out of 8 for CH and 2 out of 4 for CO, were selected for dynamic analysis. Furthermore, 1 local spike load was added in CO response study.

3G.2.4 Analysis Method

(1) Pipe Nozzle Break Load Analysis

For pipe nozzle break cases, multi-input excitation time history analyses were performed by using mode superposition method. Strain energy damping was used for this analyses.

(2) Symmetric Load Analysis

For symmetric load cases, frequency response method for n=0 harmonic was used. Hysteresis damping was considered.

(3) Asymmetric Load Analysis

For asymmetric load cases, frequency response method for n=1 harmonic was used. Hysteresis damping was considered.

(4) Analysis Parameters

The analysis parameters in terms of time/frequency steps in analysis are shown in Table 3G-1.

3G.3 Hydrodynamic Load Analysis Results

The acceleration response spectra at a few selected locations for each loading event are presented in Figures 3G-4 through 3G-108. The maximum displacements and accelerations at a few selected locations for each loading event are presented Tables 3G-2 through 3G-5.

The input excitation of suppression pool boundary horizontal loads (SRV, Chugging, and HV) was considered unidirectional which can be set at any direction in the horizontal plane, and the AP analysis was performed assuming that pipe break can be associated with any one of the vessel nozzles for each of the postulated line breaks.

The resulting response of structures considered in the analyses is thus unidirectional applicable to any azimuth angle for suppression pool loads and to the horizontal direction corresponding to the break direction for AP loads.

For subsystem analyses using floor response spectra and, if applicable, building displacement data, the input direction of the horizontal load shall be selected to result in worst subsystem response.

As an alternate approach, the horizontal input to subsystem may be taken to be the same in the two orthogonal horizontal directions.

The SRV (one) loads can be obtained by multiplying the SRV (all) loads by 0.4 and 0.3 in the horizontal and vertical directions respectively.

Table 3G-1 Analysis Parameters in Terms of Time/Frequency Steps

Load	Analysis Case	Domain Time/ Freq.	Input Time Pitch ýt (s)	Given Time Step n_1	Given Duration Time $t_1(=n_1 \cdot ýt)$ (s)	Trailing Zero Step n_2	Analysis Time Step $N(=n_1+n_2)$	Analysis Duration Time T(=Nýt) (s)	Frequency Resolution 1/T (Hz)	Transfer Function Interpolation Method: Interval	Transfer Function Interpolation Method: Method
Pipe Break											
	Definition of AP	Time	1.0×10^{-3}	2000	2.0	—	—	—	—	—	—
	Definition of F_1 ~ F4	Time	1.0×10^{-3}	2000	2.0	—	—	—	—	—	—
	A P A 1, A P A 2 A P B 1, A P B 2 A P C 1, A P C 2	Time	1.0×10^{-3}	2000	2.0	—	2000	2.0	—	—	—
SRV											
	Definition	Freq.	5×10^{-3}	150	0.75	—	—	—	—	—	—
	SRVV	Freq.	3.333×10^{-3}	150	0.50	1898	2048	6.83	0.14649	Every Step	—
	SRVH1	Freq.	4.170×10^{-3}	150	0.625	1898	2048	8.54	0.11710	Every 2 Step	Reciprocal
	SRVH2	Freq.	3.454×10^{-3}	150	0.518	1898	2048	7.074	0.14136	Every Step	—
	SRVH3	Freq.	5.175×10^{-3}	150	0.776	1898	2048	10.598	0.09436	Every 2 Step	Reciprocal
HV											
	Definition	Freq.	1×10^{-3}	2	0.002	—	—	—	—	—	—
	HVV	Freq.	1×10^{-3}	2	0.002	4094	4096	4.096	0.24414	Every Step	—
	HVH	Freq.	1×10^{-3}	2	0.002	4094	4096	4.096	0.24414	Every Step	—

Table 3G-1 Analysis Parameters in Terms of Time/Frequency Steps (Continued)

Load	Analysis Case	Domain Time/ Freq.	Input Time Pitch ýt (s)	Given Time Step n_1	Given Duration Time t_1(=n_1•ýt) (s)	Trailing Zero Step n_2	Analysis Time Step N(=n_1+n_2)	Analysis Duration Time T(=Nýt) (s)	Frequency Resolution 1/T (Hz)	Transfer Function Interpolation Method Interval	Transfer Function Interpolation Method Method
CH											
	Definition	Freq.	$1x10^{-3}$	1000	1.0	—	—	—	—	—	—
	CHV1, CHH1 CHV2, CHH2 CHV3, CHH3 CHV4, CHH4	Freq.	$1x10^{-3}$	1000	1.0	3096	4096	4.096	0.24414	Every Step	—
CO											
	COV1 Definition	Freq.	$1x10^{-3}$	2000	2.0	—	—	—	—	—	—
	COV3 Analysis	Freq.	$1x10^{-3}$	2000	2.0	2096	4096	4.096	0.24414	Every Step	—
	COV2 Definition	Freq.	$1x10^{-3}$	3000	3.0	—	—	—	—	—	—
	Analysis	Freq.	$1x10^{-3}$	3000	3.0	1096	4096	4.096	0.24414	Every Step	—

Table 3G-2 Maximum Accelerations for AP Loadings (g)

Location	Node	MSLB	FDW	RHR
Top of RPV	28	1.246	0.616	1.015
Top of Pedestal	86	0.146	0.079	0.107
Top of RSW	80	4.84	0.729	1.045
D/F Slab	85	0.233	0.065	0.117

Table 3G-3 Maximum Accelerations for Hydrodynamic Loads (g)

Location	Direction	Node	SRV	HV	CH	CO
Top of RPV	Horizontal	28	0.173	0.029	0.116	
	Vertical	1	0.273	0.002	0.057	0.517
Top of Pedestal	Horizontal	71	0.116	0.006	0.051	
	Vertical	42	0.216	0.003	0.020	0.217
Top of RSW	Horizontal	165	0.119	0.016	0.063	
	Vertical	165	0.220	0.008	0.065	0.475
D/F Slab	Horizontal	153	0.061	0.004	0.038	
	Vertical	157	0.147	0.002	0.031	0.303

Table 3G-4 Maximum Displacements for AP Loadings (mm)

Location	Node	MSLB	FDW	RHR
Top of RPV	28	1.94	0.74	1.01
Top of Pedestal	86	0.11	0.04	0.06
Top of RSW	80	1.43	0.36	0.56
D/F Slab	85	0.15	0.05	0.07

Table 3G-5 Maximum Displacements for Hydrodynamic Loads (mm)

Location	Direction	Node	SRV	HV	CH	CO
Top of RPV	Horizontal	28	0.110	0.013	0.029	
	Vertical	1	0.470	0.004	0.016	0.674
Top of Pedestal	Horizontal	71	0.069	0.003	0.012	
	Vertical	42	0.420	0.003	0.011	0.636
Top of RSW	Horizontal	165	0.065	0.006	0.001	
	Vertical	165	0.398	0.003	0.013	0.624
D/F Slab	Horizontal	153	0.039	0.002	0.006	
	Vertical	157	0.155	0.001	0.009	0.436

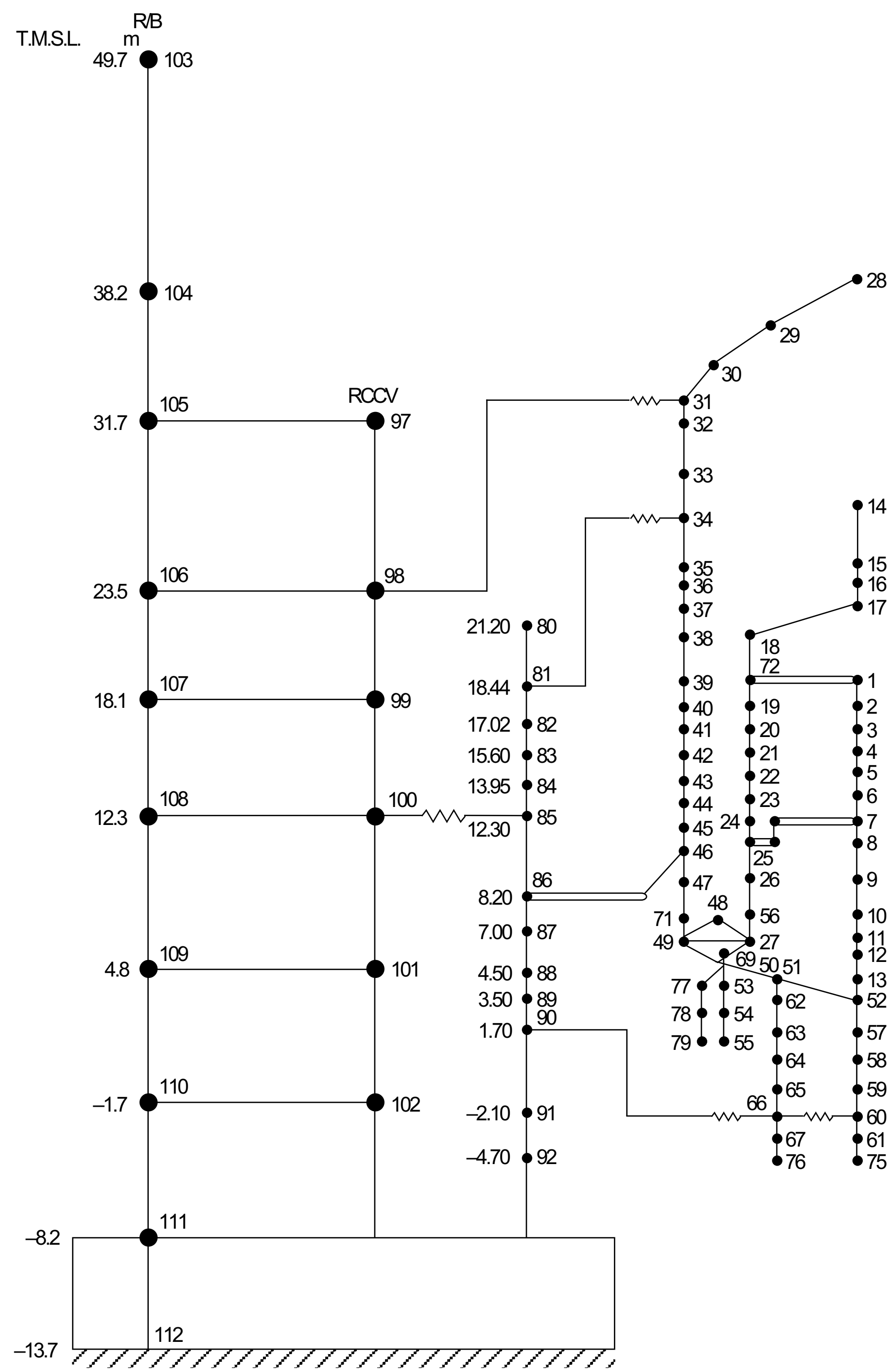

Figure 3G-1 Horizontal Beam Model for AP Load

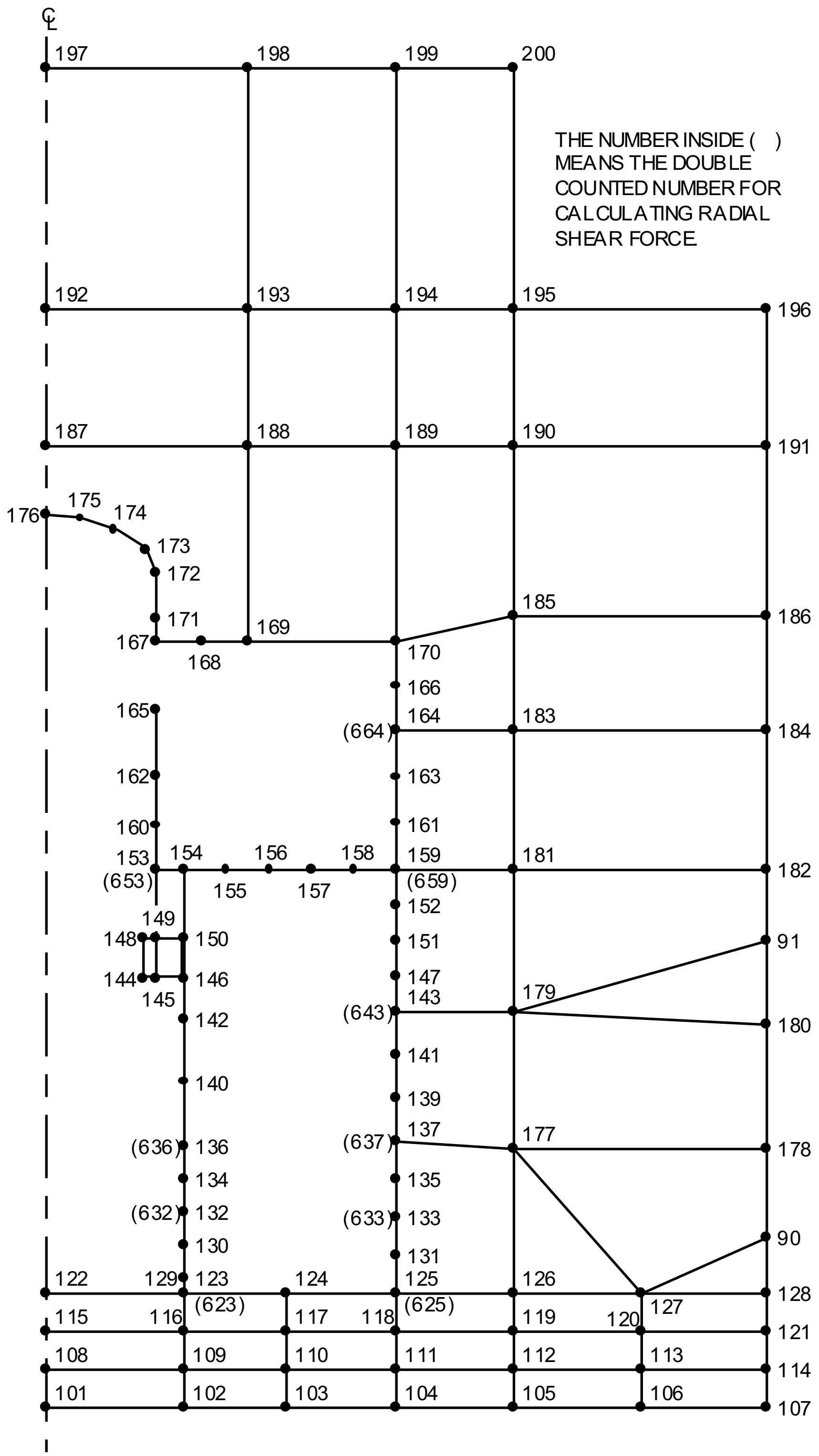

Figure 3G-2 Nodal Point (R/B Horizontal/Vertical Shell Model)

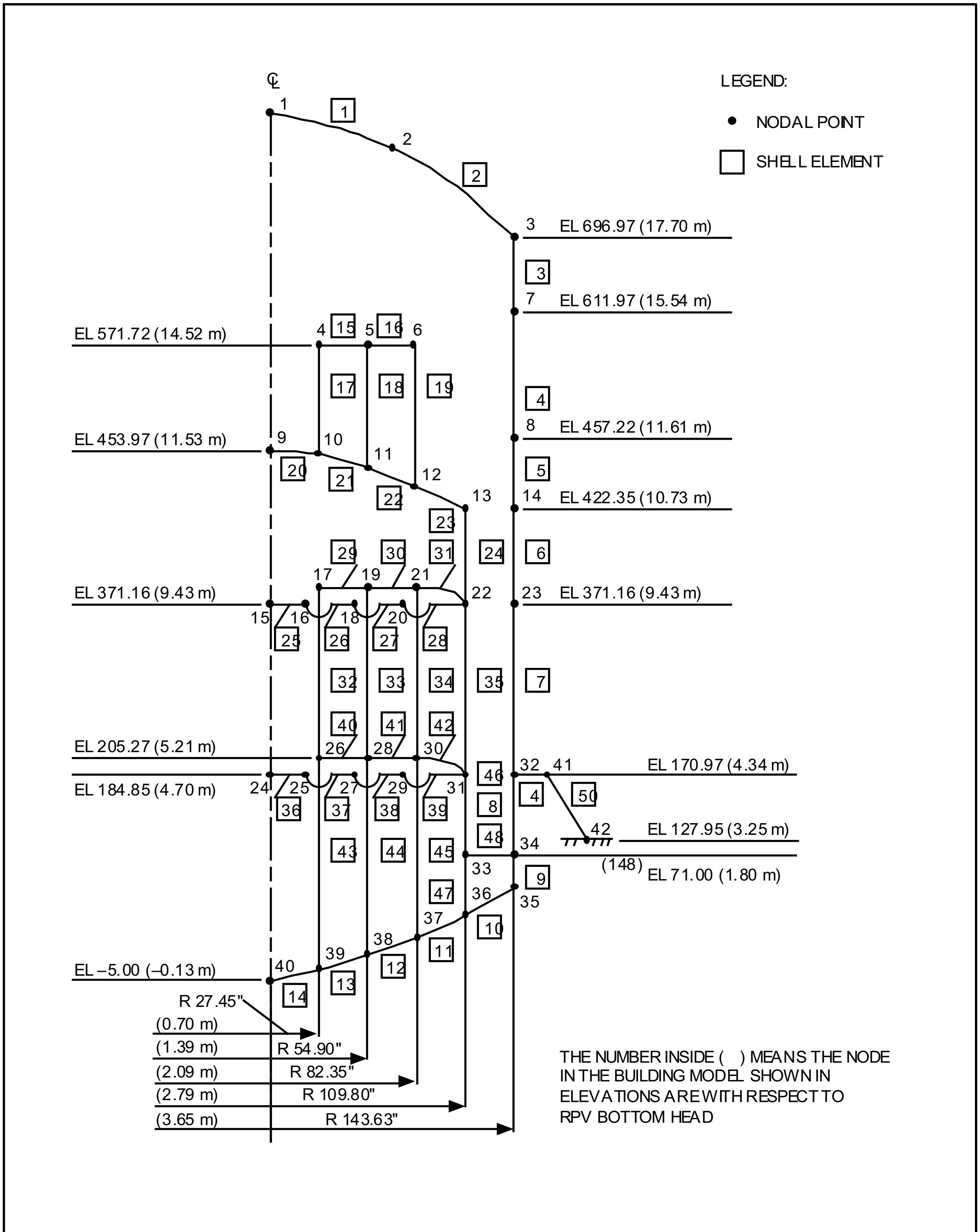

Figure 3G-3 Nodal Point No. (RPV/Internal Vertical Shell Model)

