3.7.2 **Seismic System Analysis**

This section applies to building structures that constitute primary structural systems (RB, FB, CB, and FWSC). The reactor pressure vessel (RPV) is not a primary structural component but, due to its dynamic interaction with the supporting structure, it is considered as another part of the primary system of the RB for the purpose of dynamic analysis. *Table 3.7-3 provides a summary of seismic analysis methods for primary building structures.*

3.7.2.1 **Seismic Analysis Methods**

Analysis can be performed using any of the following methods:

- time history
- response spectrum
  - singly- or multi-supported system with Uniform Support Motion (USM)
  - multi-supported system with independent support motion (ISM)
- static coefficient

3.7.2.1.1 **Time History Method**

The response of a multi-degree-of-freedom linear system subjected to external forces or uniform support excitations is represented by the following differential equations of motion in the matrix form:

$$[M]\{\ddot{u}\} + [C]\{\dot{u}\} + [K]\{u\} = \{P\} \qquad (3.7\text{-}1)$$

where,

[M]	=	mass matrix
[C]	=	damping matrix
[K]	=	stiffness matrix
{u}	=	mass matrix column vector of time-dependent relative displacements
$\{\dot{u}\}$	=	column vector of time-dependent relative velocities
$\{\ddot{u}\}$	=	column vector of time-dependent relative accelerations
{P}	=	column vector of time-dependent relative applied forces
	=	$-[M]\{\ddot{x}_g\}$ for support excitation in which $\{\ddot{x}_g\}$ is column vector of time-dependent support accelerations

The above equation can be solved by modal superposition or direct integration in the time domain, or by the complex frequency response method in the frequency domain. For the time domain solution, the numerical integration time step is sufficiently small to accurately define the dynamic excitation and to render stability and convergence of the solution up to the highest frequency (or

shortest period) of significance. For most of the commonly used numerical integration methods (such as Newmark β-method and Wilson θ-method), the maximum time step is limited to one-tenth of the shortest period of significance. The adequacy of the selected time step (Δt) is checked by ensuring that use of ½ of Δt does not change the response by more than 10%. For the frequency domain solution, the dynamic excitation time history is digitized with time steps no larger than the inverse of two times the highest frequency of significance and the frequency interval is selected to accurately define the transfer functions at structural frequencies within the range of significance.

The modal superposition method is used when the equation of motion (Equation 3.7-1) can be decoupled using the transformation,

$$\{u\}=[\phi]\{q\} \qquad (3.7\text{-}2)$$

where,

$[\phi]$	=	mode shape matrix; often mass normalized, i.e., $[\phi]^T[M][\phi] = [1]$
$\{q\}$	=	column vector of normal or generalized coordinates

Substituting Equation 3.7-2 into Equation 3.7-1 and multiplying each term by the transposition of the mode shape matrix results in the uncoupled equation of motion due to the orthogonality of the mode shapes (note that the orthogonality condition of the damping matrix is assumed). For systems subjected to base acceleration excitation, $\ddot{x}_g$, the equation of motion for the jth mode is

$$\ddot{q}_j + 2\lambda_j\omega_j\dot{q}_j + \omega_j^2 q_j = -\Gamma_j\ddot{x}_g \qquad (3.7\text{-}3)$$

where

q_j	=	generalized coordinate of jth mode
λ_j	=	damping ratio of jth mode, expressed as fraction of critical damping
ω_j	=	undamped circular frequency of jth mode
Γ_j	=	modal participation factor of jth mode
	=	$\{\phi_j\}^T[M]\{1\} / (\{\phi_j\}^T[M]\{\phi_j\})$

The final solution for each mode is obtained by the transformation from the generalized coordinates back to the physical coordinates. The total response is the superposition of the modal responses. All modes with frequencies up to the zero period acceleration (ZPA) frequency are included in the modal superposition and the residual rigid response due to the missing mass is accounted for in accordance with the methods described in Subsection 3.7.2.7.

The system equation of motion (Equation 3.7-1) can be solved directly using the direct integration method in the time domain without the need to revert to decoupling by the coordinate transformation for mode superposition.

The system equation of motion (Equation 3.7-1) can also be solved in the frequency domain using the complex frequency response method. This method requires that the transfer functions be determined first and the applied forces be transformed into frequency domain. The transfer functions can be computed directly from the system equations of motion or from the normal mode approach. The Fast Fourier Transform (FFT) algorithm is commonly used for the transformation between the time domain and frequency domain. To facilitate the FFT operation, the total number of digitized points of the excitation time history is a power of two, which can always be achieved by adding trailing zeros to the actual record. 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 in the discrete Fourier transform procedure.

For multi-supported systems subjected to ISM, the ISM method of analysis described in Subsection 3.7.2.1.2 can also be performed using the time history method.

The frequency domain solution is not used in the piping system response analysis.

3.7.2.1.2 **Response Spectrum Method**

a) Singly- or Multi-Supported System with USM

This method, applicable to singly-supported systems or multi-supported systems with USMs, is the modal superposition method described in Subsection 3.7.2.1.1 except that only the peak values of the solutions of the decoupled modal equations (Equation 3.7-3) are obtained. The maximum response in terms of the generalized coordinate for jth mode is

$$(q_j)_{max} = \Gamma_i\left(\frac{S_{aj}}{\omega_j^2}\right) \tag{3.7-4}$$

where S_{aj} is the spectral acceleration of the input spectrum corresponding to frequency ω_j for a specified damping factor. The maximum displacement of node i for mode j in the physical coordinate is

$$(\square_{ij})_{max} = \phi_{ij}(q_j)_{max} \tag{3.7-5}$$

The maximum modal displacement is then used to determine other modal response quantities, such as forces. The applicable methods of modal response combination are defined in Subsection 3.7.2.7.

b) Multi-Supported System with ISMs

This method is applicable to linear dynamic systems which are supported at two or more locations and have different excitations applied at each support. The governing equation of motion is expressed in the following partitioned matrix form:

$$\begin{bmatrix} M_a & O \\ O & M_s \end{bmatrix}\begin{Bmatrix} \ddot{U}_a \\ \ddot{U}_s \end{Bmatrix} + \begin{bmatrix} C_{aa} & C_{as} \\ C_{as} & C_{ss} \end{bmatrix}\begin{Bmatrix} \dot{U}_a \\ \dot{U}_s \end{Bmatrix} + \begin{bmatrix} K_{aa} & K_{as} \\ K_{as} & K_{ss} \end{bmatrix}\begin{Bmatrix} U_a \\ U_s \end{Bmatrix} = \begin{Bmatrix} F_a \\ F_s \end{Bmatrix} \tag{3.7-6}$$

where

U_a = displacements of active (unsupported) degrees of freedom

U_s = specified displacements of support points

M_a and M_s = diagonal mass matrices associated with active degrees of freedom and support points, respectively

O = null matrix

C_{aa} and K_{aa} = damping and stiffness matrices, respectively, associated with active degrees of freedom

C_{ss} and K_{ss} = support forces caused by unit velocities and displacements of supports, respectively

C_{as} and K_{as} = damping and stiffness matrices, respectively, denoting the coupling forces developed in the active degrees of freedom by the motion of the supports and vice versa

F_a = prescribed external 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 (•) above a variable in Equation 3.7-6. Also, the contributions of the fixed degrees of freedom have been removed in the equation. Equation 3.7-6 can be separated into two sets of equations. The first set of equations can be written as:

$$[M_s]\{\ddot{U}_s\}+[C_{ss}]\{\dot{U}_s\}+[K_{ss}]\{U_s\}+[C_{as}]\{\dot{U}_a\}+[K_{as}]\{U_a\} = \{F_s\} \tag{3.7-7}$$

and the second set as:

$$[M_a]\{\ddot{U}_a\}+[C_{aa}]\{\dot{U}_a\}+[K_{aa}]\{U_a\}+\{C_{as}]\{\dot{U}_s\}+[K_{as}]\{U_s\} = \{F_a\} \tag{3.7-8}$$

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 for the support point reaction forces (F_s). Analysis can be performed using either the time history method or response spectrum method. Additional considerations associated with the ISM response spectrum method of analysis are given in Subsection 3.7.3.9.

The response spectrum method is not used for seismic response analysis of primary building structures.

3.7.2.1.3 **Static Coefficient Method**

This is an alternative method of analysis that allows a simpler technique in return for added conservatism. This method does not require determination of natural frequencies. The response loads are determined statically by multiplying the mass value by a static coefficient equal to 1.5 times the maximum spectral acceleration at appropriate damping value of the input response spectrum. A static coefficient of 1.5 is intended to account for the effect of both multi-frequency excitation and multi-mode response for linear frame-type structures, such as members physically similar to beams and columns, which can be represented by a simple model similar to those shown to produce conservative results (References 3.7-13 and 3.7-14). A factor of less than 1.5 is used if justified. If the fundamental frequency of the structure is known, the highest spectral acceleration value at or beyond the fundamental frequency can be multiplied by a factor of 1.5 to determine the response. A factor of 1.0 instead of 1.5 can be used if the component is simple enough such that it behaves essentially as a single-degree-of-freedom system. When the component is rigid, it is analyzed statically using the ZPA as input. Structures, systems, and components are considered rigid when the fundamental frequency is equal to or greater than the frequency at which the input response spectrum returns to approximately the ZPA. Relative displacements between points of support are also considered and the resulting response is combined with the response calculated using the equivalent static method. *The static coefficient method is not used for primary building structures.*

3.7.2.2 **Natural Frequencies and Responses**

Natural frequencies and SSE responses of Category I buildings are presented in Appendix 3A.

3.7.2.3 **Procedures Used for Analytical Modeling**

The mathematical model of the structural system is developed as a stick model for seismic response analysis of primary building structures. The details of the model are determined by the complexity of the actual systems and the information required from the analysis. In the primary structural system model, the following subsystem decoupling criteria are applicable:

- If $R_m < 0.01$, decoupling can be done for any R_f.
- If $0.01 \le R_m \le 0.1$, decoupling can be done if $R_f \le 0.8$ or $R_f \ge 1.25$.
- If $R_m > 0.1$, a subsystem model should be included in the primary system model

where R_m (mass ratio) and R_f (frequency ratio) are defined as:

R_m = total mass of the supported subsystem/total mass of the supporting system

R_f = fundamental frequency of the supported subsystem/dominant frequency of the 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 dynamic input for the former is obtained by the analysis of the latter. One important exception to this procedure is the RPV, which is considered as a subsystem but is analyzed using a coupled model with the primary structure.

In general, three-dimensional models are used with six degrees of freedom assigned to each mass (node) point (i.e., three translational and three rotational). Some dynamic degrees of freedom, such as rotary inertia, can be neglected, since their contribution to the total kinetic energy of the system is small compared to the contribution from translational inertia. A two- or one-dimensional model is used if the directional coupling effect is negligible. Coupling between two horizontal motions occurs when the center of mass, the centroid, and the centroid of rigidity do not coincide. The degree of coupling depends on the amount of eccentricity and the ratio of uncoupled torsional frequency to the uncoupled lateral frequency. Structures are generally designed to keep eccentricities as small as practical to minimize lateral/torsional coupling and torsional response.

Nodal points are selected to coincide with the locations of large masses, such as floors or at heavy equipment supports, at all points where significant changes in physical geometry occur, and locations where the responses are of interest. *The mass properties in the model include all contributions expected to be present at the time of dynamic excitation, such as dead weight, fluid weight, attached piping and equipment weight, and appropriate part (25% of floor live load or minimum 75% of roof snow load, as applicable) of the live load. For design, 100% of roof snow load is used. The hydrodynamic effects of any significant fluid mass interacting with the structure are considered in modeling of the mass properties.* Masses are lumped to node points. Alternatively, the consistent mass formulation is used. The number of masses or dynamic degrees of freedom is considered adequate when additional degrees of freedom do not result in more than a 10% increase in response. Alternatively, the number of dynamic degrees of freedom is no less than twice the number of modes below the cutoff frequency in Subsection 3.7.2.1.1. For the stick models of the primary building structures, the number of dynamic degrees of freedom is no less than twice the number of modes below 50 Hz.

The RPV, including its major internal components, is analyzed together with the primary structure using a coupled RPV and supporting structural model. The RPV model is constructed following the general modeling procedures described above for the primary structures. The RPV model includes major internal components such as the fuel assemblies, control rod guide tubes, control rod drive (CRD) housings, shroud, chimney, standpipes, and steam separators. Stiffness of light components such as in-core guide tubes and housings, spargers, and their supply headers are not included in the model, but their masses are considered. For the dynamic responses of these components, floor response spectra generated from system analysis is used for subsystem analysis. Mass points are located at all points of interest such as anchors, supports, and points of discontinuity. 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. The presence of fluid and other structural components introduces a dynamic coupling effect. The hydrodynamic coupling effects caused by horizontal excitation are taken into consideration by including coupling fluid masses lumped to appropriate structural nodes at the same elevation. The details of the hydrodynamic mass derivation are given in Reference 3.7-6. In the vertical excitation, the hydrodynamic coupling effects are assumed to be negligible and the fluid masses are lumped to appropriate structural locations.

3.7.2.4 **Soil-Structure Interaction**

The seismic soil-structure interaction (SSI) analyses of the Category I buildings performed for a range of soil conditions are presented in Appendix 3A.

The site-specific SSI analyses for the RB/FB and CB were performed using either the direct method or the modified subtraction method of the SASSI2010 computer program. The subtraction method of the SASSI2010 program was not used. The SSI analysis approach and the structural models are the same as presented in Appendix 3A.

The FWSC is essentially a surface founded structure in Subsection 3.7.1.1, and there are no embedded walls for the FWSC. Therefore, the backfill requirements surrounding Seismic Category I structures are not applicable to FWSC embedded basemat (embedded 2.35 m (7.7 feet)). The FWSC is founded on fill concrete which meets the requirements for backfill underneath Seismic Category I structures. Therefore, there is no site-specific SSI analysis performed for the FWSC.

3.7.2.4.1 **Fermi 3 Site-Specific Soil-Structure Interaction Analysis**

This subsection presents the Fermi 3 site-specific SSI analyses performed in accordance with SRP 3.7.2 for the Seismic Category I RB/FB and CB. The Fermi 3 site-specific FIRS developed in Subsection 3.7.1 is in accordance with Regulatory Guide 1.208 and NRC Interim Staff Guidance(DC/COL-ISG-017) for ensuring hazard-consistent seismic input for site response and soil-structure interaction analyses. The Fermi 3 site-specific FIRS developed in Subsection 3.7.1 are fully enveloped, in all cases, by the ESBWR CSDRS. Therefore, the Fermi 3 site-specific SSI analyses were not performed to address an exceedance of the CSDRS by the FIRS; rather, the

Fermi 3 site-specific SSI analyses were performed to address the following Fermi 3 site-specific conditions:

- Partial embedment in the Bass Islands Group bedrock of the RB/FB and CB Seismic Catagory I structures, as shown on Figure 2.5.4-202 and Figure 2.5.4-203, to confirm that the design is applicable for this case.
- To demonstrate that the requirements for the backfill surrounding Seismic Catagory I structures can be neglected for RB/FB and CB with the RB/FB and CB partially embedded in the bedrock at the Fermi 3 site.

The Fermi 3 site-specific SSI analyses follow the same methodology used in the DCD for SSI analyses for the ESBWR Standard Plant using the SASSI2010 computer program. The SASSI2010 structural models are developed from the lumped-mass stick models coupled with the Fermi 3 site-specific strain compatible dynamic subsurface properties developed in Subsection 3.7.1. In the SASSI2010 model for the Fermi 3 site-specific SSI analyses, the RB/FB and CB are modeled as partially embedded into the Bass Islands Group bedrock. Cases with and without backfill above the top of the Bass Islands Group bedrock at Elevation 168.2 m (552.0 ft) NAVD 88 surrounding the RB/FB and CB have been considered. Fill concrete is used to backfill the gap between the RB/FB and CB and excavated bedrock up to the top of Bass Islands Group bedrock at Elevation 168.2 m (552.0 ft) NAVD 88, as shown on Figure 2.5.4-202 and Figure 2.5.4-203.

As shown in Table 3A.6-1, there are some models with minor modifications to evaluate the modeling effects. For the Fermi 3 SSI analyses, the most basic model, "Base" is applied.The Base Model is for uncracked concrete.

The site-specific SSI analyses results are presented and compared with the seismic responses in the following subsections to confirm the applicability of the ESBWR Standard Plant for the RB/FB and CB. Lateral wall pressures due to seismic loadings are evaluated in Subsection 3.8.4. In addition, the foundation stability and the dynamic bearing pressure demands are evaluated in Subsection 3.8.5 for the RB/FB and CB based on the Fermi 3 site-specific SSI analyses results.

3.7.2.4.1.1. **Strain Compatible Dynamic Subsurface Material Properties**

The geology of the Fermi 3 site is discussed in detail in Subsection 2.5.1. The subsurface materials encountered and the engineering properties of subsurface materials at Fermi 3 site are discussed in detail in Subsection 2.5.4.

In accordance with SRP 3.7.2, three subsurface material profiles, a best estimate (BE) profile, a lower bound (LB) profile, and an upper bound (UB) profile, were developed and used in the SSI analyses to account for variability in the subsurface materials properties at the Fermi 3 site. The development of the Fermi 3 site-specific strain compatible dynamic subsurface material properties associated with the BE, LB, and UB profiles is discussed in Subsection 3.7.1.3. The strain compatible dynamic subsurface material properties of the BE, LB, and UB subsurface profiles used in the Fermi 3 site-specific SSI analyses are provided in Table 3.7.1-206 through Table 3.7.1-211. To demonstrate that the backfill surrounding the Seismic Category I RB/FB and CB above the top of the Bass Islands Group bedrock can be neglected, separate BE, LB, and UB subsurface profiles were used for the Fermi 3 SSI analyses that separately include and do not include backfill that will be placed during construction above the Bass Islands Group bedrock at Elevation 168.2 m (552.0 ft) NAVD 88 to finished ground level grade at Elevation 179.6 m (589.3 ft) NAVD 88.

3.7.2.4.1.2. **FIRS Compatible Ground Motion Time History**

Subsection 3.7.1.1.5 describes development of the Fermi 3 site-specific ground motion time histories used in the SSI analyses. The Fermi 3 site-specific SSI analyses used three orthogonal components (two horizontal and one vertical) of a single ground motion time history that were developed to be in-column motions at the bottom of RB/FB and CB basemat levels. The site-specific ground motion time histories are compatible with the SSI FIRS developed in Subsection 3.7.1 and are used as input motions applied at the bottom of RB/FB and CB basemat levels in the Fermi 3 site-specific SSI analyses.

3.7.2.4.1.3. **Soil-Structure Interaction Analysis Method**

The Fermi 3 site-specific SSI analysis follows the methodology presented in Section 3A.5.2 using either the direct method or the modified subtraction method of the SASSI2010 computer program. The method of analysis used for site-specific SSI and SSSI analyses are shown in Table 3.7.2-201 and Table 3.7.2-202. The subtraction method of SASSI2010 program is not used for any of the site-specific SSI analyses.

As shown in Table 3.7.2-201 and Table 3.7.2-202, modified subtraction method was used for SSI analysis of the RB/FB with engineered backfill and SSSI analysis of the CB and FWSC. When using modified subtraction method, the results from the modified subtraction method models were benchmarked against the results from the direct method models to ensure that the appropriate modified subtraction method models were being utilized. These benchmark analyses were performed using the site-specific soil properties and input motions using full, half, or quarter models as follows:

- For benchmark analyses of the RB/FB, quarter model of the RB/FB was used.
- For benchmark analyses of the CB, full model of the CB was used.
- For the benchmark analyses of the FWSC, half model of the CB was used.

The SASSI2010 program uses finite elements with complex moduli for modeling the structure and foundation properties and is based on the frequency domain complex response method. The lumped mass-beam model described in Section 3A.5.1 is coupled with the soil model using site-specific strain compatible dynamic subsurface properties in SASSI2010. Structural responses in terms of accelerations, forces, and moments are computed directly. Floor response spectra are obtained from the calculated response acceleration time histories.

The SSI analyses for the three directional ground motion time history components are performed separately. The maximum co-directional responses for each of the three ground motion time history components are combined using the algebraic sum in the time domain. The SASSI2010 RB/FB and CB structural models are described in Subsection 3.7.2.4.1.4.

3.7.2.4.1.4. **Soil-Structure Interaction Analysis Structural Models**

The Fermi 3 site-specific SSI SASSI2010 structural models for the RB/FB and CB are constructed from the building stick models coupled with the foundation finite element model in the manner described in Subsection 3A.7.3. The RB/FB and CB seismic models are shown in Figure 3A.7-4 and Figure 3A.7-6, respectively. The overall Fermi 3 site-specific SSI SASSI2010 models are shown on Figure 3.7.2-201 through Figure 3.7.2-203 and Figure 3.7.2-203a through Figure 3.7.2-203c for the RB/FB, and on Figure 3.7.2-204 through Figure 3.7.2-206 and Figure 3.7.2-206a through Figure 3.7.2-206c for the CB. The Fermi 3 site-specific SSI SASSI2010 structural model configurations are the same as those shown on Figure 3A.7-8 through Figure 3A.7-10 for the RB/FB and Figure 3A.7-11 through Figure 3A.7-13 for the CB, except that the vertical and horizontal spacing of the wall and basemat nodes between grade (Elevation 4.5m in Figure 3.7.2-203 and Figure 3.7.2-206) and the foundation basemat, (i.e., the embedded portion of the RB/FB and CB), are adjusted for a closer match with the Fermi 3 site-specific subsurface profile layers and to size elements to pass frequencies up to 50Hz.

The subsurface layer thicknesses used in the RB/FB and CB Fermi 3 site-specific SSI analyses satisfy the SASSI2010 requirement that the subsurface layer thicknesses be limited to less than 20 percent of the shear wave length of the subsurface material the wave is passing through at the highest frequency of interest in the analysis (f_n). For the Fermi 3 site-specific SSI analyses, f_n is 50 Hz, except for SSI models with engineered backfill and LB subsurface profile. As shown in Table 3.7.2-201 and Table 3.7.2-202, the SSI and SSSI analysis models with engineered backfill have considered only UB and LB subsurface profiles. To keep the LB model within SASSI2010 computer code capability, the LB model layer thicknesses and mesh dimensions are kept the same as those for the corresponding UB model. For the LB models, f_n is about 19 Hz.

The SASSI2010 model X-direction and Y-direction represent plant north-south (NS) and east-west (EW) directions, respectively, at the Fermi 3 site. The SASSI2010 model Z-direction represents the vertical direction.

3.7.2.4.1.5. **Soil-Structure Interaction Analysis Cases**

The Fermi 3 site-specific SSI analyses cases are summarized in Table 3.7.2-201 and Table 3.7.2-202 for the RB/FB and CB, respectively. To account for variability in the subsurface material properties at the Fermi 3 site, the BE, LB, and UB profiles were used in the site-specific SSI analyses. Each analysis case consists of three directions of excitation (two horizontal and one vertical) applied separately to the Fermi 3 site-specific SSI SASSI2010 model. The calculated resulting co-directional floor response spectra in the X-, Y-, and Z-directions are combined using the SRSS method. The resulting co-directional structural loads responses from each direction of excitation for each case are combined using algebraic sums in the time domain to obtain the total response.

3.7.2.4.1.6. **Soil-Structure Interaction Analysis Results**

In the following subsections, the results of the Fermi 3 site-specific SSI analyses are presented and compared at key locations with the seismic design envelopes specified in Subsection 3A.9 for maximum seismic structural loads and floor response spectra.

3.7.2.4.1.6.1. **SSI Enveloping Maximum Structural Loads**

For the RB/FB model, the enveloping seismic loads from the Fermi 3 site-specific SSI analyses (herein called Fermi 3 site-specific SSI enveloping seismic loads) are presented in Table 3.7.2-203a through Table 3.7.2-203e.

The Fermi 3 site-specific SSI enveloping seismic loads for the RB/FB stick model are presented in Table 3.7.2-203a. The Fermi 3 site-specific SSI enveloping seismic loads are compared with the enveloping seismic loads provided in Table 3A.9-1a for the RB/FB stick model. Table 3.7.2-203a also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping seismic loads to the enveloping seismic loads for the RB/FB stick model. Table 3.7.2-203a shows that the Fermi 3 site-specific SSI enveloping seismic loads for the RB/FB stick model are lower than the enveloping seismic loads, with a maximum percentage ratio of approximately 67 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping seismic load is approximately 67 percent of the enveloping seismic loads used in the ESBWR Standard Plant for the RB/FB.

The Fermi 3 site-specific SSI enveloping seismic loads for the Reinforced Concrete Containment Vessel (RCCV) stick model are presented in Table 3.7.2-203b. The Fermi 3 site-specific SSI enveloping seismic loads are compared with the enveloping seismic loads provided in *Table 3A.9-1b* for the RCCV stick model. Table 3.7.2-203b also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping seismic loads to the enveloping seismic loads for the RCCV stick model. Table 3.7.2-203b shows that the Fermi 3 site-specific SSI enveloping seismic loads for

the RCCV stick model are lower than the enveloping seismic loads, with a maximum percentage ratio of approximately 68 percent. This indicates that the greatest Fermi 3 site specific SSI enveloping seismic load is approximately 68 percent of the enveloping seismic loads used in the ESBWR Standard Plant for the RCCV.

The Fermi 3 site-specific SSI enveloping seismic loads for the Vent Wall/Pedestal stick model are presented in Table 3.7.2-203c. The Fermi 3 site-specific SSI enveloping seismic loads are compared with the enveloping seismic loads provided in *Table 3A.9-1c* for the Vent Wall/Pedestal stick model. Table 3.7.2-203c also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping seismic loads to the enveloping seismic loads for the Vent Wall/Pedestal stick model. Table 3.7.2-203c shows that the Fermi 3 site-specific SSI enveloping seismic loads for the Vent Wall/Pedestal stick model are lower than the enveloping seismic loads, with a maximum percentage ratio of approximately 51 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping seismic load is approximately 51 percent of the enveloping seismic loads used in the ESBWR Standard Plant for the Vent Wall/Pedestal.

The Fermi 3 site-specific SSI enveloping seismic loads for the Reactor Shield Wall (RSW) stick model are presented in Table 3.7.2-203d. The Fermi 3 site-specific SSI enveloping seismic loads are compared with the enveloping seismic loads provided in *Table 3A.9-1d* for the RSW stick model. Table 3.7.2-203d also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping seismic loads to the enveloping seismic loads for the RSW stick model. Table 3.7.2-203d shows that the Fermi 3 site-specific SSI enveloping seismic loads for the RSW stick model are lower than the enveloping seismic loads, with a maximum percentage ratio of approximately 60 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping seismic load is approximately 60 percent of the enveloping seismic loads used in the ESBWR Standard Plant for the RSW.

The Fermi 3 site-specific SSI enveloping seismic loads for the Reactor Pressure Vessel (RPV) stick model are presented in Table 3.7.2-203e. The Fermi 3 site-specific SSI enveloping seismic loads are compared with the SSI analysis enveloping seismic loads for the RPV stick model in *Table 3A.9-1e*. Table 3.7.2-203e presents the percentage ratio of the Fermi 3 site-specific SSI enveloping seismic loads to the SSI analysis enveloping seismic loads for the RPV stick model. Table 3.7.2-203e shows that the Fermi 3 site-specific SSI enveloping seismic loads for the RPV stick model are lower than the SSI analysis enveloping seismic loads, with a maximum percentage ratio of approximately 86 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping seismic load is approximately 86 percent of the enveloping seismic loads actually used in the ESBWR Standard Plant for the RPV.

For the CB model, the Fermi 3 site-specific SSI enveloping seismic loads for CB stick model are presented in Table 3.7.2-204. The Fermi 3 site-specific SSI enveloping seismic loads are compared with the enveloping seismic loads provided in *Table 3A.9-1f* for the CB stick model. Table 3.7.2-204 also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping seismic loads to the

enveloping seismic loads for the CB stick model. Table 3.7.2-204 shows that the Fermi 3 site-specific SSI enveloping seismic loads for the CB stick model are lower than the enveloping seismic loads, with a maximum percentage ratio of approximately 72 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping seismic load is approximately 72 percent of the enveloping seismic loads used in the ESBWR Standard Plant for the CB.

The vertical loads are expressed in terms of enveloping absolute acceleration. For the RB/FB model, the enveloping maximum vertical acceleration from Fermi 3 site-specific SSI analyses based on the BE, LB, and UB subsurface profiles (herein called Fermi 3 site-specific SSI enveloping maximum vertical accelerations) are presented in Table 3.7.2-205a through Table 3.7.2-205e.

The Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the RB/FB stick model are presented in Table 3.7.2-205a. The Fermi 3 site-specific SSI enveloping maximum vertical accelerations are compared with the enveloping maximum vertical accelerations provided in Table 3A.9-3a for the RB/FB stick model. Table 3.7.2-205a also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping maximum vertical accelerations to the enveloping maximum vertical accelerations for the RB/FB stick model. Table 3.7.2-205a shows that the Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the RB/FB stick model are lower than the enveloping maximum vertical accelerations, with a maximum percentage ratio of approximately 46 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping maximum vertical acceleration is approximately 46 percent of the enveloping maximum vertical acceleration used in the ESBWR Standard Plant for the RB/FB.

The Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the RCCV stick model are presented in Table 3.7.2-205b. The Fermi 3 site-specific SSI enveloping maximum vertical accelerations are compared with the enveloping maximum vertical accelerations provided in *Table 3A.9-3b* for the RCCV stick model. Table 3.7.2-205b also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping maximum vertical accelerations to the enveloping maximum vertical accelerations for the RCCV stick model. Table 3.7.2-205b shows that the Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the RCCV stick model are lower than the enveloping maximum vertical accelerations, with a maximum percentage ratio of approximately 41 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping maximum vertical acceleration is approximately 41 percent of the enveloping maximum vertical acceleration used in the ESBWR Standard Plant for the RCCV.

The Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the Vent Wall/Pedestal stick model are presented in Table 3.7.2-205c. The Fermi 3 site-specific SSI enveloping maximum vertical accelerations are compared with the enveloping maximum vertical accelerations provided in *Table 3A.9-3c* for the Vent Wall/Pedestal stick model. Table 3.7.2-205c also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping maximum vertical accelerations to the enveloping maximum vertical accelerations for the Vent Wall/Pedestal stick model. Table

3.7.2-205c shows that the Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the Vent Wall/Pedestal stick model are lower than the enveloping maximum vertical accelerations, with a maximum percentage ratio of approximately 44 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping maximum vertical acceleration is approximately 44 percent of the enveloping maximum vertical acceleration used in the ESBWR Standard Plant for the Vent Wall/Pedestal.

The Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the RSW stick model are presented in Table 3.7.2-205d. The Fermi 3 site-specific SSI enveloping maximum vertical accelerations are compared with the enveloping maximum vertical accelerations provided in *Table 3A.9-3d* for the RSW stick model. Table 3.7.2-205d also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping maximum vertical accelerations to the enveloping maximum vertical accelerations for the RSW stick model. Table 3.7.2-205d shows that the Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the RSW stick model are lower than the enveloping maximum vertical accelerations, with a maximum percentage ratio of approximately 45 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping maximum vertical acceleration is approximately 45 percent of the enveloping maximum vertical acceleration used in the ESBWR Standard Plant for the RSW.

The Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the RB/FB Flexible Slab Oscillators are presented in Table 3.7.2-205e. The Fermi 3 site-specific SSI enveloping maximum vertical accelerations are compared with the enveloping maximum vertical accelerations provided in *Table 3A.9-3b* for the RB/FB Flexible Slab Oscillators. Table 3.7.2-205e also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping maximum vertical accelerations to the enveloping maximum vertical accelerations for the RB/FB Flexible Slab Oscillators. Table 3.7.2-205e shows that the Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the RB/FB Flexible Slab Oscillators are lower than the enveloping maximum vertical accelerations, with a maximum percentage ratio of approximately 75 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping maximum vertical acceleration is approximately 75 percent of the enveloping maximum vertical acceleration used in the ESBWR Standard Plant for the RB/FB Flexible Slab Oscillators. A conservative assessment of the responses for RB/FB horizontal oscillators showed that the Fermi 3 response spectra for these oscillators would be bounded by the response spectra used for design.

For the CB stick model, the Fermi 3 site-specific SSI enveloping maximum vertical accelerations are presented in Table 3.7.2-206. The Fermi 3 site-specific SSI enveloping maximum vertical accelerations for the CB stick model are presented in Table 3.7.2-206. The SSI enveloping maximum vertical accelerations are compared with the enveloping maximum vertical accelerations provided in *Table 3A.9-3g* for the CB stick model. Table 3.7.2-206 also presents the percentage ratio of the Fermi 3 site-specific SSI enveloping maximum vertical accelerations to the enveloping maximum vertical accelerations for the CB stick model. Table 3.7.2-206 shows that the Fermi 3

site-specific SSI enveloping maximum vertical accelerations for the CB stick model are lower than the enveloping maximum vertical accelerations, with a maximum percentage ratio of approximately 75 percent. This indicates that the greatest Fermi 3 site-specific SSI enveloping maximum vertical acceleration is approximately 75 percent of the enveloping maximum vertical acceleration used in the ESBWR Standard Plant for the CB.

3.7.2.4.1.6.2. **Comparison of the Site-Specific SSI Floor Response Spectra**

The site-specific floor response spectra for the BE, LB, and UB subsurface profiles are compared with the enveloping floor response spectra at 5 percent damping in Subsection 3A.9.2.

For the RB/FB model, the floor response spectra at 5 percent damping obtained from Fermi 3 site-specific SSI analyses (herein called Fermi 3 site-specific SSI floor response spectra at 5 percent damping) are shown on Figure 3.7.2-207a through Figure 3.7.2-207f for the X-direction, on Figure 3.7.2-208a through Figure 3.7.2-208f for the Y-direction, and on Figure 3.7.2-209a through Figure 3.7.2-209f for the vertical direction. The Fermi 3 site-specific SSI floor response spectra at 5 percent damping are compared with the Subsection 3A.9.2 enveloping floor response spectra at 5 percent damping on Figure 3.7.2-207a through Figure 3.7.2-209f (solid black lines). The Fermi 3 site-specific SSI floor response spectra at 5 percent damping at the locations presented in Subsection 3A.9.2 for the RB/FB model are considerably lower than the enveloping floor response spectra at 5 percent damping, indicating that the ESBWR Standard Plant for the RB/FB is acceptable at the Fermi 3 site.

For the CB model, Fermi 3 site-specific SSI floor response spectra at 5 percent damping are shown on Figure 3.7.2-210a and Figure 3.7.2-210b for the X-direction, on Figure 3.7.2-211a and Figure 3.7.2-211b for the Y-direction, and on Figure 3.7.2-212a and Figure 3.7.2-212b for the vertical direction. The Fermi 3 site-specific SSI floor response spectra at 5 percent damping are compared with the Subsection 3A.9.2 enveloping floor response spectra at 5 percent damping as shown on Figure 3.7.2-210a through Figure 3.7.2-212b (solid black lines). The Fermi 3 site-specific SSI floor response spectra at 5 percent damping at the locations presented in Subsection 3A.9.2 in the CB model are considerably lower than the enveloping floor response spectra at 5 percent damping, indicating that the ESBWR Standard Plant design for the CB is acceptable at the Fermi 3 site.

3.7.2.4.1.7. **Conclusions**

The Fermi 3 site-specific SSI analyses for the RB/FB and CB consider partial embedment into the Bass Islands Group bedrock. Cases both with and without taking credit for the lateral support of the backfill located above the top of Bass Islands Group bedrock (Elevation 168.2 m [552 ft] NAVD 88, Table 2.5.4-201) have been analyzed. These results show the following:

- That seismic forces, floor response spectra, and accelerations are significantly less than for the DCD design values for the ESBWR Standard Plant based on the CSDRS.

- That the factors of safety for sliding and overturning are significantly greater than the required factor of safety of 1.1 in SRP 3.8.5.
- That the dynamic bearing demands are much smaller than the allowable dynamic bearing capacity on the Bass Islands Group dolomite presented in Table 2.5.4-227.

The results from the Fermi 3 site-specific SSI analyses show that the seismic forces in members, floor response spectra, and acceleration are bounded by values for both the RB/FB and CB. In addition, Subsection 3.8.5 demonstrates that the actual factors of safety for overturning and sliding are greater than the required factors of safety.

The Fermi 3 site-specific SSI maximum soil dynamic bearing demands for the RB/FB and CB are less than the maximum dynamic bearing demands (Subsection 3.8.5). Thus, the foundation design is not impacted and is adequate.

The Fermi 3 site-specific lateral seismic soil pressures from SSI and SSSI analyses for the RB/FB and the CB are shown on Figure 3.8.4-201a through Figure 3.8.4-201h, Figure 3.8.4-202a through Figure 3.8.4-202d, and Figure 3.8.4-203a and Figure 3.8.4-203b. The lateral seismic soil pressures on embedded portions of the exterior walls exceed the lateral seismic soil pressures reported at some locations. The wall pressures are within the capacity of ESBWR DCD wall designs (Presented in Subsection 3.8.4).

Based on the Fermi 3 site-specific SSI analyses, the following conclusions apply to the Fermi 3 site:

- The DCD standard plant design (ESBWR Standard Plant) is applicable to the RB/FB and CB Seismic Category I structures at the Fermi 3 site with partial embedment into bedrock, considering cases that include and neglect the contribution of the surrounding backfill.
- The backfill requirements for the backfill above the top of the Bass Islands Group bedrock (Elevation 168.2 m [552 ft] NAVD 88) that surrounds the embedded walls of the Fermi 3 Seismic Category I structures are shown to be unnecessary. Therefore, the backfill above the top of the Bass Islands Group bedrock is not Seismic Category I backfill.
- The following Fermi 3 site-specific SSI dynamic responses using the SSI FIRS and the BE, LB, and UB subsurface profiles are less than the corresponding dynamic responses using the CSDRS:
  - Fermi 3 site-specific SSI enveloping seismic loads are less than the enveloping seismic loads. The Fermi 3 site-specific SSI enveloping seismic loads are a maximum of 86 and 72 percent of the values for the RB/FB and CB, respectively.
  - Fermi 3 site-specific SSI enveloping maximum vertical accelerations are less than the enveloping maximum vertical accelerations. The Fermi 3 site-specific SSI enveloping maximum vertical accelerations are a maximum of 75 and 75 percent of the values for the RB/FB and CB, respectively.

- Fermi 3 site-specific SSI floor response spectra are considerably less than the enveloping floor response spectra at the same locations.

- Lateral soil pressures on the embedded portions of the exterior walls exceed the lateral soil pressures reported at some locations. The wall pressures are within the capacity of ESBWR DCD wall designs (presented in Subsection 3.8.4).
- The Fermi 3 site-specific foundation stability (sliding and overturning) evaluation was performed without taking credit for the backfill located above the top of the Bass Islands Group bedrock that surrounds the embedded walls of the RB/FB and CB, and by neglecting the side frictional resistance along the sides of the basemats and the shear keys beneath the basemats. The Fermi 3 site-specific foundation stability evaluation demonstrated that the minimum Fermi 3 site-specific factors of safety for sliding and overturning for the RB/FB and CB are 1.10 for sliding and 1,733 for overturning (presented in Subsection 3.8.5).
- The Fermi 3 RB/FB and CB are stable against floatation with a minimum factor of safety of 1.86 (presented in Subsection 3.8.5).
- The dynamic bearing demands from the Fermi 3 site-specific SSI analyses are considerably below the allowable dynamic bearing capacities for the Bass Islands Group bedrock at the Fermi 3 site (presented in Subsection 3.8.5).

3.7.2.5 **Development of Floor Response Spectra**

Floor response spectra are developed from the primary structural dynamic analysis using the time history method. A direct spectra generation without resorting to time history in accordance with the method referenced in Reference 3.7-7 or equivalent is an acceptable alternative method.

Seismic floor response spectra for various damping values are generated in three orthogonal directions (two horizontal and one vertical) at various elevations and locations of interest to the design of equipment and piping. When the dynamic analyses are performed separately for each of the three components of the input motion, the resulting co-directional response spectra are combined according to the square root of the sum of the squares (SRSS) method to obtain the combined spectrum in that direction. An alternative approach to obtain co-directional floor response spectra is to perform dynamic analysis with simultaneous input of the three excitation components if those components are statistically independent. Furthermore, when the three components are mutually and statistically independent, response analysis can be performed individually and the resulting acceleration response time histories in the same direction are added algebraically for floor response spectra generation.

In the generation of floor response spectra, the spectrum ordinates are computed at frequency intervals suggested in Table 3.7.1-1 of SRP 3.7.1 plus additional frequencies corresponding to the natural frequencies of the supporting structures. Another acceptable method is to choose a set of frequencies such that each frequency is within 10% of the previous one, and add the natural

frequencies of the supporting structures to the set. Alternatively, a set of frequencies such that each frequency is within 5% of the previous one is used.

3.7.2.6 Three Components of Earthquake Motion

Earthquake motion is three-dimensional and seismic design takes into account the effects of three orthogonal components (two horizontal and one vertical) of the prescribed design earthquake. The applicable methods for combining co-directional responses caused by each of the three components are described below.

When the response spectrum method or static coefficient method of analysis is used, the maximum responses caused by each of the three components are combined by taking the SRSS of the maximum co-directional responses caused by each of the three earthquake components at a particular point of the structure or of the mathematical model. The mathematical expression is

$$R_i = \left(\sum_{j=1}^{3} R_{ij}^2 \right)^{\frac{1}{2}} \qquad (3.7\text{-}9)$$

where

R_{ij} = maximum, co-directional response of interest in direction (i) caused by excitation in direction j (j = 1, 2, 3)

R_i = total combined response of interest in direction (i) obtained by the SRSS rule to account for non-simultaneous occurrence of R_{ij}.

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 is obtained using the SRSS method to combine the maximum co-directional 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 co-directional 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. This method is used for seismic response analysis of primary building structures.

3.7.2.7 Combination of Modal Responses

This section addresses the applicable methods for the combination of modal responses and the missing mass, when the response spectrum method is used for response analysis.

The analysis methods meet the requirements in RG 1.92 Revision 2, 2006, for combining the modal responses and the missing masses, except that for piping analyses, the double sum

equation in RG 1.92 Revision 1 is used and the residual rigid response of the missing mass modes is included. The details of the equations from the Regulatory Guide are shown below.

Research since the late 1970s has shown that in the regions of amplified spectral displacement, amplified spectral velocity and amplified spectral acceleration of a spectrum, the periodic responses are dominant. Beyond amplified spectral acceleration Region CD and up to E (Refer to RG 1.92 Rev. 2 for definition of Region CD and up to E), the modal responses consist of both the periodic and rigid components. The periodic modal responses and the periodic components of modal responses are combined using the following double sum ("complete quadratic combination") equation:

$$R_{pl} = \left[\sum_{i=1}^{n}\sum_{j=1}^{n} \varepsilon_{ij} R_{pi} R_{pj}\right]^{\frac{1}{2}} \qquad (3.7\text{-}10A)$$

where R_{pl} = combined periodic response for the l^{th} component of seismic input motion (l = 1, 2, 3, for one vertical and two horizontal components), ε_{ij} = the modal correlation coefficient for modes i and j, R_{pi} = periodic response or periodic component of a response of mode i, R_{pj} = periodic response or periodic component of a response of mode j, and n = number of modes considered in the combinations of modal responses.

For completely correlated modes i and j, $\varepsilon_{ij} = 1$; for partially correlated modes i and j, $0 < \varepsilon_{ij} < 1$; for uncorrelated modes i and j, $\varepsilon_{ij} = 0$.

The modal correlation coefficients are uniquely defined, depending on the method chosen for evaluating the correlation, as follows.

Rosenblueth provided the first significant mathematical approach to the evaluation of modal correlation for seismic response spectrum analysis. It is based on the application of random vibration theory, utilizing a finite duration of white noise to represent seismic loading. A formula for calculation of the coefficient ε_{ij} as a function of modal frequencies (f_i, f_j), modal damping ratios (λ_i, λ_j), and the time duration of strong earthquake motion (t_D) was derived as follows:

$$\varepsilon_{ij} = \left[1 + \left(\frac{f'_i - f'_j}{\lambda'_i f_i + \lambda'_j f_j}\right)^2\right]^{-1} \qquad (3.7\text{-}10B)$$

where

$$f'_i = f_i[1 - \lambda_i^2]^{\frac{1}{2}}$$

$$\lambda'_i = \lambda_i + \frac{1}{\pi t_D f_i}$$

and f'_j, λ'_j are similarly defined.

The Rosenblueth double sum equation in Equation 3.7-10A, which is accepted by RG 1.92 Revision 2, is different from the double sum equation in RG 1.92 Revision 1. The RG 1.92 Revision 1 equation is as follows:

$$R_{pI} = \left[\sum_{i=1}^{n}\sum_{j=1}^{n}\varepsilon_{ij}\left|R_{pi}R_{pj}\right|\right]^{\frac{1}{2}} \quad (3.7\text{-}10C)$$

As shown, the absolute value of the responses is used in the RG 1.92 Revision 1 equation, so the product of the responses is always positive. For piping analyses, Equation 3.7-10C is used, rather than Equation 3.7-10A. The RG 1.92 Revision 1 equation used in the piping analyses provides more conservative results than the RG 1.92 Revision 2 equation. The amount of conservatism depends on the critical damping ratio used in the piping analysis. Higher damping ratio will result in more conservatism. See Subsection 3D.4.1.1 for further details of the PISYS08 piping computer program.

Appendix D to Reference 3.7-17 tabulates numerical values of ε_{ij} for the Rosenblueth formula as a function of frequency, frequency ratio, and strong motion duration time for constant modal damping of 1%, 2%, 5% and 10%. The effect of t_D is most significant at 1% damping and low frequency. For 5% and 10% damping, t_D = 10 sec and 1,000 sec produced similar values for ε_{ij} regardless of frequency. The most significant result is that ε_{ij} is highly dependent on the damping ratio; for 2%, 5% and 10% damping, ε_{ij} = 0.2, 0.5 and 0.8, respectively, at a frequency ratio of 0.9 (modal frequencies within 10%).

For modal combination involving high-frequency modes, the following procedure applies:

Step 1. Determine the modal responses only for those modes with natural frequencies less than that at which the spectral acceleration approximately returns to the ZPA of the input response spectrum (f_{ZPA}). Examples of f_{ZPA} are shown in figures 1, 2 and 3 of RG 1.92, Rev. 2. Combine such modes in accordance with the methods described above.

When applying these methods to building dynamic loads other than seismic, it is acceptable to use a ZPA cutoff frequency of 100 Hz if the spectral acceleration at 100 Hz has not returned to the ZPA of the response spectrum.

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 modes included in Step 1. This fraction d_i for each DOF i is given by the following equation:

$$d_i = \sum_{n=1}^{N}[(C_{n,j})(\phi_{n,i})] \quad (3.7\text{-}11)$$

where

n = mode number ($1, 2, \ldots, N$)

N = the number of modes included in Step 1

$\phi_{n,i}$ = eigenvector value for mode n and DOF i

j = direction of input motion

$c_{n,j}$ = participation factor for mode n in the j^{th} direction:

$$c_{n,j} = \frac{\{\phi_n\}^T [m] \{\delta_{ij}\}}{\{\phi_n\}^T [m] \{\phi_n\}}$$

where δ_{ij} is the Kronecker delta, which is 1 if DOF i is in the direction of the earthquake input motion j and 0 if DOF i is a rotation or not in the direction of the earthquake input motion j. This assumes that the three orthogonal directions of earthquake input motion are coincident with the DOF directions. Also, [m] is the mass matrix.

Next, determine the fraction of DOF mass not included in the summation of these modes:

$$e_i = d_i - \delta_{ij} \quad (3.7\text{-}12)$$

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 pseudostatic response to the inertial forces from these higher modes excited at the ZPA. The pseudostatic inertial forces associated with the summation of all higher modes for each DOF i are given by the following:

$$P_i = (ZPA)(M_i)(e_i) \quad (3.7\text{-}13)$$

where P_i is the force or moment to be applied at DOF i, M_i is the mass or mass moment of inertia associated with DOF i.

The structure is then statically analyzed for this set of pseudostatic inertial forces applied to all degrees of freedom to determine the maximum responses associated with high-frequency modes not included in Step 1.

This procedure requires the computation of individual modal responses only for lower-frequency modes. 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.

3.7.2.8 **Interaction of Non-Category I Structures with Seismic Category I Structures**

The interfaces between Seismic Category I and non-Seismic Category I SSCs are designed for the dynamic loads and displacements produced by both the Category I and non-Category I SSCs. All non-Category I SSCs meet at least one of the following requirements:

1. The collapse of any non-Category I structure, system or component does not cause the non-Category I structure, system or component to strike a Seismic Category I SSCs. SSCs

in this category are classified as Seismic Category NS. Any Seismic Category NS structure postulated to fail under SSE is located at least a distance of its height above grade from Seismic Category I structures.

2. The collapse of any non-Category I SSCs does not impair the integrity of Seismic Category I SSCs. This is 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, including missiles). SSCs in this category are classified as Seismic Category NS.

3. The non-Category I structures, systems or components are 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. SSCs in this category are classified as Seismic Category II, except the Radwaste Building.

The following subsections describe the seismic analysis methodology and design acceptance criteria for the Radwaste Building and Seismic Category II Buildings to preclude any adverse interaction with Seismic Category I structures.

The locations of structures are provided in Figure 2.1-204. Non-Category I structures within the scope of the UFSAR are addressed in the UFSAR. Non- Category I structures outside the scope of the UFSAR are located at least a distance of its height above grade from Seismic Category I structures. Thus, the collapse of any site specific 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.

For the Seismic Category II structures and Radwaste Building, Fermi 3 site-specific analyses will be performed if the backfill requirements are not met.

The locations of structures are provided in Figure 2.1-204. Non-Category I structures within the scope of the UFSAR are addressed in the UFSAR. Each site-specific non-Category I structure outside the scope of the UFSAR is located at least a distance of the structure’s height above grade from Seismic Category I structures. Thus, the collapse of any site specific 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.

The design and analysis of the Seismic Category II structures (TB, SB, and ADB) and the Seismic Category NS Radwaste Building (RW) structure will be completed as part of the detailed design phase for the ESBWR standard plant. The design and analysis for these structures will be in accordance with the ESBWR DCD, considering the soil property requirements in Tier 1 Table 5.1-1, to ensure that the acceptance criteria in Tier 1 ITAAC Tables 2.16.8-1, 2.16.9-1, 2.16.10-1, and

2.16.11-1 are met. Subsection 3.7.2.8 describes the seismic design and analysis for the TB, SB, ADB and RW structures to preclude any adverse interaction with Seismic Category I structures.

If the soil property requirements in Tier 1 Table 5.1-1 are not met, Fermi 3 site-specific seismic SSI analyses using the Fermi 3 soil properties will be performed to demonstrate the adequacy of the standard plant design for the TB, SB, ADB, and the RW structures, as follows:

- These Fermi 3 site-specific seismic SSI analyses for the TB, RW, SB, and ADB structures will be consistent with the site-specific seismic SSI analyses for the Seismic Category I RB/FB and CB structures presented in UFSAR Subsection 3.7.2.4 and will be performed using the Fermi 3 soil properties and the methodologies described in Subsections 3.7.2.8.1, 3.7.2.8.2, 3.7.2.8.3, and 3.7.2.8.4, respectively, and Appendix 3A.
- In addition to these site-specific seismic SSI analyses, site-specific seismic structure-soil-structure interaction (SSSI) analyses to evaluate any adverse effects between the TB, RW, SB, and ADB structures and adjacent Seismic Category I structures will be performed using the methodologies described in Subsections 3.7.2.8.1, 3.7.2.8.2, 3.7.2.8.3, and 3.7.2.8.4, respectively, and Appendix 3A.

Results of these site-specific seismic SSI and seismic SSSI analyses, if needed, will be discussed as part of the ITAAC completion package for the TB, RW, SB, and ADB structures to demonstrate that acceptance criteria in ITAAC Tables 2.4.15-1, 2.4.16-1, 2.4.17-1 and 2.4.18-1, respectively, are met.

3.7.2.8.1 **Turbine Building**

The Turbine Building is a Seismic Category II structure that is adjacent to the Reactor Building. The method of analysis of the Turbine Building is the same as a Seismic Category I structure including the loading cases and acceptance criteria as shown in Tables 3.8-15 and 3.8-16. The mathematical model of the structural systems for seismic analysis is either a stick model or a finite element model using the procedures in accordance with Subsection 3.7.2.3. The soil-structure interaction (SSI) analysis is performed using the soil spring/dashpot approach or the finite element approach in accordance with Appendix 3A. The generic uniform site properties are shown in Table 3A.3-1 and the layered site properties are shown in Table 3A.3-3. The effect of structure-soil-structure interaction with adjacent Seismic Category I structures is performed in the same manner as described in Subsection 3A.8.11. Seismic input motions are based on the single envelope design response spectra as defined in Table 3.7-2 with the applicable scale factor applied at the foundation level, at the bottom of the base slab.

The Turbine Building location is shown in Figure 1.1-1. The building height is shown in Figure 1.2-19. The seismic gaps between the Turbine Building and the Reactor Building are no less than the calculated maximum relative displacements between the two buildings during SSE event, considering out-of-phase motion.

3.7.2.8.2 **Radwaste Building**

The RW is designed in accordance with RG 1.143 Classification RW-IIa. The earthquake loading for the RW is full SSE instead of 1/2 SSE as shown in RG 1.143. Systems, structures and components classified as RW-IIa that are housed within the RW are designed to 1/2 SSE.

Analysis of the RW is performed in the same manner as a Seismic Category I structure including the loading cases and acceptance criteria as shown in Tables 3.8-15 and 3.8-16. The mathematical model of the structural systems for seismic analysis is either a stick model or a finite element model using the procedures of Subsection 3.7.2.3. The SSI analysis is performed using the soil spring/dashpot approach or the finite element approach in accordance with Appendix 3A. The generic uniform site properties are shown in Table 3A.3-1 and the layered site properties are shown in Table 3A.3-3. The effect of structure-soil-structure interaction with adjacent Seismic Category I structures is performed in the same manner as described in Subsection 3A.8.11. *Seismic input motions are based on the single envelope design response spectra as defined in Table 3.7-2 with the applicable scale factor, applied at the foundation level, at the bottom of the base slab.*

The RW location is shown in Figure 1.1-1. It is located at least 10 meters from the RB. The building height is shown in Figure 1.2-25.

3.7.2.8.3 **Service Building**

The Service Building is a Seismic Category II structure that is adjacent to the Reactor Building and the Fuel Building. The method of analysis of the Service Building is the same as a Seismic Category I structure including the loading cases and acceptance criteria as shown in Tables 3.8-15 and 3.8-16. The mathematical model of the structural systems for seismic analysis is either a stick model or a finite element model using the procedures of Subsection 3.7.2.3. The SSI analysis is performed using the soil spring/dashpot approach or the finite element approach in accordance with Appendix 3A. The generic uniform site properties are shown in Table 3A.3-1 and the layered site properties are shown in Table 3A.3-3. The effect of structure-soil-structure interaction with adjacent Seismic Category I structures is performed in the same manner as described in Subsection 3A.8.11. *Seismic input motions are based on the single envelope design response spectra as defined in Table 3.7-2 with the applicable scale factor, applied at the foundation level, at the bottom of the base slab.*

The Service Building location is shown in Figure 1.1-1. The seismic gaps between the Service Building and the Reactor/Fuel Building are no less than the calculated maximum relative displacements between the two buildings during an SSE event, considering out-of-phase motion.

3.7.2.8.4 **Ancillary Diesel Building**

The Ancillary Diesel Building is a Seismic Category II structure. It houses the Ancillary Diesel Generators that are classified as Criterion B under the Regulatory Treatment of Non-Safety Systems (see Section 19A). The method of analysis of the Ancillary Diesel Building is the same as

a Seismic Category I structure including the loading cases and acceptance criteria as shown in Tables 3.8-15 and 3.8-16. The mathematical model of the structural systems for seismic analysis is either a stick model or a finite element model using the procedures of Subsection 3.7.2.3. The soil-structure interaction (SSI) analysis is performed using the soil spring/dashpot approach or the finite element approach in accordance with Appendix 3A*. The generic uniform site properties are shown in Table 3A.3-1 and the layered site properties are shown in Table 3A.3-3. The effect of structure-soil-structure interaction with adjacent Seismic Category I structures is performed in the same manner as described in* Subsection 3A.8.11*. Seismic input motions are based on the single envelope design response spectra as defined in Table 3.7-2 with the applicable scale factor applied at the foundation level, bottom of the base slab.*

The Ancillary Diesel Building location is shown in Figure 1.1-1. It is located at least 15.2 meters from the Fuel Building.

3.7.2.9 **Effects of Parameter Variations on Floor Response Spectra**

Floor response spectra calculated according to the procedures described in Subsection 3.7.2.5 are peak broadened by ±15% to account for uncertainties in the structural frequencies owing to uncertainties in the material properties of the structure and soil and to approximations in the modeling techniques used in the analysis.

When, in lieu of response spectrum analysis, the calculated floor acceleration time history is used to perform a time history analysis of piping and equipment, uncertainties are accounted for by expanding and shrinking the floor acceleration time history within 1/(1±0.15) so as to change the frequency content of the time history by ±15%. In this case, multiple time history analyses are performed. Alternatively, a single synthetic time history, which matches the broadened floor response spectra, may be used.

The methods described above to account for the effect of parameter variation are applicable to seismic and other building dynamic loads.

3.7.2.10 **Use of Equivalent Vertical Static Factors**

Equivalent vertical static factors are used when the requirements for the static coefficient method in Subsection 3.7.2.1.3 are satisfied. *All Seismic Category I structures are dynamically analyzed in the vertical direction. No constant static factors are utilized.*

3.7.2.11 **Methods Used to Account for Torsional Effects**

One method of treating the torsional effects in the dynamic analysis is to carry out a dynamic analysis that incorporates the torsional degrees of freedom. For structures having negligible coupling of lateral and torsional motions, a two-dimensional model without the torsional degrees of freedom can be used for the dynamic analysis and the torsional effects are accounted for in the following manner. The locations of the center of mass are calculated for each floor. The center of rigidity and torsional stiffness are determined for each story. Torsional effects are introduced in each

story by applying a torsional moment about its center of rigidity. The torsional 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.

The seismic analysis for primary building structures is performed using a three-dimensional model including the torsional degrees of freedom.

3.7.2.12 **Comparison of Responses**

Since only the time history method is used for the dynamic analysis of Seismic Category I structures, a comparison of responses with the response spectrum method is not necessary.

3.7.2.13 **Analysis Procedure for Damping**

When the modal superposition method of analysis (either time history or response spectrum) is used for models that consist of elements with different damping properties, the composite modal damping ratio can be obtained either as stiffness-weighted:

$$\lambda_k = \frac{\{\phi\}^T[\bar{K}]\{\phi\}}{K^*} \tag{3.7-14}$$

or as mass-weighted:

$$\lambda_k = \{\phi\}^1[\bar{M}]\{\phi\} \tag{3.7-15}$$

where:

λ_k	=	*equivalent modal damping for the k-th mode*
K^*	=	$\{\phi\}^T [K] \{\phi\}$
$[K]$	=	*assembled stiffness matrix*
$[\bar{K}]$, $[\bar{M}]$	=	*modified stiffness or mass matrix constructed from element matrices formed by the product of the damping ratio for the element and its stiffness or mass matrix*
$\{\phi\}$	=	*k-th normalized modal vector.*

The composite modal damping calculated by either Equation 3.7-14 or 3.7-15 is limited to 20%. For models that take SSI into account by the lumped soil spring approach, the method defined by Equation 3.7-14 is acceptable. For fixed base model, either Equation 3.7-14 or 3.7-15 is used.

In the seismic response analysis of primary building structures described in Appendix 3A using the complex response method in the frequency domain, material damping is included in the formulation of the complex stiffness matrix:

$$[k_j^*] = [k_j](1 + 2i\lambda_j) \tag{3.7-16}$$

where

$[k_j^*]$	=	complex stiffness matrix of element j
$[k_j]$	=	stiffness matrix of element j
λ_j	=	*material damping ratio of element j*
i	=	$\sqrt{-1}$

In the seismic response analysis of primary building structures described in Appendix 3A using the time history method solved by direct integration, the damping matrix is formed by the following procedure:

(1) *First, the stiffness-weighted modal damping λ_k is calculated in accordance with Equation 3.7-14*

(2) The damping matrix that fits the relationships between the frequencies and modal damping constants above can be calculated using the following formula. (Reference 3.7-9)

$$[C] = [M][\Phi][\Lambda][\Phi]^T[M] \tag{3.7-17}$$

where,

[M]: *mass matrix*

[Φ]: *undamped characteristic mode matrix*

$$[\Lambda]: \begin{bmatrix} \Lambda_1 & & & & & & & \\ & \cdot & & & & & & \\ & & \cdot & & & & & \\ & & & \cdot & & & & \\ & & & & \Lambda_k & & & \\ & & & & & \cdot & & \\ & & & & & & \cdot & \\ & & & & & & & \cdot \\ & & & & & & & & \Lambda_n \end{bmatrix}$$

$$\Lambda_k = \frac{2\lambda_k\omega_k}{m_k}$$

λ_k: *k-th damping constant*

ω_k: *k-th undamped circular frequency*

m_k: *k-th equivalent mass*

n: *maximum mode number*

In the dynamic response analysis of containment loads described in Appendix 3F using the direct integration time history method, the damping matrix is formed by a linear combination of the mass and stiffness matrices,

$$[C] = \alpha[M] + \beta[K] \tag{3.7-18}$$

where α and β are constants. They are determined to give the required damping value as a function of the circular frequency ω, i.e.,

$$\lambda = \frac{\alpha}{2\omega} + \beta\frac{\omega}{2} \tag{3.7-19}$$

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

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 can be treated as a rigid body.

The maximum kinetic energy (E_s) can be conservatively estimated to be:

$$E_s = \frac{1}{2}\sum_i m_i[(V_h)_i^2 + (V_v)_i^2] \qquad (3.7\text{-}20)$$

where $(V_h)_i$ and $(V_v)_i$ are the maximum values of the total lateral velocity and total vertical velocity, respectively, of mass m_i, and are computed as follows:

$$\begin{aligned} |(V_h)_i| &= |(V_x)_i| + |(V_h)_g| \\ |(V_v)_i| &= |(V_z)_i| + |(V_v)_g| \end{aligned} \qquad (3.7\text{-}21)$$

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 the total mass of the structure and basemat, the potential energy required to overturn the structure is equal to:

$$E_o = m_o gh + W_p - W_b \qquad (3.7\text{-}22)$$

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 effects of embedment and buoyancy, 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 stable against overturning when the ratio of E_o to E_s is no less than 1.1 for the SSE in combination with other appropriate loads.

The Fermi 3 site-specific stability evaluation against overturning is presented in Subsection 3.8.5.

3.7.3 **Seismic Subsystem Analysis**

This section applies to Seismic Category I and Seismic Category II subsystems (equipment and piping) that are qualified to satisfy the performance requirements according to their Seismic Category I or Seismic Category II designation. Input motions for the qualification are usually in the form of floor response spectra and displacements obtained from the primary system dynamic analysis. Input motions in terms of acceleration time histories are used when needed. Dynamic qualification can be performed by analysis, testing, or a combination of both, or by the use of experience data. This section addresses the aspects related to analysis only.

3.7.3.1 **Seismic Analysis Methods**

The methods of analysis described in Subsection 3.7.2.1 *are equally applicable to equipment and piping systems.* Among the various dynamic analysis methods, the response spectrum method is used most often. For multi-supported systems analyzed by the response spectrum method, the input motions can be either the envelope spectrum with USM of all support points or the ISM at

each support. Additional considerations associated with the ISM response spectrum method of analysis are given in Subsection 3.7.3.9. For equipment analysis, refer to the requirements of Step 1 of Subsection 3.7.2.7 for ZPA cutoff frequency determination.

3.7.3.2 Determination of Number of Earthquake Cycles

The SSE is the only design earthquake considered for the ESBWR Standard Plant. To account for the cyclic effects of the more frequent occurrences of lesser earthquakes and their aftershocks, the fatigue evaluation for ASME B&PV 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 SRP 3.7.3. Alternatively, a number of fractional vibratory cycles equivalent to 20 full SSE vibratory cycles is used (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 RG 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 is used in accordance with Appendix D of IEEE-344 when followed by one full SSE.

3.7.3.3 Procedures Used for Analytical Modeling

The mathematical modeling of equipment and piping is developed according to the finite element technique following the basic modeling procedures described in Subsection 3.7.2.3 for primary systems.

3.7.3.3.1 Piping Systems

Mathematical models for Seismic Category 1 piping systems are constructed to reflect the dynamic characteristics of the system. The continuous system is modeled as an assemblage of pipe elements (straight sections, elbows, and bends) supported by hangers and anchors, and restrained by pipe guides, struts and snubbers. Pipe and hydrodynamic fluid masses are lumped at the nodes and connected by zero-mass elastic elements, which reflect the physical properties of the corresponding piping segment. The mass 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 concentrated weights on the piping systems, such as the valves, pumps, and motors, are modeled as lumped mass rigid systems if their fundamental frequencies are greater than the cutoff frequency in Subsection 3.7.2.1.1. On straight runs, mass points are located at spacing no greater than the span which would have a fundamental frequency equal to the cutoff frequency stipulated in Subsection 3.7.2.1.1, when calculated as a simply supported beam with uniformly distributed mass. The torsional effects of valve operators and other equipment with offset center of gravity with respect to the piping center line are included in the analytical model. Furthermore, all

pipe guides and snubbers are modeled so as to produce representative stiffness. The equivalent linear stiffness of the snubbers is based on certified test results provided by the vendor.

Pipe supports are designed and qualified to satisfy stiffness values used in the piping analysis. For struts and snubbers, the stiffness to consider is the combined stiffness of strut, snubber, pipe clamp and piping support steel.

In general, pipe support component weights, which are directly attached to a pipe such as a clamp, strut, snubber, and trapeze, are considered in the piping analysis. Frame type supports are designed to carry their own mass and are subjected to deflection requirements. *A maximum deflection of 1.6 mm (1/16 in.) is used for normal operating conditions, and 3.2 mm (1/8 in.) is used for abnormal conditions. For other types of supports, either it must be demonstrated that the support is dynamically rigid, or it must be demonstrated that one-half of the support mass is less than 10% of the mass of the straight pipe segment of the span at the support location, to preclude amplification. Otherwise, the contribution of the support weight amplification is added into the piping analysis. Piping supports are evaluated to include the impact of self-weight excitation on support structure and anchorage in detail along with piping analyzed loads where this effect is significant.*

The stiffness of the building steel/structure (i.e., beyond the NF jurisdictional boundary) is not considered in pipe support overall stiffness. Response spectra input to the piping system includes flexibility of the building structure. When attachment to a major building structure is not possible, any intermediate structures are included in the analysis of the pipe support.

3.7.3.3.2 **Equipment**

For dynamic analysis, equipment is represented by a lumped-mass system, which consists of discrete masses connected by zero-mass elements. The criteria used to lump masses are as follows:

- 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 number of masses or dynamic degrees of freedom is considered adequate when additional degrees of freedom do not result in more than a 10% increase in response. Alternatively, the number of dynamic degrees of freedom is no less than twice the number of modes below the cutoff frequency of Subsection 3.7.2.1.1.
- Mass is lumped at any point where a significant concentrated weight is located. Examples are the motor in the analysis of a pump stand, and the impeller in the analysis of a pump shaft.
- If the equipment has free-end overhang span whose flexibility is significant compared to the center span, a mass is lumped at the overhang span.
- *When equipment is concentrated between two existing nodes located between two supports in a finite element model, a new node is created at that location. Alternatively, the equipment mass can be concentrated at the nearest node to either side which tends to shift the natural frequency*

to the higher amplification region of the input motion response spectrum. When the approximate location of the equipment mass is shifted toward the mid-span between the supports the natural frequency is lowered and when the approximate location is shifted toward either support the natural frequency is increased. Moving the natural frequencies of the equipment into the higher amplification region of the excitation thereby conservatively increases the equipment response level.

Similarly, in the case of live loads (mobile) and variable support stiffness, the location of the load and the magnitude of the support stiffness are chosen to lower the system natural frequencies. Similar to the above discussion, this ensures conservative dynamic responses because the lowered equipment frequencies tend to be shifted to the higher amplification range of the input motion spectra. If not, the model is adjusted to give more conservative responses.

3.7.3.3.3 **Modeling of Special Engineered Pipe Supports**

Special engineered pipe supports are not used.

3.7.3.4 **Basis for Selection of Frequencies**

Where practical, in order to avoid adverse resonance effects, equipment and components are designed/selected such that their fundamental frequencies are less than half or more than twice the dominant frequencies of the support structure. Moreover, in any case, the equipment is analyzed or tested or both 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.

3.7.3.5 **Analysis Procedure for Damping**

Damping values for equipment and piping are shown in Table 3.7-1 and are consistent with RG 1.61, Revision 1. For ASME B&PV Code, Section III, Division 1 Class 1, 2, and 3, and ASME B31.1 piping systems, alternative damping values specified in Figure 3.7-37 are used. For systems made of subsystems with different damping properties, the analysis procedures described in Subsection 3.7.2.1.3 are applicable.

3.7.3.6 **Three Components of Earthquake Motion**

The applicable methods of spatial combination of responses due to each of the three input motion components are described in Subsection 3.7.2.6.

3.7.3.7 **Combination of Modal Responses**

The applicable methods of modal response combination are described in Subsection 3.7.2.7.

3.7.3.8 **Interaction of Other Systems with Seismic Category I Systems**

Each non-Category I (i.e., Seismic Category II or non-seismic) system is designed to be isolated from any Seismic Category I system by either a constraint or barrier, or is remotely located with regard to the Seismic Category I system. *If it is not feasible or practical to isolate the Seismic*

Category I system, adjacent non-Category I systems are analyzed according to the same seismic criteria as applicable to the Seismic Category I systems. For non-Category I systems attached to Seismic Category I systems, the dynamic effects of the non-Category I systems are simulated in the modeling of the Seismic Category I system. The attached non-Category I systems, up to the first anchor beyond the interface, are also designed in such a manner that during an earthquake of SSE intensity it does not cause a failure of the Seismic Category I system.

3.7.3.9 **Multiple-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 are acceptable:

- Envelope response spectrum with USM applied at all support points for each orthogonal direction of excitation; or
- ISM response spectrum at each support for each orthogonal direction of excitation.

When the ISM response spectrum method of analysis (Subsection 3.7.2.1.2) is used, a support group is defined by supports that have the same time-history input. This usually means all supports located on the same floor, or portions of a floor, of a structure. The highest response spectrum for any support in a given group is used as input for the entire support group. This approach is appropriate since the time histories for supports within each group are time phase correlated. In most cases, the support at the highest elevation has the highest response spectrum and is used for the group. For piping inside the RCCV, the responses caused by motions of supports in two groups are combined by the SRSS procedure since it has been demonstrated that the phases for the independent support motions are sufficiently uncorrelated, and the analysis results for two typical piping systems (main steam and feedwater) using the SRSS procedure are more conservative than the time history analysis method when a 10 percent margin is applied to the SRSS results. In most cases, the number of support groups can be restricted to two ISM groups (RPV and inside RCCV), but in piping analysis cases where additional support motion groups are used within the RCCV, the absolute sum procedure for an ISM analysis shall be used. For piping outside the RCCV, the absolute sum procedure for an ISM analysis shall be used.

To use the SRSS method for independent support response spectrum analysis, it is required to include 10 percent margin in the design requirements for piping stress and piping support loads to address the uncertainties that may exist from the use of the SRSS method rather than the absolute sum method for the group combination method when performing an ISM analysis.

In addition to the inertial response discussed above, the effects of relative support displacements are considered. The maximum relative support displacements are obtained from the dynamic analysis of the building, or as a conservative approximation, by using the floor response spectra. For the latter option, the maximum displacement of each support is predicted by $S_d = S_a g/\omega^2$, where S_a is the spectral acceleration in “g’s” at the high-frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and ω is the

fundamental frequency of the primary support structure in radians per second. The support displacements are imposed on the supported systems in a conservative (i.e., most unfavorable combination) manner and static analysis is performed for each orthogonal direction. The resulting responses are combined with the inertia effects by the SRSS method. Because the OBE design is not required, the displacement-induced SSE stresses due to seismic anchor motion are included in Service Level D load combinations.

In place of the response spectrum analysis, the ISM time history method of analysis is used for multi-supported systems subjected to distinct support motions, in which case both inertial and relative displacement effects are already included.

3.7.3.10 **Use of Equivalent Vertical Static Factors**

Equivalent vertical static factors are used when the requirements for the static coefficient method in Subsection 3.7.2.1.3 are satisfied.

3.7.3.11 **Torsional Effects of Eccentric Masses**

Torsional effects of eccentric masses are included for subsystems similar to that for the piping systems discussed in Subsection 3.7.3.3.1.

3.7.3.12 **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 seismic activity site.

Differential endpoint or restraint deflections cause forces and moments to be induced in 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.

When the piping analysis is performed using USM analysis, per SRP Section 3.9.2, the absolute sum method is used to combine the inertia results and the seismic anchor motion results for both piping and piping support design.

When the piping analysis is performed by ISM, the piping stresses and pipe support loads are increased by 10% when using the SRSS group combination method. With the additional 10% added to the piping stresses and the pipe support loads, the inertia and the seismic anchor motion are combined by SRSS for piping stresses and pipe support loads.

3.7.3.13 Seismic Category I Buried Piping, Conduits and Tunnels

All Seismic Category I utilities (i.e. piping, conduits, or auxiliary system components), that are routed underground are installed in concrete trenches/tunnels or in concrete duct banks in direct contact with soil.

Fire Protection System yard piping with a Seismic Category I classification is installed in reinforced concrete tunnels and covered concrete trenches near the ground surface with removable covers to facilitate maintenance and inspection access.

There are Seismic Category I conduits in four electrical duct banks from the CB to the RB. The electrical conduit are embedded in reinforced concrete duct bank in direct contact with soil.

The access tunnel, which includes walkways between and access to the RB, CB, TB, SB, and Electrical Building, is classified Seismic Category II. Since Seismic Category II structures are designed to the same criteria as Seismic Category I structures there is no impact to adjacent Seismic Category I structures.

The Radwaste Tunnel provides for pipes that transport radioactive waste to the Radwaste Building from the RB and TB. The radwaste tunnel is classified non-seismic but the structural acceptance criteria are in accordance with RG 1.143 – Safety Class RW-IIa.