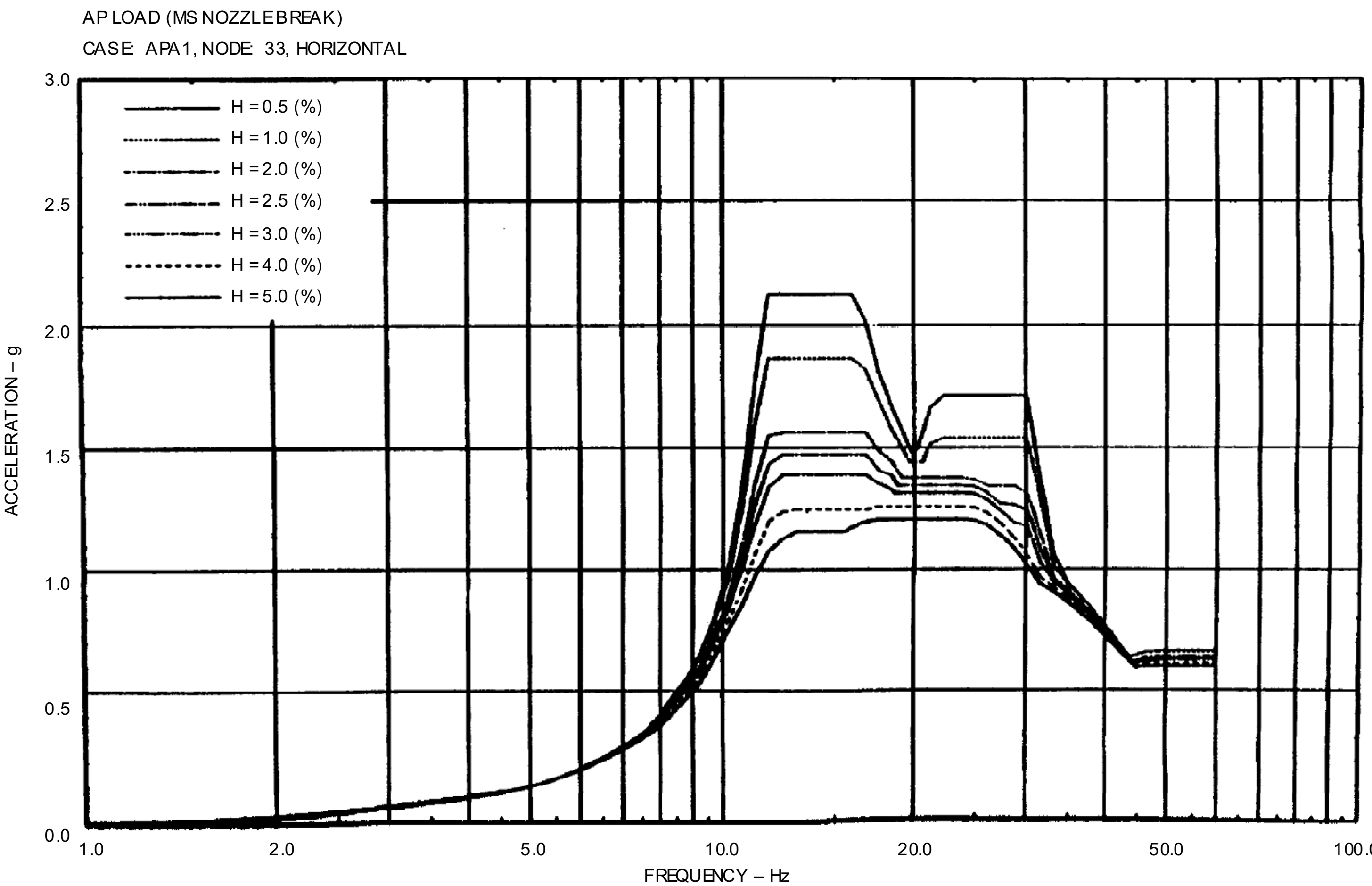

Figure 3G-4 Floor Response Spectrum—Case: APA1, Node: 33, Horizontal

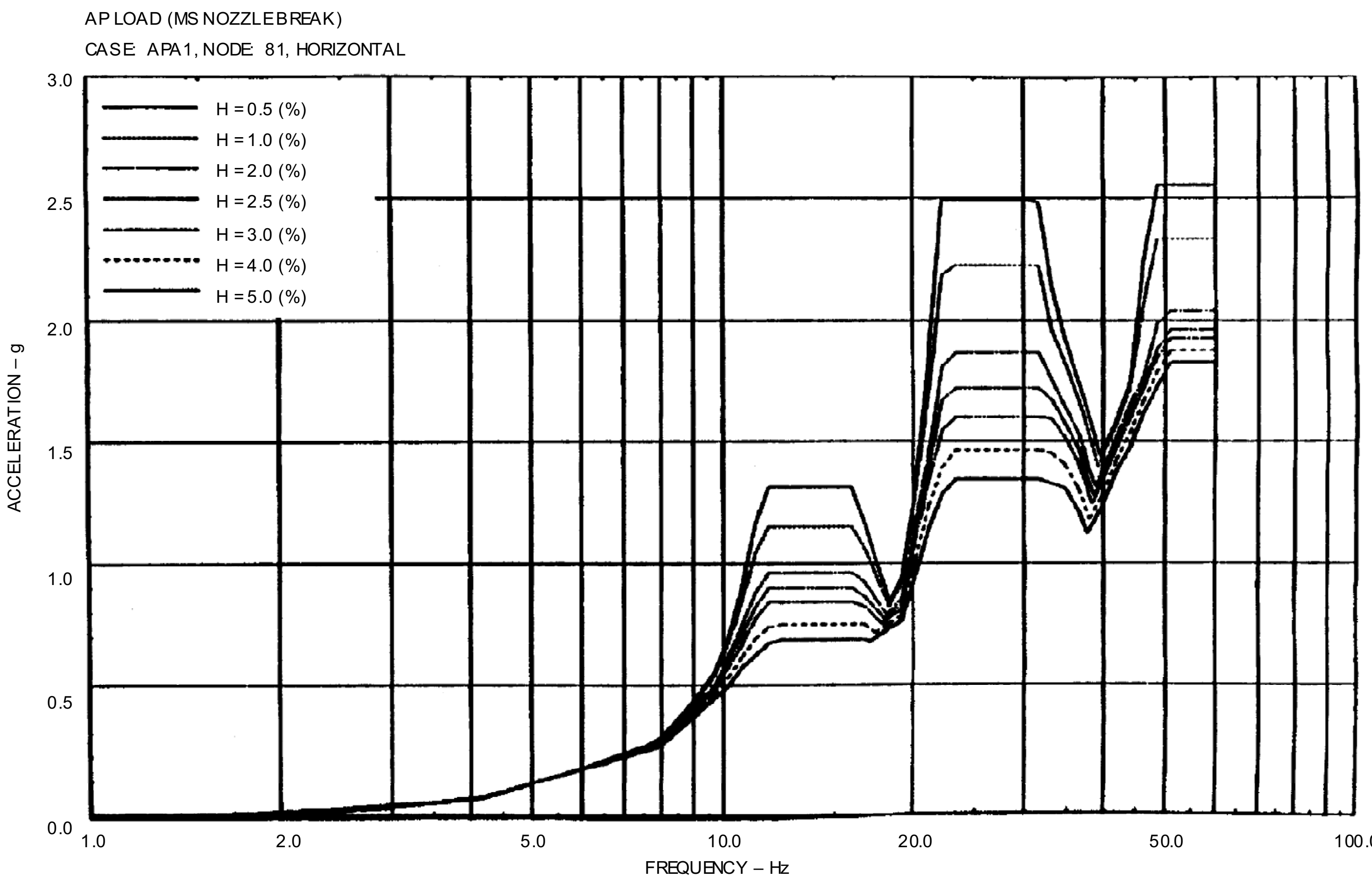

Figure 3G-5 Floor Response Spectrum—Case: APA1, Node: 81, Horizontal

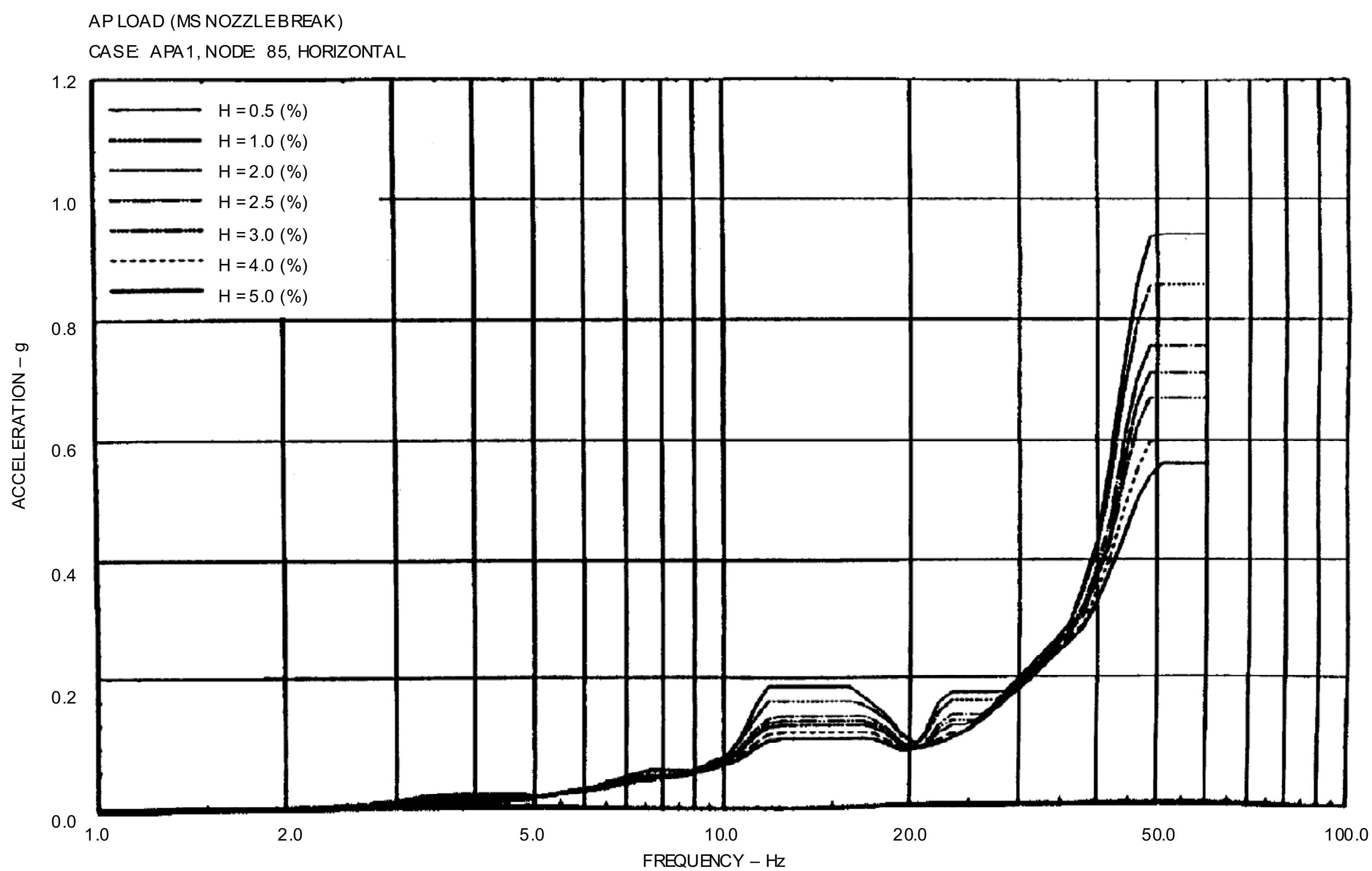

Figure 3G-6 Floor Response Spectrum—Case: APA1, Node: 85, Horizontal

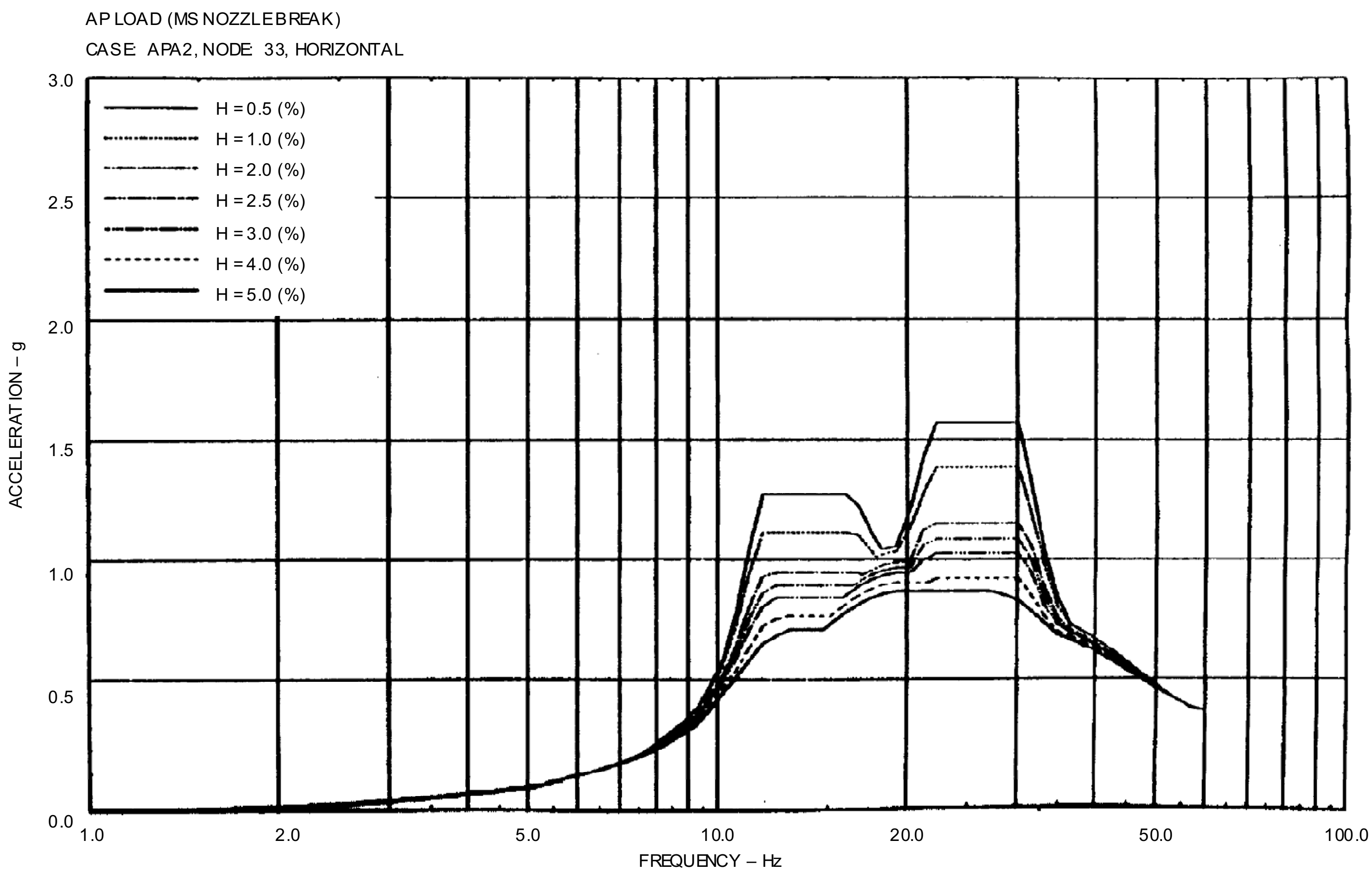

Figure 3G-7 Floor Response Spectrum—Case: APA2, Node: 33, Horizontal

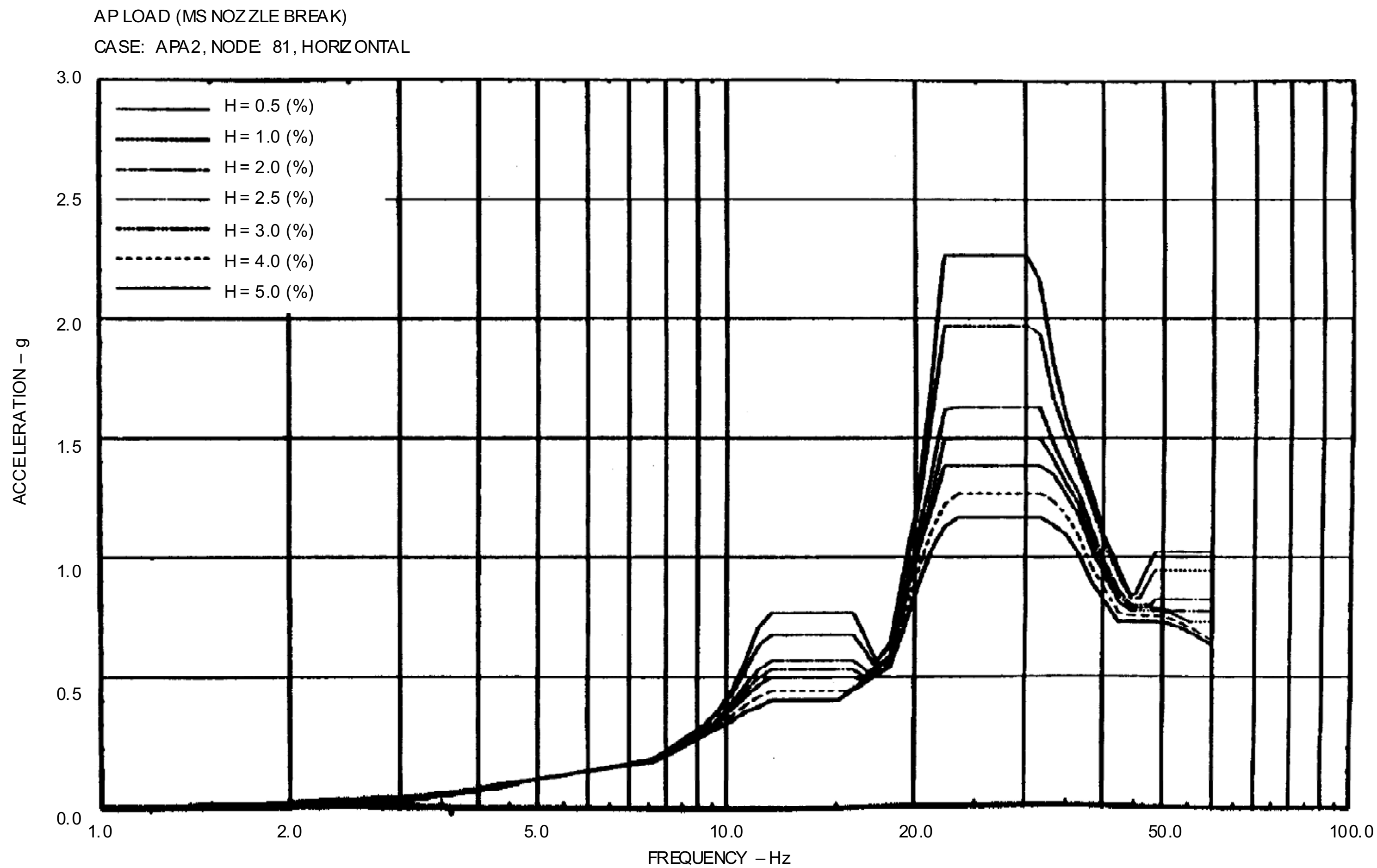

Figure 3G-8 Floor Response Spectrum—Case: APA2, Node: 81, Horizontal

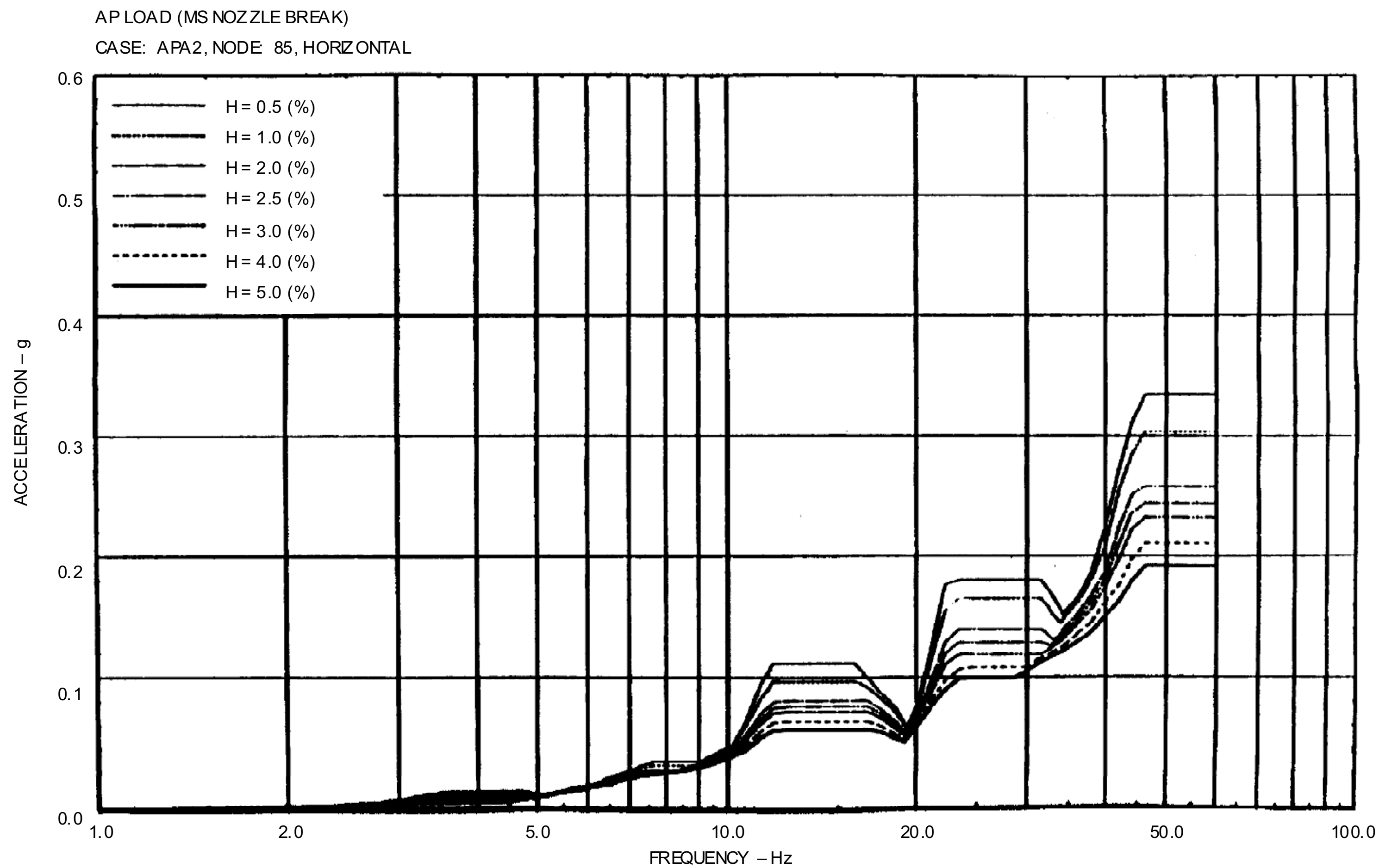

Figure 3G-9 Floor Response Spectrum—Case: APA2, Node: 85, Horizontal

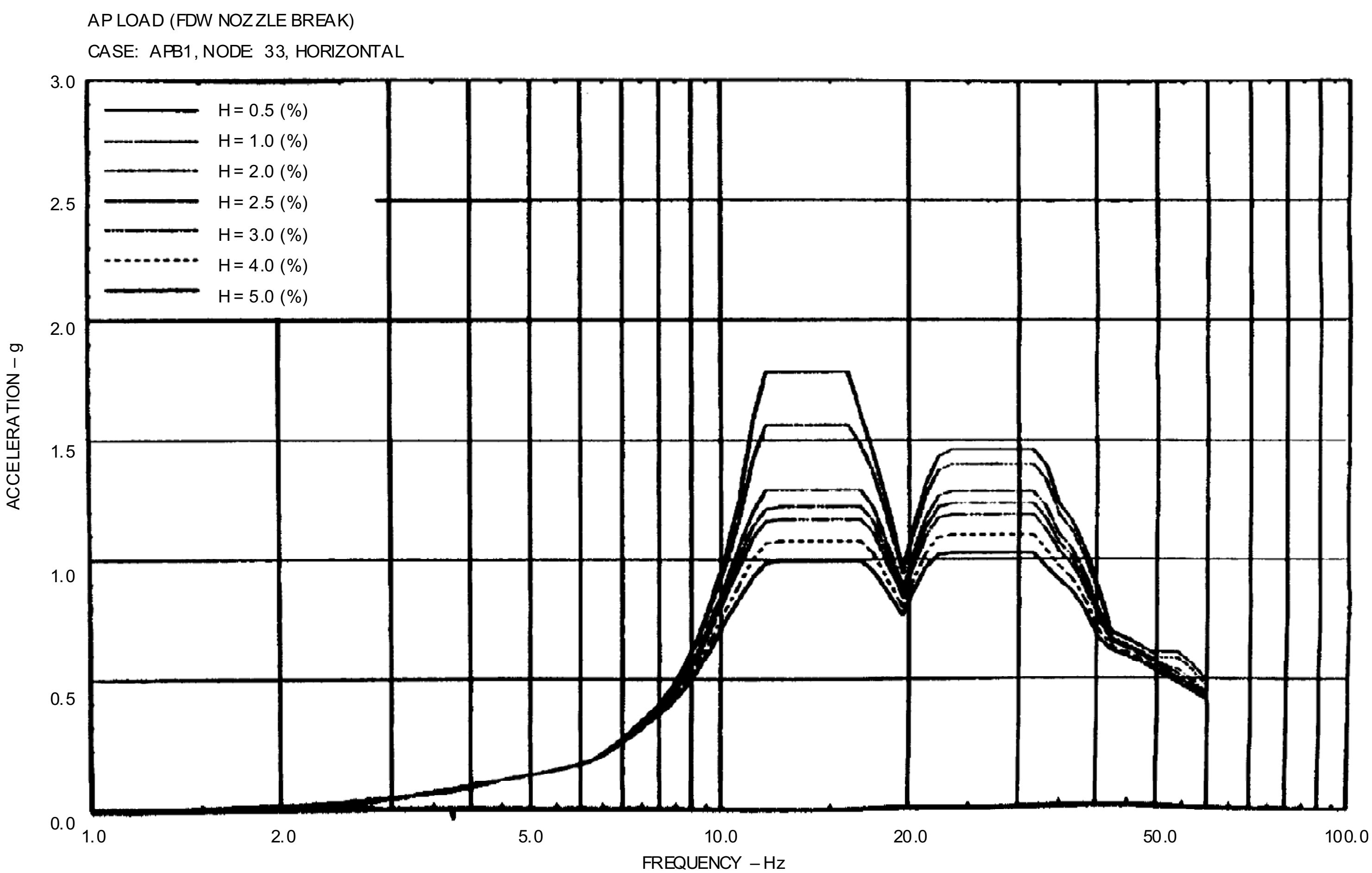

Figure 3G-10 Floor Response Spectrum—Case: APB1, Node: 33, Horizontal

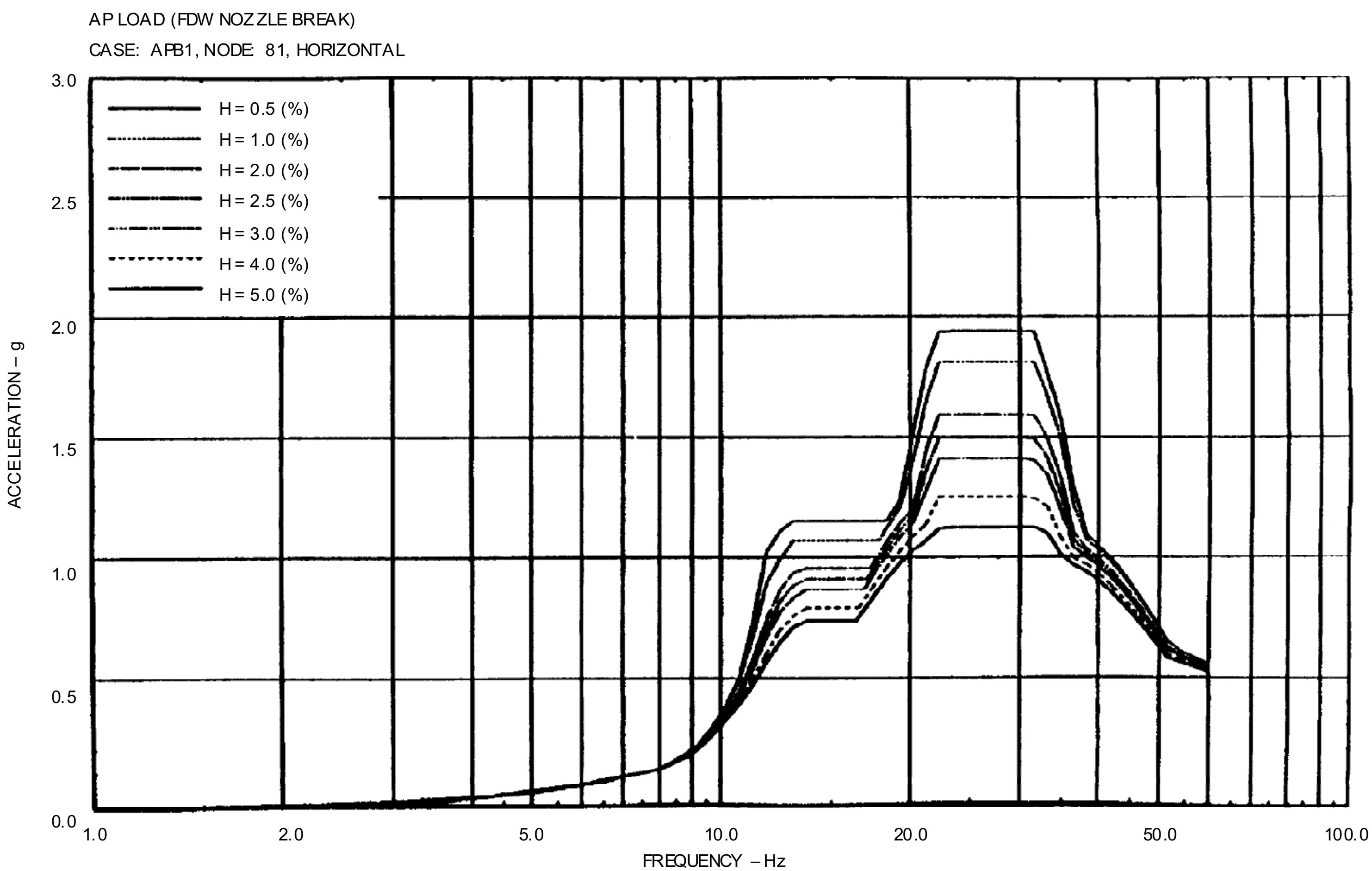

Figure 3G-11 Floor Response Spectrum—Case: APB1, Node: 81, Horizontal

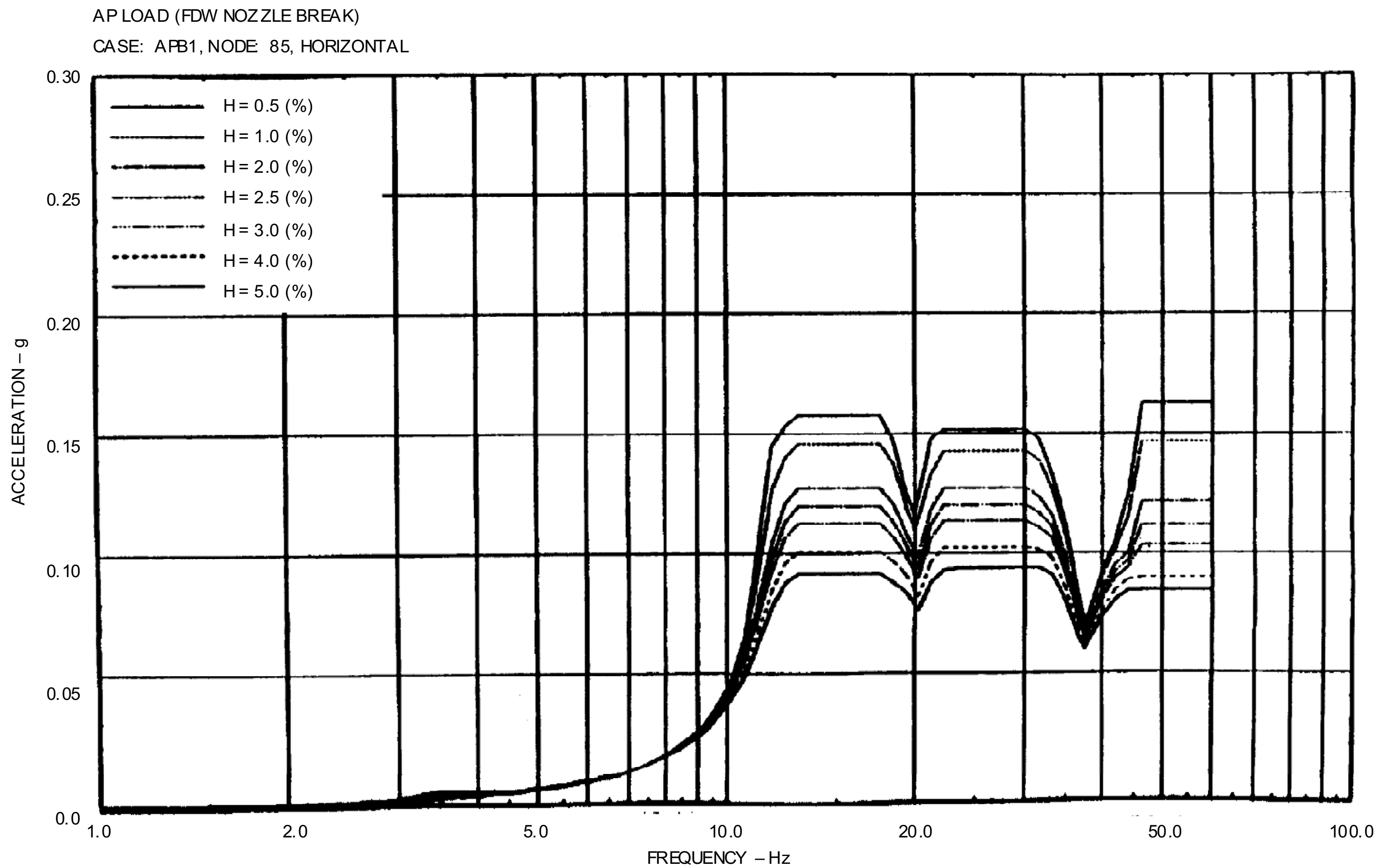

Figure 3G-12 Floor Response Spectrum—Case: APB1, Node: 85, Horizontal

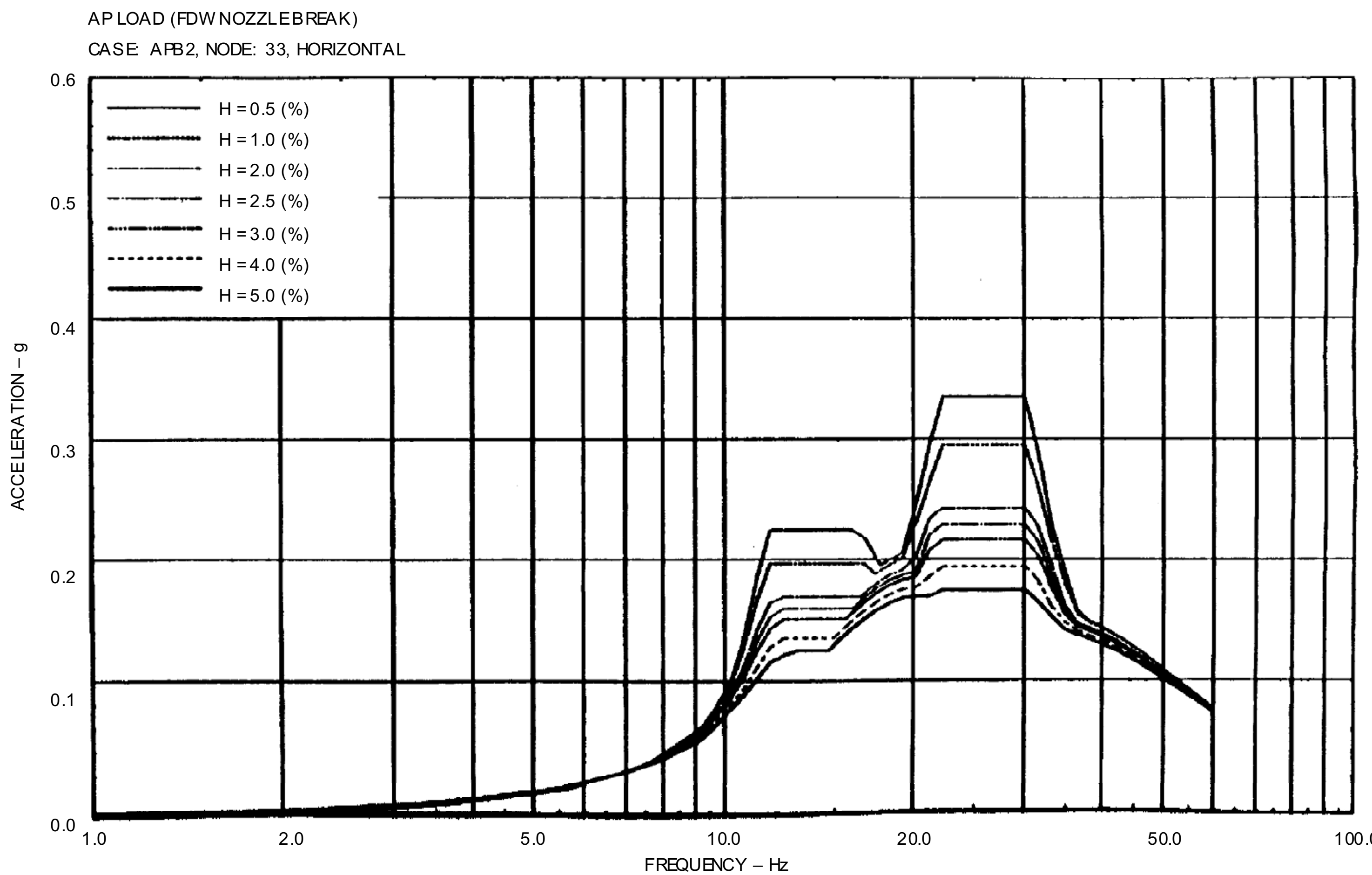

Figure 3G-13 Floor Response Spectrum—Case: APB2, Node: 33, Horizontal

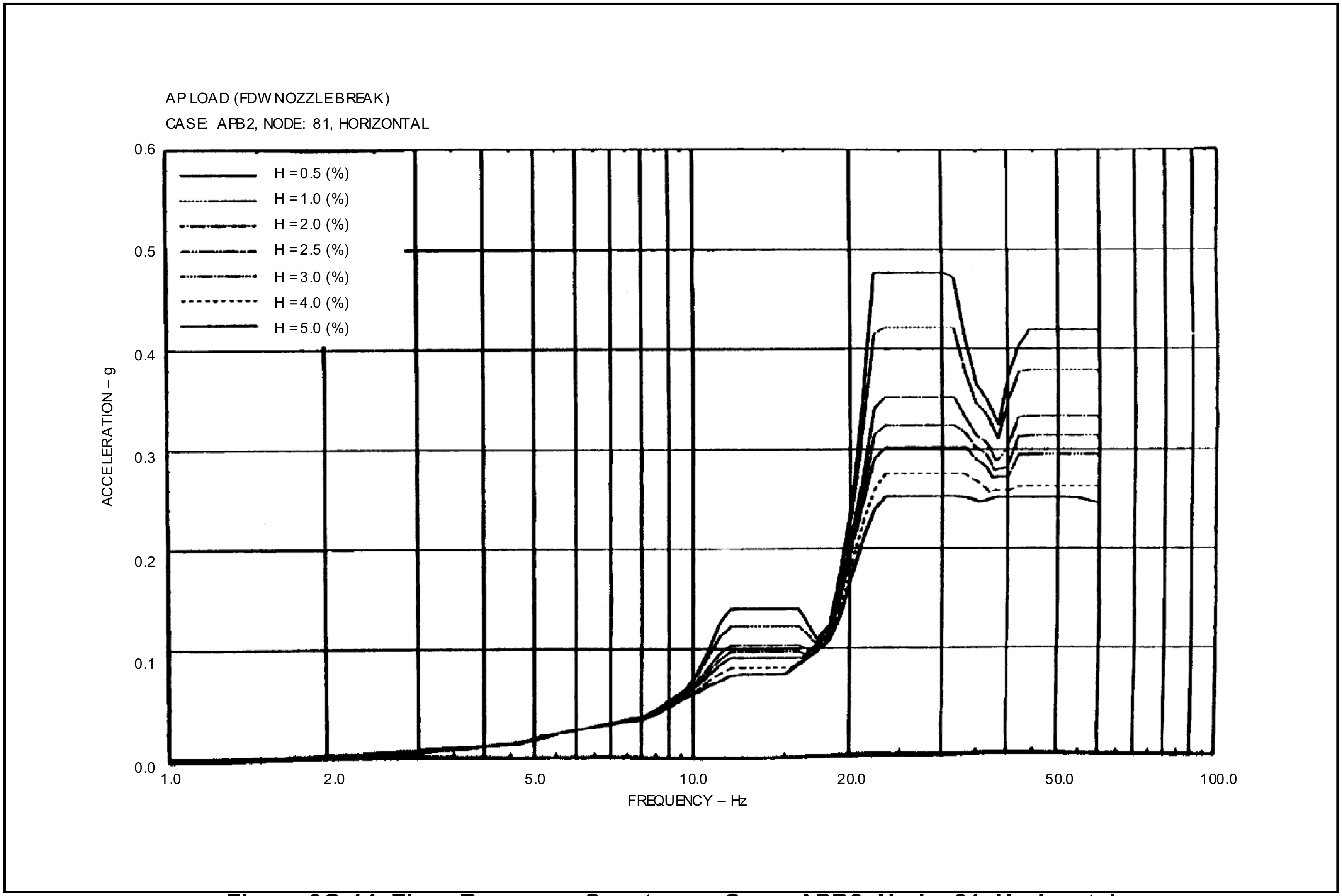

Figure 3G-14 Floor Response Spectrum—Case: APB2, Node: 81, Horizontal

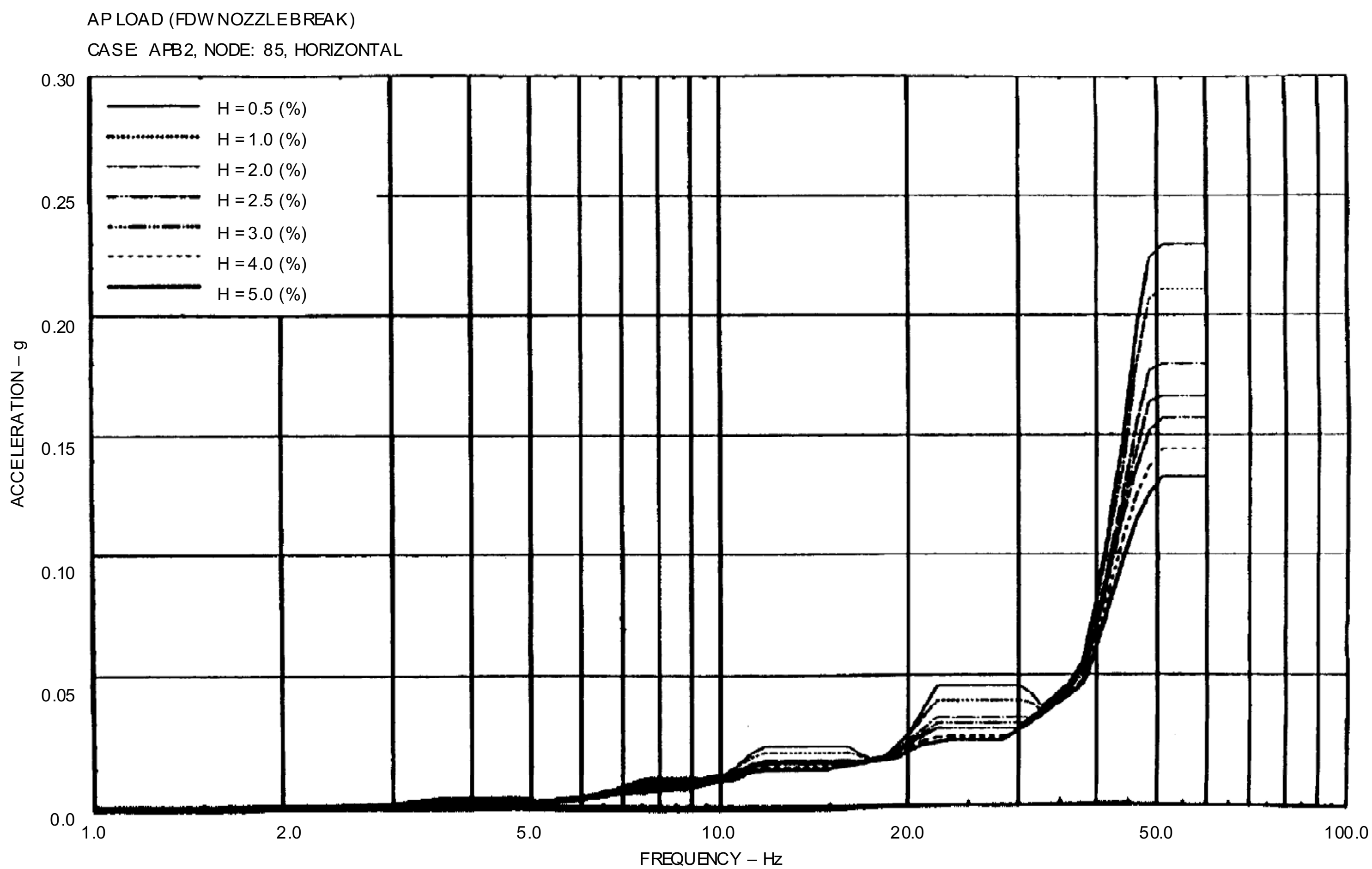

Figure 3G-15 Floor Response Spectrum—Case: APB2, Node: 85, Horizontal

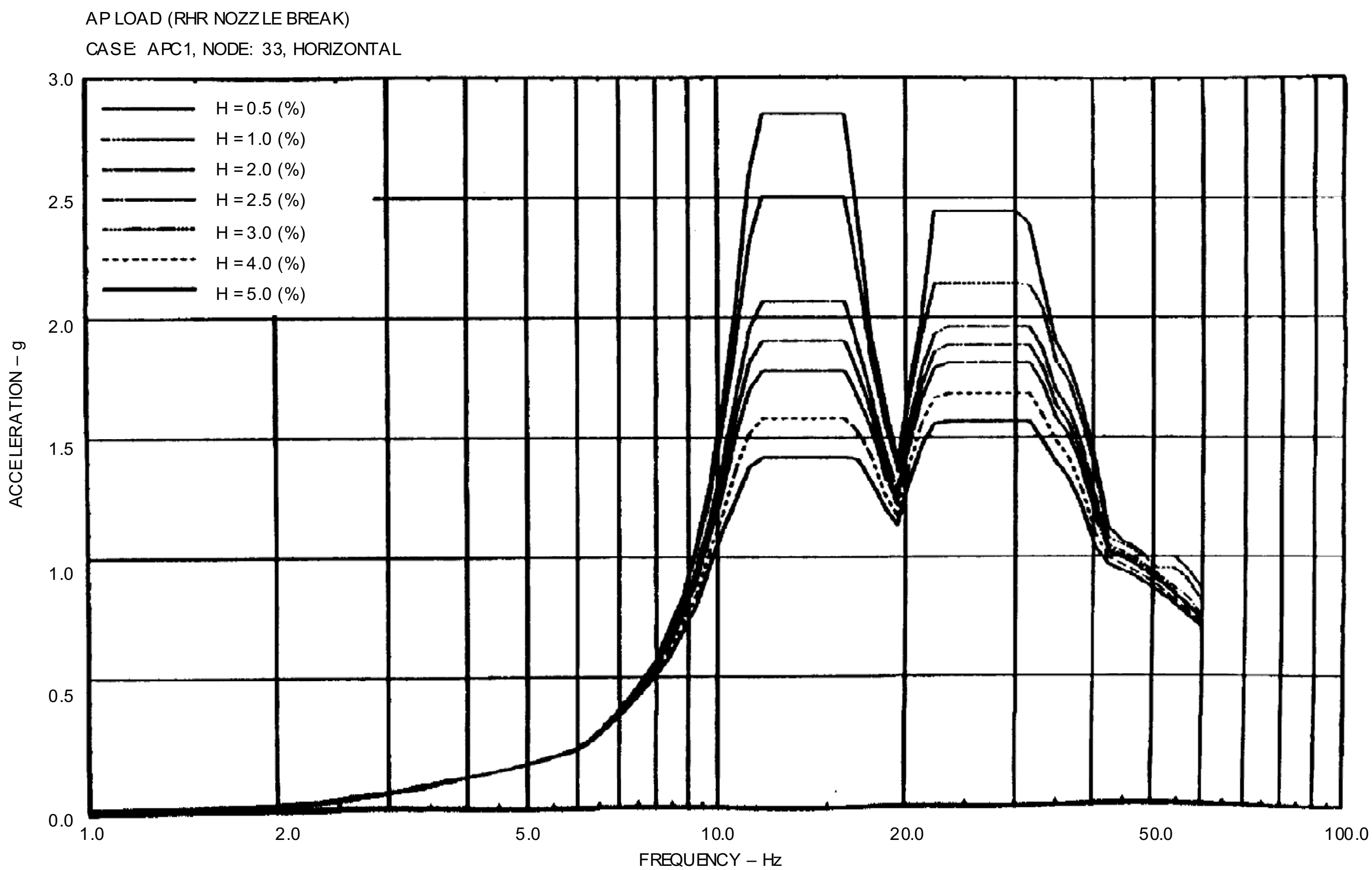

Figure 3G-16 Floor Response Spectrum—Case: APC1, Node: 33, Horizontal

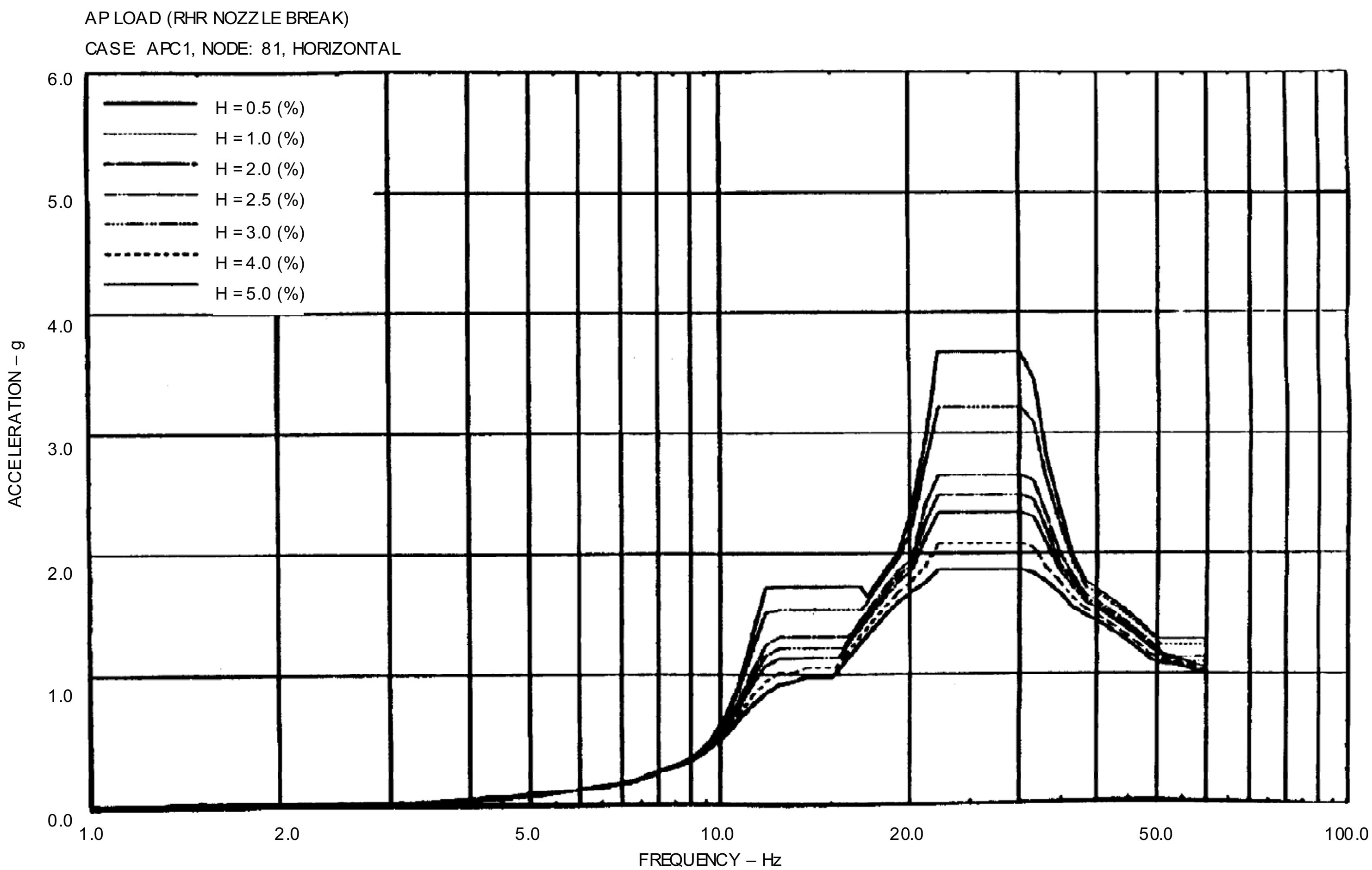

Figure 3G-17 Floor Response Spectrum—Case: APC1, Node: 81, Horizontal

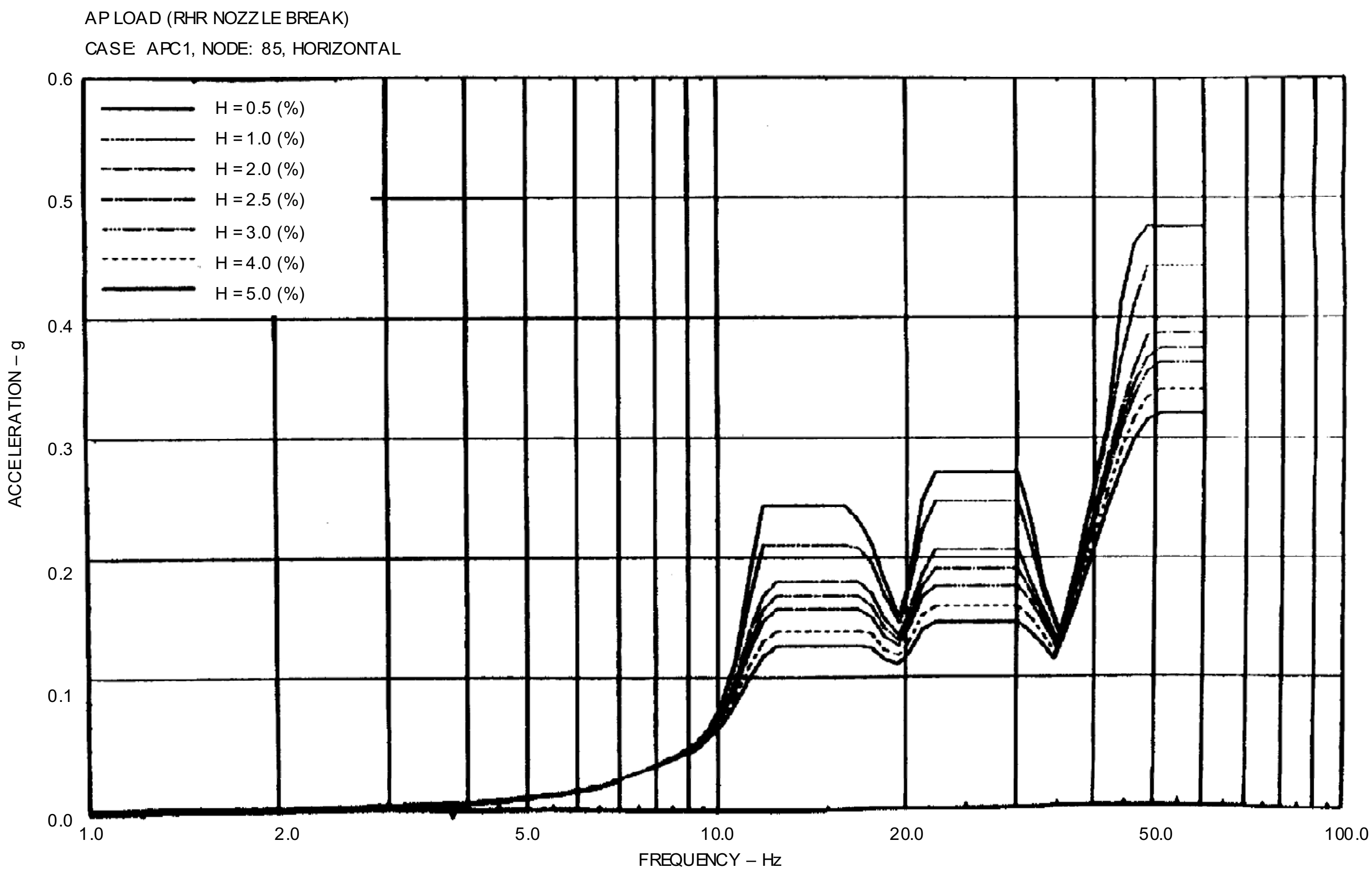

Figure 3G-18 Floor Response Spectrum—Case: APC1, Node: 85, Horizontal

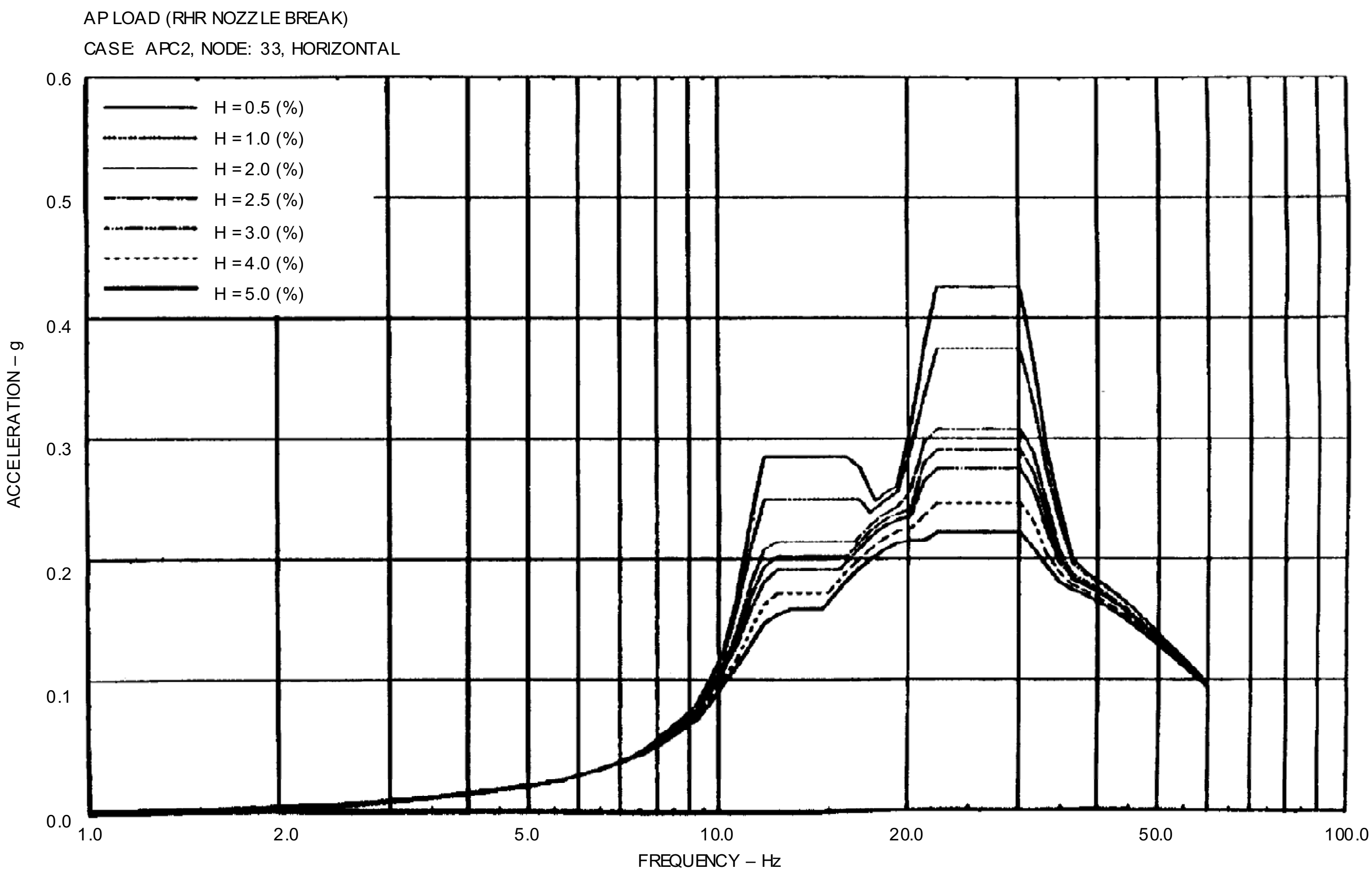

Figure 3G-19 Floor Response Spectrum—Case: APC2, Node: 33, Horizontal

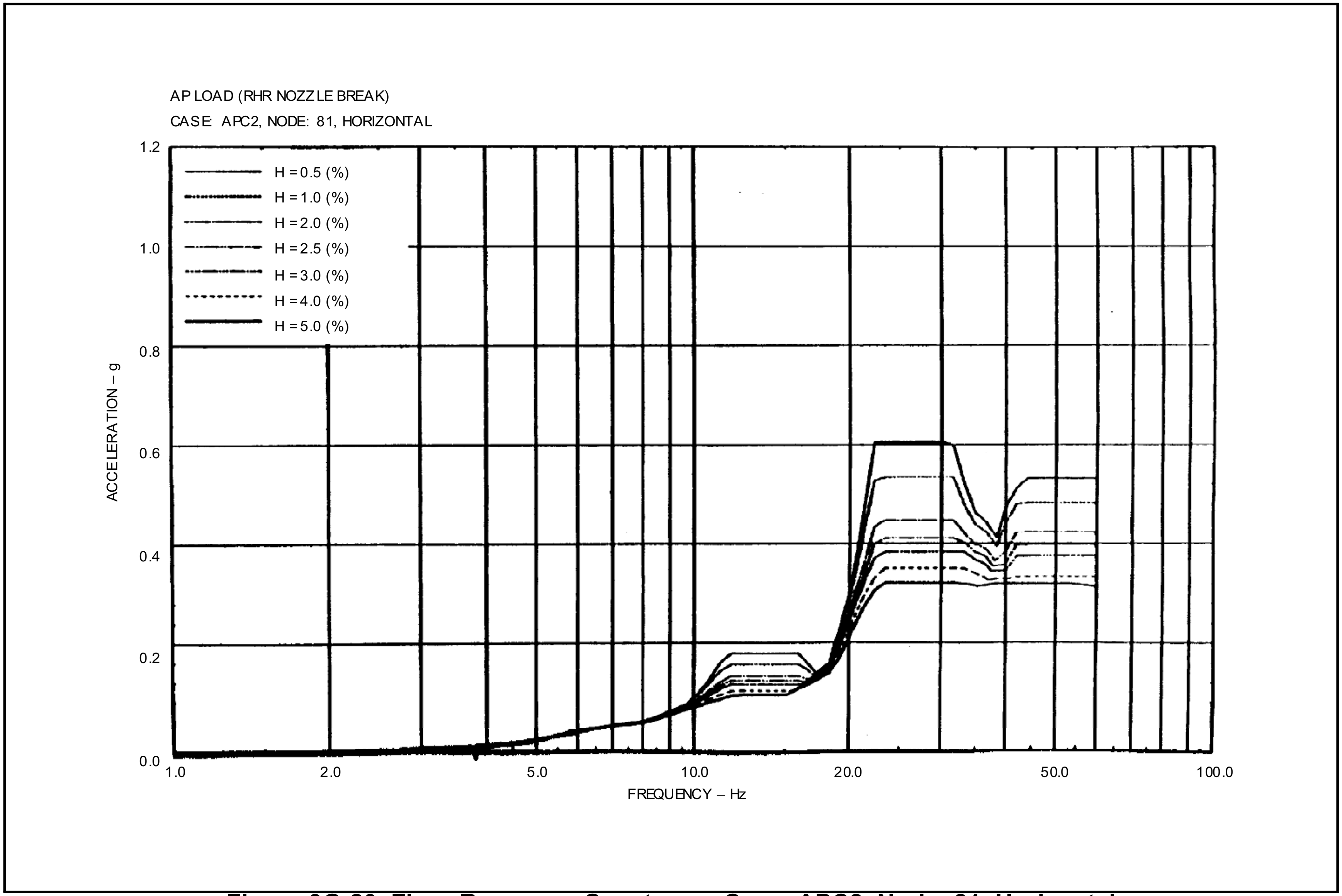

Figure 3G-20 Floor Response Spectrum—Case: APC2, Node: 81, Horizontal

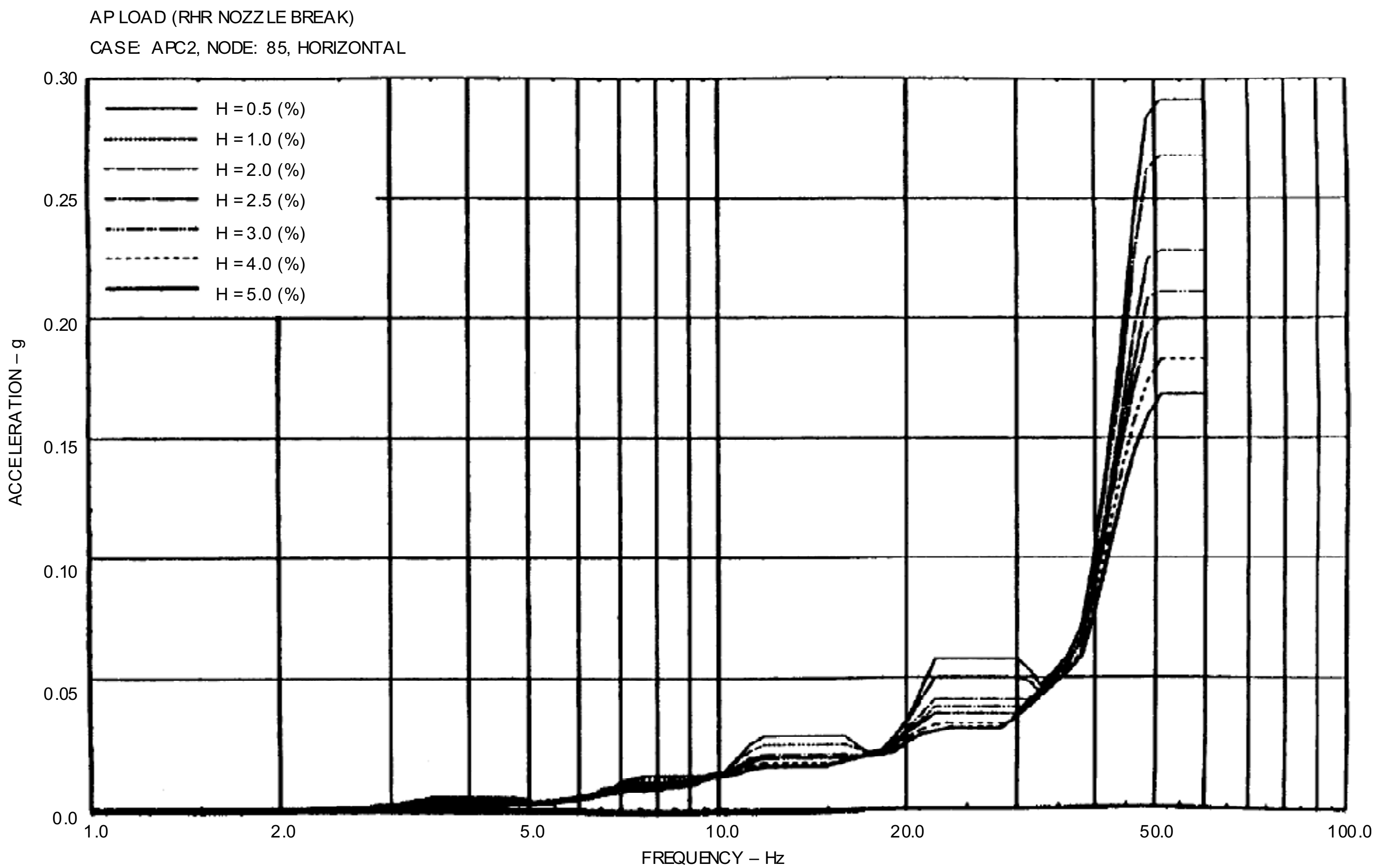

Figure 3G-21 Floor Response Spectrum—Case: APC2, Node: 85, Horizontal

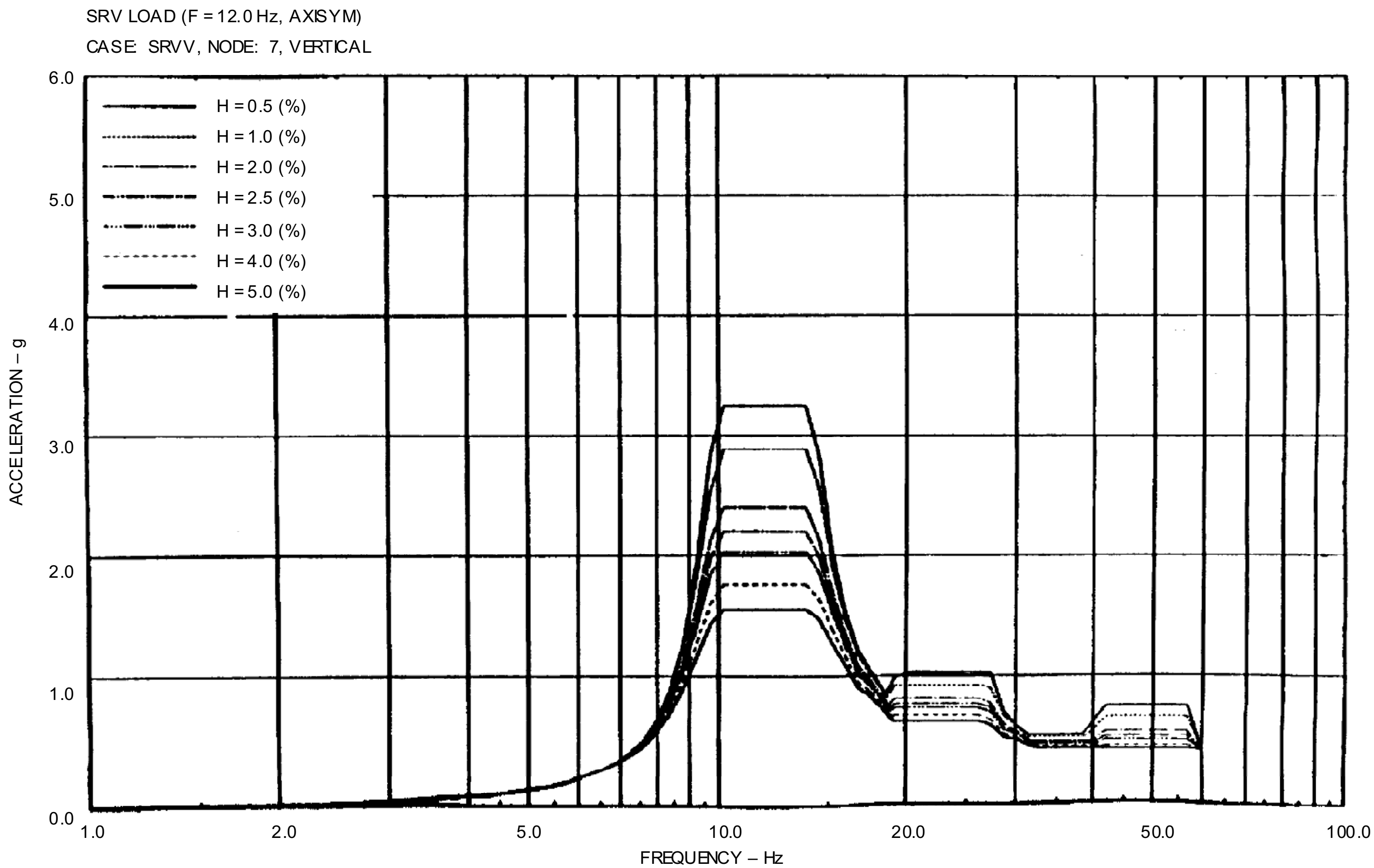

Figure 3G-22 Floor Response Spectrum—Case: SRVV, Node: 7, Vertical

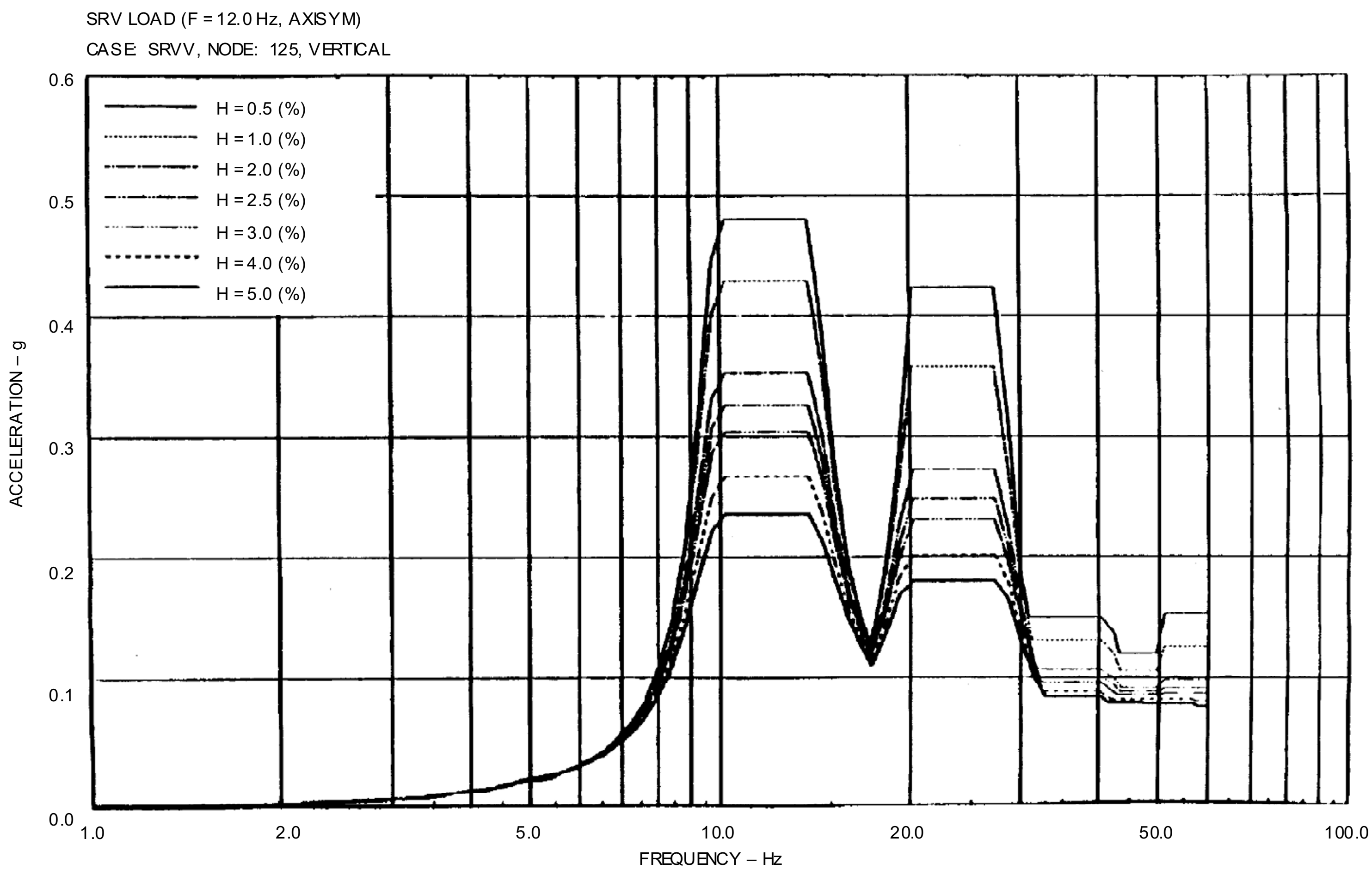

Figure 3G-23 Floor Response Spectrum—Case: SRVV, Node: 125, Vertical

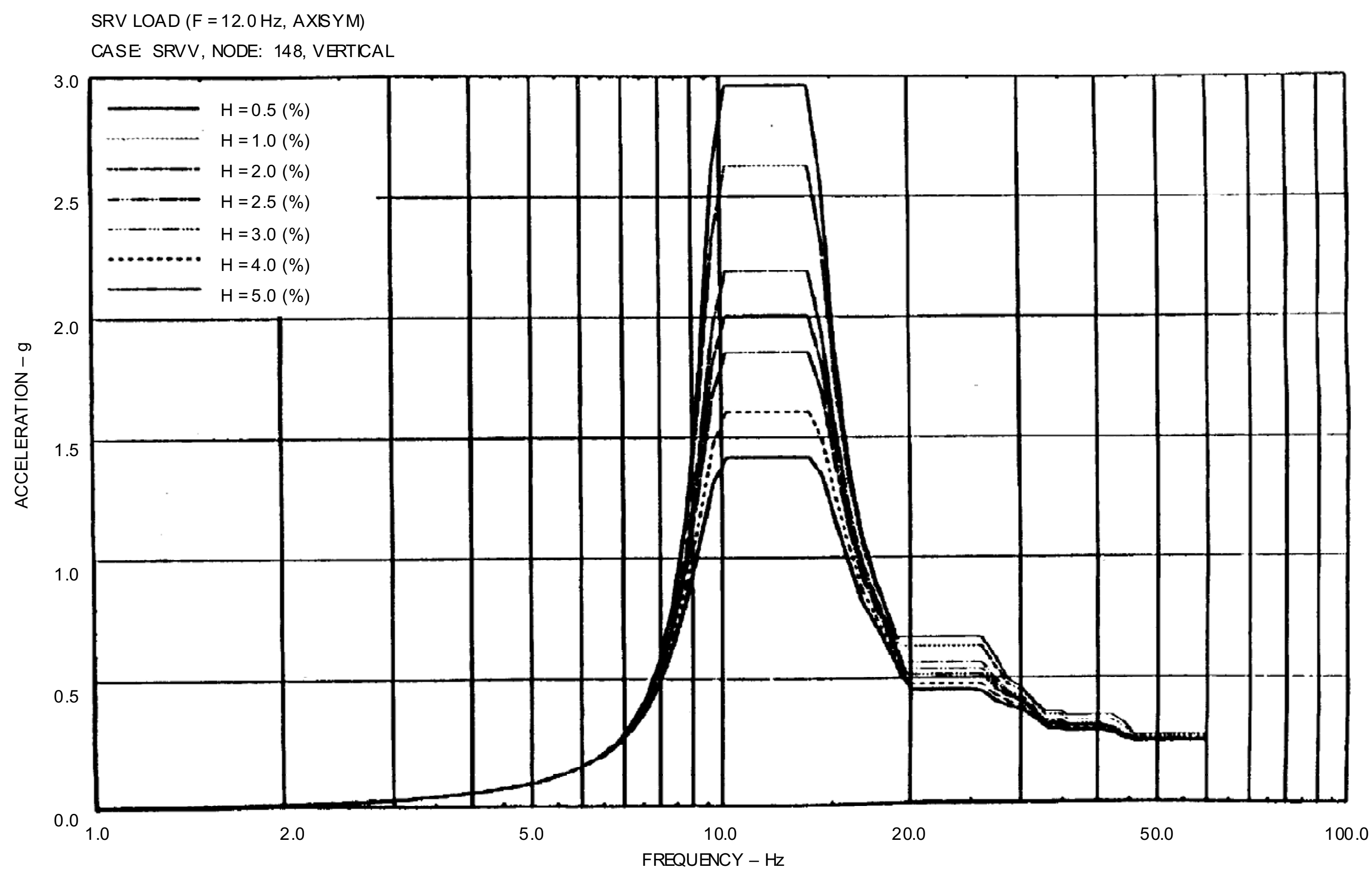

Figure 3G-24 Floor Response Spectrum—Case: SRVV, Node: 148, Vertical

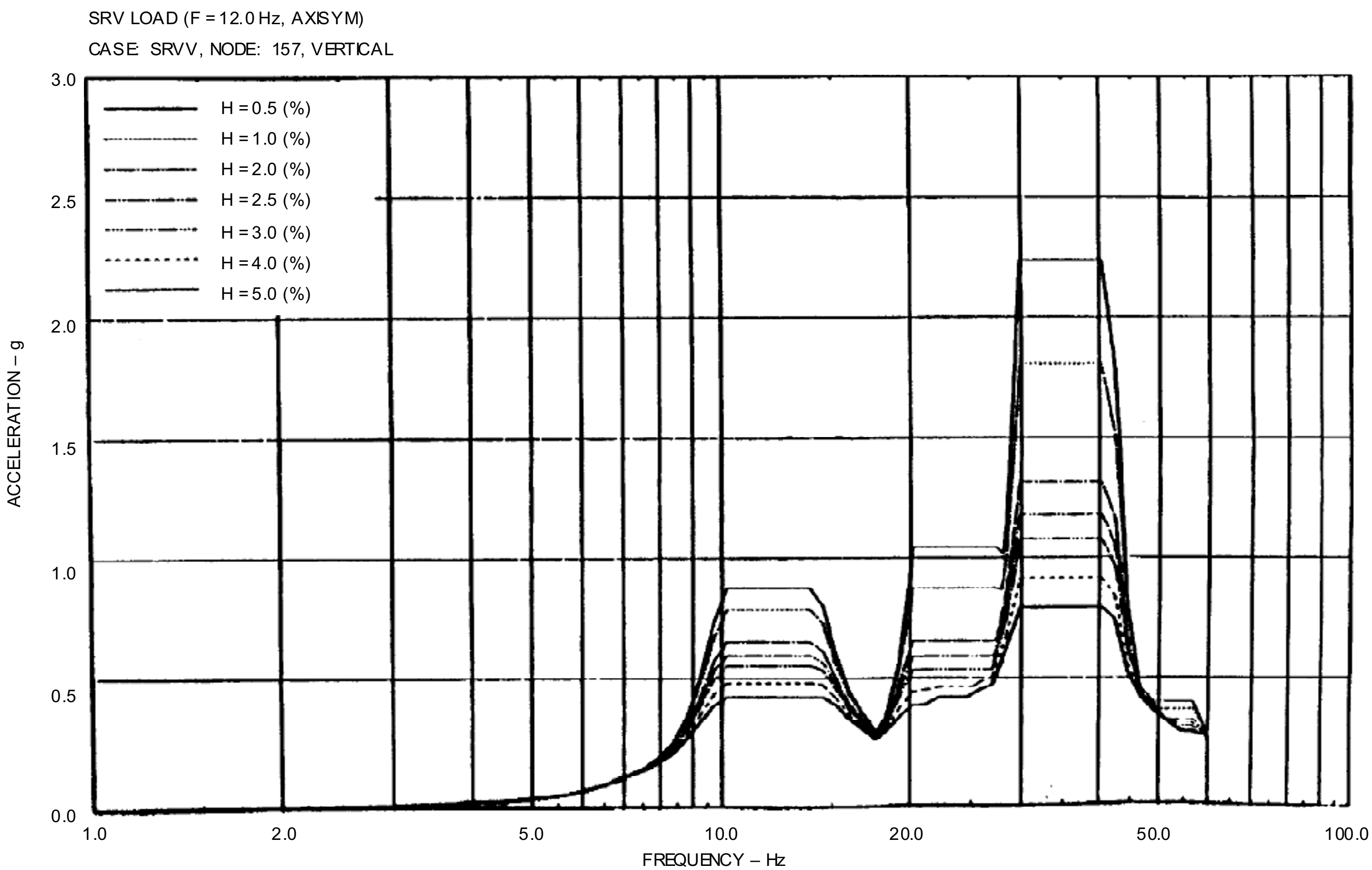

Figure 3G-25 Floor Response Spectrum—Case: SRVV, Node: 157, Vertical

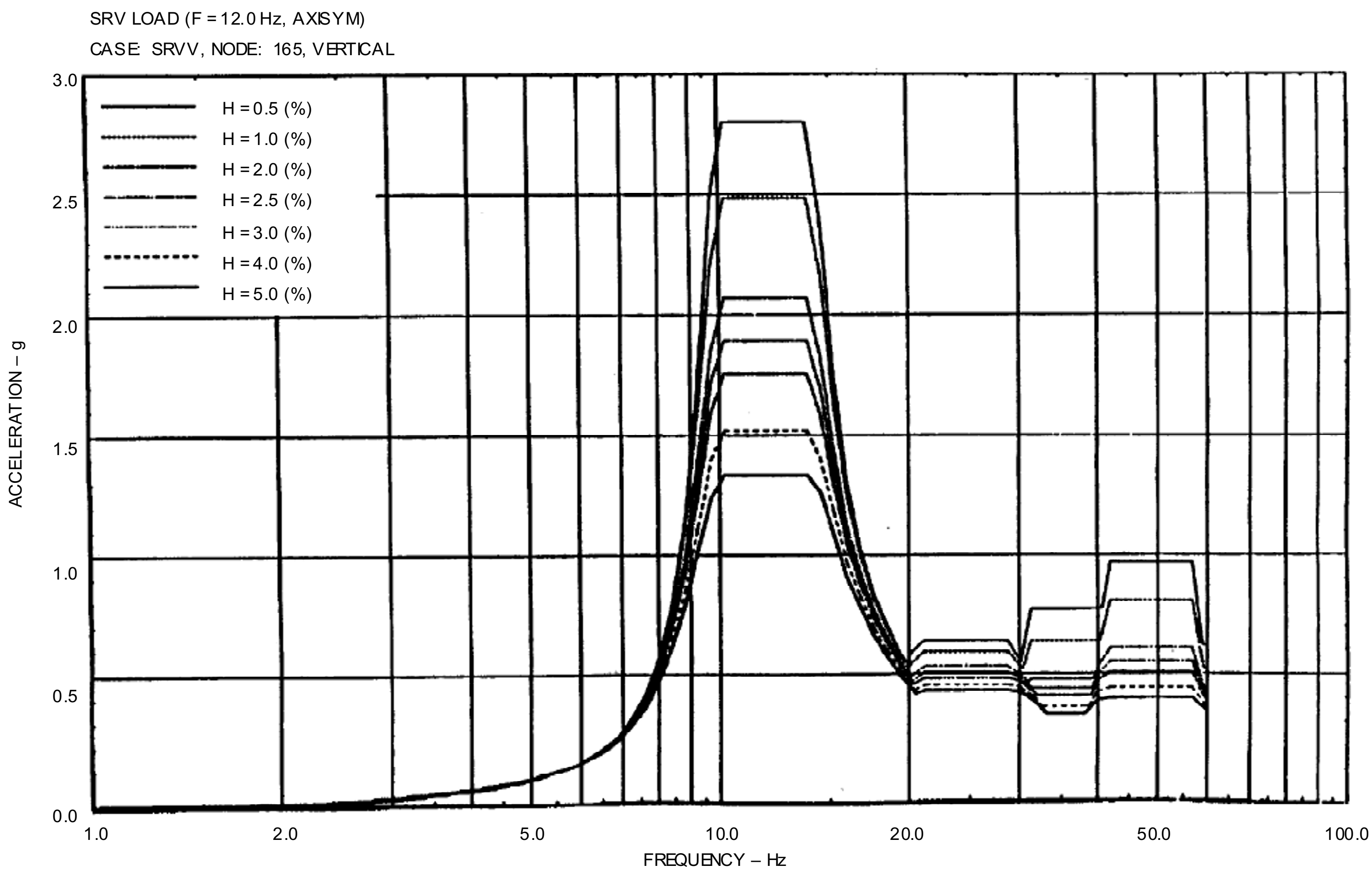

Figure 3G-26 Floor Response Spectrum—Case: SRVV, Node: 165, Vertical

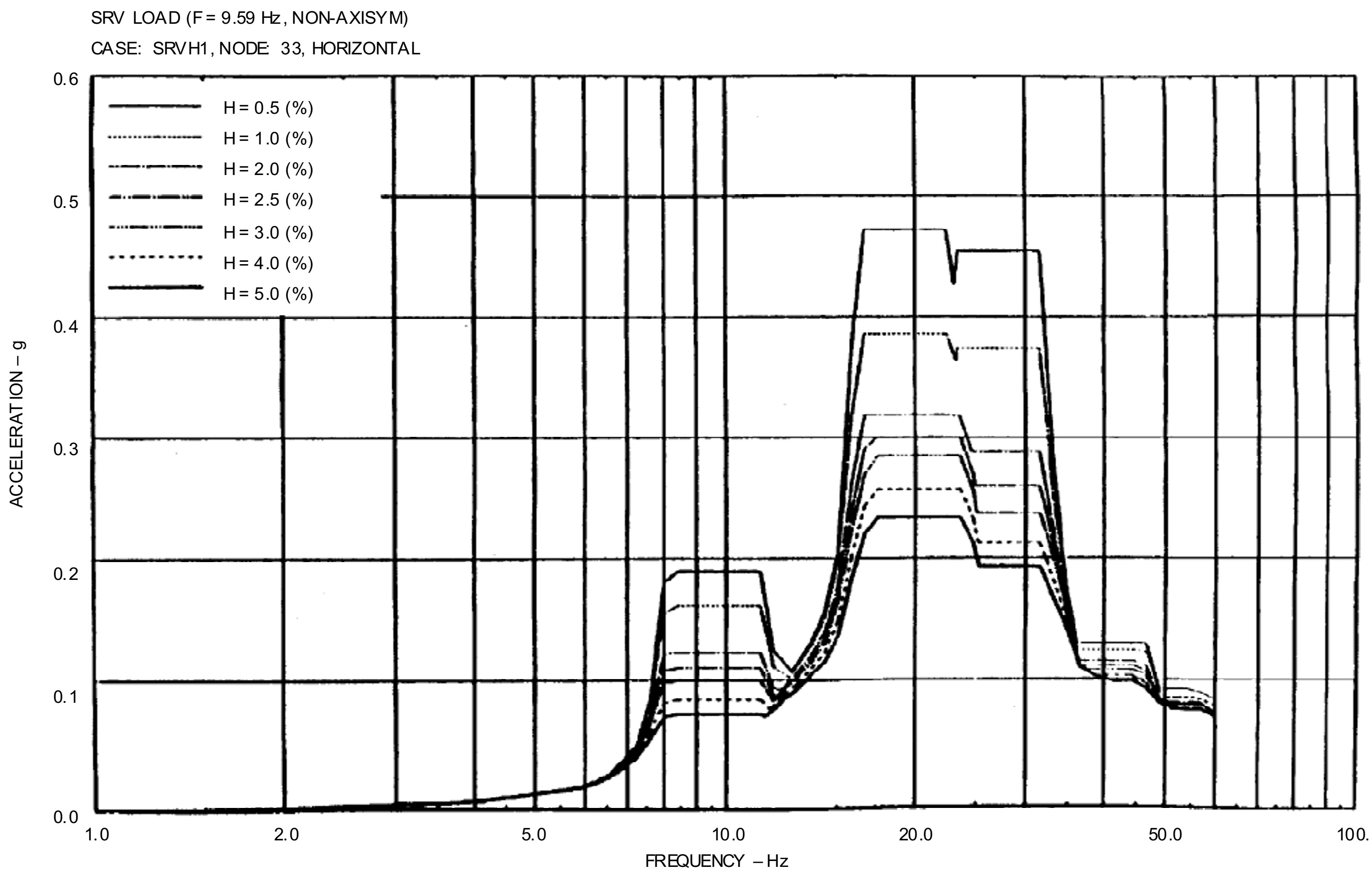

Figure 3G-27 Floor Response Spectrum—Case: SRVH1, Node: 33, Horizontal

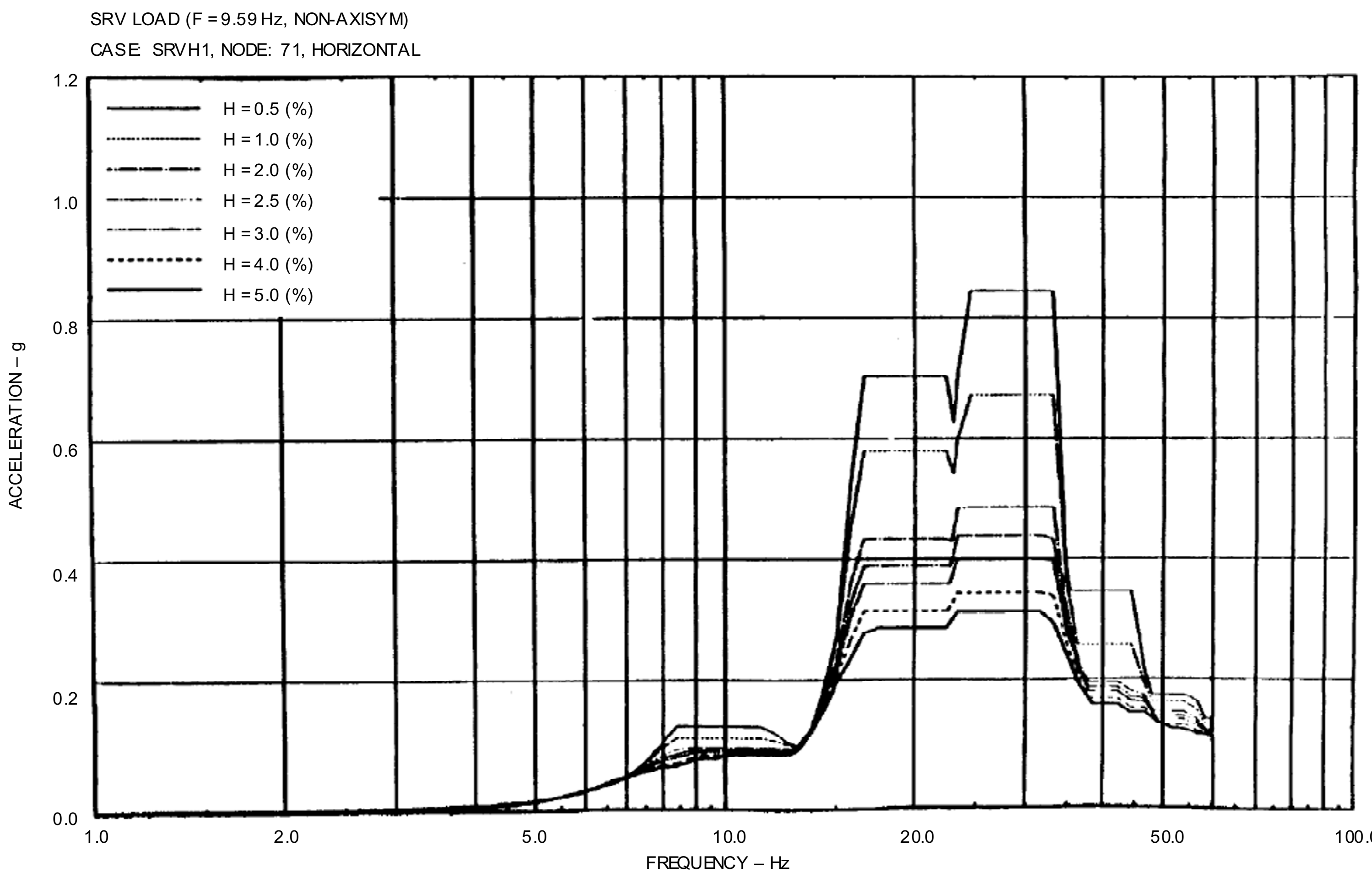

Figure 3G-28 Floor Response Spectrum—Case: SRVH1, Node: 71, Horizontal

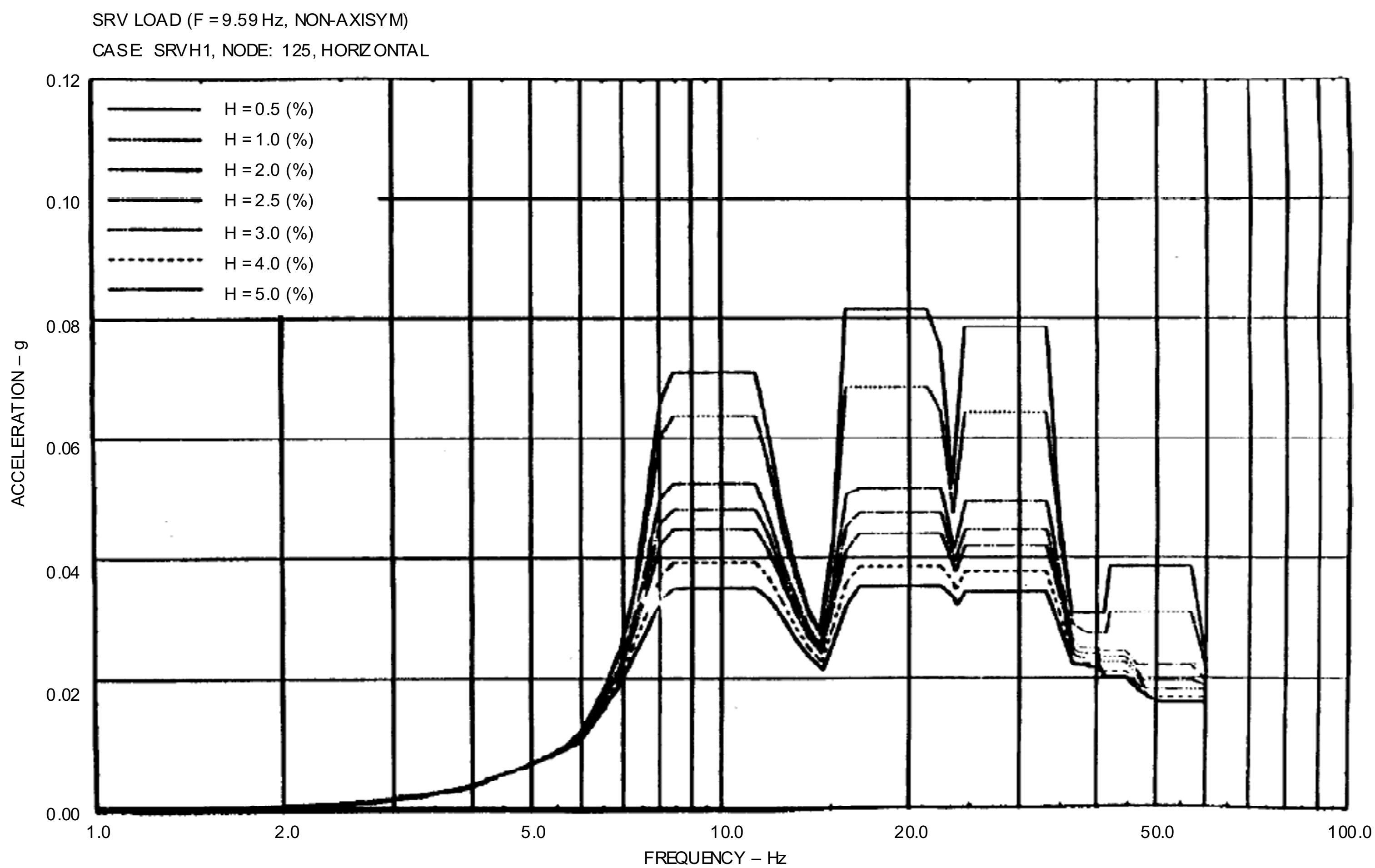

Figure 3G-29 Floor Response Spectrum—Case: SRVH1, Node: 125, Horizontal

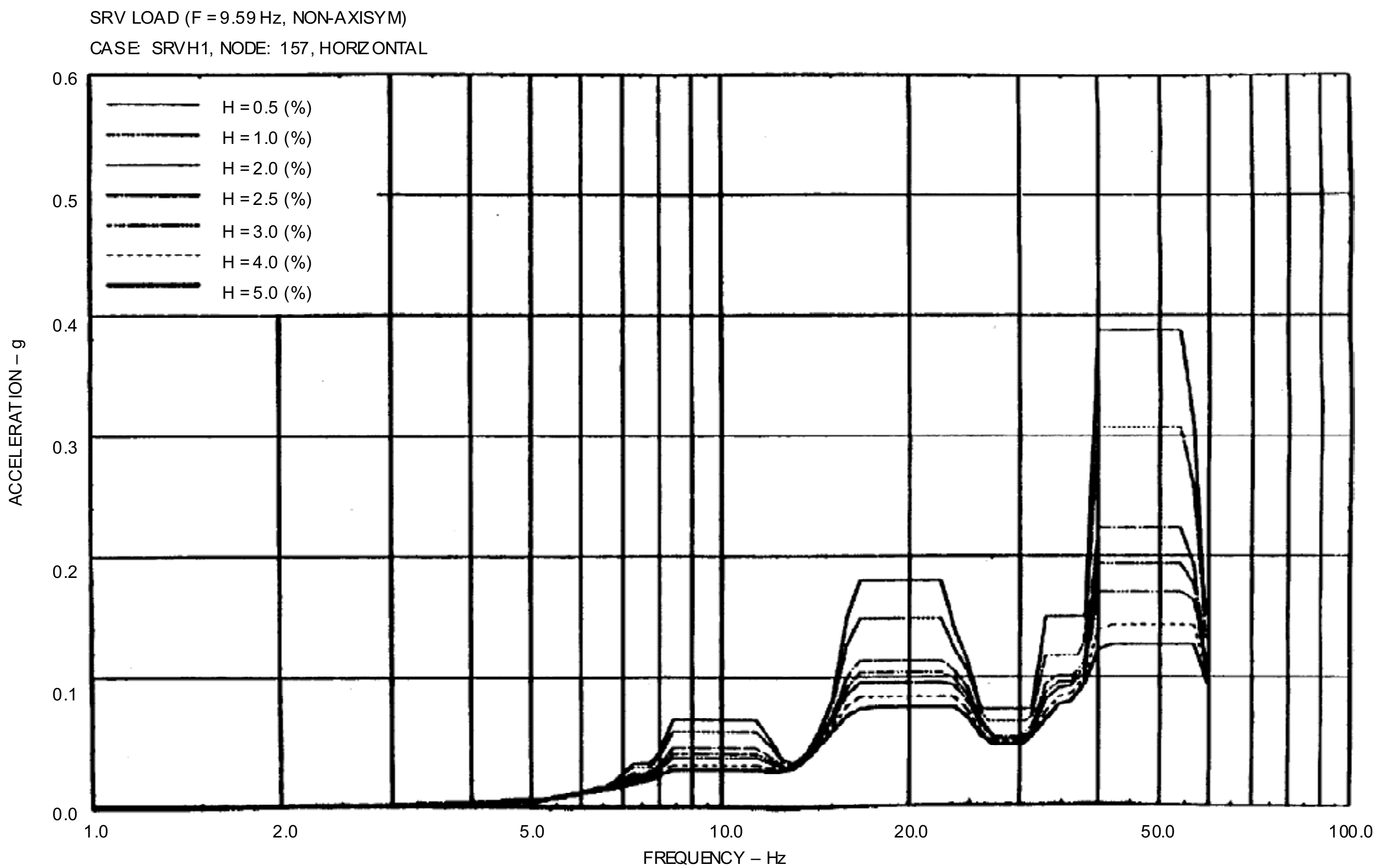

Figure 3G-30 Floor Response Spectrum—Case: SRVH1, Node: 157, Horizontal

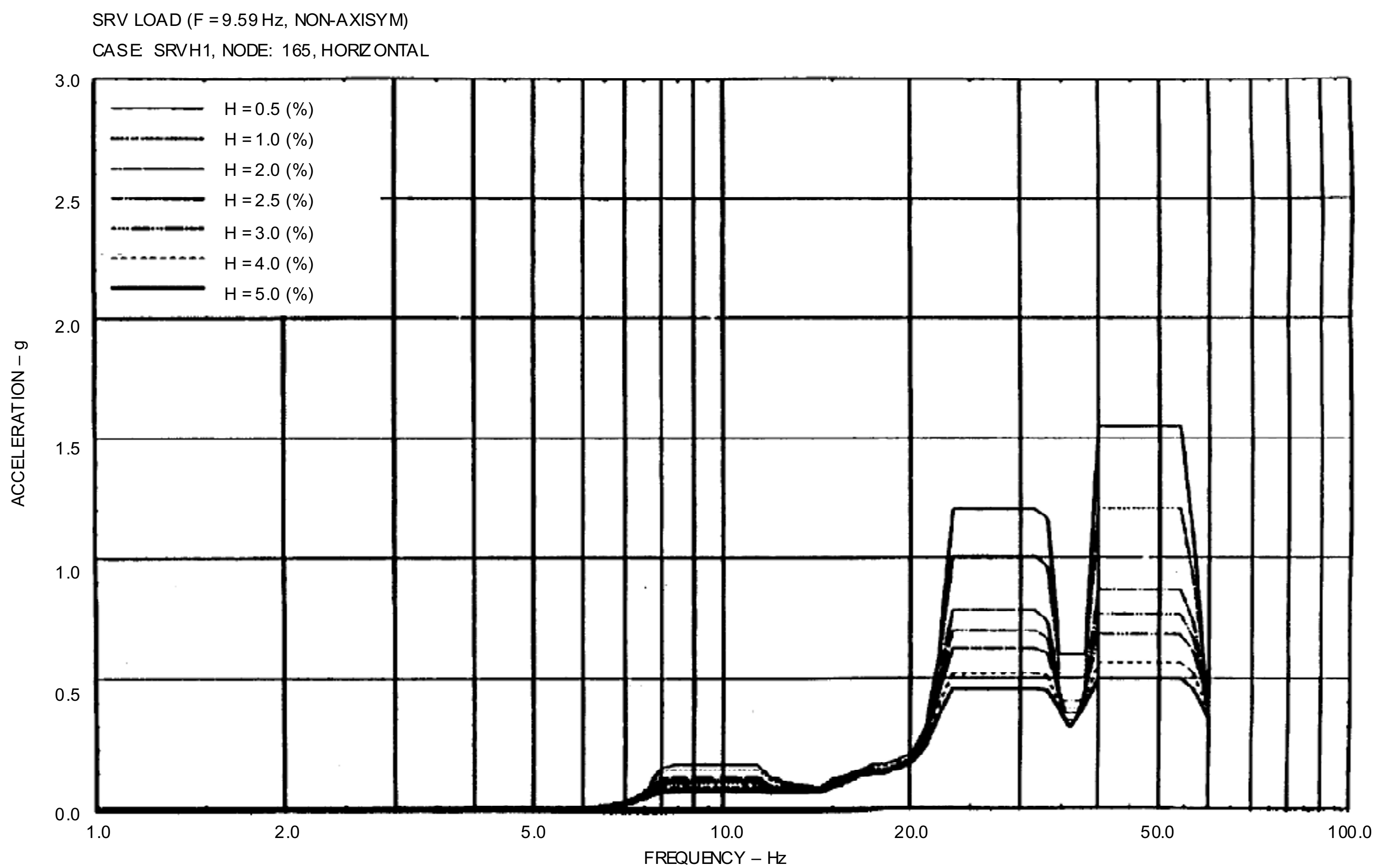

Figure 3G-31 Floor Response Spectrum—Case: SRVH1, Node: 165, Horizontal

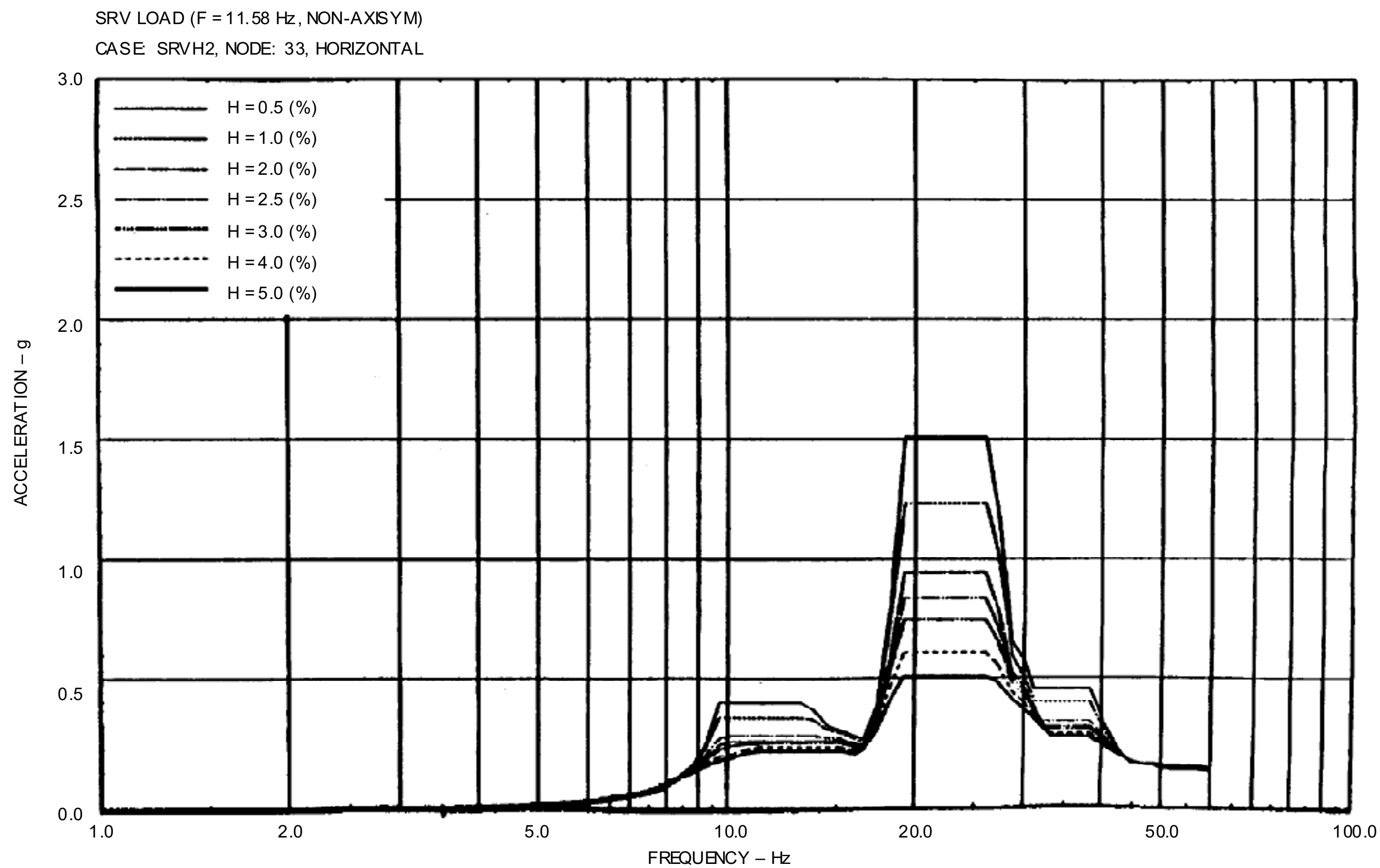

Figure 3G-32 Floor Response Spectrum—Case: SRVH2, Node: 33, Horizontal

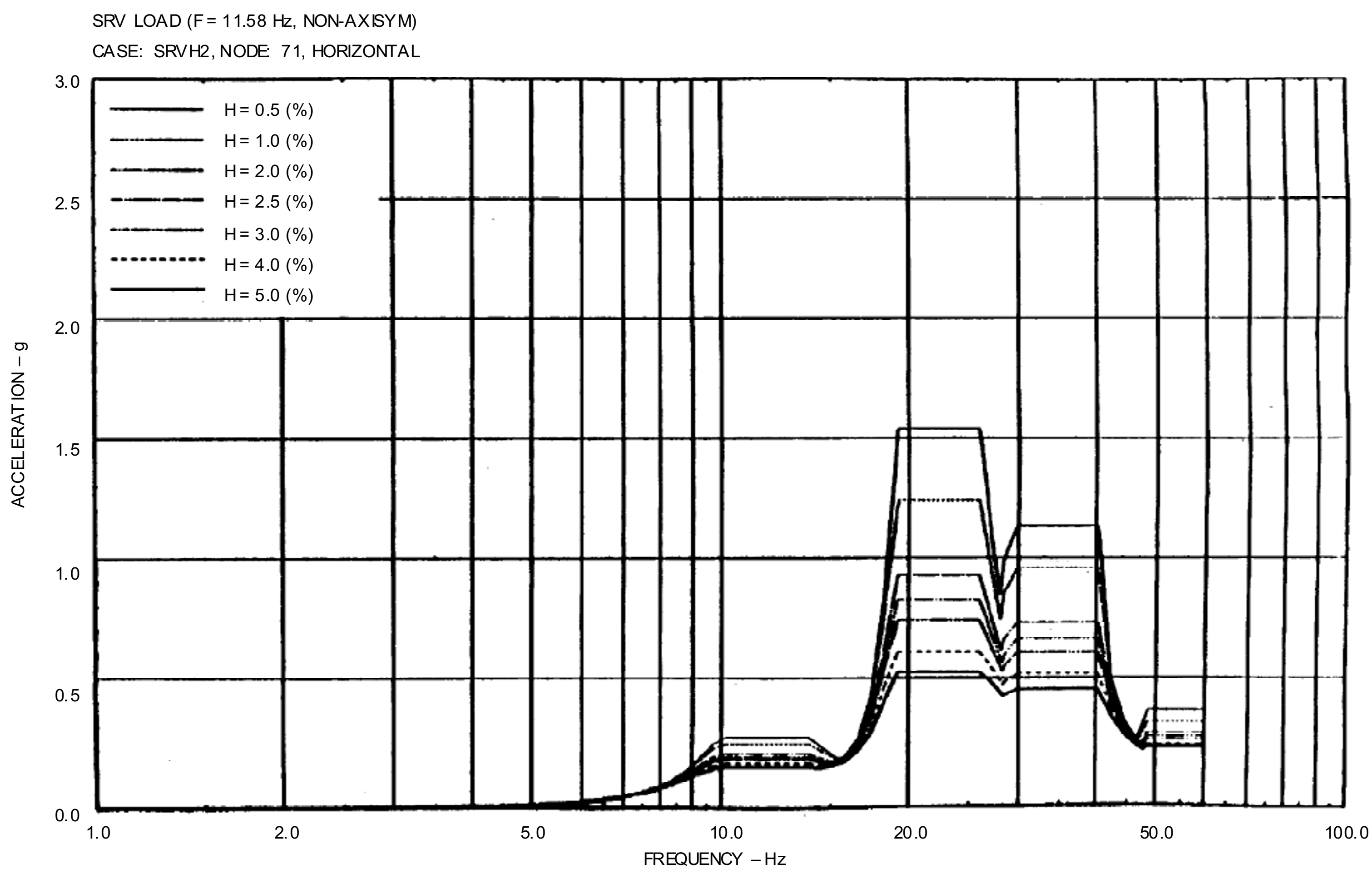

Figure 3G-33 Floor Response Spectrum—Case: SRVH2, Node: 71, Horizontal

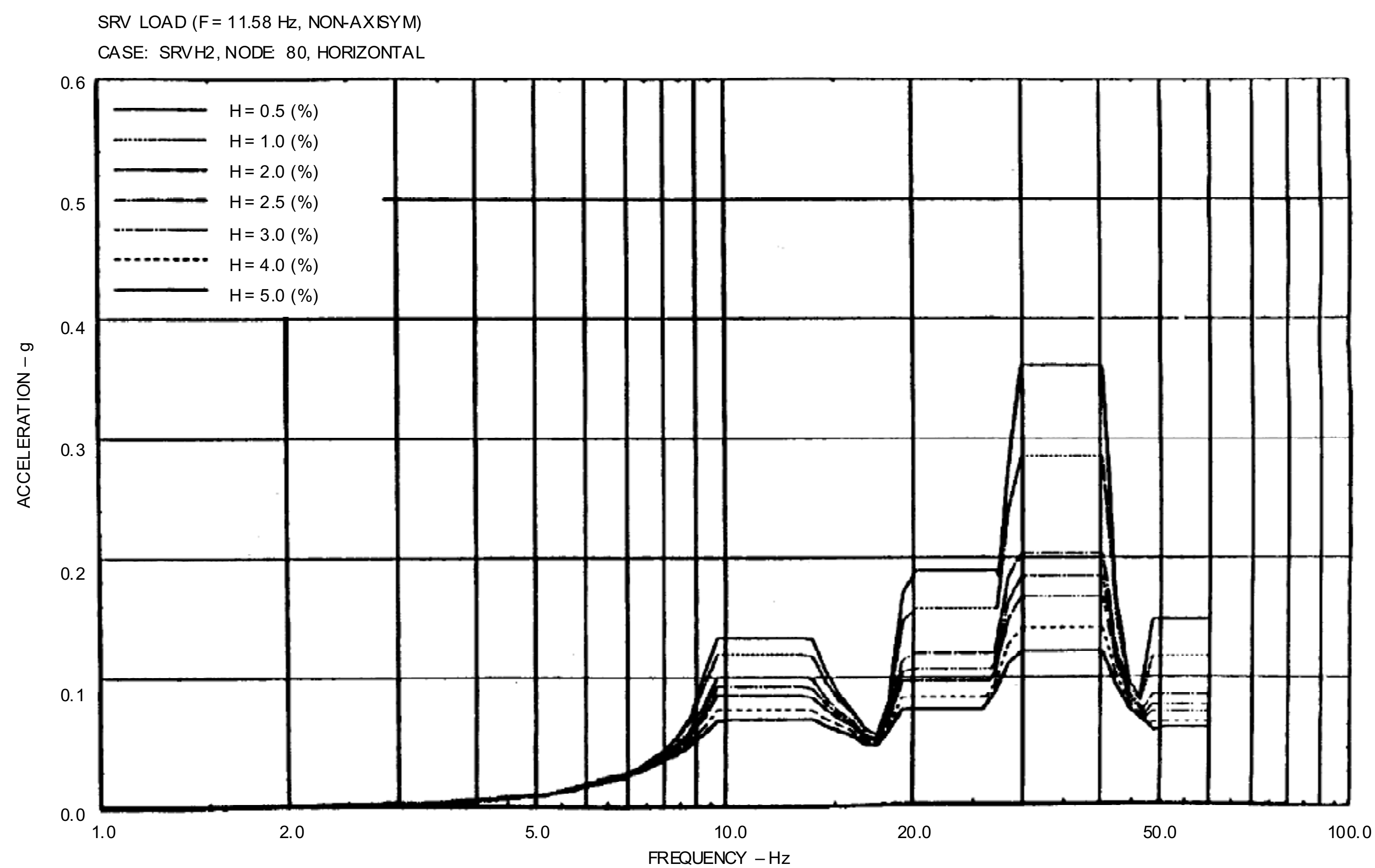

Figure 3G-34 Floor Response Spectrum—Case: SRVH2, Node: 80, Horizontal

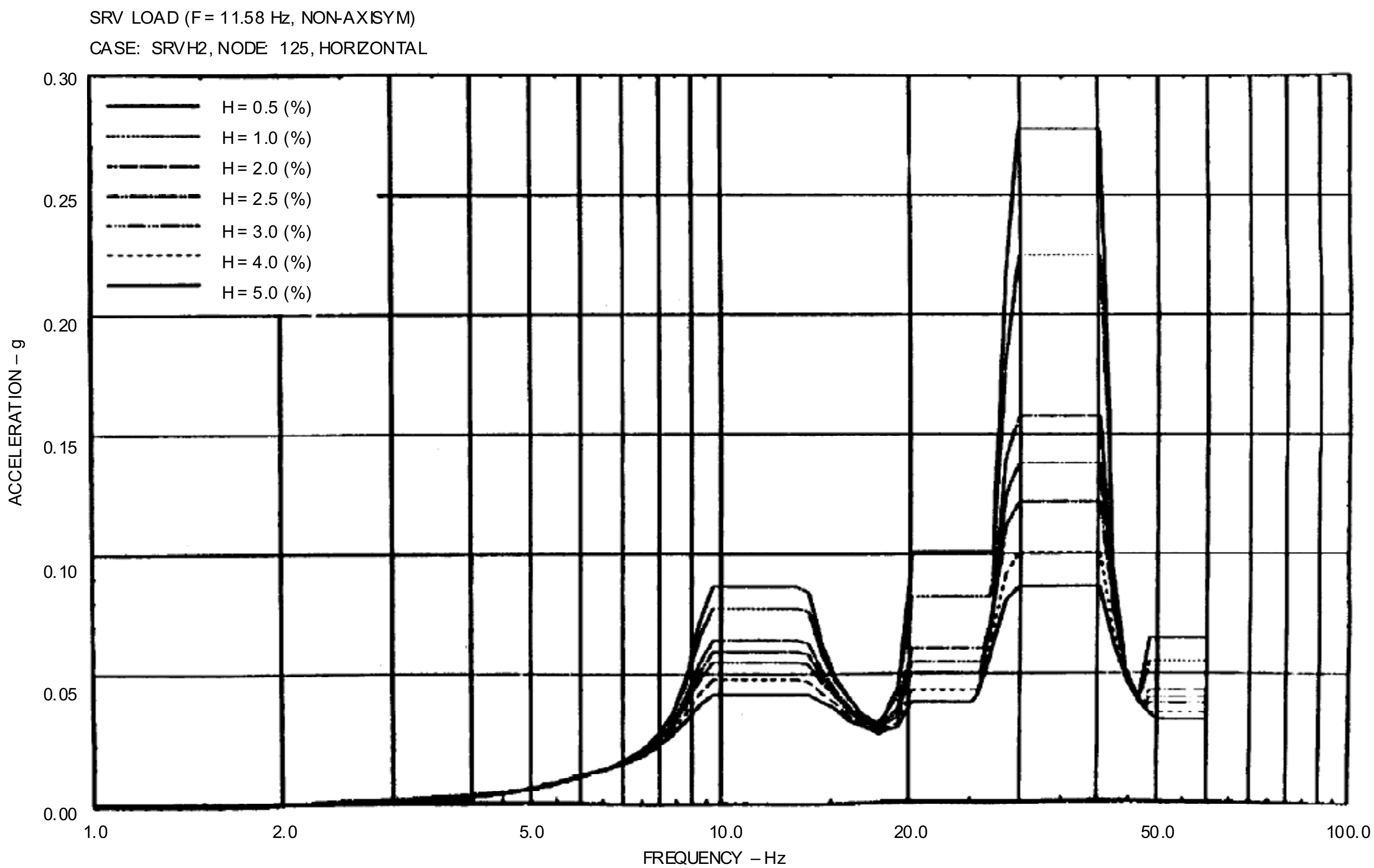

Figure 3G-35 Floor Response Spectrum—Case: SRVH2, Node: 125, Horizontal

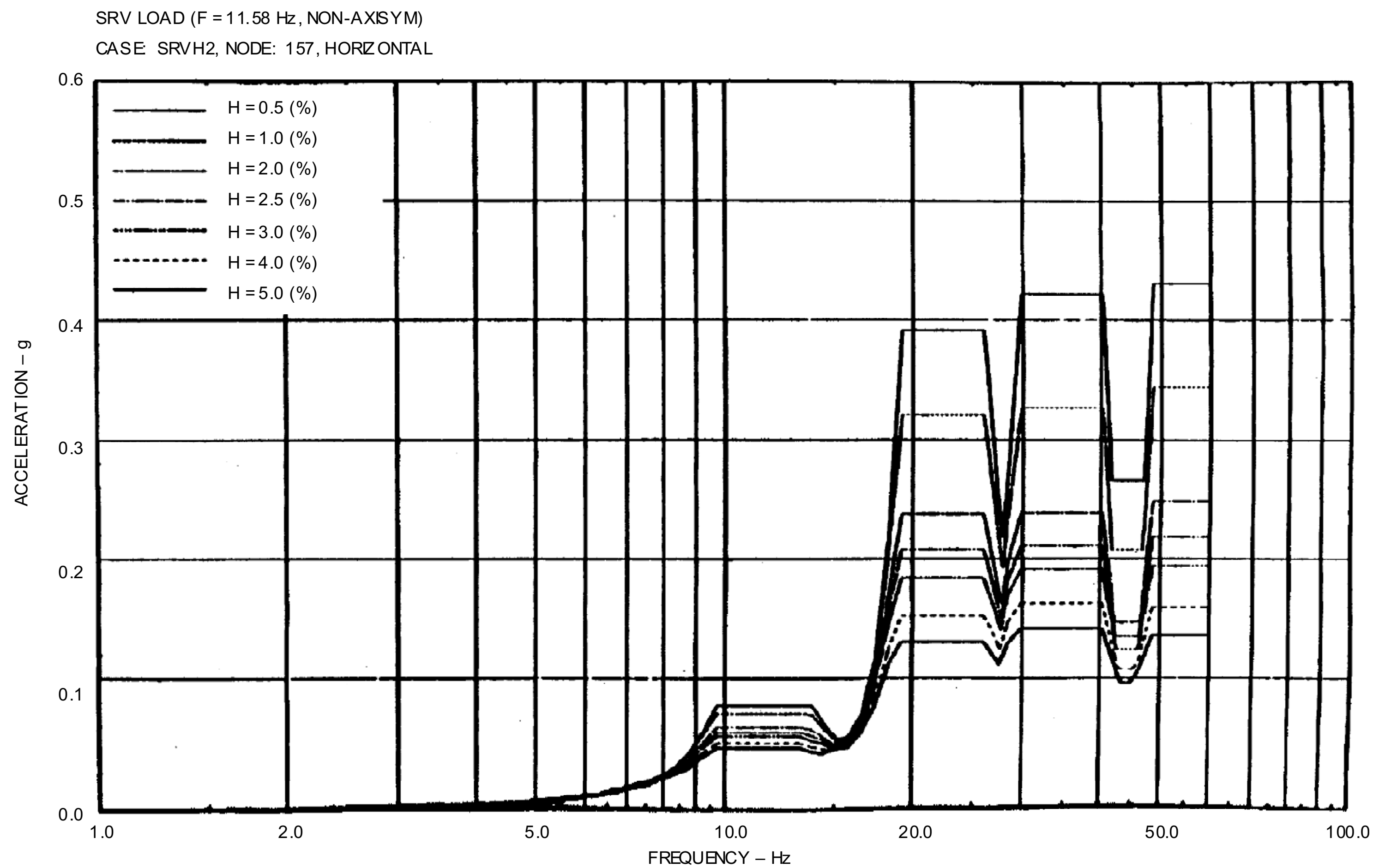

Figure 3G-36 Floor Response Spectrum—Case: SRVH2, Node: 157, Horizontal

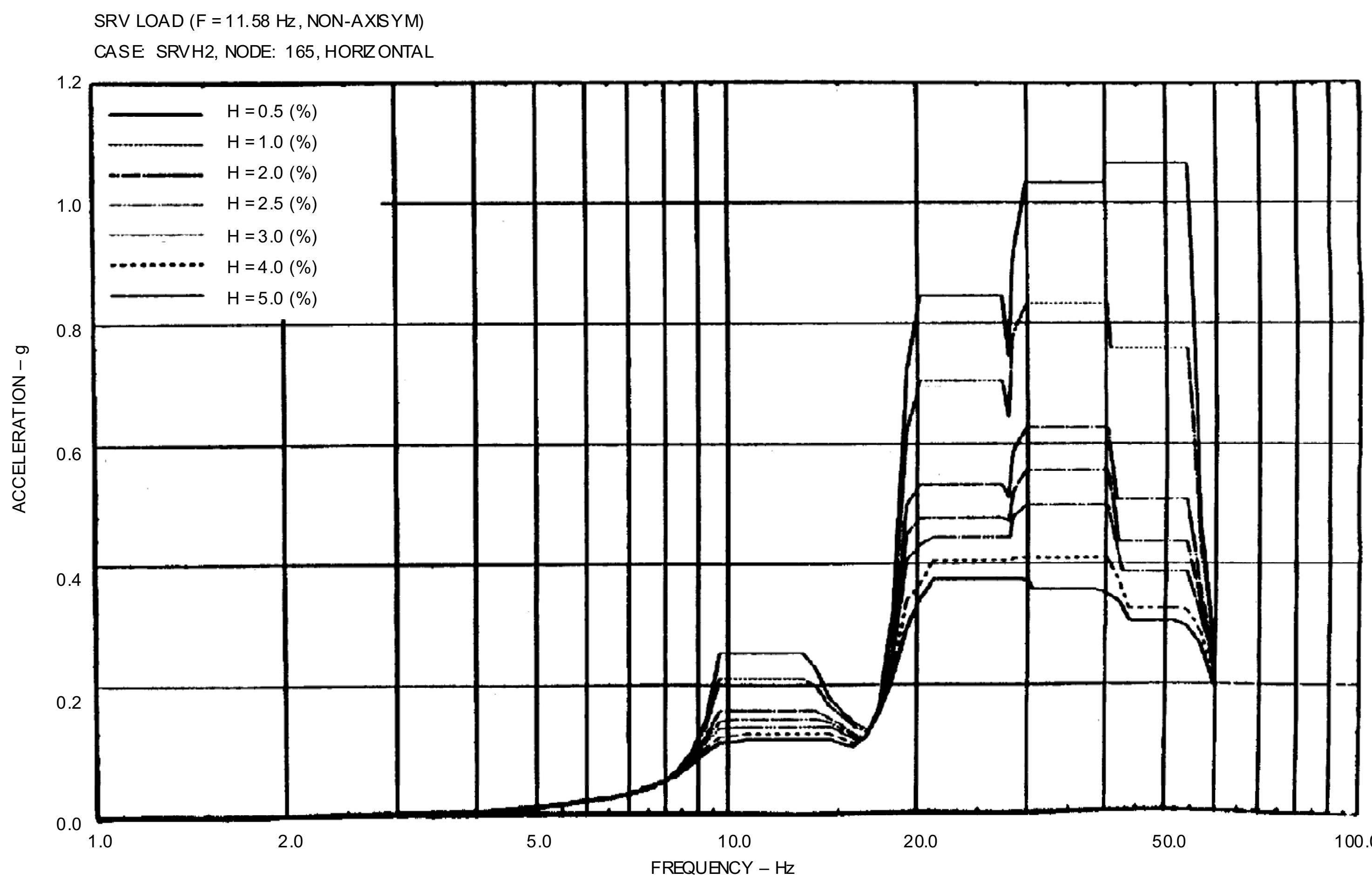

Figure 3G-37 Floor Response Spectrum—Case: SRVH2, Node: 165, Horizontal

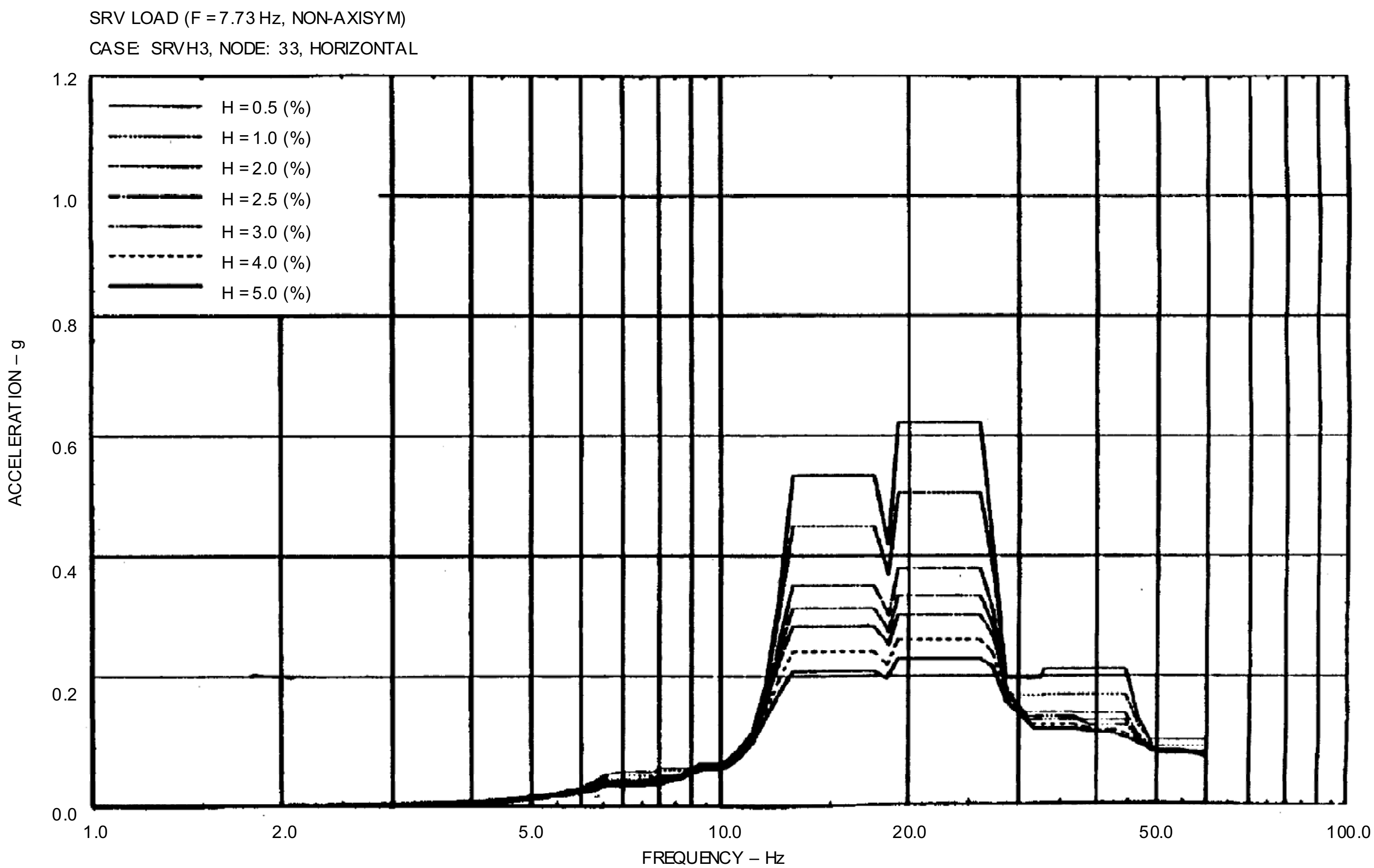

Figure 3G-38 Floor Response Spectrum—Case: SRVH3, Node: 33, Horizontal

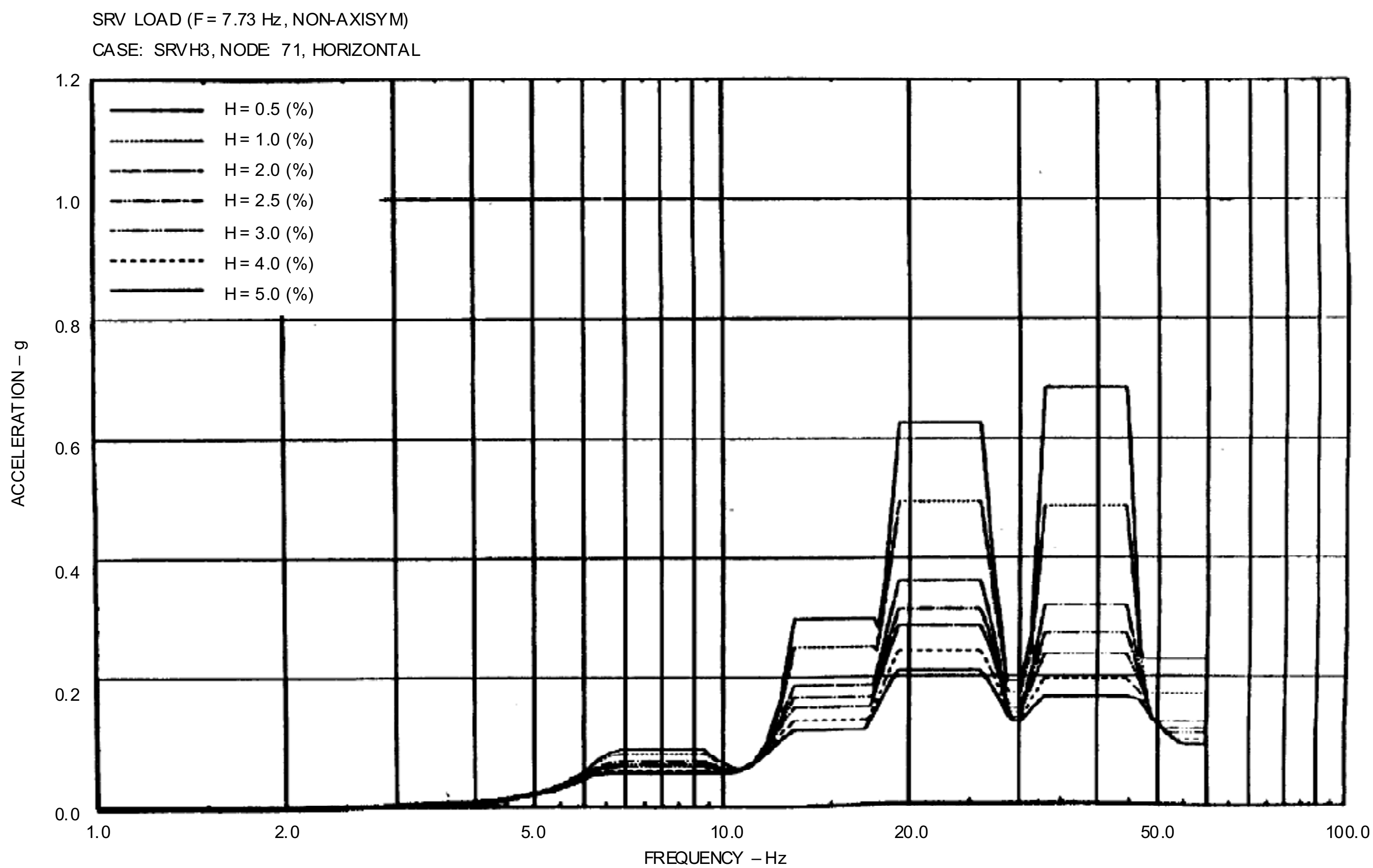

Figure 3G-39 Floor Response Spectrum—Case: SRVH3, Node: 71, Horizontal

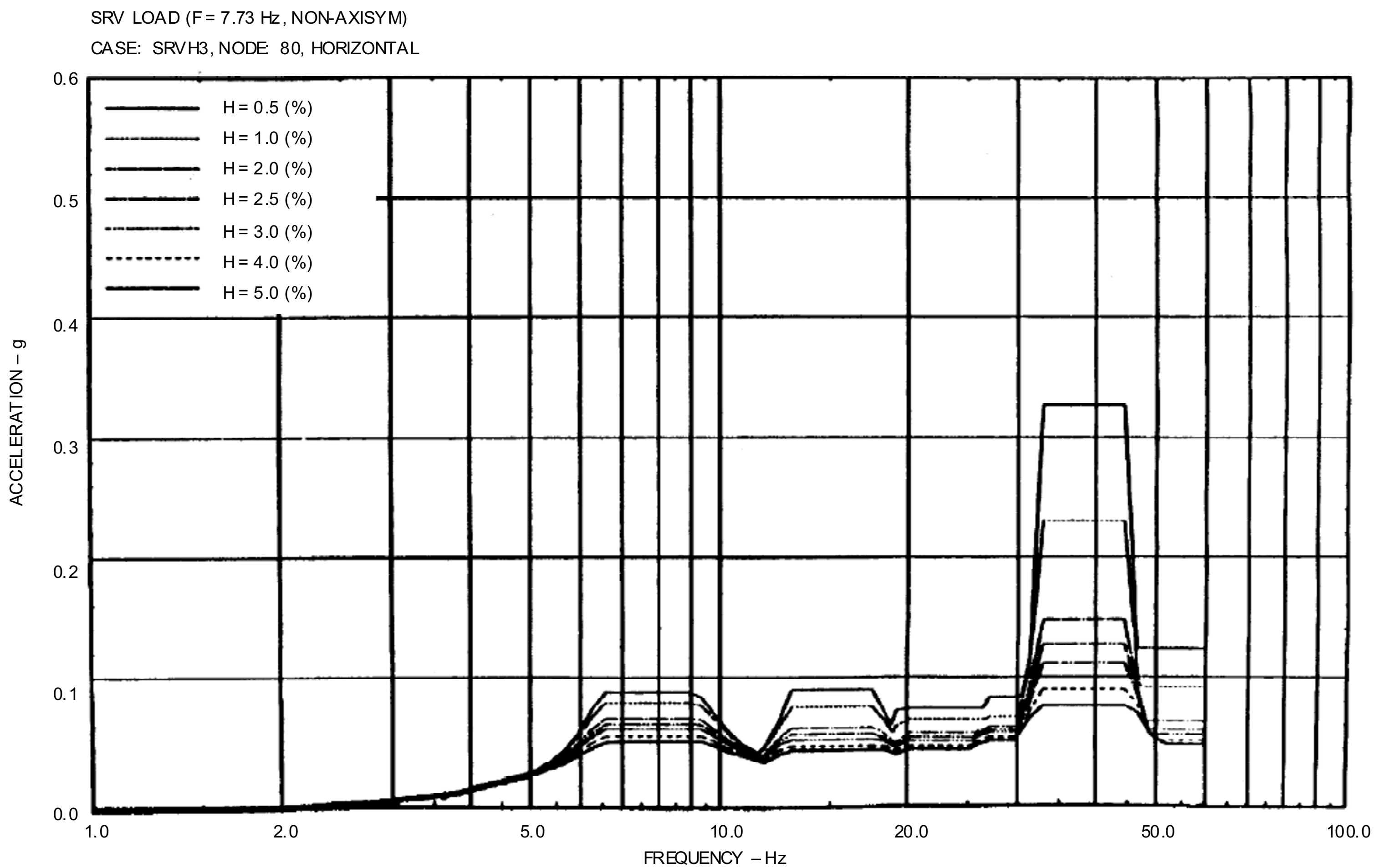

Figure 3G-40 Floor Response Spectrum—Case: SRVH3, Node: 80, Horizontal

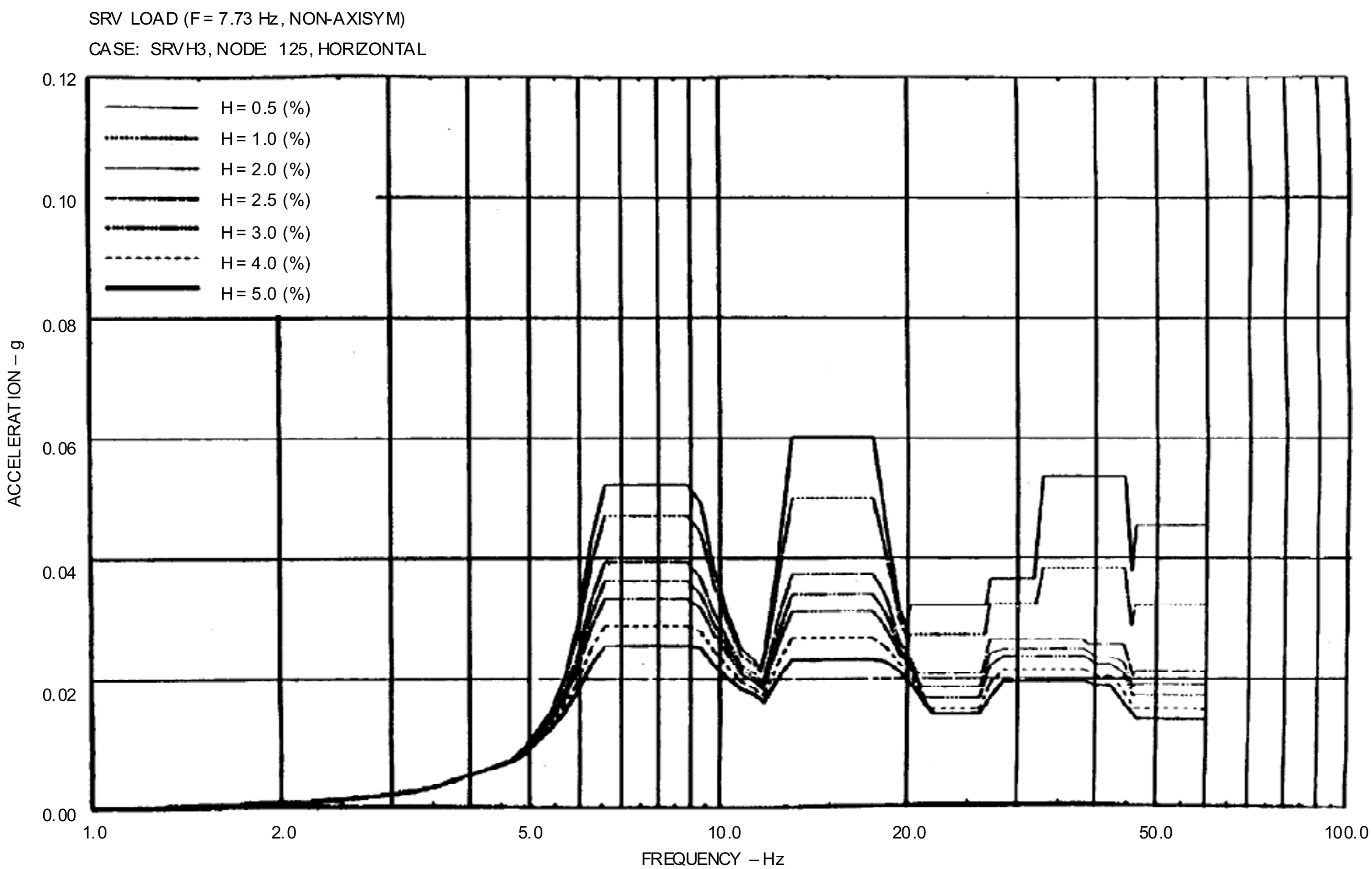

Figure 3G-41 Floor Response Spectrum—Case: SRVH3, Node: 125, Horizontal

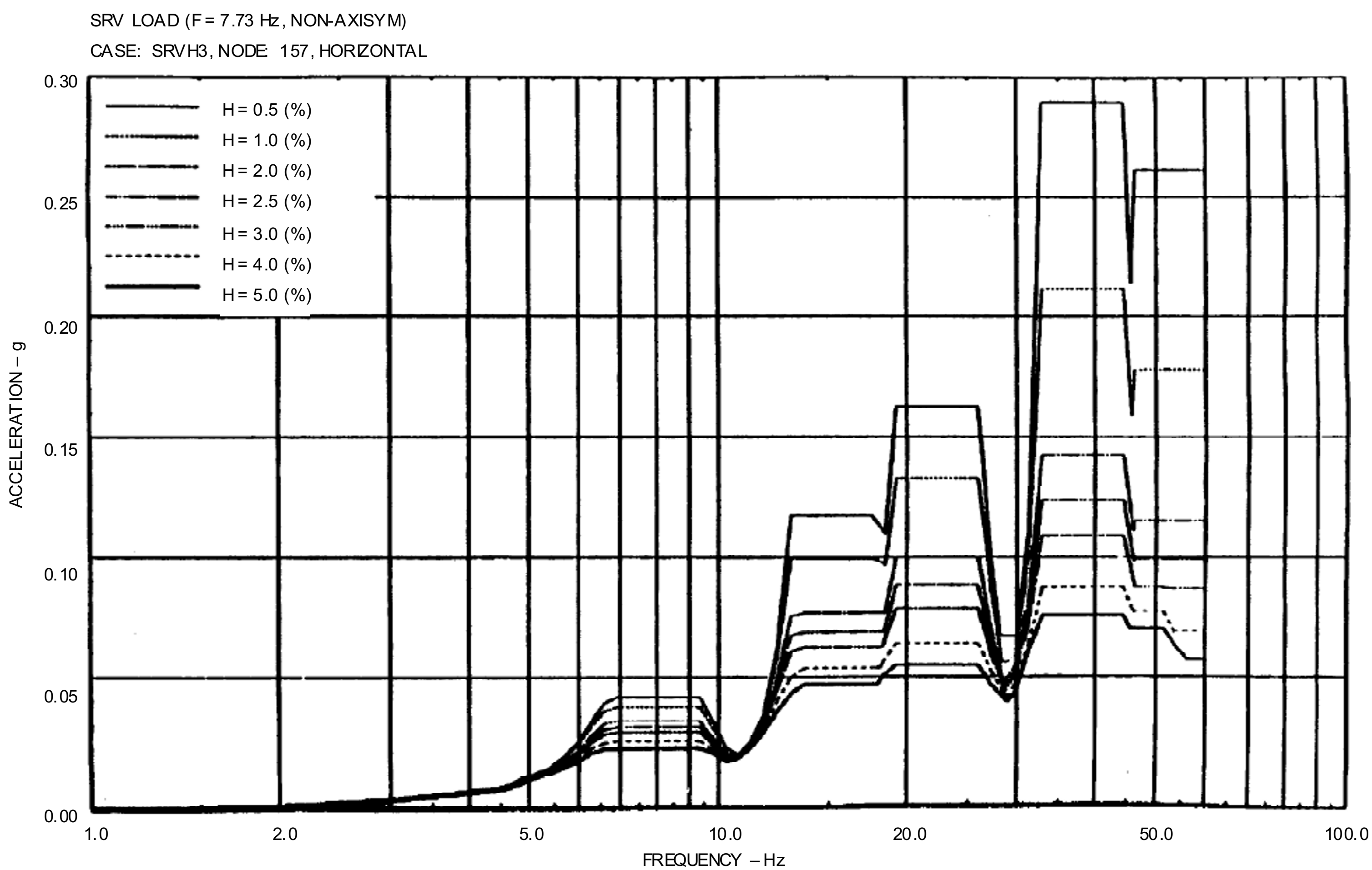

Figure 3G-42 Floor Response Spectrum—Case: SRVH3, Node: 157, Horizontal

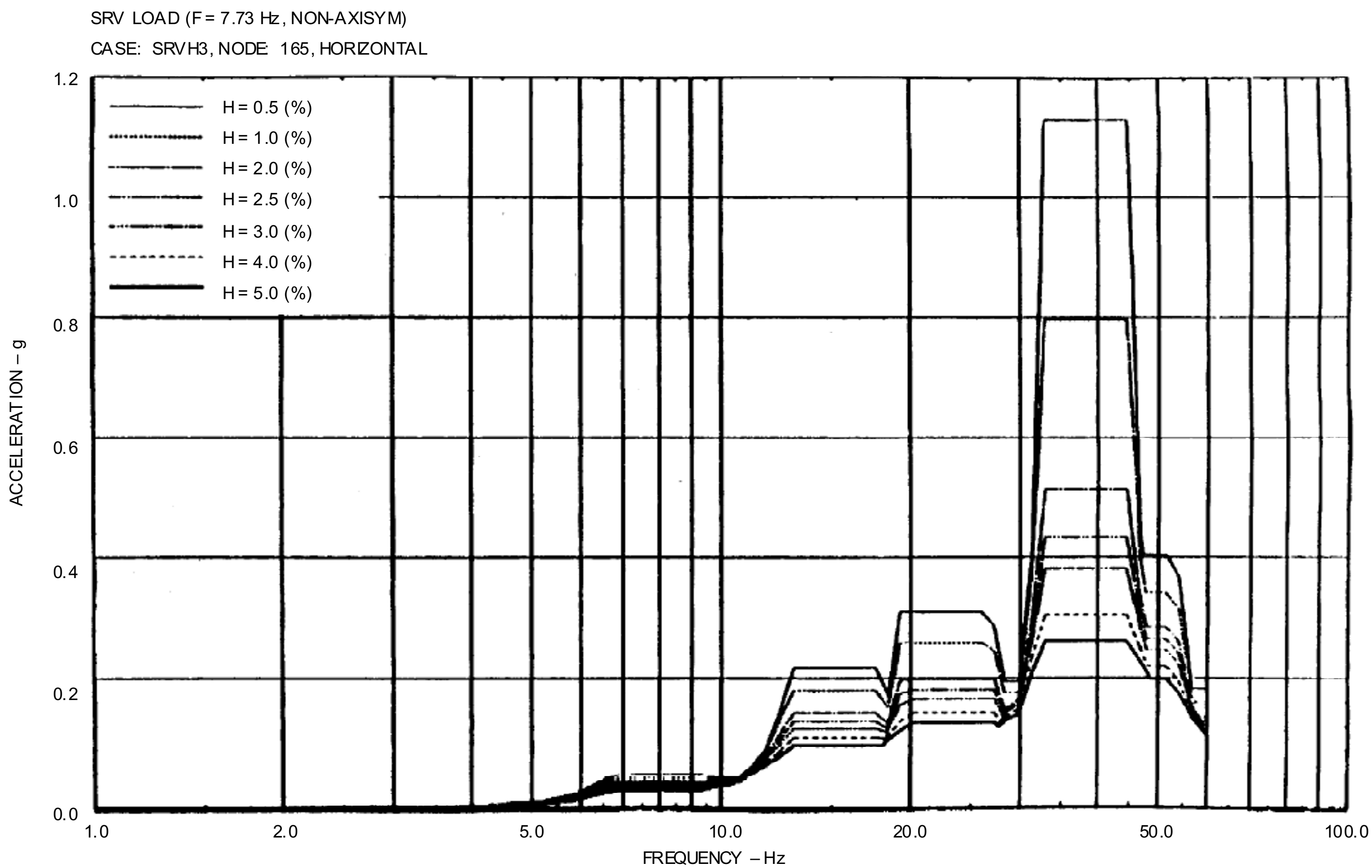

Figure 3G-43 Floor Response Spectrum—Case: SRVH3, Node: 165, Horizontal

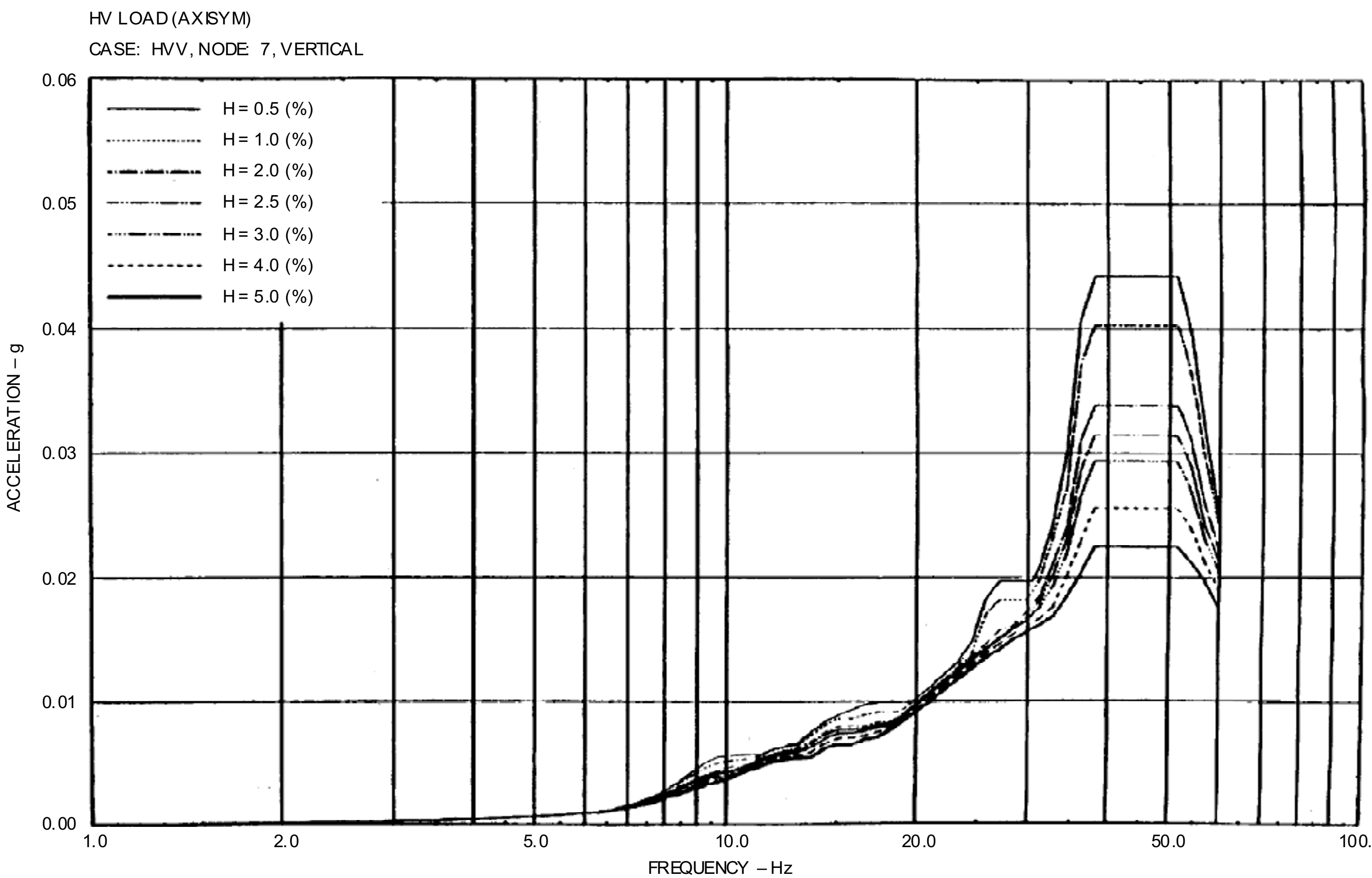

Figure 3G-44 Floor Response Spectrum—Case: HVV, Node: 7, Vertical

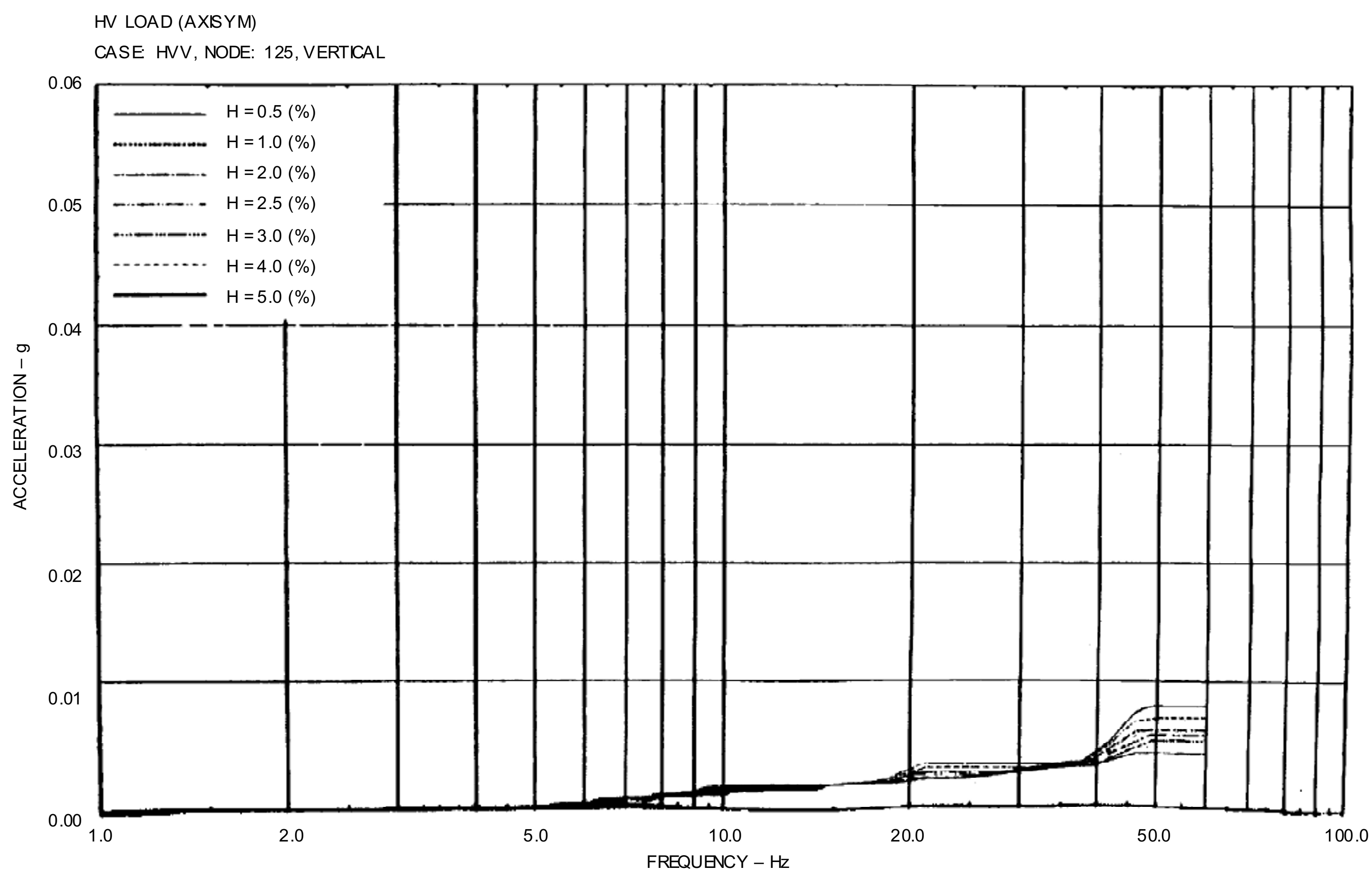

Figure 3G-45 Floor Response Spectrum—Case: HVV, Node: 125, Vertical

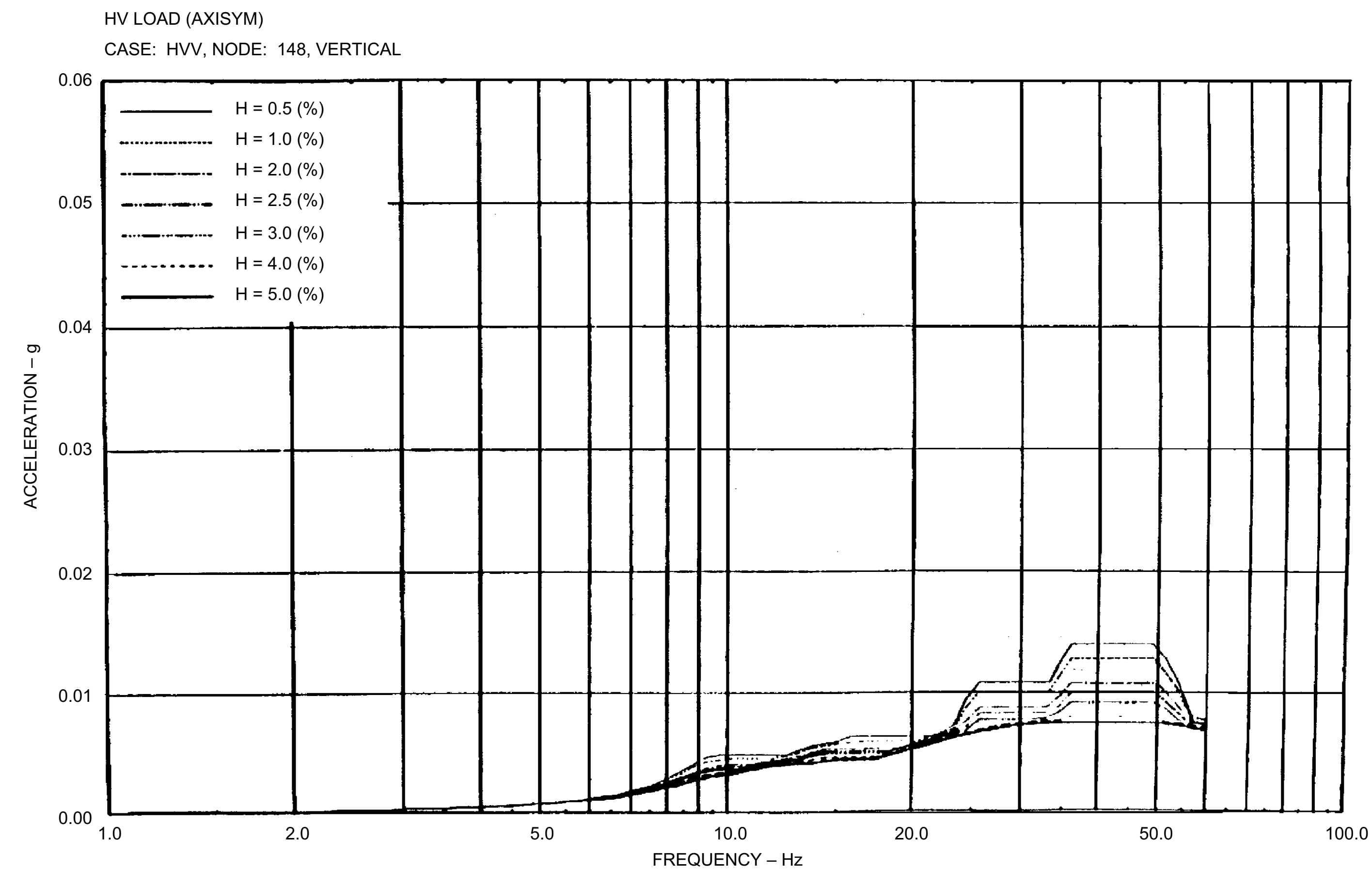

Figure 3G-46 Floor Response Spectrum—Case: HVV, Node: 148, Vertical

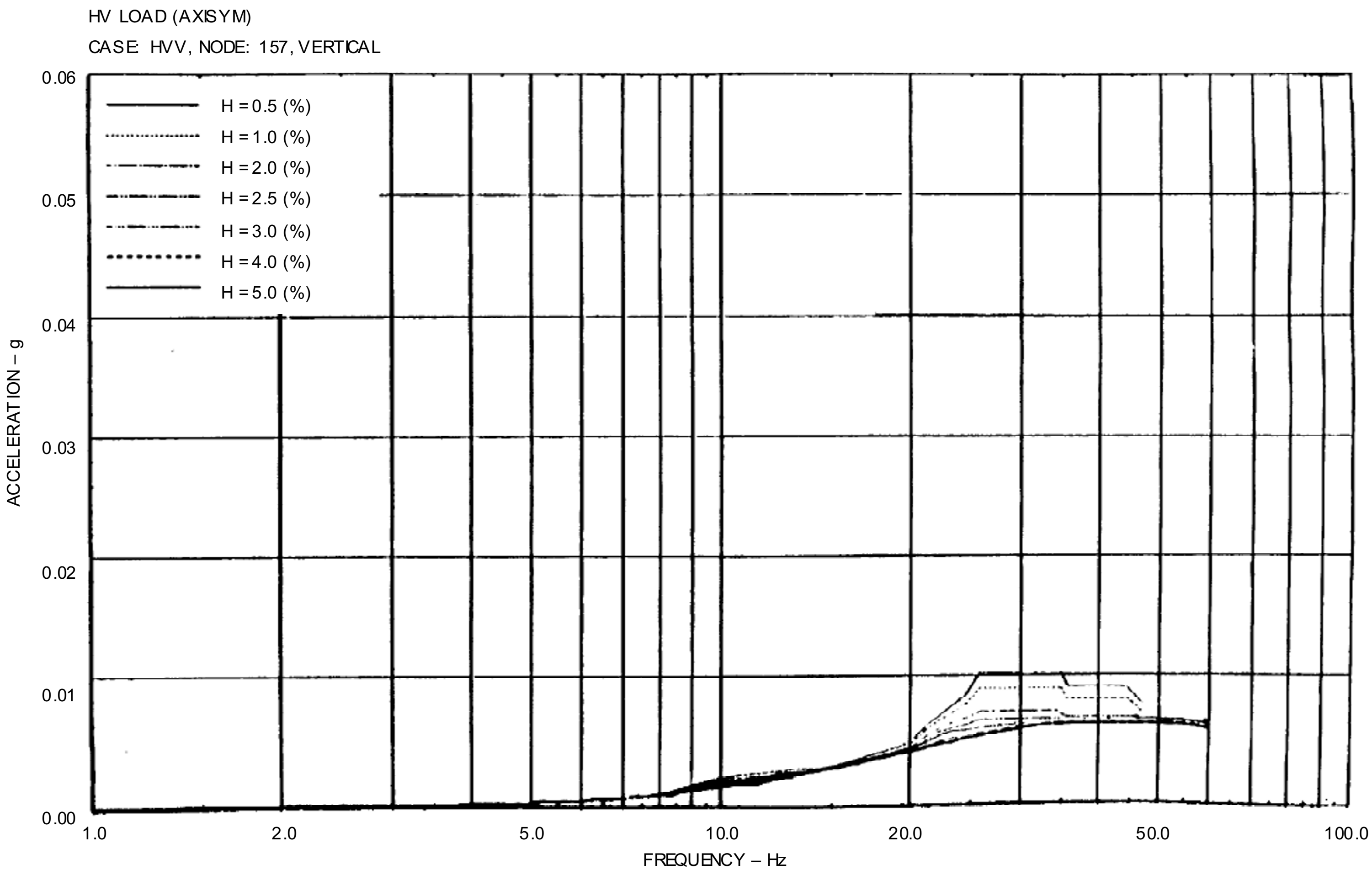

Figure 3G-47 Floor Response Spectrum—Case: HVV, Node: 157, Vertical

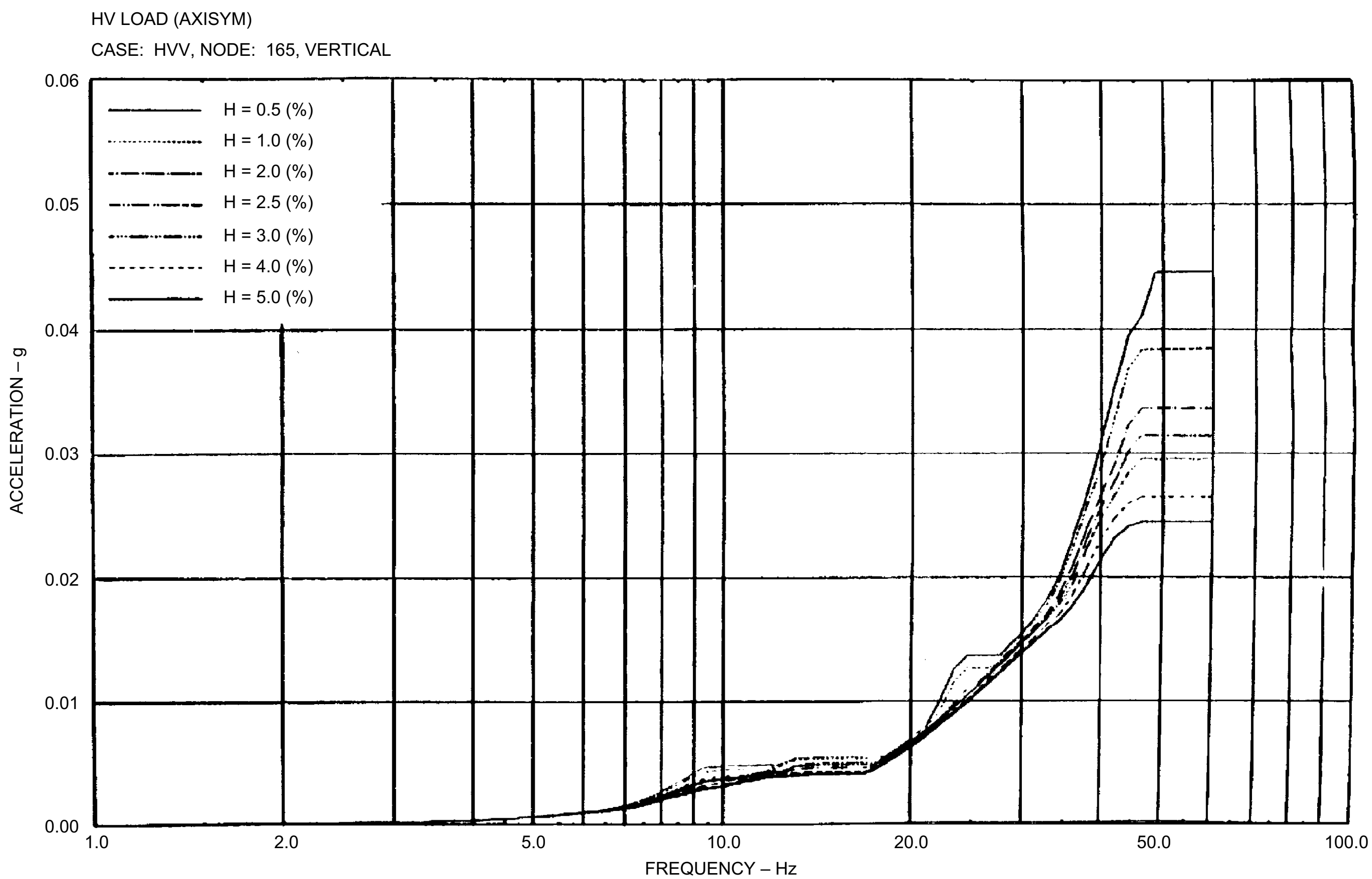

Figure 3G-48 Floor Response Spectrum—Case: HVV, Node: 165, Vertical

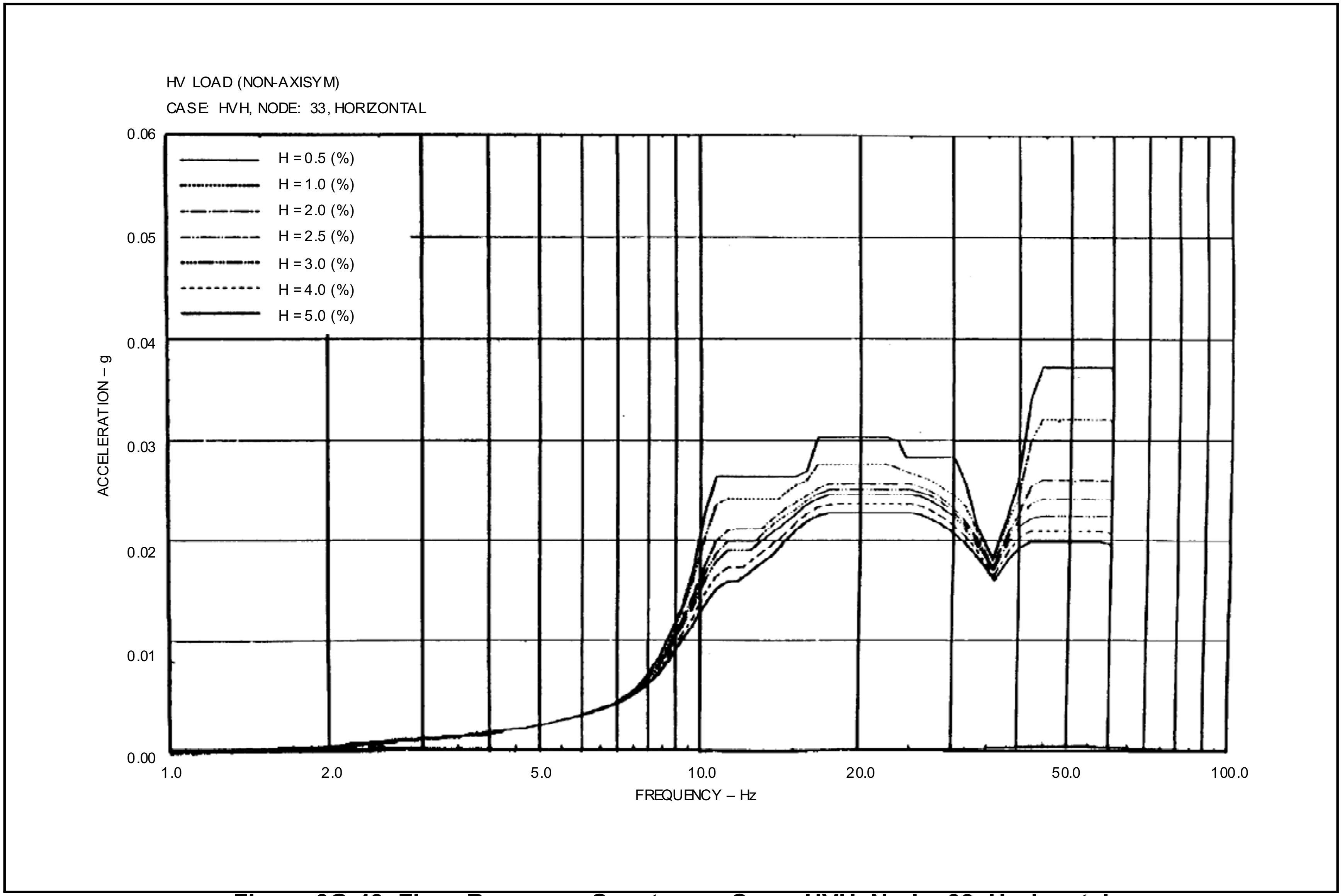

Figure 3G-49 Floor Response Spectrum—Case: HVH, Node: 33, Horizontal

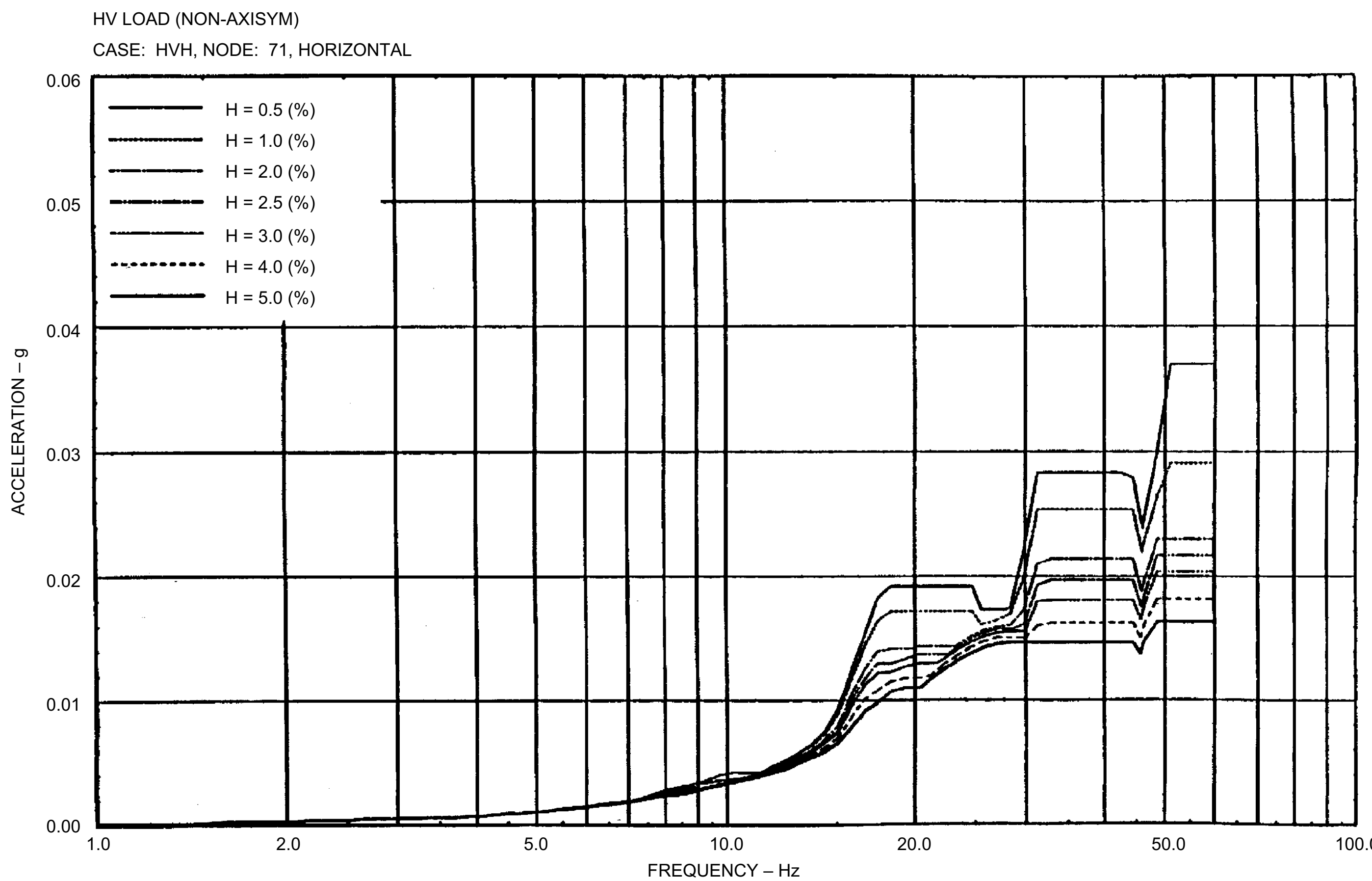

Figure 3G-50 Floor Response Spectrum—Case: HVH, Node: 71, Horizontal

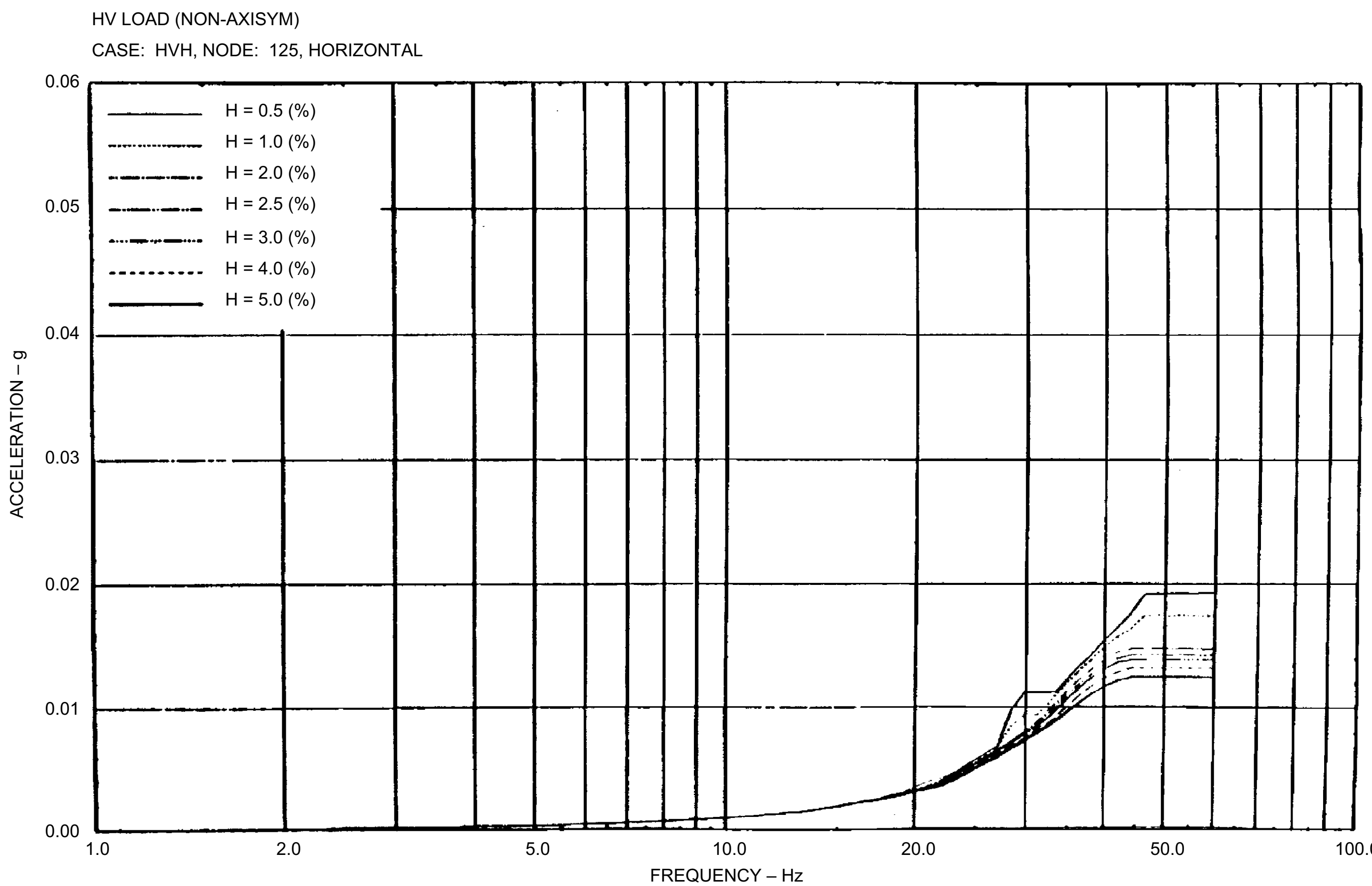

Figure 3G-51 Floor Response Spectrum—Case: HVH, Node: 125, Horizontal

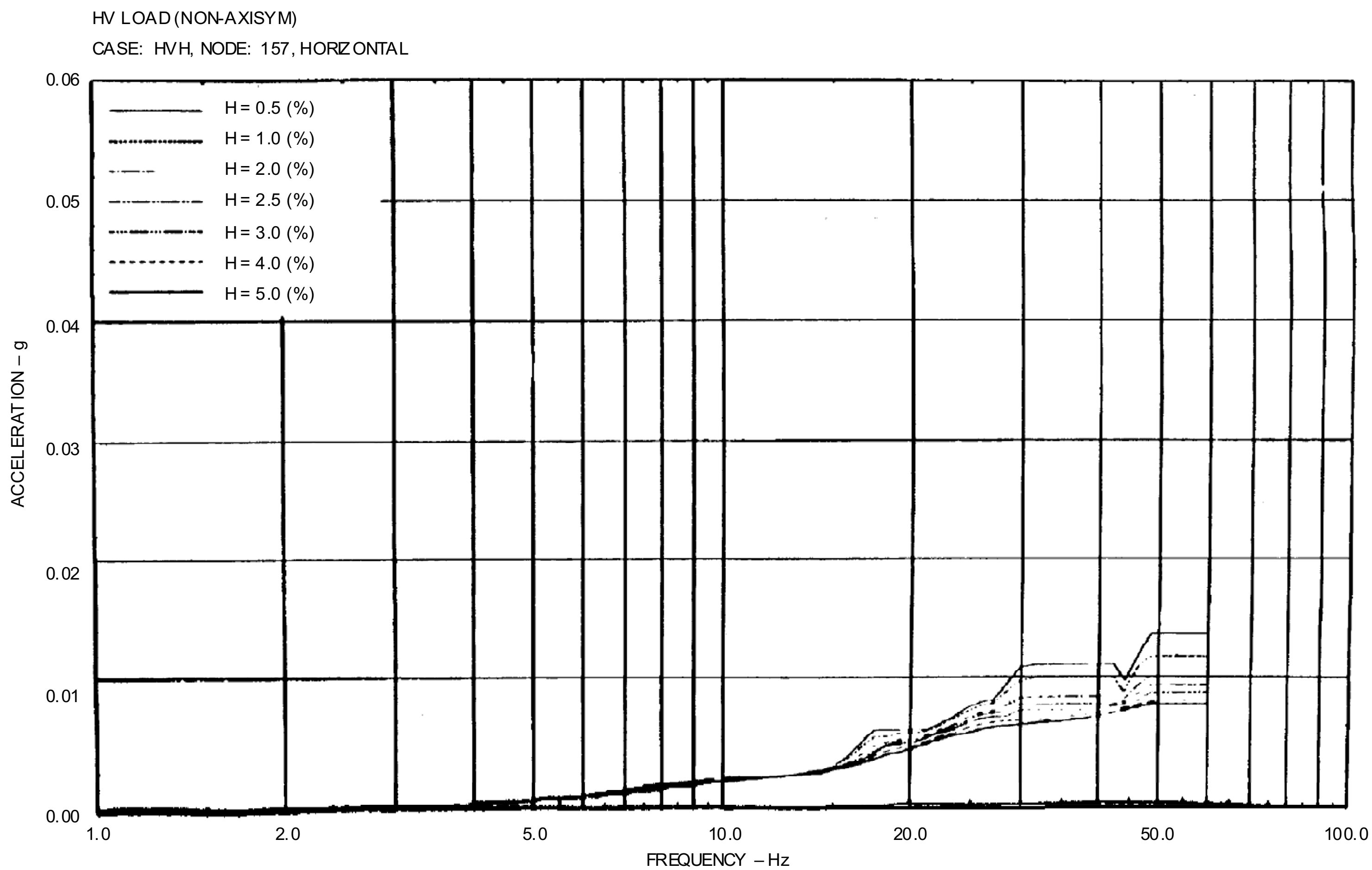

Figure 3G-52 Floor Response Spectrum—Case: HVH, Node: 157, Horizontal

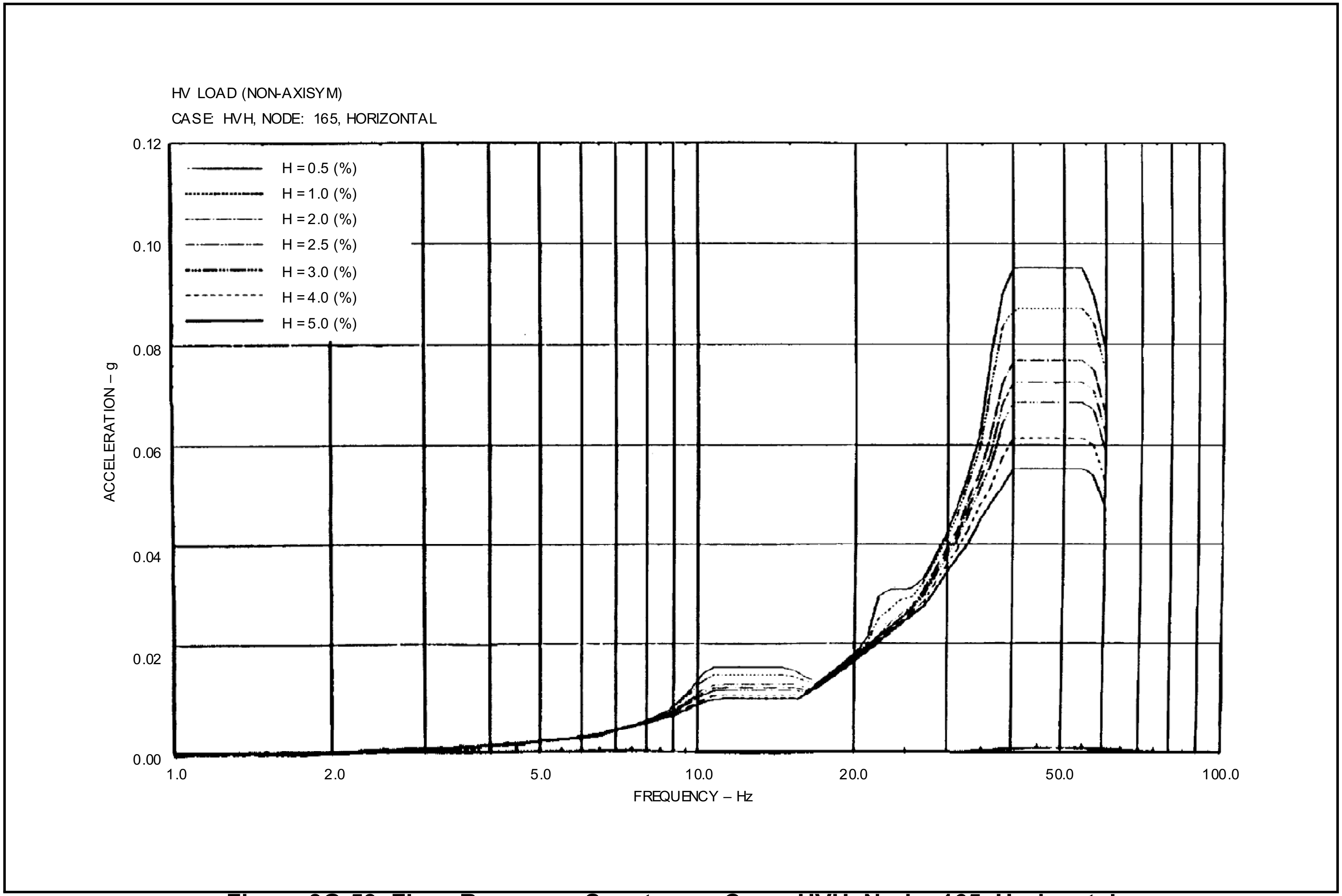

Figure 3G-53 Floor Response Spectrum—Case: HVH, Node: 165, Horizontal

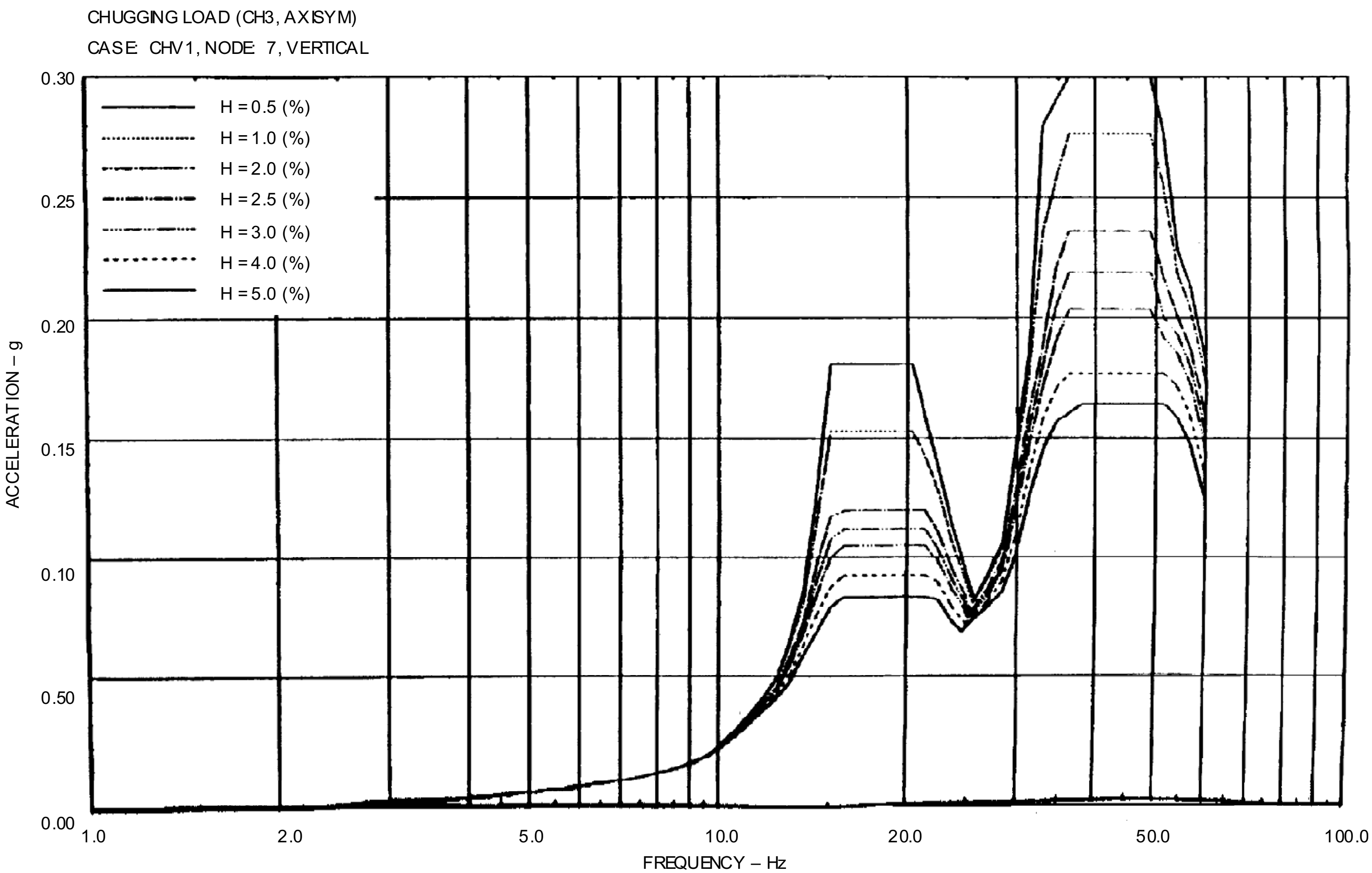

Figure 3G-54 Floor Response Spectrum—Case: CHV1, Node: 7, Vertical

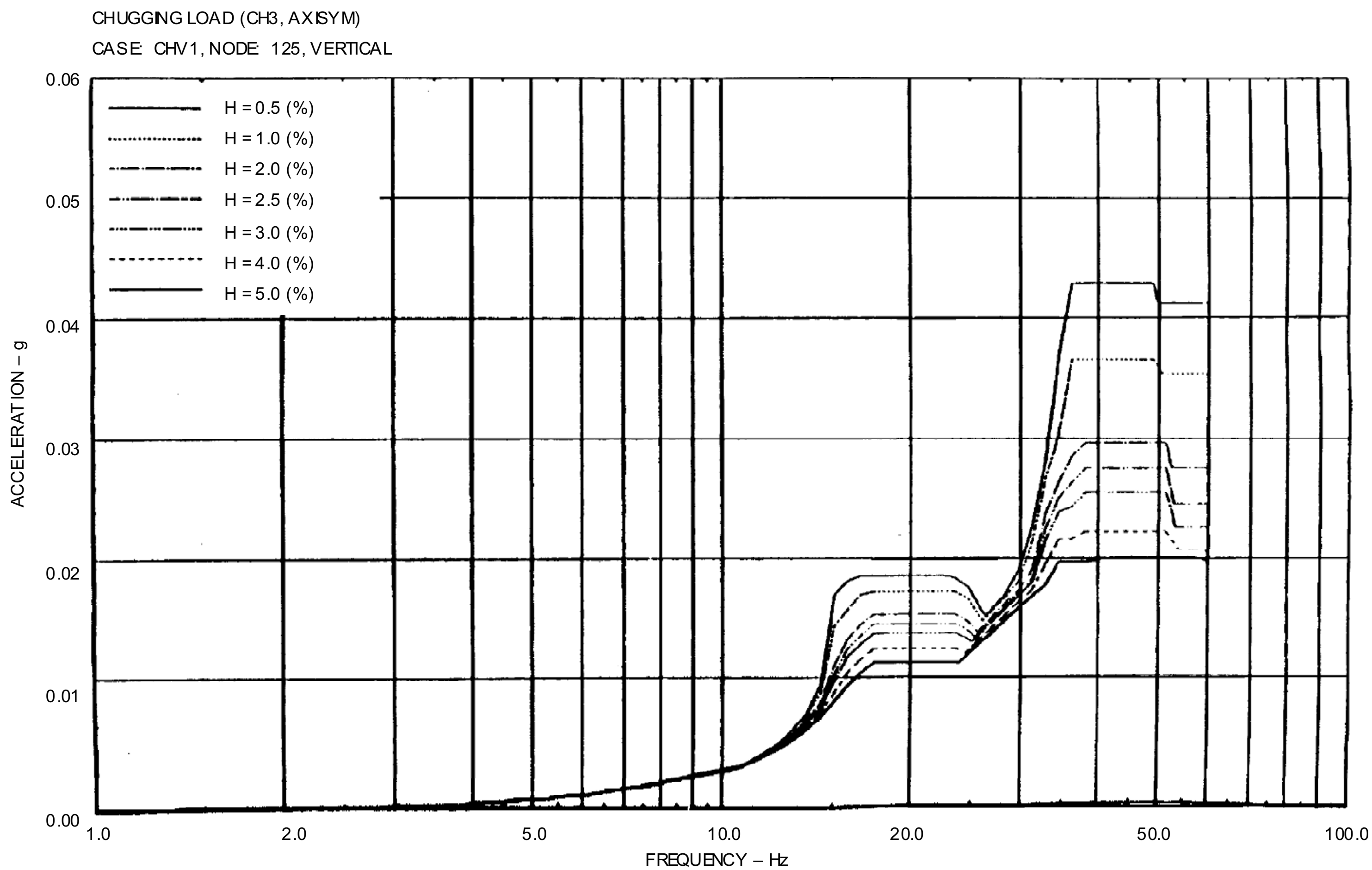

Figure 3G-55 Floor Response Spectrum—Case: CHV1, Node: 125, Vertical

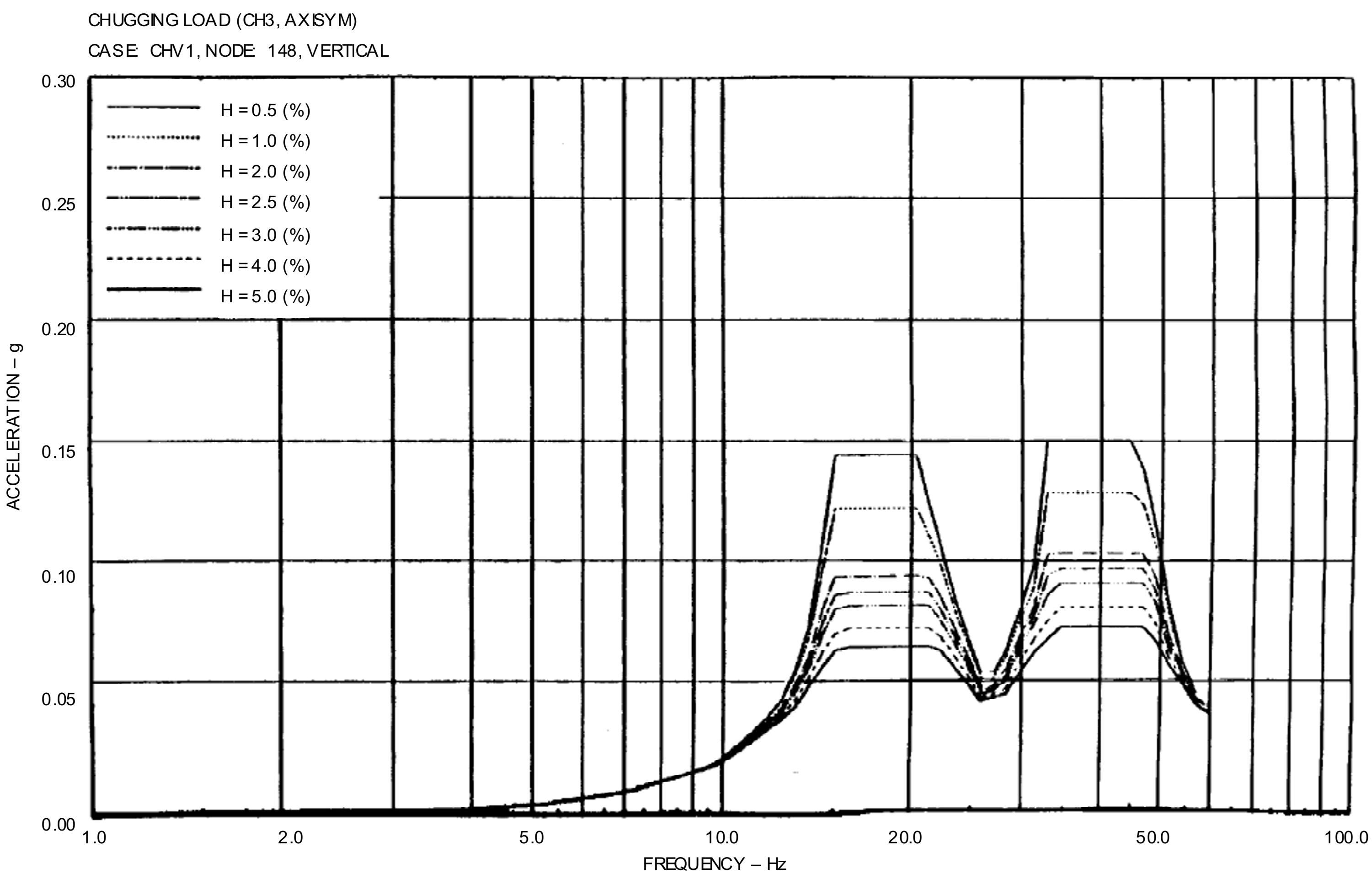

Figure 3G-56 Floor Response Spectrum—Case: CHV1, Node: 148, Vertical

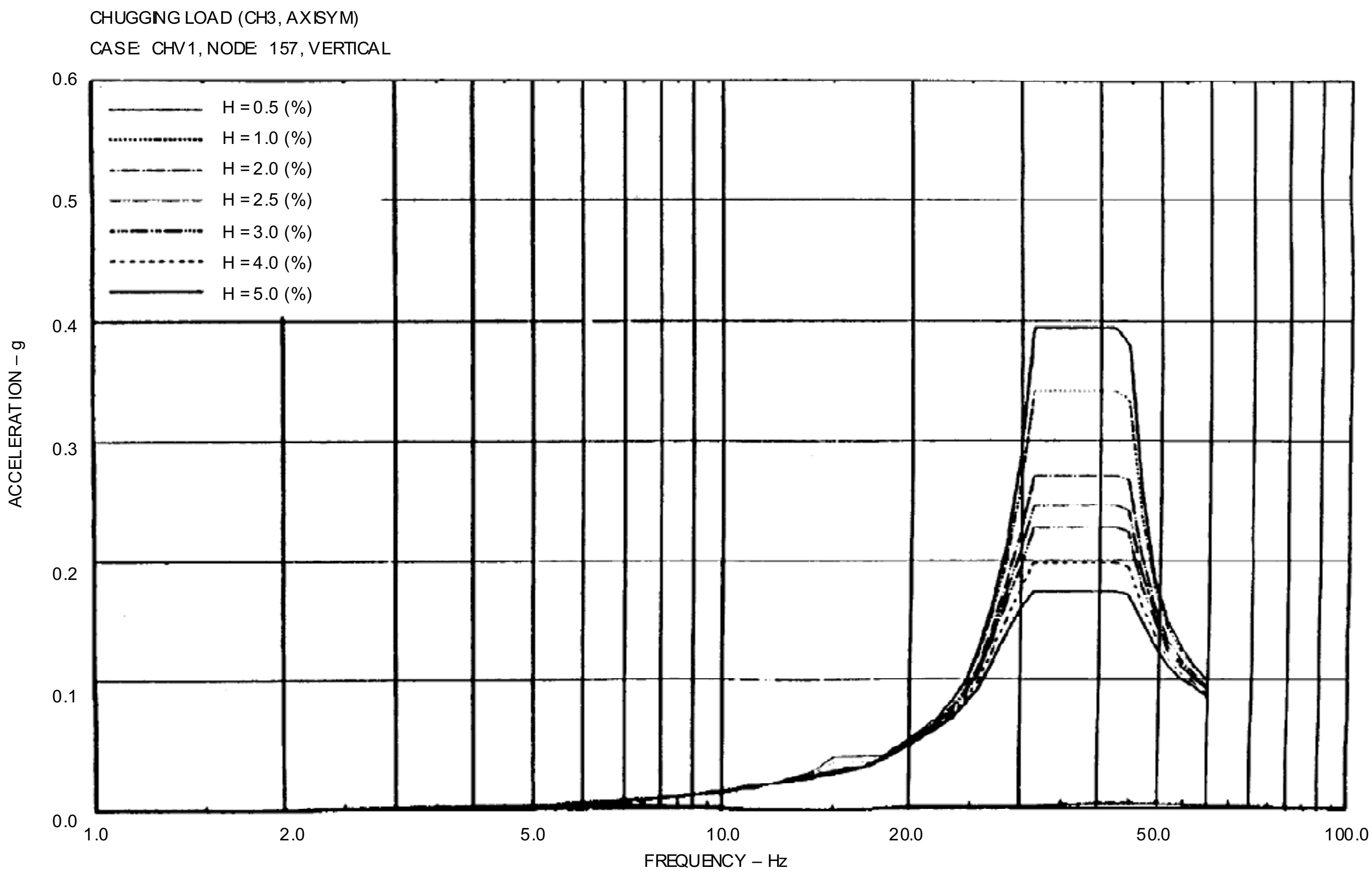

Figure 3G-57 Floor Response Spectrum—Case: CHV1, Node: 157, Vertical

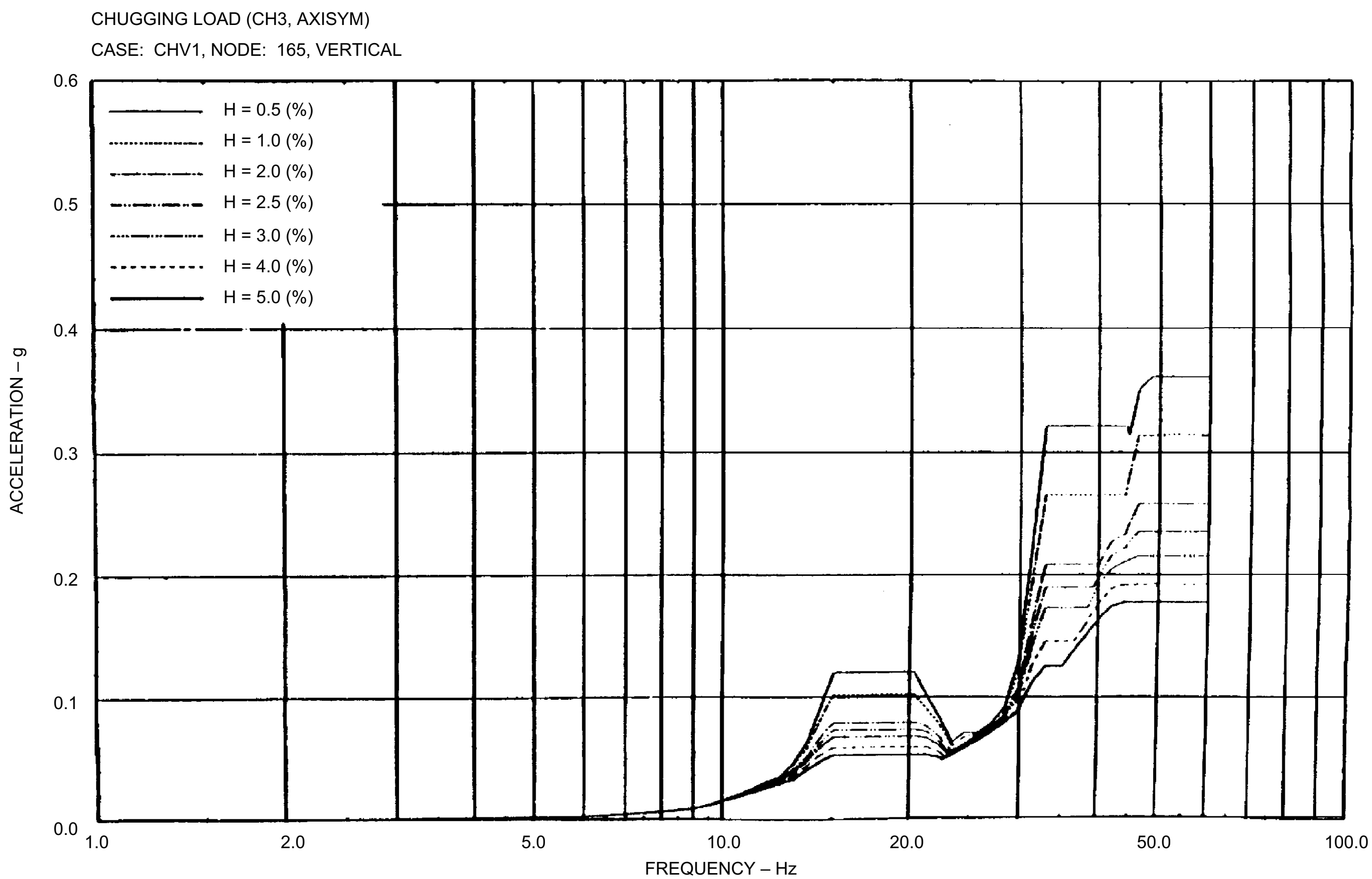

Figure 3G-58 Floor Response Spectrum—Case: CHV1, Node: 165, Vertical

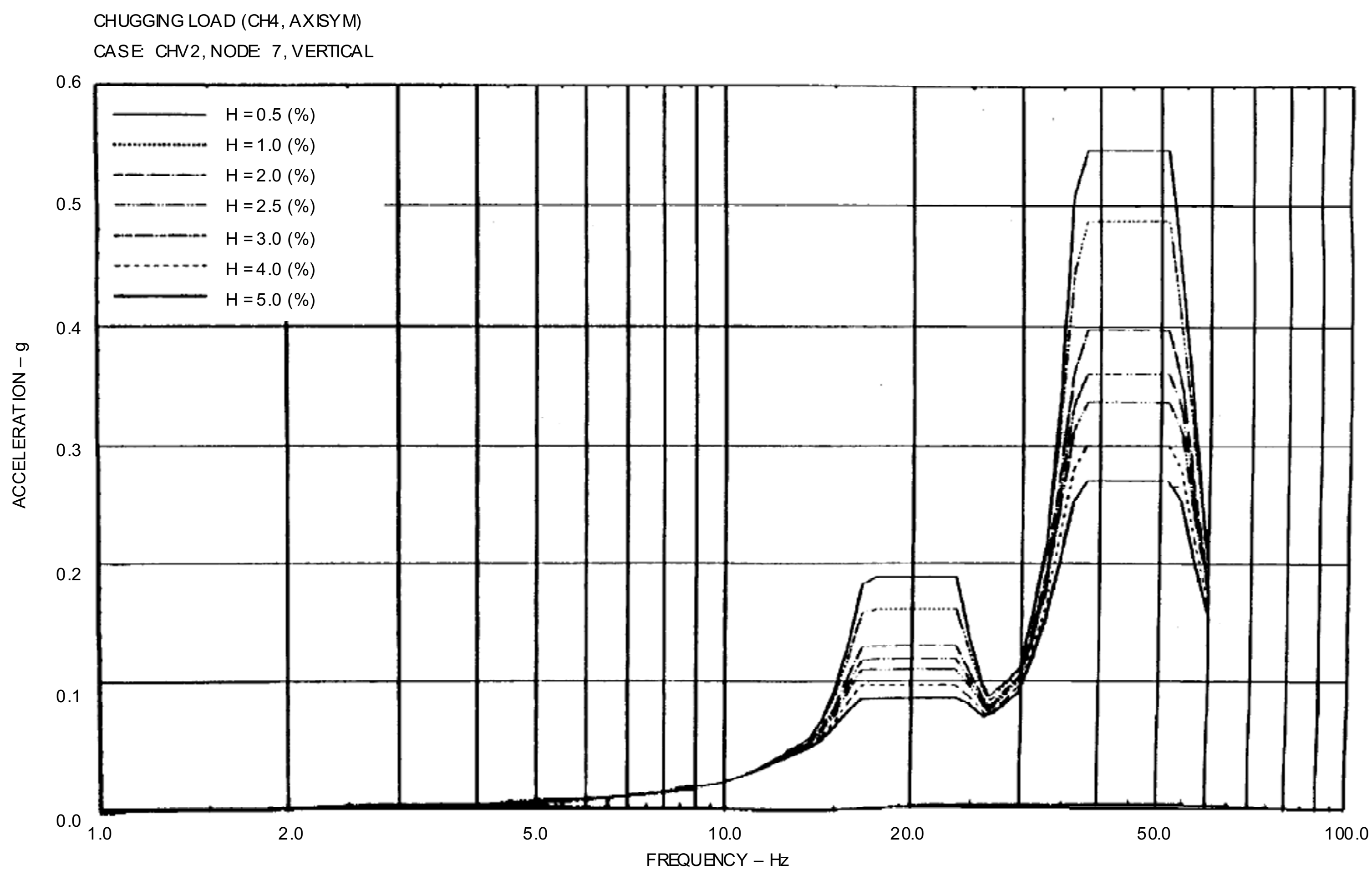

Figure 3G-59 Floor Response Spectrum—Case: CHV2, Node: 7, Vertical

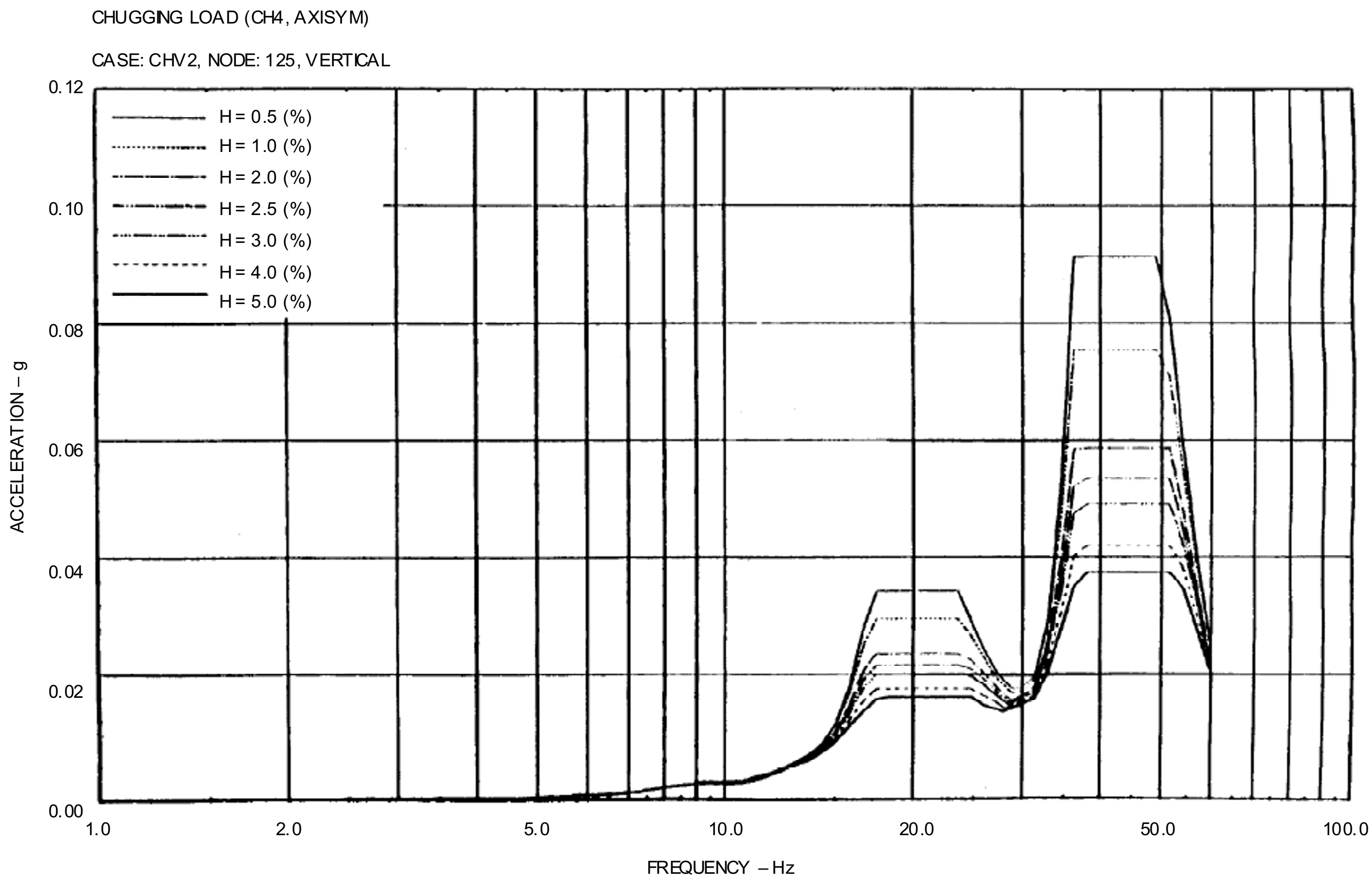

Figure 3G-60 Floor Response Spectrum—Case: CHV2, Node: 125, Vertical

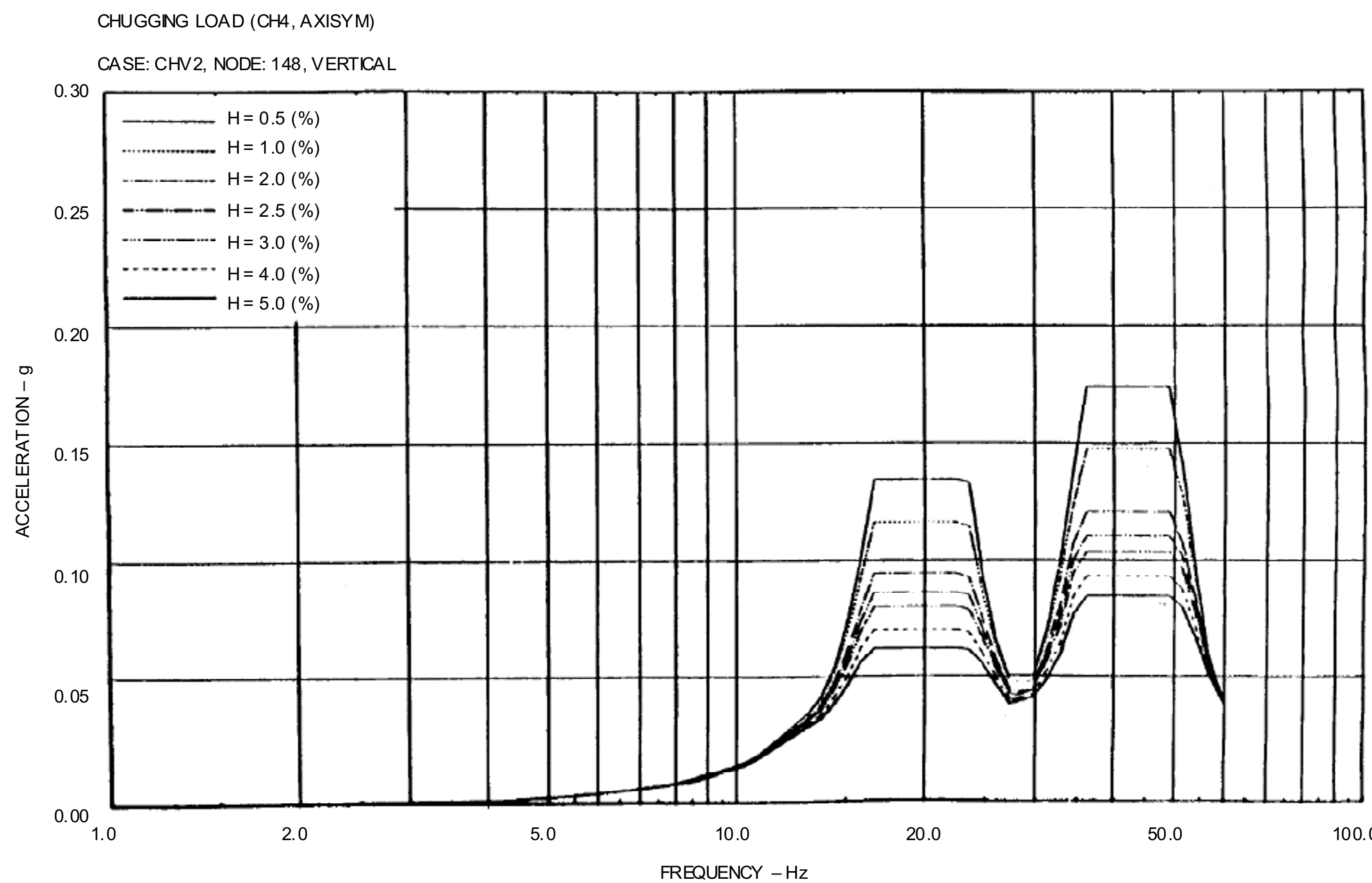

Figure 3G-61 Floor Response Spectrum—Case: CHV2, Node: 148, Vertical

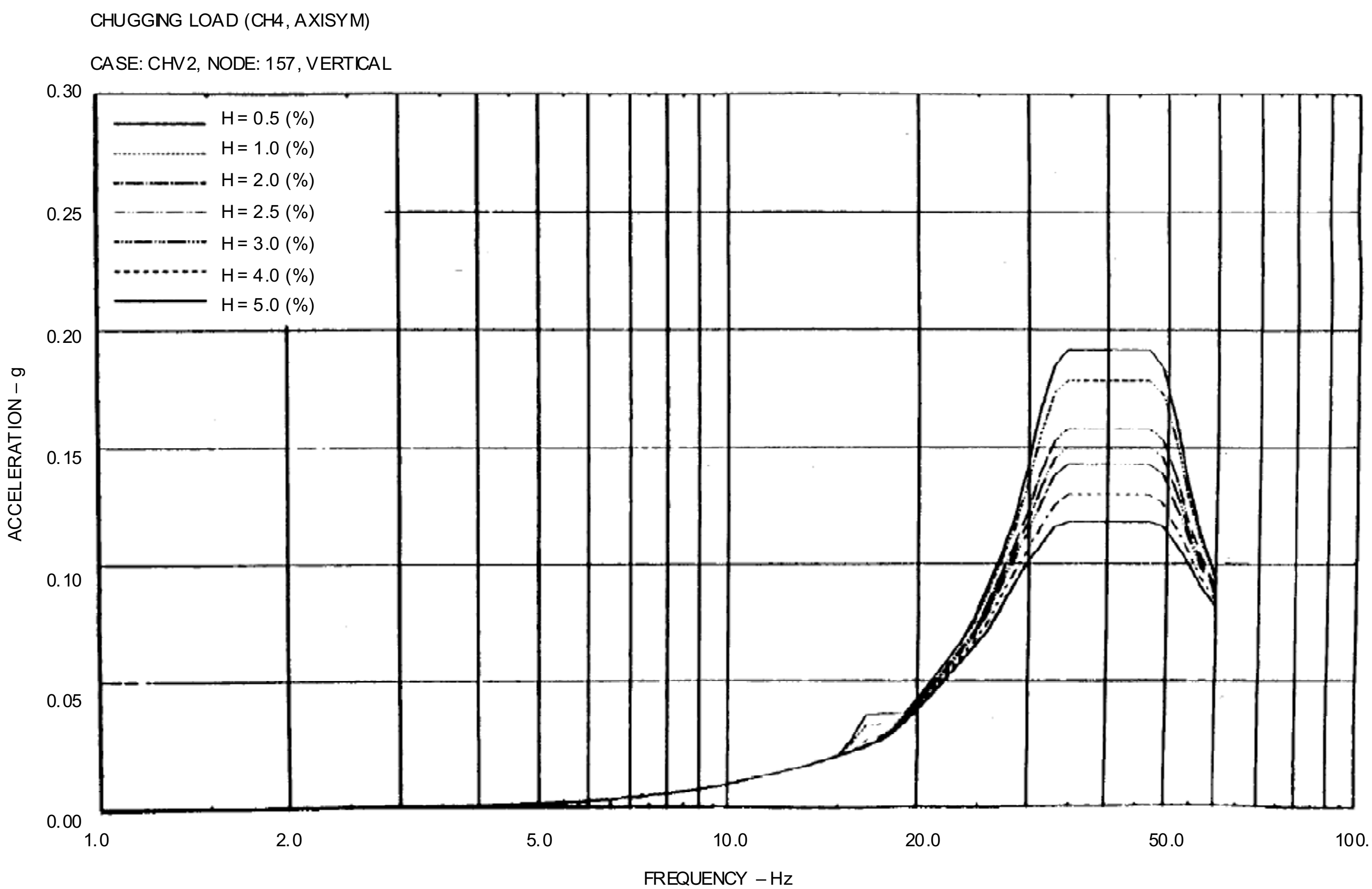

Figure 3G-62 Floor Response Spectrum—Case: CHV2, Node: 157, Vertical

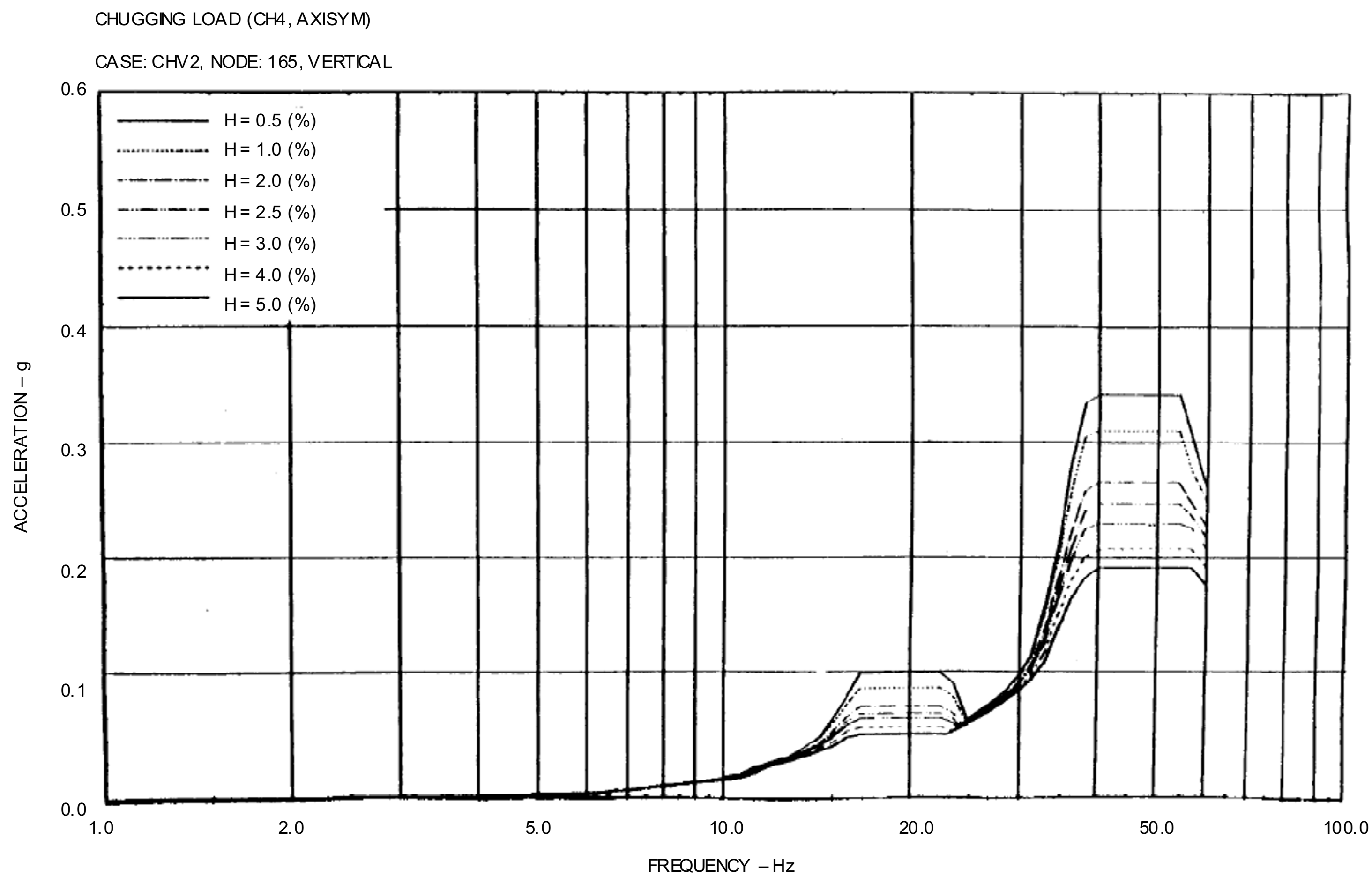

Figure 3G-63 Floor Response Spectrum—Case: CHV2, Node: 165, Vertical

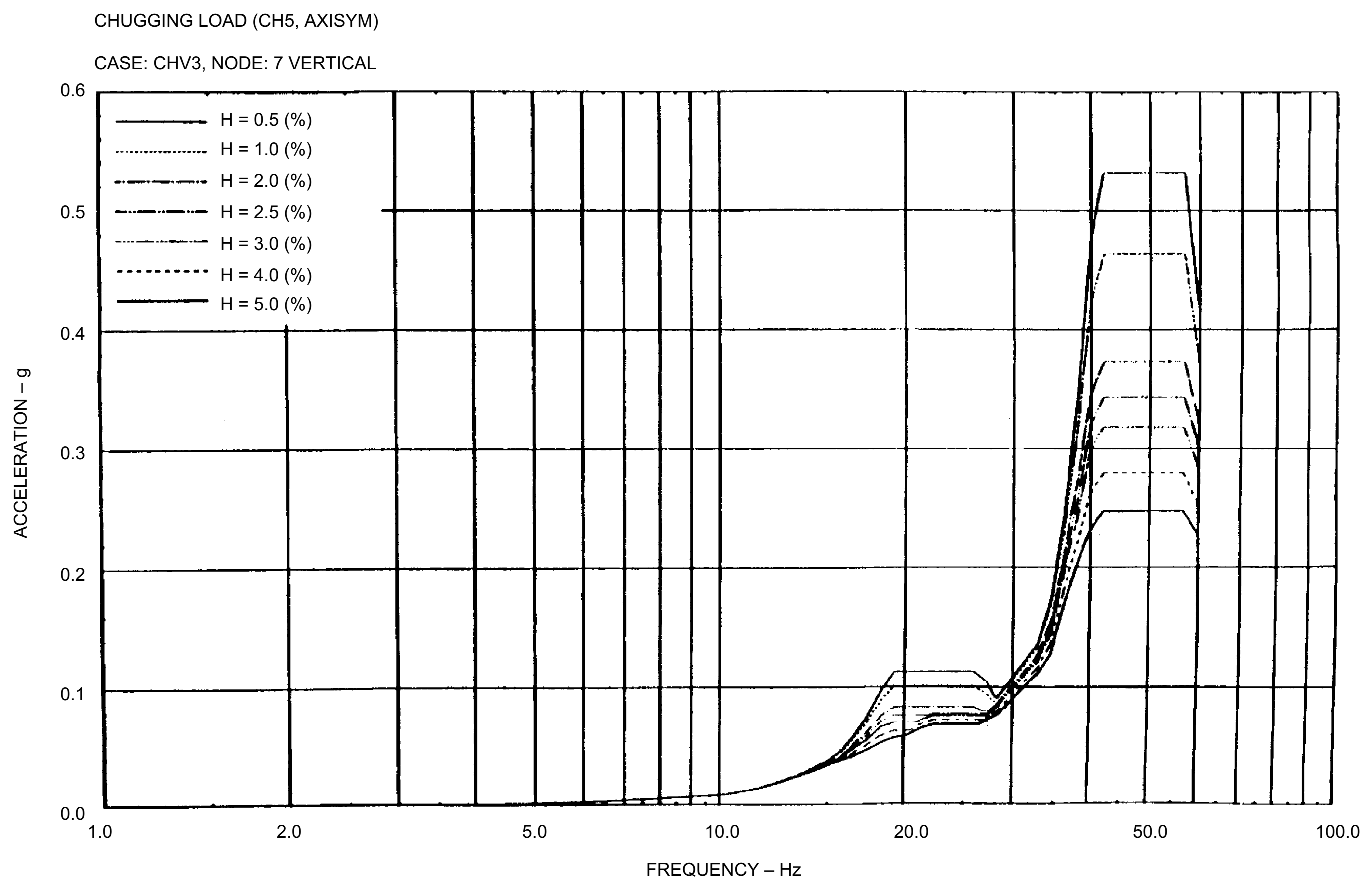

Figure 3G-64 Floor Response Spectrum—Case: CHV3, Node: 7, Vertical

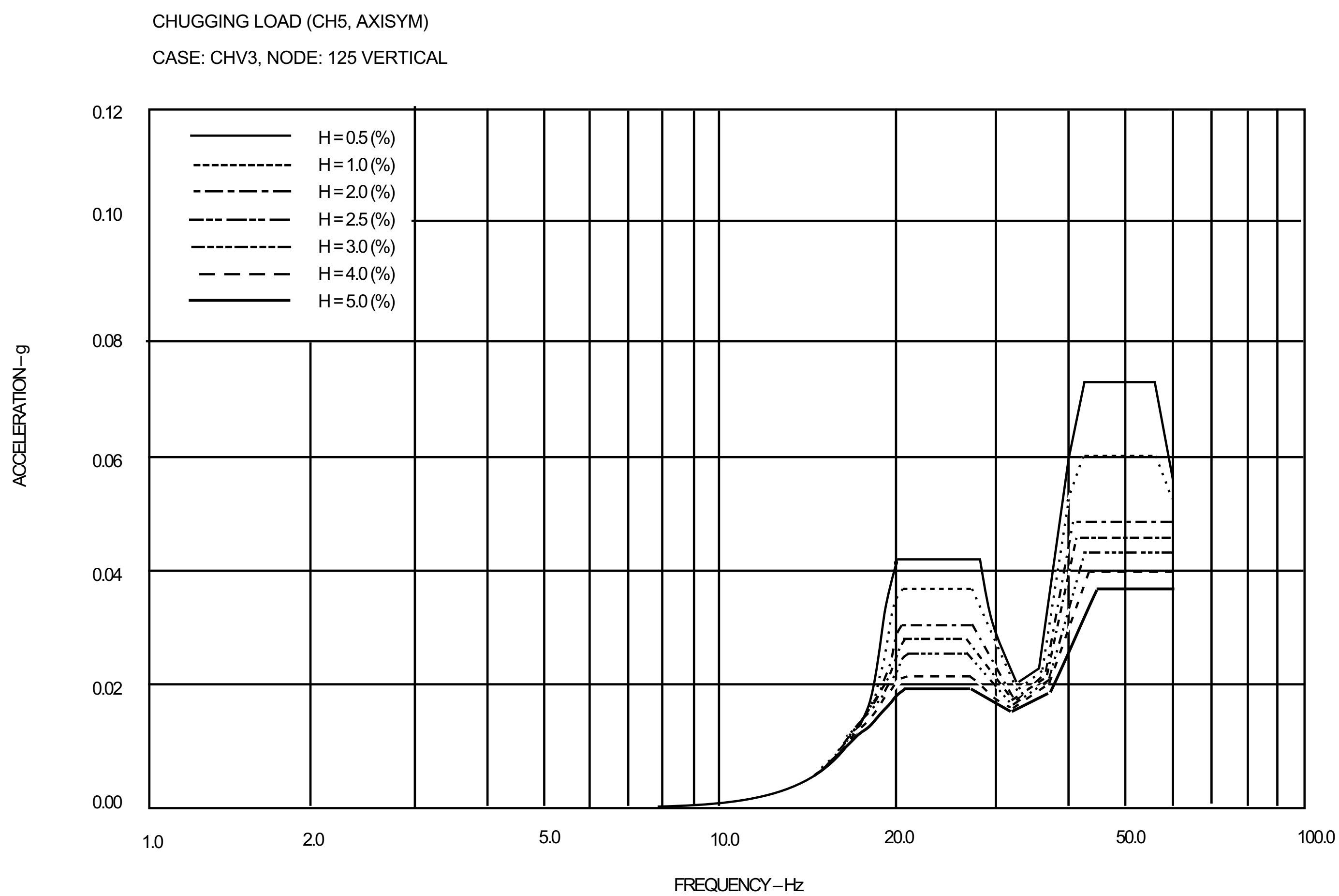

Figure 3G-65 Floor Response Spectrum—Case: CHV3, Node: 125, Vertical

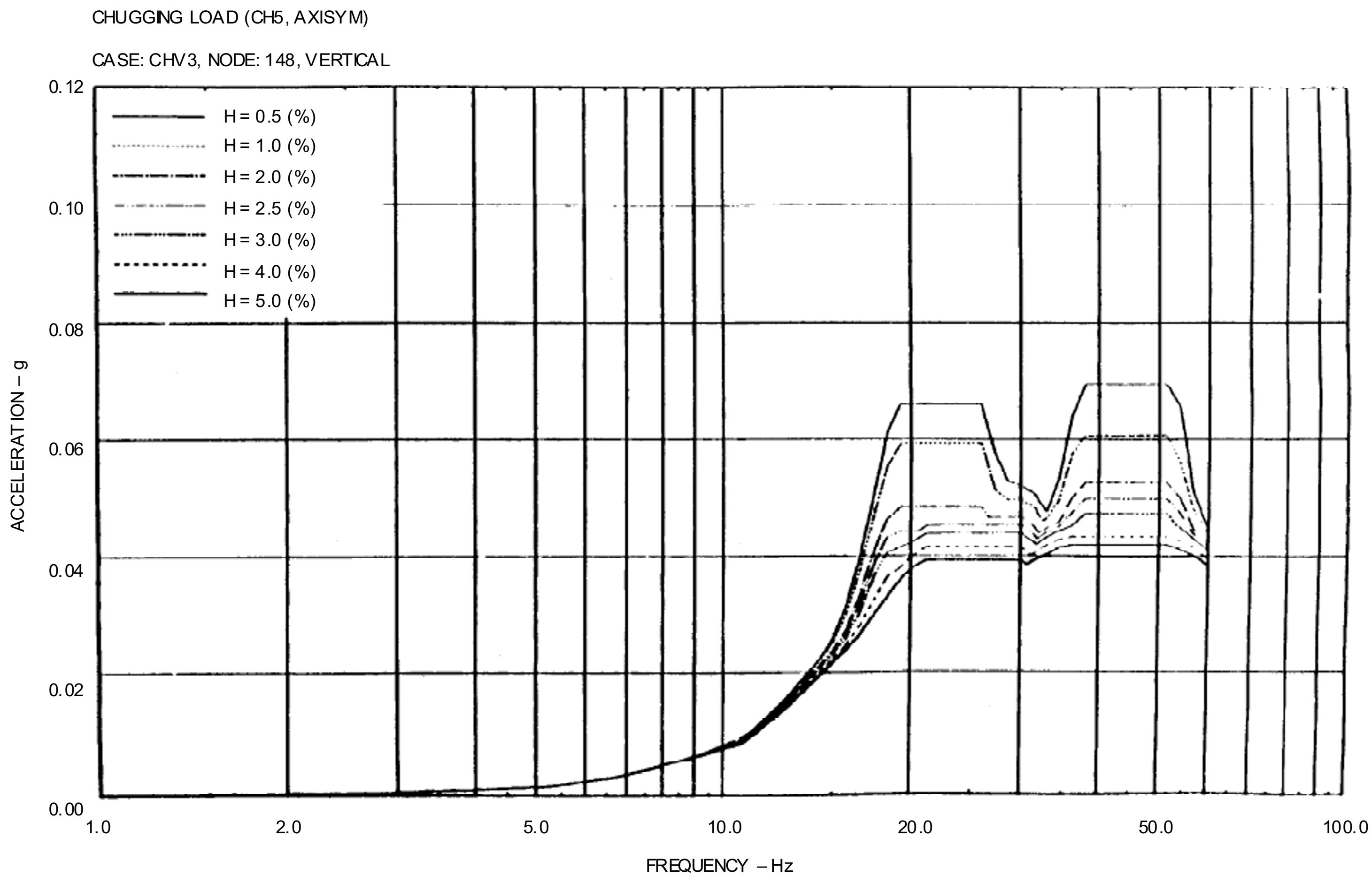

Figure 3G-66 Floor Response Spectrum—Case: CHV3, Node: 148, Vertical

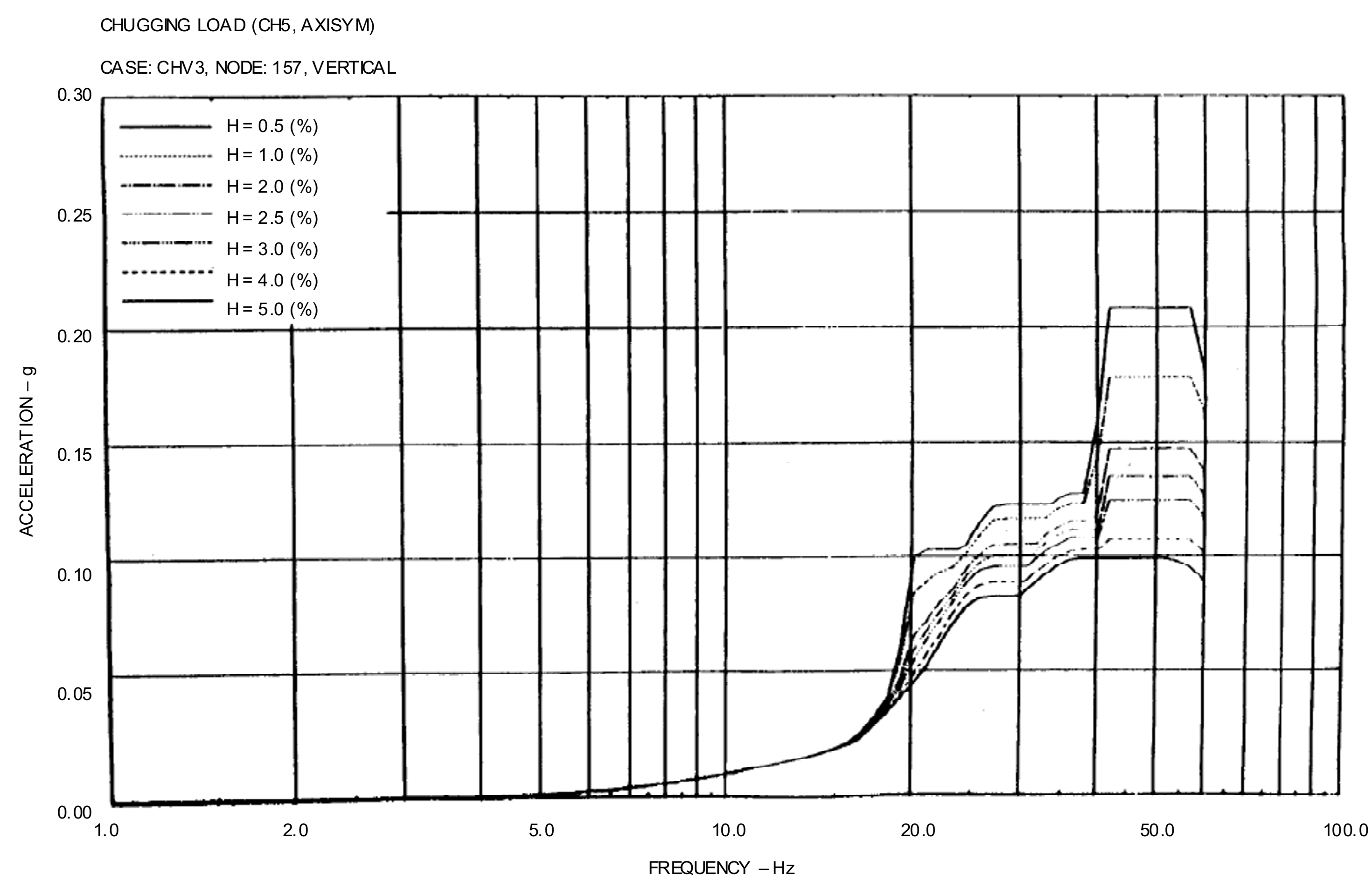

Figure 3G-67 Floor Response Spectrum—Case: CHV3, Node: 157, Vertical

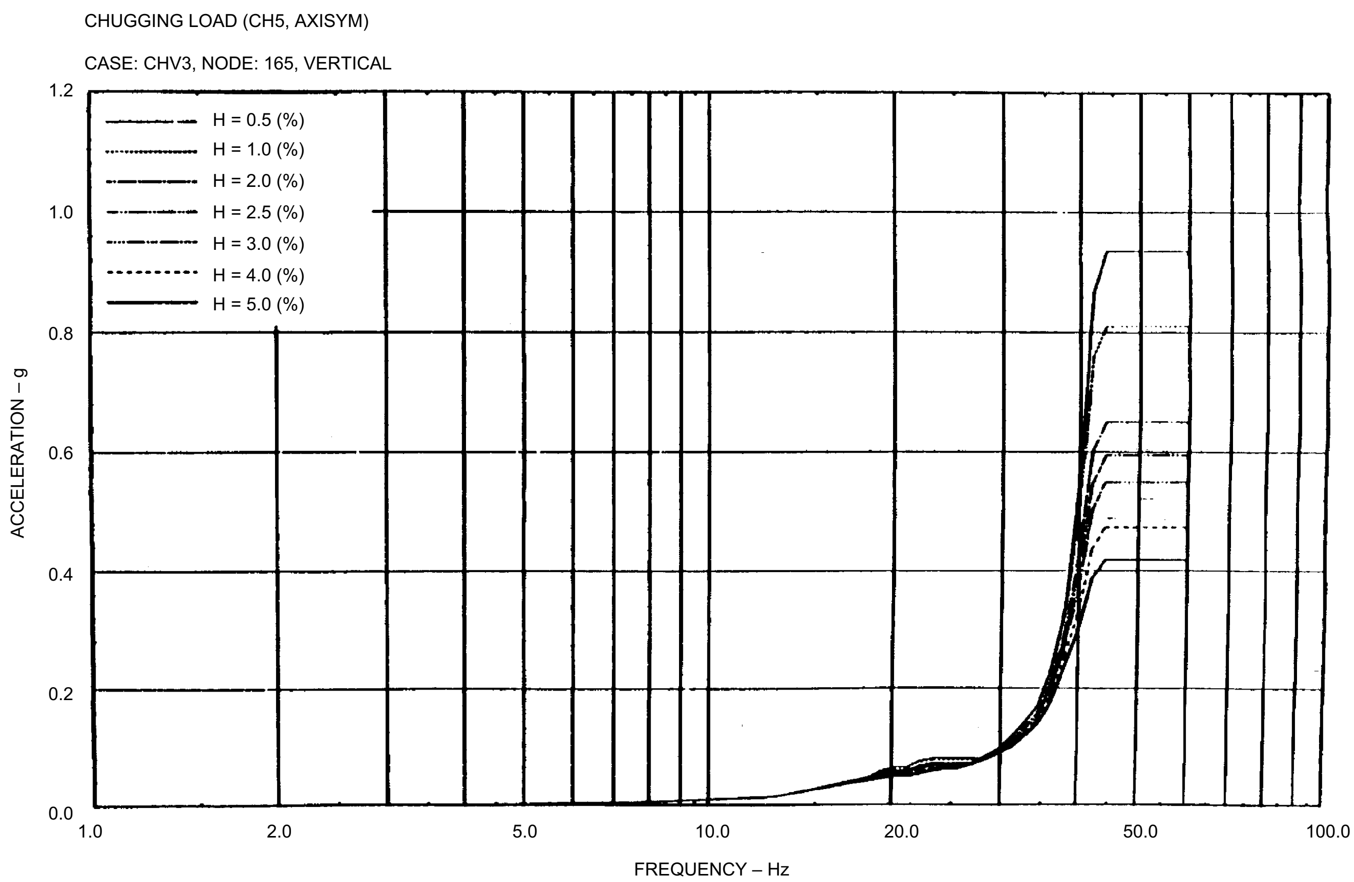

Figure 3G-68 Floor Response Spectrum—Case: CHV3, Node: 165, Vertical

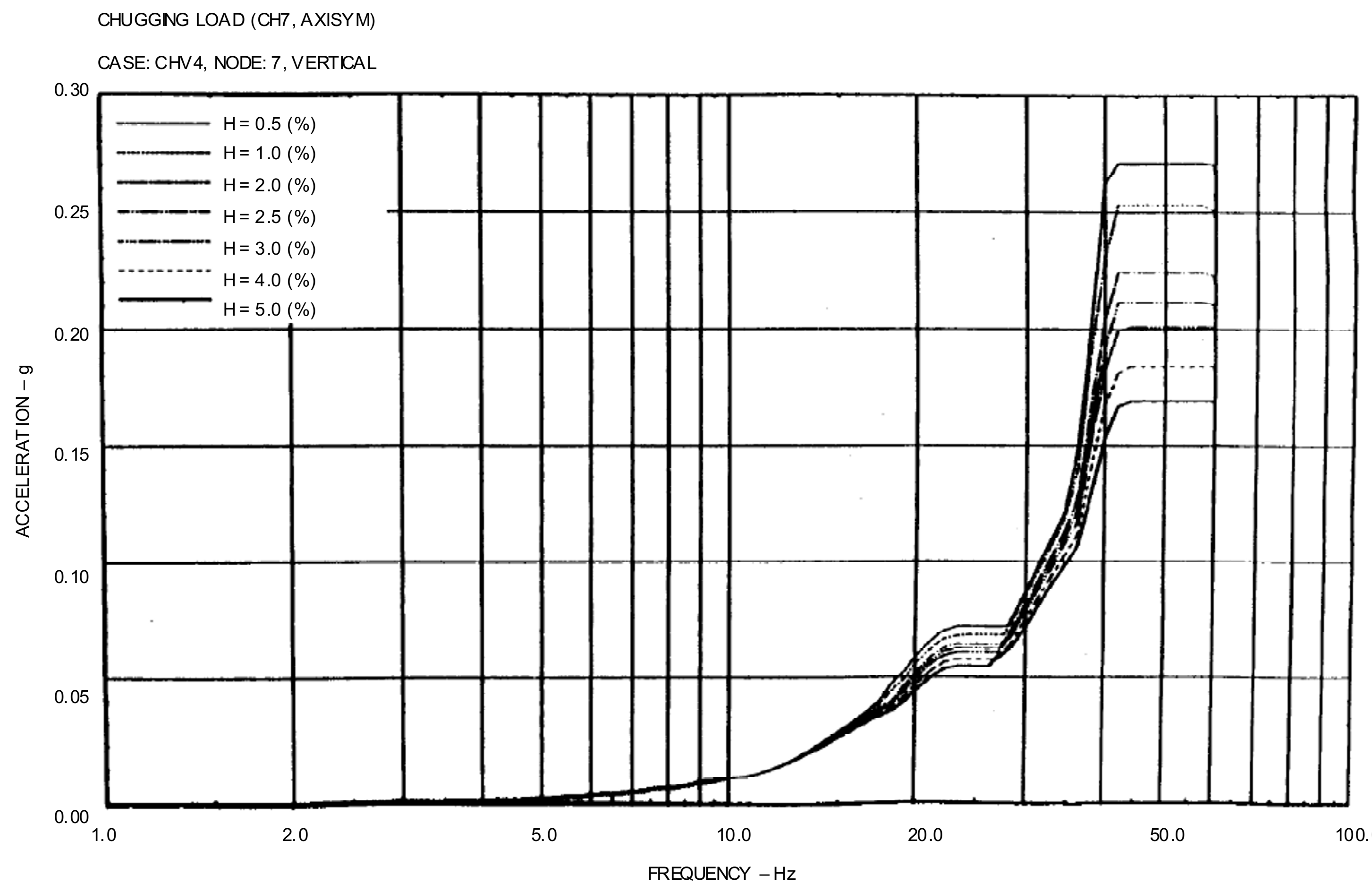

Figure 3G-69 Floor Response Spectrum—Case: CHV4, Node: 7, Vertical

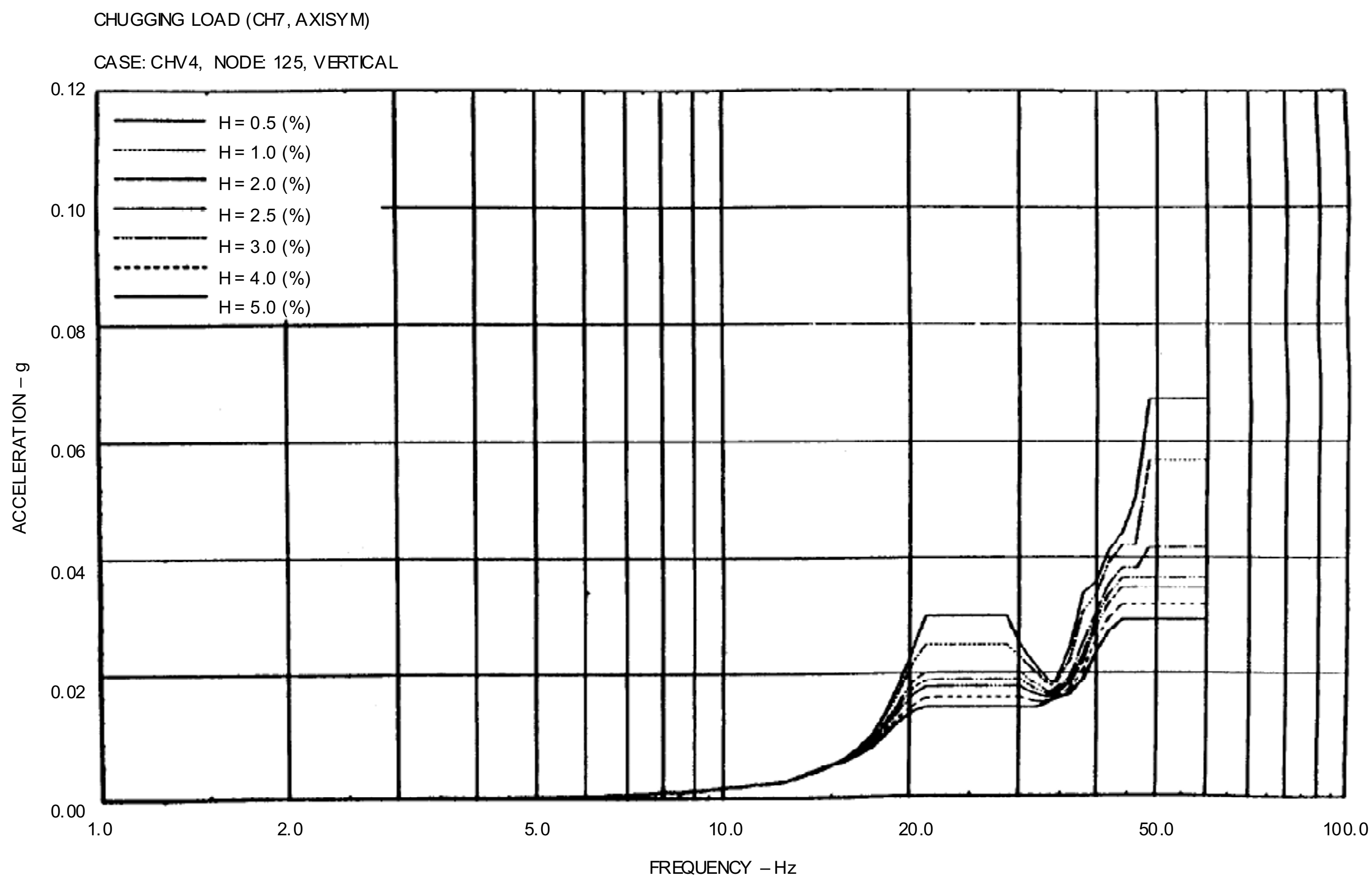

Figure 3G-70 Floor Response Spectrum—Case: CHV4, Node: 125, Vertical

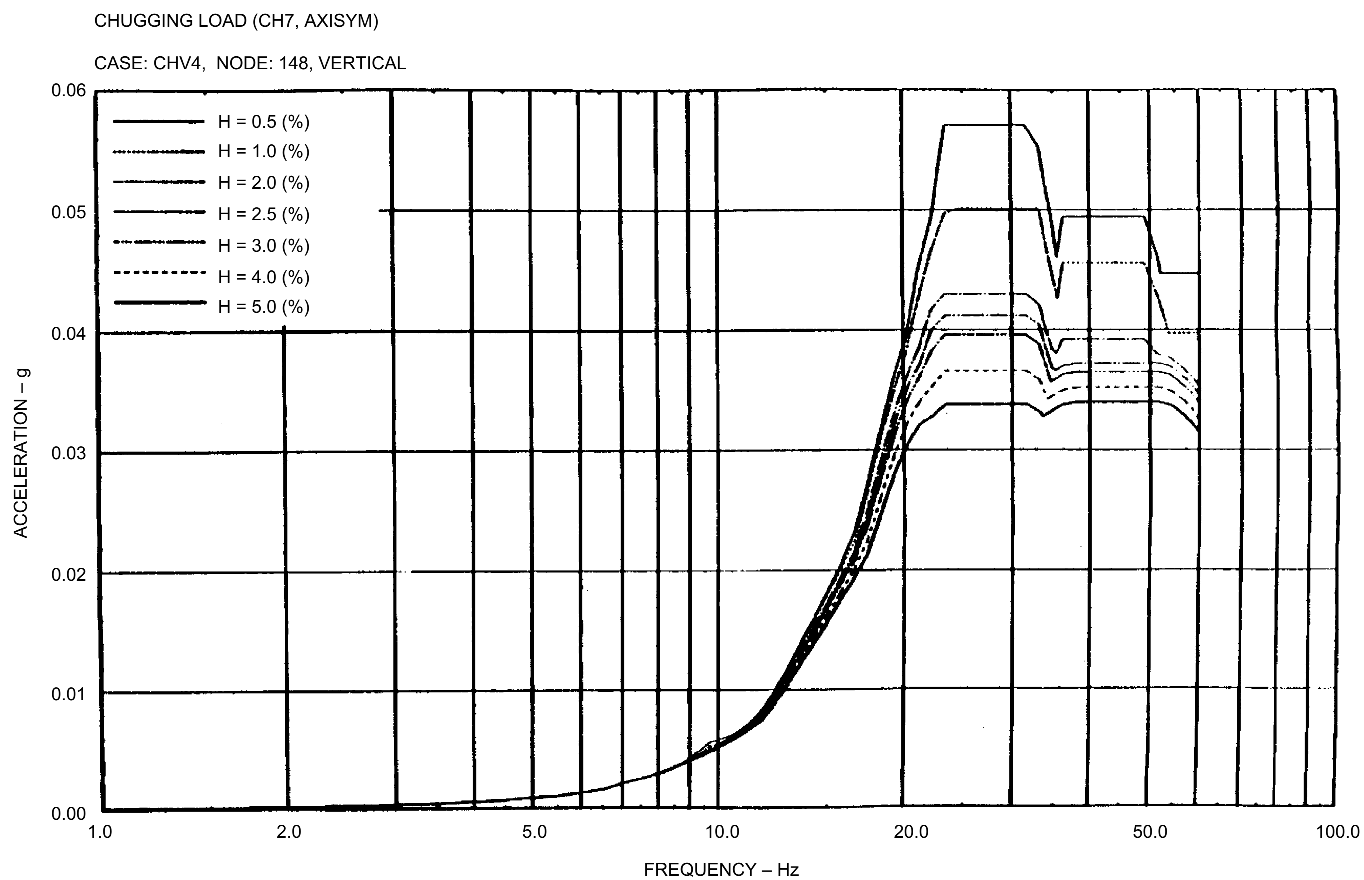

Figure 3G-71 Floor Response Spectrum—Case: CHV4, Node: 148, Vertical

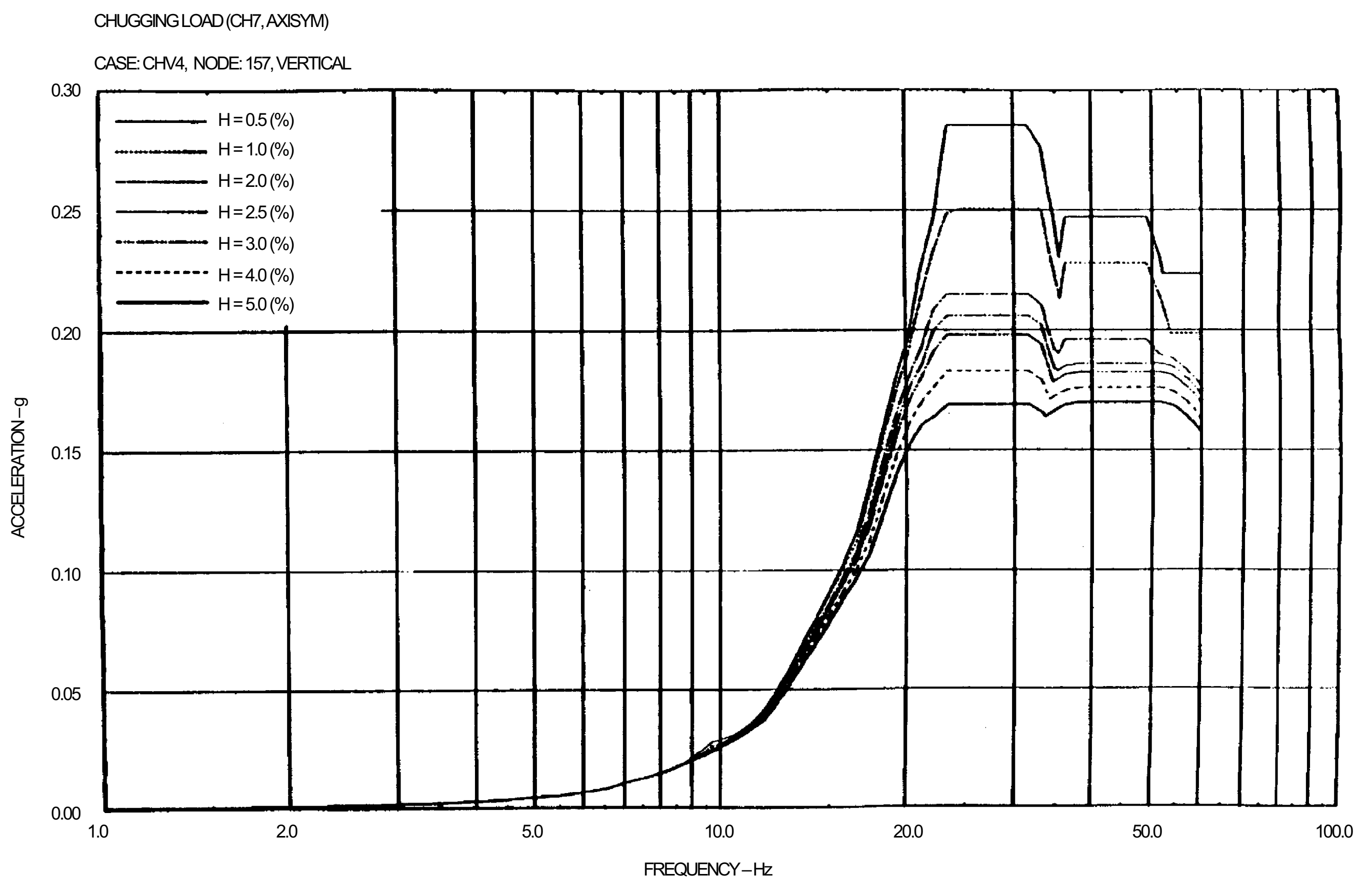

Figure 3G-72 Floor Response Spectrum—Case: CHV4, Node: 157, Vertical

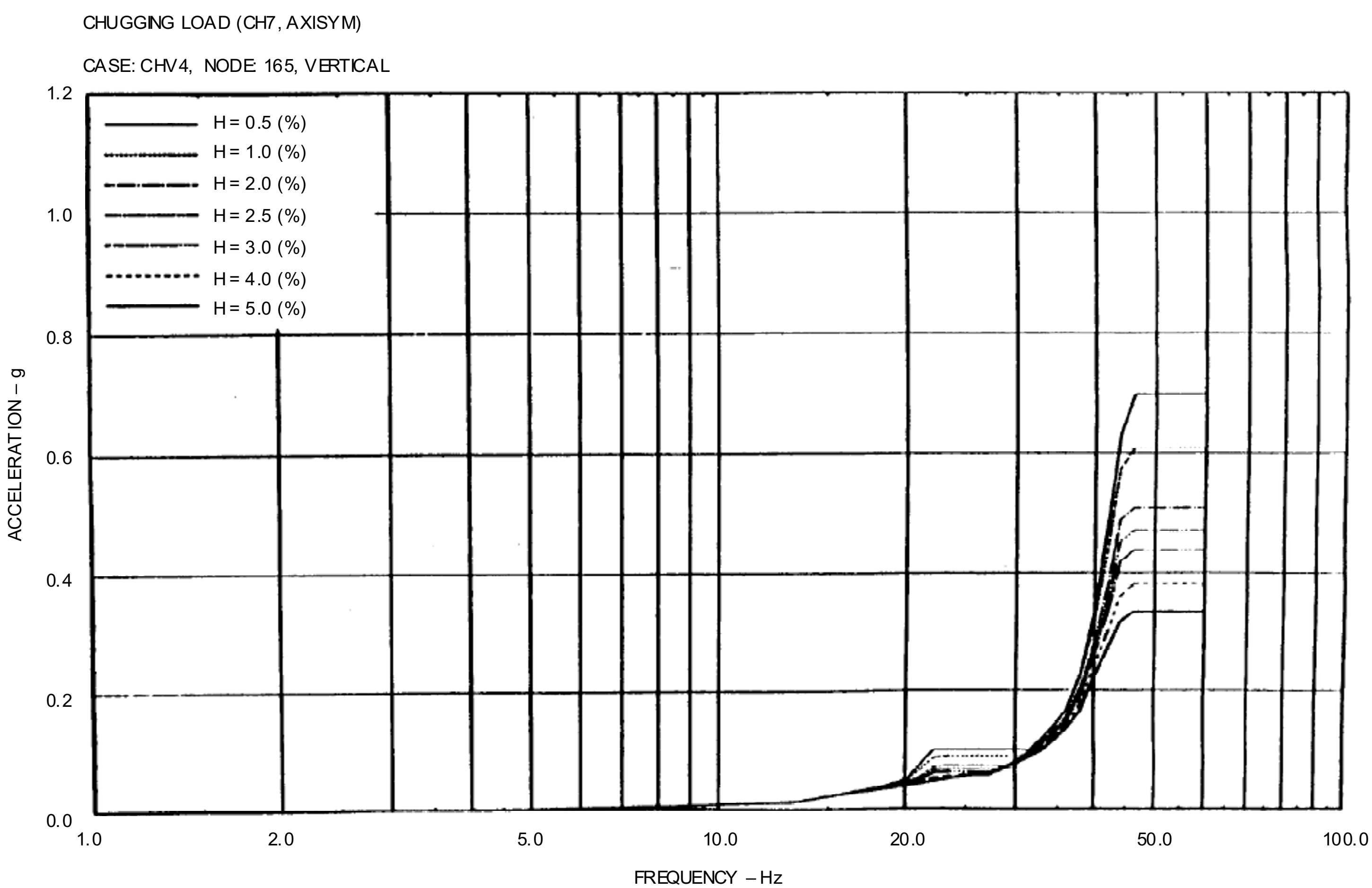

Figure 3G-73 Floor Response Spectrum—Case: CHV4, Node: 165, Vertical

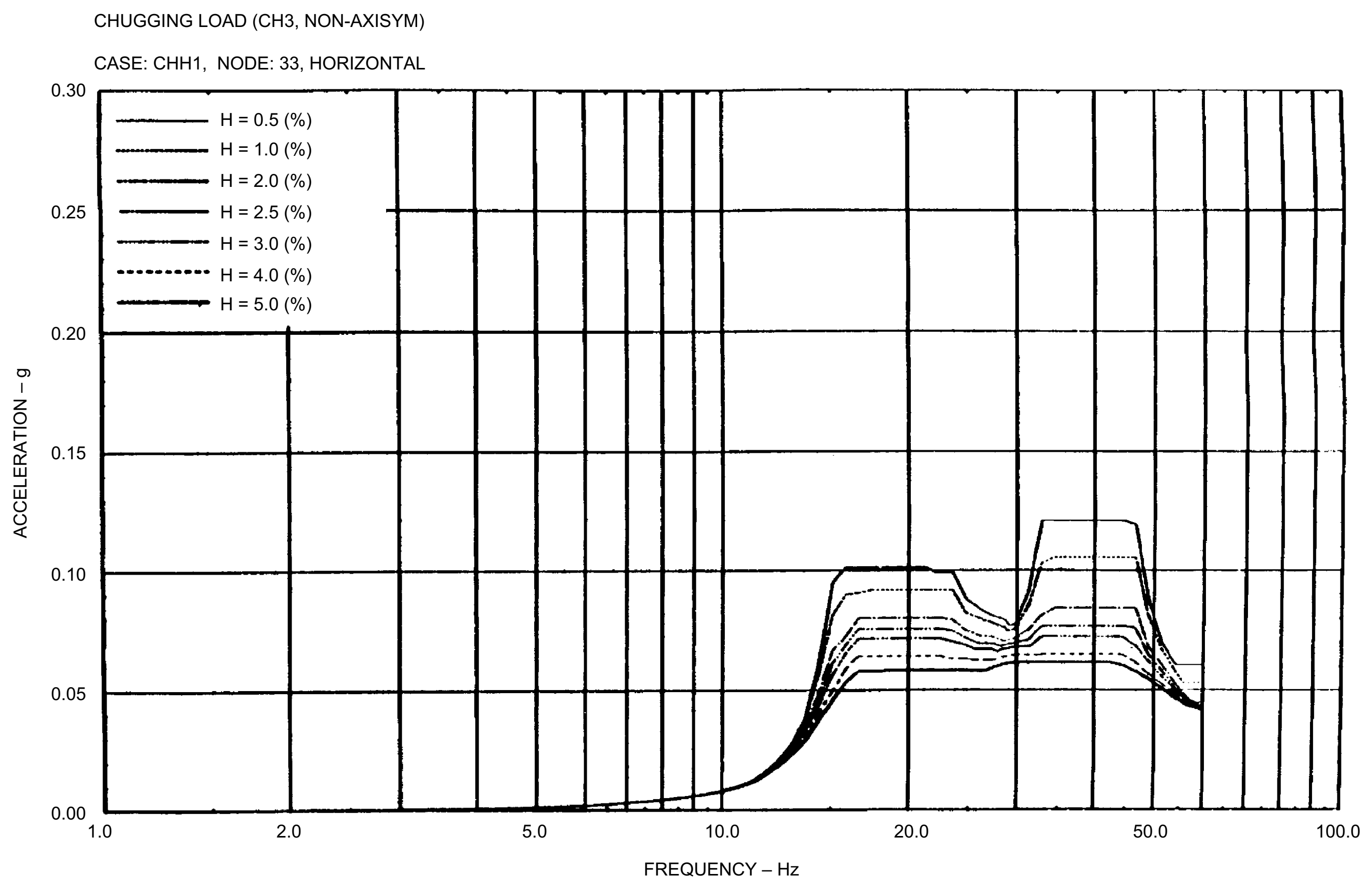

Figure 3G-74 Floor Response Spectrum—Case: CHH1, Node: 33, Horizontal

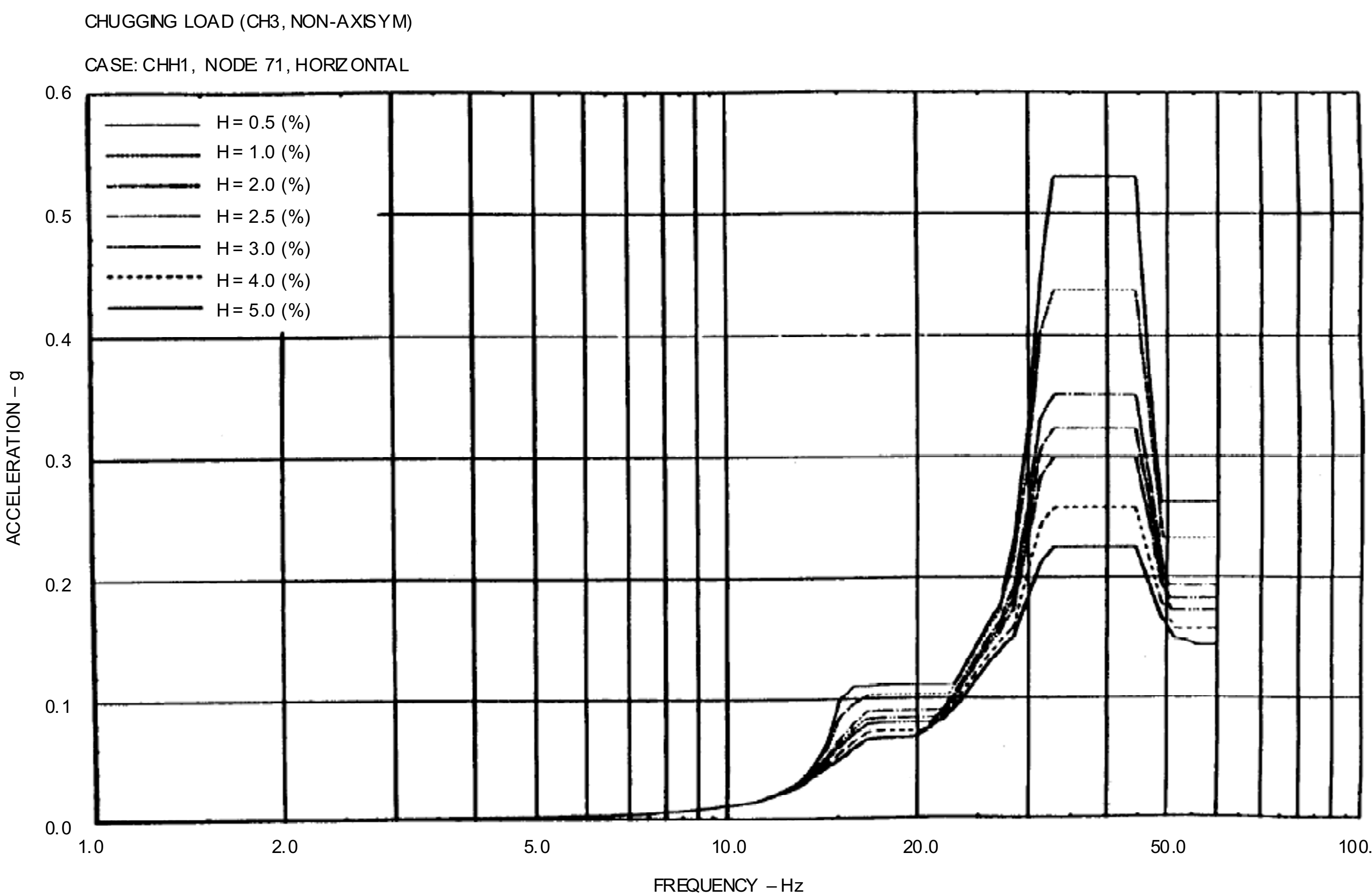

Figure 3G-75 Floor Response Spectrum—Case: CHH1, Node: 71, Horizontal

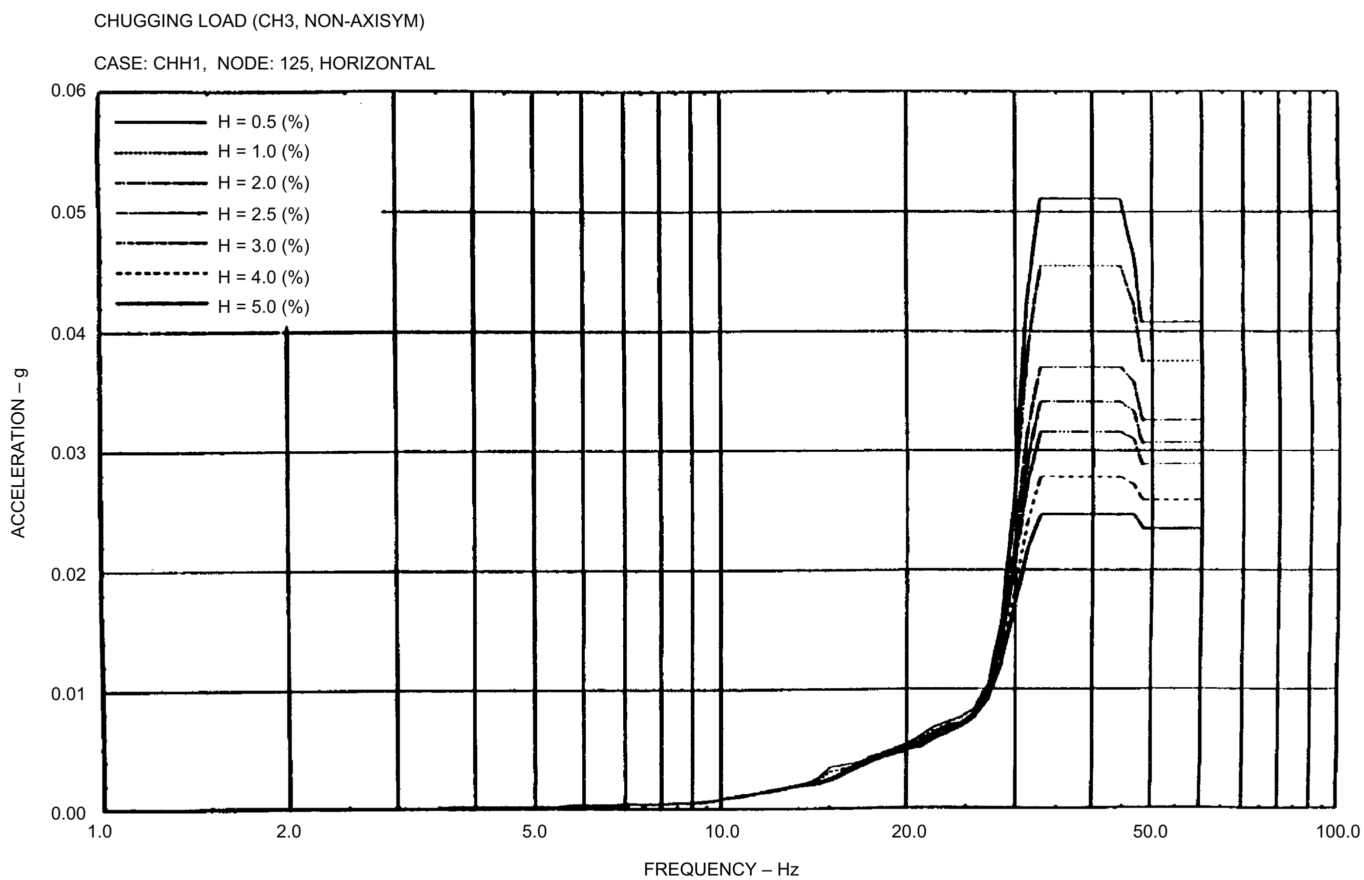

Figure 3G-76 Floor Response Spectrum—Case: CHH1, Node: 125, Horizontal

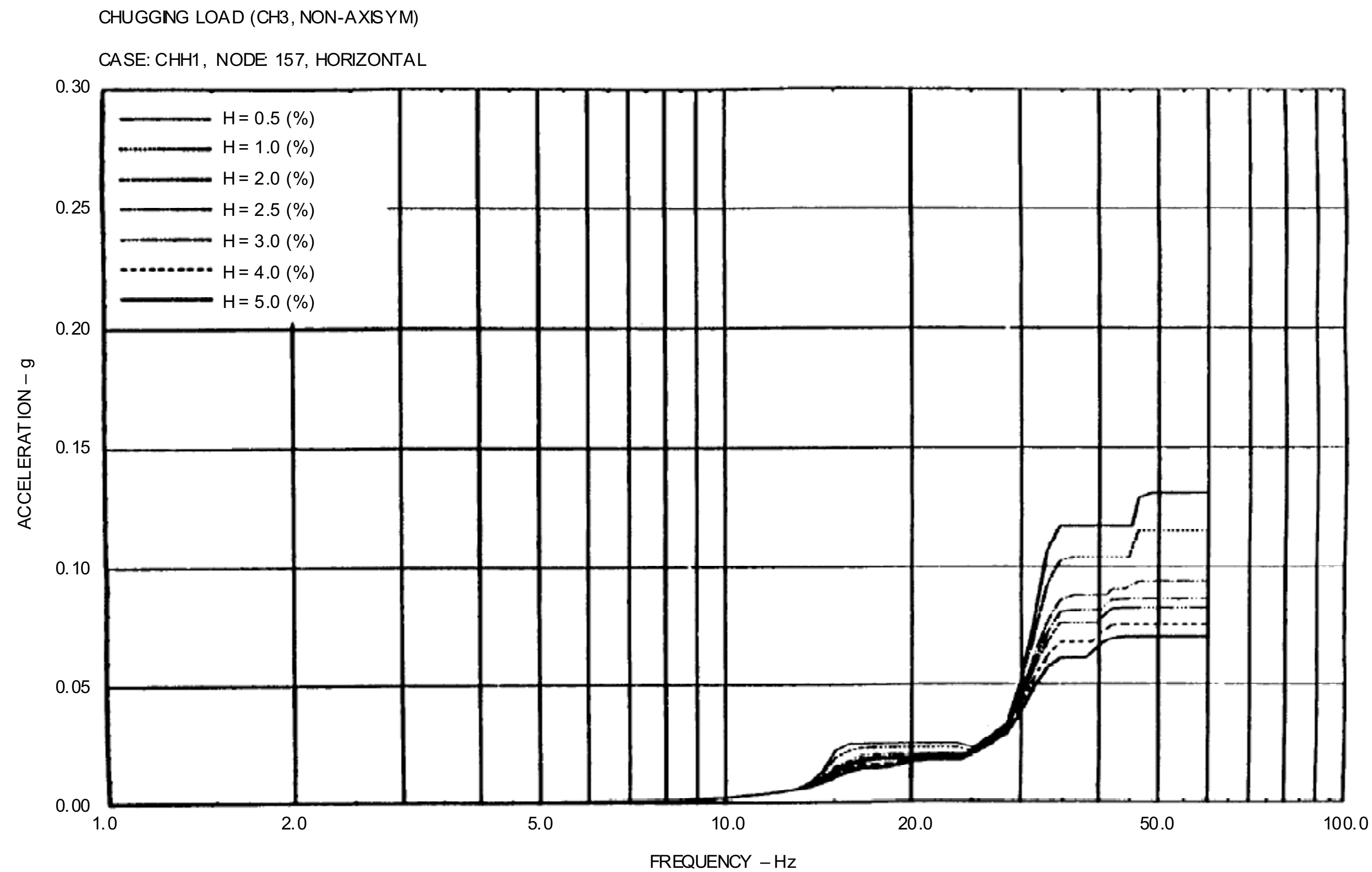

Figure 3G-77 Floor Response Spectrum—Case: CHH1, Node: 157, Horizontal

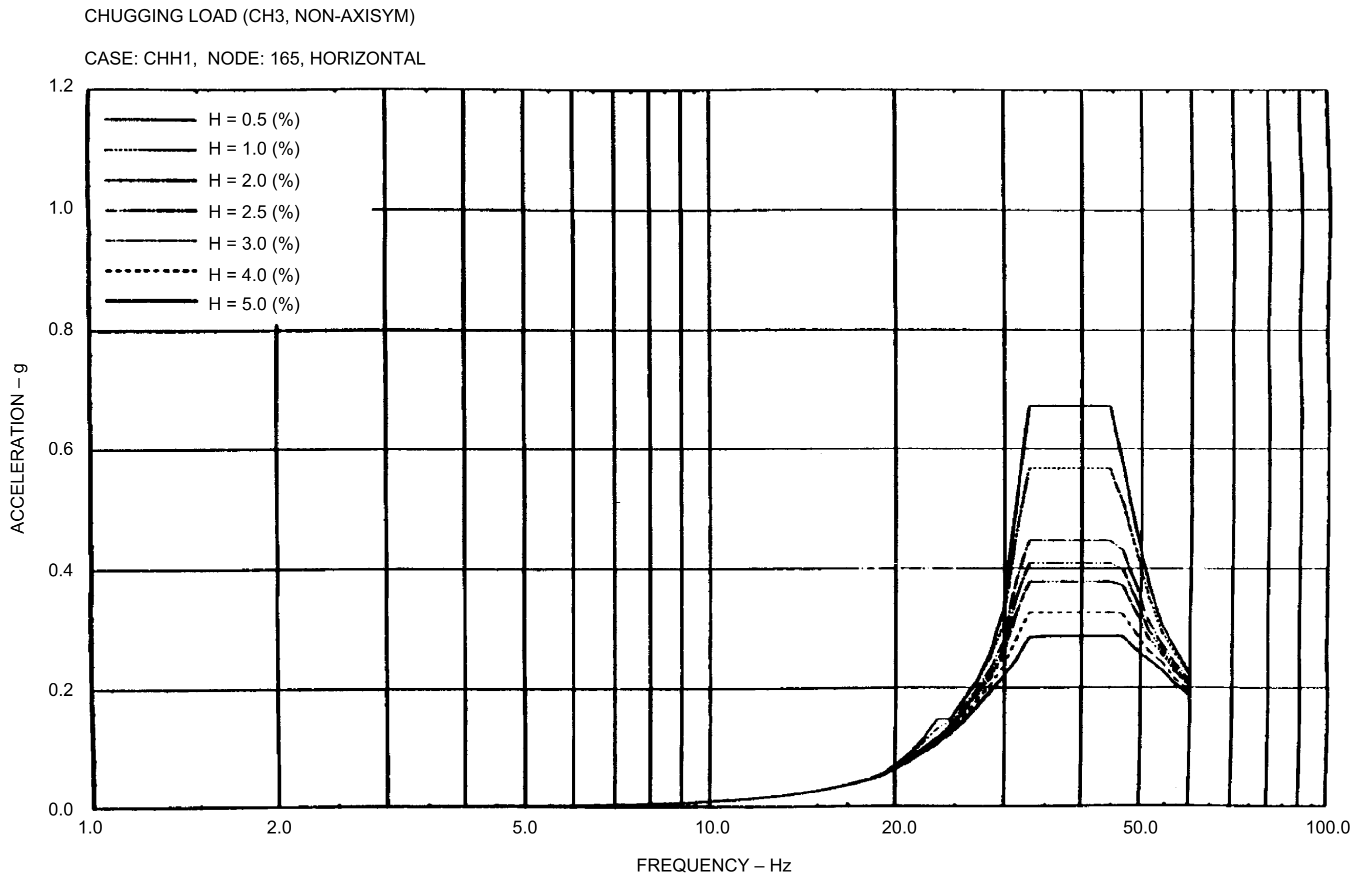

Figure 3G-78 Floor Response Spectrum—Case: CHH1, Node: 165, Horizontal

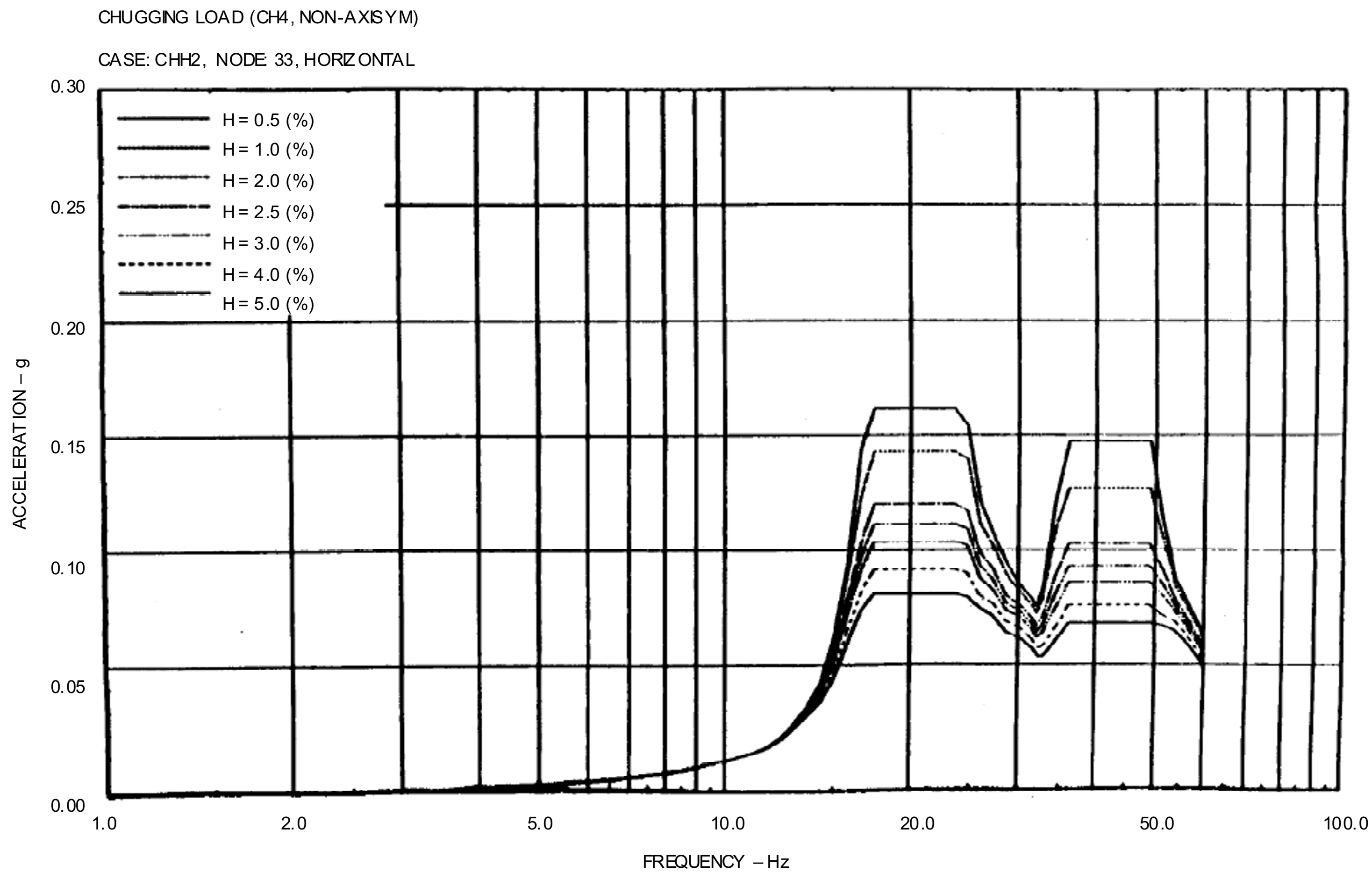

Figure 3G-79 Floor Response Spectrum—Case: CHH2, Node: 33, Horizontal

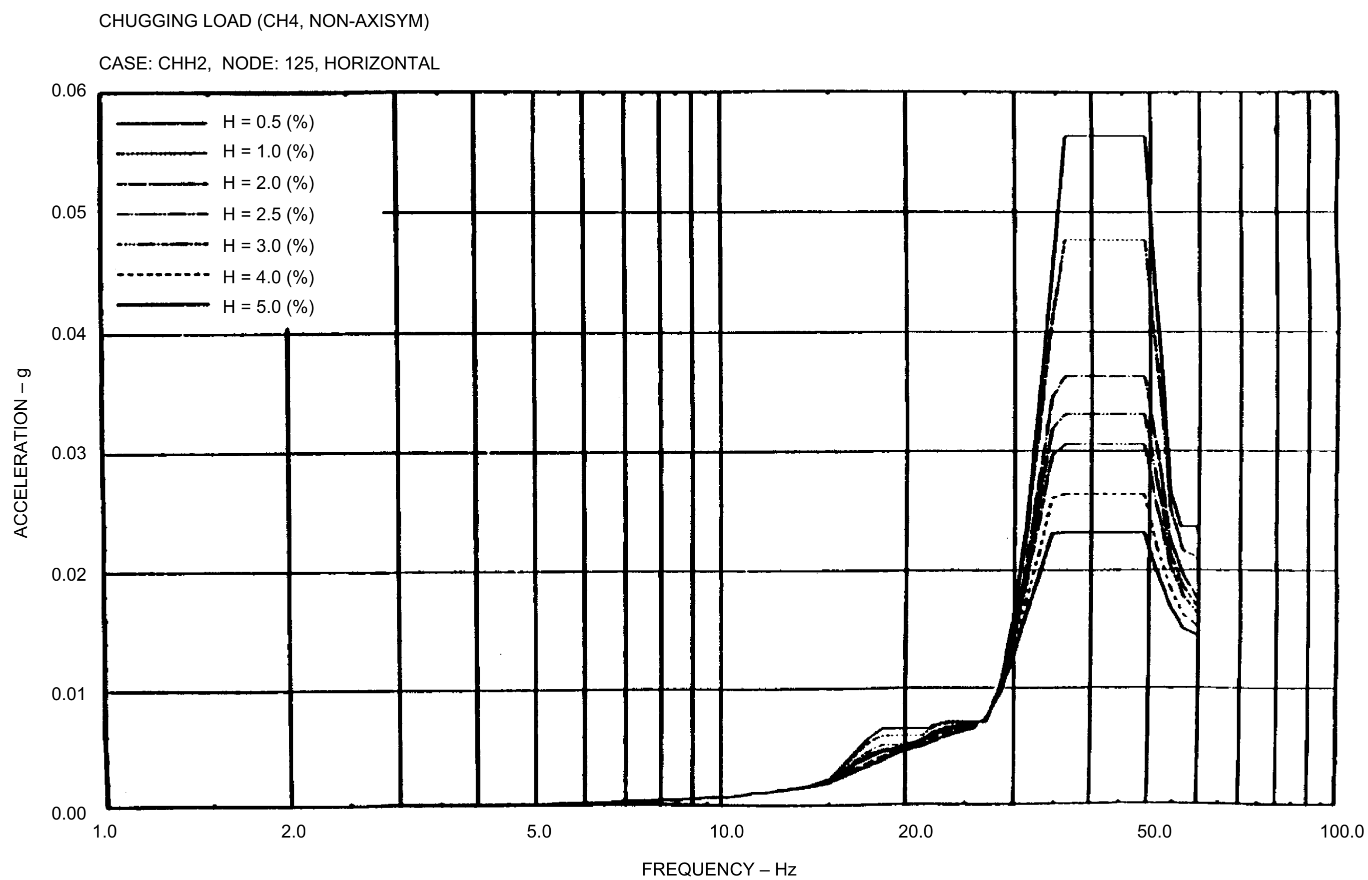

Figure 3G-80 Floor Response Spectrum—Case: CHH2, Node: 125, Horizontal

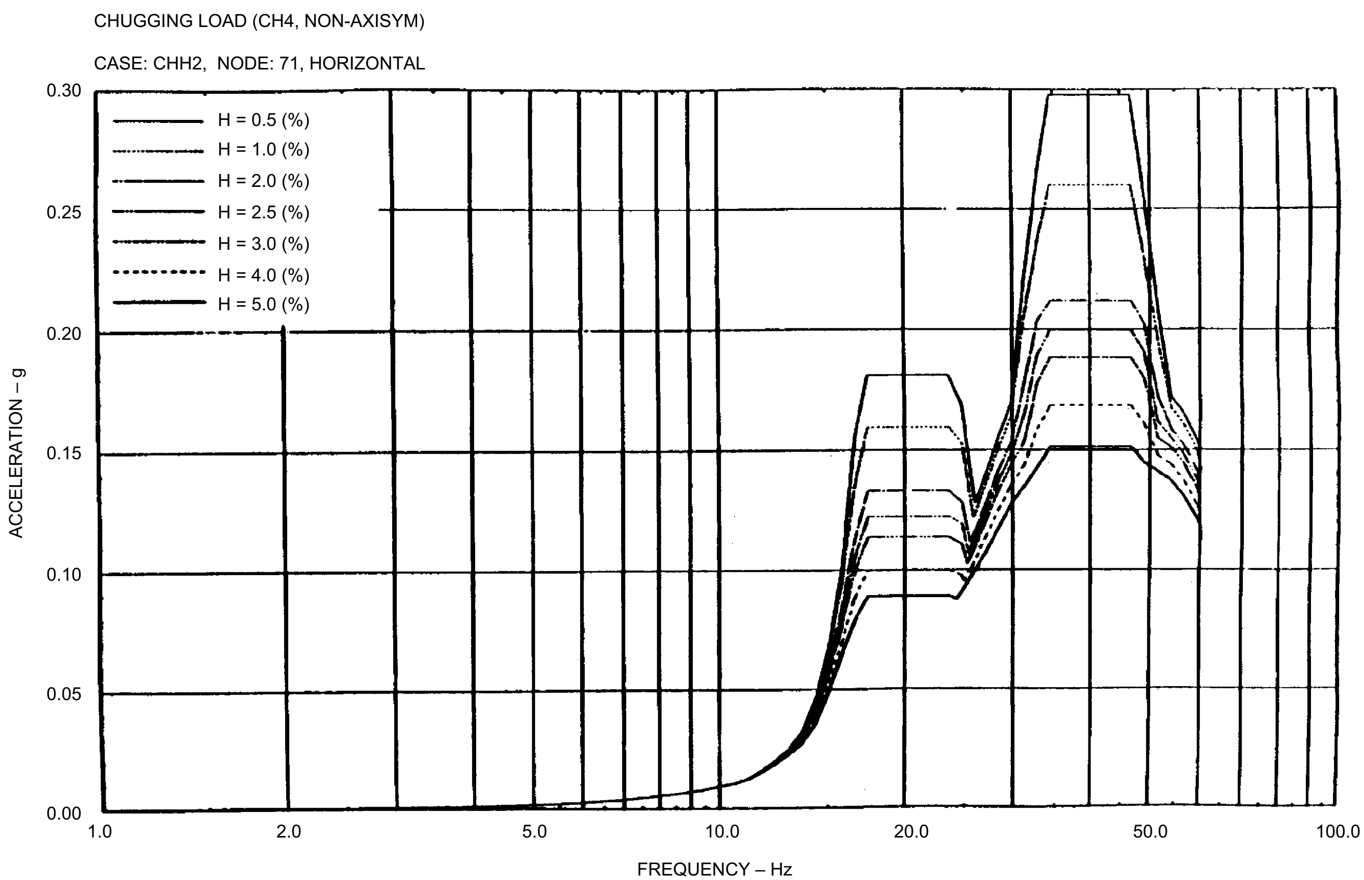

Figure 3G-81 Floor Response Spectrum—Case: CHH2, Node: 71, Horizontal

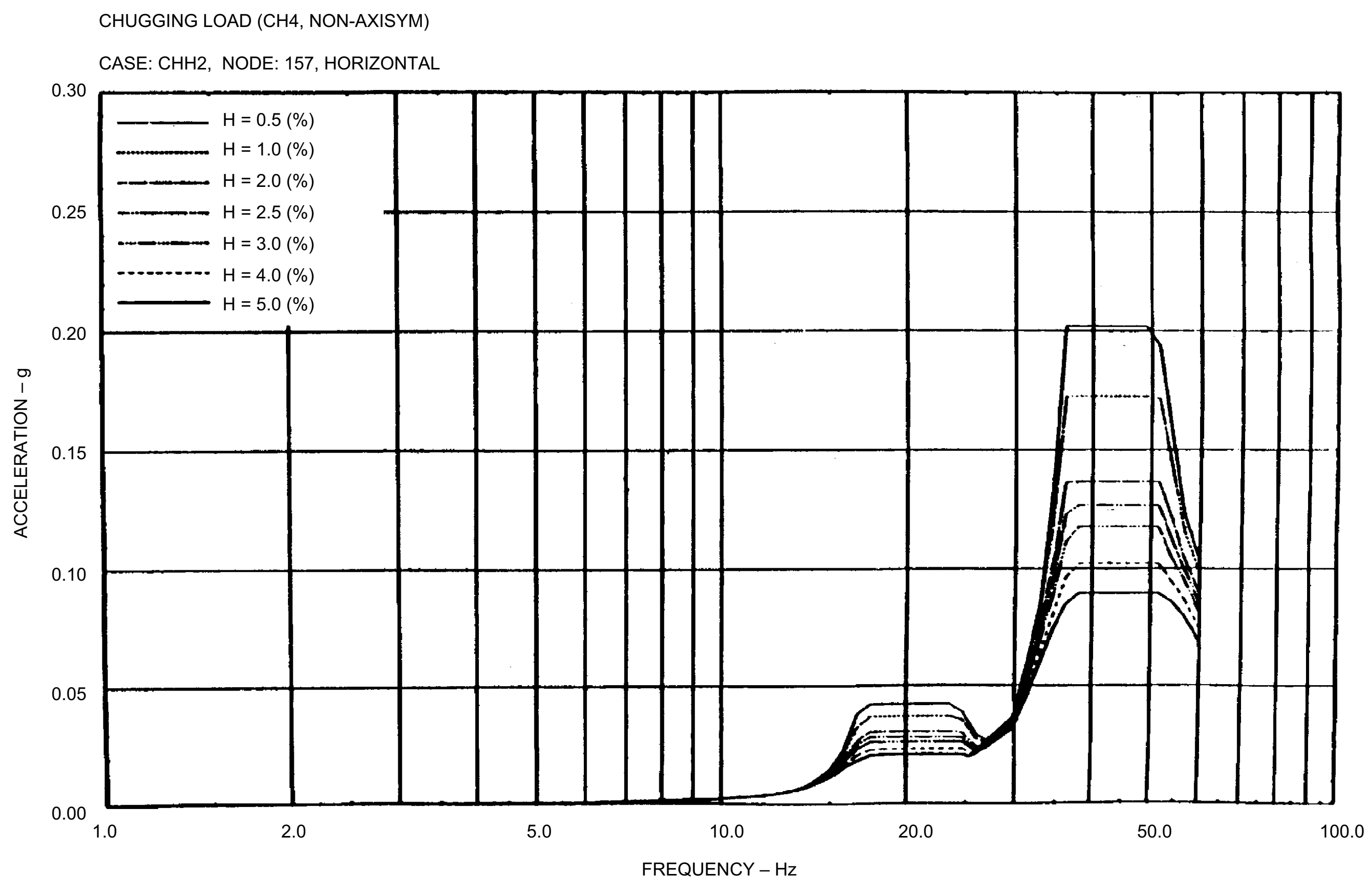

Figure 3G-82 Floor Response Spectrum—Case: CHH2, Node: 157, Horizontal

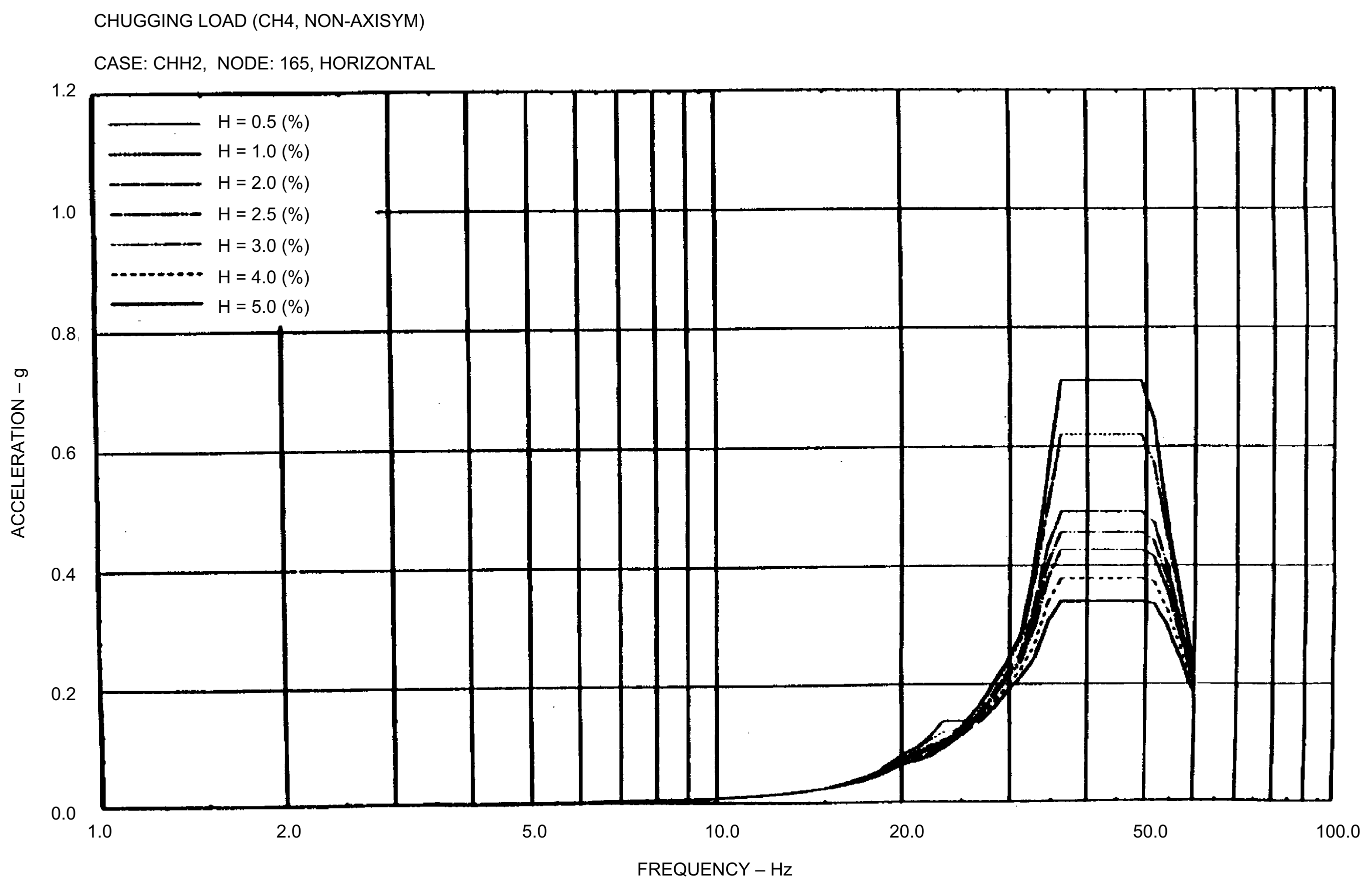

Figure 3G-83 Floor Response Spectrum—Case: CHH2, Node: 165, Horizontal

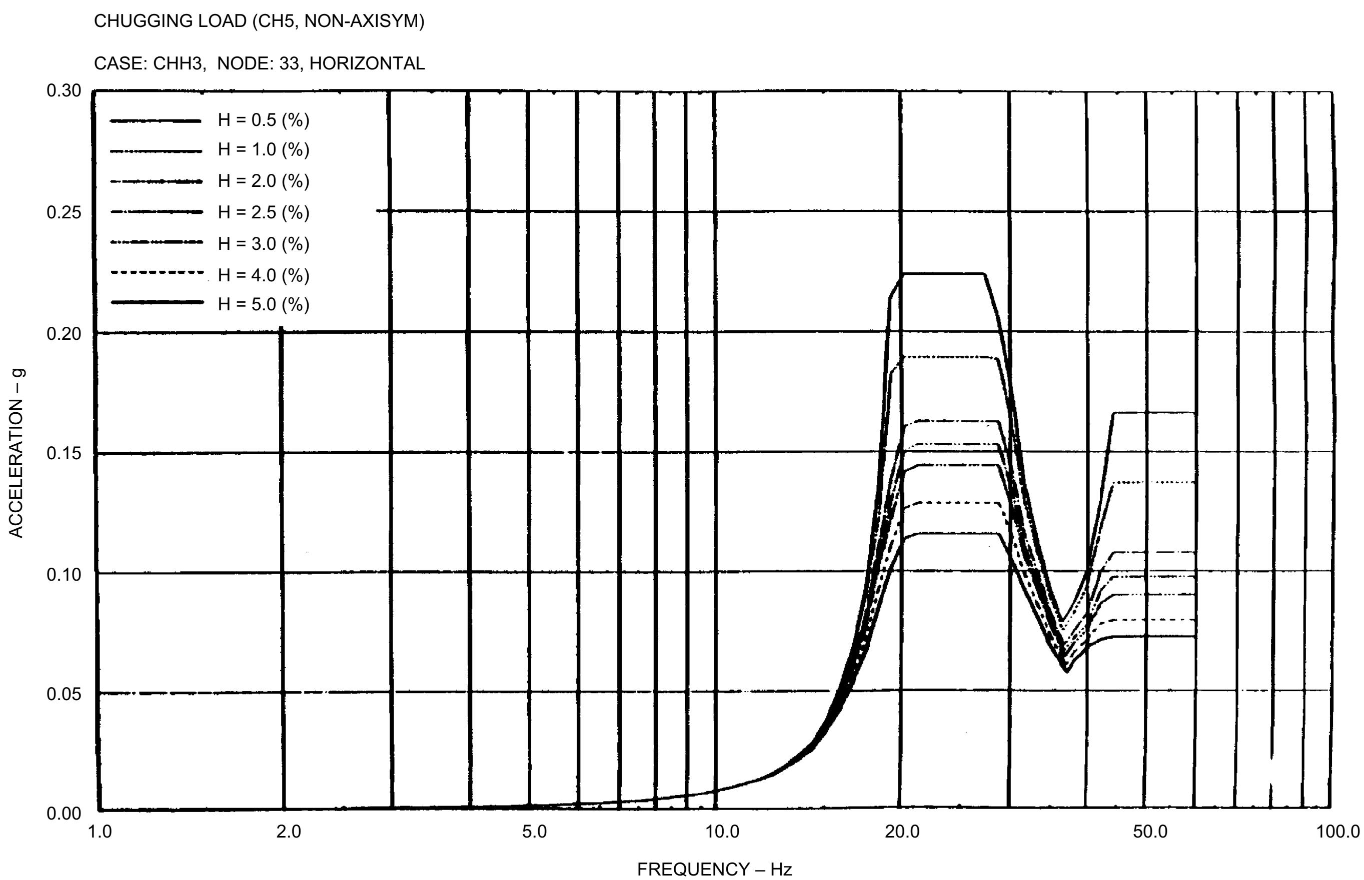

Figure 3G-84 Floor Response Spectrum—Case: CHH3, Node: 33, Horizontal

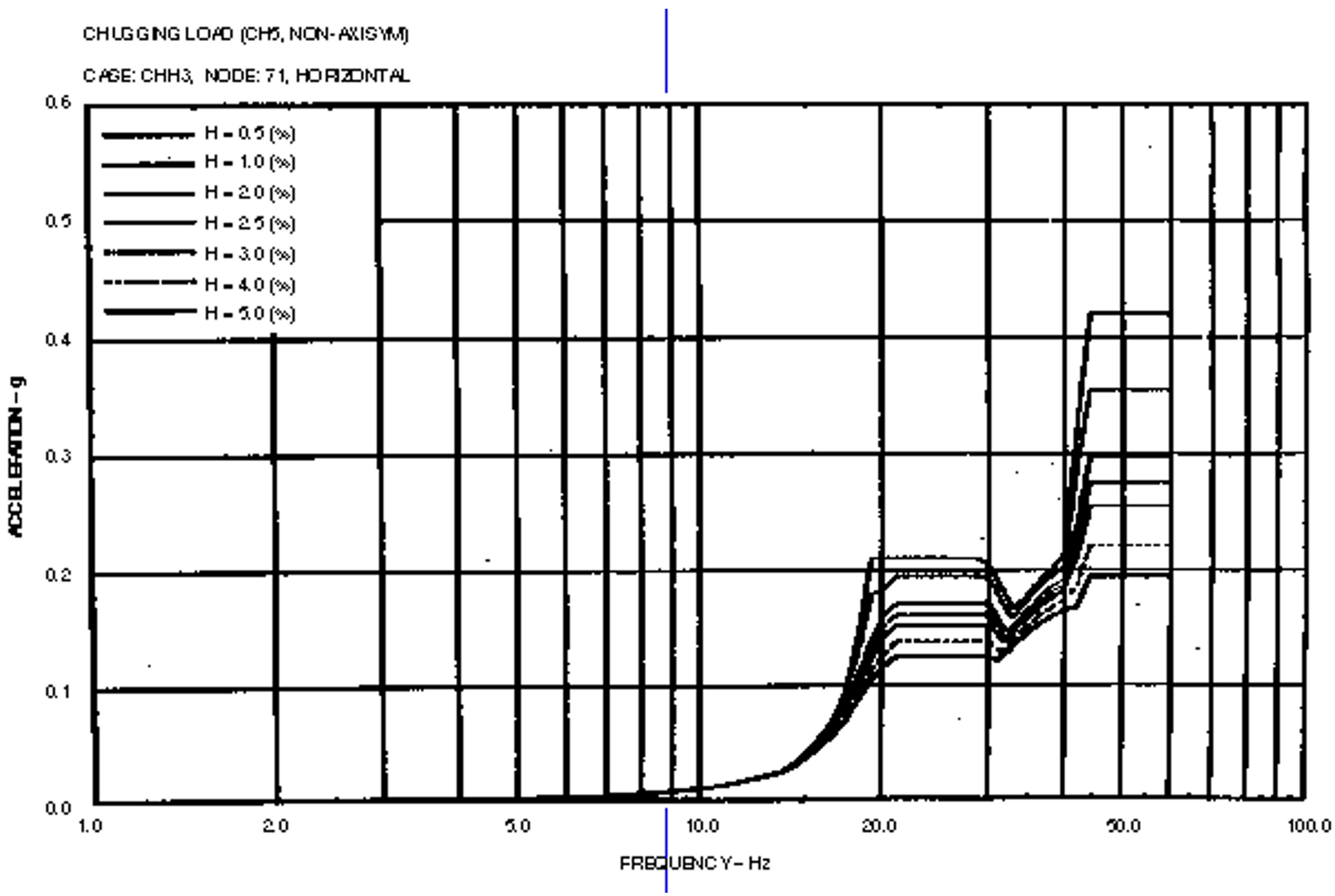

Figure 3G-85 Floor Response Spectrum—Case: CHH3, Node: 71, Horizontal

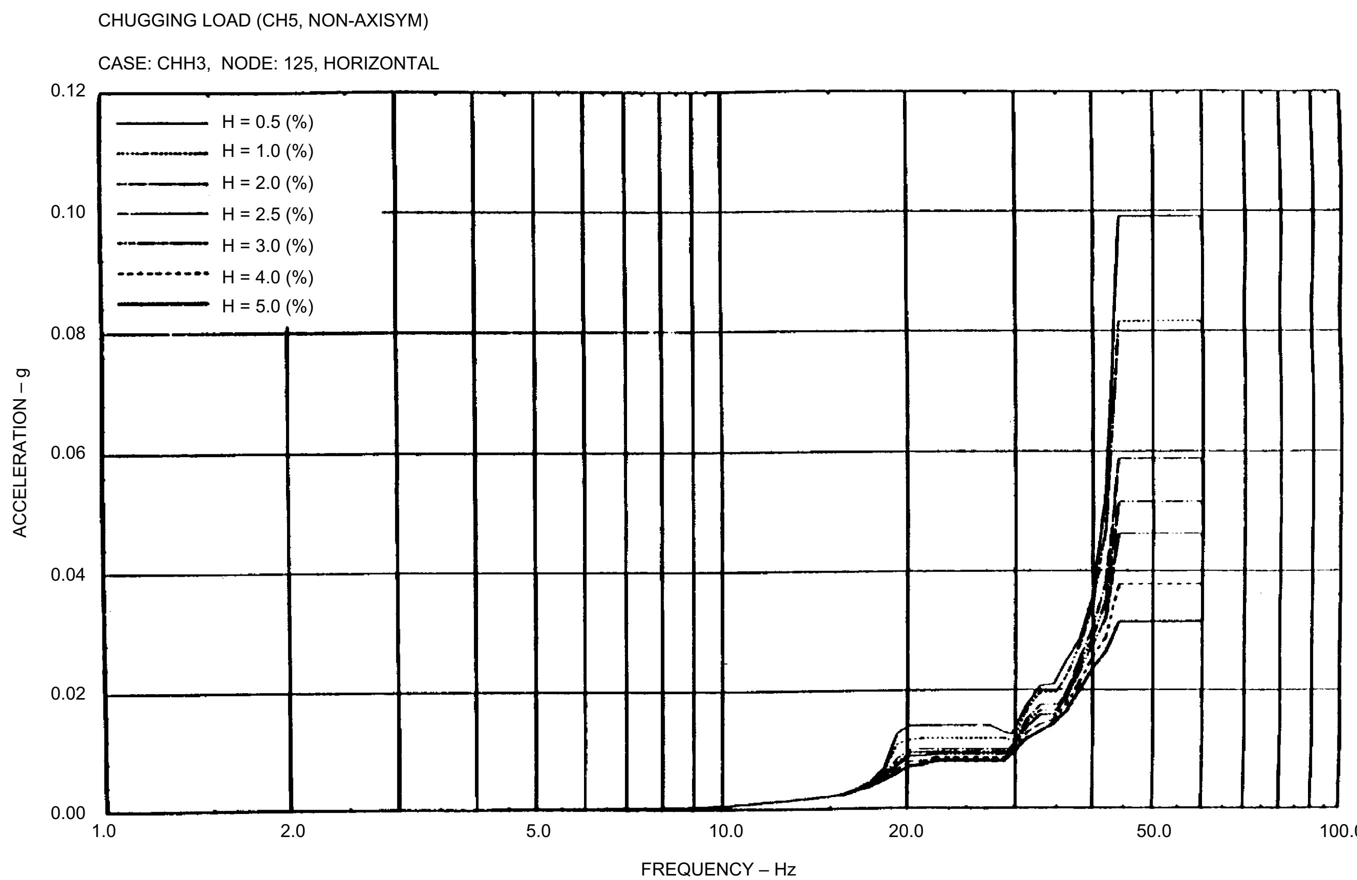

Figure 3G-86 Floor Response Spectrum—Case: CHH3, Node: 125, Horizontal

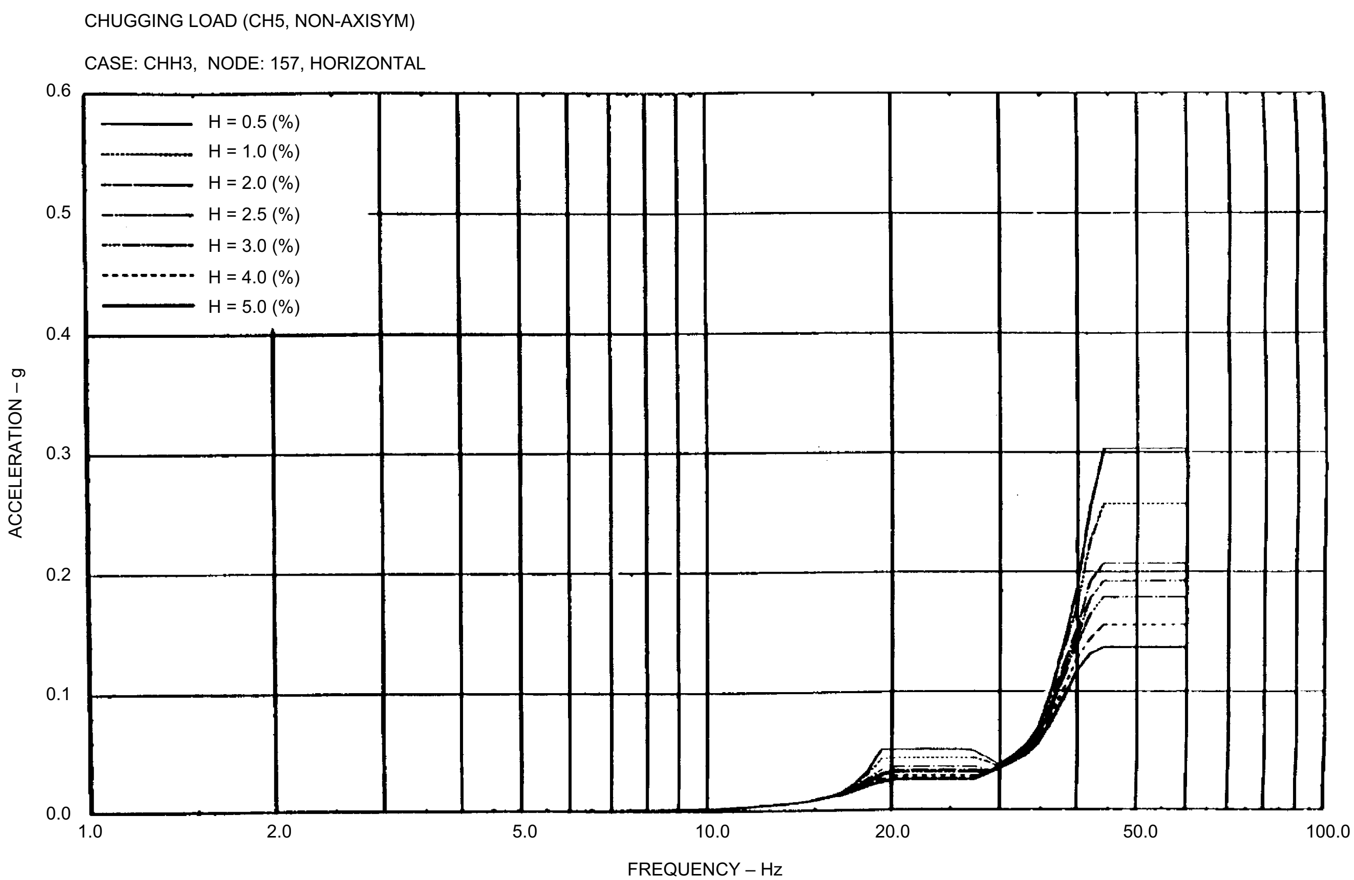

Figure 3G-87 Floor Response Spectrum—Case: CHH3, Node: 157, Horizontal

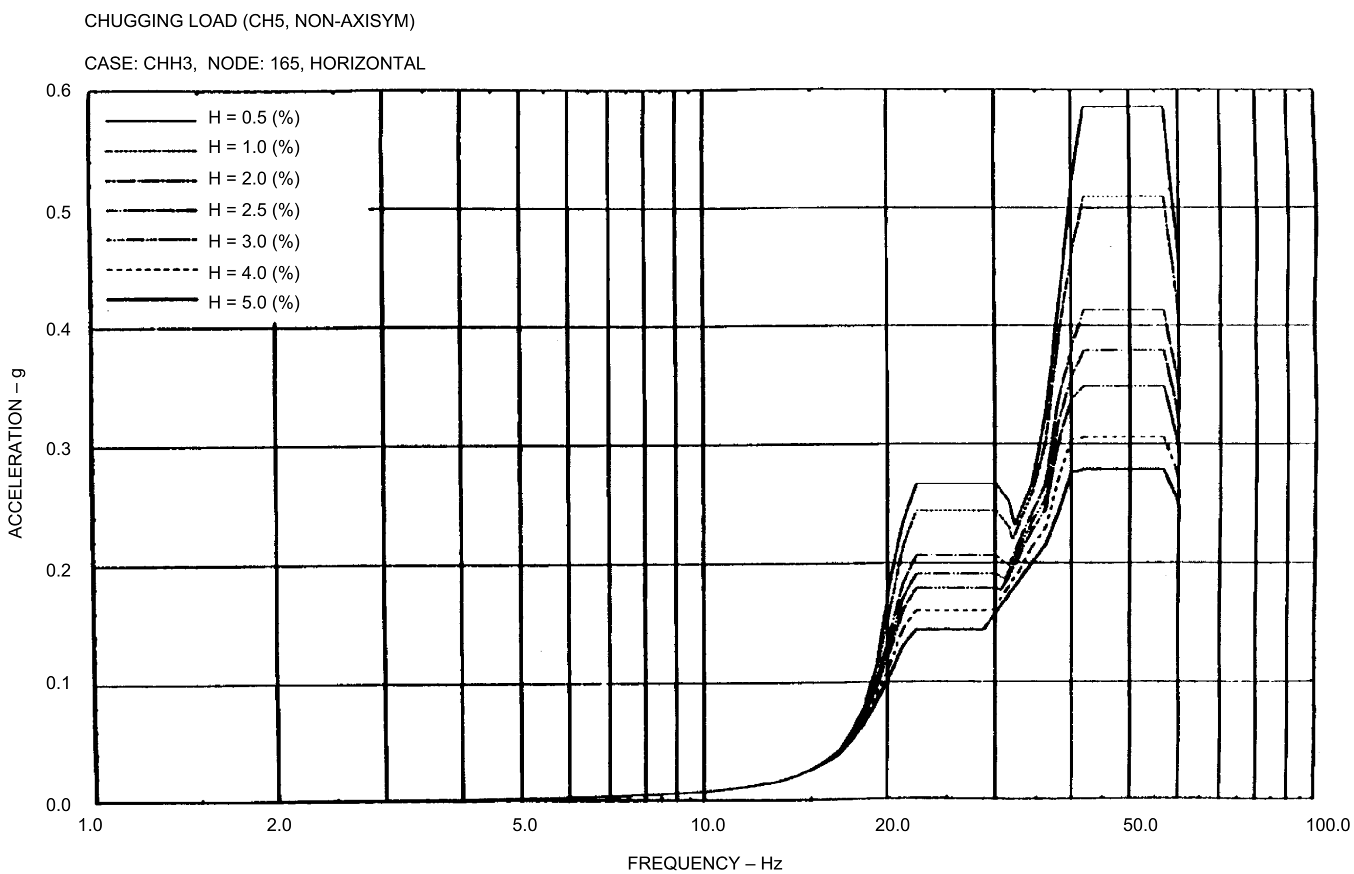

Figure 3G-88 Floor Response Spectrum—Case: CHH3, Node: 165, Horizontal

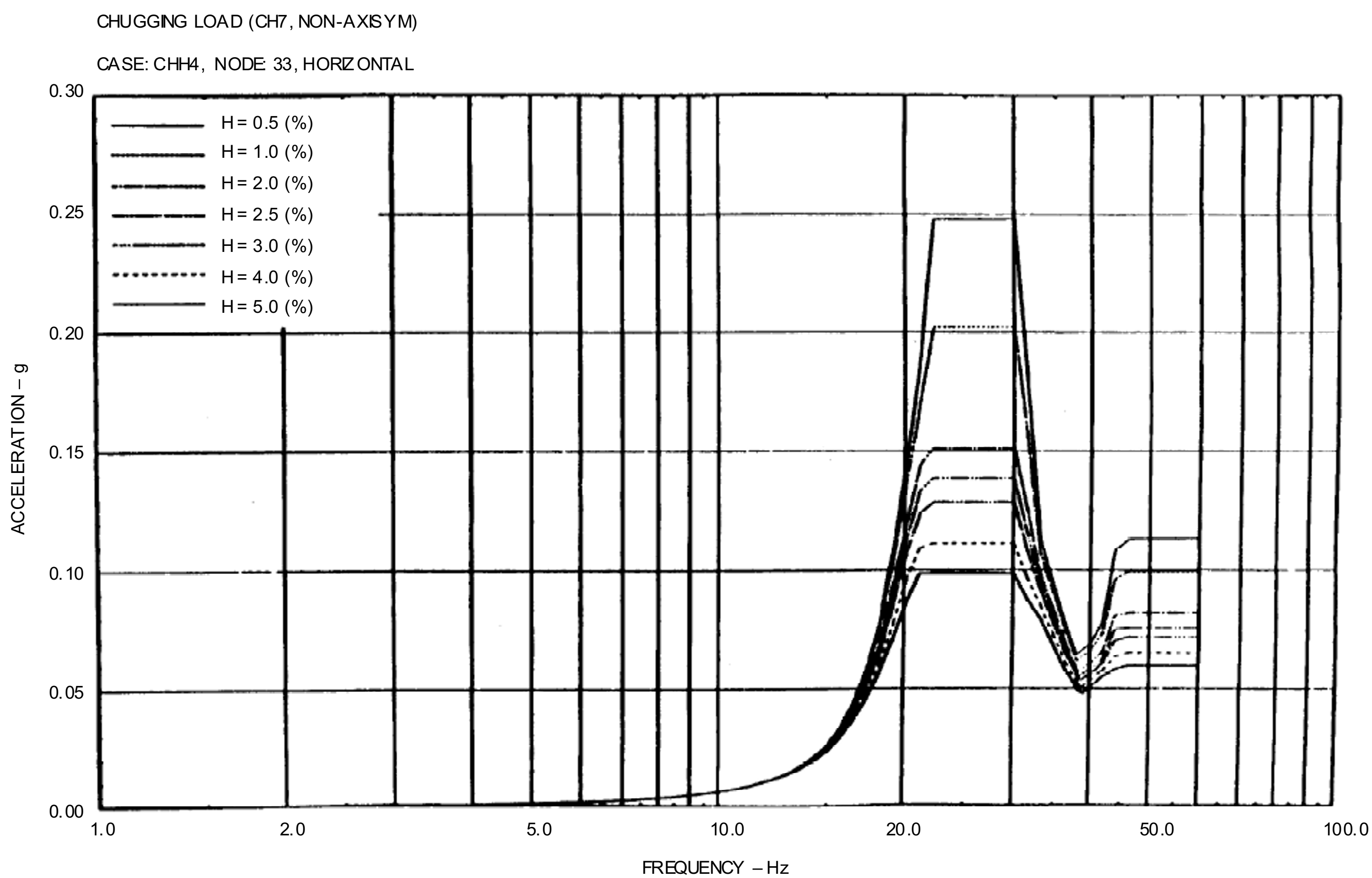

Figure 3G-89 Floor Response Spectrum—Case: CHH4, Node: 33, Horizontal

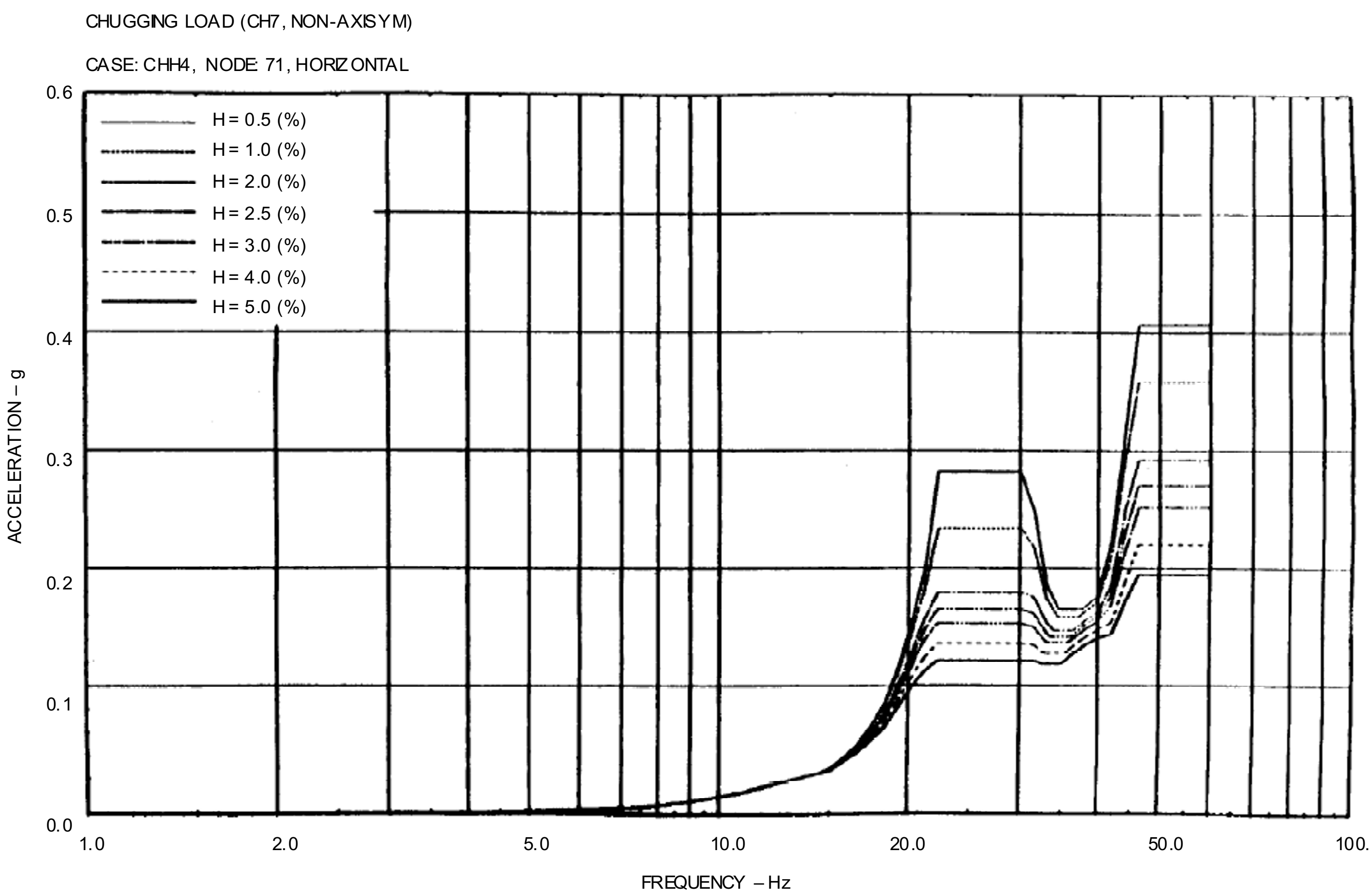

Figure 3G-90 Floor Response Spectrum—Case: CHH4, Node: 71, Horizontal

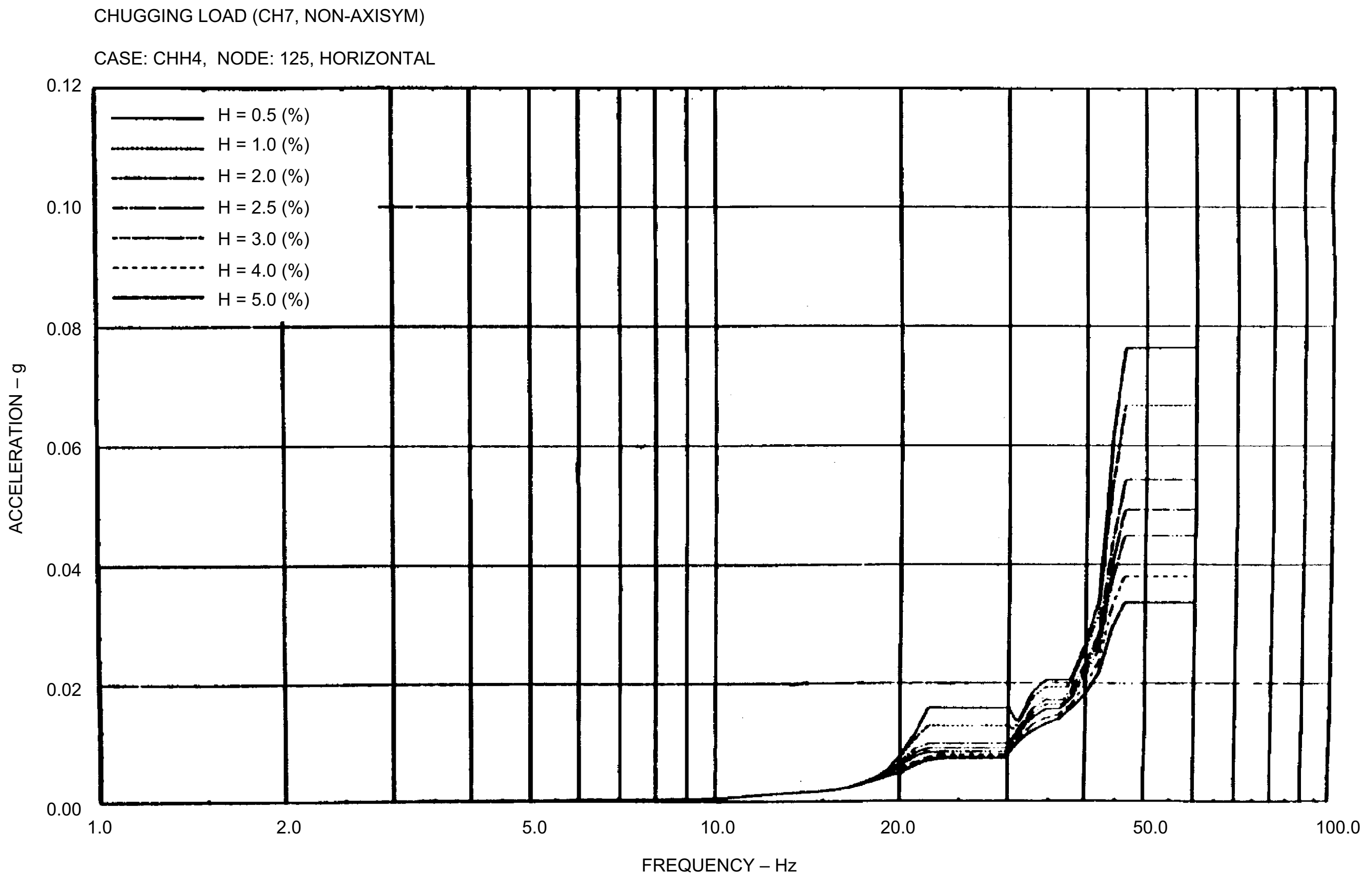

Figure 3G-91 Floor Response Spectrum—Case: CHH4, Node: 125, Horizontal

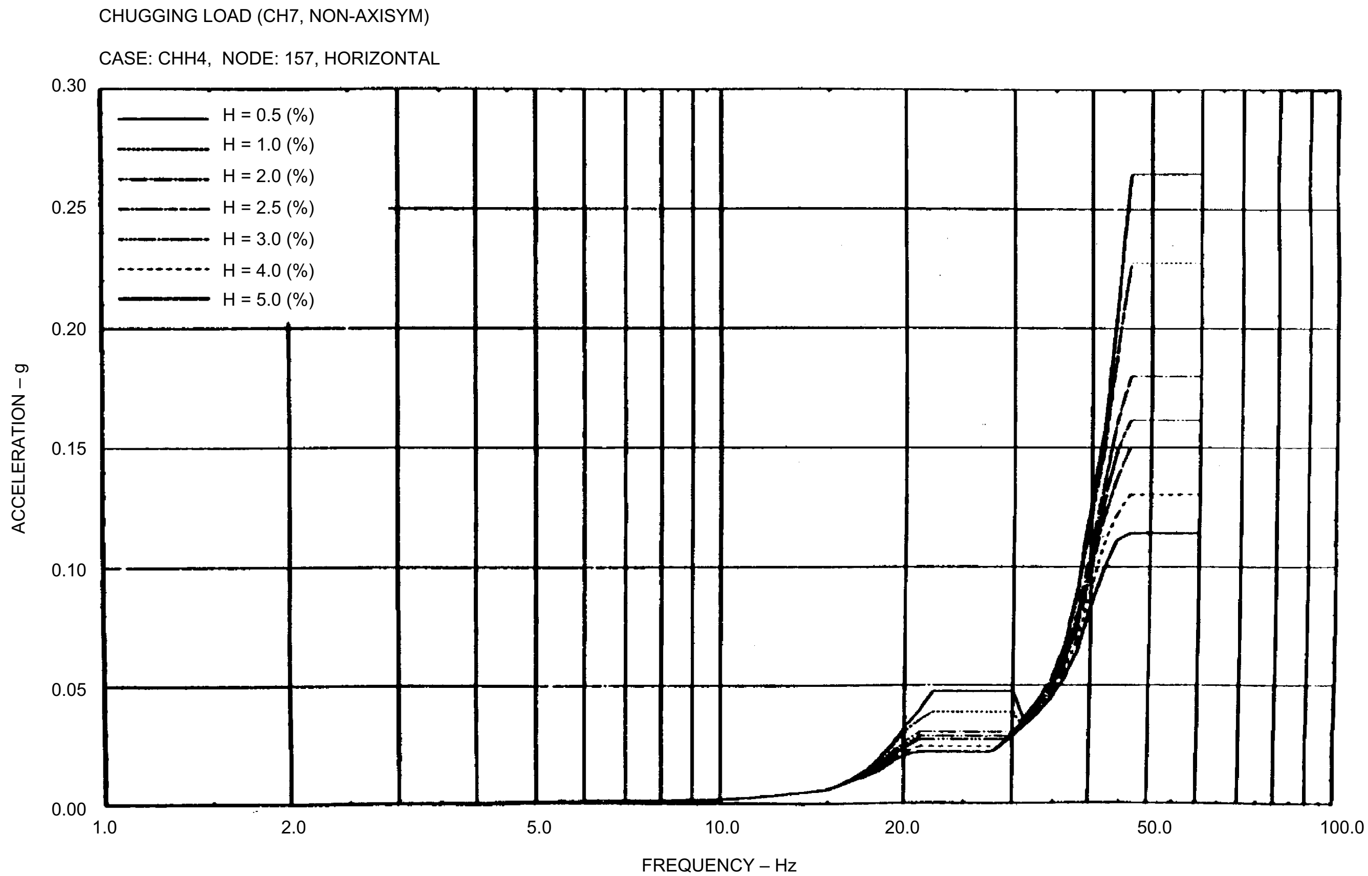

Figure 3G-92 Floor Response Spectrum—Case: CHH4, Node: 157, Horizontal

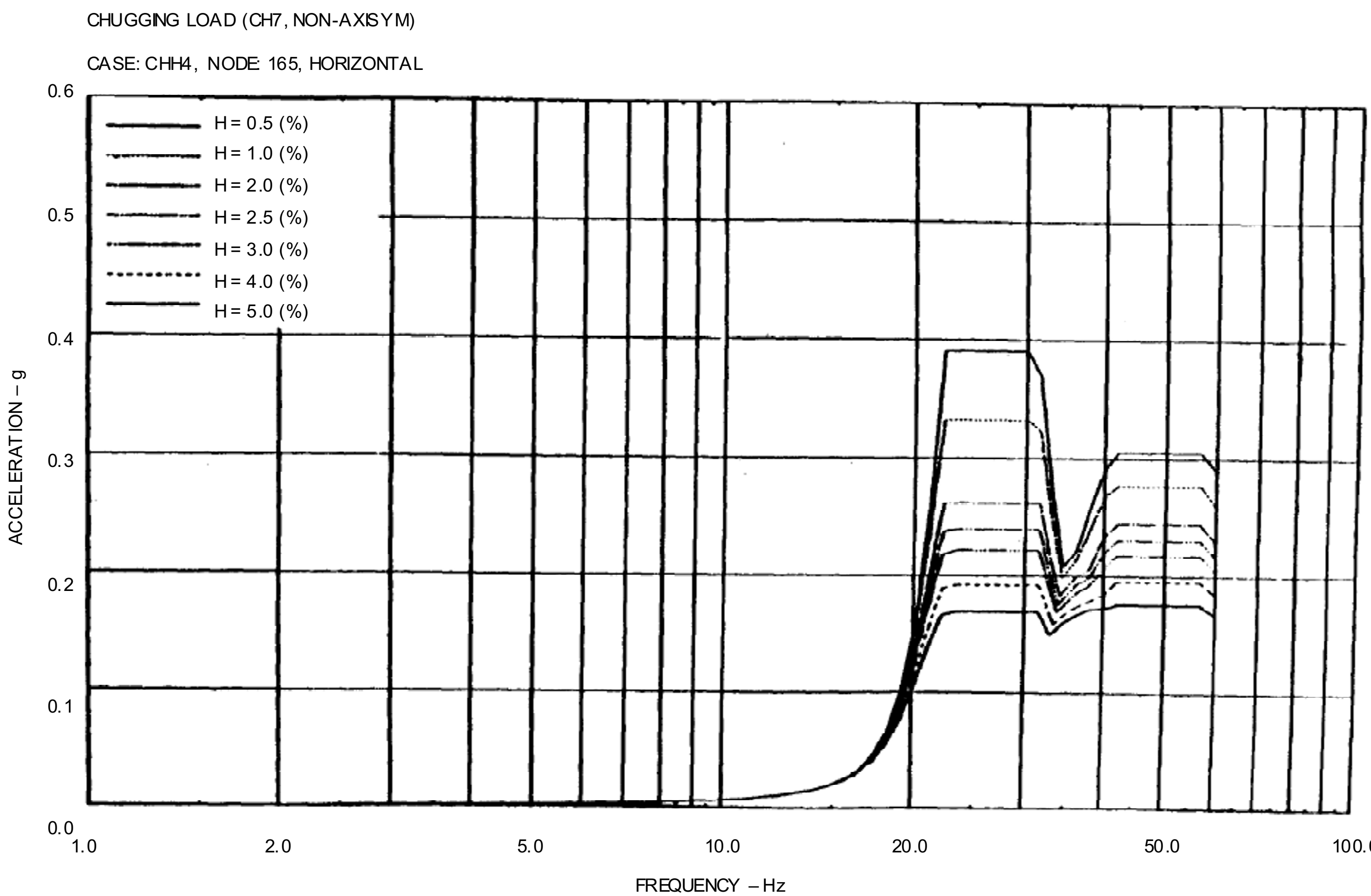

Figure 3G-93 Floor Response Spectrum—Case: CHH4, Node: 165, Horizontal

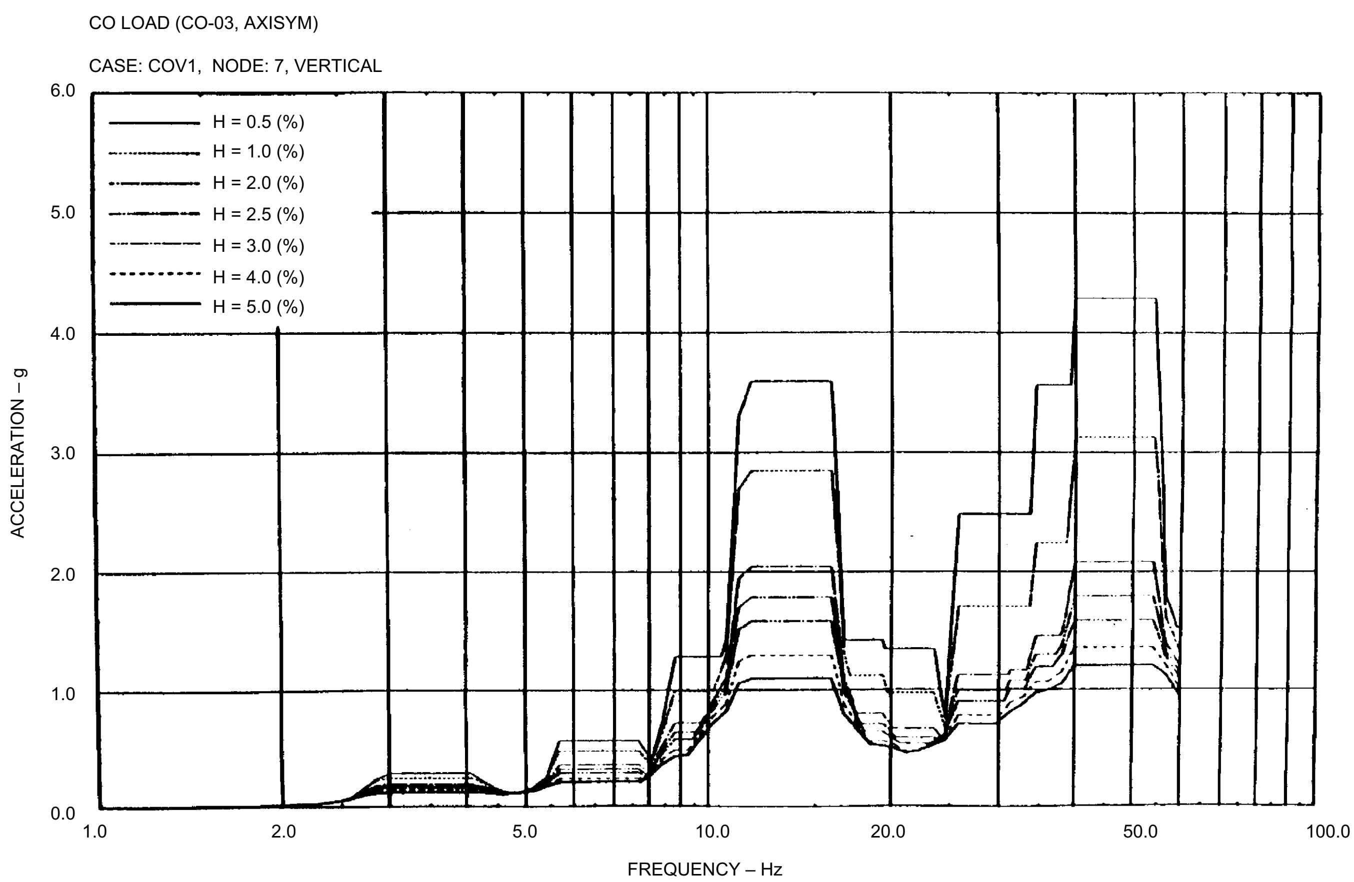

Figure 3G-94 Floor Response Spectrum—Case: COV1, Node: 7, Vertical

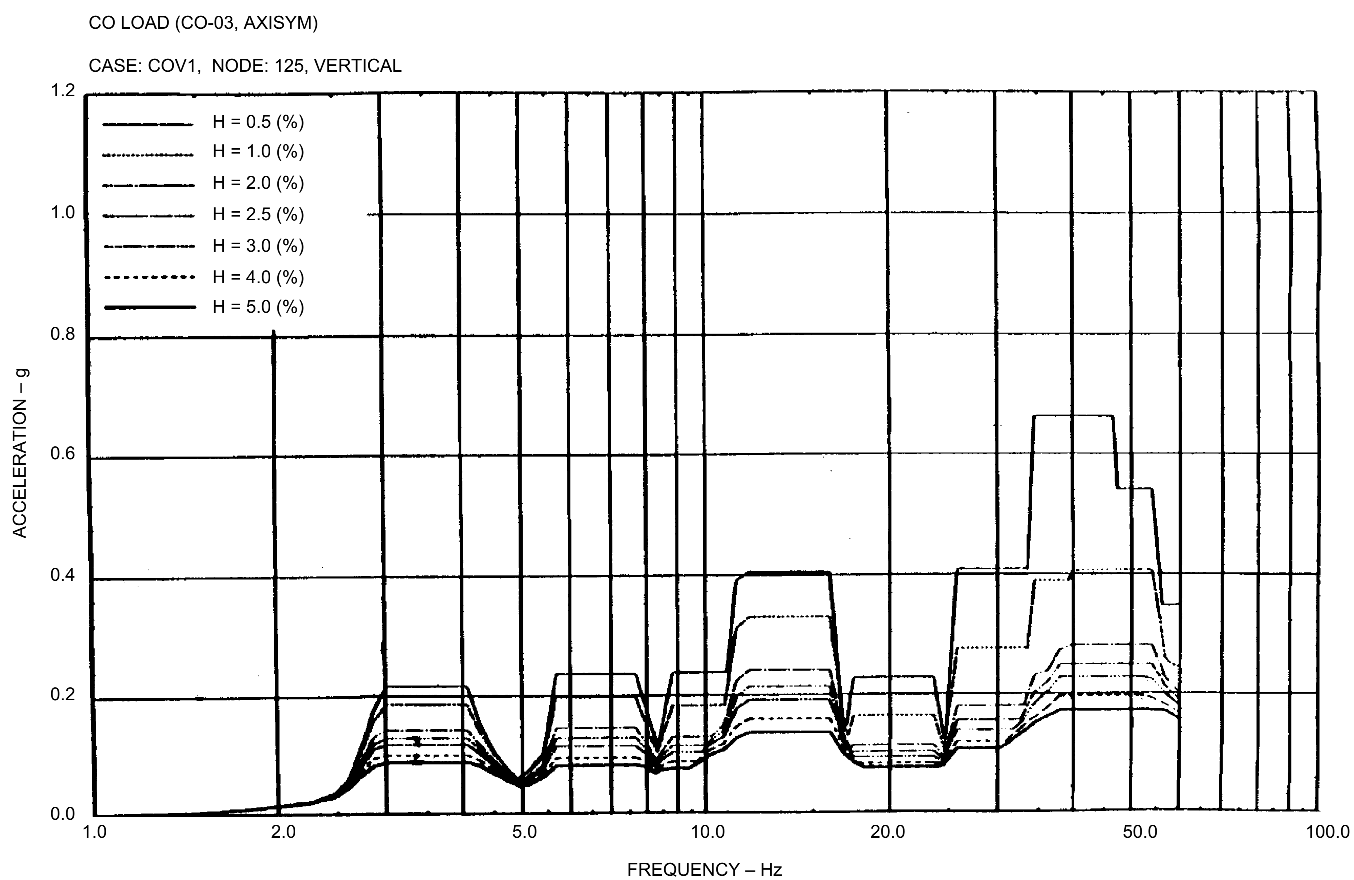

Figure 3G-95 Floor Response Spectrum—Case: COV1, Node: 125, Vertical

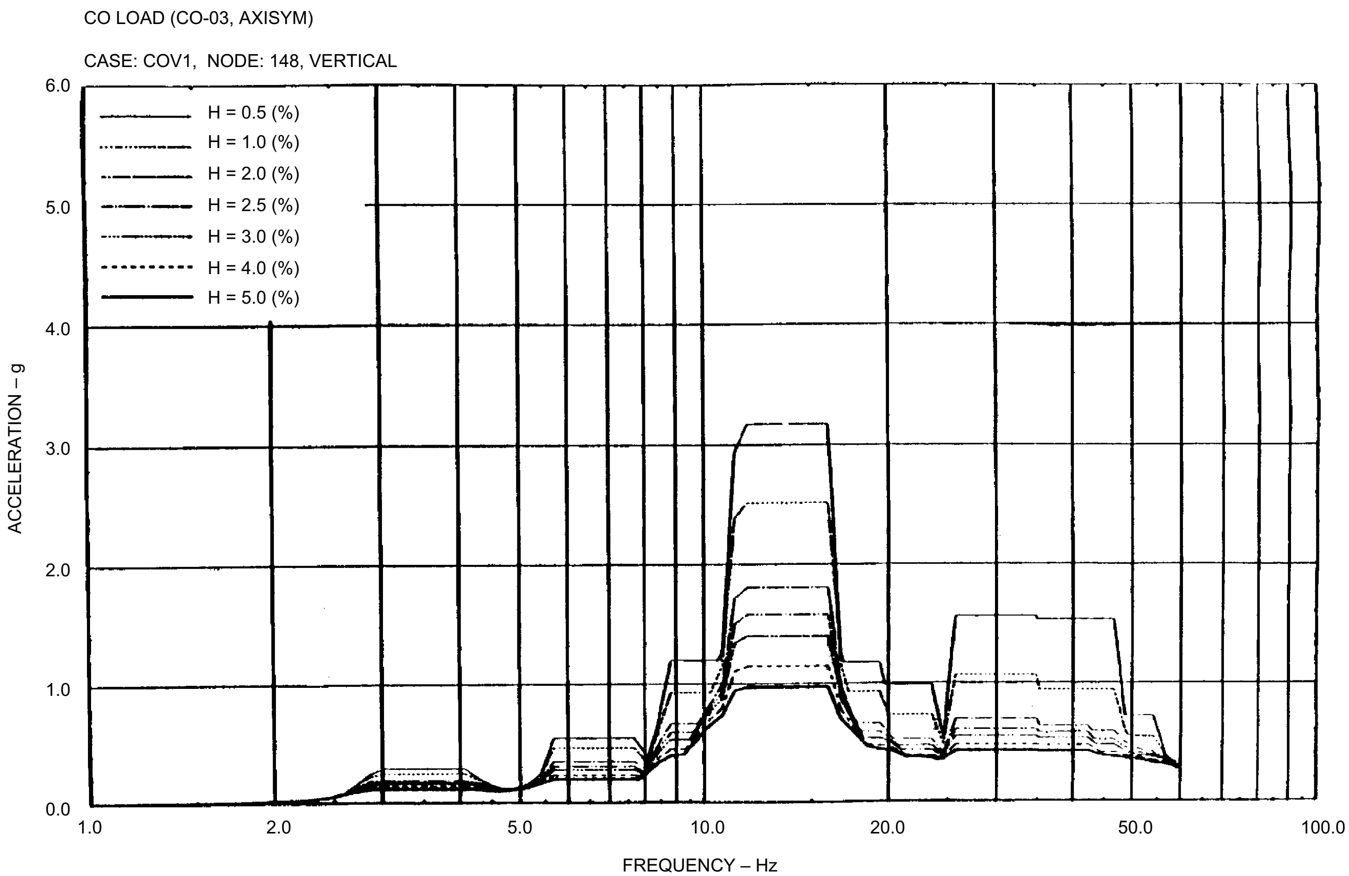

Figure 3G-96 Floor Response Spectrum—Case: COV1, Node: 148, Vertical

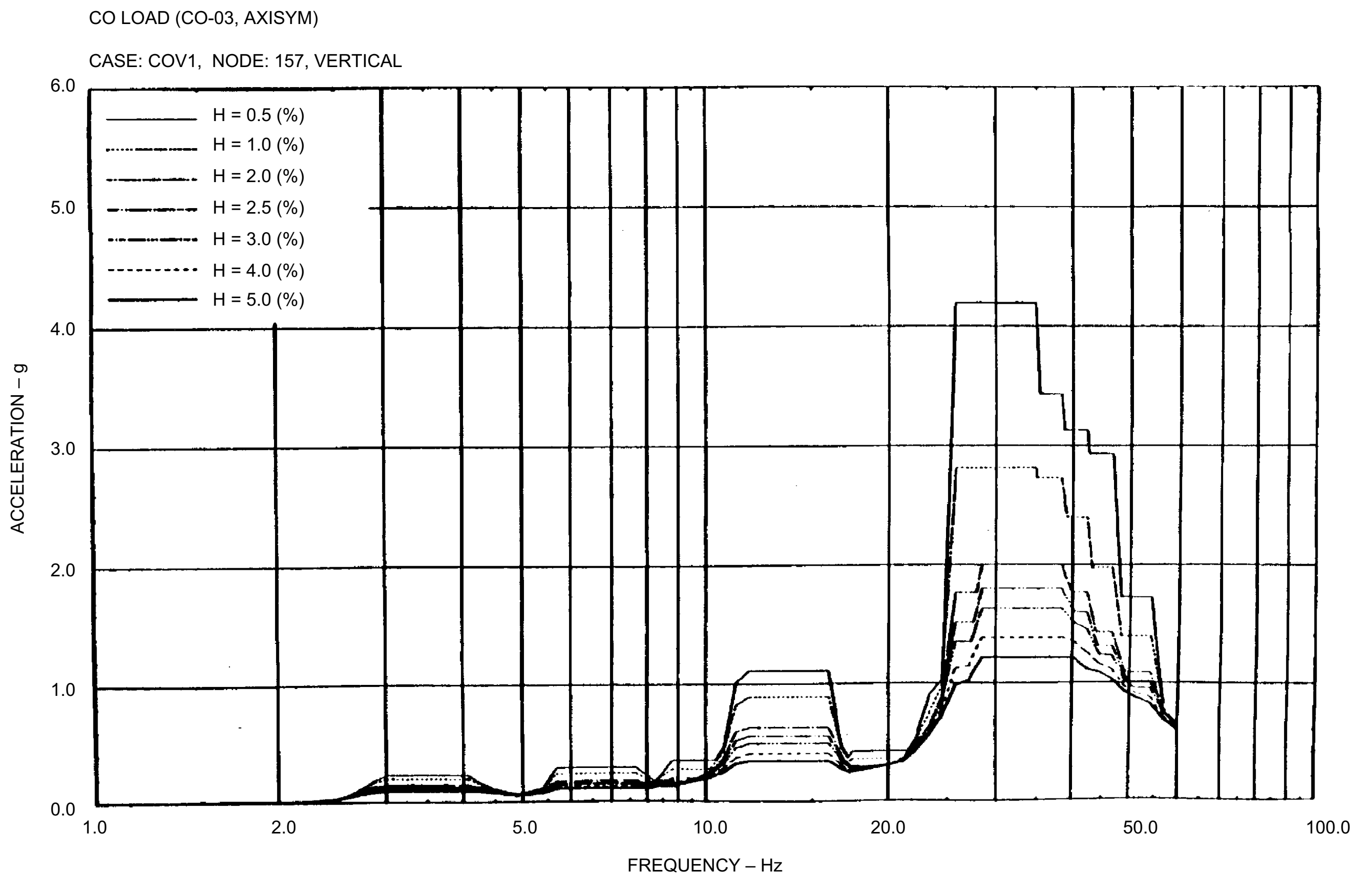

Figure 3G-97 Floor Response Spectrum—Case: COV1, Node: 157, Vertical

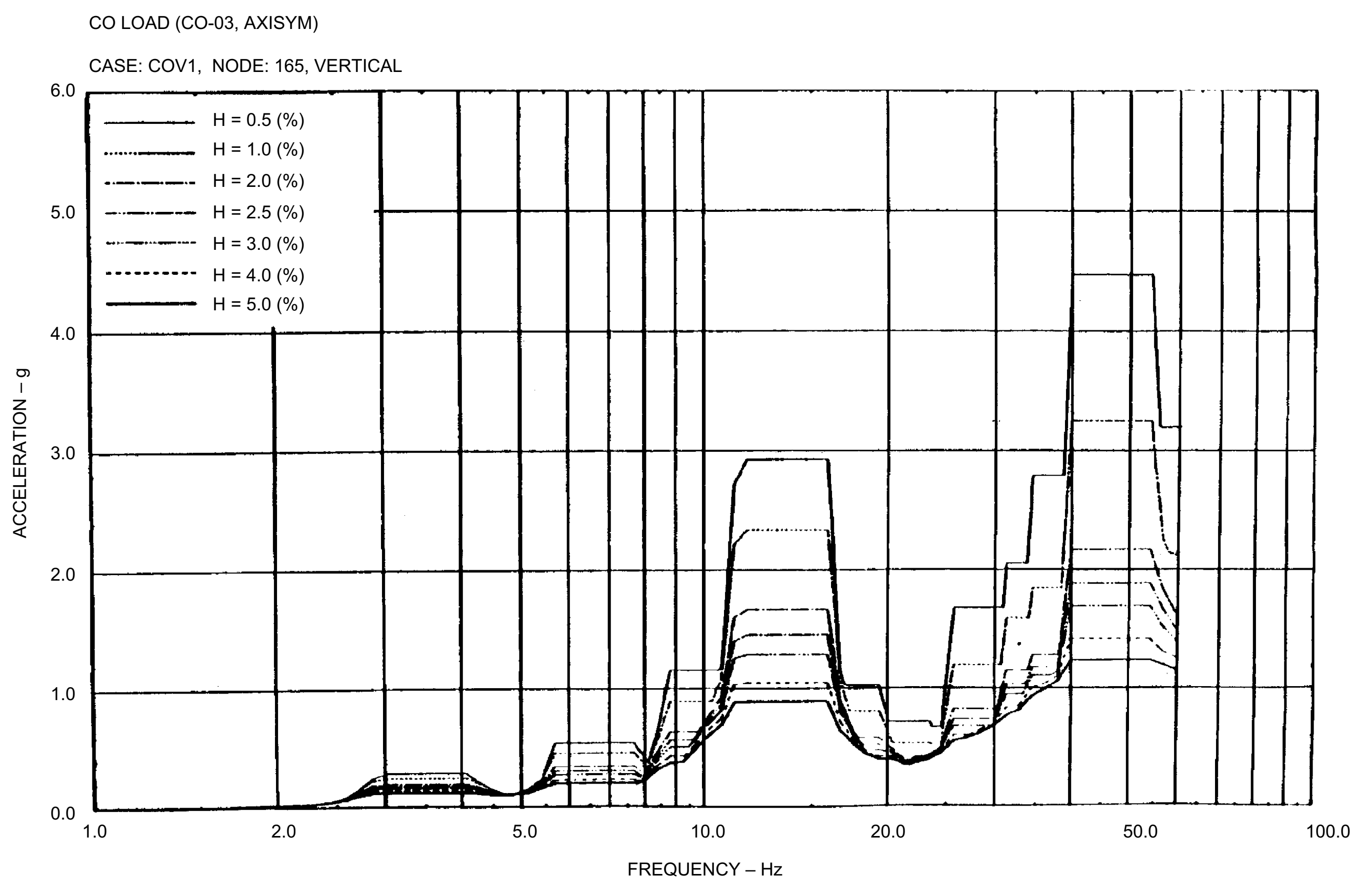

Figure 3G-98 Floor Response Spectrum—Case: COV1, Node: 165, Vertical

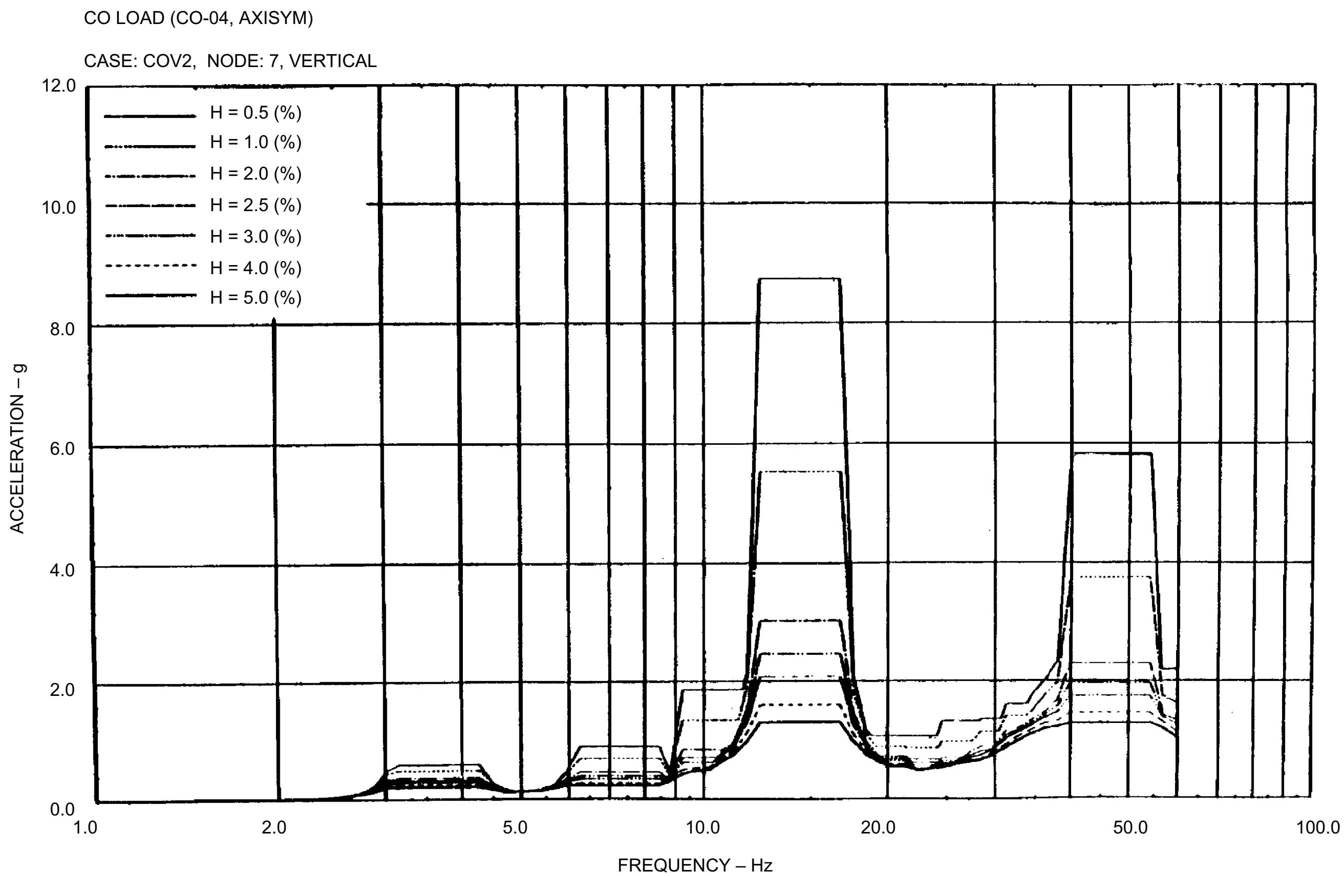

Figure 3G-99 Floor Response Spectrum—Case: COV2, Node: 7, Vertical

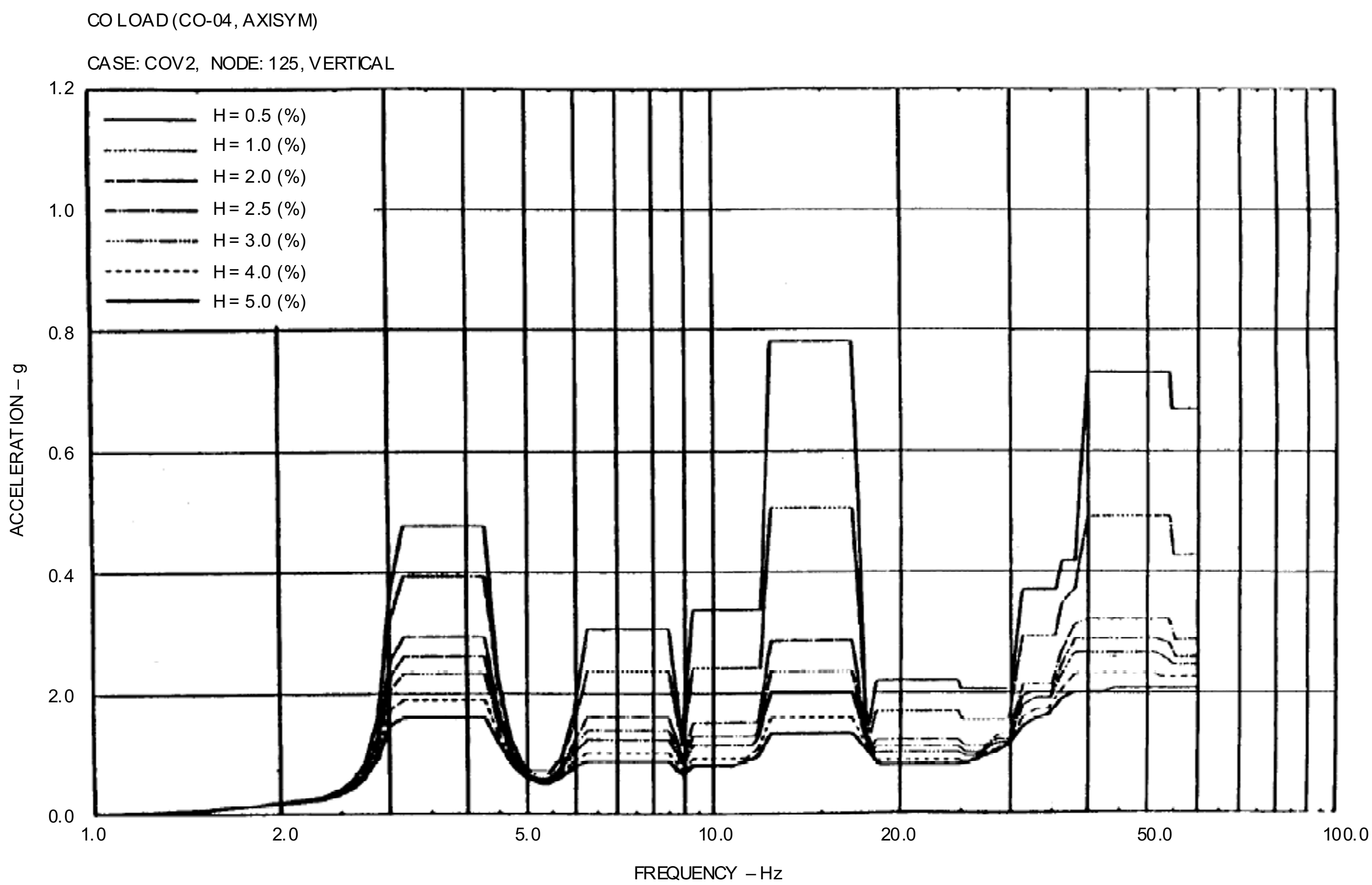

Figure 3G-100 Floor Response Spectrum—Case: COV2, Node: 125, Vertical

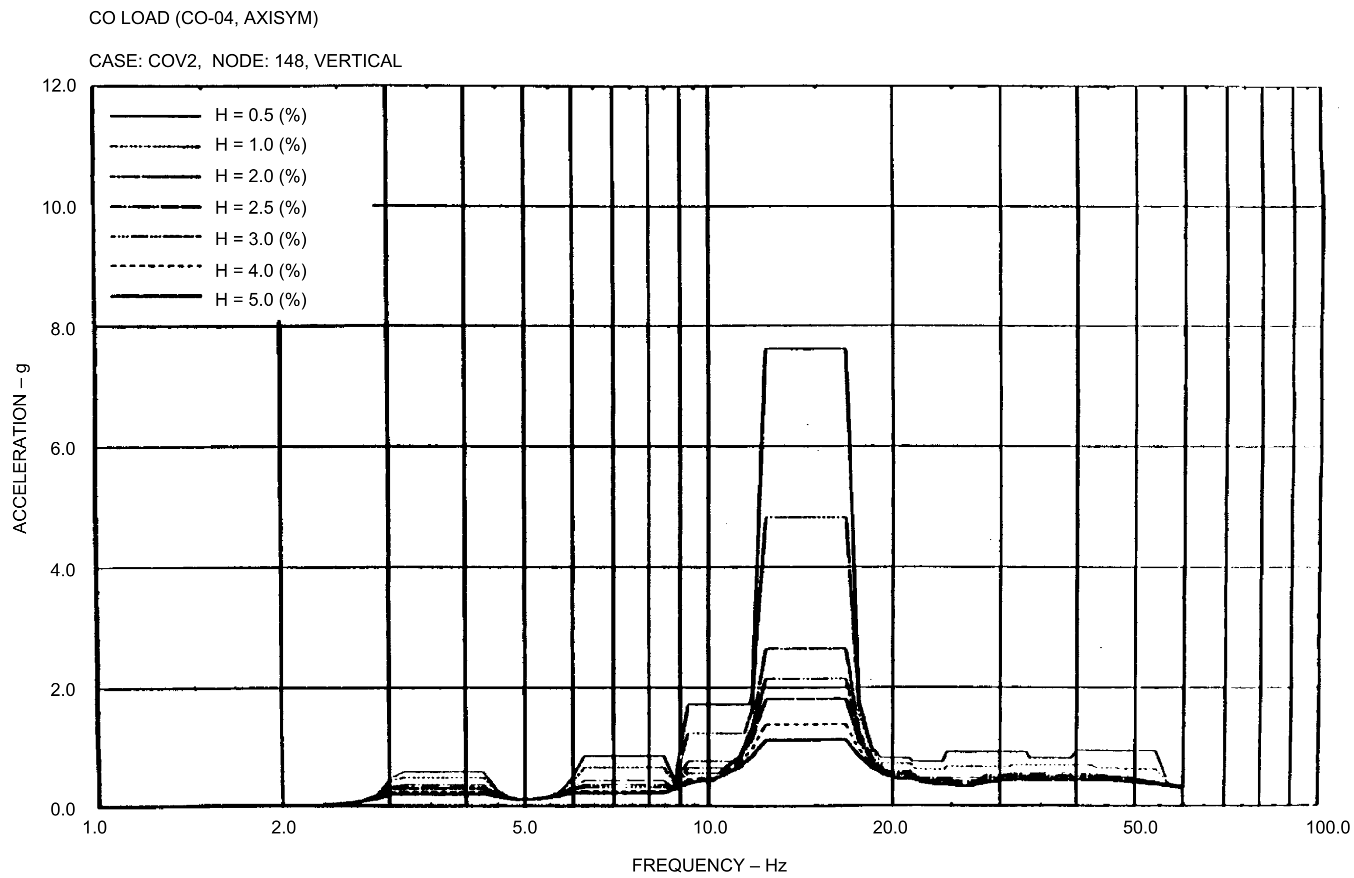

Figure 3G-101 Floor Response Spectrum—Case: COV2, Node: 148, Vertical

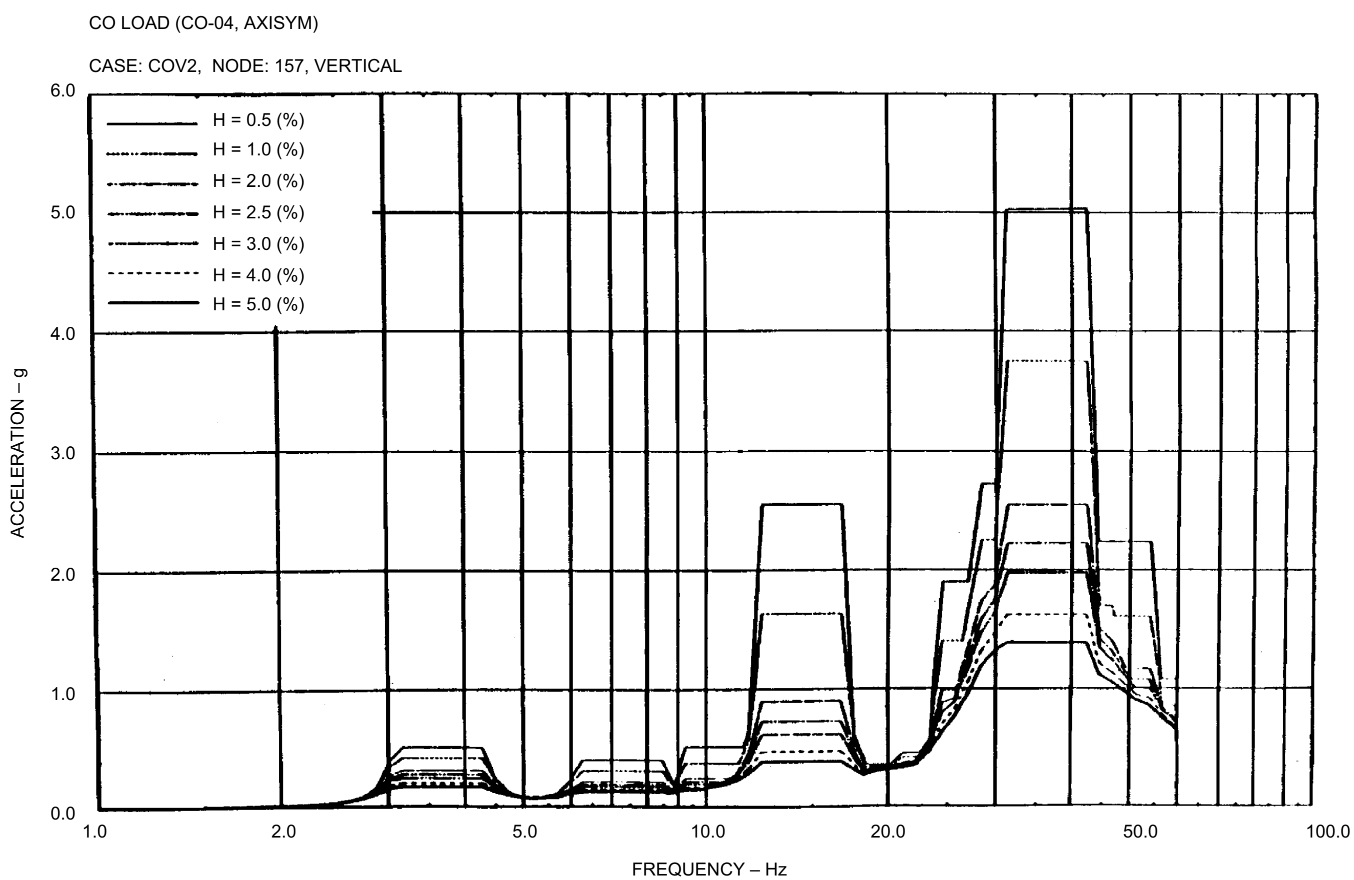

Figure 3G-102 Floor Response Spectrum—Case: COV2, Node: 157, Vertical

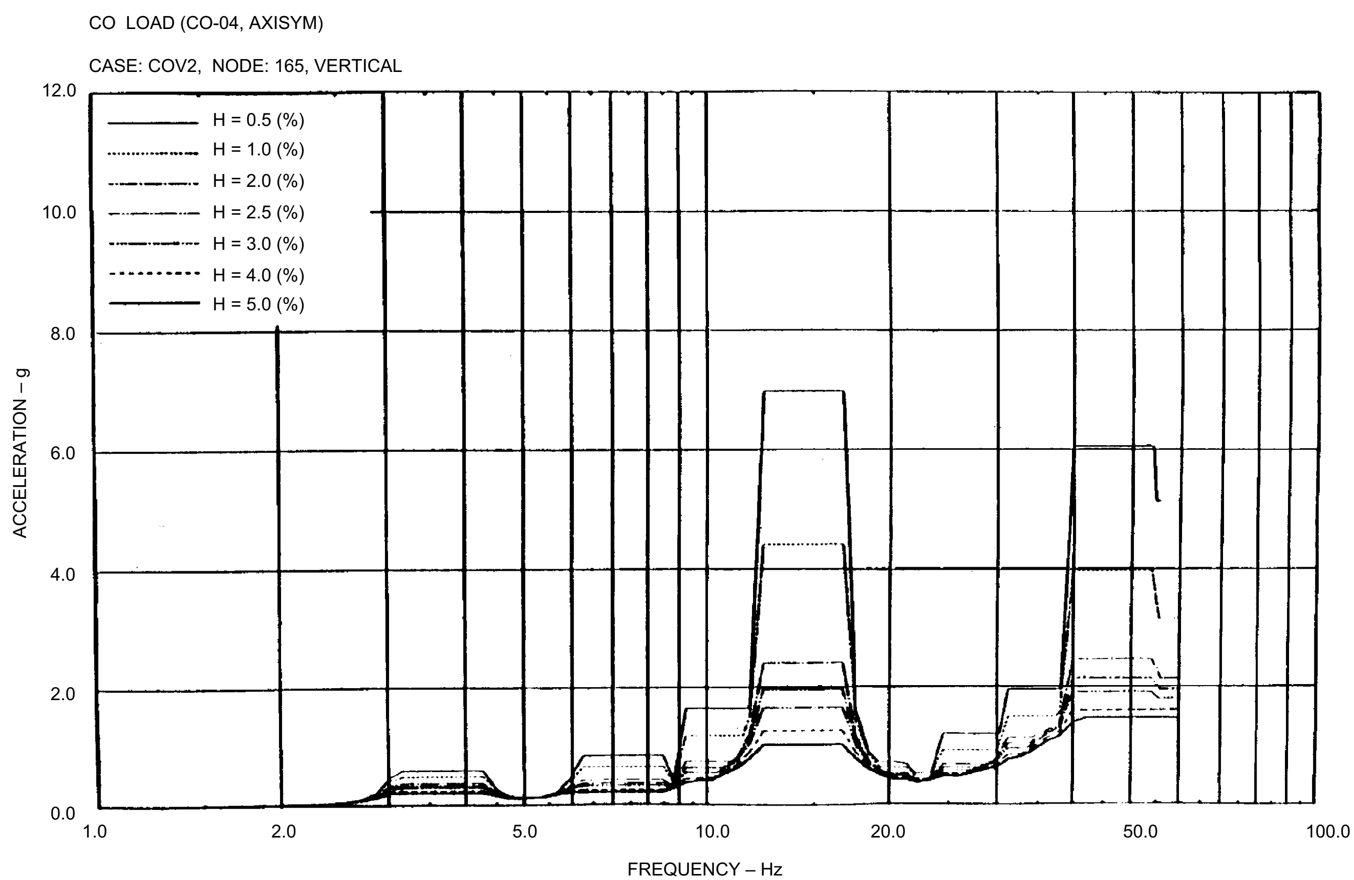

Figure 3G-103 Floor Response Spectrum—Case: COV2, Node: 165, Vertical

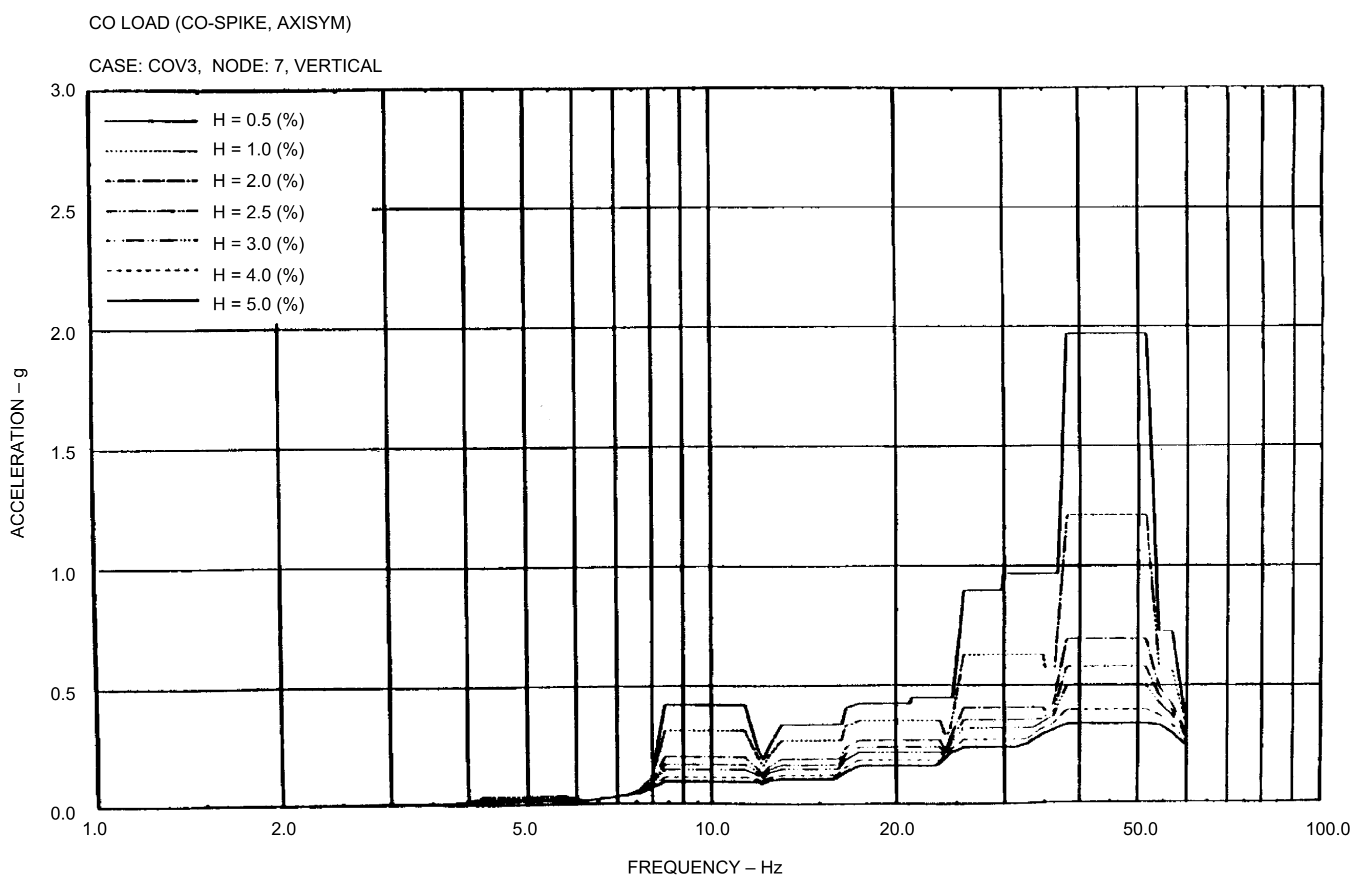

Figure 3G-104 Floor Response Spectrum—Case: COV3, Node: 7, Vertical

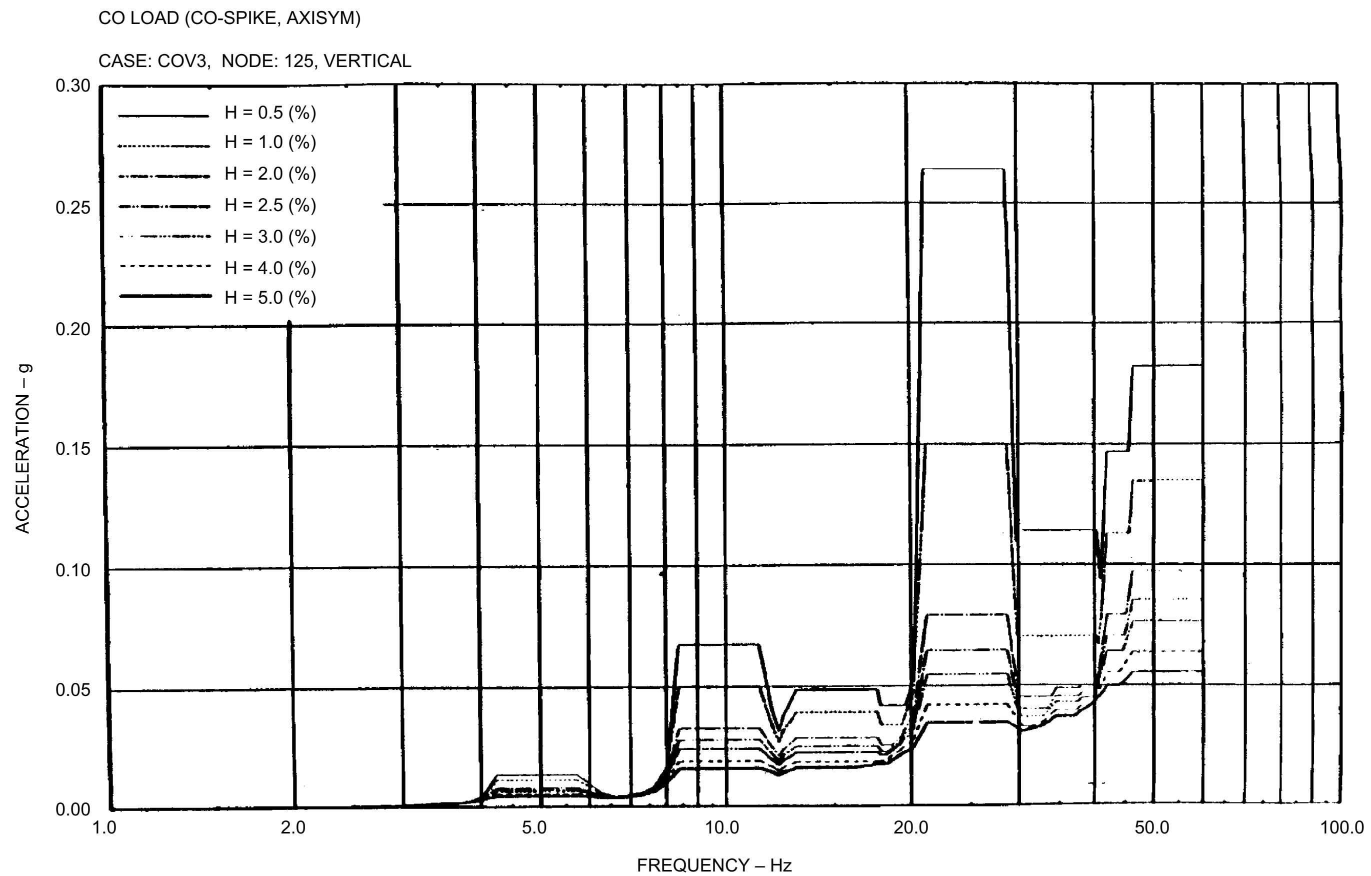

Figure 3G-105 Floor Response Spectrum—Case: COV3, Node: 125, Vertical

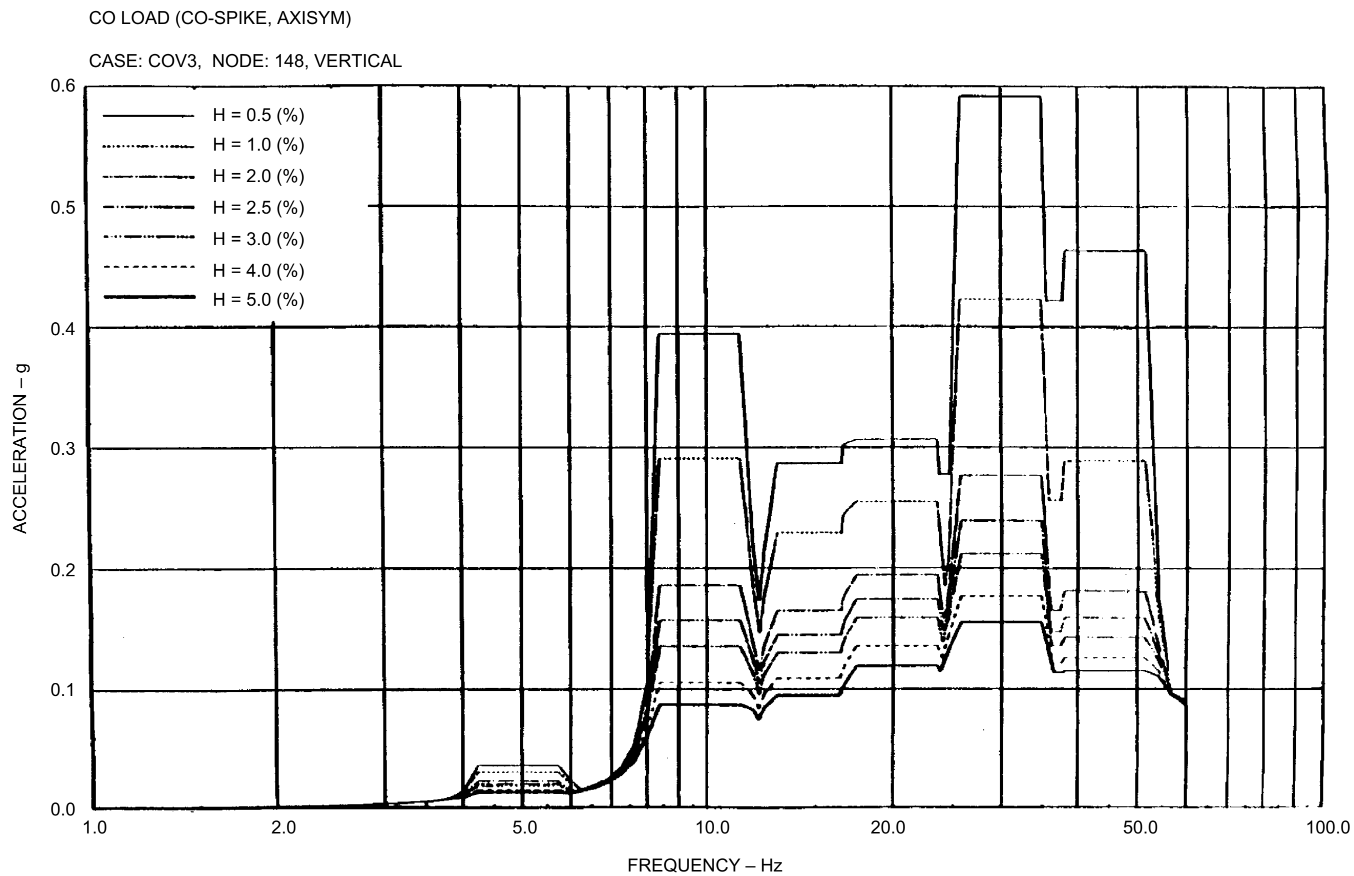

Figure 3G-106 Floor Response Spectrum—Case: COV3, Node: 148, Vertical

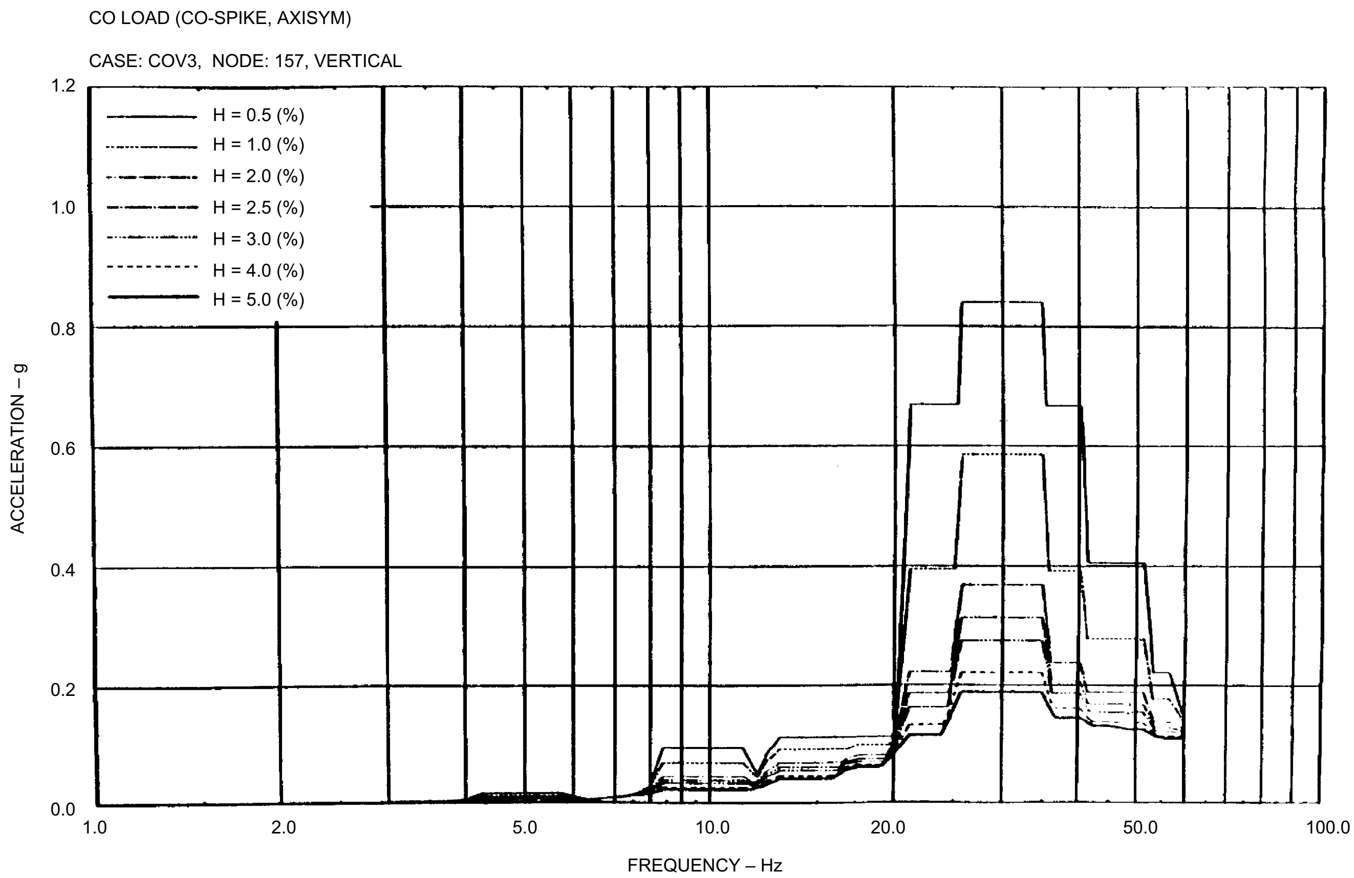

Figure 3G-107 Floor Response Spectrum—Case: COV3, Node: 157, Vertical

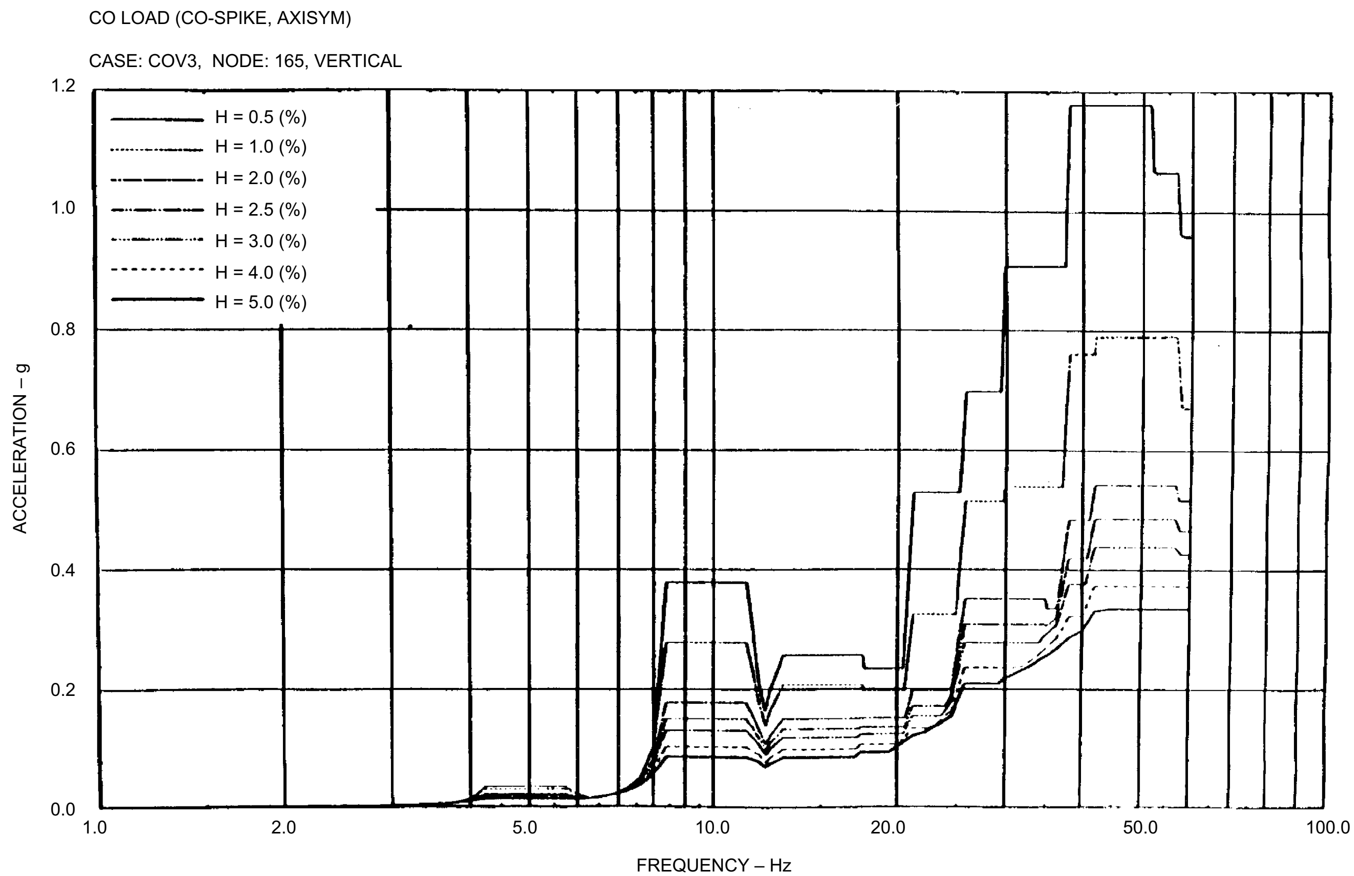

Figure 3G-108 Floor Response Spectrum—Case: COV3, Node: 165, Vertical

3H Design Details and Evaluation Results of Seismic Category I Structures

3H.1 Reactor Building

3H.1.1 Objective And Scope

The objective of this subsection is to document the structural design and analysis of the ABWR Reactor Building. The scope includes the design and analysis of the structure for normal, severe environmental, extreme environmental, abnormal, and construction loads.

This subsection addresses all applicable items included in Appendix C to USNRC Standard Review Plan, NUREG-0800, Section 3.8.4.

3H.1.2 Conclusions

The following are the major summary conclusions on the design and analysis of the Reactor Building:

- Based on the design drawings identified in Subsection 3H.1.5, stresses and strains in concrete, reinforcement, and the liner are less than the allowable stresses and strains per the applicable codes listed in Subsection 3H.1.4.1.
- The factors of safety against flotation, sliding, and overturning of the structure under various loading combinations are higher than the required minimum.
- The thickness of the roof slabs and exterior walls are more than the minimum required to preclude penetration, perforation or spalling resulting from impact of design basis tornado missiles. For extreme wind loads, the design basis tornado whose winds and missile loads bound the design basis hurricane are as specified in 3H.1.4.3.1.

3H.1.3 Structural Description

3H.1.3.1 Description of the Containment and the Reactor Building

The ABWR containment is integrated with, and fully contained within, the Reactor Building. The containment and the Reactor Building are supported by a 5.5m thick common foundation mat. The bottom of the foundation mat is embedded in the ground approximately 26m below grade. Figure 1.2-1 shows the location of the Reactor Building in relation to other plant structures. Figures 1.2-2 through 1.2-13k show the arrangement of the Reactor Building.

The containment structure is a right circular cylinder, 2m thick, with an inside radius of 14.5m and has a height of 29.5m measured from the top of the foundation mat to the bottom of the containment top slab.

The containment top slab is integral with the fuel pool girders and the containment wall. The top slab is nominally 2.2m thick. The slab thickness is increased to 2.4m beneath the fuel pool, steam dryer and steam separator pool. The top slab has an opening of 5.15m radius at center for the refueling head.

Major containment internal structures consist of the reactor pedestal, the reactor shield wall and the diaphragm floor.

The reactor pedestal is a composite steel and concrete structure which provides support for the reactor pressure vessel, the reactor shield wall, the diaphragm floor (D/F), access tunnels, horizontal vents, and the lower drywell access platforms. The pedestal consists of two concentric steel shells tied together by vertical steel stiffener plates and filled with concrete.

The diaphragm floor serves as a barrier between the drywell and the wetwell. It is a reinforced concrete circular slab, with an outside radius of 14.5m, and a thickness of 1.2m. The diaphragm floor is supported by the containment wall with a fixed-end connection and by the reactor pedestal with a hinged connection.

The internal surface of the containment is lined with a steel liner plate. The liner plate is fabricated from carbon steel except that stainless steel is used for the wetted portion of the suppression pool.

There are two 4.3m diameter access tunnels that penetrate the containment suppression pool wall and the reactor pedestal at azimuths 0° and 180°. The center lines of these tunnels are located about 7.1m above the top of the base slab.

The RPV is laterally restrained near the top of the reactor shield wall by RPV stabilizers.

The Reactor Building is a 59.6m x 56.6m reinforced concrete structure. On the periphery, there are 24 reinforced concrete columns which are connected by reinforced concrete walls. Inside the Reactor Building, there are 18 columns supporting the floors and the fuel pool girders.

The Reactor Building has six reinforced concrete floors which are monolithically connected to the containment.

The operating floor at elevation TMSL 31700 mm is connected to the fuel pool girders which are supported by the containment and the Reactor Building.

Reactor Cavity Shield Blocks as shown in Figure 3H.1-23 sit above the drywell head in the reactor cavity between the fuel pool girders. They provide missile shielding for the drywell head, and radiation shielding for maintenance personnel on the operating floor during normal plant operations.

The Reactor Building interior walls and the floor beams are not connected to the containment structure.

3H.1.4 Structural Design Criteria

3H.1.4.1 Codes

3H.1.4.1.1 Applicable Codes

(1) [*ASME/ACI 359: Boiler and Pressure Vessel Code Section III, Rules for Construction of Nuclear Power Plant Components, Division 2 Code for Concrete Reactor Vessels and Containments.*]*

(2) ACI 318: Building Code Requirements for Reinforced Concrete.

(3) [*ACI 349: Code Requirements for Nuclear Safety Related Concrete Structures.*]†

(4) AISC: Specification for Structural Steel Buildings - Allowable Stress Design (ASD) and Plastic Design.

(5) [*ANSI/AISC–N690: Specifications for the Design, Fabrication and Erection of Steel Structures for Nuclear Facilities, American Institute of Steel Construction.*]†

3H.1.4.1.2 ASME Code Jurisdictional Boundaries for the RCCV

Figure 3H.1-2 shows the ASME Section III, Division 2, Subsection CC Code jurisdictional boundaries for the concrete containment design.

3H.1.4.2 Site Design Parameters

The site design parameters are based on EPRI-ALWR requirements document for the standardized nuclear power plant (advanced). The following are some of the key design parameters:

(1) Soil Parameters:

- Minimum static bearing capacity demand: ≥718.20 kPa
- In addition for the load combinations involving seismic/dynamic loads, the dynamic bearing capacity demand shall also be met.
- Minimum shear wave velocity: 305 m/s
- Poisson's Ratio: 0.30 to 0.38
- Unit Weight: 1.9 to 2.2 t/m^3

(2) Maximum Ground Water Level:

* See Subsection 3.8.1.1.1.

† See Subsection 3.8.3.2.

– 0.61m below grade level

(3) Maximum Flood Level:

– 0.305m below grade level

(4) Maximum Snow Load:

– 2.39 kPa

(5) Design Temperatures:

– 0% exceedence values
Maximum: 46.1°C dry bulb/26.7°C coincident wet bulb.
Minimum: –40°C

– 1% exceedence values
Maximum: 37.8°C dry bulb/25°C coincident wet bulb.
Minimum: –23.3°C

(6) Seismology:

– SSE Peak Ground Acceleration (PGA): 0.30g (for both horizontal and vertical directions). SSE PGA is free field at plant grade elevation.

– SSE Response Spectra: Per Regulatory Guide 1.60

– SSE Time History: Envelope SSE response Spectra in accordance with SRP Section 3.7.1.

(7) Severe Wind:

– Basic wind speed 177 km/h (50 year recurrence interval).

– Importance Factors:
Safety related structures: 1.11
Non-safety related structures: 1.00

– Exposure Category: Exposure D

(8) Tornado:

– Maximum tornado wind speed: 483 km/h

– Maximum Rotational Speed: 386 km/h

– Maximum Translational Speed: 97 km/h

– Radius: 45.7m

– Maximum Pressure Drop: 13.83 kPa

– Maximum Rate of Pressure Drop: 8.28 kPa/s

– Missile Spectrum: See Table 2.0-1.

(9) Hurricane:

– Maximum Hurricane Wind Speed***: 257 km/h

(***See Chapter 2, Table 2.0-1 for Notes)

– Missile Spectra: Spectrum I

(10) Maximum Rainfall:

– Design rainfall is 493 mm/h. Roof parapets are furnished with scuppers to supplement roof drains, or are designed without parapets so that excessive ponding of water cannot occur. Such roof design meets the provisions of ASCE 7, Section 8.0.

3H.1.4.3 Design Loads and Loading Combinations

3H.1.4.3.1 Design Loads

3H.1.4.3.1.1 Dead Load (D) and Live Load (L) and (L_O)

Top Head wt.	60t
Normal Operating RPV wt.	1900t
Reactor shield wall	1000t
Suppression pool water depth	7.1m HWL
Tunnel (2 pieces)	135t
Equipment load on D/F slab	9.8 kPa
Fuel Pool Water depth	11.82m
Fuel elements in the fuel pool	1000t
RCCV top slab	14.32 kPa
Service floor	23.93 kPa
Other concrete floors	14.32 kPa
Stairs & platforms	4.81 kPa
Floors adjacent to equipment hatches	47.86 kPa
Fuel storage pool floor	71.79 kPa
Floor at RPV laydown	47.86 kPa
Wt. of crane	310t
Fuel Cask Load	150t

For the computation of global seismic loads, the value of floor live load is limited to the expected live load, L_o, during normal plant operation. The values of L_o are 25% of the above full floor live loads L.

Fuel pool floor shall also be designed for the following high-density storage racks loads.

Liner and Racks	= 53.0 kPa
Water	= 115.7 kPa
Consolidated Fuel	= 139.3 kPa
Total:	308.0 kPa

3H.1.4.3.1.2 Snow Load

Snow load shall be taken as 2.39 kPa. Snow load shall be reduced to 75% when snow load is combined with seismic loads.

3H.1.4.3.1.3 Wind Load (W)

Wind load based on basic wind speed of 177 km/h (Subsection 3H.1.4.2).

3H.1.4.3.1.4 Tornado Load (W_t)

For extreme wind loads, the design basis tornado winds and missile loads bound those of the design basis hurricane.

The tornado characteristics are defined in Subsection 3H.1.4.2. The tornado load, W_t is further defined by the following combinations:

$W_t = W_w$

$W_t = W_p$

$W_t = W_m$

$W_t = W_w + 0.5W_p$

$W_t = W_w + W_m$

$W_t = W_w + 0.5W_p + W_m$

where

$W_t =$ Total Tornado Load

$W_w =$ Tornado Wind Load

$W_p =$ Tornado Differential Pressure Load

$W_m =$ Tornado Missile Load

3H.1.4.3.1.5 Thermal Loads for the RCCV

- Design basis temperature: 15.5°C
- Normal Conditions:

Drywell	57°C
Suppression Pool	35°C
R/B Summer	40°C
R/B Winter	10°C

- Design Accident Conditions:

  Figure 3H.1-3 shows the section location for temperature distributions for various structural elements, and Table 3H.1-1 shows the magnitude of equivalent linear temperature distribution.

3H.1.4.3.1.6 Pressure Loads for the RCCV

3H.1.4.3.1.6.1 SIT and LOCA Loads

Table 3H.1-2 shows the structural integrity test and the LOCA (LBL and SBL/IBL) pressure loads.

3H.1.4.3.1.6.2 Condensation Oscillation (CO) and Chugging (CHUG) Loads

The condensation oscillation (CO) and chugging (CHUG) pressure loads along with Dynamic Load Factors (DLF) are provided in Table 3H.1-3. The pressure distribution for the CO loads is shown in Figure 3H.1-4, and for the chugging loads shown in Figure 3H.1-5.

3H.1.4.3.1.6.3 SRV Loads

The SRV loads along with Dynamic Load Factors (DLF) are provided in Table 3H.1-3. The pressure distributions for the SRV loads are provided in Figure 3H.1-6. The maximum SRV pressure which include a DLF of 1.8 are 171.6 kPa or –79.4 kPa.

3H.1.4.3.1.6.4 Steam Tunnel Subcompartment Pressure

The peak pressure in the R/B steam tunnel due to main steam line break is 150.0 kPa. This pressure includes a dynamic load factor of 2.0. Thermal loads need not be included due to short duration of the tunnel pressurization.

3H.1.4.3.1.6.5 Subcompartment Pressure in Other Compartments

For ABWR, the two systems, namely the Reactor Water Cleanup (CUW) system and the Reactor Core Isolation Cooling (RCIC) system are considered high energy during normal operation. The maximum pressure inside the affected subcompartments from the high energy line breaks (HELB) of these systems are determined to be 103.9 kPa. Hence, the affected subcompartments shall be evaluated for pressure of 103.9 kPa laterally applied together with other applicable loads. Since this pressure occurs gradually over many seconds, dynamic load factor need not be included. Thermal loads need not be included due to short duration of subcompartment pressurization.

3H.1.4.3.1.7 Seismic Loads

The design seismic loads are as follows:

- Figure 3H.1-8: Design Seismic Shears and Moments for Reactor Building Outer Walls
- Figure 3H.1-9: Design Seismic Shears and Moments for RCCV
- Figure 3H.1-10: Design Seismic Shears and Moments for RPV Pedestal and Reactor Shield Wall
- Table 3H.1-4: Maximum Vertical Acceleration

The seismic loads shall be composed of two perpendicular horizontal and one vertical component. The effects of the three components shall be combined based on the square root of sum of the squares (SRSS) method.

3H.1.4.3.1.8 Lateral Soil Pressure

Lateral Soil Pressure for wall design is provided in Figure 3H.1-11.

3H.1.4.3.2 Load Combinations and Acceptance Criteria

Load combinations and acceptance criteria for the various elements of the Reactor Building complex are discussed on the following subsections.

3H.1.4.3.2.1 Reinforced Concrete Containment Vessel (RCCV)

Table 3.8-1 gives detailed list of the load combinations and load factors per ASME Section III Division 2. Table 3H.1-5a loading combinations have been selected from Table 3.8-1 for the RCCV design evaluation.

3H.1.4.3.2.2 Diaphragm Floor (D/F) Slab

The diaphragm floor slab is a part of containment internal structures. The same loading combinations and acceptance criteria as shown in Subsection 3H.1.4.3.2.1 for the RCCV shall be used.

3H.1.4.3.2.3 RPV Pedestal

The RPV pedestal experiences the same loads and loading combinations as the RCCV. The loading combinations shown in Subsection 3H.1.4.3.2.1 for RCCV will also be used for the pedestal evaluation. The acceptance criteria are as per AISC for the steel components with allowable stresses limited to 0.9 Fy for factored loads. For the concrete portion of the pedestal, which is considered only to resist compressive loads, ACI 349 Code shall be used for acceptance criteria.

3H.1.4.3.2.4 Reactor Building (R/B) Concrete Structures Including Fuel Pool Girders

The loading combinations, as selected from Table 3.8-1, for the Reactor Building concrete structures including fuel pool girders are given in Table 3H.1-5b.

3H.1.4.4 Materials

3H.1.4.4.1 Concrete

- Concrete for the RCCV and the R/B including basemat shall have compressive strength, f'c = 2.76E+04 kPa, modulus of elasticity, E = 2.49E+07 kPa, Poisson’s ratio, $\mu = 0.2$, and shear modulus, G = 1.04E+07 kPa.
- Concrete fill in the pedestal; f'c = 2.76E+04 kPa, E= 2.49E+07 kPa, $\mu = 0.2$, and G = 1.04E+07 kPa.
- Concrete fill in the shield wall; f'c = 2.76E+04 kPa, E= 2.49E+07 kPa, $\mu = 0.2$, and G = 1.04E+07 kPa.

3H.1.4.4.2 Reinforcing Steel

Reinforcing steel shall be deformed billet steel conforming to ASTM A-615 grade 60. Minimum yield strength, Fy = 4.14E+05 kPa.

3H.1.4.4.3 Liner Plate

- Liner plate for RCCV in the wetted area shall be stainless steel conforming to ASME SA-240, Type 304L.
- Liner plate for the RCCV in the non-wetted area shall be 6.35 mm thick and conform to ASME SA-516 GR. 70.
- Liner Anchors: ASTM A-633 GR. C.
- Stainless steel cladding to conform to ASME SA-264.

3H.1.4.4.4 Other Materials

Other materials shall conform:

- Structural steel and connections - ASTM A-36
- High strength structural steel plates - ASTM A-572 or A441

- Bolts studs & nuts (dia, >19 mm) -ASTM A-325 or A490
- Bolts studs & nuts (dia. ≤19 mm) -ASTM A-307

3H.1.4.5 Stability Requirements

The R/B foundations shall have the following safety factors against overturning and sliding.

Load Combination	Overturning	Sliding	Floatation
D + H' + F + L_o + SSE	1.1	1.1	
D + F'			1.1

Where

D = Dead Load

F = Buoyant forces of design ground water

F' = Buoyant forces of design basis flood

H' = Lateral earth pressure

L_o = Live Load (both cases of live load having its full value and being completely absent shall be considered)

SSE = Safe Shutdown Earthquake

3H.1.5 Structural Design and Analysis Summary

3H.1.5.1 Analytical Model

The containment and the Reactor Building are analyzed as one integrated structure utilizing the finite element computer program STARDYNE. The finite element model consists of quadrilateral, triangular, beam and wedge elements. The quadrilateral and triangular elements are used to represent the slabs and walls. Beam elements are used to represent columns and beams. The model is shown in Figures 3H.1-12 to 3H.1-15.

Because the Containment and the Reactor Building are symmetrical with respect to the plane that goes through the containment vertical center line and runs parallel with the pool girders (plane of symmetry) it is considered adequate to model the 180° portion of the structure, which includes the 0°, 270° and 180° portions of the structure.