In accordance with SRP 3.7.3 (Rev. 3, March 2007), the following items are considered in the analysis and design of trenches/tunnels or concrete duct banks for Seismic Category I utilities and buried Seismic Category II and radwaste tunnels (RG 1.143 – use 1/2 SSE):

- *Two types of ground shaking-induced loadings are considered for design:*
  - *Relative deformations imposed by seismic waves traveling through the surrounding soil or by differential deformations between the soil and anchor points.*
  - *Lateral earthquake pressures and groundwater effects acting on structures.*
- *When applicable, the effects caused by local soil settlements, soil arching, etc., are considered in the analysis.*
- *Lateral earth pressures are determined in the same manner as for embedded walls below grade for Seismic Category I structures. The effect of wave propagation is accounted for in accordance with ASCE 4-98, Subsection 3.5.2 and Commentary.*
- *Longitudinal forces and strains are treated as secondary forces and strains (displacement-controlled).*
- *Longitudinal compressive strains are limited to 0.3%. The reinforcing steel added to the concrete addresses the effect of longitudinal tensile strains. Member forces are calculated per ASCE 4-98 methodology and section capacities are determined per ACI 349-01. Steel section properties are determined per AISC N690-94.*

- *Seismic input motions are based on the single envelope design response spectra as defined in Table 3.7-2 using the applicable scale factor.*
- *Primary loadings are lateral earth pressures, hydrostatic pressures, dead loads, and live loads applied concurrently with seismic excitation. Resultant stresses due to wave propagation effects and those resulting from the dynamic anchor movement are combined by the SRSS method.*
- *Expansion joints are provided between the tunnel and the connecting building to provide seismic isolation.*
- *Expansion joints along the tunnel are placed no more than 20 m (65.6 ft.) apart.*

Seismic Category I utilities and Safety Class RW-IIa radwaste piping installed in trenches or tunnels are analyzed in accordance with the standard requirements of Subsection 3.7.3. Seismic input motions for the portions located below ground are based on the single envelope design response spectra as defined in Table 3.7-2 using applicable scale factors.

3.7.3.14 Methods for Seismic Analysis of Seismic Category I Concrete Dams

There are no Seismic Category I concrete dams in the ESBWR design.

3.7.3.15 Methods for Seismic Analysis of Above-Ground Tanks

The seismic analysis of Seismic Category I above-ground tanks considers the following items:

- At least two horizontal modes of combined fluid-tank vibration and at least one vertical mode of fluid vibration are included in the analysis. The horizontal response analysis includes at least one impulsive mode in w*hich the responses of the tank shell and roof are coupled together with the portion of the fluid contents that move in unison with the shell, and the fundamental sloshing (convective) mode.*
- *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, Sa1, is then determined using this frequency and damping value for the impulsive mode. This is the same as that for the tank shell material in accordance with NUREG/CR-1161 (Reference 3.7-19). Alternatively, the maximum spectral acceleration corresponding to the relevant damping is used.*
- *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 SSI. The SSI system damping takes into account soil damping in the form of stiffness-weighted damping in accordance with Equation 3.7-14 or the complex stiffness matrix in accordance with Equation 3.7-16.*

- *In determining the spectral acceleration in the horizontal convective mode, Sa2, the fluid damping ratio is 0.5% of critical damping unless a higher value can be substantiated by experimental results.*
- *The maximum overturning moment, Mo, at the base of the tank is obtained by the modal and spatial combination methods discussed in Subsections 3.7.2.7 and 3.7.2.6, respectively. The uplift tension resulting from Mo 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 Mo, when used, must be shown to be conservative.*
- *The seismically induced hydrodynamic pressures on the tank shell at any level are determined by the modal and spatial combination methods discussed in Subsections 3.7.2.7 and 3.7.2.6, respectively. The maximum hoop forces in the tank wall are evaluated with due regard for the contribution of the vertical component of ground shaking. If the effects of SSI results in higher response, then an appropriate SSI method of analysis comparable to Reference 3.7-16 is used. The hydrodynamic pressure at any level is added to the hydrostatic pressure at that level to determine the hoop tension in the tank shell.*
- *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.*
- *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.*
- *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 Mo.*
- *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.*

The seismic SSI analysis of the Firewater Storage Tanks is described in Appendix 3A.

3.7.3.16 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. *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 are 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. Due to branch decoupling, the thermal displacements at the run pipe are combined with associated pressures and temperatures for the flexibility analyses of the branch pipe. All the stresses must meet the ASME B&PV Code

requirements. The branch pipe analysis results ensure adequate flexibility and proper design of all the restraints on the branch pipe.

2. For small bore piping defined as piping 50 mm (2 in.) and less nominal pipe size, and small branch lines 50 mm (2 in.) 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 ND-3600 in ASME B&PV Code Section III and ASME B31.1 Code, whenever the following are met:
   a. When the small bore piping handbook is serving the purpose of the design report it meets all of the ASME B&PV Code requirements for a piping design report. This includes the piping and its supports.
   b. 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 is not applied when specific information is needed on (a) magnitude of pipe and fitting 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 is not applied to piping systems that are fully engineered and installed in accordance with the engineering drawings.

3.7.3.17 Interaction of Other Piping with Seismic Category I Piping

In certain instances, Seismic Category II piping is 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. Because 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.
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 Seismic Category II subsystem; or for a distance such that there are at least two seismic restraints in each of the three orthogonal directions.

Note: The interface anchor between the seismic and non-seismic category piping is designed for the maximum load using piping reactions from both sides.

Where small Seismic Category II piping is directly attached to Seismic Category I piping, it can be decoupled from Seismic Category I piping.

For dynamic and seismic anchor motion analyses,

1. Decouple criterion is 25 to 1 in the ratio of “moment of inertia” of run pipe to branch pipe.
2. Amplified response spectra from the seismic and dynamic analyses used in the large bore piping analysis (run pipe) are applied to the small branch piping interfaces. The seismic and dynamic displacements at the connection point use the run pipe displacements.
3. Formal analysis methods and procedures similar to the main pipe should be used, or more conservative handbook analysis may also be used.
4. Branch pipe decoupling using response spectrum analysis can use one of the following options.
   a. Place the branch line close (4 times pipe diameter, for example) to large bore pipe supports.
   b. Demonstrate that the applicable pipe segment is “dynamically rigid”.
   c. Overlapping analysis. (1) Include the small bore pipe up to two supports in all three directions to the large bore pipe, (2) analyze the small bore pipe again.
   d. The dynamic analysis obtains the accelerations at the supports on both sides of the run pipe side (Aa), and side (Ab), and at the small branch at (Ac). Envelope the adjusted amplified response spectra (ARS) from both sides of the run pipe supports, (Ac/Aa) and (Ac/Ab), in all three directions and apply to the branch pipe analysis.
   e. From large bore piping analysis, obtain the ARS at the branch location to apply to the branch pipe analysis. (A referenced program is ERSIN01 user’s manual.)

The decouple criterion is 25 to 1 in the ratio of “moment of inertia” of run pipe to branch pipe. In the event that this criterion cannot be met and decoupling is also needed, then the decouple method as outlined in NUREG/CR-1980 is used. The following specific criteria from NUREG/CR-1980 are applied. In general, based on the current capability of modeling software the entire system is incorporated into one model instead of using the overlap method.

1. The overlap region has enough rigid restraints and includes enough bends in three directions to prevent the transmission of motion due to modal excitation from one end to the other and to reduce to a negligible level the sensitivity of the structure to the direction of excitation. Specifically, there are at least four rigid restraints in each of three mutually perpendicular directions in the overlap region (including the ends). For axial restraints only this requirement may be relaxed to a single restraint in any straight segment.

2. For cases where multiple spectra are involved at the different anchor points the spectrum to be used for each subsystem analysis is dependent on the rigidity of the overlap region. If the fundamental natural frequency of the overlap is demonstrated to be at least 25% higher than the highest significant forcing frequency, then the envelope spectrum of the spectra associated with the boundaries of each separate subsystem is acceptable. If this rigidity of the overlap region is not demonstrated or its frequency characteristics do not meet the criterion stated above the full system anchor-to-anchor envelope spectrum is used for all subsystems.
3. The envelope of the support forces is increased by 10% for design purposes.

3.7.4 Seismic Instrumentation

In accordance with SRP 3.7.4, the seismic instrumentation system meets the relevant requirements of GDC 2, 10 CFR 50, Appendix S, and 10 CFR 50.55a "Codes and Standards" as they relate to the capabilities and performance of the instruments to adequately measure the effects of earthquakes. Any other seismic instrumentation program, which is justified to have equivalent capabilities, may also be used. The instrumentation used for the measurements is capable of recording the effects produced by the most severe earthquakes that have been historically reported for the unique site considered and surrounding area, with sufficient margin for the limited accuracy, quantity and period of time in which historical data has been accumulated. As required in 10 CFR 50, Appendix S, instrumentation is provided so that the seismic response of safety-related nuclear plant features can be evaluated promptly after an earthquake.

[START COM 3.7-001] The seismic monitoring program described in this subsection, including the necessary test and operating procedures, will be implemented prior to receipt of fuel on site. **END COM 3.7-001]**

3.7.4.1 Comparison with Regulatory Guide 1.12

The seismic instrumentation program described in the following subsections is consistent with RG 1.12. The procedures for plant response to earthquakes follow the guidelines of the EPRI reports NP-6695 (Reference 3.7-10), NP-5930 (Reference 3.7-11) and TR-100082 (Reference 3.7-12), as permitted by RG 1.166 and RG 1.167.

3.7.4.2 Location and Description of Instrumentation

The following instrumentation and associated equipment of a solid-state digital type are used to measure plant response to earthquake motion:

- triaxial time-history accelerograph (THA): one in the free field, three in the RB and two in the CB
- recording and playback equipment
- annunciators in the main control room

The seismic instrumentation and equipment has sufficient battery capacity to sense and record 25 minutes of seismic motion over a 24-hour period. The associated battery charger is connected to a Nonsafety-related Distributed Control and Information System (N-DCIS), uninterruptible power supply (see Subsection 7.1.4) in accordance with RG 1.12. Information on the installed instruments is kept and maintained at the plant site as part of pre-earthquake planning as required by RG 1.166.

3.7.4.2.1 Time-History Accelerographs

THAs produce a record of the time-varying acceleration at the sensor location. Each triaxial acceleration sensor unit contains three accelerometers mounted in an orthogonal array (two horizontal and one vertical). All acceleration units have their principal axes oriented and aligned with the building major axes used in development of the mathematical models for seismic analysis. The acceleration sensor for each THA has a dynamic range of 1000:1 zero to peak (i.e., 0.001 g to 1.0 g) and a frequency range between 0.2 Hz to 50 Hz.

One THA is located in the free field at the finished grade. A second THA is located on the RB foundation mat. A third THA is located at the RB floor at the same elevation as finished grade elevation. A fourth THA is located at the RB operating floor. In the CB one THA is located on the foundation mat and a second THA at the main control room. The individual THAs located on each building are interconnected for common starting and common timing. The RB THAs also serve the purpose of measuring the response of the containment and its internal structures since the RB and containment are integrated. The specific THA locations on the floor are selected to maintain occupational radiation exposure as low as reasonably achievable in accordance with RG 8.8 for the location, installation, and maintenance of instrumentation.

The THA system is triggered by the accelerometer signals. The trigger is actuated whenever a threshold acceleration of not more than 0.02 g is exceeded for any of the three axes. The initial setpoint of 0.01 g can be changed once an analysis of significant plant operating data indicates that a different setpoint would provide better THA system operation.

3.7.4.2.2 Recording and Playback Equipment

Recording and playback units are provided for multiple channel recording and playback of the THA accelerometer signals. The data recorder has a dynamic range of 1000:1 and its recording speed is 200 samples per second with a 50 Hz bandwidth. The recorder is capable of recording, as a minimum, the 3 seconds prior to seismic trigger actuation, and operating continuously during the period in which the earthquake exceeds the seismic trigger threshold, plus 5 seconds minimum, beyond the last seismic trigger signal. Furthermore, the recorder is capable of a minimum of 25 minutes of continuous recording.

3.7.4.3 Control Room Operator Notification

Activation of the seismic trigger causes an audible and visual annunciation in the main control room to alert the plant operator that a felt earthquake has occurred.

The recorded THA data in the free field is processed, within four hours after the earthquake, to obtain the 5% damped response spectrum and cumulative absolute velocity for each of the three components. The cumulative absolute velocity calculations are prepared according to the procedures described in EPRI report TR-100082 (Reference 3.7-12).

3.7.4.4 Comparison of Measured and Predicted Responses

Within eight hours after the earthquake, operator actions and operator walkdown inspections are performed in accordance with the guidelines described in Reference 3.7-10, as permitted by RG 1.166, to assess the severity of the earthquake. The data from the seismic instrumentation, coupled with information obtained from a plant walkdown, is used to make the initial determination of whether the plant should be shut down, if it has not already been shut down by operational perturbations resulting from the seismic event. The plant is shut down if the walkdown inspections discover damage to equipment that would affect the safe operation of the plant, or 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 cumulative absolute velocity limit as follows:

- Response spectrum limit is exceeded if:
  - at frequencies between 2 and 10 Hz, the recorded response spectral accelerations of 5% damping exceed 1/3 of the corresponding SSE values or 0.2 g, whichever is greater; or
  - at frequencies between 1 and 2 Hz, the recorded response spectral velocities of 5% damping exceed 1/3 of the corresponding SSE values or 152.4 mm/sec (6 in/sec), whichever is greater.
- Cumulative absolute velocity limit is exceeded if the cumulative absolute velocity value calculated according to the procedures in Reference 3.7-12 is greater than 0.16 g-sec.

Following plant shutdown, post-shutdown inspections and tests are performed in accordance with Reference 3.7-10, as permitted by RG 1.167, to determine the physical condition of the plant and its readiness to resume operation. After plant is restarted (or prior to restart if the earthquake caused significant damage to the plant per Reference 3.7-10 definition), long-term evaluations are carried out for engineering assessments of plant structures and equipment using the actual event records to assure their long-term reliability in accordance with Reference 3.7-10 guidelines, as permitted by RG 1.167.

3.7.4.5 In-Service Surveillance

The seismic instrumentation operates during all modes of plant operation including periods of plant shutdown. The maintenance and repair procedures keep the maximum number of instruments in service during plant operation and shutdown. The walkdown inspection following a felt earthquake ensures the safety condition of the plant.

Each of the seismic instruments is demonstrated operable by the performance of the channel check, channel calibration, and channel functional test operations. The channel checks are 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 are performed on a monthly basis. The channel calibration are performed during each refueling. The channel functional test is performed every 6 months.

3.7.5 Site-Specific Information

(1) See Table 2.0-1 for seismology requirements of site-specific SSE ground response spectra.

(2) See Table 2.0-1 for soil properties requirements of site-specific foundation bearing capacities, minimum shear wave velocity and liquefaction potential. For sites not meeting the soil property requirements, a site-specific analysis is required to demonstrate the adequacy of the standard plant design.

3.7.6 References

3.7-1 (Deleted)

3.7-2 Dominion Nuclear North Anna, LLC, “North Anna Early Site Permit Application,” Revision 4, May 2005.

3.7-3 Exelon Generation Company, LLC, “Clinton Early Site Permit Application,” Revision 0, September 2003.

3.7-4 System Energy Resources, INC, “Grand Gulf Early Site Permit Application,” Revision 0, October 2003.

3.7-5 (Deleted)

3.7-6 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.

3.7-7 M. P. Singh, “Seismic Design Input for Secondary Systems, ASCE Mini-Conference on Civil Engineering and Nuclear Power,” Vol. II, Boston, April 1979.

3.7-8 ASCE 4-98, “Seismic Analysis of Safety-Related Nuclear Structures and Commentary.”

3.7-9 R. W. Clough et al., “Dynamics of Structure,” McGraw-Hill, 1975.

3.7-10 Electric Power Research Institute, “Guidelines for Nuclear Plant Response to an Earthquake,” EPRI NP-6695, December 1989.

3.7-11 Electric Power Research Institute, “A Criterion for Determining Exceedance of the Operating Basis Earthquake,” EPRI NP-5930, July 1988.

3.7-12 Electric Power Research Institute, “Standardization of Cumulative Absolute Velocity,” EPRI TR-100082, December 1991.

3.7-13 Stevenson, J.D., and LaPay, W.S., “Amplification Factors to be Used in Simplified Seismic Dynamic Analysis of Piping Systems,” Presented at the ASME Pressure Vessels and Piping Conference, Miami Beach, FL, June 1974.

3.7-14 Lin, C.W. and Esselman, T.C., “Equivalent Static Coefficients for Simplified Seismic Analysis of Piping Systems,” Proc., 7th International Conference on Structural Mechanics in Reactor Technology, August 1983.

3.7-15 Kennedy, R.P. and Shinozuka, M., “Recommended Minimum Power Spectral Density Functions Compatible with NRC Regulatory Guide 1.60 Response Spectrum,” January 1989, Appendix B, NUREG/CR-5347.

3.7-16 Brookhaven National Laboratory, BNL 52361, “Seismic Design and Evaluation guidelines for the Department of Energy High-Level Waste Storage Tanks and Appurtenances,” October 1995.

3.7-17 R. Morante and Y. Wang, “Reevaluation of Regulatory Guidance on Modal Response Combination Methods for Seismic response Spectrum Analysis,” NUREG/CR-6645, U.S. Nuclear Regulatory Commission, Washington, DC, December 1999.

3.7-18 "R. McGuire, W. Silva and C. Costantino, “Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines.”

3.7-19 D. Coats, “Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria,” NUREG/CR-1161, U.S. Nuclear Regulatory Commission, Washington, DC, May 1980.

Table 3.7-1 Damping Values for SSE Dynamic Analysis

Components	Percent of Critical Damping
Reinforced concrete structures	7.0
Welded and friction bolted steel assemblies/structures	4.0
Bearing bolted steel assemblies/structures	7.0
Equipment	3.0
Piping systems[1]	4.0
RPV, skirt, shroud, chimney, and separators	4.0
Control rod guide tubes and CRD housings	2.0
Fuel assemblies	6.0
Cable tray system[2]	
- maximum cable loading	10.0
- empty	7.0
- sprayed-on fire retardant or other cable-restraining mechanism	7.0
Conduit systems[2]	
- maximum cable fill	7.0
- empty	5.0
HVAC ductwork	
- companion angle	7.0
- pocket lock	10.0
- welded	4.0

Notes:

1. See Figure 3.7-37 for alternative damping values for response spectra analysis of ASME Section III, Division 1, Class 1, 2, and 3, and ASME B31.1 piping systems.
2. Notes to Table 4 of RG 1.61 Revision 1 apply.

Table 3.7-2 5%-Damped Target Spectra of Single Envelope Design Ground Motion at Foundation Level

Horizontal		Vertical	
Frequency (Hz)	**S_a (g)**	**Frequency (Hz)**	**S_a (g)**
0.1	0.023	0.1	0.015
0.25	0.141	0.25	0.094
2.5	0.939	3.5	0.894
9	0.783	9	0.783
10	0.92	10	0.724
20	1.35	20	1.11
30	1.35	30	1.24
50	1.1	50	1.21
100	0.5	100	0.5

Note:

Applicable scale factors are:

- *1.0 for embedment depths equal to or greater than 14.9 m (49 ft.)*
- *1.35 for embedment depths equal to or less than 2.35 m (7.7 ft.)*
- *Linearly interpolated between 1.0 and 1.35 for intermediate embedment depths.*

Table 3.7-3 Summary of Methods of Seismic Analysis for Primary Building Structures

Building Structure	Site Condition	SSI Model	Analysis Method	Three Components Combination	Modal Combination	Computer Program	Use of Analysis Output
Reactor Building including containment and containment internal structures	*Uniform Sites*	*3D lumped mass stick coupled with soil springs*	*Direct integration in the time domain*	*Algebraic Sum*	*n/a*	*DAC3N*	*Max. forces, moments, acceleration, floor response spectra and max. relative displacements. Interface loads with foundation medium not used.*
Reactor Building including containment and containment internal structures	*Uniform and Layered Sites*	*3D lumped mass stick coupled with soil finite elements*	*Frequency response in the frequency domain.*	*Algebraic Sum*	*n/a*	*SASSI2000*	*Max. forces, moments, acceleration, floor response spectra, max. relative displacements and interface loads with foundation medium.*
Fuel Building	*Uniform Sites*	*Integrated with the Reactor Building models*	*Direct integration in the time domain*	*Algebraic Sum*	*n/a*	*DAC3N*	*Max. forces, moments, acceleration and floor response spectra. Interface loads with foundation medium not used.*
Fuel Building	*Uniform and Layered Sites*	*Integrated with the Reactor Building models*	*Frequency response in the frequency domain.*	*Algebraic Sum*	*n/a*	*SASSI2000*	*Max. forces, moments, acceleration, floor response spectra and interface loads with foundation medium.*
Control Building	*Uniform Sites*	*3D lumped mass stick coupled with soil springs*	*Direct integration in the time domain*	*Algebraic Sum*	*n/a*	*DAC3N*	*Max. forces, moments, acceleration and floor response spectra. Interface loads with foundation medium not used.*
Control Building	*Uniform and Layered Sites*	*3D lumped mass stick coupled with soil finite elements*	*Frequency response in the frequency domain.*	*Algebraic Sum*	*n/a*	*SASSI2000*	*Max. forces, moments, acceleration, floor response spectra and interface loads with foundation medium.*
Firewater Service Complex	*Uniform Sites*	*3D lumped mass stick coupled with soil springs*	*Direct integration in the time domain*	*Algebraic Sum*	*n/a*	*DAC3N*	*Max. forces, moments, acceleration and floor response spectra. Interface loads with foundation medium not used.*
Firewater Service Complex	*Uniform and Layered Sites*	*3D lumped mass stick coupled with soil finite elements*	*Frequency response in the frequency domain*	*Algebraic Sum*	*n/a*	*SASSI2000*	*Max. forces, moments, acceleration, floor response spectra and interface loads with foundation medium.*