Two models are prepared. Both of them have exactly the same nodalization and element numbers but have different boundary conditions at the plane of symmetry. One model is for the application of symmetrical loads (symmetrical w.r.t. the plane of symmetry). The nodal points at the plane of symmetry are restrained for the following movements:

Displacement in the global Y direction
Rotation about X-axis
Rotation about Z-axis

The other model applies the boundary condition for the anti-symmetric loads. This model is applicable for load applied perpendicular to the plane of symmetry, and it is prepared specifically for the seismic loads. The nodal points in the plane of symmetry are restrained for the following movements:

Displacement in the global X direction
Displacement in the global Z direction
Rotation about Z-axis

The 6.35 mm thick liner plate is included, and is located at the pressure boundary of the containment and on top of diaphragm slab. The liner plate nodal points are connected to the containment nodal points by rigid beams. The liner plate quad elements described above are shown in Figure 3H.1-14. Pressure loads in the containment are applied on the liner plate.

The interior walls consist of load bearing and the non-load bearing walls.

The load bearing walls are between elevations TMSL −8200 and TMSL 4800. Non-load bearing walls are included only for the application and distribution of weight. These walls are represented by elements with very small modulus of elasticity and unit weight of 2.4 t/m^3.

The reactor pedestal is a composite structure consisting of steel plates and concrete. The thickness of the finite elements representing the reactor pedestal is calculated so as to account for the steel plates and concrete acting as a composite section. The effective plate element thickness and the adjusted concrete modulus of elasticity are calculated from the bending and tension stiffnesses of the reactor pedestal.

The nodal points are defined by a right hand cartesian coordinate system X1, X2, X3. This system, called the global coordinate system, has its origin located at the center of the containment at the bottom of RPV. The positive X1 axis is parallel with the fuel pool girder in the 0° direction of the containment; the X2 axis is perpendicular to the fuel pool girder in the 270° direction of the containment; the X3 axis is vertical upward. This coordinate system is shown in Figure 3H.1-12.

The forces and moments tabulated in Table 3H.1-6 through Table 3H.1-13 are based on the local coordinate system of the quadrilateral elements described above, which is based on right hand rule for numbering a quadrilateral element.

3H.1.5.2 Foundation Soil Springs

The foundation soil is represented by soil springs. The spring constants for rocking and translations are determined based on the following soil parameters:

- Shear wave velocity 305 m/s
- Unit weight 1.92 t/m^3
- Shear modulus 1.8 x 10^4 t/m^3
- Poisson's Ratio 0.38

Embedment effect was taken into consideration in the determination of the soil spring values.

The following are the values of the soil springs used in the analysis:

Vertical springs	1,398 t/m/m^2
Horizontal springs in two perpendicular directions	1,250 t/m/m^2

These spring values are multiplied by the foundation mat nodal point tributary areas to compute the spring constants assigned to the base slab nodal points. The soil springs are shown in Figure 3H.1-16.

3H.1.5.3 Loads

The following twelve basic loads are separately applied to the finite element model for static analysis:

(1) Dead Load

(2) Live Load

(3) Unit Pressure in Drywell

(4) Unit Pressure in Wetwell

(5) Condensation Oscillation (CO)

(6) Chugging Pressure

(7) Safety Relief Valve Load (SRV)

(8) Unit Pressure in Steam Tunnel

(9) Horizontal Seismic due to SSE

(10) Vertical Seismic due to SSE

(11) Thermal Load (30 Minutes)

(12) Thermal Load (6 hours)

The forces and moments obtained from STARDYNE analysis for some of the significant loads are given in Tables 3H.1-6 to 3H.1-9. Table 3H.1-6 gives the results for applied drywell pressure of 6.9 kPa. Table 3H.1-7 shows the forces and moments for 6.9 kPa applied wetwell pressure. Table 3H.1-8 lists the results for the SRSS of the three components of SSE. No signs are shown in Table 3H.1-8 since the seismic loads can change direction. Table 3H.1-9 shows the forces and moments for 6 hour thermal load case. Positive axial forces are tensile while negative forces are compressive.

The structural deformations for unit pressure in drywell, unit wetwell pressure, 6 hour thermal load and east-west direction safe shutdown earthquake are shown in Figures 3H.1-17 to 3H.1-20.

Figure 3H.1-17 shows deformations corresponding to 6.9 kPa drywell pressure.

Figure 3H.1-18 shows deformations due to 6.9 kPa pressure in wetwell.

Figure 3H.1-19 shows deformations due to 6.0 hours of thermal load application.

Figure 3H.1-20 gives deformation for 0°-180° seismic load due to SSE.

3H.1.5.4 Load Combinations

The load combinations used for design are shown in Subsection 3H.1.4.3.2.1 for the RCCV and in Subsection 3H.1.4.3.2.4 for the Reactor Building. Tables 3H.1-10 to Table 3H.1-13 show results of the critical load combinations.

Table 3H.1-10 shows the load combination results corresponding to Load Combination 1, SIT (1), which consists of D + L + P_t (1).

Table 3H.1-11 shows the results for Load Combination 8, LBL (30 min) which consists of D + L + 1.5(P_a + CO) + T_a.

Table 3H.1-12 shows the results for Load Combination 15, LBL (30 min) + SSE, which consists of D + L + P_a + CO + T_a+SSE + R_a + Y.

Table 3H.1-13 shows the load combination results corresponding to Load Combination 15a, 15b, IBL/SBL (6 hrs) + SSE, and consists of D + L + P_a + CO + SRV + T_a + SSE.

In these tables, the forces and moments are divided into three groups, for the convenience of input into the CECAP program for the rebar stress evaluation. The first row represents the forces and moments caused by Thermal Load (T_A). The second row lists the forces and moments due to loads 1 through 8 listed in Subsection 3H.1.5.3. The last row lists the resulting forces and moments due to seismic loads (SRSS).

3H.1.5.5 Structural Design

The evaluation is based on the loads, load factors and load combinations indicated in Subsection 3H.1.4.

Figure 3H.1-21 shows the location of the sections which were selected for evaluation. Bechtel computer program 'CECAP' was used for the evaluation of stresses in rebar and concrete and strains in the liner plate. The input to CECAP consists of rebar ratios, material properties, and element geometry at the section under consideration together with the forces and moments from the STARDYNE analysis, which are shown in Tables 3H.1-10 through 3H.1-13. Table 3H.1-14 lists the rebar ratios used in the evaluation. At each section, in general, three elements were analyzed at azimuth 180°, 225° and 270°. Table 3H.1-15 through 3H.1-18 show the rebar and concrete stresses at these sections for the representative elements.

Figure 3H.1-22 shows flow chart for the structural analysis and design.

Figures 3H.1-28 through 3H.1-37 present the design drawings used for the evaluation of the containment and the Reactor Building Structural design.

3H.1.5.5.1 Containment Structure

3H.1.5.5.1.1 Containment Wall

Sections 1 through 6 shown in Figure 3H.1-21 are considered critical sections for the containment wall. Maximum stress was found to be 3.72×10^5 kPa in the meridional rebar at Section 1 near the bottom of the RCCV wall due to load combinations 15a and 15b, as shown in Table 3H.1-18. The maximum stress in the circumferential rebar was found to be 3.53×10^5 kPa which also occurs at Section 1. Table 3H.1-19 shows liner plate strains. The liner maximum strain was found to be 0.00197 at Section 1, which is within allowable limits given in Table CC-3720-1, ASME Code Section III, Division 2.

3H.1.5.5.1.2 Containment Top Slab

Sections 7, 8 and 9 were examined for the Containment Top Slab. The location of these sections are shown in Figure 3H.1-21. The maximum stress of 2.53×10^5kPa was found in the horizontal rebar in the top layer at Section 8 as shown in Table 3H.1-18. Maximum Liner strain was found to be 0.000499 at Section 8 as shown in Table 3H.1-19.

3H.1.5.5.1.3 Containment Foundation Mat

Sections 10 to 14 were evaluated for the basemat within the Containment Walls and Section 18, for outside the Containment. The sections are shown in Figure 3H.1-21. The maximum rebar stress was calculated as 3.18×10^5 kPa at Section 12 just outside the RPV Pedestal and is shown in Table 3H.1-18. The liner plate maximum strain was found to be 0.000439 at Section 14 as shown in Table 3H.1-19.

3H.1.5.5.2 Containment Internal Structures

3H.1.5.5.2.1 Diaphragm Floor

Sections 15, 16, and 17, were selected for evaluation of the diaphragm floor. The sections are shown in Figure 3H.1-21. The results of the analysis are shown in Tables 3H.1-15 to 3H.1-19. The maximum stress in the radial rebar was found to be 2.45×10^5 kPa at Section 15 shown in Table 3H.1-18, whereas the maximum stress in the circumferential rebar was found to be 9.9×10^4 kPa at Section 17, as shown in Table 3H.1-17. The maximum strain in the liner plate was found to be 0.000848 cm/cm (compressive) as shown in Table 3H.1-19.

3H.1.5.5.2.2 Reactor Pedestal

Sections 19, 20 and 21 were selected for evaluation of the pedestal. These are shown in Figure 3H.1-21. The forces and moments for the load combinations are shown in Tables 3H.1-10 to 3H.1-13. The results of the analysis are shown in Table 3H.1-20. The maximum membrane stress in cylindrical steel plate was found to be 2.07×10^5 kPa and the maximum shear stress in the stiffener plates was found to be 1.84×10^5kPa, which are within the allowable stress limits.

3H.1.5.5.3 Reactor Building

Sections 22 through 34 were analyzed for the Reactor Building outside the containment. The sections are shown in Figure 3H.1-21. Sections 22 to 24 were selected for the R/B Outside Wall, Sections 25 to 29 for the spent fuel pool walls and floor and Sections 30 to 34 for the R/B Slabs.

3H.1.5.5.3.1 R/B Outside Walls

These walls resist the lateral soil pressure besides the forces and moments from the other loads. The design lateral soil pressures are shown in Figure 3H.1-11. Out-of-plane moments and shears due to these were calculated at various wall sections and added to forces and moments from other loads before the wall sections were analyzed. The moments and shears due to lateral soil pressure at rest are given in Table 3H.1-21 whereas Table 3H.1-22 gives the resulting moments and shears due to lateral soil pressure during SSE condition.

The maximum rebar stress of 3.75×10^5 kPa was found in the horizontal rebar at Section 24 as shown in Table 3H.1-18. The maximum vertical rebar stress was found to be 3.61×10^5kPa, also as shown in Table 3H.1-18.

3H.1.5.5.3.2 Fuel Pool Girders

The maximum stress of 3.39×10^5 kPa was found in the vertical rebar at Section 29 shown in Table 3H.1-17, whereas the maximum stress of 2.98×10^5 kPa was found in the horizontal rebar at Section 28 as shown in Table 3H.1-18.

3H.1.5.5.3.3 R/B Floor Slabs

Sections 30 to 32 were selected for the floor slabs at elevations TMSL –1,700, TMSL 4,800 and TMSL 12,300 (see Figure 3H.1-21) at their junction with the Containment Wall. The forces and moments at these sections are shown in Tables 3H.1-10 to 3H.1-13. The resulting rebar and concrete stresses are shown in Tables 3H.1-15 to 3H.1-18. The rebar stresses are within the allowable stress limits. The slabs at elevation TMSL – 1,700 and TMSL 4,800 were also evaluated for buckling under the lateral soil pressure loads and were found to be adequate.

3H.1.5.5.3.4 Steam Tunnel Floors

Sections 33 and 34 were analyzed for the steam tunnel. The pipe break accident pressure load was applied to the steam tunnel wall and floor elements. No thermal gradient was applied across the wall thickness.

The sections are shown in Figure 3H.1-21 The forces and moments are given in Tables 3H.1-10 to 3H.1-13. The rebar and concrete stresses are shown in Tables 3H.1-15 to 3H.1-18. The stresses are all within the allowable limits.

3H.1.5.6 Foundation Stability

The Reactor Building was evaluated for stability against overturning, sliding and floatation. The energy approach was used in calculating the factor of safety against overturning.

The factors of safety against overturning, sliding and floatation are given in Table 3H.1-23. All of these meet the acceptance criteria.

Maximum soil bearing stress was found to be 700.2 kPa due to dead plus live loads which was found to increase to 2336.0 kPa when seismic and other loads are included.

3H.1.5.7 Tornado Missile Evaluation

The minimum thickness required to prevent penetration and concrete spalling was evaluated. The US Army Technical Manual TM-5-855-1 was used to calculate penetration by a tornado missile. This result was doubled to arrive at the minimum spalling thickness. The minimum thickness required is 384 mm and 335 mm for wall and slab respectively, which are less than that provided. The winds and missile loads of design basis tornado bound those of design basis hurricane.

Table 3H.1-1 Equivalent Linear Temperature Distributions at Various Sections*

Section	Side		LOCA (30 min)		IBA/SBA (6 h)	
	T_1	T_2	T_2	T_1	T_2	T_1
			°C	°C	°C	°C
A	Soil	D/W	54.8	13.8	60.2	11.2
B	Soil	S/C Pool	37.1	14.6	41.9	12.4
C	SFP	D/W	60.5	49.3	84.2	37.6
D	Indoor	D/W	47.9	13.1	75.6	1.0
E	Indoor	S/C	38.4	13.0	53.5	6.5
F	Indoor	S/C Pool	37.4	13.6	53.5	6.5
I	S/C	D/W	52.9	36.1	84.8	43.1
M,K	D/S	D/W	49.4	12.6	71.8	2.6
J,L	SFP	D/W	60.8	49.4	81.0	40.4
X	S/C	D/W	52.9	36.1	84.8	43.1

* Note: See Figure 3H.1-3 for the Location of Sections

D/W = Drywell
S/C = Suppression Chamber
D/S = Diaphragm Slab
SFP = Spent Fuel Pool

Table 3H.1-2 SIT and LOCA Pressure Loads

Event	Pressure, P_a	
	Drywell Pressure, P_D (kPaG)	Supp. Pool Pressure, P_S (kPaG)
SIT (1)	358.90	358.90
SIT (2)	309.90	137.30
LBL (30 min)	309.90	241.25
SBL/IBL (6 hrs)	241.25	206.93

SIT = Structural Integrity Test

LBL = Large Break LOCA

IBL = Intermediate Break LOCA

SBL = Small Break LOCA

Table 3H.1-3 Hydrodynamic Loads

Hydrodynamic Loading	Pressure (kPaD)	Dynamic Load Factor (DLF)
CO	+154.95 to –154.95	1.8
Chugging (CHUG)	+35.99 to –21.18	1.5
SRV	+95.12 to –44.13	1.8

Notes: Load Distribution for Wetwell wetted surfaces are as follows:

Figure 3H.1-4: Condensation Oscillation Loads

Figure 3H.1-5: Chugging Pool Boundary Loads

Figure 3H.1-6: SRV Pool Boundary Loads

Table 3H.1-4 Maximum Vertical Acceleration

Reactor Building Outside RCCV		RCCV		Reactor Shield Wall & RPV Pedestal	
Node No.	Max. Vert. Accln. (g)	Node No.	Max. Vert. Accln. (g)	Node No.	Max. Vert. Accln. (g)
95	0.63	89	0.72	70	0.43
96	0.52	90	0.67	78	0.43
98	0.47	91	0.64	79	0.42
100	0.43	92	0.58	80	0.41
102	0.39	93	0.43	81	0.40
103	0.34	94	0.34	82	0.39
104	0.32	88	0.31	71	0.38
105	0.31	106	0.31	83	0.37
88	0.31			84	0.34
106	0.31			73	0.34
107	1.22			85	0.33
108	1.56			86	0.32
109	1.88			87	0.31
110	1.04				
111	0.54				
112	0.47				

Note: See Figure 3A-8 for Locations and Node Numbers.

Table 3H.1-5 Load Combinations, Load Factors and Acceptance Criteria for the Reinforced Concrete Containment

Notes:			
1.	For any load combination, if the effect of any load component, other than D, reduces the combined load, then the load component is deleted from the load combination.		
2.	Since P_a, P_i, P_s, T_a, SRV and LOCA are time dependent loads, their effects will be superimposed accordingly.		
3.	S	=	Allowable stress as in ASME Section III, Division 2, Subarticle CC-3430 for Service Load Combinations.
	U	=	Allowable stress as in ASME Sections III, Division 2, Subarticle CC-3420 for Factored Load Combinations.

Table 3H.1-5a Selected Load Combinations for the RCCV

No.	Event	Load Combination	Acceptance Criteria
1	SIT (1)	D + L + P_t (1)	S
1*	SIT(2)	D + L + P_t (2)	S
8	LBL (30 min)	D + L + 1.5 (P_a + CO) + T_a	U
8a,8b	IBL/SBL (6 h)	D + L + 1.5 (P_a + CO) + 1.25 SRV + T_a	U
15	LBL (30 min) + SSE	D + L + P_a + CO + T_a + SSE + R_a + Y	U
15a, 15b	IBL/SBL (6 h) + SSE	D + L + P_a + CO + SRV + T_a + SSE	U

P_t	=	Test Pressure During SIT; P_t (1) indicates test pressures of 358 kPaG in the drywell and wetwell; and P_t (2) indicates test pressures of 310 kPaG and 138 kPaG in the drywell and wetwell, respectively.
P_a, T_a, R_a	=	Containment Pressure, Temperature and Pipe Support reaction loads associated with the LOCA.
Y	=	Local effects on the containment due to DBA. These include Y_r (restraint reaction), Y_J (jet impingement) and Y_m (missile impact). These are to be considered only for those events which are not eliminated through application of Leak-Before-Break (LBB).
S	=	Allowable stress as in ASME Section III, Division 2, Subarticle CC-3430 for Service Load Combinations.
U	=	Allowable stress as in ASME Section III, Division 2, Subarticle CC-3420 for Factored Load Combinations.

Table 3H.1-5b Selected Load Combinations for the Reactor Building

Event	Concrete Structures Including Fuel Pool Girders Load Combination	Acceptance Criteria
	1.4D + 1.7L	U
	D + L + T_o + R_o + SRV + W_t	U
	D + L + P_t (1)	
LBL (30 min)	D + L + 1.5 (P_a + CO) + T_a	U
LBL/SBL (6 h)	D + L + 1.5 (P_a + CO) + 1.25 SRV + T_a	U
LBL (30min) + SSE	D + L + P_a + CO + T_a + SSE + R_a + Y	U
IBL/SBL (6 h) + SSE	D + L + P_a + CO + SRV + T_a + SSE	U
W_t =	Tornado loading including effect of missile impact.	
R_o =	Pipe reaction during normal operating condition.	
T_o =	Normal operating thermal load.	
U =	Section strength required to resist design loads based on strength design methods described in ACI-349 Code.	
Notes:	1) L includes lateral earth pressure on the external walls. 2) The design basis tornado winds and missile loads bound those of the design basis hurricane.	

Table 3H.1-6 Results of "Stardyne" Analysis for Unit Drywell Pressure: 6.9 kPa [Pd]

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
1	RCCV Wetwell Bottom	1	185.18	-7.18E+02	2.06E+04	-4.06E+02	-1.49E+03	-7.79E+03	6.32E+00	1.05E+01	1.94E+03
		5	225.80	-3.43E+02	2.13E+04	-1.45E+03	-1.45E+03	-7.78E+03	5.09E+01	1.75E+00	1.87E+03
		9	265.09	2.06E+02	1.98E+04	-1.82E+02	-1.82E+02	-7.83E+03	1.64E+01	-3.5E+00	1.82E+03
2	RCCV Wetwell Mid-Height	37	185.18	-8.73E+02	2.23E+04	-3.62E+02	-2.12E+02	-1.11E+03	8.63E+00	-3.5E+00	-5.88E+02
		41	225.80	-10.52E+02	2.08E+04	-2.54E+03	5.31E+01	-7.80E+02	7.98E+01	1.40E+01	-7.37E+02
		45	265.09	.21E-02	1.76E+04	-4.11E+02	-2.63E+02	-9.77E+02	8.72E+00	-1.92E+01	-7.58E+02
3	RCCV Wetwell Top	91	185.18	1.95E+04	2.12E+04	7.19E+02	1.83E+03	1.21E+04	-7.21E+01	-3.85E+01	5.53E+03
		95	225.80	1.68E+04	2.24E+04	-3.28E+03	1.83E+03	1.06E+04	-6.06 E+01	-1.15E+02	4.93E+03
		99	265.09	1.72E+04	1.32E+04	-7.23E+02	1.88E+03	1.03E+04	-1.70E+01	-1.40E+01	4.99E+03
4	RCCV Drywell Bottom	109	185.18	2.71E+04	4.25E+04	2.28E+03	-3.14E+03	-9.96E+03	-1.37E+02	-8.75E+00	9.70E+03
		113	225.80	2.49E+04	4.75E+04	-5.96E+03	-2.50E+03	-1.02E+04	-1.10E+02	-8.22E+01	1.20E+04
		117	265.09	2.48E+04	3.50E+04	-9.54E+02	-2.29E+03	1.03E+04	-4.07E+01	7.00E+00	1.21E+04
5	RCCV Drywell Mid-Height	127	185.18	2.64E+04	3.08E+04	2.75E+03	-4.97E+02	2.99E+03	4.45E+01	5.25E+01	-2.66E+03
		131	225.80	2.48E+04	4.72E+04	-7.30E+03	1.35E+03	8.62E+03	2.58E+02	9.80E+01	-5.95E+02
		135	265.09	2.46E+04	3.34E+04	-7.03E+02	1.89E+03	9.10E+03	-4.83E+01	-7.00E+01	-4.04E+02

Table 3H.1-6 Results of "Stardyne" Analysis for Unit Drywell Pressure: 6.9 kPa [Pd] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
6	RCCV Drywell Top	163	185.18	1.78E+04	2.74E+04	2.30E+03	-6.69E+03	-3.35E+04	5.06E+01	-2.03E+02	-1.55E+04
		167	225.80	1.09E+04	3.99E+04	-9.49E+03	-8.97E+03	-4.0E+04	6.65E+02	-3.06E+02	-2.01E+04
		171	265.09	8.55E+03	3.04E+04	-1.07E+03	-6.79E+03	-3.60E+04	1.47E+02	5.07E+01	-1.91E+04
7	RCCV Top Slab @ RCCV Wall	1616	185.18	1.81E+04	1.58E+04	2.13E+04	1.94E+03	-2.57E+04	1.11E+02	7.35E+01	1.86E+04
		1620	225.81	1.92E+03	3.25E+04	3.25E+04	9.55E+03	-1.70E+04	-4.88E+03	-7.02E+03	3.36E+04
		1624	265.09	3.92E+03	2.00E+04	2.00E+04	1.09E+03	-2.95E+04	6.24E+02	2.45E+02	2.24E+04
8	RCCV Top Slab @ Center	1634	185.18	7.35E+03	9.01E+03	9.01E+03	1.79E+04	1.89E+04	1.01E+03	-2.66E+02	4.80E+03
		1638	225.36	3.29E+03	1.07E+04	1.07E+04	3.05E+04	3.77E+04	2.94E+02	2.88E+02	4.64E+03
		1642	265.00	-7.41E+03	1.86E+04	1.86E+04	1.61E+04	1.90E+04	-6.42E+02	-1.44E+01	2.66E+03
9	RCCV Top Slab @ Drywell Head Opening	1652	185.00	-9.02E+03	-4.04E+03	3.60E+03	2.21E+04	-2.28E+04	1.82E+03	-2.39E+03	2.83E+04
		1656	225.00	3.98E+03	-1.83E+03	2.10E+04	1.43E+03	1.39E+04	1.30E+04	-2.49E+03	1.72E+04
		1660	265.00	-2.33E+04	-2.99E+02	-4.04E+04	2.24E+04	-1.74E+04	-3.95E+03	1.24E+03	2.24E+04
10	Basemat Cavity @ Center	929	270.00	1.92E+04	1.75E+04	1.75E+00	-2.24E+05	-2.17E+05	-5.43E+00	1.40E+01	-3.85E+04
11	Basemat Inside RPV Pedestal	926	192.04	1.83E+04	1.86E+04	8.60E+02	-1.95E+05	-1.9E+05	4.41E+03	-2.57E+03	-1.36E+04
		939	270.00	1.91E+04	1.75E+04	-1.05E+01	-2.09E+05	-1.89E+05	-1.33E+01	5.25E+00	-1.74E+04

Table 3H.1-6 Results of "Stardyne" Analysis for Unit Drywell Pressure: 6.9 kPa [Pd] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
12	Basemat Outside RPV Pedestal	908	185.01	6.55E+03	-2.70E+03	1.33E+02	-1.04E+05	-1.27E+05	9.68E+01	-1.30E+03	-4.40E+04
		912	225.00	7.37E+03	-3.42E+03	7.97E+02	-1.08E+05	-1.24E+05	-1.50E+03	-1.94E+02	-4.38E+04
		916	264.99	8.04E+03	-4.10E+03	1.30E+02	-1.07E+05	-1.24E+05	-2.50E+02	-1.75E+01	-4.27E+04
13	Basemat Between RCCV & RPV Pedestal	890	185.18	3.80E+03	2.10E+01	1.19E+02	-7.45E+04	-1.74E+04	-4.09E+02	9.81E+01	3.10E+04
		894	225.36	4.76E+03	-7.70E+02	8.21E+02	-7.85E+04	-1.65E+04	-5.13E+02	-7.32E+02	-3.03E+04
		898	265.00	5.57E+03	-1.37E+03	9.98E+01	-7.85E+04	-1.85E+04	1.93E+02	-4.55E+01	-2.96E+04
14	Basemat Inside RCCV	872	185.18	2.81E+03	1.01E+03	5.25E+01	-3.97E+04	4.53E+04	-5.50E+02	-1.77E+02	-2.41E+04
		876	225.81	3.86E+03	2.61E+02	9.74E+02	-4.55E+04	4.59E+04	8.62E+01	2.91E+02	-2.36E+04
		880	265.09	4.76E+03	-2.10E+02	8.76E+01	-4.45E+04	3.99E+04	5.41E+02	-1.51E+02	2.25E+04
15	S/P Slab @ RPV	1376	185.01	1.67E+04	1.11E+03	-1.51E+02	5.08E+01	-2.54E+03	-1.01E+02	-1.03E+02	1.84E+04
		1380	225.00	1.59E+04	1.87E+03	-1.36E+03	1.00E+01	-2.73E+03	-1.05E+02	1.49E+02	1.82E+04
		1384	264.99	1.52E+04	2.11E+03	-3.36E+02	8.30E+01	-2.61E+03	-1.72E+01	-5.43E+01	1.82E+04
16	S/P Slab @ Center	1358	185.18	1.27E+04	4.22E+03	-3.03E+02	-6.49E+03	-1.92E+04	2.13E+01	8.76E+00	-5.45E+03
		1362	225.36	1.18E+04	5.17E+03	-9.18E+02	-6.35E+03	-1.90E+04	-2.03E+02	-2.52E+02	-5.64E+03
		1366	265.00	1.18E+04	6.08E+03	-1.52E+02	-6.50E+03	-1.90E+04	-3.40E+01	-1.93E+01	-5.59E+03
17	S/P Slab @ RCCV	1340	185.18	1.22E+04	5.29E+03	-4.01E+02	1.65E+03	1.75E+04	-1.47E+01	5.43E+01	-1.91E+04
		1344	225.81	1.08E+04	6.80E+03	-2.61E+02	1.94E+03	1.82E+04	-8.68E+01	1.02E+02	-1.93E+04
		1348	265.09	1.18E+04	7.72E+03	-1.21E+02	1.78E+03	1.81E+04	-1.21E+01	-1.23E+01	-1.93E+04

Table 3H.1-6 Results of "Stardyne" Analysis for Unit Drywell Pressure: 6.9 kPa [Pd] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
18	Basemat Outside RCCV	946	185.18	6.50E+02	2.66E+03	8.06E+01	5.52E+04	-2.14E+04	6.09E+03	-2.64E+03	2.36E+02
		982	227.93	3.03E+03	-6.55E+02	6.29E+02	-2.24E+04	4.85E+04	-1.49E+04	-9.37E+02	-9.81E+02
		986	265.09	4.13E+03	-9.18E+02	-5.60E+02	-2.57E+04	4.58E+04	6.49E+03	6.51E+02	-7.36E+01
19	RPV Pedestal Bottom	199	185.01	1.44E+04	-3.75E+04	-3.87E+02	-2.15E+03	-1.08E+04	1.41E+01	1.58E+01	8.81E+03
		204	235.01	1.46E+04	-3.63E+04	-1.44E+02	-2.15E+03	-1.07E+04	-1.83E+01	-1.40E+01	8.79E+03
		208	275.01	1.47E+04	3.61E+04	7.01E+00	-2.14E+03	-1.06E+04	3.69E+00	1.05E+01	8.74E+03
20	RPV Pedestal Center	235	185.01	4.69E+04	-3.63E+04	3.55E+02	8.41E+02	4.22E+03	8.90E+02	-1.58E+01	-8.51E+02
		240	235.01	4.69E+04	-3.64E+04	-6.48E+01	8.05E+02	4.12E+03	-3.11E+00	8.76E+00	-8.83E+02
		244	275.01	4.69E+04	-3.66E+04	-1.33E+02	8.41E+02	4.11E+03	5.34E+01	-7.01E+00	8.98E+02
21	RPV Pedestal Top	289	185.01	3.56E+04	-3.68E+04	-1.63E+02	-6.63E+03	-2.50E+04	2.04E+02	1.21E+03	-3.87E+03
		294	235.01	3.38E+04	-3.71E+04	1.24E+03	-4.67E+03	-2.48E+04	-3.59E+02	-6.36E+02	-3.54E+03
		298	275.01	3.29E+04	-3.73E+04	-7.02E+02	-3.67E+03	-2.50E+04	1.39E+02	1.06E+03	-3.45E+03
22	R/B Outside Wall @ Base	397	183.86	1.23E+05	-5.94E+02	-2.08E+02	-2.53E+02	-1.07E+03	-1.64E+01	1.75E+01	1.45E+02
		404	221.17	-1.91E+02	-1.30E+03	-2.82E+02	9.30E+01	-1.30E+02	-6.98E+01	6.48E+01	8.06E+01
		405	224.57	-1.58E+02	-1.36E+03	7.36E+01	1.05E+02	-1.25E+02	3.37E+01	-7.18E+01	8.05E+01
		410	242.10	1.77E+02	-3.06E+02	3.91E+02	-2.17E+02	-1.01E+03	1.53E+02	-3.50E+01	2.10E+02
		415	267.23	4.03E+02	9.11E+01	-1.40E+01	-3.99E+02	-1.80E+03	2.11E+01	4.77E+01	3.92E+02

Table 3H.1-6 Results of "Stardyne" Analysis for Unit Drywell Pressure: 6.9 kPa [Pd] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
23	R/B Outside Wall @ Center	508	183.86	3.40E+02	7.71E+01	-3.85E+01	3.99E+01	3.51E+01	1.22E+01	2.63E+01	2.57E+02
		511	204.28	1.14E+02	3.33E+01	2.19E+02	-1.42E+01	-1.48E+02	1.15E+02	1.05E+01	1.51E+02
		516	224.57	-1.24E+02	-1.17E+03	-5.04E+02	7.38E+01	-5.96E+01	-2.20E+01	-6.3E+01	-4.2E+01
		521	242.10	2.71E+02	5.08E+01	-6.74E+02	5.78E+01	-2.75E+02	-1.51E+02	8.76E+00	-2.8E+01
24	R/B Outside Wall @ Grade	689	183.86	9.98E+02	-4.73E+02	-9.1E+01	3.52E+01	-3.19E+02	-2.17E+01	-2.1E+01	3.85E+01
		693	209.68	8.35E+02	-3.01E+02	-3.51E+02	-5.07E+01	-8.76E+01	-4.23E+00	7.88E+01	1.93E+01
		696	221.17	1.75E+02	-7.00E+00	-3.08E+02	-4.80E+01	1.07E+01	-2.13E+01	-3.15E+01	-2.45E+01
		701	237.99	7.62E+02	1.23E+02	3.47E+02	1.01E+01	1.10E+02	1.95E+01	-5.25E+00	-8.23E+01
		707	267.23	1.96E+03	6.13E+01	2.66E+02	1.17E+02	1.94E+02	2.53E+01	1.23E+01	-1.02E+02
25	Fuel Pool Wall @ Base	2561	184.31	1.42E+03	7.22E+02	-1.31E+02	2.61E+02	-3.50E+02	-1.18E+02	9.11E+01	-6.83E+01
		2562	192.75	5.36E+02	-6.62E+02	-2.28E+02	-2.63E+01	2.03E+02	-1.87E+02	3.54E+02	-4.31E+02
26	Fuel Pool Slab @ Center	3153	186.10	-3.50E+03	4.94E+03	-1.35E+03	2.05E+03	-3.02E+03	6.23E+02	-7.22E+02	-2.38E+02
		3158	197.05	-7.19E+03	4.27E+03	-5.62E+03	2.99E+03	9.30E+02	1.16E+03	-2.15E+02	-2.52E+03
27	Fuel Pool Girder @ Drywell Head Opening	2742	295.00	6.58E+03	-2.45E+04	-2.24E+04	6.76E+03	4.72E+04	3.76E+03	-4.96E+03	-1.32E+04
		2826	295.00	7.43E+04	3.71E+03	-1.35E+02	1.20E+03	4.94E+03	-7.96E+-3	-1.40E+03	-5.31E+03

Table 3H.1-6 Results of "Stardyne" Analysis for Unit Drywell Pressure: 6.9 kPa [Pd] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
28	Fuel Pool Girder @ RCCV Wall	2733	208.92	1.26E+04	4.98E+04	6.89E+03	1.28E+-3	3.25E+02	-4.94E+02	4.8E+02	6.73E+02
		2856	208.92	-1.40E+04	9.57E+03	1.41E+04	1.35E+03	9.23E+02	1.43E+03	-6.52E+02	-5.75E+02
29	Fuel Pool Girder @ Deep End	2567	199.68	8.10E+02	-4.23E+02	7.14E+03	-1.18E+03	-2.38E+03	8.81E+01	-2.43E+02	5.32E+02
		2655	199.68	9.01E+04	-1.35E+03	4.16E+03	-9.12E+02	-1.36E+03	-5.96E+02	7.0E+00	2.22E+02
		2730	199.68	6.05E+04	-2.03E+03	-2.65E+03	-6.76E+02	-1.01E+03	-1.22E+03	3.18E+02	1.06E+02
		2815	199.68	7.21E+03	-7.12E+02	-4.27E+03	6.23E+02	-4.34E+02	-1.76E+03	1.59E+03	1.06E+02
30	R/B Floor @ El-21.98 ft. @ RCCV	1086	185.18	7.91E+02	-1.73E+03	2.27E+02	-4.49E+02	-1.20E+02	-3.87E+01	-4.51E+02	-1.75E+00
		1109	213.85	9.01E+02	-1.67E+03	1.69E+02	-3.23E+02	-8.41E+01	5.78E+01	-1.19E+02	6.47E+01
		1125	265.09	-1.04E+03	1.23E+03	2.38E+02	-1.14E+02	-4.80E+02	-1.60E+01	-5.25E+01	-4.97E+02
31	R/B Floor @ El-0.67 ft. @ RCCV	1215	185.18	-2.97E+02	3.62E+02	-1.1E+02	-3.60E+02	-8.76E+01	-2.87E+01	-2.99E+02	-1.22E+01
		1239	213.85	-7.7E+02	2.38E+02	7.35E+01	-2.32E+02	5.25E+01	4.71E+01	-8.05E+01	5.25E+01
		1251	265.09	6.47E+02	-2.8E+02	-1.87E+02	-7.47E+01	-3.29E+02	-1.30E+01	-4.02E+01	-3.29E+02
32	R/B Floor @ El-23.95 ft. @ RCCV	1400	354.82	-7.22E+03	7.80E+03	1.45E+03	1.47E+03	-1.56E+02	-1.15E+02	-7.78E+02	2.43E+02
		1426	326.15	-4.58E+03	6.28E+03	-1.43E+03	1.15E+03	2.25E+02	3.53E+01	3.64E+02	4.11E+02
		1442	274.91	7.36E+03	-3.93E+03	9.97E+02	-1.20E+02	1.01E+03	-7.5E+01	-1.03E+02	8.05E+01
		2484	331.09	-4.83E+03	1.75E+04	1.05E+04	2.30E+02	-1.17E+03	-2.08E+02	-4.21E+02	1.27E+03
		2544	331.09	-9.53E+03	2.38E+04	4.67E+03	8.18E+02	1.29E+03	-2.56E+02	-3.85E+02	4.53E+02

Table 3H.1-6 Results of "Stardyne" Analysis for Unit Drywell Pressure: 6.9 kPa [Pd] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