Figure 3.7-1 Horizontal SSE Design Spectra, Generic Site

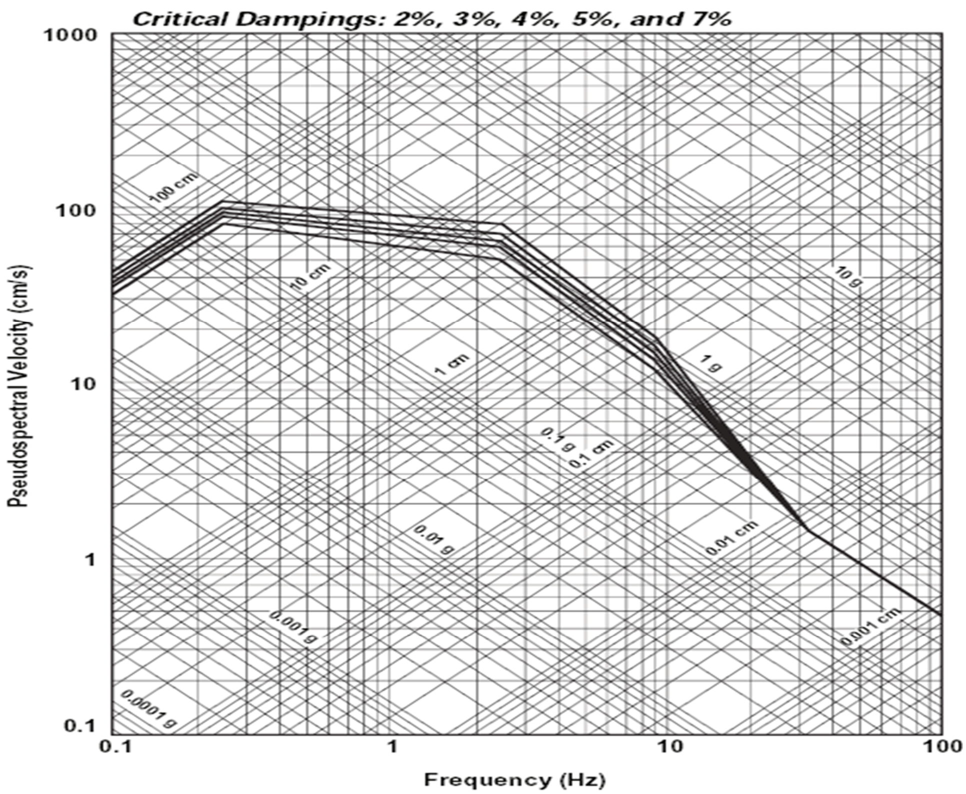


Figure 3.7-2 Vertical SSE Design Spectra, Generic Site

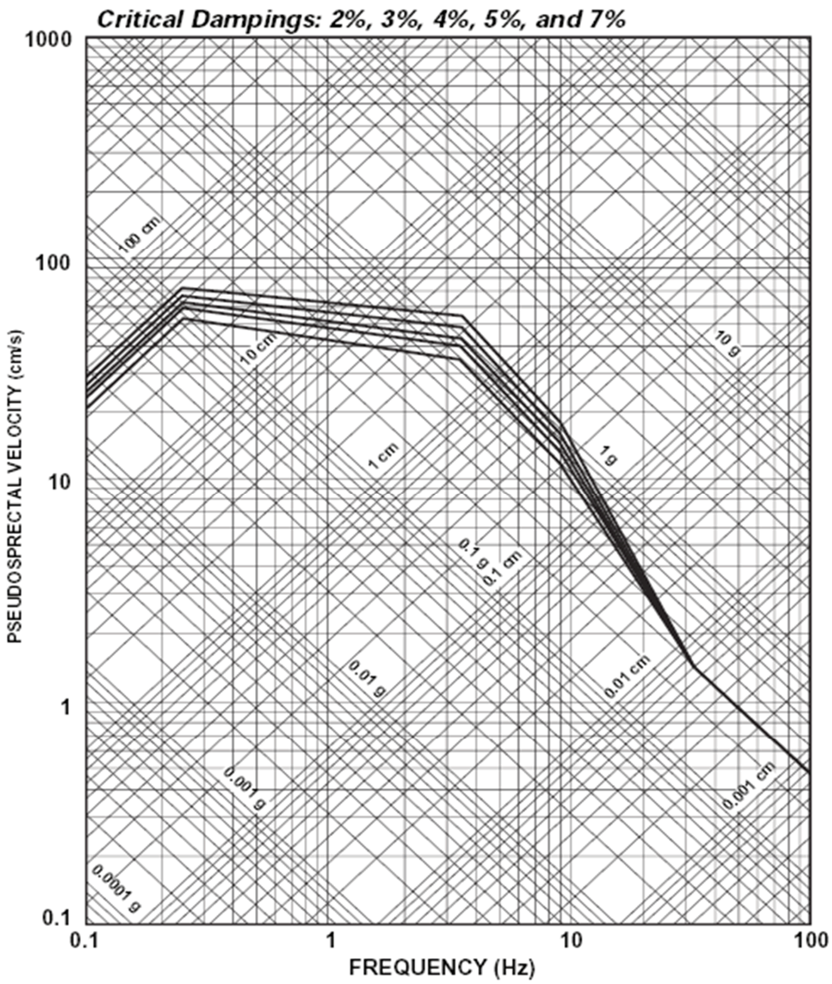


Figure 3.7-3 Horizontal, H1 Component Time History, Generic Site

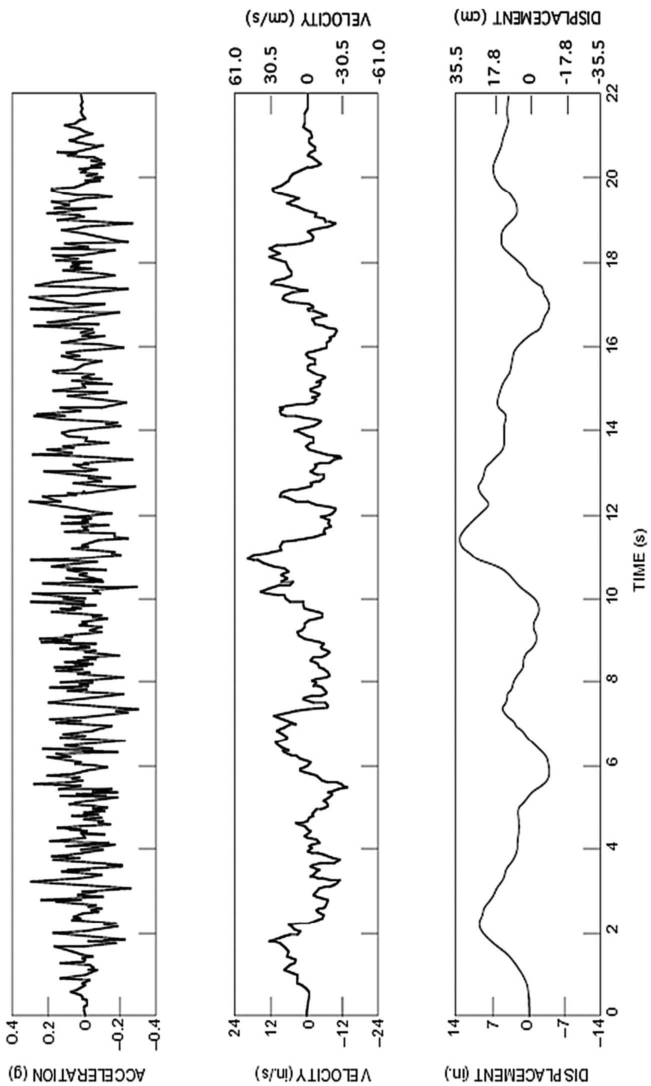


Figure 3.7-4 Horizontal, H2 Component Time History, Generic Site

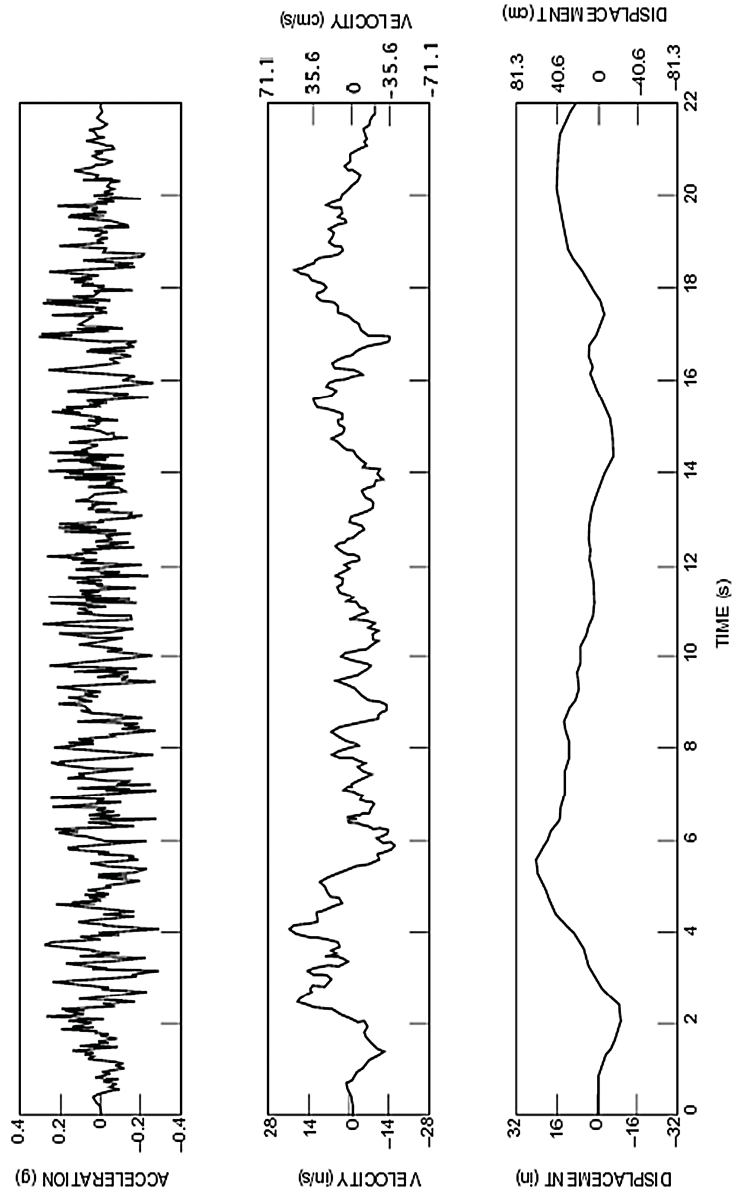


Figure 3.7-5 Vertical, Component Time History, Generic Site

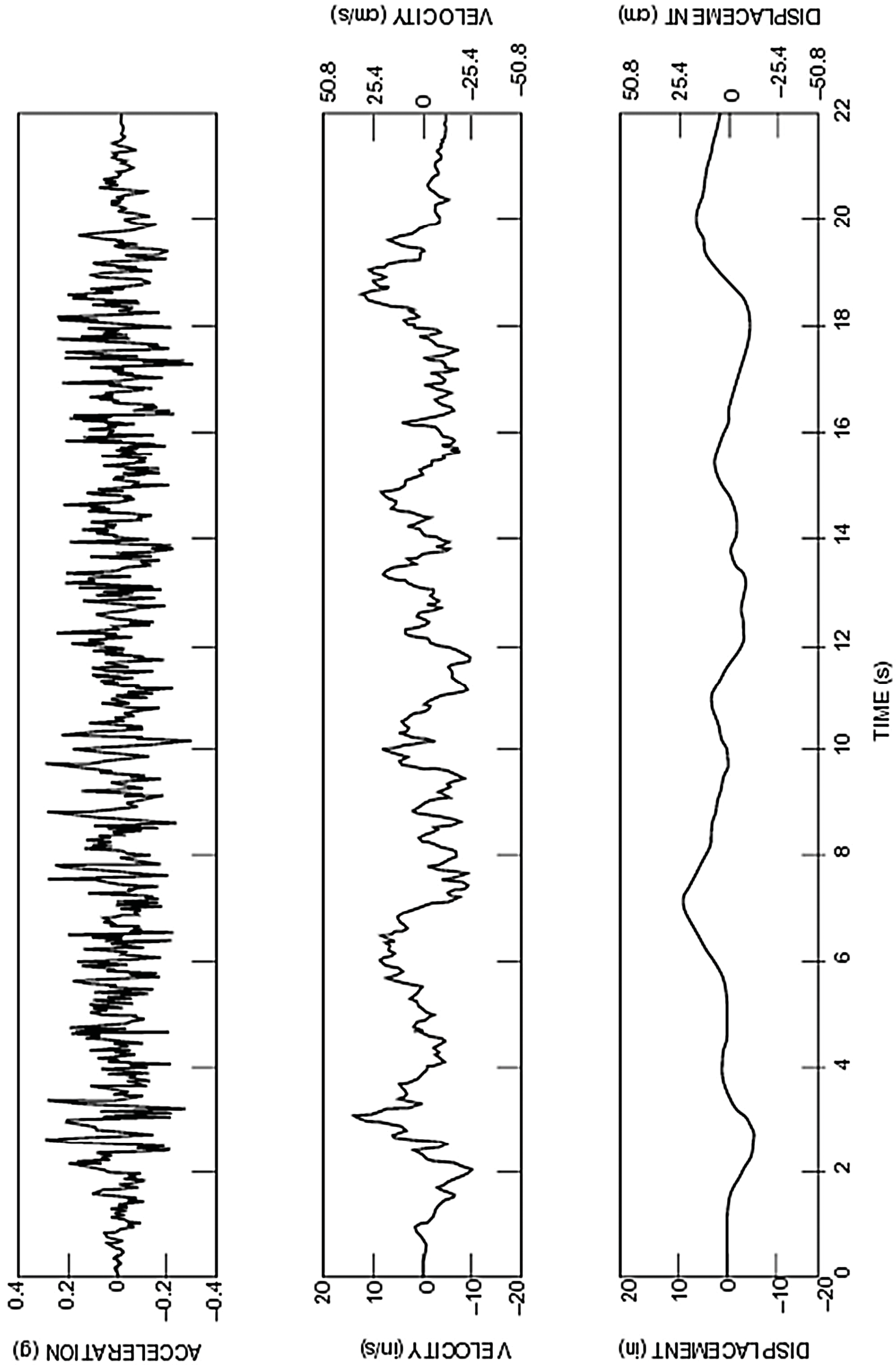


Figure 3.7-6 2% Damped Response Spectra, H1 Component, Generic Site

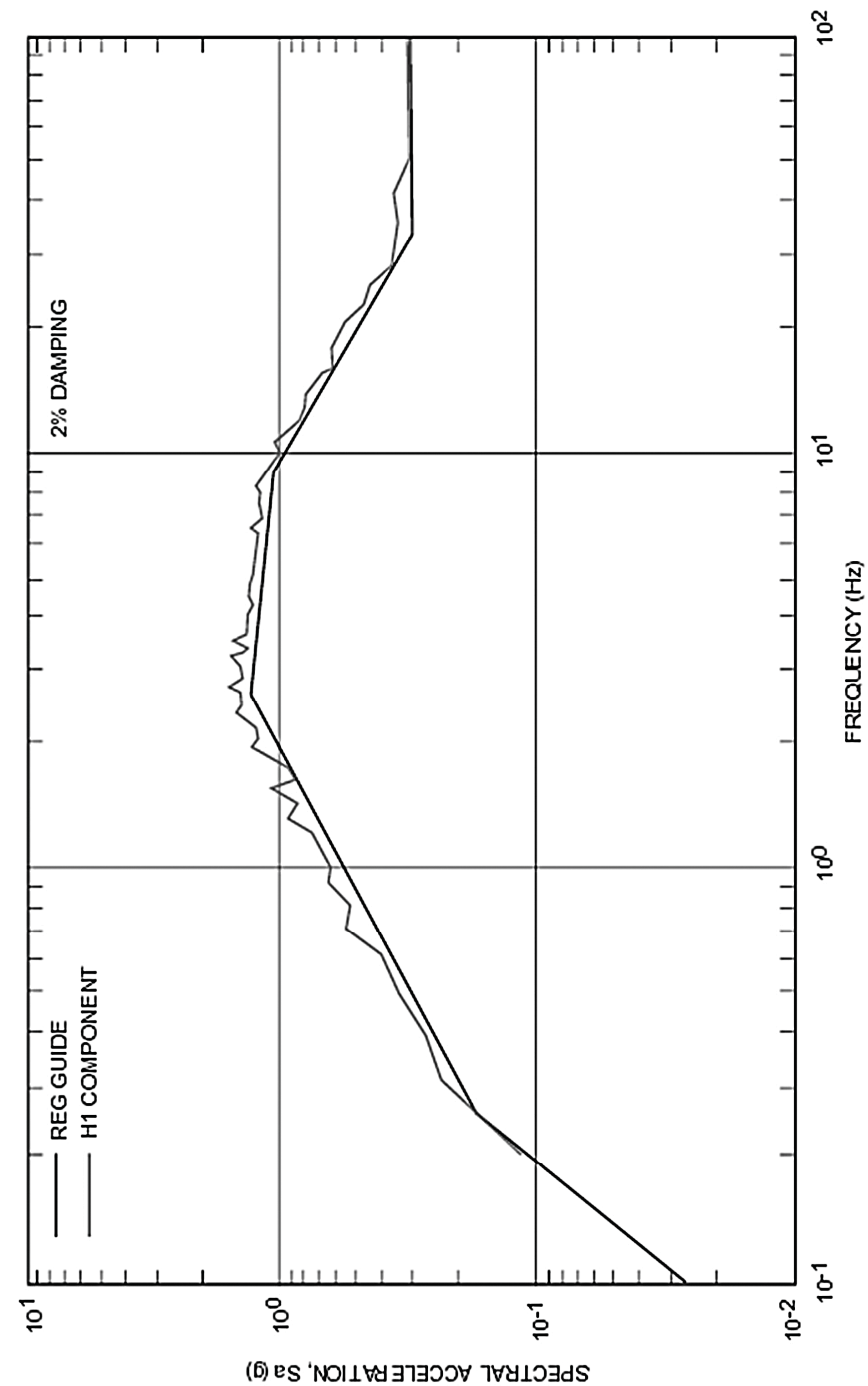


Figure 3.7-7 3% Damped Response Spectra, H1 Component, Generic Site

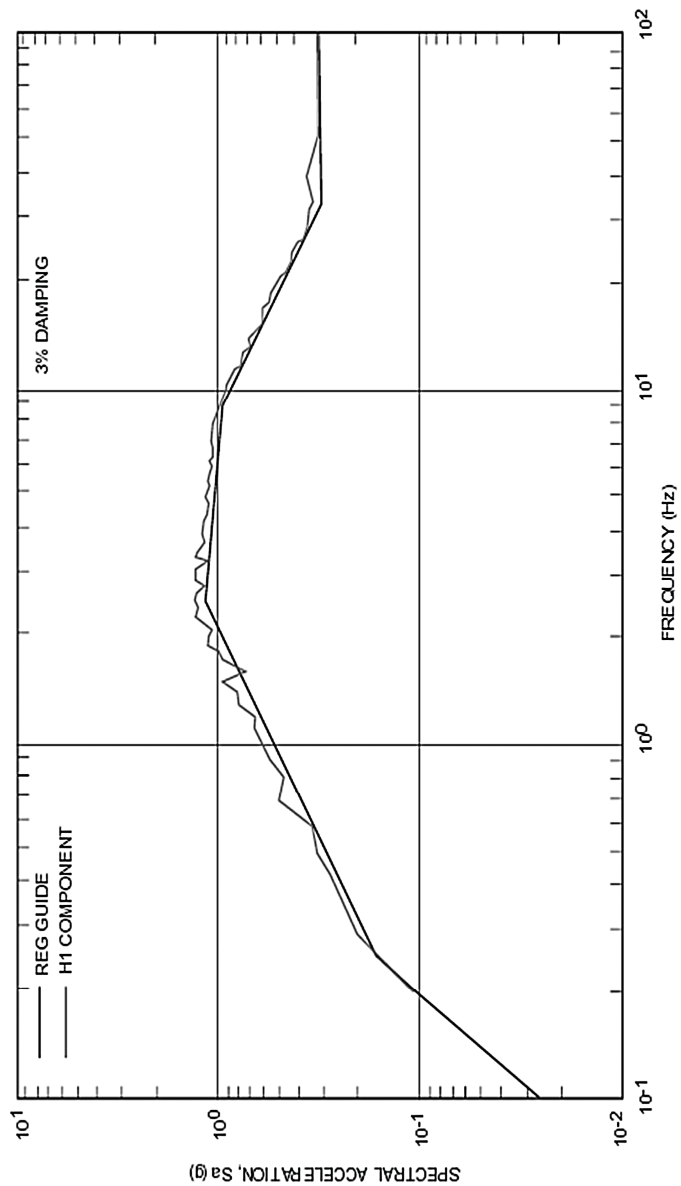


Figure 3.7-8 4% Damped Response Spectra, H1 Component, Generic Site

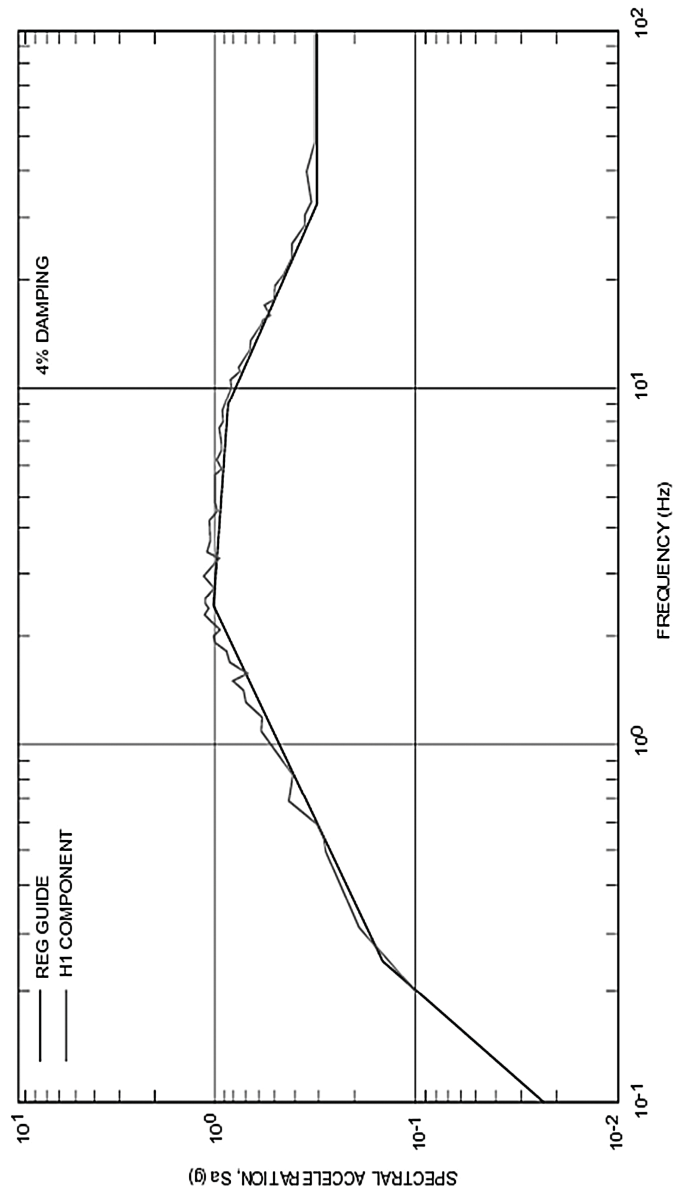


Figure 3.7-9 5% Damped Response Spectra, H1 Component, Generic Site

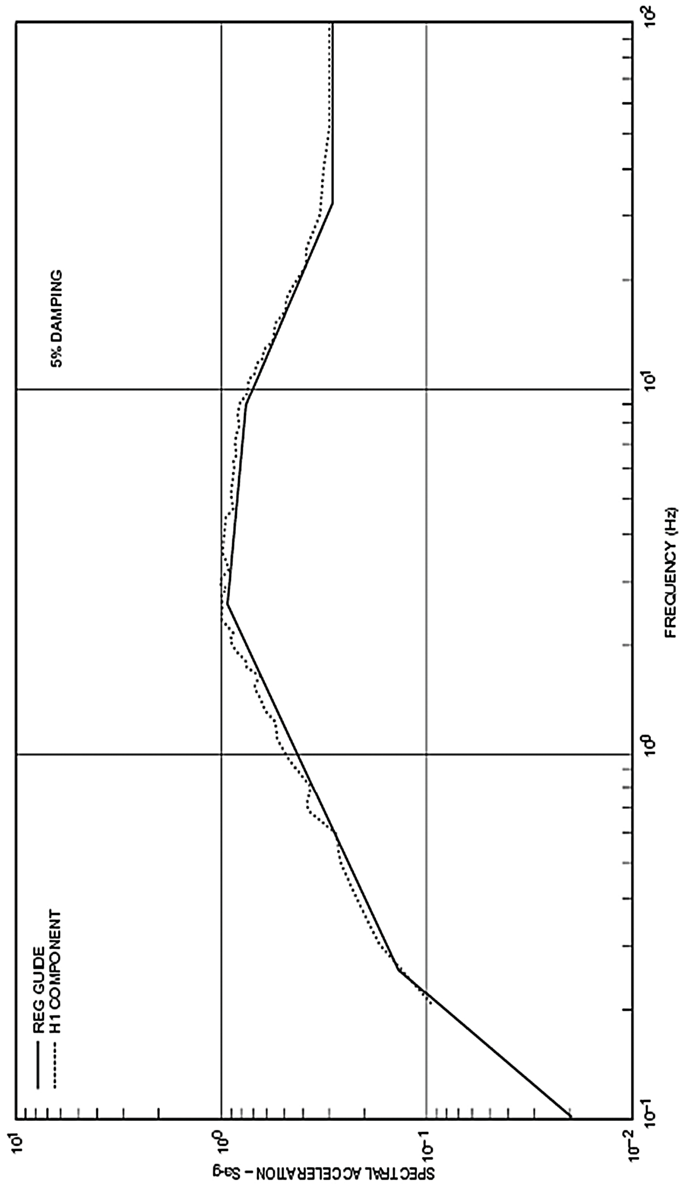


Figure 3.7-10 7% Damped Response Spectra, H1 Component, Generic Site

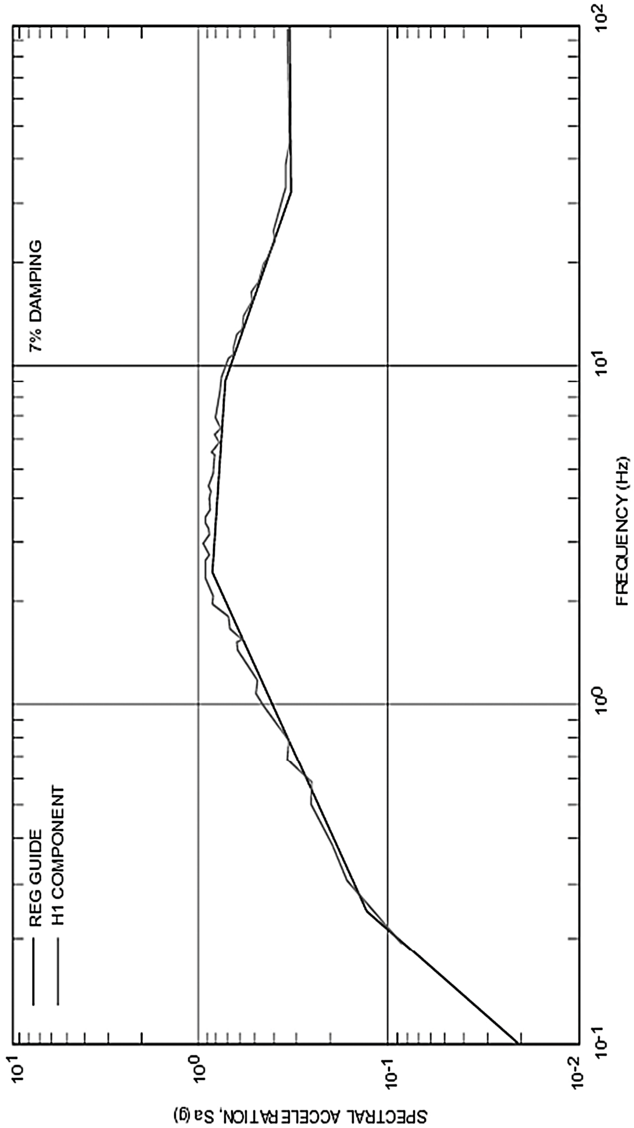


Figure 3.7-11 2% Damped Response Spectra, H2 Component, Generic Site

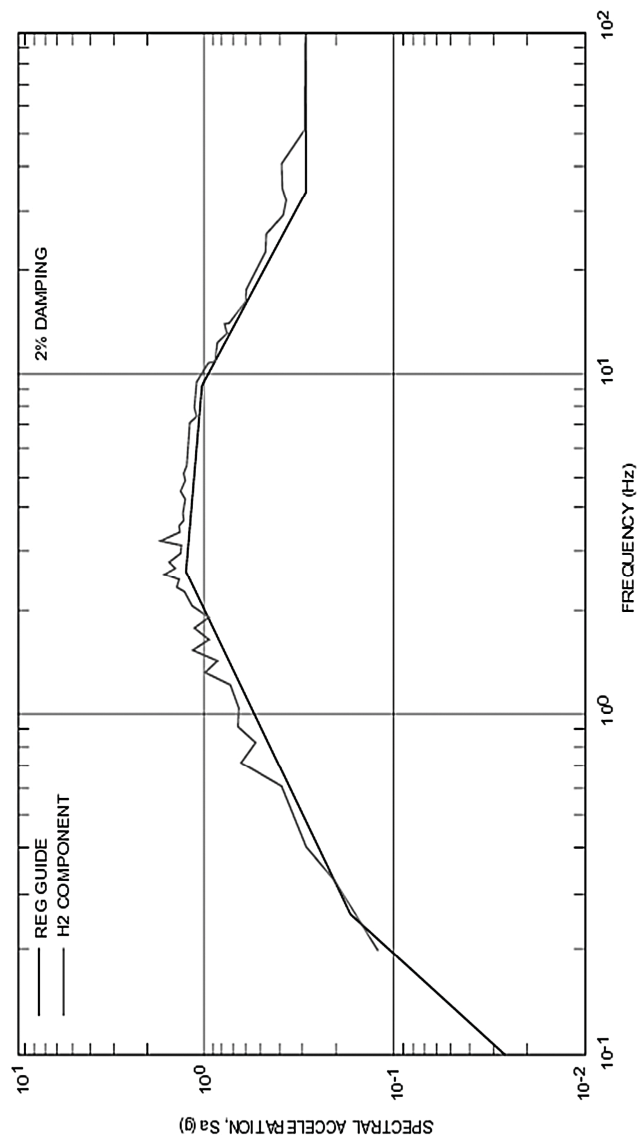


Figure 3.7-12 3% Damped Response Spectra, H2 Component, Generic Site

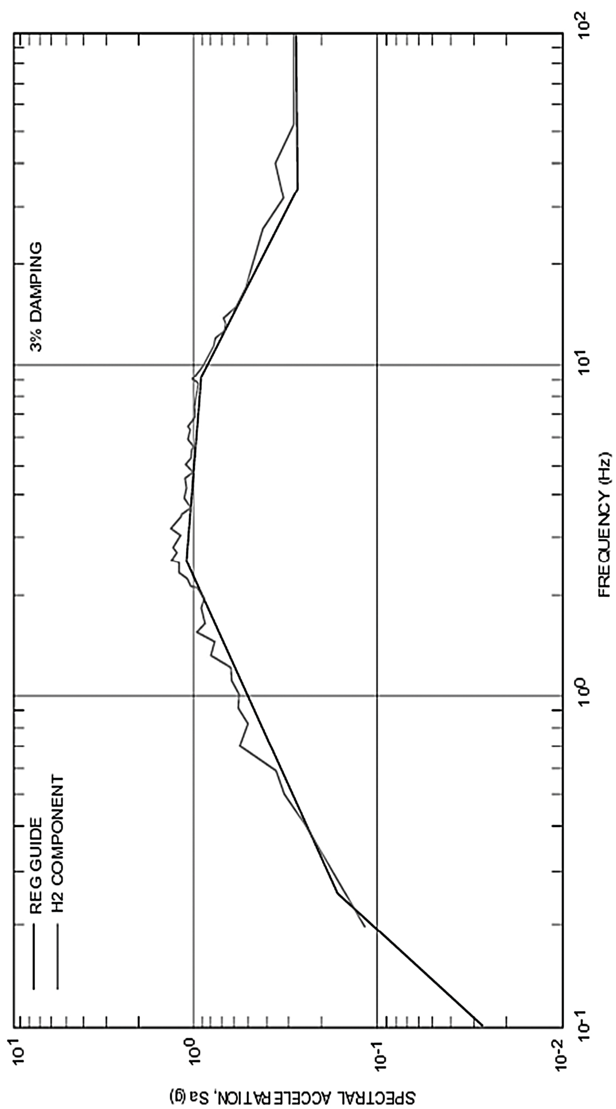


Figure 3.7-13 4% Damped Response Spectra, H2 Component, Generic Site

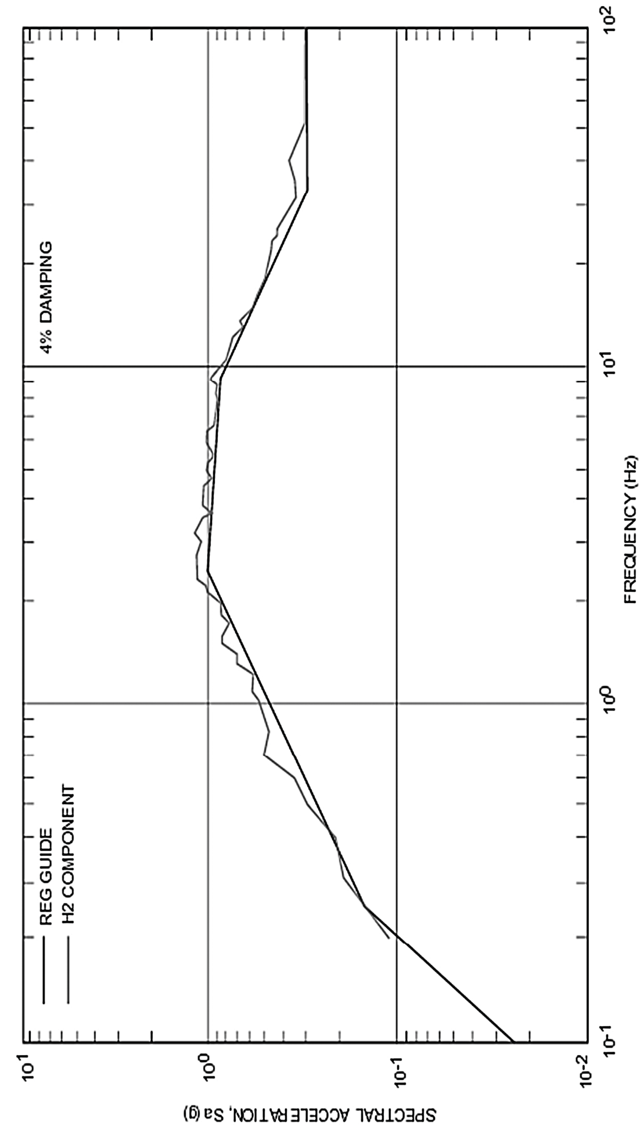


Figure 3.7-14 5% Damped Response Spectra, H2 Component, Generic Site

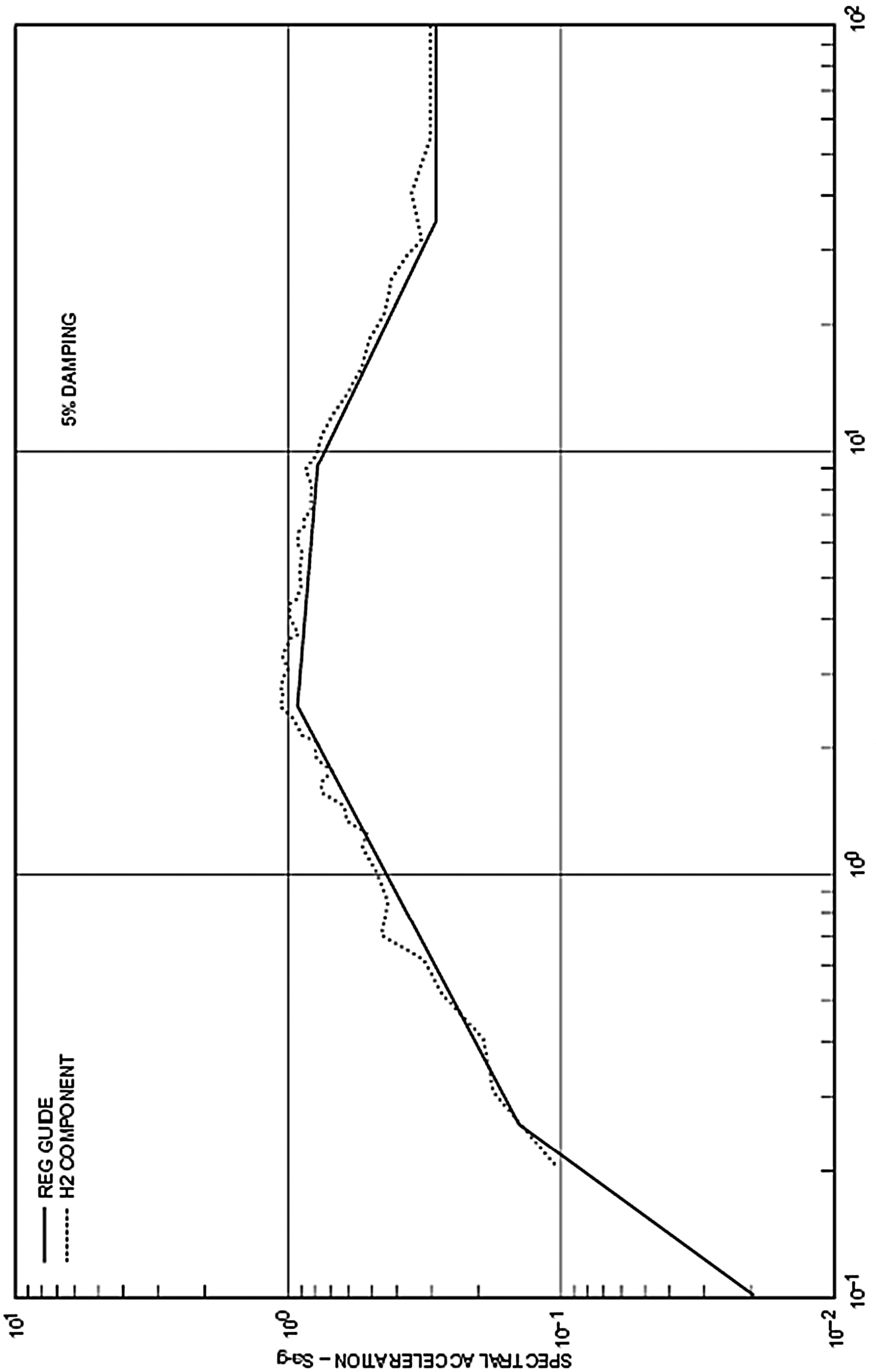


Figure 3.7-15 7% Damped Response Spectra, H2 Component, Generic Site

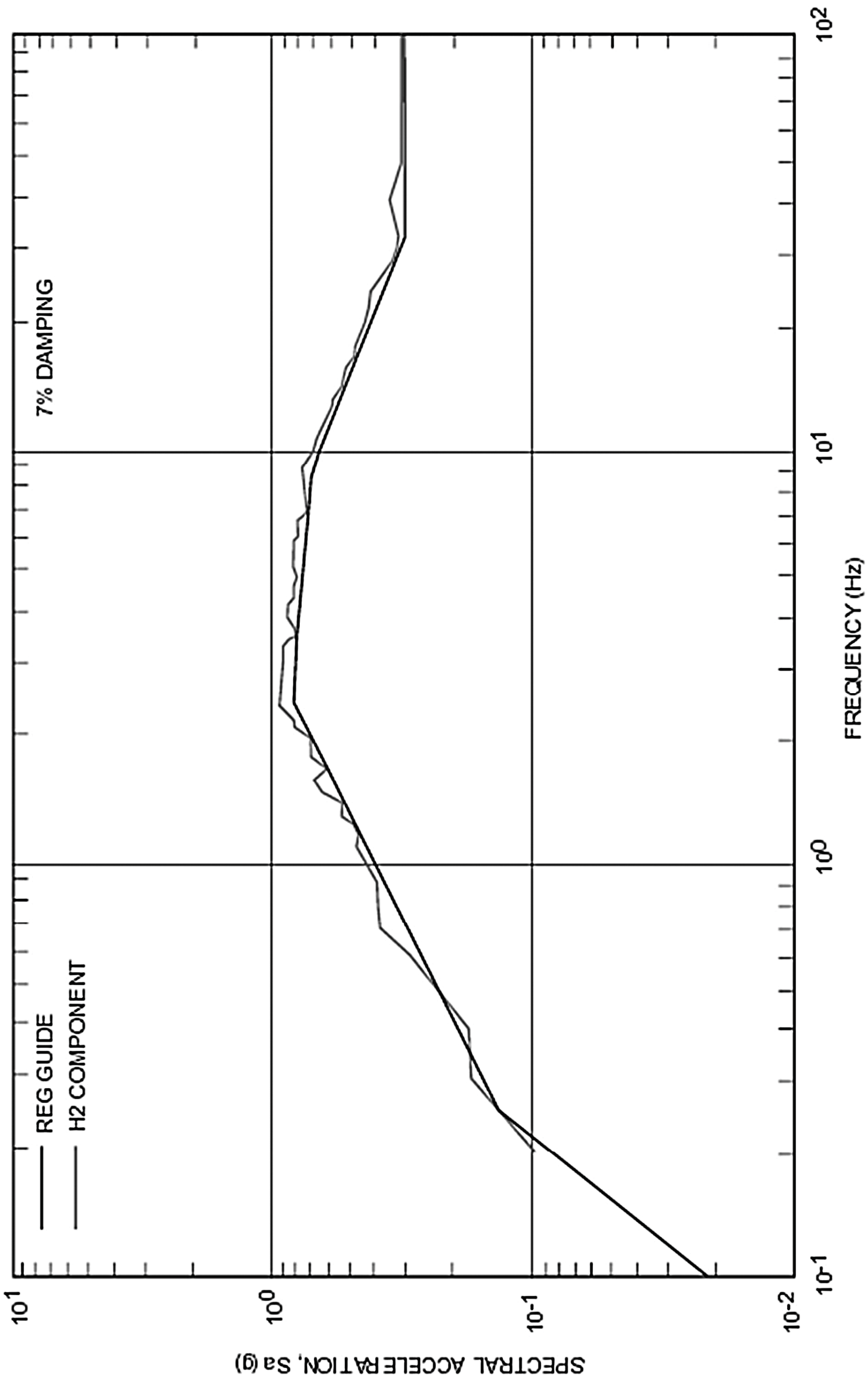


Figure 3.7-16 2% Damped Response Spectra, Vertical Component, Generic Site

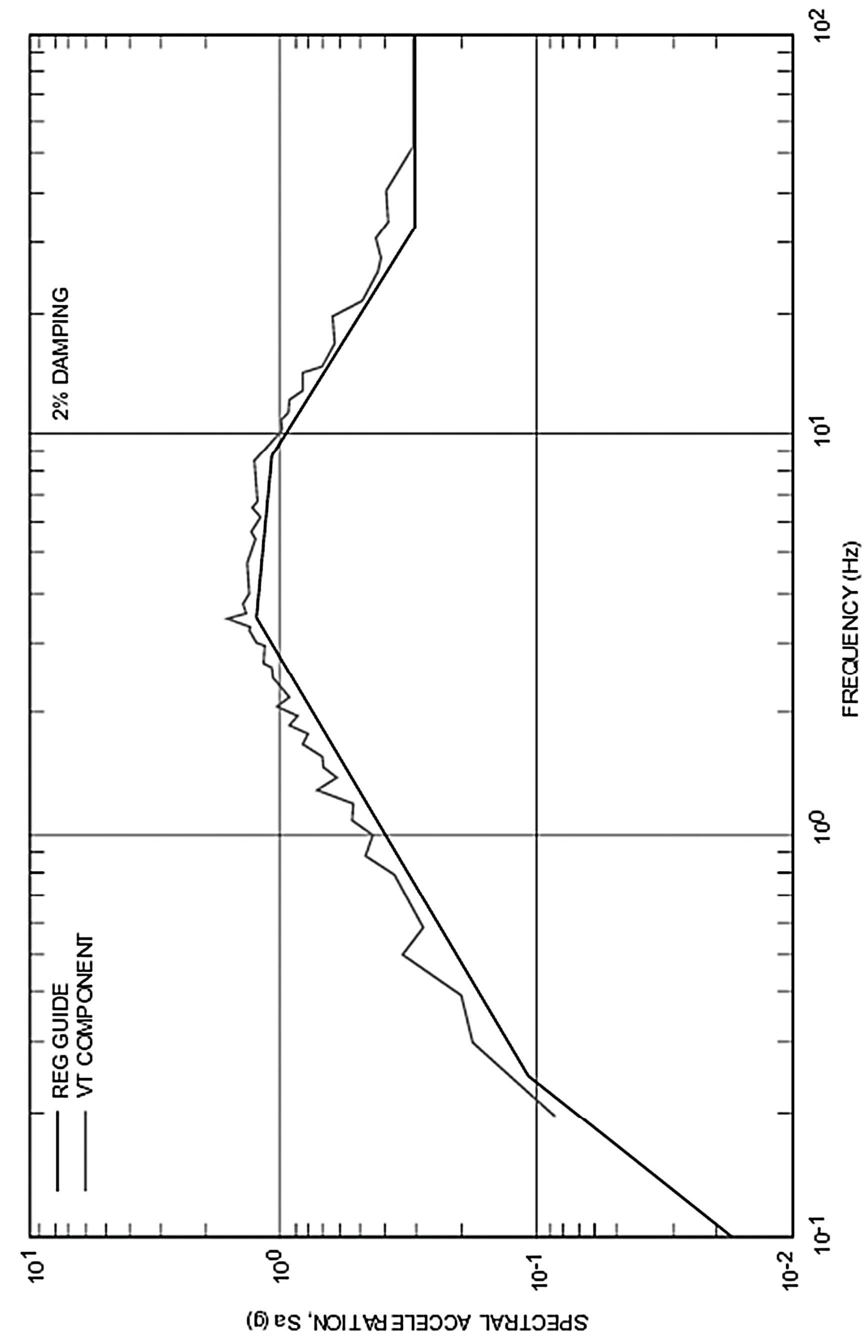


Figure 3.7-17 3% Damped Response Spectra, Vertical Component, Generic Site

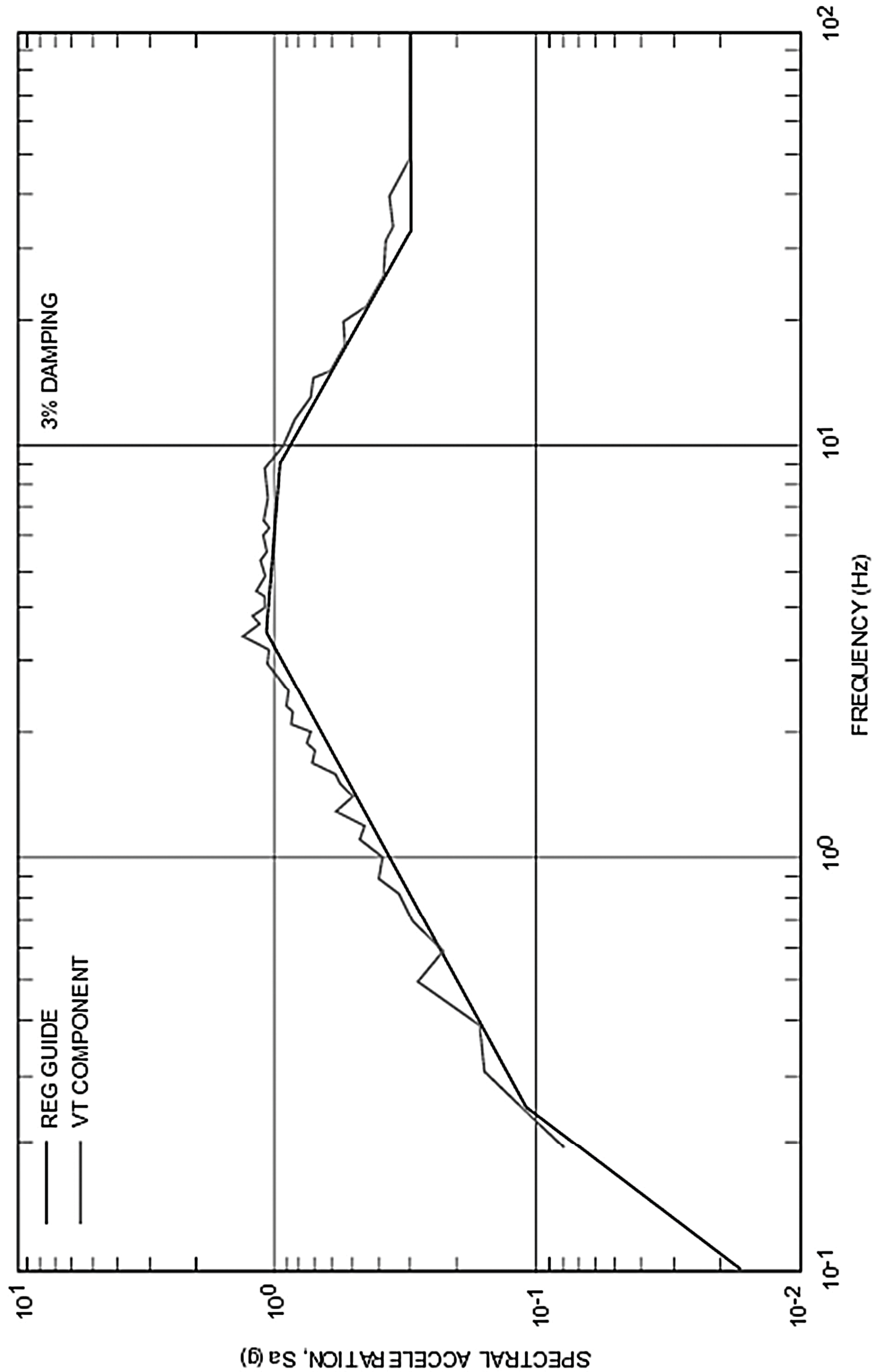


Figure 3.7-18 4% Damped Response Spectra, Vertical Component, Generic Site

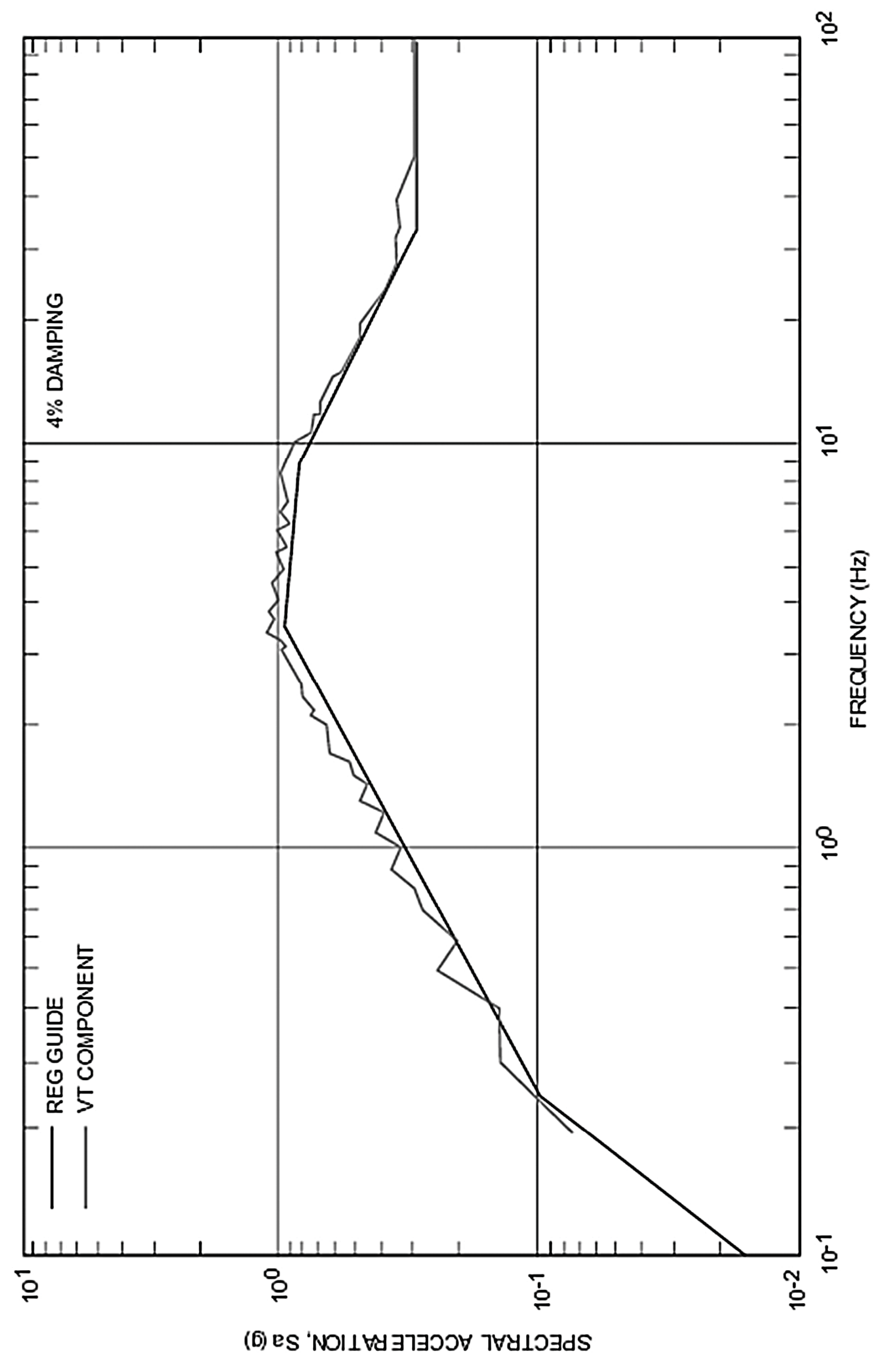


Figure 3.7-19 5% Damped Response Spectra, Vertical Component, Generic Site

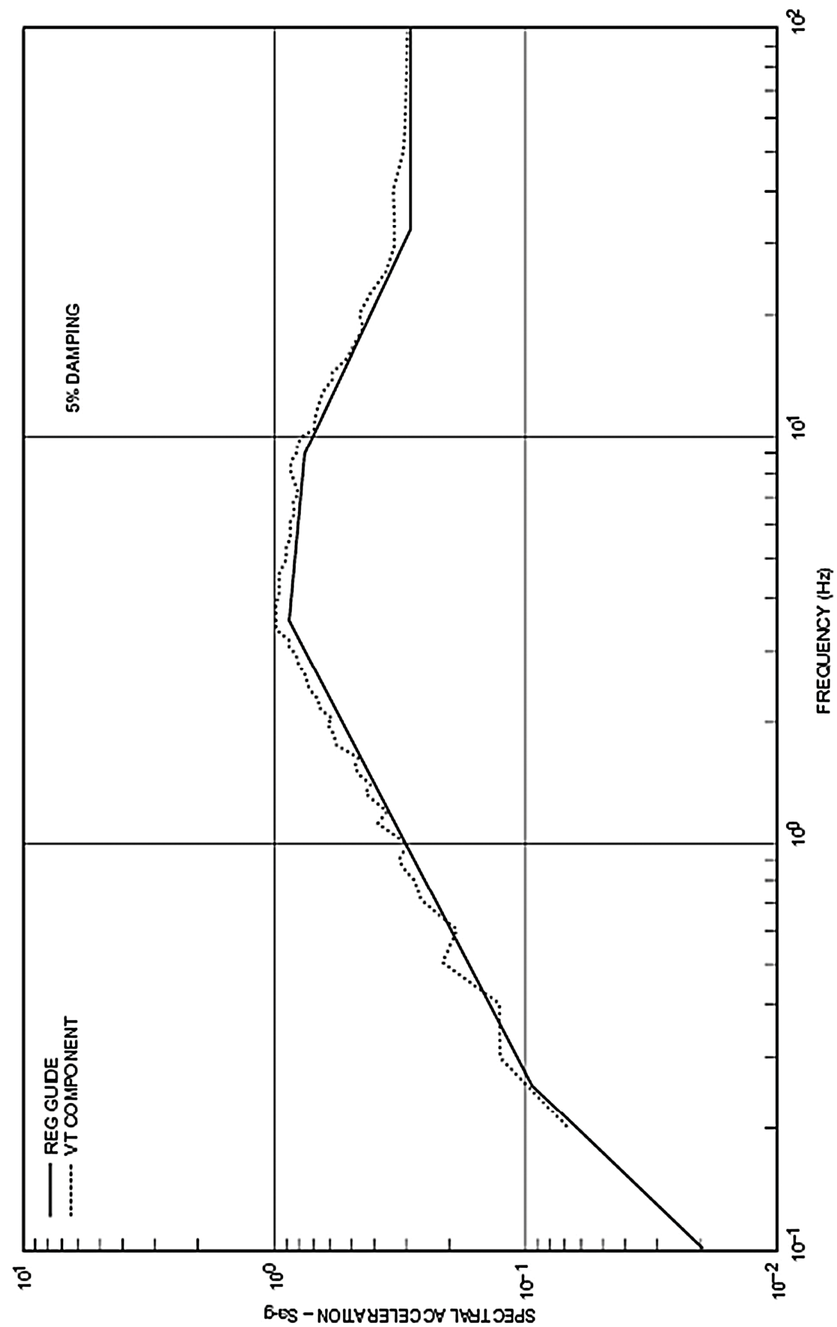


Figure 3.7-20 7% Damped Response Spectra, Vertical Component, Generic Site

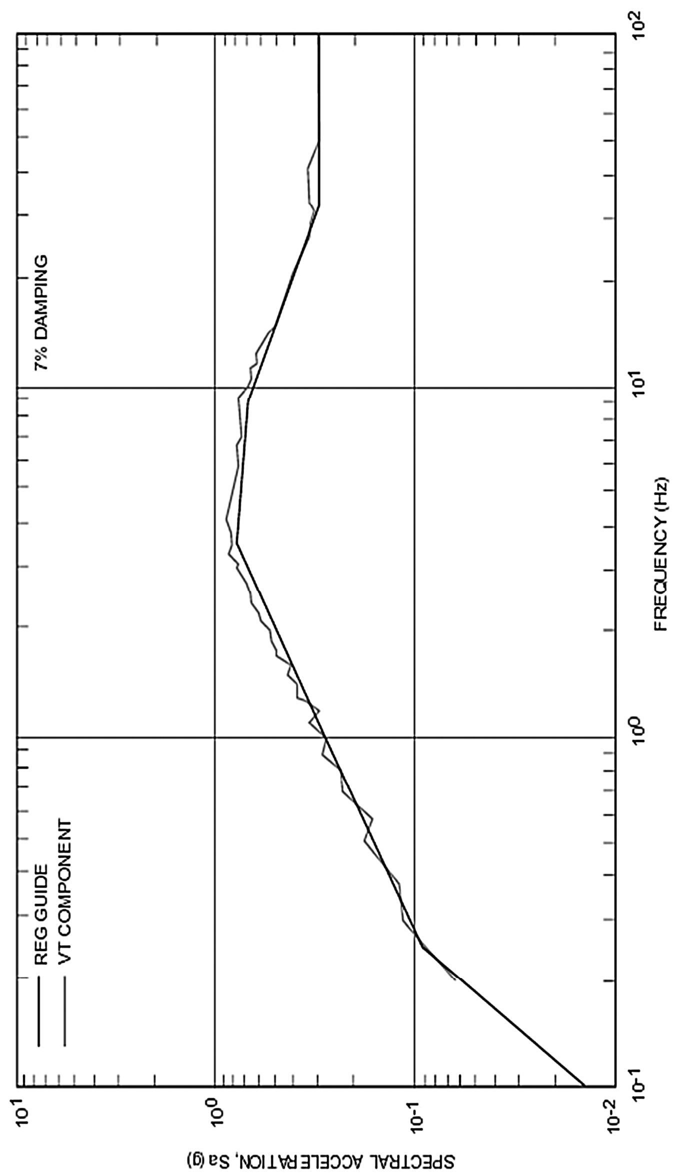


Figure 3.7-21 Power Spectral Density Function, H1 Component, Generic Site

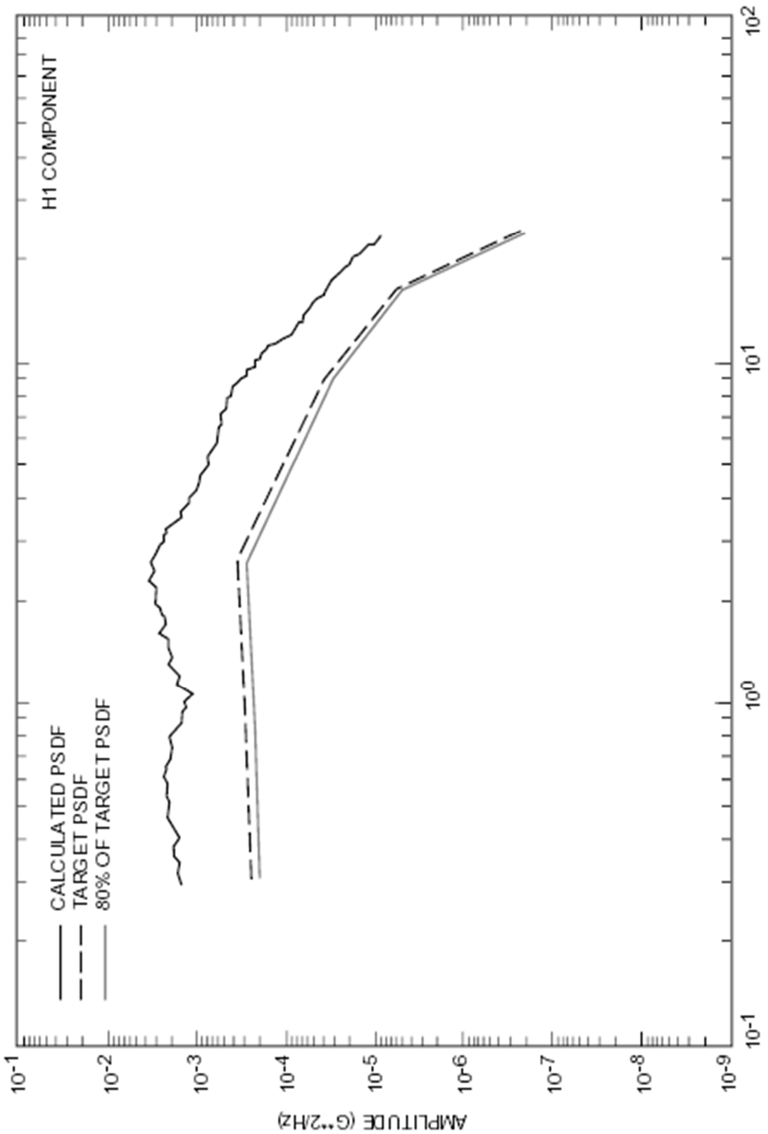


Figure 3.7-22 Power Spectral Density Function, H2 Component, Generic Site

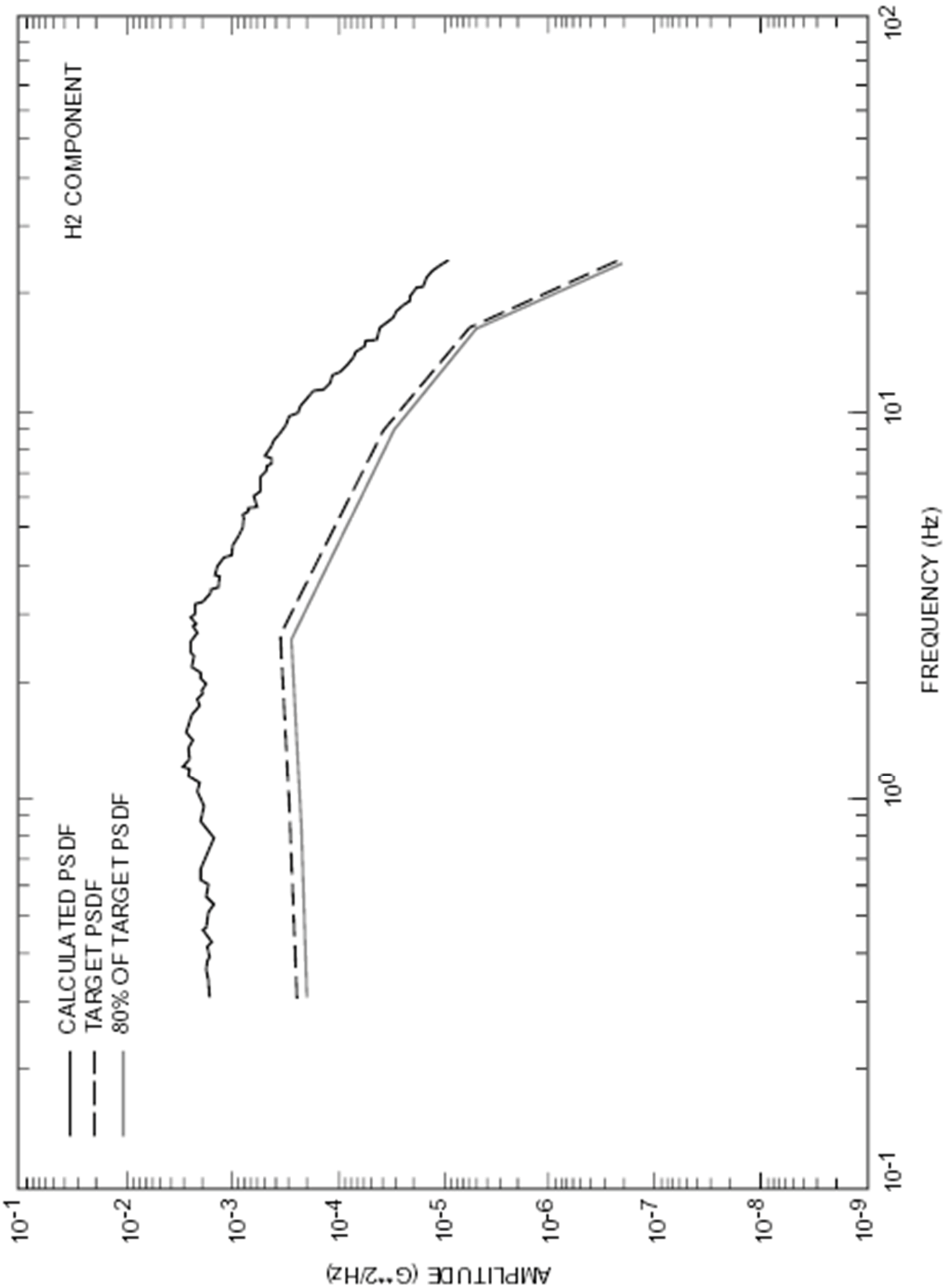


Figure 3.7-23 Power Spectral Density Function, Vertical Component, Generic Site

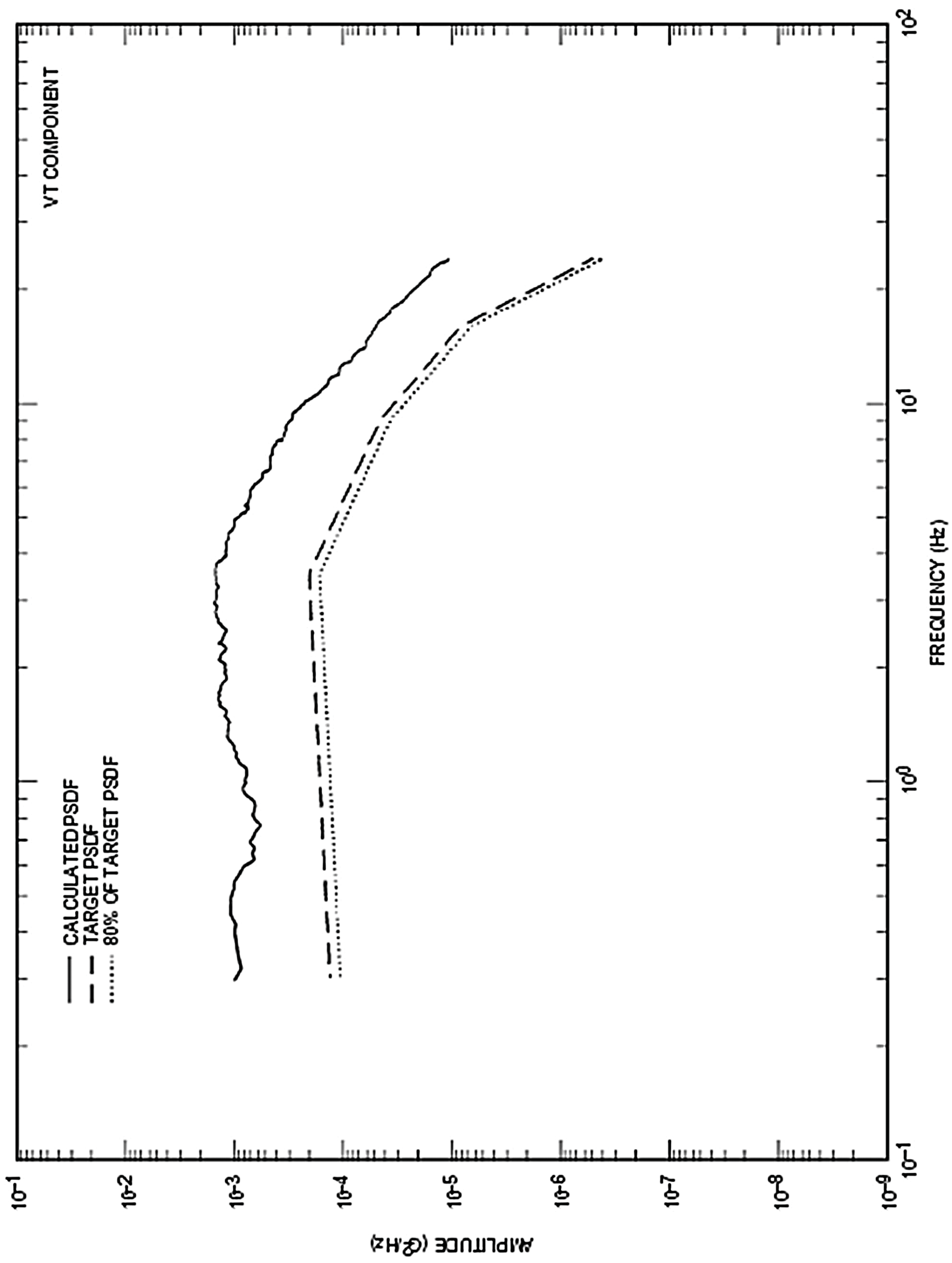
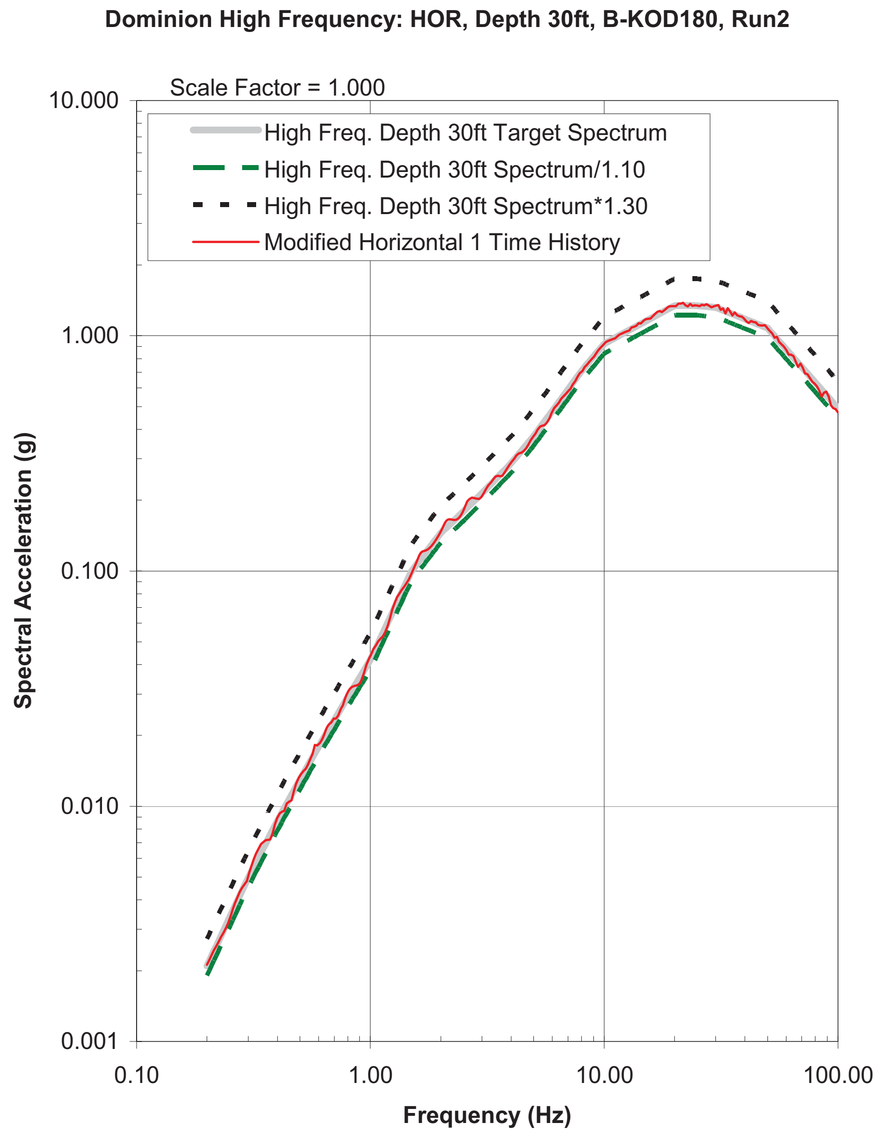


Figure 3.7-24 North Anna ESP Horizontal H1 Target Spectrum at ESBWR CB Base



Note: B-KOD is the Livermore earthquake of January 27, 1980 recorded at the Kodak Building Site in San Ramon, California. “HOR” is “Horizontal”.

Figure 3.7-25 North Anna ESP Horizontal H1 Time Histories at ESBWR CB Base

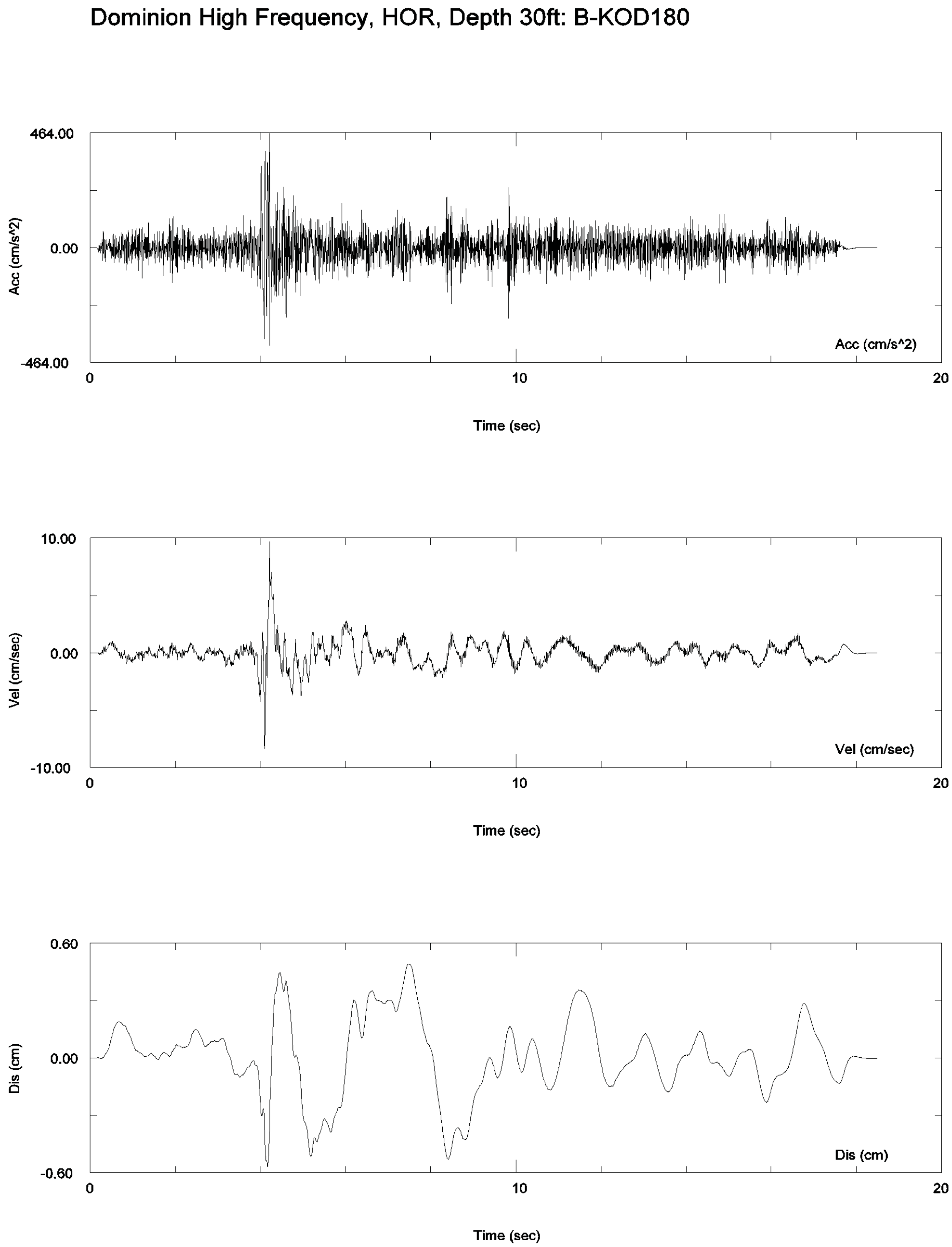
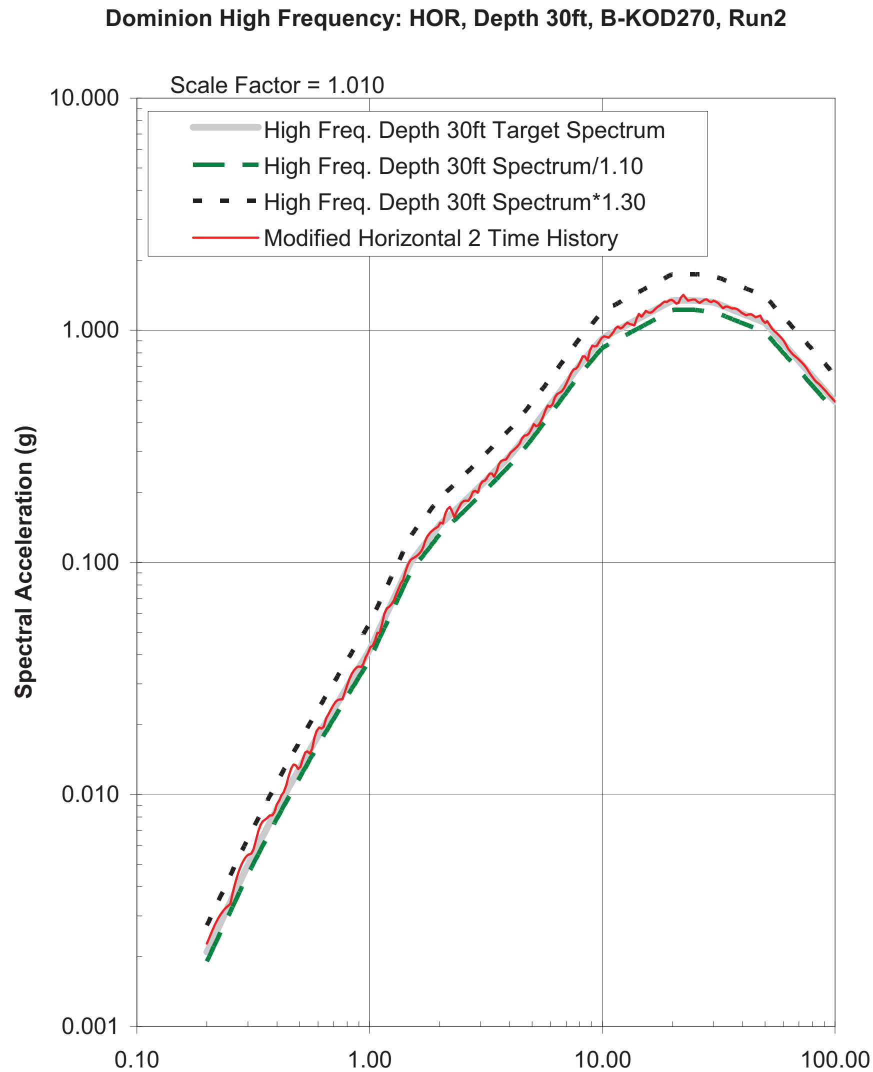


Figure 3.7-26 North Anna ESP Horizontal H2 Target Spectrum at ESBWR CB Base



Note: B-KOD is the Livermore earthquake of January 27, 1980 recorded at the Kodak Building Site in San Ramon, California. “HOR” is “Horizontal”.

Figure 3.7-27 North Anna ESP Horizontal H2 Time Histories at ESBWR CB Base

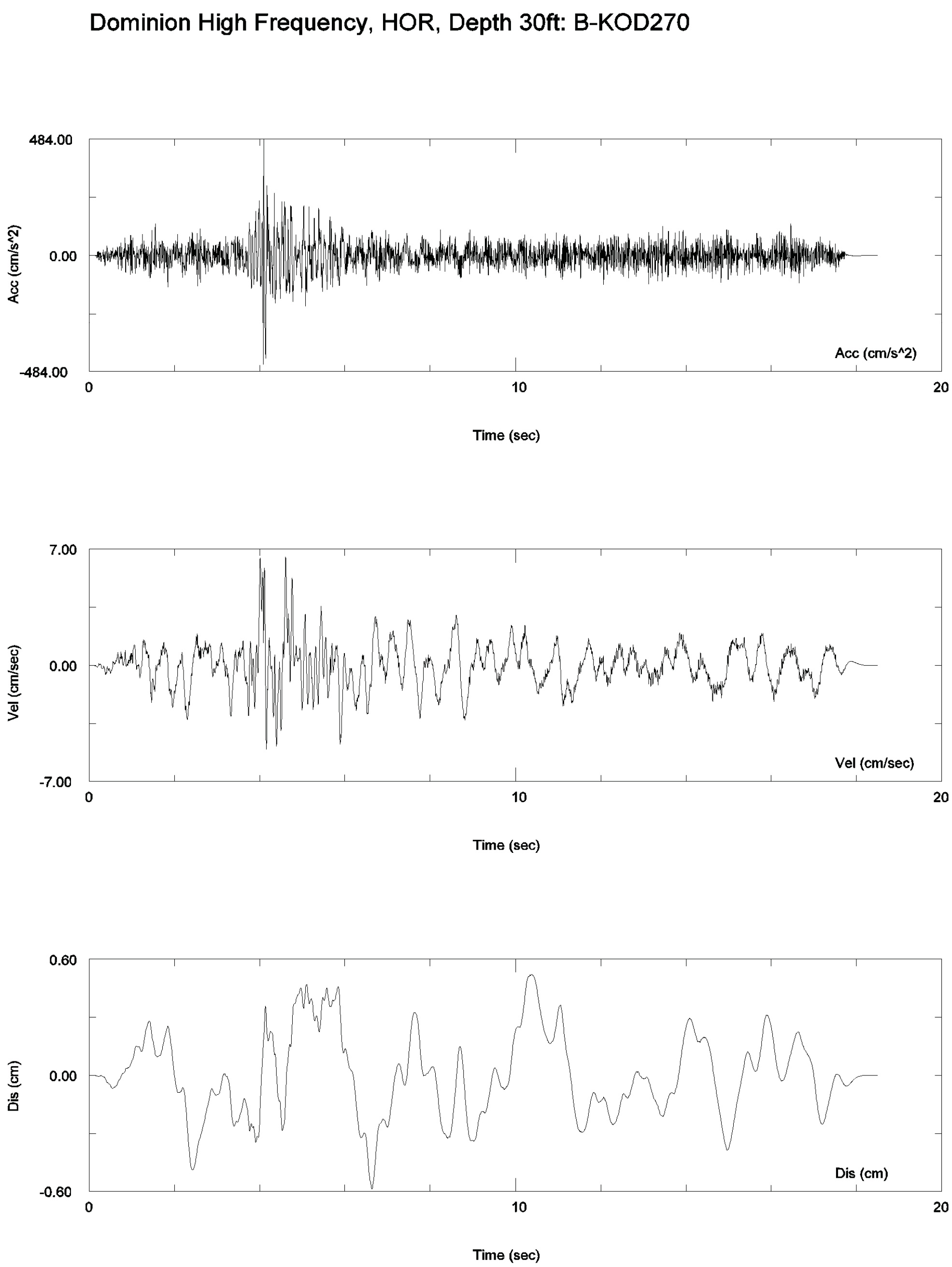
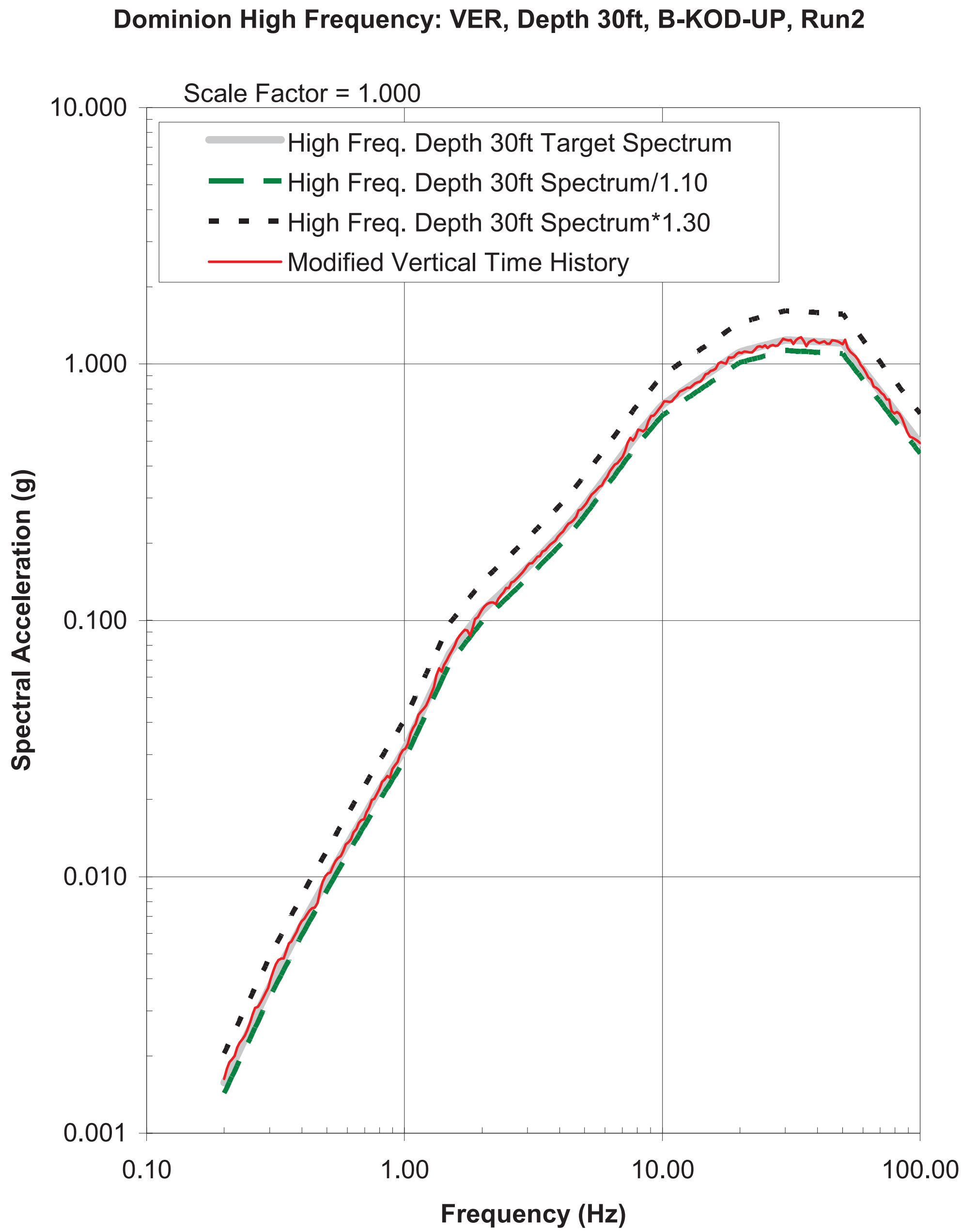


Figure 3.7-28 North Anna ESP Vertical Target Spectrum at ESBWR CB Base



Note: B-KOD is the Livermore earthquake of January 27, 1980 recorded at the Kodak Building Site in San Ramon, California. “VER” is “Vertical”.

Figure 3.7-29 North Anna ESP Vertical Time Histories at ESBWR CB Base

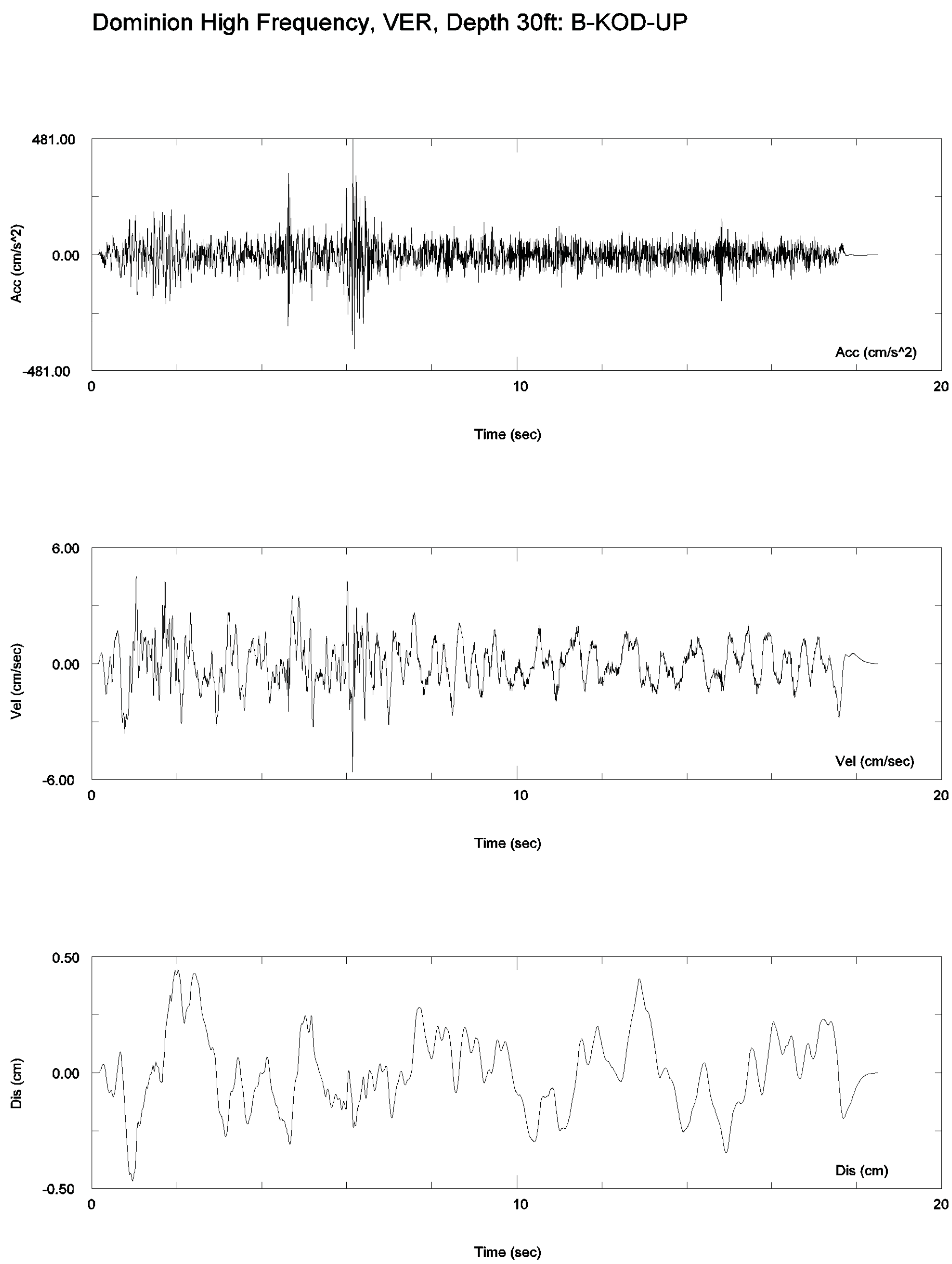
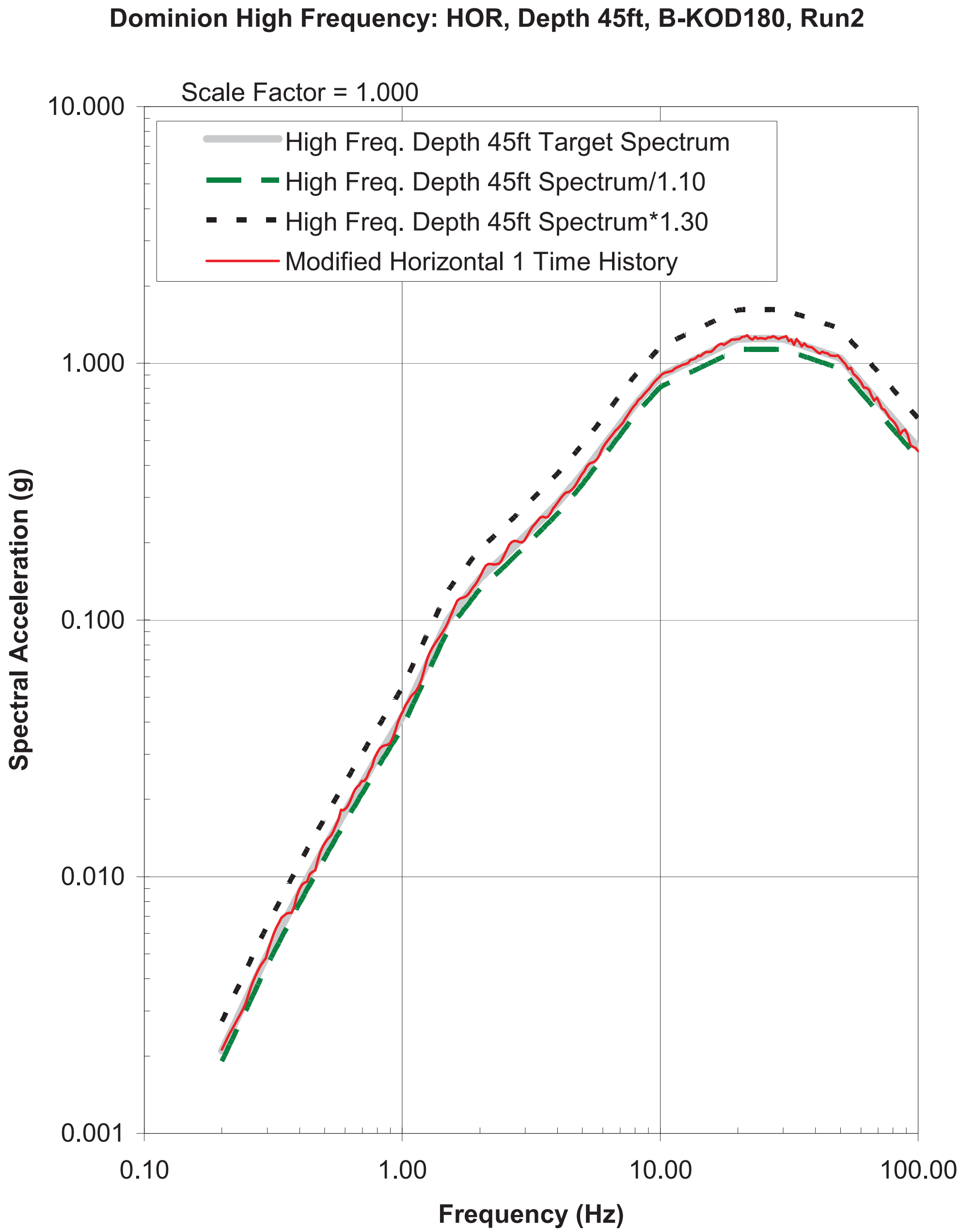


Figure 3.7-30 North Anna ESP Horizontal H1 Target Spectrum at ESBWR RB/FB Base



Note: B-KOD is the Livermore earthquake of January 27, 1980 recorded at the Kodak Building Site in San Ramon, California. “HOR” is “Horizontal”.

Figure 3.7-31 North Anna ESP Horizontal H1 Time Histories at ESBWR RB/FB Base

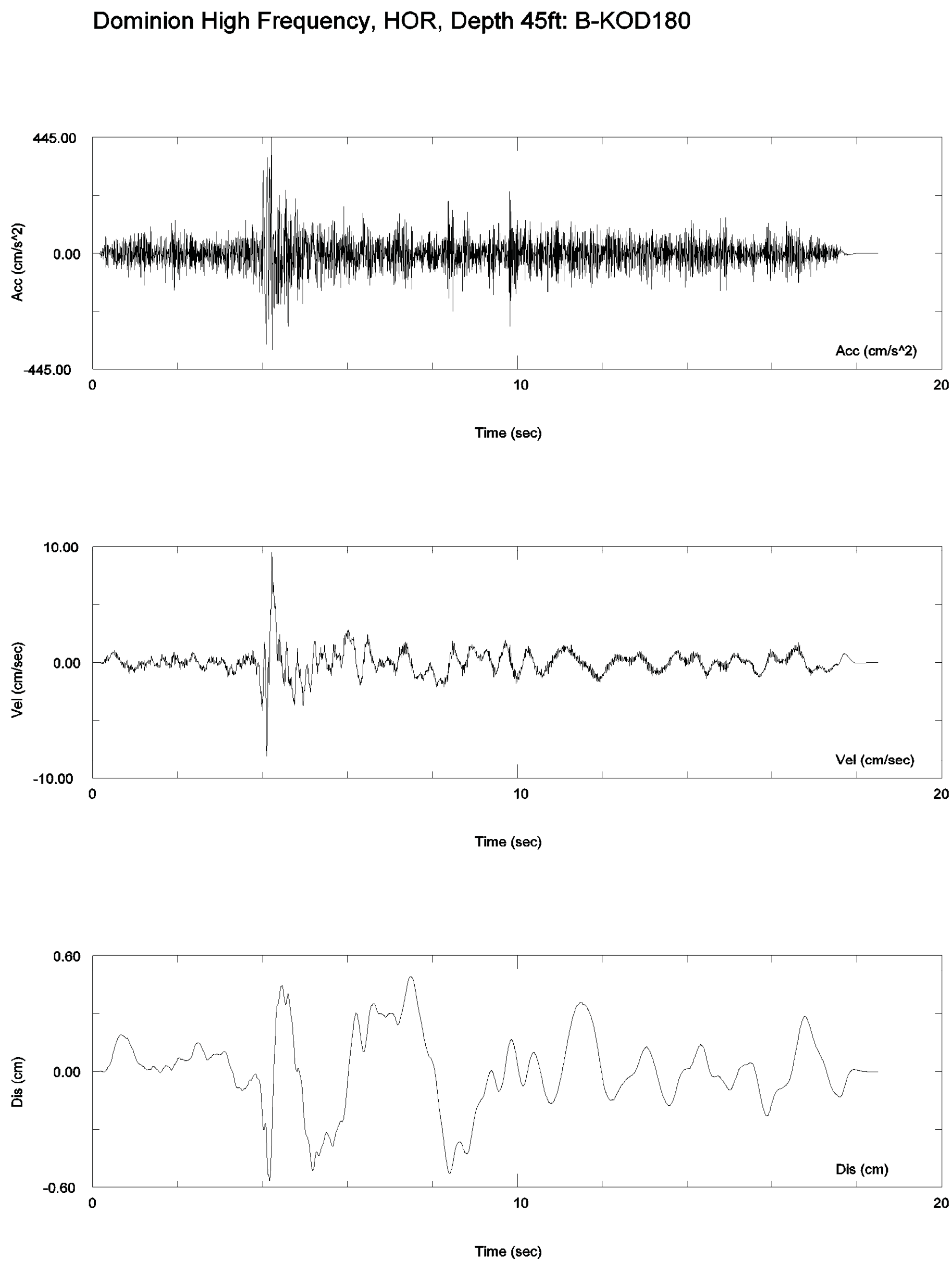
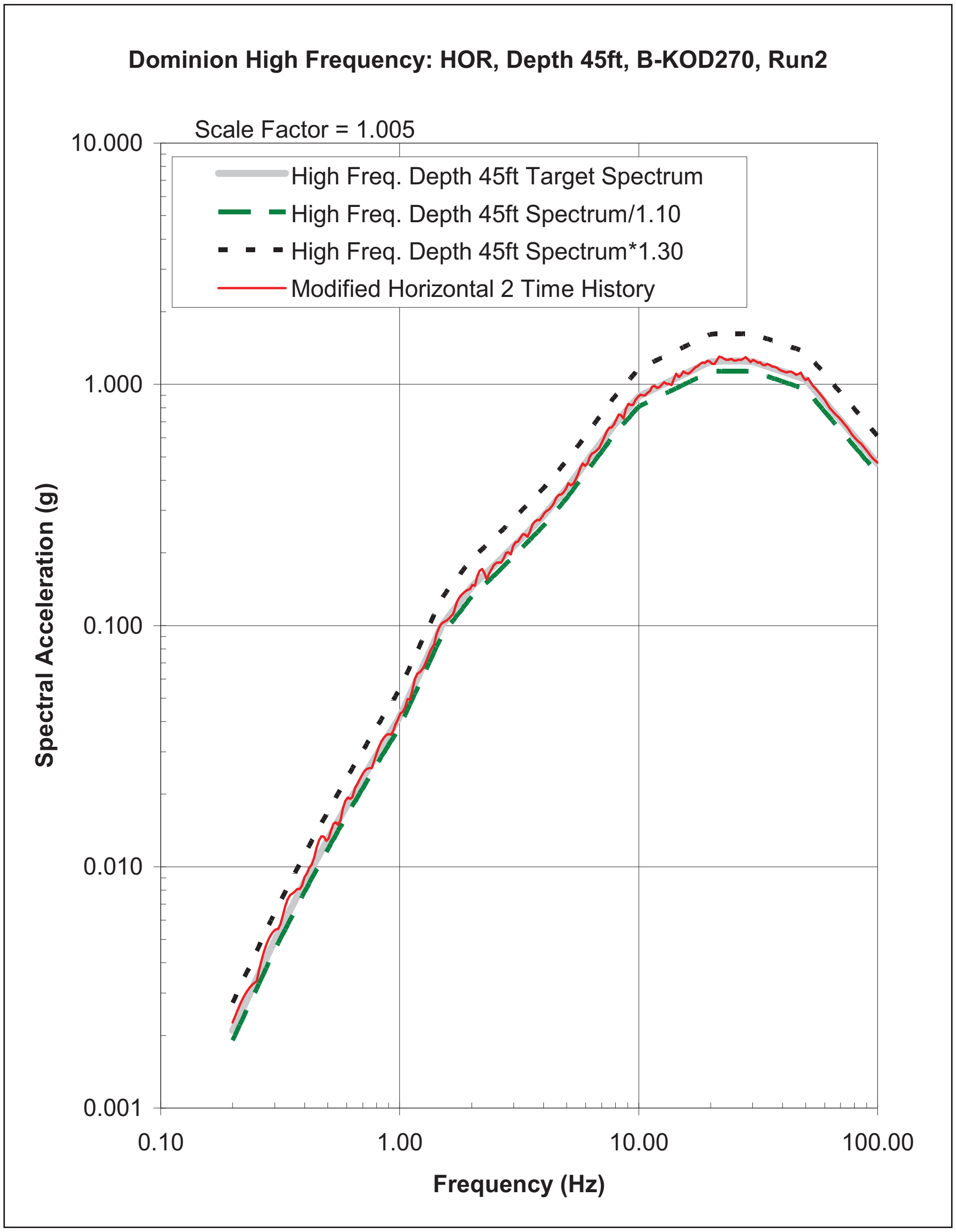


Figure 3.7-32 North Anna ESP Horizontal H2 Target Spectrum at ESBWR RB/FB Base



Note: B-KOD is the Livermore earthquake of January 27, 1980 recorded at the Kodak Building Site in San Ramon, California. “HOR” is “Horizontal”.

Figure 3.7-33 North Anna ESP Horizontal H2 Time Histories at ESBWR RB/FB Base

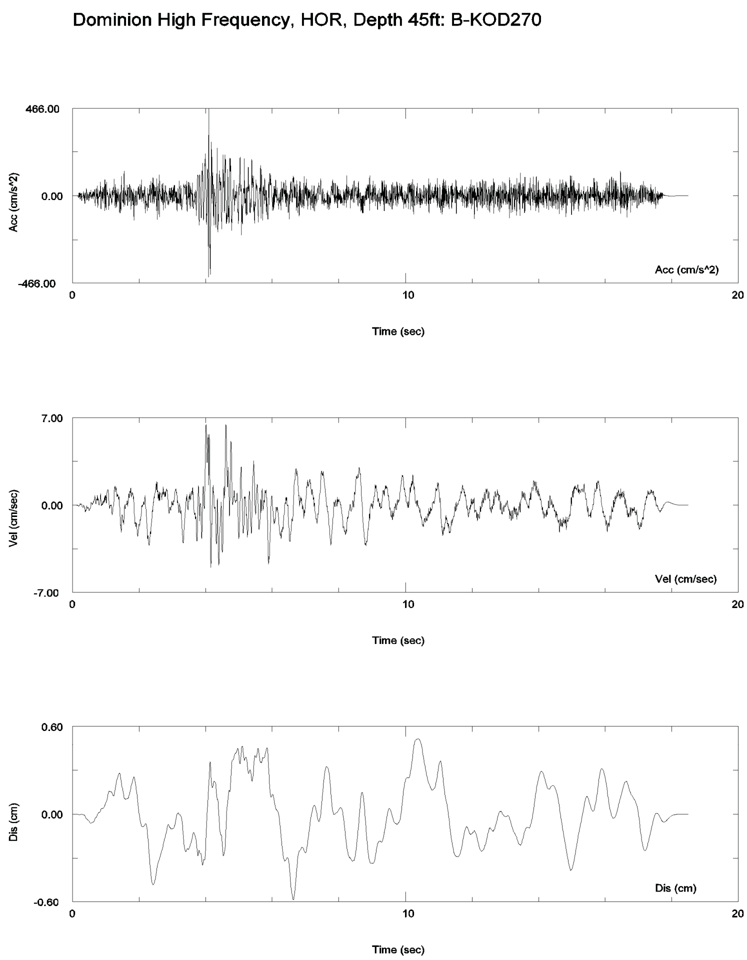
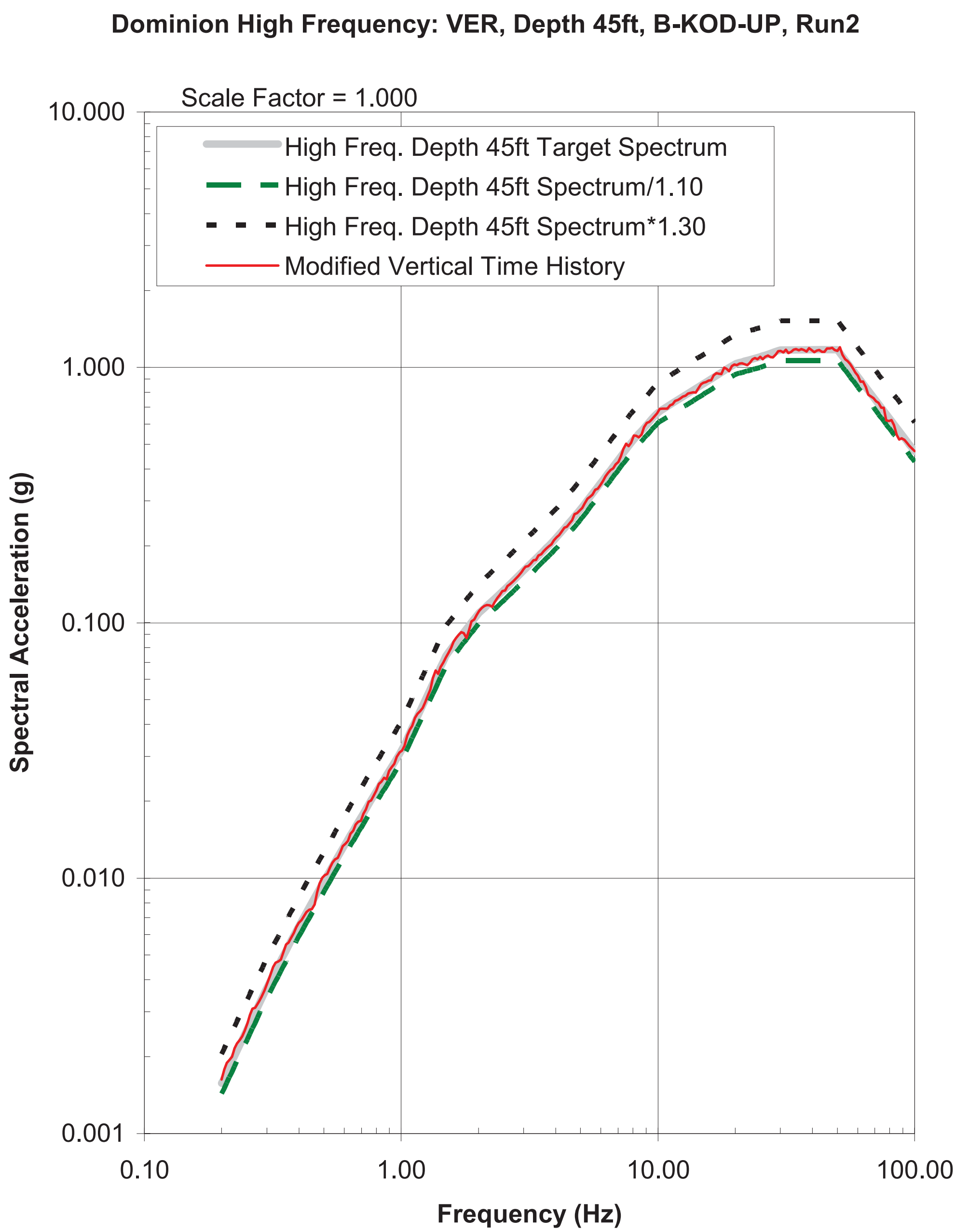


Figure 3.7-34 North Anna ESP Vertical Target Spectrum at ESBWR RB/FB Base



Note: B-KOD is the Livermore earthquake of January 27, 1980 recorded at the Kodak Building Site in San Ramon, California. “VER” is “Vertical”.

Figure 3.7-35 North Anna ESP Vertical Time Histories at ESBWR RB/FB Base

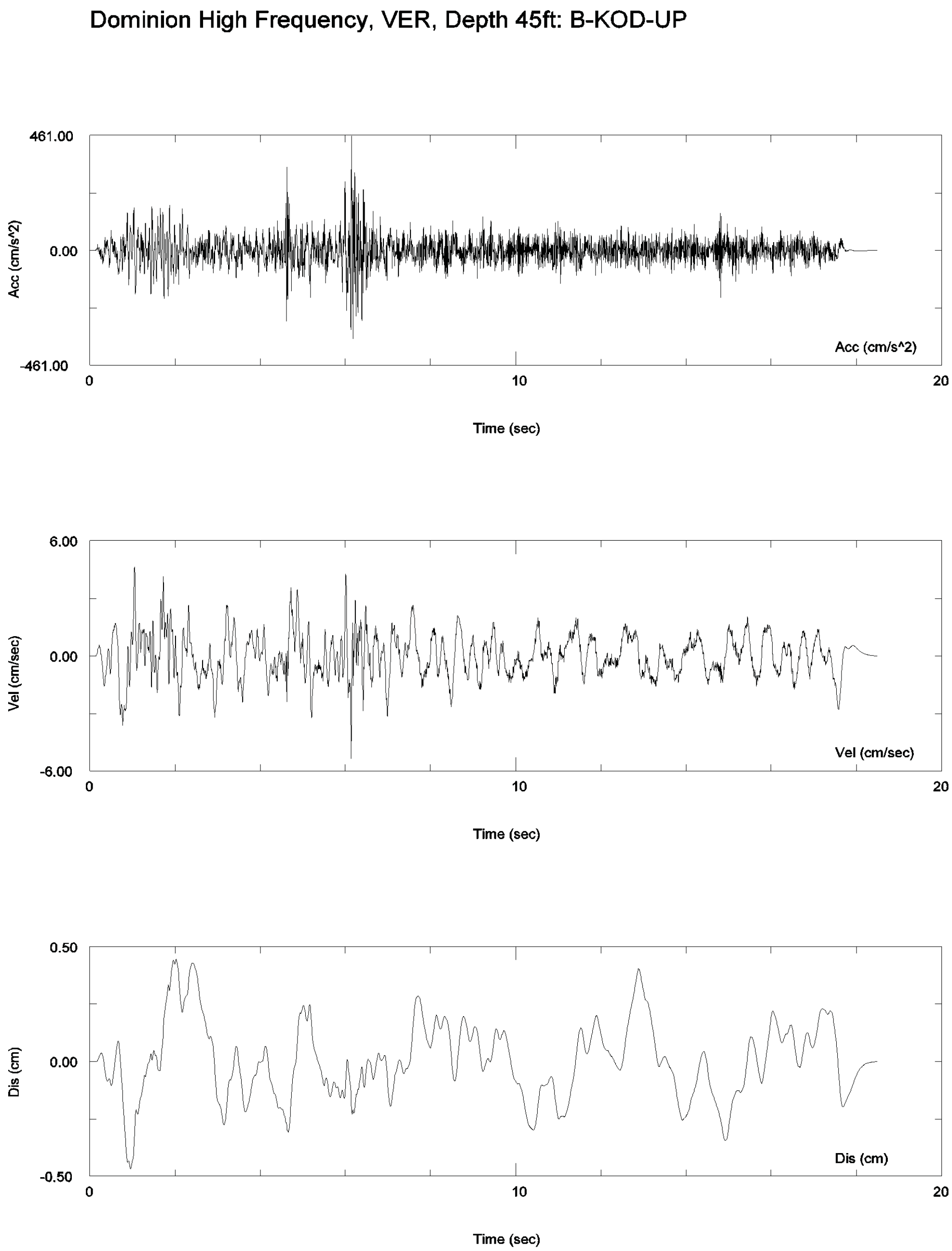
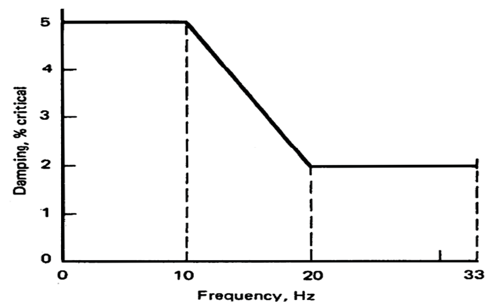


Figure 3.7-36 Not used.

Figure 3.7-37 Alternative Damping Values for Response Spectra Analysis of ASME B&PV Code, Section III, Division 1 Class 1, 2, and 3, and ASME B31.1 Piping Systems



Notes:

As an alternative to the response spectra analysis using an envelope of the SSE response spectra at all support points (based on uniform support motion), frequency-dependent damping values shown in Figure 3.7-37 is used, subject to the following restrictions:

1. When frequency-dependent damping is used, it is done completely and consistently. (For equipment other than piping, damping values specified in Regulatory Guide 1.61 are to be used.)
2. The specified damping values are used only in those analyses in which current seismic spectra and procedures have been employed. Such use is to be limited to response spectral analyses. The acceptance of the use of the specified damping values with other types of dynamic analyses (e.g., time-history analyses or independent support motion method) requires further justification.
3. When used for reconciliation work or support optimization of existing designs, the effects of increased motion on existing clearances and on-line mounted equipment should be checked.
4. Frequency-dependent damping is not appropriate for analyzing the dynamic response of piping systems using supports designed to dissipate energy by yielding.
5. Frequency-dependent damping is not applicable to piping in which stress corrosion cracking has occurred, unless a case-specific evaluation is provided and reviewed by the NRC staff.
6. The damping values specified are applicable in analyzing piping response for seismic and other dynamic loads filtering through building structures in high frequency range beyond 33 Hz

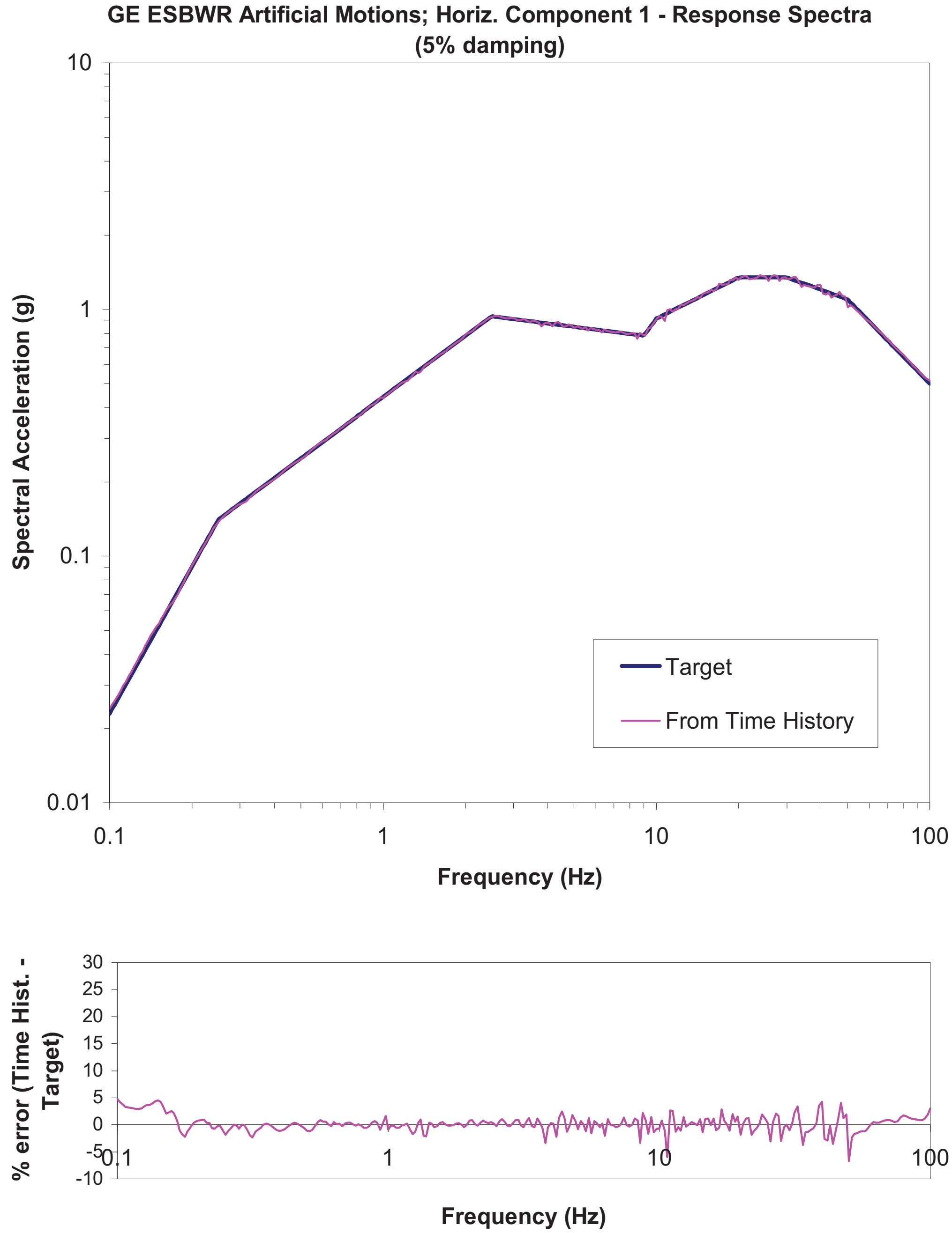


Figure 3.7-38. Single Envelope Spectrum Match - H1 Component

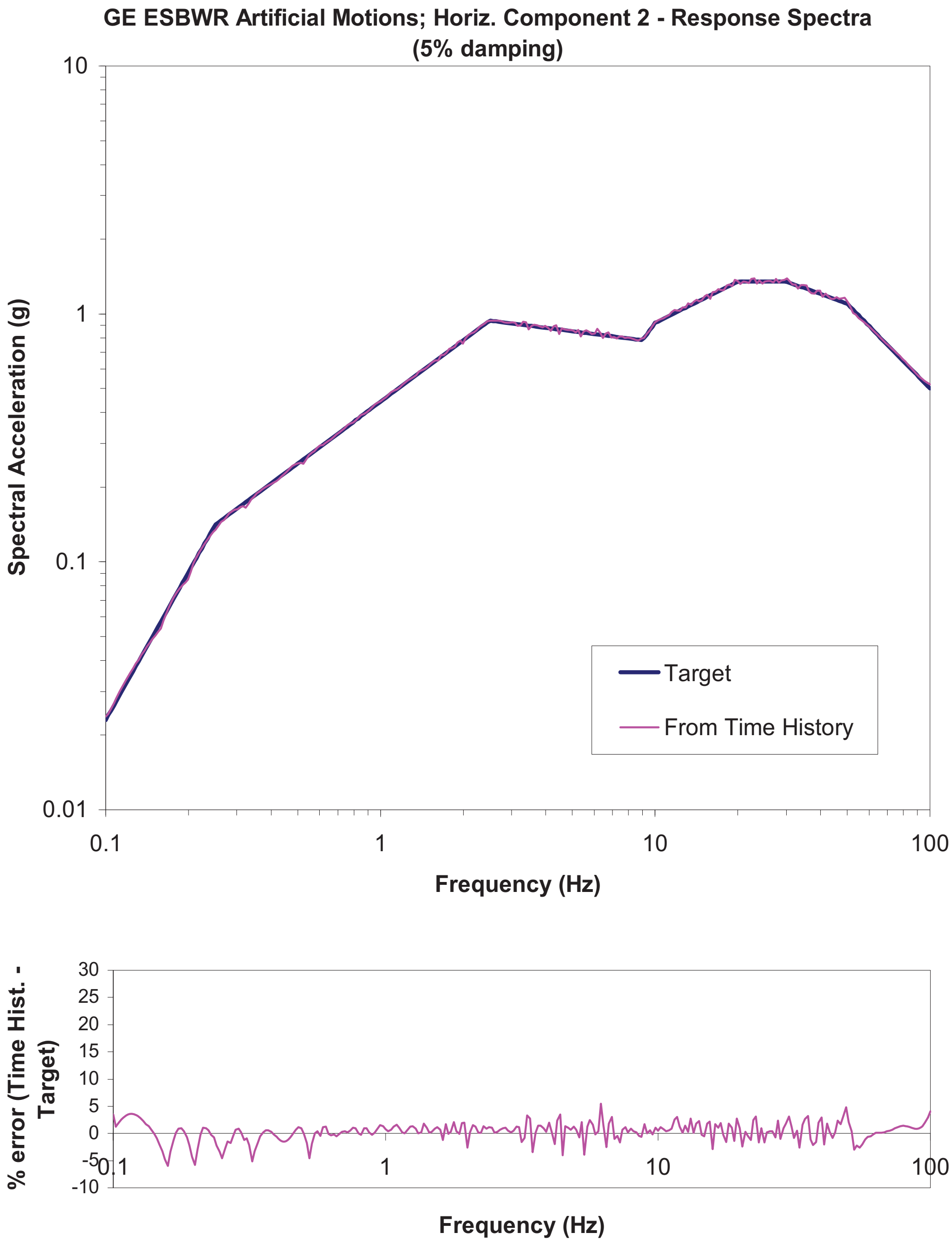


Figure 3.7-39. Single Envelope Spectrum Match - H2 Component

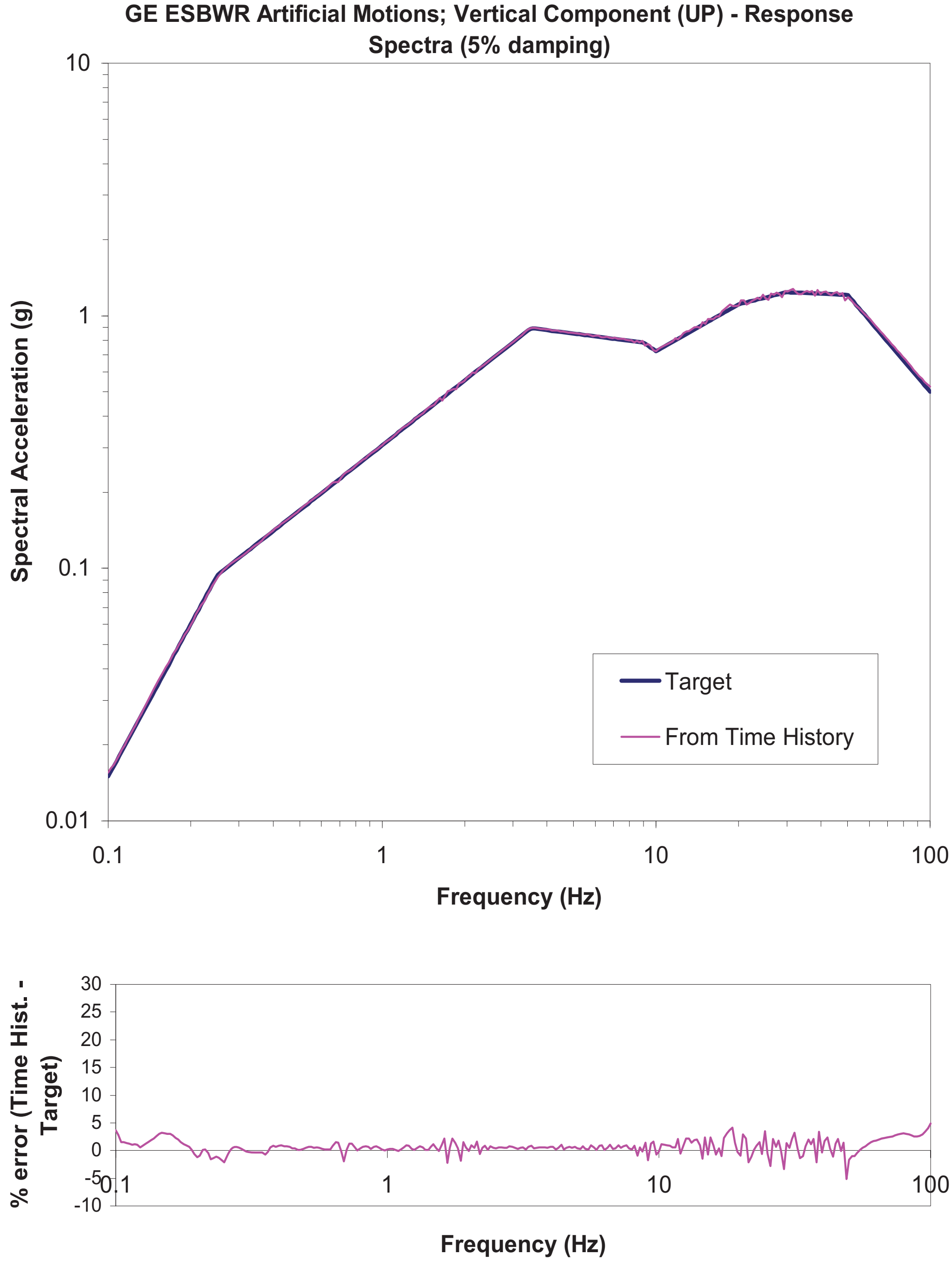


Figure 3.7-40. Single Envelope Spectrum Match - Vertical Component

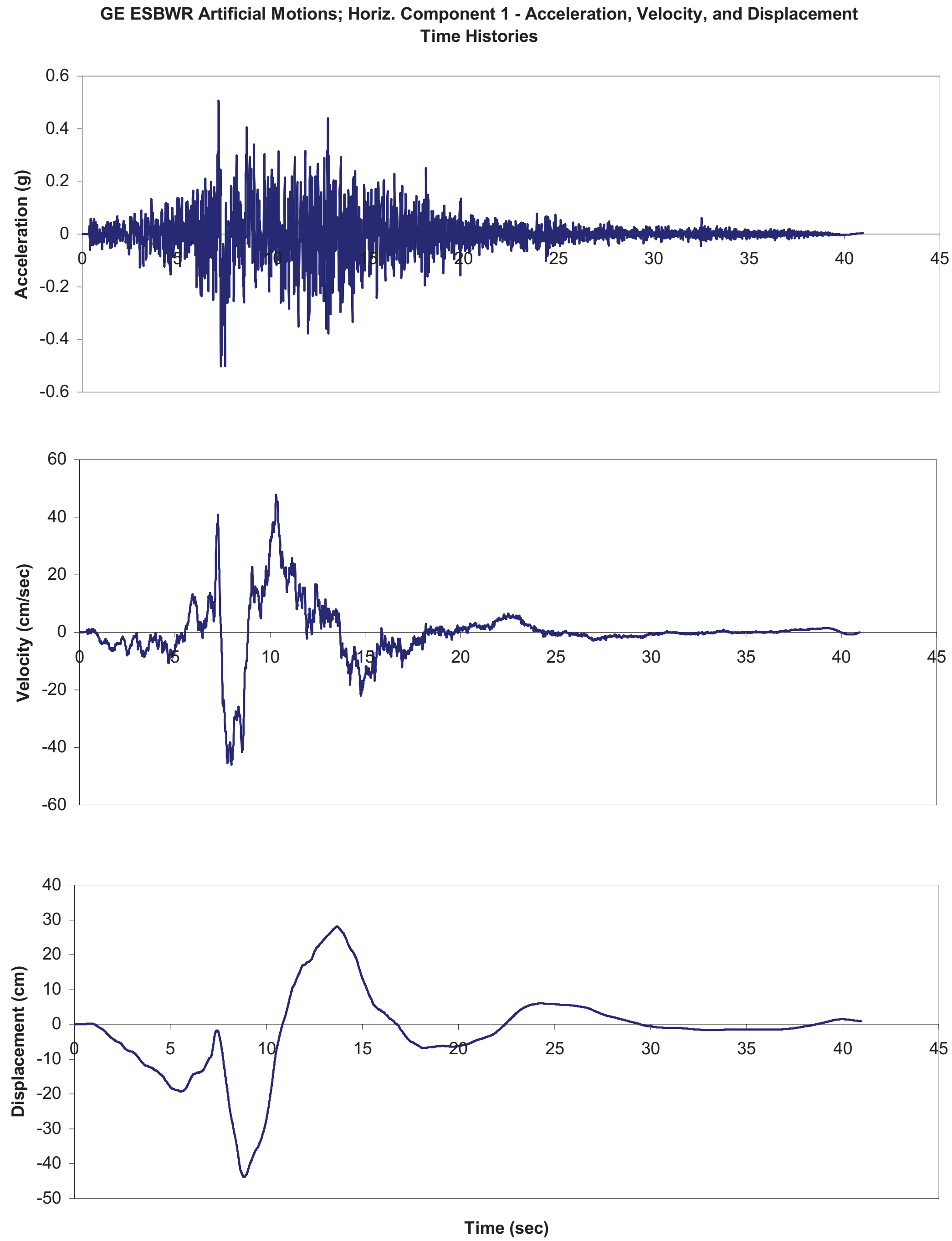


Figure 3.7-41. Single Envelope Time Histories - H1 Component

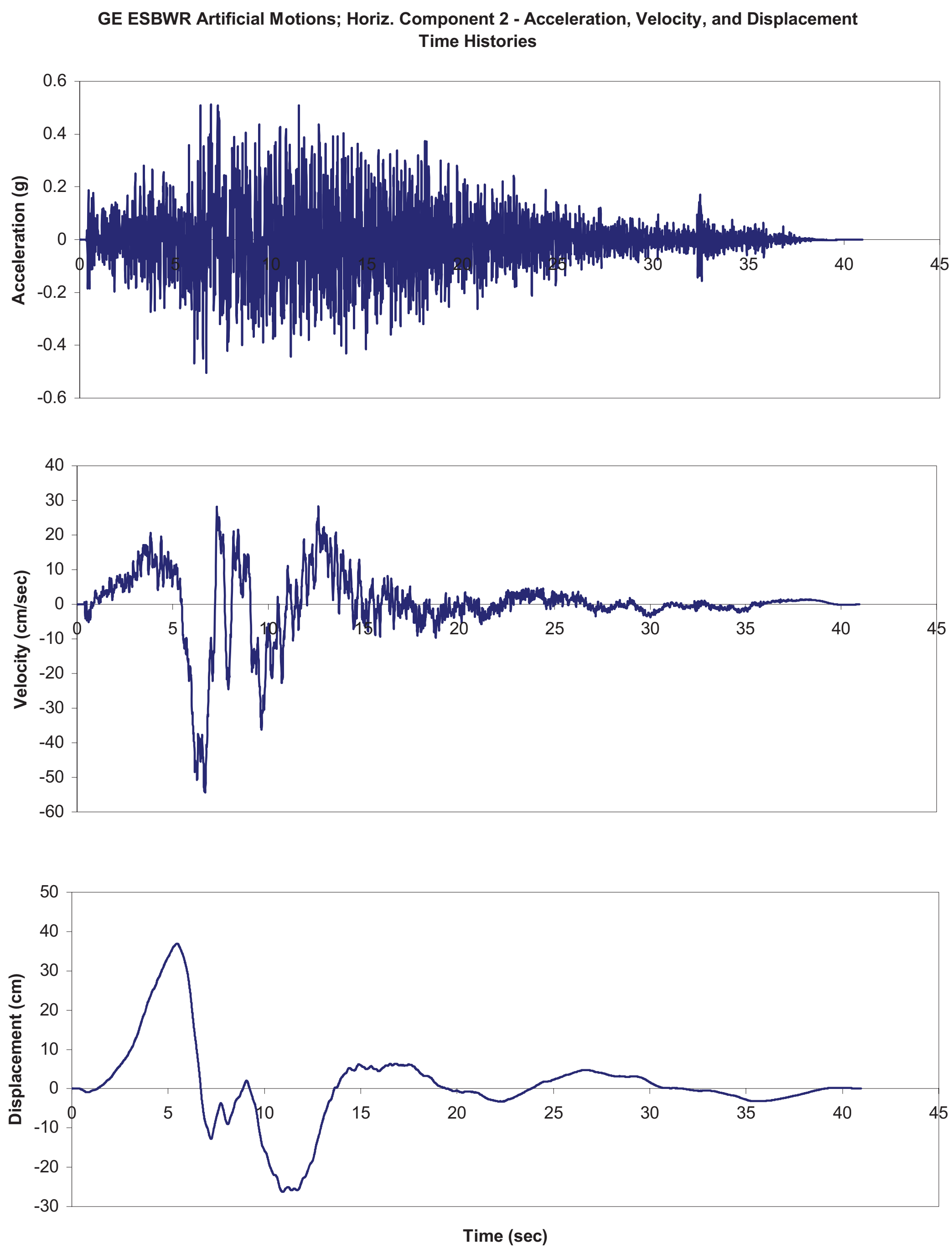


Figure 3.7-42. **Single Envelope Time Histories - H2 Component**

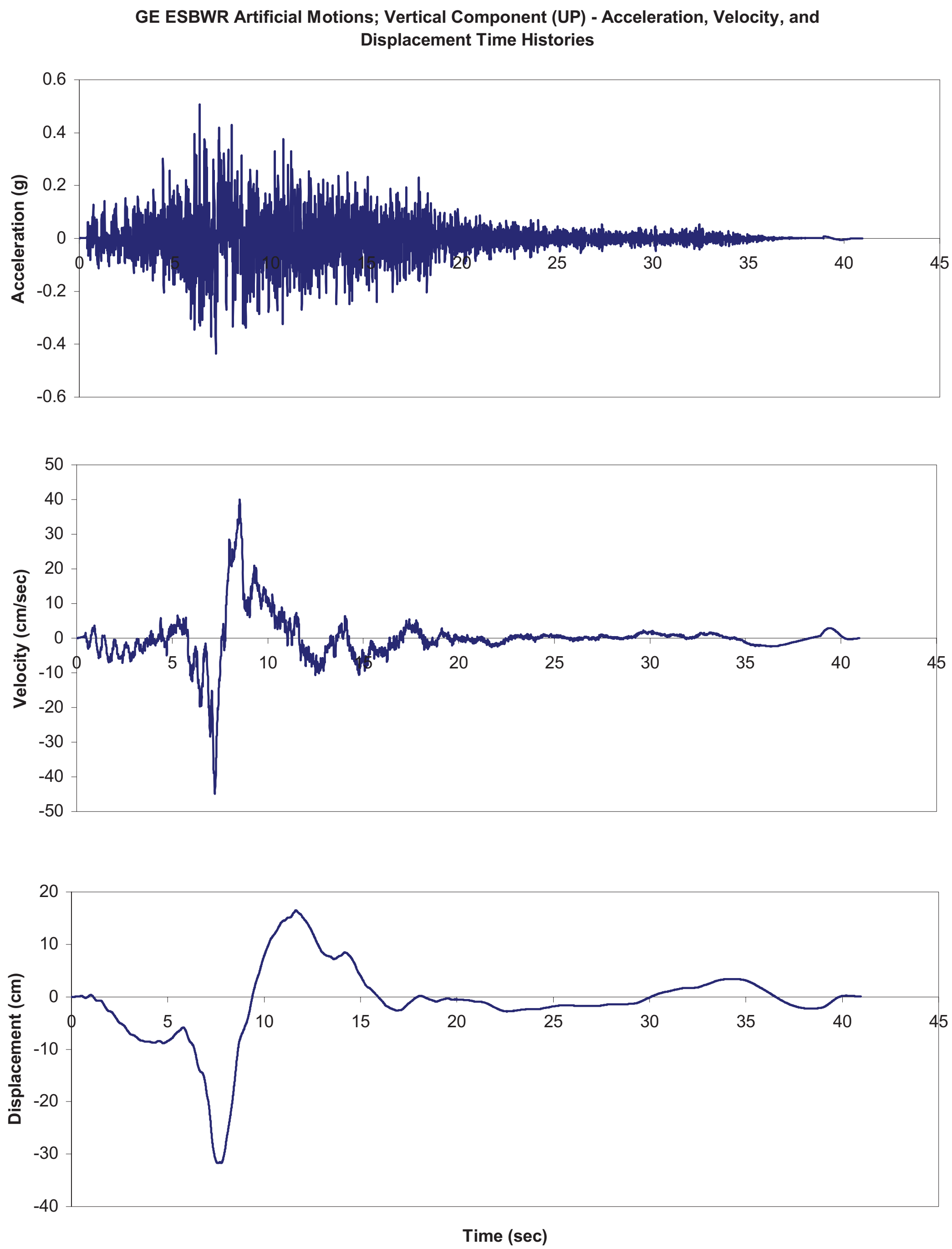


Figure 3.7-43. Single Envelope Time Histories - Vertical Component

Table 3.7.2-201 RB/FB Soil-Structure Interaction Analysis Cases

Building	Case ID No.	Model[1] (DCD)	Model	SASSI2010 Method of Analysis	Input Motion	Subsurface Profile		
						UB	BE	LB
RB/FB	RBFB1UB-DM	Base	SSI Without Engineered Backfill	DM	FIRS	✓	--	--
	RBFB1BE-DM		SSI Without Engineered Backfill	DM		--	✓	--
	RBFB1LB-DM		SSI Without Engineered Backfill	DM		--	--	✓
	RBFB2UB-MSM		SSI With Engineered Backfill	MSM		✓	--	--
	RBFB2LB-MSM		SSI With Engineered Backfill	MSM		--	--	✓

Note [1]: As shown in the Table 3A.6-1, there are some models with minor modifications to evaluate the modeling effects. For the Fermi 3 SSI analyses, the most basic model, “Base” is applied.

BE = Best estimate

LB = Lower bound

UB = Upper bound

Table 3.7.2-202 CB Soil-Structure Interaction Analysis

Building	Case ID No.	Model[1] (DCD)	Model	SASSI2010 Method of Analysis	Input Motion	Subsurface Profile UB	Subsurface Profile BE	Subsurface Profile LB
CB	CB1UB-DM	Base	SSI Without Engineered Backfill	DM	FIRS	✓	--	--
	CB1BE-DM		SSI Without Engineered Backfill	DM		--	✓	--
	CB1LB-DM		SSI Without Engineered Backfill	DM		--	--	✓
	CB2UB-DM		SSI With Engineered Backfill	DM		✓	--	--
	CB2LB-DM		SSI With Engineered Backfill	DM		--	--	✓
	CB3UB-DM		SSSI With Engineered Backfill	DM		✓	--	--
	CB3LB-DM		SSSI With Engineered Backfill	DM		--	--	✓
	CB4-FWSC1UB-MSM		SSSI With Engineered Backfill	MSM		✓	--	--
	CB4-FWSC1LB-MSM		SSSI With Engineered Backfill	MSM		--	--	✓

Note [1]: As shown in the Table 3A.6-1, there are some models with minor modifications to evaluate the modeling effects. For the Fermi 3 SSI analyses, the most basic model, “Base” is applied.

BE = Best estimate

LB = Lower bound

UB = Upper bound

Table 3.7.2-203a Ratio with DCD Enveloping Seismic Loads: RB/FB Stick

			Fermi 3 enveloping SSI seismic load					Enveloping seismic load ratio (SSI / DCD)				
			Shear force (MN)		Bending moment (MN-m)			Shear force		Bending moment		
Elevation (m)	Element No.	Node No.	X-Dir	Y-Dir	X-Dir	Y-Dir	Torsion (MN-m)	X-Dir	Y-Dir	X-Dir	Y-Dir	Torsion
52.4	1110	110			994.4	975.3				61%	54%	
		109	89.5	97.4	2235.4	2547.8	670.9	59%	62%	52%	57%	49%
34.0	1109	109			2929.2	3357.8				52%	61%	
		108	105.8	97.2	3644.5	3978.3	1130.5	55%	64%	56%	63%	47%
27.0	1108	108			3760.4	4589.9				49%	65%	
		107	250.4	221.9	4880.2	5536.1	2127.3	59%	55%	54%	64%	64%
22.5	1107	107			5069.2	5904.2				51%	64%	
		106	279.6	251.4	6448.9	7043.4	4091.1	58%	54%	56%	62%	67%
17.5	1106	106			6968.1	7337.7				56%	61%	
		105	300.5	297.0	8149.6	8342.5	3353.8	56%	53%	59%	60%	66%
13.57	1105	105			8406.8	8597.5				59%	60%	
		104	319.2	314.8	9846.0	9785.6	3535.9	56%	52%	59%	58%	67%
9.06	1104	104			10062.8	10000.2				59%	58%	
		103	339.9	327.3	11477.1	11199.3	3819.4	56%	50%	59%	57%	64%
4.65	1103	103			7412.4	6390.9				39%	32%	
		102	390.9	283.8	9408.5	7788.9	4794.1	47%	33%	41%	32%	42%
-1.00	1102	102			9925.3	8280.6				42%	33%	
		101	420.2	292.3	12033.1	9724.1	5238.7	48%	31%	44%	33%	45%
-6.40	1101	101			7529.3	6445.1				27%	21%	
		2	223.7	165.0	8536.2	6850.0	3453.9	24%	16%	26%	19%	30%

Table 3.7.2-203b Ratio with DCD Enveloping Seismic Loads: RCCV Stick

			Fermi 3 enveloping SSI seismic load					Enveloping seismic load ratio (SSI / DCD)				
			Shear force (MN)		Bending moment (MN-m)			Shear force		Bending moment		
Elevation (m)	Element No.	Node No.	X-Dir	Y-Dir	X-Dir	Y-Dir	Torsion (MN-m)	X-Dir	Y-Dir	X-Dir	Y-Dir	Torsion
34.0	1209	209			94.4	222.8				48%	38%	
		208	87.7	106.5	606.5	949.1	16.9	64%	58%	57%	63%	47%
27.0	1208	208			828.9	1427.5				49%	56%	
		206	102.1	128.7	1516.5	2630.7	1236.3	62%	52%	51%	60%	68%
17.5	1206	206			1599.7	2824.7				48%	60%	
		205	134.1	135.0	2111.6	3327.8	1326.7	58%	47%	51%	58%	67%
13.57	1205	205			2182.8	3434.9				50%	58%	
		204	147.0	149.6	2772.8	4035.4	1479.5	56%	46%	51%	56%	68%
9.06	1204	204			2903.0	4161.4				52%	55%	
		203	162.2	162.2	3605.5	4781.9	1678.0	53%	44%	53%	54%	64%
4.65	1203	203			3737.8	4931.4				53%	54%	
		202	83.9	83.9	4213.6	5407.1	1295.8	37%	29%	53%	51%	45%
-1.00	1202	202			4361.6	5574.3				54%	52%	
		201	114.0	104.4	4978.5	6126.1	1328.3	42%	32%	53%	49%	45%
-6.40	1201	201			5070.2	6222.6				53%	49%	
		2	60.8	45.6	5309.2	6374.5	579.6	23%	15%	49%	45%	30%

Table 3.7.2-203c Ratio with DCD Enveloping Seismic Loads: Vent Wall/Pedestal Stick

Elevation (m)	Element No.	Node No.	Fermi 3 enveloping SSI seismic load					Enveloping seismic load ratio (SSI / DCD)				
			Shear force (MN)		Bending moment (MN-m)		Torsion (MN-m)	Shear force		Bending moment		Torsion
			X-Dir	Y-Dir	X-Dir	Y-Dir		X-Dir	Y-Dir	X-Dir	Y-Dir	
17.50	701	701			27.2	26.8				35%	32%	
		702	8.8	10.6	38.4	29.7	21.1	25%	29%	34%	22%	18%
14.50	702	702			39.5	40.1				33%	27%	
		703	9.7	10.4	53.2	57.5	21.7	27%	26%	24%	22%	18%
11.50	703	703			51.8	59.0				23%	22%	
		704	11.2	12.2	66.2	92.6	22.3	30%	29%	19%	24%	19%
8.50	704	704			68.0	93.8				20%	24%	
		705	12.3	12.9	80.2	107.6	22.9	33%	29%	21%	25%	19%
7.4625	705	705			74.9	104.8				21%	24%	
		706	7.0	6.6	86.7	120.2	11.4	17%	16%	19%	23%	11%
4.65	1303	303			211.6	274.3				36%	44%	
		377	16.5	16.5	240.7	302.4	63.9	50%	37%	40%	45%	45%
2.4165	1377	377			297.2	372.2				41%	46%	
		302	24.3	23.8	359.1	430.8	77.7	51%	36%	46%	47%	45%
-1.00	1302	302			332.2	395.7				40%	41%	
		376	33.4	29.9	379.8	443.4	66.4	51%	37%	41%	42%	45%
-2.75	1376	376			380.0	443.5				41%	42%	
		301	33.5	30.0	497.3	544.7	66.4	51%	37%	45%	41%	45%
-6.40	1301	301			466.6	541.4				41%	40%	
		2	25.1	18.0	554.0	585.4	34.9	24%	15%	33%	30%	30%

Table 3.7.2-203d Ratio with DCD Enveloping Seismic Loads: RSW Stick

Elevation (m)	Element No.	Node No.	Fermi 3 enveloping SSI seismic load: Shear force (MN) X-Dir	Shear force (MN) Y-Dir	Bending moment (MN-m) X-Dir	Bending moment (MN-m) Y-Dir	Torsion (MN-m)	Enveloping seismic load ratio (SSI / DCD): Shear force X-Dir	Shear force Y-Dir	Bending moment X-Dir	Bending moment Y-Dir	Torsion
24.18	707	707			1.0	1.0				48%	59%	
		708	1.5	1.2	7.1	5.4	0.2	50%	44%	54%	44%	50%
20.2	708	708			10.4	8.1				57%	48%	
		709	7.1	4.6	40.5	24.0	0.7	49%	37%	51%	35%	50%
15.775	709	709			42.8	24.8				52%	35%	
		710	8.2	5.2	79.4	47.7	1.0	47%	36%	50%	36%	53%
11.35	710	710			79.9	48.0				50%	35%	
		711	8.7	6.3	114.8	72.6	1.0	44%	38%	49%	37%	42%
7.4625	711	711			74.6	98.1				38%	53%	
		712	18.0	17.0	103.9	140.3	12.1	44%	48%	36%	56%	52%
4.65	712	713			46.9	57.6				37%	43%	
		714	7.2	7.2	55.9	69.6	13.7	50%	37%	42%	46%	45%
2.4165	713	713			1.5	1.4				42%	44%	
		714	0.5	0.5	1.2	1.2	0.1	33%	38%	41%	44%	50%
1.96	714	714			1.1	1.0				41%	42%	
		715	0.3	0.3	0.3	0.2	0.05	33%	43%	60%	44%	50%

Table 3.7.2-203e Ratio with DCD Enveloping Seismic Loads: RPV Stick

Location	Element No.	Node No.	Fermi 3 enveloping SSI seismic load					Enveloping seismic load ratio (SSI / DCD)				
			Axial (MN)	Shear force (MN)		Bending moment (MN-m)		Axial	Shear force		Bending moment	
				X-Dir	Y-Dir	X-Dir	Y-Dir		X-Dir	Y-Dir	X-Dir	Y-Dir
Shroud Bottom	844	845	4.6			11.7	6.6	54%			72%	46%
		846	4.6	5.0	2.6	15.8	7.3	54%	70%	37%	74%	42%
RPV Support	871	815	10.7			83.6	57.2	42%			58%	42%
		711	10.7	16.0	10.4	74.5	54.3	42%	86%	58%	53%	40%

Table 3.7.2-204 Ratio with DCD Enveloping Seismic Loads: CB Stick

			Fermi 3 enveloping SSI seismic load					Enveloping seismic load ratio (SSI / DCD)				
			Shear force (MN)		Bending moment (MN-m)			Shear force		Bending moment		
Elevation (m)	Element No.	Node No.	X-Dir	Y-Dir	X-Dir	Y-Dir	Torsion (MN-m)	X-Dir	Y-Dir	X-Dir	Y-Dir	Torsion
13.80	6	6			78.9	68.1				49%	55%	
		5	19.0	20.8	127.7	126.7	20.1	57%	71%	51%	64%	27%
9.06	5	5			173.9	159.7				48%	58%	
		4	36.1	39.4	304.1	316.2	42.6	68%	72%	53%	71%	33%
4.65	4	4			183.8	109.8				25%	20%	
		3	49.8	50.4	476.6	427.8	79.8	66%	63%	42%	43%	45%
-2.00	3	3			384.4	426.0				31%	41%	
-7.40		2	58.9	64.7	679.0	678.2	148.1	47%	65%	43%	44%	60%

Table 3.7.2-205a Ratio with DCD Enveloping Maximum Vertical Acceleration: RB/FB

Elev. (m)	Node No.	Stick model	Fermi 3 enveloping SSI max. vertical acceleration (g)	Enveloping max. vertical acceleration ratio (SSI / DCD)
52.40	110	RB/FB	0.46	36%
34.00	109	RB/FB	0.36	43%
27.00	108	RB/FB	0.34	46%
22.50	107	RB/FB	0.29	40%
17.50	106	RB/FB	0.29	40%
13.57	105	RB/FB	0.28	38%
9.06	104	RB/FB	0.29	40%
4.65	103	RB/FB	0.28	36%
-1.00	102	RB/FB	0.26	34%
-6.40	101	RB/FB	0.25	37%
-11.50	2	RB/FB	0.23	36%
-15.50	1	RB/FB	0.23	45%

Table 3.7.2-205b Ratio with DCD Enveloping Maximum Vertical Acceleration: RCCV

Elev. (m)	Node No.	Stick model	Fermi 3 enveloping SSI max. vertical acceleration (g)	Enveloping max. vertical acceleration ratio (SSI / DCD)
34.00	209	RCCV	0.37	41%
27.00	208	RCCV	0.35	40%
17.50	206	RCCV	0.30	41%
13.57	205	RCCV	0.28	36%
9.06	204	RCCV	0.26	41%
4.65	203	RCCV	0.24	35%
-1.00	202	RCCV	0.21	36%
-6.40	201	RCCV	0.22	37%

Table 3.7.2-205c Ratio with DCD Enveloping Maximum Vertical Acceleration: VW/Pedestal

Elev. (m)	Node No.	Stick model	Fermi 3 enveloping SSI max. vertical acceleration (g)	Enveloping max. vertical acceleration ratio (SSI / DCD)
17.50	701	VW	0.33	30%
14.50	702	VW	0.32	31%
11.50	703	VW	0.32	35%
8.50	704	VW	0.31	40%
7.4625	705	VW	0.31	44%
4.65	706, 303	Pedestal	0.28	42%
2.4165	377	Pedestal	0.26	41%
-1.00	302	Pedestal	0.22	37%
-2.753	376	Pedestal	0.21	41%
-6.40	301	Pedestal	0.22	43%

Table 3.7.2-205d Ratio with DCD Enveloping Maximum Vertical Acceleration: RSW

Elev. (m)	Node No.	Stick model	Fermi 3 enveloping SSI max. vertical acceleration (g)	Enveloping max. vertical acceleration ratio (SSI / DCD)
24.18	707	RSW	0.40	41%
20.20	708	RSW	0.40	42%
15.775	709	RSW	0.37	45%
11.35	710	RSW	0.34	45%
7.4625	711	RSW	0.31	44%
4.65	712	RSW	0.28	42%
2.4615	713	RSW	0.26	41%
1.96	714	RSW	0.26	41%
-0.80	715	RSW	0.26	40%

Table 3.7.2-205e Ratio with DCD Enveloping Maximum Vertical Acceleration: RB/FB Flexible Slab Oscillators (Sheet 1 of 2)

Elev. (m)	Node No.	Stick model	Fermi 3 enveloping SSI max. vertical acceleration (g)	Enveloping max. vertical acceleration ratio (SSI / DCD)
52.40	9101	Oscillator	0.46	38%
	9102	Oscillator	0.84	46%
	9103	Oscillator	1.50	48%
	9104	Oscillator	1.15	47%
	9105	Oscillator	0.86	37%
	9106	Oscillator	1.46	49%
	9107	Oscillator	1.15	41%
	9108	Oscillator	0.93	36%
34.00	9091	Oscillator	0.51	39%
	9092	Oscillator	0.49	45%
27.00	9081	Oscillator	0.46	40%
	9082	Oscillator	0.45	45%
	9083	Oscillator	0.46	42%
	9084	Oscillator	0.50	38%
	9085	Oscillator	0.41	43%
22.50	9071	Oscillator	0.72	45%
	9072	Oscillator	0.98	75%
	9073	Oscillator	0.96	47%
	9074	Oscillator	0.59	45%
	9075	Oscillator	0.54	47%
17.50	9061	Oscillator	0.76	42%
	9062	Oscillator	0.96	65%
	9063	Oscillator	0.39	47%
	9064	Oscillator	0.86	47%
	9065	Oscillator	0.47	33%

Table 3.7.2-205e Ratio with DCD Enveloping Maximum Vertical Acceleration: RB/FB Flexible Slab Oscillators (Sheet 2 of 2)

Elev. (m)	Node No.	Stick model	Fermi 3 enveloping SSI max. vertical acceleration (g)	Enveloping max. vertical acceleration ratio (SSI / DCD)
13.57	9051	Oscillator	0.39	48%
	9052	Oscillator	0.50	34%
9.06	9041	Oscillator	0.37	42%
	9042	Oscillator	0.55	38%
4.65	9031	Oscillator	0.74	63%
	9032	Oscillator	0.41	42%
	9033	Oscillator	0.60	59%
	9034	Oscillator	0.74	49%
	9035	Oscillator	0.47	34%
-1.00	9021	Oscillator	0.58	52%
	9022	Oscillator	0.78	54%
	9023	Oscillator	0.50	50%
	9024	Oscillator	0.45	51%
	9025	Oscillator	0.53	40%
	9026	Oscillator	0.77	49%
	9027	Oscillator	0.36	41%
-6.40	9011	Oscillator	0.50	55%
	9012	Oscillator	0.51	55%
	9013	Oscillator	0.60	44%

Table 3.7.2-206 Ratio with DCD Enveloping Maximum Vertical Acceleration: CB

Elev. (m)	Node No.	Stick model	Fermi 3 enveloping SSI max. vertical acceleration (g)	Enveloping max. vertical acceleration ratio (SSI / DCD)
13.8	6	CB	0.37	37%
9.06	5	CB	0.34	40%
4.65	4	CB	0.30	41%
-2	3	CB	0.23	40%
-7.4	2	CB	0.21	40%
-10.4	1	CB	0.21	40%
13.8	9001	Oscillator	1.21	55%
	9002	Oscillator	0.76	56%
	9003	Oscillator	1.02	71%
9.06	9101	Oscillator	1.05	53%
	9102	Oscillator	0.71	56%
	9103	Oscillator	1.07	75%
4.65	9201	Oscillator	0.55	42%
	9202	Oscillator	0.84	59%
-2	9301	Oscillator	0.65	47%

Figure 3.7.2-201 SASSI2010 Plate Element for RB/FB Basemat (RBFB1)

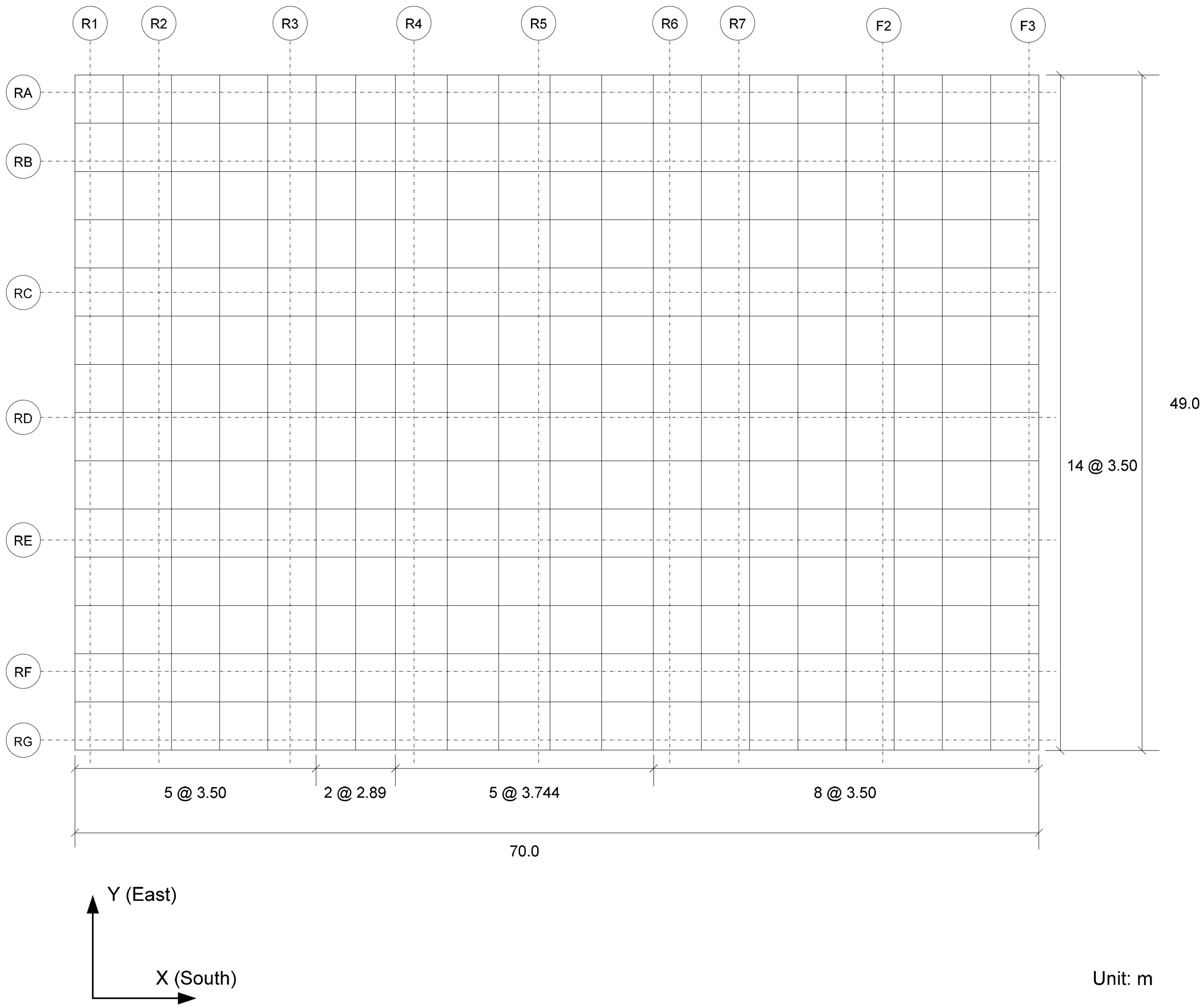
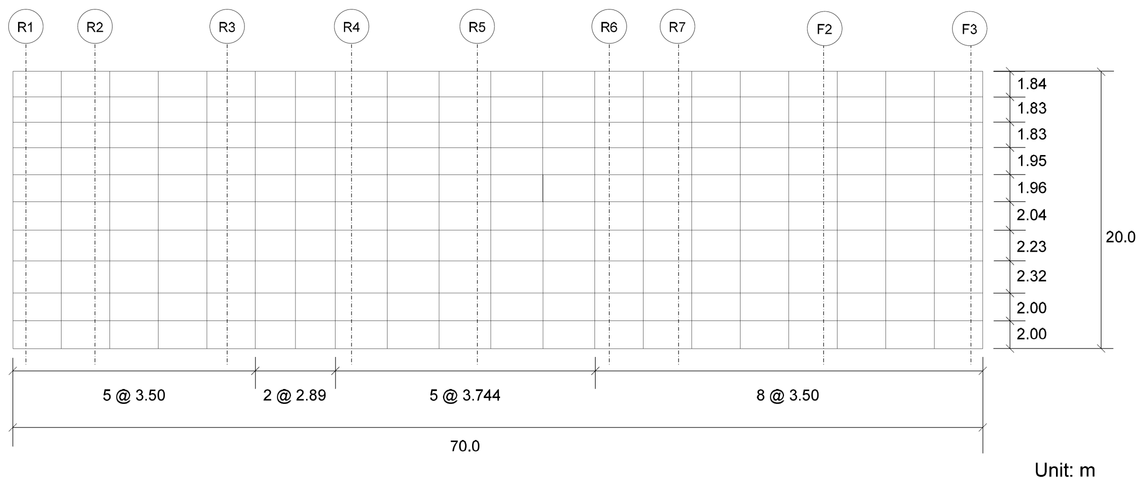
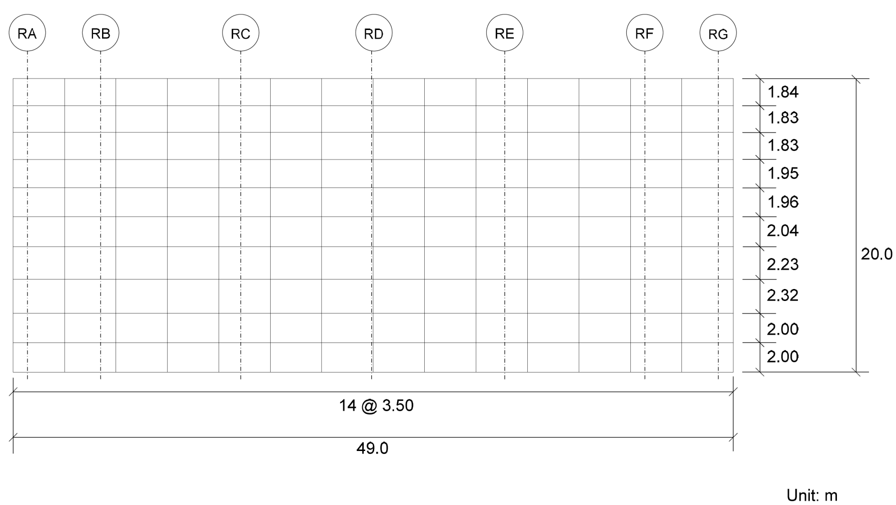


Figure 3.7.2-202 SASSI2010 Plate Elements for RB/FB Exterior Walls (RBFB1)



(a) Walls on Column Rows RA and RG



(b) Walls on Column Rows R1 and F3

Figure 3.7.2-203 Overview of SASSI2010 SSI RB/FB Model (RBFB1)

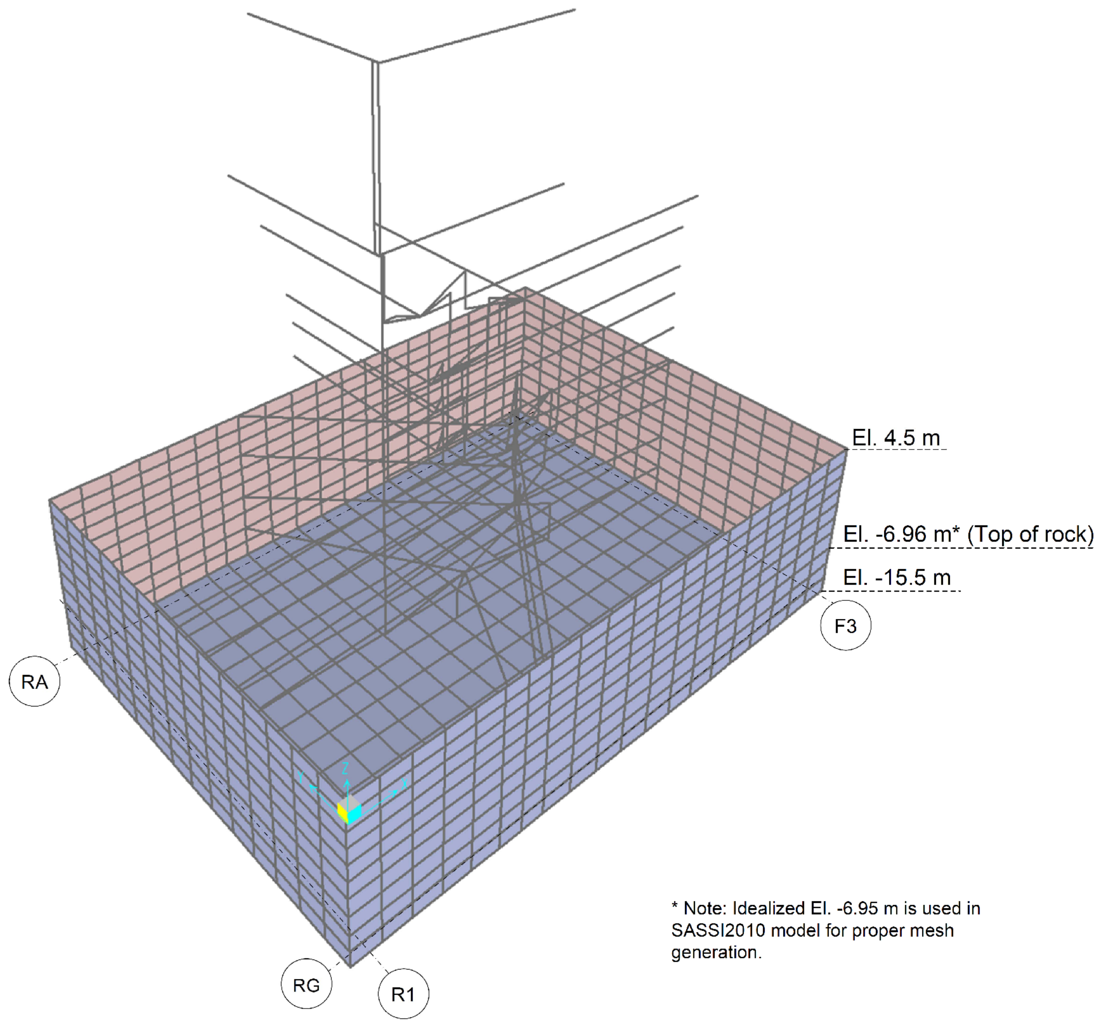


Figure 3.7.2-203a SASSI2010 Plate Element for RB/FB Basemat (RBFB2)

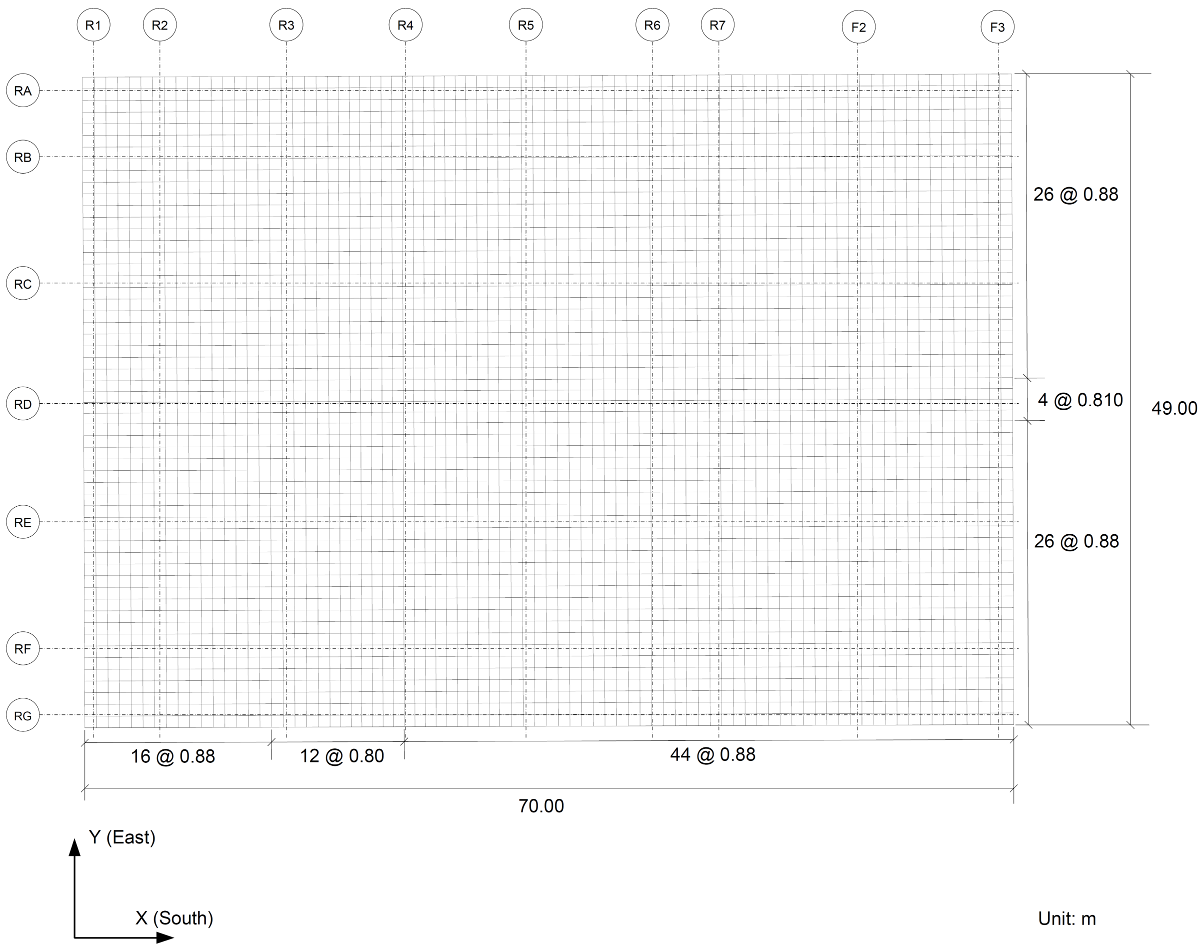
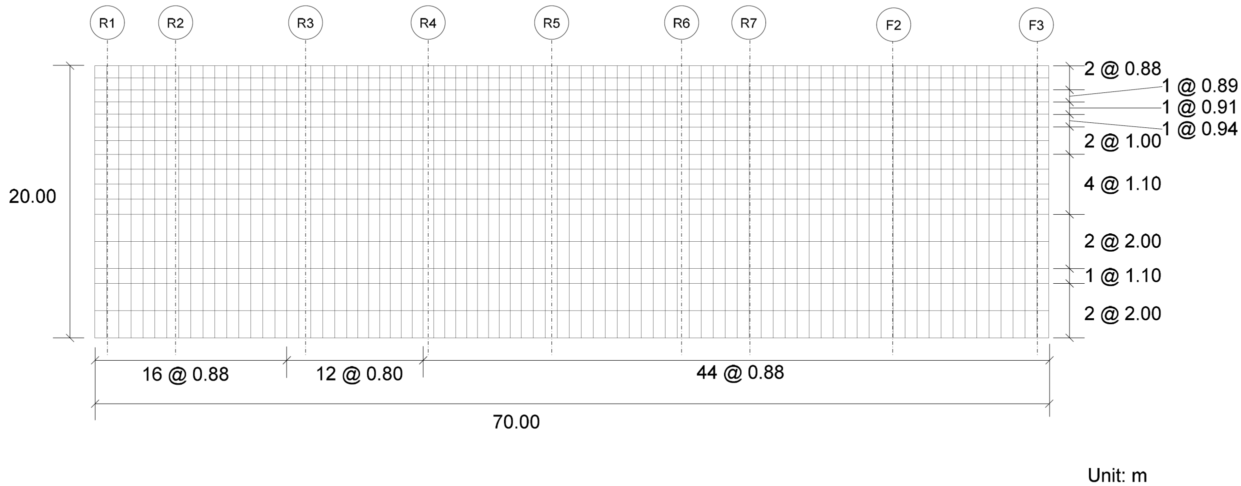
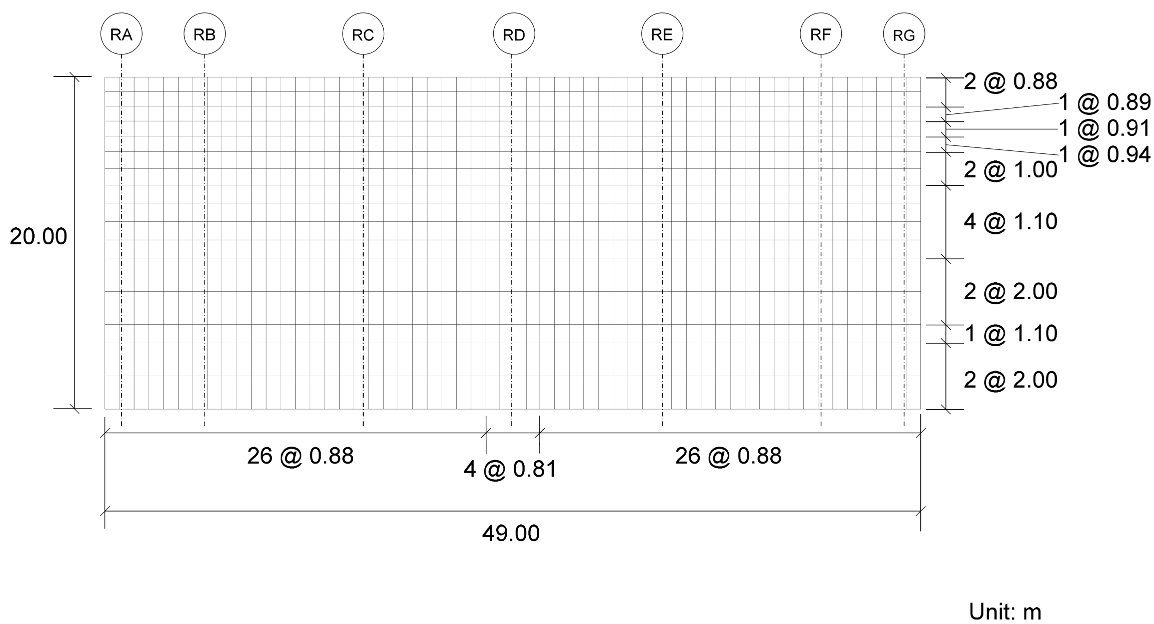


Figure 3.7.2-203b SASSI2010 Plate Element for RB/FB Exterior Walls (RBFB2)



(a) Walls on Column Rows RA and RG



(b) Walls on Column Rows R1 and F3

Figure 3.7.2-203c Overview of SASSI2010 SSI RB/FB Model (RBFB2)

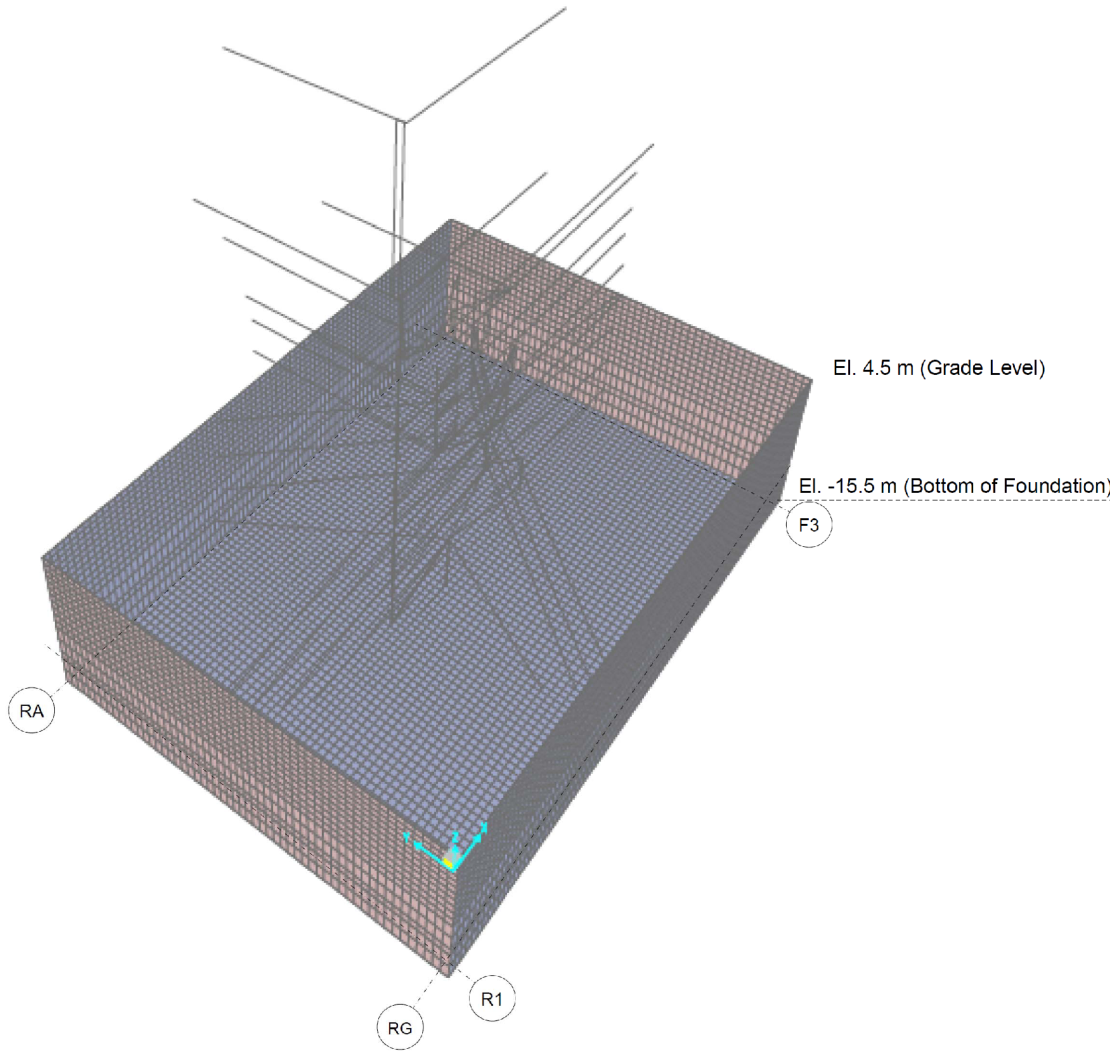


Figure 3.7.2-204 SASSI2010 Plate Elements for CB Basemat (CB1)

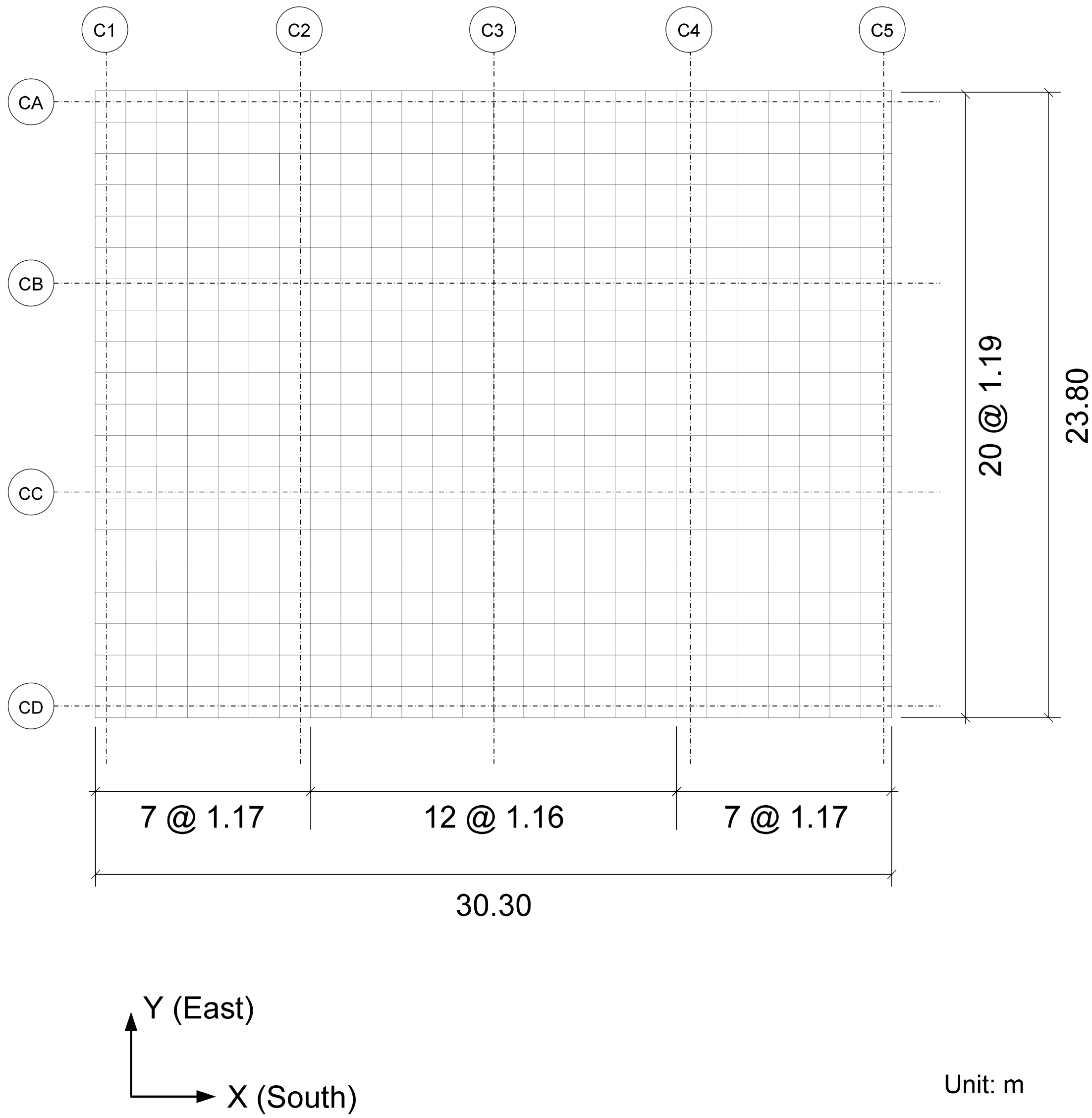
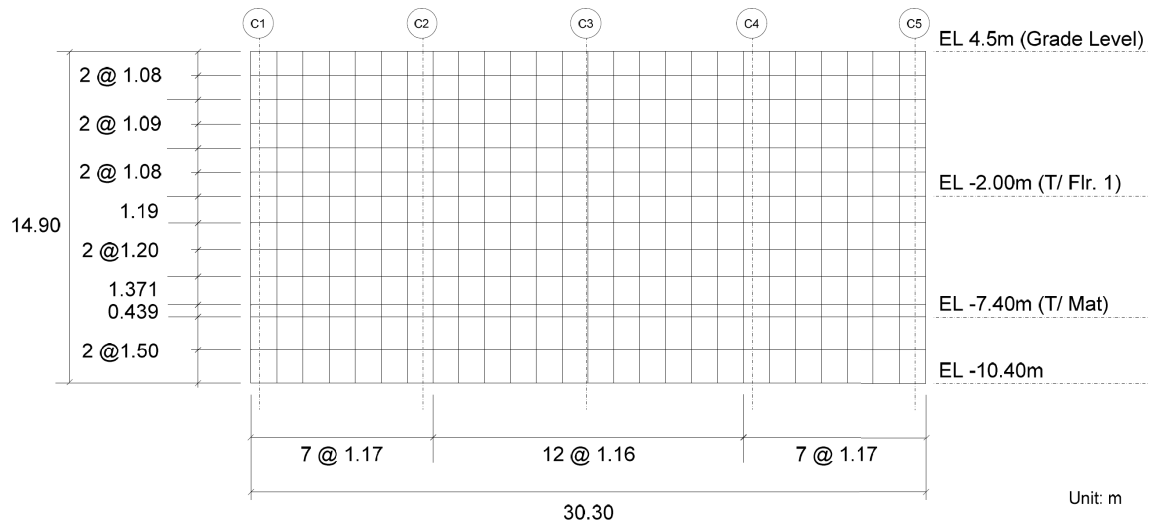
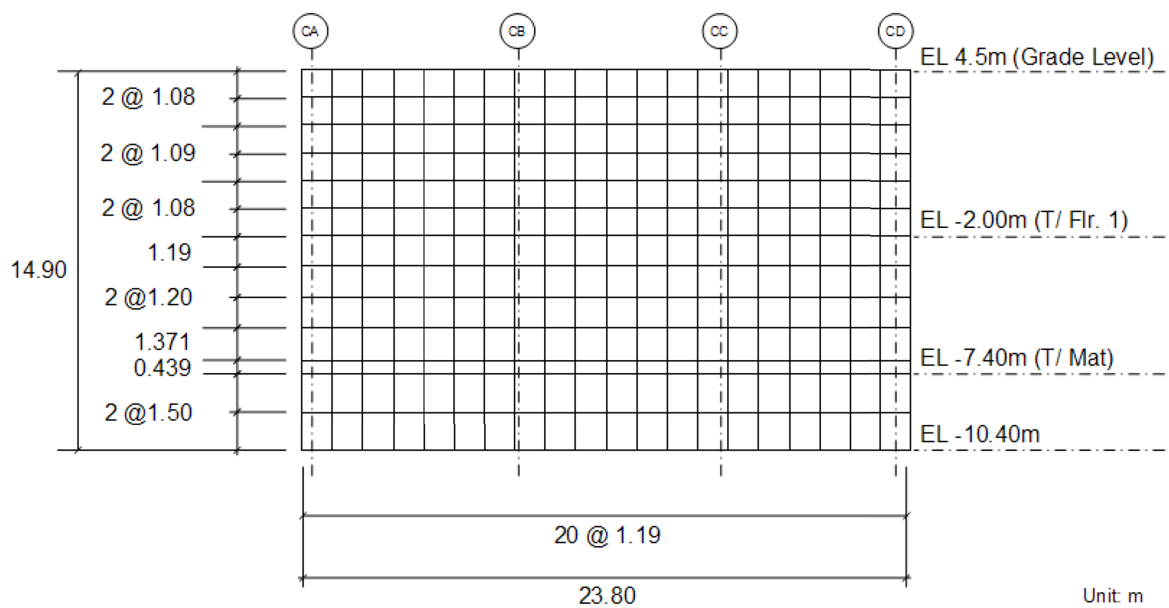


Figure 3.7.2-205 SASSI2010 Plate Elements for CB Exterior Walls (CB1)



(a) Walls on Column Rows CA and CD



(b) Walls on Column Rows C1 and C5

Figure 3.7.2-206 Overview of CB SASSI2010 SSI Model (CB1)

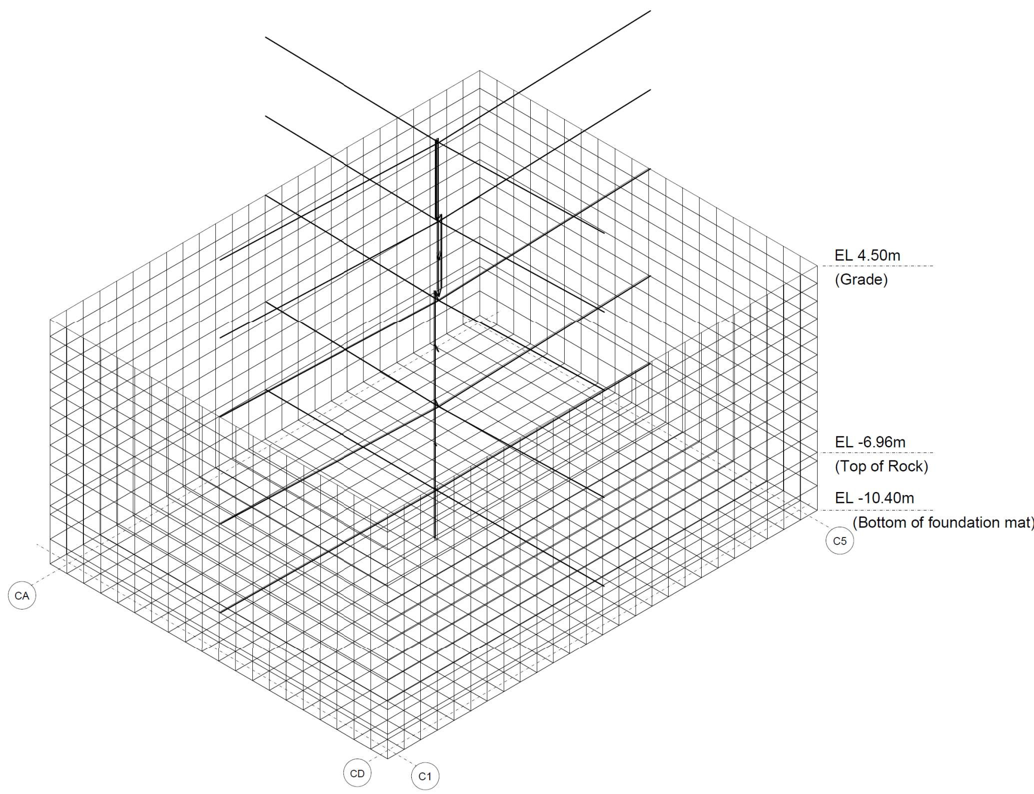


Figure 3.7.2-206a SASSI2010 Plate Element for CB Basemat (CB2)

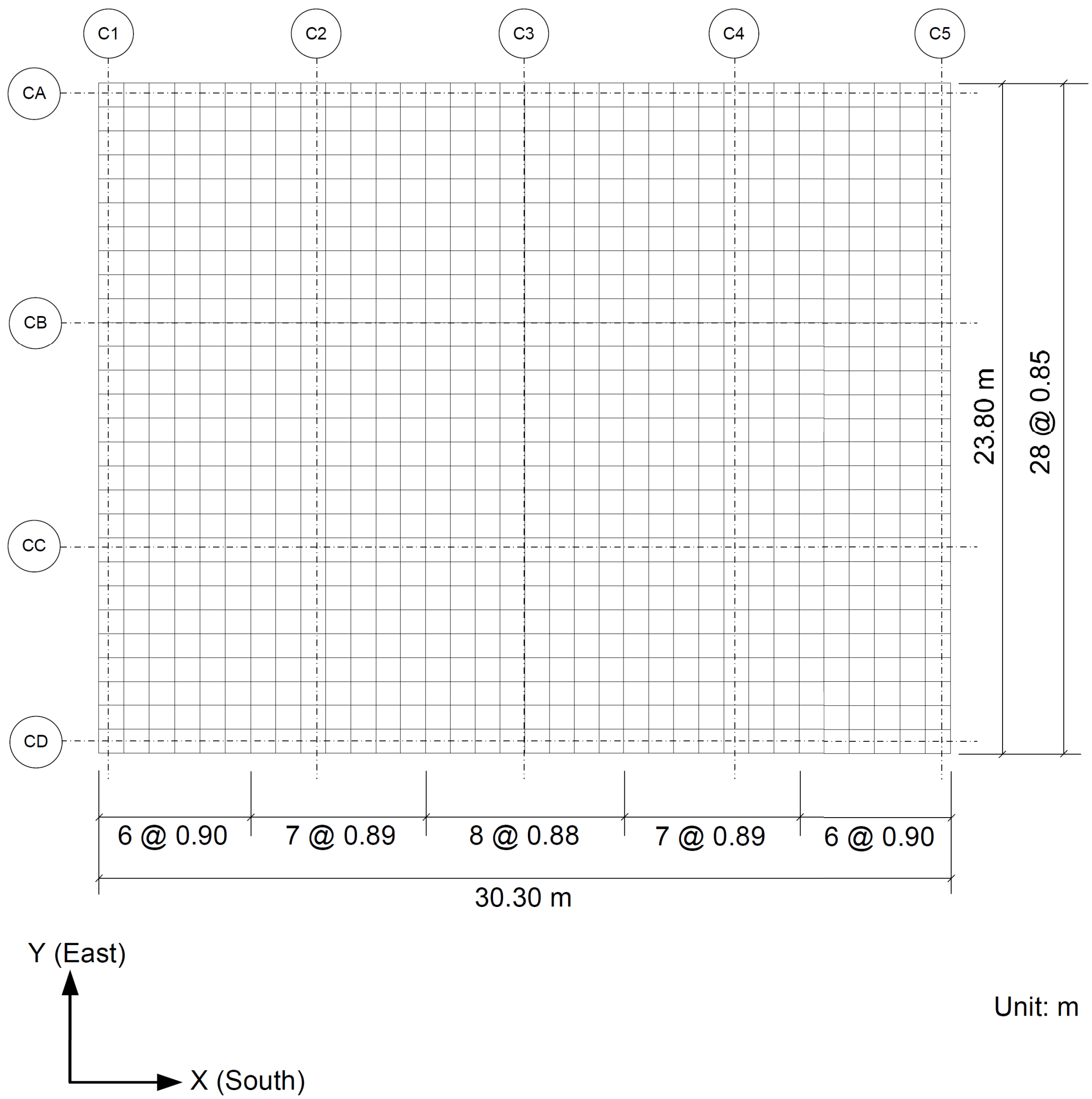
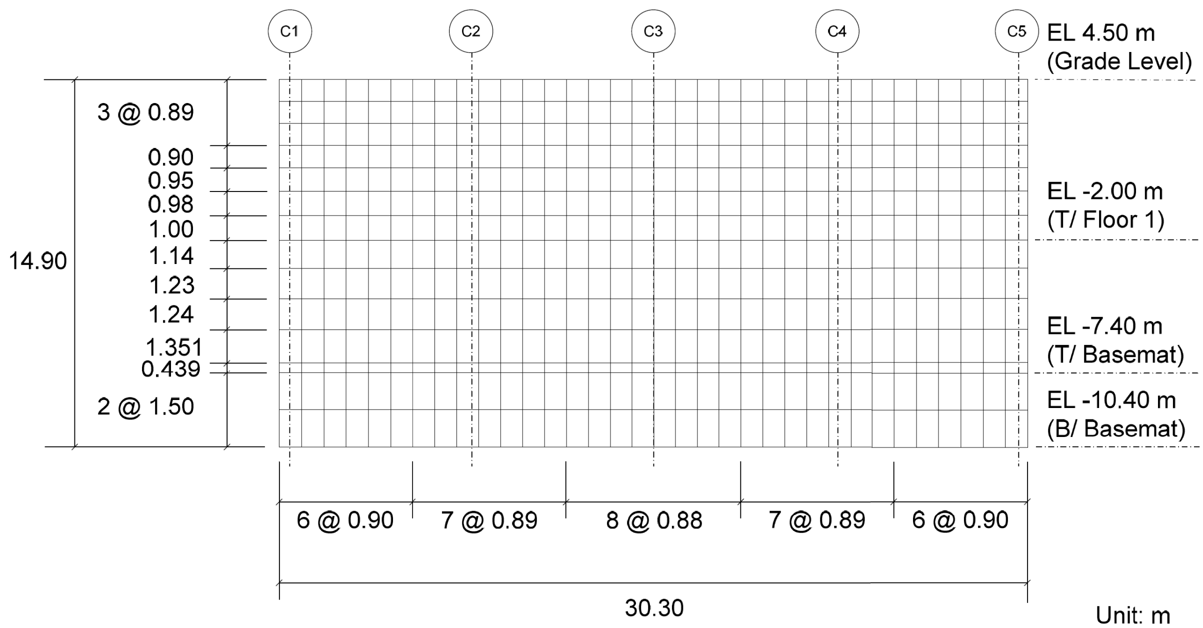
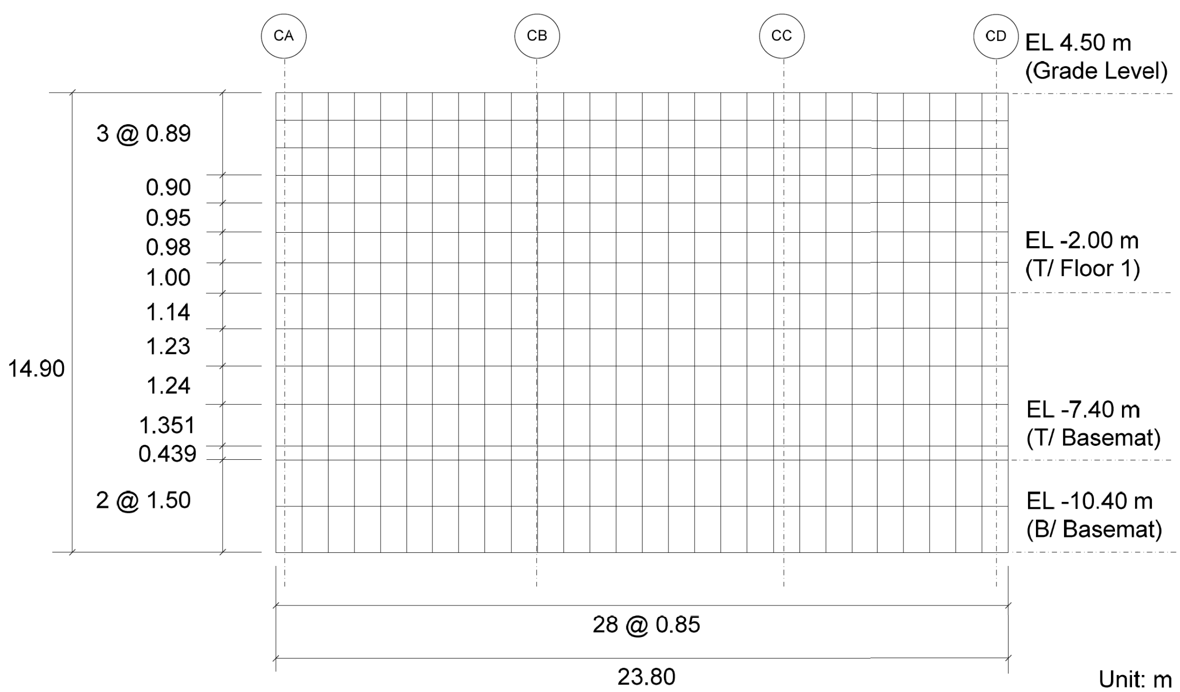


Figure 3.7.2-206b SASSI2010 Plate Element for CB Exterior Walls (CB2)



(a) Walls on Column Rows CA and CD



(b) Walls on Column Rows C1 and C5

Figure 3.7.2-206c Overview of SASSI2010 SSI CB Model (CB2)