33	Steam Tunnel Floor	1403	354.77	-2.77E+03	2.36E+03	7.73E+02	4.39E+02	-4.01E+02	-1.40E+02	1.13E+02	-9.45E+01
		1417	345.29	-3.51E+03	1.80E+03	2.53E+03	7.61E+02	4.80E+02	-2.73E+02	-2.46E+02	-3.32E+02
		2487	340.93	-2.24E+03	6.26E+02	-4.98E+03	-2.77E+02	-6.14E+02	-1.42E+01	1.20E+02	4.21E+02
		2547	340.93	1.03E+03	1.92E+03	-6.03E+03	-3.89E+01	4.49E+02	6.27E+01	3.85E+01	3.62E+02
34	Steam Tunnel Roof	1673	356.34	3.95E+02	3.27E+03	3.88E+02	1.43E+03	-3.72E+02	-3.88E+02	1.22E+03	-5.25E+00
		1695	337.09	1.23E+03	4.84E+02	6.44E+02	1.19E+01	4.94E+01	3.47E+00	-1.22E+01	1.05E+01
		1745	265.90	7.17E+02	-5.67E+02	-2.45E+02	-5.74E+00	6.81E+00	-8.9E+01	1.4E+01	1.75E+00
		1750	301.04	4.28E+02	5.28E+02	9.13E+02	7.65E+00	1.68E+01	-1.08E+01	8.75E+01	1.23E+01
		1754	319.38	1.17E+02	4.84E+02	2.8E+01	2.4E+01	-3.29E+01	2.09E+00	1.75E+01	1.75E+00

Table 3H.1-7 Results of "Stardyne" Analysis for Unit Wetwall Pressure: 6.9 kPa [Pw]

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
1	RCCV Wetwell Bottom	1	185.18	2.60E+04	2.22E+04	5.25E+01	-2.02E+03	-7.30E+03	3.93E+01	-1.93E+01	5.97E+03
		5	225.80	2.64E+04	2.41E+04	5.18E+02	-2.45E+03	-7.36E+03	-1.04E+02	-1.93E+01	5.92E+03
		9	265.09	2.58E+04	2.34E+04	-4.03E+01	-2.13E+03	-8.19E+03	-3.13E+01	-3.5E+00	6.43E+03
2	RCCV Wetwell Mid–Height	37	185.18	6.41E+04	2.24E+04	3.20E+02	1.52E+02	5.48E+03	-5.18E+00	-2.70E+02	3.87E+03
		41	225.80	6.65E+04	2.31E+04	-1.19E+02	-1.15E+03	5.20E+03	-7.49E+01	1.10E+02	3.91E+03
		45	265.09	6.42E+04	2.46E+04	-4.40E+02	4.71E+02	6.43E+03	-2.38E+03	-2.38E+01	3.89E+03
3	RCCV Wetwell Top	91	185.18	3.12E+04	2.23E+04	-3.68E+01	3.22E+03	-1.25E+04	-5.57E+01	-3.85E+01	-1.12E+04
		95	225.80	3.07E+04	2.41E+04	4.9E+02	-3.65E+03	-1.31E+04	-9.76E+01	-5.6E+01`	-1.13E+04
		99	265.09	2.87E+04	2.48E+04	-1.62E+02	-3.20E+03	-1.42E+04	2.77E+01	3.5E+00	-1.20E+04
4	RCCV Drywell Bottom	109	185.18	8.34E+03	-4.97E+03	-1.57E+02	6.41E+02	4.22E+03	-5.55E+00	-4.03E+01	-4.52E+03
		113	225.80	8.60E+03	-3.23E+03	2.01E+03	8.83E+-2	4.57E+03	-2.95E+02	1.14E+02	-2.98E+03
		117	265.09	8.45E+03	-1.25E+03	4.26E+02	1.08E+03	4.93E+03	-7.01E-01	1.4E+01	-2.84E+03
5	RCCV Drywell Mid–Height	127	185.18	2.33E+03	-4.55E+03	-3.41E+02	-1.41E+03	-6.60E+01	1.54E+01	-8.93E+01	-3.47E+03
		131	225.80	3.09E+03	-2.39E+03	1.80E+03	6.88E+01	-1.71E+03	-3.72E+01	1.80E+-2	-1.45E+03
		135	265.09	4.54E+03	-1.46E+03	3.87E+02	-1.89E+02	-1.32E+02	-6.2E-01	-1.58E+01	-1.65E+03
6	RCCV Drywell Top	163	185.18	9.12E+02	-8.18E+02	-5.13E+01	5.64E+02	4.21E+02	4.21E+00	-1.58E+01	6.53E+02
		167	225.80	4.9E+01	-1.13E+03	-1.19E+02	-1,20E+02	-6.54E+02	2.77E+02	-2.8E+01	5.44E+02
		171	265.09	-2.73E+02	-9.75E+02	2.45E+01	3.51E+02	-1.96E+03	-1.75E+01	7.0E+00	-2.99E+02

Table 3H.1-7 Results of "Stardyne" Analysis for Unit Wetwall Pressure: 6.9 kPa [Pw] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
7	RCCV Top Slab @ RCCV Wall	1616	185.18	1.14E+03	-6.48E+02	-2.87E+02	4.79E+02	8.94E+02	-1.31E+01	-4.9E+01	-3.60E+02
		1620	225.81	6.53E+02	-2.38E+02	-7.18E+01	1.48E+01	-5.07E+01	3.31E+02	2.8E+02	-6.51E+02
		1624	265.09	1.47E+02	6.43E+02	8.06E+01	1.03E+01	-8.22E+01	-4.79E+01	-3.33E+01	2.45E+01
8	RCCV Top Slab @ Center	1634	185.18	8.09E+02	-3.16E+02	-1.73E+02	6.25E+01	-1.56E+02	-6.14E+01	-5.43E+01	-2.77E+02
		1638	225.36	2.17E+02	4.08E+02	-1.19E+02	-5.02E+02	-5.90E+02	2.20E+02	2.29E+02	9.1E+01
		1642	265.00	2.50E+02	4.19E+02	4.72E+01	1.40E+02	-8.50E+01	2.67E+00	1.05E+01	4.2E+01
9	RCCV Top Slab @ Drywell Head Opening	1652	185.00	1.10E+03	6.48E+01	-4.38E+01	-2.16E_02	-2.23E+02	-2.91E+01	2.63E+01	1.49E+02
		1656	225.00	4.39E+02	9.98E+01	-5.78E+01	-1.87E+02	1.51E+02	-7.90E+01	-1.75E+01	-8.93E+01
		1660	265.00	6.88E+02	1.17E+02	3.40E+02	3.21E+02	1.46E+01	1.69E+01	-4.2E+01	-4.73E+01
10	Basemat Cavity @ Center	929	270.00	-3.08E+03	-2.99E+03	-5.25E+00	2.27E+04	2.57E+04	-1.02E+01	3.33E+01	-1.65E+02
11	Basemat Inside RPV Pedestal	926	192.04	-3.08E+03	-3.08E+03	7.88E+01	2.47E+04	2.49E+04	-9.46E+02	-4.92E+02	-4.87E+02
		939	270.00	-3.06E+03	-2.96E+03	02.1E+01	2.37E+04	2.69E+04	-1.83E+01	1.23E+01	-6.97E+02
12	Basemat Outside RPV Pedestal	908	185.01	4.97E+03	1.65E+04	4.03E+01	1.06E+04	6.32E+03	-2.41E+02	3.85E+01	2.15E+04
		912	225.00	5.30E+03	1.63E+04	-7.88E+01	9.94E+03	8.00E+03	01.21E+03	02.97E+01	2.19E+04
		916	264.99	5.10E+03	1.68E+04	-1.51E+02	9.57E+03	9.39E+03	-2.22E+02	-4.55E+01	2.22E+04

Table 3H.1-7 Results of "Stardyne" Analysis for Unit Wetwall Pressure: 6.9 kPa [Pw] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
13	Basemat Between RCCV & RPV Pedestal	890	185.18	7.81E+03	1.37E+04	5.95E+01	2.58E+03	-1.60E+04	-1.60E+02	-2.45E+01	-2.05E+03
		894	225.36	8.30E+03	1.34E+04	7.36E+01	1.41E+03	-1.53E+04	-1.09E+03	-7.88E+01	-1.70E+03
		898	265.00	7.93E+03	1.41E+04	-1.58E+02	7.67E+02	-1.49E+04	-5.27E+01	-8.05E+01	-1.48E+03
14	Basemat Inside RCCV	872	185.18	8.91E+03	1.27E+04	1.40E+01	9.87E+03	2.31E+04	1.29E+02	5.25E+01	-1.73E+04
		876	225.81	9.52E+03	1.21E+04	1.40E+02	8.53E+03	2.31E+04	-8.27E+02	-1.98E+02	-1.71E+04
		880	265.09	9.19E+03	1.32E+04	-2.07E+02	7.90E+04	2.30E+04	8.38E+01	-3.32E+01	-1.70E+04
15	S/P Slab @ RPV	1376	185.01	6.01E+03	1.77E+04	-4.61E+02	-2.62E+03	-5.12E+03	1.57E+02	1.35E+02	-2.24E+04
		1380	225.00	4.90E+03	1.77E+04	-2.46E+03	-2.71E+03	-5.12E+03	7.03E+00	-4.03E+01	-2.22E+04
		1384	264.99	2.86E+03	1.89E+04	-6.29E+02	-2.75E+03	-5.30E+03	2.65E+01	9.46E+01	-2.22E+04
16	S/P Slab @ Center	1358	185.18	9.60E+03	1.39E+04	-1.58E+02	5.66E+03	2.21E+04	-1.97E+01	8.75E-00	2.68E+03
		1362	225.36	8.83E+03	1.45E+04	-1.94E+03	5.64E+03	2.20E+04	1.18E+02	2.05E+02	2.73E+03
		1366	265.00	7.14E+03	1.63E+04	-6.70E+02	5.60E+03	2.22E+04	1.20E+01	1.58E+01	2.66E+03
17	S/P Slab @ RCCV	1340	185.18	9.93E+03	1.26E+04	-8.05E+01	-1.21E+03	-1.37E+04	-1.83E+01	5.25E+00	2.05E+04
		1344	225.81	9.46E+03	1.30E+04	-1.66E+03	-1.17E+03	-1.34E+04	-9.97E+01	1.58E+01	2.03E+04
		1348	265.09	8.42E+03	1.53E+04	-6.03E+02	-1.14E+03	-1.32E+04	-2.64E+01	1.05E+01	2.03E-04

Table 3H.1-7 Results of "Stardyne" Analysis for Unit Wetwall Pressure: 6.9 kPa [Pw] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
18	Basemat Outside RCCV	946	185.18	4.9E+01	6.15E+03	-5.44E+92	2.51E+03	6.13E+03	-8.85E+02	-1.41E+03	-5.29E+02
		982	227.93	5.99E+03	1.12E+02	1.92E+03	4.12E+93	4.07E+02	2.29E+02	-2.35E+02	3.87E+02
		986	265.09	6.58E+03	2.31E+02	-9.81E+02	4.05E+03	-9.03E+02	-4.21E+02	1.52E+02	-9.53E+02
19	RPV Pedestal Bottom	199	185.01	-1.55E+04	4.12E+04	.00	9.79E+01	4.89E+03	-1.90E+00	-1.41E+01	-6.08E+03
		204	235.01	-1.54E+04	4.12E+04	-3.60E+02	1.01E+03	5.03E+03	-5.46E+00	2.1E+01	-6.15E+03
		208	275.01	1.54E+04	4.19E+04	-1.57E+01	1.02E+03	5.16E+03	9.7E-01	-1.05E+01	-6.20E+03
20	RPV Pedestal Center	235	185.01	-4.76E+04	4.19E+04	1.90E+02	-5.20E+02	-2.74E+03	-2.66E-01	1.58E+01	2.29E+02
		240	235.01	-4.76E+04	4.12E+04	-6.79E+02	-5.65E+02	-2.92E+03	-5.06E+00	-5.08E+01	1.87E+02
		244	275.01	-4.76E+04	4.10E+04	1.71E+02	-6.58E+02	-3.03E+03	-4.88E-01	3.15E+01	1.70E+02
21	RPV Pedestal Top	289	185.01	-4.31E+03	4.29E+04	5.81E+02	3.60E+03	2.90E+04	-3.48E+02	-7.40E+02	1.16E+04
		294	235.01	-7.34E+03	4.20E+04	6.31E+02	6.67E+03	2.89E+04	-2.90E+02	1.71E+03	1.18E+04
		298	275.01	-5.14E+03	4.17E+04	2.8E+02	8.50E+03	2.90E+04	4.16E+00	-9.02E+02	1.21E+04
22	R/B Outside Wall @ Base	397	183.86	2.56E+03	1.02E+03	7.7E+01	-2.39E+02	-1.86E+01	3.81E+01	1.92E+01	9.05E+02
		404	221.17	1.21E+02	1.46E+03	6.54E+02	-3.23E+02	-3.91E+02	1.09E+02	1.57E+01	5.77E+01
		405	224.57	1.96E+02	1.37E+03	-1.31E+03	-2.94E+02	-3.49E+02	-1.10E+02	5.25E+00	5.25E+00
		410	242.10	2.22E+03	7.89E+02	-1.73E+03	-3.95E+02	-1.42E+03	-4.19E+02	-1.05E+02	5.28E+02
		415	267.23	4.16E+03	8.38E+02	-1.45E+02	-1.99E+02	-2.52E+03	-5.96E+01	7.88E+01	1.26E+03

Table 3H.1-7 Results of "Stardyne" Analysis for Unit Wetwall Pressure: 6.9 kPa [Pw] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
23	R/B Outside Wall @ Center	508	183.86	5.67E+03	1.12E+03	04.7E+03	8.01E+02	1.99E+03	-1.59E+02	2.1E+02	5.25E+00
		511	204.28	3.34E+03	1.28E+03	-4.02E+01	2.59E+02	1.39E+03	-1.08E+02	-6.1E+01	9.4E+-1
		516	242.57	1.00E+03	3.85E+03	06.05E+02	-9.17E+02	1.20E+02	2.57E+01	4.86E+02	1.75E+02
		521	242.10	3.65E+03	2.01E+02	-5.72E+02	1.45E+02	1.43E+03	1.42E+03	2.13E+02	8.75E+00
24	R/B Outside Wall @ Grade	689	183.86	1.29E+03	1.11E+03	7.7E+01	-6.63E+01	-4.54E+02	-1.94E+01	-1.75E+01	1.06E+02
		693	209.68	4.15E+02	1.27E+03	-8.62E+02	4.21E+01	-8.01E+01	6.27E+00	8.75E+00	-6.3E+01
		696	221.17	-1.4E+02	2.74E+02	-3.85E+02	1.98E+01	-2.96E+01	6.27E+00	3.15E+01	1.22E+01
		701	237.99	4.32E+02	3.60E+02	5.89E+02	2.98E+01	-6.81E+01	6.68E+00	1.22E+01	-1.75E+02
		707	267.23	1.18E+03	1.4E+02	3.69E+02	4.25E+01	1.06E+02	-1.65E+01	5.42E+01	-1.12E+02
25	Fuel Pool Wall @ Base	2561	184.31	1.05E+03	1.83E+02	-1.19E+02	6.99E+02	1.55E+03	1.89E+01	2.62E+01	-5.40E+02
		2562	192.75	6.09E+02	2.59E+02	-2.06E+02	9.48E+01	1.15E+03	-2.87E+01	-3.60E+02	-4.86E+02
26	Fuel Pool Slab @ Center	3153	186.10	-2.20E+03	-6.12E+01	-7.33E+02	1.38E+03	1.3E+03	-1.02E+02	7.68E+02	3.74E+02
		3158	197.05	-3.76E+03	-1.96E+02	1.94E+03	1.25E+03	9.08E+01	-6.90E+02	3.65E+02	7.14E+02
27	Fuel Pool Girder @ Drywell Head Opening	2742	295.00	2.59E+03	1.05E+02	–5.80E+02	–2.27E+02	–1.50E+02	–1.75E+02	–3.85E–01	2.28E+03
		2826	295.00	5.66E+03	0.89E+02	–3.31E+02	–2.56E+02	–7.48E+01	–1.85E+02	9.81E+01	–1.33E+02

Table 3H.1-7 Results of "Stardyne" Analysis for Unit Wetwall Pressure: 6.9 kPa [Pw] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
28	Fuel Pool Girder @ RCCV Wall	2733	208.92	3.29E+03	–2.63E+03	1.96E+02	–4.23E+01	1.09E+01	5.03E+01	–2.10E+01	7.00E+00
		2856	208.92	7.39E+03	–4.75E+02	1.05E+03	–1.30E+02	–2.18E+01	1.81E+02	–1.28E+02	–8.40E+01
29	Fuel Pool Girder @ Deep End	2567	199.68	–2.05E+03	–4.33E+02	3.45E+03	–5.12E+01	3.87E+02	–1.58E+02	4.55E+01	–1.14E+02
		2655	199.68	–2.73E+02	–4.15E+02	3.90E+03	–8.01E+01	9.75E+01	–3.56E+01	1.80E+02	–5.95E+01
		2730	199.68	1.72E+03	–9.46E+01	3.64E+03	–4.76E+01	–1.14E+02	–1.68E+01	1.10E+02	–1.23E+01
		2815	199.68	4.50E+03	2.63E+01	1.75E+03	–3.73E+01	–8.59E+01	–1.19E+02	1.91E+02	3.68E+01
30	R/B Floor @ El–21.98 ft. @ RCCV	1086	185.18	–1.30E+04	1.01E+04	–3.55E+03	–1.05E+02	–6.72E+00	–1.51E+01	–6.65E+01	2.63E+01
		1109	213.85	–1.45E+04	1.16E+04	2.71E+03	–1.27E+02	4.45E–01	2.08E+01	–2.45E+01	2.10E+01
		1125	265.09	1.05E+04	–1.36E+04	–2.59E+03	–1.35E+01	–1.88E+02	–7.5E+00	–1.58E+01	–1.45E+02
31	R/B Floor @ El–0.67 ft. @ RCCV	1215	185.18	–1.01E+04	9.81E+03	–2.68E+03	4.05E+02	9.52E+01	3.06E+01	2.99E+02	1.23E+01
		1239	213.85	–1.10E+04	1.00E+04	2.36E+03	2.72E+02	5.61E+01	–4.04E+01	6.83E+01	–5.43E+01
		1251	265.09	8.97E+03	–1.10E+04	–2.17E+03	1.05E+02	5.21E+02	1.73E+01	6.48E+01	5.45E+02
32	R/B Floor @ El–23.95 ft. @ RCCV	1400	354.82	–4.80E+03	6.09E+03	1.03E+03	1.94E+03	7.38E+02	–1.91E+02	–1.05E+03	6.12E+01
		1426	326.15	–2.40E+03	3.79E+03	–3.43E+03	7.65E+02	3.07E+02	7.92E+01	–2.28E+02	–2.18E+02
		1442	274.91	5.34E+03	–1.40E+03	1.08E+03	5.87E+02	2.64E+03	2.71E+00	–5.95E+01	2.45E+02
		2484	331.09	–3.73E+03	–3.39E+03	–7.17E+03	1.37E+02	5.38E+02	–1.02E+02	–1.57E+01	–4.09E+02
		2544	331.09	–5.88E+03	–2.34E+03	–6.75E+03	–4.00E+01	–1.26E+01	–9.35E+01	2.46E+02	1.92E+01

Table 3H.1-7 Results of "Stardyne" Analysis for Unit Wetwall Pressure: 6.9 kPa [Pw] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
33	Steam Tunnel Floor	1403	354.77	–8.74E+02	3.13E+03	6.46E+02	–6.72E+02	–5.70E+02	8.94E+01	–3.53E+02	–1.38E+02
		1417	345.29	–4.92E+02	2.15E+03	2.29E+03	2.57E+01	2.10E+02	1.14E+02	7.0E+01	–5.54E+02
		2487	340.93	–3.83E+02	7.22E+02	–5.04E+03	–2.69E+02	–4.54E+02	1.23E+01	–3.5E+01	3.16E+02
		2547	340.93	–2.81E+02	1.15E+03	–5.53E+03	–7.43E+01	2.61E+02	–5.70E+01	–4.55E+01	2.24E+02
34	Steam Tunnel Roof	1673	356.34	8.03E+02	7.68E+02	–1.47E+02	–2.56E+02	8.10E+01	6.94E+01	–1.56E+02	1.05E+01
		1695	337.09	1.59E+02	3.60E+02	2.53E+02	9.35E–01	1.99E+01	1.42E+00	–3.5E+00	2.1E+01
		1745	265.90	–6.12E+01	1.45E+02	1.22E+02	1.38E+01	–1.76E+01	1.46E+01	4.9E+01	–7.0E+00
		1750	301.04	4.72E+01	1.97E+02	2.27E+01	–5.83E+00	–5.47E+00	5.07E+00	1.75E+00	–7.0E+00
		1754	319.38	5.95E+01	1.26E+02	9.8E+01	1.60E+01	2.42E+01	3.87E+00	1.05E+01	1.75E+00

Table 3H.1-8 Results of "Stardyne" Analysis for Safe Shutdown Earthquake (SSE) SRSS of Three Components [Ess]

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
1	RCCV Wetwell Bottom	1	185.18	3.08E+06	1.00E+07	5.28E+06	4.05E+05	2.15E+06	1.83E+05	5.22E+04	9.50E+05
		5	225.80	3.50E+06	1.14E+07	6.44E+06	4.22E+05	2.28E+08	2.37E+05	9.16E+04	1.02E+06
		9	265.09	3.36E+06	1.04E+07	7.18E+06	–3.56E+05	2.25E+08	2.85E+05	5.90E+04	1.03E+06
2	RCCV Wetwell Mid–Height	37	185.18	4.88E+05	8.04E+06	6.27E+06	8.60E+04	3.63E+05	1.97E+05	3.26E+04	5.56E+04
		41	225.80	3.00E+05	8.33E+06	6.62E+06	1.49E+05	6.47E+05	2.47E+05	7.35E+04	1.85E+05
		45	265.09	4.92E+05	7.94E+06	6.89E+06	3.49E+05	7.28E+05	2.18E+05	1.78E+05	2.02E+05
3	RCCV Wetwell Top	91	185.18	8.50E+05	6.53E+06	5.90E+06	1.39E+05	4.36E+05	1.89E+05	3.69E+04	2.61E+05
		95	225.80	1.07E+06	5.97E+06	6.66E+06	3.19E+04	5.94E+05	2.31E+05	5.72E+04	3.15E+05
		99	265.09	7.73E+05	5.23E+06	6.43E+06	2.26E+05	9.00E+05	1.43E+05	2.45E+04	3.81E+05
4	RCCV Drywell Bottom	109	185.18	1.83E+06	5.88E+06	5.37E+06	1.25E+05	5.11E+05	2.28E+05	6.11E+04	1.31E+06
		113	225.80	1.70E+06	4.91E+06	6.21E+06	3.94E+05	7.93E+05	3.20E+05	8.88E+04	3.29E+05
		117	265.09	4.53E+05	4.08E+06	5.26E+06	2.19E+05	1.12E+06	1.18E+05	2.41E+04	4.17E+05
5	RCCV Drywell Mid–Height	127	185.18	2.80E+06	5.83E+06	5.36E+06	5.77E+05	3.63E+06	2.13E+05	9.45E+04	1.73E+06
		131	225.80	2.69E+06	4.27E+06	5.97E+06	4.30E+05	2.27E+05	2.84E+05	9.21E+04	3.61E+05
		135	265.09	7.82E+05	3.10E+06	4.85E+06	5.74E+04	1.25E+05	1.76E+05	9.01E+04	4.88E+05

Table 3H.1-8 Results of "Stardyne" Analysis for Safe Shutdown Earthquake (SSE) SRSS of Three Components [Ess] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
6	RCCV Drywell Top	163	185.18	2.19E+06	2.21E+05	3.47E+06	3.45E+05	8.37E+05	1.14E+05	7.86E+04	7.70E+05
		167	225.80	2.65E+06	2.46E+06	4.04E+06	4.71E+05	2.21E+06	1.19E+05	7.47E+04	8.47E+05
		171	265.09	2.56E+06	1.48E+06	3.87E+06	4.48E+05	2.33E+06	1.90E+05	8.18E+04	9.77E+05
7	RCCV Top Slab @ RCCV Wall	1616	185.18	2.53E+06	1.05E+06	1.32E+06	2.57E+05	8.01E+05	7.49E+04	2.03E+05	2.10E+05
		1620	225.81	2.71E+06	8.96E+05	1.44E+06	8.22E+05	7.06E+05	5.58E+05	7.31E+05	2.09E+06
		1624	265.09	2.71E+06	3.22E+05	1.36E+06	2.67E+05	1.87E+06	1.97E+05	7.67E+04	1.03E+06
8	RCCV Top Slab @ Center	1634	185.18	1.66E+06	6.86E+05	1.33E+06	2.80E+05	3.49E+05	2.99E+05	2.17E+05	1.88E+05
		1638	225.36	2.15E+06	6.21E+05	1.76E+06	1.88E+06	2.56E+06	5.70E+06	3.69E+05	7.13E+05
		1642	265.00	2.02E+06	4.02E+05	1.24E+06	5.58E+05	1.86E+06	1.01E+05	1.76E+05	1.06E+06
9	RCCV Top Slab @ Drywell Head Opening	1652	185.00	5.76E+05	1.82E+05	6.09E+05	3.28E+05	1.74E+05	4.04E+05	5.21E+04	1.36E+05
		1656	225.00	9.00E+05	1.07E+06	6.48E+05	2.67E+05	3.00E+05	2.20E+05	2.61E+05	2.22E+05
		1660	265.00	1.97E+06	1.83E+05	2.93E+05	5.22E+05	6.01E+05	1.61E+05	1.03E+05	2.44E+05
10	Basemat Cavity @ Center	929	270.00	3.55E+05	2.00E+05	3.48E+05	8.24E+06	8.25E+06	4.38E+05	2.63E+06	2.66E+06

Table 3H.1-8 Results of "Stardyne" Analysis for Safe Shutdown Earthquake (SSE) SRSS of Three Components [Ess] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
11	Basemat Inside RPV Pedestal	926	192.04	1.84E+06	1.39E+06	7.83E+05	1.48E+07	1.50E+07	1.24E+06	1.68E+06	3.62E+06
		939	270.00	1.45E+06	2.04E+05	1.38E+06	1.13E+07	1.23E+07	8.28E+04	9.95E+05	2.82E+06
12	Basemat Outside RPV Pedestal	908	185.01	1.83E+06	1.27E+06	1.50E+06	7.82E+06	1.37E+07	4.53E+06	1.60E+06	3.66E+06
		912	225.00	2.76E+06	3.16E+05	2.82E+05	7.26E+06	1.42E+07	4.50E+06	3.29E+05	3.13E+06
		916	264.99	2.20E+06	1.40E+06	1.11E+04	7.53E+06	1.31E+07	5.03E+06	2.02E+05	2.61E+06
13	Basemat Between RCCV & RPV Pedestal	890	185.18	2.48E+06	8.54E+05	2.69E+06	5.94E+06	5.55E+06	3.09E+06	7.54E+05	1.87E+06
		894	225.36	3.77E+06	4.99E+05	1.06E+06	5.73E+06	7.52E+06	2.52E+06	7.33E+05	1.65E+06
		898	265.00	3.08E+06	9.73E+05	2.40E+06	6.35E+06	7.46E+06	2.45E+06	5.88E+05	1.33E+06
14	Basemat Inside RCCV	872	185.18	3.30E+06	6.13E+05	3.72E+06	5.06E+06	4.06E+06	2.59E+06	6.80E+05	2.08E+06
		876	225.81	4.80E+06	1.05E+06	1.58E+06	4.37E+06	8.54E+06	2.42E+06	6.50E+05	2.19E+06
		880	265.09	4.21E+06	6.36E+05	3.35E+06	6.26E+06	7.92E+06	1.69E+06	3.80E+05	2.25E+06
15	S/P Slab @ RPV	1376	185.01	5.61E+05	8.14E+05	1.40E+05	4.10E+05	1.11E+06	8.70E+04	7.92E+04	5.45E+05
		1380	225.00	7.10E+05	2.96E+05	5.09E+05	3.39E+04	9.22E+05	1.06E+05	8.74E+04	4.3E+05
		1384	264.99	4.97E+05	4.08E+05	2.17E+05	2.81E+05	7.60E+05	1.19E+05	1.08E+05	3.44E+05

Table 3H.1-8 Results of "Stardyne" Analysis for Safe Shutdown Earthquake (SSE) SRSS of Three Components [Ess] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
16	S/P Slab @ Center	1358	185.18	6.08E+05	9.73E+05	1.7E+05	1.43E+05	2.57E+05	8.71E+04	6.60E+03	3.47E+05
		1362	225.36	8.60E+05	2.33E+05	3.60E+05	1.44E+05	1.76E+05	1.07E+05	8.50E+03	2.76E+05
		1366	265.00	5.38E+05	5.22E+05	5.03E+05	1.31E+05	1.14E+05	1.29E+05	1.04E+04	2.16E+05
17	S/P Slab @ RCCV	1340	185.18	8.32E+05	1.02E+06	2.15E+05	2.03E+05	1.00E+06	1.14E+04	3.10E+04	2.68E+05
		1344	225.81	9.84E+05	2.17E+05	3.20E+05	1.46E+05	7.46E+05	4.08E+04	6.26E+04	1.99E+05
		1348	265.09	6.54E+05	5.91E+05	6.32E+05	1.16E+05	5.65E+05	4.01E+04	2.97E+04	1.62E+05
18	Basemat Outside RCCV	946	185.18	2.19E+06	3.55E+06	8.79E+05	3.89E+06	5.48E+06	3.02E+06	7.00E+06	1.19E+06
		982	227.93	4.95E+06	1.29E+06	3.15E+06	3.54E+06	6.53E+06	2.98E+06	1.00E+06	7.57E+06
		986	265.09	4.95E+06	2.08E+06	1.65E+06	6.72E+06	9.36E+06	1.88E+06	7.71E+05	6.41E+06
19	RPV Pedestal Bottom	199	185.01	1.73E+06	8.90E+06	1.95E+06	3.58E+05	1.77E+06	2.56E+04	3.77E+04	7.82E+05
		204	235.01	1.72E+06	8.34E+06	2.45E+06	3.16E+05	1.64E+06	1.61E+04	5.86E+04	7.27E+05
		208	275.01	1.58E+06	7.95E+06	2.53E+06	3.20E+05	1.55E+06	2.50E+04	4.39E+04	6.87E+05
20	RPV Pedestal Center	235	185.01	5.04E+05	5.25E+06	2.83E+06	5.88E+04	1.14E+05	1.01E+04	2.49E+04	6.38E+04
		240	235.01	4.57E+05	4.90E+06	3.09E+06	7.12E+04	1.08E+05	8.52E+03	2.61E+04	5.51E+04
		244	275.01	4.48E+05	4.75E+06	3.23E+06	7.00E+04	1.08E+05	8.84E+03	3.13E+04	6.01E+04

Table 3H.1-8 Results of "Stardyne" Analysis for Safe Shutdown Earthquake (SSE) SRSS of Three Components [Ess] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
21	RPV Pedestal Top	289	185.01	9.64E+05	7.01E+05	1.58E+06	7.06 E+05	1.35E+06	4.54E+04	4.63E+04	6.16E+05
		294	235.01	1.12E+06	7.42E+05	1.83E+06	8.16E+05	1.06E+06	8.56E+04	1.44E+05	3.95E+05
		298	275.01	1.03E+06	7.80E+05	1.84E+06	3.57E+05	8.94E+05	4.29E+04	9.83E+04	3.82E+05
22	R/B Outside Wall @ Base	397	183.86	2.07E+06	5.05E+06	4.36E+06	1.27E+05	5.22E+05	4.37E+04	6.53E+03	1.22E+05
		404	221.17	4.36E+05	3.47E+06	2.52E+06	2.00E+05	5.18E+05	4.63E+04	1.00E+05	1.46E+05
		405	224.57	5.27E+05	3.43E+06	2.68E+06	1.66E+05	5.02E+05	3.92E+04	1.30E+05	1.70E+05
		410	242.10	2.47E+06	5.77E+06	4.18E+06	2.17E+05	7.38E+05	3.87E+04	3.33E+04	1.99E+05
		415	267.23	2.52E+06	3.36E+06	4.77E+06	1.99E+05	9.78E+05	5.71E+04	3.51E+04	3.11E+05
23	R/B Outside Wall @ Center	508	183.86	2.02E+05	4.81E+06	4.71E+06	2.56E+04	1.71E+04	1.85E+04	2.13E+04	3.84E+04
		511	204.28	1.86E+05	4.88E+06	4.50E+06	3.10E+04	8.87E+04	3.95E+04	5.18E+03	2.72E+04
		516	224.57	2.79E+05	3.68E+06	3.82E+06	1.00E+04	4.95E+04	9.65E+04	9.93E+04	6.52E+04
		521	242.10	5.00E+05	3.49E+06	4.45E+06	6.10E+04	1.19E+05	5.35E+04	3.09E+04	4.25E+04
		526	267.23	1.15E+06	6.22E+05	–1.01E+05	-2.98E+05	-6.45E+05	1.68E+04	4.62E+04	–8.95E+03
24	R/B Outside Wall @ Grade	689	183.86	3.90E+05	2.99E+06	2.34E+06	3.56E+04	1.00E+05	2.24E+04	1.22E+04	8.52E+04
		693	209.68	4.44E+05	3.50E+06	2.84E+06	8.70E+04	1.09E+05	1.83E+04	5.18E+04	1.75E+05
		696	221.17	1.11E+05	2.90E+06	2.64E+06	2.47E+04	4.05E+04	2.93E+04	2.14E+04	4.29E+04
		701	237.99	6.77E+05	1.70E+06	2.46E+06	2.14E+04	4.05E+04	1.18E+04	2.36E+04	7.02E+04
		707	267.23	8.60E+05	1.25E+06	2.60E+06	3.00E+04	1.57E+05	2.00E+04	3.38E+04	1.40E+05

Table 3H.1-8 Results of "Stardyne" Analysis for Safe Shutdown Earthquake (SSE) SRSS of Three Components [Ess] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
25	Fuel Pool Wall @ Base	2561	184.31	1.52E+06	3.68E+06	1.75E+06	6.50E+05	2.08E+06	1.58E+05	3.74E+04	6.00E+05
		2562	192.75	6.79E+05	1.85E+06	2.22E+06	2.99E+05	1.20E+06	2.02E+05	3.58E+05	3.92E+05
26	Fuel Pool Slab @ Center	3153	186.10	1.32E+06	1.77E+06	5.03E+05	4.31E+06	2.69E+06	4.58E+05	1.59E+06	7.35E+05
		3158	197.05	2.40E+06	1.64E+06	5.47E+05	1.86E+06	5.34E+05	1.21E+06	1.06E+06	1.56E+06
27	Fuel Pool Girder @ Drywell Head Opening	2742	295.00	6.71E+05	1.09E+06	1.26E+06	7.86E+05	3.79E+06	3.16E+05	1.20E+05	3.88E+05
		2826	295.00	2.32E+06	5.27E+05	1.18E+06	6.68E+05	2.29E+06	1.17E+06	8.12E+05	9.53E+05
28	Fuel Pool Girder @ RCCV Wall	2733	208.92	9.89E+05	2.87E+06	2.50E+06	3.20E+05	8.84E+05	9.58E+04	3.74E+05	2.81E+05
		2856	208.92	2.00E+06	1.96E+06	2.55E+06	9.10E+05	9.08E+05	5.16E+05	4.44E+05	2.45E+05
29	Fuel Pool Girder @ Deep End	2567	199.68	1.53E+06	3.14E+06	1.46E+06	4.51E+05	1.53E+06	2.15E+05	1.06E+05	5.30E+05
		2655	199.68	8.05E+05	2.09E+06	1.80E+06	3.10E+05	3.86E+05	3.30E+05	1.63E+05	2.42E+05
		2730	199.68	1.15E+06	9.24E+05	2.26E+06	4.07E+05	2.49E+05	4.91E+05	2.02E+05	9.06E+04
		2815	199.68	1.95E+06	2.41E+05	1.58E+06	6.31E+05	1.94E+05	7.60E+05	6.92E+05	2.08E+04
30	R/B Floor @ El–21.98 ft. @ RCCV	1086	185.18	7.13E+05	3.70E+05	1.28E+06	4.32E+05	9.67E+04	2.95E+04	5.30E+05	5.85E+04
		1109	213.85	7.48E+05	6.18E+05	4.84E+05	2.47E+05	5.44E+04	4.39E+04	1.13E+05	4.76E+04
		1125	265.09	2.32E+05	1.14E+06	2.25E+05	7.58E+04	3.50E+05	1.52E+04	4.24E+04	3.87E+05

Table 3H.1-8 Results of "Stardyne" Analysis for Safe Shutdown Earthquake (SSE) SRSS of Three Components [Ess] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
31	R/B Floor @ El–0.67 ft. @ RCCV	1215	185.18	1.22E+05	3.66E+05	5.19E+04	1.77E+05	2.11E+05	1.92E+04	1.70E+05	1.35E+04
		1239	213.85	8.67E+04	3.19E+05	1.19E+05	1.35E+05	2.95E+04	2.22E+04	5.92E+04	3.10E+04
		1251	265.09	1.91E+05	2.87E+05	1.13E+05	4.53E+04	2.00E+04	9.03E+03	2.82E+04	2.33E+05
32	R/B Floor @ El–23.95 ft. @ RCCV	1400	354.82	1.65E+06	1.26E+06	3.83E+05	1.22E+06	3.51E+05	1.88E+05	7.35E+05	1.20E+05
		1426	326.15	3.28E+05	1.93E+06	1.41E+06	7.39E+05	2.48E+05	4.83E+04	2.79E+05	1.30E+05
		1442	274.91	6.82E+05	6.04E+05	8.92E+04	3.39E+05	1.60E+06	3.91E+04	6.46E+04	1.20E+06
		2484	331.09	1.86E+06	1.53E+06	2.96E+06	1.00E+05	4.40E+05	3.60E+04	1.60E+05	3.26E+05
		2544	331.09	4.84E+05	1.97E+06	3.02E+06	5.24E+04	1.71E+05	3.31E+04	8.58E+04	7.69E+04
33	Steam Tunnel Floor	1403	354.77	1.33E+06	1.01E+06	4.85E+05	2.90E+05	3.94E+05	7.44E+04	7.02E+04	9.22E+04
		1417	345.29	1.66E+06	8.05E+05	7.01E+05	3.66E+05	4.25E+05	1.01E+05	8.96E+04	2.92E+05
		2487	340.93	1.00E+06	5.48E+05	1.72E+06	1.52E+05	5.79E+05	2.99E+04	6.68E+04	1.52E+05
		2547	340.93	2.92E+05	1.71E+06	1.87E+06	9.90E+04	2.33E+05	7.00E+04	4.41E+04	1.20E+05
34	Steam Tunnel Roof	1673	356.34	2.17E+05	6.56E+05	6.71E+05	3.81E+05	1.11E+05	7.99E+04	1.22E+05	5.02E+04
		1695	337.09	1.68E+05	5.30E+05	2.05E+05	7.97E+03	3.97E+04	1.00E+04	3.72E+03	7.18E+04
		1745	265.90	4.60E+05	3.01E+05	3.45E+05	1.82E+04	7.93E+04	1.60E+04	5.90E+04	2.90E+04
		1750	301.04	3.44E+05	3.60E+05	5.61E+05	3.37E+04	5.76E+04	3.70E+03	1.44E+04	6.61E+04
		1754	319.38	2.09E+05	2.52E+05	3.12E+05	3.93E+04	3.74E+04	6.00E+03	1.80E+04	5.51E+04

Table 3H.1-9 Results of "Stardyne" Analysis for Thermal Loads (6 Hours) [T_{A-II}]

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
1	RCCV Wetwell Bottom	1	185.18	–3.33E+06	–3.06E+06	–7.87E+03	5.46E+06	5.53E+06	3.44E+03	–9.53E+03	2.68E+05
		5	225.80	–3.06E+06	–2.37E+06	–1.63E+05	5.45E+06	5.55E+06	–1.31E+04	–1.24E+04	2.77E+05
		9	265.09	–3.15E+06	–2.93E+06	–7.31E+04	5.50E+06	5.42E+06	1.70E+02	3.19E+03	3.29E+05
2	RCCV Wetwell Mid-Height	37	185.18	–3.77E+06	–2.61E+06	–3.23E+04	5.40E+06	4.80E+06	9.11E+03	–3.42E+04	2.38E+05
		41	225.80	–3.52E+06	–2.55E+06	–3.14E+05	5.25E+06	4.83E+06	–1.26E+04	1.82E+04	2.29E+05
		45	265.09	–3.87E+06	–2.73E+06	–1.14E+05	5.55E+06	4.97E+06	–2.13E+04	1.71E+04	2.73E+05
3	RCCV Wetwell Top	91	185.18	4.00E+06	–2.19E+06	–6.95E+04	5.79E+06	7.41E+06	–3.50E+03	–5.50E+04	1.06E+06
		95	225.80	3.87E+06	–2.68E+06	9.60E+03	5.41E+06	6.95E+06	–4.40E+04	–4.85E+04	8.90E+05
		99	265.09	2.40E+06	–2.77E+06	3.03E+04	5.78E+06	6.62E+06	–3.40E+01	–8.58E+02	6.37E+05
4	RCCV Drywell Bottom	109	185.18	1.07E+05	–2.46E+06	–3.33E+04	9.03E+06	1.03E+07	–1.26E+04	–3.75E+04	–2.17E+06
		113	225.80	1.91E+05	3.21E+06	8.47E+05	8.75E+06	1.04E+07	–1.19E+05	1.62E+04	–1.72E+06
		117	265.09	–1.53E+05	–2.39E+06	2.62E+05	9.32E+06	1.06E+07	–1.15E+04	1.74E+04	–1.32E+06
5	RCCV Drywell Mid-Height	127	185.18	–1.50E+06	–2.26E+06	–2.57E+05	7.89E+06	4.44E+06	4.46E+04	7.37E+04	–2.33E+06
		131	225.80	–1.44E+06	–3.53E+06	1.43E+06	7.90E+06	5.68E+06	–7.43E+04	–2.26E+03	–1.84E+06
		135	265.09	–3.54E+06	–2.14E+06	3.64E+05	8.63E+06	6.75E+06	4.75E+04	5.74E+04	–1.68E+06

Table 3H.1-9 Results of "Stardyne" Analysis for Thermal Loads (6 Hours) [T_{A-II}] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
6	RCCV Wetwell Top	163	185.18	–2.04E+06	–1.27E+06	1.15E+05	7.05E+06	10.50E+06	2.63E+04	6.62E+04	2.22E+06
		167	225.80	–1.34E+07	–1.36E+06	2.09E+06	6.54E+06	9.57E+06	3.43E+05	–4.43E+05	4.47E+05
		171	265.09	–1.46E+07	–1.54E+06	1.43E+05	6.31E+06	7.81E+06	5.23E+04	–3.70E+04	–9.60E+05
7	RCCV Top Slab @ RCCV Wall	1616	185.18	–7.55E+05	–2.49E+06	–1.41E+04	8.19E+06	11.33E+06	6.56E+04	5.60E+04	–1.51E+06
		1620	225.81	–7.70E+05	–2.64E+06	–5.97E+04	8.83E+06	1022E+06	1.66E+05	3.10E+05	–1.16E+06
		1624	265.09	–3.82E+06	–1.59E+06	–1.48E+05	9.30E+06	9.65E+06	–1.63E+04	–6.25E+04	1.02E+05
8	RCCV Top Slab @ Center	1634	185.18	–6.01E+06	–3.38E+06	5.63E+05	6.41E+06	5.29E+06	–1.31E+05	–1.73E+05	–1.75E+06
		1638	225.36	–8.09E+06	–1.32E+07	–3.19E+06	5.39E+06	6.58E+06	–2.51E+05	7.79E+05	–9.58E+05
		1642	265.00	–4.06E+06	–2.50E+06	7.07E+05	9.89E+06	9.80E+06	1.27E+05	–4.59E+04	1.40E+05
9	RCCV Top Slab @ Drywell Head Opening	1652	185.18	–1.01E+07	–1.47E+06	1.06E+05	5.52E+06	4.46E+05	–3.11E+04	4.71E+04	2.80E+05
		1656	225.00	–3.66E+06	–11.81E+06	–3.00E+05	4.54E+06	5.00E+06	4.78E+05	1.50E+06	2.03E+06
		1660	265.00	–1.83E+07	–5.53E+06	1.48E+06	9.36E+06	2.78E+06	–7.84E+05	7.55E+06	–4.38E+05
10	Basemat Cavity @ Center	929	270.00	–8.30E+06	–7.89E+06	3.68E+03	–4.03E+07	–3.87E+07	–8.25E+03	1.21E+04	7.55E+02
11	Basemat Inside RPV Pedestal	926	192.04	–8.12E+06	–8.19E+06	–6.65E+04	–3.98E+07	–3.98E+07	–7.20E+05	–2.80E+05	2.26E+04
		939	270.00	–8.23E+06	–7.83E+06	1.51E+04	–4.01E+07	–3.87E+07	–2.61E+04	3.71E+03	3.15E+04

Table 3H.1-9 Results of "Stardyne" Analysis for Thermal Loads (6 Hours) [T_{A-II}] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
12	Basemat Outside RPV Pedestal	908	185.01	9.63E+04	–1.19E+07	2.18E+04	–2.00E+07	–3.52E+07	–1.80E+05	9.44E+04	–1.04E+06
		912	225.00	3.85E+05	–1.19E+07	–1.54E+05	–2.04E+07	–3.41E+07	–6.28E+05	–2.61E+04	–8.39E+05
		916	264.99	1.84E+05	–1.13E+07	–6.19E+04	2.05E+07	–3.36E+07	–8.49E+04	–1.67E+04	–6.32E+05
13	Basemat Between RCCV & RPV Pedestal	890	185.18	–3.16E+06	–8.75E+06	4.63E+04	–2.27E+07	–2.94E+07	–1.62E+05	–3.12E+04	–6.85E+05
		894	225.36	–2.64E+06	–9.00E+06	–7.11E+04	–2.37E+07	–2.86E+07	–5.31E+05	–3.31E+04	–5.52E+05
		898	265.00	–2.80E+06	–8.19E+06	–7.53E+04	–2.36E+07	–2.90E+07	2.46E+04	–3.82E+04	–4.12E+05
14	Basemat Inside RCCV	872	185.18	–4.52E+06	–7.49E+06	1.51E+05	–2.33E+07	–2.68E+07	9.12E+03	2.56E+04	–5.18E+05
		876	225.81	–3.68E+06	–7.95E+06	1.91E+05	–2.44E+07	–2.62E+07	–3.82E+05	–2.19E+05	–3.68E+05
		880	265.09	–3.70E+06	–6.90E+06	–8.58E+04	–2.42E+07	–2.71E+07	1.62E+05	–9.16E+03	–2.84E+05
15	S/P Slab @ RPV	1376	185.01	–8.58E+06	–9.60E+06	–1.53E+05	–1.38E+06	–1.56E+05	–1.56E+04	8.79E+04	8.20E+05
		1380	225.00	–8.81E+06	–10.18E+06	–1.11E+06	–1.39E+06	–2.40E+05	–1.90E+04	2.33E+04	8.20E+05
		1384	264.99	–1.00E+06	–9.02E+06	–3.37E+04	–1.48E+07	–3.15E+05	–1.26E+03	–8.30E+04	7.16E+05
16	S/P Slab @ Center	1358	185.18	–8.98E+06	–9.52E+06	5.69E+04	–1.72E+06	–2.42E+06	2.50E+03	–1.69E+03	5.94E+05
		1362	225.36	–8.82E+06	–10.09E+06	–9.20E+05	–1.73E+06	–2.40E+06	–2.11E+04	–1.93E+03	5.56E+05
		1366	265.00	–1.02E+06	–8.49E+06	–4.75E+05	–1.75E+06	–2.33E+06	–1.17E+04	–1.68E+04	5.29E+05
17	S/P Slab @ RCCV	1340	185.18	–8.98E+06	–9.26E+06	1.26E+05	–2.28E+06	–3.71E+06	–3.34E+02	6.44E+03	4.62E+05
		1344	225.81	–9.01E+06	–10.34E+06	–7.71E+05	–2.27E+06	–3.66E+06	–4.71E+04	2.77E+04	4.48E+05
		1348	265.09	–9.95E+06	–8.17E+06	–4.21E+05	–2.26E+06	–3.47E+06	–1.73E+04	1.82E+02	4.06E+05