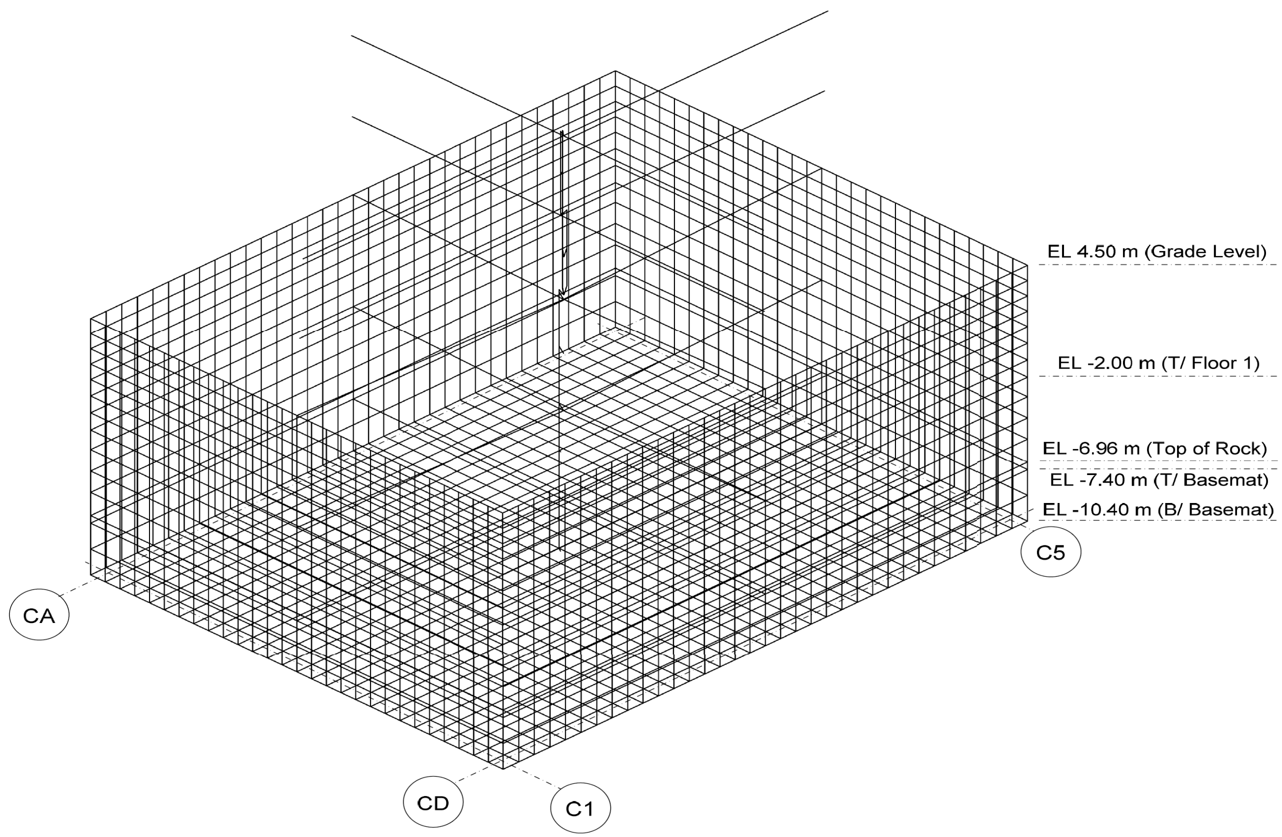


Figure 3.7.2-207a Comparison of Floor Response Spectra - RB/FB Refueling Floor in X-Direction

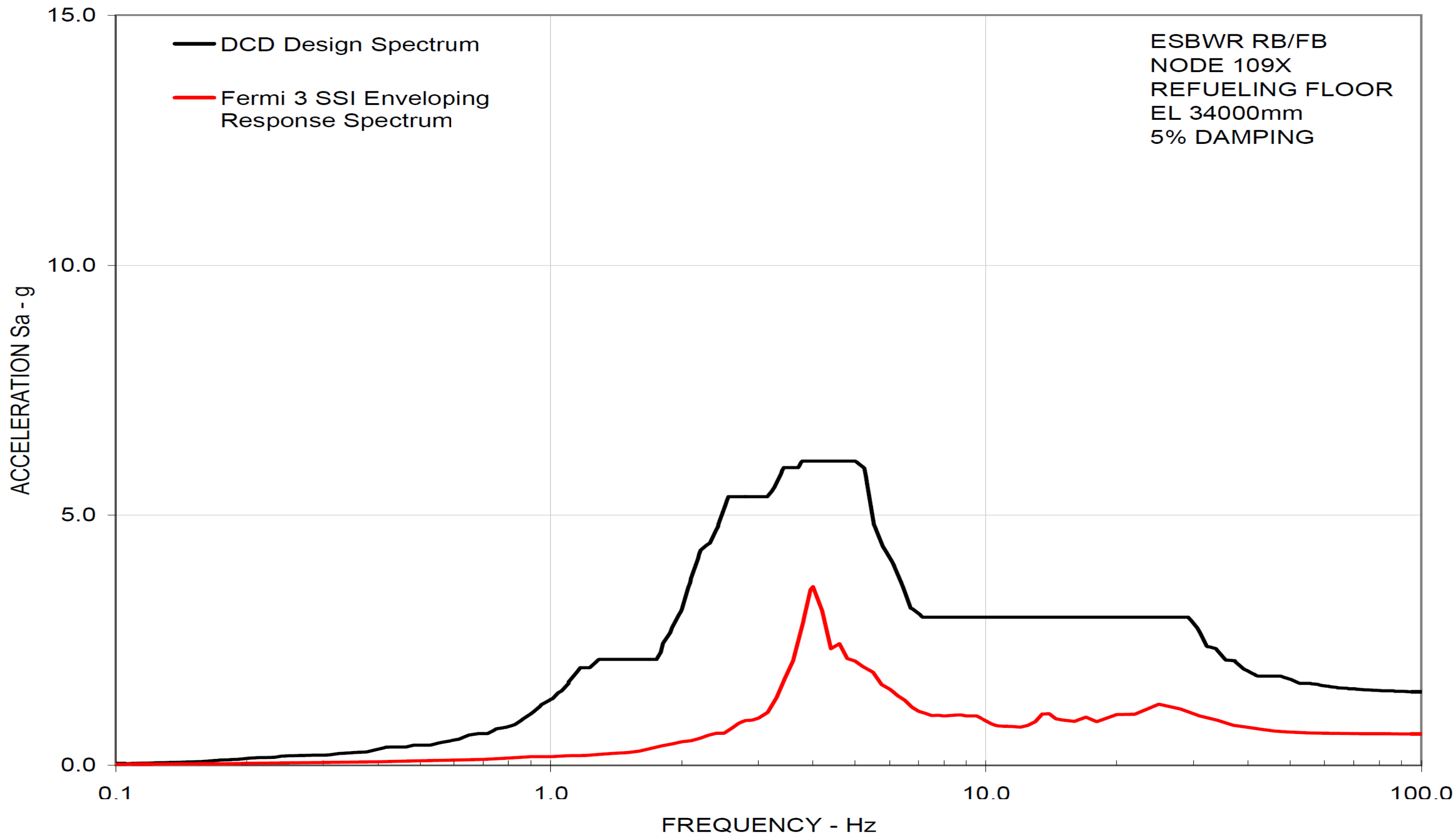


Figure 3.7.2-207b Comparison of Floor Response Spectra - RCCV Top Slab in X-Direction

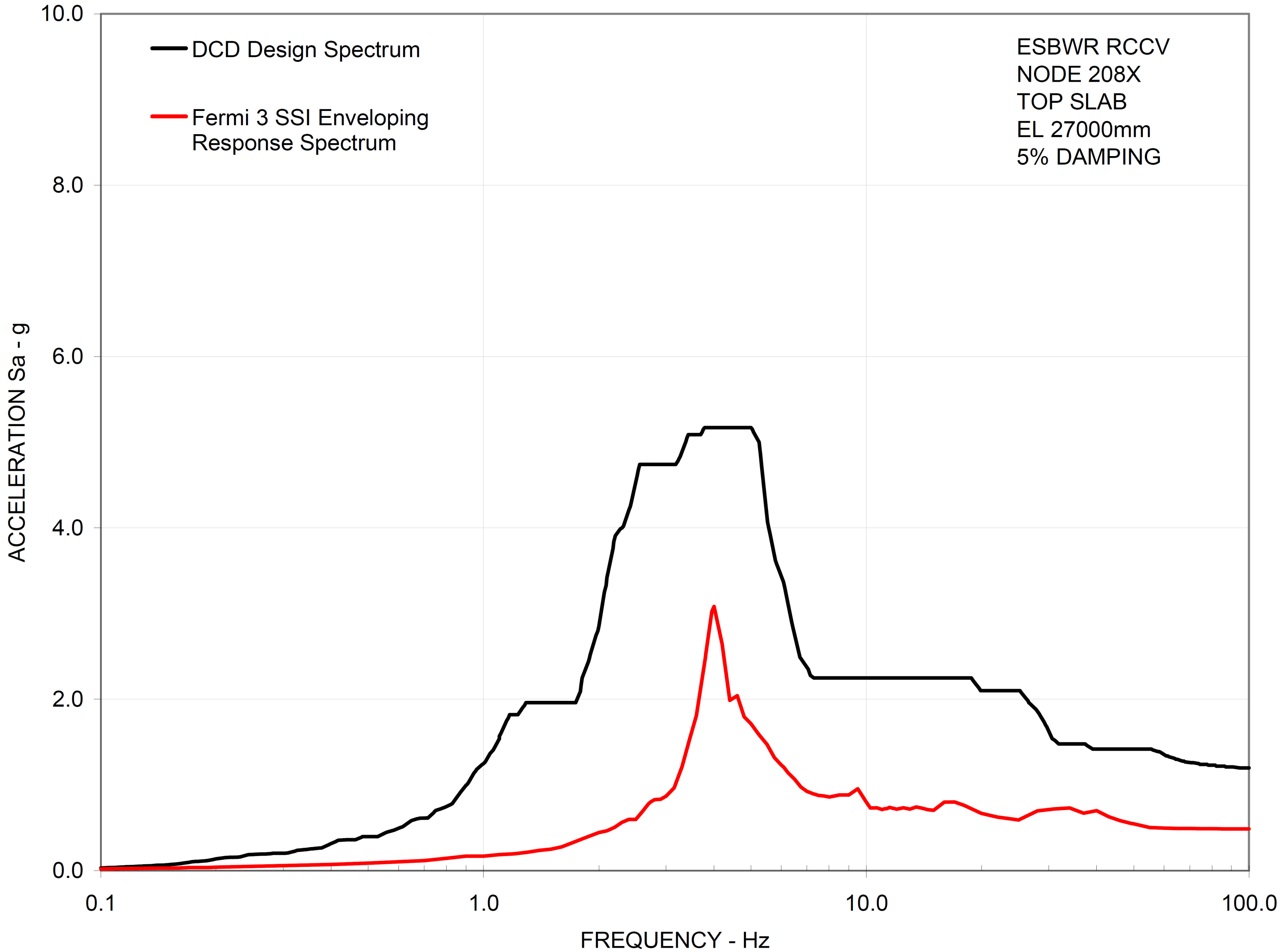


Figure 3.7.2-207c Comparison of Floor Response Spectra - Vent Wall Top in X-Direction

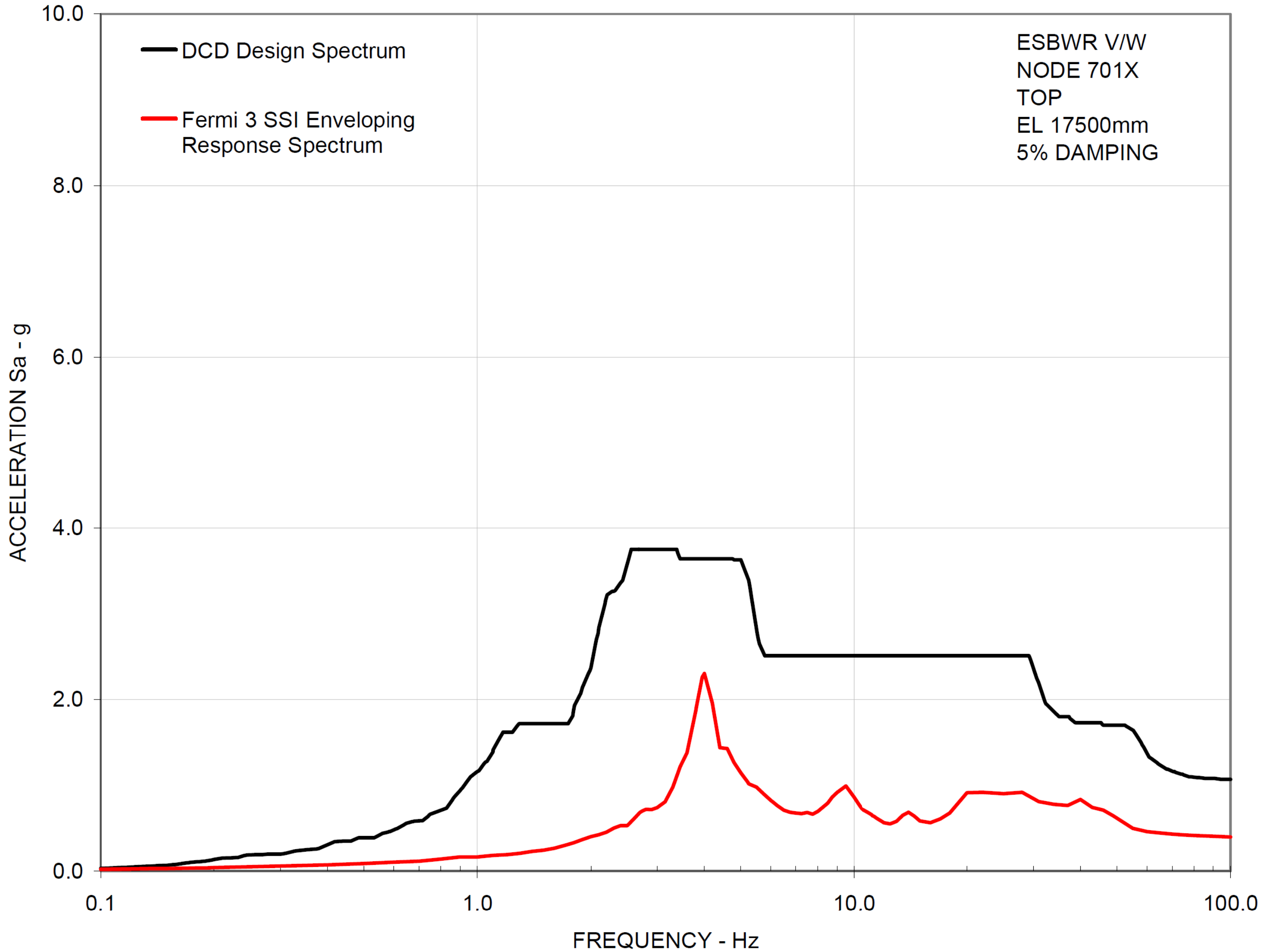


Figure 3.7.2-207d Comparison of Floor Response Spectra - RSW Top in X-Direction

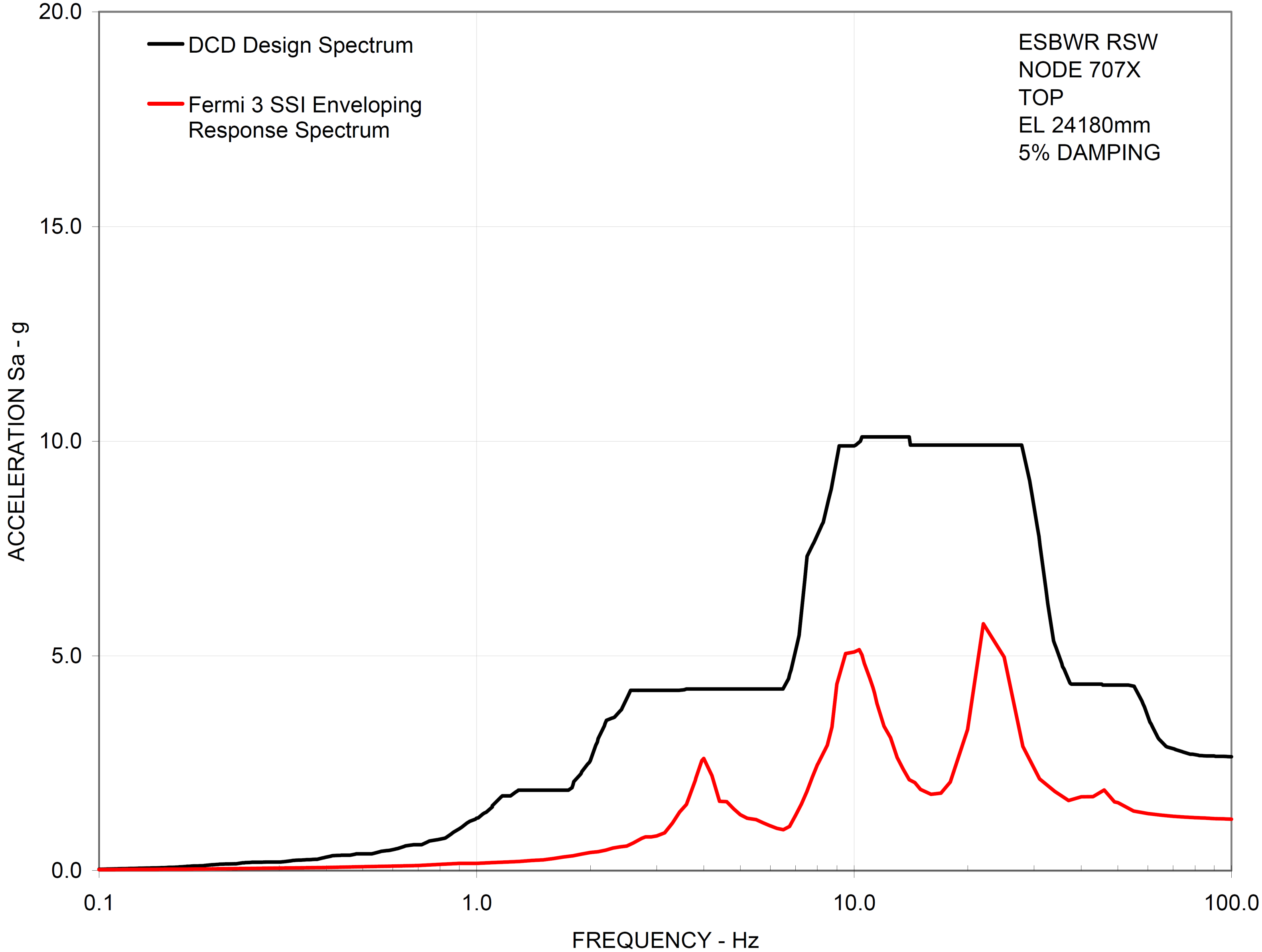


Figure 3.7.2-207e Comparison of Floor Response Spectra - RPV Top in X-Direction

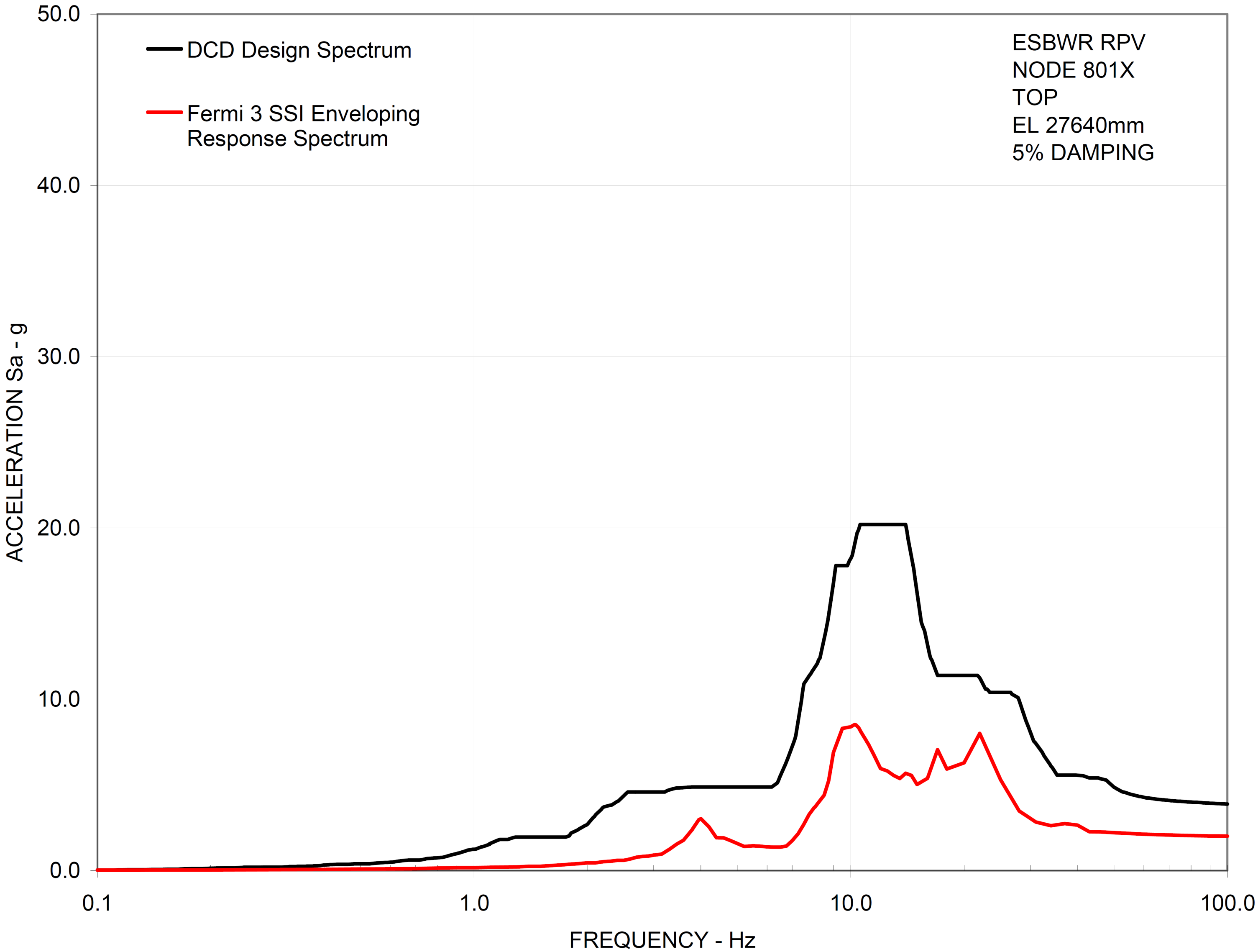


Figure 3.7.2-207f Comparison of Floor Response Spectra - RB/FB Basemat in X-Direction

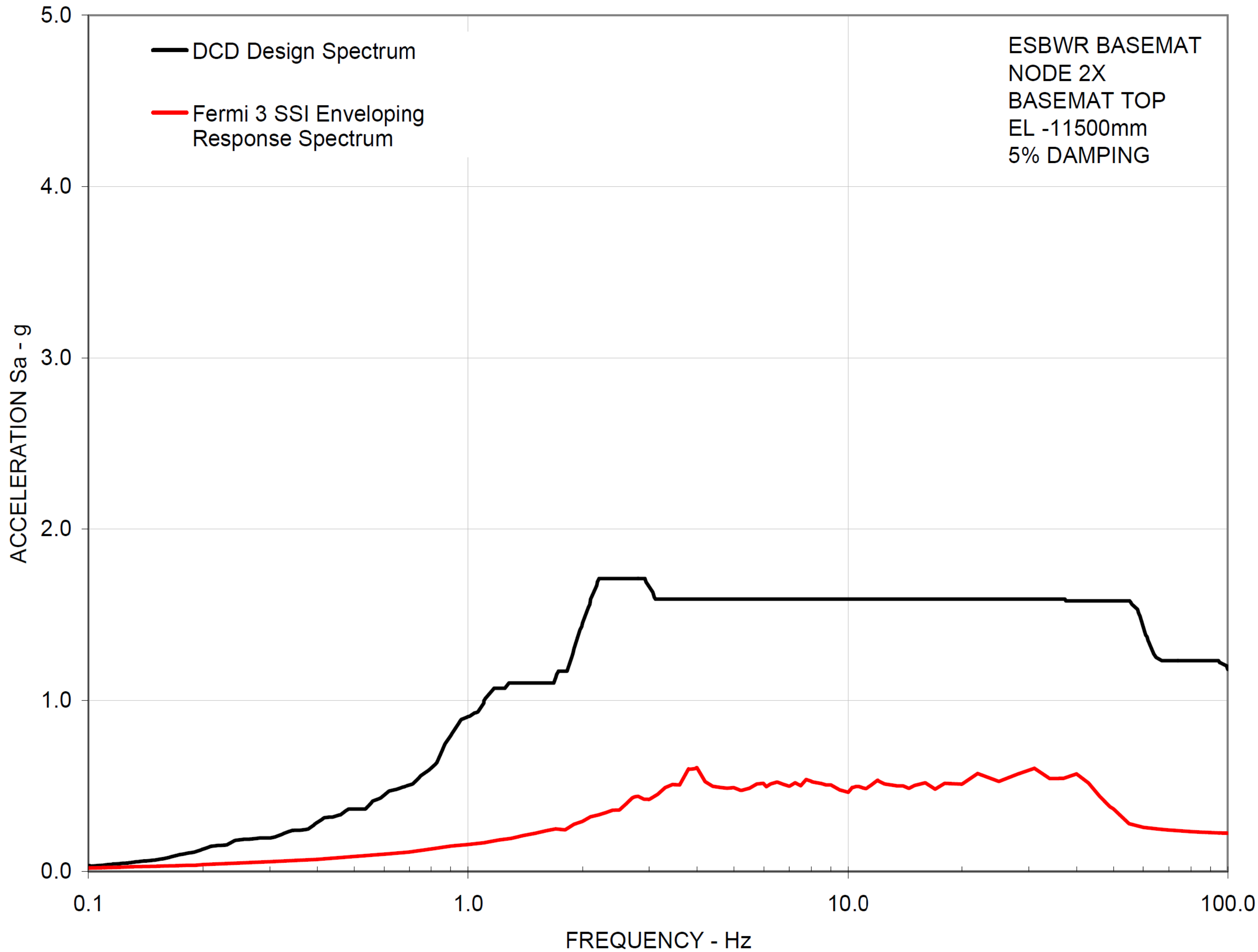


Figure 3.7.2-208a Comparison of Floor Response Spectra - RB/FB Refueling Floor in Y-Direction

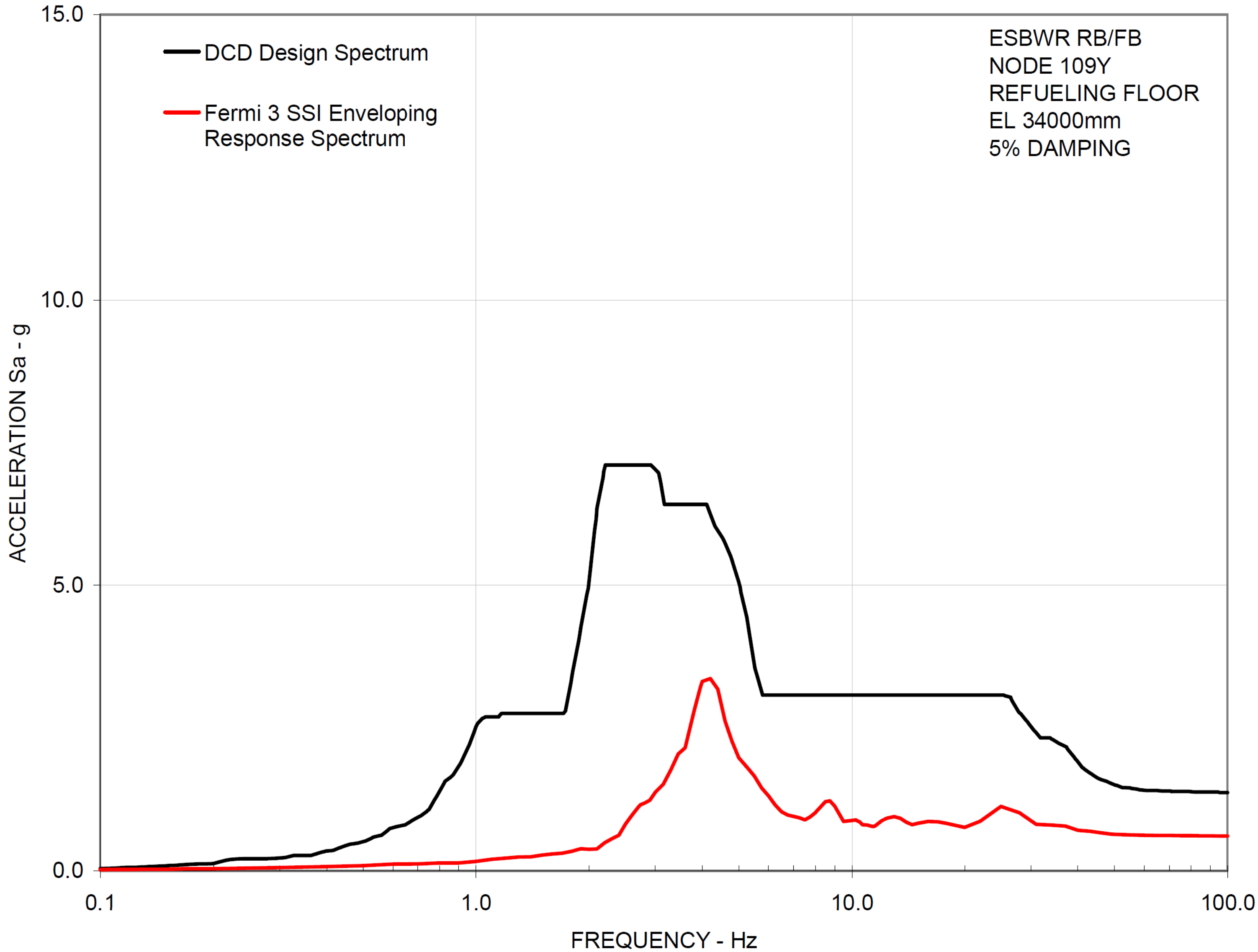


Figure 3.7.2-208b Comparison of Floor Response Spectra - RCCV Top Slab in Y-Direction

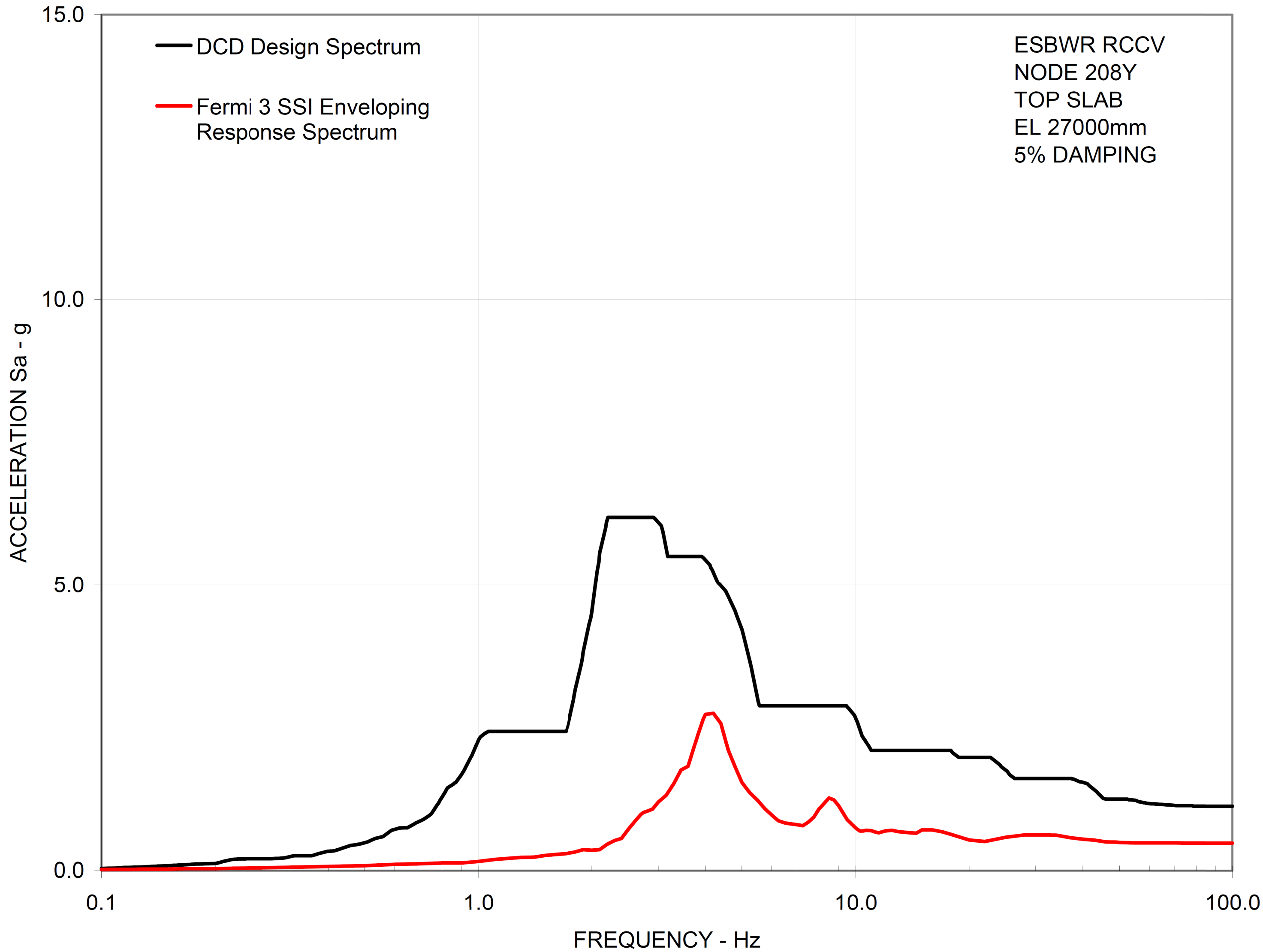


Figure 3.7.2-208c Comparison of Floor Response Spectra - Vent Wall Top in Y-Direction

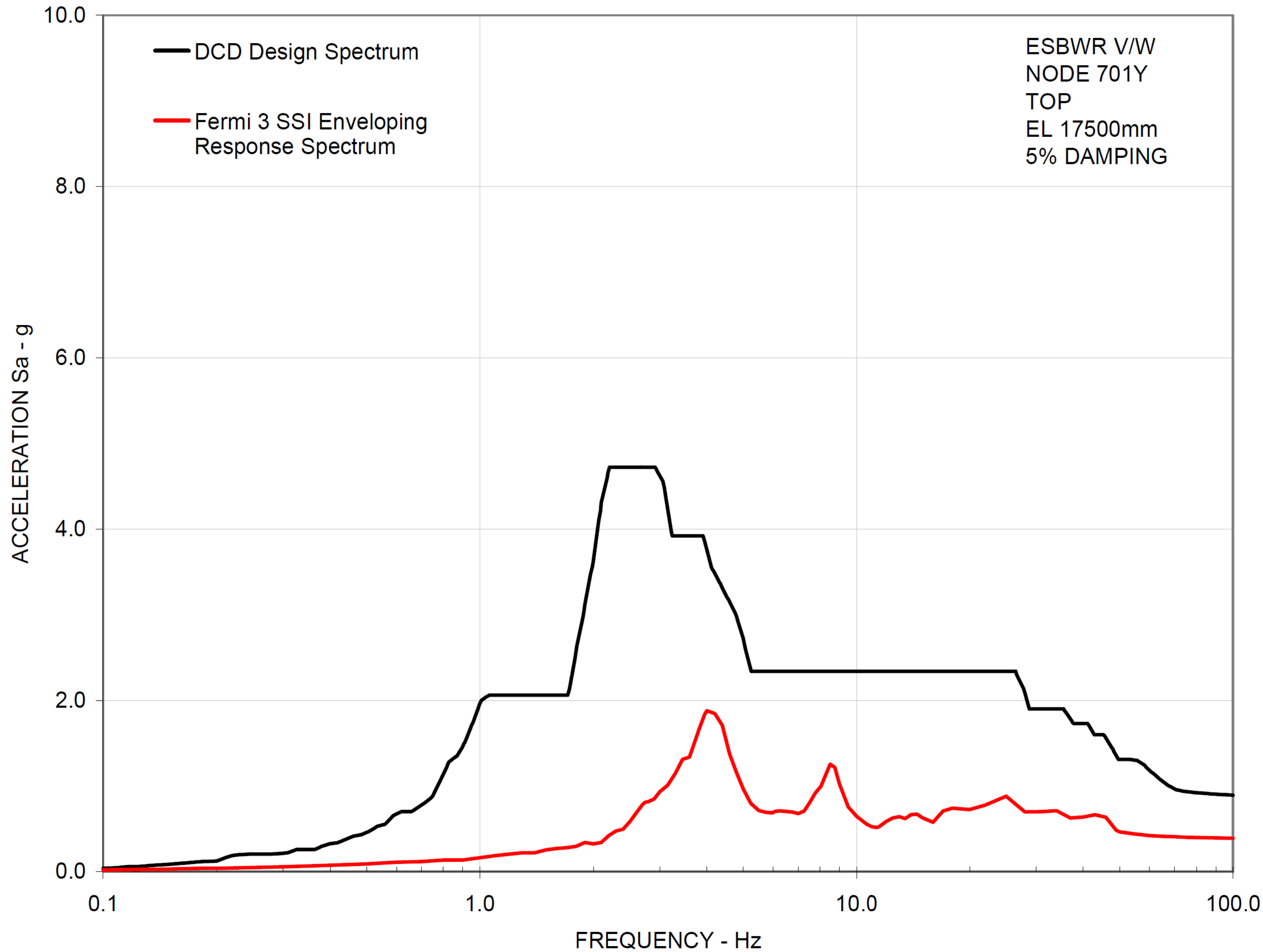


Figure 3.7.2-208d Comparison of Floor Response Spectra - RSW Top in Y-Direction

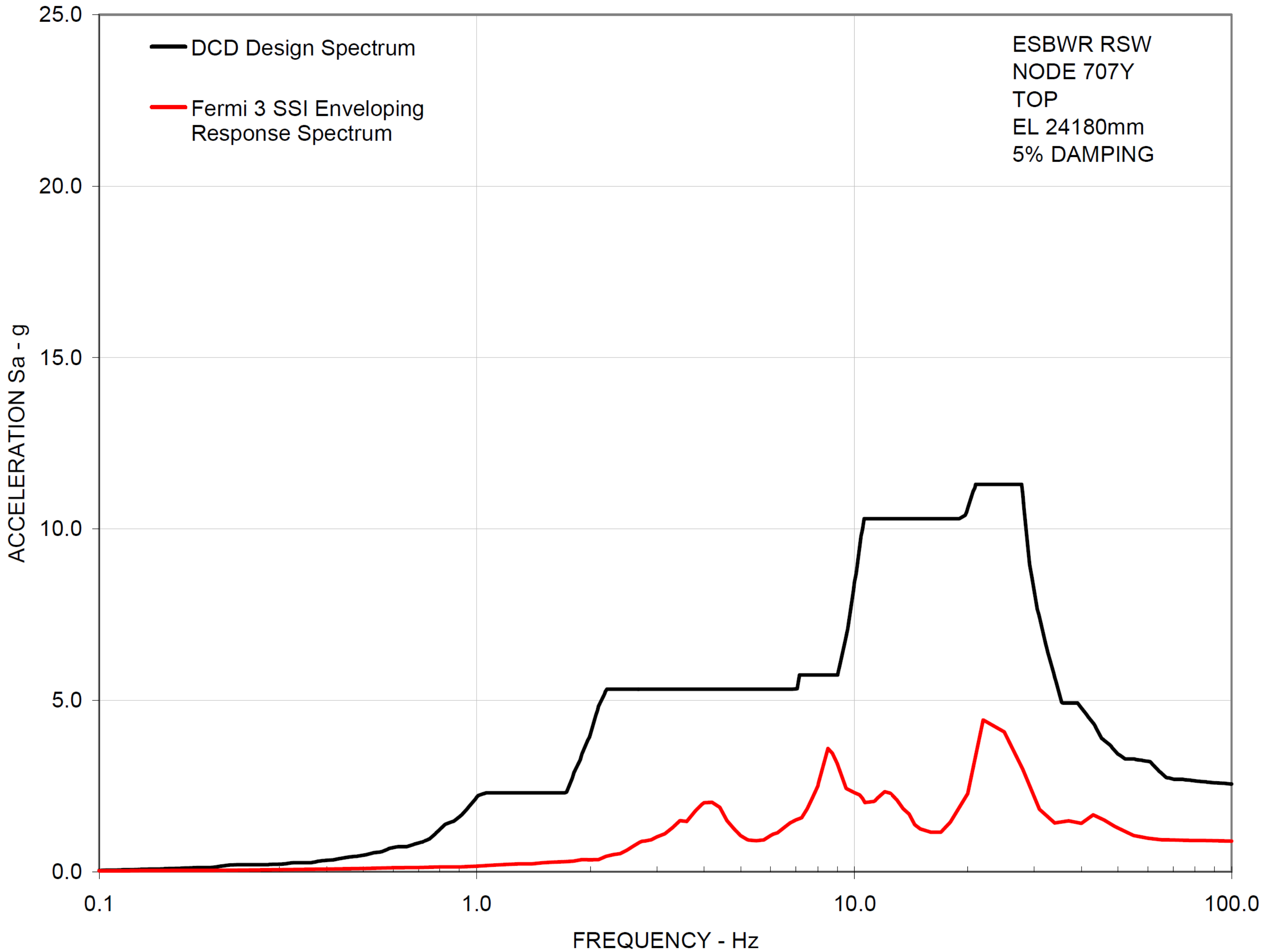


Figure 3.7.2-208e Comparison of Floor Response Spectra - RPV Top in Y-Direction

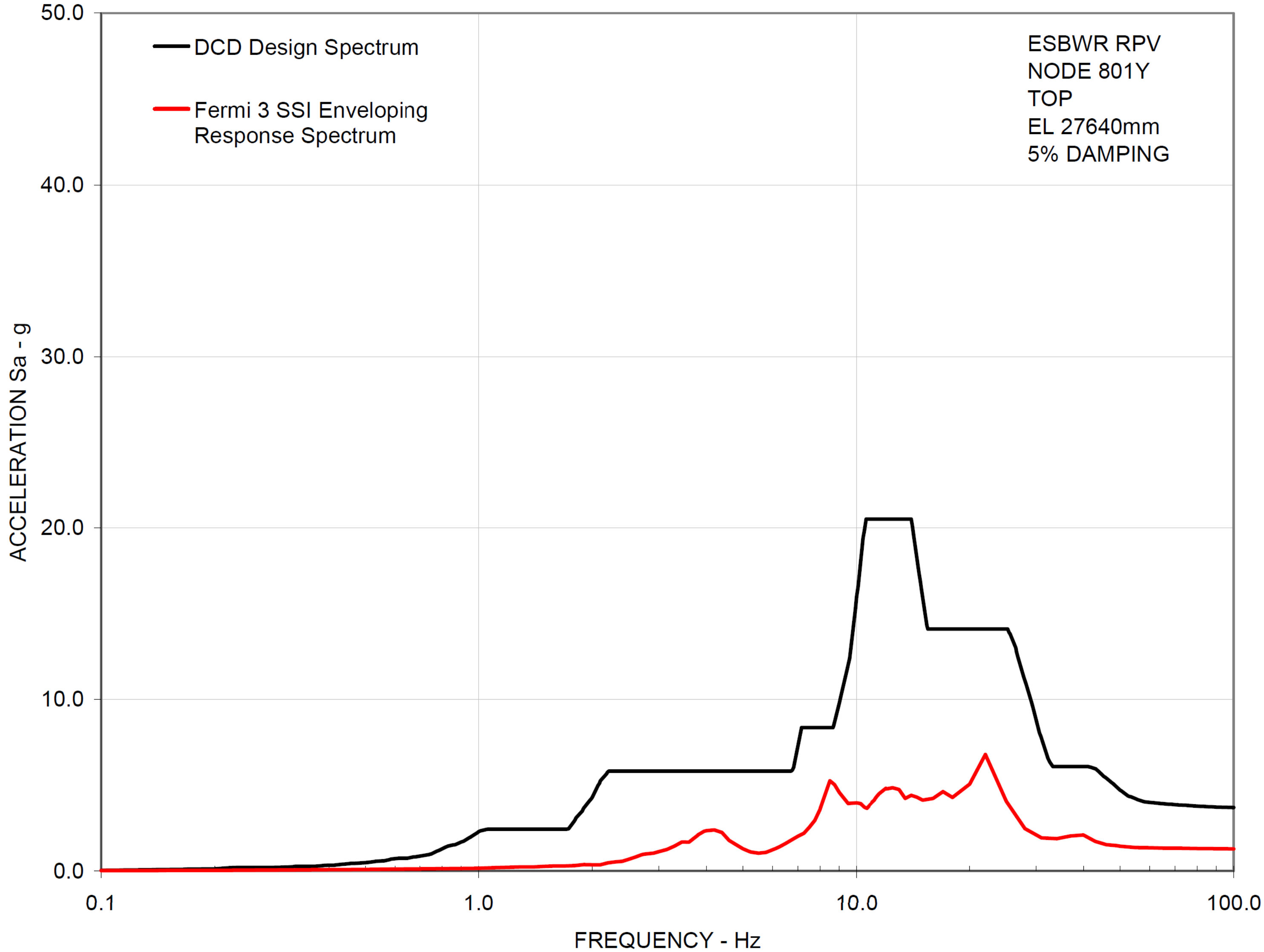


Figure 3.7.2-208f Comparison of Floor Response Spectra - RB/FB Basemat in Y-Direction

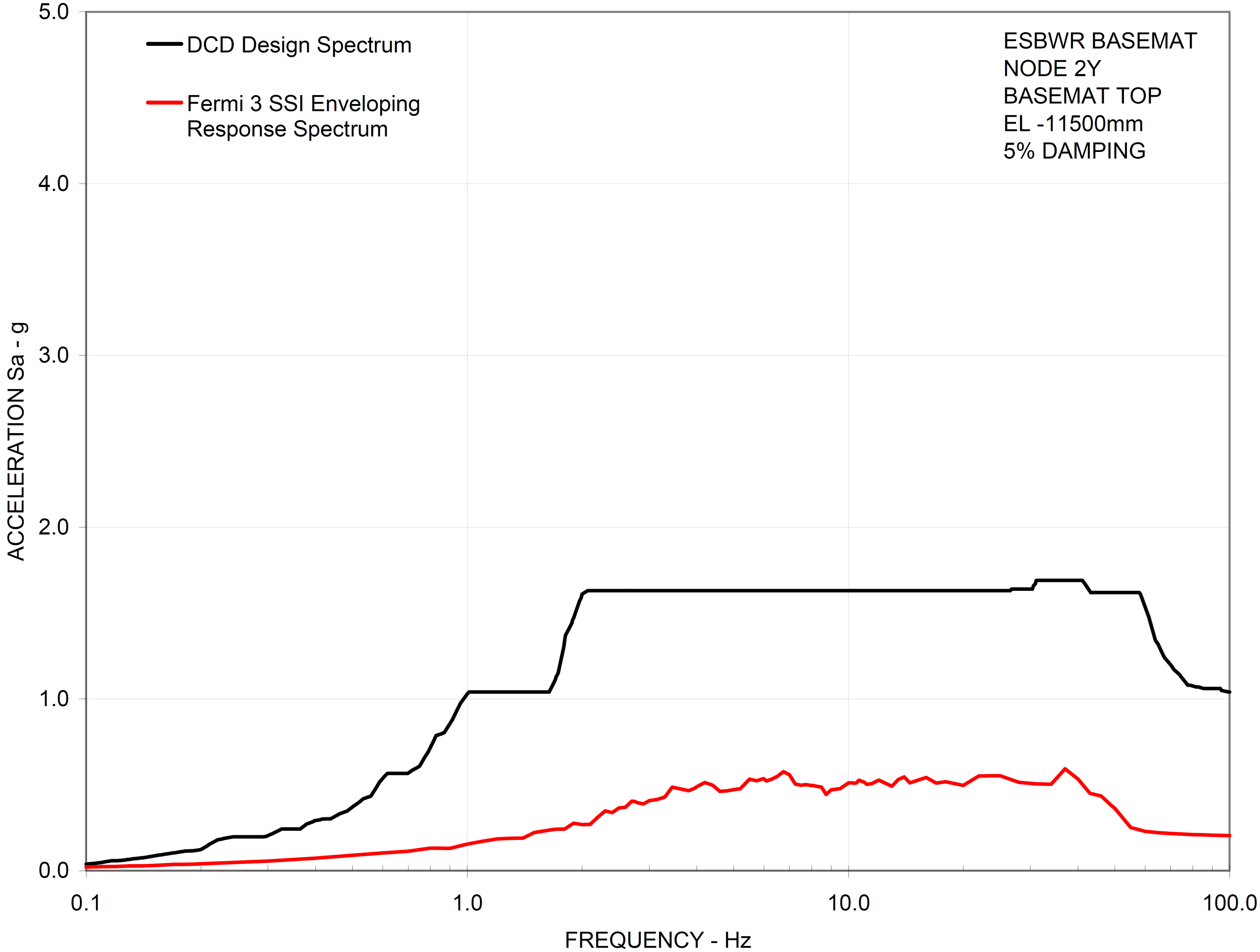


Figure 3.7.2-209a Comparison of Floor Response Spectra - RB/FB Refueling Floor in Z-Direction

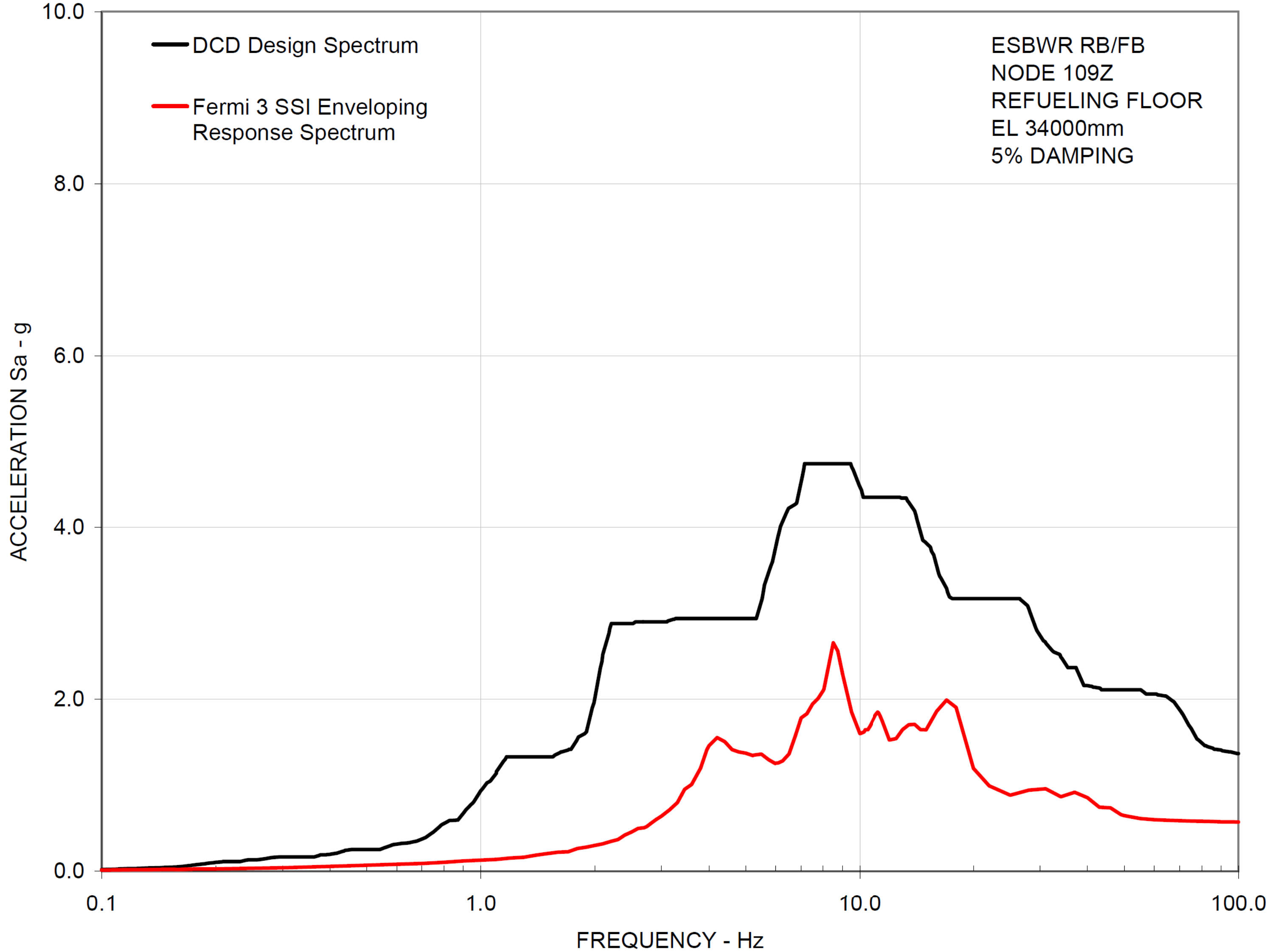


Figure 3.7.2-209b Comparison of Floor Response Spectra - RCCV Top Slab in Z-Direction

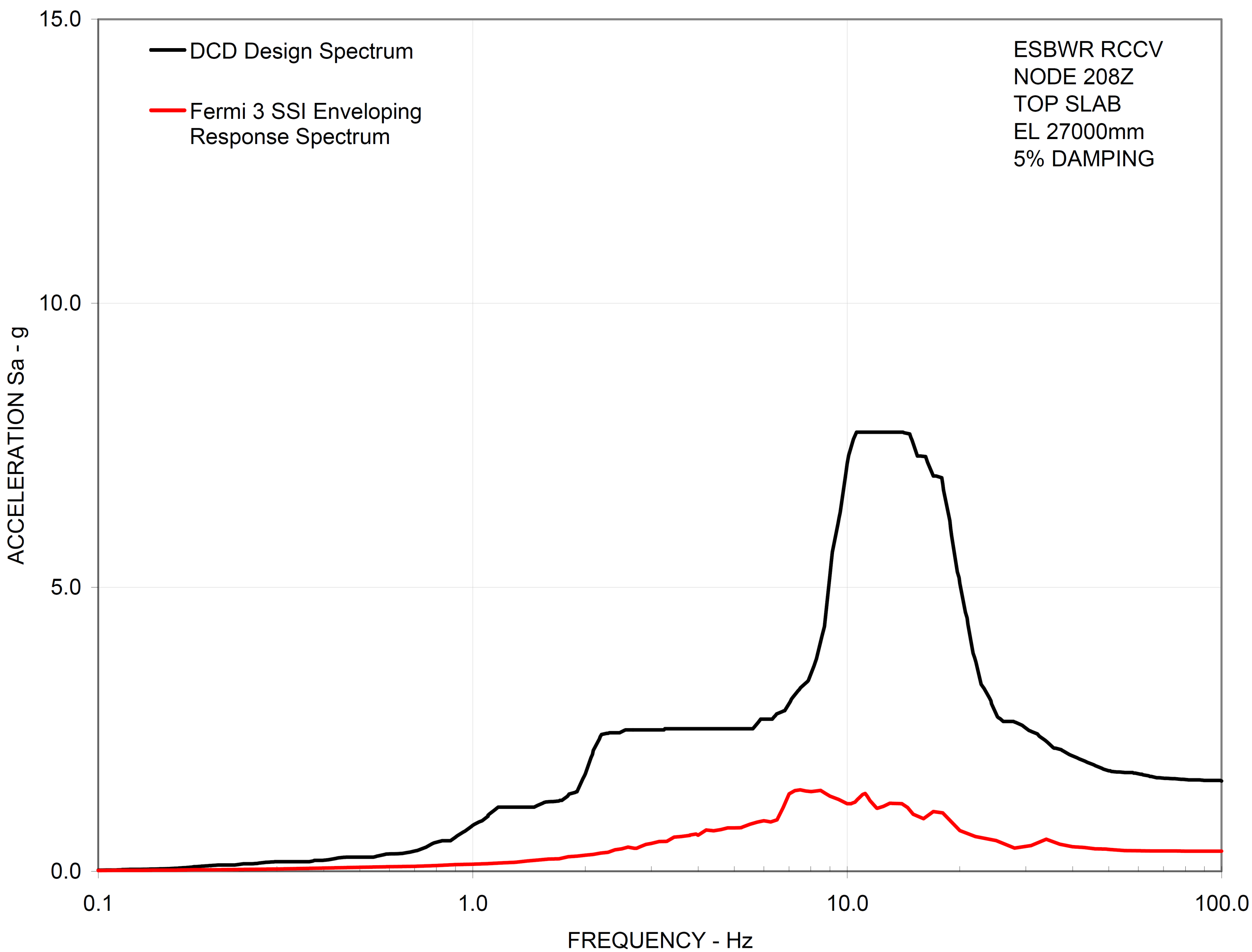


Figure 3.7.2-209c Comparison of Floor Response Spectra - Vent Wall Top in Z-Direction

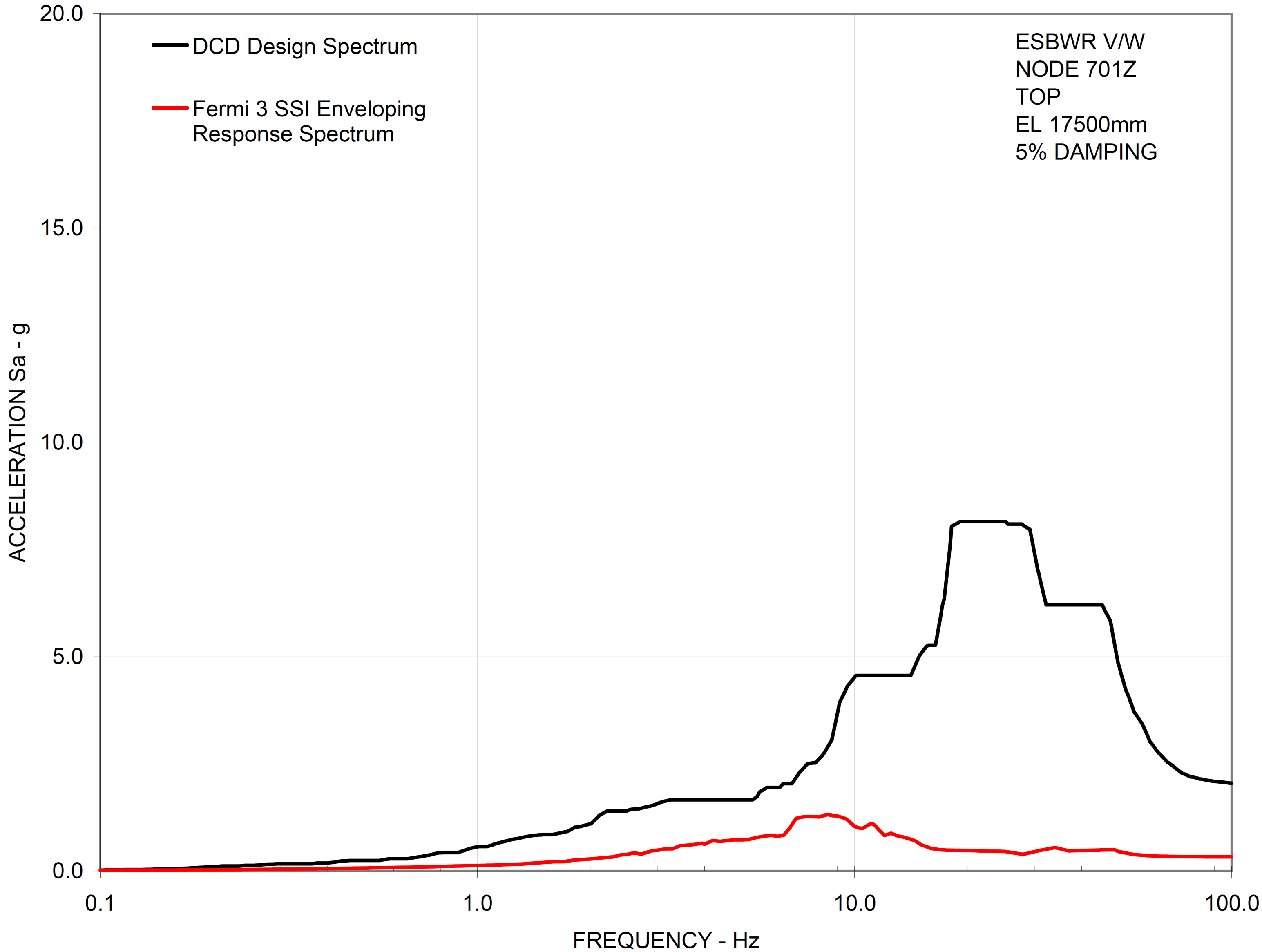


Figure 3.7.2-209d Comparison of Floor Response Spectra - RSW Top in Z-Direction

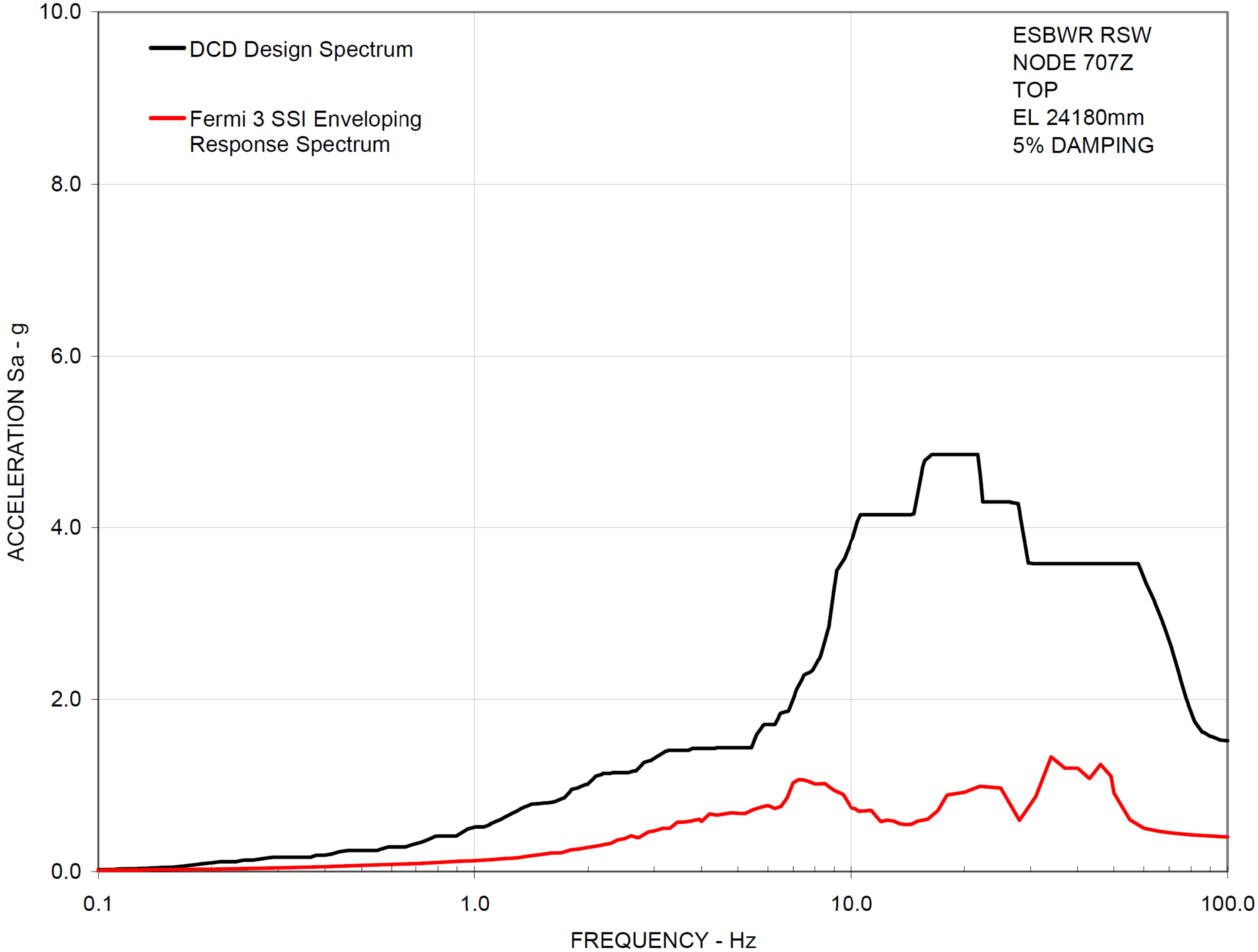


Figure 3.7.2-209e Comparison of Floor Response Spectra - RPV Top in Z-Direction

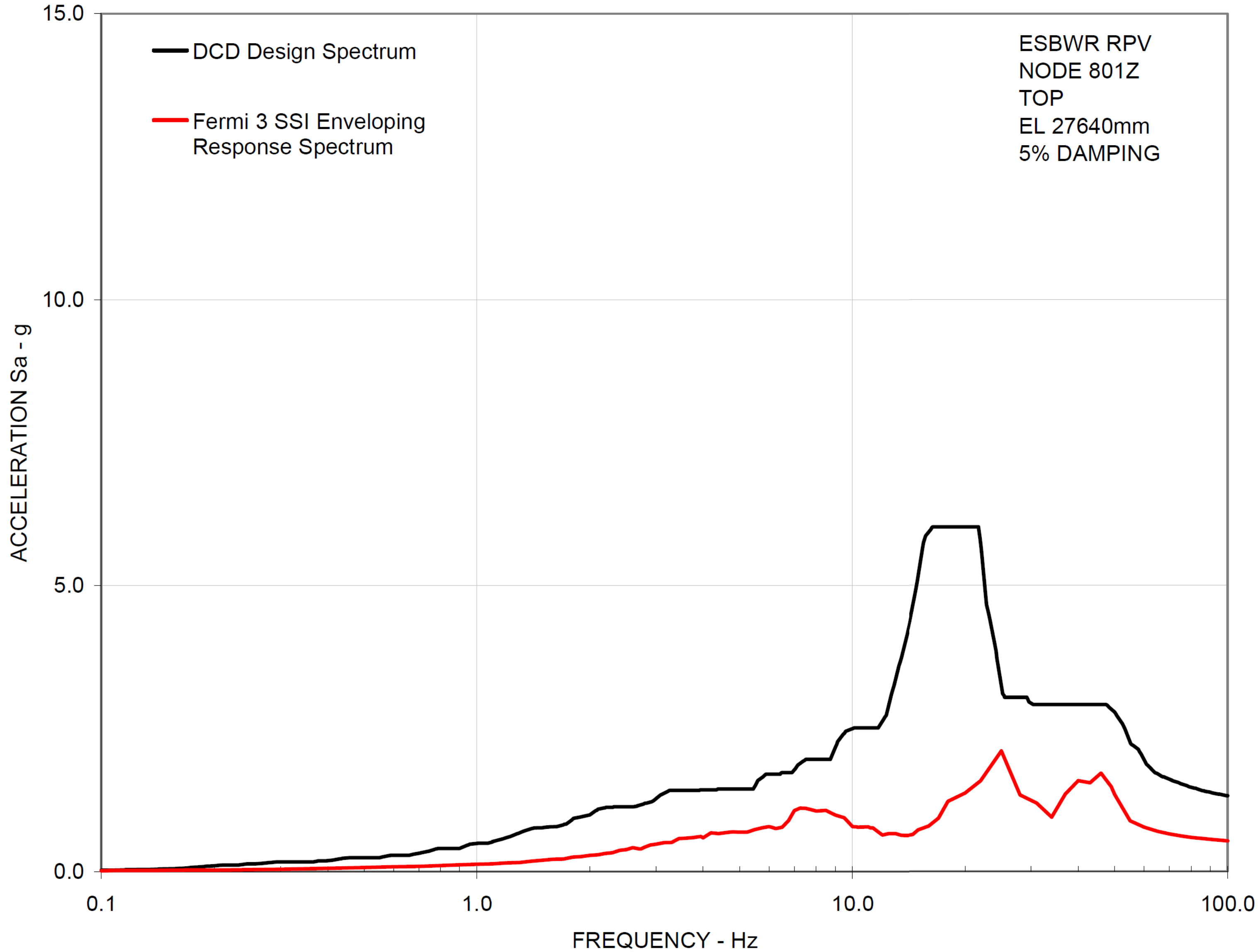


Figure 3.7.2-209f Comparison of Floor Response Spectra - RB/FB Basemat in Z-Direction

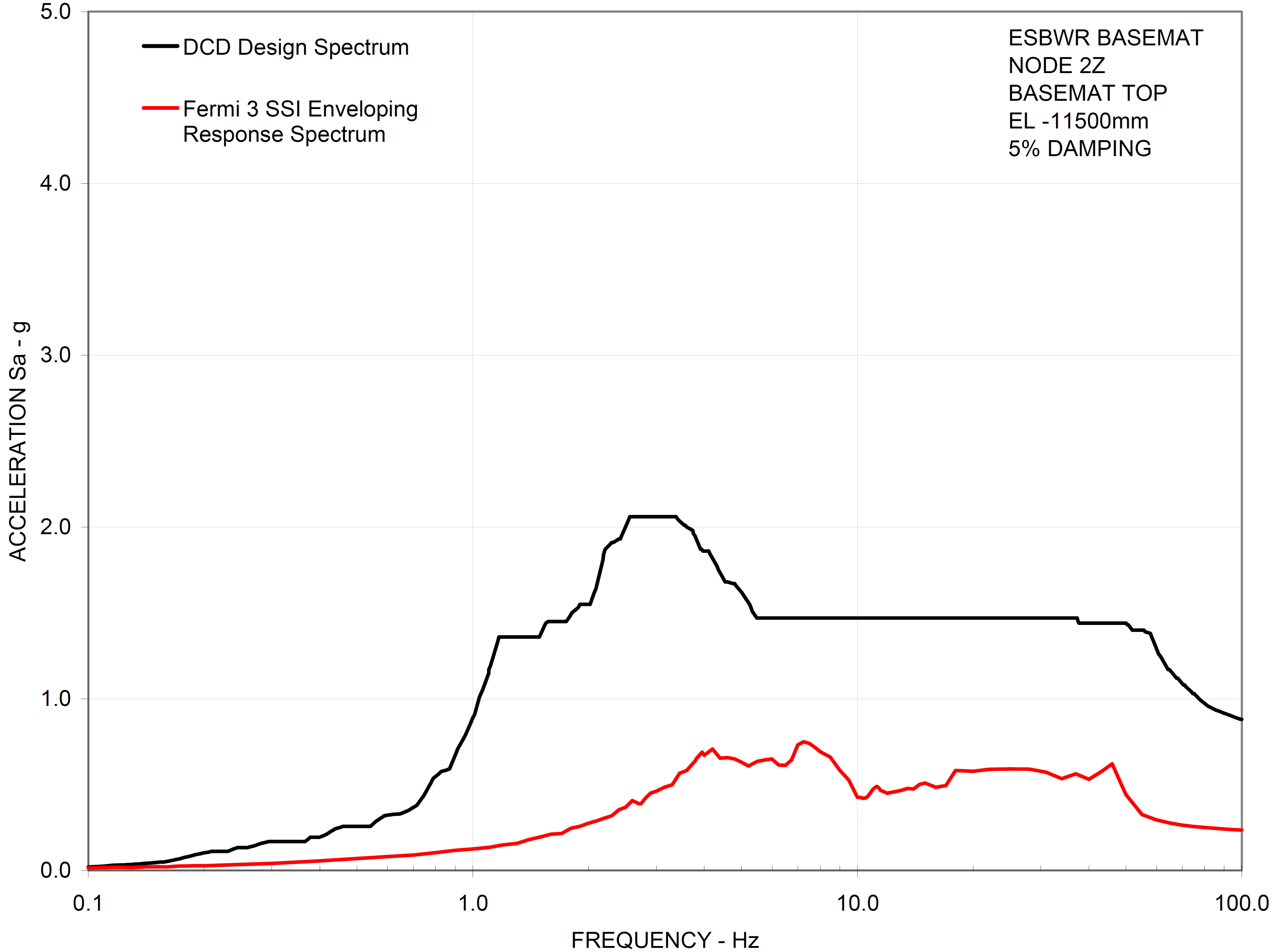


Figure 3.7.2-210a Comparison of Floor Response Spectra - CB Top in X-Direction

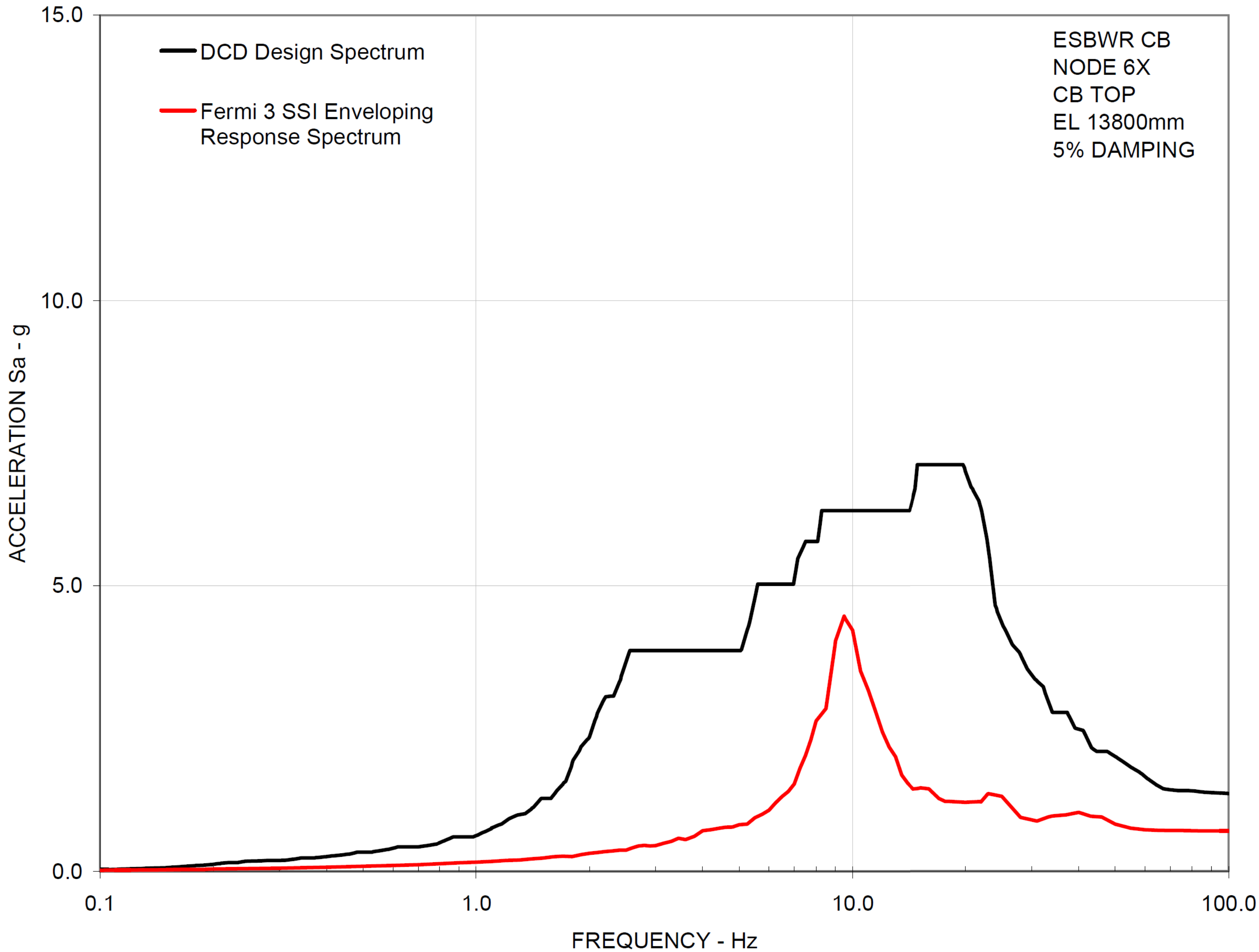


Figure 3.7.2-210b Comparison of Floor Response Spectra - CB Basemat in X-Direction

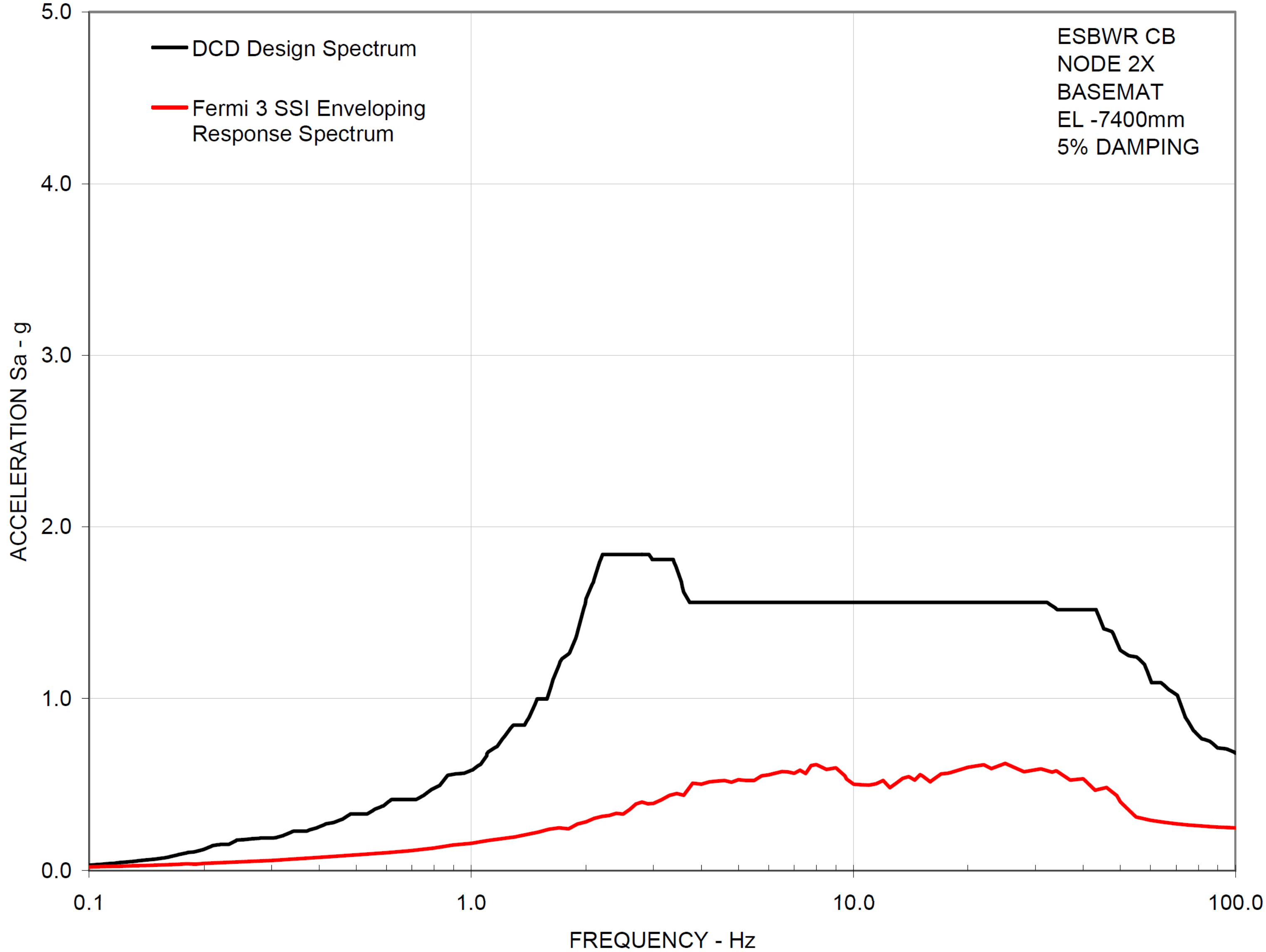


Figure 3.7.2-211a Comparison of Floor Response Spectra - CB Top in Y-Direction

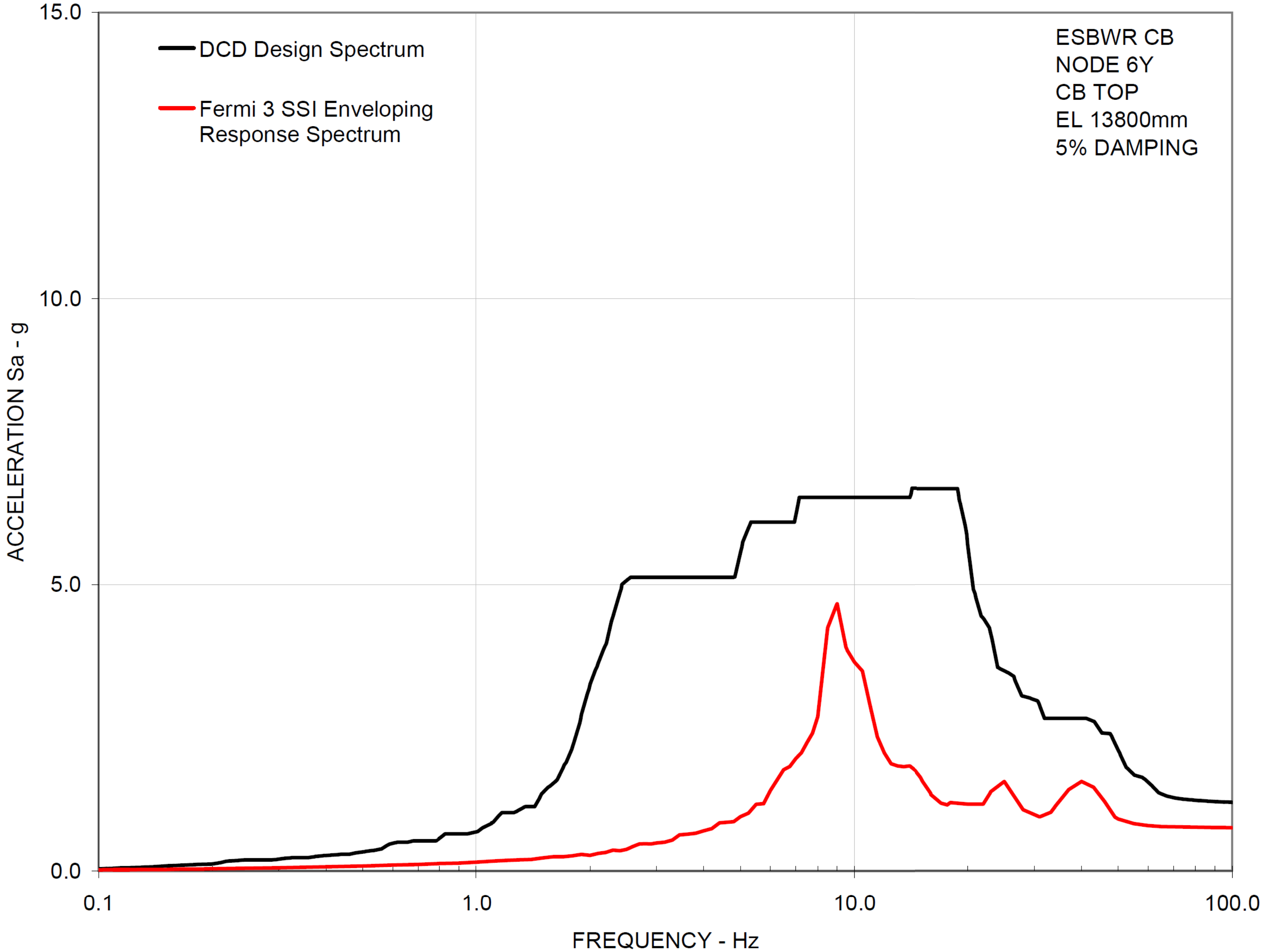


Figure 3.7.2-211b Comparison of Floor Response Spectra - CB Basemat in Y-Direction

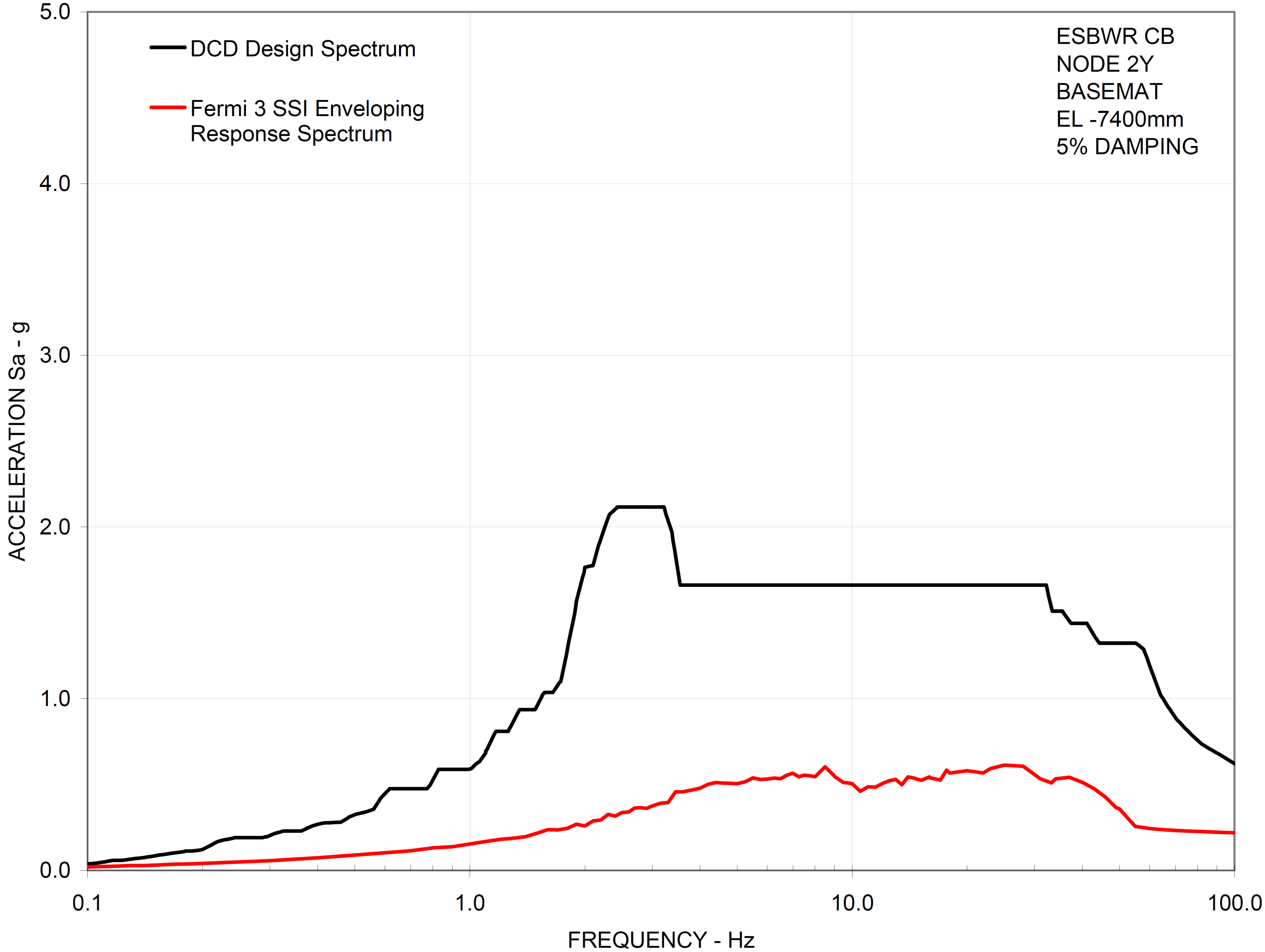


Figure 3.7.2-212a Comparison of Floor Response Spectra - CB Top in Z-Direction

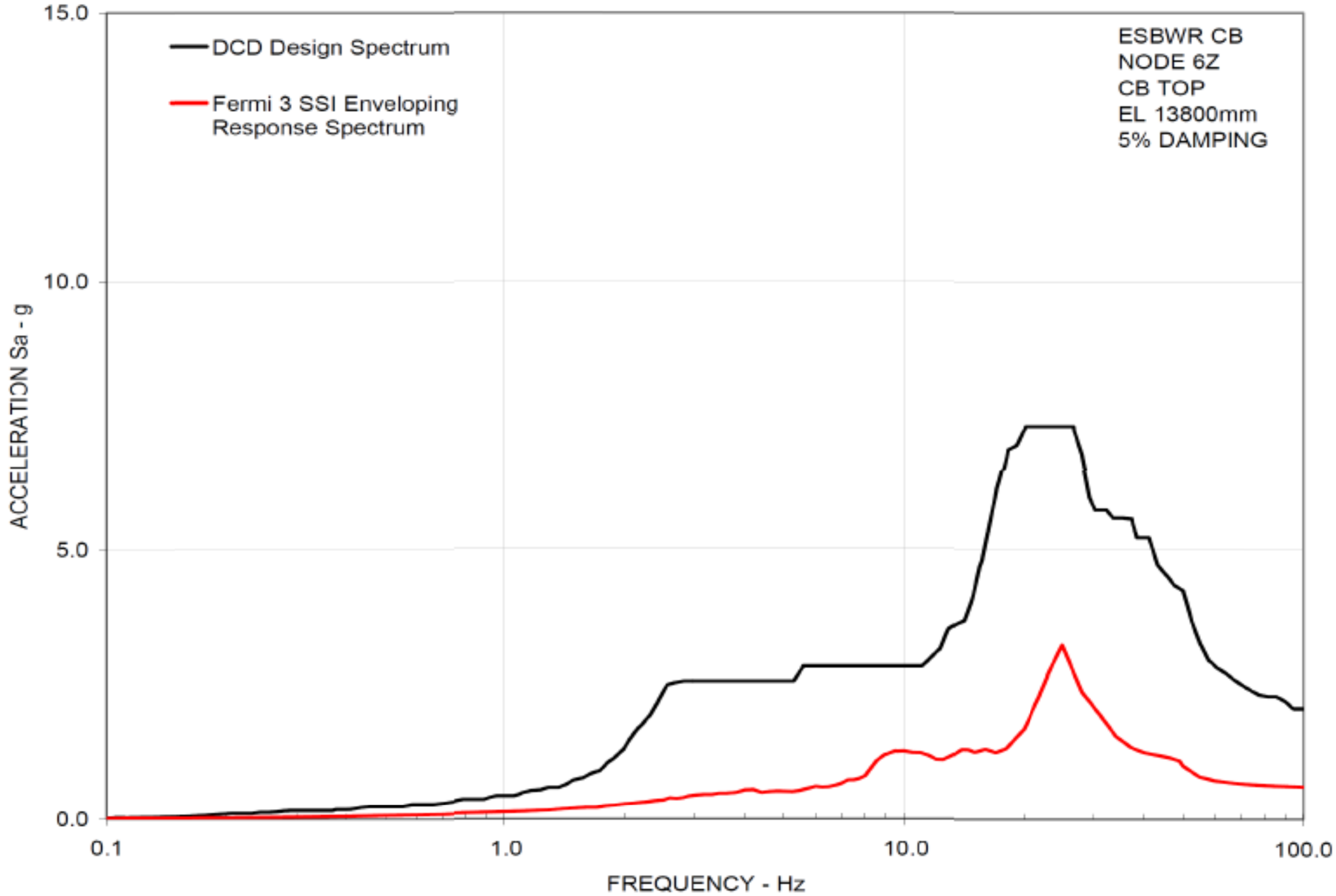
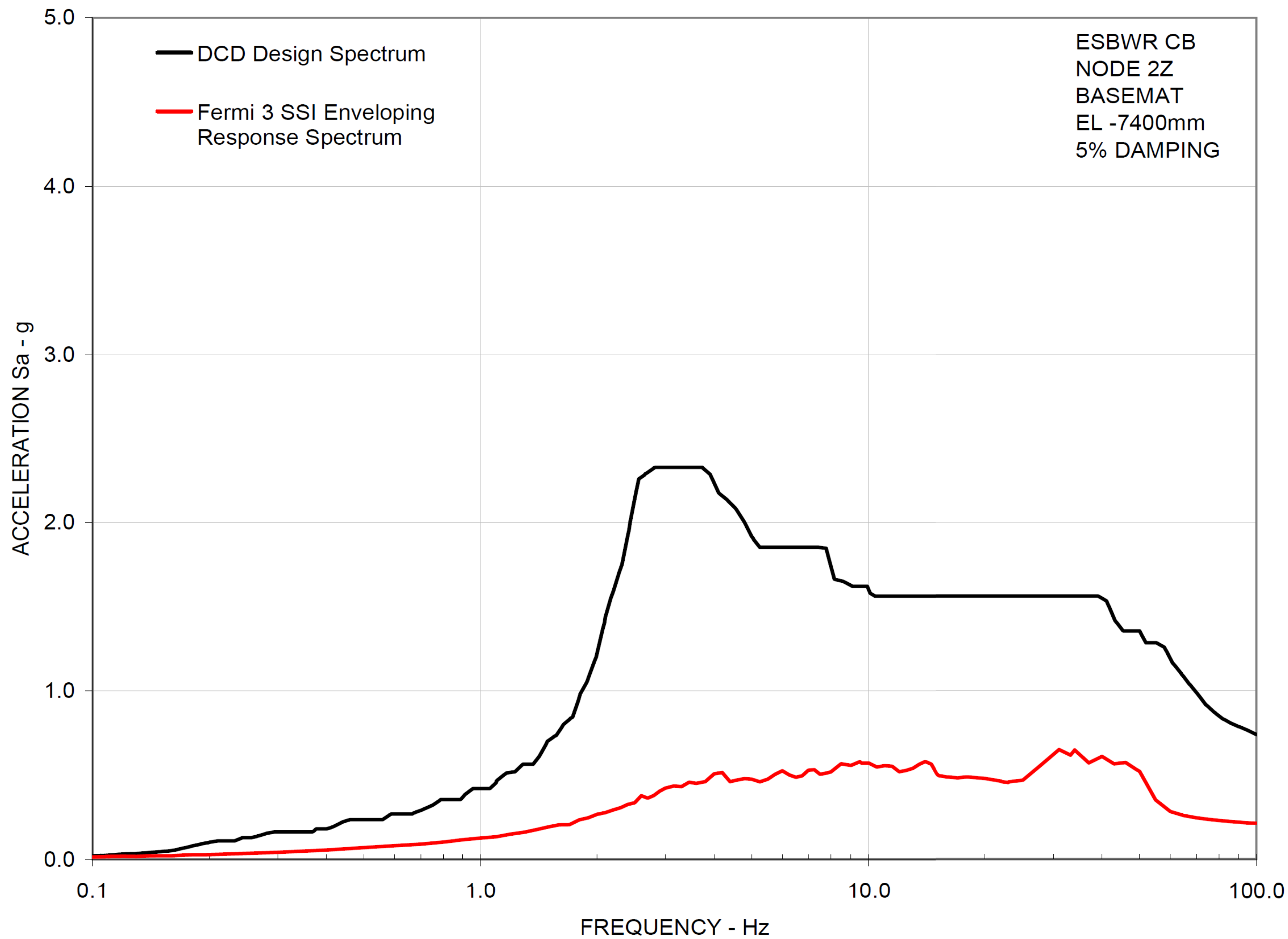


Figure 3.7.2-212b Comparison of Floor Response Spectra - CB Basemat in Z-Direction



3.8 Seismic Category I Structures

The Seismic Category I structures include the Concrete Containment, Reactor Building (RB), Control Building (CB), Fuel Building (FB) and Firewater Service Complex (FWSC).

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 loss-of-coolant-accident (LOCA). The containment structure is totally enclosed by the RB. 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 functional 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 Appendix 3G Section 3G.1. Appendix 3G Section 3G.1 contains a more detailed description of the containment and the analytical models, inputs, analytical procedures, figures, results from controlling load combinations, components with controlling concrete stresses, reinforcement stresses, and liner strains for the concrete containment vessel.

The containment is a low-leakage reinforced concrete structure with an internal steel liner in the drywell and wetwell to serve as a leaktight membrane. The containment is a cylindrical shell structure, which consists of the Reactor Pressure Vessel (RPV) pedestal, the containment cylindrical wall, the top slab, the suppression pool slab and the foundation mat. The containment is divided by the diaphragm floor and the vent wall into a drywell (upper and lower) and a wetwell. The top slab of the concrete containment is an integral part of the Isolation Condenser/Passive Containment Cooling System (IC/PCCS) pools (including expansion pools), the buffer pool, which is also used to store the dryer, and the equipment storage pool, which is also used to store the chimney partitions and the separator. The pool girders, which serve as barriers of the pools, rigidly connect the top slab and the RB walls. The RB floors that surround the containment walls and walls that are under the suppression pool floor slab are also integrated structurally with the concrete containment. The containment foundation mat is continuous with the RB foundation mat, and the FB as well. The containment and the structures integrated with the containment are constructed of cast-in-place, reinforced concrete.

The configuration of the containment is shown in Figure 3.8-1.. Additional peripheral volumes for anchoring of the containment reinforcements are considered within the code jurisdictional boundary and constructed in accordance with the rules of ASME Code Section III, Division 2. The boundaries of the additional peripheral volumes are determined based on the required development lengths of

containment reinforcements. *The key dimensions of the containment are summarized in Table 3.8-1.*

The containment foundation mat is a flat plate (see Table 3.8-13 and Figure 3.8-1.). 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 RPV pedestal reinforcement.

The containment wall and the RPV pedestal are right circular cylinders. 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 vicinity of the wall intersection with the foundation mat, the top slab and the suppression pool slab, around major piping penetrations, equipment hatches and personnel airlocks. Figure 3.8-2 shows a sketch of reinforcement in the reinforced concrete containment vessel (RCCV) wall around equipment hatches and personnel airlocks.

The containment top slab and the suppression pool slab are circular plates which have uniform thickness.

The reinforcement of the top slab and the suppression pool slab consist of top and bottom layers of main reinforcement and vertical tie bars for shear reinforcement. The top and bottom layers of main reinforcement are arranged in a rectangular grid in the top slab. The main reinforcement of the suppression pool slab is arranged in the radial and circumferential directions.

Regarding steel members such as structural steel shapes, piping supports or commodity supports attached to the exterior containment, Figure 3.8-4 provides a typical external containment plate support with embedment.

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 wetwell and Gravity-Driven Cooling System (GDCS) pools.

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 in Appendix 3G Subsection 3G.1.5.4. The liner plate is thickened locally and additional anchorage is provided at major structural attachments such as penetration sleeves, structural beam brackets, the vent wall, RPV support bracket and the Safety Relief Valve (SRV) quencher support connection to the suppression pool slab, and the diaphragm floor connection to the containment wall. Figure 3.8-5 shows the typical detail for the

quencher anchorage. The design forces of liner plates are obtained from the analysis directly, and the anchorage design is performed in accordance with ACI 349-01 Appendix B.

Regarding steel members such as structural steel shapes, piping supports or commodity supports inside containment, Figure 3.8-3 shows a typical support plate with anchors embedded in the concrete containment and integrally welded to the containment liner. The dimensions of the plate and the number of anchors depend on the loads for each support. They are designed in accordance with ANSI/AISC N690 and ACI 349 Appendix B.

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 wetwell and lower drywell is placed after the slab concrete is in place.

3.8.1.1.3 **Containment Boundary**

The jurisdictional boundary for application of Section III, Division 2 of the ASME Code to the concrete containment is shown in Figure 3.8-1. The boundary extends to the:

(1) Outside diameter of the RPV pedestal from the foundation mat to the suppression pool floor slab.

(2) Outside diameter of the containment wall from the suppression pool floor slab to containment top slab.

(3) Basemat circular plate under the RPV pedestal (the foundation basemat is a single basemat for the RB, the FB and the concrete containment).

(4) Suppression pool slab from the inside diameter of the RPV pedestal to the outside diameter of the containment wall.

(5) Containment top slab from the drywell head opening to the outside diameter of the containment.

The above are included in the ASME Code jurisdictional boundary for design, material, fabrication, inspection, testing, stamping, etc., requirements of the code. However, any other structural components which are integral with the containment structure are treated the same as the containment as far as loads and loading combinations are concerned in the design. Similarly, the RB 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 vent wall and diaphragm floor slab, which partition the containment into drywell and wetwell, are not part of the containment boundary. The vent wall and the diaphragm floor slab, steel structures filled with concrete, are designed according to codes given in Subsection 3.8.3.

Those portions of the structure outside the indicated Code jurisdictional boundary are designed, analyzed and constructed as indicated in Subsection 3.8.4. The analytical model includes the

containment, RB, FB and all the integrally connected structures and therefore includes continuity effects in the analysis.

3.8.1.2 Applicable Codes, Standards, and Specifications

The design, fabrication, construction, testing, and in-service inspection of the concrete 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, "Domestic Licensing of Production and Utilization Facilities."

3.8.1.2.2 Construction Codes of Practice

Table 3.8-9 Items 1 and 3.

3.8.1.2.3 General Design Criteria, Regulatory Guides, and Industry Standards

1. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants", Criteria 1, 2, 4, 16 and 50. Conformance is discussed in Section 3.1.
2. *Table 3.8-9 Items 29, 30, 31 and 33*
3. Industry Standards

Only nationally recognized industry standards such as those published by the ASTM and the ANSI as referenced by the Applicable Codes, Standards, and Regulations are used.

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 that vary in intensity and occurrence, such as forces exerted by the lateral pressure of soil. Live load for structures inside the containment is 9.6 kPa (200 psf) during outages and laydown operations. The loads are applied to the containment interior floors, except the suppression pool floor slab.
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 that 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 SRVs 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 (SIT) or Integrated Leak Rate Test (ILRT).

2. T_t — Thermal effects and loads during the SIT or ILRT.

3.8.1.3.3 **Severe Environmental Loads**

W —Loads indirectly transmitted 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 including pool sloshing loads.

2. W' —Loads indirectly transmitted by the tornado specified in Section 3.3.

3.8.1.3.5 **Abnormal Plant Loads**

1. R_a — Pipe reactions (including R_o) from thermal conditions generated by a LOCA.

2. T_a — Thermal effects (including T_o) and loads generated by a LOCA.

3. P_a — Design accident pressure load within the containment generated by a LOCA, based upon the calculated peak pressure with an appropriate margin.

4. Y — Local effects on the containment due to a LOCA. The local effects 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 Design Basis Accident (DBA). The time-dependent nature of the load and the ability of the containment to deform beyond yield is 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 is 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., plastic or elastic), together with the ability of the containment to deform beyond yield, is considered in establishing the structural capacity necessary to resist the impact.

5. 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.
6. 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.
7. PS — Pool swell bubble pressure on the suppression pool boundary due to a LOCA.
8. DET - Detonation Loading specifically on the ICS and PCCS Condensers, including drain piping and vent piping. This loading can occur sporadically after a LOCA during the first 72 hours.

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 3.8-2. Table 3.8-2 complies with Table CC-3230-1 of the ASME Code Section III Division 2 Subsection CC.

Loads and load combinations listed in Table 3.8-2 are used for the design of the steel liner and liner anchors, but the load factor for all loads in the load combinations is 1.0.

As for seismic loads, the maximum co-directional responses to each of the excitation components are combined by the SRSS method.

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 NASTRAN described in Appendix 3C. The containment, RB and FB layout utilizes an integrated structural system. The structure is idealized as a three-dimensional assemblage of beam elements, and isoparametric membrane-bending plate elements.

The finite element analysis (FEA) model of the containment, RB and FB includes the whole structure. The details of the global FEA model are described in Appendix 3G Subsection 3G.1.4.1.

The foundation soil is simulated by a set of horizontal and vertical springs. The soil spring constraints are calculated based on the properties of the soil spring used in the soil–structure interaction (SSI) analysis model, which is described in Appendix 3A. The constraints by soil surrounding the RB and FB are neglected in the FEA model.

3.8.1.4.1.1.1 **Nonaxisymmetrical Loads**

Nonaxisymmetrical loads imposed on the containment and its connected structures, each of which may bear different kinds of loadings, include the following as defined in Subsection 3.8.1.3:

1. Tornado wind (indirect).
2. Design wind (indirect).
3. SSE.
4. Local pipe rupture forces, including local compartmental pressures from ruptured pipes in compartments inside or outside the containment.
5. LOCA hydrodynamic pressures in the suppression pool.
6. SRV discharge in the suppression pool.
7. Loadings from embedded steel brackets in the wall and top slab.

The containment wall is shielded from the design wind and tornado by the RB, which completely encloses the structure. Forces from the design wind and tornado are transmitted directly to the containment wall through the RB connections.

The LOCA and SRV hydrodynamic pressures on the suppression pool boundaries as described in Appendix 3B are applied as equivalent static pressures equal to the dynamic peak value times dynamic load factor (DLF). The LOCA and SRV dynamic analyses are described in Appendix 3F.

3.8.1.4.1.1.2 **Axisymmetrical Loads**

Axisymmetrical loads imposed on the containment and its connected structures 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. Local dead and live loads from embedded brackets, treated as axisymmetrical loads for overall structural response.
6. Dead and live loads from internal structures imposed on the suppression pool slab.
7. Normal operating thermal gradients.
8. Abnormal plant thermal gradients.
9. Preoperational test pressure.
10. Abnormal plant pressure loads (including those from high energy line breaks).
11. Normal external pressure load.
12. SRV discharge to suppression pool.
13. LOCA hydrodynamic pressures in the suppression pool.

The LOCA and SRV hydrodynamic pressures on the suppression pool boundaries as described in Appendix 3B are applied as equivalent static pressures equal to the dynamic peak value times DLF. The LOCA and SRV dynamic analyses are described in Appendix 3F.

3.8.1.4.1.1.3 Major Penetrations

The major penetrations in the concrete containment include: (1) the drywell head, (2) the upper drywell equipment and personnel hatches, (3) the lower drywell equipment and personnel hatches, (4) the wetwell access hatch, and (5) the main steam and feedwater pipe penetrations. The global model includes all major penetrations. The state of stress and behavior of the containment around these openings is determined by the use of analytical numerical techniques. The penetrations are included in the global FEA model integrating the containment, RB and FB, described in Subsection 3.8.1.4.1.1.

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 is performed using as-built properties.

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 from the analysis done using the computer program NASTRAN, as described in Subsection 3.8.1.4.1.1. The global

analysis considers the major structural configurations, including RCCV with the internal steel components, the RB with floor connections to the RCCV, and the basemat, using plate element modeling and linear material assumptions. The selected sections from the global model used for the section sizing design calculations are described in Subsection 3G.1.5.4.

The SSDP-2D program module, described in Appendix 3C, is used to determine the extent of concrete cracking at these sections and the resulting concrete and rebar stresses. The SSDP-2D 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 SSDP-2D assume that the concrete is isotropic and linearly elastic but with zero tensile strength. The methods used in SSDP-2D can also account for the reduced thermal forces and moments due to concrete cracking when the option of thermal cracking is selected. However, the redistribution of section forces and moments that occurs due to concrete cracking under thermal loads is not calculated by the SSDP-2D procedure. To account for the concrete cracking effects and redistribution of forces and moments from thermal loads, the procedure described in Subsection 3.8.1.4.1.3 is used and the option of thermal cracking in SSDP-2D is not selected.

The input data for the SSDP-2D program consist of the membrane forces, shear forces and bending moments calculated by the NASTRAN linear analysis. The section forces and moments from thermal loads under LOCA are scaled according to the procedure in the next subsection before combination with the other load cases. The areas of the reinforcing steel in terms of steel area to concrete cross-section ratio are based on the design shown in Appendix 3G. The evaluation of containment structural adequacy is shown in Subsection 3.8.1.5.

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 anchor design considers deviations in geometry due to fabrication and erection tolerances; however, strains associated with construction-related liner deformations are excluded when calculating liner strains for the service and factored load combinations according to ASME Code Section III, Division 2, Subarticle CC-3720. 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.

3.8.1.4.1.3 **Concrete Cracking Considerations**

For thermal loads, the effects of concrete cracking must be considered in developing the internal forces and moments in the section. For these loads, concrete cracking relieves the thermal stress, as well as redistributes the internal forces and moments on the sections from those obtained from a linear analysis. For the LOCA thermal loads, a half-symmetric, 3D continuum element model is used to evaluate the redistribution of forces due to concrete cracking. This analysis is performed with the ABAQUS/ANACAP-U software, which is described in Appendix 3C. A linear analysis, using the solid element model, is first performed as a baseline analysis with benchmarking to the linear

plate element design model using NASTRAN. A nonlinear, concrete cracking analysis is then performed under the same thermal loading conditions. In each case, the section forces and moments are calculated from the section stresses.

For each section force component for each of the critical design-basis sections, the ratio of the section force from the cracking analysis to that of the linear analysis is computed for the critical time points following the LOCA. These “thermal ratios” are then used to multiply the section forces obtained from the linear design model for section internal force and moment due to LOCA thermal loads before combining with the other loads according to the load combination condition. In general, the thermal ratios are less than one where the thermal stresses from the linear analysis are high because of the relief and redistribution of stress as the concrete cracks. In some cases, the thermal ratio may be greater than one because of the redistribution of the section forces and moments due to concrete cracking. This typically occurs at sections where the thermal stresses from the linear analysis are low and a small increase in stress develops from redistribution in the non-linear analysis. The section forces and moments from the non-linear analysis can also be used directly.

3.8.1.4.1.4 **Corrosion Prevention**

Type 304L stainless steel or clad carbon steel plate is 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 the southern United States under normal operation. The following conditions can cause the pool temperature to rise above normal:

1. Reactor core isolation mode: pool temperature can rise 17°C (62°F) above normal for a total of 165 days during the 60-year lifetime.
2. Suppression pool cooling mode: pool temperature can rise 17°C (62°F) 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 (600°F) for 60 years is 0.12 mm (4 mils). The major concern has involved the air/water interface area where pitting is most likely to occur. The 0.12 mm (4 mils) corrosion allowance is a small fraction of the stainless steel thickness, which is a nominal 2.5 mm (98 mils) 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 is required, as well as the use of the Fuel

and Auxiliary Pool Cooling System (FAPCS) to maintain water quality standards. The FAPCS (Subsection 9.1.3) also acts to maintain purity levels.

The wetted surfaces and water-to-air interface area of the suppression pool are monitored for general corrosion and local pitting in accordance with the ASME B&PV Code, Section XI, Subsection IWE, by the In-service Inspection (ISI) Program described in Subsection 3.8.1.7.3.

3.8.1.4.2 Ultimate Capacity of the Containment

A deterministic analysis is performed to determine the ultimate capacity of the containment and the details are described in Appendix 19B. A probabilistic analysis for containment pressure fragility is also performed and the details are described in Appendix 19C.

3.8.1.5 Structural Acceptance Criteria

For evaluation of the adequacy of the concrete 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 ESBWR) are shown in Table 3.8-3.

The allowable tangential shear strength provided by orthogonal reinforcement without inclined reinforcement for concrete with 34.5 MPa (5000 psi) specified compressive strength is limited to 4.88 MPa (708 psi) for factored load combinations. Inclined reinforcement is not used to resist tangential shear in the ESBWR containment.

3.8.1.6 Material, Quality Control and Special Construction Techniques

Materials used in construction of the containment are in accordance with RG 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 (psi)
Top Slab	41.4 (6000)
Containment excluding Top Slab	34.5 (5000)
Foundation Mat	27.6 (4000)

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 alkalis calculated as sodium oxide plus 0.658 % 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.

   Type V cement is used, or other suitable means are employed, to prevent sulfate attack and concrete deterioration at locations where concrete is in contact with soils having more than 0.20% water soluble sulfate (as SO_4) of ground-water with a sulfate concentration exceeding 1500 ppm.

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 specified in ASME Code Section III, Division 2, Subarticle CC-2223.1. 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 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°C (70°F). Type D is used when average ambient air temperature for the daylight period is 21°C (70°F) and above. Pozzolans, if used, conform to Specification for Coal Fly Ash and Raw or Calcined Natural Pozzolans for Use in Concrete (ASTM C-618), except that the loss on ignition is 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 (Standard Practice for Selecting Proportions for Normal, Heavyweight, and Mass 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 Standard Specification for Deformed and Plain Carbon-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 used in construction of the containment are in accordance with the Article CC-2500 of ASME Code Section III, Division 2, and augmented by the requirements of RG 1.136.

The materials conform to the requirements of the Articles CC-2500 through CC-2700 ASME Code Section III, Division 2. The liner plate is of the following type and grade:

Carbon Steel:	*ASME SA-516 Gr.-70 or ASTM A-709 HPS 70W †*
Carbon Steel with Stainless Clad:	*ASME SA-264 (SA-516 Gr. -70 + SA-240 Type 304L or ASTM A-709 HPS 70W † + SA-240 Type 304L)*
Stainless Steel:	*ASME SA-240 Type 304L*

† ASME Code Case N-763

Dimensional tolerances for the erection of the liner plate and appurtenances are detailed in the construction specifications based on the structure geometry, liner stability, concrete strength and the construction methods to be used and ASME requirements. The liner plate anchorages are designed for the loads indicated in Subsection 3.8.1.3.

3.8.1.6.5 Quality Control

Quality Control (QC) 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 Liner and Appurtenances

Welding activities conform to the requirements of Section III of the ASME Code. The required nondestructive examinations (NDEs) and acceptance criteria are provided in Table 3.8-5.

3.8.1.7 Testing and In-service Inspection Requirements

3.8.1.7.1 Structural Integrity Pressure Test

A SIT of the containment structure is performed in accordance with Article CC-6000 of ASME Code Section III, Division 2 and RG 1.136, after completion of the containment construction. The design pressure is 310 kPaG (45 psig). The drywell and wetwell are tested simultaneously at a pressure of 356.8 kPaG (51.8 psig). This is 115% of the design pressure. Next a differential pressure test of 277.5 kPad (40 psid) is conducted between the drywell and the wetwell. The drywell pressure is greater than the wetwell pressure during the differential pressure test. This test differential pressure is 115% of the design-differential pressure. At no time during the SIT the maximum drywell pressure of 356.8 kPaG (51.8 psig) is exceeded.

During these tests, the wetwell, GDCS pools, IC/PCCS pools (including expansion pools), reactor well, Equipment Storage pool, and Fuel Buffer 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-6370 of ASME Code Section III, Division 2.

3.8.1.7.2 Preoperational and In-Service Integrated Leak Rate Test

Preoperational and in-service integrated leak rate testing is discussed in Subsection 6.2.6.

3.8.1.7.3 Preservice and In-service Inspection

3.8.1.7.3.1 Scope

This subsection describes the preservice and ISI Program requirements for the Containment Structure, ASME B&PV Code, Class CC and MC pressure retaining components and their integral attachments. It describes those programs implementing the requirements of the ASME B&PV Code Section XI (ASME Section XI). Subsection IWE of ASME Section XI applies to Class MC and metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments. Subsection IWL of ASME Section XI applies to the Class CC reinforced concrete.

The design to perform preservice inspection is in compliance with the requirements of the ASME Section XI, 2001 Edition with 2003 Addenda. The preservice and inservice inspection program plans are based on the ASME Section XI, Edition and Addenda specified in accordance with 10 CFR 50, Section 50.55a. The Containment Structure is designed to provide access for the examinations required by ASME Section XI, IWE-2500 and IWL-2500. The actual Edition of ASME Section XI to be used is specified based on the procurement date of the component per 10 CFR 50, Section 50.55a. The ASME Code requirements discussed in this section are provided for information and are based on the 2001 Edition of ASME Section XI with 2003 Addenda.

3.8.1.7.3.2 Exclusions

During the detailed design phase, the goal is to minimize the number of inaccessible areas in order to reduce the number of exclusion areas. Furthermore, remote tooling is used in high radiation areas where feasible.

Portions of the Containment Structure are excluded from preservice and inservice examination requirements of ASME Section XI, Subsections IWE and IWL as follows:

1. For Class MC components and metallic shell and penetration liners of Class CC components and their integral attachments:
    a. Vessels, parts, and appurtenances outside the boundaries of the containment system as defined in the Design Specifications.
    b. Embedded or inaccessible portions of containment vessels parts, and appurtenances that meet the requirements of the Edition and Addenda of ASME Section III used for construction.
    c. Portions of containment vessels, parts and appurtenances that become embedded or inaccessible as a result of vessel repair/replacement activities if the prerequisites for exemption of inaccessible surface areas under ASME Section XI, IWE-1232 and IWE-5220 are satisfied.
    d. Piping, pumps, and valves that are part of the containment system, or which penetrate or are attached to the containment vessel. These components are examined in

accordance with the ASME Section XI requirements, i.e., Subsection IWB or IWC, applicable to their classification as defined in the Design Specification.

2. For Class CC reinforced concrete, those portions of the concrete surface that are covered by the liner, foundation material, or backfill, or are otherwise obstructed by adjacent structures, components, parts, or appurtenances.

3.8.1.7.3.3 Preservice Examination

The preservice examinations are performed prior to plant startup but after the Structural Integrity Pressure Test. Visual examinations are performed after the application of any required protective coatings. The preservice examinations include those examinations listed in ASME Section XI, Table IWE-2500-1, IWL-2510 and Table IWL-2500-1.

3.8.1.7.3.4 In-service Inspection Schedule

The inservice inspection interval for Class MC components and metallic shell and penetration liners of Class CC components and their supports conform to Inspection Program B as described in ASME Section XI, IWE-2412. Except where deferral is permitted by ASME Section XI, IWE-2500-1, the percentages of examinations completed within each period of the interval are to correspond to Table IWE-2412-1. The diaphragm floor and vent wall receive a visual, VT-3, examination once during each inspection interval.

The in-service inspection of Class CC reinforced concrete are performed at 1, 3, and 5 years after the completion of the Structural Integrity Pressure Test and every 5 years thereafter in accordance with ASME Section XI, IWL-2410 and Table IWL-2500-1.

3.8.1.7.3.5 Pressure Tests

The pressure testing (leakage testing) of the Containment Structure are conducted in accordance with 10 CFR 50, Appendix J. In addition, the leakage test requirements of ASME Section XI, IWE-5000 and IWL-5000 are applied following repair/replacement activities as defined by the ASME Code.

3.8.1.7.3.6 Qualification of Examination Personnel

Personnel performing preservice and inservice examinations of the containment system are qualified in accordance with the applicable requirements of the ASME Section XI. Personnel performing visual examination types VT-1, VT-3, and ultrasonic examination are qualified in accordance with Section XI, IWA-2300. Personnel performing detailed visual examination and general visual examination of concrete are qualified in accordance with IWA-2300 to perform examinations as described in IWL-2300.

3.8.1.7.3.7 **Visual Examination Methodology**

Visual examination types VT-1 and VT-3 are conducted in accordance with ASME Section XI, IWA-2210. When performing examinations remotely, the requirements of Table IWA-2210-1 are modified in order to extend maximum specified direct examination distance and decrease the minimum illumination, provided that the conditions or indications for which the examination is being conducted can be detected at the chosen distance and illumination.

3.8.1.7.3.8 **Visual Examination of Surfaces**

The type VT-1 examination is used to conduct the detailed examination required for visible containment surfaces requiring augmented examination in accordance with ASME Section XI, Table IWE-2500-1, Examination Category E-C, Item E4.11. The type VT-3 examination is used to conduct the general visual examinations required for wetted surfaces of submerged areas and accessible surfaces of BWR ventilation systems as required by Table IWE-2500-1, Examination Category E-A, Items E1.12 and E1.20, respectively. Other surfaces are examined as specified by ASME Section XI, Tables IWE-2500-1 or IWL-2500-1, as applicable.

3.8.1.7.3.9 **Visual Examination of Bolted Connections**

The type VT-3 examination is used to conduct the general visual examination of pressure retaining bolted connections that are part of the accessible surface areas identified by ASME Section XI, Table IWE-2500-1, Examination Category E-A, Item E1.11. That VT-3 examination is conducted at least once during each inspection interval as defined by IWE-2412. The bolting is disassembled to perform the VT-3 examination; however, as an alternative to a rigid inspection schedule, the VT-3 visual examination is performed whenever the bolting is disassembled for any reason. Where flaws or degradation are identified during a VT-3 examination, a type VT-1 examination is performed.

3.8.1.7.3.10 **Ultrasonic Examination**

The ultrasonic thickness measurements used for surfaces requiring augmented examination in accordance with ASME Section XI, Table IWE-2500-1, Examination Category E-C, Item E4.12, are conducted using a technique demonstrated on a calibration standard. Methods such as those described in ASTM E 797, Standard Practice for Measuring Thickness by Manual Ultrasonic Pulse-Echo Contact Method, are acceptable. The ultrasonic thickness measurements are performed for both Class MC components and metallic shell and penetration liners of Class CC components if augmented examination is necessary under the provisions of ASME Section XI, IWE-1240.

3.8.1.7.3.11 **Acceptance Criteria**

The acceptance standards of the material specification or IWB-3517.1 are used for the evaluation of bolting. For other preservice and inservice examinations, the requirements of IWE-3000 for ASME Class MC components and metallic liners or IWL-3000 for ASME Class CC components are

used for evaluation. The ultrasonic acceptance standard of IWE-3511.3 for ASME Class MC components is applied to metallic liners of Class CC components.

3.8.1.7.3.12 Evaluation of Inaccessible Areas

During operation, areas inaccessible for examination for acceptability are evaluated if conditions exist in accessible areas that indicate the presence of or result in the degradation of the inaccessible areas. For each such area identified, the following information is included in the In-service Inspection Summary report required by ASME Section XI, IWA-6000:

(1) A description of the type and estimated extent of degradation, and the conditions that led to the degradation.

(2) An evaluation of each area and the result of the evaluation.

(3) A description of necessary corrective actions.

3.8.2 Steel Components of the Reinforced Concrete Containment

3.8.2.1 Description of the Steel Containment Components

The ESBWR has a RCCV as described in Subsection 3.8.1. This section describes the following steel components of the concrete containment vessel:

1. Personnel Air Locks
2. Equipment Hatches
3. Penetrations
4. Drywell Head
5. Passive Containment Cooling System (PCCS) Condenser

3.8.2.1.1 Personnel Air Locks

Two personnel air locks with an inside diameter sufficient to provide 1850 mm (6 ft. 13/16 in.) high by 750 mm (2 ft. 5-1/2 in.) 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.

Lock and swing of the doors is by manual and automatic means. The locks extend radially outward from the RCCV into the RB and are supported by the RCCV only. The minimum clear horizontal distance not impaired by the door swing is 1850 mm (6 ft. 13/16 in.).

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.

3.8.2.1.2 **Equipment Hatch**

Three equipment hatches are provided. One of these serves the upper drywell and another serves the lower drywell. The third equipment hatch provides personnel and equipment access to the wetwell 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 station, 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.

3.8.2.1.3 **Penetrations**

In addition to the personnel airlocks, equipment hatches and drywell head, other steel components of the concrete containment vessel include piping and electrical penetrations. The major piping penetrations are associated with main steam and feedwater lines. Electrical penetrations are described in Subsection 8.3.1. A summary of various containment penetrations is given in Section 6.2. 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 FEA with boundaries extending to a region where the discontinuity effects of the opening are negligible.

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. With regard to the local areas of concrete around high energy penetrations, thermal analyses have been carried out to demonstrate that concrete temperature limits in ASME Section III, Division 2, CC-3440 are satisfied. In all cases the concrete temperature is lower than 93°C (200°F) for normal operation,

and lower than 177°C (350°F) for accident condition. The sleeve length for hot penetrations is designed to meet these temperature requirements.

Figures 3.8-6, 3.8-7, 3.8-8, 3.8-9, 3.8-10 and 3.8-11 show the typical details for the containment mechanical and electrical penetrations.

3.8.2.1.4 **Drywell Head**

A 10,400 mm (34 ft. 1-7/16 in.) 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. This structure is shown in Appendix 3G Figure 3G.1-51. The drywell head is designed for removal during reactor refueling and for replacement prior to reactor operation using the RB 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.

There is water in the reactor well above the drywell head during normal operation. The height of water is found in Table 3G.1-4. The stainless steel clad thickness for the drywell head is 2.5 mm (98 mils) and is determined in accordance with NB-3122.3 requirements so that it results in negligible change to the stress in the base metal.

There are six support brackets attached to the inner surface of the drywell head circumferentially to support the head on the operating floor during refueling. These support brackets have no stiffening effect and do not resist loads when the head is in the installed configuration.

To provide a leak resistant refueling seal, a structural seal plate with an attached compressible-bellows sealing mechanism between the Reactor Vessel and Upper Drywell opening is utilized. The Refueling Seal is a continuous gusseted radial plate that is anchored to the Drywell opening in the Top floor slab. The radial plate surrounds the RPV with a radial gap opening to allow for thermal radial expansion of the RPV. A circumferential radial bracket from the RPV connects to a circumferential bellows that is also connected to the underside of the Drywell opening plate, thus providing a refueling seal, and allowing for axial thermal expansion of the RPV.

3.8.2.1.5 **PCCS Condenser**

There are six PCCS Condensers located in the PCCS subcompartment pools. The condensers form an integral part of the containment boundary while the pool structure and pool water are outside containment. The PCCS Condensers are described in Subsection 6.2.2, and their structural evaluation is documented in Appendix 3G.

3.8.2.2 **Applicable Codes, Standards, Specifications and Regulatory Guides**

3.8.2.2.1 **Codes, Standards and Regulatory Guides**

In addition to the documents specified in Subsection 3.8.1.2.2, the following code, standard and regulatory guide apply:

(1) *ASME B&PV Code, Section III, Division 1, Nuclear Power Plant Components, Subsection NE, Class MC and Code Case N-284.*

(2) *ANSI/AISC N690-1994s2 (2004) Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities.*

(3) *Regulatory Guide 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.*

3.8.2.2.2 Code Classification

The steel components of the RCCV are classified as Class MC in accordance with Subarticle NCA-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 NCA-4000 and NCA-5000 of ASME Code Section III. Structural steel attachments beyond the boundaries established for the steel components of the RCCV are designed, fabricated, and constructed according to the AISC N690-94, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities."

3.8.2.3 Loads and Load Combinations

The applicable loads are described in Subsection 3.8.1.3 and load combinations are shown in Table 3.8-4.

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 (Design by Formula) of ASME Code Section III. For the configurations and loadings that are not explicitly treated in Subarticle NE-3130, the design is in accordance with the applicable Subarticles designated in paragraphs (b) and (d) of Subarticle NE-3130 of ASME Code Section III.

The design of nonpressure-resisting parts is performed in accordance with the general practices of the AISC N690-94, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities."

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 is analyzed using a finite element computer program and/or manual calculation based on handbook formulas and tables. 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, NE-3200 and NE-3300 of ASME Code Section III, Division 1.

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 and/or manual calculation is 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 Sub-articles NE-3130, NE-3200 and NE-3300 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 **Other 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 required analysis and associated stress intensity limits are in accordance with Subarticle NE-3200 of ASME Code Section III, Division 1, including fatigue evaluation as required.

Main Steam and Feedwater penetrations are analyzed using the finite element method of analysis for applicable loads and load combinations. The resulting stresses meet the acceptance criteria stipulated in Subarticle NE-3200 of ASME Code Section III, Division 1, including fatigue evaluation as required.

3.8.2.4.1.4 **Drywell Head**

The drywell head, consisting of shell, flanged closure and drywell-head anchor system, is analyzed using a finite-element stress analysis computer program or manual calculation. 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, NE-3200 and NE-3300 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 compressive stress values in accordance with Subarticle NE-3222 of ASME Code Section III, Division 1, or Code Case N-284.

3.8.2.4.1.5 **PCCS Condenser**

The PCCS condensers are composed of two modules consisting of drum-and-tube type heat exchangers using horizontal upper and lower drums connected with multiple vertical tubes. Two identical modules are coupled to form one PCCS heat exchanger unit. The condenser assembly forms an integral part of the containment boundary and is submerged in the water of an IC/PCCS pool subcompartment. The pool water lies outside the containment boundary. Three sleeves containing the feed line, return line and drain lines pass through the RCCV Top Slab. The condenser, the lines connected to the condenser, and the sleeves are part of the containment boundary. Figure 3.8-7 shows the typical configuration for these passages through the RCCV Top Slab and Table 3.8-17 lists each of these passages and their function.

The PCCS condenser is anchored to the RCCV Top Slab and is laterally supported by a 3D steel frame structure that transmits the horizontal dynamic forces to the RCCV Top Slab.

The PCCS condenser is subjected to various combinations of piping reactions, mechanical, thermal, detonation pressure, and seismic loads including sloshing. The resulting forces due to various load combinations are combined with the effects of differential pressures.

A finite-element analysis model supplemented with hand calculation is used to determine the stresses in the different components of the PCCS condenser and supports. Details of this analysis, including relevant drawings and results, can be found in Reference 3.8-1. The PCCS condenser parts conform to the design requirements of Subarticles NE-3200 and NE-3300 of ASME Code, Section III, Subsection NE (Class MC). The PCCS condenser support is evaluated in accordance with the ASME Code, Section III, Subsection NF.

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 maximum 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", "Abnormal/Severe Environmental" and "Abnormal/Extreme Environmental" in Table 3.8-2 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 Level A, B, C and D conditions are summarized in Table 3.8-4.

Stability against compression buckling is assured by an adequate factor of safety.

The allowable stress limits used in the design and analysis of non-pressure-resisting components are in accordance with Subsection 3.8.2.2.1 Item (2).

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The steel pressure retaining components of the RCCV meet the requirements of Article NE-2120 of ASME Section III. The principal materials for the RCCV locks, hatches, penetrations, drywell head, and PCCS condensers are as follows:

- *Plate (SA-240 type 304L, SA-516 grade 60 or 70)*
- *Pipe (seamless SA-333 grade 1 or 6; or SA-106 grade B or SA-312 type 304L or Welded SA-671 Gr CC70)*
- *Forgings (SA-182 grade FXM-19, SA-336 F316)*
- *General Tubing (SA-213 grade TP304L)*
- *PCCS Condenser Tubing (SA-312 grade XM-19)*
- *Bolting (SA-193-B8, SA-437 Gr B4B bolts or SA-564 Gr 630. Nuts shall conform to SA-194 or to the requirements for nuts in the specification for the bolting material to be used.)*
- *Clad (SA-240 type 304L)*

3.8.2.7 Testing and In-service Inspection Requirements

Testing and In-service Inspection Requirements 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 conform to requirements of Section III of the ASME Code. The required NDE and acceptance criteria are provided in Table 3.8-5.

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 structural steel. The containment internal structures include the following:

- Diaphragm floor
- Vent wall
- GDCS pool walls
- Reactor shield wall (RSW)
- RPV support brackets
- Miscellaneous platforms

The containment internal structures consist of the diaphragm floor slab, vent wall, GDCS pool walls, RSW, and the RPV support bracket. These structures are shown in the general arrangement drawings in Appendix 3G Subsection 3G.1.

The diaphragm floor slab acts as a barrier between the drywell and the wetwell. The diaphragm floor slab is supported on the reinforced concrete containment wall at its outer periphery and on the vent wall at its inner periphery. The diaphragm floor slab is a structural steel design. The space between the floor slab top and bottom plates is filled with concrete. The slab is supported by a system of radial beams spaced evenly all around and spanning between the vent wall structure and the reinforced concrete containment wall.

The vent wall structure is also a structural steel design consisting of two concentric carbon steel cylinders connected together by vertical web plates evenly all around. The vent wall structure is anchored at the bottom into the RPV pedestal and is restrained at the top by the diaphragm floor slab. The cylindrical annulus carries 12 vent pipes and 12 SRV downcomer pipes with sleeves,

from the drywell into the suppression pool. The space in the cylindrical annulus is filled with concrete.

There are three GDCS pools supported on top of the diaphragm floor slab. The pools on one side are contained by the reinforced concrete containment wall and on the other side by structural steel walls.

The RSW is a thick steel cylindrical structure that surrounds the RPV. It is supported by the RPV support brackets. The function of the RSW is to attenuate radiation emanating from the RPV. In addition, the RSW provides structural support for the RPV stabilizer and the RPV insulation. Openings are provided in the RSW to permit the routing of necessary piping to the RPV and to permit in-service inspection of the RPV and piping.

Appendix 3G Appendix 3G.1 contains the detail design and analysis information for these internal structures.

3.8.3.1.1 **Diaphragm Floor**

The diaphragm floor serves as a barrier between the drywell and the wetwell. It is a concrete-fill steel slab having steel plates at the top and bottom surfaces, with an outside diameter of 18.0 m (59 ft. 5/8 in.), and a thickness of 0.6 m (23-5/8 in.).

The diaphragm floor is supported by the vent wall and the containment wall. The connection of the diaphragm floor to the containment wall is a fixed support.

Carbon steel plates, 25 mm (1 in.) thick, are provided on the top and bottom of the diaphragm floor. The plates prevent bypass flow of steam from the upper drywell to the wetwell air space during a LOCA.

3.8.3.1.2 **RPV Support Bracket**

The eight (8) RPV support brackets are located at the junction of RPV pedestal and vent wall structure. These brackets are made of structural steel and they provide structural support to the RPV as well as the RSW. See Appendix 3G Subsection 3G.1.5.4.2.4.

3.8.3.1.3 **Reactor Shield Wall**

The RSW is supported by the RPV support bracket and surrounds the RPV. Its function is to attenuate radiation emanating from the Reactor Vessel. In addition, the RSW provides structural support for the Reactor Vessel stabilizer, the reactor vessel insulation, some of the drywell equipment, in-service inspection catwalks 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 made of structural steel and is shaped as a right cylinder. The plate thickness varies along the elevation and is 160 mm (6-5/16 in.), 210 mm (8-1/4 in.), and 260 mm (10-1/4 in.) and the inside of the wall is 4.646 m (15 ft. 2-7/8 in.) radius.

3.8.3.1.4 **Vent Wall**

The vent wall structure is made up of two concentric carbon steel cylindrical plates connected together by vertical web plates at 15 degrees on centers. The cylindrical structure has an inner and outer diameter of 13.2 m (43 ft. 3-11/16 in.) and 16.7 m (54 ft. 9-1/2 in.) respectively with overall height of 12.85 m (42 ft. 1-7/8 in.). The vent wall structure is anchored at the bottom into the RPV pedestal and is restrained at the top by the diaphragm floor at elevation 17500.

The cylindrical annulus carries twelve 1.20 m (3 ft. 11-1/4 in.) O.D. vent pipes and twelve SRV discharge pipes with sleeves, from the drywell into the suppression pool; and three lines of the drywell cooling system. The space in the cylindrical annulus is filled with concrete. The wetted surface of the outer cylinder is covered with stainless steel cladding to prevent corrosion.

3.8.3.1.5 **Gravity Driven Cooling System Pool**

There are three GDCS pools supported on top of the diaphragm floor.

The pools on one side are contained by the RCCV wall and on the other side by walls made of structural steel.

The GDCS pool walls away from the RCCV are made of carbon steel plates lined with stainless steel cladding and backed up with vertical and horizontal steel structural framing system.

3.8.3.1.6 **Miscellaneous Platforms**

Miscellaneous platforms are designed to allow access and to provide support for equipment and piping. The platforms consist of steel beams and open grating to facilitate movement of air and liquids in case of pipe breaks. Platforms are classified as Seismic Category I structures when they support safety-related functions. Otherwise they are classified as Seismic Category II. Similarly, other miscellaneous structural components inside containment that do not support safety-related functions are classified as Seismic Category II.

3.8.3.1.7 **Miscellaneous Commodities**

See Subsection 3.8.4.1.6 for Cable trays, Conduits, and their supports. See Subsection 3.8.4.1.7 for Heating, Ventilation and Air Conditioning (HVAC) ducts and their supports.

3.8.3.2 Applicable Codes, Standards, and Specifications

The design and construction of the concrete and steel internal structures of the containment conform to the applicable codes, standards, specifications, and regulations listed in Table 3.8-6 except where specifically stated otherwise.

Structure or Component	Specific Reference Number in Table 3.8-6
Diaphragm Floor	1-12, 15-20
RPV Support Bracket	15-20
Vent wall	1-12, 15-20
Reactor Shield Wall	15-20
GDCS Pool Wall	15-20
Miscellaneous Platforms	15-20

Anchorage of steel internal structures complies with RG 1.199.

3.8.3.3 Loads and Load Combinations

3.8.3.3.1 Load Definitions

The loads and applicable load combinations for which a containment internal 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 RSW is also designed to the Annulus Pressurization loads, which are loads and pressures directly on the RSW caused by a rupture of a pipe within the reactor vessel shield wall annulus region.

3.8.3.3.2 Load Combination

The load combinations and associated acceptance criteria for steel internal structures of the containment are listed in Table 3.8-7.

3.8.3.4 Design and Analysis Procedures

The design of steel internal structures is performed in accordance with the general practice of the ANSI/AISC N690. See Table 3.8-7 for more details. The effects of concrete cracking of the containment structure on the accidental thermal stresses in the containment internal structures are accounted for in the form of thermal ratios as described in Subsection 3.8.1.4.1.3.

See Subsection 3.8.3.7 for accessibility to equipment, valves, instrumentation, welds, supports, etc. for operation, inspection or removal.

3.8.3.4.1 **Diaphragm Floor**

The diaphragm 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 top plate, bottom plate and support beams. The radial support beams are welded to the diaphragm floor, so they form an integral structure.

3.8.3.4.2 **Reactor Pressure Vessel Support Bracket**

The RPV support bracket 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 from RPV support and RSW are resisted by the integral action of eight separate brackets equally spaced circumferentially. The RPV support is described in Subsection 5.3.3.2.2.

3.8.3.4.3 **Reactor Shield Wall**

The RSW 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 including those from the RPV stabilizer are resisted by the thick steel cylinder supported by the RPV support bracket.

3.8.3.4.4 **Vent Wall**

The vent wall 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 cylinders with connecting ribs.

3.8.3.4.5 **Gravity Driven Cooling System Pool**

The GDCS pool wall 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 wall plate and support beams.

3.8.3.4.6 **Miscellaneous Platforms**

The miscellaneous platforms are considered as additional mass in the finite-element model described in Subsection 3.8.1.4.1.1. The platform design is based on the elastic method.

3.8.3.5 **Structural Acceptance Criteria**

3.8.3.5.1 **Diaphragm Floor**

The structural acceptance criteria for the diaphragm floor are in accordance with ANSI/AISC N690. See Table 3.8-7 for more details.

3.8.3.5.2 Reactor Pressure Vessel Support Bracket

The structural acceptance criteria for the RPV support bracket are in accordance with ANSI/AISC N690. See Table 3.8-7 for more details.

3.8.3.5.3 Reactor Shield Wall

The structural acceptance criteria for the RSW are in accordance with ANSI/AISC N690. See Table 3.8-7 for more details.

3.8.3.5.4 Vent Wall

The structural acceptance criteria for the vent wall are in accordance with ANSI/AISC N690. See Table 3.8-7 for more details.

3.8.3.5.5 Gravity Driven Cooling System Pool

The structural acceptance criteria for the GDCS pool are in accordance with ANSI/AISC N690. See *Table 3.8-7* for more details.

3.8.3.5.6 Miscellaneous Platforms

The structural acceptance criteria for safety-related platforms are in accordance with ANSI/AISC N690. See Table 3.8-7 *for more details. The same criteria are used for nonsafety-related platforms for design purposes only.*

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Diaphragm Floor

The materials conform to all applicable requirements of ANSI/AISC N690 and ACI 349 and comply with the following:

Item	***Specification***
Top and bottom plate	*ASTM A-709 HPS 70W †*
Support beam	*ASTM A-709 HPS 70W †*
Internal stiffeners	*ASTM A-36 or A-709 HPS 70W †*
Concrete fill	*f ’c = 34.5 MPa (5000 psi)*
Stainless cladding for wetted surface of top plate	*ASTM A-240 Type 304L*

† ASME Code Case N-763

Different material choices are available from the specifications listed above.

3.8.3.6.2 Reactor Pressure Vessel Support Bracket

The steel plate materials conform to all applicable requirements of ANSI/AISC N690 and comply with ASTM A-516 Gr. 70 or A-709 HPS 70W (ASME Code Case N-763). Materials are chosen depending on the thickness of each part.

3.8.3.6.3 Reactor Shield Wall

The materials conform to all applicable requirements of ANSI/AISC N690 and comply with the following:

Materials are chosen depending on the thickness of each part.

Item	***Specification***
Cylinder Plate	*ASTM A-516 Gr. 70 or ASTM A-668 Gr. F or Gr. G or A-709 HPS 70W (ASME Code Case N-763)*

Different material choices are available from the specification listed above.

3.8.3.6.4 Vent Wall

The materials conform to all applicable requirements of ANSI/AISC N690 and ACI 349 and comply with the following:

Item	***Specification***
Inner and outer cylinders (excluding the portions submerged in the suppression pool)	*ASTM A-709 HPS 70W †*
Internal stiffeners	*ASTM A-36 or A-709 HPS 70W †*
Concrete fill	*f 'c= 34.5 MPa (5000 psi)*
Outer shell submerged in the suppression pool	*ASTM A-709 HPS 70W † with A-240 Type 304L clad*
Vent Pipe	*ASTM A-240 Type 304L*

† ASME Code Case N-763

Different material choices are available from the specifications listed above.

3.8.3.6.5 Gravity Driven Cooling System Pool

The materials conform to all applicable requirements of ANSI/AISC N690 and comply with the following:

Item	*Specification*
Pool wall plate	*ASTM A-709 HPS 70W † with A-240 Type 304L Clad*
Structural support beam	*ASTM A-709 HPS 70W †, ASTM A-709 HPS 70W † with A-240 Type 304L Clad*
Stiffeners	*ASTM A-36*

† ASME Code Case N-763

3.8.3.6.6 Miscellaneous Platforms

The materials conform to all applicable requirements of ANSI/AISC N690 for safety-related and ANSI/AISC 360 for nonsafety-related and comply with the following:

Item	*Specification*
Structural steel and connections	*ASTM A-36, ASTM A-992 Wide Flanges, A-500 Gr B-Tube Steel*
High strength structural steel plates	*ASTM A-572, Gr. 50 or Gr. 65 (ASME Code Case N-632)*
High strength bolting material [dia. ≥ 19 mm (3/4 in)]	*ASTM A-325*
Carbon steel bolting and threaded rod material [dia. < 19 mm (3/4 in)]	*ASTM A-307*

3.8.3.7 Testing and In-service Inspection Requirements

A formal program of testing and in-service inspection is not planned for the internal structures except the diaphragm floor and vent wall. The other internal structures are not directly related to the functioning of the containment system; therefore, no testing or inspection is performed.

However, during the operating life of the plant the condition of these other internal structures is monitored per 10 CFR 50.65 in accordance with Section 1.5 of RG 1.160.

Testing and in-service inspection of the diaphragm floor and vent wall are directly related to the functioning of the containment system and are discussed in Subsection 3.8.1.7.

Space Control is exercised in the ESBWR by means of a 3D model. It is the means by which interference checking and space control is accomplished. It includes all safety-related and nonsafety-related SSCs. Items are added to the model as it is being developed by stages depending on criticality to the plant and construction sequence of the item. Accessibility to equipment, valves, instrumentation, welds, supports, etc. for operation, inspection or removal is characterized by sufficient space to allow unobstructed access and reach of site personnel. Therefore, aisles, platforms, ladders, handrails, etc. are reviewed as the components are laid out.

Interferences with access ways, doorways, walkways, truck ways, lifting wells, etc. are constantly monitored.

This method of configuration control is maintained and documented during the plant layout process. Remote tooling is considered only if for some layout reasons the required inspection could not be carried out otherwise.

3.8.3.8 Welding Methods and Acceptance Criteria for Structural and Building Steel

Welding activities are performed with written procedures, and with the requirements of the AISC Manual of Steel Construction. The visual acceptance criteria comply with AWS Structural Welding Code D1.1 and Nuclear Construction Issues Group Standard, “Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants”, NCIG-01 (also known as EPRI NP-5380).

3.8.4 Other Seismic Category I Structures

Other Seismic Category I structures which are not inside the containment and which constitute the ESBWR Standard Plant are RB, CB, FB and FWSC. Figure 1.1-1 shows the spatial relationship of these buildings. Although the RW that houses non safety-related facilities is not a Seismic Category I structure, it is designed to meet requirements as defined in RG 1.143 under Safety Class RW-IIa. The seismic design of the Radwaste Building is full SSE instead of 1/2 SSE as shown in Table 2 of RG 1.143. The RB and FB are built on a common foundation mat and structurally integrated into one building. The FWSC consists of Firewater Storage Tank (FWS) and Fire Pump Enclosure (FPE) structures that share a common basemat. The other structures in close proximity to these structures are the Turbine Building (TB) and Service Building. The Ancillary Diesel Building is located at least 15.2 m (49 ft. 10-3/8 in.) from the Fuel Building. They are structurally separated from the other ESBWR Standard Plant buildings. *Seismic gaps are provided with no less than the calculated maximum relative displacement during SSE event, considering out-of-phase motion between independent Nuclear Island buildings to eliminate seismic interaction.*

Among the Seismic Category I structures within the ESBWR Standard Plant, other than the containment structure, only the RB contains certain rooms that have high-energy pipes, and therefore these rooms are more structurally demanding. The main steam tunnel walls protect the RB from potential impact by rupture of the high-energy main steam pipes that extend to the TB. Thus the RB walls of the main steam tunnel are designed to accommodate the pipe support forces and the environmental conditions during and after the postulated high-energy pipe break. Longitudinal pipe breaks required by BTP 3-4 of SRP 3.6.2 are postulated inside the main steam tunnel and cause a slight pressurization that is used for environmental qualification. See Subsection 6.2.3.2 for the main steam tunnel functional design.

The ESBWR Standard Plant does not contain underground Seismic Category I pipelines that are directly buried in the ground (i.e. all are contained in concrete trenches/tunnels or concrete duct bank) or masonry wall construction.

Removable shield blocks consisting of metallic forms filled with grout or concrete designed to Seismic Category II requirements are used. The shield blocks are provided with removable structural steel frame also designed to Seismic Category II requirements to prevent the shielding blocks from sliding or tipping under seismic events.

3.8.4.1 Description of the Structures

3.8.4.1.1 Reactor Building Structure

Key dimensions of the RB are summarized in Table 3.8-8.

The RB encloses the concrete containment and its internal systems, structures, and components. Located above the concrete containment in the RB are the IC/PCCS pools (including expansion pools), the buffer pool, which is also used to store the dryer, and the equipment storage pool, which is also used to store the chimney partitions and the separator. Main Steam and Feedwater lines are routed to the TB through the Main Steam Tunnel in the RB as described in Subsection 3.8.4. The RB is a Seismic Category I structure.

The RB is a rigid box type shear wall building constructed of reinforced concrete. Vertical loads are carried by a system of external walls box-shaped surrounding a large cylindrical shaped concrete containment. Lateral loads are resisted by external shear walls as well as the internal concentric cylindrical structure.

These structures are tied together by a system of internal concrete bearing walls and concrete floor slabs.

The load resisting characteristic of the building is that of a concrete box type shear wall structure.

The summary report for the RB is in Appendix 3G Section 3G.1. This report contains a description of the RB, the loads, load combinations, reinforcement stresses, and concrete reinforcement details for the basemat, seismic walls, and floors.

3.8.4.1.2 Control Building

The CB is adjacent to but structurally independent of the RB (see Figures 1.2-2 through 1.2-5 and 1.2-11). The key dimensions of the CB are summarized in Table 3.8-8.

The CB houses the safety-related electrical, control and instrumentation equipment, the control room for the RB and TB and the CBVS equipment. The CB is a Seismic Category I structure that houses control equipment and operation personnel.

The CB is a reinforced concrete box type shear wall structure consisting of walls and slabs and is supported on a foundation mat. Steel framing is composite with concrete slab and is used to support the slabs for vertical loads. The CB is a shear wall structure designed to accommodate all seismic loads with its walls and connected floors. Therefore, frame members such as beams or columns are designed to resist vertical loads and to accommodate deformations of the walls in case of earthquake conditions.

The summary report for the CB is in Appendix 3G Section 3G.2. This report contains a description of the CB, the loads, load combinations, reinforcement stresses, and concrete reinforcement details for the basemat, seismic walls, and floors.

3.8.4.1.3 **Fuel Building**

The FB is integrated with the RB, sharing a common wall between the RB and the FB and a large common foundation mat (see Section 1.2). The key dimensions of the FB are summarized in Table 3.8-8.

The FB houses the spent fuel pool facilities and their supporting system and HVAC equipment. The FB is a Seismic Category I structure except for the penthouse that houses HVAC equipment. The penthouse is a Seismic Category II structure.

The FB is a reinforced concrete box type shear wall structure consisting of walls and slabs and is supported on a foundation mat. Concrete or steel framing is composite with a concrete slab and is used to support the slabs for vertical loads. The FB is a shear wall structure designed to accommodate all seismic loads with its walls and connected floors. Therefore, frame members such as beams or columns are designed to resist vertical loads and to accommodate deformations of the walls in case of earthquake conditions.

The summary stress report for the FB is in Appendix 3G Section 3G.3. This report contains a description of the FB, the loads, load combinations, reinforcement stresses, and concrete reinforcement details for the basemat, seismic walls, and floors.