Table 3H.1-9 Results of "Stardyne" Analysis for Thermal Loads (6 Hours) [T_{A-II}] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
18	Basemat Outside RCCV	946	185.18	–6.74E+06	1.20E+07	–1.30E+06	–1.76E+07	–1.97E+06	–1.38E+06	–3.15E+06	–7.58E+05
		982	227.93	1.05E+07	–5.95E+06	4.10E+06	–4.50E+06	–1.59E+07	2.25E+06	4.45E+05	–2.31E+06
		986	265.09	1.26E+07	–6.04E+06	–1.77E+06	–3.07E+06	–1.90E+07	–9.83E+05	–2.12E+05	–3.05E+06
19	RPV Pedestal Bottom	199	185.11	–9.00E+06	–1.23E+06	–3.59E+04	4.94E+06	4.49E+06	–1.91E+03	1.52E+02	1.57E+06
		204	235.01	–8.98E+06	–1.06E+06	–1.30E+05	4.98E+06	4.58E+06	–3.41E+03	8.48E+03	1.531E+06
		208	275.01	–8.93E+06	–8.88E+05	–5.24E+04	4.98E+06	4.63E+06	4.97E+02	3.57E+02	1.51E+06
20	RPV Pedestal Center	235	185.01	2.98E+06	–6.76E+05	–2.38E+05	4.94E+06	4.54E+06	0.153E+02	1.91E+04	–2.36E+05
		240	235.01	2.96E+06	–1.01E+06	1.70E+04	4.94E+06	4.45E+06	–3.09E+03	–3.57E+04	–2.54E+05
		244	275.01	2.96E+06	–1.14E+06	–2.45E+05	4.94E+06	4.41E+06	1.08E+03	2.40E+04	–2.63E+05
21	RPV Pedestal Top	289	185.01	–7.79E+06	–1.28E+06	–4.10E+05	9.79E+06	7.03E+06	–1.11E+05	–9.61E+05	–5.60E+06
		294	235.01	–8.91E+06	–1.56E+06	1.09E+06	11.04E+06	7.16E+06	–8.68E+04	1.44E+06	–5.50E+06
		298	275.01	–9.95E+06	–1.77E+06	–6.43E+05	12.02E+06	7.16E+06	–6.00E+04	–1.04E+06	–5.32E+06
22	R/B Outside Wall @ Base	397	183.86	2.01E+06	8.60E+05	–1.48E+05	–4.76E+05	–8.41E+05	–3.10E+03	6.41E+03	8.61E+04
		404	221.17	1.98E+05	–1.93E+05	–2.45E+05	–4.43E+05	–5.56E+05	–2.62E+04	1.63E+04	2.82E+04
		405	224.57	2.84E+05	–2.28E+05	–2.87E+05	–4.18E+05	–5.00E+05	5.70E+03	–6.25E+03	5.43E+03
		410	242.10	1.84E+06	6.51E+05	–1.43E+05	–5.21E+05	–7.79E+05	–1.81E+03	–8.84E+03	8.42E+04
		415	267.23	3.29E+06	1.09E+06	9.79E+04	–4.49E+05	–9.34E+05	–6.68E+03	1.48E+04	1.30E+05

Table 3H.1-9 Results of "Stardyne" Analysis for Thermal Loads (6 Hours) [T_{A-II}] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
23	R/B Outside Wall @ Center	508	183.86	2.56E+05	9.60E+05	–7.14E+03	–3.63E+05	–6.90E+05	1.91E+04	–1.31E+04	6.29E+03
		511	204.28	–1.10E+05	6.69E+05	1.91E+05	–4.36E+05	–7.61E+05	3.72E+04	7.79E+03	–1.33E+04
		516	224.57	–6.36E+04	1.05E+06	–1.15E+06	–4.49E+05	–4.33E+05	–5.61E+04	–3.82E+04	–5.02E+04
		521	242.10	4.15E+05	3.43E+05	–9.35E+05	–4.72E+05	–7.65E+05	–8.28E+04	1.98E+04	–7.78E+04
24	R/B Outside Wall @ Grade	689	183.86	6.30E+06	2.82	–2.82	1.31	1.46	4.72	2.75	–6.25
		693	209.68	4.62E+06	2.14	–6.00	1.05E+06	1.36E+06	1.42E+04	9.75E+04	1.45E+04
		696	221.17	1.84E+06	3.15E+06	–7.99E+05	1.02E+06	1.35E+06	4.23E+04	–3.64E+05	–2.52E+05
		701	237.99	5.41E+06	1.09E+05	–7.27E+04	1.36E+05	1.32E+05	2.81E+03	–2.85E+04	–1.26E+04
		707	267.23	6.80E+06	1.12E+06	3.42E+05	1.38E+06	1.36E+06	1.76E+04	3.43E+04	–4.19E+04
25	Fuel Pool Wall @ Base	2561	184.31	–2.59E+06	–1.30E+06	1.37E+05	3.03E+06	4.41E+06	–6.10E+04	–6.02E+04	–7.44E+05
		2562	192.75	–2.42E+06	–1.86E+06	1.00E+06	2.23E+06	3.77E+06	–6.14E+04	–4.64E+05	–6.74E+05
26	Fuel Pool Slab @ Center	3153	186.10	–3.56E+06	5.29E+06	–1.15E+06	–5.07E+06	–5.12E+06	–2.14E+05	8.13E+02	1.28E+05
		3158	197.05	–5.88E+05	4.45E+06	–3.05E+06	–4.81E+06	–3.60E+06	3.62E+04	–2.28E+05	–6.69E+05
27	Fuel Pool Girder @ Drywell Head Opening	2742	295.00	2.68E+06	1.93E+06	–1.15	7.03E+06	3.75E+06	–4.49E+05	1.30E+06	–7.00E+04
		2826	295.00	3.34E+06	9.53E+05	9.25E+05	3.95E+06	2.24E+06	–7.21E+04	–5.27E+04	–1.82E+06

Table 3H.1-9 Results of "Stardyne" Analysis for Thermal Loads (6 Hours) [T_{A-II}] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
28	Fuel Pool Girder @ RCCV Wall	2733	208.92	–3.94E+06	–2.59E+06	5.59E+06	3.72E+06	3.24E+06	1.40E+05	–2.22E+05	–5.25E+05
		2856	208.92	9.60E+06	2.50E+06	1.09E+06	9.75E+06	–2.60E+05	2.60E+05	–3.73E+05	–5.92E+05
29	Fuel Pool Girder @ Deep End	2567	199.68	–2.08E+06	5.99E+05	2.49E+06	2.84E+06	3.02E+06	–4.98E+05	–3.29E+05	5.17E+05
		2655	199.68	–3.06E+06	6.41E+05	2.77E+06	3.07E+06	3.90E+06	–5.12E+05	3.36E+05	1.22E+05
		2730	199.68	–1.43E+06	1.20E+06	2.35E+06	3.23E+06	3.11E+06	3.67	2.22E+05	–2.61E+05
		2815	199.68	5.71E+06	6.13E+05	4.99E+05	3.55E+06	9.61E+05	–8.14E+04	2.59E+05	–3.66E+05
30	R/B Floor @ El– 21.98 ft. @ RCCV	1086	185.18	–1.38E+06	1.27E+06	–3.26E+05	8.41E+04	–1.11E+04	1.54E+03	1.77E+05	3.40E+04
		1109	213.85	–1.43E+06	1.32E+06	3.73E+05	–7.21E+03	–2.60E+04	–6.59E+03	1.64E+04	–5.53E+03
		1125	265.09	1.26E+06	–1.39E+06	–3.49E+05	–2.51E+04	1.23E+04	1.27E+03	1.00E+04	6.58E+04
31	R/B Floor @ El–0.67 ft. @ RCCV	1215	185.18	–9.68E+05	1.74E+05	–3.57E+05	–1.33E+04	2.57E+03	1.20E+03	3.17E+04	1.65E+03
		1239	213.85	–1.14E+06	1.84E+06	2.38E+05	–8.10E+03	2.46E+03	2.13E+03	4.75E+03	3.40E+03
		1251	265.09	1.55E+06	–9.95E+05	–4.40E+05	6.90E+03	2.06E+04	–4.42E+02	5.18E+03	4.94E+04
32	R/B Floor @ El– 23.95 ft. @ RCCV	1400	354.82	–6.95E+06	7.99E+06	1.60E+06	–4.81E+05	–1.85E+05	2.66E+04	–2.61E+05	5.27E+04
		1426	326.15	–5.62E+06	6.65E+06	–1.71E+06	–1.35E+05	–1.72E+04	–1.09E+05	3.27E+05	2.14E+05
		1442	274.91	6.70E+06	–4.85E+06	9.70E+05	5.61E+04	5.79E+05	8.01E+03	–1.56E+04	6.11E+05
		2484	331.09	–2.64E+06	3.71E+06	–1.59E+06	5.56E+05	1.96E+05	–8.37E+04	–4.27E+05	–1.26E+04
		2544	331.09	–2.17E+06	5.06E+06	–2.33E+06	5.34E+05	2.85E+05	–8.46E+04	1.66E+04	3.68E+04

Table 3H.1-9 Results of "Stardyne" Analysis for Thermal Loads (6 Hours) [T_{A-II}] (Continued)

Section	Location	Element #	Azimuth	F_x (N_{hh}) N/m	F_y(N_{mm}) N/m	F_{xy}(N_{hm}) N/m	M_x(M_{hh}) N·m/m	M_y(M_{mm}) N·m/m	M_{xy}(M_{mh}) N·m/m	Q_x(N_{rh}) N/m	Q_y(N_{rm}) N/m
33	Steam Tunnel Floor	1403	354.77	–1.84E+06	2.82E+06	7.78E+05	–1.19E+05	–2.58E+05	–4.90E+04	1.56E+05	–5.17E+04
		1417	345.29	–1.37E+06	1.45E+06	2.07E+06	–8.90E+04	–1.45E+04	–1.18E+05	–9.75E+04	–3.15E+04
		2487	340.93	–9.35E+05	–1.34E+05	–1.96E+06	–1.86E+05	–4.39E+03	–3.06E+04	–1.86E+04	1.11E+05
		2547	340.93	–2.35E+05	–7.27E+05	–2.29E+06	–1.10E+05	3.55E+05	2.68E+04	–1.27E+05	1.03E+05
34	Steam Tunnel Roof	1673	356.34	–1.39E+06	5.48E+05	4.75E+05	–8.94E+05	4.90E+04	1.66E+05	–4.75E+05	3.75E+04
		1695	337.09	–5.22E+05	–3.42E+05	1.03E+06	–4.67E+02	2.53E+04	4.85E+03	3.06E+03	2.61E+04
		1745	265.90	1.29E+05	–8.32E+05	2.47E+04	–8.05E+03	–1.01E+05	1.46E+04	5.22E+04	–4.78E+04
		1750	301.04	–5.17E+05	–3.68E+05	1.77E+06	–2.52E+04	–7.83E+04	6.10E+03	4.24E+03	–3.36E+04
		1754	319.38	–8.14E+05	–9.51E+05	2.29E+06	–1.26E+04	–1.61E+04	3.38E+03	–2.19E+03	–5.83E+03

Table 3H.1-10 Load Combination 1

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 1								
EL # 5 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 5 1-8	8.35E+05	–4.16E+06	–5.91E+04	–1.47E+05	–3.27E+05	–1.76E+03	2.39E+03	4.31E+05
EL # 5 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 2								
EL # 41 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 41 1-8	3.78E+06	–3.31E+06	1.24E+05	–4.73E+04	5.28E+05	–1.08E+04	–2.27E+03	1.03E+05
EL # 41 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 3								
EL # 95 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 95 1-8	1.80E+06	–2.45E+06	4.93E+05	–1.28E+05	–4.30E+05	–7.64E+04	–3.73E+04	–5.32E+05
EL # 95 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 4								
EL # 113 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 113 1-8	1.07E+06	–1.89E+06	9.18E+05	6.73E+04	4.92E+04	–1.33E+05	5.15E+04	2.60E+05
EL # 113 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 5								
EL # 131 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 131 1-8	8.74E+05	–1.41E+06	8.15E+05	1.97E+05	1.65E+05	–3.53E+04	5.19E+04	–2.9E+05
EL # 131 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 6								
EL # 167 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 167 1-8	7.16E+05	–1.83E+05	2.5E+05	–2.02E+05	–9.51E+05	8.25E+04	8.21E+02	–6.94E+05
EL # 167 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-10 Load Combination 1 (Continued)

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 7								
EL # 1620 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1620 1-8	7.58E05	1.37E+06	4.9E+03	7.76E+04	–5.8E+05	4.74E+04	1.43E+05	2.0E+05
EL # 1620 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 8								
EL # 1638 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1638 1-8	7.04E+05	1.14E+06	–2.68E+05	2.22E+05	2.38E+05	2.13E+05	1.75E+05	7.0E+04
EL # 1638 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 9								
EL # 1656 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1656 1-8	6.11E+05	5.17E+05	9.73E05	–2.21E+05	3.48E+05	4.67E+05	–1.6E+05	7.47E+05
EL # 1656 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 10								
EL # 922 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 929 1-8	5.44E+05	5.85E+05	–3.25E+03	–1.01E+06	–3.6E+05	–8.94E+03	4.67E+04	9.69E+04
EL # 929 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 11								
EL # 926 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 926 1-8	5.75E+05	5.66E+05	2.33E+03	–1.53E+06	–1.81E+06	–3.97E+05	1.59E+05	4.0E+05
EL # 926 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 12								
EL # 912 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 912 1-8	5.18E+05	5.07E+05	–3.96E+04	1.12E+04	–2.37E+05	–3.1E+05	1.6E+04	–1.22E+06
EL # 912 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 13								
EL # 894 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 894 1-8	5.36E+05	4.83E+05	–2.36E+04	1.2E+06	1.79E+06	–2.78E+05	4.43E+04	–3.19E+05
EL # 894 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-10 Load Combination 1 (Continued)

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 14								
EL # 876 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 876 1-8	5.03E+05	4.79E+05	–1.66E+04	2.19E+06	1.48E+06	–2.96E+05	6.25E+04	3.58E+05
EL # 876 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 15								
EL # 1380 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1380 1-8	1.27E+06	1.02E+06	–6.14E+05	1.15E+05	1.36E+05	6.22E+03	7.93E+02	1.41E+05
EL # 1380 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 16								
EL # 1362 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1362 1-8	1.2E+06	1.1E+06	–4.99E+05	6.4E+04	–4.8E+04	5.97E+03	–2.56E+03	–8.27E+03
EL # 1362 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 17								
EL # 1344 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1344 1-8	1.19E+06	1.15E+06	–4.28E+05	6.86E+03	–1.06E+5	–3.99E+04	5.52E+04	6.78E+04
EL # 1344 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 18								
EL # 982 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 982 1-8	2.29E+05	–4.96E+05	8.02E+04	3.26E+06	7.07E+05	–7.71E+04	3.46E+05	–2.69E+06
EL # 982 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 22								
EL # 415 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 415 1-8	–1.96E+05	–2.34E+06	–2.07E+05	8.82E+04	3.36E05	–9.5E+03	3.44E+03	–7.03E+03
EL # 415 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 23								
EL # 516 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 516 1-8	–2.82E+04	–1.05E+06	2.99E+05	–2.76E+04	3.74E+04	–1.49E+04	–6.76E+02	1.34E+04
EL # 516 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-10 Load Combination 1 (Continued)

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 24								
EL # 707 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 707 1-8	2.64E+05	–7.48E+05	–6.24E+03	1.03E+04	–1.93E+03	5.26E+03	–3.96E+02	5.18E+04
EL # 707 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 25								
EL # 2561 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 2561	1.27E+06	2.03E+06	3.81E+05	–1.23E+05	–8.57E+05	–2.68E+04	1.54E+04	4.15E+05
EL # 2561	.00	.00	.00	.00	.00	.00	.00	.00
Section 26								
EL # 3158 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 3158 1-8	–4.91E+04	7.26E+05	–9.12E+05	–5.92E+05	3.11E+05	–7.45E+05	7.66E+05	–8.81E+05
EL # 3158 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 27								
EL # 2826 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 2826 1-8	4.21E+06	–4.93E+05	–3.16E+05	5.19E+04	–5.04E+05	–2.07E+05	–3.29E+04	–1.68E+05
EL # 2826 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 28								
EL # 2733 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 2733 1-8	1.47E+06	–4.96E+05	1.66E+06	5.49E+04	–6.22E+04	3.94E+04	–8.97E+04	8.76E+04
EL # 2733 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 29								
EL # 2567 TA	.00	.00	.00	.00	.00	.00	.00	.00
El # 2567 1-8	5.58E+05	1.66E+06	1.38E+06	–2.13E+05	–8.58E+05	–8.46E+04	–3.8E+04	4.48E+05
EL # 2567 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 30								
EL # 1109 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1109 1-8	–6.85E+05	1.02E+06	3.78E+04	4.64E+04	1.59E+04	1.45E+04	8.27E+04	–1.18E+04
EL # 1109 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-10 Load Combination 1 (Continued)

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 31								
EL # 1239 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1239 1-8	–5.55E+05	7.45E+05	1.8E+05	6.20E+04	1.48E+04	3.21E+03	6.31E+04	–1.67E+04
EL # 1239 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 32								
EL # 1442 TA	.00	.00	.00	.00	.00	.00	.00	.00
El # 1442 1-8	6.22E+05	–2.70E+05	1.44E+05	2.86E+05	1.24E+06	–3.1E+04	–4.43E+04	9.93E+05
EL # 1442 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 33								
EL # 2487 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 2487 1-8	–1.09E+05	9.91E+04	–1.03E+06	1.37E+04	8.51E+04	1.77E+04	4.15E+03	–1.59E+04
EL # 2487 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 34								
EL # 1750 TA	.00	.00	.00	.00	.00	.00	.00	.00
EL # 1750 1-8	2.07E+05	9.30E+03	4.96E+04	–3.55E+04	–4.14E+04	1.07E+03	9.84E+03	5.11E+04
EL # 1750 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-11 Load Combination 8

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 1								
EL # 5 TA	–1.41E+06	–1.83E+06	–1.34E+05	2.79E+06	3.09E+06	–1.06E+04	–7.43E+03	6.22E+04
EL # 5 1-8	1.50E+06	–3.39E+06	–5.36E+04	–2.99E+05	–1.05E+06	–3.2E+03	2.25E+03	9.93E+05
EL # 5 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 2								
EL # 41 TA	–2.52E+06	–1.94E+06	–2.48E+05	2.63E+06	2.4E+06	–4.92E03	1.04E+04	2.7E+05
EL # 41 1-8	4.46E+06	–2.58E+06	3.74E+04	–6.37E+04	5.07E+05	–5.63E+03	2.96E+03	5.62E+03
EL # 41 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 3								
EL # 95 TA	1.57E+06	–1.96E+06	–6.03E+04	2.97E+05	3.56E+06	–1.48E+04	–2.36E+04	3.34E+05
EL # 95 1-8	2.17E+06	–1.73E+06	2.89E+05	–1.01E+05	–2.16E+5	–5.63E+04	–3.08E+04	–4.03E+05
EL # 95 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 4								
EL # 113 TA	–6.48E+05	–2.11E+06	7.92E+05	4.09E+06	5.05E+06	–3.11E+04	7.02E+03	–9.28E+05
EL # 113 1-8	1.62E+06	–8.16E+05	5.62E+05	–3.03E+04	–5.35E04	–9.81E+04	3.51E+04	5.15E+05
EL # 113 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 5								
EL # 131 TA	–1.42E+06	–1.8E+06	9.61E+05	3.58E+06	2.44E06	6.67E+04	–6.02E+04	–1.08E+06
EL # 131 1-8	1.53E+06	–4.2E+05	4.63E+05	1.62E+05	3.38E+05	–1.83E+04	3.12E+04	–2.43E+05
EL # 131 SE	00	.00	.00	.00	.00	.00	.00	.00
Section 6								
EL # 167 TA	–3.33E+4	–8.23E+05	9.81E+05	5.64E+06	7.01E+06	3.17E+05	–2.92E+05	9.89E+05
EL # 167 1-8	9.52E+05	5.56E+05	1.09E+05	–3.6E+05	–1.58E+06	6.4E+04	1.56E+02	–1.02E+06
EL # 167 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 7								
EL # 1620 TA	–2.24E+06	–4.11E+06	–4.83E+05	6.08E+06	8.50E+06	–1.06E+05	5.88E+05	–9.39E+05
EL # 1620 1-8	7.28E+05	1.9E+06	–8.76E+04	2.00E+05	–8.71E+05	–5.77E+04	6.17E+03	7.93E+05
EL # 1620 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-11 Load Combination 8 (Continued)

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 8								
EL # 1638 TA	–7.08E+06	–1.29E+07	–2.75E+06	9.74E+05	2.79E+06	–5.93E+05	4.40E+05	–1.04E+06
EL # 1638 1-8	7.38E+05	1.31E+06	–4.42E+05	7.33E+05	8.55E+05	1.91E+05	1.40E+05	1.28E+05
EL # 1638 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 9								
EL # 1656 TA	–3.88E+06	–1.06E+07	–4.16E+05	2.57E+05	1.18E+06	–3.28E+05	3.15E+05	7.50E+05
EL # 1656 1-8	6.07E+05	5.16E+05	1.29E+06	–1.84E+05	5.48E+05	6.92E+05	–2.03E+05	1.02E+06
EL # 1656 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 10								
EL # 929 TA	–1.09E+07	–1.06E+07	3.78E+03	–3.21E+07	–3.08E+07	–6.79E+03	1.48E+04	–9.32E+02
EL # 929 1-8	1.13E+06	9.99E+05	–2.73E+03	–1.10E+07	–1.01E+07	–7.71E+03	4.35E+04	9.38E+04
EL # 929 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 11								
EL # 926 TA	–1.08E+07	–1.08E+07	–6.03E+04	–3.17E+07	–3.17E+07	–5.01E+05	–2.64E+05	1.47E+04
EL # 926 1-8	1.08E+06	1.09E+06	7.63E+04	–1.13E+07	–1.17E+07	–5.40E+05	2.54E+05	3.85E+05
EL # 926 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 12								
EL # 912 TA	–1.17E+05	–1.08E+07	–1.31E+05	–1.47E+07	–2.64E+07	–4.82E+03	–1.27E+04	–5.46E+05
EL # 912 1-8	1.23E+06	1.68E+06	3.47E+04	–5.81E+06	–7.94E+06	–3.45E+05	2.15E+03	–2.00E+06
EL # 912 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 13								
EL # 894 TA	–2.70E+06	–8.30E+06	–7.71E+04	–1.76E+07	–2.22E+07	4.40E+05	–6.44E+03	–3.54E+05
EL # 894 1-8	1.37E+06	1.53E+06	3.88E+04	–4.09E+06	–2.25E+06	–2.78E+05	–5.91E+03	–1.64E+06
EL # 894 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 14								
EL # 876 TA	–3.72E+06	–7.45E+06	1.05E+05	–1.83E+07	–2.05E+07	–3.63E+05	–1.69E+05	–2.22E+05
EL # 876 1-8	1.39E+06	1.48E+06	4.03E+04	–1.48E+06	1.94E+06	–2.38E+05	5.71E+04	–1.38E+06
EL # 876 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-11 Load Combination 8 (Continued)

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 15								
EL # 1380 TA	–5.03E+06	–6.30E+06	–7.29E+05	–4.46E+05	4.25E+05	–1.58E+04	1.70E+04	5.29E+05
EL # 1380 1-8	1.53E+06	1.14E+06	–4.63E+05	3.16E+04	–9.79E+04	4.55E+03	1.42E+03	3.27E+05
EL # 1380 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 16								
EL # 1362 TA	–5.15E+06	–6.21E+06	–6.26E+05	–6.31E+05	–1.02E+06	–1.67E+04	–1.48E+03	3.63E+05
EL # 1362 1-8	1.42E+06	1.24E+06	–3.75E+05	–7.58E+04	–2.95E+05	3.17E+03	–7.21E+03	–1.54E+05
EL # 1362 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 17								
EL # 1344 TA	–5.32E+06	–6.31E+06	–5.50E+05	–9.77E+05	–1.84E+06	–2.94E+04	1.29E+04	2.91E+05
EL # 1344 1-8	1.40E+06	1.30E+06	–3.16E+05	5.26E+04	3.36E+05	–3.26E+04	4.82E+04	–2.67E+05
EL # 1344 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 18								
EL # 982 TA	9.54E+06	–5.48E+06	3.40E+06	–2.66E+06	–1.33E+07	1.86E+06	3.89E+05	–1.73E+06
EL # 982 1-8	7.96E+05	–5.13E+05	2.71E+05	2.60E+05	4.25E+05	–6.12E+05	4.70E+05	–3.72E+06
EL # 982 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 22								
EL # 415 TA	2.82E+06	8.14E+05	7.29E+04	–4.25E+05	–7.34E+05	–3.64E+03	1.02E+04	7.78E+04
EL # 415 1-8	–1.13E+05	–2.34E+06	–2.25E+05	3.45E+04	4.63E+04	–7.67E+03	1.27E+04	9.58E+04
EL # 415 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 23								
EL # 516 TA	–9.23E+04	9.95E+05	–9.33E+05	–4.22E+05	–4.13E+05	–4.81E+04	–4.29E+04	–4.55E+04
EL # 516 1-8	–4.22E+04	–1.26E+06	3.57E+05	–2.14E+04	3.85E+04	–1.20E+04	–5.14E+03	1.19E+04
EL # 516 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 24								
EL # 707 TA	6.34E+06	1.11E+06	2.66E+05	1.35E+06	1.34E+06	1.80E+04	1.77E+04	–2.26E+04
EL # 707 1-8	2.93E+05	–7.56E+05	–2.92E+04	6.90E+03	–1.93E+04	7.79E+03	–4.72E+03	6.36E+04
EL # 707 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-11 Load Combination 8 (Continued)

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 25								
EL # 2561 TA	–2.45E+06	–1.45E+06	–2.78E+04	4.11E+06	5.21E+06	–1.04E+05	–8.46E+04	–7.22E+05
EL # 2561 1-8	1.21E+06	2.03E+06	3.91E+05	–1.95E+05	–1.03E+06	–2.99E+04	1.55E+04	4.73E+05
EL # 2561 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 26								
EL # 3158 TA	–9.02E+05	2.05E+06	–1.15E+06	–5.21E+06	–4.13E+06	–1.49E+05	–4.29E+04	–4.50E+05
EL # 3158 1-8	2.42E+05	9.40E+05	–8.74E+05	–6.76E+05	2.92E+05	–6.49E+05	7.15E+05	–9.86E+05
EL # 3158 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 27								
EL # 2826 TA	1.67E+06	–5.24E+04	3.75E+04	5.07E+06	3.00E+06	–5.16E+03	–4.41E+05	–1.68E+06
EL # 2826 1-8	4.61E+06	–5.63E+05	–5.06E+05	8.91E+04	–4.22E+05	–3.03E+05	–6.78E+04	–2.30E+05
EL # 2826 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 28								
EL # 2733 TA	–6.06E+06	–1.34E+06	3.77E+06	5.21E+06	4.29E+06	1.68E+04	3.73E+04	–6.95E+05
EL # 2733 1-8	1.34E+06	5.95E+05	1.50E+06	7.42E+06	–7.08E+04	2.28E+04	–7.93E+04	1.00E+05
EL # 2733 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 29								
EL # 2567 TA	1.28E+05	6.58E+05	2.05E+06	4.54E+06	4.16E+06	–4.49E+05	–3.47E+05	3.84E+05
EL # 2567 1-8	7.74E+05	1.69E+06	1.12E+06	–2.39E+05	–9.33E+05	–6.06E+04	–4.54E+04	4.71E+05
EL # 2567 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 30								
EL # 1109 TA	–1.10E+06	1.08E+06	2.63E+05	8.90E+03	–2.58E+04	–6.85E+03	1.43E+04	–1.11E+04
EL # 1109 1-8	–1.06E+06	1.11E+06	1.40E+05	1.61E+04	6.78E+03	1.93E+04	6.78E+04	–6.15E+03
EL # 1109 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 31								
EL # 1239 TA	–7.62E+05	1.32E+06	1.87E+05	7.21E+03	5.20E+03	–1.13E+03	8.67E+03	–4.38E+02
EL # 1239 1-8	–5.82E+05	7.68E+05	1.70E+05	4.25E+04	9.57E+03	6.26E+03	5.58E+04	–1.23E+04
EL # 1239 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-11 Load Combination 8 (Continued)

	F_x (N_{hh}) N/m	F_y (N_{mm}) N/m	F_{xy} (N_{hm}) N/m	M_x (M_{hh}) N•m/m	M_y (M_{mm}) N•m/m	M_{xy} (N_{hm}) N•m/m	Q_{xz} (N_{hr}) N/m	Q_{yz} (N_{mr}) N/m
Section 32								
EL # 1442 TA	4.20E+06	–3.27E+06	6.37E+05	4.23E+04	5.16E+05	8.27E+03	–1.03E+04	5.67E+05
EL # 1442 1-8	8.09E+05	–2.99E+05	1.47E+05	2.35E+05	1.07E+06	–2.95E+04	–3.91E+04	8.57E+05
EL # 1442 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 33								
EL # 2487 TA	–4.15E+05	–2.03E+05	–1.14E+06	–1.21E+05	7.43E+04	2.62E+04	–3.75E+04	1.37E+04
EL # 2487 1-8	–6.06E+04	7.66E+04	–6.99E+05	1.30E+04	9.09E+04	1.24E+04	1.01E+04	–1.76E+04
EL # 2487 SE	.00	.00	.00	.00	.00	.00	.00	.00
Section 34								
EL # 1750 TA	–5.55E+05	–4.59E+05	1.57E+06	–2.15E+04	–6.36E+04	4.85E+03	2.85E+03	–2.85E+04
EL # 1750 1-8	2.28E+05	2.02E+03	6.00E+04	–3.46E+04	–4.13E+04	2.87E+02	9.55E+03	5.21E+04
EL # 1750 SE	.00	.00	.00	.00	.00	.00	.00	.00

Table 3H.1-12 Load Combination 15

	F_x (Nhh) N/m	F_y (Nmm) N/m	F_{xy} (Nhm) N/m	M_x (Mhh) N-m/m	M_y (Mmm) N-m/m	M_{xy} (Mhm) N-m/m	Q_{xz} (Nhr) N/m	Q_{yz} (Nmr) N/m
SECTION 1								
EL # 5 TA	–1.41E+06	–1.83E+06	–1.34E+05	2.78E+06	3.09E+06	–1.06E+04	–7.43E+03	6.22E+04
EL # 5 1-8	8.30E+05	–4.43E+06	–3.92E+04	–1.82E+05	–5.43E+05	–1.78E+03	1.93E+03	6.71E+05
EL # 5 SE	3.50E+06	1.14E+07	6.44E+06	4.21E+05	2.28E+06	2.37E+05	9.16E+04	1.02E+06
SECTION 2								
EL # 41 TA	–2.52E+06	–1.94E+06	–2.48E+05	2.63E+06	2.40E+06	–4.92E+03	1.04E+04	2.70E+05
EL # 41 1-8	3.10E+06	–3.59E+06	1.12E+05	–3.90E+04	4.37E+05	–7.45E+03	–9.37E+02	–1.70E+04
EL # 41 SE	2.99E+05	8.33E+06	6.62E+06	1.49E+05	6.47E+05	2.47E+05	7.35E+04	1.85E+05
SECTION 3								
EL # 95 TA	1.57E+06	–1.96E+06	–6.03E+04	2.97E+06	3.56E+06	–1.48E+04	–2.36E+04	3.34E+05
EL # 95 1-8	1.22E+06	–2.77E+06	4.06E+05	–7.81E+04	–2.46E+05	–6.02E+04	–2.99E+04	–3.35E+05
EL # 95 SE	1.07E+06	5.98E+06	6.66E+06	3.19E+04	5.94E+05	2.31E+05	5.72E+04	3.15E+05
SECTION 4								
EL # 113 TA	–6.48E+05	–2.11E+06	7.92E+05	4.09E+06	5.05E+06	–3.11E+04	7.02E+03	–9.28E+05
EL # 113 1-8	8.59E+05	–1.94E+06	7.50E+05	3.02E+04	–4.48E+04	–1.03E+05	4.00E+04	2.73E+05
EL # 113 SE	1.70E+06	4.91E+06	6.22E+06	3.94E+05	7.92E+05	3.19E+05	8.89E+04	3.29E+05
SECTION 5								
EL # 131 TA	–1.42E+06	–1.80E+06	9.61E+05	3.58E+06	2.44E+06	6.67E+04	–6.02E+04	–1.08E+06
EL # 131 1-8	8.29E+05	–1.53E+06	6.76E+05	1.49E+05	1.60E+05	–2.78E+04	3.67E+04	–2.23E+05
EL # 131 SE	2.69E+06	4.27E+06	5.97E+06	4.30E+05	2.27E+05	2.81E+05	9.21E+04	3.61E+05
SECTION 6								
EL # 167 TA	–3.37E+06	–8.23E+05	9.81E+05	5.64E+06	7.00E+06	3.17E+05	–2.92E+05	9.89E+05
EL # 167 1-8	6.83E+05	–3.63E+05	3.23E+05	–1.50E+05	–6.67E+05	5.38E+04	6.17E+03	–5.74E+05
EL # 167 SE	2.65E+06	2.46E+06	4.04E+06	4.71E+05	2.21E+06	1.19E+05	7.47E+04	8.48E+05
SECTION 7								
EL # 1620 TA	–2.24E+06	–4.11E+06	–4.83E+05	6.08E+06	8.49E+06	–1.06E+05	5.88E+05	–9.39E+05
EL # 1620 1-8	6.94E+05	1.16E+06	1.06E+05	–6.49E+03	–4.78E+05	5.61E+04	1.69E+05	2.40E+04
EL # 1620 SE	2.71E+06	8.96E+05	1.44E+06	8.22E+05	7.06E+05	5.58E+05	7.31E+05	2.09E+06
SECTION 8								
EL # 1638 TA	–7.08E+06	–1.29E+07	–2.75E+06	9.74E+05	2.79E+06	–5.93E+05	4.40E+05	–1.04E+06
EL # 1638 1-8	6.66E+05	1.06E+06	–1.18E+05	4.28E+04	5.59E+03	1.99E+05	1.43E+05	2.67E+04
EL # 1638 SE	2.15E+05	6.21E+05	1.76E+06	1.88E+06	2.56E+06	5.70E+05	3.69E+05	7.13E+05
SECTION 9								
EL # 1656 TA	–3.88E+06	–1.06E+07	–4.16E+05	2.57E+05	1.18E+06	–3.28E+05	3.15E+05	7.50E+05
EL # 1656 1-8	5.31E+05	5.47E+05	8.19E+05	–2.17E+05	2.37E+05	3.77E+05	–1.47E+05	6.33E+05
EL # 1656 SE	8.90E+05	1.07E+06	6.48E+05	2.67E+05	3.00E+05	2.20E+05	2.61E+05	2.22E+05
SECTION 10								
EL # 929 TA	–1.09E+07	–1.06E+07	3.78E+03	–3.21E+07	–3.08E+07	–6.79E+03	1.48E+04	–9.32E+02
EL # 929 1-8	6.55E+05	6.09E+05	–2.78E+03	–4.17E+06	–3.54E+06	–7.85E+03	4.38E+04	1.64E+05
EL # 929 SE	3.55E+05	2.00E+05	3.48E+05	8.23E+06	8.24E+06	4.38E+05	2.63E+06	2.66E+06

Table 3H.1-12 Load Combination 15 (Continued)

	F_x (Nhh) N/m	F_y (Nmm) N/m	F_{xy} (Nhm) N/m	M_x (Mhh) N-m/m	M_y (Mmm) N-m/m	M_{xy} (Mhm) N-m/m	Q_{xz} (Nhr) N/m	Q_{yz} (Nmr) N/m
SECTION 11								
EL # 926 TA	–1.08E+07	–1.08E+07	–6.03E+04	–3.17E+07	–3.17E+07	–5.01E+05	–2.64E+05	1.47E+04
EL # 926 1-8	6.46E+05	6.46E+05	3.54E+04	–5.09E+06	–5.52E+06	–5.52E+05	2.75E+05	6.34E+05
EL # 926 SE	1.84E+06	1.39E+06	7.83E+05	1.47E+07	1.50E+07	1.24E+06	1.68E+06	3.62E+06
SECTION 12								
EL # 912 TA	–1.17E+05	–1.08E+07	–1.31E+05	–1.47E+07	–2.64E+07	–4.82E+05	–1.27E+04	–5.46E+05
EL # 912 1-8	7.73E+05	1.06E+06	–2.53E+03	–2.17E+06	–3.36E+06	–2.86E+05	1.06E+04	–1.36E+06
EL # 912 SE	2.77E+06	3.16E+05	2.82E+05	7.26E+06	1.42E+07	4.50E+06	3.29E+05	3.13E+06
SECTION 13								
EL # 894 TA	–2.70E+06	–8.30E+06	–7.71E+04	–1.76E+07	–2.22E+07	–4.40E+05	–6.44E+03	–3.54E+05
EL # 894 1-8	8.67E+05	9.64E+05	2.52E+03	–9.81E+05	–3.54E+05	–2.50E+05	2.49E+04	–6.42E+05
EL # 894 SE	3.77E+06	4.99E+05	1.07E+06	5.73E+06	7.52E+06	2.52E+06	7.33E+05	1.65E+06
SECTION 14								
EL # 876 TA	–3.55E+06	–7.45E+06	1.05E+05	–1.83E+07	–2.05E+07	–3.63E+05	–1.69E+05	–2.22E+05
EL # 876 1-8	8.62E+05	9.31E+05	2.08E+03	3.81E+05	5.92E+05	–2.44E+05	5.73E+04	9.70E+04
EL # 876 SE	4.80E+06	1.05E+06	1.58E+06	4.37E+06	8.54E+06	2.42E+06	6.50E+05	2.19E+06
SECTION 15								
EL # 1380 TA	–5.03E+06	–6.30E+06	–7.29E+05	–4.46E+05	4.25E+05	–1.58E+04	1.70E+04	5.29E+05
EL # 1380 1-8	1.08E+06	7.59E+05	–4.47E+05	1.04E+05	1.16E+05	6.80E+03	–6.74E+02	3.35E+05
EL # 1380 SE	7.10E+05	2.96E+05	5.09E+05	3.36E+05	9.22E+05	1.06E+05	8.75E+04	4.30E+05
SECTION 16								
EL # 1362 TA	–5.15E+06	–6.21E+06	–6.25E+05	–6.31E+05	–1.02E+06	–1.67E+04	–1.48E+03	3.63E+05
EL # 1362 1-8	9.87E+05	8.53E+05	–3.67E+05	–1.68E+04	–2.65E+05	5.56E+03	–4.84E+03	–5.52E+04
EL # 1362 SE	8.60E+05	2.33E+05	3.60E+05	1.44E+05	1.76E+05	1.07E+05	8.51E+03	2.76E+05
SECTION 17								
EL # 1344 TA	–5.32E+06	–6.31E+06	–5.50E+05	–9.76E+05	–1.84E+06	–2.94E+04	1.29E+04	2.91E+05
EL # 1344 1-8	9.74E+05	9.07E+05	–3.20E+05	2.39E+04	1.06E+05	–3.18E+04	4.85E+04	–1.74E+05
EL # 1344 SE	9.84E+05	2.17E+05	3.20E+05	1.46E+05	7.45E+05	4.08E+04	6.26E+04	1.99E+05
SECTION 18								
EL # 982 TA	9.54E+06	–5.48E+06	3.40E+06	–2.66E+06	–1.33E+07	1.85E+06	3.89E+05	–1.73E+06
EL # 982 1-8	4.51E+05	–4.98E+05	1.63E+05	1.63E+06	–3.29E+05	–1.78E+05	4.49E+05	–3.37E+06
EL # 982 SE	4.95E+06	1.29E+06	3.15E+06	3.54E+06	6.53E+06	2.97E+06	1.00E+06	7.57E+06
SECTION 22								
EL # 415 TA	2.82E+06	8.14E+05	7.29E+04	–4.25E+05	–7.34E+05	–3.64E+03	1.02E+04	7.78E+04
EL # 415 1-8	–2.20E+05	–2.35E+06	–2.16E+05	6.27E+04	2.18E+05	–7.61E+03	7.44E+03	3.29E+04
EL # 415 SE	2.52E+06	3.37E+06	4.77E+06	1.99E+05	9.78E+05	5.71E+04	3.51E+04	3.11E+05
SECTION 23								
EL # 516 TA	–9.23E+04	9.95E+05	–9.33E+05	–4.22E+05	–4.13E+05	–4.80E+04	–4.29E+04	–4.55E+04
EL # 516 1-8	–5.29E+04	–1.24E+06	3.57E+05	–8.82E+03	3.71E+04	–1.30E+04	–1.10E+04	1.01E+04
EL # 516 SE	2.78E+05	3.68E+06	3.82E+06	1.00E+05	4.95E+04	9.65E+04	9.93E+04	6.52E+04

Table 3H.1-12 Load Combination 15 (Continued)

	F_x (Nhh) N/m	F_y (Nmm) N/m	F_{xy} (Nhm) N/m	M_x (Mhh) N-m/m	M_y (Mmm) N-m/m	M_{xy} (Mhm) N-m/m	Q_{xz} (Nhr) N/m	Q_{yz} (Nmr) N/m
SECTION 24								
EL # 707 TA	6.34E+06	1.11E+06	2.66E+05	1.35E+06	1.33E+06	1.80E+04	1.77E+04	–2.26E+04
EL # 707 1-8	2.29E+05	–7.57E+05	–3.26E+04	5.26E+03	–1.87E+04	6.79E+03	–4.43E+03	6.34E+04
EL # 707 SE	8.60E+05	1.35E+06	2.61E+06	3.00E+04	1.57E+05	1.99E+04	3.39E+04	1.40E+05
SECTION 25								
EL # 2561 TA	–2.45E+06	–1.45E+06	–2.78E+04	4.11E+06	5.20E+06	–1.04E+05	–8.46E+04	–7.22E+05
EL # 2561 1-8	1.19E+06	2.01E+06	3.92E+05	–1.87E+05	–9.91E+05	–2.71E+05	1.34E+04	4.64E+05
EL # 2561 SE	1.52E+06	3.68E+05	1.75E+06	6.49E+05	2.08E+06	1.58E+95	3.74E+04	6.01E+05
SECTION 26								
EL # 3158 TA	–9.02E+05	2.04E+06	–1.15E+06	–5.20E+06	–4.13E+06	–1.49E+05	–4.29E+04	–4.50E+05
EL # 3158 1-8	3.35E+05	7.98E+05	–7.56E+05	–7.21E+05	2.81E+05	–6.89E+05	7.30E+05	–9.20E+05
EL # 3158 SE	2.40E+06	1.64E+06	5.47E+05	1.86E+06	5.34E+05	1.21E+06	1.06E+06	1.56E+06
SECTION 27								
EL # 2826 TA	1.67E+06	–5.24E+04	3.75E+04	5.07E+06	3.00E+06	–5.16E+03	–4.41E+05	–1.68E+06
EL # 2826 1-8	3.09E+06	–4.77E+05	–2.03E+05	6.04E+04	–5.34E+05	–1.29E+05	–3.37E+04	–1.15E+05
EL # 2826 SE	2.32E+06	5.27E+05	1.18E+06	6.87E+05	2.29E+06	1.17E+06	8.12E+05	9.53E+05
SECTION 28								
EL # 2733 TA	–6.06E+06	–1.34E+06	3.77E+06	5.20E+06	4.29E+06	1.68E+04	3.73E+04	–6.95E+05
EL # 2733 1-8	1.11E+06	–5.88E+05	1.39E+06	4.62E+04	–7.38E+04	3.60E+04	–9.08E+04	8.43E+04
EL # 2733 SE	9.89E+05	2.87E+06	2.50E+06	3.20E+05	8.84E+05	9.58E+04	3.74E+05	2.81E+05
SECTION 29								
EL # 2567 TA	1.28E+05	6.58E+05	2.05E+06	4.54E+06	4.16E+06	–4.49E+05	–3.47E+05	3.84E+05
EL # 2567 1-8	7.23E+05	1.70E+06	1.02E+06	–2.09E+05	–8.73E+05	–6.73E+04	–3.95E+04	4.56E+05
EL # 2567 SE	1.53E+06	3.14E+06	1.46E+06	4.51E+05	1.53E+06	2.15E+05	1.06E+05	5.29E+05
SECTION 30								
EL # 1109 TA	–1.10E+06	1.08E+06	2.63E+05	8.90E+03	–2.58E+04	–6.85E+03	1.43E+04	–1.11E+04
EL # 1109 1-8	–7.03E+05	9.10E+05	1.02E+06	3.40E+04	1.13E+04	1.64E+04	7.53E+04	–9.54E+03
EL # 1109 SE	7.48E+05	6.18E+05	4.85E+05	2.48E+05	5.43E+04	4.38E+04	1.13E+05	4.76E+04
SECTION 31								
EL # 1239 TA	–7.62E+05	1.32E+06	1.87E+05	7.21E+03	5.20E+03	–1.13E+03	8.67E+03	–4.38E+02
EL # 1239 1-8	–3.68E+05	5.82E+05	1.31E+05	4.83E+04	1.12E+04	5.13E+03	5.84E+04	–1.37E+04
EL # 1239 SE	8.67E+04	3.19E+05	1.19E+05	1.35E+05	2.95E+04	2.22E+04	5.92E+04	3.10E+04
SECTION 32								
EL # 1442 TA	4.20E+06	–3.27E+06	6.37E+05	4.23E+04	5.16E+05	8.27E+03	–1.03E+04	5.67E+05
EL # 1442 1-8	5.26E+05	–1.97E+05	1.09E+05	2.44E+05	1.06E+06	–2.88E+04	–3.80E+04	8.59E+05
EL # 1442 SE	6.82E+05	6.04E+05	8.92E+04	3.39E+05	1.60E+06	3.91E+04	6.46E+04	1.20E+06
SECTION 33								
EL # 2487 TA	–4.15E+05	–2.03E+05	–1.14E+06	–1.21E+05	7.43E+04	–2.62E+04	–3.75E+04	1.37E+04
EL # 2487 1-8	–3.13E+04	6.07E+04	–6.35E+05	2.27E+04	1.07E+05	1.42E+04	6.64E+03	–2.98E+04
EL # 2487 SE	1.00E+06	5.48E+05	1.72E+06	1.52E+05	5.79E+05	2.99E+04	6.69E+04	1.52E+05
SECTION 34								
EL # 1750 TA	–5.17E+05	–3.68E+05	1.77E+06	–2.52E+04	–7.83E+04	6.09E+03	4.24E+03	–3.36E+04
EL # 1750 1-8	2.13E+05	–2.07E+04	2.94E+04	–3.47E+04	–4.17E+04	4.81E+02	9.78E+03	5.19E+04
EL # 1750 SE	3.44E+05	3.60E+05	5.61E+05	3.37E+04	5.76E+04	3.69E+03	1.45E+04	6.61E+04

Table 3H.1-13 Load Combination 15a and 15b

	F_x (Nhh) N/m	F_y (Nmm) N/m	F_{xy} (Nhm) N/m	M_x (Mhh) N-m/m	M_y (Mmm) N-m/m	M_{xy} (Mhm) N-m/m	Q_{xz} (Nhr) N/m	Q_{yz} (Nmr) N/m
SECTION 1								
EL # 5 TA	–3.06E+06	–2.37E+06	–1.63E+05	5.45E+06	5.55E+06	–1.31E+04	–1.24E+04	2.77E+05
EL # 5 2-10	1.04E+06	–4.59E+06	–1.52E+04	–2.15E+05	–6.84E+05	–2.81E+03	1.82E+03	8.64E+05
EL # 5 SE	3.50E+06	1.14E+07	6.44E+06	4.21E+05	2.28E+06	2.37E+05	9.16E+04	1.02E+06
SECTION 2								
EL # 41 TA	–3.52E+06	–2.55E+06	–3.14E+05	5.25E+06	4.83E+06	–1.26E+04	1.82E+04	2.94E+05
EL # 41 2-10	3.15E+06	–3.75E+06	1.19E+05	–4.15E+04	4.26E+05	–6.27E+03	4.38E+02	–7.83E+04
EL # 41 SE	2.99E+05	8.33E+06	6.62E+06	1.49E+05	6.47E+05	2.47E+05	7.35E+04	1.85E+05
SECTION 3								
EL # 95 TA	3.87E+06	–2.68E+06	9.61E+03	5.40E+06	6.94E+06	–4.40E+04	–4.85E+04	8.90E+05
EL # 95 2-10	9.39E+05	–2.97E+06	3.73E+05	–7.95E+04	–2.70E+05	–5.05E+04	–2.51E+04	–3.01E+05
EL # 95 SE	1.07E+06	5.98E+06	6.66E+06	3.19E+04	5.94E+05	2.31E+05	5.72E+04	3.15E+05
SECTION 4								
EL # 113 TA	1.91E+05	–3.21E+06	8.48E+05	8.74E+06	1.07E+07	–1.19E+05	1.62E+04	–1.72E+06
EL # 113 2-10	6.31E+05	–2.27E+06	6.92E+05	2.71E+04	–1.78E+04	–8.52E+04	3.42E+04	1.94E+05
EL # 113 SE	1.70E+06	4.91E+06	6.22E+06	3.94E+05	7.92E+05	3.19E+05	8.89E+04	3.29E+05
SECTION 5								
EL # 131 TA	–1.44E+06	–3.53E+06	1.43E+06	7.89E+06	5.68E+06	–7.43E+04	–2.26E+03	–1.84E+06
EL # 131 2-10	6.74E+05	–1.88E+06	6.43E+05	1.13E+05	9.90E+04	–2.50E+04	2.90E+04	–1.87E+05
EL # 131 SE	2.69E+06	4.27E+06	5.97E+06	4.30E+05	2.27E+05	2.81E+05	9.21E+04	3.61E+05
SECTION 6								
EL # 167 TA	–1.34E+07	–1.36E+06	2.09E+06	6.54E+06	9.57E+06	3.43E+05	–4.43E+05	4.47E+05
EL # 167 2-10	5.99E+05	–7.07E+05	4.21E+05	–6.68E+04	–2.69E+05	3.42E+04	1.10E+04	–3.82E+05
EL # 167 SE	2.65E+06	2.46E+06	4.04E+06	4.71E+05	2.21E+05	1.19E+05	7.47E+04	8.48E+05
SECTION 7								
EL # 1620 TA	–7.70E+05	–2.64E+06	–5.97E+04	8.83E+06	1.02E+07	1.66E+05	3.10E+05	–1.16E+06
EL # 1620 2-10	6.47E+05	8.48E+05	2.20E+05	–1.12E+05	–3.19E+05	9.14E+04	2.26E+05	–2.80E+05
EL # 1620 SE	2.71E+06	8.96E+05	1.44E+06	8.22E+05	7.06E+05	5.58E+05	7.31E+05	2.09E+06
SECTION 8								
EL # 1638 TA	–8.09E+06	–1.32E+07	–3.19E+06	5.39E+06	6.58E+06	–2.51E+05	7.79E+05	–9.58E+05
EL # 1638 2-10	6.25E+05	9.47E+05	6.06E+04	–2.44E+05	–3.56E+05	1.94E+05	1.23E+05	–2.57E+04
EL # 1638 SE	2.15E+06	6.21E+05	1.76E+06	1.88E+06	2.56E+06	5.70E+05	3.69E+05	7.13E+05
SECTION 9								
EL # 1656 TA	–3.66E+06	–1.18E+07	–3.00E+05	4.54E+06	4.95E+06	4.77E+05	1.50E+06	2.03E+06
EL # 1656 2-10	4.63E+05	5.75E+05	6.04E+05	–2.25E+05	9.11E+04	2.41E+05	–1.22E+05	4.64E+05
EL # 1656 SE	9.00E+05	1.07E+06	6.48E+05	6.00E+04	3.00E+05	2.20E+05	2.61E+05	2.22E+05
SECTION 10								
EL # 929 TA	–8.30E+06	–7.89E+06	3.68E+03	–4.03E+07	–3.86E+07	–8.25E+03	1.21E+04	7.55E+02
EL # 929 2-10	6.18E+05	5.29E+05	–2.52E+03	–4.70E+06	–4.09E+06	–7.20E+03	4.20E+04	2.27E+05
EL # 929 SE	3.55E+05	2.00E+05	3.48E+05	8.23E+06	8.24E+06	4.38E+05	2.63E+06	2.66E+06

Table 3H.1-13 Load Combination 15a and 15b (Continued)

	F_x (Nhh) N/m	F_y (Nmm) N/m	F_{xy} (Nhm) N/m	M_x (Mhh) N-m/m	M_y (Mmm) N-m/m	M_{xy} (Mhm) N-m/m	Q_{xz} (Nhr) N/m	Q_{yz} (Nmr) N/m
SECTION 11								
EL # 926 TA	–8.12E+06	–8.19E+06	–6.66E+04	–3.97E+07	–3.98E+07	–7.19E+05	–2.80E+05	2.26E+04
EL # 926 2-10	5.90E+05	5.94E+05	5.17E+04	–6.00E+06	–6.54E+06	–6.79E+05	3.58E+05	8.53E+05
EL # 926 SE	1.84E+06	1.39E+06	7.83E+05	1.47E+07	1.50E+07	1.24E+06	1.68E+06	3.62E+06
SECTION 12								
EL # 912 TA	3.85E+05	–1.19E+07	–1.54E+05	–2.04E+07	–3.40E+07	–6.28E+05	–2.61E+04	–8.39E+05
EL # 912 2-10	9.33E+05	1.53E+06	1.55E+04	–2.82E+06	–4.45E+06	–2.71E+05	8.27E+03	–1.08E+06
EL # 912 SE	2.77E+06	3.16E+05	2.82E+05	7.26E+06	1.42E+07	4.50E+06	3.29E+05	3.13E+06
SECTION 13								
EL # 894 TA	–2.64E+06	–9.00E+06	–7.11E+04	–2.37E+07	–2.86E+07	–5.31E+05	–3.31E+04	–5.52E+05
EL # 894 2-10	1.13E+06	1.35E+06	1.48E+04	–1.84E+06	–1.60E+06	–2.37E+05	1.68E+04	–6.77E+05
EL # 894 SE	3.77E+06	4.99E+05	1.06E+06	5.73E+06	7.52E+06	2.52E+06	7.33E+05	1.65E+06
SECTION 14								
EL # 876 TA	–3.68E+06	–7.95E+06	1.91E+05	–2.44E+07	–2.62E+07	–3.82E+05	–2.19E+05	–3.68E+05
EL # 876 2-10	1.13E+06	1.29E+06	9.46E+03	–3.76E+05	–3.77E+04	–2.19E+05	5.08E+04	–3.34E+05
EL # 876 SE	4.80E+06	1.05E+06	1.58E+06	4.37E+06	8.54E+06	2.42E+06	6.50E+05	2.19E+06
SECTION 15								
EL # 1380 TA	–8.81E+06	–1.02E+07	–1.11E+05	–1.39E+06	–2.40E+05	–1.90E+04	2.33E+04	8.20E+05
EL # 1380 2-10	9.03E+05	6.82E+05	–3.52E+05	8.44E+04	9.47E+04	7.77E+03	–2.61E+03	2.31E+05
EL # 1380 SE	7.10E+05	2.96E+05	5.09E+05	7.56E+04	9.22E+05	1.06E+05	8.75E+04	4.30E+05
SECTION 16								
EL # 1362 TA	–8.82E+06	–1.01E+07	–9.20E+05	–1.73E+06	–2.40E+06	–2.11E+04	–1.93E+03	5.55E+05
EL # 1362 2-10	8.38E+05	7.49E+05	–2.93E+05	1.18E+03	–1.71E+05	7.10E+03	–3.69E+03	–3.59E+04
EL # 1362 SE	8.60E+05	2.33E+05	3.60E+05	1.44E+05	1.76E+05	1.07E+05	8.51E+03	2.76E+05
SECTION 17								
EL # 1344 TA	–9.01E+06	–1.03E+07	–7.71E+05	–2.27E+06	–3.66E+06	–4.71E+04	2.77E+04	4.48E+05
EL # 1344 2-10	8.31E+05	7.88E+05	–2.62E+05	1.67E+04	5.61E+04	–2.70E+04	4.40E+04	–1.01E+05
EL # 1344 SE	9.84E+05	2.17E+05	3.20E+05	1.46E+05	7.45E+05	4.08E+04	6.26E+04	1.99E+05
SECTION 18								
EL # 982 TA	1.05E+07	–5.95E+06	4.10E+06	–4.49E+06	–1.59E+07	2.25E+06	4.45E+05	–2.31E+06
EL # 982 2-10	6.13E+05	–4.94E+05	2.24E+05	8.15E+05	–1.27E+06	–1.44E+05	5.16E+05	–3.77E+06
EL # 982 SE	4.94E+06	1.29E+06	3.15E+06	3.54E+06	6.53E+06	2.97E+06	1.00E+06	7.57E+06
SECTION 22								
EL # 415 TA	3.29E+06	1.09E+06	9.79E+04	–4.49E+05	–9.34E+05	–6.67E+03	1.48E+04	1.30E+05
EL # 415 2-10	–2.07E+05	–2.35E+06	–2.23E+05	4.82E+04	1.38E+05	–6.98E+03	1.02E+04	6.56E+04
EL # 415 SE	2.52E+06	3.37E+06	4.77E+06	1.99E+05	9.78E+05	5.71E+04	3.51E+04	3.11E+05
SECTION 23								
EL # 516 TA	–6.36E+04	1.05E+06	–1.15E+06	–4.49E+05	–4.33E+0-5	–5.60E+04	–3.82E+04	–5.03E+04
EL # 516 2-10	–6.07E+04	–1.32E+06	3.94E+05	–3.70E+03	3.77E+04	–1.14E+04	–1.39E+04	9.38E+03
EL # 516 SE	2.78E+05	3.68E+06	3.82E+06	1.00E+05	4.95E+04	9.65E+04	9.93E+04	6.52E+04

Table 3H.1-13 Load Combination 15a and 15b (Continued)

	F_x (Nhh) N/m	F_y (Nmm) N/m	F_{xy} (Nhm) N/m	M_x (Mhh) N-m/m	M_y (Mmm) N-m/m	M_{xy} (Mhm) N-m/m	Q_{xz} (Nhr) N/m	Q_{yz} (Nmr) N/m
SECTION 24								
EL # 707 TA	6.80E+06	1.12E+06	3.42E+05	1.37E+06	1.36E+06	1.76E+04	3.43E+04	–4.19E+04
EL # 707 2-10	2.03E+05	–7.61E+05	–4.81E+04	1.81E+03	–2.93E+04	7.49E+03	–6.65E+03	7.03E+04
EL # 707 SE	8.60E+05	1.25E+06	2.61E+06	3.00E+04	1.57E+05	1.99E+04	3.39E+04	1.40E+05
SECTION 25								
EL # 2561 TA	–2.59E+06	–1.30E+06	1.37E+05	3.02E+06	4.40E+06	–6.09E+04	–6.02E+04	–7.44E+05
EL # 2561 2-10	1.13E+06	2.00E+06	3.99E+05	–2.24E+05	–1.06E+06	–2.65E+04	1.18E+04	4.91E+05
EL # 2561 SE	1.52E+06	3.68E+06	1.75E+06	6.49E+05	2.08E+06	1.58E+05	3.74E+04	6.01E+05
SECTION 26								
EL # 3158 TA	–5.88E+05	4.45E+06	–3.05E+06	–4.80E+06	–3.60E+06	3.62E+04	–2.28E+05	–6.69E+05
EL # 3158 2-10	5.88E+05	8.15E+05	–6.39E+05	–8.10E+05	2.58E+05	–6.65E+05	7.11E+05	–9.25E+05
EL # 3158 SE	2.40E+06	1.64E+06	5.47E+05	1.86E+06	5.34E+05	1.21E+06	1.06E+06	1.56E+06
SECTION 27								
EL # 2826 TA	3.34E+06	9.53E+05	9.25E+05	3.95E+06	2.24E+06	–7.21E+05	–5.27E+04	–1.82E+06
EL # 2826 2-10	2.02E+06	–4.45E+05	–5.95E+04	5.72E+04	–5.80E+05	–3.77E+04	–2.56E+04	–5.32E+04
EL # 2826 SE	2.32E+06	5.27E+05	1.18E+06	6.68E+05	2.29E+06	1.17E+06	8.12E+05	9.53E+05
SECTION 28								
EL # 2733 TA	–3.94E+06	–2.59E+06	5.59E+06	3.71E+06	3.23E+06	1.40E+05	–2.22E+05	–5.25E+05
EL # 2733 2-10	8.35E+05	–9.46E+05	1.21E+06	3.34E+04	–8.25E+04	3.71E+04	–9.43E+04	7.84E+04
EL # 2733 SE	9.89E+05	2.87E+06	2.50E+06	3.20E+05	8.84E+05	9.58E+04	3.74E+05	2.81E+05
SECTION 29								
EL # 2567 TA	–2.08E+06	5.99E+05	2.49E+06	2.84E+06	3.02E+06	–4.98E+05	–3.29E+05	5.17E+05
EL # 2567 2-10	8.07E+05	1.72E+06	7.88E+05	–2.00E+05	–8.67E+05	–5.83E+04	–3.88E+04	4.58E+05
EL # 2567 SE	1.53E+06	3.14E+06	1.46E+06	4.51E+05	1.53E+06	2.15E+05	1.06E+05	5.29E+05
SECTION 30								
EL # 1109 TA	–1.43E+06	1.32E+06	3.73E+05	–7.21E+03	–2.60E+04	–6.58E+03	1.64E+04	–5.53E+03
EL # 1109 2-10	–8.36E+05	9.40E+05	8.71E+04	2.84E+04	9.07E+03	1.72E+04	7.14E+04	–8.54E+03
EL # 1109 SE	7.48E+05	6.18E+05	4.85E+05	2.48E+05	5.43E+04	4.38E+04	1.13E+05	4.76E+04
SECTION 31								
EL # 1239 TA	–1.14E+06	1.84E+06	2.38E+05	–8.10E+03	2.46E+03	2.13E+03	4.75E+03	3.40E+03
EL # 1239 2-10	–3.07E+05	5.41E+05	1.14E+05	4.33E+04	9.75E+03	5.75E+03	5.65E+04	–1.26E+04
EL # 1239 SE	8.67E+04	3.19E+05	1.19E+05	1.35E+05	2.95E+04	2.22E+04	5.92E+04	3.10E+04
SECTION 32								
EL # 1442 TA	6.71E+06	–4.85E+06	9.70E+05	5.60E+04	5.78E+05	8.01E+03	–1.56E+04	6.11E+05
EL # 1442 2-10	4.53E+05	–1.37E+05	8.87E+04	2.22E+05	9.60E+05	–2.70E+04	–3.84E+04	7.89E+05
EL # 1442 SE	6.82E+05	6.04E+05	8.92E+04	3.39E+05	1.60E+06	3.91E+04	6.46E+04	1.20E+06
SECTION 33								
EL # 2487 TA	–9.35E+05	–1.34E+05	–1.96E+06	–1.86E+05	–4.39E+03	–3.06E+04	–1.86E+04	1.11E+05
EL # 2487 2-10	2.66E+04	3.77E+04	–3.96E+05	2.83E+04	1.22E+05	1.22E+04	7.21E+03	–3.90E+04
EL # 2487 SE	1.00E+06	5.48E+05	1.72E+06	1.52E+05	5.79E+05	2.99E+04	6.69E+04	1.52E+05

Table 3H.1-14 Rebar Ratios Used in the Analysis

Section #	Location	Reinforcing Steel %				
		Meridional		Hoop		
		Inside Face/Top	Outside Face/ Bottom	Inside Face/Top	Outside Face/ Bottom	Shear Ties
1	RCCV Wetwell Bottom	1.333	1.205	0.859	0.859	0.50
2	RCCV Wetwell Mid-Height	1.333	1.205	0.938	0.938	0.50
3	RCCV Wetwell Top	1.333	1.205	1.290	1.290	0.50
4	RCCV Drywell Bottom	1.333	1.205	1.721	1.721	0.50
5	RCCV Drywell Mid-Height	1.111	1.005	1.143	1.143	0.50
6	RCCV Drywell Top	1.111	1.005	1.143	1.143	0.50
7	RCCV Top Slab @ RCCV Wall	0.841	0.630	0.630	0.630	0.46
8	RCCV Top Slab @ Center	0.841	0.841	0.630	0.630	0.46
9	RCCV Top Slab @ Drywell Head Opening	1.262	1.262	0.630	0.630	0.91
10	Basemat Cavity @ Center	0.329	0.329	0.329	0.329	0.55
11	Basemat Inside RPV Pedestal	0.276	0.276	0.288	0.288	0.55
12	Basemat Outside RPV Pedestal	0.631	0.631	0.309	0.309	0.55
13	Basemat Between RCCV & RPV Pedestal	0.439	0.439	0.495	0.495	0.55
14	Basemat Inside RCCV	0.483	0.483	0.495	0.495	0.55
15	D/F Slab @ RPV	1.563	1.563	1.612	1.612	0.52
16	D/F Slab @ Center	0.943	0.943	1.074	1.074	0.31
17	D/F Slab @ RCCV	1.131	1.131	1.074	1.074	0.25
18	Basemat Outside RCCV	0.308	0.614	0.308	0.614	0.83
22	R/B Outside Wall @ Base	0.996	0.996	0.996	0.996	0.30

Table 3H.1-14 Rebar Ratios Used in the Analysis (Continued)

Section #	Location	Reinforcing Steel %				
		Meridional		Hoop		
		Inside Face/Top	Outside Face/ Bottom	Inside Face/Top	Outside Face/ Bottom	Shear Ties
23	R/B Outside Wall @ Center	1.059	1.059	1.059	1.059	0.30
24	R/B Outside Wall @ Grade	1.270	1.270	1.270	1.270	0.10
25	Fuel Pool Wall @ Base	0.851	0.851	0.851	0.851	0.20
26	Fuel Pool Slab @ Center	0.784	0.784	0.946	0.946	0.38
27	Fuel Pool Girder @ Drywell Head Opening	0.591	0.591	0.591	0.591	0.20
28	Fuel Pool Girder @ RCCV Wall	0.968	0.968	0.968	0.968	0.20
29	Fuel Pool Girder @ Deep End	0.968	0.968	0.968	0.968	0.20
30	R/B Floor @ El. –6.70m@ RCCV	1.409	1.409	1.409	1.409	0.20
31	R/B Floor @ El. –0.20m @ RCCV	1.692	1.692	1.692	1.692	0.20
32	R/B Floor @ El. 7.30m @ RCCV	0.908	0.908	0.605	0.605	0.30
33	Steam Tunnel Floor	1.211	1.211	1.211	1.211	0.50
34	Steam Tunnel Roof	1.129	1.129	1.129	1.129	0.50

Table 3H.1-15 Rebar and Concrete Stresses Due to Load Combination 1

Section #	Location	Element #	Azimuth (Deg.)	Calculated Reinforcing Steel Stresses (MPa)					Reinforcing Steel Allowable Stress (MPa)	Concrete Stress (MPa)	
				Inside Face/Top		Outside Face/ Bottom		Shear Ties			
				Merid.	Circum.	Merid.	Circum.			Calculated	Allowable
1	RCCV Wetwell Bottom	5	225.80	–10.69	30.27	–14.82	15.65	2.00	310.26	–1.896	–16.55
2	RCCV Wetwell Mid-Height	41	225.80	–14.48	107.91	–6.55	87.91	0.00	310.26	–2.020	–16.55
3	RCCV Wetwell Top	95	225.80	0.34	43.30	–7.65	31.23	22.06	310.26	–1.613	–16.55
4	RCCV Drywell Bottom	113	225.80	0.14	23.65	–0.69	20.48	15.44	310.26	–1.476	–16.55
5	RCCV Drywell Mid-Height	131	225.80	2.62	26.06	6.41	30.61	9.17	310.26	–1.158	–16.55
6	RCCV Drywell Top	167	225.80	34.00	22.48	9.17	15.38	45.51	310.26	–1.682	–16.55
7	RCCV Top Slab @ RCCV Wall	1620	225.81	23.17	30.06	68.19	29.44	25.03	310.26	–1.027	–16.55
8	RCCV Top Slab @ Center	1638	225.36	33.79	33.51	33.92	35.16	12.89	310.26	–0.903	–16.55
9	RCCV Top Slab @ Drywell Head Opening	1656	225.00	45.23	51.78	29.65	44.75	25.79	310.26	–1.855	–16.55
10	Basemat Cavity @ Center	929	270.00	11.10	6.14	14.76	19.03	4.07	310.26	–0.690	–16.55
11	Basemat Inside RPV Pedestal	926	192.04	12.76	8.76	32.20	26.82	9.93	310.26	–0.552	–16.55
12	Basemat Outside RPV Pedestal	912	225.00	24.68	17.24	24.62	15.93	28.61	310.26	–0.552	–16.55
13	Basemat Between RCCV & RPV Pedestal	894	225.36	28.20	20.48	6.62	3.86	8.83	310.26	–0.648	–16.55
14	Basemat Inside RCCV	876	225.81	24.41	27.10	7.58	–1.03	11.10	310.26	–0.696	–16.55
15	D/F Slab @ RPV	1380	225.00	44.33	45.37	42.40	45.92	21.17	310.26	–1.007	–16.55
16	D/F Slab @ Center	1362	225.36	49.99	56.95	79.57	66.60	–0.28	310.26	–0.848	–16.55
17	D/F Slab @ RCCV	1344	225.81	41.58	52.40	68.88	63.30	8.41	310.26	–1.090	–16.55
18	Basemat Outside RCCV	982	227.93	51.30	46.20	36.13	2.69	50.33	310.26	–1.331	–16.55

Table 3H.1-16 Rebar and Concrete Stresses Due to Load Combination 8

Section #	Location	Element #	Azimuth (Deg.)	Calculated Reinforcing Steel Stresses (MPa)					Reinforcing Steel Allowable Stress (MPa)	Concrete Stress (MPa)	
				Inside Face/Top		Outside Face/Bottom		Shear Ties			
				Merid.	Circum.	Merid.	Circum.			Calculated	Allowable
1	RCCV Wetwell Bottom	5	225.80	–28.75	25.23	1.79	45.30	27.86	372.32	–4.59	–23.44
2	RCCV Wetwell Mid-Height	41	225.80	–33.10	93.29	13.72	119.14	7.65	372.32	–5.27	–23.44
3	RCCV Wetwell Top	95	225.80	–31.30	38.75	25.03	73.57	11.58	372.32	–5.32	–23.44
4	RCCV Drywell Bottom	113	225.80	–34.27	–3.10	86.05	81.64	15.17	372.32	–8.09	–23.44
5	RCCV Drywell Mid-Height	131	225.80	2.07	–1.86	63.30	105.98	38.68	372.32	–4.77	–23.44
6	RCCV Drywell Top	167	225.80	19.03	–18.20	37.58	52.68	53.37	372.32	–4.35	–23.44
7	RCCV Top Slab @ RCCV Wall	1620	225.81	–39.16	–2.69	142.17	88.74	17.24	372.32	–9.89	–23.44
8	RCCV Top Slab @ Center	1638	225.36	–53.99	64.68	–9.03	95.91	18.00	372.32	–7.72	–23.44
9	RCCV Top Slab @ Drywell Head Opening	1656	225.00	–34.82	–8.83	–16.13	–8.14	17.79	372.32	–5.14	–23.44
10	Basemat Cavity @ Center	929	270.00	–22.96	–25.93	86.53	95.84	5.86	372.32	–4.68	–23.44
11	Basemat Inside RPV Pedestal	926	192.04	–25.79	–26.06	116.32	109.77	6.27	372.32	–5.84	–23.44
12	Basemat Outside RPV Pedestal	912	225.00	–38.20	–4.34	92.12	123.01	24.55	372.32	–7.12	–23.44
13	Basemat Between RCCV and RPV Pedestal	894	225.36	19.93	–8.89	67.98	70.47	41.65	372.32	–2.46	–23.44
14	Basemat Inside RCCV	876	225.81	22.41	–6.21	38.89	51.23	36.34	372.32	–1.35	–23.44
15	D/F Slab @ RPV	1380	225.00	–16.62	22.62	–29.86	–11.65	22.00	372.32	–4.92	–23.44
16	D/F Slab @ Center	1362	225.36	–42.40	–23.10	–4.69	–6.00	10.41	372.32	–8.80	–23.44
17	D/F Slab @ RCCV	1344	225.81	–45.37	–26.61	1.03	–5.45	13.31	372.32	–9.86	–23.44
18	Basemat Outside RCCV	982	227.93	52.26	271.04	98.05	166.24	82.88	372.32	–4.32	–23.44
22	R/B Outside Wall @ Base	415	267.23	51.64	59.64	–10.20	6.62	80.46	372.32	–4.77	–23.44
23	R/B Outside Wall @ Center	516	224.57	74.19	66.54	23.44	15.38	90.39	372.32	–3.59	–23.44
24	R/B Outside Wall @ Grade	707	267.23	–21.37	67.02	68.60	229.47	16.48	372.32	–4.01	–23.44
25	Fuel Pool Wall @ Base	2561	184.31	48.75	–5.58	74.95	74.88	45.85	372.32	–3.31	–23.44

Table 3H.1-16 Rebar and Concrete Stresses Due to Load Combination 8 (Continued)

Section #	Location	Element #	Azimuth (Deg.)	Calculated Reinforcing Steel Stresses (MPa)					Reinforcing Steel Allowable Stress (MPa)	Concrete Stress (MPa)	
				Inside Face/Top		Outside Face/Bottom		Shear Ties			
				Merid.	Circum.	Merid.	Circum.			Calculated	Allowable
26	Fuel Pool Slab @ Center	3158	197.05	107.70	26.82	132.45	126.04	149.83	372.32	–6.70	–23.44
27	Fuel Pool Girder @ Drywell Head Opening	2826	295.00	46.06	123.00	61.16	161.55	140.04	372.32	–1.50	–23.44
28	Fuel Pool Girder @ RCCV Wall	2733	208.92	57.99	–5.58	180.10	183.20	38.34	372.32	–8.43	–23.44
29	Fuel Pool Girder @ Deep End	2567	199.68	109.98	57.30	180.72	180.79	103.01	372.32	–6.27	–23.44
30	R/B Floor @ El. –6.70 m @ RCCV	1109	213.85	99.29	–8.89	99.22	–17.31	–0.28	372.32	–2.53	–23.44
31	R/B Floor @ El. –0.20 m @ RCCV	1239	213.85	97.56	5.10	86.19	–12.89	0.83	372.32	–2.45	–23.44
32	R/B Floor @ El. 7.30 m @ RCCV	1442	274.91	82.74	213.4	–7.86	160.38	135.35	372.32	–6.45	–23.44
33	Steam Tunnel Floor	2487	340.93	45.09	48.61	41.78	26.61	–0.28	372.32	–2.51	–23.44
34	Steam Tunnel Roof	1750	301.04	74.47	118.59	8.89	–1.52	14.62	372.32	–11.33	–23.44

Table 3H.1-17 Rebar and Concrete Stresses Due to Load Combination 15

Section #	Location	Element #	Azimuth (Deg.)	Calculated Reinforcing Steel Stresses (MPa)					Reinforcing Steel Allowable Stress (MPa)	Concrete Stress (MPa)	
				Inside Face/Top		Outside Face/Bottom		Shear Ties			
				Merid.	Circum.	Merid.	Circum.			Calculated	Allowable
1	RCCV Wetwell Bottom	5	225.80	218.15	260.29	334.27	324.07	193.82	372.32	–9.267	–23.44
2	RCCV Wetwell Mid-Height	41	225.80	164.10	188.79	277.32	282.28	67.16	372.32	–8.226	–23.44
3	RCCV Wetwell Top	95	225.80	120.25	164.65	253.87	232.29	63.99	372.32	–7.812	–23.44
4	RCCV Drywell Bottom	113	225.80	97.84	97.29	283.80	208.42	78.81	372.32	–8.680	–23.44
5	RCCV Drywell Mid-Height	131	225.80	199.13	176.37	291.04	289.80	147.07	372.32	–8.970	–23.44
6	RCCV Drywell Top	167	225.80	38.47	59.43	353.16	258.98	55.16	372.32	–8.708	–23.44
7	RCCV Top Slab @ RCCV Wall	1620	225.81	201.68	236.22	43.58	37.78	127.21	372.32	–5.868	–23.44
8	RCCV Top Slab @ Center	1638	225.36	15.79	193.96	–47.71	124.39	88.67	372.32	–9.377	–23.44
9	RCCV Top Slab @ Drywell Head Opening	1656	225.00	1.17	122.32	–26.89	123.70	34.68	372.32	–5.109	–23.44
10	Basemat Cavity @ Center	929	270.00	9.31	–7.86	107.29	117.35	70.88	372.32	–6.171	–23.44
11	Basemat Inside RPV Pedestal	926	192.04	43.04	–19.51	255.60	223.47	99.22	372.32	–8.991	–23.44
12	Basemat Outside RPV Pedestal	912	225.00	–59.78	41.44	174.79	257.46	62.54	372.32	–12.990	–23.44
13	Basemat Between RCCV and RPV Pedestal	894	225.36	–53.09	31.85	166.58	170.86	28.82	372.32	–9.619	–23.44
14	Basemat Inside RCCV	876	225.81	–34.54	68.88	177.68	191.82	45.92	372.32	–8.453	–23.44
15	D/F Slab @ RPV	1380	225.00	17.58	–25.93	–38.89	–4.96	58.75	372.32	–8.805	–23.44
16	D/F Slab @ Center	1362	225.36	–38.68	44.96	25.10	92.74	42.33	372.32	–10.143	–23.44
17	D/F Slab @ RCCV	1344	225.81	–60.06	46.75	78.53	98.53	24.48	372.32	–16.410	–23.44
18	Basemat Outside RCCV	982	227.93	216.43	317.86	276.63	253.12	224.16	372.32	–6.440	–23.44
22	R/B Outside Wall @ Base	415	267.23	342.68	306.83	190.78	220.30	321.10	372.32	–9.839	–23.44
23	R/B Outside Wall @ Center	516	224.57	316.62	238.43	216.37	152.17	293.38	372.32	–8.226	–23.44
24	R/B Outside Wall @ Grade	707	267.23	96.60	174.17	272.42	371.16	183.00	372.32	–7.005	–23.44
25	Fuel Pool Wall @ Base	2561	184.31	116.32	44.89	320.07	211.40	116.53	372.32	–5.454	–23.44

Table 3H.1-17 Rebar and Concrete Stresses Due to Load Combination 15 (Continued)

Section #	Location	Element #	Azimuth (Deg.)	Calculated Reinforcing Steel Stresses (MPa)					Reinforcing Steel Allowable Stress (MPa)	Concrete Stress (MPa)	
				Inside Face/Top		Outside Face/Bottom		Shear Ties			
				Merid.	Circum.	Merid.	Circum.			Calculated	Allowable
26	Fuel Pool Slab @ Center	3158	197.05	220.09	118.04	231.81	226.43	353.23	372.32	–10.053	–23.44
27	Fuel Pool Girder @ Drywell Head Opening	2826	295.00	98.94	196.65	155.83	257.18	300.21	372.32	–4.192	–23.44
28	Fuel Pool Girder @ RCCV Wall	2733	208.92	167.69	92.74	312.76	274.42	198.64	372.32	–10.446	–23.44
29	Fuel Pool Girder @ Deep End	2567	199.68	202.92	122.46	339.30	269.59	210.92	372.32	–7.219	–23.44
30	R/B Floor @ El. –6.70 m @ RCCV	1109	213.85	152.45	109.35	149.62	–13.24	40.61	372.32	–5.302	–23.44
31	R/B Floor @ El. –0.20 m @ RCCV	1239	213.85	127.63	60.61	83.36	–24.48	6.27	372.32	–5.419	–23.44
32	R/B Floor @ El. 7.30 m @ RCCV	1442	274.91	229.40	278.49	10.07	158.03	300.62	372.32	–12.521	–23.44
33	Steam Tunnel Floor	2487	340.93	97.08	134.59	118.04	76.81	10.41	372.32	–4.951	–23.44
34	Steam Tunnel Roof	1750	301.04	41.71	55.37	86.81	86.81	46.33	372.32	–9.239	–23.44

Table 3H.1-18 Rebar and Concrete Stresses Due to Load Combinations 15a and 15b

Section #	Location	Element #	Azimuth (Deg.)	Calculated Reinforcing Steel Stresses (MPa)					Reinforcing Steel Allowable Stress (MPa)	Concrete Stress (MPa)	
				Inside Face/Top		Outside Face/Bottom		Shear Ties			
				Merid.	Circum.	Merid.	Circum.			Calculated	Allowable
1	RCCV Wetwell Bottom	5	225.80	187.82	201.75	371.09	352.20	219.67	372.32	–10.515	–23.44
2	RCCV Wetwell Mid-Height	41	225.80	112.46	121.77	304.62	315.31	51.23	372.32	–8.019	–23.44
3	RCCV Wetwell Top	95	225.80	59.64	137.35	315.86	276.21	75.43	372.32	–10.308	–23.44
4	RCCV Drywell Bottom	113	225.80	23.65	53.16	367.02	258.77	100.11	372.32	–13.645	–23.44
5	RCCV Drywell Mid-Height	131	225.80	130.52	110.25	318.07	349.23	151.97	372.32	–9.784	–23.44
6	RCCV Drywell Top	167	225.80	15.17	–47.02	319.17	183.48	27.37	372.32	–13.362	–23.44
7	RCCV Top Slab @ RCCV Wall	1620	225.81	195.27	226.29	77.50	60.06	161.14	372.32	–3.854	–23.44
8	RCCV Top Slab @ Center	1638	225.36	146.66	252.29	–73.50	31.23	20.62	372.32	–14.831	–23.44
9	RCCV Top Slab @ Drywell Head Opening	1656	225.00	22.96	85.29	–45.78	40.34	59.78	372.32	–8.322	–23.44
10	Basemat Cavity @ Center	929	270.00	14.55	–3.31	138.10	155.07	76.26	372.32	–6.770	–23.44
11	Basemat Inside RPV Pedestal	926	192.04	52.54	–19.65	282.07	283.80	109.00	372.32	–10.452	–23.44
12	Basemat Outside RPV Pedestal	912	225.00	–75.29	19.38	206.37	316.90	52.68	372.32	–15.534	–23.44
13	Basemat Between RCCV and RPV Pedestal	894	225.36	–59.09	33.85	216.99	197.75	43.58	372.32	–12.004	–23.44
14	Basemat Inside RCCV	876	225.81	–39.78	61.57	211.33	213.75	55.50	372.32	–9.784	–23.44
15	D/F Slab @ RPV	1380	225.00	129.00	–49.09	245.19	20.62	42.06	372.32	–12.011	–23.44
16	D/F Slab @ Center	1362	225.36	–81.02	–54.95	5.65	–5.03	26.89	372.32	–18.244	–23.44
17	D/F Slab @ RCCV	1344	225.81	–73.50	–42.33	96.67	32.68	27.72	372.32	–22.650	–23.44
18	Basemat Outside RCCV	982	227.93	228.09	328.06	299.66	271.73	242.84	372.32	–7.089	–23.44
22	R/B Outside Wall @ Base	415	267.23	360.33	316.41	193.68	222.85	327.86	372.32	–10.170	–23.44
23	R/B Outside Wall @ Center	516	224.57	327.72	246.29	221.88	158.03	295.45	372.32	–8.460	–23.44
24	R/B Outside Wall @ Grade	707	267.23	96.87	177.62	273.25	373.92	194.44	372.32	–7.102	–23.44
25	Fuel Pool Wall @ Base	2561	184.31	128.59	60.26	309.24	183.61	120.52	372.32	–4.268	–23.44

Table 3H.1-18 Rebar and Concrete Stresses Due to Load Combinations 15a and 15b (Continued)

Section #	Location	Element #	Azimuth (Deg.)	Calculated Reinforcing Steel Stresses (MPa)					Reinforcing Steel Allowable Stress (MPa)	Concrete Stress (MPa)	
				Inside Face/Top		Outside Face/Bottom		Shear Ties			
				Merid.	Circum.	Merid.	Circum.			Calculated	Allowable
26	Fuel Pool Slab @ Center	3158	197.05	277.25	156.72	288.56	262.01	374.81	372.32	–10.287	–23.44
27	Fuel Pool Girder @ Drywell Head Opening	2826	295.00	149.21	220.22	194.23	271.94	343.65	372.32	–3.689	–23.44
28	Fuel Pool Girder @ RCCV Wall	2733	208.92	139.21	123.21	322.07	297.52	260.70	372.32	–10.646	–23.44
29	Fuel Pool Girder @ Deep End	2567	199.68	228.98	120.94	311.17	203.54	223.05	372.32	–7.122	–23.44
30	R/B Floor @ El. –6.70 m @ RCCV	1109	213.85	139.69	1.03	189.96	86.19	38.82	372.32	–4.640	–23.44
31	R/B Floor @ El. –0.20 m @ RCCV	1239	213.85	143.62	53.92	103.01	–24.89	8.00	372.32	–5.274	–23.44
32	R/B Floor @ El. 7.30 m @ RCCV	1442	274.91	218.02	355.44	0.97	232.71	288.56	372.32	–12.776	–23.44
33	Steam Tunnel Floor	2487	340.93	73.02	99.01	180.92	138.66	53.71	372.32	–5.964	–23.44
34	Steam Tunnel Roof	1750	301.04	49.58	64.19	98.53	92.67	46.40	372.32	–10.294	–23.44

Table 3H.1-19 Containment Liner Plate Strains (Max)

Load Comb. #	Struct. Comp.	Calculated Strain						Allowable Strain	
								Compress.	Tension
		Containment Wall							
	Section #	1	2	3	4	5	6		
	Element #	5	41	95	113	131	167		
1		0.000237	0.000771	0.000337	0.000210	0.000208	0.000301	−0.00400	0.00400
8		0.000200	0.000656	0.000282	−0.000313	−0.000142	0.000101	−0.00400	0.01000
15a,15b		0.001970	0.001060	0.001040	0.000309	0.001090	−0.000547	−0.01400	0.01000

Table 3H.1-19 Containment Liner Plate Strains (Max) (Continued)

Load Comb. #	Struct. Comp.	Calculated Strain								Allowable Strain Compress.	Allowable Strain Tension
		Containment Top Slab			Basemat Slab						
	Section #	7	8	9	10	11	12	13	14		
	Element #	1620	1638	1656	929	926	912	894	876		
1		0.000441	0.000384	0.000372	0.000134	0.000302	0.000200	0.000041	0.000043	–0.00400	0.00400
8		0.000017	0.000499	0.000004	–0.000163	–0.000206	–0.000252	0.000103	0.000123	–0.01400	0.01000
15a,15b		0.000341	0.000149	0.000318	0.000099	0.000259	0.000196	0.000291	0.000439	–0.01400	0.01000

Table 3H.1-19 Containment Liner Plate Strains (Max) (Continued)

Load Comb. #	Struct. Comp.	Calculated Strain						Allowable Strain Compress.	Allowable Strain Tension
		Diaphragm Floor Slab							
	Section #	**15**	**16**	**17**					
	Element #	**1380**	**1362**	**1344**					
1		0.000481	0.000833	0.000744				–0.00400	0.00400
1*		0.000408	0.001150	0.000512				–0.00400	0.00400
8		0.000017	–0.000328	–0.000365				–0.01400	0.01000
15a,15b		0.000044	–0.000658	–0.000848				–0.01400	0.01000

Table 3H.1-20 Stresses in Pedestal

Section	Critical Load Case	Resultant Membrane Stress in Steel Jacket (MPa)		Resultant Shear Stress in Stiffener Plate (MPa)	
		Calculated	Allowable	Calculated	Allowable
19	15a,15b	167.00	310.26	62.06	189.61
20	15a,15b	146.86	310.26	6.83	189.61
21	15a,15b	206.50	310.26	184.10	189.61

Table 3H.1-21 Maximum Moments and Shears in Walls Due to Lateral Soil Pressure (At-Rest Condition)

Section	M_x (N•m/m)	M_y (N•m/m)	V (N/m)
22	1.21E+06	7.95E+05	1.15E+06
23	8.19E+05	5.40E+05	7.81E+05
At El. 12.0m	5.31E+05	3.47E+05	4.57E+05

Table 3H.1-22 Maximum Moments and Shears in Walls Due to SSE Soil Pressure (SSE Condition)

Section	M_x (N•m/m)	M_y (N•m/m)	V (N/m)
22	1.62E+06	1.06E+06	1.54E+06
23	1.49E+06	9.85E+05	1.43E+06
At El. 12.0m	1.69E+06	1.11E+06	1.45E+06

Table 3H.1-23 Factors of Safety for Foundation Stability*

Load Combination	Overturning		Sliding		Floatation	
	Req'd.	Actual	Req'd.	Actual	Req'd.	Actual
D + F’					1.1	2.43
$D + L_o + F + H + E_{ss}$	1.1	490	1.1	1.11		

Here:

F = Buoyant Forces from Design Ground Water (0.61m Below Grade)

F’ = Buoyant Forces from Design Basis Flood (0.3m Below Grade)

H = Lateral Soil Pressure

L_o = Live Load Acting During an Earthquake (Zero Live Load is Considered).

E_{ss} = SSE Load

D = Dead Load

* Based on the calculation for shear forces due to tornado loads, it was found that it is less than 10% of the shear forces due to the seismic effects. Hence it was concluded that the load combinations comprising of wind and tornado loadings will not be the governing load combinations for the evaluation of overturning and sliding effects of the R/B stability and therefore, were not evaluated.

Figure 3H.1-1 (Refer to Figure 1.2-1)

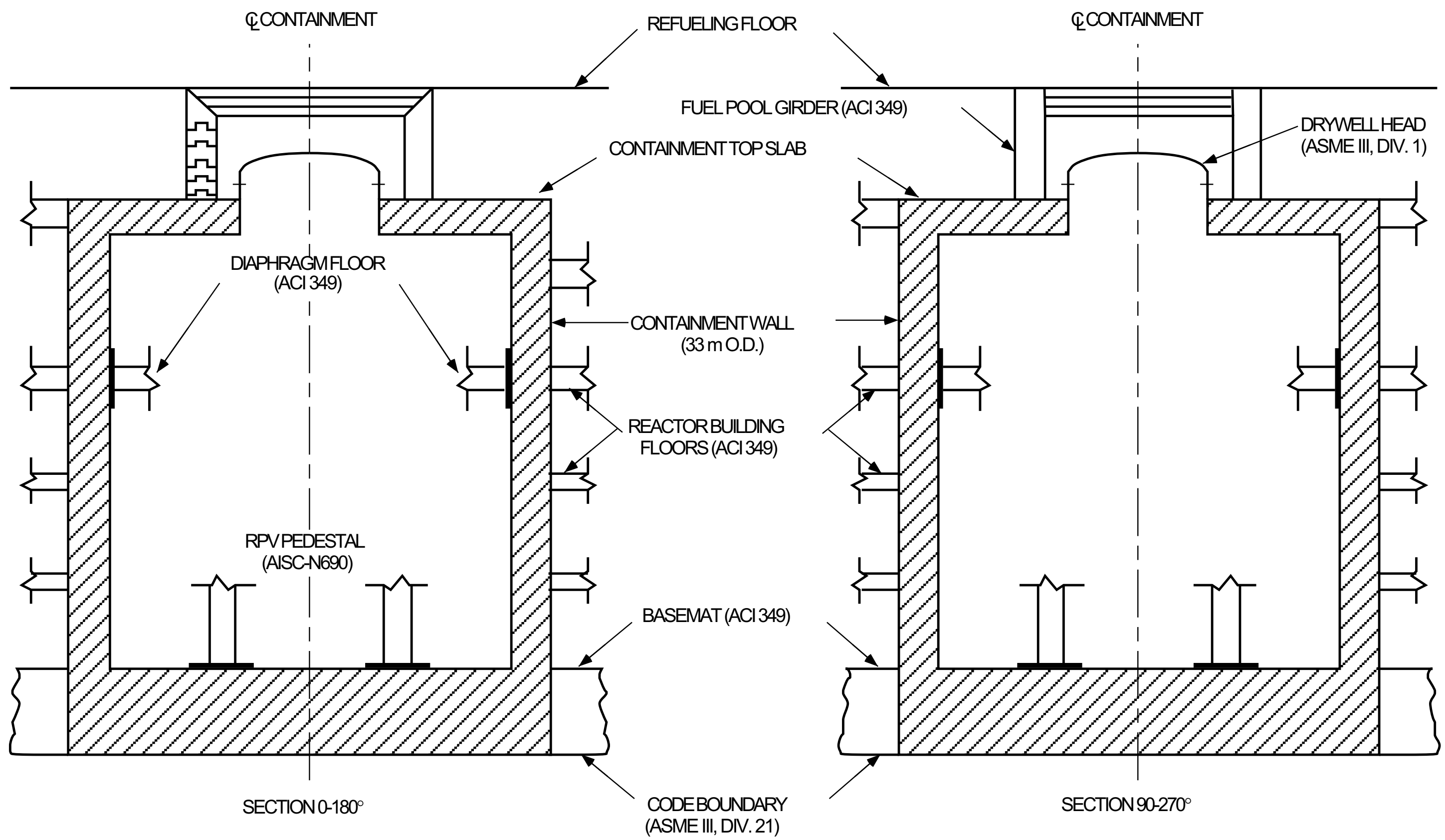

Figure 3H.1-2 Containment Structure ASME Code Jurisdictional Boundary

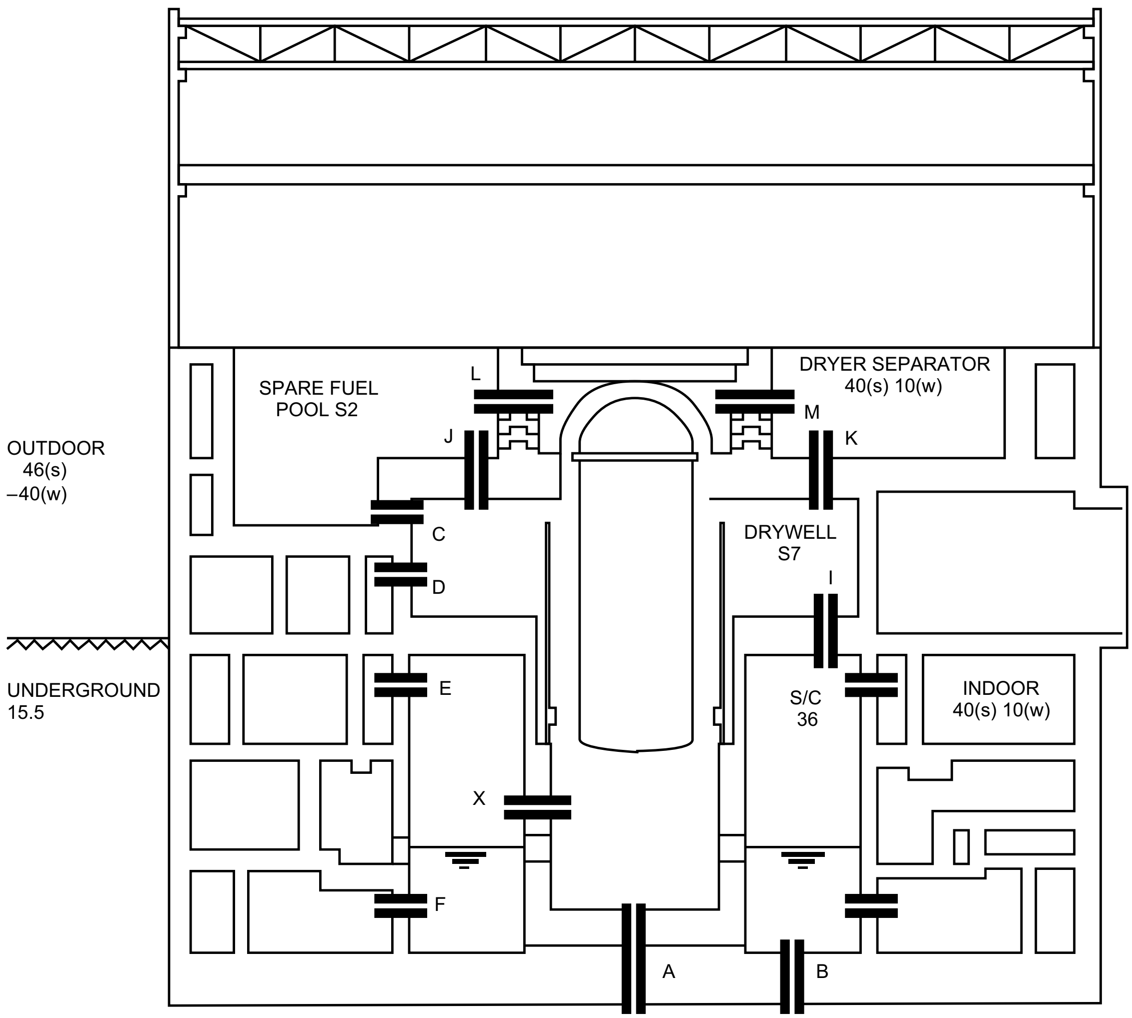

Figure 3H.1-3 Normal Operating Temperature (°C) and Sections Location for Thermal Distribution Analysis

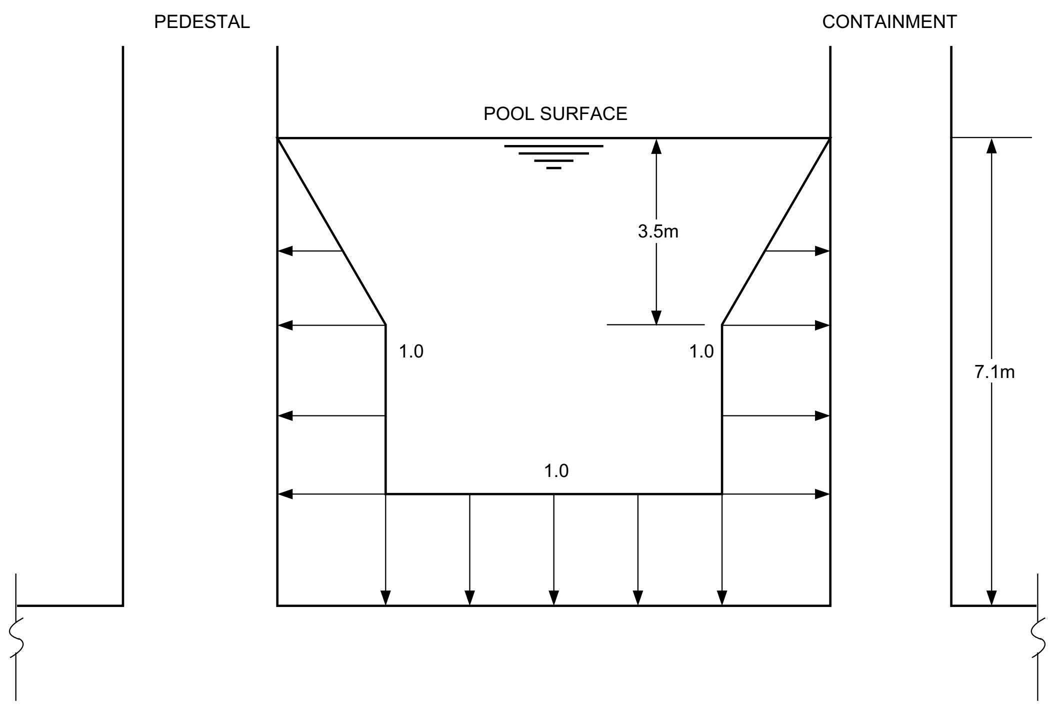

Figure 3H.1-4 Distribution of Condensation-Oscillation (CO) Pressure

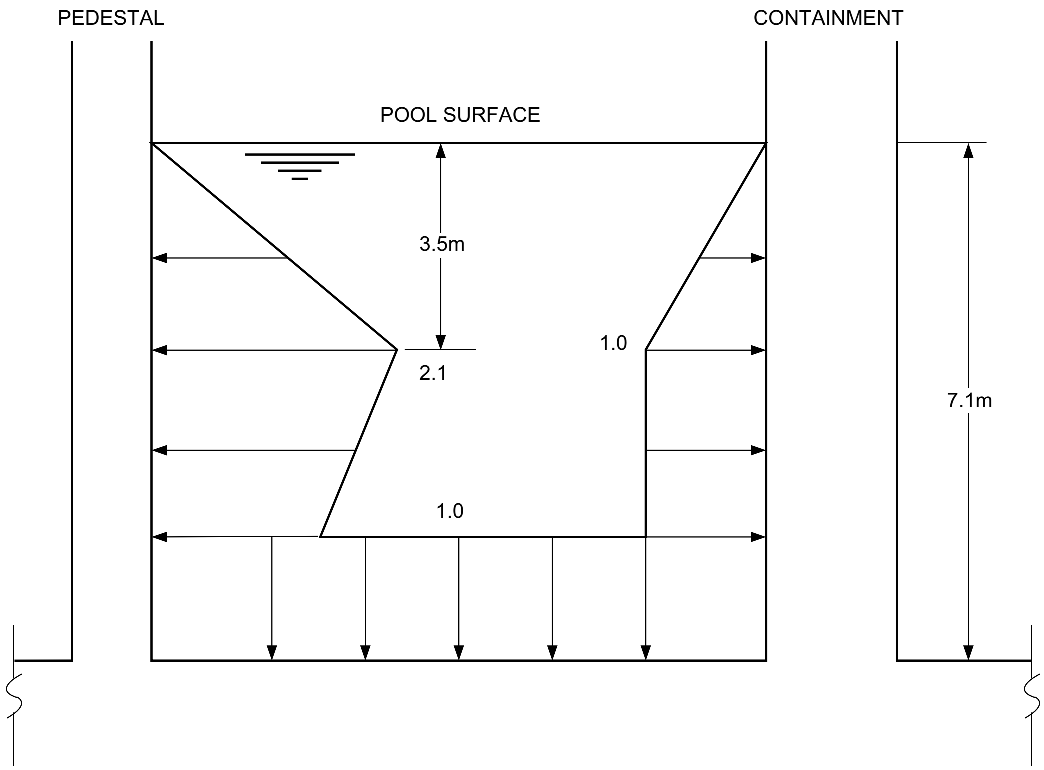

Figure 3H.1-5 Distribution of Chugging Pressure

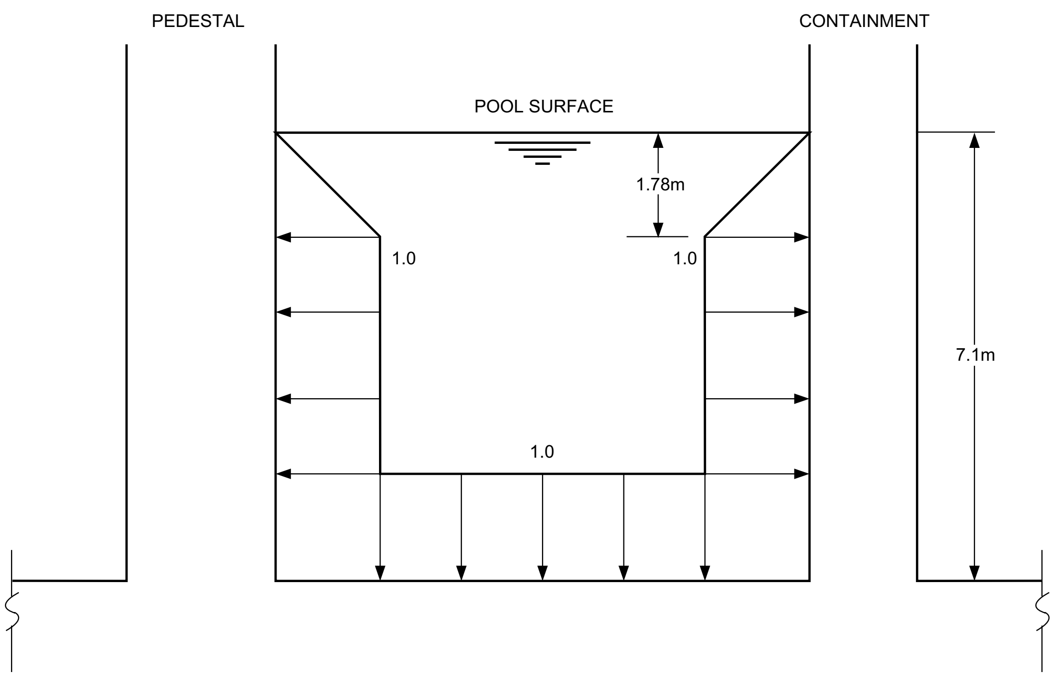

Figure 3H.1-6 Distribution of Safety-Relief Valve (SRV) Actuation Pressure

Figure 3H.1-7 Not Used

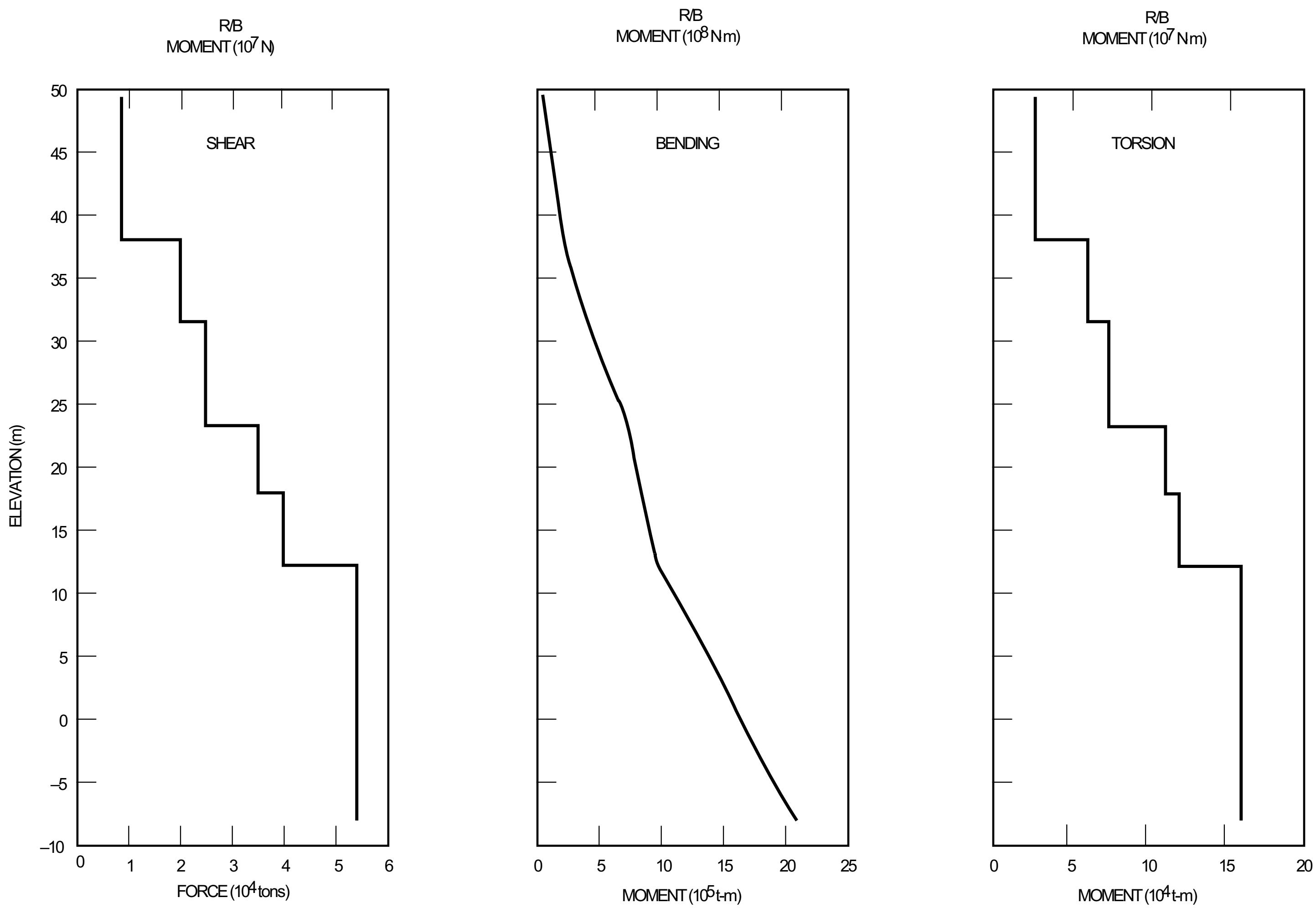

Figure 3H.1-8 Design Seismic Shears and Moments for Reactor Building Outer Walls

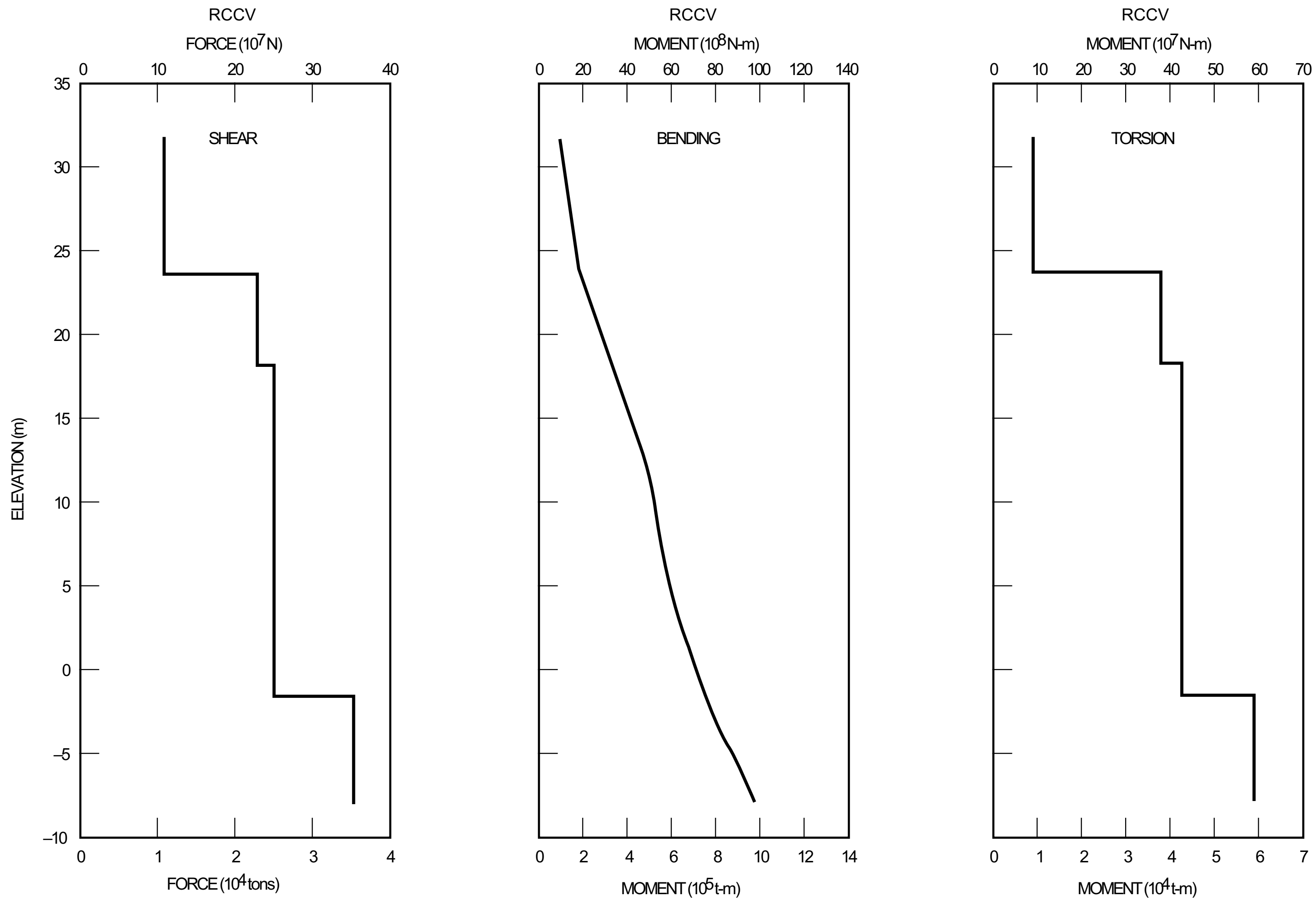

Figure 3H.1-9 Design Seismic Shears and Moments for RCCV

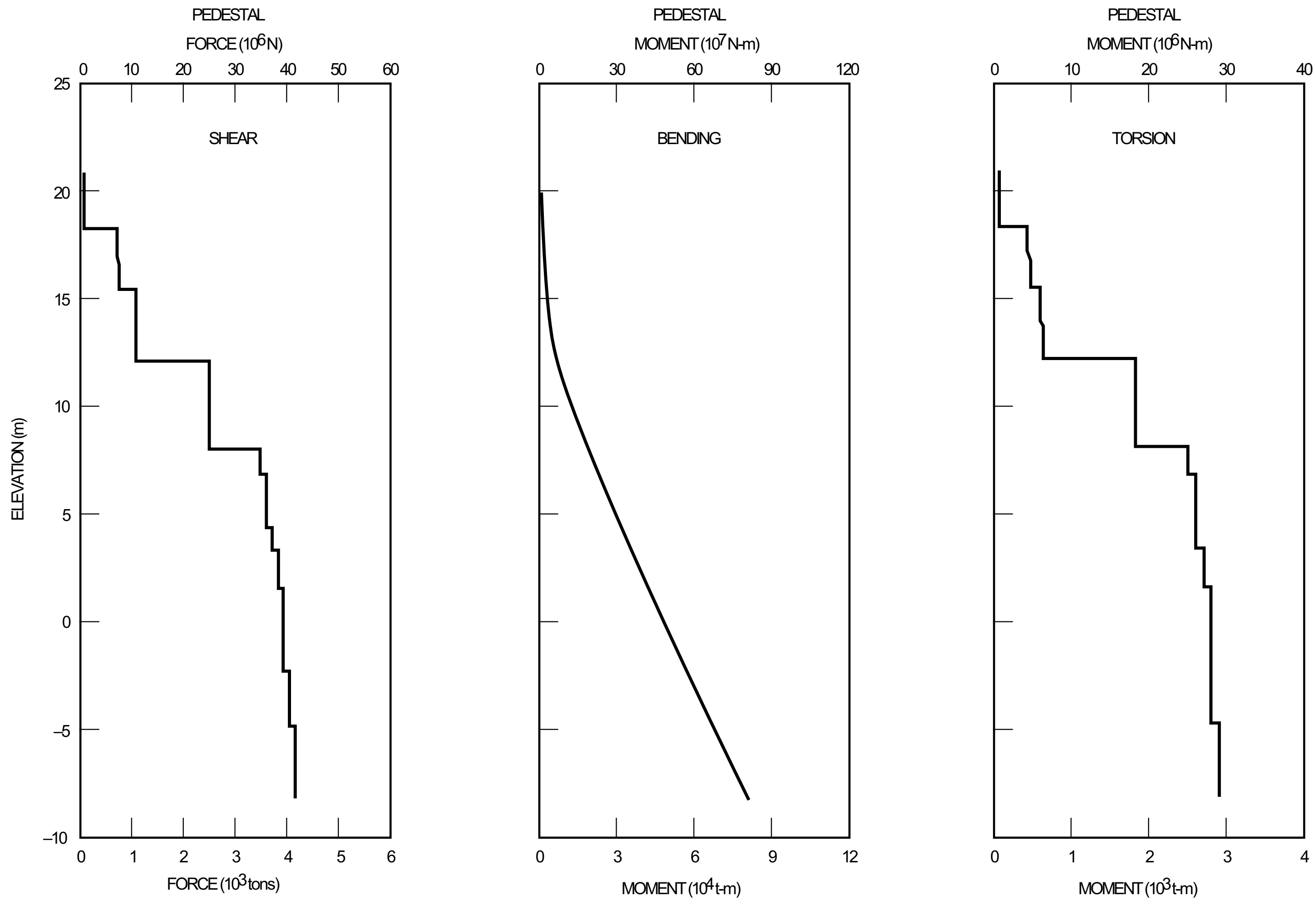

Figure 3H.1-10 Design Seismic Shears and Moments for RPV Pedestal Reactor Shield Wall

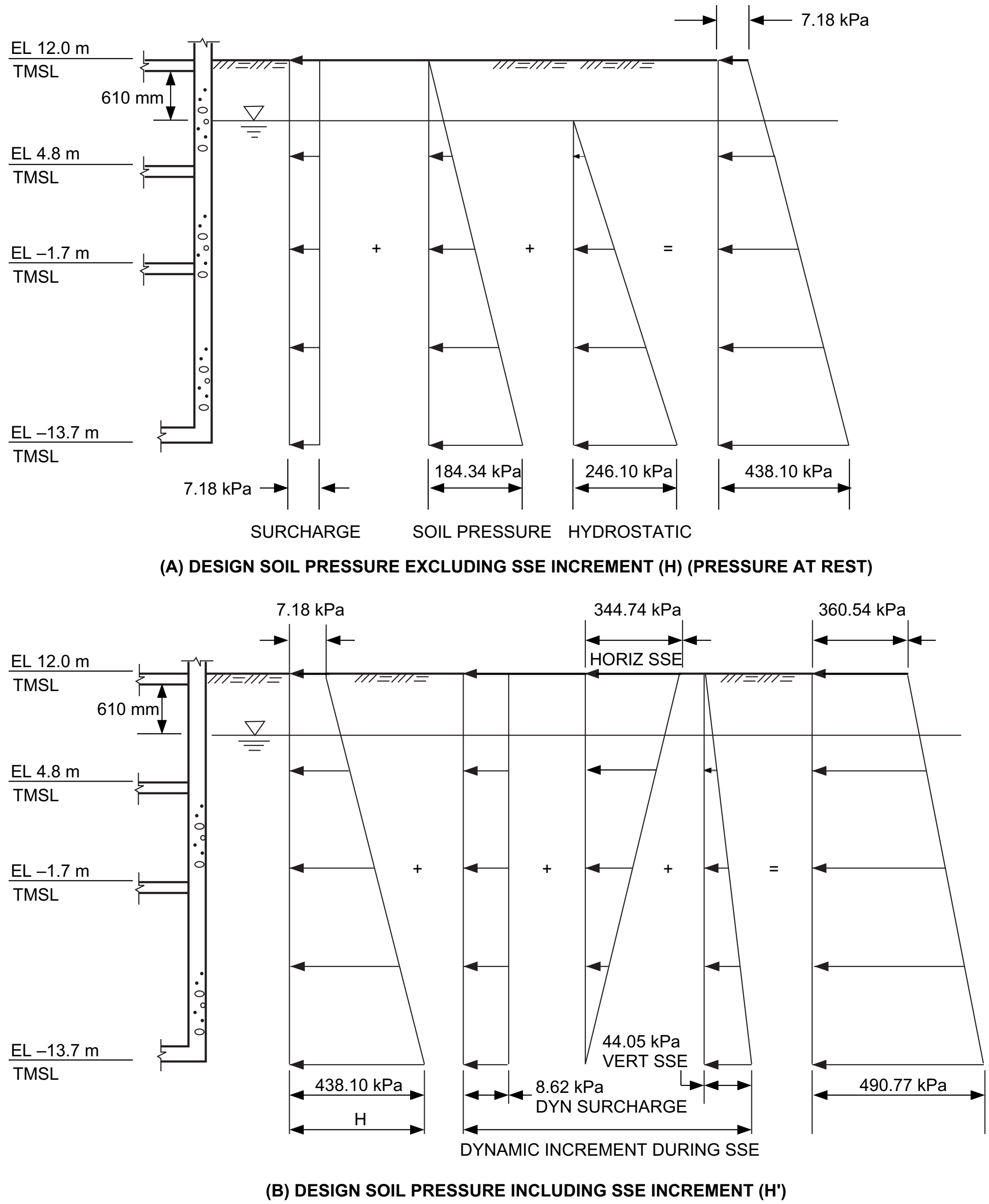

Figure 3H.1-11 Design Lateral Soil Pressures for RB Outerwalls

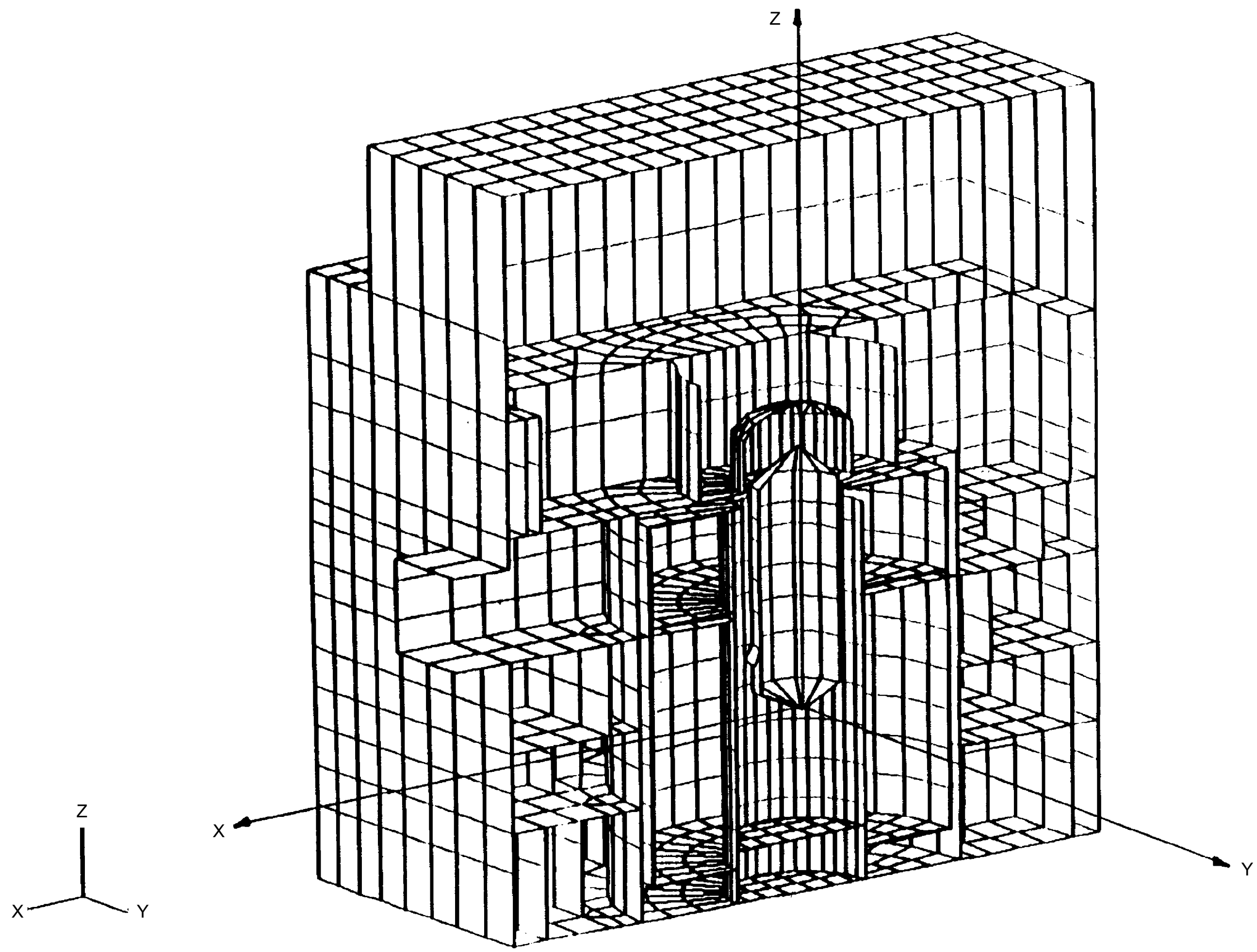

Figure 3H.1-12 F.E.M. Isometric View of Model Representing Half of Structure

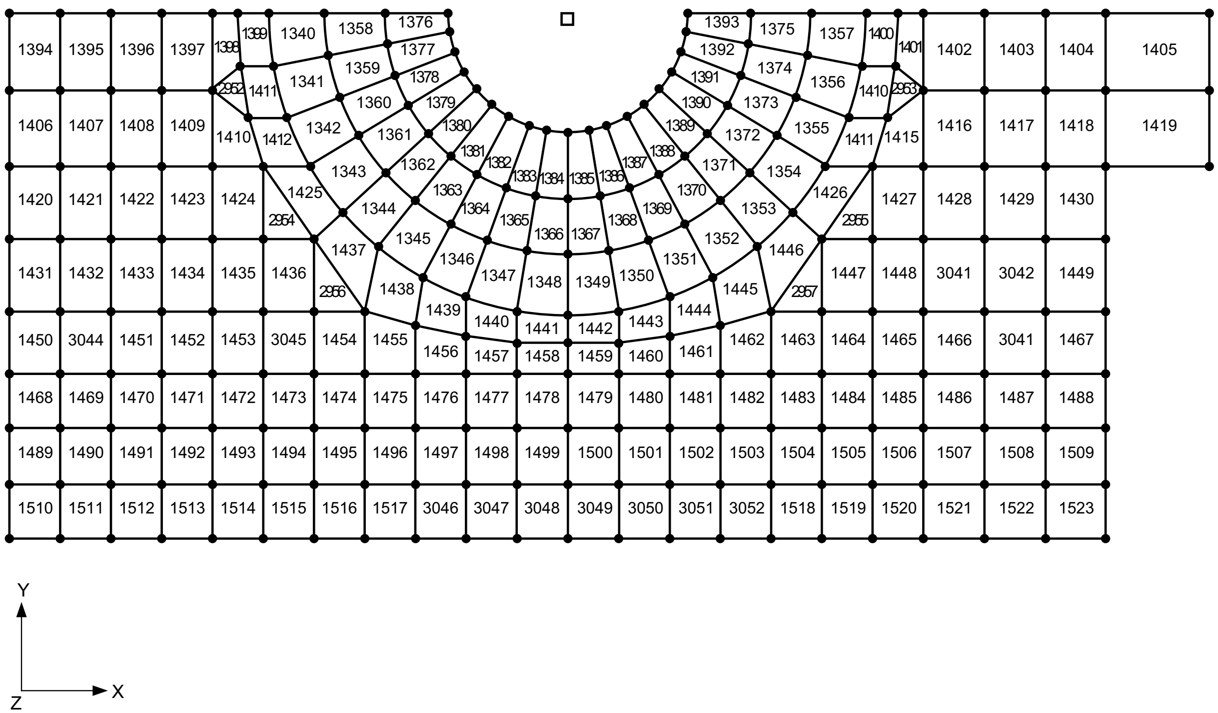

Figure 3H.1-13 F.E.M. Location of Elements at Diaphragm Floor Elevation

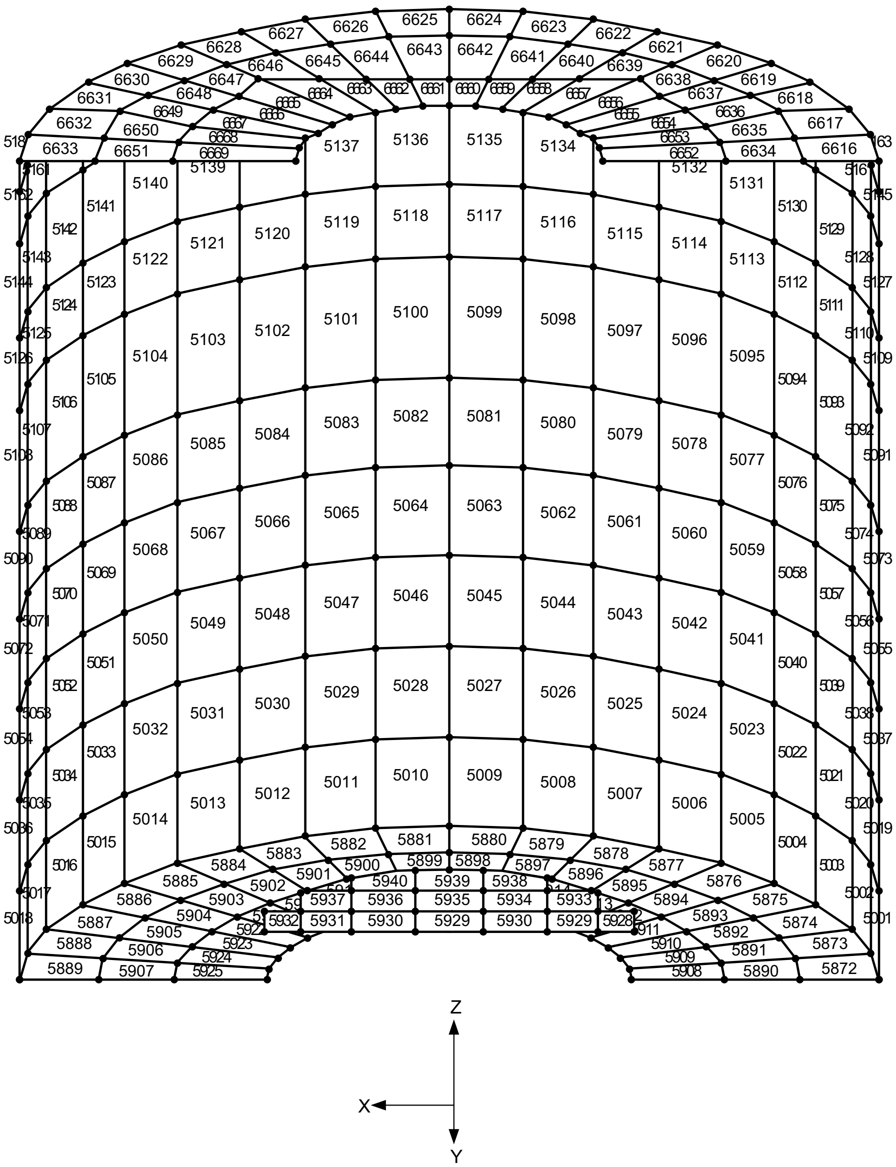

Figure 3H.1-14 F.E.M. Isometric View of Liner Plate

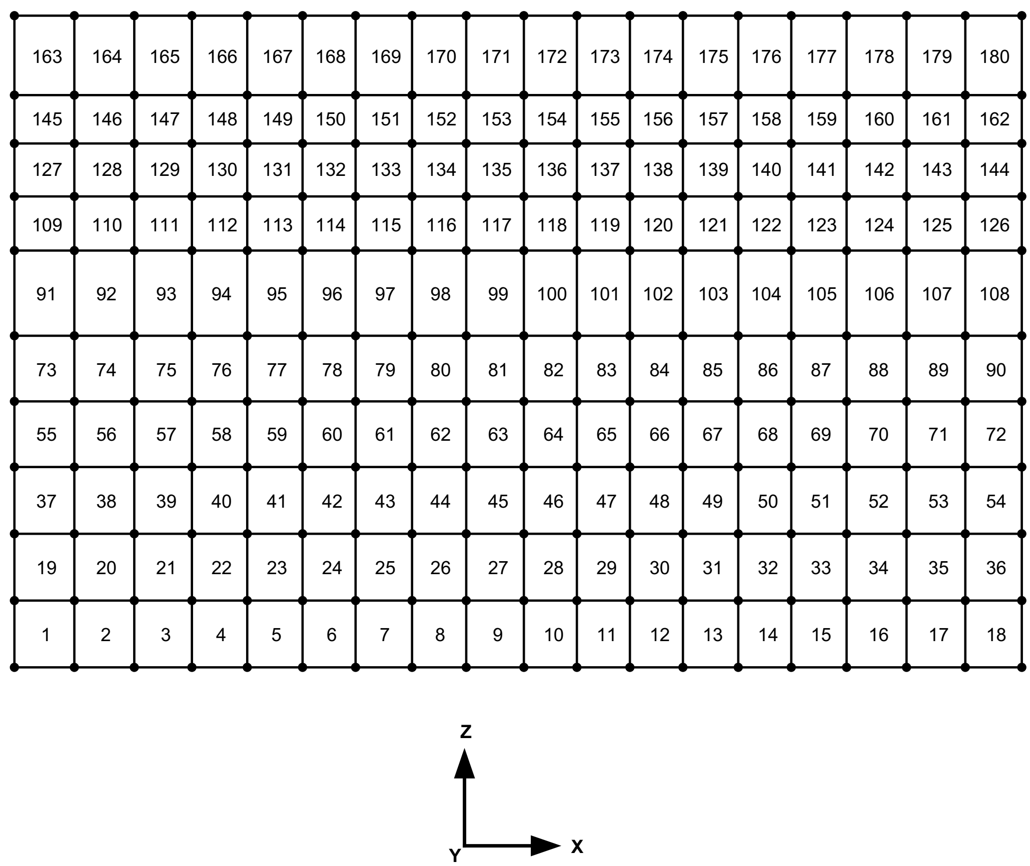

Figure 3H.1-15 F.E.M. Developed Elevation of RCCV Wall

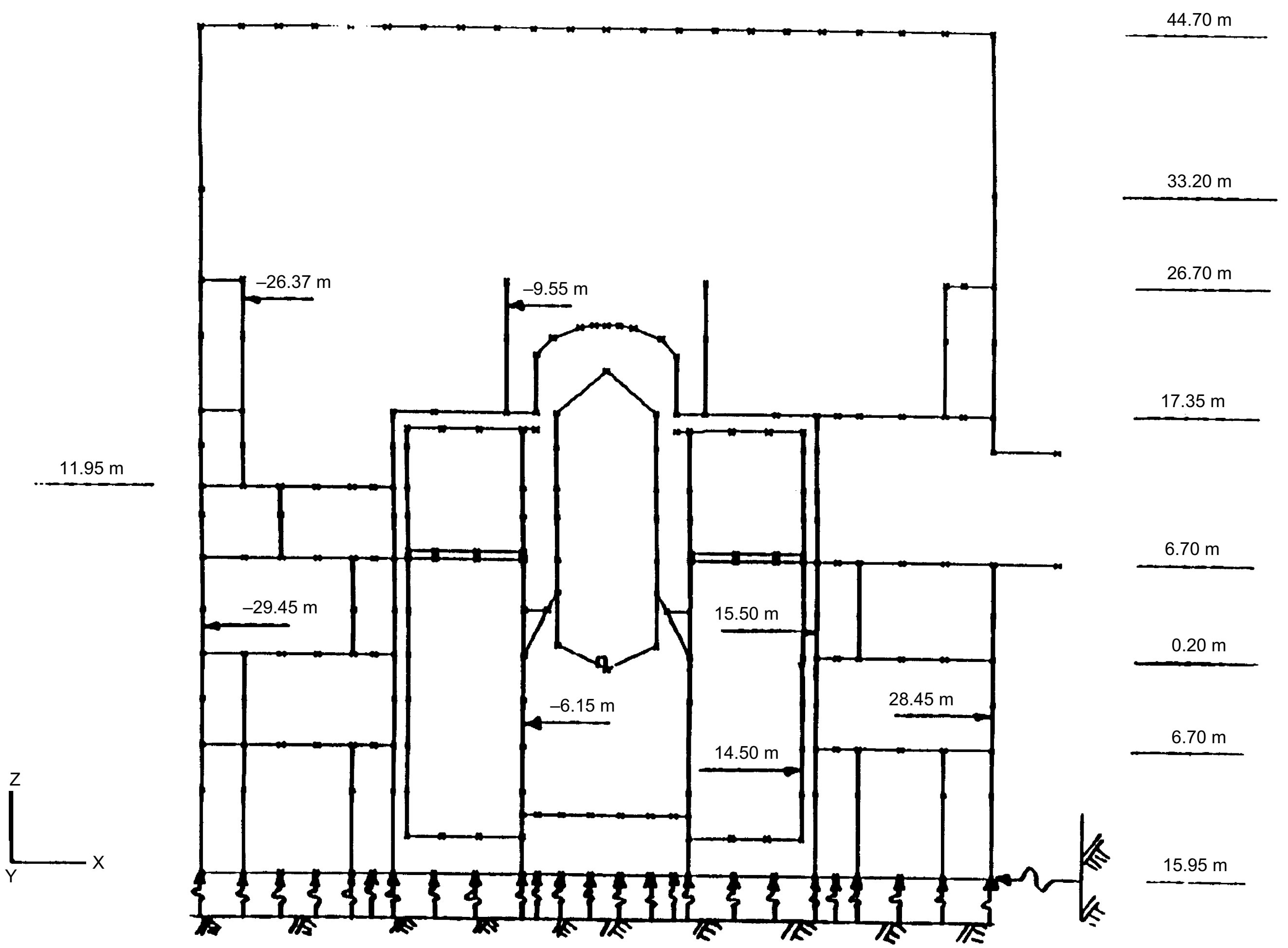

Figure 3H.1-16 Section 0°–180° Soil Springs

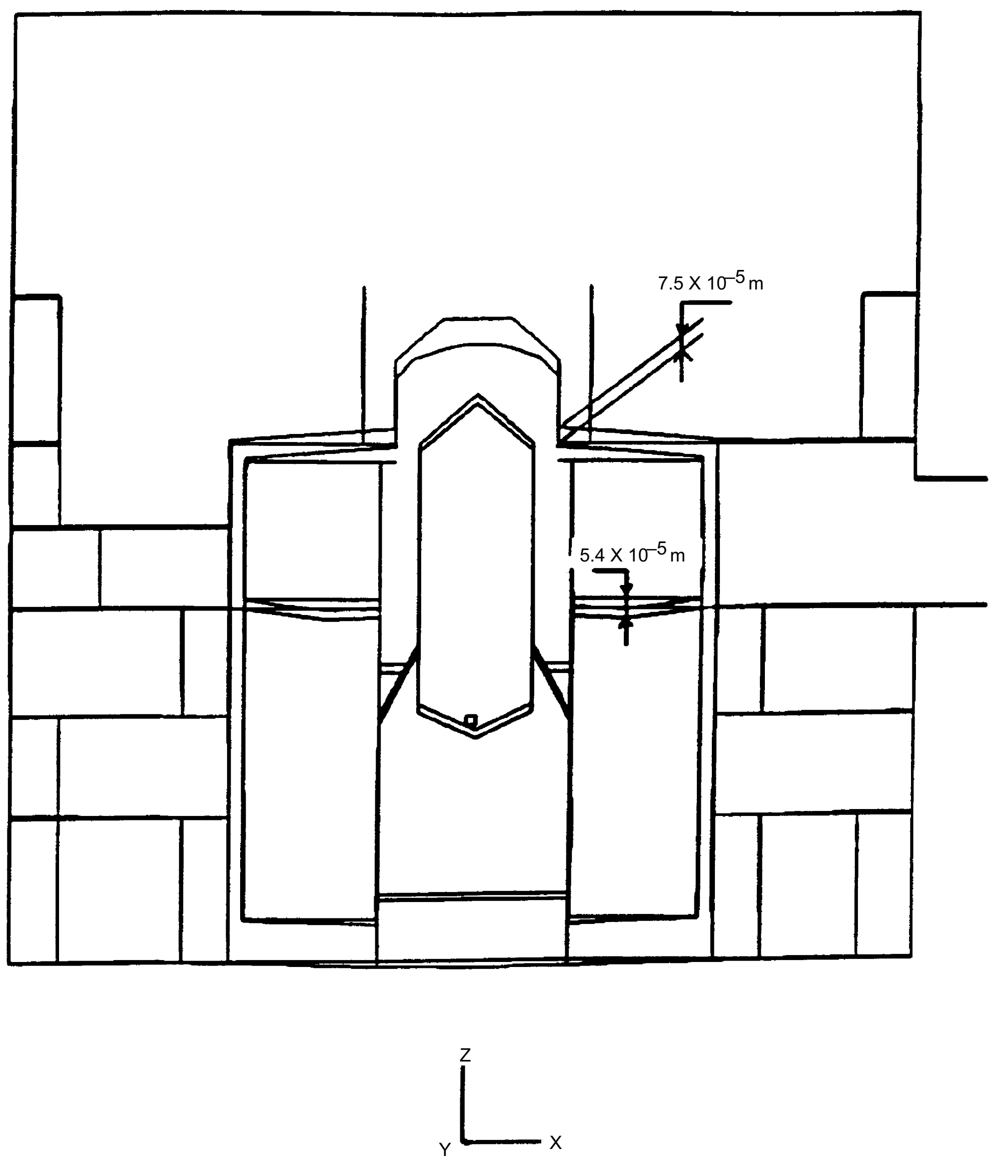

Figure 3H.1-17 Deformed Shape—Drywell Pressure (6.9 kPa)