3.8.4.1.4 **Firewater Service Complex**

The FWSC consists of two FWS and a FPE that share a common basemat. Each FWS is designed with a cylindrical reinforced concrete wall and a dome-shaped reinforced concrete roof and is capable of storing 2082 m3 (550000 gallons) of water. The FPE is a reinforced concrete box type structure with shear walls and a roof slab that provides enclosure and protection for pumps and tanks. The FWS is lined with a stainless steel plate to prevent leakage of the stored water. The liner plate is not designed to provide structural integrity to the FWS. The key dimensions of the FWSC are summarized in Table 3.8-8.

The summary stress report for the FWSC is in Appendix 3G Section 3G.4. This report contains a description of the FWSC, the loads, load combinations, reinforcement stresses, and concrete reinforcement details for the basemat, seismic walls, and roofs.

3.8.4.1.5 **Radwaste Building**

The RW is shown in Section 1.2.

The RW is a reinforced concrete box type structure consisting of walls and slabs and is supported on a foundation mat. The key dimensions of the RW are summarized in Table 3.8-8.

The RW houses the equipment and floor drain tanks, sludge phase separators, resin hold up tanks, detergent drain tanks, a concentrated waste tank, chemical drain collection tank, associated pumps and systems for the radioactive liquid and solid waste treatment systems.

The RW is a non-seismic category structure. The RW is designed according to the safety classifications defined in RG 1.143 Category RW-IIa. The seismic design of the Radwaste Building is full SSE instead of 1/2 SSE as shown in Table 2 of RG 1.143. The tornado wind loads follow Table 2.0-1 of the DCD Tier 2. Classification RW-IIa SSCs housed within the building are designed to 1/2 SSE.

3.8.4.1.6 Seismic Category I Cable Trays, Cable Tray Supports, Conduits, and Conduit Supports

Electrical cables are carried on continuous horizontal and vertical runs of steel trays or through steel conduits. The tray and conduit locations are based on the requirements of the electrical cable network. Trays or conduits are supported at intervals by supports made of hot or cold rolled steel sections. The supports are attached to walls, floor, and ceilings of structures as required by the arrangement. The type of support and spacing is determined by allowable tray or conduit spans which are governed by rigidity and stress. Bracing is provided where required. Dynamic Analysis methods are described in Section 3.7. The loads, loading combinations, and allowable stresses are in accordance with applicable codes, standards, and regulations consistent with Tables 3.8-6 and 3.8-9. Analysis methods follow those presented in Sections 3.7 and 3.8. Design and location requirements for conduit and cable tray supports are also specified in Subsections 3.9.2 and 3.10.3.2.

3.8.4.1.7 Seismic Category I HVAC Ducts and HVAC Duct Supports

HVAC duct locations and elevations are based on the requirements of the HVAC system. HVAC ducts are made of steel sheet metal and are supported at intervals by supports made of hot or cold rolled steel sections. The supports are attached to walls, floor, and ceilings of structures as required by the arrangement. The type of support and spacing is determined by allowable duct spans that are governed by rigidity and stress. Bracing is provided where required. Dynamic Analysis methods are described in Section 3.7. The loads, loading combinations, and allowable stresses are in accordance with applicable codes, standards, and regulations consistent with Tables 3.8-6 and 3.8-9. Analysis methods follow those presented in Sections 3.7 and 3.8. Design and location requirements for HVAC Ducts and HVAC Duct supports are also specified in Subsections 3.9.2, 9.4.1, 9.4.2 and 9.4.6.

3.8.4.2 Applicable Codes, Standards, and Specifications

3.8.4.2.1 Reactor Building

The major portion of the RB outside Containment structure is not subjected to the abnormal and severe accident conditions associated with a containment. *Applicable documents for the RB design are shown in Table 3.8-9, except items 4, 11, 30 and 32.*

3.8.4.2.2 Control Building

Applicable documents for the CB design are the same as the RB, which are listed in Table 3.8-9.

3.8.4.2.3 Fuel Building

Applicable documents for the FB design are same as the RB, which are listed in Table 3.8-9. Applicable documents for the spent fuel racks and associated structures are specified in Subsection 9.1.2.

3.8.4.2.4 Radwaste Building

Applicable codes, standards, specifications and regulations used in the design and construction of RW are items 1, 2, and 32 listed in Table 3.8-9.

3.8.4.2.5 Welding of Pool Liners

All pool liner welds, including the spent fuel pool liner welds, are visually inspected before starting any other NDE method. The visual weld acceptance criteria are defined in AWS Structural Welding Code, D1.1. In accordance with approved procedures, the welded seams of the liner plate are inspected by:

- Liquid Penetrant Examinations. To be carried out on all liner plate butt, fillet, corner and tee welds in accordance with ASME, Section V, Article 6 requirements. The acceptance criteria are in accordance with the requirements of ASME Section III, NE-5352.
- Helium sniffer test or vacuum box technique in accordance with ASME Section V, Article 10 requirements. Any evidence of leakage is unacceptable.

After construction is finished, each isolated pool is leak tested.

The liner welds for all pools outside of the RCCV, including the spent fuel pool, are backed by leak chase channels and a leak detection system to monitor any leakage during plant operation. The leak chase channels are grouped according to the different pool areas and direct any leakage to area drains. This allows both leak detection and determination of where leaks originate. The functioning of the leak chase channels are checked prior to completion of the pool liner installation.

For the floor area of the FB spent fuel pool liner that is not occupied by fuel storage racks or other equipment, the liner plates above the leak chase channels have a stainless steel reinforcing strip of material to protect against puncture from dropped objects such as a fuel assembly. The liner plates above the leak chase channels in the RB buffer pool deep pit floor do not require a reinforcing strip

of material since the RB buffer pool deep pit floor is fully occupied by high density fuel storage racks or other equipment. These racks or other equipment will shield the RB buffer pool deep pit floor from impacts from dropped objects such as a fuel assembly.

The legs of the free standing spent fuel storage racks rest on bearing pads, which rest on embedded plates that are integral with the fuel building spent fuel pool floor liner plate. The bearing pads are not welded to the embedded plates. The embedded plate carries the rack support reaction loads to the fuel building concrete mat foundation. The embedded plates are anchored to the concrete using anchor studs.

To prevent damage to the fuel building spent fuel pool floor liner plate, a clear gap of at least the maximum displacement under all evaluated loads, including SSE, plus 25 mm (1 in) between all sides of the bearing pad and the edge of the embedded plate will be provided.

3.8.4.2.6 Firewater Service Complex

Applicable documents for the FWSC design are the same as the RB, which are listed in *Table 3.8-9*.

3.8.4.3 Loads and Load Combinations

3.8.4.3.1 Reactor Building

3.8.4.3.1.1 Loads and Notations

This section presents only the loads that are applied to the RB directly. Other loads, which are applied to the RCCV only but have effects on RB structures because of common foundation mat, like P_a and T_a, are also considered in the RB design.

Loads and notations are as follows:

D = Dead load of the structure and equipment plus any other permanent loads, including vertical and lateral pressures of liquids.

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 floor slabs – 4.8 kPa (100 psf).
- Concrete roofs – 2.9 kPa (60 psf).
- Construction live load on floor framing in addition to dead weight of floor – 2.4 kPa (50 psf).

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 is omitted from areas occupied by equipment whose weight is specifically included in dead load. Live load is not omitted under equipment where access is provided, for instance, an elevated tank on four legs.

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 ESBWR design, 25% of full live load L (designated as L_o), is used in the load combinations that include seismic loads.

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.

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.

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.)

P_a = Accident pressure at main steam tunnel due to high energy line break.

F = Internal pressures resulting from flooding of compartments.

E' = SSE loads as defined in Section 3.7 including SSE-induced hydrodynamic pressures in pools. The impulsive and convective pressures may be combined by the SRSS method.

T_o = Thermal effects — load effects induced by normal thermal gradients existing through the RB 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 listed in *Table 3.8-10*. The stress free temperature for the design is 15.5°C (59.9°F).

T_a = Thermal effects (including T_o) which may occur during a design accident.

H = Loads caused by static or seismic earth pressures and water in soil.

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 is taken as 0.9, if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the coefficient for that load is taken as zero.

The safety-related concrete structure is designed using the loads, load combinations, and load factors listed in Table 3.8-15. Because a number of concrete structures in the RB are integrally connected with the concrete containment, the load combinations for the concrete containment, which are listed in Table 3.8-2, are additionally considered in the design of the RB concrete structures. The maximum co-directional responses to each of the excitation components for seismic loads are combined by the SRSS method as described in Subsection 3.8.1.3.6.

3.8.4.3.1.3 Load Combinations for Steel Members

The safety-related steel structure is designed using the loads, load combinations, and load factors listed in Table 3.8-16. The maximum co-directional responses to each of the excitation components for seismic loads are combined by the SRSS method as described in Subsection 3.8.1.3.6.

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

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. In addition, because the CB is structurally separated from the concrete containment, the load combinations for the concrete containment do not apply to the CB design. The live loads and temperature loads are as follows:

- All concrete floors except for HVAC room – 4.8 kPa (100 psf)
- Concrete floors in HVAC room – 2.9 kPa (60 psf)
- Concrete roof – 2.9 kPa (60 psf)
- Construction live load on floor framing in addition to dead weight of floor – 2.4 kPa (50 psf)

The temperatures during normal operating conditions are shown in Table 3.8-11. The temperatures during abnormal operating conditions are shown in Table 3H-10 and are associated with a postulated loss of HVAC function.

3.8.4.3.3 Fuel Building

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. The accident thermal load, T_a, includes the thermal effects of boiling water at 104ºC (219ºF) in the spent fuel pool

which may occur due to loss of FAPCS cooling function. The live loads and temperature loads are as follows:

- All concrete floors except for HVAC room – 4.8 kPa (100 psf)
- Concrete floors in HVAC room – 2.9 kPa (60 psf)
- Concrete roof – 2.9 kPa (60 psf)
- Construction live load on floor framing in addition to dead weight of floor – 2.4 kPa (50 psf)

The temperatures during normal operating conditions are shown in Table 3.8-12.

The spent fuel pool structure (reinforced concrete floor and walls, and steel liner) is evaluated for loads imposed by the spent fuel storage racks in combination with other applicable loads in accordance with the load combinations and acceptance criteria defined in Table 3.8-15. Table 3.8-15 is also applicable to steel liners except that the load factors in load combinations are 1.0 and acceptance criteria are in accordance with ASME Section III Division 2 CC-3700.

3.8.4.3.4 Radwaste Building

Loads and load combinations for the RW are described in Subsection 3.7.2.8.2.

3.8.4.3.5 Firewater Service Complex

Refer to the loads, notations, and combinations established in Subsection 3.8.4.3.1, except that fluid pressure F, accident pressure P_a, accident thermal T_a, accident pipe reactions R_a and pipe break loads Y_r, Y_j, Y_m do not exist. In addition, because the FWSC is structurally separated from the concrete containment, the load combinations for the concrete containment do not apply to the FWSC design. The live loads and temperature loads are as follows:

- All concrete floors (except FWS areas) - 4.8 kPa (100 psf)
- Concrete roof - 2.9 kPa (60 psf)
- Construction live load on floor framing in addition to dead weight of floor - 2.4 kPa (50 psf)

The temperatures during normal operating conditions are shown in Table 3.8-18.

3.8.4.4 Design and Analysis Procedures

3.8.4.4.1 Reactor Building, Control Building and Fuel Building

The RB, CB and FB are analyzed using the linear elastic finite element computer program NASTRAN described in Appendix 3C.

As described in Subsection 3.8.4.1.3, the RB and FB is integrated into one building. Therefore, the Reactor Building/Fuel Building (RB/FB) structure is analyzed using a common FEA model, which includes the RB and FB and also the concrete containment. The model is described in Appendix 3G Section 3G.1.4.1.

The FEA model of the CB includes the entire structure. The details of the FEA model of the CB are described in Appendix 3G Section 3G.2.4.1

The foundation soil is simulated by a set of horizontal and vertical springs in each model. The soil spring constraints are calculated based on the properties of the soil spring used in the SSI analysis model, which is described in Appendix 3A. The constraints by soil surrounding the buildings are conservatively neglected in the FEA models.

3.8.4.4.2 Radwaste Building

The RW is described in Subsection 3.8.4.1.5. The design is in accordance with the criteria in Table 3.8-9 Item 32 for Safety Class RW-IIa.

3.8.4.4.3 Firewater Service Complex

As described in Subsection 3.8.4.1.4, the FWSC consists of two FWS and a FPE that share a common basemat. Therefore, the FWSC structures are analyzed using a common FEA model, which includes the two FWS and a FPE. The model is described in Appendix 3G Section 3G.4.4.1.

The foundation soil is simulated by a set of horizontal and vertical springs in the model. The soil spring constraints are calculated based on the properties of the soil spring used in the soil-structure interaction (SSI) analysis model, which is described in Appendix 3A.

3.8.4.5 Structural Acceptance Criteria

3.8.4.5.1 Reactor Building

The acceptance criteria for the design of the safety-related reinforced concrete structure are included in Table 3.8-15. "U" in Table 3.8-15 is the section strength required to resist design loads based on the strength design method described in Table 3.8-9 item 1 and in SRP 3.8.4 Section II.3. For the acceptance criteria for the load combinations in Table 3.8-2, which is also applicable to the RB concrete design, refer to Subsection 3.8.1.5.

The RB is designed to the more limiting acceptance criteria of the ASME Section III, Division 2, Subsection CC and ACI 349-01. The relevant acceptance criteria are allowable compressive stress in concrete, allowable tensile and compressive stresses in reinforcing steel, and allowable transverse shear stress for the design of RB concrete elements. For a combination of axial force and bending moment, the acceptance criteria of ASME Section III, Division 2, Subsection CC are more limiting than ACI 349-01 and are applied in the RB design. The acceptance criteria for transverse shear are essentially the same between ASME Section III, Division 2, Subsection CC and ACI 349-01. Therefore, the ACI-349-01 acceptance criteria for transverse shear are applied in the RB design. The aforementioned acceptance criteria is also applicable to the additional peripheral volumes for anchoring the containment reinforcement, which are shown in Figure 3.8-1..

The acceptance criteria for the design of the safety-related steel structure are included in Table 3.8-16. Allowable elastic working stress, S, is the allowable stress limit specified in Part 1 of ANSI/AISC N690.

The design criteria preclude excessive deformation of the RB.

3.8.4.5.2 Control Building

The acceptance criteria for the design of the safety-related reinforced concrete structure are included in Table 3.8-15. "U" in Table 3.8-15 is the section strength required to resist design loads based on the strength design method described in Table 3.8-9 item 1 and in SRP 3.8.4 Section II.3.

The acceptance criteria for the design of the safety-related steel structure are included in Table 3.8-16. Allowable elastic working stress, S, is the allowable stress limit specified in Part 1 of ANSI/AISC N690. The design criteria preclude excessive deformation of the CB.

3.8.4.5.3 Fuel Building

The acceptance criteria for the design of the FB are same as the RB in Subsection 3.8.4.5.1.

3.8.4.5.4 Radwaste Building

Structural acceptance criteria and materials criteria for the RW is in accordance with Item 32 in Table 3.8-9 for Safety Class RW-IIa.

3.8.4.5.5 Firewater Service Complex

The acceptance criteria for the design of the FWSC are the same as the CB, which is discussed in Subsection 3.8.4.5.2.

3.8.4.5.6 Exterior Wall Design

The Fermi 3 site-specific exterior wall designs for the RB/FB and CB are evaluated against lateral earth pressures based on the results from the Fermi 3 site-specific SSI and SSSI analyses for the RB/FB and CB presented in Subsection 3.7.2.4.1.

Figure 3.8.4-201a through Figure 3.8.4-201h show the lateral seismic soil pressures on the walls of the RB/FB from Fermi 3 SSI analyses, the ESBWR DCD design soil pressures, and the ESBWRDCDwall capacity passive pressures.

In some of the cases shown in Figure 3.8.4-201a through Figure 3.8.4-201h, the lateral seismic soil pressures on the walls of the RB/FB from Fermi 3 SSI analyses exceed the ESBWRDCDdesign soil pressures. For these cases, the induced out-of-plane bending moments and shear forces in the walls due to the seismic soil pressures from the Fermi 3 SSI analyses are bounded by either the corresponding induced out-of-plane bending moments and shear forces in the walls due to the ESBWRDCDdesign soil pressures or the corresponding induced out-of-plane bending moments and shear forces in the walls due to the ESBWRDCD wall capacity passive pressures. In the design

of the exterior walls, the wall capacity passive pressures are combined with the at-rest soil pressures for wall design.

Figure 3.8.4-202a through Figure 3.8.4-202d and Figure 3.8.4-203a and Figure 3.8.4-203b show the lateral seismic soil pressures on the walls of the CB from Fermi 3 SSI and SSSI analyses and the ESBWRDCD design soil pressures.

In some of the cases shown in Figure 3.8.4-202a through Figure 3.8.4-202d, the lateral seismic soil pressures on the walls of the CB from Fermi 3 SSI analyses exceed the ESBWRDCD design soil pressures. For these cases, the induced out-of-plane bending moments and shear forces in the walls due to the seismic soil pressures from the Fermi 3 SSI analyses are bounded by the corresponding induced out-of-plane bending moments and shear forces in the walls due to the ESBWRDCD design soil pressures.

3.8.4.6 **Material, Quality Control and Special Construction Techniques**

This subsection contains information related to the materials, QC and special construction techniques used in the construction of other Seismic Category I structures.

3.8.4.6.1 **Concrete**

The specified compressive strength of concrete at 28 days, or earlier, is 4000 psi for the foundation mat and 5000 psi for other structures. Concrete material is the same as described in Subsection 3.8.1.6.1 with the following exception: Concrete is batched and placed according to ACI 349-01.

3.8.4.6.2 **Reinforcing Steel**

Reinforcing steel is the same as in Subsection 3.8.1.6.2.

3.8.4.6.3 **Splices of Reinforcing Steel**

Splices of reinforcing steel are the same as in Subsection 3.8.1.6.3 except that placing and splicing is in accordance with ACI 349-01. Welding of reinforcing bars complies with all the applicable requirements of ASME Code Section III, Division 2.

3.8.4.6.4 **Quality Control**

QC is the same as in Subsection 3.8.1.6.5 except that the Construction Specifications reference ACI 349-01 and applicable Regulatory Guides. *For welding of reinforcing bars, inspection and documentation requirements conform to ASME Code Section III, Division 2 also.*

3.8.4.6.5 **Special Construction Techniques**

No special construction techniques are employed other than that some of the components, such as rebar cages, are pre-assembled and lifted into place.

3.8.4.6.6 Structural Steel Including Plates

Structural steel conforms to ASTM A-36, A-500 Gr. B HSS, A-572 Gr. 50 or A-992 W Shapes. Plates conform to ASTM A-36 or ASME SA-516 Gr. 70.

3.8.4.7 Testing and In-Service Inspection Requirements

Other Seismic Category I structures are monitored per NUREG-1801 and 10 CFR 50.65 as clarified in RG 1.160, in accordance with Section 1.5 of RG 1.160.

3.8.5 Foundations

This section describes foundations for all Seismic Category I structures of the ESBWR Standard Plant.

3.8.5.1 Description of the Foundations

The RB including the containment and FB are built on a common foundation mat as described in Subsection 3.8.4. The foundation of the CB is separated from the foundation of the RB and FB.

The foundation of the RB and FB is a rectangular reinforced concrete mat. Its key dimensions are shown in Table 3.8-13. The foundation mat is constructed of cast-in-place conventionally reinforced concrete. It supports the RB, the FB, the containment structure, and other internal structures. The containment structure foundation is defined as within the perimeter or the exterior surface of the containment structure. The containment foundation mat details are discussed in Subsection 3.8.1.1.1.

The CB foundation is rectangular reinforced concrete mat. The key dimensions are included in Table 3.8-13.

The foundation for Category I structures is contained in the summary stress reports for their respective buildings. The RB foundation is contained in Appendix 3G Section 3G.1, the CB foundation is in Appendix 3G Section 3G.2, and the FB foundation is in Appendix 3G Section 3G.3. The summary stress report contains a section detailing safety factors against sliding, over turning, and flotation.

As described in Subsection 3.8.4.1.4, the FWSC consists of two FWS and a FPE that share a common basemat. The foundation of the FWSC is separated from the foundations of the RB/FB and CB. *The foundation of the FWSC is a rectangular reinforced concrete mat. Its key dimensions are shown in Table 3.8-13.* The foundation mat is constructed of cast-in-place conventionally reinforced concrete. It supports the two FWS and their contents, FPE and other associated SSCs. Details of the foundation design and analysis for the FWSC, including foundation stability evaluation are contained in Appendix 3G Section 3G.4.

3.8.5.2 Applicable Codes, Standards and Specifications

The applicable codes, standards and specifications for the containment foundation and for the other Seismic Category I foundations are the same as those for their respective superstructures consistent with SRP 3.8.5 Section II.2.

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.

The jurisdictional boundary for application of Section III, Division 2 of the ASME Code to the concrete containment foundation is discussed in Subsection 3.8.1.1.3.

3.8.5.3 Loads and Load Combinations

The loads and load combinations for the containment foundation and for the other Seismic Category I foundations are the same as those for their respective superstructures with additional foundation stability requirements consistent with SRP 3.8.5 Section II.3.

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 3.8-14.

3.8.5.4 Design and Analysis Procedures

The foundations of Seismic Category I structures are analyzed using the 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 foundation mats for the Concrete Containment, RB/FB, CB and FWSC are analyzed using the linear elastic finite element computer program NASTRAN as described in Subsections 3.8.1.4.1.1, 3.8.4.4.1 and 3.8.4.4.3. The type of finite elements used to model the foundation mat is the thick

shell type of elements that account for out-of-plane shear deformation also. The foundation mat resists out-of-plane forces applied from superstructures and foundation soil. Bending moments in the foundation mat are evaluated for the resultant out-of-plane forces. The foundation soil is modeled with elastic springs and connected to the foundation mat elements in the FEA model. By means of using this method, the SSI is considered in the foundation design, and the requirement of SRP 3.8.5 II 4.a is satisfied.

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 worst case scenario for foundation base mat design is the soft soil because it is subject to largest deformation. From the NASTRAN analysis the results are scanned for the highest loads in the mat sections and are selected for checking the section. This enveloping of most severe loading is done for all loading considered in the analysis. In order to confirm the appropriateness of this condition, basemat deformation and sectional moment are compared between the soft soil case [Vs = 300 m/sec (984 ft/sec)] and the hard rock case [Vs = 1700 m/sec (5577 ft/sec)]. Basemat deformation for the soft soil condition is much larger than that of the hard rock condition. Bending moments for the soft soil are larger than those for the hard rock with few exceptions. The higher bending moments at some locations for the hard rock site have no effect on the design because they are much less than the maximum moments of the soft soil site on which rebar sizing is based.

In the global FEA model the soil springs are assumed to be two-way springs capable of withstanding compression and tension. To evaluate the effect of potential uplift of the basemat under seismic loads, the soil springs, once in tension, are removed through an iterative process. This iterative process is continued until there are no more springs in tension. The analysis results confirmed the adequacy of the basemat design. Details are provided in Appendix 3G Section 3G.1.5.5.1.

The selected waterproofing material for the bottom of the basemat is a chemical crystalline powder that is added to the mud mat mixture forming a water proof barrier when cured. No membrane waterproofing is used under the foundations in the ESBWR.

The standard ESBWR design is developed using a range of soil conditions as detailed in Appendix 3A. The minimum requirements for the physical properties of the site-specific subgrade materials are furnished in Table 2.0-1. Stability of subsurface materials and foundations are addressed in Table 2.0-2, Subsection 2.5.4. Settlement of the foundations, and differential settlement between foundations for the site-specific foundations medium, is calculated, and safety-related systems (i.e., piping, conduit, etc.) 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 is evaluated.

A detailed description of the analytical and design methods for the foundations of the RB including the containment, CB, FB and FWSC is included in Appendix 3G.

3.8.5.5 Structural Acceptance Criteria

The structural acceptance criteria for the containment foundation and for the other Seismic Category I foundations are the same as those for their respective superstructures with additional foundation stability requirements consistent with SRP 3.8.5 Section II.5.

The main structural criteria for the containment portion of the foundation are to provide 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 RB, CB, FB and FWSC foundations are described in Subsection 3.8.4.5.

The allowable factors of safety of the ESBWR structures for overturning, sliding, and flotation are included in Table 3.8-14. The calculated factors of safety are shown in Appendix 3G 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:

$$FS = (F_{ub} + F_{us} + F_r + F_{us}' + F_r')/(F_v + F_o)$$

Notations are as follows:

F_{ub} = Friction resistance force provided at the potential sliding plane.

F_{us} = Skin friction resistance force provided by basemat side parallel to the direction of motion.

F_r = Lateral resistance pressure along the wall and basemat opposite to the direction of motion, provided that the wall capacity passive pressure is not exceeded.

F_{us}' = Skin friction resistance force provided by shear key side parallel to the direction of motion (when shear keys are used).

F_r' = Lateral resistance pressure along the shear key opposite to the direction of motion (when shear keys are used).

F_v = Base shear at the basemat bottom.

F_o = Lateral soil force due to surcharge load of adjacent structure, as applicable.

The sliding evaluation is performed for two orthogonal horizontal directions separately. In each direction the horizontal SSE shear and vertical SSE force at the base are combined in a time consistent manner at each time step when the input motions are statistically independent. Alternately, the maximum horizontal SSE base shear may be combined with the maximum vertical

SSE force acting upward. The total vertical load at the base takes into account the dead loads and buoyancy force.

The factor of safety against flotation is defined as:

$$FS = F_{DL}/F_B$$

Notations are as follows:

F_{DL} = Downward force due to dead load.

F_B = Upward force due to buoyancy.

3.8.5.5.1 Foundation Stability

The Fermi 3 site-specific foundation stability for the RB/FB and CB are evaluated against overturning, sliding, and floatation based on the results from the Fermi 3 site-specific SSI analyses for the RB/FB and CB presented in Subsection 3.7.2.4.1. The stability calculations against overturning, sliding, and floatation are executed according to the procedure presented in Subsection 3.8.5.5.

The factor of safety against overturning due to earthquake loading is determined by the energy approach described in Subsection 3.7.2.14. The calculated Fermi 3 site-specific factors of safety against overturning based on the Fermi 3 site-specific SSI for the RB/FB and CB are shown in Table 3.8.5-201 and Table 3.8.5-202, respectively. It is shown that the Fermi 3 site-specific factors of safety against overturning for the RB/FB and CB are 2,262 and 1,733 (greater than 1.1 as required by SRP 3.8.5), respectively. These factors of safety indicate that the Fermi 3 RB/FB and CB are stable against overturning.

The Fermi 3 site-specific sliding evaluation is performed using forces generated during the Fermi 3 site-specific SSI analyses, which neglects the backfill above the top of the bedrock and with the RB/FB and CB in firm contact with the bedrock using fill concrete as backfill in the gap between the RB/FB and CB, and the bedrock up to the top of Bass Islands Group bedrock at Elevation 168.2 m (552.0 ft) NAVD 88. The gap between the RB/FB and CB up to the top of the Bass Islands Group bedrock at Elevation 168.2 m (552.0 ft) NAVD 88 is also filled with fill concrete. As the Fermi 3 site-specific SSI neglects the backfill above the top of the bedrock, forces associated with the backfill are not included in the sliding analysis; therefore, the bedrock alone supplies the resistance to sliding of both the RB/FB and the CB. In the sliding evaluation for the Fermi 3 RB/FB and CB, the following skin friction resistance forces are neglected:

1. F_{us} = Skin friction resistance force provided by basemat side parallel to the direction of motion (i.e., F_{us} = 0)
2. F_{us}' = Skin friction resistance force provided by shear key side parallel to the direction of motion (when shear keys are used (i.e., F_{us}' = 0).

The calculated Fermi 3 site-specific factors of safety against sliding for the RB/FB and CB are shown in Table 3.8.5-201 and Table 3.8.5-202, respectively. The Fermi 3 site-specific factors of safety against sliding for the RB/FB and CB are 1.22 and 1.10 (equal to or greater than minimum factor of safety of 1.1 as required by SRP 3.8.5), respectively. With the exception of the CB with no engineered backfill above the top of the Bass Islands Group bedrock, the sliding stability safety factors are based on available friction at the bottom of the foundation. For CB with no engineered backfill, in addition to the base friction, lateral bearing resistance along the CB foundation sides by the in-situ rock or concrete fill between the CB and RB/FB is required to meet the minimum required sliding safety factor of 1.10. The concrete fill between the CB and the RB/FB is capable of providing the required lateral bearing resistance through the friction between the bottom of the concrete fill and the top of the in-situ rock below. The in-situ Bass Islands Group bedrock is also found capable of providing the required lateral bearing resistance. These factors of safety indicate that the Fermi 3 RB/FB and CB are stable against sliding.

The sliding of the FWSC was evaluated using the driving forces (the base shear time history forces) based on the governing factor of safety cases from the ESBWR DCD SSI analysis results without crediting the backfill surrounding the basemat. The sliding evaluation also includes the fill concrete below the FWSC in which the shear keys are embedded. The presence of the shear keys results in potential failure occurring within the fill concrete. The fill concrete was evaluated in accordance with ACI 318 and the corresponding portions of ACI 349 considering the following:

- Failure of the fill concrete in compression from lateral pressure applied by the shear keys. This potential failure condition is checked using the ACI 318 Section 22.5.5.
- Failure through the fill concrete at or below the base of the shear keys considering the maximum amount of shear resistance from shear-friction reinforcement allowed in ACI 318, Section 11.6 and the corresponding portions of ACI 349, Section 11.7.

The resulting factor of safety against sliding is greater than 15, which is greater than the minimum factor of safety of 1.1 as required by SRP 3.8.5. During detailed design, the amount of shear-friction reinforcement in the fill concrete is selected to provide a minimum factor of safety of 1.1 against sliding for the FWSC, which provides a minimum 10 percent design margin for the shear-friction reinforcement.

The calculated Fermi 3 site-specific factors of safety against floatation for the RB/FB and CB are shown in Table 3.8.5-201 and Table 3.8.5-202, respectively. It is shown that the Fermi 3 site-specific factors of safety against floatation for the RB/FB and CB are 3.50 and 1.86 (greater than minimum factor of safety of 1.1 as required by SRP 3.8.5), respectively. These factors of safety indicate that the Fermi 3 RB/FB and CB are stable against floatation.

3.8.5.5.2 Soil Bearing Pressures

The maximum soil dynamic bearing pressure demand at the Fermi 3 site for the BE, UB, and LB subsurface profiles, based on the results from the Fermi 3 site-specific SSI analyses for the RB/FB

and CB presented in Subsection 3.7.2.4.1, are evaluated using the Modified Energy Balance Method according to the Appendix 3G Section 3G.1.5.5.

The Fermi 3 site-specific SSI maximum dynamic soil bearing pressure demands are summarized in Table 3.8.5-203 for the RB/FB and CB. The Fermi 3 site-specific SSI maximum soil bearing pressure (maximum toe pressure) demands are all less than the maximum dynamic bearing demands for both the RB/FB and the CB in Table 3G.1-58 and Table 3G.2-27, respectively.

The Fermi 3 site-specific SSI maximum dynamic soil bearing pressure demands presented in Table 3.8.5-203 for the RB/FB and the CB are compared with the Fermi 3 site-specific allowable bearing capacity under the dynamic condition in Table 2.5.4-227 in Subsection 2.5.4.10. It is confirmed that the Fermi 3 site-specific maximum dynamic soil bearing pressure demands for the RB/FB and CB are less than the allowable bearing capacity under dynamic condition presented in Table 2.5.4-227 in Subsection 2.5.4.10.

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 for the containment foundation mat and Subsection 3.8.4.6 for the foundations of the other Seismic Category I structures.

3.8.5.7 **Testing and In-Service Inspection Requirements**

The foundations of Seismic Category I structures are monitored per NUREG-1801 and 10 CFR 50.65 as clarified in Section 1.5 of RG 1.160.

3.8.6 **Special Topics**

3.8.6.1 **Foundation Waterproofing**

The selected waterproofing material for the bottom of the basemat is a chemical crystalline powder that is added to the mud mat mixture forming a water proof barrier when cured. For the vertical edges of the basemat, spray-type crystalline waterproofing compound will be applied in accordance with manufacturer application procedures. No membrane waterproofing is used under the foundations in ESBWR.

Contraction joints are made after the mud mat concrete is poured to control cracks. The width and spacing of the contraction joints follow the common practice for pavements. The spray-type crystalline waterproofing compound applied on the top surface of the mud mat will fill up cracks in the mud mat that have been formed. After application of the crystalline waterproofing compound, which has a self-healing capability up to a 0.4 mm crack width, this waterproofing compound will be able to eliminate cracks in the mud mat concrete.

The type of the waterproofing system applied to the exterior walls is sheet-applied barrier materials described in Section 4.2.1.4 of ACI 515.1R-79 (revised 1985) (e.g. non-vulcanized butyl rubber

sheet). The minimum thickness of the waterproofing sheet is 2.0 mm. Two layers of sheets are applied to the exterior walls below grade.

3.8.6.2 Site-Specific Physical Properties and Foundation Settlement

See Table 2.0-1 for soil properties requirements of site-specific foundation bearing capacities, minimum shear wave velocity, liquefaction potential, angle of internal friction and maximum settlement values for Seismic Category I buildings.

For sites not meeting the soil property requirements, a site-specific analysis is required to demonstrate that site-specific conditions are enveloped by the standardized design.

3.8.6.3 Structural Integrity Pressure Result

See Tier 1 Table 2.15.1-2 for the SIT of the containment structure, which is an ITAAC item.

3.8.6.4 Identification of Seismic Category I Structures

See Subsections 3.8.1, 3.8.2, 3.8.3 and 3.8.4 for identification of Seismic Category I structures.

3.8.6.5 Foundation Mud Mat

The mud mat is designed as structural plain concrete in accordance with ACI 318-05. The specified compressive strength of concrete at 28 days, or earlier, is 17.3 MPa (2500 psi) for the mud mat. The thickness of the mud mat is no less than 200 mm (8 in.). The performance testing requirements for the mud mat are those delineated in ACI 318-05. The mud mat construction is performed in accordance with the same standards and requirements as the basemat. The top surface of the mudmat is intentionally roughened in accordance with ACI 349-01 Subsection 11.7.9 requirement.

In order to ensure that the failure surface can only occur within the soil below the mud mat and to justify the use of a 0.7 coefficient of friction in the sliding evaluation, troughs are provided on the ground surface before the mud mat is poured. The size of the troughs is approximately 150 mm (6 in) wide and 100 mm (4 in) deep. They are arranged in a grid pattern with no larger than a 2.5 m (8.2 ft) spacing distributed over the footprint of the mud mat.

3.8.7 References

3.8-1 GE Hitachi Nuclear Energy, “ESBWR ICS and PCCS Condenser Combustible Gas Mitigation and Structural Evaluation,” NEDE-33572P, Class II (Proprietary), Revision 3, September 2010; NEDO-33572, Revision 3, Class I (Non-proprietary), September 2010.

Table 3.8-1 Key Dimensions of Concrete Containment

Portion	Dimension	Notes
Foundation mat	*Thickness = 5.1 m*	
Containment wall	*Thickness = 2.0 m*	
	Inside radius = 18.0 m	
	Height = 19.95 m	*From the top of the suppression pool slab to the bottom of the top slab*
RPV pedestal (Part of Lower Containment)	*Thickness = 2.5 m*	
	Inside radius = 5.6 m	
	Height = 15.05 m	*From the top of the foundation mat to the top of the suppression pool slab*
Top slab	*Thickness = 2.4 m*	
Suppression pool slab	*Thickness = 2.0 m*	

SI to U.S. Customary units conversion (SI units are the controlling units and U.S. Customary units are for reference only):

1 m = 3.28 ft

Table 3.8-2 Load Combinations, Load Factors and Acceptance Criteria for the Reinforced Concrete Containment[1],[2],[3],[7]

Description	No.	Load Conditions																Acceptance Criteria[6]
		D	L	P_t	P_o	P_a	T_t	T_o	T_a	E'	W	W'	R_o	R_a	Y[4]	SRV	LOCA	
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		S
Factored Severe Environmental	4	1.0	1.3		1.0			1.0			1.5		1.0			1.3		U
Extreme	5	1.0	1.0		1.0			1.0		1.0			1.0			1.0		U
Environmental	6	1.0	1.0		1.0			1.0				1.0	1.0			1.0		U
Abnormal	7	1.0	1.0			1.5			1.0					1.0		1.25	Note[5]	U
	8	1.0	1.0			1.0			1.0					1.25		1.0	Note[5]	U
	9	1.0	1.0			1.25			1.0					1.0		1.25	Note[5]	U
Abnormal/Severe Environmental	10	1.0	1.0			1.25			1.0		1.25			1.0		1.0	Note[5]	U
Abnormal/ Extreme Environmental	11	1.0	1.0			1.0			1.0	1.0				1.0	1.0	1.0	Note[5]	U

(1) The loads are described in Subsection 3.8.1.3 and acceptance criteria in Subsection 3.8.1.5.

(2) 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.

(3) Because P_a, T_a, SRV and LOCA are time-dependent loads, their effects are superimposed accordingly.

(4) Y includes Y_j, Y_m and Y_r.

(5) LOCA loads, CO, CHUG and Pool Swell (PS) are time-dependant loads for which DLF may be used. The sequence of occurrence is given in Appendix 3B. The load factor for LOCA loads is the same as the corresponding pressure load P_a. LOCA loads include hydrostatic pressure (with a load factor of 1.0) due to containment flooding.

(6) S = Allowable Stress as in ASME Section III, Div. 2, Subsection CC-3430 for Service Load Combination. U = Allowable Stress as in ASME Section III, Div. 2, Subsection CC-3420 for Factored Load Combination.

(7) The peak responses of dynamic loads do not occur at the same instant. SRSS method to combine peak dynamic responses is acceptable for concrete structures.

Table 3.8-3 Major Allowable Stresses in Concrete and Reinforcing Steel

	Concrete		Reinforcing Steel	
	Compression	***Tangential Shear***	***Tension***	
Service Load Combination	***Foundation*** *12.4 MPa (For primary case)* *16.6 MPa* *(For primary plus secondary case)* ***Top Slab*** *18.6 MPa (For primary case)* *24.8 MPa* *(For primary plus secondary case)* ***Others*** *15.5 MPa (For primary case)* *20.7 MPa* *(For primary plus secondary case)*	*(1) Provided by concrete* $v_c = 0$ *(2) Provided by orthogonal reinforcement* $v_{so} = 0.415\sqrt{f'_c} = 2.18\ MPa$ *(For foundation)* *= 2.67 MPa* *(For top slab)* *= 2.44 MPa* *(For others)*	*206.8 MPa* *273.0 MPa* *310.2 MPa*	*(For primary case)* *(For primary plus secondary case)* *(For test pressure case)*
Factored Load Combination	***Foundation*** *20.7 MPa (For primary case)* *23.5 MPa* *(For primary plus secondary case)* ***Top Slab*** *31.1 MPa (For primary case)* *35.2 MPa* *(For primary plus secondary case)* ***Others*** *25.9 MPa (For primary case)* *29.3 MPa* *(For primary plus secondary case)*	*(1) Provided by concrete* $v_c = 0$ *(2) Provided by orthogonal reinforcement* $v_{so} = 0.830\sqrt{f'_c} = 4.36\ MPa$ *(For foundation)* *= 5.34 MPa* *(For top slab)* *= 4.88 MPa* *(For others)*	*372.2 MPa*	

Note: f'_c is in MPa

SI to U.S. Customary units conversion (SI units are the controlling units and U.S. Customary units are for reference only):

1 MPa = 145.038 psi

Table 3.8-4 Load Combination, Load Factors and Acceptance Criteria for Steel Containment Components of the RCCV[(1), (2), (3)]

Service Level	No	Load Combination [(1)]																	Acceptance Criteria			
		D	L	P_t	P_o	P_a	T_t	T_o	T_a	E'	W	W'	R_o	R_a	Y[(4)]	SRV[(12)]	DET [(12)]	LOCA [(5),(12)]	P_m	P_L	P_L+P_b [(8)]	P_L+P_b+Q
Test Condition	1	1.0	1.0	1.0			1.0												0.75 S_y	1.15Sy	1.15Sy [(11)]	N/A [(10)]
Design Condition	2	1.0	1.0			1.0			1.0					1.0					1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	N/A
Level A, B [(9)]	3	1.0	1.0		1.0			1.0					1.0						1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	3.0 S_{m1}
	4	1.0	1.0		1.0			1.0								1.0						
	5	1.0	1.0			1.0			1.0					1.0				1.0				
	6	1.0	1.0			1.0			1.0					1.0		1.0		1.0				
Level C [(6)]	7	1.0	1.0		1.0			1.0		1.0			1.0						1.2 S_{mc} or* 1.0 S_y	1.8 S_{mc} or* 1.5S_y	1.8 S_{mc} or* 1.5S_y	N/A
	8	1.0	1.0			1.0			1.0	1.0				1.0		1.0		1.0				
	9	1.0	1.0			1.0			1.0	1.0				1.0				1.0				
	12[13]	1.0	1.0						1.0	1.0							1.0					
Level D [(7)]	10	1.0	1.0			1.0			1.0	1.0				1.0	1.0	1.0		1.0	S_f	1.5S_f	1.5S_f	N/A
	11	1.0	1.0			1.0			1.0	1.0				1.0	1.0			1.0				
		(Deleted)																				

Notes:

(1) *The loads are described in* Subsection 3.8.1.3.

(2) *For any load combination, if the effects of any load component (other than D) reduces the combined load, then the load component is deleted from the load combination.*

(3) *P_a, T_a, SRV and LOCA are time-dependent loads. The sequence of occurrence is given in* Appendix 3B.

(4) *Y includes Y_j, Y_m and Y_r.*

(5) *LOCA loads include CO, CHUG and PS. They are time-dependent loads. The sequence of occurrence is given in* Appendix 3B. *LOCA loads include hydrostatic pressure (with a load factor of 1.0) due to containment flooding.*

(6) *Limits identified by (*) indicate a choice of the larger of the two.*

(7) *S_f is 85% of the general primary membrane allowable permitted in Appendix F, ASME B&PV Code, Section III. In the application of Appendix F, S_{m1}, if applicable, is as specified in Section II, Part D, Subpart 1, Tables 2A and 2B of ASME B&PV Code, which is the same as S_m.*

(8) *Values shown are for a rectangular section. See NE-3221.3(d) for other than a solid rectangular section.*

(9) *The allowable stress intensity S_{m1} is the S_m listed in Section II, Part D, Subpart 1, Tables 2A and 2B of the ASME B&PV Code. The allowable stress intensity S_{mc} is 1.1 times the Sm listed in Section II, Part D, Subpart 1, Tables 1A and 1B of the ASME B&PV Code, except that S_{mc} does not exceed 90% of the material's yield strength at temperature shown in Section II, Part D, Subpart 1, Tables Y-1 of the ASME B&PV Code.*

(10) *N/A = No evaluation required.*

(11) *Bending and General Membrane P_m+P_b.*

(12) *The peak responses of dynamic loads do not occur at the same instant. SRSS method to combine peak dynamic responses is acceptable for steel structures.*

(13) These loads are applicable only to the PCCS condenser.

Table 3.8-5 Welding Activities and Weld Examination Requirements for Containment Vessel

Component	Weld Type	NDE Requirements
Steel components[1] (no concrete backing, ASME Section III, Division 1, Subsection NE)	Category A, Butt welds (Long'l)	RT[2]
	Category B, Butt welds (Circ.)	RT[2][3]
	Category C, Butt welds	RT[2]
	Category C, Nonbutt welds	UT or MT or PT
	Category D, Butt welds	RT[2]
	Category D, Nonbutt welds	UT or MT or PT
	Structural attachment welds a) Butt welds b) Nonbutt welds	 RT[2] UT or MT or PT
	Special welds, Weld metal cladding	PT
Containment liner[4] (with concrete backing, ASME Section III, Division 2, Subsection CC)	Category A, Butt welds (Long'l)	RT[5]
	Category B, Butt welds (Circ.)	RT[5]
	Category D, Butt welds	RT[5]
	Category D, Nonbutt welds	UT or MT or PT[6]
	Categories E, F, G, J, and Full Penetration H	UT or MT or PT[6]
	Structural attachment welds	MT or PT
	Special welds, Weld metal cladding	PT

Notes:

(1) Welded joint locations of the Categories are shown in Figure NE-3351-1 of the ASME Section III.
Welding activities and welding examinations comply with the provisions of the ASME Section III Subsection NE.
Backing bars are not used in weld joints in flued-head containment penetration assemblies or other penetration sleeves and process piping.

(2) When the joint detail does not permit radiographic examination, UT plus MT or PT is substituted as permitted by ASME Section III, Division 1, Subarticle NE-5280.

(3) Surface examination of the root pass and completed weld is substituted in electrical penetration assemblies for RT per ASME Section III, Division 1, Subarticles NE-3352.2 (b) and NE-5280.

(4) Welded joint locations of the Categories are shown in Figure CC-3831-1 of the ASME Section III.

Welding activities and welding examinations comply with the provisions of the ASME Section III Subsection CC.

(5) *RT is used for welds without backup bars. For welds with backup bars MT or UT is used.*

(6) *Only for austenitic welds, liquid penetrant shall be substituted for magnetic particle examination.*

LEGEND:

RT – Radiographic Examination PT – Liquid Penetrant Examination

MT – Magnetic Particle Examination UT – Ultrasonic Examination

Table 3.8-6 Codes, Standards, Specifications, and Regulations Used in the Design and Construction of Seismic Category I Internal Structures of the Containment (Sheet 1 of 2)

Specification Reference Number	Specification or Standard Designation	Title
1	ACI 301-05	Specifications for Structural Concrete
2	ACI 347-04	Guide to Formwork for Concrete
3	ACI 305R-99	Hot Weather Concreting
4	ACI 211.1-91	Standard Practice for Selecting Proportions for Normal, Heavy Weight and Mass Concrete
5	ACI 315-99	Details and Detailing of Concrete Reinforcement
6	ACI 306.1-90	Standard Specification for Cold Weather Concreting (Reapproved 2002)
7	ACI 309R-05	Guide for the Consolidation of Concrete
8	ACI 308.1-98	Standard Specification for Curing Concrete
9	ACI 212.3R-04	Chemical Admixtures for Concrete
10	ACI 214R-02	Evaluation of Strength Test Results of Concrete
11	ACI 311.5-04	Guide for Concrete Plant Inspection and Testing of Ready-Mixed Concrete
12	ACI 304R-00	Guide for Measuring, Mixing, Transporting, and Placing Concrete
13	ACI 349-01/349R-01	*Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary*
14	Not Used.	
15	*ANSI/AISC N690-1994 (R2004) and S2*	*Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities and Supplement No. 2*(1)
16	AWS D1.1/D1.1M 2004	Structural Welding Code – Steel (AWS D1.1/D1.1M) Rev. 05
17	EPRI NP-5380, 1987	NCIG-01 - Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, Rev. 2, Sep. 1987.
18	ANSI/ASME NQA-1-1983	Quality Assurance Program Requirements for Nuclear Facilities, 1983 Edition with NQA-1a-1983 Addenda, (Reference Section 17.0)
19	Regulatory Guide 1.54	*Service Level I, II and III Protective Coatings Applied to Nuclear Power Plants, Rev. 1, July 2000.*
20	*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, Rev. 1 and Draft 2.*
21	*Regulatory Guide 1.136*	*Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments, Rev. 3, March 2007.*
22	*Regulatory Guide 1.142*	*Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments), Nov. 2001.*
23	*Regulatory Guide 1.199*	*Anchoring Components and Structural Supports in Concrete, November 2003.*

Table 3.8-6 Codes, Standards, Specifications, and Regulations Used in the Design and Construction of Seismic Category I Internal Structures of the Containment (Sheet 2 of 2)

Specification Reference Number	Specification or Standard Designation	Title
24	(Deleted)	
25	ASME/ANSI AG-1-2003	Code on Nuclear Air and Gas Treatment
26	AISI CF02-1	AISI Specification for the Design of Cold-Formed Steel Structural Members, AISI 2001 Edition and 2004 Supplement
27	SMACNA 1481, Third Edition, 2005	HVAC Duct Construction Standards-Metal and Flexible
28	IEEE-344-1987	Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations

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
EPRI	Electric Power Research Institute
IEEE	Institute of Electrical and Electronics Engineers, Inc.
NCIG	Nuclear Construction Issues Group
SMACNA	Sheet Metal and Air Conditioning Contractors' National Association

Note:

1. To comply with NUREG-1503, Appendix G, NRC Position on the use of ANSI/AISC N690 (1984), for impact and impulsive loads, the ductility factors μ in Table Q1.5.8.1 are replaced with the ductility factors in Appendix A to SRP Subsection 3.5.3.

Table 3.8-7 Load Combination, Load Factors and Acceptance Criteria for Steel Structures Inside the Containment[1],[2]

Category	No.	Load Combination														Acceptance Criteria[5]
		D	L	P_o	P_a	T_o	T_a	E'	W	W'	R_o	R_a	Y[4]	SRV[6],[7]	LOCA[6],[7]	
Normal	1	1.0	1.0	1.0												S
	2	1.0	1.0	1.0		1.0					1.0					$S^{(a)}$
Severe Environmental	3	1.0	1.0	1.0					1.0					1.0		S
	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		$S^{(a)}$
	6	1.0	1.0	1.0		1.0		1.0			1.0			1.0		$S^{(a)}$
Extreme Environmental	7	1.0	1.0	1.0		1.0				1.0	1.0			1.0		$1.6S^{(b)(c)}$
	8	1.0	1.0	1.0		1.0		1.0			1.0			1.0		$1.6S^{(b)(c)}$
Abnormal	9	1.0	1.0		1.0		1.0					1.0		1.0	Note[3]	$1.6S^{(b)(c)}$
	9a	1.0	1.0		1.0		1.0							1.0	Note[3]	$1.6S^{(b)(c)}$
Abnormal/Severe Environmental	10	1.0	1.0		1.0		1.0					1.0	1.0	1.0	Note[3]	$1.6S^{(b)(c)}$
Abnormal/Extreme Environmental	11	1.0	1.0		1.0		1.0	1.0				1.0	1.0	1.0	Note[3]	$1.7S^{(b)(c)}$

(1) The loads are described in Subsection 3.8.3.3 and acceptance criteria in Subsection 3.8.3.5.

(2) For any load combination, where any load reduces the effects of other loads, the corresponding coefficient for that load is taken as 0.9 if it can be demonstrated that the load is always present or occur simultaneously with the other loads. Otherwise, the coefficient for that load is taken as zero.

(3) LOCA loads, such us CO, CHUG and PS are time-dependant loads. The sequence of occurrence is given in Appendix 3B. The load factor for LOCA loads is the same as the corresponding Pressure Load P_a. The maximum values of P_a, T_a, R_a, Y including an appropriate DLF are used, unless an appropriate time history analysis is performed to justify otherwise. LOCA includes Annulus Pressurization loads and effects. LOCA loads include hydrostatic pressure (with a load factor of 1.0) due to containment flooding.

(4) Y includes Y_j, Y_m and Y_r.

(5) Allowable elastic working stress (S) is the allowable stress limit specified in Part 1 of ANSI/AISC N-690-1994-s2 (2004).

(a) For primary plus secondary stress, the allowable limits are increased by a factor of 1.5.

(b) Stress limit coefficient in shear does not exceed 1.4 in members and bolts.

(c) The Stress limit coefficient where axial compression exceeds 20% of normal allowable, is 1.5 for load combinations 7, 8, 9, 9a and 10, and be 1.6 for load combination 11.

(6) *Other loads such as jet loads and drag loads associated with SRV and LOCA hydrodynamic loads are applicable to submerged structures and those above suppression pool water surface. Methodology for calculation of these loads is given in the ESBWR Containment Load Definition (NEDE-33261P).*

(7) *The peak responses of dynamic loads do not occur at the same instant. SRSS method to combine peak dynamic responses is acceptable for steel structures.*

Table 3.8-8 Key Dimensions of RB, CB, FB, RW and FWSC

Building	Dimension		Notes
Reactor Building	Story	Six stories (above grade) Three stories (below grade)	
	Plan	49.0 m × 49.0 m (below EL 34.0 m) 49.0 m x 39.0 m (above EL 34.0 m)	
	Height	64.2 m	From the top of the foundation mat †
Control Building	Story	Two stories (above grade) Two stories (below grade)	
	Plan	30.3 m × 23.8 m	
	Height	21.20 m	From the top of the foundation mat †
Fuel Building	Story	One story (above grade) Three stories (below grade)	Excluding the penthouse
	Plan	21.0 m × 49.0 m	
	Height	34.0 m	From the top of the foundation mat † (excluding the penthouse)
Radwaste Building	Story	Two stories (above grade) Two stories (below grade)	
	Plan	66.0 m x 33.8 m	
	Height	26.0 m	From the top of the foundation mat (excluding the penthouse)
Firewater Service Complex	Story	One story (above grade)	
	Plan	52.0 m x 20.0 m	
	Height	15.1 m (FWS) 3.6 m (FPE)	From the top of the foundation mat †

† *For relative location of Grade to top of mat see* Table 3.8-13

SI to U.S. Customary units conversion (SI units are the controlling units and U.S. Customary units are for reference only):

1 m = 3.28 ft

Table 3.8-9 Codes, Standards, Specifications, and Regulatory Guides Used in the Design and Construction of Seismic Category I Structures (Sheet 1 of 2)

Specification Reference Number	Specification or Standard Designation	Title
1	*ACI 349-01/349R-01*	*Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary*
2	*ANSI/AISC N690-1994 (R2004) & S2*	*Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities and Supplement No. 2*[(1)]
3	*ASME-2004*	*Boiler and Pressure Vessel Code Section III, Division 2, Subsection CC*
4	*ASME-2004*	*Boiler and Pressure Vessel Code Section III, Subsection NE, Division 1, Class MC*
5	ANSI/ASME NQA-1-1983	Quality Assurance Program Requirements for Nuclear Facilities, 1983 Edition with NQA-1a-1983 Addenda, (Reference Section 17.0)
6	AWS D1.1/D1.1M 2004	Structural Welding Code - Steel
7	AWS D1.4 -98	Structural Welding Code - Reinforcing Steel (AWS D1.1/D1.1M) Rev. 05
8	AWS D1.6-99	Structural Welding Code for Stainless Steel
9	ASCE 4-98	Seismic Analysis of Safety-Related Nuclear Structures
10	ASCE 7-02	Minimum Design Loads for Buildings and Other Structures
11	AISC 360-05	2005 AISC Specification for Structural Steel Building
12	SSPC-PA-1-00	Paint Application Specification No. 1, Shop, Field and Maintenance Painting of Steel
13	SSPC-PA-2-04	Paint Application Specification No. 2, Measurement of Dry Coating Thickness with Magnetic Gages
14	SSPC-SP-1-82	Surface Preparation Specification No. 1, Solvent Cleaning
15	SSPC-SP-5-00	Surface Preparation Specification No. 5, White Metal Blast Cleaning
16	SSPC-SP-6-00	Surface Preparation Specification No. 6, Commercial Blast Cleaning
17	SSPC-SP-10-00	Surface Preparation Specification No. 10, Near-White Blast Cleaning
18	ASME 2004	Boiler and Pressure Vessel Code Section II
19	Not Used	
20	*Regulatory Guide 1.28*	*Quality Assurance Program Requirements (Design and Construction), Aug. 1985*
21	*Regulatory Guide 1.29*	*Seismic Design Classification, Sep. 1978*
22	*Regulatory Guide 1.31*	*Control of Ferrite Content in Stainless Steel Weld Metal, Apr. 1978*
23	*Regulatory Guide 1.44*	*Control of the Use of Sensitized Stainless Steel, May 1973*
24	*Regulatory Guide 1.54*	*Service Level I, II and III Protective Coatings Applied to Nuclear Power Plants, Rev. 1, July 2000*
25	*Regulatory Guide 1.60*	*Design Response Spectra for Seismic Design of Nuclear Power Plants, Dec. 1973*
26	*Regulatory Guide 1.61*	*Damping Values for Seismic Design of Nuclear Power Plants, Rev. 1*

Table 3.8-9 Codes, Standards, Specifications, and Regulatory Guides Used in the Design and Construction of Seismic Category I Structures (Sheet 2 of 2)

Specification Reference Number	Specification or Standard Designation	Title
27	*Regulatory Guide 1.69*	*Concrete Radiation-Shields for Nuclear Power Plants, Dec. 1973*
28	*Regulatory Guide 1.76*	*Design Basis Tornado for Nuclear Power Plants, Apr. 1974*
29	*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, Rev. 1*
30	*Regulatory Guide 1.136*	*Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments, Rev. 3, March 2007.*
31	*Regulatory Guide 1.142*	*Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments), Nov. 2001*
32	*Regulatory Guide 1.143*	*Design Guidance for Radioactive Waste Management Systems, Structures and Components installed in Light Water Cooled Nuclear Power Plants, Nov. 2001*(2)
33	*Regulatory Guide 1.199*	*Anchoring Components and Structural Supports in Concrete, November 2003.*
34	(Applicable ASTM Specifications for Materials and Standards)	
35	(Deleted)	
36	ASME/ANSI AG-1-2003	Code on Nuclear Air and Gas Treatment
37	AISI CF02-1	AISI Specification for the Design of Cold-Formed Steel Structural Members, AISI 2001 Edition and 2004 Supplement
38	SMACNA 1481, Third Edition, 2005	HVAC Duct Construction Standards-Metal and Flexible
39	IEEE-344-1987	Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations
40	Regulatory Guide 1.57	*Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components, March 2007.*

Explanation of Abbreviation

ACI	American Concrete Institute	AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute	ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers	ASME	American Society for Mechanical Engineer
AWS	American Welding Society	IEEE	Institute of Electrical and Electronics Engineers, Inc.
SMACNA	Sheet Metal and Air Conditioning Contractors' National Association		

SSPC Steel Structures Painting Council

See Subsections 3.8.1.2 and 3.8.3.2 for Applications.

Notes:

(1) To comply with NUREG-1503, Appendix G, NRC Position on the use of ANSI/AISC N690 (1984), for impact and impulsive loads, the ductility factors μ in Table Q1.5.8.1 are replaced with the ductility factors in Appendix A to SRP Subsection 3.5.3.

(2) The seismic design of the Radwaste Building is full SSE instead of 1/2 SSE as shown in Table 2 of RG 1.143. The tornado wind loads follow Table 2.0-1 of the DCD Tier 2. Classification RW-IIa SSCs housed within the building are designed to 1/2 SSE.

Table 3.8-10 Temperatures During Operating Conditions (RB)

Region	Summer Operation	Winter Operation
RB rooms outside containment	40°C	10°C
Main steam tunnel	57°C	57°C
IC/PCCS pools (including expansion pools) Reactor Well Equipment Storage pool Fuel Buffer pool	43°C	43°C
Exterior	46.1°C †	-40.0°C
Ground	15.5°C	15.5°C

† steady state; 47.2°C allowed for short duration.

SI to U.S. Customary units conversion (SI units are the controlling units and U.S. Customary units are for reference only):

1°C = (°F - 32)/1.8

Table 3.8-11 Temperatures During Operating Conditions (CB)

Region	Summer Operation	Winter Operation
Main control room DCIS room	21°C	21°C
HVAC room	30°C	10°C
Exterior	46.1°C †	-40.0°C
Ground	15.5°C	15.5°C

† *steady state; 47.2°C allowed for short duration.*

SI to U.S. Customary units conversion (SI units are the controlling units and U.S. Customary units are for reference only):

1°C = (°F - 32)/1.8

Table 3.8-12 Temperatures During Operating Conditions (FB)

Region	Summer Operation	Winter Operation
Room	*40°C*	*10°C*
Spent fuel pool	*48.9°C*	*48.9°C*
Exterior	*46.1°C †*	*-40.0°C*
Ground	*15.5°C*	*15.5°C*

† steady state; 47.2°C allowed for short duration.

SI to U.S. Customary units conversion (SI units are the controlling units and U.S. Customary units are for reference only):

1°C = (°F - 32)/1.8

Table 3.8-13 *Key Dimensions of Foundations*

Building	Dimension	Notes
Reactor Building *Fuel Building*	*Plan 70.0 m × 49.0 m*	*A common foundation of RB and FB*
	Thickness = 4.0 m	*The thickness is increased to 5.1 m at the containment portion, and 5.5 m at the spent fuel pool portion.*
	Top of foundation = 16 m below grade	
Control Building	*Plan 30.3 m × 23.8 m*	
	Thickness = 3.0 m	
	Top of foundation = 11.9 m below grade	
Firewater Service Complex	*Plan 52 m x 20 m*	*A common foundation of FWS and FPE*
	Thickness = 2.5 m	
	Top of foundation = 0.15 m above grade	

SI to U.S. Customary units conversion (SI units are the controlling units and U.S. Customary units are for reference only):

1 m = 3.28 ft

Table 3.8-14 Load Combinations and Factor of Safety for Foundation Design

Load Combination		Overturning	Sliding	Flotation
1	D + H + W	1.5	1.5	--
2	D + H + E’	1.1	1.1	--
3	D + H + W_t	1.1	1.1	--
4	D + F’	--	--	1.1

Nomenclature:

D: Dead Load

H: Lateral Earth Pressure

W: Wind Load

E’: Basic SSE Seismic Load

W_t: Tornado Wind

F’: Buoyant force of the design basis flood

Table 3.8-15 Load Combinations, Load Factors and Acceptance Criteria for the Safety-Related Reinforced Concrete Structures[(1),(2),(3)]

		Load Combination													Acceptance Criteria[(5)]
Category	No.	D	F	L[(6)]	H	P_a	T_o	T_a	E'	W	W_t	R_o	R_a	Y[(4)]	
Normal	1	1.4	1.4	1.7	1.7							1.7			U
	2	1.05	1.05	1.3	1.3		1.3					1.3			U
Severe	3	1.4	1.4	1.7	1.7					1.7		1.7			U
Environmental	4	1.05	1.05	1.3	1.3		1.3			1.3		1.3			U
	5	1.2	1.2							1.7					U
Extreme	6	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	1.0	1.0	1.5		1.0					1.0		U
Abnormal/Extreme Environmental	9	1.0	1.0	1.0	1.0	1.0		1.0	1.0				1.0	1.0	U

(1) The loads are described in Subsection 3.8.4.3 and acceptance criteria in Subsection 3.8.4.5. The effects of SRV and LOCA dynamic loads that originate inside the containment are considered as applicable.

(2) For any load combination, where any load reduces the effects of other loads, the corresponding coefficient for that load is taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the coefficient for that load is taken as zero.

(3) Because P_a and T_a are time-dependent loads, their effects are superimposed accordingly.

(4) Y includes Y_j, Y_m and Y_r. The maximum value of Y including an appropriate DLF is used, unless an appropriate time history analysis is performed to justify otherwise.

(5) U = Required section strength based on the strength design method per ACI 349-01.

(6) The normal winter precipitation roof load is considered as a normal live load for all load combinations. The extreme winter precipitation roof load is considered as an extreme live load for the extreme environmental and abnormal/extreme environmental load combinations without concurrent seismic or tornado loads.

Table 3.8-16 Load Combinations, Load Factors and Acceptance Criteria for the Safety-Related Steel Structures(1),(2),(3)

Category	Load Combination												Acceptance Criteria(5)
	No.	$D^{(6)}$	$L^{(7)}$	P_a	T_o	T_a	E'	W	W_t	R_o	R_a	$Y^{(4)}$	
Normal	1	1.0	1.0										S
	2	1.0	1.0		1.0					1.0			S (a)
Severe	3	1.0	1.0					1.0					S
Environmental	4	1.0	1.0		1.0			1.0		1.0			S (a)
Extreme	5	1.0	1.0		1.0		1.0			1.0			1.6S (b)(c)
Environmental	6	1.0	1.0		1.0				1.0	1.0			1.6S (b)(c)
Abnormal	7	1.0	1.0	1.0		1.0					1.0		1.6S (b)(c)
Abnormal/Extreme Environmental	8	1.0	1.0	1.0		1.0	1.0				1.0	1.0	1.7S (b)(c)

(1) The loads are described in Subsection 3.8.4.3 and acceptance criteria in Subsection 3.8.4.5. The effects of SRV and LOCA dynamic loads that originate inside the containment are considered as applicable.

(2) For any load combination, where any load reduces the effects of other loads, the corresponding coefficient for that load is taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the coefficient for that load is taken as zero.

(3) Because P_a and T_a are time-dependent loads, their effects are superimposed accordingly.

(4) Y includes Y_j, Y_m and Y_r. The maximum values of Y including an appropriate DLF are used, unless an appropriate time history analysis is performed to justify otherwise.

(5) Allowable elastic working stress (S) is the allowable stress limit specified in Part 1 of AISC N690-1994-s2 (2004).

(a) For primary plus secondary stress, the allowable limits are increased by a factor of 1.5.

(b) Stress limit coefficient in shear does not exceed 1.4 in members and bolts.

(c) Stress limit coefficient, where axial compression exceeds 20% of nominal allowable, is 1.5 for load combinations 5, 6, and 7, and 1.6 for load combination 8.

(6) Dead Load includes settlements.