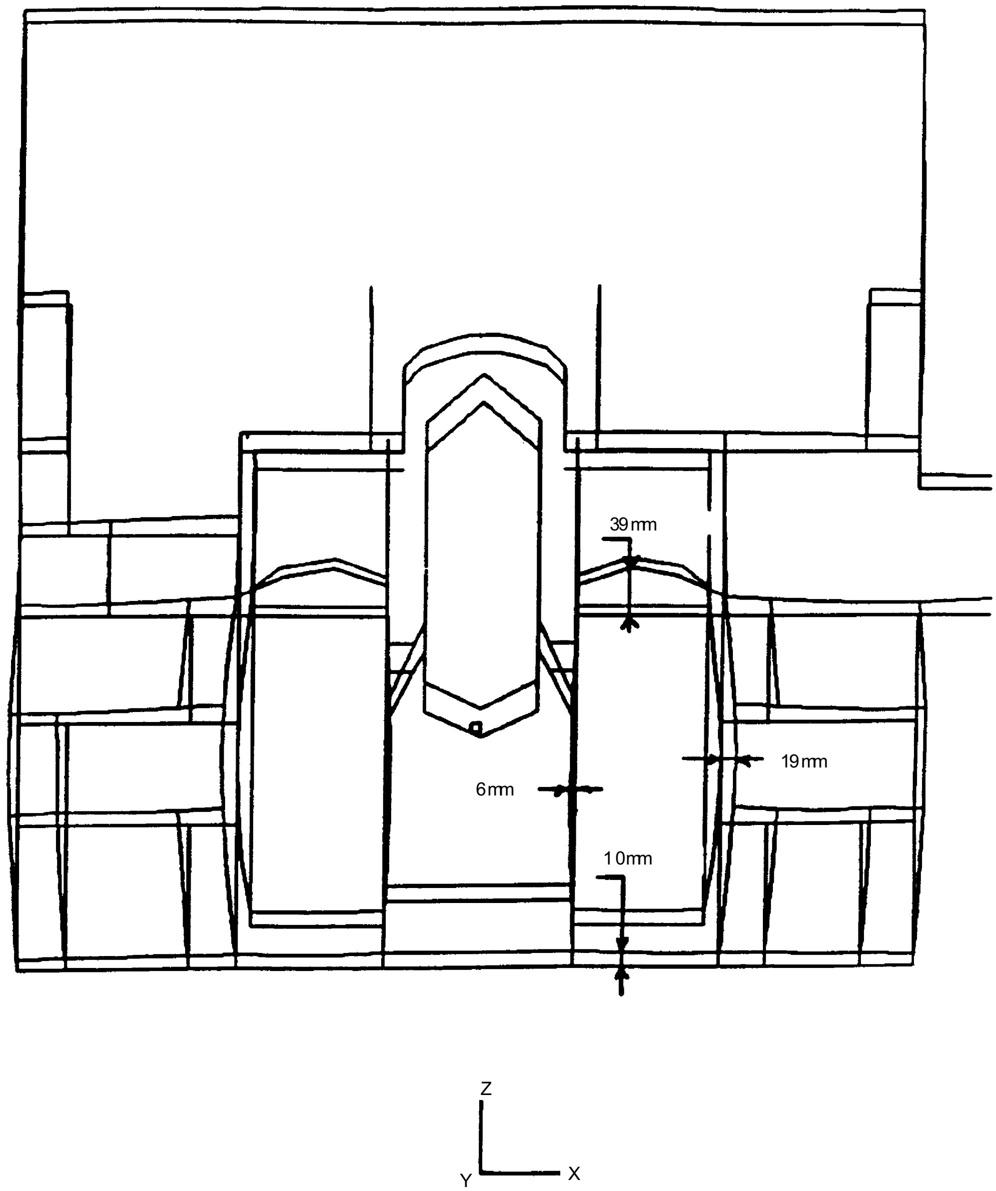

Figure 3H.1-18 Deformed Shape—Wetwell Pressure (6.9 kPa)

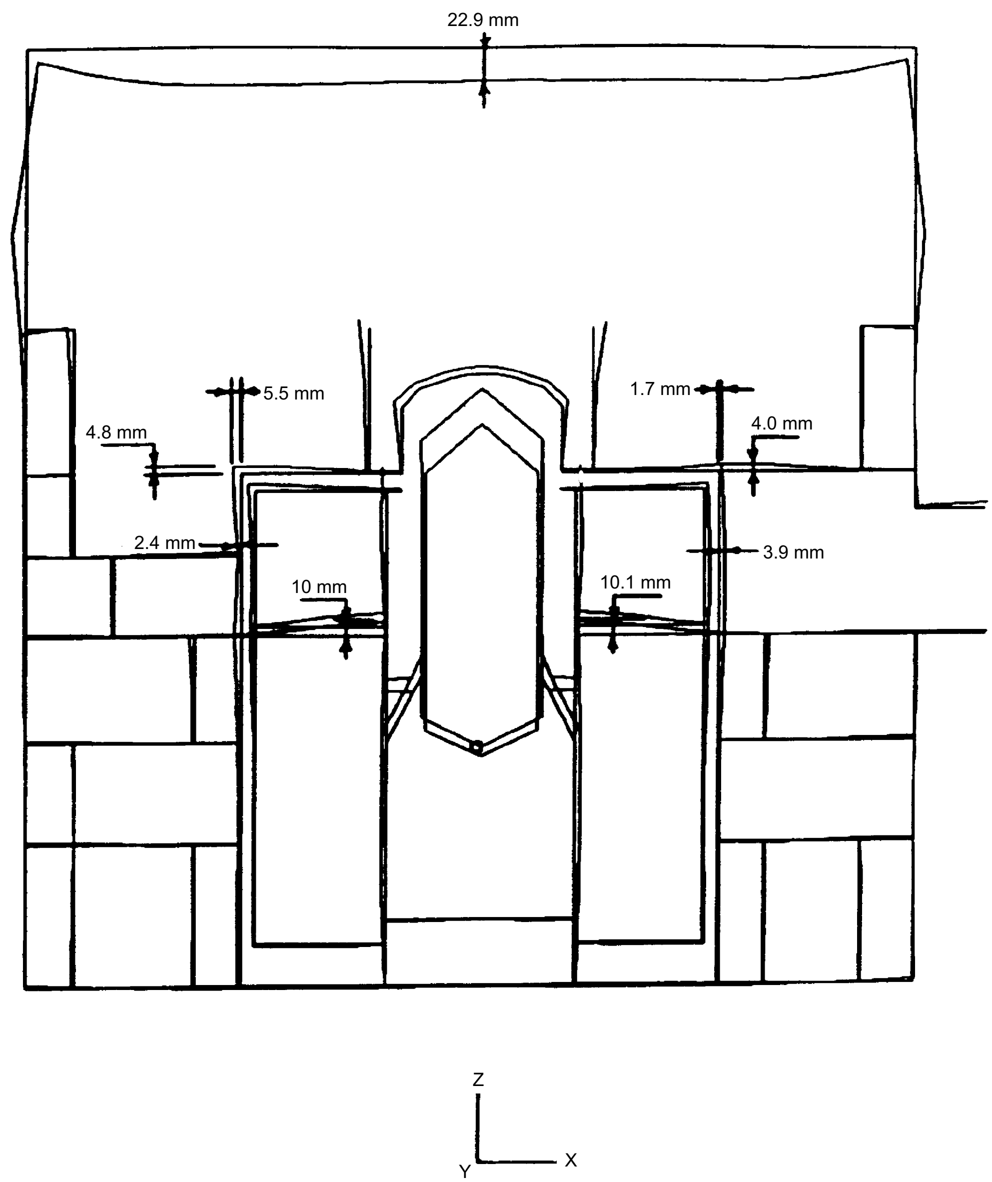

Figure 3H.1-19 Deformed Shape—Thermal Load (6 Hours)

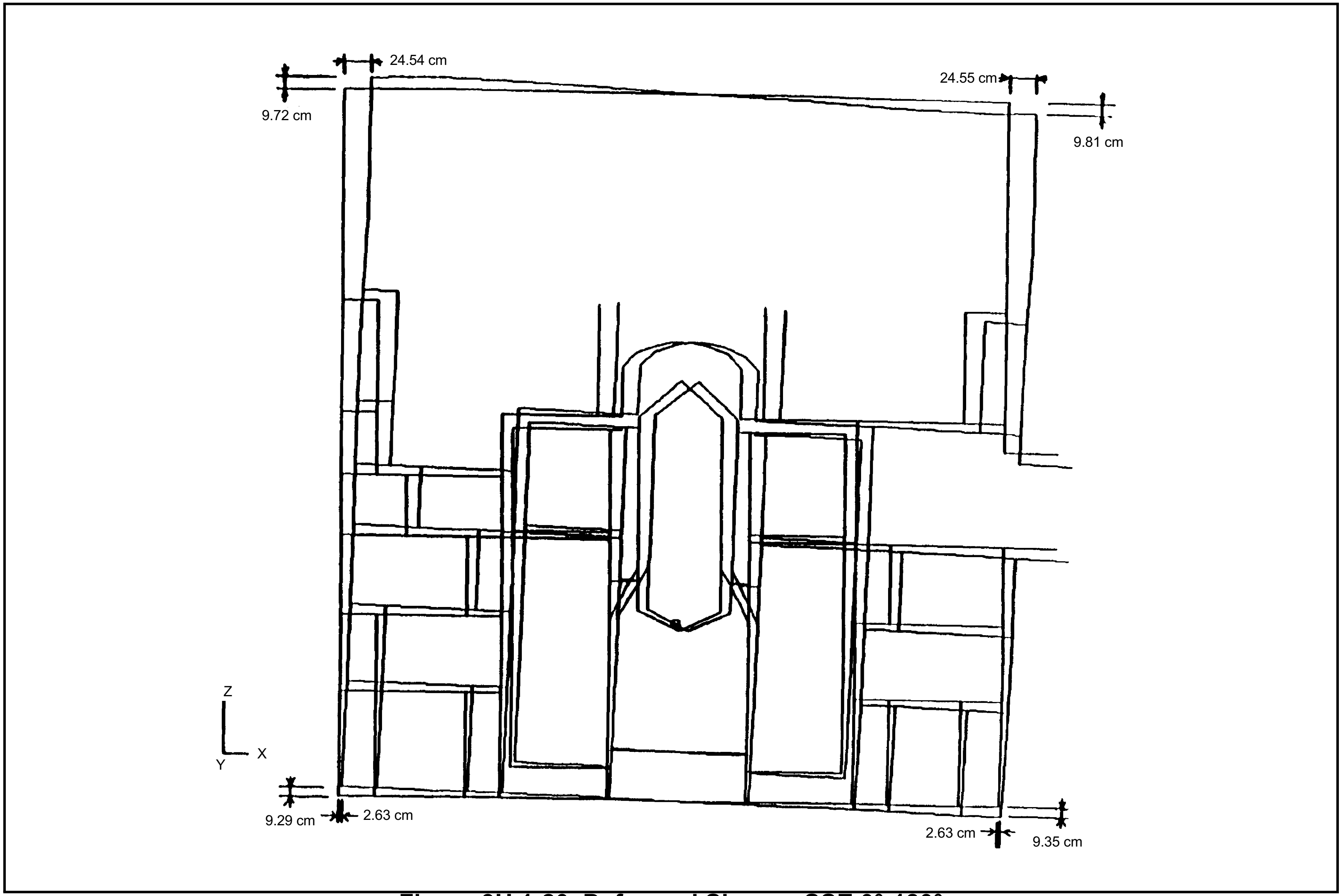

Figure 3H.1-20 Deformed Shape—SSE 0°-180°

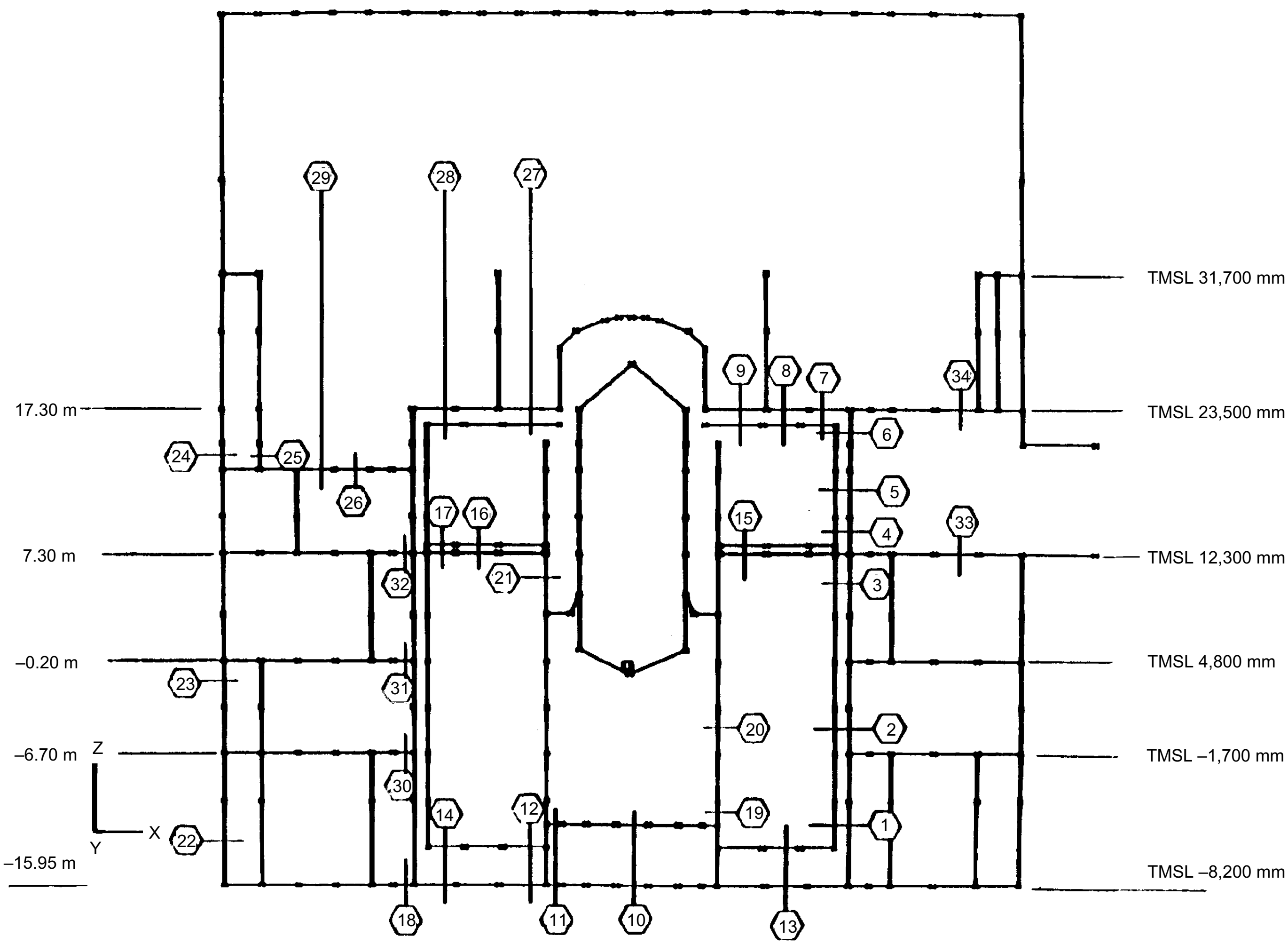

Figure 3H.1-21 Section Considered for Analysis

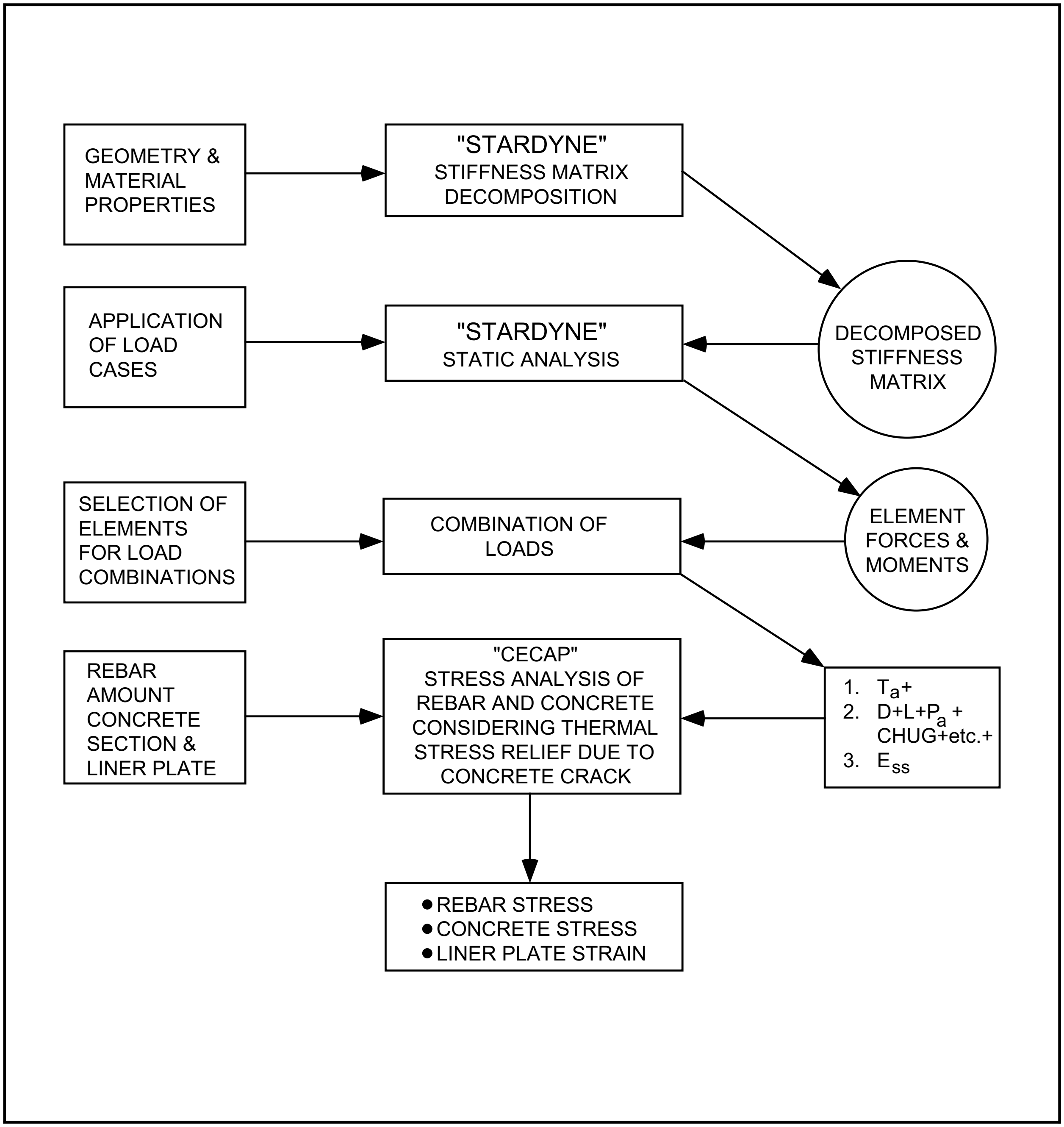

Figure 3H.1-22 Flow Chart for Structural Analysis and Design

The following figures are not used in the DCD:

Figure 3H.1-24

Figure 3H.1-25

Figure 3H.1-26

Figure 3H.1-27

The following figures are located in Chapter 21:

Figure 3H.1-23 Reactor Building Reactor Cavity Shield Blocks

Figure 3H.1-28 Configuration of RPV Pedestal

Figure 3H.1-29 Rebar Arrangement of F/P Girder and Slab (1/2)

Figure 3H.1-30 Containment Structure Wall Reinforcement

Figure 3H.1-31 Containment Structure Opening Reinforcement

Figure 3H.1-32 Containment Structure Opening Reinforcement

Figure 3H.1-33 Containment Structure Top Slab Reinforcement

Figure 3H.1-34 Reactor Building Foundation Reinforcement (Sheet 1)

Figure 3H.1-35 Reactor Building Foundation Reinforcement (Sheet 2)

Figure 3H.1-36 Diaphragm Floor Reinforcement

Figure 3H.1-37 List of Seismic Wall Sections

3H.2 Control Building

3H.2.1 Objective and Scope

The objective of this subsection is to document the structural design and analysis of the ABWR Control Building. The scope includes the design and analysis of the structure for the normal, severe environmental, extreme environmental, abnormal, and construction loads.

This subsection addresses all applicable items included in Appendix C to USNRC Standard Review Plan, NUREG-0800, Section 3.8.4.

3H.2.2 Conclusions

The following are the major summary conclusions on the design and analysis of the Control Building:

- Based on the design drawings identified in Subsection 3H.2.5, stresses in concrete, reinforcement, structural steel, and steel deck are less than the allowable stresses per the applicable codes listed in Subsection 3H.2.4.1.
- The factors of safety against flotation, sliding, and overturning of the structure under various loading combinations are higher than the required minimum.
- The thickness of the roof slabs and exterior walls are more than the minimum required to preclude penetration, perforation, or spalling resulting from impact of design basis tornado/hurricane missiles.
- The building has been evaluated for the design basis tornado/hurricane. Welded studs are provided for the roof structural steel to provide required resistance for negative pressure.

3H.2.3 Structural Description

The Control Building is a Seismic Category I structure which houses the control room, computer facility, electrical panels, electrical switchgear, Reactor Building cooling water facilities, electrical battery and motor control center (MCC) rooms, and HVAC facilities. The main steam tunnel from the Reactor Building to the Turbine Building is located in the top portion of the Control Building.

The Control Building is a 56m long xþ24m wide structure that is 30.4m high above the top of the 3m thick base mat. It consists of six floors, four of which are below grade. The total building embedment is 23.2m. It is a reinforced concrete structure consisting of walls and slabs and is supported by a mat foundation. Steel framing is used to support the slabs for construction loads. Steel deck is used as formwork to support the slabs during construction.

Figure 1.2-1 shows the location of the Control Building in relation to other plant structures. Figures 1.2-14 through 1.2-22 show the arrangement of the building.

3H.2.4 Structural Design Criteria

3H.2.4.1 Design Codes

Reinforced concrete is designed by the strength design method [*in accordance with ACI 349*][3] as augmented by USNRC Regulatory Guide 1.142.

Structural steel is designed by the allowable stress design method [*in accordance with the ANSI/AISC-N690.*]*

3H.2.4.2 Site Design Parameters

The following are some of the key design parameters.

3H.2.4.2.1 Soil Parameters

- Minimum shear wave velocity: 305 m/s
- Poisson ratio: 0.3 to 0.38
- Unit weight: 1.9 to 2.2 t/m^3
- Liquefaction potential: None
- Minimum Static Soil Bearing Capacity Demand: ≥ 718.20 kPa

3H.2.4.2.2 Design Ground Water Level

Design ground water level is at 0.61m below grade level.

3H.2.4.2.3 Design Flood Level

Design flood level is at 0.305m below grade level.

3H.2.4.2.4 Maximum Snow Load

Design snow load is 2.39 kPa.

3H.2.4.2.5 Maximum Rainfall

Design rainfall is 493 mm/h. Roof parapets are furnished with scuppers to supplement roof drains, or, are designed without parapets so that excessive ponding of water cannot occur. Such roof design meets the provisions of ASCE 7, Section 8.0.

3 See Subsection 3.8.3.2.

3H.2.4.2.6 Design Temperatures

- Maximum: 38°C
- Minimum: –23°C
- Stress Free Temperature: 15.1°C

3H.2.4.3 Design Loads and Load Combinations

3H.2.4.3.1 Normal Loads

Normal loads are those which are encountered during normal plant startup, operation, and shutdown.

3H.2.4.3.1.1 Dead Loads (D)

Dead loads include weight of the structure, permanent equipment, and other permanent static loads. An additional minimum allowance of 2.39 kPa uniform load is made for dead loads due to piping, raceways and HVAC duct work.

Figure 3H.2-11 shows the dead loads used in the design exclusive of concrete weight.

3H.2.4.3.1.2 Live Loads (L and Lo)

Live loads include floor and roof area loads, movable loads and laydown loads. The live loads, L, used in the design are as follows

- Roof 1.42 kPa[1]
- Floor at elevation 18250 mm TMSL 4.79 kPa
- Floor at elevation 3500 mm TMSL 19.15 kPa
- All other floors 11.97 kPa

1 Snow load controls design of the roof.

Figure 3H.2-12 shows the live loads, L, used in the design.

For computation of global seismic loads, the value of live load is limited to the expected live load, L_o, during normal plant operation, which was taken as 25% of the above live loads L_o.

However, in the load combinations involving seismic loads, the valves of L_o used for the design are as follows:

■ Roof	0.36 kPa[1]
■ Floor at elevation 18250 mm TMSL	0
■ Floor at elevation 3500 mm TMSL	14.37 kPa
■ All other floors	7.18 kPa

1 Snow load controls design of the roof.

Figure 3H.2-13 shows the live loads L_o used in the design.

3H.2.4.3.1.3 Snow Load

A value of 2.39 kPa is used for snow live load (L). The snow load (L_o) is reduced to 75%, when it is combined with seismic loads. Figures 3H.2-12 and 3H.2-13 show the snow loads used in the design.

3H.2.4.3.1.4 Lateral Soil Pressures (H and H')

The following soil parameters are used in the computation of lateral soil pressures:

■ Dry unit weight:	1.9 to 2.2 t/m^3
■ Shear wave velocity:	305 m/s
■ Internal friction angle:	30° to 40°

A uniform surcharge load of 0.215 MPa is estimated from the Turbine Building. Conservatively, the same surcharge load is used in the computation of lateral soil pressures on each exterior wall. This adequately compensates for surcharge from the more lightly loaded Service Building.

The dynamic lateral soil pressure increment due to SSE is calculated per the Mononobe-Okabe method.

The design of the structure is based on the at rest soil pressure. Figure 3H.2-14 shows the at rest lateral soil pressure H (excluding the dynamic soil pressure increment) and H' (including the dynamic soil pressure increment). Active and passive soil pressures are used in the evaluation of the structure. Figure 3H.2-15 shows these pressures.

3H.2.4.3.1.5 Normal Thermal Load (T_o)

The normal operating temperatures used in the design are as follows:

- Inside steam tunnel: 57°C
- Below steam tunnel: 18°C
- On either side of steam tunnel: 10°C
- Outside Control Building: 38°C Max.
  –23°C Min.

3H.2.4.3.2 Severe Environmental Loads

The severe environmental load considered in the design is that generated by wind. The following parameters are used in the computation of wind loads:

- Basic wind speed: 177 km/h
- Exposure: D
- Importance factor: 1.11

Wind loads are calculated per the provisions of ASCE 7. Figure 3H.2-16 shows wind loads used in the design.

3H.2.4.3.3 Extreme Environmental Loads

Extreme environmental loads consist of loads generated by the tornado and safe shutdown earthquake and, for extreme wind loads, the design basis tornado whose winds and missile loads bound the design basis hurricane, as specified in 3H.1.4.3.1.

3H.2.4.3.3.1 Tornado Loads (W_t)

The following tornado load effects are considered in the design:

- Wind pressure (W_w)
- Differential pressure (W_p)
- Missile impact (W_m)

Parameters used in computation of tornado loads are as follows:

- Maximum wind speed: 483 km/h
- Maximum rotational speed: 386 km/h
- Maximum translational speed: 97 km/h
- Radius of maximum rotational speed: 45.7m
- Differential pressure: 13.83 kPa
- Pressure differential rate: 8.28 kPa/s
- Missile spectrum: See Table 2.0-1

(1) Tornado Wind Pressure (W_w)

Tornado wind pressures are computed using the procedure described in Bechtel Topical Report BC-TOP-3-A. The topical report uses the methods and procedures contained in ASCE 7 with the following exceptions:

(a) Wind velocity and wind pressures are constant with height.

(b) Wind velocity and wind pressures vary with horizontal distance from the center of the tornado.

(c) Wind pressures are determined by multiplying the wind pressure at the radius of maximum wind by the size coefficient.

(d) The gust factor is unity.

(e) ASCE criteria for determining wind pressures on building components and cladding are not applicable.

(2) Tornado Differential Pressure (W_p)

The Control Building is designed as not vented, i.e. without major openings. The differential pressure causes suction on exterior walls and roof of the structure.

(3) Tornado Missile Impact (W_m)

Tornado missile impact effects on the structure can be assessed in the following two ways:

(a) Local damage in terms of penetration, perforation, and spalling.

(b) Structural response in terms of deformation limits, strain energy capacity, structural integrity and structural stability.

For the Control Building, the tornado missile effect has been assessed in terms of the local damage.

(4) Tornado Load Combinations

Tornado load effects are combined per USNRC Standard Review Plan, NUREG-0800 Section 3.3.2 as follows:

$$W_t = W_w$$

$$W_t = W_p$$

$$W_t = W_m$$

$$W_t = W_w + 0.5\ W_p$$

$$W_t = W_w + 1.0\ W_m$$

$$W_t = W_w + 0.5\ W_p + 1.0\ W_m$$

Figure 3H.2-17 shows distribution of tornado loads due to combined W_w and W_p.

3H.2.4.3.3.2 Safe Shutdown Earthquake Loads (E')

The SSE loads are applied in three directions, viz. the two horizontal directions and the vertical direction. The total structural response is predicted by combining the applicable maximum co-directional responses by "the square root of the sum of the squares (SRSS)" method.

The SSE loads are based on a free-field peak ground acceleration value of 0.3g at plant grade elevation. The loads consist of story shears, torsional moments, and overturning moments. The loads are distributed to the resisting walls in proportion to their rigidities. The SSE loads for the Control Building are provided in Table 3H.2-1.

3H.2.4.3.4 Abnormal Loads

Abnormal loads are loads generated by postulated accidents.

3H.2.4.3.4.1 Accident Pressure Load (P_a)

This load is caused by a break in the main steam line. The pressure is equal to 150.31 kPa including a dynamic load factor of 2 and is applied to the walls, floor and ceiling (i.e. underside of roof) of the steam tunnel as a static load.

Figure 3H.2-18 shows accident pressure load used in the design.

3H.2.4.3.4.2 Accident Thermal Load (T_a)

Due to the events short duration, T_o values discussed in Subsection 3H.2.4.3.1.5 are used for T_a.

3H.2.4.3.4.3 Accident Hydrostatic Load (F_a)

This load is caused by accidental flooding of any one of the three compartments of basement (EL −8200 mm TMSL). The design flood water level reaches the ceiling. For the purpose of design, the central compartment is flooded. The flood causes hydrostatic pressure on walls on grid lines A, D, 3, and 5, and vertical pressure on the base mat.

Figure 3H.2-19 shows accident hydrostatic load used in the design.

3H.2.4.3.5 Construction Loads

3H.2.4.3.5.1 Steel Deck

The steel deck supporting wet concrete is designed for the weight of concrete plus 2.39 kPa uniformly distributed load.

3H.2.4.3.5.2 Structural Steel Framing

The steel beams supporting the deck are designed for the weight of concrete plus 4.81 kPa uniformly distributed and a 22.26 kN concentrated load placed anywhere on the span of major beams to maximize moment and shear.

3H.2.4.3.6 Load Combinations

The load combinations and structural acceptance criteria are consistent with the provisions of Section 3.8.4 of USNRC Standard Review Plan NUREG-0800.

3H.2.4.3.6.1 Notations

S = Allowable stress for allowable stress design method

U = Required strength for strength design method

D = Dead load

L = Live load or snow load

L_o = Live load or snow load concurrent with earthquake

H = Lateral soil pressure

H' = Lateral soil pressure including dynamic increment

W = Wind load

W_t = Tornado load

E’ = Safe shutdown earthquake load

P_a = Accident pressure load

F_a = Accident hydrostatic load

T_o = Normal thermal load

T_a = Accident thermal load

3H.2.4.3.6.2 Structural Steel

S = D + L

S = D + L + W

1.6S = D + L + W_t

1.6S = D + L_o + E’

3H.2.4.3.6.3 Reinforced Concrete

U = 1.4D + 1.7L + 1.7H

U = 1.4D + 1.7L + 1.7H + 1.7W

U = 1.2D + 1.7W

U = 0.75 (1.4D + 1.7L + 1.7H + 1.7T_o)

U = 0.75 (1.4D + 1.7L + 1.7H +1.7W + 1.7T_o)

U = 1.0D + 1.0L + 1.0H + 1.0W_t + 1.0T_o

U = 1.0D + 1.0L + 1.0H + 1.5P_a + 1.0T_a

U = 1.0D + 1.0L_o + 1.0H'+ 1.0P_a + 1.0E’ + 1.0T_a

U = 1.0D + 1.0L_o + 1.0H'+ 1.0F_a + 1.0E’ + 1.0T_a

3H.2.4.4 Materials

Structural materials used in the design and their properties are as follows:

3H.2.4.4.1 Concrete

Concrete conforms to the requirements of ACI 349. Its design properties are:

- Compressive strength (f'c) =27.58 MPa
- Modulus of elasticity = 24.81 GPa
- Shear modulus = 10.59 GPa
- Poisson's ratio = 0.18

3H.2.4.4.2 Reinforcement

Deformed billet-steel reinforcing bars are considered in the design. Reinforcement conforms to the requirement of ASTM A615. Its design properties are:

- Yield strength = 414 MPa
- Tensile strength = 621 MPa

3H.2.4.4.3 Structural Steel

High strength, low-alloy structural steel conforms to ASTM A572, Grade 50. The steel design properties are:

- Yield strength = 345 MPa
- Tensile strength = 448 MPa

3H.2.4.4.4 Anchor Bolts

Material for anchor bolts conforms to the requirements of ASTM A36. Its design properties are:

- Yield strength = 248 MPa
- Tensile strength = 400 MPa

3H.2.4.5 Stability Requirements

The stability requirements are based on the provisions of Section 3.8.5 of the USNRC Standard Review Plan, NUREG-0800.

The following minimum factors of safety are provided against overturning, sliding and flotation:

L_o = Live load concurrent with earthquake (both cases of live load having its full value and being completed absent shall be considered)

F = Buoyant Force of Design Ground Water

F′ = Buoyant Force of Design Basis Flood

Load Combination	Overturning	Sliding	Flotation
D = F’	-	-	1.1
D + F + W + H	1.5	1.5	-
D + F + W_t + H	1.1	1.1	-
D + L_o + F + H’ + E’	1.1	1.1	-

The Control Building base shear and overturning moments for purpose of stability evaluation are given in Table 3H.2-2.

3H.2.5 Structural Design and Analysis Summary

3H.2.5.1 Analytical Model

A three dimensional finite element model representing half of the structure, i.e., between grid lines A & D and 1 & 4, was developed to perform the static analysis.

Figure 3H.2-20 shows the analytical model. Walls, floor slabs and roof slabs are represented by quadrilateral plate elements. The size of the plate elements vary from a minimum of 1.47m x 1.85m to a maximum of 2.0m x 2.2m. Linear elastic beam elements are used to represent the columns.

The 0° – 180° axis represents the plane of symmetry for the model. Nodal restraints are used to define the boundary conditions along this plane.

The foundation soil is represented by vertical and horizontal springs. The embedment effect is considered in the computation of the spring constants.

The model consists of a total of 1953 nodes, 2028 plate elements, 22 beam elements, and 240 soil springs.

3H.2.5.2 Analysis

A three dimensional finite element model was developed as described above for the structural evaluation of the Control Building. The STARDYNE computer program was used for the analysis.

The foundation soil is represented by vertical and horizontal springs.

Reinforced concrete floor slab and its supporting structural steel framing beam and columns are used to resist vertical loads such as dead load, live load, and equipment loads. Floor slabs act as diaphragms to transmit lateral loads to the exterior walls. Exterior walls act as shear walls and are used to resist lateral loads, like soil earth pressure, seismic loads, wind loads and tornado loads. Minimum thicknesses for the exterior walls and roof are provided to preclude concrete penetration, perforation, and spalling and to prevent local damage due to the tornado generated missiles. For extreme wind loads, the design basis tornado whose winds and missile loads bound the design basis hurricane are as specified in 3H.1.4.3.1.

All loads as described in Subsection 3H.2.4 are considered. The horizontal SSE seismic loads, as described in Subsection 3H.2.4.3.3.2, are the equivalent static loads as provided in Table 3H.2-1. The horizontal seismic loads applied at different elevations of the structural model are given as shear forces and moments. These forces are distributed to the nodal points at the appropriate elevations in proportion to the nodal point masses. The vertical SSE seismic loads are applied as pressure loads which are computed by multiplying the acceleration ‘g’ values by the total floor masses. The horizontal torsional moment has been considered by calculating the center of mass in each direction. A distance equal to 5% of the building dimension perpendicular to the direction of the force for each level multiplied by the mass generated the torsional moment.

Velocity pressure loading due to wind and tornado is determined by using the method and procedures contained in ASCE 7. Velocity pressure is assumed not to vary with height. All significant openings are considered sealed, i.e. the structure is non-vented.

Loads from Subsection 3H.2.4 are applied to the model as plate pressure and nodal loads and these loads are combined in accordance with the load combinations described in Subsection 3H.2.4.3.6.

3H.2.5.3 Structural Design

The Control Building is essentially a reinforced concrete structure consisting of walls and slabs and is supported by a mat foundation. Steel framing is used to support the slabs. Steel deck is used as form work to support the slabs during construction.

The reinforced concrete elements of the structure along with the steel framing form the vertical load resisting system. The vertical loads are carried by the slabs and steel framing to the walls and columns. The walls and columns transmit the loads to the base mat which then transfers them to the foundation soil.

The lateral load resisting system is composed of only the reinforced concrete elements. The roof and floor slabs act as diaphragms to transfer the lateral loads to the walls. The loads are transmitted to the base mat from the walls and then to the foundation soil. The design evaluation of the Control Building structure is divided into the following parts:

- Reinforced concrete elements
- Structural steel framing

- Steel deck
- Stability evaluation

Criteria described in Subsection 3H.2.4 are used in the structural design.

Figures 3H.2-21 through 3H.2-30 present the design drawings used for the evaluation of the Control Building.

3H.2.5.3.1 Reinforced Concrete Elements

The reinforced concrete portion of the structure is comprised of the following elements:

- Base mat
- Floor and roof slabs
- Walls

The strength design method is used for design of the elements. The ACI 349 and the NRC Regulatory Guide 1.142 govern the design. All the loads and load combinations listed in Subsection 3H.2.4 are considered in the design. Concrete of 27.58 MPa compressive strength (f'c) and deformed billet steel reinforcement of 414 MPa yield strength (f_y) are considered in the design. The structural design of these elements are discussed in detail in the following subsections:

3H.2.5.3.1.1 Base Mat

The design forces, concrete thickness, required reinforcing, and provided reinforcement are shown in Table 3H.2-3. The design of the base mat is checked for the following:

- Transfer of lateral forces from the wall bases to the mat
- Punching shear under columns and walls.

3H.2.5.3.1.2 Floor and Roof Slabs

The design forces, concrete thickness, required reinforcement, and provided reinforcement are shown in Table 3H.2-3.

Tornado missile effect has been assessed in terms of local damage. Minimum thickness of slab precludes penetration, perforation, or spalling. The actual thickness of the structural slab, i.e. the total slab thickness, less the 76 mm steel deck, is 400 mm, which is greater than the minimum 335 mm thickness of the roof required to preclude spalling due to Spectrum I missiles. To resist pullout due to tornado suction, welded studs are used to anchor the roof slab at elevation 22,000 mm TMSL to the steel framing.

3H.2.5.3.1.3 Walls

The exterior walls, i.e., walls on column lines A, D, 1, and 7 are divided into two segments for design purposes:

- Top segment between roof and elevation 17,150 mm TMSL
- Bottom segment between elevation 17,150 mm TMSL and the basemat.

The two segments differ in thickness. The design forces, and concrete thickness, required reinforcement, and provided reinforcement are shown in Table 3H.2-4.

Tornado/hurricane missile effect on exterior walls above grade has been assessed for local damage. The minimum design wall thickness is 600 mm, which is greater than the 384 mm minimum wall thickness required to preclude penetration, perforation, and spalling.

The interior walls are located on column lines 3 and 5. Table 3H.2-4 shows the design forces, concrete thickness, required reinforcement, and provided reinforcement.

3H.2.5.3.2 Structural Steel Framing

Structural steel framing consists of beams and columns. The steel framing with deck is required to support the steam tunnel floor and roof slabs when the concrete is wet. Once the concrete has attained its design strength, the slab will resist the load, and the steel framing is then redundant.

High-strength, low-alloy ASTM A572 steel with a yield strength (f_y) of 345 MPa is considered in the design. The choice of this high-strength steel over ASTM A36 steel is that it enables the use of less shallow beam sections, thereby increasing head room. Steel framing supporting floors at elevations 13,100 mm TMSL and 18,250 mm TMSL and roof at elevation 22,750 mm TMSL are encased in concrete to increase headroom. Columns are also of ASTM A572 steel so that smaller sections can be used.

Connections of steel framing supporting roof slabs at elevation 22,000 mm TMSL are designed to resist pullout due to tornado/hurricane suction.

Steel beams are supported by the concrete walls and steel columns. The columns are supported by base plates attached to the base mat by ASTMþA36 anchor bolts. The columns are designed as concentrically loaded compression members. The anchor bolts provided are nominal since they are not subjected to shear or tension.

The allowable stress design method is used for design of structural steel. AISC S-335 and ANSI/AISC-N690 govern the design. The number of different steel section sizes is kept to a minimum to optimize fabrication cost.

3H.2.5.3.3 Steel Deck

The steel deck is used as form work to support the wet concrete of the roof and floor slabs. The deck is designed for construction loads discussed in Subsection 3H.2.4.3.5. The depth of the deck is kept to a minimum to maximize floor to ceiling height. A 76 mm deck is used to support all slabs except those at elevations 18,250 mm TMSL and 22,750 mm TMSL, where a 114 mm deck is utilized. The steel deck conforms to ASTM A 446, grade A, and is galvanized.

3H.2.5.3.4 Stability Evaluation

The stability of the Control Building is evaluated for the various load combinations listed in Subsection 3H.2.4.5. The factors of safety are shown in Table 3H.2-5.

Table 3H.2-1 Control Building SSE Loads

TMSL Elev. (m)	Node	0°–180° Shear (t)	0°–180° Moment (MN·m)	Torsion (MN·m)	Vertical Accel. (g)
–8.2	102	18,100	4550	497	0.31
–2.15	103	18,100	3472	497	0.34
3.5	104	18,100	2471	497	0.39
7.9	105	14,600	1648	401	0.44
12.3	106	10,600	959	291	0.48
17.2	107	4,900	405	134	0.50

TMSL Elev. (m)	Node	90°–270° Shear (t)	90°–270° Moment (MN·m)	Torsion (MN·m)
–8.2	102	18,400	4835	506
–2.15	103	18,400	3736	506
3.5	104	18,400	2913	506
7.9	105	14,000	1961	385
12.3	106	9,600	1216	264
17.2	107	4,300	586	117

Table 3H.2-2 Control Building Base Shear and Overturning Moments for Stability Evaluation

Axis	Shear (t)	Moment (MN·m)
0°–180°	18,100	4550
90°–270°	18,400	4835

Table 3H.2-3 Base Mat, Floor and Roof Slabs–Design Forces and Reinforcement

Element No.	Load Combination	Thick-ness mm (in.)	Design Loads (See Notes 1 and 2) Axial About 90°–270° Axis	Axial About 0°–180° Axis	Shear In-Plane	Shear Out-of-Plane	Flexure About 90°–270° Axis	Flexure About 0°–180° Axis	Reinforcement Top & Bottom Each Way (cm^2/m) Requ-ired	Top & Bottom Each Way Actual	Shear Ties (cm^2/m) Requ-ired	Shear Ties Actual	Remarks
200	$D+L_o+H'+F_a+E'+T_a$	3000 (118)	9.93E+05 (68,074)	–1.38E+06 (–94,349)	1.37E+06 (93,946)	1.14E+06 (78,019)	5.77E+06 (1,296,979)	7.15E+06 (1,607,610)	89.3	101.6	<17.6	21.2	Basemat (El. –8200 mm TMSL)
66	$D+L_o+H'+F_a+E'+T_a$	3000 (118)	–5.63E+06 (–388,064)	–4.11E+06 (–281,586)	2.14E+06 (146,698)	5.80E+06 (397,354)	4.27E+06 (960,113)	7.10E+06 (1,595,485)	<89.3	101.6	17.6	21.2	
Slab at El. –2150 mm: Envelope Values	$D+L_o+H'+F_a+E'+T_a$ $D+L_o+H'+F_a+E'+T_a$	400 (16)	–4.09E+06 (–280,560)	–5.63E+06 (–385,459)	2.59E+06 (177,475)	Nominal	Nominal	Nominal	32.4	33.0	None	None	Slabs at El 2150, 3500, 7900, 12300 and 17150 mm TMSL
1691	$D+L_0+H'+F_a+E'+T_a$	1600 (63)	–6.72E+06 (–460,253)	–1.61E+06 (–110,141)	4.52E+06 (309,792)	3.40E+06 (233,050)	4.69E+05 (105,337)	3.28E+06 (738,328)	110.2	112.8	<20.1	21.2	Steam Tunnel Floor (El. 18250 mm TMSL)
1999	$D+L_0+H'+F_a+E'+T_a$	1600 (63)	–3.54E+06 (–242,458)	–7.55E+05 (–51,744)	4.55E+06 (311,875)	3.40E+06 (233,318)	4.78E+05 (107,564)	1.88E+06 (423,067)	<110.2	112.8	20.1	21.2	
2030	$D+L+H+1.5P_a+T_a$	1600 (63)	3.24E+06 (221,693)	1.24E+05 (8,521)	1.41E+05 (9,657)	5.58E+05 (38,257)	7.86E+05 (176,656)	1.01E+07 (2,268,556)	294.9 <183.3	338.7 203.2	<24.8	33.0	Steam Tunnel Roof (El. 22750 mm TMSL)
2168	$D+L+H+1.5P_a+T_a$	1600 (63)	9.33E+05 (63,921)	2.12E+06 (145,555)	1.07E+05 (7,338)	4.99E+05 (34,191)	7.41E+06 (1,666,253)	6.97E+06 (1,567,707)	<294.9 183.3	338.7 203.2	<24.8	33.0	* in X1 Direction ** in X2 Direction
2335	$D+L+H+1.5P_a+T_a$	1600 (63)	6.10E+06 (417,934)	1.02E+06 (69,821)	2.33E+06 (159,533)	4.06E+06 (278,208)	1.45E+06 (326,505)	9.66E+05 (217,243)	<294.9 <183.3	338.7 203.2	24.8	33.0	

Notes: 1. The values of axial forces and shears are shown in N/m and those for flexure are shown in N•m/m. The corresponding values in lb/ft and lb-ft/ft are shown in parenthesis.
2. Positive axial forces are tensile; negative axial forces are compressive.

Table 3H.2-4 Walls–Design Forces and Reinforcement

Element No.	Load Combination	Thickness mm (in)	Design Loads (See Notes 1 and 2): Axial Horizontal	Axial Vertical	Shear In-Plane	Shear Out-of-Plane	Flexure About Horizontal Axis	Flexure About Vertical Axis	Reinforcement: Vertical ea face, (cm^2/m) Required	Vertical Actual	Horizontal ea face, (cm^2/m) Required	Horizontal Actual	Shear Ties (cm^2/m) Required	Shear Ties Actual	Remarks
2809	$D+L_0+H'+P_a+E'+T_a$	600 (24)	1.39E+06 (95,021)	1.70E+06 (116,256)	1.28E+06 (87,830)	9.20E+04 (6,302)	4.39E+05 (98,657)	3.49E+05 (78,551)	<83.7	84.8	55.0	56.8	None	None	Walls on grid lines A and D — portion between El. 17150 and roof
2781	$D+L_0+H'+P_a+E'+T_a$	600 (24)	1.17E+06 (80,438)	2.38E+06 (163,363)	1.09E+06 (74,995)	6.25E+04 (4,280)	6.25E+05 (140,611)	3.78E+05 (84,878)	83.7	84.8	<55.0	56.8	None	None	
2730	$D+L_0+H'+P_a+E'+T_a$	1000 (39)	5.71E+06 (391,238)	–3.29E+05 (–22,559)	1.04E+06 (71,232)	3.87E+05 (26,510)	1.29E+06 (289,467)	1.02E+06 (229,060)	<119.1	127.2	122.7	127.2	<19.9	21.2	Walls on grid lines A and D — portion between basemat (El. –8200) and El. 17150 *Between finished grade and El. 17150 only.
3448	$D+L_0+H'+P_a+E'+T_a$	1000 (39)	3.85E+06 (263,693)	–1.82E+06 (–124,454)	3.45E+06 (236,275)	4.39E+05 (30,092)	1.31E+06 (295,419)	1.10E+06 (248,020)	119.1	127.2	<122.7	127.2	<19.9	21.2	
2474	$D+L_0+H'+P_a+E'+T_a$	1000 (39)	–8.49E+05 (–58,188)	1.17E+06 (80,237)	2.32E+06 (158,928)	1.97E+06 (134,669)	8.31E+05 (186,864)	2.33E+05 (52,426)	<119.1	127.2	<43.0	56.8	19.9	21.2	
2437	$D+L_0+H'+P_a+E'+T_a$	1000 (39)	–1.95E+06 (–133,728)	–1.66E+05 (–11,370)	2.38E+06 (163,094)	1.27E+06 (86,688)	6.90E+04 (15,505)	9.27E+05 (208,403)	<123.4	127.2	43.0	56.8	<19.9	21.2	
3133	$D+L_0+H'+P_a+E'+T_a$	1000 (39)	–9.25E+05 (–63,356)	2.24E+06 (153,821)	2.65E+06 (181,507)	3.41E+05 (23,392)	1.83E+06 (410,942)	4.26E+05 (95,659)	123.4	127.2	<43.0	56.8	<19.9	21.2	
3757	$D+L_0+H'+F_a+E'+T_a$	600 (24)	8.12E+05 (55,608)	1.05E+06 (71,702)	1.11E+06 (76,070)	2.25E+05 (15,429)	5.09E+05 (114,398)	3.07E+05 (68,961)	58.7	66.1	49.2	49.6	None	None	Walls on grid lines 1 and 7 — portion between El. 17150 and roof
3751	$D+L_0+H'+F_a+E'+T_a$	1000 (39)	2.24E+06 (153,283)	3.52E+05 (24,098)	1.44E+06 (98,582)	1.04E+05 (7,116)	9.97E+05 (224,210)	9.97E+05 (223,990)	95.4	101.8	71.2	84.7	<18.0	21.2	Walls on grid lines 1 and 7 — portion between basemat (El. -8200) and El. 17150
3605	$D+L_0+H'+F_a+E'+T_a$	1000 (39)	–1.49E+06 (–102,211)	1.28E+06 (87,965)	2.69E+06 (184,195)	2.22E+05 (15,221)	1.44E+06 (323,639)	3.25E+05 (73,149)	95.4	101.8	<71.2	84.7	<18.0	21.2	
3632	$D+L_0+H'+F_a+E'+T_a$	1000 (39)	–2.80E+06 (–191,587)	1.17E+06 (80,237)	1.96E+06 (134,198)	1.82E+06 (124,723)	6.66E+05 (149,606)	2.40E+04 (5,401)	<95.4	101.8	<71.2	84.7	18.0	21.2	

Notes:
1. The values of axial forces and shears are shown in N/m and those for flexure are shown in N•m/m. The corresponding values in lb/ft and lb-ft/ft are shown in parenthesis.
2. Positive axial forces are tensile; negative axial forces are compressive.

Table 3H.2-4 Walls–Design Forces and Reinforcement (Continued)

Ele-ment No.	Load Combination	Thick-ness mm (in)	Design Loads (See Notes 1 and 2)						Reinforcement						Remarks
			Axial		Shear		Flexure		Vertical ea face, (cm²/m)		Horizontal ea face, (cm²/m)		Shear Ties (cm²/m)		
			Horizontal	Vertical	In-Plane	Out-of-Plane	About Horizontal Axis	About Vertical Axis	Requ-ired	Actual	Requ-ired	Actual	Requ-ired	Actual	
3912	D+L_0+H′+P_a+E′+T_a	1600 (63)	5.98E+06 (409,450)	1.05E+07 (721,728)	6.08E+06 (416,774)	7.36E+05 (50,427)	2.18E+06 (490,749)	3.25E+05 (73,149)	249.7	254.4	<189.7	203.5	<10.6	12.7	Walls on grid lines 3 and 5
3912	D+L_0+H′+P_a+E′+T_a	1600 (63)	5.98E+06 (409,450)	1.05E+07 (721,728)	6.08E+06 (416,774)	7.36E+05 (50,427)	2.18E+06 (490,749)	3.25E+05 (73,149)	249.7	254.4	189.7	203.5	<10.6	12.7	
3948	D+L+H+1.5P_a+E′+T_a	1600 (63)	–3.06E+06 (–209,462)	–2.29E+06 (–157,114)	2.42E+06 (165,715)	1.99E+06 (136,349)	204,262 (101,236)	3.81E+05 (85,738)	<249.7	254.4	<189.7	203.5	10.6	12.7	

Notes: 1. The values of axial forces and shears are shown in N/m and those for flexure are shown in N•m/m. The corresponding values in lb/ft and lb-ft/ft are shown in parenthesis.
2. Positive axial forces are tensile; negative axial forces are compressive.

Table 3H.2-5 Stability Evaluation–Factors of Safety

Load Combination	Overturning		Sliding		Flotation	
	Required	Actual	Required	Actual	Required	Actual
D+F’	–	–	–	–	1.1	1.42
D+F+H+W	1.5	2.79	1.5	2.74	–	–
D+F+H+W_t	1.1	2.66	1.1	2.69	–	–
D+L_o+F+H’+E’**	1.1	123[1]	1.1	1.14	–	–

1 Based on the energy technique

** Zero live load is considered.

Figure 3H.2-1 Not Used

Figure 3H.2-2 Not Used

Figure 3H.2-3 Not Used

Figure 3H.2-4 Not Used

Figure 3H.2-5 Not Used

Figure 3H.2-6 Not Used

Figure 3H.2-7 Not Used

Figure 3H.2-8 Not Used

Figure 3H.2-9 Not Used

Figure 3H.2-10 Not Used

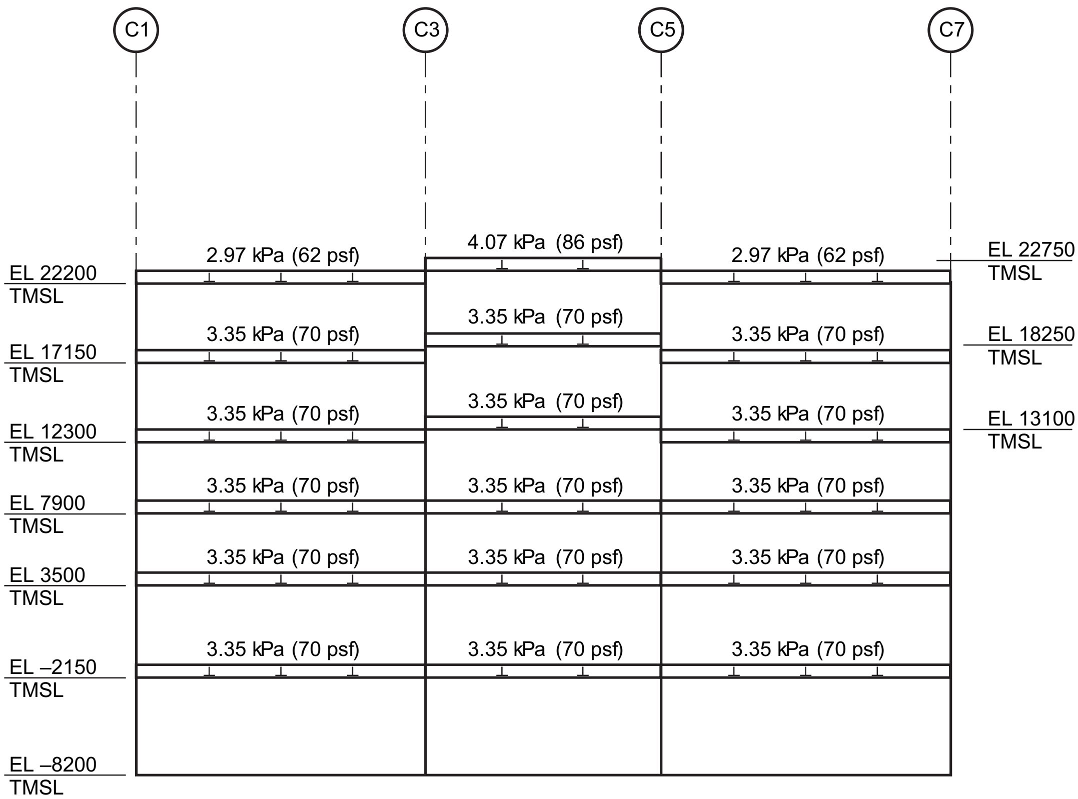

Figure 3H.2-11 Dead Load (D)

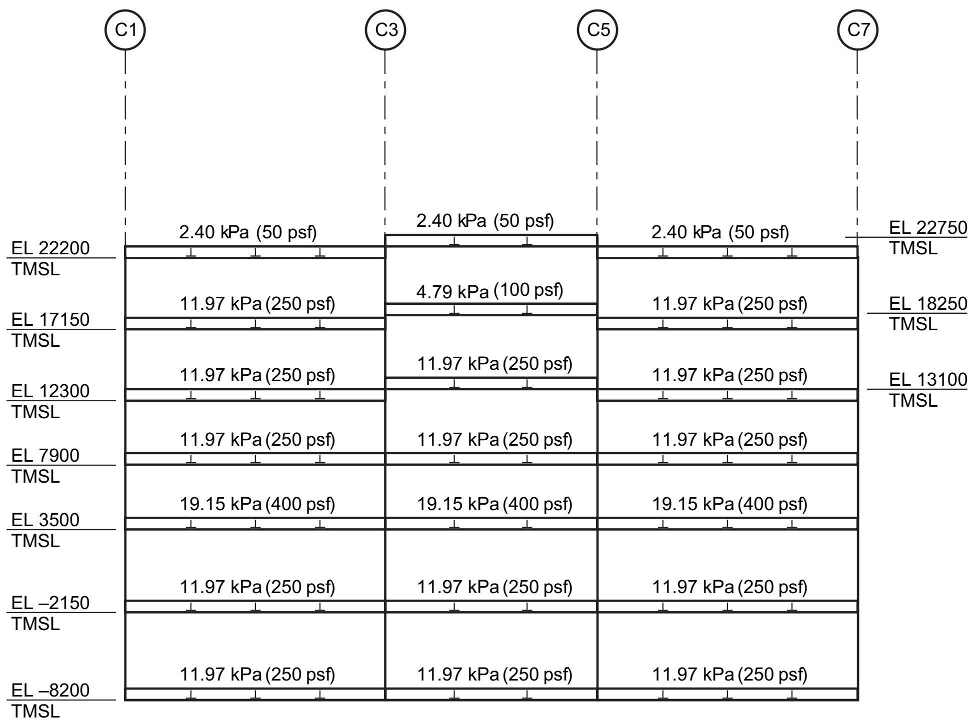

Figure 3H.2-12 Live and Snow Loads (L)

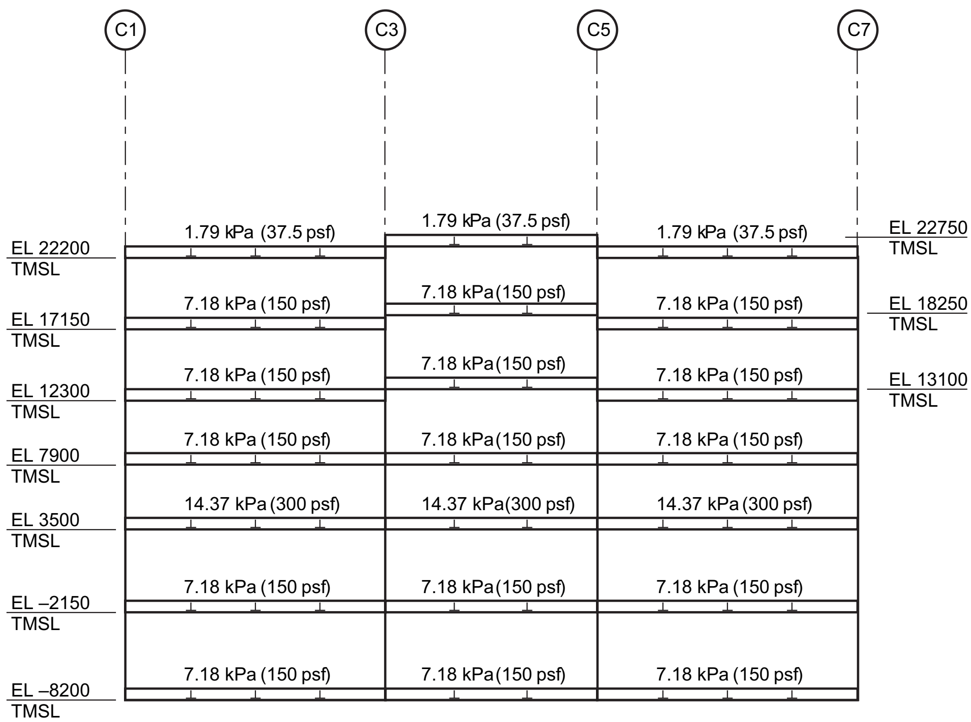

Figure 3H.2-13 Live and Snow Loads During SSE (L_o)

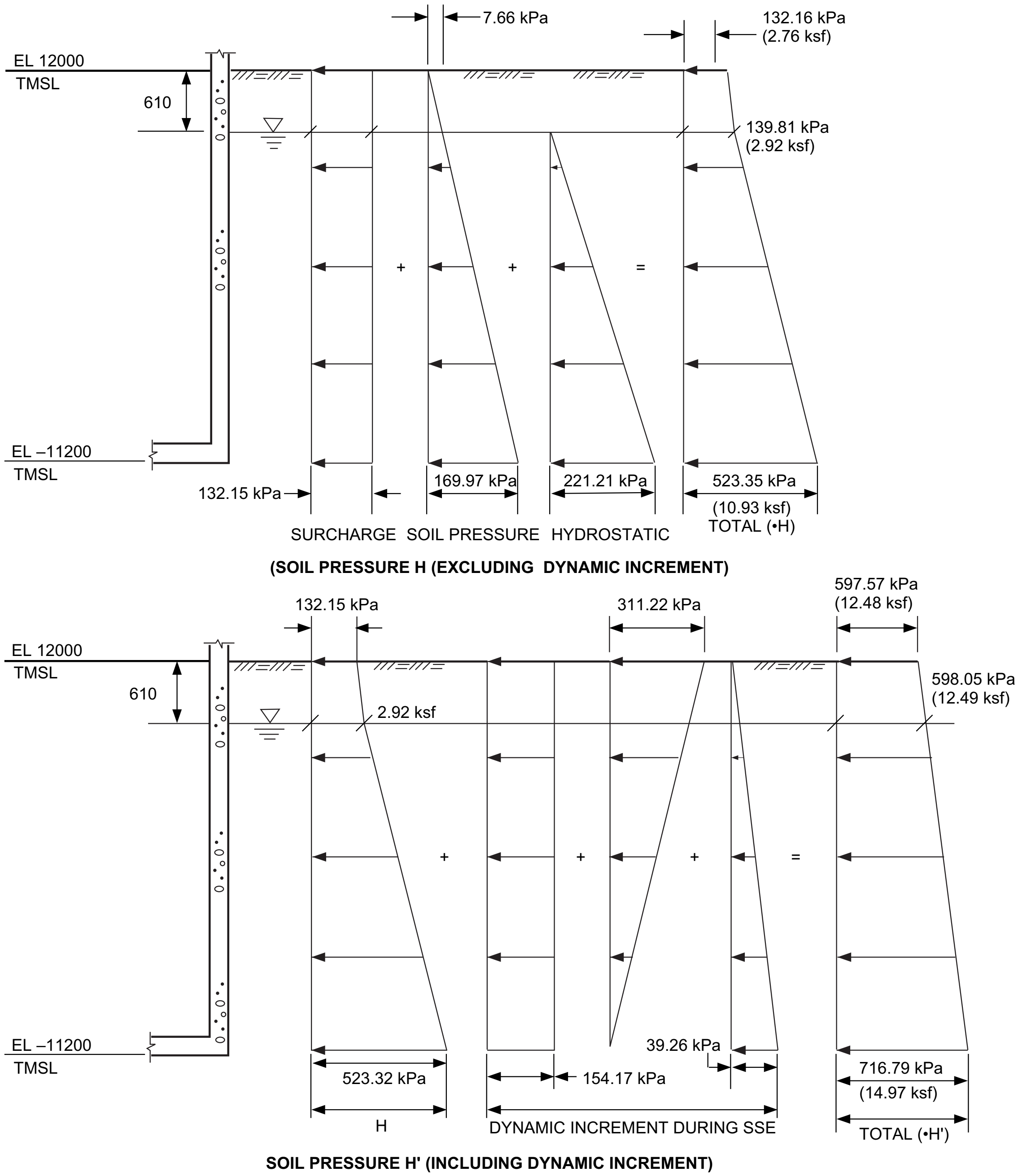

Figure 3H.2-14 At Rest Lateral Soil Pressures on Walls (H and H')

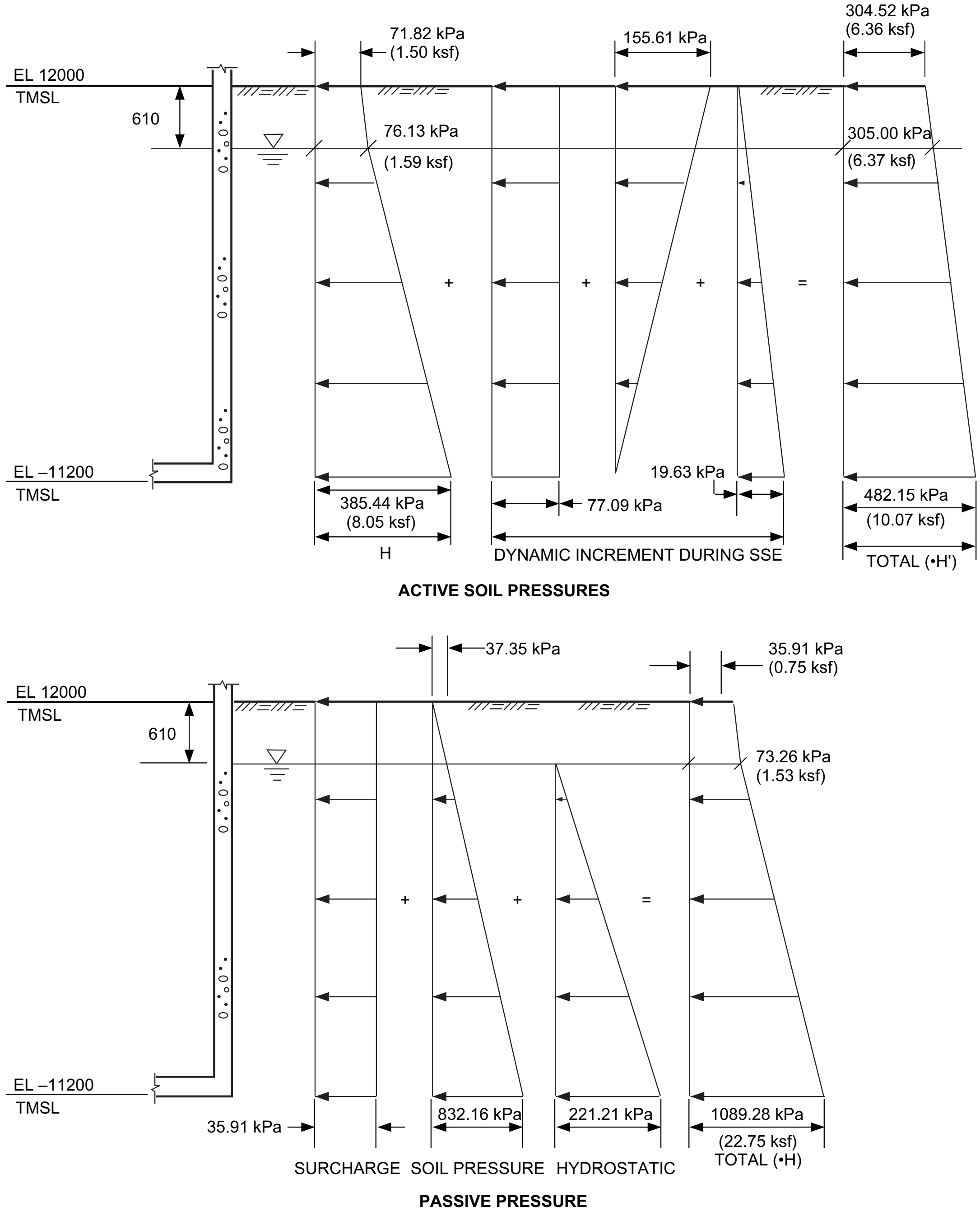

Figure 3H.2-15 Active and Passive Lateral Soil Pressures on Walls

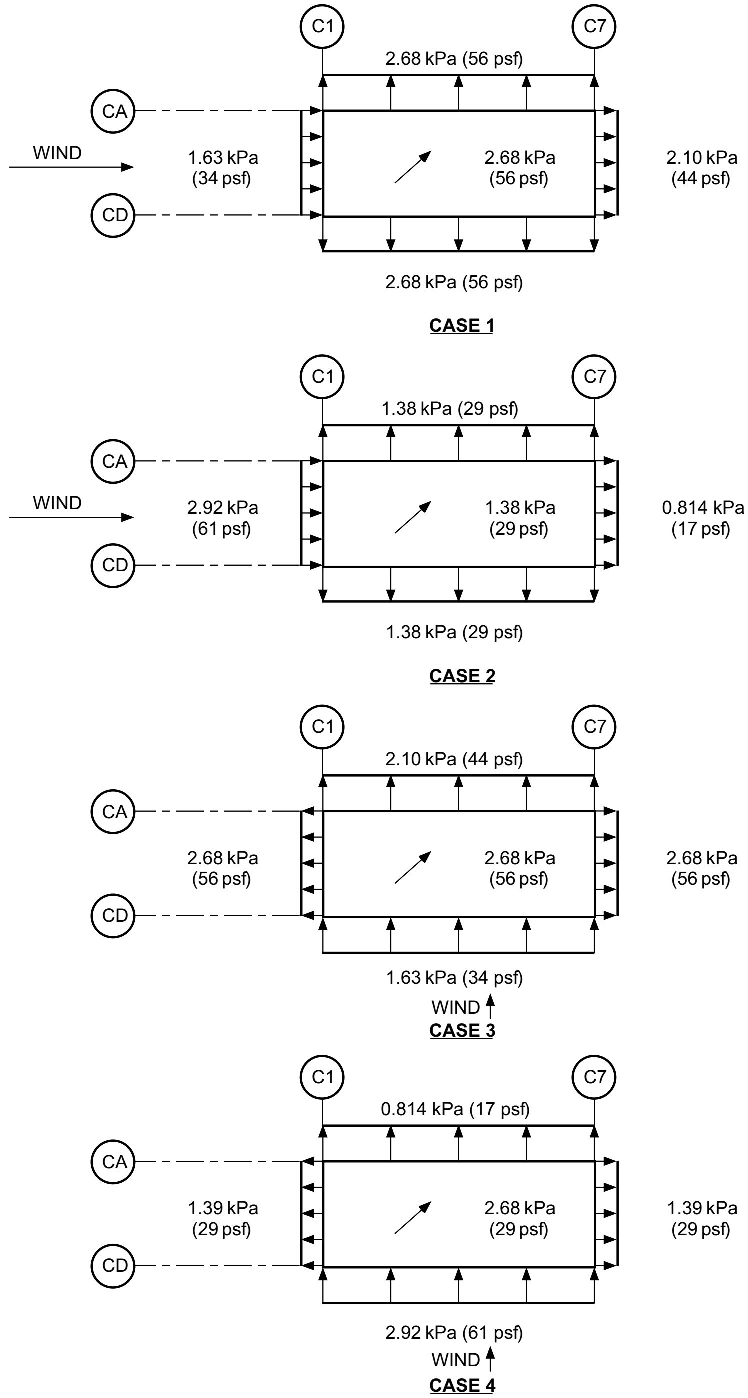

Figure 3H.2-16 Wind Loads (W)

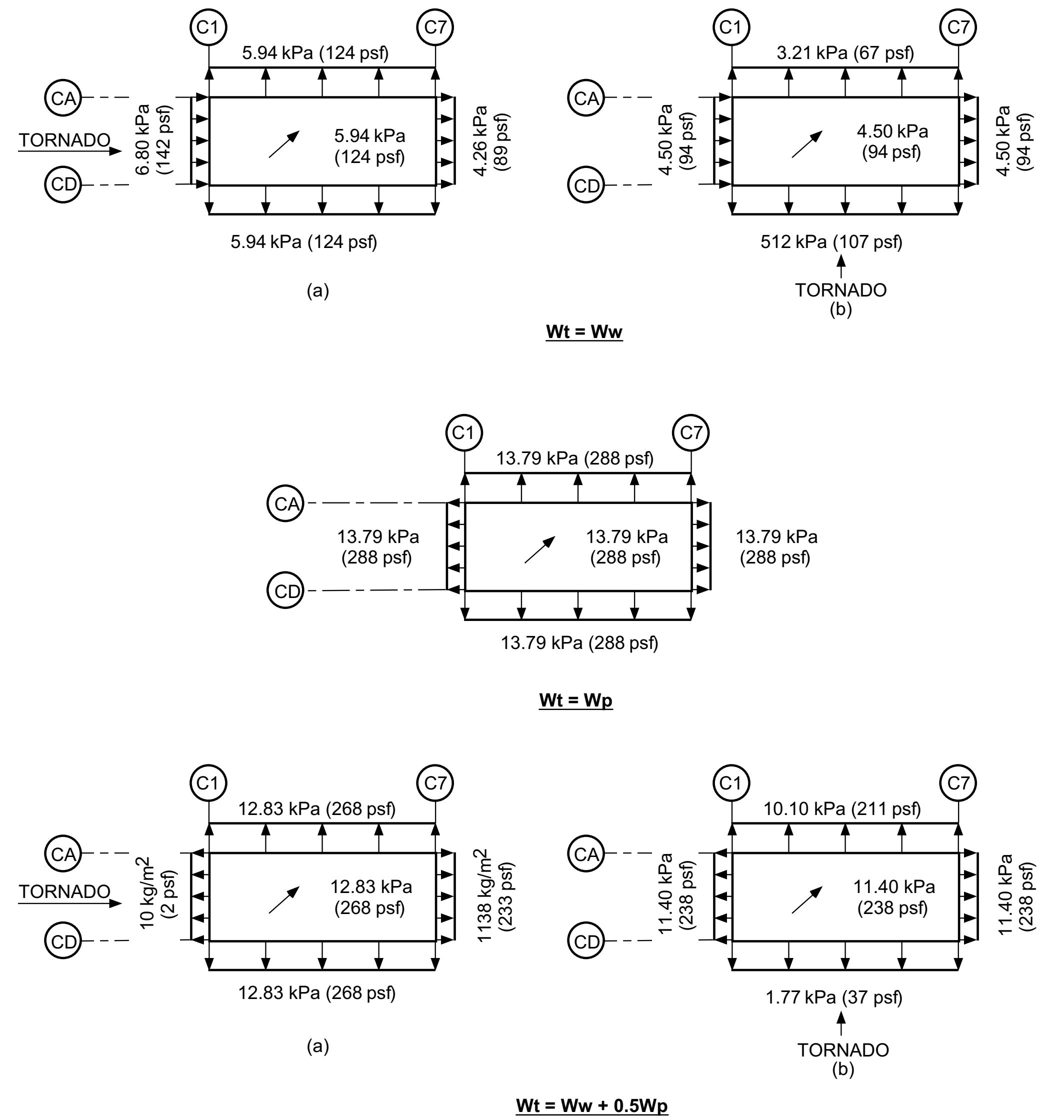

Figure 3H.2-17 Tornado Loads (W_t)

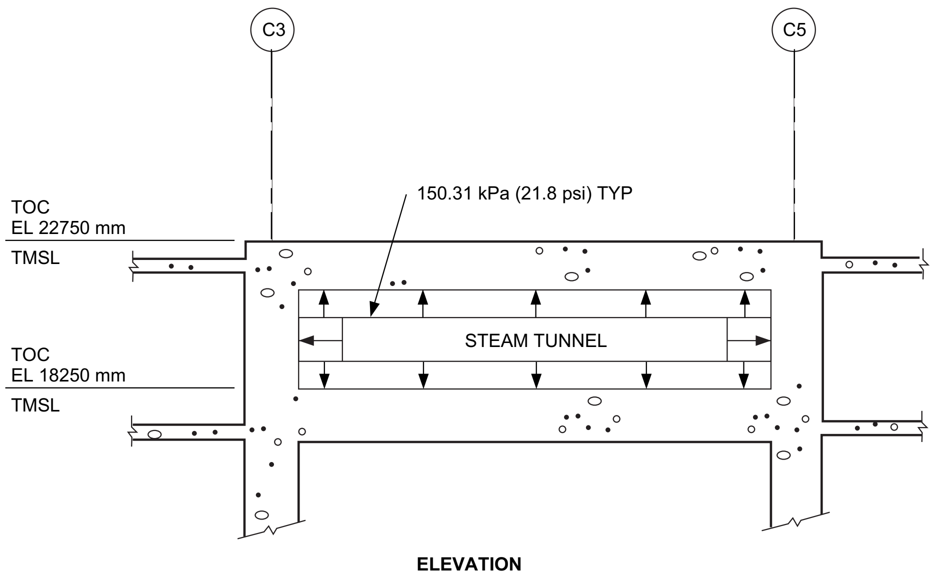

Figure 3H.2-18 Accident Pressure Load (P_a)

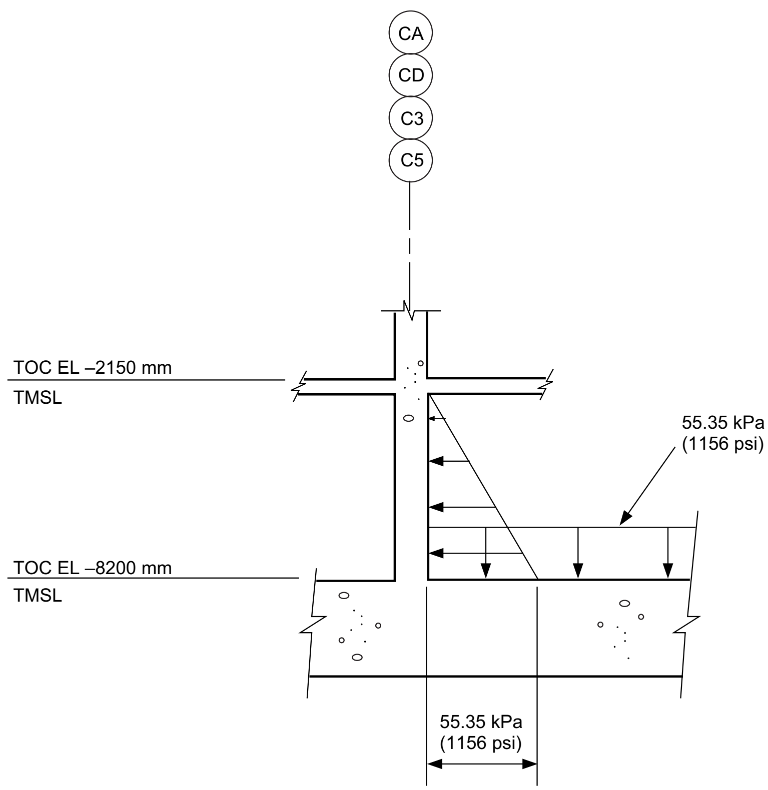

Figure 3H.2-19 Accident Hydrostatic Load (F_a)

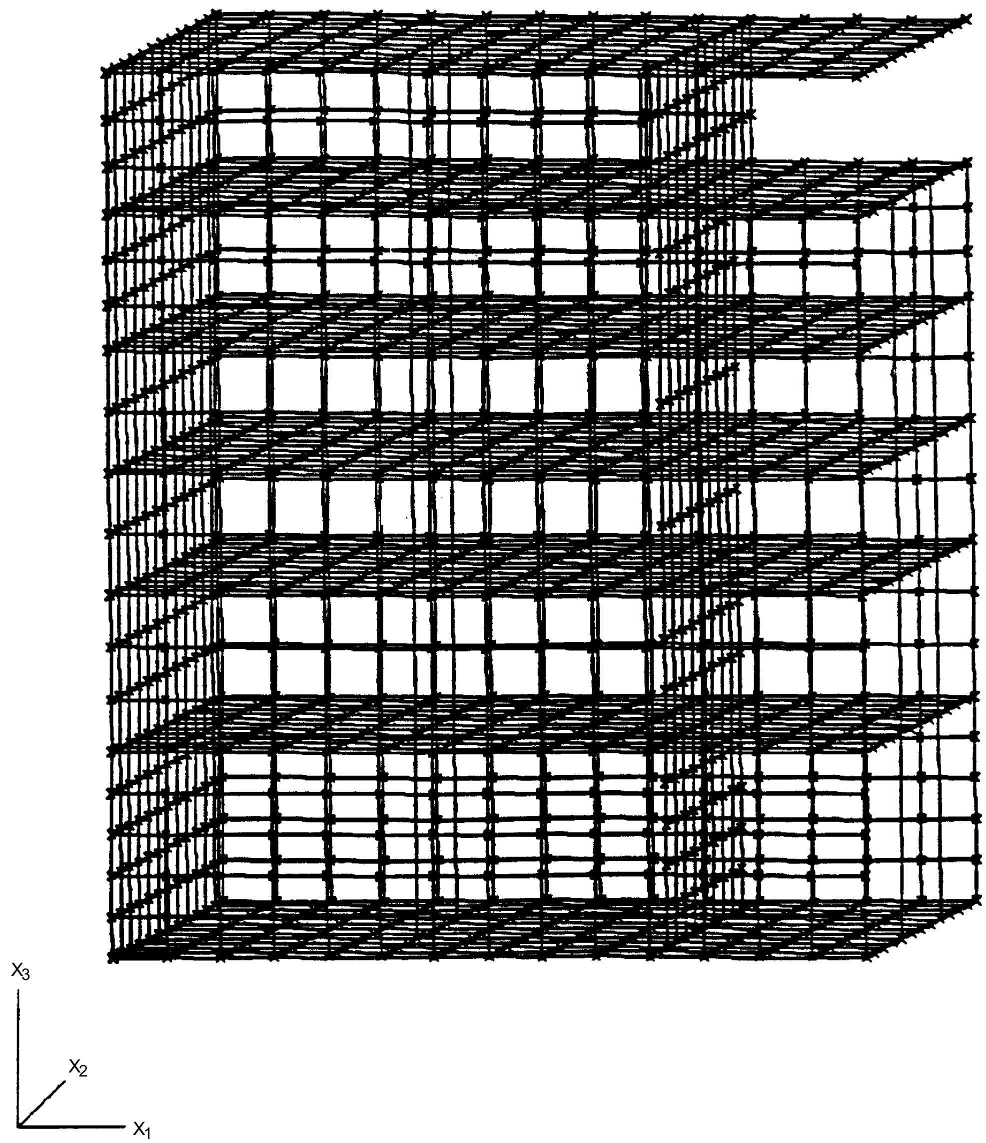

Figure 3H.2-20 Control Building Static Analysis Model

The following figures are located in Chapter 21:

3H.3 Radwaste Building

3H.3.1 Objective Scope

The objective of this subsection is to document the structural design and analysis of the ABWR Radwaste Building. The scope includes the design and analysis of the basemat and exterior walls of the building for the normal, severe environmental, extreme environmental and construction loads.

This subsection addresses all applicable items included in Appendix C to USNRC Standard Review Plan, NUREG-0800, Section 3.8.4

3H.3.2 Conclusions

The following are the major summary conclusions on the design and analysis of the Radwaste Building:

- Based on the design drawings identified in Subsection 3H.3.5, stresses in concrete, reinforcement, and structural steel are less than the allowable stresses per the applicable codes listed in Subsection 3H.3.4.1.
- The factors of safety against flotation, sliding, and overturning of the structure under various loading combinations are higher than the required minimum.
- The building has been evaluated for the design basis tornado/hurricane. Welded studs are provided for the roof structural steel to provide required resistance for negative pressure.

3H.3.3 Structural Description

The basemat and exterior walls of the Radwaste Building which are at and below grade are Seismic Category I structures. The portions which are above grade are not Seismic Category I structures; however major structural concrete walls and slabs of these above grade structures are designed to resist Seismic Category I loads. The Radwaste Building houses the liquid, solid, and gaseous radwaste treatment and storage facilities, offgas monitoring, and maintenance areas.

The Radwaste Building is a 60.4m long (0°-180° direction) x 41.2m wide (90°-270° direction) structure that is 29.5m in height above the top of the 2.5m thick basemat. It consists of four floors, two of which are below grade. The total building embedment is 16m. The Radwaste Building is a reinforced concrete structure consisting of walls and slabs and is supported by a mat foundation. Steel framing is used to support the slabs for construction loads. Steel deck is used as form work to support the slabs during construction.

The location of the Radwaste Building in relation to other plant structures is shown on the Figure 1.2-1. Figures 1.2-23a through 1.2-23e show the arrangement of the building.

3H.3.4 Structural Design Criteria

3H.3.4.1 Design Codes

Reinforced concrete is designed by the strength design method [*in accordance with ACI 349*][4] as augmented by USNRC Regulatory Guide 1.142.

Structural steel is designed by the allowable stress design method in accordance with the AISC Manual of Steel Construction, for non-safety applications and [*by ANSI/AISC-N690 for Safety Class 3 applications.*]*

3H.3.4.2 Site Design Parameters

The following are some of the key design parameters.

3H.3.4.2.1 Soil Parameters

Minimum shear wave velocity:	305 m/s
Poisson's ratio:	0.3 to 0.38
Unit weight:	1.9 to 2.2 t/m^3
Liquefaction potential:	None
Minimum Static Soil Bearing Capacity Demand:	≥718.20 kPa

3H.3.4.2.2 Design Ground Water Level

Design ground water level is at 0.61m below grade level.

3H.3.4.2.3 Design Flood Level

Design basis flood level is at 0.305m below grade level.

3H.3.4.2.4 Maximum Snow Load

Design snow load is 2.39 kPa.

3H.3.4.2.5 Maximum Rainfall

Design rainfall intensity is 493 mm/h. Roof parapets are furnished with scuppers to supplement roof drains, or, are designed without parapets so that excessive pounding of water cannot occur. Such roof design meets the provisions of ASCE 7, Section 8.0.

4 See Subsection 3.8.3.2.

3H.3.4.2.6 Design Temperatures

Maximum Ambient External:	38°C
Minimum Ambient External:	–23°C
Stress Free Temperature:	15.5°C
Building Internal Temperature:	Not controlling design

3H.3.4.3 Design Loads And Load Combinations

3H.3.4.3.1 Normal Loads

Normal loads are those which are encountered during normal plant startup, operation, and shutdown.

3H.3.4.3.1.1 Dead Loads (D)

Dead loads include weight of the structure, permanent equipment and other permanent static loads. An additional minimum allowance of 2.39 kPa uniform load is made for dead loads due to piping, raceways and HVAC duct work.

3H.3.4.3.1.2 Live Loads (L and L_o)

Live loads include floor and roof area loads, movable loads and laydown loads. The live loads, L, used in the design are as follows:

Roof	1.42E-03 MPa
Floors	1.20E-02 MPa

Live load is omitted from areas occupied by equipment.

For computation of global seismic loads, the value of live load is limited to the expected live load, L_o, during normal plant operation which is taken as 25% of the above live loads, L_o. However, in the load combinations involving seismic loads the L_o values used for the design are as follows:

Roof	3.63E-04 MPa
Floors	7.18E-03 MPa

3H.3.4.3.1.3 Snow Load

A value of 2.39 kPa is used for snow live load (L). The snow load (L_o) is reduced to 75%, when it is combined with seismic loads.

3H.3.4.3.1.4 Lateral Soil Pressures (H and H')

The following soil parameters are used in the computation of lateral soil pressures:

Dry unit weight:	1.9 to 2.2 t/m^3
Shear wave velocity:	305 m/s
Internal friction angle:	30° to 40°

The dynamic lateral soil pressure increment due to SSE is calculated in accordance with the Mononobe-Okabe method.

The design of the structure is based on the at rest soil pressure. Figure 3H.3-1 shows the at rest lateral soil pressure H (excluding the dynamic soil pressure increment) and H' (including the dynamic soil pressure increment). Active and passive soil pressures are used in the evaluation of the structure. Figure 3H.3-2 shows these pressures.

3H.3.4.3.2 Severe Environmental Loads

The severe environmental load considered in the design is that generated by wind. The following parameters are used in the computation of wind loads.

Basic wind speed:	177 km/h
Exposure:	D
Importance factor:	1.1

Wind loads are calculated per the provisions of ASCE 7.

3H.3.4.3.3 Extreme Environmental Loads

Extreme environmental loads consist of loads generated by the tornado and safe shutdown earthquake. The design basis tornado whose wind and missile loads bound the design basis hurricane are as specified in 3H.1.4.3.1.

3H.3.4.3.3.1 Tornado Loads (W_t)

The following tornado load effects are considered in the design:

- Wind pressure (W_w)
- Differential pressure (W_p)
- Missile Load (W_m)

Parameters used in computation of tornado loads are as follows:

Maximum wind speed:	483 km/h
Maximum rotational speed:	386 km/h
Maximum translational speed:	97 km/h
Radius of maximum rotational speed	45.7m
Differential pressure:	13.83 kPa
Rate of pressure differential:	8.28 kPa/s
Missile Spectrum:	See Table 2.0-1

(1) Tornado Wind Pressure (W_w)

Tornado wind pressures are computed using the procedure described in Bechtel Topical Report BC-TOP-3-A. The topical report uses the methods and procedures contained in ASCE 7 with the following exceptions:

(a) Wind velocity and wind pressures are constant with height.

(b) Wind velocity and wind pressures vary with horizontal distance from the center of the tornado.

(c) Wind pressures are determined by multiplying the wind pressure at the radius of maximum wind by the size coefficient.

(d) The gust factor is unity.

(e) ASCE criteria for determining wind pressures on building components and cladding are not applicable.

(2) Tornado Differential Pressure (W_p)

The Radwaste Building is designed as unvented, i.e. without major openings. The differential pressure causes suction on exterior walls and roof of the structure.

(3) Tornado Missile Impact (W_m)

Tornado missile impact effects on the structure can be assessed in the following two ways:

(a) Local damage in terms of penetration, perforation, and spalling.

(b) Structural response in terms of deformation limits, strain energy capacity, structural integrity and structural stability.

(4) Tornado Load Combinations

Tornado load effects are combined as follows:

$$W_t = W_w$$

$$W_t = W_p$$

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

3H.3.4.3.3.2 Safe Shutdown Earthquake Loads (E)

The SSE Loads are applied in three directions, viz. the two horizontal directions and the vertical direction. The total structural response is predicted by combining the applicable maximum codirectional responses by "the square root of the sum of the squares (SRSS)" method.

The SSE loads are based on a free-field peak ground acceleration value of 0.3g at plant grade elevation. The loads consist of story shears, torsional moments, and overturning moments. The loads are distributed to the resisting walls in proportion to their rigidities. The SSE loads for the Radwaste Building are provided in Table 3H.3-1.

3H.3.4.3.4 Construction Loads

3H.3.4.3.4.1 Steel Deck

The steel deck supporting wet concrete is designed for the weight of concrete plus 2.39 kPa uniformly distributed load.

3H.3.4.3.4.2 Structural Steel Framing

The steel beams supporting the deck are designed for the weight of concrete plus 4.805 kPa uniformly distributed and a 22.26 kN concentrated load placed anywhere on the span of major beams to maximize moment and shear.

3H.3.4.3.5 Load Combinations

3H.3.4.3.5.1 Notations

S = Allowable stress for allowable stress design method

U = Required strength for strength design method

D = Dead load

L = Live load or snow load

L_o = Live load or snow load concurrent with earthquake

H = Lateral soil pressure

H' = Lateral soil pressure including dynamic (SSE) increment

W = Wind load

W_t = Tornado load

E = Safe shutdown earthquake load

3H.3.4.3.5.2 Structural Steel

$S = D + L$

$S = D + L + W$

$1.6S = D + L + W_t$

$1.6S = D + L_o + E$

3H.3.4.3.5.3 Reinforced Concrete

$$U = 1.4D + 1.7L + 1.7H$$

$$U = 1.4D + 1.7L + 1.7H + 1.7W$$

$$U = 1.0D + 1.0L + 1.0H + 1.0W_t$$

$$U = 1.0D + 1.0L_o + 1.0H' + 1.0E$$

3H.3.4.4 Materials

Structural materials used in the design and their properties are as follows:

3H.3.4.4.1 Concrete

Concrete conforms to the requirements of ACI 349. Its design properties are:

Compressive stress (f'_c)	=	27.58 MPa
Modulus elasticity	=	2.48E+04 MPa
Shear modulus	=	1.06E+04 MPa
Poisson’s ratio	=	0.18

3H.3.4.4.2 Reinforcement

Deformed billet-steel reinforcing bars are considered in the design. Reinforcement conforms to the requirement of ASTM A615. Its design properties are:

Yield stress	=	4.14E+02 MPa
Tensile strength	=	6.21E+02 MPa

3H.3.4.4.3 Structural Steel

High strength, low-alloy structural steel conforms to ASTM A572, Grade 50. The steel design properties are:

Yield stress	=	3.45E+02 MPa
Tensile strength	=	4.48E+02 MPa

Connection bolts conform generally to ASTM A325 for bearing type connections.

Welded connection material conforms to AWS D1.0 for E70XX electrodes.

3H.3.4.4.4 Anchor Bolts

Material for anchor bolts conforms to the requirements of ASTM A36. Its design properties are:

Yield stress = 2.48E+02 MPa

Tensile strength = 4.00E+02 MPa

Alternate anchor bolt material conforms to the requirements of ASTM A307. The minimum tensile strength of this material is 4.14E+02 MPa.

3H.3.4.5 Stability Requirements

The following minimum factors of safety are provided against overturning, sliding and flotation:

Load Combination	Overturning	Sliding	Flotation
D + F'	—	—	1.1
D + F + W + H	1.5	1.5	—
D + F + W_t+ H	1.1	1.1	—
D + L_o+ F + H'+ E'	1.1	1.1	—

F = Buoyant Force Due To Design Ground Water

F' = Buoyant Force Due To Design Basis Flood

3H.3.5 Structural Design and Analysis Summary

3H.3.5.1 Analytical Model

A three dimensional finite element model, as shown in Figures 3H.3-3 through 3H.3-8, was developed using the STARDYNE program. The Radwaste Building is idealized as a 3-D assemblage of linear elastic beams and rectangular plate elements. Interior structural steel columns are represented by beam elements and concrete base mat, exterior walls and floor slabs are represented by rectangular plate elements. The foundation soil is represented by soil spring elements. The model consists of 1715 nodes, 60 beam elements, 1728 plate elements and 741 nodal soil spring elements.

3H.3.5.2 Analysis

A three dimensional finite element model developed as described above for the structural evaluation of the Radwaste Building. The STARDYNE computer program was used for the analysis.

The foundation soil is represented by vertical and horizontal springs.

Reinforced concrete floor slab and its supporting structural steel framing beam and columns are used to resist vertical loads such as dead load, live load, and equipment loads. Floor slabs act as diaphragms to transmit lateral loads to the exterior walls. Exterior walls act as shear walls and are used to resist lateral loads, like soil earth pressure, seismic loads, wind loads and tornado loads. The roof slab and exterior walls above grade are non-Seismic Category I structures and are designed for the wind loads given in Subsection 3H.3.4.3.2, however they are only checked for tornado loads given in Subsection 3H.3.4.3.3.1 such that they do not collapse and damage Category I portion of the structure.

All loads as described in Subsection 3H.3.4 are considered. The horizontal SSE seismic loads, as described in Subsection 3H.3.4.3.3.2, are the equivalent static loads as provided in Table 3H.3-1. The horizontal seismic loads applied at different elevations of the structural model are given as shear forces and moments. These forces are distributed to the nodal points at the appropriate elevations in proportion to the nodal point masses. The vertical SSE seismic loads are applied as pressure loads which are computed by multiplying the acceleration ‘g’ values by the total floor masses. The horizontal torsional moment has been considered in the analysis. Such torsions are provided in Table 3H.3-1.

Velocity pressure loading due to wind and tornado is determined by using the method and procedures contained in ASCE 7-88. Velocity pressure is assumed not to vary with height. All significant openings are considered sealed, i.e. the structure is non-vented.

For design, the CECAP computer program was used for the evaluation of stresses in rebar and concrete. CECAP is a Bechtel Computer Program, “Concrete Element Cracking Analysis.” The input to CECAP consists of rebar ratios, material properties and element geometry of the section under consideration together with the forces and moments from the STARDYNE analysis for a critical load combination. The thermal loads are not critical and therefore not considered in the structural design.

Loads from Subsection 3H.3.4 are applied to the model as plate pressure and nodal loads and these loads are combined in accordance with the load combinations described in Subsection 3H.3.4.3.5.

3H.3.5.3 Structural Design

3H.3.5.3.1 Reinforced Concrete and Structural Steel

Forces and moments in critical elements of various components of the Radwaste Building such as base mat and exterior walls computed from the STARDYNE analysis are summarized in Tables 3H.3-2 through 3H.3-6. Load combinations are summarized in Tables 3H.3-7 through 3H.3-9 with the section locations and the element coordinate system defined in Figure 3H.3-9 and 3H.3-10, respectively. From these load combinations the base mat and exterior walls are designed and the resultant reinforcing steel and concrete stresses are computed and are shown on Tables 3H.3-10 through 3H.3-12. The summary of reinforcing steel in the base mat and exterior walls below grade is provided in the Table 3H.3-13. The summary of minimum and maximum reinforcement ratios is provided in Table 3H.3-14. The summary of structural steel safety margins is provided in Table 3H.3-15.

Figures 3H.3-11 to 3H.3-16 present the design drawings used for the evaluation of the Radwaste Building.

3H.3.5.3.2 Stability Evaluation

The Radwaste Building foundation stability has been checked based on the shears and overturning moments provided in Table 3H.3-1. The foundation provides the following minimum factors of safety when subjected to SSE loads, tornado loads, lateral earth pressures, and buoyant forces due to maximum flood level:

Load Combination	Overturning	Sliding	Flotation
D + F’	—	—	1.3
D + F + H + W_t	1.97	2.63	—
D + F + H'+ E’	112.0	1.36	—

The factor of safety associated with overturning due to SSE loading (E) is computed using the energy approach. No live load is considered combination with SSE. The wind load is smaller than tornado load and the corresponding safety factors will be higher.

3H.3.5.3.3 Maximum Soil Bearing Pressure

Maximum soil bearing pressures, as computed using conventional method, for various load combinations are summarized in Table 3H.3-16

Table 3H.3-1 Radwaste Building Design Seismic Loads

Node	Elevation TMSL (m)	Bending Moments (N·m)	Web Shears (t)	Torsion (N·m)
90°-270° Direction: Forces and Moments in Beam Elements				
1	–1.50	2.40E + 09	1.16E + 08	2.31E + 08
2	4.80	2.40E + 09	1.16E + 08	2.31E + 08
3	12.30	1.42E + 09	9.32E + 07	1.86E + 08
4	21.00	3.31E + 08	3.43E + 07	6.86E + 07
5	28.00	1.33E + 08	3.43E + 07	6.86E + 07
0°-180° Direction: Forces and Moments in Beam Elements				
1	–1.50	2.20E + 09	1.29E + 08	3.36E + 08
2	4.80	2.20E + 09	1.29E + 08	3.36E + 08
3	12.30	1.20E + 09	1.03E + 08	2.68E + 08
4	21.00	2.65E + 08	3.63E + 07	9.43E + 07
5	28.00	1.86E + 07	3.63E + 07	9.43E + 07

Vertical Direction: Maximum Acceleration

Node	Elevation TMSL (m)	Acceleration (g)
1	–1.50	0.3
2	4.80	0.3
3	12.30	0.31
4	21.00	0.41
5	28.00	0.49

Table 3H.3-2
Forces and Moments in Critical Elements for Dead Load (D)

				Forces in (N/m)					Moments in (N•m/m)		
Section	Location	(Region)	El. #	F_x	F_y	F_{xy}	F_{xz}	F_{yz}	M_x	M_y	M_{xy}
0 (BASEMAT)	90°	(BMT–1)	55	–1.47E+5	–1.69E+5	5.84E+3	–6.77E+5	–2.76E+5	–9.56E+4	–3.71E+5	–5.70E+5
	90°	(BMT–1)	146	–1.49E+5	–1.66E+5	4.38E+3	–2.57E+5	–1.61E+4	–1.36E+6	–9.15E+5	7.45E+5
	270°	(BMT–1)	126	–5.11E+4	–5.84E+4	2.92E+3	5.95E+5	5.25E+4	–2.10E+5	–2.68E+5	–7.34E+4
	0°	(BMT–1)	9	–3.06E+5	–1.82E+5	–3.50E+4	–7.30E+3	5.91E+5	–7.56E+4	9.43E+4	2.18E+4
	Middle	(BMT–2)	52	–1.08E+5	–9.34E+4	–5.84E+4	2.85E+5	–1.07E+5	–1.23E+6	–1.14E+6	9.19E+5
	Middle	(BMT–2)	60	–1.61E+5	–1.78E+5	–2.04E+4	2.87E+5	2.41E+5	–7.08E+5	–1.52E+6	–2.12E+5
	Middle	(BMT–2)	75	–1.59E+5	–1.59E+5	–4.38E+3	–1.04E+5	–1.02E+4	–1.55E+6	–1.34E+6	–3.46E+5
	Middle	(BMT–2)	129	–1.53E+5	–1.66E+5	4.38E+3	–1.07E+5	1.75E+4	–1.57E+6	–1.37E+6	3.39E+5
	Middle	(BMT–2)	178	–1.01E+5	–1.08E+5	5.25E+4	2.90E+5	1.08E+5	–1.26E+6	–1.20E+6	–9.34E+5
1 (EXT. WALL)	90°	(1A)	2108	–1.94E+5	–1.00E+6	–5.84E+3	4.38E+3	1.65E+5	1.56E+5	8.15E+5	–1.69E+4
	90°	(1A)	2109	–1.78E+5	–9.95E+5	2.92E+3	7.30E+3	1.50E+5	1.44E+5	7.37E+5	–3.07E+4
	270°	(1B)	2405	–1.26E+5	–8.49E+5	7.30E+3	–1.61E+4	–5.40E+4	1.14E+5	5.43E+5	–2.98E+4
	270°	(1B)	2406	–1.31E+5	–8.51E+5	4.38E+3	–4.38E+3	–5.40E+4	1.16E+5	5.63E+5	–1.02E+4
	270°	(1B)	2408	–1.30E+5	–8.61E+5	–4.38E+3	1.61E+4	–5.55E+4	1.15E+5	5.48E+5	2.94E+4
	0°	(1A)	1508	–2.58E+5	–7.98E+5	2.19E+4	–1.46E+3	–1.69E+5	1.58E+5	8.74E+5	–1.33E+3
	0°	(1B)	1512	–1.88E+5	–8.79E+5	–1.14E+5	–1.02E+4	–1.62E+5	1.42E+5	8.14E+5	–7.56E+3
	0°	(1B)	1513	–1.61E+5	–9.06E+5	–9.78E+4	–1.17E+4	–1.31E+5	1.40E+5	7.37E+5	–6.23E+3
2 (EXT. WALL)	270°	(2B)	2424	1.63E+5	4.23E+4	–1.42E+5	–1.17E+5	–4.09E+4	–1.26E+5	5.03E+4	2.62E+4
3 (EXT. WALL)	90°	(2A)	2134	9.05E+4	–7.02E+5	1.23E+5	1.46E+3	1.12E+5	1.73E+4	2.58E+4	3.25E+4
	270°	(2B)	2425	1.17E+5	–2.33E+4	1.17E+5	7.44E+4	–2.92E+3	–1.47E+5	4.89E+3	1.11E+4
4 (EXT. WALL)	270°	(2B)	2441	8.32E+4	–5.92E+5	5.55E+4	5.84E+3	–4.38E+4	6.45E+4	1.63E+5	1.25E+4
	270°	(2B)	2442	7.15E+4	–6.06E+5	1.90E+4	1.46E+3	–4.52E+4	6.45E+4	1.77E+5	4.45E+3
	0°	(2A)	1565	4.96E+4	–5.47E+5	–1.68E+5	–1.17E+4	3.50E+4	–5.47E+4	–1.53E+5	–6.23E+3
5 (Ext. Wall)	90°	(2A)	2458	9.92E+4	–3.93E+5	–1.46E+5	–1.46E+4	–2.33E+4	2.36E+4	3.38E+4	–5.56E+4
	270°	(2B)	2155	1.02E+5	–6.04E+5	2.33E+4	0.0	–3.21E+4	–1.42E+4	–7.70E+4	8.90E+2
	270°	(2B)	2460	4.23E+4	–1.46E+5	–5.84E+4	–2.63E+4	1.90E+4	–7.56E+4	–7.12E+3	–3.02E+4
	0°	(2A)	1579	1.28E+5	–4.25E+5	2.33E+5	1.46E+3	3.65E+4	–1.33E+4	–7.65E+4	8.90E+2
6 (Ext. Wall)	0°	(2A)	1597	2.13E+5	–3.20E+5	2.15E+5	1.46E+3	3.65E+4	1.78E+3	1.42E+4	2.67E+3
7 (Ext. Wall)	270°	(3)	2478	3.65E+4	–3.14E+5	1.61E+4	–1.46E+3	5.84E+3	–5.78E+3	–3.20E+4	8.90E+2
	0°	(3)	1614	–5.98E+4	–7.92E+5	7.44E+4	–0.0	4.38E+3	–1.78E+3	–9.34E+3	–0.0
	0°	(3)	1620	–5.55E+4	–8.35E+5	–8.17E+4	–4.38E+3	2.92E+3	–2.22E+3	–1.33E+4	2.22E+3
8 (Ext. Wall)	270°	(4)	2491	2.33E+4	–2.51E+5	–1.02E+4	0.0	5.84E+3	–3.56E+3	–1.42E+4	0.0
9 (Ext. Wall)	0°	(4)	1650	–7.88E+4	–6.71E+5	–6.57E+4	0.0	4.38E+3	2.22E+3	1.73E+4	8.90E+2
10 (Ext. Wall)	0°	(4)	1673	3.79E+4	–1.50E+5	2.29E+5	0.0	1.17E+4	–4.00E+3	–2.31E+4	–2.22E+3
11 (Ext. Wall)	180°	(4)	1998	1.46E+3	–6.13E+4	–2.92E+3	–2.92E+3	–2.92E+3	–8.90E+2	2.22E+3	–1.78E+3
12 (Ext. Wall)	0°	(4)	1705	2.92E+3	–6.57E+4	–1.36E+5	0.0	1.61E+4	7.56E+4	4.27E+4	–0.0

Table 3H.3-3
Forces and Moments in Critical Elements for Live Loads (L)

				Forces in (N/m)					Moments in (N•m/m)		
Section	Location	(Region)	El. #	F_x	F_y	F_{xy}	F_{xz}	F_{yz}	M_x	M_y	M_{xy}
0 (BASEMAT)	90°	(BMT–1)	55	–1.31E+4	–4.82E+4	–1.02E+4	–2.38E+5	–3.79E+4	–2.27E+5	–1.13E+5	–5.43E+4
	90°	(BMT–1)	146	–1.02E+4	–3.50E+4	7.30E+3	–4.38E+3	1.61E+4	–5.93E+5	–1.96E+5	1.10E+5
	270°	(BMT–1)	126	4.38E+3	–4.52E+4	0.0	7.88E+4	0.0	–1.40E+5	3.11E+3	3.51E+4
	0°	(BMT–1)	9	0.0	–1.02E+4	–1.46E+4	–2.77E+4	2.04E+5	–7.83E+4	–2.22E+5	7.56E+4
	Middle	(BMT–2)	52	2.92E+3	–7.30E+3	–2.92E+3	1.90E+5	–2.39E+5	1.16E+4	–1.16E+4	6.45E+4
	Middle	(BMT–2)	60	1.46E+3	–4.38E+3	–1.02E+4	3.71E+5	3.24E+5	1.47E+5	–1.91E+4	–6.63E+4
	Middle	(BMT–2)	75	–7.30E+3	–2.33E+4	–2.92E+3	–4.38E+3	–1.75E+4	–2.83E+5	–4.35E+5	–1.20E+4
	Middle	(BMT–2)	129	–5.84E+3	–2.63E+4	2.92E+3	–4.38E+3	1.61E+4	–2.83E+5	–4.34E+5	1.51E+4
	Middle	(BMT–2)	178	5.84E+3	–1.17E+4	–0.0	1.90E+5	2.39E+5	1.11E+4	–8.01E+3	–6.81E+4
1 (EXT. WALL)	90°	(1A)	2108	–6.71E+4	–3.74E+5	–1.31E+4	1.46E+3	1.61E+4	2.49E+4	1.34E+5	–1.78E+3
	90°	(1A)	2109	–6.71E+4	–3.82E+5	–2.04E+4	1.46E+3	1.46E+4	2.54E+4	1.28E+5	–4.45E+3
	270°	(1B)	2405	–5.25E+4	–1.87E+5	–2.92E+3	–1.46E+3	2.92E+3	1.78E+3	8.01E+3	–4.45E+2
	270°	(1B)	2406	–4.96E+4	–1.82E+5	–0.0	–0.0	4.38E+3	8.90E+2	4.45E+3	0.0
	270°	(1B)	2408	–5.11E+4	–1.87E+5	2.92E+3	1.46E+3	2.92E+3	1.78E+3	8.01E+3	4.45E+2
	0°	(1A)	1508	–5.69E+4	–3.33E+5	–4.38E+3	0.0	–1.02E+4	2.14E+4	1.14E+5	2.22E+3
	0°	(1B)	1512	–1.46E+3	–2.61E+5	–3.36E+4	–0.0	–5.84E+3	9.34E+3	6.36E+4	2.67E+3
	0°	(1B)	1513	1.31E+4	–2.19E+5	–1.02E+4	–1.46E+3	–4.38E+3	7.56E+3	5.03E+4	–4.45E+2
2 (EXT. WALL)	270°	(2B)	2424	1.75E+4	–1.01E+5	–4.96E+4	–1.17E+4	–5.84E+3	–1.65E+4	2.67E+3	2.67E+3
3 (EXT. WALL)	90°	(2A)	2134	1.61E+4	–3.52E+5	1.61E+4	0.0	8.76E+3	1.42E+4	5.34E+4	6.23E+3
	270°	(2B)	2425	1.90E+4	–1.05E+5	5.40E+4	1.02E+4	–1.46E+3	–2.45E+4	–2.67E+3	–8.90E+2
4 (EXT. WALL)	270°	(2B)	2441	2.92E+3	–1.85E+5	5.84E+3	0.0	2.92E+3	8.01E+3	2.62E+4	–0.0
	270°	(2B)	2442	–1.46E+3	–1.82E+5	1.46E+4	–0.0	2.92E+3	6.67E+3	2.76E+4	0.0
	0°	(2A)	1565	1.02E+4	–1.98E+5	–6.71E+4	–2.92E+3	4.09E+4	–2.18E+4	–7.38E+4	–4.89E+3
5 (Ext. Wall)	90°	(2A)	2458	2.04E+4	–1.72E+5	–3.21E+4	–2.92E+3	4.38E+3	1.02E+4	2.62E+4	–5.34E+3
	270°	(2B)	2155	1.75E+4	–2.70E+5	4.38E+3	0.0	–3.50E+4	–0.0	–0.0	0.0
	270°	(2B)	2460	2.63E+4	–1.14E+5	–4.23E+4	–1.61E+4	4.38E+3	–2.31E+4	2.22E+3	–8.90E+2
	0°	(2A)	1579	2.63E+4	–2.19E+5	6.57E+4	5.84E+3	4.38E+4	4.45E+3	1.82E+4	4.45E+2
6 (Ext. Wall)	0°	(2A)	1597	4.52E+4	–2.01E+5	6.71E+4	1.46E+3	4.52E+4	2.27E+4	1.31E+5	7.56E+3
7 (Ext. Wall)	270°	(3)	2478	8.76E+3	–1.11E+5	–0.0	–0.0	7.30E+3	–3.11E+3	–1.65E+4	0.0
	0°	(3)	1614	–2.77E+4	–3.58E+5	1.46E+3	–1.46E+3	1.90E+4	–5.78E+3	–3.69E+4	–4.45E+2
	0°	(3)	1620	–3.50E+4	–3.59E+5	–1.75E+4	–1.46E+3	1.75E+4	–5.34E+3	–3.69E+4	8.90E+2
8 (Ext. Wall)	270°	(4)	2491	7.30E+3	–1.07E+5	–5.84E+3	–0.0	7.30E+3	8.90E+2	4.00E+3	0.0
9 (Ext. Wall)	0°	(4)	1650	–8.90E+4	–4.41E+5	–9.92E+4	2.92E+3	2.04E+4	9.34E+3	7.34E+4	4.89E+3
10 (Ext. Wall)	0°	(4)	1673	–4.38E+4	–5.84E+4	1.71E+5	2.92E+3	2.92E+4	–1.56E+4	–9.65E+4	–7.56E+3
11 (Ext. Wall)	180°	(4)	1998	–8.76E+3	–2.19E+4	–2.92E+4	2.92E+3	–2.92E+3	1.78E+3	–1.78E+3	–2.67E+3
12 (Ext. Wall)	0°	(4)	1705	–7.30E+4	–4.23E+4	–1.07E+5	1.46E+3	3.36E+4	7.56E+3	4.67E+4	–3.56E+3

Table 3H.3-4
Forces and Moments in Critical Elements for Soil Pressure @ Rest (H)

				Forces in (N/m)					Moments in (N•m/m)		
Section	Location	(Region)	El. #	F_x	F_y	F_{xy}	F_{xz}	F_{yz}	M_x	M_y	M_{xy}
0 (BASEMAT)	90°	(BMT–1)	55	–1.17E+6	–1.00E+6	6.71E+4	–2.48E+4	6.28E+4	8.12E+5	2.22E+5	1.14E+5
	90°	(BMT–1)	146	–1.16E+6	–9.84E+5	–5.84E+4	–4.09E+4	–1.17E+4	6.39E+5	2.78E+5	–1.15E+5
	270°	(BMT–1)	126	–1.76E+6	–1.67E+6	3.06E+4	1.97E+5	–1.75E+4	2.57E+6	5.00E+5	3.07E+4
	0°	(BMT–1)	9	–1.31E+6	–1.18E+6	1.55E+5	–3.36E+4	3.50E+4	1.07E+5	8.21E+5	7.38E+4
	Middle	(BMT–2)	52	–1.39E+6	–1.32E+6	2.96E+5	6.57E+4	4.82E+4	1.02E+6	7.05E+5	–1.91E+5
	Middle	(BMT–2)	60	–1.20E+6	–1.14E+6	5.40E+4	–1.90E+4	3.36E+4	6.98E+4	3.00E+5	2.62E+4
	Middle	(BMT–2)	75	–1.18E+6	–1.03E+6	–1.02E+4	–4.52E+4	1.17E+4	4.41E+5	2.11E+5	4.98E+4
	Middle	(BMT–2)	129	–1.18E+6	–1.02E+6	–3.36E+4	–4.38E+4	–1.75E+4	4.54E+5	2.31E+5	–3.60E+4
	Middle	(BMT–2)	178	–1.37E+6	–1.30E+6	–2.64E+5	6.28E+4	–5.11E+4	1.05E+6	7.42E+5	1.89E+5
1 (EXT. WALL)	90°	(1A)	2108	–4.52E+5	3.36E+4	1.11E+5	–4.38E+3	5.62E+5	3.11E+4	1.89E+5	6.67E+3
	90°	(1A)	2109	–4.33E+5	4.82E+4	1.61E+5	–8.76E+3	5.63E+5	2.98E+4	1.86E+5	1.25E+4
	270°	(1B)	2405	–6.86E+5	–1.09E+5	1.90E+4	3.94E+4	–1.13E+6	2.35E+5	1.48E+6	4.40E+4
	270°	(1B)	2406	–6.99E+5	–1.24E+5	–1.31E+4	1.02E+4	–1.14E+6	2.54E+5	1.53E+6	1.25E+4
	270°	(1B)	2408	–6.86E+5	–1.01E+5	–7.15E+4	–3.94E+4	–1.13E+6	2.34E+5	1.48E+6	–4.31E+4
	0°	(1A)	1508	–5.18E+5	–1.75E+4	1.58E+5	4.38E+3	–5.69E+5	3.25E+4	1.83E+5	6.67E+3
	0°	(1B)	1512	–5.34E+5	3.06E+4	1.07E+5	7.88E+4	–6.00E+5	1.10E+5	3.03E+5	9.87E+4
	0°	(1B)	1513	–5.60E+5	1.46E+4	8.46E+4	9.63E+4	–7.24E+5	1.02E+5	5.41E+5	1.34E+5
2 (EXT. WALL)	270°	(2B)	2424	–8.03E+5	–3.06E+5	–1.07E+5	5.17E+5	1.02E+5	6.05E+5	–5.52E+4	–2.62E+5
3 (EXT. WALL)	90°	(2A)	2134	–6.44E+5	1.01E+5	2.00E+5	–8.76E+3	–3.82E+5	–6.98E+4	–1.23E+5	9.34E+3
	270°	(2B)	2425	–9.65E+5	–3.39E+5	1.61E+4	–6.90E+5	7.15E+4	8.78E+5	–9.12E+4	1.04E+5
4 (EXT. WALL)	270°	(2B)	2441	–5.71E+5	–3.50E+4	1.02E+4	–4.38E+3	1.26E+5	–4.23E+5	–1.57E+6	–4.00E+3
	270°	(2B)	2442	–5.60E+5	–5.40E+4	–1.17E+4	0.0	1.23E+5	–3.80E+5	–1.62E+6	–8.90E+2
	0°	(2A)	1565	–4.38E+5	3.06E+4	–1.17E+5	7.30E+4	–2.13E+5	1.38E+5	–3.74E+4	1.87E+4
5 (Ext. Wall)	90°	(2A)	2458	–5.52E+5	1.46E+3	7.30E+3	1.04E+5	2.67E+5	–3.67E+5	–7.12E+5	2.64E+5
	270°	(2B)	2155	–4.98E+5	7.73E+4	0.0	1.46E+3	–8.76E+3	–5.16E+4	–2.38E+5	–2.67E+3
	270°	(2B)	2460	–5.71E+5	–2.06E+5	4.67E+4	4.41E+5	–1.05E+5	6.59E+5	–3.11E+3	2.31E+5
	0°	(2A)	1579	–4.35E+5	1.31E+4	–8.90E+4	–7.30E+3	5.84E+3	–4.85E+4	–2.54E+5	–9.34E+3
6 (Ext. Wall)	0°	(2A)	1597	–4.26E+5	8.76E+3	–7.30E+4	4.38E+3	1.15E+5	–1.87E+4	–1.03E+5	–1.38E+4
7 (Ext. Wall)	270°	(3)	2478	–1.93E+5	4.38E+3	–1.02E+4	2.92E+3	–4.09E+4	3.96E+4	2.22E+5	–2.67E+3
	0°	(3)	1614	–7.15E+4	6.86E+4	1.17E+4	1.46E+3	–8.76E+3	6.23E+3	4.40E+4	–2.22E+3
	0°	(3)	1620	–8.46E+4	1.01E+5	–2.19E+4	1.75E+4	–7.30E+3	9.34E+3	7.38E+4	–9.79E+3
8 (Ext. Wall)	270°	(4)	2491	–1.18E+5	–1.17E+4	8.76E+3	–0.0	–3.94E+4	2.09E+4	1.03E+5	–0.0
9 (Ext. Wall)	0°	(4)	1650	–3.06E+4	1.17E+4	–4.82E+4	–0.0	–8.76E+3	–8.90E+3	–4.00E+3	4.45E+2
10 (Ext. Wall)	0°	(4)	1673	–8.76E+4	–1.31E+4	7.30E+3	1.46E+3	4.38E+3	–3.56E+3	–1.51E+4	3.11E+3
11 (Ext. Wall)	180°	(4)	1998	–4.38E+3	2.77E+4	1.46E+3	2.92E+3	–1.46E+3	2.22E+3	–1.33E+3	–4.45E+2
12 (Ext. Wall)	0°	(4)	1705	–5.84E+3	–2.92E+3	–1.46E+3	–0.0	2.92E+3	8.90E+2	5.34E+3	4.45E+2

Table 3H.3-5
Forces and Moments in Critical Elements for Seismic Soil Pressure (H')

				Forces in (N/m)					Moments in (N•m/m)		
Section	Location	(Region)	El. #	F_x	F_y	F_{xy}	F_{xz}	F_{yz}	M_x	M_y	M_{xy}
0 (BASEMAT)	90°	(BMT–1)	55	–1.56E+6	–1.28E+6	2.31E+5	–3.94E+4	9.34E+4	1.25E+6	3.93E+5	1.58E+5
	90°	(BMT–1)	146	–1.59E+6	–1.27E+6	–1.36E+5	–7.00E+4	–1.46E+4	9.60E+5	4.41E+5	–1.73E+5
	270°	(BMT–1)	126	–2.80E+6	–2.69E+6	2.77E+4	4.09E+5	–4.09E+4	5.09E+6	1.02E+6	8.10E+4
	0°	(BMT–1)	9	–1.93E+6	–1.53E+6	3.12E+5	–7.73E+4	4.67E+4	1.93E+5	1.31E+6	1.70E+5
	Middle	(BMT–2)	52	–2.08E+6	1.93E+6	5.05E+5	1.26E+5	9.05E+4	1.99E+6	1.35E+6	–3.03E+5
	Middle	(BMT–2)	60	–1.74E+6	–1.50E+6	1.50E+5	–3.06E+4	5.69E+4	1.02E+5	5.04E+5	4.36E+4
	Middle	(BMT–2)	75	–1.64E+6	–1.34E+6	4.52E+4	–7.00E+4	2.19E+4	6.72E+5	3.57E+5	5.78E+4
	Middle	(BMT–2)	129	–1.64E+6	–1.33E+6	–6.13E+4	–7.00E+4	–2.33E+4	6.74E+5	3.60E+5	–6.14E+4
	Middle	(BMT–2)	178	–2.07E+6	–1.92E+6	–4.87E+5	1.24E+5	–9.05E+4	2.00E+6	1.35E+6	3.11E+5
1 (EXT. WALL)	90°	(1A)	2108	–7.08E+5	5.84E+4	2.26E+5	–1.02E+4	7.89E+5	6.85E+4	4.03E+5	1.42E+4
	90°	(1A)	2109	–6.60E+5	7.00E+4	3.56E+5	–1.75E+4	7.91E+5	6.58E+4	3.89E+5	2.40E+4
	270°	(1B)	2405	–1.19E+6	–2.17E+5	1.24E+5	8.61E+4	–2.02E+6	5.08E+5	3.20E+6	9.47E+4
	270°	(1B)	2406	–1.22E+6	–2.50E+5	3.36E+4	2.33E+4	–2.04E+6	5.53E+5	3.31E+6	2.58E+4
	270°	(1B)	2408	–1.19E+6	–2.17E+5	–1.43E+5	–8.61E+4	–2.02E+6	5.08E+5	3.20E+6	–9.52E+4
	0°	(1A)	1508	–8.36E+5	–2.19E+4	3.18E+5	8.76E+3	–8.00E+5	7.21E+4	4.00E+5	1.47E+4
	0°	(1B)	1512	–8.80E+5	9.78E+4	1.78E+5	1.72E+5	–8.61E+5	2.37E+5	6.48E+5	2.16E+5
	0°	(1B)	1513	–9.38E+5	5.98E+4	1.07E+5	2.12E+5	–1.14E+6	2.21E+5	1.16E+6	2.94E+5
2 (EXT. WALL)	270°	(2B)	2424	–1.34E+6	–5.72E+5	–2.23E+5	8.65E+5	2.07E+5	1.22E+6	3.91E+4	–5.60E+5
3 (EXT. WALL)	90°	(2A)	2134	–1.33E+6	1.78E+5	5.02E+5	–2.33E+4	–6.28E+5	–8.36E+4	3.34E+4	4.94E+4
	270°	(2B)	2425	–1.88E+6	–7.50E+5	1.58E+5	–1.36E+6	1.52E+5	1.85E+6	–1.13E+5	3.09E+5
4 (EXT. WALL)	270°	(2B)	2441	–1.34E+6	–4.38E+3	1.11E+5	–7.30E+3	3.79E+4	–9.63E+5	–3.60E+6	–4.89E+3
	270°	(2B)	2442	–1.32E+6	–3.65E+4	3.50E+4	0.0	3.36E+4	–8.67E+5	–3.72E+6	–4.45E+2
	0°	(2A)	1565	–1.03E+6	1.26E+5	–2.67E+5	1.59E+5	–7.06E+5	2.70E+5	–2.51E+5	3.60E+4
5 (Ext. Wall)	90°	(2A)	2458	–1.44E+6	3.79E+4	–4.82E+4	2.31E+5	5.03E+5	–9.37E+5	–2.08E+6	5.65E+5
	270°	(2B)	2155	–1.27E+6	2.50E+5	–2.63E+4	1.46E+3	5.55E+4	–2.16E+5	–1.08E+6	–4.45E+3
	270°	(2B)	2460	–1.64E+6	–7.22E+5	5.55E+4	1.22E+6	–2.50E+5	1.57E+6	–1.54E+5	4.76E+5
	0°	(2A)	1579	–1.13E+6	8.90E+4	–2.38E+5	–1.61E+4	–6.71E+4	–2.11E+5	–1.12E+6	–1.69E+4
6 (Ext. Wall)	0°	(2A)	1597	–1.24E+6	8.46E+4	–1.88E+5	5.84E+3	5.22E+5	–1.04E+5	–5.55E+5	–2.98E+4
7 (Ext. Wall)	270°	(3)	2478	–5.85E+5	1.27E+5	–3.65E+4	4.38E+3	–9.34E+4	8.98E+4	5.06E+5	–5.78E+3
	0°	(3)	1614	–2.00E+5	4.09E+5	1.46E+3	1.46E+3	–2.04E+4	1.56E+4	1.17E+5	–5.78E+3
	0°	(3)	1620	–2.26E+5	5.09E+5	–2.92E+4	3.94E+4	–1.75E+4	2.31E+4	1.83E+5	–2.00E+4
8 (Ext. Wall)	270°	(4)	2491	–3.87E+5	5.98E+4	3.94E+4	–0.0	–8.90E+4	4.76E+4	2.37E+5	0.0
9 (Ext. Wall)	0°	(4)	1650	–1.20E+5	1.50E+5	–1.84E+5	–1.46E+3	–2.19E+4	–1.33E+3	–4.00E+3	1.33E+3
10 (Ext. Wall)	0°	(4)	1673	–3.46E+5	–2.63E+4	2.33E+4	–4.38E+3	8.76E+3	–7.12E+3	–3.11E+4	7.12E+3
11 (Ext. Wall)	180°	(4)	1998	–7.30E+3	5.11E+4	2.92E+3	7.30E+3	–2.92E+3	4.89E+3	–3.56E+4	–4.45E+2
12 (Ext. Wall)	0°	(4)	1705	–1.21E+5	–4.38E+3	–2.92E+3	–0.0	5.84E+3	1.78E+3	9.34E+3	8.90E+2

Table 3H.3-6
Forces and Moments in Critical Elements for Seismic Load (E)

				Forces in (N/m)					Moments in (N•m/m)		
Section	Location	(Region)	El. #	F_x	F_y	F_{xy}	F_{xz}	F_{yz}	M_x	M_y	M_{xy}
0 (BASEMAT)	90°	(BMT–1)	55	2.90E+5	1.48E+6	1.89E+6	9.79E+5	3.06E+5	5.48E+5	9.02E+5	8.42E+5
	90°	(BMT–1)	146	5.06E+5	1.08E+6	1.76E+6	4.06E+5	2.54E+5	2.34E+6	1.77E+6	9.68E+5
	270°	(BMT–1)	126	1.20E+5	1.80E+6	2.03E+6	7.87E+5	5.11E+5	6.32E+5	5.16E+5	1.26E+6
	0°	(BMT–1)	9	4.67E+6	1.15E+5	1.67E+6	2.71E+5	7.18E+5	6.71E+5	1.27E+6	7.51E+5
	Middle	(BMT–2)	52	9.59E+5	6.39E+5	1.22E+6	2.83E+5	2.38E+5	2.48E+6	2.67E+6	5.91E+5
	Middle	(BMT–2)	60	1.47E+6	4.42E+5	8.68E+5	3.59E+5	3.68E+5	1.50E+6	2.84E+6	4.25E+5
	Middle	(BMT–2)	75	5.62E+5	8.64E+5	1.70E+6	9.19E+4	3.24E+5	2.72E+6	1.82E-6	9.52E+5
	Middle	(BMT–2)	129	5.12E+5	8.49E+5	1.70E+6	9.63E+4	3.25E+5	2.72E+6	1.84E+6	9.43E+5
	Middle	(BMT–2)	178	9.22E+5	6.30E+5	1.23E+6	2.83E+5	2.36E+5	2.50E+6	2.72E+6	5.91E+5
1 (EXT. WALL)	90°	(1A)	2108	9.79E+5	1.30E+6	2.14E+6	1.61E+4	1.50E+5	1.31E+5	6.81E+5	6.67E+4
	90°	(1A)	2109	9.06E+5	1.45E+6	1.99E+6	1.61E+4	1.63E+5	1.53E+5	7.22E+5	5.60E+4
	270°	(1B)	2405	9.41E+5	1.21E+6	1.93E+6	3.36E+4	5.69E+4	1.07E+5	5.00E+5	8.67E+4
	270°	(1B)	2406	9.94E+5	1.11E+6	2.01E+6	3.65E+4	4.67E+4	8.58E+4	4.56E+5	9.74E+4
	270°	(1B)	2408	1.01E+6	1.12E+6	2.01E+6	3.65E+4	4.67E+4	8.54E+4	4.55E+5	9.70E+4
	0°	(1A)	1508	2.32E+6	1.01E+6	2.02E+6	1.17E+4	1.75E+4	3.42E+4	1.09E+5	3.96E+4
	0°	(1B)	1512	2.03E+6	1.22E+6	1.51E+6	1.90E+4	7.59E+4	7.61E+4	4.07E+5	4.45E+4
	0°	(1B)	1513	1.85E+6	1.34E+6	1.26E+6	1.61E+4	7.44E+4	9.39E+4	4.54E+5	3.65E+4
2 (EXT. WALL)	270°	(2B)	2424	1.59E+5	2.07E+5	1.19E+6	1.24E+5	4.52E+4	1.09E+5	1.05E+5	4.14E+4
3 (EXT. WALL)	90°	(2A)	2134	9.06E+5	1.45E+6	1.99E+6	1.61E+4	1.63E+5	1.53E+5	7.22E+5	5.60E+4
	270°	(2B)	2425	1.18E+5	3.91E+5	1.28E+6	7.73E+4	1.31E+4	1.32E+5	3.56E+4	6.49E+4
4 (EXT. WALL)	270°	(2B)	2441	1.62E+5	8.27E+5	1.79E+6	1.31E+4	4.23E+4	5.87E+4	1.16E+5	6.67E+3
	270°	(2B)	2442	1.02E+5	8.04E+5	1.84E+6	1.46E+4	3.94E+4	4.58E+4	1.20E+5	1.51E+4
	0°	(2A)	1565	1.43E+5	8.76E+5	1.76E+6	7.30E+3	3.36E+4	3.02E+4	8.85E+4	3.02E+4
5 (Ext. Wall)	90°	(2A)	2458	1.94E+5	6.46E+5	1.57E+6	1.17E+4	2.92E+4	3.51E+4	1.96E+4	4.85E+4
	270°	(2B)	2155	3.60E+5	7.44E+5	1.94E+6	7.30E+3	2.77E+4	1.38E+4	1.21E+5	1.07E+4
	270°	(2B)	2460	5.25E+4	4.19E+5	1.43E+6	2.77E+4	2.33E+4	6.72E+4	1.69E+4	6.54E+4
	0°	(2A)	1579	1.57E+6	1.06E+6	1.88E+6	4.38E+3	3.36E+4	7.56E+3	4.71E+4	2.09E+4
6 (Ext. Wall)	0°	(2A)	1597	3.36E+6	1.14E+6	1.79E+6	1.17E+4	3.65E+4	2.67E+4	8.76E+4	1.25E+4
7 (Ext. Wall)	270°	(3)	2478	2.93E+5	3.60E+5	1.31E+6	1.46E+3	1.02E+4	6.67E+3	4.09E+4	1.78E+3
	0°	(3)	1614	1.07E+6	2.29E+6	1.56E+6	1.46E+3	2.33E+4	6.67E+3	4.54E+4	4.89E+3
	0°	(3)	1620	7.87E+5	2.44E+6	1.14E+6	2.92E+3	2.33E+4	5.34E+3	3.91E+4	6.23E+3
8 (Ext. Wall)	270°	(4)	2491	3.53E+5	2.69E+5	1.29E+6	1.46E+3	1.02E+4	1.33E+3	1.33E+4	2.22E+3
9 (Ext. Wall)	0°	(4)	1650	8.01E+5	1.12E+6	1.20E+6	2.92E+3	2.48E+4	1.38E+4	8.63E+4	4.89E+3
10 (Ext. Wall)	0°	(4)	1673	2.79E+6	6.06E+5	7.46E+5	5.84E+3	2.77E+4	1.91E+4	8.32E+4	9.79E+3
11 (Ext. Wall)	180°	(4)	1998	1.82E+5	4.82E+4	5.41E+5	7.30E+3	2.92E+3	8.45E+3	4.00E+3	3.56E+3
12 (Ext. Wall)	0°	(4)	1705	2.38E+6	9.78E+4	4.99E+5	2.92E+3	3.21E+4	9.34E+3	5.60E+4	4.00E+3

Table 3H.3-7
Forces and Moments in Critical Elements for Load Combination: 1.4D + 1.7L + 1.7H

				Forces in (N/m)					Moments in (N•m/m)		
Section	Location	(Region)	El. #	F_x	F_y	F_{xy}	F_{xz}	F_{yz}	M_x	M_y	M_{xy}
0 (BASEMAT)	90°	(BMT–1)	55	–2.22E+6	–2.02E+6	1.04E+5	–1.39E+6	–3.43E+5	8.61E+5	–3.32E+5	–6.96E+5
	90°	(BMT–1)	146	–2.21E+6	–1.97E+6	–8.03E+4	–4.36E+5	–1.46E+4	–1.83E+6	–1.14E+6	1.04E+6
	270°	(BMT–1)	126	–3.06E+6	–2.99E+6	5.69E+4	1.30E+6	4.52E+4	3.83E+6	4.80E+5	9.79E+3
	0°	(BMT–1)	9	–2.65E+6	–2.28E+6	1.88E+5	–1.17E+5	1.23E+6	–5.69E+4	1.15E+6	2.85E+5
	Middle	(BMT–2)	52	–2.51E+6	–2.39E+6	4.16E+5	8.33E+5	–4.74E+5	3.96E+4	–4.22E+5	1.07E+6
	Middle	(BMT–2)	60	–2.26E+6	–2.19E+6	4.67E+4	9.98E+5	9.47E+5	–6.22E+5	–1.65E+6	–3.65E+5
	Middle	(BMT–2)	75	–2.25E+6	–2.02E+6	–2.77E+4	–2.29E+5	–2.19E+4	–1.90E+6	–2.26E+6	–4.20E+5
	Middle	(BMT–2)	129	–2.22E+6	–2.01E+6	–4.67E+4	–2.31E+5	2.33E+4	–1.90E+6	–2.27E+6	4.39E+5
	Middle	(BMT–2)	178	–2.47E+6	–2.38E+6	–3.78E+5	8.36E+5	4.71E+5	3.47E+4	–4.27E+5	–1.10E+6
1 (EXT. WALL)	90°	(1A)	2108	–1.15E+6	–1.98E+6	1.58E+5	0.0	1.21E+6	3.14E+5	1.69E+6	–1.51E+4
	90°	(1A)	2109	–1.10E+6	–1.96E+6	2.44E+5	0.0	1.19E+6	2.95E+5	1.57E+6	–2.89E+4
	270°	(1B)	2405	–1.43E+6	–1.69E+6	3.65E+4	4.38E+4	–1.99E+6	5.62E+5	3.29E+6	3.29E+4
	270°	(1B)	2406	–1.46E+6	–1.71E+6	–1.75E+4	1.17E+4	–2.00E+6	5.96E+5	3.39E+6	7.56E+3
	270°	(1B)	2408	–1.44E+6	–1.70E+6	–1.20E+5	–4.23E+4	–1.99E+6	5.61E+5	3.29E+6	–3.11E+4
	0°	(1A)	1508	–1.34E+6	–1.71E+6	2.90E+5	5.84E+3	–1.22E+6	3.12E+5	1.73E+6	1.29E+4
	0°	(1B)	1512	–1.17E+6	–1.62E+6	–3.50E+4	1.20E+5	–1.26E+6	4.01E+5	1.76E+6	1.62E+5
	0°	(1B)	1513	–1.15E+6	–1.62E+6	–1.02E+4	1.46E+5	–1.42E+6	3.83E+5	2.04E+6	2.18E+5
2 (EXT. WALL)	270°	(2B)	2424	–1.11E+6	–6.33E+5	–4.64E+5	6.95E+5	–1.07E+5	8.24E+5	–1.82E+4	–4.03E+5
3 (EXT. WALL)	90°	(2A)	2134	–9.40E+5	–1.41E+6	5.40E+5	–1.31E+4	–4.77E+5	–6.98E+4	–8.23E+4	7.21E+4
	270°	(2B)	2425	–1.44E+6	–7.87E+5	2.85E+5	–1.05E+6	–1.14E+5	1.25E+6	–1.53E+5	1.90E+5
4 (EXT. WALL)	270°	(2B)	2441	–8.48E+5	–1.20E+6	1.04E+5	1.46E+3	1.58E+5	–6.16E+5	–2.39E+6	1.07E+4
	270°	(2B)	2442	–8.57E+5	–1.25E+6	8.76E+3	2.92E+3	1.50E+5	–5.45E+5	–2.46E+6	4.89E+3
	0°	(2A)	1565	–6.57E+5	–1.05E+6	–5.49E+5	1.04E+5	–2.45E+5	1.21E+5	–4.03E+5	1.56E+4
5 (Ext. Wall)	90°	(2A)	2458	–7.65E+5	–8.38E+5	–2.47E+5	1.49E+5	4.29E+5	–5.74E+5	–1.12E+6	3.63E+5
	270°	(2B)	2155	–6.74E+5	–1.17E+6	4.09E+4	2.92E+3	–1.18E+5	–1.08E+5	–5.12E+5	–3.56E+3
	270°	(2B)	2460	–8.70E+5	–7.49E+5	–7.73E+4	6.84E+5	–1.44E+5	9.75E+5	–1.16E+4	3.49E+5
	0°	(2A)	1579	–5.15E+5	–9.57E+5	2.87E+5	–1.46E+3	1.36E+5	–9.34E+4	–5.07E+5	–1.33E+2
6 (Ext. Wall)	0°	(2A)	1597	–3.50E+5	–7.76E+5	2.90E+5	1.02E+4	3.23E+5	9.34E+3	6.72E+4	–7.12E+3
7 (Ext. Wall)	270°	(3)	2478	–2.63E+5	–6.20E+5	2.92E+3	1.46E+3	–4.96E+4	5.43E+4	3.04E+5	2.67E+3
	0°	(3)	1614	–2.54E+5	–1.60E+6	1.27E+5	–0.0	2.48E+4	–2.22E+3	–8.90E+2	–4.89E+3
	0°	(3)	1620	–2.80E+5	–1.61E+6	–1.81E+5	2.33E+4	2.48E+4	4.00E+3	4.36E+4	–1.16E+4
8 (Ext. Wall)	270°	(4)	2491	–1.56E+5	–5.52E+5	–8.76E+3	–0.0	–4.82E+4	3.20E+4	1.61E+5	4.45E+2
9 (Ext. Wall)	0°	(4)	1650	–3.14E+5	–1.67E+6	–3.41E+5	4.38E+3	2.63E+4	1.73E+4	1.42E+5	1.07E+4
10 (Ext. Wall)	0°	(4)	1673	–1.71E+5	–3.33E+5	6.25E+5	2.92E+3	7.30E+4	–3.83E+4	–2.22E+5	–1.07E+4
11 (Ext. Wall)	180°	(4)	1998	–2.04E+4	–7.73E+4	–5.25E+4	7.30E+3	–1.02E+4	5.78E+3	–2.22E+3	–7.56E+3
12 (Ext. Wall)	0°	(4)	1705	–1.31E+5	–1.68E+5	–3.77E+5	2.92E+3	8.61E+4	2.54E+4	1.48E+5	–5.34E+3

Table 3H.3-8
Forces and Moments in Critical Elements for Load Combination (D + L + H’ ± E)Max. Tension

				Forces in (N/m)					Moments in (N•m/m)		
Section	Location	(Region)	El. #	F_x	F_y	F_{xy}	F_{xz}	F_{yz}	M_x	M_y	M_{xy}
0 (BASEMAT)	90°	(BMT–1)	55	–1.43E+6	–2.19E+4	2.11E+6	–1.93E+6	–5.25E+5	1.47E+6	–9.91E+5	–1.31E+6
	90°	(BMT–1)	146	–1.24E+6	–3.88E+5	–1.89E+6	–7.37E+5	–2.69E+5	–3.34E+6	–2.44E+6	1.65E+6
	270°	(BMT–1)	126	–2.73E+6	–9.95E+5	2.06E+6	1.87E+6	5.22E+5	5.37E+6	1.27E+6	1.31E+6
	0°	(BMT–1)	9	2.43E+6	–1.61E+6	1.93E+6	–3.85E+5	1.56E+6	7.10E+5	2.45E+6	1.02E+6
	Middle	(BMT–2)	52	–1.22E+6	–1.39E+6	1.66E+6	8.81E+5	–4.92E+5	3.26E+6	2.86E+6	1.27E+6
	Middle	(BMT–2)	60	–4.33E+5	–1.24E+6	9.88E+5	9.86E+5	9.91E+5	–1.96E+6	–3.88E+6	–6.59E+5
	Middle	(BMT–2)	75	–1.25E+6	–6.58E+5	1.73E+6	–2.71E+5	–3.28E+5	–3.88E+6	–3.24E+6	–1.25E+6
	Middle	(BMT–2)	129	–1.29E+6	–6.76E+5	–1.76E+6	–2.77E+5	3.36E+5	–3.90E+6	–3.29E+6	1.24E+6
	Middle	(BMT–2)	178	–1.24E+6	–1.41E+6	–1.67E+6	8.86E+5	4.93E+5	3.25E+6	2.86E+6	–1.28E+6
1 (EXT. WALL)	90°	(1A)	2108	8.76E+3	–1.75E+4	2.35E+6	–2.04E+4	1.12E+6	3.81E+5	2.03E+6	–7.12E+4
	90°	(1A)	2109	1.46E+3	1.42E+5	2.33E+6	–2.33E+4	1.12E+6	3.88E+5	1.98E+6	–6.72E+4
	270°	(1B)	2405	–4.25E+5	–4.82E+4	2.06E+6	1.04E+5	–2.13E+6	7.31E+5	4.25E+6	1.51E+5
	270°	(1B)	2406	–4.10E+5	–1.72E+5	2.04E+6	5.40E+4	–2.13E+6	7.55E+5	4.34E+6	1.13E+5
	270°	(1B)	2408	–3.98E+5	–3.94E+4	–2.09E+6	–1.02E+5	–2.13E+6	7.30E+5	4.25E+6	–1.50E+5
	0°	(1A)	1508	1.17E+6	–1.42E+5	2.35E+6	2.04E+4	–9.97E+5	2.86E+5	1.50E+6	5.52E+4
	0°	(1B)	1512	9.59E+5	1.82E+5	1.54E+6	1.81E+5	–1.10E+6	4.64E+5	1.93E+6	2.56E+5
	0°	(1B)	1513	7.65E+5	2.79E+5	–1.26E+6	2.16E+5	–1.35E+6	4.63E+5	2.40E+6	3.24E+5
2 (EXT. WALL)	270°	(2B)	2424	–1.00E+6	–4.25E+5	–1.61E+6	8.61E+5	2.06E+5	1.18E+6	1.98E+5	–5.73E+5
3 (EXT. WALL)	90°	(2A)	2134	–9.79E+5	2.13E+5	2.36E+6	–2.63E+4	–6.44E+5	–8.45E+4	1.72E+5	1.21E+5
	270°	(2B)	2425	–1.63E+6	–4.87E+5	1.61E+6	–1.35E+6	1.61E+5	1.81E+6	–1.45E+5	3.84E+5
4 (EXT. WALL)	270°	(2B)	2441	–1.09E+6	4.67E+4	1.96E+6	–1.46E+4	–4.38E+4	–9.49E+5	–3.53E+6	1.47E+4
	270°	(2B)	2442	–1.15E+6	–2.04E+4	1.90E+6	1.75E+4	–4.96E+4	–8.42E+5	–3.63E+6	1.91E+4
	0°	(2A)	1565	–8.27E+5	2.55E+5	–2.26E+6	1.52E+5	–6.63E+5	2.23E+5	–5.66E+5	5.56E+4
5 (Ext. Wall)	90°	(2A)	2458	–1.13E+6	1.20E+5	–1.79E+6	2.23E+5	5.14E+5	–9.39E+5	–2.04E+6	5.53E+5
	270°	(2B)	2155	–7.94E+5	1.20E+5	1.94E+6	1.02E+4	–3.94E+4	–2.45E+5	–1.28E+6	–1.42E+4
	270°	(2B)	2460	–1.52E+6	–5.63E+5	–1.48E+6	1.20E+6	–2.47E+5	1.53E+6	–1.76E+5	5.11E+5
	0°	(2A)	1579	5.92E+5	5.01E+5	1.94E+6	–1.46E+4	4.67E+4	–2.27E+5	–1.23E+6	–3.65E+4
6 (Ext. Wall)	0°	(2A)	1597	2.37E+6	7.02E+5	1.89E+6	2.04E+4	6.39E+5	–1.06E+5	–4.97E+5	–3.25E+4
7 (Ext. Wall)	270°	(3)	2478	–2.47E+5	6.28E+4	–1.33E+6	5.84E+3	–9.05E+4	8.76E+4	4.98E+5	–6.23E+3
	0°	(3)	1614	7.79E+5	1.55E+6	1.64E+6	1.46E+3	2.77E+4	1.47E+4	1.17E+5	–1.11E+4
	0°	(3)	1620	4.71E+5	1.76E+6	–1.27E+6	3.79E+4	2.77E+4	2.09E+4	1.72E+5	–2.31E+4
8 (Ext. Wall)	270°	(4)	2491	–2.92E+3	–2.92E+4	1.31E+6	–1.46E+3	–8.61E+4	4.63E+4	2.40E+5	2.22E+3
9 (Ext. Wall)	0°	(4)	1650	5.15E+5	1.58E+5	–1.55E+6	4.38E+3	2.77E+4	2.36E+4	1.73E+5	1.16E+4
10 (Ext. Wall)	0°	(4)	1673	2.44E+6	3.71E+5	1.17E+6	–5.84E+3	7.73E+4	–4.63E+4	–2.34E+5	–1.25E+4
11 (Ext. Wall)	180°	(4)	1998	1.68E+5	1.46E+4	–5.72E+5	1.46E+4	–1.17E+4	1.42E+4	–7.12E+3	–8.90E+3
12 (Ext. Wall)	0°	(4)	1705	2.18E+6	–1.31E+4	–7.46E+5	4.38E+3	8.90E+4	2.62E+4	1.54E+5	–6.67E+3

Table 3H.3-9
Forces and Moments in Critical Elements for Load Combination (D + L + H’ ± E)Max. Compression

				Forces in (N/m)					Moments in (N•m/m)		
Section	**Location**	**(Region)**	**El. #**	F_x	F_y	F_{xy}	F_{xz}	F_{yz}	M_x	M_y	M_{xy}
0 (BASEMAT)	90°	(BMT–1)	55	–2.01E+6	–2.97E+6	2.11E+6	–1.93E+6	–5.25E+5	1.47E+6	–9.91E+5	–1.31E+6
	90°	(BMT–1)	146	–2.26E+6	–2.55E+6	–1.89E+6	–7.37E+5	–2.69E+5	–3.34E+6	–2.44E+6	1.65E+6
	270°	(BMT–1)	126	–2.97E+6	–4.59E+6	2.06E+6	1.87E+6	5.22E+5	5.37E+6	1.27E+6	1.31E+6
	0°	(BMT–1)	9	–6.91E+6	–1.84E+6	1.93E+6	–3.85E+5	1.56E+6	7.10E+5	2.45E+6	1.02E+6
	Middle	(BMT–2)	52	–3.14E+6	–2.67E+6	1.66E+6	8.81E+5	–4.92E+5	3.26E+6	2.86E+6	1.27E+6
	Middle	(BMT–2)	60	–3.36E+6	–2.13E+6	9.88E+5	9.86E+5	9.91E+5	–1.96E+6	–3.88E+6	–6.59E+5
	Middle	(BMT–2)	75	–2.37E+6	–2.39E+6	1.73E+6	–2.71E+5	–3.28E+5	–3.88E+6	–3.24E+6	–1.25E+6
	Middle	(BMT–2)	129	–2.32E+6	–2.37E+6	–1.76E+6	–2.77E+5	3.36E+5	–3.90E+6	–3.29E+6	1.24E+6
	Middle	(BMT–2)	178	–3.09E+6	–2.67E+6	–1.67E+6	8.86E+5	4.93E+5	3.25E+6	2.86E+6	–1.28E+6
1 (EXT. WALL)	90°	(1A)	2108	–1.95E+6	–2.62E+6	2.35E+6	–2.04E+4	1.12E+6	3.81E+5	2.03E+6	–7.12E+4
	90°	(1A)	2109	–1.81E+6	–2.75E+6	2.33E+6	–2.33E+4	1.12E+6	3.88E+5	1.98E+6	–6.72E+4
	270°	(1B)	2405	–2.31E+6	–2.46E+6	2.06E+6	1.04E+5	–2.13E+6	7.31E+5	4.25E+6	1.51E+5
	270°	(1B)	2406	–2.40E+6	–2.39E+6	2.04E+6	5.40E+4	–2.13E+6	7.55E+5	4.34E+6	1.13E+5
	270°	(1B)	2408	–2.35E+6	–2.49E+6	–2.09E+6	–1.02E+5	–2.13E+6	7.30E+5	4.25E+6	–1.50E+5
	0°	(1A)	1508	–3.48E+6	–2.17E+6	2.35E+6	2.04E+4	–9.97E+5	2.86E+5	1.50E+6	5.52E+4
	0°	(1B)	1512	–3.10E+6	–2.27E+6	1.54E+6	1.81E+5	–1.10E+6	4.64E+5	1.93E+6	2.56E+5
	0°	(1B)	1513	–2.93E+6	–2.41E+6	–1.26E+6	2.16E+5	–1.35E+6	4.63E+5	2.40E+6	3.24E+5
2 (EXT. WALL)	270°	(2B)	2424	–1.32E+6	–8.39E+5	–1.61E+6	8.61E+5	2.06E+5	1.18E+6	1.98E+5	–5.73E+5
3 (EXT. WALL)	90°	(2A)	2134	–1.46E+6	–1.97E+6	2.36E+6	–2.63E+4	–6.44E+5	–8.45E+4	1.72E+5	1.21E+5
	270°	(2B)	2425	–1.86E+6	–1.27E+6	1.61E+6	–1.35E+6	1.61E+5	1.81E+6	–1.45E+5	3.84E+5
4 (EXT. WALL)	270°	(2B)	2441	–1.42E+6	–1.61E+6	1.96E+6	–1.46E+4	–4.38E+4	–9.49E+5	–3.53E+6	1.47E+4
	270°	(2B)	2442	–1.35E+6	–1.63E+6	1.90E+6	1.75E+4	–4.96E+4	–8.42E+5	–3.63E+6	1.91E+4
	0°	(2A)	1565	–1.11E+6	–1.50E+6	–2.26E+6	1.52E+5	–6.63E+5	2.23E+5	–5.66E+5	5.56E+4
5 (Ext. Wall)	90°	(2A)	2458	–1.51E+6	–1.17E+6	–1.79E+6	2.23E+5	5.14E+5	–9.39E+5	–2.04E+6	5.53E+5
	270°	(2B)	2155	–1.51E+6	–1.37E+6	1.94E+6	1.02E+4	–3.94E+4	–2.45E+5	–1.28E+6	–1.42E+4
	270°	(2B)	2460	–1.63E+6	–1.40E+6	–1.48E+6	1.20E+6	–2.47E+5	1.53E+6	–1.76E+5	5.11E+5
	0°	(2A)	1579	–2.55E+6	–1.63E+6	1.94E+6	–1.46E+4	4.67E+4	–2.27E+5	–1.23E+6	–3.65E+4
6 (Ext. Wall)	0°	(2A)	1597	–4.34E+6	–1.58E+6	1.89E+6	2.04E+4	6.39E+5	–1.06E+5	–4.97E+5	–3.25E+4
7 (Ext. Wall)	270°	(3)	2478	–8.33E+5	–6.58E+5	–1.33E+6	5.84E+3	–9.05E+4	8.76E+4	4.98E+5	–6.23E+3
	0°	(3)	1614	–1.36E+6	–3.04E+6	1.64E+6	1.46E+3	2.77E+4	1.47E+4	1.17E+5	–1.11E+4
	0°	(3)	1620	–1.10E+6	–3.13E+6	–1.27E+6	3.79E+4	2.77E+4	2.09E+4	1.72E+5	–2.31E+4
8 (Ext. Wall)	270°	(4)	2491	–7.09E+5	–5.65E+5	1.31E+6	–1.46E+3	–8.61E+4	4.63E+4	2.40E+5	2.22E+3
9 (Ext. Wall)	0°	(4)	1650	–1.09E+6	–2.08E+6	–1.55E+6	4.38E+3	2.77E+4	2.36E+4	1.73E+5	1.16E+4
10 (Ext. Wall)	0°	(4)	1673	–3.15E+6	–8.41E+5	1.17E+6	–5.84E+3	7.73E+4	–4.63E+4	–2.34E+5	–1.25E+4
11 (Ext. Wall)	180°	(4)	1998	–1.97E+5	–8.03E+4	–5.72E+5	1.46E+4	–1.17E+4	1.42E+4	–7.12E+3	–8.90E+3
12 (Ext. Wall)	0°	(4)	1705	–2.57E+6	–2.10E+5	–7.46E+5	4.38E+3	8.90E+4	2.62E+4	1.54E+5	–6.67E+3

Table 3H.3-10 Rebar and Concrete Stresses for Load Combination: 1.4D + 1.7L + 1.7H

				Maximum Stresses in the Reinforcing Steel (MPa)						Allowable Stress (MPa)	
				Inside Face		Outside Face			Max. Conc. Stress		
Section	Location	(Region)	El. #	Vertical[1] MPa	Horizontal[2] MPa	Vertical[1] MPa	Horizontal[2] MPa	Shear Ties MPa	MPa	Steel MPa	Conc. MPa
0 (Basemat)	90°	(BMT–1)	55	0.0	7.6	–5.5	2.1	77.9	–2.1	372	–23
	90°	(BMT–1)	146	17.9	28.3	–13.1	–19.3	4.1	–4.8	372	–23
	270°	(BMT–1)	126	–9.0	–28.3	–5.5	64.8	49.6	–6.2	372	–23
	0°	(BMT–1)	9	–1.4	–6.2	15.9	–7.6	56.5	–2.1	372	–23
	Middle	(BMT–2)	52	–2.8	–2.8	–8.3	–3.4	50.3	–2.1	372	–23
	Middle	(BMT–2)	60	26.2	19.3	0.7	9.7	152.4	–2.8	372	–23
	Middle	(BMT–2)	75	29.0	14.5	–20.0	–17.2	5.5	–4.1	372	–23
	Middle	(BMT–2)	129	24.8	15.9	–20.0	–17.2	4.8	–4.1	372	–23
	Middle	(BMT–2)	178	–2.1	–2.1	–8.3	–2.8	49.6	–2.1	372	–23
1 (Ext. Wall)	90°	(1A)	2108	–33.1	–9.7	102.0	0.7	85.5	–9.7	372	–23
	90°	(1A)	2109	–29.6	–9.0	95.8	2.8	88.9	–9.0	372	–23
	270°	(1B)	2405	–33.1	–15.9	161.3	5.5	149.6	–15.9	372	–23
	270°	(1B)	2406	–35.2	–17.2	166.2	9.0	149.6	–16.5	372	–23
	270°	(1B)	2408	–33.1	–15.2	161.3	7.6	151.0	–15.9	372	–23
	0°	(1A)	1508	–28.3	–10.3	119.3	4.1	89.6	–9.7	372	–23
	0°	(1B)	1512	–20.7	–12.4	75.8	6.9	90.3	–9.0	372	–23
	0°	(1B)	1513	–24.1	–11.7	90.3	12.4	102.0	–11.0	372	–23
2 (Ext. Wall)	270°	(2B)	2424	2.8	–4.1	31.0	137.9	88.3	–9.0	372	–23
3 (Ext. Wall)	90°	(2A)	2134	–0.7	–2.1	–4.1	–4.8	44.8	–2.1	372	–23
	270°	(2B)	2425	0.7	0.0	6.2	160.6	156.5	–10.3	372	–23
4 (Ext. Wall)	270°	(2B)	2441	181.3	44.1	–53.1	–18.6	12.4	–13.8	372	–23
	270°	(2B)	2442	106.9	33.1	–40.7	–13.1	15.9	–12.4	372	–23
	0°	(2A)	1565	13.8	2.1	–9.0	6.9	37.2	–2.8	372	–23
5 (Ext. Wall)	90°	(2A)	2458	63.4	104.1	–14.5	–6.2	44.1	–9.7	372	–23
	270°	(2B)	2155	7.6	–1.4	–15.9	–5.5	2.8	–2.8	372	–23
	270°	(2B)	2460	0.0	0.7	12.4	145.5	107.6	–9.7	372	–23
	0°	(2A)	1579	17.9	4.8	–15.9	–4.1	4.1	–3.4	372	–23
6 (Ext. Wall)	0°	(2A)	1597	0.0	1.4	3.4	2.1	31.7	–0.7	372	–23
7 (Ext. Wall)	270°	(3)	2478	–24.1	–6.2	47.6	3.4	0.0	–7.6	372	–23
	0°	(3)	1614	–16.5	–2.1	–17.2	–2.8	0.0	–2.1	372	–23
	0°	(3)	1620	–17.9	–1.4	–14.5	–0.7	0.0	–2.8	372	–23
8 (Ext. Wall)	270°	(4)	2491	–16.5	–3.4	24.8	1.4	0.0	–4.8	372	–23
9 (Ext. Wall)	0°	(4)	1650	–27.6	–4.1	–12.4	–2.1	0.0	–4.8	372	–23
10 (Ext. Wall)	0°	(4)	1673	108.9	66.9	15.9	24.8	0.0	–6.9	372	–23
11 (Ext. Wall)	180°	(4)	1998	0.0	0.7	2.8	6.9	0.0	–0.7	372	–23
12 (Ext. Wall)	0°	(4)	1705	7.6	13.1	83.4	46.9	0.0	–5.5	372	–23

Notes:
1. 0° – 180° for Section 0 (Basemat)
2. 90° – 270° for Section 0 (Basemat)

Key:
= Regional Maximum

Table 3H.3-11 Rebar and Concrete Stresses for Load Combination (D + L + H' ± E)$_{Max. Tension}$

Section	Location	(Region)	El. #	Maximum Stresses in the Reinforcing Steel (MPa) — Inside Face — Vertical[1] MPa	Inside Face — Horizontal[2] MPa	Outside Face — Vertical[1] MPa	Outside Face — Horizontal[2] MPa	Shear Ties MPa	Max. Conc. Stress MPa	Allowable Stress (MPa) — Steel MPa	Allowable Stress (MPa) — Conc. MPa
0 (Basemat)	90°	(BMT–1)	55	148.9	122.0	95.8	130.3	221.3	–4.1	372	–23
	90°	(BMT–1)	146	217.9	122.0	–6.2	–17.2	29.0	–9.0	372	–23
	270°	(BMT–1)	126	42.1	4.1	171.0	245.5	159.3	–10.3	372	–23
	0°	(BMT–1)	9	42.1	117.9	197.9	242.7	111.7	–5.5	372	–23
	Middle	(BMT–2)	52	–17.9	–14.5	178.6	184.1	56.5	–9.0	372	–23
	Middle	(BMT–2)	60	165.5	144.8	21.4	53.8	243.4	–7.6	372	–23
	Middle	(BMT–2)	75	217.2	219.9	–11.7	–20.7	41.4	–9.7	372	–23
	Middle	(BMT–2)	129	219.3	217.2	–11.7	–20.7	37.2	–9.7	372	–23
	Middle	(BMT–2)	178	–20.0	–15.9	173.1	196.5	62.7	–9.0	372	–23
1 (Ext. Wall)	90°	(1A)	2108	53.1	82.7	279.2	161.3	147.5	–11.0	372	–23
	90°	(1A)	2109	54.5	82.0	288.2	162.0	151.0	–11.0	372	–23
	270°	(1B)	2405	–9.7	37.9	271.0	159.3	177.2	–20.0	372	–23
	270°	(1B)	2406	–15.2	40.7	284.8	170.3	180.0	–21.4	372	–23
	270°	(1B)	2408	–11.7	40.7	284.8	173.7	179.3	–20.7	372	–23
	0°	(1A)	1508	70.3	122.0	251.0	204.1	144.8	–9.0	372	–23
	0°	(1B)	1512	5.5	46.9	162.0	175.1	92.4	–10.3	372	–23
	0°	(1B)	1513	14.5	71.7	148.9	104.8	152.4	–11.0	372	–23
2 (Ext. Wall)	270°	(2B)	2424	13.1	17.2	130.3	302.0	117.9	–13.8	372	–23
3 (Ext. Wall)	90°	(2A)	2134	123.4	123.4	149.6	130.3	176.5	–4.8	372	–23
	270°	(2B)	2425	24.8	17.2	71.7	340.6	202.7	–17.2	372	–23
4 (Ext. Wall)	270°	(2B)	2441	221.3	219.9	–31.0	11.0	21.4	–16.5	372	–23
	270°	(2B)	2442	230.3	202.7	–35.9	9.7	20.0	–17.9	372	–23
	0°	(2A)	1565	156.5	121.3	111.0	132.4	205.5	–4.8	372	–23
5 (Ext. Wall)	90°	(2A)	2458	180.6	275.8	–7.6	10.3	35.9	–15.9	372	–23
	270°	(2B)	2155	197.9	148.9	–1.4	20.7	6.2	–5.5	372	–23
	270°	(2B)	2460	45.5	47.6	33.1	219.3	271.7	–13.1	372	–23
	0°	(2A)	1579	226.1	244.8	22.1	111.0	5.5	–4.8	372	–23
6 (Ext. Wall)	0°	(2A)	1597	188.2	332.3	135.8	292.3	184.8	–4.1	372	–23
7 (Ext. Wall)	270°	(3)	2478	–0.7	35.2	194.4	166.2	0.0	–13.1	372	–23
	0°	(3)	1614	108.2	171.7	202.0	244.8	0.0	–6.2	372	–23
	0°	(3)	1620	72.4	97.2	214.4	208.9	0.0	–5.5	372	–23
8 (Ext. Wall)	270°	(4)	2491	56.5	72.4	238.6	184.1	0.0	–9.7	372	–23
9 (Ext. Wall)	0°	(4)	1650	84.1	133.8	264.1	241.3	0.0	–6.9	372	–23
10 (Ext. Wall)	0°	(4)	1673	232.4	276.5	114.5	267.5	0.0	–6.9	372	–23
11 (Ext. Wall)	180°	(4)	1998	46.2	46.2	69.6	85.5	0.0	–2.1	372	–23
12 (Ext. Wall)	0°	(4)	1705	64.8	189.6	174.4	312.3	0.0	–6.2	372	–23

Notes:
1. 0° – 180° for Section 0 (Basemat)
2. 90° – 270° for Section 0 (Basemat)

Key:
[shaded cell] = Regional Maximum

Table 3H.3-12 Rebar and Concrete Stresses for Load Combination (D + L + H' ± E)$_{\text{Max. Compression}}$

				Maximum Stresses in the Reinforcing Steel (MPa)						Allowable Stress (MPa)	
				Inside Face		Outside Face					
Section	Location	(Region)	El. #	Vertical[1] MPa	Horizontal[2] MPa	Vertical[1] MPa	Horizontal[2] MPa	Shear Ties MPa	Max. Conc. Stress MPa	Steel MPa	Conc. MPa
0 (Basemat)	90°	(BMT–1)	55	53.1	77.9	23.4	95.1	185.5	–4.1	372	–23
	90°	(BMT–1)	146	137.2	77.9	–21.4	–25.5	17.2	–9.7	372	–23
	270°	(BMT–1)	126	–6.9	–7.6	63.4	197.2	134.4	–10.3	372	–23
	0°	(BMT–1)	9	–6.9	–22.1	97.9	–4.1	95.8	–5.5	372	–23
	Middle	(BMT–2)	52	–25.5	–28.3	115.1	126.2	49.0	–9.0	372	–23
	Middle	(BMT–2)	60	114.5	48.3	–4.1	–2.8	168.2	–6.9	372	–23
	Middle	(BMT–2)	75	147.5	162.0	–24.1	–28.3	31.7	–9.7	372	–23
	Middle	(BMT–2)	129	152.4	168.9	–24.1	–29.0	34.5	–10.3	372	–23
	Middle	(BMT–2)	178	–26.9	–28.3	119.3	122.0	46.2	–9.0	372	–23
1 (Ext. Wall)	90°	(1A)	2108	–26.2	12.4	166.2	72.4	98.6	–12.4	372	–23
	90°	(1A)	2109	–21.4	15.2	155.8	73.1	101.4	–11.7	372	–23
	270°	(1B)	2405	–44.1	1.4	226.1	94.5	151.0	–21.4	372	–23
	270°	(1B)	2406	–44.8	–0.7	228.9	92.4	152.4	–42.1	372	–23
	270°	(1B)	2408	–44.8	–1.4	224.1	92.4	150.3	–21.4	372	–23
	0°	(1A)	1508	–16.5	–10.3	130.3	28.3	87.6	–9.7	372	–23
	0°	(1B)	1512	–26.9	–17.9	93.1	32.4	67.6	–11.0	372	–23
	0°	(1B)	1513	–35.9	–17.9	95.8	–4.1	100.7	–13.1	372	–23
2 (Ext. Wall)	270°	(2B)	2424	8.3	6.2	88.3	284.1	100.7	–14.5	372	–23
3 (Ext. Wall)	90°	(2A)	2134	45.5	64.1	62.7	67.6	137.2	–4.8	372	–23
	270°	(2B)	2425	15.9	9.0	57.9	312.3	186.2	–16.5	372	–23
4 (Ext. Wall)	270°	(2B)	2441	188.9	189.6	–46.2	2.1	21.4	–17.9	372	–23
	270°	(2B)	2442	183.4	168.9	–47.6	3.4	20.0	–17.2	372	–23
	0°	(2A)	1565	91.0	82.7	51.7	93.8	172.4	–4.8	372	–23
5 (Ext. Wall)	90°	(2A)	2458	157.9	240.6	–20.7	2.8	33.1	–16.5	372	–23
	270°	(2B)	2155	130.3	87.6	–24.8	–2.8	8.3	–7.6	372	–23
	270°	(2B)	2460	31.7	37.2	20.0	205.5	266.8	–13.8	372	–23
	0°	(2A)	1579	100.0	33.8	–29.0	–15.9	7.6	–7.6	372	–23
6 (Ext. Wall)	0°	(2A)	1597	31.0	–20.7	2.1	–24.1	97.2	–4.8	372	–23
7 (Ext. Wall)	270°	(3)	2478	–18.6	11.7	151.7	104.1	0.0	–13.8	372	–23
	0°	(3)	1614	–35.2	–13.8	–17.9	–2.1	0.0	–7.6	372	–23
	0°	(3)	1620	–39.3	–13.1	–15.9	–0.7	0.0	–6.9	372	–23
8 (Ext. Wall)	270°	(4)	2491	22.1	29.6	162.0	107.6	0.0	–9.0	372	–23
9 (Ext. Wall)	0°	(4)	1650	–25.5	7.6	39.3	55.2	0.0	–8.3	372	–23
10 (Ext. Wall)	0°	(4)	1673	64.8	–26.2	–8.3	–37.9	0.0	–8.3	372	–23
11 (Ext. Wall)	180°	(4)	1998	33.8	24.1	54.5	53.8	0.0	–2.1	372	–23
12 (Ext. Wall)	0°	(4)	1705	0.7	–26.9	80.0	–19.3	0.0	–6.2	372	–23

Notes:
1. 0° – 180° for Section 0 (Basemat)
2. 90° – 270° for Section 0 (Basemat)

Key:
[shaded] = Regional Maximum

Table 3H.3-13 Summary of Reinforced Steel

Concrete Structures: fc' = 4000 psi (25.58 MPa)				
Structure	**Thickness m**	**Main Reinfocement**	**Shear Ties**	**Governing Load Combination**
Basemat:				
Region BMT-1	2.5	#18 @ 22.86 cm E.W. – T&B	#8 @ 22.86 cm x 68.58 cm	D + L + H' + E
Region BMT-2	2.5	#18 @ 22.86 cm E.W. – T&B	#8 @ 22.86 cm x 68.58 cm	D + L + H' + E
Floor Slabs:				
• El. 21000 mm	0.3	#8 @ 30.48 cm E.W. – T&B	—	D + L + E
• El. 12300 mm	0.3	#8 @ 30.48 cm E.W. – T&B	—	D + L + E
• El. 4800 mm	0.3	#8 @ 30.48 cm E.W. – T&B	—	D + L + E
Roof Slab:	0.3	#8 @ 30.48 cm E.W. – T&B	—	1.4D + 1.7L
Exterior Wall:				
• Below Grade				
Region 1A	1.2	#18 @ 22.86 cm Vert. (E.F.) #18 @ 22.86 cm Horiz. (E.F.)	#8 @ 22.86 cm x 45.72 cm	D + L + H' + E
Region 1B	1.2	2 – #18 @ 22.86 cm Vert. (E.F.) #18 @ 22.86 cm Horiz. (E.F.)	#8 @ 22.86 cm x 45.72 cm	D + L + H' + E
Region 2A	1.2	#18 @ 22.86 cm Vert. (E.F.) #18 @ 45.72 cm Horiz. (E.F.)	#8 @ 45.72 cm x 45.72 cm	D + L + H' + E
Region 2B	1.2	2 – #18 @ 22.86 cm Vert. (E.F.) #18 @ 45.72 cm Horiz. (E.F.)	#8 @ 45.72 cm x 45.72 cm	D + L + H' + E
• Above Grade				
Region 3	0.6	#18 @ 22.86 cm Vert. (E.F.) #18 @ 45.72 cm Horiz. (E.F.)	—	D + L + H' + E
Region 4	0.6	#18 @ 45.72 cm Vert. (E.F.) #18 @ 45.72 cm Horiz. (E.F.)	—	D + L + H' + E

Table 3H.3-14 Summary of Reinforcing Steel Ratios

Concrete Structures: fc' = 4000 psi (27.58 MPa)						
			Reinforcement Ratio			
			Provided		Allowable	
Structure	**Thick-ness m**	**Main Reinforcement**	**Minimum**	**Maximum**	**Minimum**	**Maximum**
Basemat:						
Region BMT-1	2.5	#18 @22.86 cm E.W. - T&B	0.0049	0.0049	0.0033	0.0213
Region BMT-2	2.5	#18 @22.86 cm E.W. - T&B	0.0049	0.0049	0.0033	0.0213
Floor Slabs:						
• El. 21000 mm	0.3	#8 @ 30.48 cm E.W. - T&B	0.0063	0.0063	0.0033	0.0213
• El. 12300 mm	0.3	#8 @ 30.48 cm E.W. - T&B	0.0063	0.0063	0.0033	0.0213
• El. 4800 mm	0.3	#8 @ 30.48 cm E.W. - T&B	0.0063	0.0063	0.0033	0.0213
Roof Slab:	0.3	#8 @ 30.48 cm E.W. - T&B	0.0063	0.0063	0.0033	0.0213
Exterior Wall:						
• Below Grade						
Region 1A	1.2	#18 @ 22.86 cm Vert. (E.F.) #18 @ 22.86 cm Horiz. (E.F.)	0.0109	0.0109	0.0033	0.0213
Region 1B	1.2	2 - #18 @ 22.86 cm Vert. (E.F.) #18 @ 22.86 cm Horiz. (E.F.)	0.0116	0.0232	0.0033	0.0213
Region 2A	1.2	#18 @ 22.86 cm Vert. (E.F.) #18 @ 45.72 cm Horiz. (E.F.)	0.0054	0.0109	0.0033	0.0213
Region 2B	1.2	2 - #18 @ 22.86 cm Vert. (E.F.) #18 @ 45.72 cm Horiz. (E.F.)	0.0058	0.0232	0.0033	0.0213
• Above Grade						
Region 3	0.6	#18 @ 22.86 cm Vert. (E.F.) #18 @ 45.72 cm Horiz. (E.F.)	0.0121	0.0242	0.0033	0.0213
For members with compression reinforcement, the portion of Pb equalized by compression reinforcement need not be reduced by the 0.75 factor.						

Notes:

1. Minimum reinforcement ratio is based on $200/f_y$ per ACI 359 Code, Section 10.5.
2. Maximum reinforcement ratio is based on 0.75 P_b ACI 359 Code, Section 10.3.3.
   P_b = Reinforcement ratio producing balanced strain condition.

Table 3H.3-14 Summary of Reinforcing Steel Ratios

Concrete Structures: fc' = 4000 psi (27.58 MPa)						
			Reinforcement Ratio			
			Provided		Allowable	
Structure	Thick-ness m	Main Reinforcement	Minimum	Maximum	Minimum	Maximum
Region 4	0.6	#18 @ 45.72 cm Vert. (E.F.) #18 @ 45.72 cm Horiz. (E.F.)	0.0121	0.0121	0.0033	0.0213
For members with compression reinforcement, the portion of Pb equalized by compression reinforcement need not be reduced by the 0.75 factor.						

Notes:

1. Minimum reinforcement ratio is based on $200/f_y$ per ACI 359 Code, Section 10.5.
2. Maximum reinforcement ratio is based on 0.75 P_b ACI 359 Code, Section 10.3.3.
   P_b = Reinforcement ratio producing balanced strain condition.

Table 3H.3-15 Summary of Structural Steel Safety Margins

ASTM A572 Gr. 50 (Fy = 50 ksi (345.19 MPa)			
Structure	**Size**	**Safety Margin = Capacity/Demand**	**Governing Load Combination**
Roof Steel Beams:			
	W27 x 84	3.7	D + W_t
	W27 x 114	1.03	D + W_t
	W27 x 146	1.1 to 1.6	D + W_t
	W33 x 201	1.5	D + W_t
Floor Steel Beams:			
El. 21000 mm	W27 x 84	1.4	D + L + E
	W27 x 146	1.06 to 1.31	D + L + E
	W27 x 194	1.01 to 1.2	D + L + E
	W36 x 359	1.01	D + L + E
El. 12300 mm	W27 x 84	1.4	D + L + E
	W27 x 146	1.06 to 1.31	D + L + E
	W27 x 194	1.01 to 1.2	D + L + E
	W36 x 359	1.01	D + L + E
El. 4800 mm	W27 x 84	1.4	D + L + E
	W27 x 146	1.06 to 1.31	D + L + E
	W27 x 194	1.01 to 1.2	D + L + E
	W36 x 359	1.01	D + L + E
Columns:			
Above Grade:	W14 x 398	1.51	D + L + H’ + E
Below Grade:	W14 x 730	1.49	D + L + H’ + E

Table 3H.3-16 Maximum Soil Bearing Pressures

Load Case Name	Load Combination	Max. Soil Bearing Pressure kPa
Service		
90°–270°	D + L + H	73.75
0°–180°	D + L + H	73.75
Wind		
90°–270°	D + L + H + W_X	107.78
0°–180°	D + L + H + W_Y	123.07
Tornado (Wind Only)		
90°–270°	D + L + H + W_{WX}	116.11
0°–180°	D + L + H + W_{WY}	132.49
Tornado (Wind + 1/2PD)		
90°–270°	D + L + H + W_{WX} + 1/2 W_{wp}	109.34
0°–180°	D + L + H + W_{WY} + 1/2 W_{wp}	127.98
Tornado (Press. Drop)		
90°–270°	D + L + H + W_{WP}	92.57
0°–180°	D + L + H + W_{WP}	107.87
Seismic		
90°–270°; E_Z (down)	D + L + H’ + E_X + E_Z	237.71
0°–180°; E_Z (down)	D + L + H’ + E_y + E_Z	404.43

Note: For extreme wind loads, the design basis tornado whose winds and missile loads bound the design basis hurricane are as specified in 3H.1.4.3.1.

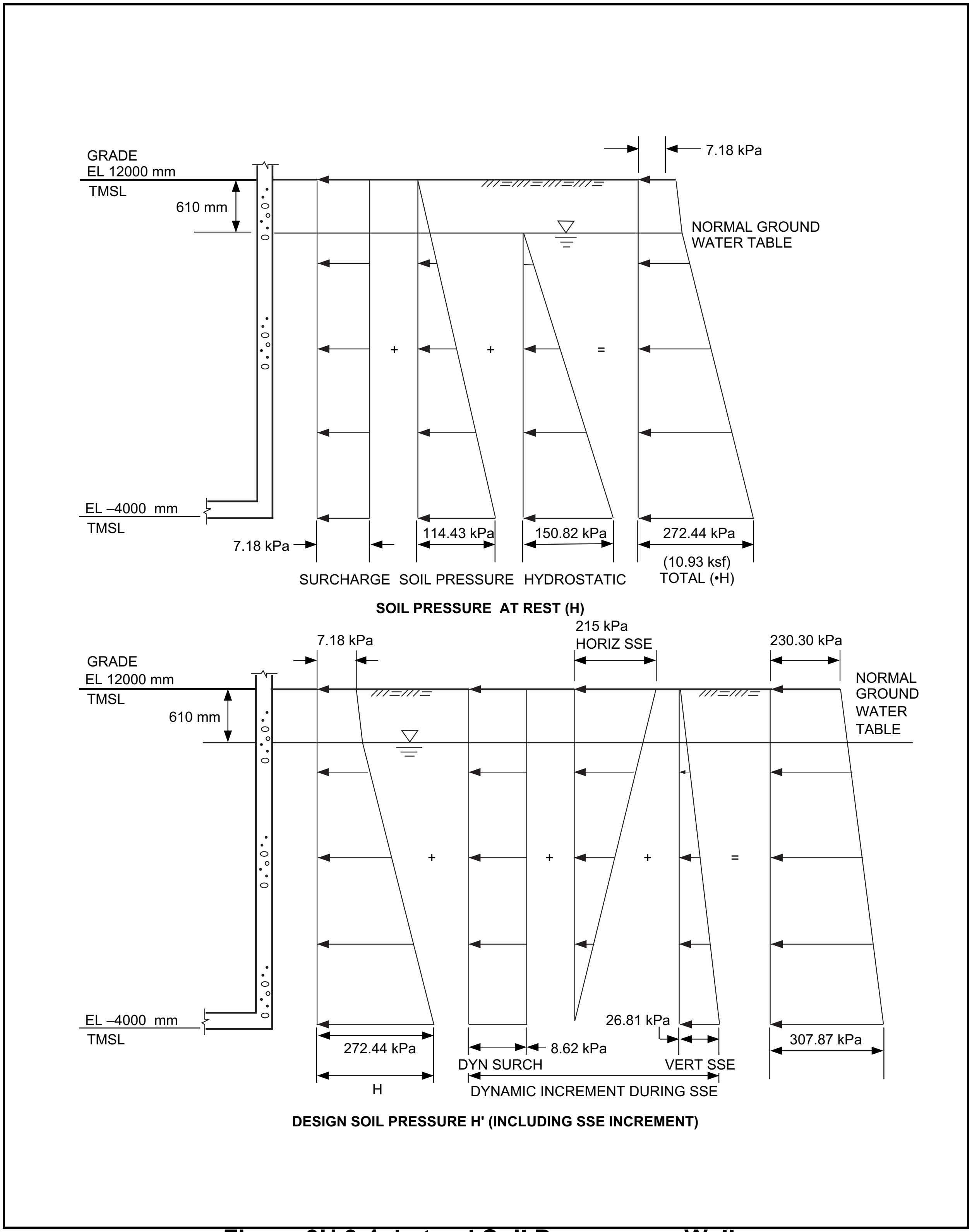

Figure 3H.3-1 Lateral Soil Pressure on Walls

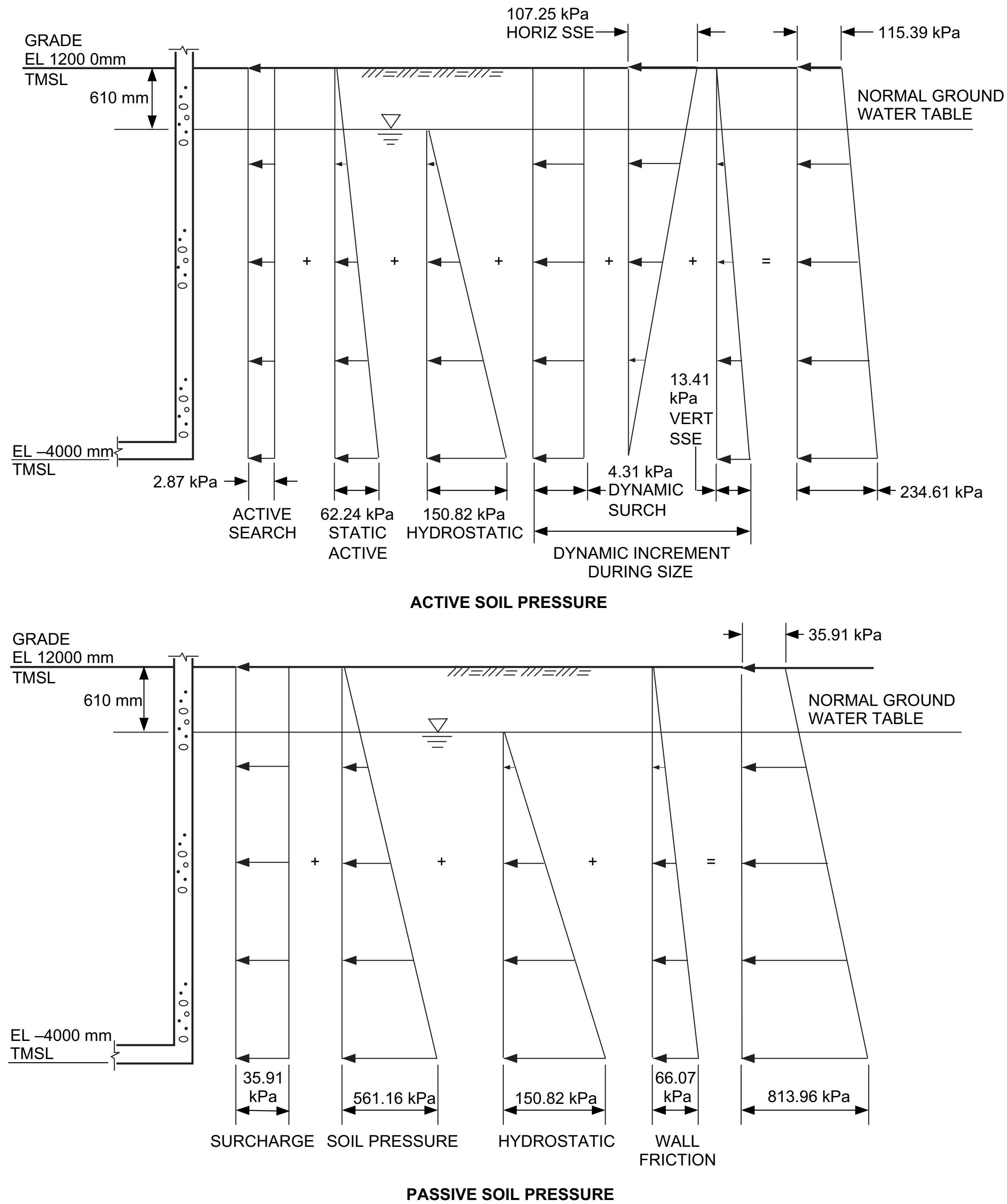

Figure 3H.3-2 Active and Passive Lateral Soil Pressures on Walls

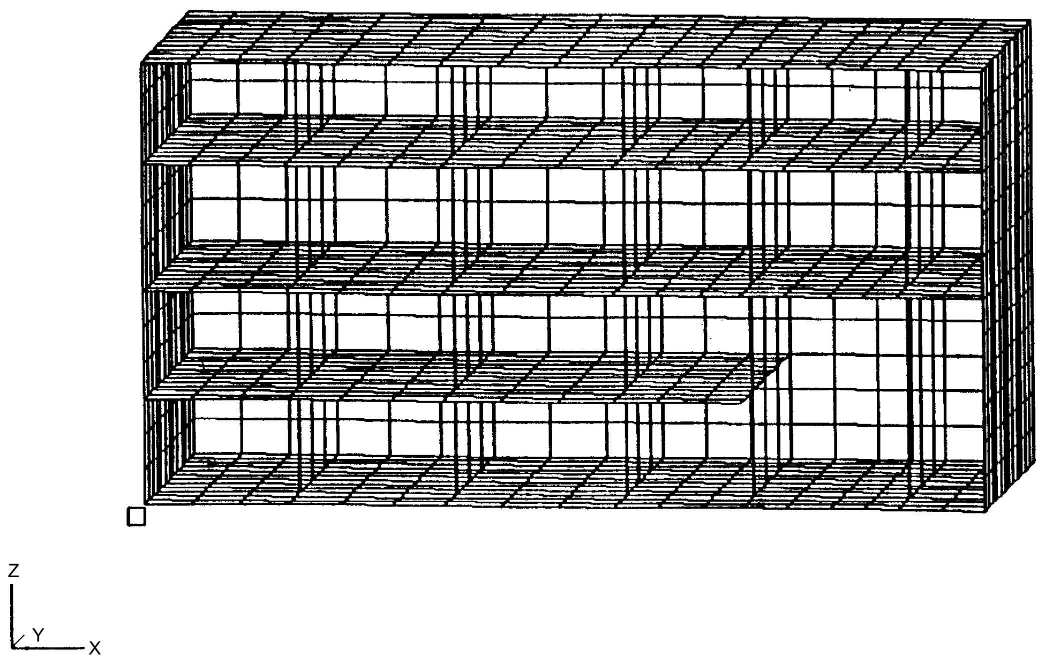

Figure 3H.3-3 Stardyne Modal of Radwaste Building (Front Wall Removed for Clarity)

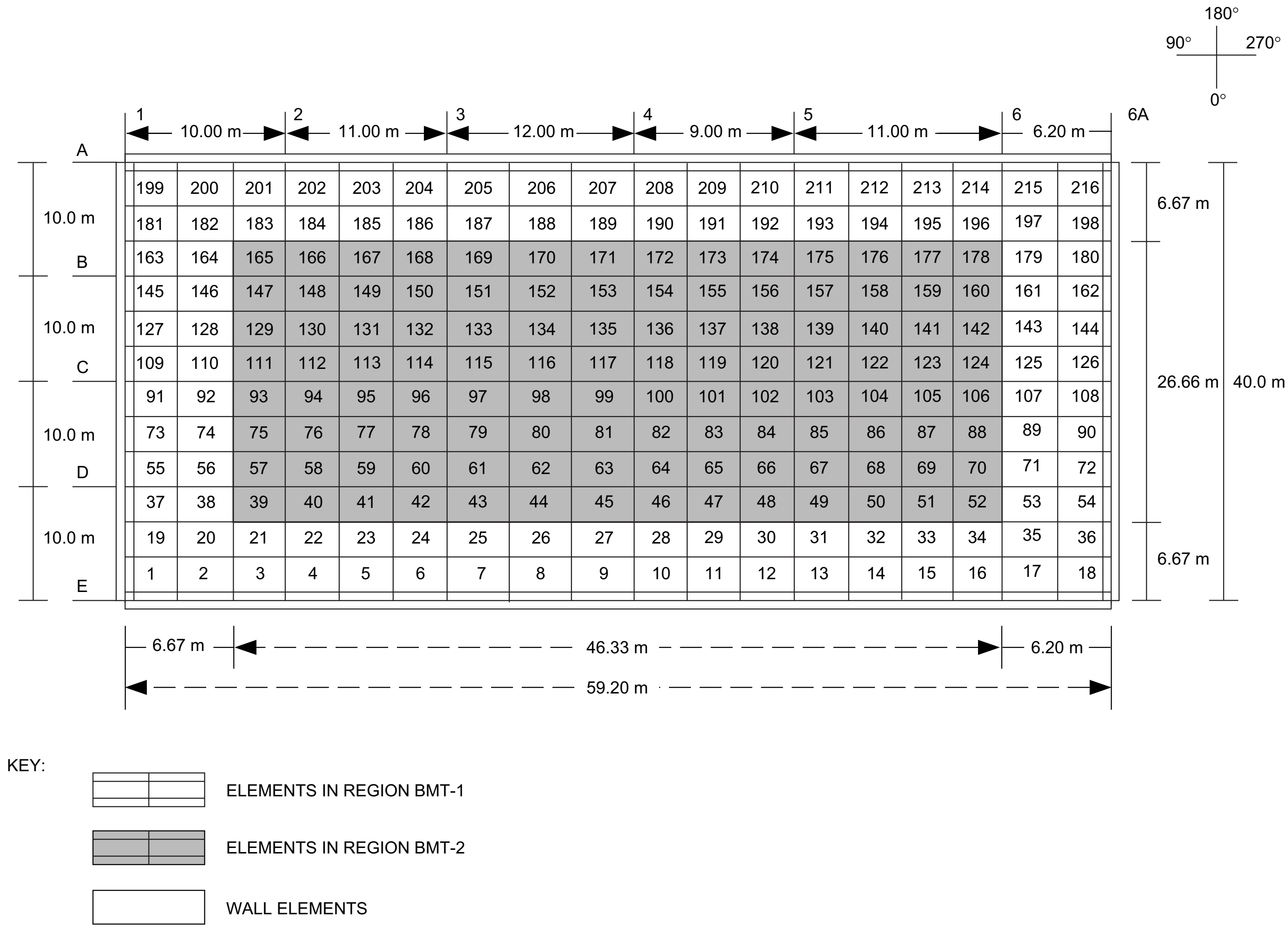

Figure 3H.3-4 Basemat Element Reinforcing Regions

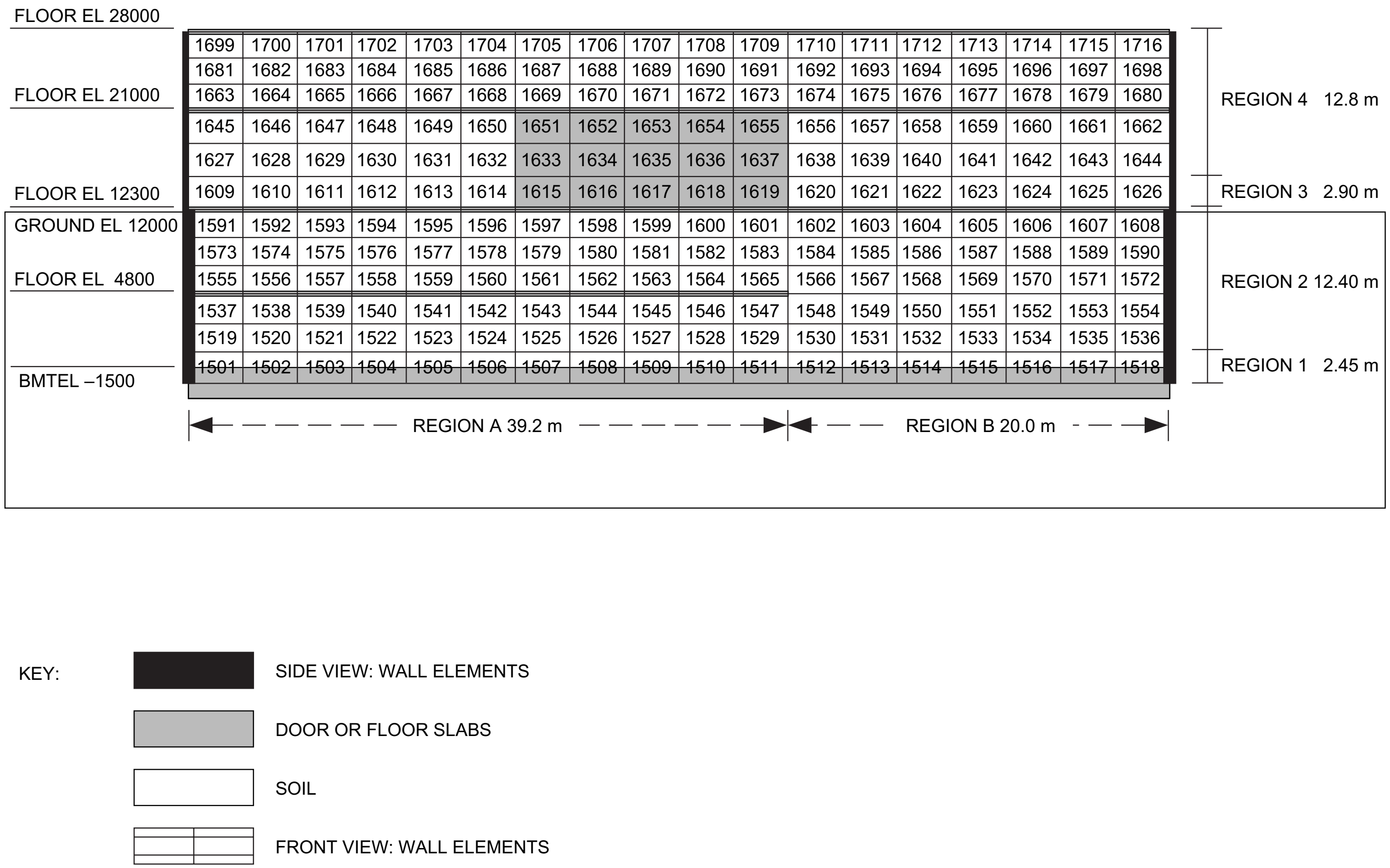

Figure 3H.3-5 Wall Elements — 0° Wall

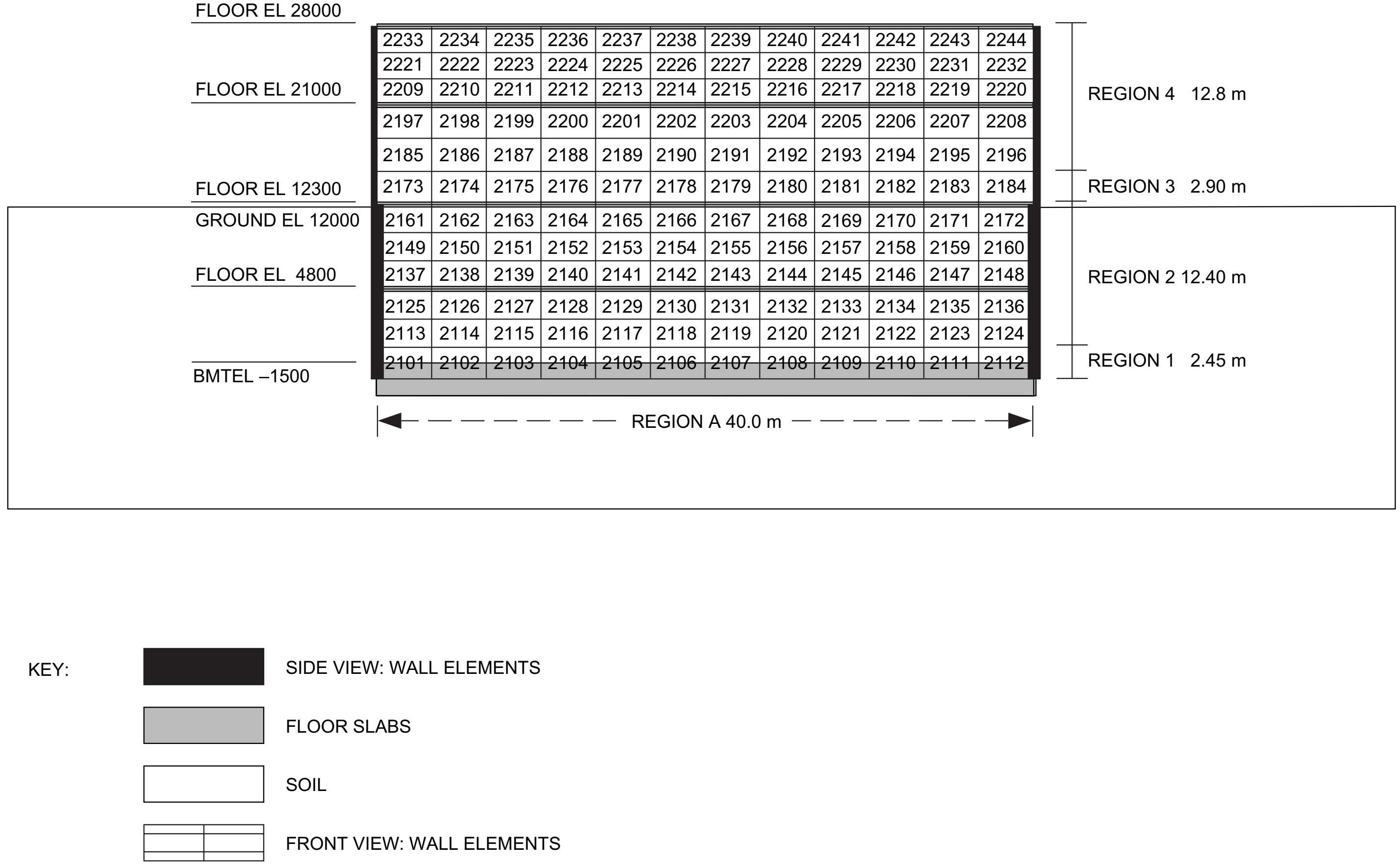

Figure 3H.3-6 Wall Elements — 90° Wall

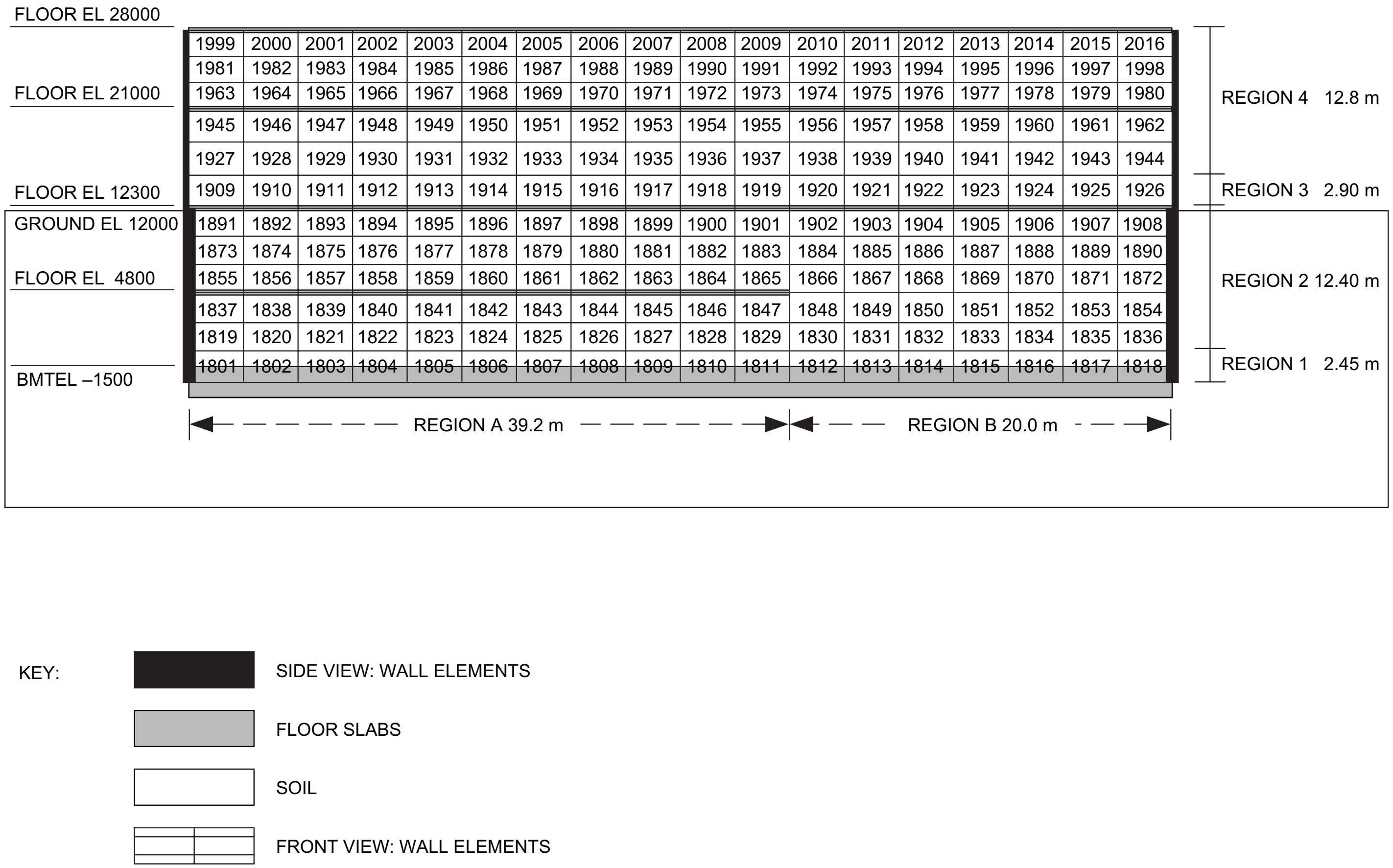

Figure 3H.3-7 Wall Elements — 180° Wall

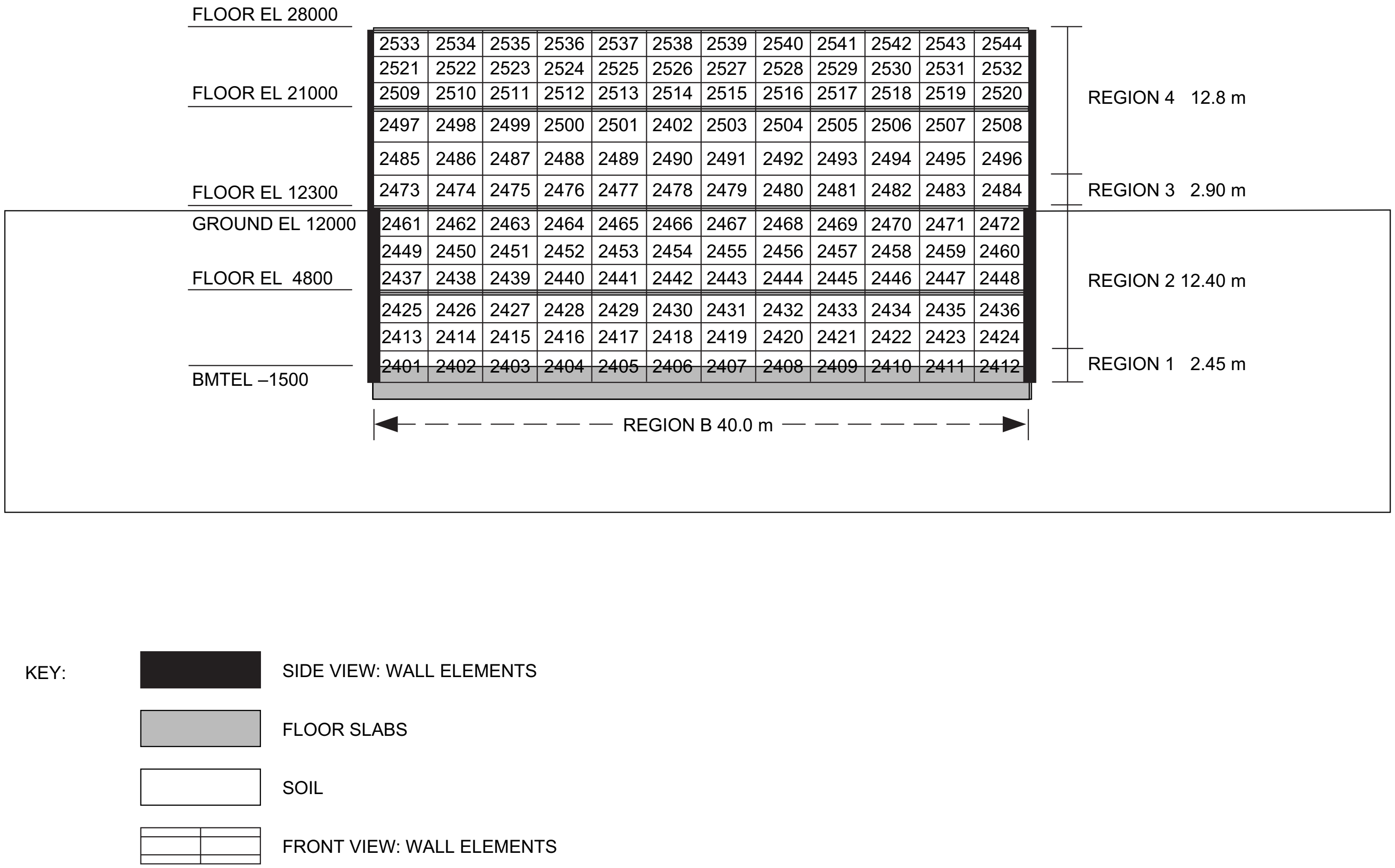

Figure 3H.3-8 Wall Elements — 270° Wall

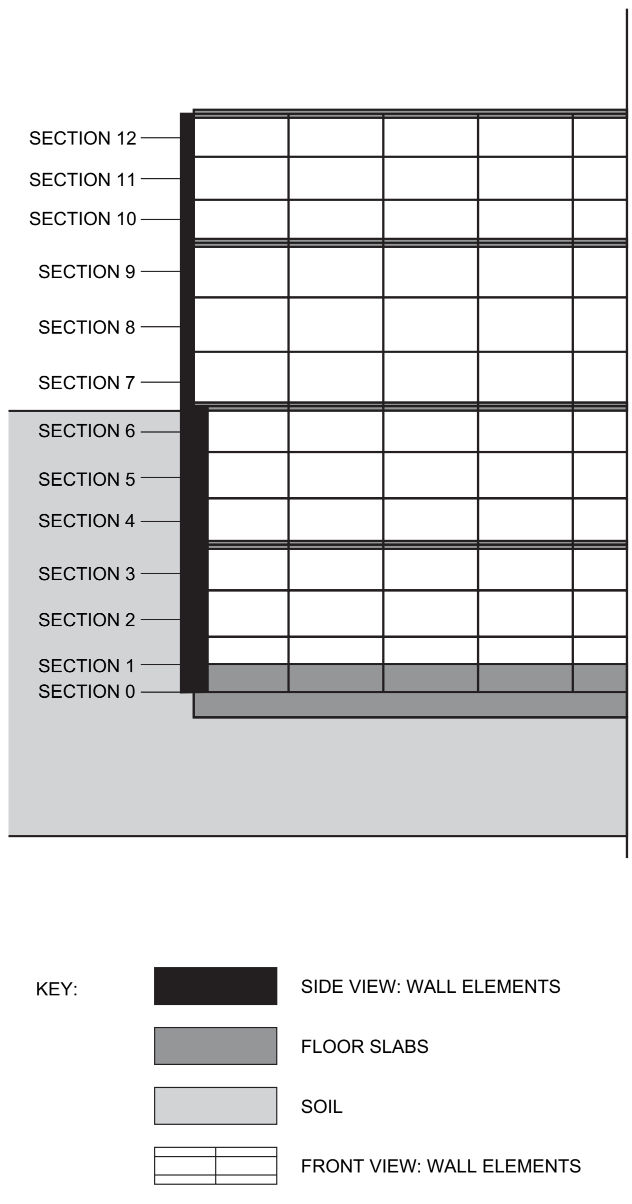

Figure 3H.3-9 Section Locations

LOCAL TO GLOBAL AXIS CONVERSION				
LOCAL AXIS	BASEMAT	WALLS (BY DIRECTION) 90° – 270°	WALLS (BY DIRECTION) 0° – 180°	COLUMNS
X	90° – 270°	90° – 270°	0° – 180°	90° – 270°
Y	0° – 180°	VERTICAL	VERTICAL	0° – 180°
Z	VERTICAL	0° – 180°	90° – 270°	VERTICAL

FORCE AND MOMENT DESCRIPTION	
STARDYNE LABEL	EXPLANATION OF STARDYNE LABEL
Fx	AXIAL FORCE, LOCAL X DIRECTION
Fy	AXIAL FORCE, LOCAL Y DIRECTION
Fxy	SHEAR, LOCAL XY PHASE
Fxz	SHEAR, LOCAL XZ PHASE
Mx	MOMENT, ABOUT LOCAL Y AXIS
My	MOMENT, ABOUT LOCAL X AXIS
Mxy	WARPING MOMENT
Fyz	SHEAR, LOCAL YZ PLANE

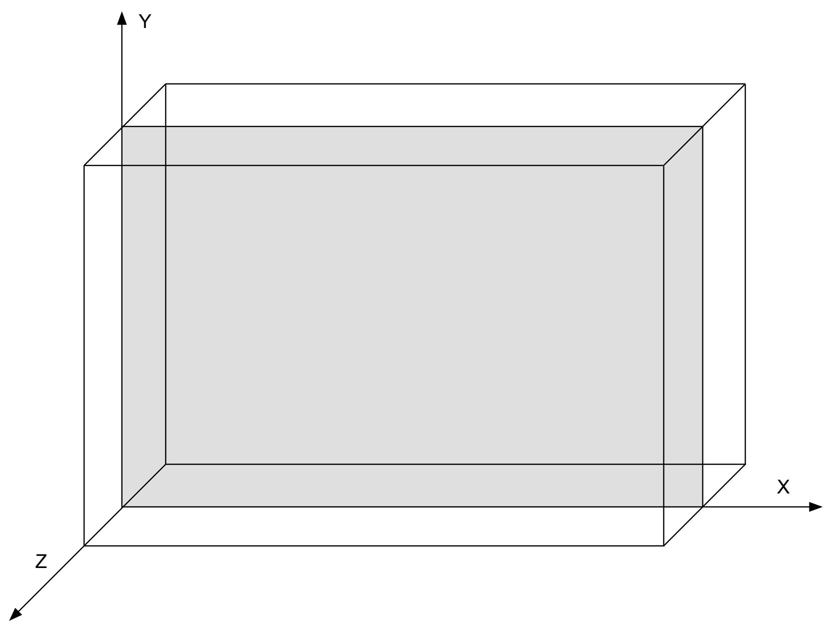

Figure 3H.3-10 Element Coordinate System

The following figures are located in Chapter 21:

3H.4 Structural Evaluation of R/B Compartment Walls Due to HELB

3H.4.1 Objective

The objective of this evaluation is to assess the structural adequacy of the Reactor Building compartment walls which may be subjected to pressure loads due to a high energy line break (HELB) of the reactor water cleanup lines (CUW) or the reactor core isolation lines (RCIC).

3H.4.2 Evaluation Approach

The Reactor Building subcompartment walls (shown in Figures 3H.4-1 through 3H.4-7), which may be subjected to HELB pressurization loads, were divided into two types.

Type 1 Walls
These are the walls which also act as shear walls in resisting seismic loads. These walls (S1 through S5) were modeled in the STARDYNE finite element analysis model and their stiffness were taken into consideration in the SASSI soil structure interaction analysis. These walls are rigidly connected with the floor slabs and are continuous in a vertical plane from the basemat upwards.

Type 2 Walls
These are the walls which are not shear walls. They are connected to the floor slabs with hinged connections.

For structural evaluation, the critical combination considered was:

$U = D + L + P_a + SSE$.

P_a, the compartment pressure load due to high energy line break (HELB) was defined to be 0.104 MPaD.

SSE loads, for Type 1 walls were obtained from the STARDYNE analysis of the Reactor Building. For the Type 2 walls, SSE loads were determined based on floor accelerations from the SASSI analysis.

3H.4.3 Analytical Results

The results are presented in Tables 3H.4-1 and 3H.4-2. Table 3H.4-1 shows reinforcing steel requirements for the Type 1 (shear) walls. Table 3H.4-2 shows reinforcing steel requirements for the Type 2 (non-shear) walls.

3H.4.4 Removable Walls

For areas where removal of a portion of the wall becomes necessary to provide access for equipment servicing/replacement such as for the CUW compartment at El. –1700, conceptual design has been developed using removable precast concrete blocks which are held together and anchored to the non-removable portion of the wall by bolts. This is shown in Figure 3H.4-8. Heaviest block weighs approximately 2.5 metric tons. The bolts are sized and precast blocks are reinforced to 0.104 MPaD subcompartment pressure. Twenty five-millimeter diameter A-36 bolts are found to be adequate. The required reinforcing in the concrete blocks is shown in Table 3H.4-3.

3H.4.5 Conclusions

Based on the results of this evaluation, it can be concluded that the wall thickness, as provided, are adequate to resist the compartment pressure loads due to a high energy line break. The associated rebar requirements are summarized in Table 3H.4-1 and 3H.4-2.

It is also concluded that, to provide access for equipment removal such as for the CUW compartment at El. –1700 (see Figure 3H.4-3), a removable concrete block concept as shown in Figure 3H.4-8 is feasible.

Table 3H.4-1 Design of Type 1 (Shear) Walls Exposed to HELB Loadings

Wall #	Thickness (m)	Height (m)	Length (m)	Main Reinforcing (E.W. & E.F.)			Shear Reinforcing		
				As_{REQ} (cm^2/m)	As_{PROV} (cm^2/m)		Av_{REQ} (cm^2/cm^2)	As_{PROV} (cm^2/cm^2)	
S1a	0.90	3.00	5.60	83.6	84.7	#18 @ 0.305 m	0.0008	0.0031	#6 @0.305 m x 0.305 m
S1b	0.90	1.90	5.60	81.9	84.7	#18 @ 0.305 m	0.0008	0.0031	#6 @ 0.305 m x 0.305 m
S2	0.60	5.70	8.50	70.1	72.6	#18 @ 0.356 m	None	None	None
S3	0.80	5.70	5.90	55.0	56.5	#18 @ 0.457 m	None	None	None
S4	0.60	5.70	8.50	76.2	84.7	#18 @ 0.305 m	None	None	None
S5	0.80	5.70	6.70	72.4	72.6	#18 @ 0.357 m	0.0008	0.0022	#6 @ 0.357 m x 0.357 m

Table 3H.4-2 Design of Type 2 (Non-Shear) Walls Exposed to HELB Loadings

Thickness (m)	Max. Height (m)	Area of Main Reinforcing Steel (E.W. & E.F.)				Shear Reinforcing	
		Calculated (cm^2/m)	Code Min (cm^2/m)	Required (cm^2/m)	Provided (cm^2/m)	Av_{REQ} (cm^2/cm^2)	As_{PROV}
0.25	3.00	20.1	7.0	20.1	25.2 #8 @ 0.203 m	None	None
0.40	3.20	12.7	11.4	12.7	16.7 #8 @ 0.305 m	None	None
0.50	5.80	33.4	14.8	33.4	49.5 #11 @ 0.203 m	None	None
0.55	6.50	38.5	16.5	38.5	49.5 #11 @ 0.203 m	0.000	None
0.60	7.50	47.2	18.2	47.2	49.5 #11 @ 0.203 m	0.000	None
0.70	6.50	31.1	21.4	31.1	33.0 #11 @ 0.305 m	None	None
0.80	5.80	22.9	24.8	24.8	25.2 #8 @ 0.203 m	None	None
0.90	5.80	21.4	27.9	27.9	33.0 #11 @ 0.305 m	None	None
1.00	6.50	24.8	31.3	31.3	33.0 #11 @ 0.305 m	None	None
1.10	3.30	7.8	34.7	34.7	49.5 #11 @ 0.203 m	None	None
1.20	5.80	19.0	37.9	37.9	49.5 #11 @ 0.203 m	None	None
1.30	6.50	22.4	41.3	41.3	49.5 #11 @ 0.203 m	None	None

Notes:

1. $As_{CODE\ MIN}$ is based on ρ_{MIN} = 200/fy from ACI 349-90 Section 10.5.1.

Table 3H.4-3 Design of Concrete Blocks for Removable Wall Exposed to HELB Loadings

Thickness (m)	Unsup. Len. (m)	Height (m)	Area of Main Reinforcing Steel (E.F.)			Shear Reinforcing	
			Calculated (cm^2/m)	Code Min (cm^2/m)	Provided (cm^2/m)	Av_{REQ} (cm^2/cm^2)	As_{PROV}
0.43	3.00	0.60	20.3	24.6	66.9 4 #8’s	None	#3 Ties @ 0.305
0.43	3.00	0.40	13.5	16.5	33.4 2 #8’s	None	#3 Ties @ 0.305

Notes:

1. $As_{CODE\ MIN}$ is based on ρ_{MIN} = 200/fy from ACI 349-90 Section 10.5.1.
2. Reinforcing is being calculated for the outer blocks; however, this reinforcing will conservatively be used for all blocks. This will provide reinforcing that is more than sufficient to resist all handling loads that the blocks will be exposed to.

Figure 3H.4-1 Location of Walls Exposed to HELB, El. –8200 mm

Figure 3H.4-2 Location of Walls Exposed to HELB, El. –5100 mm

Figure 3H.4-3 Location of Walls Exposed to HELB, El. –1700 mm

Figure 3H.4-4 Location of Walls Exposed to HELB, El. 1500 mm

Figure 3H.4-5 Location of Walls Exposed to HELB, El. 4800 mm

Figure 3H.4-6 Location of Walls Exposed to HELB, El. 8500 mm

Figure 3H.4-7 Location of Walls Exposed to HELB, El. 12300 mm

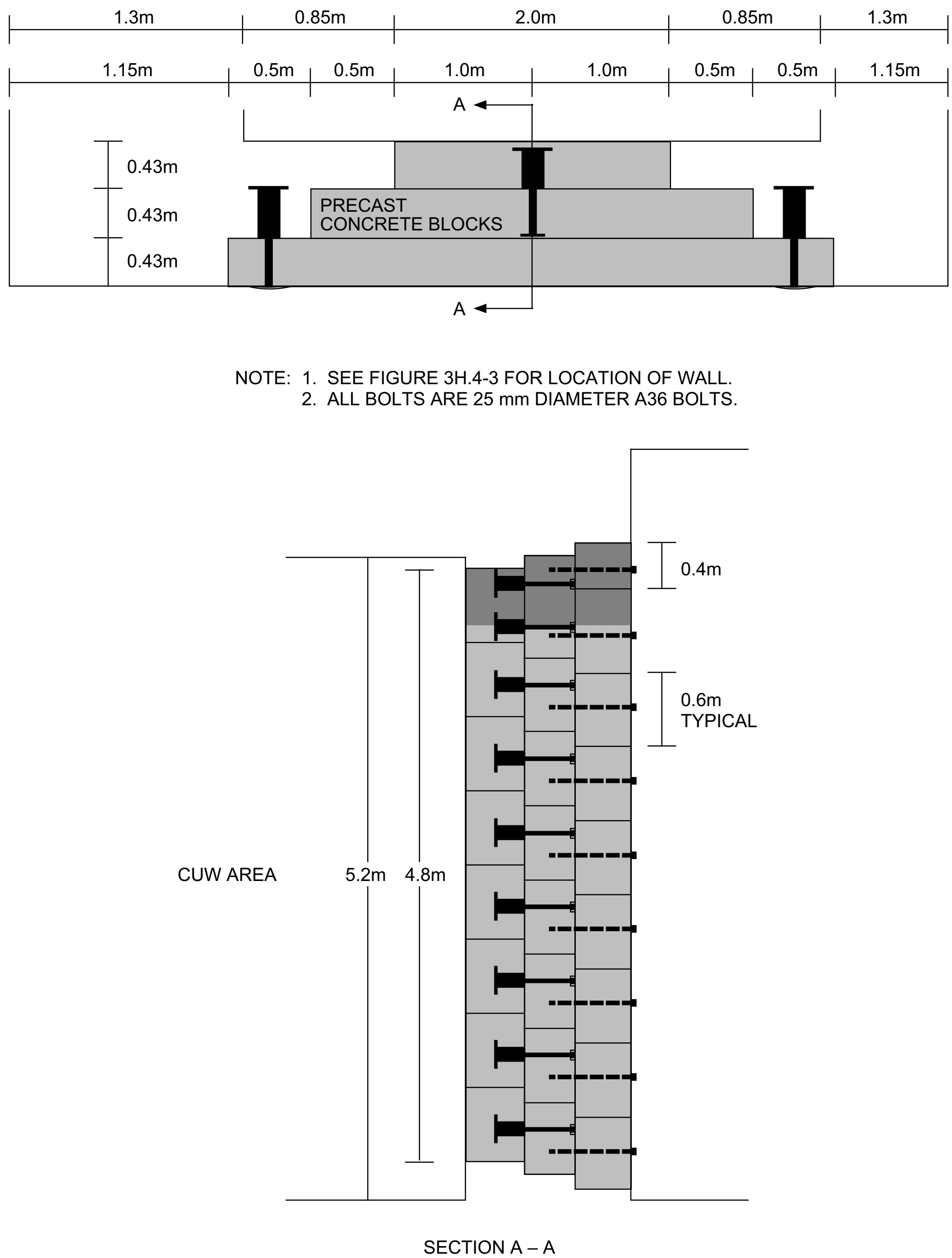

Figure 3H.4-8 Removable Precast Concrete Blocks

3H.5 Structural Analysis Reports

3H.5.1 Structural Analysis Report For The Reinforced Concrete Containment and the Containment Internal Structures

A structural analysis report will be prepared. It will document the following activities associated to the construction materials and as-built dimensions of the building:

(1) Review of construction records for material properties used in construction (i.e., in-process testing of concrete properties and procurement specifications for structural steel and reinforcing bars).

(2) Inspection of as-built building dimensions.

For material properties and dimensions, assess compliance of the as-built structure with design requirements in Subsections 3.8.1 and 3.8.3 and in the Detail design documents.

Construction deviations and design changes will be assessed to determine appropriate disposition.

This disposition will be accepted "as-is," provided the following acceptance criteria are met:

- The structural design meets the acceptance criteria and load combinations of Subsections 3.8.1 and 3.8.3.
- The dynamic responses (i.e., spectra, shear forces, axial forces and moments) of the as-built building are bounded by the spectra in Appendices 3A and 3G.

Depending upon the extent of the deviation or design changes, compliance with the acceptance criteria can be determined by either:

(a) Analyses or evaluations of construction deviations and design changes, or

(b) The design basis analyses will be repeated using the as-built condition.

3H.5.2 Structural Analysis Report For The Steel Containment

A structural analysis report will be prepared. It will document the following activities associated to the construction materials and as-built dimensions of the building:

(1) Review of construction records for material properties used in construction (i.e., procurement specifications for structural steel).

(2) Inspection of as-built building dimensions.

For material properties and dimensions, assess compliance of the as-built structure with design requirements in the Subsection 3.8.2 and in the detail design documents.

Construction deviations and design changes will be assessed to determine appropriate disposition.

This disposition will be accepted "as-is," provided the following acceptance criteria are met:

- The structural design meets the acceptance criteria and load combinations of Subsection 3.8.2.
- The dynamic responses (i.e., spectra, shear forces, axial forces and moments) of the as-built structure are bounded by the spectra in Appendices 3A and 3G.

Depending upon the extent of the deviation or design changes, compliance with the acceptance criteria can be determined by either:

(a) Analyses or evaluations of construction deviations and design changes, or

(b) The design basis analyses will be repeated using the as-built condition.

3H.5.3 Structural Analysis Report For The Reactor Building, Control Building and Radwaste Building Substructure (Including Seismic Category I Tunnels)

A structural analysis report will be prepared. It will document the following activities associated to the construction materials and as-built dimensions of the building:

(1) Review of construction records for material properties used in construction (i.e., in-process testing of concrete properties and procurement specifications for structural steel and reinforcing bars).

(2) Inspection of as-built building dimensions.

For material properties and dimensions, assess compliance of the as-built structure with design requirements in the Subsection 3.8.4 and in the detail design documents.

Construction deviations and design changes will be assessed to determine appropriate disposition.

This disposition will be accepted "as-is," provided the following acceptance criteria are met:

- The structural design meets the acceptance criteria and load combinations of Subsection 3.8.4.
- The dynamic responses (i.e., spectra, shear forces, axial forces and moments) of the as-built structure are bounded by the spectra in Appendices 3A and 3G.
- The as-built piping configuration as it relates to HELB design pressures has been accounted for in accepting the as-built building deviations.

Depending upon the extent of the deviation or design changes, compliance with the acceptance criteria can be determined by either:

(a) Analyses or evaluations of construction deviations and design changes, or

(b) The design basis analyses will be repeated using the as-built condition.

3H.5.4 Structural Analysis Report For The Reactor Building, Control Building and Radwaste Building Foundations

A structural analysis report will be prepared. It will document the following activities associated to the construction materials and as-built dimensions of the building:

(1) Review of construction records for material properties used in construction (i.e., in-process testing of concrete properties and procurement specifications for structural steel and reinforcing bars).

(2) Inspection of as-built building dimensions.

For material properties and dimensions, assess compliance of the as-built structure with design requirements in the Subsection 3.8.5 and in the detail design documents.

Construction deviations and design changes will be assessed to determine appropriate disposition.

This disposition will be accepted "as-is," provided the following acceptance criteria are met:

- The structural design meets the acceptance criteria and load combinations of Subsection 3.8.5.
- The dynamic responses (i.e., spectra, shear forces, axial forces and moments) of the as-built structure are bounded by the spectra in Appendices 3A and 3G.

Depending upon the extent of the deviation or design changes, compliance with the acceptance criteria can be determined by either:

(a) Analyses or evaluations of construction deviations and design changes, or

(b) The design basis analyses will be repeated using the as-built condition.

3H.5.5 Structural Analysis Report For The Turbine Building

The T/B is not classified as a Seismic Category I structure. However, the building is designed such that damage to safety-related functions does not occur under seismic loads corresponding to the safe shutdown earthquake (SSE) ground acceleration.

A structural analysis report will be prepared. It will document the following activities associated to the construction materials and as-built dimensions of the building:

(1) Review of construction records for material properties used in construction (i.e., in-process testing of concrete properties and procurement specifications for structural steel and reinforcing bars).

(2) Inspection of as-built building dimensions.

For material properties and dimensions, assess compliance of the as-built structure with design requirements in the Uniform Building Code (UBC) and in Table 3.2-1 and paragraph 3.7.3.16.

Construction deviations and design changes will be assessed to determine appropriate disposition.

This disposition will be accepted "as-is," provided the following acceptance criteria are met:

- The structural design meets the acceptance criteria and load combinations of the UBC code.

Depending upon the extent of the deviation or design changes, compliance with the acceptance criteria can be determined by either:

(a) Analyses or evaluations of construction deviations and design changes, or

(b) The design basis analysis will be repeated using the as-built condition.

3H.6 Summary of Key Structural Design Features

An assessment of the effects on the ABWR for the beyond design basis impact of a large, commercial aircraft has been performed in accordance with 10 CFR 50.150(a). A summary of the assessment can be found in Appendix 19G. Information that supports detailed design used in the AIA assessment is provided in NEDE-33875P, "Aircraft Impact Assessment, Licensing Basis Information and Design Details for Key Design Features" (Reference 19G-3). NEDE-33875P captures the strengthening measures configured as part of the design enhancements for Aircraft Impact Assessment.

This appendix describes the key structural design features of the ABWR that were identified in that assessment.

(1) Structural configuration of Spent Fuel Pool (SFP) within Reactor Building precludes direct strike on SFP. The spent-fuel pool is a reinforced concrete structure with a 6.4mm (minimum) thick ASTM A-240 Type 304L stainless steel liner (see DCD Section 9.1.2.1.3). The SFP walls are strengthened as described in NEDE-33875P (Reference 19G-3) to ensure the integrity of the SFP is maintained.

(2) Structural configuration of primary containment (RCCV) within Reactor Building precludes direct strike on containment, and structural design of RCCV ensures that RCCV is not perforated.

(3) Shield blocks over drywell head are configured to fully resist secondary impacts from concrete debris, aircraft wreckage, and falling crane components to protect integrity of drywell head. The reactor cavity shield blocks are shown in Figure 3H.1-23.

(4) Interior partition walls are thickened and strengthened as shown in NEDE-33875P (Reference 19G-3) to limit physical damage to interior partition walls.

(5) Reinforced Concrete Sliding Barriers with structural capacity equivalent to the surrounding wall are provided for the 6 large openings on 1F (Figure 1.2-8) to limit physical damage to exterior wall.

(6) Protective awnings for the HVAC exhaust openings on 2F (Figure 1.2-9) are sized to provide structural capacity equivalent to the corresponding exterior wall to prevent unabated wreckage through these openings.

(7) Protective awnings for the HVAC intake openings on 3F (Figure 1.2-10) are sized to provide structural capacity equivalent to that provided in Table 3-2 of NEI 07-13 for exterior walls (Reference 19G-1).

(8) Deleted.

(9) Control Building Annex exterior walls are made of reinforced concrete and are at least 450mm thick.

(10) The Service Building exterior wall running in the North-South direction immediately adjacent to the Control Building is a reinforced concrete wall of 900mm minimum thickness.

(11) Turbine Building reinforced concrete exterior wall adjacent to the Control Building (south wall) from column line T6 to T9 up to elevation 22750mm is at least 900mm thick.

(12) R/B exterior walls on the East, West, and South sides are strengthened with enhanced reinforcement as described in NEDE-33875P (Reference 19G-3).

3I Equipment Qualification Environmental Design Criteria

3I.1 Introduction

This appendix specifies the minimum set of plant environmental conditions, which envelope the actual environments expected over the plant life, for which safety-related systems and equipment are to be designed and qualified. The plant conditions considered in defining the environmental conditions are normal operating including anticipated abnormal operating and test, and accident conditions including post accident operations. The accident condition considered is a hypothesized single event (not reasonably expected during the course of plant operation) that has the potential to cause severe environmental conditions for safety-related equipments.

The primary environmental parameters addressed are: pressure, temperature, relative humidity, radiation and chemical conditions. Safety-related systems and equipment are to be designed and qualified for the environmental conditions specified in this appendix. The parameters specified in this appendix do not include margins that may be required to satisfy applicable codes and standards for equipment qualification. The radiation data specified in this appendix is intended to provide a conservative basis for equipment qualification and is not intended to limit or justify personnel access.

Table 3I-1 is a cross reference of environmental data tables for normal and accident thermodynamic and radiation parameters and significant plant compartments and buildings.

3I.2 Plant Zones

3I.2.1 Inside Primary Containment

The primary containment vessel (PCV) is subdivided into three thermodynamic and four radiation zones to represent the enveloping levels of the environmental parameters as shown in Figure 3I-1. Accident conditions reduce the number of enveloping levels of the environmental parameters.

Equipment located in these zones is shown on the primary containment arrangement Figures 1.2-2 through 1.2-3c and 1.2-13a through 1.2-13k. Safety class equipment is identified on the system P&ID and IED drawings for inside and outside the primary containment as separated by the indicated containment penetrations.

3I.2.2 Outside Primary Containment

The area outside the PCV includes:

(1) Reactor Building (secondary containment)

(2) Reactor Building (clean zone outside secondary containment)

(3) Control Building

(4) Turbine Building

The R/B primary and secondary containment boundaries and major equipment zones in the secondary containment are shown on the arrangement Figures 1.2-4 through 1.2-12. The building arrangement figures in Chapter 21 are overlaid with fire areas separation and equipment (Subsection 9A.4) and radiation level areas and equipment (Subsection 12.3). The table of contents for Chapter 21 provides a means to cross reference the overlaid information using the figures with the same elevation as noted in the titles. Major equipment zones are shown on the Control Building (C/B) arrangements, Figures 1.2-14 through 1.2-22, and the Turbine Building (T/B) arrangement, Figures 1.2-24 through 1.2-31. The one zone in the T/B locates the equipment portions of the Reactor Protection System. The diesel generator fuel and oil supply systems (tanks and transfer pumps) are underground outside of the R/B and are not affected by accident conditions.

3I.3 Environmental Conditions Parameters

3I.3.1 Plant Normal Operating Conditions

3I.3.1.1 Pressure, Temperature and Relative Humidity

Tables 3I-2 through 3I-2 define the thermodynamic environment conditions (pressure, temperature, and relative humidity) for areas inside and outside primary containment vessel during plant normal operating conditions including anticipated test and abnormal occurrences. The areas outside the primary containment vessel include: (1) Reactor Building (secondary containment, and clean zones), (2) Control Building, and (3) Turbine Building. The drywell cooling system controls the thermal environment within the drywell, thoroughly mixes the inerting gas, and condenses steam from leaks of the primary coolant to support the Leak Detection and Isolation System.

3I.3.1.2 Radiation

Tables 3I-7 through 3I-11 define the radiation environment conditions for areas inside and outside the primary containment vessel during plant normal operating conditions including anticipated test and abnormal occurrences. The areas outside the primary containment vessel include: (1) Reactor Building (secondary containment and clean zones), (2) Control Building, and (3) Turbine Building.

The dose values denote the integrated (over 60 years.) radiation field to which equipment may be exposed.

The COL applicant should review and revise, as necessary, the radiation environment conditions given in Tables 3I-7 through 3I-11 based upon as designed and as procured equipment (see Subsection 3I.3.3.1 for COL license information).

3I.3.2 Plant Accident Conditions

3I.3.2.1 Pressure, Temperature, and Relative Humidity

Tables 3I-12 through 3I-2 define the thermodynamic environment conditions for areas inside and outside the primary containment vessel during plant accident conditions including post accident periods. The areas outside the primary containment vessel include: (1) Reactor Building (secondary containment, clean zones, and main steam tunnel), and (2) Control Building. These environmental conditions are due to various postulated pipe ruptures outside the primary containment vessel in systems connected to the reactor coolant (steam or water) pressure boundary. The environmental conditions in the secondary containment are dominated by breaks in the CUW and RCIC systems. A discussion of the analyses of these breaks is provided in Subsection 6.2.3.3.1.4.

3I.3.2.2 Radiation

Tables 3I-16 through 3I-19 define the radiation environment conditions inside and outside the primary containment vessel during plant accident conditions, including post-accident periods. The areas outside the primary containment vessel include: (1) Reactor Building (secondary containment), and (2) Control Building. The dose values denote the integrated dose for six months to which equipment may be exposed.

The COL applicant should review and revise, as necessary, the radiation environment conditions given in Tables 3I-16 through 3I-19 based upon as designed and as procured equipment (see Subsection 3I.3.3.1 for COL license information).

3I.3.2.3 Water Quality and Submergence

Reactor water quality characteristics for the design basis LOCAs inside primary containment are as follows:

(1) pH = 5.3 to 8.9

(2) Conductivity $\leq 2.0\ \mu$ S/cm

(3) $\leq$ 8 ppm O_2, $\leq$ 1 ppm CO_2

(4) $\leq$ 1 ppm dissolved salts available to deposit as dry salts upon evaporation from hot surfaces.

(5) $\leq$ 150 ppb undissolved solids

(6) $\leq$ 60 ppb dissolved H_2 arising from $\leq$ 4.0% volume of H_2O in containment atmosphere.

A 1600 micrometer particle (maximum diameter) sized containment spray with a flow density of (approximately) 1.0 liter/s per square meter may be initiated at ten minutes following a loss-of-coolant accident (LOCA) signal and continuing for up to 100 days, for areas inside primary containment vessel (drywell and wetwell). The plant design includes provisions for drainage to prevent submergence of essential equipment in the upper drywell during spray operation. Essential equipment located in the lower drywell will be qualified for submergence.

Water quality characteristics for normal operations are listed in Table 5.2-5 for reactor water and Subsection 5.2.3.2.2 (Reference 5.2-9) for ECCS water systems.

3I.3.3 COL License Information

3I.3.3.1 Radiation Environment Conditions

The COL applicant should review and revise, as necessary, the radiation environment conditions given in Tables 3I-7 through 3I-11 and Tables 3I-16 through 3I-19 based upon as designed and as procured equipment (Subsections 3I.3.1.2 and 3I.3.2.2).

Table 3I-1 Plant Environment Location and Condition Cross Reference of Table Numbers

[1]Location / [2]Condition	Primary Containment	Secondary Containment	Clean Zone Outside Secondary Containment	Control Building	Turbine Building
Normal					
(a) Thermodynamic	3I-2	3I-2	3I-2	3I-2	3I-2
(b) Radiation	3I-7	3I-8	3I-8	3I-10	3I-11
Accidents					
(a) Thermodynamic	3I-12	3I-13	3I-2	3I-2	
(b) Radiation	3I-16	3I-17	3I-18	3I-19	

1. Specific zones are located on arrangement drawings, and typical equipment is identified on P&ID and IED design drawings referenced by Figure numbers on each page.
2. Test and abnormal environments are included with normal conditions.

Table 3I-2 Thermodynamic Environment Conditions Inside Primary Containment Vessel Plant Normal Operating Conditions[1]

No.	Plant Zone/Typical Equipment	Pressure[2] kPaG	Temperature °C	Relative Humidity
a-1	Upper drywell and lower area of lower drywell [Figs. 1.2-3, 1.2-3a/5.1-3]	–3.43 13.73	Max 65 Min 10 Ave 57	Max 90 Min 10
a-2	Upper area of lower drywell [Figs. 1.2-3b/11.2-1]	–3.43 13.73	Max 57[4] Min 10	Max 90 Min 10
a-3	Wetwell area (suppression pool and nitrogen space) [Figs. 1.2-3c/6.2-39/7.6-11]	–3.43 13.73	Max 35[3] Min 10	Nor 100

1. The primary containment atmosphere is nitrogen
2. Primary containment atmosphere will be pressurized to 279.49 kPaG during integrated leak rate test, for less than 3 days. (Test)
3. The temperature of suppression pool water may reach 49°C during reactor isolation. (Abnormal)
4. CRD housing area

Table 3I-3 Thermodynamic Environment Conditions Inside Reactor Building (Secondary Containment) Plant Normal Operating Conditions

Plant Zone/Typical Equipment	Pressure[1] kPaG	Temperature °C	Relative Humidity
General floor area (not otherwise noted) /Similar Equipment	0	Max 40 Min 10	Max 90 Min 10
RHR pump rooms [Figs. 1.2-4/5.4-10]	0	Max 40[2] Min 10	Max 90 Min 10
RCIC pump room [Figs. 1.2-4/5.4-8]			
HPCF pump rooms [Figs. 1.2-4/6.3-7]			
FPC pump room [Figs. 1.2-9/9.1-1]			
SGTS rooms [Figs. 1.2-10/6.5-1]			
MS tunnel room [Figs. 1.2-8/5.1-3]	0	Max 60 Min 10	Max 90 Min 10
Divisional valve rooms [Figs. 1.2-8/ ECCS]	0	Max 60 Min 10	Max 90 Min 10
Instrument rack rooms [Figs. 1.2-6/ ECCS]	0	Max 40 Min 10	Max 90 Min 10
CUW heat exchanger rooms (Figs. 1.2-4 and 5.4-12)	0	Max 50 Min 10	Max 90 Min 10

1. The indicated (positive or negative) pressure will be maintained. Pressure difference will not be controlled.
2. During pump operation (test running, etc.) this temperature will be a Max. 66°C. The frequency of this maximum temperature occurrence is assumed 2 hours/month (test) or 90 days/year in RHR room (abnormal) and 2 hours/month in the other rooms.

Table 3I-4 Thermodynamic Environment Conditions Inside Reactor Building (Outside Secondary Containment) Plant Normal Operating Conditions

Plant Zone/Typical Equipment	Pressure[1] kPaG	Temperature °C	Relative Humidity
Clean zone outside secondary containment area (not otherwise noted) [Figs. 6.2-26/6.7-1]	0	Max 40 Min 10	Max 90 Min. 10
Diesel generator rooms [Figs. 1.2-8/9.5-6]	0	Max 50 Min 10	Max 90 Min 10
SGTS Monitor room [Figs. 1.2-8/6.5-1]	0	Max 40 Min 10	Max 90 Min 10

1. The indicated (positive or negative) pressure will be maintained. Pressure difference will not be controlled.

Table 3I-5 Thermodynamic Environment Conditions Inside Control Building Plant Normal Operating Conditions

Plant Zone/Typical Equipment	Pressure[1] kPaG	Temperature °C	Relative Humidity
Control Building rooms (unless otherwise noted) [Figs. 1.2-15/9.2-1a]	0	Max 40 Min 10	Max 90 Min 10
Main control and computer rooms [Figs. 1.2-15/18C7-1]	0	Max 30 Min 10	Max 60 Min 10
Control Building HVAC equipment rooms [Figs. 1.2-15/9.2-1a}	0	Max 40 Min 5	Max 90 Min 10

1. The indicated (positive or negative) pressure will be maintained. Pressure difference will not be controlled.

Table 3I-6 Thermodynamic Environment Conditions Inside Turbine Building Plant Normal Operating Conditions

Plant Zone/Typical Equipment	Pressure[1] kPaG	Temperature °C	Relative Humidity
Main steam stop valve area [Figs. 1.2-25/7.2-9	0	Max 60 Min 10	Max 90 Min 10

1. The indicated (positive or negative) pressure will be maintained. Pressure difference will not be controlled.

Table 3I-7 Radiation Environment Conditions Inside Primary Containment Vessel Plant Normal Operating Conditions

		Operating Dose Rate			Integrated Dose[1] and Neutron Fluence		
No.	Plant Zone/Typical Equipment	Gamma (Gy/h)[2]	Beta (Gy/h)[3]	Neutron/cm^2-s	Gamma(Gy)	Beta (Gy)	Neutron /cm^2
b-1	Upper drywell area [Figs. 1.2-3/ 5.1-3]	0.2	Neg	$6x10^4$	1E+5	Neg	$1x10^{14}$
b-2	Upper area of lower drywell [Figs. 1.2-3a/ 5.1-3]	0.2 (See Note 4)	Neg	$2x10^4$	1E+5	Neg	$4x10^{13}$
b-3	Lower area of lower drywell [Figs.1.2-3b/ 11.2-1]	0.15	Neg	$1x10^4$	8E+4	Neg	$2x10^{13}$
b-4	Wetwell area (suppression pool and airspace) [Figs. 1.2-3c/ 6.2-39, 7.6-11]	<0.01	Neg	$2x10^2$	5E+3	Neg	$4x10^{11}$

1. Integration time based upon 1.5 year cycles at 18 months operations at 95% availability over 60 years.
2. Operating dose rate at 100% rated power and 30 cm away from the radiation source.
3. Beta dose rates negligible (neg.), primarily due to Ar-41 and typically only in area between vessel and shield wall.
4. Gamma dose rate directly under vessel. Dose rate will increase to 90 Gy/h inside shield directly opposite core mid-plane.

Table 3I-8 Radiation Environment Conditions Inside Reactor Building (Secondary Containment) Plant Normal Operating Conditions

	Operating Dose Rate		Integrated Dose[1]	
Plant Zone/Typical Equipment	**Gamma (Gy/h)[2]**	**Beta (Gy/h)[3]**	**Gamma (Gy)**	**Beta (Gy)**
General floor area (not otherwise noted)/Similar equipment	5E-5	Neg	3E+1	Neg
RHR rooms [Figs. 1.2-4/5.4-10]	3E-4	Neg	2E+2	Neg
RCIC room [Figs. 1.2-4/5.4-8]	5E-5/2E-2 (See Note 4)	Neg	3E+1	Neg
HPCF rooms [Figs. 1.2-4/6.3-7]	5E-5	Neg	3E+1	Neg
SGTS rooms [Figs. 1.2-10/6.5-1]	5E-5	Neg	3E+1	Neg
CUW room				
Heat exchanger [Fig. 1.2-4]	2E-1	Neg	1E+5	Neg
Pump room [Fig. 1.2-4]	6E-3	Neg	3E+3	Neg
Filter demin/tank room [Fig. 1.2-6]	2	Neg	1E+6	Neg
MS tunnel [Figs. 1.2-8/5.1-3]	4E-2	Neg	2E+4	Neg
Divisional valve rooms [Figs. 1.2-8/ECCS]	5E-4	Neg	3E+2	Neg
Instrument rack rooms [Figs. 1.2-6/ECCS]	<1E-5	Neg	5E+0	Neg

1. Integration time based upon 1.5 year cycles at 18 months operations at 95% availability over 60 years.
2. Operating dose rate at 100% rated power and 30 cm away from the radiation source.
3. Beta dose rates negligible (neg.), generally less than 0.001 m Gy/h.
4. During system check out dose rate will increase to larger number.

Table 3I-9 Radiation Environment Conditions Inside Reactor Building (Outside Secondary Containment) Plant Normal Operating Conditions

	Operating Dose Rate		Integrated Dose[1]	
Plant Zone/Typical Equipment	Gamma[2] (Gy/h)	Beta [3](Gy/h)	Gamma (Gy)	Beta (Gy)
Clean zone outside secondary containment area (not otherwise noted) [Figs 6.2-26/ 6.7-1]	6E-6	Neg	3	Neg
Monitor room [Figs. 1.2-8/ 6.5-1]	5E-5	Neg	30	Neg

1. Integration time based upon 1.5 year cycles at 18 months operations at 95% availability over 60 years.
2. Operating dose rate at 100% rated power and 30 cm away from the radiation source.
3. Beta dose rates negligible (neg.), generally less than 0.001 m Gy/h.

Table 3I-10 Radiation Environment Conditions Inside Control Building Plant Normal Operating Conditions

	Operating Dose Rate		Integrated Dose[1]	
Plant Zone/Typical Equipment	Gamma[2] (Gy/h)	Beta[3](Gy/h)	Gamma (Gy)	Beta (Gy)
Main control room, battery and HVAC rooms [Fig 1.2-15]	6E-6	Neg	3	Neg
RCW pump and heat exchanger room [Fig 1.2-15]	5E-5	Neg	27	Neg

1. Integration time based upon 60 years.
2. Operating dose rate at 100% rated power and 30 cm away from the radiation source.
3. Beta dose rates negligible (neg.), generally less than 0.001 m Gy/h.

Table 3I-11 Radiation Environment Conditions Inside Turbine Building Plant Normal Operating Conditions

Plant Zone/Typical Equipment	Operating Dose Rate Gamma[2] (Gy/h)	Operating Dose Rate Beta[3] (Gy/h)	Integrated Dose[1] Gamma (Gy)	Integrated Dose[1] Beta (Gy)
Main steam stop valve area [Figs 1.2-25/7.2-9]	0.1	Neg	5E+4	Neg

1. Integration time based upon 1.5 year cycles at 18 months operations at 95% availability over 60 years.
2. Operating dose rate at 100% rated power and 30 cm away from the radiation source.
3. Beta dose rates negligible (neg.), generally less than 0.001 m Gy/h.

Table 3I-12 Thermodynamic Environment Conditions Inside Primary Containment Vessel Plant Accident Conditions

No.	Plant Zone/Typical Equipment		Time[1] 3(h)	6(h)	1 (day)	100 (day)
a-1 & a-2	Drywell area [Figs. 1.2-3, 1.2-3a, 1.2-3b/5.1-3, 11.2-1]	Temperature(°C)	171	160	121	93
		Pressure (kPaG)	– 13.73 ~309.89	– 13.73 ~309.89	0 ~172.60	0 ~138.27
		Humidity(%)	Steam	Steam	100	100
a-3	Wetwell area [Figs. 1.2-3c/6.2-39, 7.6-11]	Temperature(°C)	122	122	122	100
		Pressure (kPaG)	– 13.73 ~309.89	– 13.73 ~309.89	0 ~172.60	0 ~138.27
		Humidity(%)	100	100	100	100

1. Time” defines the period after LOCA. For example, “3(h)” means 3 hours after the occurrence of LOCA, and “1(day)” means time period between 6 hours after LOCA and 24 hours after LOCA.

Table 3I-13 Thermodynamic Environment Conditions Inside Reactor Building (Secondary Containment) Plant Accident Conditions[1]

Plant Zone/Typical Equipment		Time[2] 1 (h)	6 (h)	12 (h)	100 (day)
Control rod drive hydraulic system (scram etc. of hydraulic control unit) [Figs. 1.2-4/4.6-8]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97 Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
Control rod hydraulic pumps (scram system of hydraulic control unit) [Figs. 1.2-4/4.6-8]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
RCIC (valves except isolation valves, assemblies, cable, turbine, pipe spaces, corridor) [Figs. 1.2-4/5.4-8]	Temperature (°C) Pressure (kPaG) Humidity (%)	142 102.97[3] Steam	142 102.97[3] Steam	66 3.43 100	66 0 90 Max
RCIC turbine electric control system[3,6] [Figs. 1.2-5/5.4-8]	Temperature (°C) Pressure (kPaG) Humidity (%)	142 102.97[3] Steam	142 102.97[3] Steam	66 3.43 100	66 0 90 Max
RHR (LPFL, cooling system at S/D, containment cooling, service water system) valve, pump (motor, seal cooler) instrument control electric equipment (including cable and sources of electricity) [Figs. 1.2-4/5.4-10]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
HPCF pump, motor (seal cooler) instrument, control electric equipment (including cable and sources of electricity) [Figs. 1.2-4/6.3-7]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
Neutron monitor system[6], (cable of IRM, preamplifier, drive relay panel, cable of LPRM) [Figs. 1.2-3b/7.6-1]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90Max
Leak detection installation (steam water)[4,6] (instrument, sources of electricity) instrument and sources of electricity for surveillance after accident [Figs. 1.2-6/5.2-8]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max

Table 3I-13 Thermodynamic Environment Conditions Inside Reactor Building (Secondary Containment) Plant Accident Conditions[1] (Continued)

Plant Zone/Typical Equipment		Time[2] 1 (h)	6 (h)	12 (h)	100 (day)
FPC (cooling system, SPCU [makeup water system] valve, pump motor, heat exchanger, instrument, control electric equipment) cable sources of electricity, pipe spaces [Figs. 1.2-9/9.1-1]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
CUW (Pump, valve, non-regen. and regen. heat exchangers, pipe spaces, filter demin filter demin. valve rooms) corridor [Figs. 1.2-4/5.4-12]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
FCS[6] valves including isolation valve (recombiner instrument, controls), electrical equipment (power source cables) [Figs. 1.2-8/6.2-40]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
Mainsteam Tunnel (Outside secondary containment)					
MS isolation valve[5] MS drain isolation valve Nitrogen line isolation valve[5,6] Process water line isolation valve[5,6] [Figs 1.2-2, 1.2-3, 1.2-3a, 5.1-3]	Temperature (°C) Pressure (kPaG) Humidity (5)	171 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
Feedwater isolation valve [5] [Figs 1.2-2, 1.2-3, 1.2-3a/5.1-3]	Temperature (°C) Pressure (kPaG) Humidity (%)	171 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
RCIC, check valve (inside MS tunnel) [Figs. 1.2-2, 1.2-3, 1.2-3a/5.4-8]	Temperature (°C) Pressure (kPaG) Humidity (%)	171 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max

1. Systems or components located in the Reactor Building outside the secondary containment or in other buildings and required to support the equipment listed in this table during accident condition will be qualified to the conditions specified in the equipment qualification design criteria table for the respective area or building.
2. Time means the time from the occurrence of LOCA.
3. The 102.97 kPaG equipment qualification pressure specified is the structural design basis for the respective rooms (see Subsection 6.2.3.3) in which this equipment is located and not the saturation pressure associated with the equipment qualification temperature.

4. Safety-related motor control centers, power centers, metal clad switchgear, and remote multiplexing units in the reactor building are located outside the secondary containment in the emergency electrical equipment rooms.
5. Valve assemblies and cables required for valve operation are included.
6. 100°C may be applied in the case that adequate separation in the arrangement is ensured and there is no possibility of exposure to steam environment.

Table 3I-14 Thermodynamic Environment Conditions Inside Reactor Building (Outside Secondary Containment) Plant Accident Conditions

Plant Zone/Typical Equipment	Pressure[1] kPaG	Temperature °C	Relative Humidity
Clean zone outside secondary containment (not otherwise noted) [Figs. 6.2-26/ 6.7-1]	0	Max 50 Min 10	Max 90 Min 10
Diesel generator room [Figs. 1.2-8/9.5-6]	0	Max 50 Min 10	Max 90 Min 10
Monitor room [Figs. 1.2-8/6.5-1]	0	Max 50 Min 10	Max 90 Min 10

1. The indicated (positive or negative) pressure will be maintained. Pressure difference will not be controlled.

Table 3I-15 Thermodynamic Environment Conditions Inside Control Building Plant Accident Conditions

Plant Zone/Typical Equipment	Pressure[1] kPaG	Temperature °C	Relative Humidity
Control Building (not otherwise noted) [Figs. 1.2-15/9.2-1a]	0	Max 50	Max 90 Min 10
Main control, computer, and battery rooms [Figs. 1.2-15/18C-1]	0	Max 30 Min 21	Max 60
Control Building HVAC equipment rooms Figs. 1.2-15/9.2-1a]	0	Max 50	Max 90 Min 10

1. The indicated (positive or negative) pressure will be maintained. Pressure difference will not be controlled.

Table 3I-16 Radiation Environment Conditions Inside Primary Containment Design Basis Accident Conditions

		LOCA Dose Rate		Integrated Dose[1]	
Plant Zone/Typical Equipment	Accident	Gamma (Gy/h)	Beta (Gy/h)	Gamma (Gy)	Beta (Gy)
Drywell [Figs. 1.2-3, 1.2-3a/5.1-3]	15.6.5	2E+5	2E+6	2E+6	2E+7
Wetwell [Figs. 1.2-3c/6.2-39, 7.6-11]	15.6.5	3E+5	4E+6	3E+6	5E+7

1. Integrated dose is summed over a six month period for Accident Case 15.6.5.

Table 3I-17 Radiation Environment Conditions Inside Reactor Building Design Basis Accident (Secondary Containment)

		LOCA Dose Rate		Integrated Dose[1]	
Plant Zone/Typical Equipment	Accident	Gamma (Gy/h)	Beta (Gy/h)	Gamma (Gy)	Beta (Gy)
General floor area [Fig. 1.2-4]	15.6.5	8E-2	2E+0	2E+1	3E+2
RHR room [Figs. 1.2-4/5.4-10]	15.6.5	2E+3	1E+5	6E+5	8E+7
RCIC room [Figs. 1.2-4/5.4-8]	15.6.2	7E-2	1E+0	9E-1	3E+1
HPCF room [Figs. 1.2-4/6.3-7]	15.6.5	1E+3	6E+4	4E+5	5E+7
SGTS room [Figs. 1.2-10/6.5-1]	15.6.5	2E+4	2E+0	3E+7	3E+2
MS tunnel [Figs. 1.2-8/5.1-3]	15.6.4	9E-1	7E+0	2E+0	9E+0
Divisional valve room [Figs 1.2-5/ECCS]	15.6.5	2E+3	2E+5	8E+5	2E+8
Instrument rack room [Figs. 1.2-6/ECCS]	15.6.5	3E-2	2E+0	5E+0	5E+2

1. Integrated dose is summed over a six month period for Accident Case 15.6.5, 6 hours for 15.6.2, and 2 hours for 15.6.4.

Table 3I-18 Radiation Environment Conditions Inside Reactor Building Design Basis Accident Conditions (Outside Secondary Containment)

Plant Zone/Typical Equipment	Accident	LOCA Dose Rate Gamma (Gy/h)	LOCA Dose Rate Beta (Gy/h)	Integrated Dose[1] Gamma (Gy)	Integrated Dose[1] Beta (Gy)
Clean zone outside secondary containment area (not otherwise noted) [Figs. 6.2-26/6.7-1]	15.6.5	8E-5	2E-3	2E-2	3E-1
Monitor room [Figs. 1.2-8/6.5-1]	15.6.5	8E-5	2E-3	2E-2	3E-1

1. Integrated dose is summed over a six month period for Accident Case 15.6.5.

Table 3I-19 Radiation Environment Conditions Inside Control Building Design Basis Accident Conditions

Plant Zone/Typical Equipment	Accident	LOCA Dose Rate Gamma (Gy/h)	LOCA Dose Rate Beta (Gy/h)	Integrated Dose[1] Gamma (Gy)	Integrated Dose[1] Beta (Gy)
Main Control Room and Process computer room[4]	15.6.5	1.0E-3	1.0E-2	8.0E-2	2.0E+0
HVAC Rooms, Level 17150mm[4]	15.6.5	5.0E-3 to 1.0E-1[2]	5.0E-2	3.0E+1 to 1.0E+2[2]	3.0E+2
All other areas[4]	15.6.5	5.0E-3 to 5.0E-2[3]	5.0E-2	3.0E+1 to 3.0E+2[3]	3.0E+2

1. Integration dose is summed over a six month period for Accident Case 15.6.5.
2. Highest dose rates closer to CR-HVAC Filter Units.
3. Highest dose rates closer to the RBCW Units and HVAC intake units.
4. Refer to Figure 1.2-15.

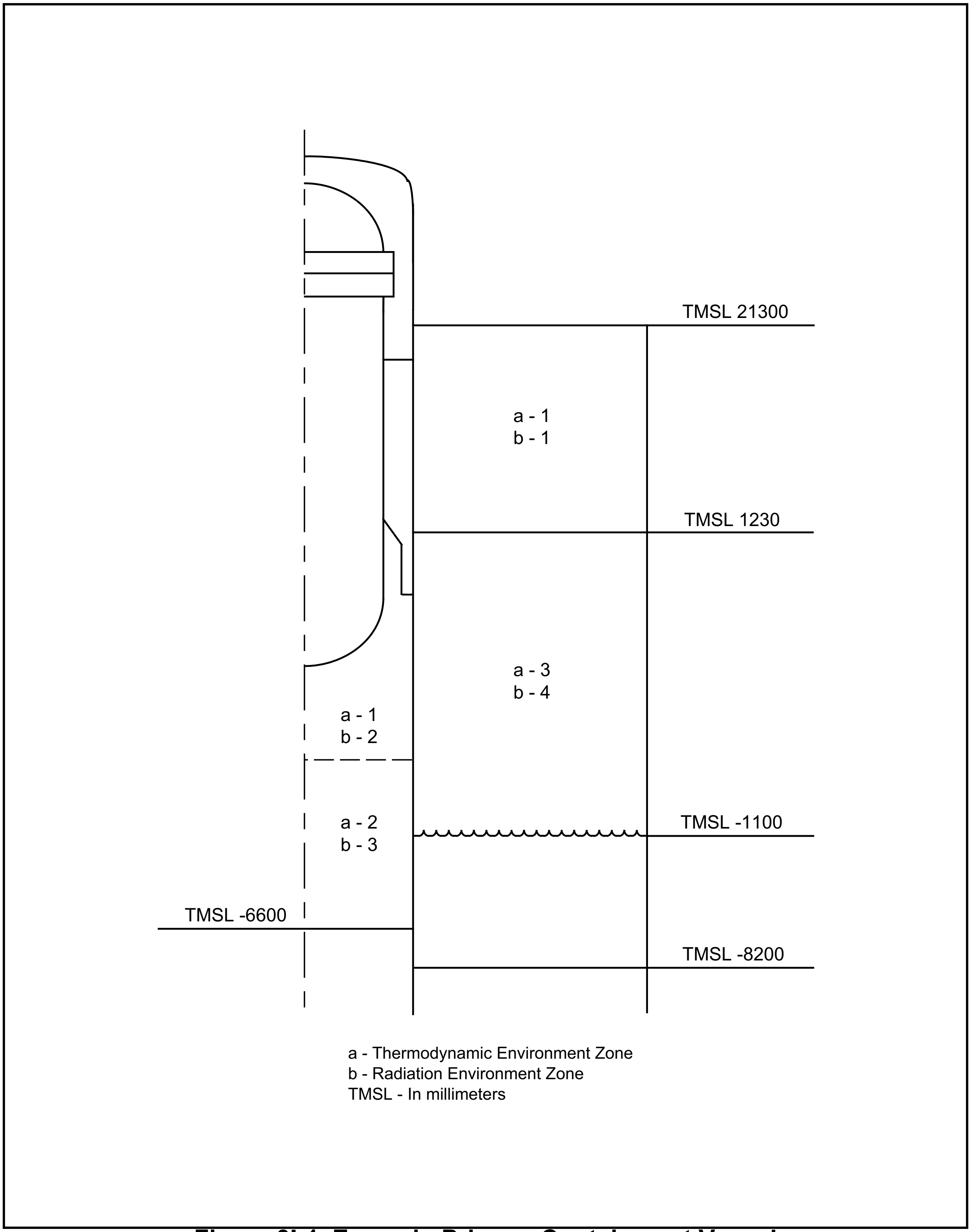

Figure 3I-1 Zones in Primary Containment Vessel

3J Not Used

3K Designated NEDE-24326-1-P Material Which May Not Change Without Prior NRC Staff Approval

[*This Appendix presents the necessary NEDE-24326-1-P (Reference 3.9-6 or 3.11-2) material for identifying the material by italics which may not change without prior NRC Staff approval.*]*

3K.1 General Requirements for Dynamic Testing (4.4.2.5.1†)

(a) [***Mounting*** – *Specimens to be tested will be mounted in a manner that adequately simulates the installed configuration or as described in the applicable GE mounting documentation.* Mounting will be specified in the PPQS.

(b) ***Monitoring*** – *Sufficient monitoring equipment will be used to evaluate the performance of the specimen before, during, and after the test. Monitoring product is used to allow determination of applied vibration levels and equipment responses. The location of monitoring sensors* shall be specified by the PPQS and *will be documented in the test report.*

When required by the PPQS*, the response of the product will be measured using accelerometers. When required* by the PPQS*, the accelerometers shall be located at a sufficient number of locations on the product to define the mode shapes and/or frequencies which would be required to allow dynamic qualification of individual safety-related components and devices, to support analytical extrapolation of test results, or to verify frequency requirements.*

(c) ***Exploratory Tests*** – *Exploratory vibration tests may be performed on the product to aid in the determination of the test method that will best qualify or determine the dynamic characteristics of the product. If it can be shown that the equipment is not resonant at any frequency within the expected frequency range, it may be considered a rigid body and tested according to methods and procedures discussed in Subsection 4.4.2.5.6† or analyzed according to the methods of Subsection 4.4.4.1.4.5†.*

If the product contains a single resonance or multiple resonances, one of the methods outlined in Subsection 4.4.2.5.3† will be used to qualify the product t by test.

The exploratory test may be performed in the form of a low-level, continuous sinusoidal sweep at a rate no greater than 1 octave per minute over the frequency range equal to or greater than that to which the equipment is to be qualified. All resonances will be recorded for use in determining the test method to be used or the dynamic characteristics of the equipment. If the configuration of the product is such that critical natural frequencies cannot be ascertained, dynamic qualification will be accomplished by testing by the Response Spectrum method as specified in Paragraph 4.4.2.5.3.6†. An acceptable alternative qualification method is a fragility test as described in Subsection 4.4.2.5.7†.

* See Subsection 3.10.

† Refers to section numbers of NEDE-24326-1-P.

(d) ***Dynamic Event Aging Tests*** *– The dynamic tests simulate the effect of five (5) upset events* and in-service hydrodynamic loads having a long duration in order to simulate dynamic event aging followed by one (1) faulted event.† The dynamic tests are performed on aged products unless otherwise justified.*]‡

There are two hydrodynamic loads that have long durations: Safety Relief valve (SRV) and Chugging. The first step in considering these long duration hydrodynamic loads is to obtain Required Response Spectra (RRS) data for the worst SRV and for Chugging events. These spectra should not include any other loads. Having obtained the appropriate RRS's, the duration of SRV testing is determined by multiplying the number of SRV actuations by 0.5 second. The number of SRV actuations is given in Table 3.9-1.*ƒ*

Chugging tests will have a 15 minute duration.

Since Chugging is a post LOCA phenomena, Chugging will only be applied to equipment which is required to function post LOCA. SRV will be applied to all equipment located in areas where hydrodynamic loads exist.

The test sequence to be used when addressing long term hydrodynamic loads will be:

(1) Vibration aging (if required)

(2) SRV cycles (duration as above)

(3) 5 Upset events* (0.5 SSE**+ hydrodynamic) (30 seconds each)

(4) 1 Faulted event (SSE + hydrodynamic) (30 second duration)

(5) Chugging (15 minute duration)

[*Since most testing will be biaxial rather than triaxial, the above sequence and durations will be applied twice with the equipment being rotated 90 degrees on the table between the two tests.*]‡

[*The Test Response Spectra (TRS) will envelop the RRS as specified in 4.2.2.a(6)*††.]‡

* *Upset Event - 0.5 SSE (in lieu of the OBE specified by NEDE-24326-1-P), or alternatively, as described in Subsection 3.7.3.2.*

† *Faulted Event - The SSE combined with appropriate hydrodynamic loads.*

‡ See Section 3.10.

ƒ Table 3.9-1 to be used in lieu of the SRV actuations specified by NEDE-24326-1-P.

** 0.5 SSE to be used in lieu of the OBE specified by NEDE-24326-1-P.

†† Refers to section numbers of NEDE-24326-1-P.

For SRV tests, the TRS will be examined to assure that motion cycles are equal to or greater than 4X the number of SRV actuations.

(e) [***Loading*** *– Dynamic tests will be performed with the product subjected to nominal operating service conditions. If significant, normal operating loads such as electrical, mechanical, pressure, and thermal will be included. Where normal operating loads cannot be included in the dynamic tests, supplemental analysis will be used to qualify the product for those effects.*]*

3K.2 Product and Assembly Testing (4.4.2.5.2†)

(a) [*Products will be tested simulating nominal operating conditions.*‡]* The product shall be mounted on the shaker table as stated in Paragraph 4.4.2.5.1(a)† . If the product is intended to be mounted on a panel, the panel will be included in the test mounting.

Alternatively, the response at the product mounting location may be measured in the assembly test as specified in Paragraph 4.4.2.5.1(a)†. Then the product will be mounted directly to the shake table, with the dynamic input being that which was determined at the product mounting location.

3K.3 Multiple-Frequency Tests (4.4.2.5.3†)

(a) **General** – [*When the dynamic ground motion has not been strongly filtered, the mounting location retains the broadband characteristics. In this case, multi-frequency testing is applicable to dynamic qualification.*]*

(b) **Response Spectrum Test** – Testing shall be performed by applying artificially generated input excitation to the product, the amplitude of which is controlled in 1/3 octave or narrower bands. [*The excitation will be controlled to provide a test response spectrum (TRS) which meets or exceeds the required response spectrum (RRS). The peak value of the input excitation equals or exceeds the zero period acceleration (ZPA) of the RRS.*]*

3K.4 Single- and Multi-axis Tests (4.4.2.5.4†)

[*Single-axis tests may be allowed if the tests are designed to conservatively reflect the dynamic event at the equipment mounting locations or if the product being tested can be shown to respond independently in each of the three orthogonal axis or otherwise withstand the dynamic event at its mounting location.*

* See Section 3.10.

† Refers to section number of NEDE-24326-1-P.

‡ *In addition, dynamic coupling between interacting equipment will be considered.*

If the preceding considerations do not apply, multi-axis testing will be used. The minimum is biaxial testing with simultaneous inputs in a principal horizontal axis and the vertical axis. Independent random inputs are preferred, and, if used, the test will be performed in two steps with the equipment rotated 90° in the horizontal plane for the second step. If independent random inputs are not used (such as with single frequency tests), four tests would be run; first, with the inputs in phase; second, with one input 180° out of phase; third, with the equipment rotated 90° horizontally and the inputs in phase; and, finally, with the same equipment orientation as in the third step but with one input 180° out of phase.]*

3K.5 Single Frequency Tests (4.4.2.5.6†)

[*If it can be shown that the products, as defined in R.G.1.92 has no resonances, or only one resonance, or if resonances are widely spaced and do not interact to reduce the fragility level in the frequency range of interest or, if otherwise justified, single frequency tests may be used to fully test the product.*]*

3K.6 Damping (4.4.2.5.7†)

[*The product damping value used for dynamic qualification shall be established. See Section 3.5 of IEEE-344†.‡*]*

3K.7 Qualification Determination (4.4.3.3†)

[*In order for equipment to be qualified by reason of operating experience, documented data will be available confirming that the following criteria have been met:*

(a) *the product providing the operating experience is identical or justifiably similar to the equipment to be qualified;*

(b) *the product providing the operating experience has operated under all service conditions which equal or exceed, in severity, the service conditions and performance requirements for which the product is to be qualified; and*

(c) *the installed product must, in general, be removed from service and subjected to partial type testing to include the dynamic and design basis event environments for which the product is to be qualified.*]*

3K.8 Dynamic Qualification by Analysis (4.4.4.1.4†)

(a) The analytical procedures described in this section may be used for dynamic qualification of products.

* See Section 3.10.

† Refers to section numbers of NEDE-24326-1-P.

‡ Also see subsections 3.7.3.8.1.7, 3.9.2.2, 3.9.3 and 3.10.2.

(b) Many factors control the design of a qualification program. Paragraphs 4.2.2.c(3)*and 4.2.2.d(1)* provide general guidelines on dynamic analysis techniques. Analytical techniques and modeling assumptions will, when possible, be based on a correlation of the analytical approach with testing or operating experience performed on similar equipment or structures. [*Analysis may be used as a qualification method for the following conditions:*

(1) *if maintaining structural integrity is the only required assurance of the safety function,*]†

(2) if the response of the equipment is linear or has a simple nonlinear behavior which can be predicted by conservative analytical methods, or

(3) if the product is too large to test.

3K.9 Required Response Spectra (4.4.4.1.4.6.2*)

(a) [*The required response spectra that define the dynamic criteria for the location(s) of the product under consideration are to be given in the PPQs. If the equipment under consideration is attached to the structural system at more than one location, then the dynamic analysis performed takes into consideration the different response spectra at the different support locations. The effect of multiple support attachment points or multiple locations of the particular product can also be accounted for by selecting a single spectrum which will effectively produce the critical maximum responses due to different accelerations existing at different points.*]† This may be conservatively accomplished by enveloping the response spectra for the different applicable locations. Alternatively, actual multi-support excitation effects may be taken into account by performing a multi-support excitation analysis.

3K.10 Time History Analysis (4.4.4.1.4.6.3*)

Time history analysis will be performed when conditions arise invalidating the response spectrum method of analysis due to nonlinear phenomena, or when generation of in-equipment response spectra or a more exact result is desired. To integrate or differentiate, the analysis will be done by an applicable numerical integration technique. The largest time step used in the analysis will be 1/10 of the period of the highest significant mode of vibration of the equipment. [*The dynamic input will be the time history motion at the equipment support location.*]† For products supported at several locations, the responses will be determined by simultaneous excitations using appropriate time history input at each support location. The scaled time

* Refers to section number of NEDE-24326-1-P

† See Section 3.10.

interval will be varied as per Paragraph 4.4.2.a(6)*. If the product frequency is within the range of the supporting structure, then a time interval will be chosen such that the peak of the response spectrum shall be at the product resonance frequency. The total time interval range will be provided with the time history.

* Refers to section number of NEDE-24326-1-P.

3L Evaluation of Postulated Ruptures in High Energy Pipes

3L.1 Background and Scope

An evaluation of the dynamic effects of fluid dynamic forces resulting from postulated ruptures in high energy piping systems is required by SRP 3.6.1 and 3.6.2. The criteria for performing this evaluation is defined in Subsections 3.6.1 and 3.6.2 of this Tier 2 and in the Standard Review Plans and ANS 58.2 which are referenced in the Tier 2.

This Appendix defines an acceptable procedure for performing these evaluations. The procedure is based on use of analytical methodology, computer programs and pipe whip restraints used by GE, but it is intended to be applicable to other computer programs and to pipe whip restraints of alternate design. Applicability of alternate programs will be justified by the COL.

The evaluation is performed in four major steps:

(1) Identify the location of the postulated rupture and whether the rupture is postulated as circumferential or longitudinal.

(2) Select the type and location of the pipe whip restraints.

(3) Perform a complete system dynamic analysis or a simplified dynamic analysis of the ruptured pipe and its pipe whip restraints to determine the total movement of the ruptured pipe, the loads on the pipe, strains in the pipe whip restraint, and the stresses in the penetration pipe.

(4) Evaluate safety-related equipment that may be impacted by the ruptured pipe or the target of the pipe rupture jet impingement.

The criteria for locations where pipe ruptures must be postulated and the criteria for defining the configuration of the pipe rupture are defined in Subsection 3.6.2. Also defined in Subsection 3.6.2 are: (1) the fluid forces acting at the rupture location and in the various segments of the ruptured pipe, (2) the jet impingement effects including jet shape and direction and jet impingement load.

The high energy fluid systems are defined in Subsection 3.6.2.1.1 and identified in Tables 3.6-3 and 3.6-4. Essential systems, components and equipments, or portions thereof, specified in Tables 3.6-1 and 3.6-2 are to be protected from pipe break effects which would impair their ability to facilitate safe shutdown of the plant.

The information contained in Subsections 3.6.1 and 3.6.2 and in the SRPs and ANS 58.2 is not repeated in this Appendix.

3L.2 Identification of Rupture Locations and Rupture Geometry

3L.2.1 Ruptures in Containment Penetration Area.

Postulation of pipe ruptures in the portion of piping in the containment penetration area is not allowed. This includes the piping between the inner and outer isolation valves. Therefore, examine the final stress analysis of the piping system and confirm that, for all piping in containment penetration areas, the design stress and fatigue limits specified in Subsection 3.6.2.1.4.2 are not exceeded.

3L.2.2 Ruptures in Areas other than Containment Penetration.

(1) Postulate breaks in Class 1 piping in accordance with Subsection 3.6.2.1.4.3.

(2) Postulate breaks in Classes 2 and 3 piping in accordance with Subsection 3.6.2.1.4.4.

(3) Postulate breaks in seismically analyzed non-ASME Class piping in accordance with the above requirements for Classes 2 and 3 piping.

3L.2.3 Determine the Type of Pipe Break

Determine whether the high energy line break is longitudinal or circumferential in accordance with Subsection 3.6.2.1.6.1.

3L.3 Design and Selection of Pipe Whip Restraints

3L.3.1 Make Preliminary Selection of Pipe Whip Restraint

The load carrying capability of the GE U-Bar pipe whip restraint is determined by the number, size, bend radius and the straight length of the U-bars. The pipe whip restraint must resist the thrust force at the pipe rupture location and the impact force of the pipe. The magnitude of these forces is a function of the pipe size, fluid, and operating pressure.

A preliminary selection of one of the standard GE pipe whip restraints is made by matching the thrust force at the rupture location with a pipe whip restraint capable of resisting this thrust force. This is done by access to the large database contained in the GE REDEP computer file. This file correlates the pipe size and the resulting thrust force at the pipe rupture with the U-bar pipe whip restraints designed to carry the thrust force. REDEP then supplies the force/deflection data for each pipe whip restraint.

3L.3.2 Prepare Simplified Computer Model of Piping-Pipe Whip Restraint System.

Prepare a simplified computer model of piping system as described in Subsection 3L.4.2.1 and as shown in Figures 3L-1 and 3L-2. Critical variables are length of pipe, type of end condition, distance of pipe from structure and location of the pipe whip restraint. Locate the pipe whip restraint as near as practical to the ruptured end of the pipe but establish location to minimize interference to Inservice Inspection.

3L.3.3 Run "Pipe Dynamic Analysis" (PDA)

Run the PDA computer program using the following input:

(1) The information from the simplified piping model, including pipe length, diameter, wall thickness and pipe whip restraint location.

(2) Piping information such as pipe material type, stress/strain curve and pipe material mechanical properties.

(3) Pipe whip restraint properties such as force-deflection data and elastic plastic displacements.

(4) Force time-history of the thrust at the pipe rupture location.

3L.3.4 Select Pipe Whip Restraint for Pipe Whip Restraint Analysis.

PDA provides displacements of pipe and pipe whip restraint, pipe whip U-bar strains, pipe forces and moments at fixed end, time at peak load and lapsed time to achieve steady state using thrust load and pipe characteristics.

Check displacements at pipe broken end and at pipe whip restraint and compare loads on the piping and strains of pipe whip restraint U-bars with allowable loads and strains. If not satisfied with output results rerun PDA with different pipe whip restraint parameters.

3L.4 Pipe Rupture Evaluation

3L.4.1 General Approach

There are several analytical approaches that may be used in analyzing the pipe/pipe whip restraint system for the effects of pipe rupture. This procedure defines two acceptable approaches.

(1) **Dynamic Time-History Analysis With Simplified Model**—A dynamic time history analysis of a portion of a piping system may be performed in lieu of a complete system analysis when it can be shown to be conservative by test data or by comparison with a more complete system analysis. For example, in those cases where pipe stresses in the containment penetration region need not be calculated, it is acceptable to model only a portion of the piping system as a simple cantilever with a fixed or pinned end or as a beam with both ends fixed or with one end pinned and one end fixed.

When a circumferential break is postulated, the pipe system is modeled as a simple cantilever, the thrust load is applied opposite the fixed (or pinned) end and the pipe whip restraint acts between the fixed (or pinned) end and the thrust load. It is then assumed that all deflection of the pipe is in one plane. As the pipe moves a resisting bending moment in the pipe is created and later a restraining force at the pipe whip restraint. Pipe movement stops when the resisting moments about the fixed (or pinned) end exceed the applied thrust moment.

When a longitudinal break is postulated, the pipe system has both ends supported. To analyze this case, two simplifications are made to allow the use of the cantilever model described above. First, an equivalent point mass is assumed to exist at D (Figure 3L-2) instead of pipe length DE. The inertia characteristics of this mass, as it rotates about point B, are calculated to be identical to those of pipe length DE, as it rotates about point E. Second, an equivalent resisting force is calculated (from the bending moment-angular deflection relationships for end DE) for any deflection for the case of a built-in end. This equivalent force is subtracted from the applied thrust force when calculating the net energy.

See Figures 3L-1 and 3L-2 for the models described above.

(2) **Dynamic Time-History Analysis with Detailed Piping Model**—In many cases it is necessary to calculate stresses in the ruptured pipe at locations remote from the pipe whip restraint location. For example, the pipe in the containment penetration area must meet the limits of SRP 3.6.2. In these cases it is required that the ruptured piping, the pipe supports, and the pipe whip restraints be modeled in sufficient detail to reflect its dynamic characteristics. A time-history analysis using the fluid forcing functions at the point of rupture and the fluid forcing functions of each pipe segment is performed to determine deflections, strains, loads to structure and equipment and pipe stresses.

3L.4.2 Procedure For Dynamic Time-History Analysis With Simplified Model

3L.4.2.1 Modeling of Piping System:

For many piping systems, all required information on their response to a postulated pipe rupture can be determined by modeling a portion of the piping system as a cantilever with either a fixed or pinned end. The fixed end model, as shown in Figure 3L-1, is used for piping systems where the stiffness of the piping segment located between A and B is such that the slope of the pipe length, BD, at B, will be approximately zero. The pinned end model, as shown in Figure 3L-1, is used for piping systems where the slope of the pipe length, BD, at B, is much greater than zero. The pinned end model is also used whenever it is not clear that the pipe end is fixed.

A simplified cantilever model may also be used for a postulated longitudinal break in a pipe supported at both ends, as shown in Figure 3L-2. The pipe can have both ends fixed or have pinned end at B and a fixed end at E, as shown in Figure 3L-2. Subsection 3L.4.1(1) discusses the simplification techniques used to allow the use of a cantilever model. A fixed end is used when rotational stiffness of the piping at that location is such that the slope of the pipe at that end is approximately zero. A pinned end is used when the pipe slope at that end is much greater than zero. If it is not clear whether an end is fixed or pinned, the end condition giving more conservative results should be assumed.

The pipe whip restraint is modeled as two components acting in series; the restraint itself and the structure to which the restraint is attached. The restraint and piping behave as determined by an experimentally or analytically determined force-deflection relationship. The structure deflects as a simple linear spring of representative spring constant.

The model must account for the maximum clearance between the restraint and the piping. The clearance is equal to the maximum distance from the pipe during normal operation to the position of the pipe when the pipe whip restraint starts picking up the rupture load. This simplified model is not used if the piping has snubbers or restraints strong enough to affect the pipe movement following a postulated rupture.

3L.4.2.2 Dynamic Analysis of Simplified Piping Model

When the thrust force (as defined in Subsection 3.6.2.2.1) is applied at the end of the pipe, rotational acceleration will occur about the fixed (or pinned) end. As the pipe moves, the net rotational acceleration will be reduced by the resisting bending moment at the fixed end and by the application of the restraining force at the pipe whip restraint. The kinetic energy will be absorbed by the deflection of the restraint and the bending of the pipe. Movement will continue until equilibrium is reached. The primary acceptance criteria is the pipe whip restraint deflection or strain must not exceed the design strain limit of 50% of the restraint material ultimate uniform strain capacity.

The analysis may be performed by a general purpose computer program with capability for nonlinear time-history analysis such as ANSYS, or by a special purpose computer program especially written for pipe rupture analysis such as the GE computer program, “Pipe Dynamic Analysis” (PDA).

3L.4.3 Procedure For Dynamic Time-History Analysis Using Detailed Piping Model

3L.4.3.1 Modeling of Piping System

In general, the rules for modeling the ruptured piping system are the same as the modeling rules followed when performing seismic/dynamic analysis of Seismic Category I piping. These rules are outlined in Subsection 3.7.3.3. The piping, pipe supports and pipe whip restraints are modeled in sufficient detail to reflect their dynamic characteristics. Inertia and stiffness effects of the system and gaps between piping and the restraints must be included.

If the snubbers or other seismic restraints are included in the piping model they should be modeled with the same stiffness used in the seismic analysis of the pipe. However, credit for seismic restraints cannot be taken if the applied load exceeds the Level D rating.

The pipe whip restraints are modeled the same as for the simplified model described in Subsection 3L.4.2.1. For piping designed with the GE U-Bar pipe whip restraints, the selected size and dimensions, and the resulting force-deflection and elastic/plastic stiffness is first determined according to the procedure previously defined in Section 3L.3.

3L.4.3.2 Dynamic Analysis using Detail Piping Model

The pipe break nonlinear time-history analysis can be performed by the ANSYS, or other NRC approved non-linear computer programs. The force time histories acting at the break location and in each of the segments of the ruptured pipe are determined according to the criteria defined in ANS 58.2. The time step used in the analysis must be sufficiently short to obtain convergence of the solution. (GE has shown that for a rupture of the main steam pipe a time step of .001 seconds is adequate for convergence.) The analysis must not stop until the peak of the dynamic load and the pipe response are over.

The primary acceptance criteria are: (1) The piping stresses between the primary containment isolation valves are within the allowable limits specified in Subsection 3.6.2.1.4.2, (2) the pipe whip restraint loads and displacements due to the postulated break are within the design limits, and (3) specified allowable loads on safety-related valves or equipment to which the ruptured piping is attached are not exceeded.

3L.5 Jet Impingement on Essential Piping

Postulated pipe ruptures result in a jet of fluid emanating from the rupture point. Safety-related systems and components require protection if they are not designed to withstand the results of the impingement of this jet. Subsection 3.6.2.3.1 provides the criteria and procedure for: (1) defining the jet shape and direction, (2) defining the jet impingement load, temperature and impingement location and (3) analysis to determine effects of jet impingement on safety-related equipment.

The paragraphs below provide some additional criteria and procedure for the analysis required to determine the effects of jet impingement on piping.

(1) Jet impingement is a faulted load and the primary stresses it produces in the piping must be combined with the stresses caused by SSE to meet the faulted stress limits for the designated ASME class of piping.

(2) If a pipe is subjected to more than one jet impingement load, each jet impingement load is applied independently to the piping system and the load which supplies the largest bending moment at each node is used for evaluation.

(3) A jet impingement load may be characterized as a two part load applied to the piping system—a dynamic portion when the applied force varies with time and a static portion which is considered steady state.

For the dynamic load portion, when static analysis methods are used, apply a dynamic load factor of 2. Snubbers are assumed to be activated. Stresses produced by the dynamic load portion are combined by SRSS with primary stresses produced by SSE.

For the static load portion, snubbers are not activated and stresses are combined with SSE stresses by absolute sum

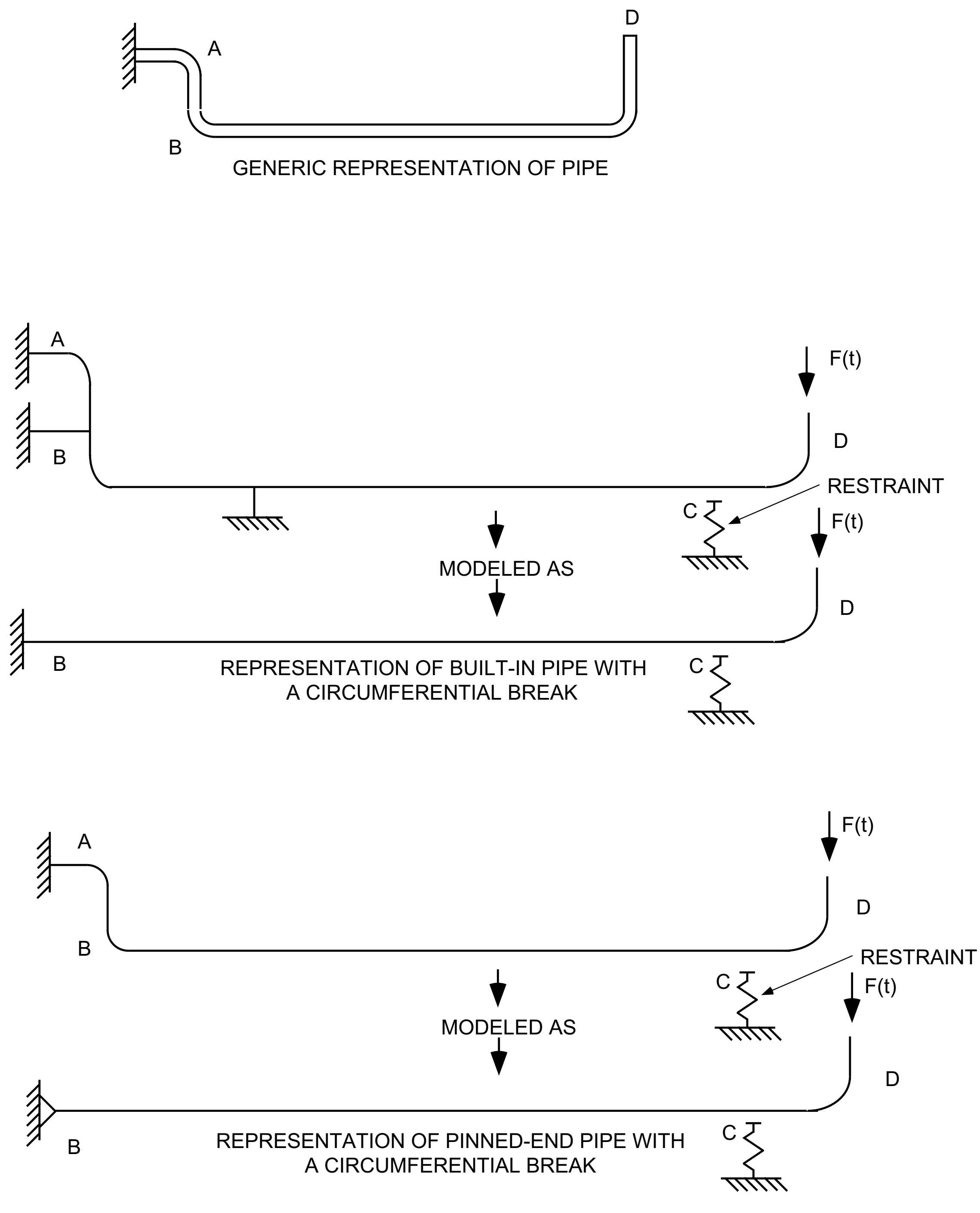

Figure 3L-1 Simplified Piping Models

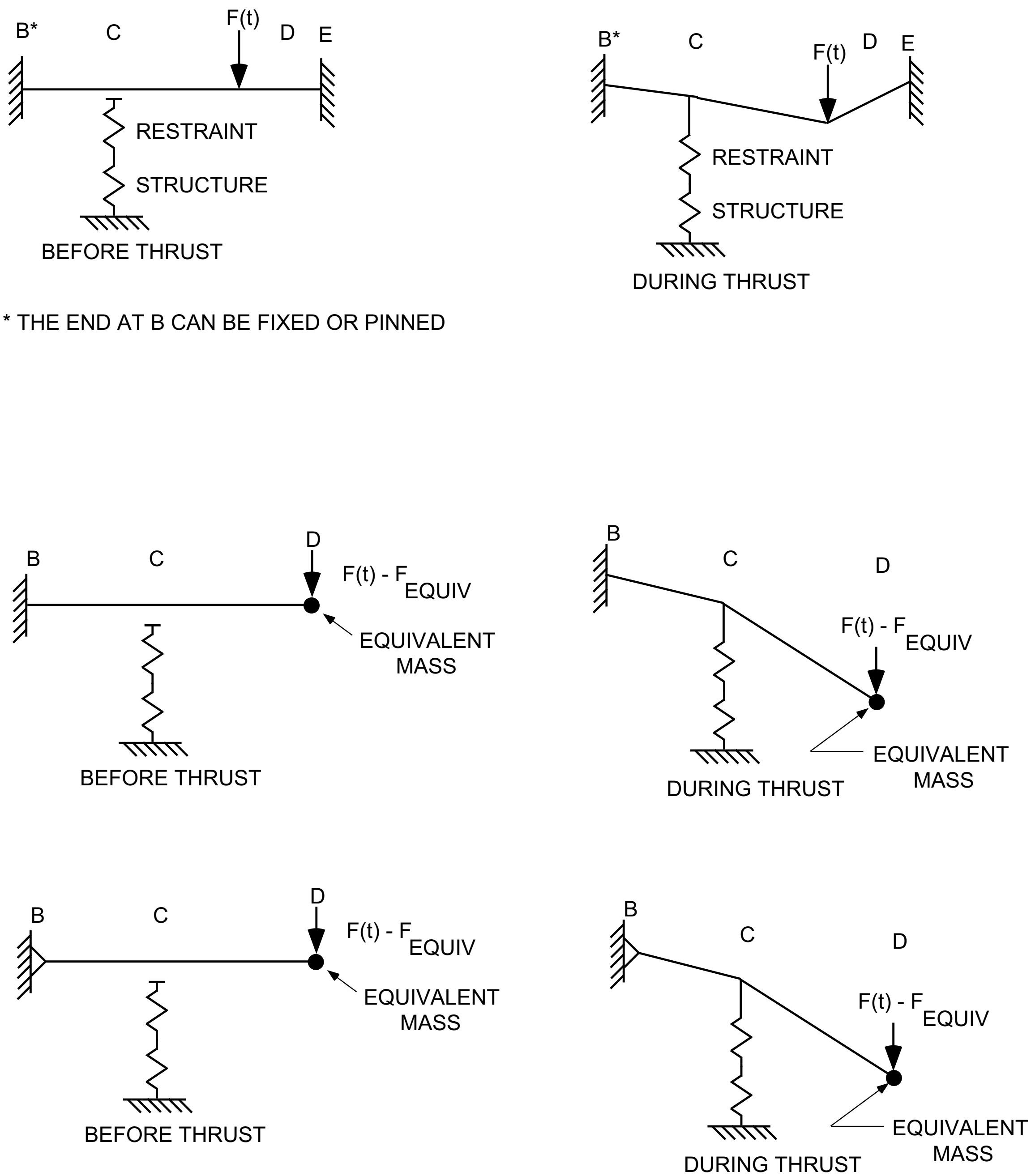

Figure 3L-2 Representation of Pipe With Both Ends Supported With a Longitudinal Break

Figure 3L-3 Not Used

3M Resolution Of Intersystem Loss Of Coolant Accident For ABWR

3M.1 Introduction

An intersystem loss of coolant accident (ISLOCA) is postulated to occur when a series of failures or inadvertent actions occur that allow the high pressure from one system to be applied to the low design pressure of another system, which could potentially rupture the pipe and release coolant from the reactor system pressure boundary. This may also occur within the high and low pressure portions of a single system. Future ALWR designs like the ABWR are expected to reduce the possibility of a LOCA outside the containment by designing to the extent practicable all piping systems, major system components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) to an ultimate rupture strength (URS) at least equal to the full RCPB pressure. The general URS criteria was recommended by the Reference 1 and the NRC Staff recommended specific URS design characteristics by Reference 2.

3M.2 ABWR Regulatory Requirements

In SECY-90-016 and SECY-93-087 (References 3 and 4), the NRC staff resolved the ISLOCA issue for advanced light water reactor plants by requiring that low-pressure piping systems that interface with the reactor coolant pressure boundary be designed to withstand reactor pressure to the extent practicable. However, the staff believes that for those systems that have not been designed to withstand full reactor pressure, evolutionary ALWRs should provide (1) the capability for leak testing the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are de-energized and (3) high-pressure alarms to warn main control room operators when rising reactor pressure approaches the design pressure of attached low-pressure systems or when both isolation valves are not closed. The staff noted that for some low-pressure systems attached to the RCPB, it may not be practical or necessary to provide a higher system ultimate pressure capability for the entire low-pressure connected system. The staff will evaluate such exceptions on a case-by-case basis during specific design certification reviews.

GE provided a proposed implementation of the issue resolution for the ABWR in Reference 5 and again in Reference 6. The staff in the Civil Engineering and Geosciences Branch of the Division of Engineering completed its evaluation of the Reference 5 proposal. Specifically, as reported by Reference 2 and summarized below, the staff has evaluated the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure should the low-pressure system be subjected to full RCPB pressure.

Reference 2 found that for the ABWR the design pressure for the low-pressure piping systems that interface with the RCPB should be equal to 0.4 times the normal operating RCPB pressure of 7.07 MPaG, the minimum wall thickness of low-pressure piping should be no less than that of a standard weight pipe, and that Class 300 valves are adequate . The design is to be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subarticle NC/ND-

3600. Furthermore, the staff will continue to require periodic surveillance and leak rate testing of the pressure isolation valves per Technical Specification requirements as a part of the ISI program.

3M.3 Boundary Limits of URS

Guidance given by Reference 3 provides provision for applying practical considerations for the extent to which systems are upgraded to the URS design pressure. The following items form the basis of what constitutes practicality and set forth the test of practicality used to establish the boundary limits of URS for the ABWR:

(1) It is impractical to consider a disruptive open flow path from reactor pressure to a low pressure sink. A key assumption to understanding the establishment of the boundary limits from this practicality basis is that only static pressure conditions are considered. Static conditions are assumed when the valve adjacent to a low pressure sink remains closed. Thus, the dynamic pressurization effects accompanied by violent high flow transients and temperature escalations are precluded that would occur if the full RCPB pressure was connected directly to the low pressure sink. As a consequence, the furthest downstream valve in such a path is assumed closed so that essentially all of the static reactor pressure is contained by the URS upgraded region.

(2) It is impractical to design or construct large tank structures to the URS design pressure that are vented to atmosphere and have a low design pressure. Tanks included in this category are:

(a) Condensate storage tank,

(b) SLC main tank,

(c) LCW receiving tank,

(d) HCW receiving tank,

(e) FPC skimmer surge tank, and

(f) FPC spent fuel storage pool and cask pit.

(g) Condensate hotwell

These are termed low pressure sinks for the purposes of this report. See Table 3M-1 for approximate sizes of these tanks as an indication of the impracticality of increasing the design pressure. The size of these tanks would result in an unnecessary dollar cost burden to increase their design pressure to the URS value. The small tanks in Table 3M-1 are greater than 3 meters in height and diameter. (For perspective, remember the “3 meter board” at the swimming pool is the high dive.) The large condensate storage tank, if constructed with its height equal to the diameter, is approximately as tall as a four story building. The FPC System's tank, pool, and pit (Table 3M-1) have no top cover and are open to the large refueling floor (bay), so that their pressure can not be increased above the static head for which they are designed.

(3) It is impractical to design piping systems that are connected to low pressure sink features to the URS design pressure when the piping is always locked open to a low pressure sink by locked open valves. These piping sections are extensions of the low pressure sink and need no greater design pressure than the low pressure sink to which they are connected.

In summary, the following low pressure sinks are protected by an adjacent closed valve and are impractical to design to the URS design pressure.

(1) **Suppression Pool —** Provides a normal low pressure sink, approximately 4.9 kPaG above atmospheric for its interfacing systems and the first closed valve is at least 2.83 MPaG rated. The suppression pool is designed to Seismic Category I.

(2) **Condensate Storage Tank—** Vented to atmosphere and its locked open valves insure it is a low pressure sink for its interfacing systems. The first closed valve of each interfacing system with URS upgrade is at least 2.82 MPaG rating.

(3) **SLC main tank**—Vented to atmosphere with the first closed valve at least 2.82 MPaG rating. The SLC main tank is designed to Seismic Category I.

(4) **LCW Receiving Tank -** Vented to atmosphere, and the first closed valve is at least 2.82 MPaG and one of the dual tank's inlet valves is locked open.

(5) **HCW Receiving Tank—** Vented to atmosphere, and the first closed valve is at least 2.82 MPaG and one of the dual tank's inlet valves is locked open.

(6) **FPC Skimmer Surge Tank**— The Fuel Pool Cooling Cleanup System's skimmer surge tank is open to the near atmospheric pressure of the refueling floor. The first closed valve is at least 2.82 MPaG rated. The FPC skimmer surge tank is designed to Seismic Category I.

(7) **FPC Spent Fuel Storage Pool and Cask Pit—** The Fuel Pool Cooling Cleanup System's spent fuel storage pool and cask pit is open to the near atmospheric pressure of the refueling floor. The first closed valve is at least 2.82 MPaG rated. The FPC spent fuel storage pool and cask pit is designed to Seismic Category I.

(8) **Condensate Hotwell**— During reactor high pressure operation, the hotwell operates at a vacuum pressure.

3M.4 Evaluation Procedure

The pressure of each system piping boundary on all of the ABWR P&ID's was reviewed to identify where changes were needed to provide URS protection. Where low pressure piping interfaces with higher pressure piping connected to piping with reactor coolant at reactor pressure, design pressure values were increased to2.82 MPaG. The low pressure piping boundaries were upgraded to URS pressures and extend to the last closed valve connected to piping interfacing a low pressure sink, such as the suppression pool, condensate storage tank or other open configuration (identified pool or tank). Some upgraded boundaries were located at normally open valves, but the upgrading would be needed if the non-normal closed condition occurred. Each interfacing system's piping was reviewed for upgrading. For some systems, with low pressure piping and normally open valves, the valves were changed to lock open valves to insure an open piping pathway from the last URS boundary to the tank or low pressure sink.

Typical systems for this upgrade include the:

(1) Radwaste LCW and HCW receiving tank piping,

(2) Fuel Pool Cooling System's RHR interface piping connected to the skimmer surge tanks,

(3) Condensate Storage System's tank locked open supply valves,

(4) Makeup Water Condensate and Makeup Water Purified Systems with locked open valves and pump bypass piping to the Condensate Storage Tank.

All test, vent and drain piping was upgraded where it interfaces with the piping upgraded to URS pressure. Similarly, all instrument and relief valve connecting piping was upgraded.

3M.5 Systems Evaluated

The following fourteen systems, interfacing directly or indirectly with the RCPB, were evaluated.

	Tier 2 Figure No.
1. Residual Heat Removal (RHR) System	5.4-10
2. High Pressure Core Flooder (HPCF) System	6.3-7
3. Reactor Core Isolation Cooling (RCIC) System	5.4-8
4. Control Rod Drive (CRD) System	4.6-8
5. Standby Liquid Control (SLC) System	9.3-1
6. Reactor Water Cleanup (CUW) System	5.4-12
7. Fuel Pool Cooling Cleanup (FPC) System	9.1-1
8. Nuclear Boiler (NB) System	5.1-3
9. Reactor Recirculation (RRS) System	5.4-4
10. Makeup Water (Condensate) (MUWC) System,	9.2-4
11. Makeup Water (Purified) (MUWP) System.	9.2-5
12. Radwaste System	11.2-2
(LCW Receiving Tank, HCW Receiving Tank).	
13. Condensate and Feedwater (CFS) System	10.4-6
14. Sampling (SAM) System	—

Attachment 3MA contains a system-by-system evaluation of potential reactor pressure application to piping and components, discussing the URS boundary and listing the upgraded components. For some systems, certain regions of piping and components not upgraded are also listed.

3M.6 Piping Design Pressure for URS Compliance

Guidelines for URS compliance were established by Reference 2, which concluded that for the ABWR:

(1) The design pressure for the low-pressure piping systems that interface with the RCPB pressure boundary should be equal to 0.4 times the normal operating RCPB pressure of 7.07 MPaG, and

(2) The minimum wall thickness of the low-pressure piping should be no less than that of a standard weight pipe.

3M.7 Applicability of URS Non-piping Components

Reference 2 also provided the NRC Staff's position that:

(1) The remaining components in the low-pressure systems should also be designed to a design pressure of 0.4 times the normal operating reactor pressure. This is accomplished in Tier 2 by the revised boundary symbols of the P&IDs to the 2.82 MPaG design pressure, which includes all the piping and components associated with the boundary symbols. A stated parameter (e.g., design pressure) of a boundary symbol on the P&ID applies to all the piping and components on the P&ID that extend away from the boundary symbol, including along any branch line, until another boundary symbol occurs on the P&ID. The components include flanges, and pump seals, etc. as shown on the P&ID.

ABWR heat exchangers are not affected by ISLOCA upgrades to the URS design pressure. The following heat exchangers are in systems evaluated for ISLOCA, but the heat exchangers were not upgraded.

(a) The Reactor Water Cleanup System (CUW) heat exchangers are designed for the high reactor pressure already above the URS.

(b) The Residual Heat Removal (RHR) heat exchanger are designed for 3.43 MPaG on the tube side which exceeds the 2.82 MPaG URS design pressure. The heat exchanger’s tube side carries the reactor water. The shell side which carries the Reactor Building Cooling Water (RCW) system’s cooling water which has a design pressure of 1.37 MPaG, which is the same as the RCW design pressure. Since the heat exchanger’s tube side is designed well above the URS design pressure, an over pressurization failure was not assumed that would apply reactor pressure to the shell side.

(c) The Fuel Pool Cooling and Cleanup (FPC) System heat exchangers are isolated from a potential exposure to reactor pressure so that no upgrade was applicable.

(2) A Class 300 valve is adequate for ensuring the pressure of the low-pressure piping system under full reactor pressure. The rated working pressure for Class 300 valves varies widely depending on material and temperature (ASME/ANSI B16.34). However, as a lower limit bounding condition, within the material group that includes the stainless steels, the lowest working pressure is 2.86 MPaG at 204 °C , which exceeds the URS of 2.82 MPaG. For lower temperatures the working pressure increases. The material group that includes the carbon steels has working pressures above this value. More typical working pressure values at 93°C range between 4.12 MPaG to 4.81 MPaG.

3M.8 Results

The results of this work are shown by the markups of the enclosed P&IDs, which are Tier 2 figures. The affected sheets are listed below

System	Tier 2 Figure No.	Affected Sheet Nos.
1. Residual Heat Removal (RHR) System	5.4-10	1, 2, 3, 4, 6, 7
2. High Pressure Core Flooder (HPCF) System	6.3-7	1, 2
3. Reactor Core Isolation Cooling (RCIC) System	5.4-8	1, 3
4. Control Rod Drive (CRD) System	4.6-8	1, 3
5. Standby Liquid Control (SLC) System	9.3-1	1
6. Reactor Water Cleanup (CUW) System	5.4-12	1, 3
7. Fuel Pool Cooling and Cleanup (FPC) System	9.1-1	1, 2
8. Nuclear Boiler (NB) System	5.1-3	1, 5
9. Reactor Recirculation (RRS) System	5.4-4	1
10. Makeup Water (Condensate) (MUWC) System	9.2-4	1
11. Makeup Water (Purified) (MUWP) System	9.2-5	1, 2, 3
12. Radwaste System (LCW Receiving Tank, HCW Receiving Tank)	11.2-2	1, 3, 7
13. Condensate and Feedwater (CSF) System	10.4-6	
14. Sampling (SAM) System		

Also, see Attachment A for more detail.

The design pressure of the following two tanks was upgraded as a result of the evaluations performed in Attachment 3MA.

SLC test tank

RCIC turbine barometric condenser tank

3M.9 Valve Misalignment Due To Operator Error

An important result to observe is that because of the widespread application of the URS boundary for the ABWR design as compared to previously constructed BWRs, misalignment of valves due to operator error is a contributor to ISLOCA that has no known consequence. The ABWR design with the ISLOCA URS applied for the boundary described by this appendix and its attachement, has extended the increased design pressure (URS) over the full extent of regions that could potentially experience reactor pressure, so that operator misaligned valves will not expose piping to reactor pressure not designed to the URS pressure.

The ISLOCA issue that has been dealt with for existing BWRs, where valve misalignment due to operator error was a significant contributor to ISLOCA considerations, had to use the design pressures used for plant construction that were accepted before ISLOCA issues were considered. As a result, operator error of valve misalignment could possibly result in situations where high pressure might occur in piping regions design pressures below the current accepted URS design pressure (2.82 MPaG).

3M.10 Additional Operational Considerations

The periodic surveillance testing of the ECCS injection valves that interface with the reactor coolant system might lead to ISLOCA conditions if their associated testable check valve was stuck open. To avoid this occurrence, the RHR, HPCF, and RCIC motor operated injection valves will only be tested during low pressure shutdown operation. This practice follows from the guidance given by Reference 3, page 8, paragraph 7.

Although the following is not a new design feature, the RHR shutdown cooling suction line containment isolation valves are also only tested during shutdown operation. These valves are interlocked against opening for reactor pressure greater than the shutdown cooling setpoint approximately 0.93 MPaG.

3M.11 Summary

Based on the NRC staff's new guidance cited in References 1 through 4, the ABWR is in full compliance. For ISLOCA considerations, a design pressure of 2.82 MPaG and pipe having a minimum wall thickness equal to standard grade has been provided as an adequate margin with respect to the full reactor operating pressure of 7.07 MPaG by applying the guidance recommended by Reference 2. This design pressure was applied to the low pressure piping at their boundary symbols on the P&IDs, and therefore, impose the requirement on the associated piping, valves, pumps, tanks, instrumentation and all other equipment shown between boundary symbols. Notes were added to each URS upgraded P&ID requiring pipe to have a minimum wall thickness equal to standard grade and requiring valves with a design pressure of 2.82 MPaG or greater to be a minimum of Class 300. Upgrading revisions were made to 13 systems.

3M.12 References

(1) Dino Scaletti, NRC, to Patrick Marriott, “GE, Identification of New Issues for the General Electric Company Advanced Boiling Water Reactor Review”, September 6, 1991

(2) Chester Poslusny, NRC, to Patrick Marriott, “GE, Preliminary Evaluation of the Resolution of the Intersystem Loss-of-Coolant Accident (ISLOCA) Issue for the Advanced Boiling Water Reactor (ABWR) - Design Pressure for Low-Pressure Systems”, December 2, 1992, Docket No. 52-001

(3) James M. Taylor, NRC, to The Commissioners, SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements”, Jan. 12, 1990

(4) James M. Taylor, NRC, to The Commissioners, SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs”, April 2, l993

(5) Jack Fox, GE, to Chet Poslusny, NRC, “Proposed Resolution of ISLOCA Issue for ABWR”, October 8, 1992

(6) Jack Fox, GE, to Chet Poslusny, NRC, “Resolution of Intersystem Loss of Coolant Accident for ABWR”, April 30, 1993.

Table 3M-1 Low Pressure Sink Component Sizes

Tank Name	Volume m^3	Diameter m	Height m	Length m	Width m	Design Pressure MPaG	Note
Condensate storage tank	2110	13.9	13.9			1.37	(1)
SLC main tank	32	3.44	3.44			SWH	(1)
LCW receiving tank	430	8.18	8.18			0.98	(1)
HCW receiving tank	45	3.85	3.85			0.98	(1)
FPC skimmer surge tank	30	2.3	7.2			SWH	
FPC spent fuel storage pool	2960		11.8	17.9	14.0	SWH	
FPC cask pit	121		11.8	3.2	3.2	SWH	
Condensate hotwell	7800		20	30	13		

Notes:

(1) Diameter and height calculated from volume based on diameter = height.

SWH = Static water head

3MA System Evaluation For ISLOCA

3MA.1 General Comments About the Appendix

This Attachment discusses each of the systems evaluated in detail, presented in the order listed in the Appendix, and following a repetitive outline format.

The first section, "Upgrade Description," describes the changes made to the system and the reasons for placement of the URS boundary.

The second section, "Downstream Interfaces," discusses the systems that interface with the subject system, that could potentially be pressurized by reactor pressure passed through (downstream) the subject system. Each downstream system is dispositioned as being either not applicable for URS upgrading or applicable and the topic of another Attachment 3MA section.

The third section, "Upgraded Components," provides a detailed listing of the components upgraded to the URS design pressure. Also, to indicate some components were not inadvertently overlooked, some components are shown as "No change." The listings are grouped in sections that describe a particular pressure travel path. This grouping may include more than the system of the subject section to detail the path to the tank or sink in which the pressure is dissipated after crossing the last closed valve at the URS boundary.

3MA.2 Residual Heat Removal System

3MA.2.1 Upgrade Description

The RHR System pump suction piping was low pressure and has been upgraded to the URS design pressure. The RHR has two suction sources, one from the suppression pool and the other from the RPV as used for shutdown cooling. The suction piping also includes the keep-fill pump and its piping.

The URS boundary was terminated at the last valve before the suppression pool, which is valve E11-F001. The suppression pool is a large structure, designed to 0.310 MPaG and impractical to upgrade to the URS design pressure.

The other suction branch to the RPV is not a URS boundary because it interfaces to the high pressure RPV. The only portions of the RHR System that are not upgraded to the URS design pressure is unobstructed piping to the suppression pool.

3MA.2.2 Downstream Interfaces

Other systems are listed below that interface with RHR and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

- Makeup Water (Condensate) System upstream of the injection valve for the purpose of providing a filling and flushing water source. Another interface with MUWC is between the pair of valves to the FPC System. The MUWC System is discussed in Section 3MA.11, where it is explained how certain MUWC upgrades were made that provide an open path to the CST. The MUWC line cannot be pressurized because of the open communication to the CST, and the CST is vented to atmosphere. There is no source to pressurize the MUWC line because of closed valves in the RHR System's URS region.
- High Conductivity Waste (Radwaste) for drainage located up stream of the pump suction. HCW upgrades are discussed in the Radwaste System, Section 3MA.13.
- Low Conductivity Waste, (Radwaste) at the end of a branch off of the loop B mainline down stream of the RHR heat exchanger. The LCW upgrades are discussed in the Radwaste System, Section 3MA.13.
- Sampling System at the outlet of the RHR heat exchanger. The Sampling System's design pressure exceeds the URS design pressure without upgrade.
- Fuel Pool Cooling and Cleanup System on an RHR System discharge branch. FPC System upgrades are discussed in Section 3MA.8.
- Flammability Control System branches off the main discharge line downstream of the branch that returns to the suppression pool. The FCS design pressure exceeds the URS design pressure without upgrade.
- The Fire Protection System and the fire truck connections provide water for the Alternating Current (AC) Independent Water Addition piping of RHR loops B and C upstream of the RPV injection lines, wetwell and drywell spray lines, and spent fuel pool lines. The Fire Protection System piping is designed for 1.37 MPaG and each line is protected from over pressure by two locked closed block and bleed valves, RHR-F101B/C and RHR-F102B/C, and drain pipes between these valves which are vented to the HCW sump in the Reactor Building. This design very effectively prevents reactor pressure from reaching the Fire Protection System. No upgrade to URS is practical or appropriate for the extensive piping of the Fire Protection System since the system function is not related to ISLOCA nor is its interconnection a normal plant operational pathway.

3MA.2.3 Upgraded Components — RHR System

A detailed listing of the components upgraded for the RHR System follows, including identification of those interfacing system components not requiring upgrade.

RESIDUAL HEAT REMOVAL SYSTEM, Tier 2 Figure 5.4-10, Sheets 1 through 7.

RHR Subsystem A suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	RHR Pump C001A	3.43 MPaG, 182°C, 3B, As	No change
	450A-RHR-002 Pipe	2.82 MPaG, 182°C, 3B, As	Was 1.37 MPaG
	20A-RHR-701 Pipe	2.82 MPaG, 182°C, 3B, As	Was 1.37 MPaG
	20A-RHR-F701A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX002A Press.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-D002A Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-700 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F700A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PI001A Press.I	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-018 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F026A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-F001A MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	450A-RHR-001 Pipe	0.310 MPaG, 104°C,3B,As	No change
	450A-RHR-D001A Suct.Str.	0.310 MPaG, 104°C,3B,As	No change

RHR Subsystem A suction piping from the reactor pressure vessel.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	350A-RHR-011 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	350A-RHR-F012A MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-032 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F042A Rel.Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-707 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F712A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PT009A Press.T	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	350A-RHR-011 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-504 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-F508A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-030 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 100A-RHR-031 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 100A-RHR-F041A Check V	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 100A-RHR-F040A Valve.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

* To LCW funnel drain to LCW Sump.
** To MUW (Condensate) Stem interface.

RHR Subsystem A discharge fill pump suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	40A-RHR-C002A Pump	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-015 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-F022A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-D008A Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-708 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F713A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX010APress.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-017 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F025A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-D009A RO	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem A discharge from relief valves and test line valve directly to the suppression pool without restriction.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	250A-RHR-008 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-025 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-014 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-037 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-033 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-021 Pipe	0.310 MPaG, 104°C,3B,As	No change
Sheet 2	250A-RHR-008 Pipe	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

RHR Subsystem A flushing line interface at branch discharging to feedwater.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	100A-MUWC-134 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 2	100A-RHR -F032A Valve	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -026 Pipe	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F033A Check V.	3.43 MPaG, 182°C,3B,As	No change

RHR Subsystem A flushing line interface at suction shutdown branch from RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	100A-MUWC-133 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 3	100A-RHR -F040A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -031 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -F041A Check V.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem B suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	RHR Pump C001B	3.43 MPaG, 182°C,3B,As	No Change
	450A-RHR-102 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-731 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F701B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX002B Press.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-D002B Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-730 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F700B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PI001B Press.I	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-124 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F026B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-F001B MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	450A-RHR-101 Pipe	0.310 MPaG, 104°C,3B,As	No change
	450A-RHR-D001B Suct.Str.	0.310 MPaG, 104°C,3B,As	No change

RHR Subsystem B suction piping from the reactor pressure vessel.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	350A-RHR-111 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	350A-RHR-F012B MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-139 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F042B Rel.Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-737 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F712B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PT009B Press.T	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	350A-RHR-111 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-534 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-F508B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-137 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 300A-RHR-114 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 300A-RHR-F016B Valve LC	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-138 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-F041B Check Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-F040B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

* To LCW funnel drain to LCW Sump. ** To FPC System interface.
*** To MUW (Condensate) System interface.

RHR Subsystem B discharge fill pump suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	40A-RHR-C002B Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-121 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-F022B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-D008B Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-738 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F713B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX010BPress.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-123 Pipe	2.82 MPaG, 182°C182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F025B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-D009B RO	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem B flushing line interface at branch discharging to RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	100A-MUWC-137 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 5	100A-RHR -F032B Valve	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -132 Pipe	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F033B Check V.	3.43 MPaG, 182°C,3B,As	No change

RHR Subsystem B flushing line interface at suction of shutdown branch from RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	100A-MUWC-136 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 2	100A-RHR -F040B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	100A-RHR -138 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -F041B Check V.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem B discharge from relief valves and test line valve directly to the suppression pool without restriction.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	250A-RHR-109 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-131 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-120 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-145 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-140 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-127 Pipe	0.310 MPaG, 104°C,3B,As	No change
Sheet 2	250A-RHR-109 Pipe	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

RHR Subsystem B interface with Radwaste System.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	150A-RHR-129 Pipe	3.43 MPaG, 182°C,3B,As	No change
	150A-LCW-F006 Valve	2.82 MPaG, 66°C,4D,B	Was 0.981 MPaG
	150A-LCW-CS Pipe	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS Pipe	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS Valve LO	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS AO Valve	0.981 MPaG, 66°C,4D,B	No change
	* LCW Collector Tank A	0 MPaG, 66°C,4D,B	No change
	200A-LCW-CS Valve LO	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS AO Valve	0.981 MPaG, 66°C,4D,B	No change
	*LCW Collector Tank B	0 MPaG, 66°C,4D,B	No change

* Each LCW collector tank is served by the HVAC tank vent system exhausting tank air through filter to RW Stack.

RHR Subsystem B outdoor fire truck connection in RHR pump discharge pipe to RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 7	100A-RHR -F103B Valve	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR -F104B Check V.	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR - Pipe	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR - Pipe	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR -F100B Check V.	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR -F101B Key Lock V.	3.43 MPaG, 182°C,3B,As	New Item
	100A-RHR - Pipe	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR - Pipe	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR -F790B Globe V.	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR -PI-099B Press I	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR - Pipe	3.43 MPaG, 182°C,3B,As	New Item
	* 20A-RHR -F592B Globe V. LO	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR - Pipe	3.43 MPaG, 182°C,3B,As	New Item
	** 20A-RHR -F591B Globe V. NC	3.43 MPaG, 182°C,3B,As	New Item
	100A-RHR -F102B Key Lock V.	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR -FE-100B Flow El.	3.43 MPaG, 182°C,3B,As	New Item
	*** 300A-RHR -105 Pipe	3.43 MPaG, 182°C,3B,As	No change

* Funnel drain to LCW sump in Reactor Building.
** Test valve.
*** Injection pipe to RPV at outboard isolation valve MO F-005B.

RHR Subsystem C suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 6	RHR Pump C001C	3.43 MPaG, 182°C,3B,As	No change
	450A-RHR-202 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-761 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F701C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX002C Press.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-D002C Temp.Str	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-760 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F700C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PI001C Press.I	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-225 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F026C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-F001C MO Vlv	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	450A-RHR-201 Pipe	0.310 MPaG, 104°C,3B,As	No change
	450A-RHR-D001C Suct. Str.	0.310 MPaG, 104°C,3B,As	No change

RHR Subsystem C suction piping from the reactor pressure vessel.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 6	350A-RHR-212 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	350A-RHR-F012C MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-240 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F042C Rel.Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-767 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F712C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PT009C Press.T	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	350A-RHR-212 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-564 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-F508C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-238 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 300A-RHR-215 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 300A-RHR-F016C Valve LC	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-239 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-F041C Check Vlv.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-F040C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

* To LCW funnel drain to LCW Sump. ** To FPC System interface.
*** To MUW (Condensate) System interface.

RHR Subsystem C discharge fill pump suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 6	40A-RHR-C002C Pump	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-222 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-F022C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-D008C Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-768 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F713C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX010C Press.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-224 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F025C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-D009C RO	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem C discharge from relief valves and test line valve direct to the suppression pool without restriction.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	250A-RHR-209 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-232 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-221 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-246 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-241 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-228 Pipe	0.310 MPaG, 104°C,3B,As	No change
Sheet 2	250A-RHR-209 Pipe	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

RHR Subsystem C flushing line interface at branch discharge to RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	100A-MUWC-140 Pipe	1.37 MPaG, 66°C,4D,B	No change
	100A-RHR -F032C Valve	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -233 Pipe	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F033C Check V.	3.43 MPaG, 182°C,3B,As	No change

RHR Subsystem C flushing line interface at suction of shutdown branch from RPV.

Reference	Components	Press./Temp./Design Seismic Class	Remarks
Sheet 2	100A-MUWC-140 Pipe	1.37 MPaG, 66°C,4D,B	No change
	100A-RHR -F040C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -239 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -F041C Check V.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem C outdoor fire truck connection in RHR pump discharge pipe to RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 7	100A-RHR -F103 Valve	2.82 MPaG, 66°C,7E,C	Was 1.57 MPaG
	100A-RHR -F104 Check V.	2.82 MPaG, 66°C,7E,C	Was 1.57 MPaG
	100A-RHR -249 Pipe	2.82 MPaG, 66°C,7E,C	Was 1.57 MPaG
	100A-RHR -247 Pipe	2.82 MPaG, 66°C,7E,C	Was 1.57 MPaG
	100A-RHR -F100 Check V.	2.82 MPaG, 66°C,7E,C	Was 1.57 MPaG
	100A-RHR -F101 Key Lock V.	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -248 Pipe	3.43 MPaG, 182°C,3B,As	No change
	20A-RHR -769 Pipe	3.43 MPaG, 182°C,3B,As	No change
	20A-RHR -F790 Globe V.	3.43 MPaG, 182°C,3B,As	No change
	20A-RHR -PI-099 Press I	3.43 MPaG, 182°C,3B,As	No change
	20A-RHR -570 Pipe	3.43 MPaG, 182°C,3B,As	No change
	* 20A-RHR -F592 Globe V.LO	3.43 MPaG, 182°C,3B,As	No change
	20A-RHR -571 Pipe	3.43 MPaG, 182°C,3B,As	No change
	** 20A-RHR -F591 Globe V.NC	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F102 Key Lock V.	3.43 MPaG, 182°C,3B,As	No change
	20A-RHR -FE-100 Flow El.	3.43 MPaG, 182°C,3B,As	No change
	*** 300A-RHR -205 Pipe	3.43 MPaG, 182°C,3B,As	No change

* Funnel drain to LCW sump in Reactor Building.
** Test valve.
*** Injection pipe to RPV at outboard isolation valve MO F-005C.

No other low Press. components of the RHR System were identified for upgrading to the higher design Press. as shown on the marked P & ID's. Interface with the LCW Reactor Building sump which is vented to atmosphere, is through open funnel drains with low Press. piping to the sump.

3MA.3 High Pressure Core Flooder System

3MA.3.1 Upgrade Description

The HPCF System pump suction piping was low pressure and has been upgraded to the URS design pressure. The HPCF has two suction sources, the primary source being the condensate storage tank (CST) and the suppression pool as an alternate.

The URS boundary was terminated at the last HPCF valve in the pipeline to the CST, E22-F001. Beyond this valve, the pipeline is open to the CST except for three locked open maintenance valves in parallel adjacent to the CST. The pipeline to the CST is a large pipe (50.8 cm) providing a common supply to the HPCF, RCIC, and SPCU System, Because of the open communication to the CST, and the CST is vented to atmosphere, this line cannot be pressurized. The CST is a large structure, impractical to upgrade to the URS design pressure.

The URS boundary was terminated at the last valve before the suppression pool, which is valve E22-F006 and is normally closed. The suppression pool is a large structure, impractical to upgrade to the URS design pressure. The only portions of the HPCF System that are not upgraded to the URS design pressure is unobstructed piping to the suppression pool.

3MA.3.2 Downstream Interfaces

Other systems are listed below that interface with HPCF and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability is ISLOCA is given.

- Makeup Water (Condensate) System upstream of the injection valve for the purpose of providing the piping keep-fill water source and a filling and flushing water source. The MUWC System is discussed in Section 3MA.11, where it is explained how certain MUWC changes were made that provide an open path to the CST. The MUWC line cannot be pressurized because of the open communication to the CST, and the CST is vented to atmosphere.

There is no source to pressurize the MUWC line because of closed valves in the HPCF System's URS region.

- High Conductivity Waste System for drainage is located downstream of CST suction valve. HCW is discussed in Section 3MA.13.

3MA.3.3 Upgraded Components — HPCF System

A detailed listing of the components upgraded for the HPCF System follows, with identification of interfacing system components not requiring upgrade.

HIGH PRESSURE CORE FLOODER SYSTEM, Tier 2 Figure 6.3-7, Sheets 1 and 2.

HPCF Subsystem B suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 2	400A-HPCF-006 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-701 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-F701B Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-PX004B Press. Pt.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-D001B Temp. Str.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-010 Pipe	2.82 MPaG, 104°C,3B,As	Was 1.37 MPaG
	20A-HPCF-700 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-F700B Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-PI001B Press.Ind.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-PT002B Press.Trn.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-PT003B Press.Trn.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	25A-HPCF-023 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	25A-HPCF-F020B Relief V.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-F007B Check V.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-025 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-F022B T.Valve&cap	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-F006B MO.Valve	2.82 MPaG, 104°C,3B,As	Was 1.37 MPaG
	400A-HPCF-009 Pipe	0.310 MPaG, 104°C,3B,As	No change
	400A-HPCF-D003B Suction Str.	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

HPCF Subsystem B suction piping from the Condensate Storage Tank.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 2	400A-HPCF-006 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-018 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-F012B Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-F011B Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-017 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-F002B Check V.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-F001B MO.Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-005 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	500A-HPCF-004 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	400A-HPCF-105 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	200A-HPCF-015 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	200A-HPCF-016 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	* 300A-HPCF-001 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	* 300A-HPCF-002 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	* 300A-HPCF-003 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change

* Connects to lock open valves at condensate storage tank vented to atmosphere.

HPCF Subsystem B test and minimum flow piping to the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 2	80A-HPCF-014 Pipe	0.310 MPaG, 104°C,3B,As	No change
	200A-HPCF-012 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-HPCF-024 Pipe	0.310 MPaG, 104°C,3B,As	No change
	250A-RHR- 109 Pipe	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

HPCF Subsystem B keep fill line interface.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	20A-MUWC-135 Pipe	1.37 MPaG, 66°C,4D,B	No change
	25A-HPCF-F013B Valve	10.79 MPaG, 100°C,3B,As	No change
	25A-HPCF-D006B R0	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-019 Pipe	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-020 Pipe	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-F016B Valve	10.79 MPaG, 100°C,3B,As	No change

HPCF Subsystem C suction piping from the suppression pool and condensate storage tank.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 2	400A-HPCF-106 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-801 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-F701C Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-P004C Press. Pt.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-D001C Temp. Str.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-800 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-F700C Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-PI001C Press.Ind.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-PT002C Press.Trn.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-PT003C Press.Trn.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-106 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-118 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-F012C Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-F011C Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-117 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	50A-HPCF-F002C Check V.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-F001C MO.Valve	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-110 Pipe	0.310 MPaG, 104°C,3B,As	No change
	25A-HPCF-123 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	25A-HPCF-F020C Relief V.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-F007C Check V.	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-125 Pipe	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	20A-HPCF-F022C T.Valve&cap	2.82 MPaG, 100°C,3B,As	Was 1.37 MPaG
	400A-HPCF-F006B MO.Valve	2.82 MPaG, 104°C,3B,As	Was 1.37 MPaG
	400A-HPCF-109 Pipe	0.310 MPaG, 104°C,3B,As	No change
	400A-HPCF-D003C Suction Str.	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

HPCF Subsystem C test and minimum flow piping to the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 2	80A-HPCF-114 Pipe	0.310 MPaG, 104°C,3B,As	No change
	200A-HPCF-112 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-HPCF-124 Pipe	0.310 MPaG, 104°C,3B,As	No change
	250A-RHR- 209 Pipe	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

HPCF Subsystem C keep fill line interface.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	20A-MUWC-138 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 1	25A-HPCF-F013C Valve	10.79 MPaG, 100°C,3B,As	No change
	25A-HPCF-D006C R0	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-119 Pipe	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-120 Pipe	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-F016C Valve	10.79 MPaG, 100°C,3B,As	No change

* Connects to locked open valves from condensate storage tank which is vented to atmosphere.

3MA.4 Reactor Core Isolation Cooling System

3MA.4.1 Upgrade Description

The RCIC System pump suction piping was low pressure and has been upgraded to the URS design pressure. The RCIC has two suction sources, the primary source being the CST and the suppression pool as an alternate.

The URS boundary was terminated at the last RCIC valve in the pipeline to the CST, E51-F001. Beyond this valve, the pipeline is open to the CST except for three locked open maintenance valves in parallel adjacent to the CST. The pipeline to the CST is a large pipe (500A) providing a common supply to the RCIC, HPCF, and SPCU System. Because of the open communication to the CST, and the CST is vented to atmosphere, this line cannot be pressurized. The CST is a large structure, impractical to upgrade to the URS design pressure.

The URS boundary was terminated at the last valve before the suppression pool, which is valve E510F006 and is normally closed. The suppression pool is a large structure, impractical to upgrade to the URS design pressure. The only portions of the RCIC System that are not upgraded to the URS design pressure is unobstructed piping to the suppression pool.

3MA.4.2 Downstream Interfaces

Other systems are listed below that interface with RCIC and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

- Makeup Water (Condensate) System upstream of the injection valve for the purpose of providing the piping keep-fill water source and a filling and flushing water source. The MUWC System is discussed in Section 3MA.11.
- High Conductivity Waste System for drainage is located downstream of CST suction check valve. HCW is discussed in Section 3MA.13.
- Reactor Core Isolation Cooling System shares common CST suction. The CST suction has open communication to the CST, and the CST is vented to atmosphere; this line cannot be pressurized and was not practical to upgrade to the URS design pressure.
- Suppression Pool Cleanup System shares common CST suction. The CST suction has open communication to the CST, and the CST is vented to atmosphere; this line cannot be pressurized and was not practical to upgrade to the URS design pressure.

3MA.4.3 Upgraded Components — RCIC System

A detailed listing of the components upgraded for the RCIC System follows, including identification of those interfacing system components not requiring upgrade.

REACTOR CORE ISOLATION COOLING SYSTEM, Tier 2 Figure 5.4-8, Sheets 1 & 3.

RCIC pump suction piping

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	200A-RCIC-001-W Pipe	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-703-W Pipe	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-F701 Valve	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-PX015 P.Test	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	200A-RCIC-D001 Strainer	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-700-W Pipe	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-F700 Valve	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-PT001 P.Trans	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-701-W Pipe	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-702-W Pipe	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-PI003 P.Ind.	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	20A-RCIC-PT002 P.Trans	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	50A-RCIC-018-W Pipe	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	50A-RCIC-F017 R.Valve	2.82 MPaG, 104°C,3B,As	Was 1.37 MPaG
	200A-RCIC-F002 T.Check	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	200A-RCIC-F060 Valve	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	* 200A-RCIC-F001 MO Valve	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	** 200A-RCIC-005-W Pipe	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	** 200A-RCIC-F007 Check V.	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	** 20A-RCIC-025-W Pipe	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
	** 20A-RCIC-F027 T.Valve	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG
Sheet 1	** 200A-RCIC-F006 MO Valve	2.82 MPaG, 104°C,3B,As	Was 1.37 MPaG

* HPCF Interface Piping 200A-HPCF-015-S, 1.37 MPaG, 66°C,B (S1,S2), As (open pathway to Condensate Storage Tank with LO valves).

** Suction Piping from Suppression Pool Interface 200A-RCIC-004-W, 0.310 MPaG, 104°C, 3B, As.

RCIC discharge from relief valves and test line valve direct to the suppression pool without restriction.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	50A-RCIC-009-W Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RCIC-019-W Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RCIC-007-W Pipe	0.310 MPaG, 104°C,3B,As	No change
	250A-RHR- 008 Pipe	0.310 MPaG, 104°C,3B,As	No change
Sheet 1	Suppression Pool		

ABWR High Press. Core Flooder System, Tier 2 Figure 6.3-7, components interfacing with RCIC System are not upgraded because this is the open pathway to the condensate storage tank vented to the atmosphere.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	200A-HPCF-015-W Pipe	1.37 MPaG, 66°C,B (S1,S2), As	No change
	400A-HPCF-105-W Pipe	1.37 MPaG, 66°C,B (S1,S2), As	No change
	500A-HPCF-004-W Pipe	1.37 MPaG, 66°C,B (S1,S2), As	No change
	300A-HPCF-001-W Pipe	1.37 MPaG, 66°C,B (S1,S2), As	No change
	300A-HPCF-002-W Pipe	1.37 MPaG, 66°C,B (S1,S2), As	No change
	300A-HPCF-003-W Pipe	1.37 MPaG, 66°C,B (S1,S2), As	No change

ABWR Makeup Water System (Condensate), Tier 2 Figure 9.2-4, components interfacing with HPCF System are not upgraded due to the open pathway to the condensate storage tank vented to the atmosphere.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	300A-MUWC-F100 Valve	1.37 MPaG, 66°C,B (S1,S2), As	No change
	300A-MUWC-F101 Valve	1.37 MPaG, 66°C,B (S1,S2), As	No change
	300A-MUWC-F102 Valve	1.37 MPaG, 66°C,B (S1,S2), As	No change
	300A-MUWC-100 Pipe		
	Static Hd,	66°C,B (S1,S2), As	No change
	300A-MUWC-101 Pipe		
	Static Hd,	66°C,B (S1,S2), As	No change
	300A-MUWC-102 Pipe		
	Static Hd,	66°C,B (S1,S2), As	No change
	Condensate Storage Tank,	66°C,4D, Non-seismic	No change

RCIC turbine condensate piping to the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	250A-RCIC-037-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	20A-RCIC-720-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	20A-RCIC-F722 Valve	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	20A-RCIC-PI012 P.Ind.	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	350A-RCIC-Cond. Pot	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	350A-RCIC-038-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	20A-RCIC-721-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	20A-RCIC-F723 Valve	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	20A-RCIC-722-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	20A-RCIC-PT013A P.Trans	8.62 MPaG, 302°C,3B,As	Was 1.37 MPaG
	20A-RCIC-PT013E P.Trans	8.62 MPaG, 302°C,3B,As	Was 1.37 MPaG
	** 25A-RCIC-051-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	** 25A-RCIC-F051 Valve	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	** 25A-RCIC-D012 Strainer	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	** 25A-RCIC-D013 S.Trap	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	** 25A-RCIC-F052 Valve	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
Sheet 3	** 25A-RCIC-052-S Pipe	2.82 MPaG, 184°C,4D,As	Was 0.981 MPaG
Sheet 1	350A-RCIC-F038 Check	8.62 MPaG, 302°C,3B,As	Was 1.37 MPaG
	20A-RCIC-053-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	20A-RCIC-F053 T.Valve	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	350A-RCIC-F039 Valve	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	A-RCIC-F069 T.Valve	2.82 MPaG, 184°C,3B,As	Was 10.981 MPaG
	350A-RCIC-039-S Pipe	0.981 MPaG, 184°C,3B,As	No change
Sheet 1	Suppression Pool		

* Vent via Rupture Disks.

** RCIC Turbine Condensate Piping to the Barometric Condenser.

RCIC vacuum tank condensate piping to the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	50A-RCIC-Vacuum Pump	2.82 MPaG, 121°C,4D,As	Was 0.755 MPaG
	50A-RCIC-044-S Pipe	2.82 MPaG, 88°C,4D,As	Was 0.310 MPaG
	50A-RCIC-067-S Pipe	2.82 MPaG, 88°C,4D,As	Was 0.310 MPaG
	50A-RCIC-PCV Valve	2.82 MPaG, 121°C,4D,As	Was 0.755 MPaG
Sheet 3	20A-RCIC-068-S Pipe	2.82 MPaG, 121°C,4D,As	Was 0.981 MPaG
Sheet 1	50A-RCIC-F046 Check V.	2.82 MPaG, 104°C,3B,As	Was 0.310 MPaG
	20A-RCIC-057-S Pipe	2.82 MPaG, 104°C,3B,As	Was 0.310 MPaG
	20A-RCIC-F059 T.Valve	2.82 MPaG, 104°C,3B,As	Was 0.310 MPaG
	50A-RCIC-F047 MO Valve	2.82 MPaG, 104°C,3B,As	Was 0.310 MPaG
	50A-RCIC-045-S Pipe	0.981 MPaG, 104°C,3B,As	No change
Sheet 1	Suppression Pool		

RCIC steam drains from trip and throttle valve piping and turbine to condensate chamber.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	* 20A-RCIC-063-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	* 20A-RCIC-061-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG
	** 20A-RCIC-064-S Pipe	8.62 MPaG, 302°C,3B,As	Was 0.981 MPaG

* RCIC Trip and Throttle Valve leakoffs are piped to Condensing Chamber.
** RCIC Turbine Condensate Drain connects to the Condensing Chamber

RCIC turbine valve leakoffs are piped to the barometric condenser.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	* 25A-RCIC-058-S Pipe	2.82 MPaG, 184°C,4D,As	Was 0.981 MPaG
	** 25A-RCIC-059-S Pipe	2.82 MPaG, 184°C,4D,As	Was 0.981 MPaG
	Barometric Condenser	2.82 MPaG, 184°C,4D,As	Was 0.755 MPaG
	*** 25A-RCIC-065-S Pipe	2.82 MPaG, 184°C,4D,As	Was 0.755 MPaG
	25A-RCIC-Relief Valve	2.82 MPaG, 121°C,4D,As	Was 0.755 MPaG
	25A-RCIC-066-S Pipe	0 MPaG, 121°C,4D,As	No change

* RCIC Trip and Throttle Valve Stem leakoff is piped to the Barometric
** RCIC Turbine Governor Valve Stem is piped to the to Barometric Condenser.
*** Barometric Condenser Press. relief and piping.

RCIC pump cooling water piping for pump and turbine lube oil coolers.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	50A-RCIC-011-W Pipe	2.82 MPaG,77°C,3B,As	Was 0.863 MPaG
	50A-RCIC-028-W Pipe	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	50A-RCIC-F030 Relief V.	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	50A-RCIC-029-W Pipe	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	20A-RCIC-713-W Pipe	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	20A-RCIC-PX018 Press	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	50A-RCIC-Turb.LO Cooler	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	50A-RCIC-Pump LO Cooler	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	15A-RCIC-TX019 Temp.Pt.	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	20A-RCIC-714-W Pipe	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	20A-RCIC-F714 Valve	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	20A-RCIC-PX020 Press.Pt.	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	15A-RCIC-012-W Pipe	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	15A-RCIC-013-W Pipe	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	15A-RCIC-014-W Pipe	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
	15A-RCIC-015-W Pipe	2.82 MPaG, 77°C,3B,As	Was 0.863 MPaG
Sheet 3	Barometric Condenser	2.82 MPaG, 121°C,4D,As	Was 0.755 MPaG

RCIC vacuum tank and condensate pump piped to RCIC pump suction pipe.

Reference	Components	Press./Temp./Design/ Seismic Class	Remark
Sheet 3	RCIC Vacuum Tank	2.82 MPaG, 77°C,4D,As	Was 0.755 MPaG
	RCIC Press. Switch H	2.82 MPaG, 121°C,4D,As	Was 0.755 MPaG
	RCIC Level Switch H	2.82 MPaG, 121°C,4D,As	Was 0.755 MPaG
	RCIC Level Switch L	2.82 MPaG, 121°C,4D,As	Was 0.755 MPaG
	RCIC Cond. Pump	2.82 MPaG, 88°C,4D,As	Was 1.37 MPaG
	50A-RCIC-F014 Check V.	2.82 MPaG, 88°C,4D,As	Was 1.37 MPaG
	50A-RCIC-016-W Pipe	2.82 MPaG, 88°C,4D,As	Was 1.37 MPaG
	20A-RCIC-715-W Pipe	2.82 MPaG, 88°C,4D,As	Was 1.37 MPaG
	20A-RCIC-F715 Valve	2.82 MPaG, 88°C,4D,As	Was 1.37 MPaG
	20A-RCIC-PX021 Press.Pt.	2.82 MPaG, 88°C,4D,As	Was 1.37 MPaG
	50A-RCIC-F015 Valve	2.82 MPaG, 88°C,3B,As	Was 1.37 MPaG
Sheet 3	50A-RCIC-017-W Pipe	2.82 MPaG, 88°C,3B,As	Was 1.37 MPaG
	50A-RCIC-030-W Pipe	2.82 MPaG, 88°C,3B,As	Was 1.37 MPaG
	50A-RCIC-F031 MO Valve	2.82 MPaG, 88°C,3B,As	Was 1.37 MPaG
	50A-RCIC-F032 AO Valve	2.82 MPaG, 88°C,3B,As	Was 1.37 MPaG
	20A-RCIC-032-W Pipe	2.82 MPaG, 88°C,3B,As	Was 1.37 MPaG
	20A-RCIC-F034 T.Valve	2.82 MPaG, 88°C,3B,As	Was 1.37 MPaG
	* 50A-RCIC-F016 Check	2.82 MPaG, 77°C,3B,As	Was 1.37 MPaG

* 50A-RCIC-017 Pipe connects with RCIC pump suction 200A-RCIC-001-W Pipe on sheet 1 upgraded to 2.82 MPaG.

Sheet 2: Valve gland leak off piping.

Branch piping from RCIC steam supply isolation valves FO-035,inside primary containment and FO-036 outside primary containment to VGL Radwaste Treatment System.

Reference	Components	Press./Temp./Design/ Press. Rating	Remarks
Sh 2,I-11	25A-RCIC-506-S Pipe	8.62 MPaG, 302°C,1A,As	Reactor Press.
I-7	25A-RCIC-507-S Pipe	8.62 MPaG, 302°C,1A,As	Reactor Press.

Sheet 2: Instrument piping from RCIC steam supply piping to PT-009, PI-010 and level switch LS-011.

Reference	Components	Press./Temp./Design/ Press. Rating	Remarks
Sh 2,H-6	20A-RCIC-716-S Pipe	8.62 MPaG, 302°C,1A,As	Reactor Press.
H-7	20A-RCIC-717-S Pipe	8.62 MPaG, 302°C,1A,As	Reactor Press.
G-5	20A-RCIC-718-S Pipe	8.62 MPaG, 302°C,1A,As	Reactor Press.
F-5	0A-RCIC-719-S Pipe	8.62 MPaG, 302°C,1A,As	Reactor Press.

No other low Press. components of the RCIC System were identified for upgrading to the higher design Press. as shown on the marked P & ID's. Interface with the LCW Reactor Building sump which is vented to atmosphere, is through open funnel drains with low Press. piping to the sump.

3MA.5 Control Rod Drive System

3MA.5.1 Upgrade Description

The CRD System interfaces with the reactor in a manner that makes low pressure piping over pressurization very unlikely. The minimum failure path from the reactor to the low pressure piping has three check valves in series and the second check valve is 1.27 cm in size. This path is from the purge flow channels of the CRD, out through the first check valve in the CRD housing, through the purge supply line that has the second 1.27 cm check valve, and to the pump discharge check valve. An alternate path through the accumulator charging line has additionally the normally closed scram valve, and this path is less likely for failure, therefore not considered. The path from the pump discharge, back through the pump to its suction, and back through the suction lines to the condensate storage tank or the condensate feedwater source is an open path. The open pump suction pipeline is a minimum 100 mm diameter except for a 150 mm diameter attachment to the Condensate Storage Tank. The CRD pumps run continuously while the reactor is at operating pressure, which prevents reactor pressure from reaching the low pressure piping unless for the unlikely case when both CRD pumps have failed. Therefore, an ISLOCA condition from a 12.7 mm diameter source could only occur when three check valves in series fail open at the same time both CRD pumps have failed. The ISLOCA guidelines do not provide credit for this rare condition, so the low pressure piping has been upgraded to the URS design criteria over the entire low pressure piping region of the CRD system. The suction path through the Makeup Water System (Condensate) to the Condensate Storage Tank from the CRD interface is an open path whose design pressure was not upgraded to URS design criteria. The piping design of the primary suction path through the Condensate, Feedwater and Condensate Extraction System has not been established, but if a check valve is in the path, the design pressure up to and including the check valve will be the URS design pressure.

The normal key assumption to this evaluation, as stated in the Boundary Limits of URS section above, that the valve adjacent to a low pressure sink remains closed, means that the pump discharge check valve remains closed as a given. however, this valve is in the high pressure piping, which is unique for the CRD system. according to this accepted line of reasoning, the low pressure piping would not have to be upgraded because it would not experience the high reactor pressure. However, the low pressure piping has been upgraded in response to reference 1's guidance that states "for all interfacing systems and components which do not meet the full RCS URS criteria, justification is required, which must include engineering feasibility; not solely a risk benefit analysis." Upgrading the low pressure piping is feasible and was done.

3MA.5.2 Downstream Interfaces

Other systems are listed below that interface with the CRD system and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

- Reactor Water Cleanup System at the output of the filter units. The RWCU design pressure exceeds the URS design pressure without upgrade.
- Reactor Recirculation System purge water supplied by the CRD system, has a 8.83 mPaGdesign pressure, which exceeds the URS design pressure and needs no upgrade.
- Makeup Water (Condensate) System provides pump suction from and system return to the CST. The MUWC system is discussed in Section 3MA.11, where it is explained how certain MUWC upgrades were made that provide an open path to the CST. This line cannot be pressurized because of the open communication to the CST, and the CST is vented to atmosphere. There is no source to pressurize the MUWC line because of closed pump discharge check valves in the CRD system's URS region.
- Condensate, Feedwater and Air Extraction system provides pump suction from the turbine building condensate supply. This system is not part of the Tier 2 design scope, but it is expected to be an open path to a large source similar to the MUWC system. Because of the open path the CFAE system was not upgraded.
- Sampling system at the output of the filter units. The Sampling systems's design pressure exceeds the URS design pressure without upgrade.
- Nuclear Boiler system at a branch off of the CRD purge line provides the water for conducting RPV hydro tests and the 9.81 MPaG design pressure exceeds the URS design pressure and needs no upgrade.

3MA.5.3 Upgraded Components — CRD System

A detailed listing of the components upgraded for the CRD System follows, including identification of those interfacing system components not requiring upgrade.

CONTROL ROD DRIVE SYSTEM, Tier 2 Figure 4.6-8, Sheets 1, 2 & 3.

CRD pump suction piping Condensate, Feedwater and Condensate Air Extraction System or Condensate Storage Tank of the Makeup Water System (Condensate).

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Note 14	100A-CFDWAO-Fxxx Valve	4.12 MPaG, 66°C,B,(S1,S2),As	No change
Sheet 1	100A-CRD-001-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	150A-MUWC-F103 Valve LO	1.37 MPaG, 66°C,B,(S1,S2),As	No change
	150A-CRD-002-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
Sheet 1	Condensate Storage Tank,	66°C,4D, Non-seismic	No change
	50A-MUWC-F103 Valve	1.37 MPaG, 66°C,6D,C	Lock Open
	50A-MUWC-103 Pipe	Static Hd, 66°C,6D,C	No change
	50A-CRD-033-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	50A-CRD-032-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-F001A Gate V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-003-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-D001A Filter	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-500-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-F500A Valve NC	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-501-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-F501A Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-004-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-F002A Gate V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-F001B Gate V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-005-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-D001B Filter	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-502-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-F500B Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-503-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-F501B Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-006-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-F002B Gate V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-007-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-700-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-F700 Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-DPT001 Diff P T	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-F701 Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-701-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-F003A Gate V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-008-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	25A-CRD-504-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	25A-CRD-F004A Safe.RV	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-702-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-F702A Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-PI002A Press I	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-PT003A Press T	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-C001A Pump	3.43 MPaG, 66°C,6D,C	No change
	* CRD-F502A Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG

	* A-CRD-505-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F503A Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F504A Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* A-CRD-506-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* A-CRD-507-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F505A Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F506A Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-F003B Gate V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	100A-CRD-010-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	25A-CRD-508-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	25A-CRD-F004B Relief V.	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-703-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	20A-CRD-F702B Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-PI002B Press I	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-PT003B Press T	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	CRD-C001B Pump	3.43 MPaG, 66°C,6D,C	No change
	* A-CRD-509-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F502B Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F503B Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* A-CRD-510-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F504B Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F505B Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* A-CRD-511-S Pipe	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG
	* CRD-F506B Globe V	2.82 MPaG, 20°C,6D,C	Was 1.37 MPaG

* Size dependent on pump requirements.

CRD interface from pump discharge to the MUWC System condensate storage tank.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	50A-CRD-034-S Pipe	18.63 MPaG, 66°C,6D,C	No change
	50A-CRD-F022 Globe V	18.63 MPaG, 66°C,6D,C	No change
	50A-CRD-035-S Pipe	18.63 MPaG, 66°C,6D,C	No change
	50A-CRD-F023 Globe V	18.63 MPaG, 66°C,6D,C	No change
	50A-MUWC-xxx-S Pipe	1.37 MPaG, 66°C,6D,C	No change
	50A-MUWC-Fxxx Globe V	1.37 MPaG, 66°C,6D,C	No change
	Condensate Storage Tank	66°C,Non-seismic	No change

CRD interface from pump discharge to the RRS System.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	20A-CRD-036-S Pipe	18.63 MPaG, 66°C,4C,B	No change
	20A-CRD-F024 Globe V	18.63 MPaG, 66°C,4C,B	No change
	20A-CRD-F025 Globe V	18.63 MPaG, 66°C,4C,B	No change

CRD interface from pump discharge to the CUW System.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	20A-CRD-037-S Pipe	18.63 MPaG, 66°C,4C,B	No change
	20A-CRD-F026 Globe V	18.63 MPaG, 66°C,4C,B	No change
	20A-CRD-F027 Globe V	18.63 MPaG, 66°C,4C,B	No change

No other low pressure components of the Control Rod Drive System were identified for upgrading to the higher design pressure as shown on the marked P & ID's. Interface with the LCW Reactor Building sump which is vented to atmosphere, is through open funnel drains with low pressure piping to the sump.

3MA.6 Standby Liquid Control System

3MA.6.1 Upgrade Description

The SLC System interfaces with the reactor through the HPCF injection piping inside the drywell. The leakage path includes three 40A check valves in series with normally closed motor operated valves in addition to the positive displacement pumps piped in parallel. A 40A nominal pipe size test pipe from the pump discharge piping to the test tank has two normally closed valves in series. All of these valves have leakage test features. Short monthly pump operating tests produce demineralized water flow through the test tank.

The 100A pump suction piping between the pumps, the storage tank outlet valve, the test tank and the MUWP System interface is upgraded to URS design criteria. The SLC storage tank is vented to atmosphere and serves as the pressure release sink connecting to the outermost pump suction piping valves.

All low pressure instrumentation, pressure relief, drain piping and valving are upgraded to URS design criteria to reduce the level of pressure challenge to these components.

3MA.6.2 Downstream interfaces.

Other systems are listed below that interface with the SLC System and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

- MUWP System 80A piping interface occurs at the SLC check valve connected to a branch off the test loop suction pipe. This SLC branch piping consists of a normally closed flushing valve and a normally open 20A suction piping pressurizing valve to prevent borated solution migrating to the SLC injection pump suction piping. Refer to Section 3MA.12 for upgrade information on the MUWP System.

- MUWP System also provides the makeup water to the SLC System storage tank through block and bleed valves and a piping drain to a portable container to prevent leakage of additional makeup into the SLC storage tank which could dilute the borate solution in the tank.

3MA.6.3 Upgraded Components — SLC System

A detailed listing of the components upgraded for the SLC System follows, including identification of those interfacing system components not requiring upgrade.

STANDBY LIQUID CONTROL SYSTEM, Tier 2 Figure 9.3-1, Sheet 1.

SLC Injection Pump A suction piping from the SLC storage tank.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	SLC-C001A Pump	10.79 MPaG, 66°C,2B,A	No Change
	SLC-F003A Relief V.	10.79 MPaG, 66°C,2B,A	No Change
	50A-SLC Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	100A-SLC-F002A Valve LO	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	100A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	100A-SLC-F001A Valve MO	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	* SLC-A001 Storage Tk.	Static Hd., 66°C,2B,A	No Change

SLC Injection Pump B suction piping from the SLC storage tank.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	SLC-C001B Pump	10.79 MPaG, 66°C,2B,A	No Change
	SLC-F003B Relief V.	10.79 MPaG, 66°C,2B,A	No Change
	50A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	100A-SLC-F002B Valve LO	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	100A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	20A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	20A-SLC-F500 Valve	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	100A-SLC-F001B Valve MO	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	* SLC-A001 Storage TK.	Static Hd., 66°C,2B,A	No Change

* SLC Storage Tank is vented to atmosphere.

SLC test tank piping.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	** 40A-SLC-F011 Valve LC	10.79 MPaG, 66°C,2B,A	Was ATP
	40A-SLC-SS Pipe	10.79 MPaG, 66°C,2B,A	Was 1.37 MPaG
	SLC-A002 Test Tank	2.82 MPaG, 66°C,2B,A	Was STH
	100A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	100A-SLC-F012 Valve LC	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	25A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	SLC-F026 Relief V.	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	20A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	100A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG

** ATP is atmospheric pressure.

SLC interface with MUWP for makeup and pressurization of suction piping from tank. (Pressure higher than static head of SLC storage tank.)

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	80A-MUWP-F163 Valve LO	1.37 MPaG, 66°C,4D,C	No change
	80A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	SLC-F013 Check V.	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	80A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	80A-SLC-F014 Valve LC	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	80A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	20A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	20A-SLC-F020 Valve LO	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	20A-SLC-D002 RO	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	20A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG

SLC storage tank interface with MUWP for purified makeup water.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	80A-MUWP-F163 Valve LO	1.37 MPaG, 66°C,4D,C	No change
	80A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	SLC-F013 Check V.	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	80A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	25A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	25A-SLC-F015 Valve LC	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	20A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	20A-SLC-F505 Valve NO	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	25A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	Was 1.37 MPaG
	25A-SLC-F023 Valve LC	2.82 MPaG, 66°C,2B,A	Was 1.37 MPaG
	25A-SLC-SS Pipe	2.82 MPaG, 66°C,2B,C	No change
	*SLC-A001 Storage TK.	Static Head, 66°C,2B,A	No change

* SLC Storage Tank is vented to atmosphere.

3MA.7 Reactor Water Cleanup System

3MA.7.1 Upgrade Description

The Reactor Water Cleanup system (CUW) is a high pressure system that is almost totally designed above the URS design pressure. One pipe connecting to radwaste was upgraded. It is the pipe downstream of valve G31-F023 shown at zone E-14 of Figure 5.4-12, sheet 3. The interface symbol is labeled “LCW Collector Tank”.

3MA.7.2 Downstream Interfaces

A system is listed below that interfaces with CUW and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

- Low Conductivity Waste, (Radwaste) connects to a branch of the CUW filter/demineralizer discharge, as described in 3MA.7.1 above. There is not a practical reason to upgrade this interface in CUW as discussed in the Radwaste system, Section 3MA.13.

3MA.7.3 Upgraded Components — CUW System

A detailed listing of the components upgraded for the CUW System follows, including identification of those interfacing system components not requiring upgrade.

REACTOR WATER CLEANUP SYSTEM, Tier 2 Figure 5.4-12, Sheets 1, 2 and 3.

CUW System interface with Radwaste System

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	150A-CUW-F023 Valve MO	10.20 MPaG, 66°C,4D,C	No change
	150A-CUW-31-CS Pipe	2.82 MPaG, 66°C,6D,C	Was 0.981 MPaG
	* LCW Collector Tank A	0 MPaG	No change

3MA.8 Fuel Pool Cooling Cleanup System

3MA.8.1 Upgrade Description

The fuel pool cooling system interfaces with the RHR system at two locations that could possibly expose the FPC system to reactor pressure. One location is the discharge from the FPC to RHR in the line downstream from the skimmer surge tank; the other location is the RHR return to the FPC in the line to the reactor well. See Figure 9.1-1a, upper right and left hand corners respectively.

Upgrading of components and new pipeline with testable check valve FPC-F105 and gate valve FPC-F106 were added to the first interface of the discharge from the FPC to the RHR. This new line has the gate valve locked open with the check valve's flow direction into the skimmer surge tank and provides an open path into the skimmer surge tank from valves RHR-F016B and RHR-F016C. Valve FPC-F029 has an open path to the skimmer surge tank as provided by the existing design. This new line and its two new valves provides an open path to the skimmer surge tank that prevents FPC-F031 from overpressure. Valve FPC-F031 is open only during the mode of draining the dryer/separator pool or the reactor well pool, at which time the new locked open valve FPC-F106 must be closed, otherwise water could be pumped back into the surge tank. Closing FPC-F106 does not jeopardize ISLOCA protection because the reactor is shutdown during this mode. All the piping between the FPC valves, FPC-F029, FPC-F031, and FPC-F106 and the RHR valves, RHR-F016B and RHR-F016C, were upgraded to the URS design pressure of 2.82 MPaG.

The second interface, the RHR return to the FPC in the line to the reactor well, was not upgraded because of the continuous open path to the spent fuel storage pool and cask pit. Valves FPC-F093 and FPC-F017 are always locked open and provide an open path from the RHR valves, RHR-F015B and RHR-F015C, to the spent fuel storage pool and cask pit.

3MA.8.2 Downstream Interfaces

The fuel pool cleanup system has no further downstream system interfaces that could allow reactor pressure from RHR to proceed further than the FPC system.

3MA.8.3 Upgraded Components — FPC System

A detailed listing of the components upgraded for the FPC System follows, including identification of those interfacing system components not requiring upgrade.

FUEL POOL COOLING AND CLEANUP SYSTEM, Tier 2 Figure 9.1-1, Sheets 1, 2 and 3.

FPC System interface with makeup from RHR System or SPCU System.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	250A-RHR-F015C Valve MO	3.43 MPaG, 182°C,3B,As	No change
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,3B,4C,A(S2)	No change
	250A-RHR-F015B Valve MO	3.43 MPaG, 182°C,3B,As	No change
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-F094 Check Valve	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	20A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	20A-FPC-F506 Valve	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-F022 Valve LO	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-F023 Check Valve	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	REACTOR WELL		No change
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-F093 Valve LO	1.57 MPaG, 66°C,4C,A(S2)	No change
	80A-SPCU F014 Valve MO	1.57 MPaG, 66°C,4D,B	No change
	80A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	80A-FPC-F091 Check Valve	1.57 MPaG, 66°C,4C,A(S2)	No change
	80A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	80A-FPC-D011 RO	1.57 MPaG, 66°C,4C,A(S2)	No change
	80A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-F016 Check Valve	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-F017 Valve LO	1.57 MPaG, 66°C,4C,A(S2)	No change
	250A-FPC-F018 Check Valve	1.57 MPaG, 66°C,4C,A(S2)	No change
	SPENT FUEL STORAGE POOL		

FPC System interface with suction of RHR System for cooling.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	300A-RHR-F016C Valve MO	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	300A-FPC-SS Pipe	2.82 MPaG, 66°C,4C,B(S1,S2)	Was 1.37 MPaG
	300A-RHR-F016B Valve MO	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	300A-FPC-SS Pipe	2.82 MPaG, 66°C,4C,B(S1,S2)	Was 1.37 MPaG
	300A-FPC-F029 Valve NC	2.82 MPaG, 66°C,4C,B(S1,S2)	Was 1.37 MPaG
	300A-FPC-SS Pipe	Static Hd. atg, 66°C,4C,B(S1,S2)	No change
* SPENT FUEL STORAGE POOL			
	250A-FPC-SS Pipe	2.82 MPaG, 66°C,4C,B(S1,S2)	Was 1.37 MPaG
	250A-FPC-F031 Valve NC	2.82 MPaG, 66°C,4C,B(S1,S2)	Was 1.37 MPaG
	250A-FPC-SS Pipe	1.57 MPaG, 66°C,6D,C	No change
** FILTER DEMINERALIZER			No change
	300A-FPC-SS Pipe	2.82 MPaG, 66°C,4C,B(S1,S2)	New Branch
	300A-FPC-F105 Check Valve	2.82 MPaG, 66°C,4C,B(S1,S2)	New Valve
	300A-FPC-SS Pipe	2.82 MPaG, 66°C,4C,B(S1,S2)	New Branch
	300A-FPC-F106 Valve LO	2.82 MPaG, 66°C, 4C,B(S1,S2)	New Valve
	300A-FPC-SS Pipe	Static Hd. MPaG, 66°C,3B,A(S2)D	New Branch
	*** SKIMMER SURGE TANK		No change

* FPC Valve F029 is open only for fuel pool cooling mode B (maximum heat load operation with RHR System B or C operating in parallel with FPC System).

** FPC Valve F031 is open only for fuel pool cooling mode B (refueling when Dryer/Separator Pool is drained and pumped to Radwaste LCW collector tank by RHR System B or C).

*** FPC Valve F031 leakage is directed to skimmer surge tank through a lock open valve and a check valve into skimmer surge tank.

3MA.9 Nuclear Boiler System

3MA.9.1 Upgrade Description

The NBS piping and instrumentation are designed for reactor pressure. One low pressure level transmitter and level indicator with the associated piping and two normally closed globe valves are upgraded to URS design criteria. This level instrumentation is used to measure the level in the reactor well during refueling and is selected for the required sensitivity. A relief valve downstream of the two normally closed globe valves discharges to a LCW funnel drain to the Reactor Building LCW sump.

3MA.9.2 Downstream Interfaces

Other systems are listed below that interface with the NBS and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

CRD, RCIC, RPV, RHR, HPCF, CUW, MS, are high pressure interfaces of the NBS and RW(LCW, HCW, VG) are low pressure interfaces of the NBS. Interfacing systems at high pressure have low pressure interfaces addressed in their specific system listings.

3MA.9.3 Upgraded Components — NBS System

A detailed listing of the components upgraded for the NBS System follows.

NUCLEAR BOILER SYSTEM, Tier 2 Figure 5.1-3, Sheets 1 & 5.

Refueling level transmitter piping.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	20A-NBS-F708 Relief V	2.82 MPaG, 20°C,1A, As	Was 0.686 MPaG
	* 20A-NBS-LT004 Level XT	2.82 MPaG, 20°C,1A, As	Was 0.686 MPaG
	20A-NBS-Interconn. Pipe	2.82 MPaG, 20°C,1A, As	Was 0.686 MPaG

* LT-004 must be low pressure rated for level sensitivity during refueling.

Other fluid piping components of the NBS System are rated for reactor pressure, except the main steam drain header interface with the Condensate and Feedwater System piping to be designed for at least 2.82 MPaGand other drains including valve gland leakage, LCW and HCW funnel drains to the drywell equipment drain sump.

3MA.10 Reactor Recirculation System

3MA.10.1 Upgrade Description

Ten Reactor Internal Recirculation Pumps (RIP) are installed around the perimeter of the reactor vessel and operate at reactor pressure.

3MA.10.2 Downstream Interfaces

Other systems are listed below that interface with the RRS System and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

- MUWP System interfaces with each reactor recirculation pump to provide RIP casing makeup water. Another MUWP System interface exists during refueling or maintenance shutdown to provide water for the RIP shaft inflatable seal subsystem. Pressure upgrades are required for the interfacing components of the MUWP System.

- RCW System interfaces with each RRS RIP motor cooling subsystem through a heat exchanger designed for 8.62 MPaG .and utilizes RCW water for cooling the RIP motors. No upgrade is needed for the RCW System connecting piping designed to 1.37 MPaG.

- CRD System piping connects to ten RIP motor purge subsystems. Control Rod Drive System Tier 2 Figure 4.6-8, sheet 1 at C-2, the 20A-CRD-036 pipe and 20A-CRD-F025 valve interface with the 20A-RRS-003A pipe connecting to the ten RIP motors. No upgrade is required because the design pressure for both the CRD and RRS is 18.63 MPaG.

- RWS Open funnel drain piping connects to the LCW and HCW sumps in the drywell.

- MUWP Makeup Water System (Purified) Tier 2 Figure 9.2-5 shows other components interfacing with RRS System. These are not upgraded because they are part of the open pathway to the Condensate Storage Tank which is vented to the atmosphere. Another MUWP System interface is connected to a portable inflatable shaft seal pump and tank only during refueling or when the reactor is shut down for maintenance.

3MA.10.3 Upgraded Components — RRS System

A detailed listing of the components upgraded for the RRS System follows, including identification of those interfacing system components not requiring upgrade.

REACTOR RECIRCULATION SYSTEM Tier 2 Figure 5.4-4, Sheets 1 & 2.

RRS interface with MUWP System for Reactor Internal Pump (RIP) casing makeup water.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	20A-RRS-502A-K Pipes	8.62 MPaG,302°C, 4A,As	No change
	20A-RRS-F504A-K Valves NC	8.62 MPaG,302°C, 4A,As	No change
	15A-MUWP-189-198 Pipes	2.82 MPaG, 66°C, 6D,C	Was 1.37 MPaG
	50A-MUWP-185 Pipe	2.82 MPaG, 66°C, 6D,C	Was 1.37 MPaG
	50A-MUWP-F142 Check Valve	2.82 MPaG,171°C, 3B,As	Was 1.37 MPaG
	50A-MUWP-184 Pipe	2.82 MPaG,171°C, 3B,As	Was 1.37 MPaG
	50A-MUWP-F141 Valves NC	2.82 MPaG,171°C, 3B,As	Was 1.37 MPaG
	50A-MUWP-183 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	80A-MUWP-181 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	80A-MUWP-F140 Valve LO	1.37 MPaG, 66°C, 6D,C	No change
	125A-MUWP-101 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	125A-MUWP-F101 Valve LO	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-602 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F602 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-601 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F601 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-FQ102 Flow Integr.	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-801 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F801 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-800 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F800 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-PX101 Press. Pt.	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-600 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F600 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F100 Valve LO	1.37 MPaG, 66°C, 6D,C	No change
	125A-MUWP-102 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	125A-MUWP-F102 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	150A-MUWP-xxx Pipe	1.37 MPaG, 66°C, 6D,C	No change
	150A-MUWP-xxx Pipe	1.37 MPaG, 66°C, 6D,C	No change
	50A-RRS-Fxxx Check Valve	1.37 MPaG, 66°C, 6D,C	No change
	Condensate Storage Tank,	66°C, 6D,Non-seismic	No change

3MA.11 Makeup Water System Condensate

3MA.11.1 Upgrade Description

The MUWC System has extensive system interfaces throughout the plant for makeup water to fill systems and serve flushing connections. The extent of the piping and the size of the Condensate Storage Tank of the MUWC System makes it impractical to upgrade. Instead valves are changed to lock open type to create a clear path from the URS boundary to the Condensate Storage Tank which is vented to atmosphere.

3MA.11.2 Downstream Interfaces

HPCF System is a downstream interface of the MUWC System at three outlets of the Condensate Storage Tank. The CRD piping is not upgraded to the URS design pressure because the maximum static head is 0.159 MPaG. The first closed valve of the HPCF System suction piping is upgraded to URS design pressure based on data provided in Section 2.

CRD System 150A suction piping interfaces with Condensate Storage Tank.

Other interfaces include the HPCF System fill line, RHR flushing lines, CRD makeup and discharge, and MUWP System are not upgraded due to the impractical nature of upgrades for such an extensive piping system with lock open type valves and open piping paths to the vented condensate storage tank.

All MUWC valves between the interfacing system connections and the Condensate Storage Tank are lock open type to provide an open pathway to relieve pressure to this tank which is vented to the atmosphere.

3MA.11.3 Upgraded Components — MUCW System

A detailed listing of the components upgraded for the MUWC System follows, including identification of those interfacing system components not requiring upgrade.

MAKEUP WATER SYSTEM (CONDENSATE) Tier 2 Figure 9.2-4, Sheets 1.

HPCF Subsystem B keep fill line interface.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	* 50A-MUWC-135 Pipe	1.37 MPaG, 66°C,4D,B	No change
	25A-HPCF-F013B Valve LO	1.37 MPaG, 100°C,3B,As	No change
	25A-HPCF-D006B R0	1.37 MPaG, 100°C,3B,As	No change
	25A-HPCF-019 Pipe	1.37 MPaG, 100°C,3B,As	No change
	50A-HPCF-F016B Valve	1.37 MPaG, 100°C,3B,As	No change
	50A-HPCF-F014B Check V.	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-F015B Check V.	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-020 Pipe	10.79 MPaG, 100°C,3B,As	No change

HPCF Subsystem C keep fill line interface.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	* 50A-MUWC-138 Pipe	1.37 MPaG, 66°C,4D,B	No change
	25A-HPCF-F013C Valve LO	1.37 MPaG, 100°C,3B,As	No change
	25A-HPCF-D006C R0	1.37 MPaG, 100°C,3B,As	No change
	25A-HPCF-119 Pipe	1.37 MPaG, 100°C,3B,As	No change
	50A-HPCF-F016C Valve	1.37 MPaG, 100°C,3B,As	No change
	50A-HPCF-F014C Check V.	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-F015C Check V.	10.79 MPaG, 100°C,3B,As	No change
	50A-HPCF-120 Pipe	10.79 MPaG, 100°C,3B,As	No change

MUWC System interface with HPCF System.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	300A-HPCF-001 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	300A-HPCF-002 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
	300A-HPCF-003 SS Pipe	1.37 MPaG, 66°C,B(S1,S2)	No change
Sheet 1	300A-MUWC-F100 Valve LO	1.37 MPaG, 66°C,4D,B	No change
	300A-MUWC-F101 Valve LO	1.37 MPaG, 66°C,4D,B	No change
	300A-MUWC-F102 Valve LO	1.37 MPaG, 66°C,4D,B	No change
	300A-MUWC-100 Pipe	Static Hd. 66°C,4D,B	No change
	300A-MUWC-101 Pipe	Static Hd. 66°C,4D,B	No change
	300A-MUWC-102 Pipe	Static Hd. 66°C,4D,B	No change

RHR Subsystem A flushing line interface at branch discharging to feedwater.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	* 100A-MUWC-134 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 3	100A-RHR -F032A Valve	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -026 Pipe	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F033A Check V.	3.43 MPaG, 182°C,3B,As	No change

RHR Subsystem A flushing line interface at suction shutdown branch from RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	* 100A-MUWC-133 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 2	100A-RHR -F040A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -031 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -F041A Check V.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem B flushing line interface at branch discharging to RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	* 100A-MUWC-137 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 5	100A-RHR -F032B Valve	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -132 Pipe	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F033B Check V.	3.43 MPaG, 182°C,3B,As	No change

RHR Subsystem B flushing line interface at suction of shutdown branch from RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	* 100A-MUWC-136 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 2	100A-RHR -F040B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -138 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -F041B Check V.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem C flushing line interface at branch discharge to RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	* 100A-MUWC-140 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 2	100A-RHR -F032C Valve	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -233 Pipe	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F033C Check V.	3.43 MPaG, 182°C,3B,As	No change

RHR Subsystem C flushing line interface at suction of shutdown branch from RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	* 100A-MUWC-139 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 2	100A-RHR-F040C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR-239 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR-F041C Check V.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

* Makeup Water System (Condensate) piping designed with open pathway to Condensate Storage Tank.

MUWC System changes and upgrades.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	150A-MUWC-F131 Valve LO	1.37 MPaG, 66°C,4D,B	No change
	250A-MUWC-F111 Valve LO	1.37 MPaG, 66°C,4D,B	No change
	250A-MUWC-F110 Valve LO	1.37 MPaG, 66°C,4D,B	No change
	** 250A-MUWC-110 Pipe	1.37 MPaG, 66°C,4D,B	No change

** Interface with new MUWC System pump minimum flow bypass pipe with check valve and LO service valves connecting to Condensate Storage Tank.

MUWC System interface with MUWP.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	150A-WUMP-101 SS Pipe	1.37 MPaG, 66°C,4D,C	No change
	150A-WUMP-Fxxx SS Valve LO	1.37 MPaG, 66°C,4D,C	No change
	150A-WUMP-Fxxx SS Check V.	1.37 MPaG, 66°C,4D,C	No change
	Condensate Storage Tank		

MUWC interface with the CRD System pump suction piping.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	150A-CRD-002-S Pipe	2.82 MPaG, 20°C,4D,B	Was 1.37 MPaG
Sheet 1	150A-MUWC-Fxxx LO Valve	1.37 MPaG, 66°C,4D,B	Lock Open
	150A-MUWC-xxx Pipe	1.37 MPaG, 66°C,4D,B	No change
	150A-MUWC-Fxxx LO Valve	1.37 MPaG, 66°C,4D,B	Lock Open
	150A-MUWC-xxx Pipe	1.37 MPaG, 66°C,4D,B	No change
	150A-MUWC-Fxxx LO Valve	1.37 MPaG, 66°C,4D,B	Lock Open
	150A-MUWC-xxx Pipe	Static Hd, 66°C,4D,B	No change
	Condensate Storage Tank,	66°C,4D, Non-seismic	No change

MUWC interface with the CRD System pump discharge piping.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	50A-CRD-034-S Pipe	18.63 MPaG, 20°C,4C,B	No change
	50A-CRD-F021 Valve MO	18.63 MPaG, 20°C,4C,B	No change
	50A-CRD-F022Valve	18.63 MPaG, 20°C,4C,B	No change
	50A-CRD-035-S Pipe	18.63 MPaG, 20°C,4C,B	No change
	50A-CRD-F023 Valve	18.63 MPaG, 20°C,4C,B	No change
Sheet 1	50A-MUWC-F10 Valve	1.37 MPaG, 66°C,4D,B	Lock Open
	50A-MUWC-xxx Pipe	Static Hd, 66°C,4D,B	No change
	Condensate Storage Tank,	66°C,4D, Non-seismic	No change

3MA.12 Makeup Water System Purified

3MA.12.1 Upgrade Description

The MUWP System is not upgraded due to the extensive nature of the piping distribution, but instead all valves between the interface of potential reactor pressure sources and the Condensate Storage Tank are changed to the lock open type. This provides a clear path for the release of pressure to the Condensate Storage Tank which is vented to atmosphere. The potential reactor pressure sources are the SLC System makeup seal, the RRS ten RIP casing makeup water connections, and shaft (RIP) inflatable seal capped connections. The piping and valves connected to the RIPs within the primary containment were upgraded to the URS design pressure.

3MA.12.2 Downstream Interfaces

The Makeup Water System Purified System has no further downstream system interfaces that could allow reactor pressure to proceed further than the MUWP System.

3MA.12.3 Upgraded Components — MUWP System

A detailed listing of the components upgraded for the MUWP System follows, including identification of those interfacing system components not requiring upgrade.

MAKEUP WATER SYSTEM (PURIFIED) Tier 2 Figure 9.2-5, Sheets 1, 2 and 3.

MUWP interface with the SLC System makeup seal and storage tank fill line.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	80A-SLC -F013 Check Valve	2.82 MPaG, 66°C, 4A,A	No change
	80A-MUWP-F019 Valve LO	1.37 MPaG, 66°C, 6D,C	No change
Sheet 2	80A-MUWP-F163 Valve LO	1.37 MPaG, 66°C, 6D,C	No change
	80A-MUWP-217 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	80A-MUWP-214 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	80A-MUWP-F162 Valve LO	1.37 MPaG, 66°C, 6D,C	No change
	100A-MUWP-180 Pipe	1.37 MPaG, 66°C, 6D,C	No change
Sheet 1	125A-MUWP-101 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	125A-MUWP-F101 Valve LO	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-602 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F602 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-601 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F601 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-FQ102 Flow Integr.	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-801 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F801 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-800 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F800 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-PX101 Press. Pt.	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-600 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	20A-MUWP-F600 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
	125A-MUWP-F100 Valve LO	1.37 MPaG, 66°C, 6D,C	No change
	125A-MUWP-102 Pipe	1.37 MPaG, 66°C, 6D,C	No change
	125A-MUWP-F102 Valve NC	1.37 MPaG, 66°C, 6D,C	No change
Sheet 3	150A-MUWP-xxx Pipe	1.37 MPaG, 66°C, 6D,C	No change
	150A-MUWP-Fxxx Check Valve	1.37 MPaG, 66°C, 6D,C	No change
	150A-MUWP-xxx Pipe	Static Head, 66°C, 6D,C	No change
	Condensate Storage Tank,	66°C, 4D,Non-seismic	No change

MUWP System interface with RRS for Reactor Internal Pump (RIP) casing makeup water.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	15A-RRS-502A-K Pipes	8.62 MPaG,302°C, 4A,As	No change
	15A-RRS-F504A-K Valves NC	8.62 MPaG,302°C, 4A,As	No change
	15A-MUWP-189-198 Pipes	2.82 MPaG, 66°C, 4D,C	Was 1.37 MPaG
	50A-MUWP-185 Pipe	2.82 MPaG, 66°C, 4D,C	Was 1.37 MPaG
	50A-MUWP-F142 Check Valve	2.82 MPaG,171°C, 3B,As	Was 1.37 MPaG
	50A-MUWP-184 Pipe	2.82 MPaG,171°C, 3B,As	Was 1.37 MPaG
	50A-MUWP-F141 Valves NC	2.82 MPaG,171°C, 3B,As	Was 1.37 MPaG
	50A-MUWP-183 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	80A-MUWP-181 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	80A-MUWP-F140 Valve LO	1.37 MPaG, 66°C, 4D,C	No change
	125A-MUWP-101 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	125A-MUWP-F101 Valve LO	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-602 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-F602 Valve NC	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-601 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-F601 Valve NC	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-FQ102 Flow Integr.	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-801 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-F801Valve NC	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-800 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-F800 Valve NC	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-PX101 Press. Pt.	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-600 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-F600 Valve NC	1.37 MPaG, 66°C, 4D,C	No change
	20A-MUWP-F100 Valve LO	1.37 MPaG, 66°C, 4D,C	No change
	125A-MUWP-102 Pipe	1.37 MPaG, 66°C, 4D,C	No change
	125A-MUWP-F102 Valve NC	1.37 MPaG, 66°C, 4D,C	No change
	150A-MUWP-xxx Pipe	1.37 MPaG, 66°C, 4D,C	No change
	150A-MUWP-xxx Pipe	1.37 MPaG, 66°C, 4D,C	No change
	150A-RRS-Fxxx Check Valve	1.37 MPaG, 66°C, 4D,C	No change
	150A-MUWP-xxx Pipe	Static Head, 66°C, 4D,C	No change
	Condensate Storage Tank,	66°C, 4D,Non-seismic	No change

3MA.13 Radwaste System

3MA.13.1 Upgrade Description

The Radwaste System LCW and HCW inlet piping header connects to each interfacing system at a valve. The header is not upgraded because it is an open pathway to the collector tanks. The dual LCW tanks rotate the fill mode one at a time through a level controlled AO valve at the inlet of each tank. The maintenance valve is a lock open type. The dual HCW tanks operate similarly to the LCW tanks.

3MA.13.2 Downstream Interfaces

Other systems are listed below that interface with the RW System and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

There are no downstream interfaces because the LCW and HCW collector tanks and associated piping are all at atmospheric pressure since the HVAC System tank exhaust vents each tank.

3MA.13.3 Upgraded Components — RW System

A detailed listing of the components upgraded for the RW System follows, including identification of those interfacing system components not requiring upgrade.

RADWASTE SYSTEM, GE Proprietary Drawing 103E1634, Sheets 1, 3 and 7.

RW LCW Subsystem interface with the RHR System.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	50A-RHR 129 Pipe	3.43 MPaG, 182°C,3B,As	No change
Sheet 3	150A-LCW-F006Valve	2.82 MPaG, 66°C,4D,B	Was 0.981 MPaG
	150A-LCW-CS Pipe	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS Pipe	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW Valve LO	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-F001A AO Valve	0.981 MPaG, 66°C,4D,B	No change
	* LCW Collector Tank A	0 MPaG, 66°C,4D,B	No change
	200A-LCW- Valve LO	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-F001B AO Valve	0.981 MPaG, 66°C,4D,B	No change
	* LCW Collector Tank B	0 MPaG, 66°C,4D,B	No change

* Each LCW collector tank has HVAC tank vent system exhausting tank air through filter to RW Stack.

RW HCW interface with the RHR System, Subsystem A.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	50A-RHR 018 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	150A-RHR-F026A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 7	150A-HCW-SS Valve	2.82 MPaG, 66°C, 4D,B	Was 0.981 MPaG
	150A-HCW-SS Pipe	0.981 MPaGg, 66°C,4D,B	No change
	150A-HCW-SS Valve LO	0.981 MPaG, 66°C,4D,B	No change
	150A-HCW-F003A Valve AO	0.981 MPaG, 66°C,4D,B	No change
	* HCW Collector Tank A	0 MPaG, 66°C,4D,B	No change
	150A-HCW-SS Valve LO	0.981 MPaG, 66°C,4D,B	No change
	150A-HCW-F002B Valve	0.981 MPaG, 66°C,4D,B	No change
	* HCW Collector Tank B	0 MPaG, 66°C,4D,B	No change

* Each HCW collector tank has HVAC tank vent system exhausting tank air through filter to RW Stack.

RW HCW interface with the RHR System, Subsystem B.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	50A-RHR 124 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F026B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 7	150A-HCW-SS Valve	2.82 MPaG, 66°C,4D,B	Was 0.981 MPaG
	150A-HCW-SS Pipe	0.981 MPaG, 66°C,4D,B	No change
	150A-HCW-SS Valve LO	0.981MPaG, 66°C,4D,B	No change
	150A-HCW-F003A Valve AO	0.981 MPaG, 66°C,4D,B	No change
	*HCW Collector Tank A	0 MPaG, 66°C,4D,B	No change
	150A-HCW-SS Valve LO	0.981 MPaG, 66°C,4D,B	No change
	150A-HCW-F002B Valve	0.981 MPaG, 66°C,4D,B	No change
	*HCW Collector Tank B	0 MPaG, 66°C,4D,B	No change

*Each HCW collector tank has HVAC tank vent system exhausting tank air through filter to RW Stack.

RW HCW interface with the RHR System, Subsystem C.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	50A-RHR 225 Pipe	2.82 MPaG, 182°C, 3B, As	Was 1.37 MPaG
	50A-RHR-F-26C Valve	2.82 MPaG, 182°C, 3B, As	Was 1.37 MPaG
Sheet 7	150A-HCW-SS Valve	2.82 MPaG, 66°C,4D,B	Was 1.37 MPaG
	150A-HCW-SS Pipe	0.981 MPaG, 66°C, 4D,B	No change
	150A-HCW-SS Valve LO	0.981 MPaG, 66°C, 4D,B	No change
	150A-HCW-F003A Valve AO	0.981 MPaG, 66°C, 4D,B	No change
	* HCW Collector Tank A	0 MPaG, 66°C, 4D,B	No change
	150A-HCW-SS Valve LO	0.981 MPaG, 66°C, 4D,B	No change
	150A-HCW-F002B Valve	0.981 MPaG, 66°C, 4D,B	No change
	* HCW Collector Tank B	0 MPaG, 66°C, 4D,B	No change

*Each HCW collector tank has HVAC tank vent system exhausting tank air through filter to RW Stack.

3MA.14 Condensate and Feedwater (CFS) System

3MA.14.1 Upgrade Description

The CFS System provides high pressure feedwater to the reactor, and all of the system is designed for high pressure except the condensate pumps suction. The high pressure design includes the condensate polishing (hollow fiber filters and demineralizers) units. The transition to low pressure occurs from the condensate suction into the LP condenser shell (hotwell). The hotwell is a low pressure sink. The last closed valve in the path from the reactor is the condensate pumps discharge check valve. The piping to the condensate pumps suction can remain below the URS design pressure because it connects to the low pressure heat sink hotwell. The maintenance block valves in the condensate pump suction lines were upgraded to a LOCK OPEN status.

3MA.14.2 Downstream Interfaces

None

3MA.14.3 Upgraded Components

The maintenance block valves in the condensate pump suction lines were upgraded to a LOCK OPEN status.

3MA.15 Sampling (SAM) System

3MA.15.1 Upgrade Description

The Sampling System receives water from several of the above systems, and an analysis, as presented below, resulted in not requiring any pressure upgrades. The following interfaces include all of the potential links of SAM to the reactor pressure, and since none of the individual portions need upgrading, SAM as a whole was not upgraded.

(1) **RHR Interface**: Samples can be taken downstream of the RHR heat exchanger, which is from a pipeline with a design pressure of 3.43 MPaG. The SAM System is designed for pressures at least as great as the point in the interfacing system where the sample is obtained. Therefore, the URS design pressure of 2.82 MPaG is exceeded and no upgrade required for this portion of SAM.

(2) **SLC Interface**: Samples can be taken from the SLC main tank, which is one of the low pressure sinks. Therefore, no upgrade is required for this portion of SAM.

(3) **CUW Interface**: Samples can be taken from the inlet and outlet of the filter demineralizer units, which are designed for full reactor pressure. The SAM System is designed for pressures at least as great as the point in the interfacing system where the sample is obtained. Therefore, the URS design pressure of 2.82 MPaG is exceeded and no upgrade required for this portion of SAM.

(4) **FPC Interface**: Samples can be taken from the inlet to the filter demineralizer units and from the heat exchanger outlet. The pipeline sample points have a design pressure of 1.57 MPaG ; however, this region of the FPC System did not need upgrading to the URS design pressure. Therefore, no upgrade is required for this portion of SAM.

(5) **NBS Interface**: Samples can be taken from the points within the NBS which are designed for full reactor pressure. The SAM System is designed for pressures at least as great as the point in the interfacing system where the sample is obtained. Therefore, the URS design pressure of 2.82 MPaG is exceeded and no upgrade required for this portion of SAM.

(6) **MUWP Interface**: Samples can be taken from a point within the MUWP System located in the turbine building that did not need upgrading to the URS design pressure. Therefore, no upgrade is required for this portion of SAM.

(7) **Rad Waste Interface**: Samples can be taken from the LCW and HCW collector tanks, which are low pressure sinks. Therefore, no upgrade is required for this portion of SAM.

3MA.15.2 Downstream Interfaces

None

3MA.15.3 Upgraded Components

None