(7) *The normal winter precipitation roof load is considered as a normal live load for all load combinations. The extreme winter precipitation roof load is considered as an extreme live load for the extreme environmental and abnormal/extreme environmental load combinations without concurrent seismic or tornado loads.*

Table 3.8-17 PCCS Passages Through RCCV Top Slab

Passage Number	Description	RCCV Sector
0001	Condenser Steam Inlet Line A	I
0007	Condenser Condensate + Vent Line A1	I
0008	Condenser Condensate + Vent Line A2	I
0002	Condenser Steam Inlet Line B	I/III
0009	Condenser Condensate + Vent Line B1	I/III
0010	Condenser Condensate + Vent Line B2	I/III
0003	Condenser Steam Inlet Line C	III
0011	Condenser Condensate + Vent Line C1	III
0012	Condenser Condensate + Vent Line C2	III
0004	Condenser Steam Inlet Line D	II
0013	Condenser Condensate + Vent Line D1	II
0014	Condenser Condensate + Vent Line D2	II
0005	Condenser Steam Inlet Line E	II/IV
0015	Condenser Condensate + Vent Line E1	II/IV
0016	Condenser Condensate + Vent Line E2	II/IV
0006	Condenser Steam Inlet Line F	IV
0017	Condenser Condensate + Vent Line F1	IV
0018	Condenser Condensate + Vent Line F2	IV

Notes:

1. All PCCS Passages are located in the RCCV Top Slab.

Table 3.8-18 Temperatures During Operating Conditions (FWSC)

Region	Summer Operation	Winter Operation
Water & Air in FWS	43.0°C	4.5°C
FPE Interior	26.7°C	4.5°C
Exterior	46.1°C	-40.0°C
Ground	15.5°C	15.5°C

SI to U.S. Customary units conversion (SI units are the controlling units and U.S. Customary units are for reference only):

1°C = (°F - 32)/1.8

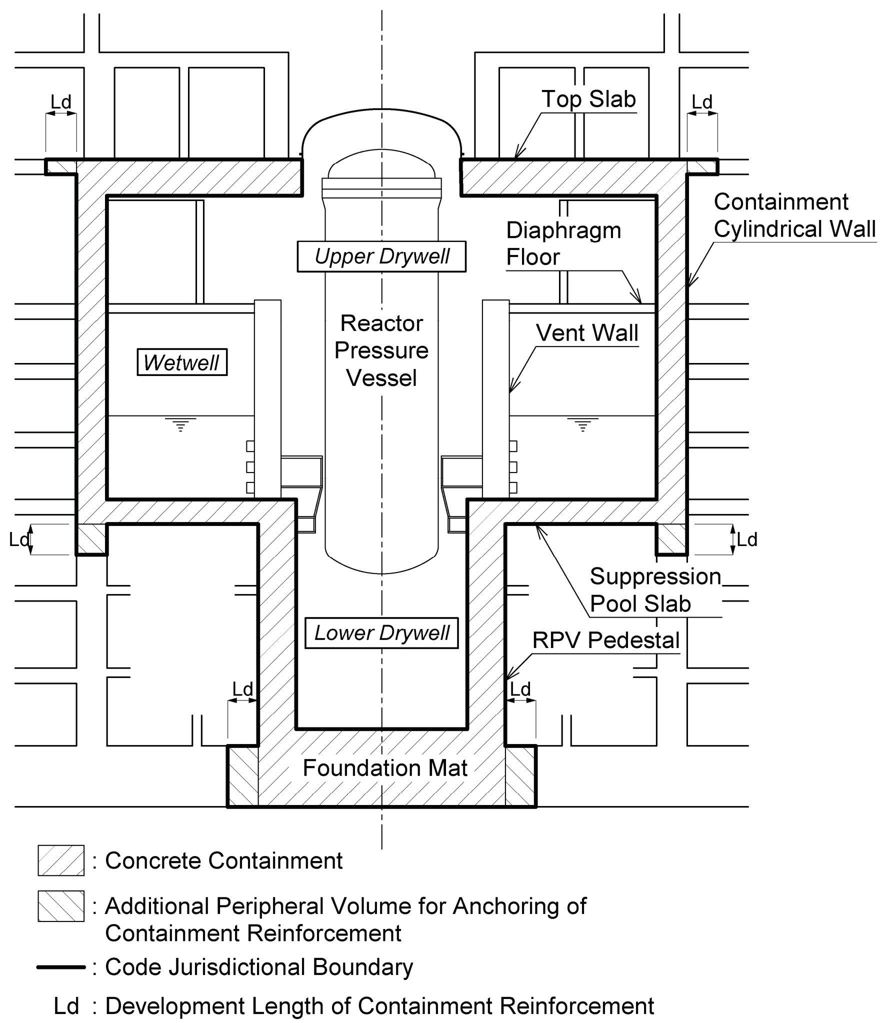


Figure 3.8-1. Configuration of Concrete Containment

Figure 3.8-2 Schematic of Reinforcements in RCCV Wall Around Equipment Hatch/Personnel Airlock Opening

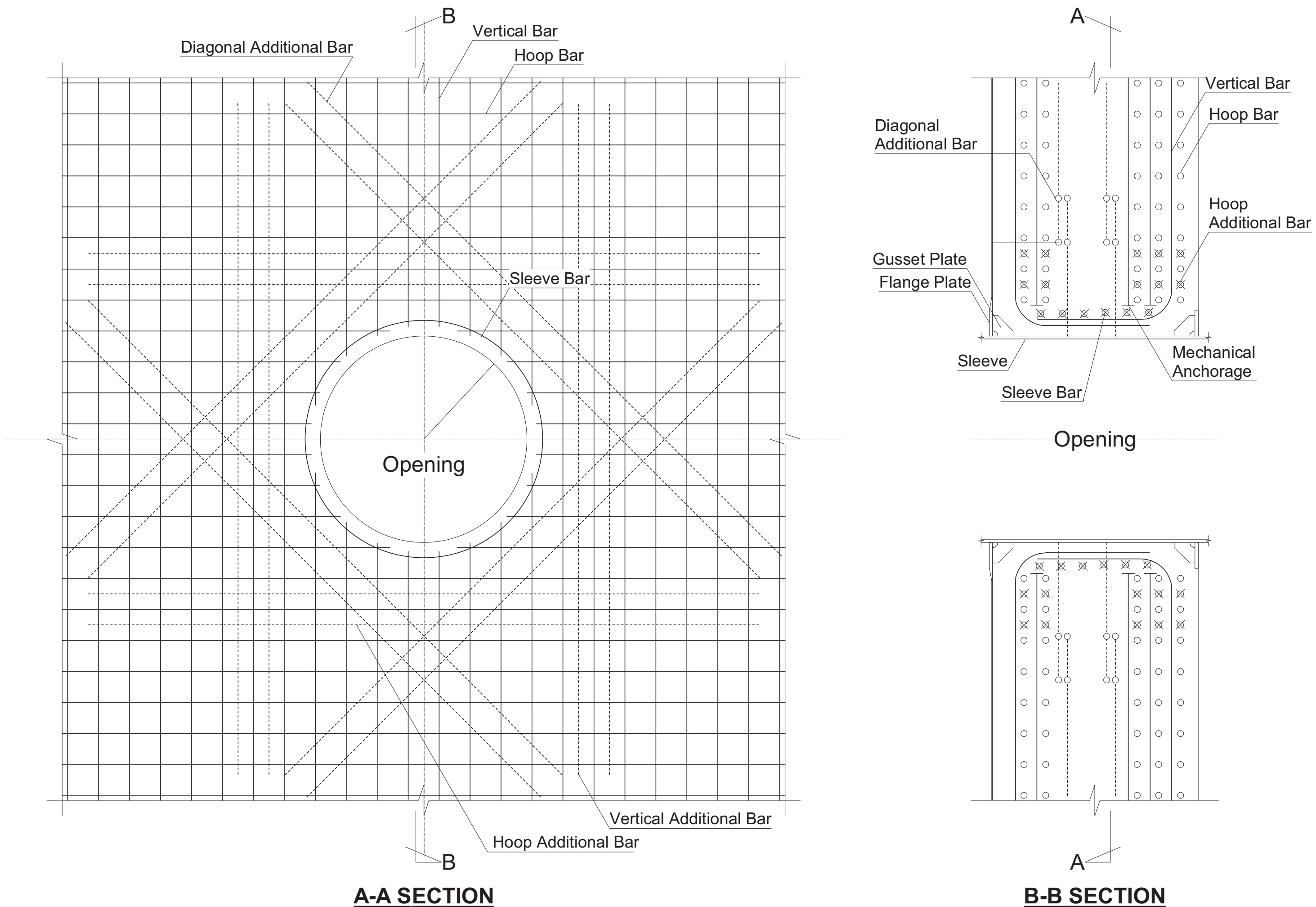


Figure 3.8-3 Typical Internal Containment Plate Support with Embedment Integral with Containment Liner

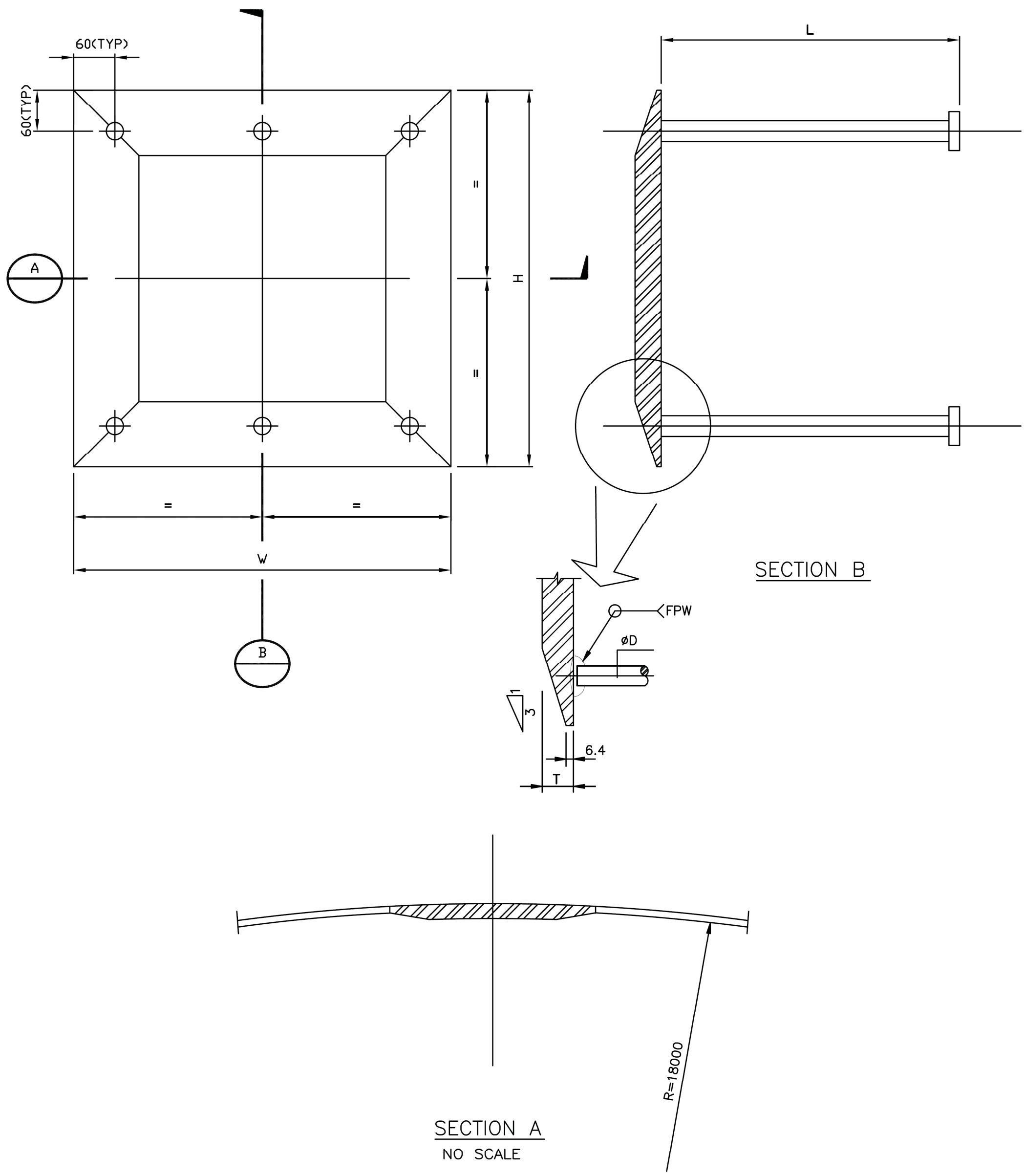


Figure 3.8-4 Typical External Containment Plate Support with Embedment

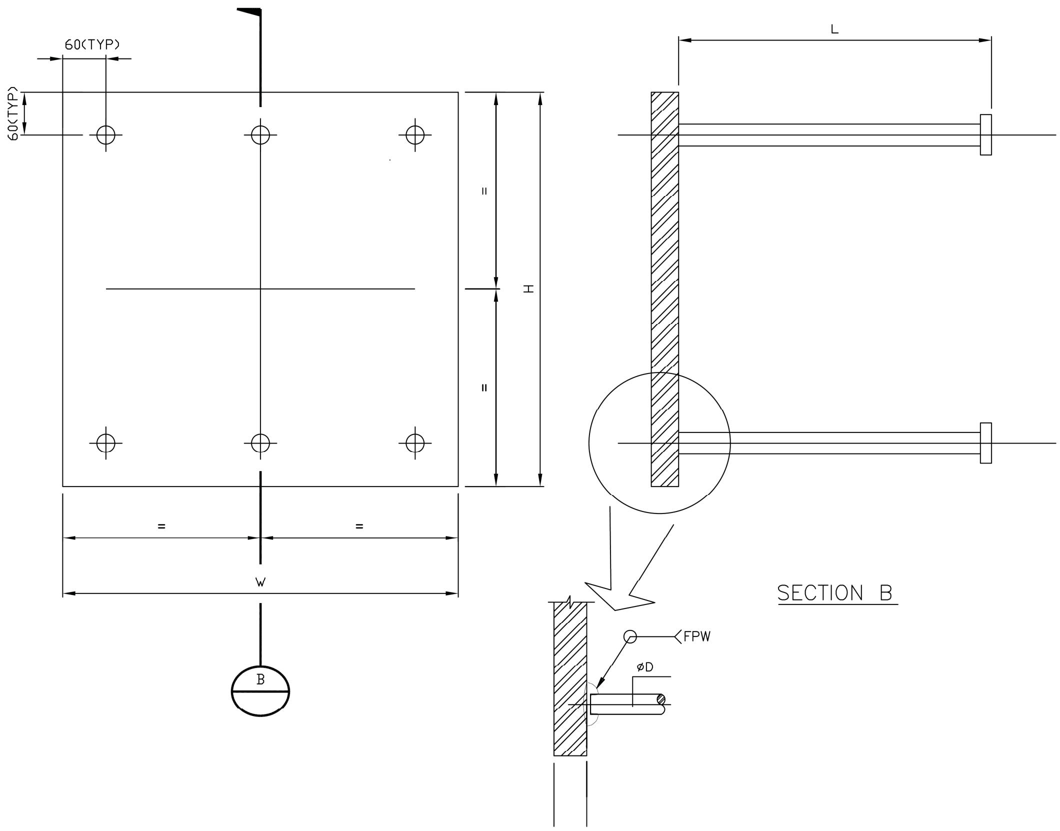


Figure 3.8-5 Quencher Anchorage

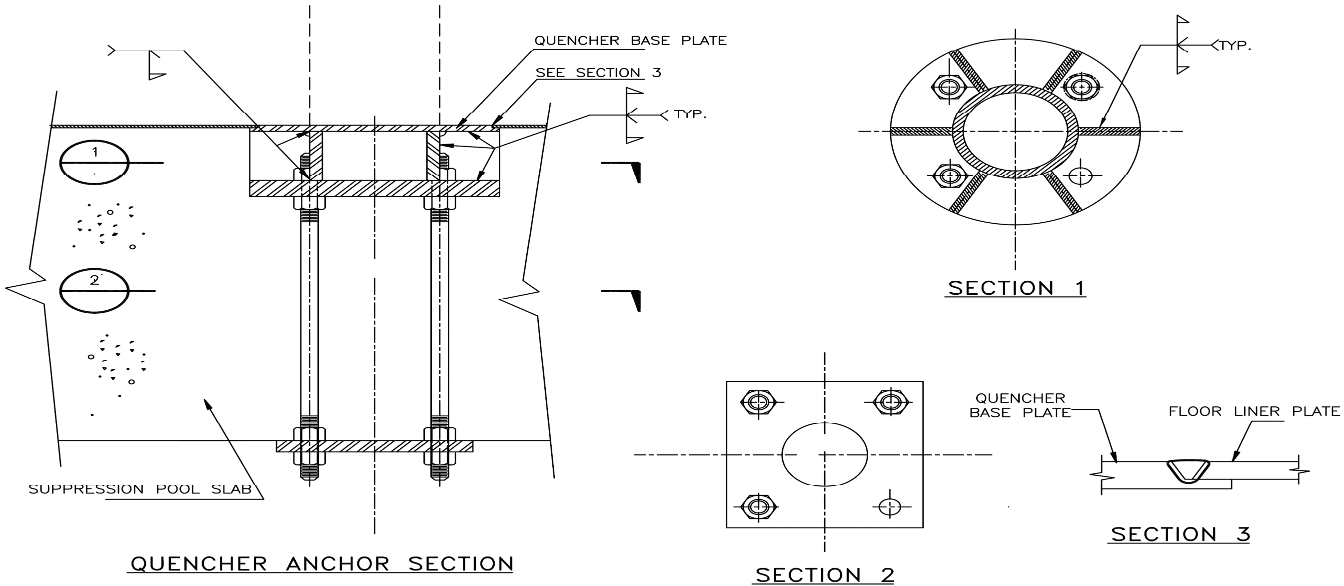


Figure 3.8-6 RCCV Wall High-Energy Penetration

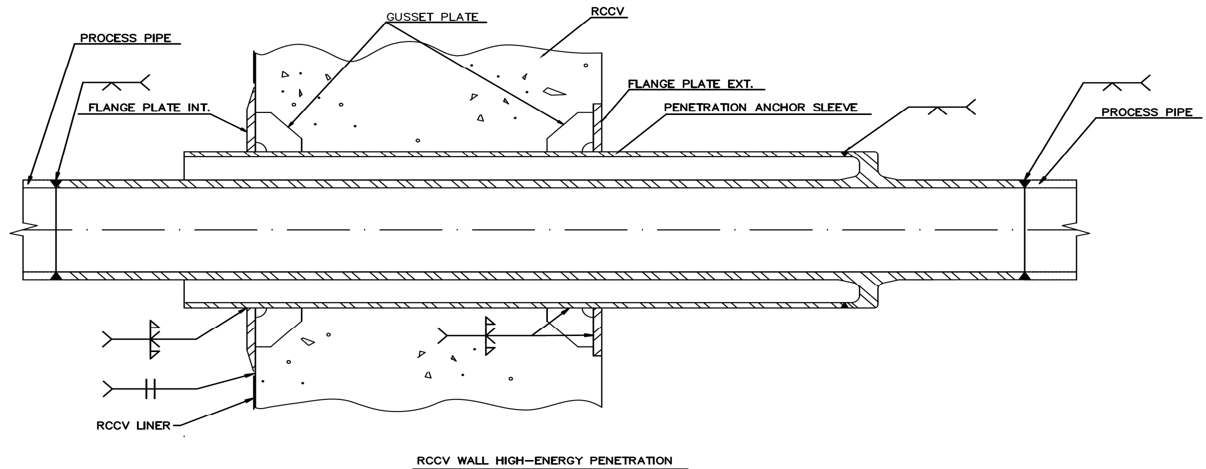


Figure 3.8-7 Typical RCCV Top Slab Penetration and PCCS Passages

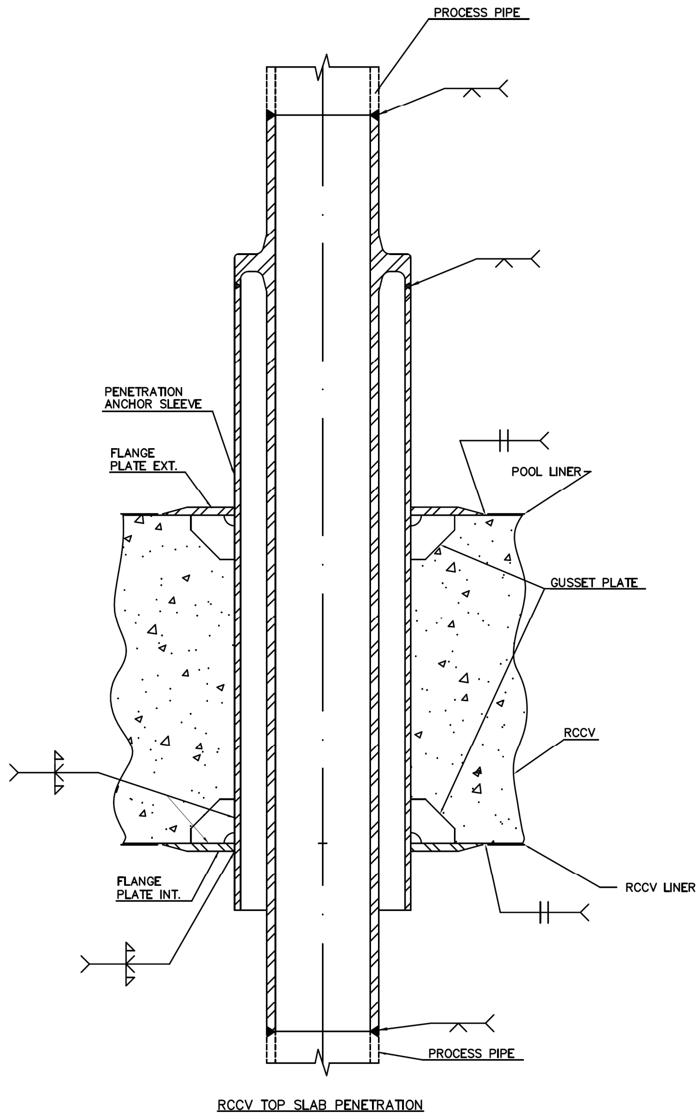


Figure 3.8-8 RCCV Low-Energy Penetration

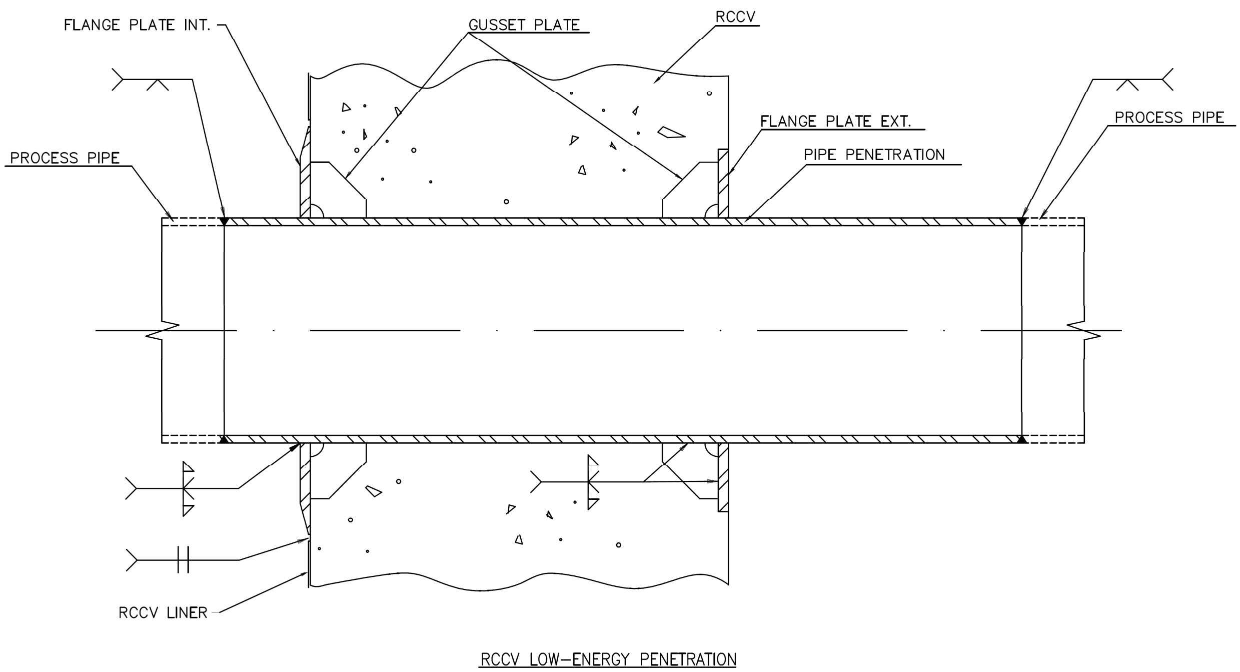


Figure 3.8-9 RCCV Multiple Penetration

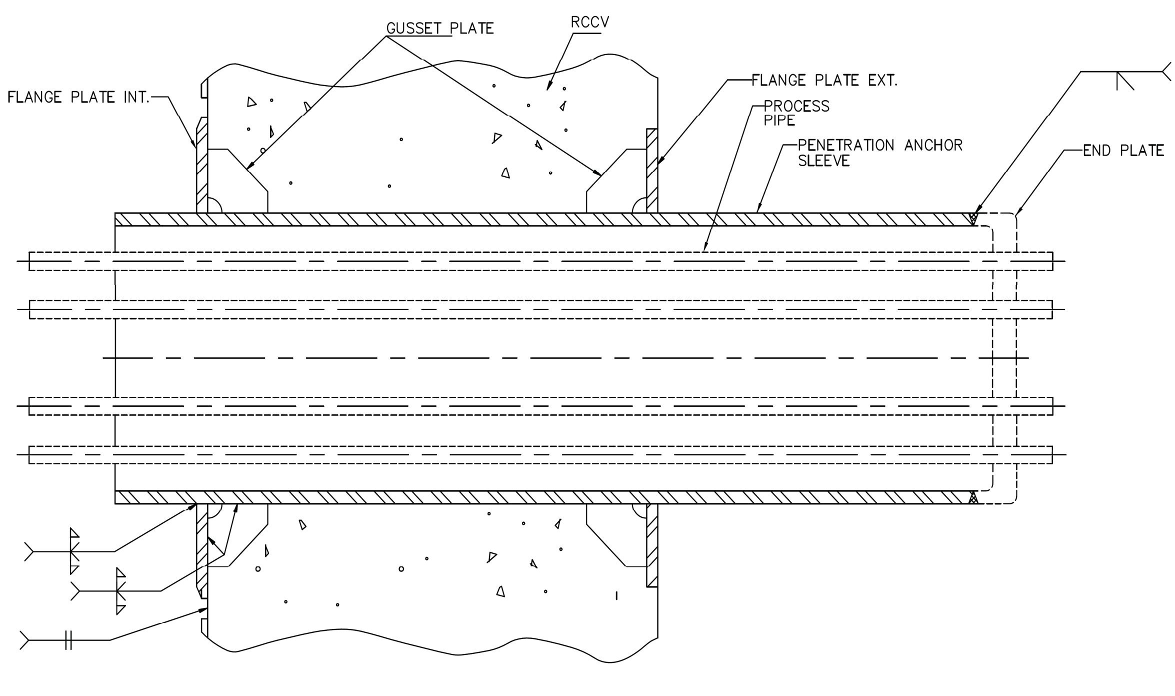


Figure 3.8-10 RCCV Electrical Penetration

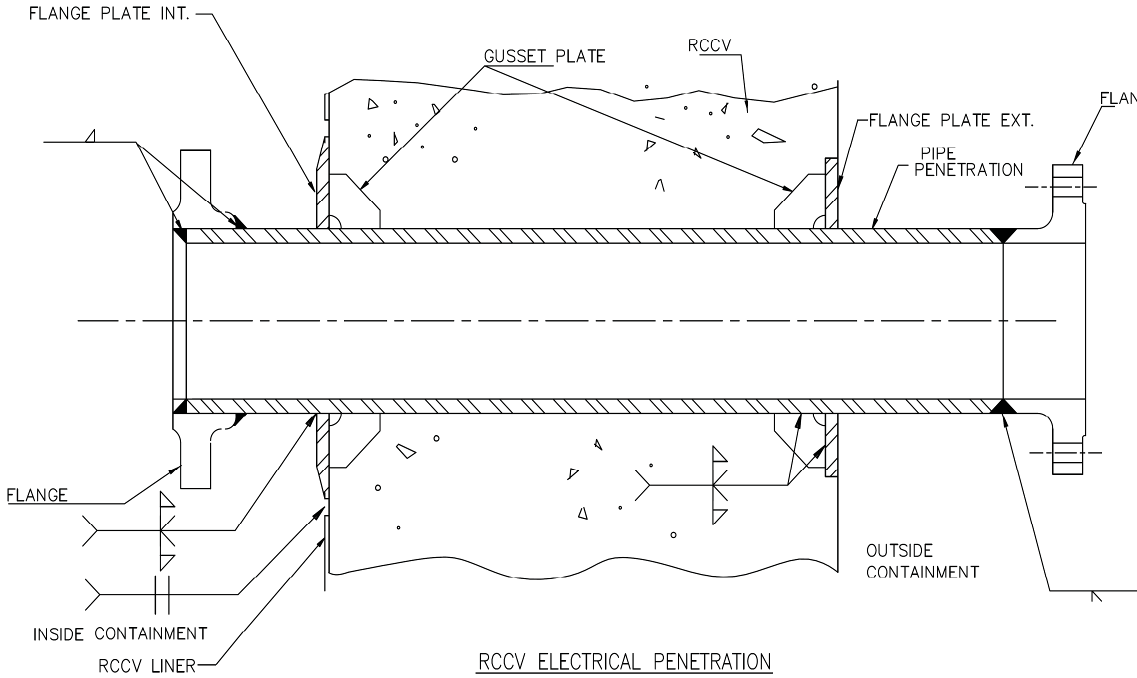


Figure 3.8-11 RCCV Spare Penetration

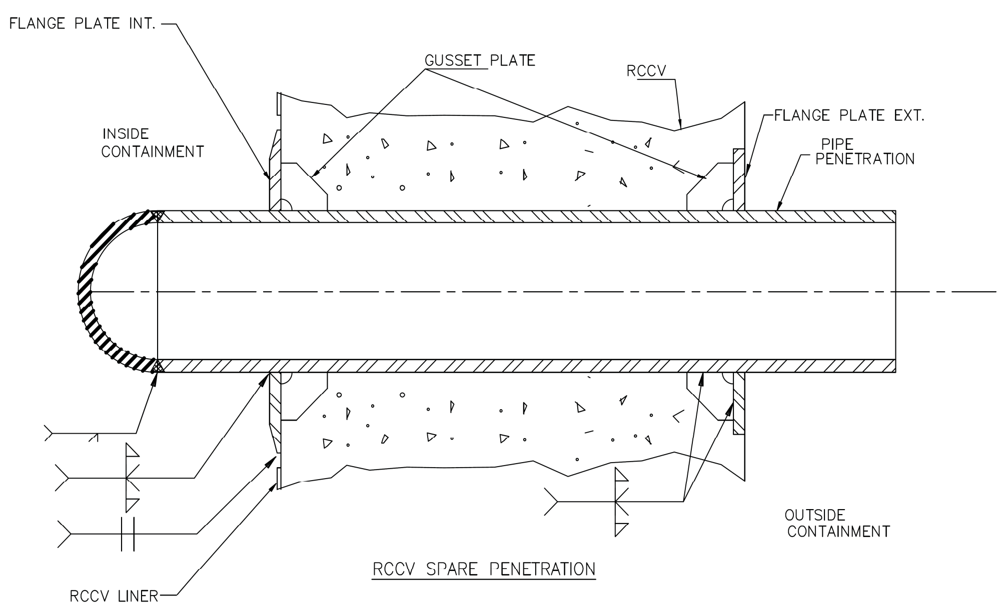


Figure 3.8-12 (Deleted)

Table 3.8.5-201 Factors of Safety for RB/FB Foundation Stability

	Overturning		Sliding		Flotation	
Load Combination	**SRP 3.8.5 Minimum FS**	**Calculated FS**	**SRP 3.8.5 Minimum FS**	**Calculated FS**	**SRP 3.8.5 Minimum FS**	**Calculated FS**
D+H+E'	1.1	2,262	1.1	1.22	--	--
D+F'	--	--	--	--	1.1	3.50

Where,

D=Dead Load

H=Lateral Soil Pressure

E'=Safe Shutdown Earthquake

F'=Buoyant forces of design basis flood

FS=Factor of Safety

Table 3.8.5-202 Factors of Safety for CB Foundation Stability

	Overturning		Sliding		Flotation	
Load Combination	SRP 3.8.5 Minimum FS	Calculated FS	SRP 3.8.5 Minimum FS	Calculated FS	SRP 3.8.5 Minimum FS	Calculated FS
D+H+E'	1.1	1,733	1.1	1.10	--	--
D+F'	--	--	--	--	1.1	1.86

Where,
D=Dead Load
H=Lateral Soil Pressure
E'=Safe Shutdown Earthquake
F'=Buoyant forces of design basis flood
FS=Factor of Safety

Table 3.8.5-203 Maximum Soil Dynamic Bearing Pressure Demand for RB/FB and CB

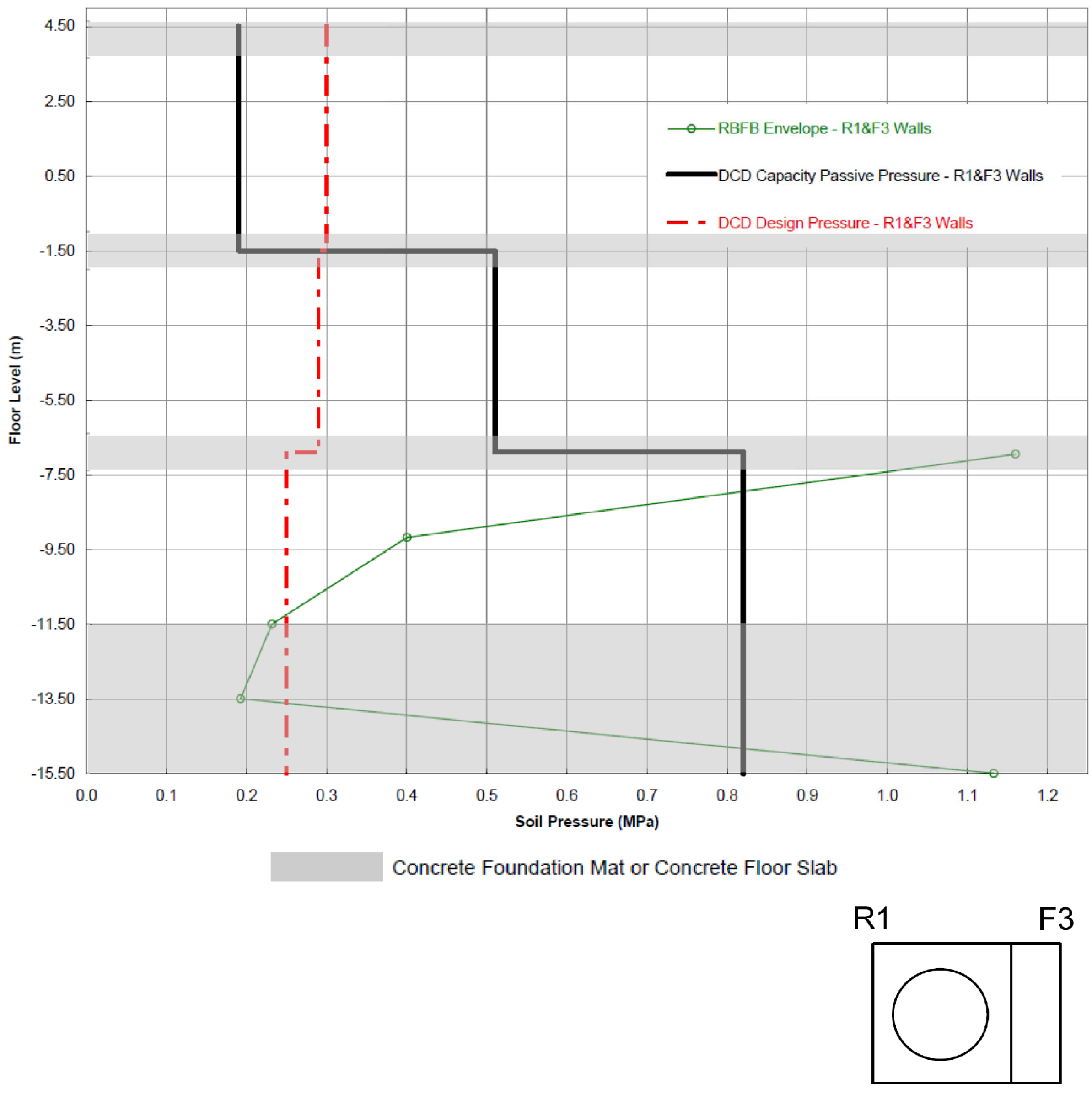
	Dynamic Bearing Pressure Demand	
	RB/FB	CB
Subsurface Condition	Fermi 3 Site-Specific SSI (Static + FIRS[(1)])	Fermi 3 Site-Specific SSI (Static + FIRS[(1)])
Fermi 3 Lower Bound Subsurface Profile	1,913 KPa ($39,954\ lbf/ft^2$)	791 KPa ($16,520\ lbf/ft^2$)
Fermi 3 Best Estimate Subsurface Profile	1,970 KPa ($4\ 1,144\ lbf/ft^2$)	823 KPa ($17,189\ lbf/ft^2$)
Fermi 3 Upper Bound Subsurface Profile	2,053 KPa ($42,878\ lbf/ft^2$)	853 KPa ($17,815\ lbf/ft^2$)

Notes:

1. FIRS is the SSI FIRS developed in Subsection 3.7.1.

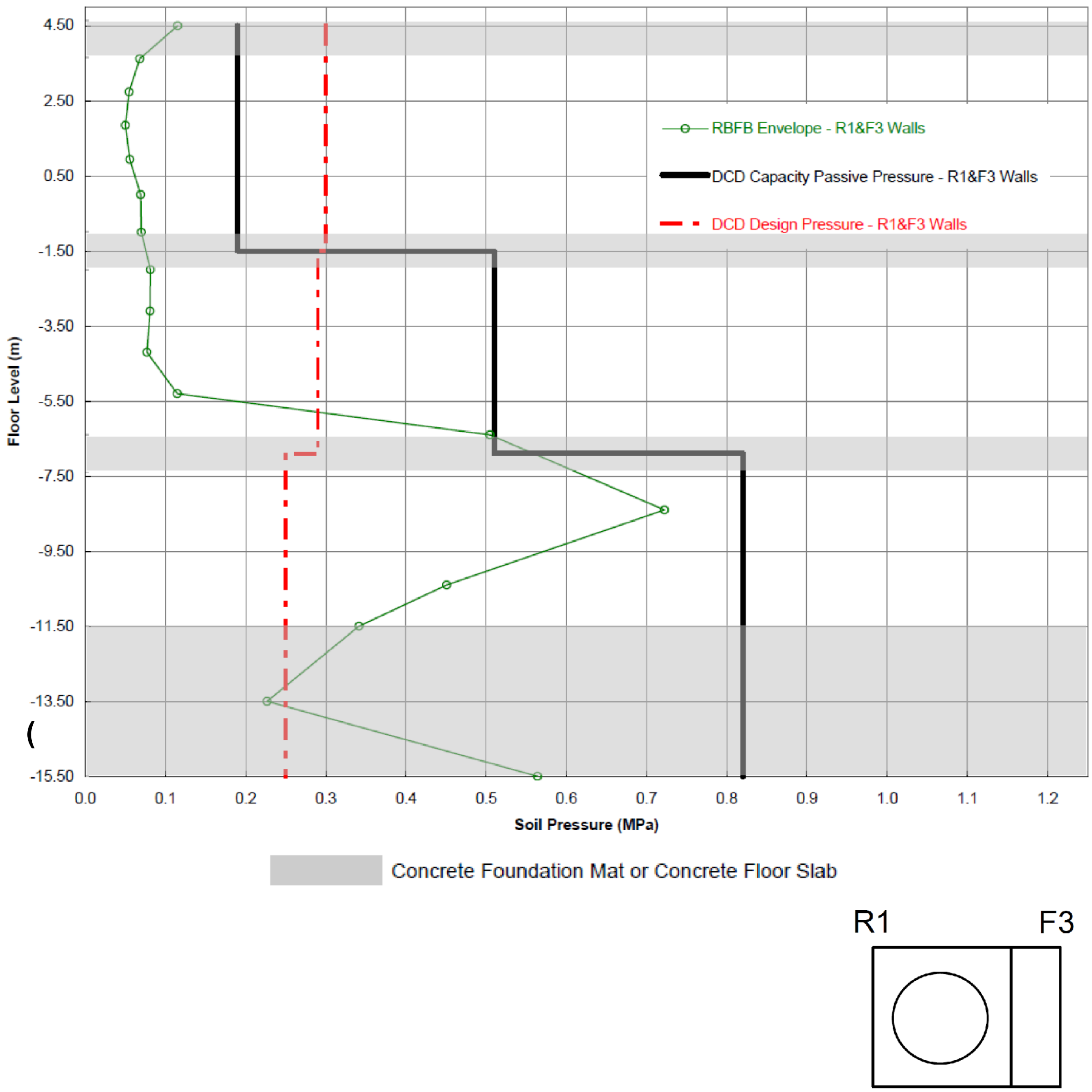
   KPa = kilopascal

Figure 3.8.4-201a SSI Lateral Soil Pressure RB/FB



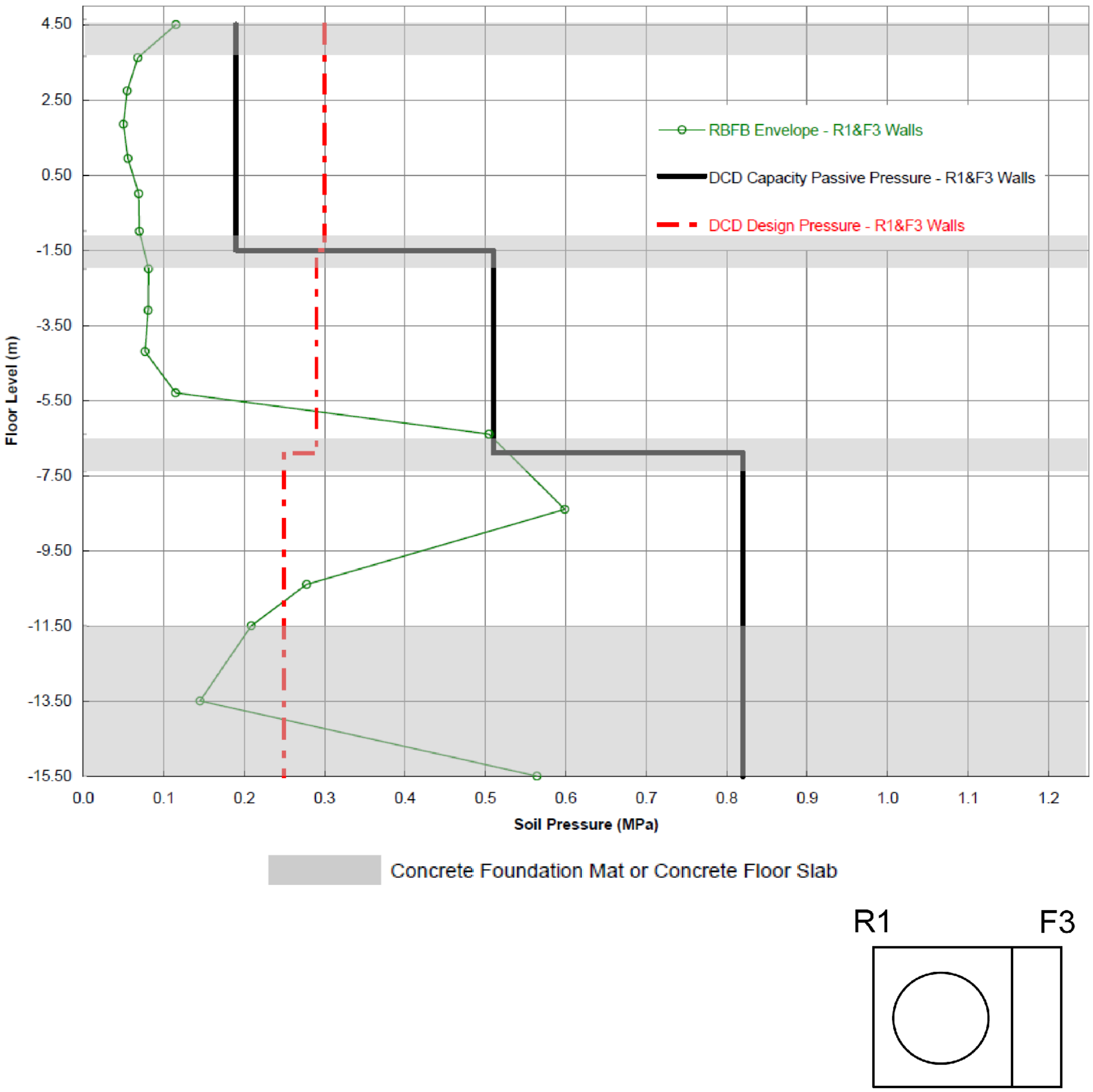
(a) Licensing Basis Envelope – Walls on Column Rows R1 and F3

Figure 3.8.4-201b SSI Lateral Soil Pressure RB/FB



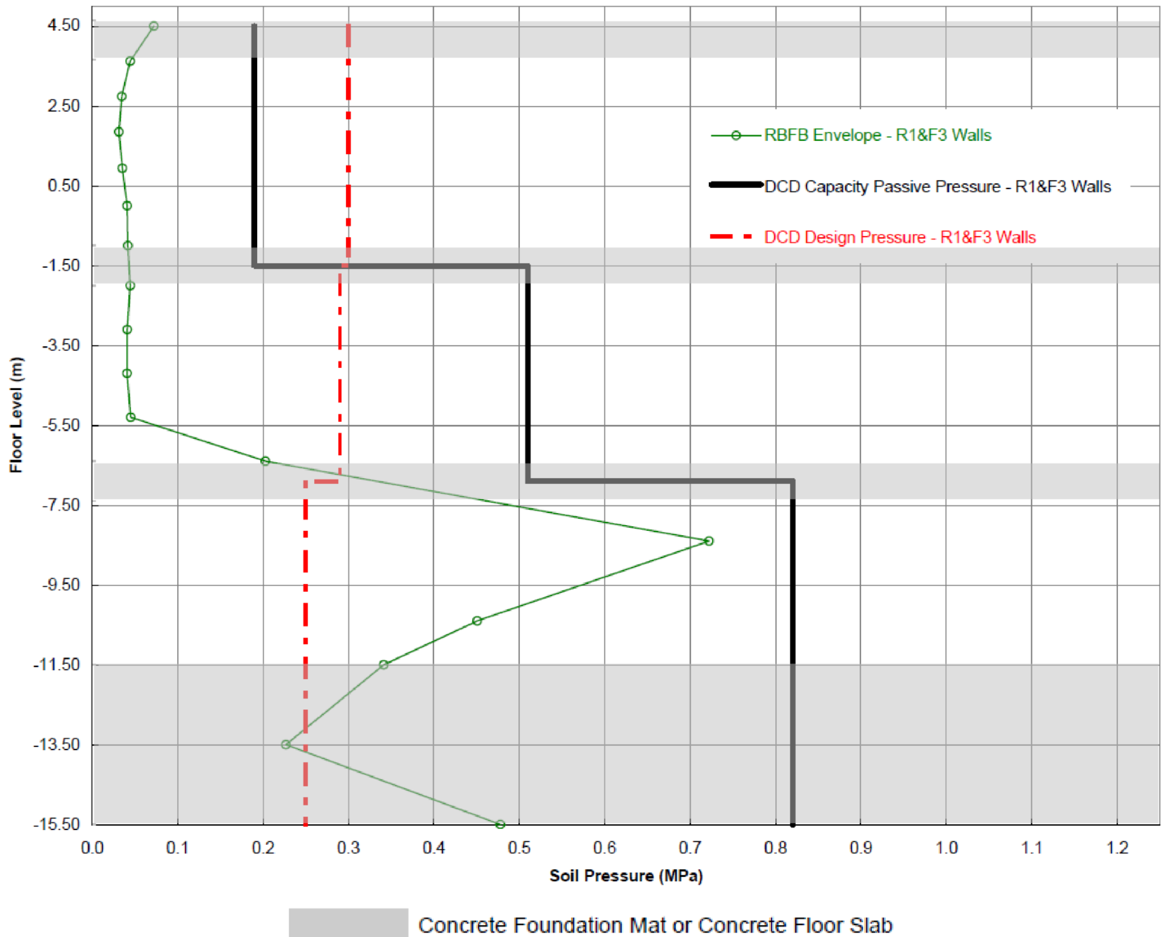
(b) Engineered Backfill Envelope – Walls on Column Rows R1 and F3

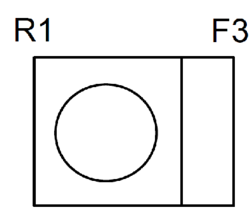
Figure 3.8.4-201c SSI Lateral Soil Pressure RB/FB



(c) Engineered Backfill – UB Rock/Soil Profile – Walls on Column Rows R1 and F3

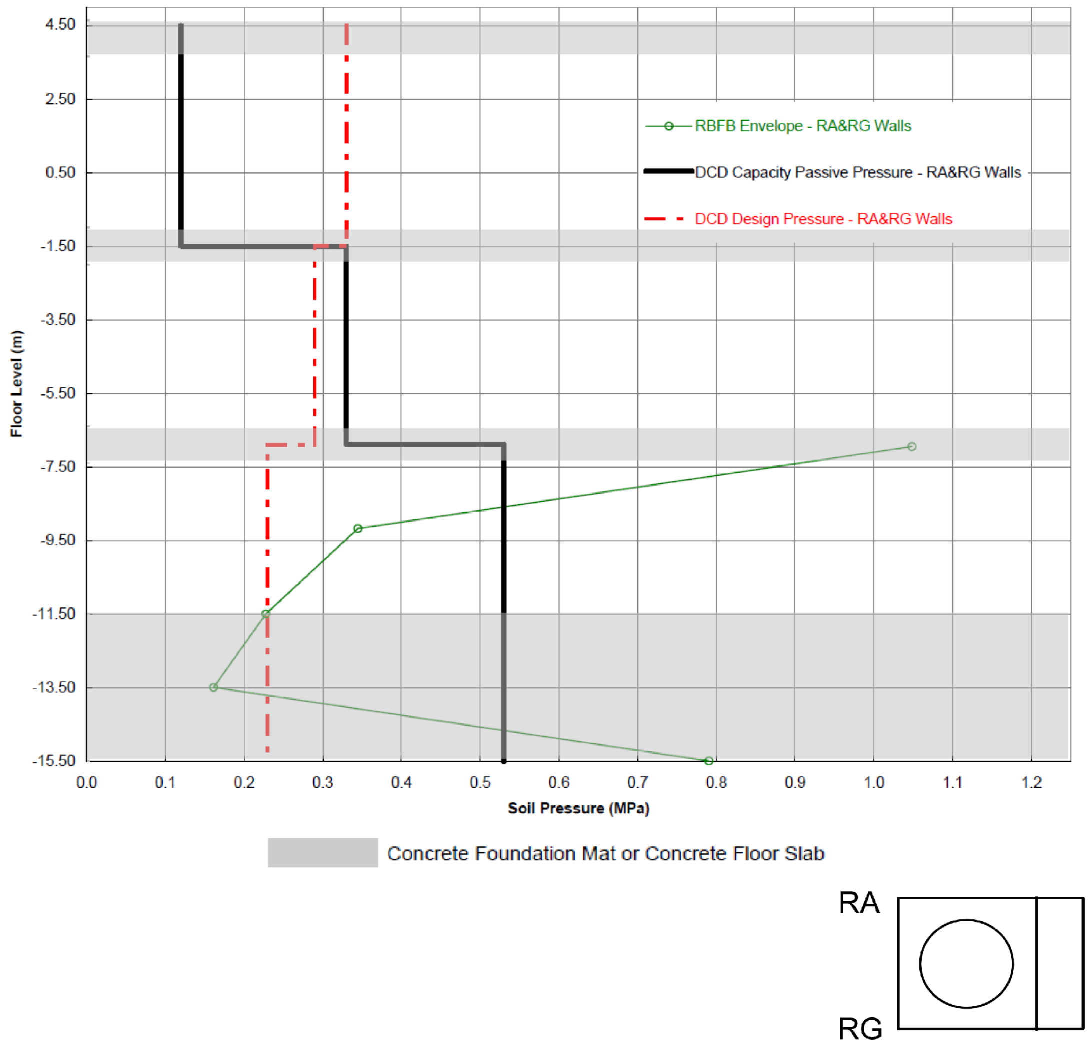
Figure 3.8.4-201d SSI Lateral Soil Pressure RB/FB





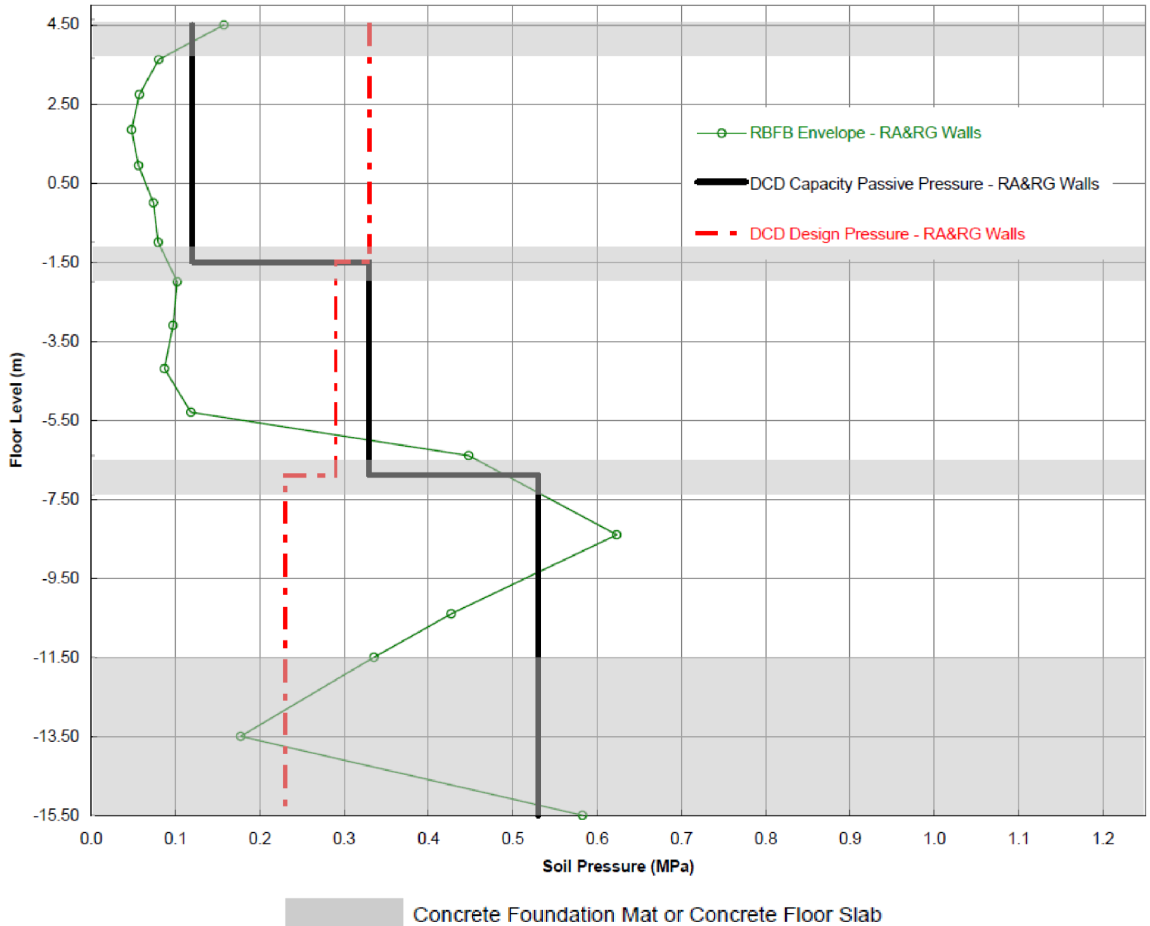
(d) Engineered Backfill – LB Rock/Soil Profile – Walls on Column Rows R1 and F3

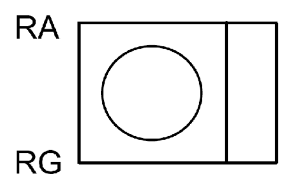
Figure 3.8.4-201e SSI Lateral Soil Pressure RB/FB



(e) Licensing Basis Envelope – Walls on Column Rows RA and RG

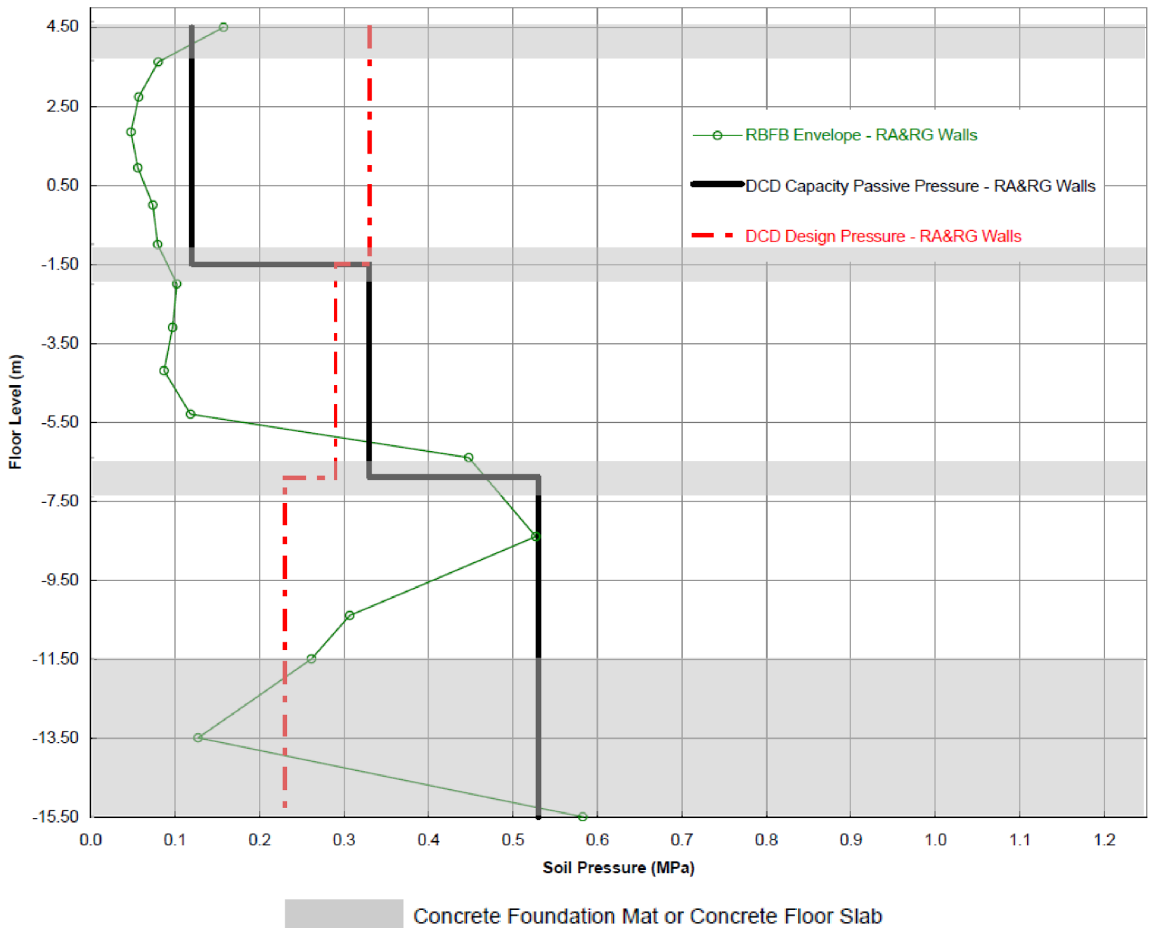
Figure 3.8.4-201f SSI Lateral Soil Pressure RB/FB

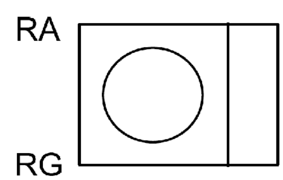




(f) Engineered Backfill Envelope – Walls on Column Rows RA and RG

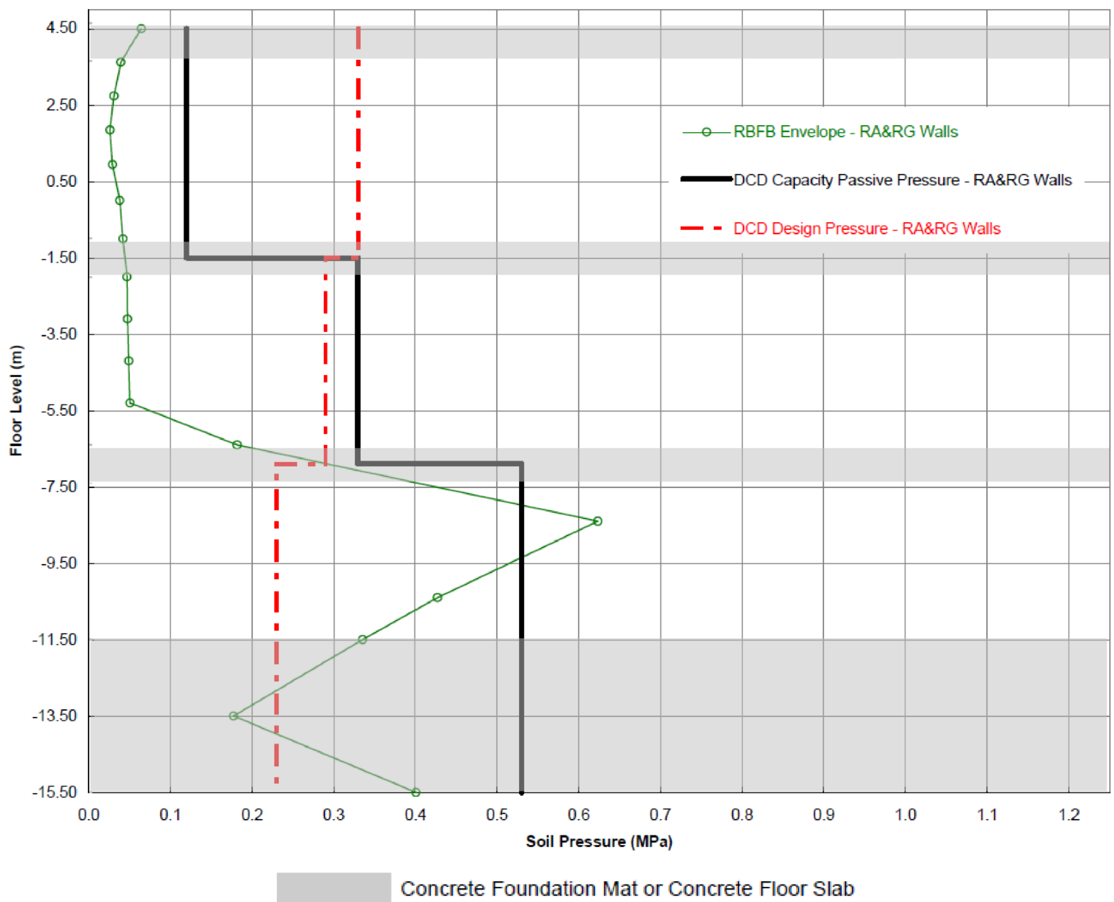
Figure 3.8.4-201g SSI Lateral Soil Pressure RB/FB

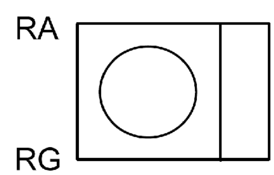




(g) Engineered Backfill – UB Rock/Soil Profile – Walls on Column Rows RA and RG

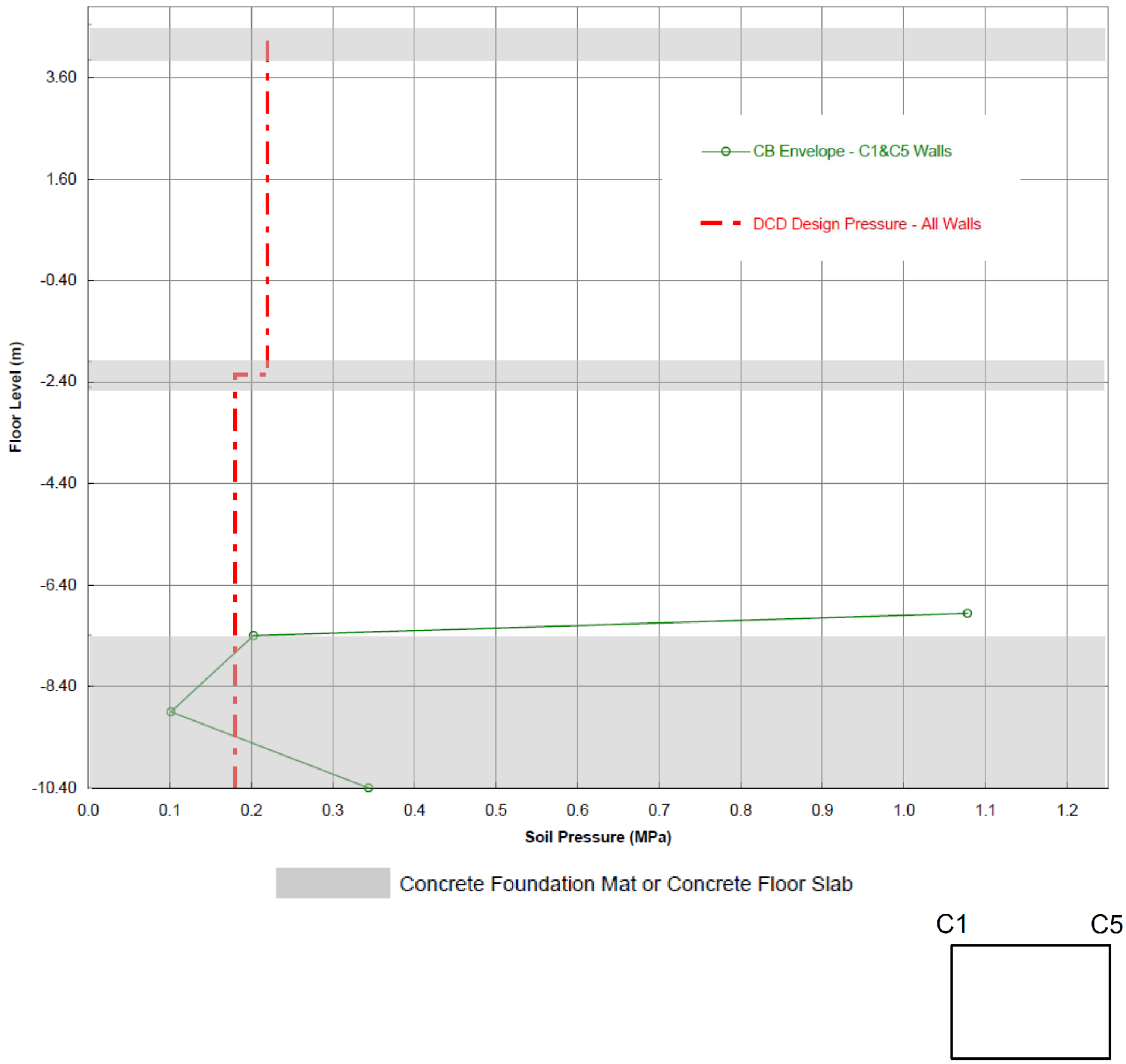
Figure 3.8.4-201h SSI Lateral Soil Pressure RB/FB





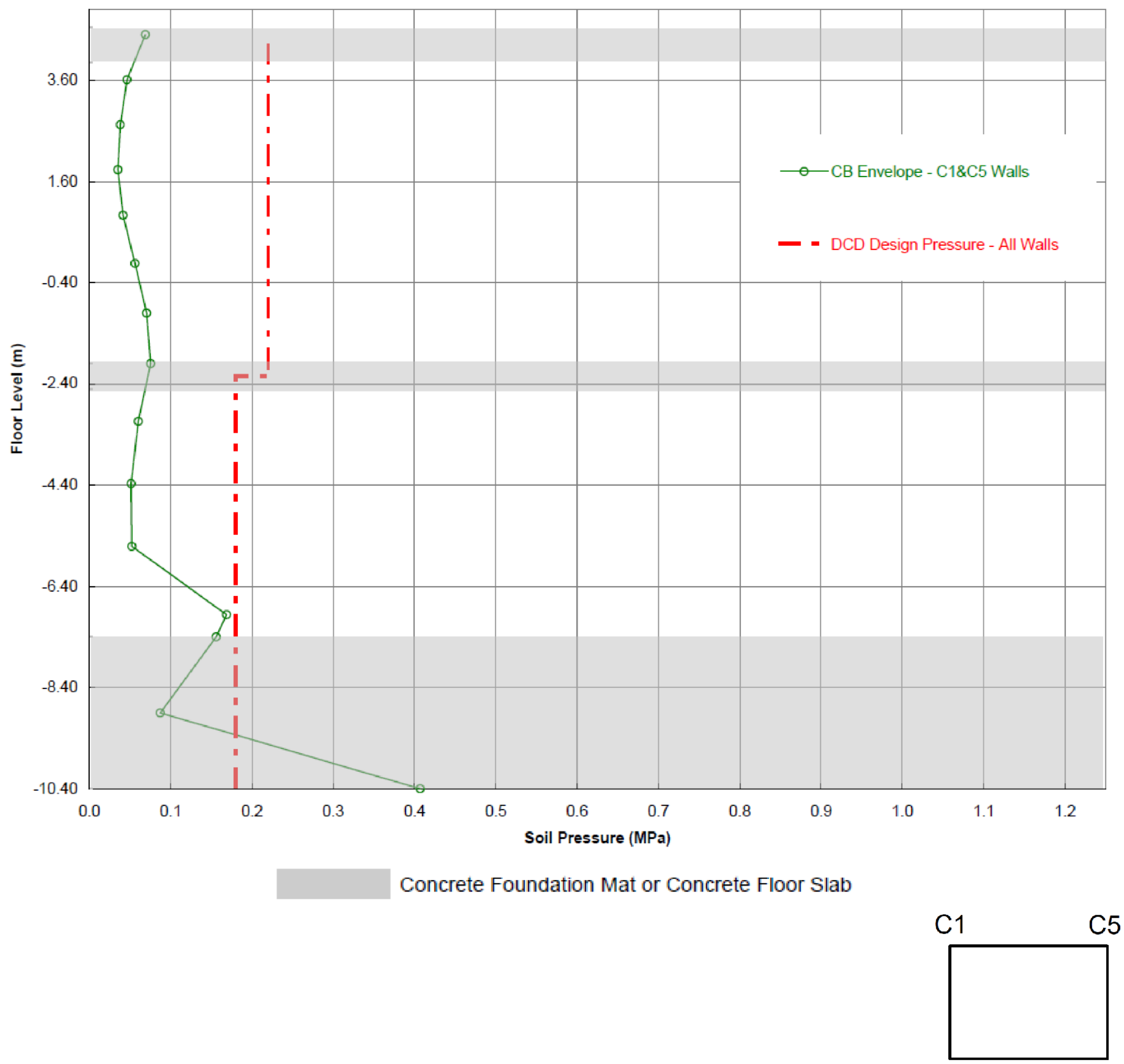
(h) Engineered Backfill – LB Rock/Soil Profile – Walls on Column Rows RA and RG

Figure 3.8.4-202a SSI Lateral Soil Pressure CB



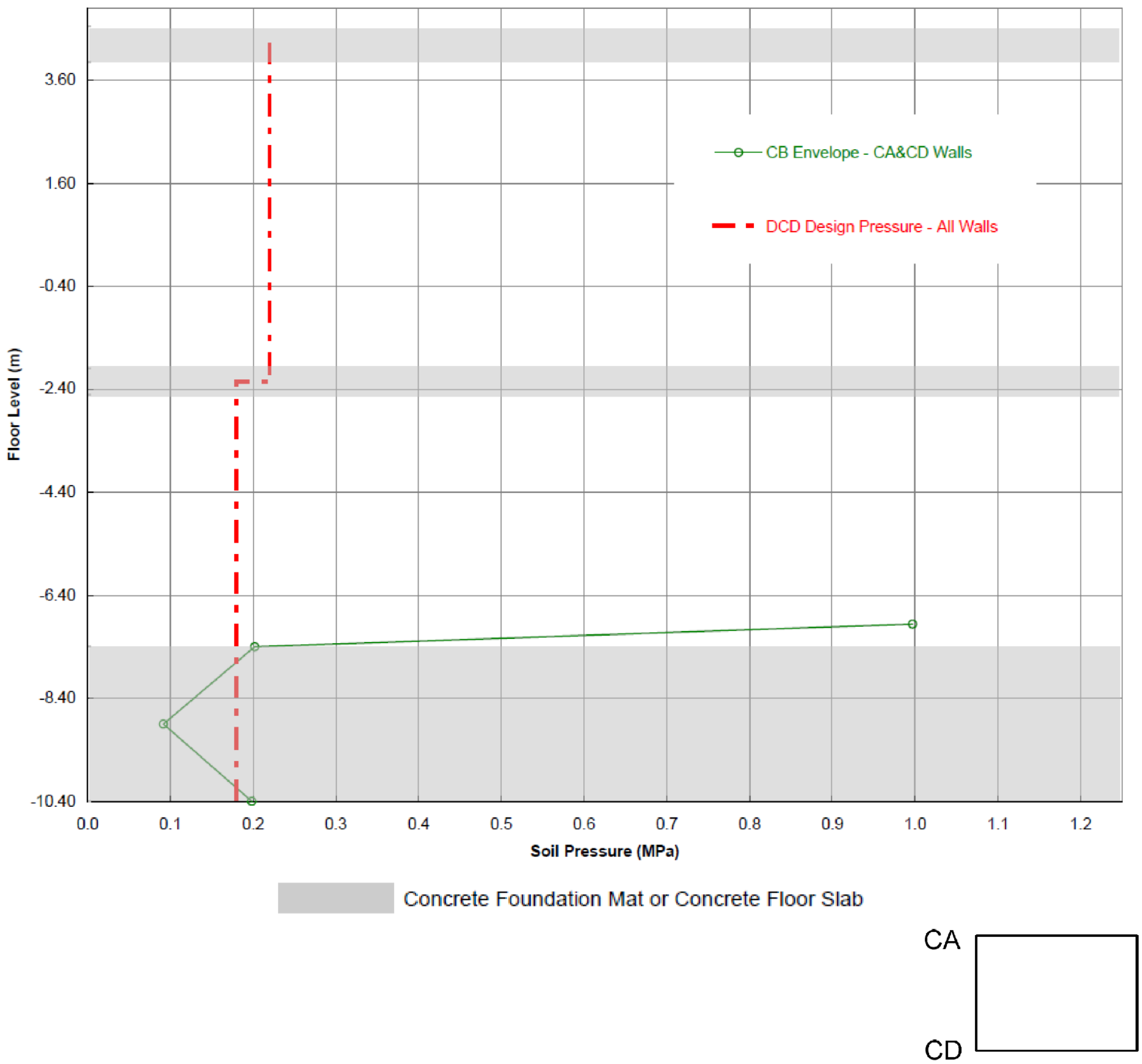
(a) Licensing Basis Envelope – Walls on Column Rows C1 and C5

Figure 3.8.4-202b SSI Lateral Soil Pressure CB



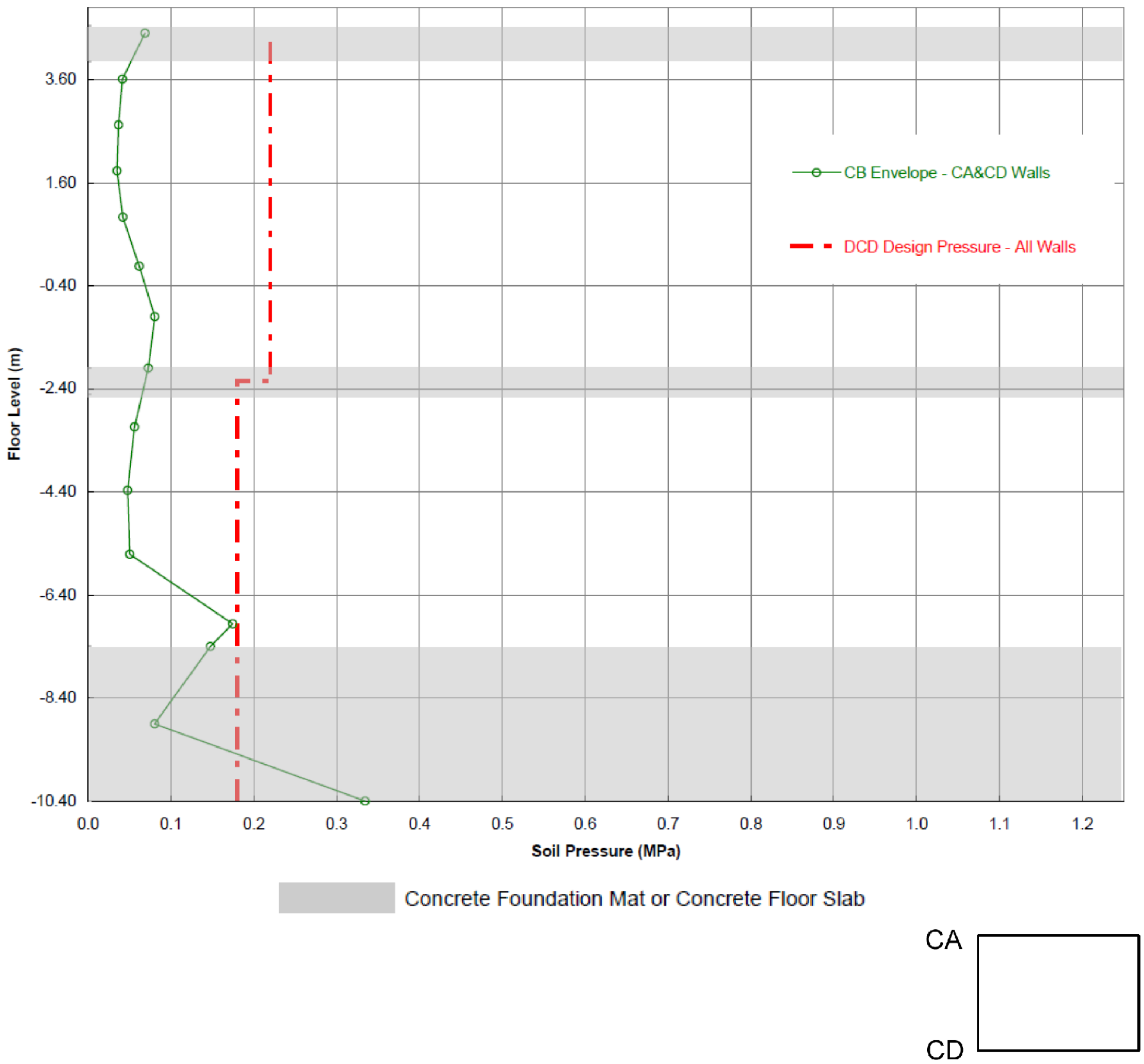
(b) Engineered Backfill Envelope – Walls on Column Rows C1 and C5

Figure 3.8.4-202c SSI Lateral Soil Pressure CB



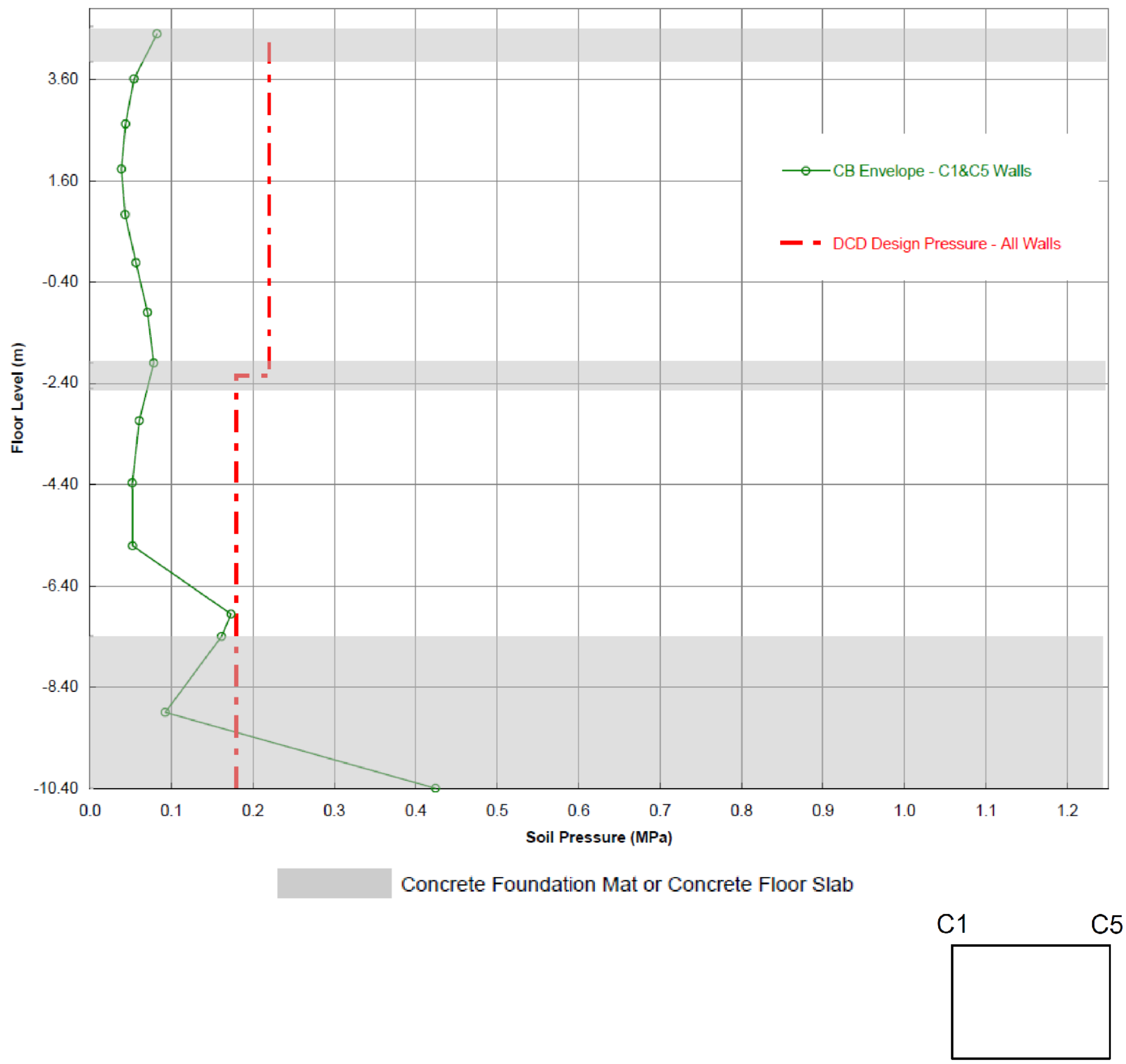
(c) Licensing Basis Envelope – Walls on Column Rows CA and CD

Figure 3.8.4-202d SSI Lateral Soil Pressure CB



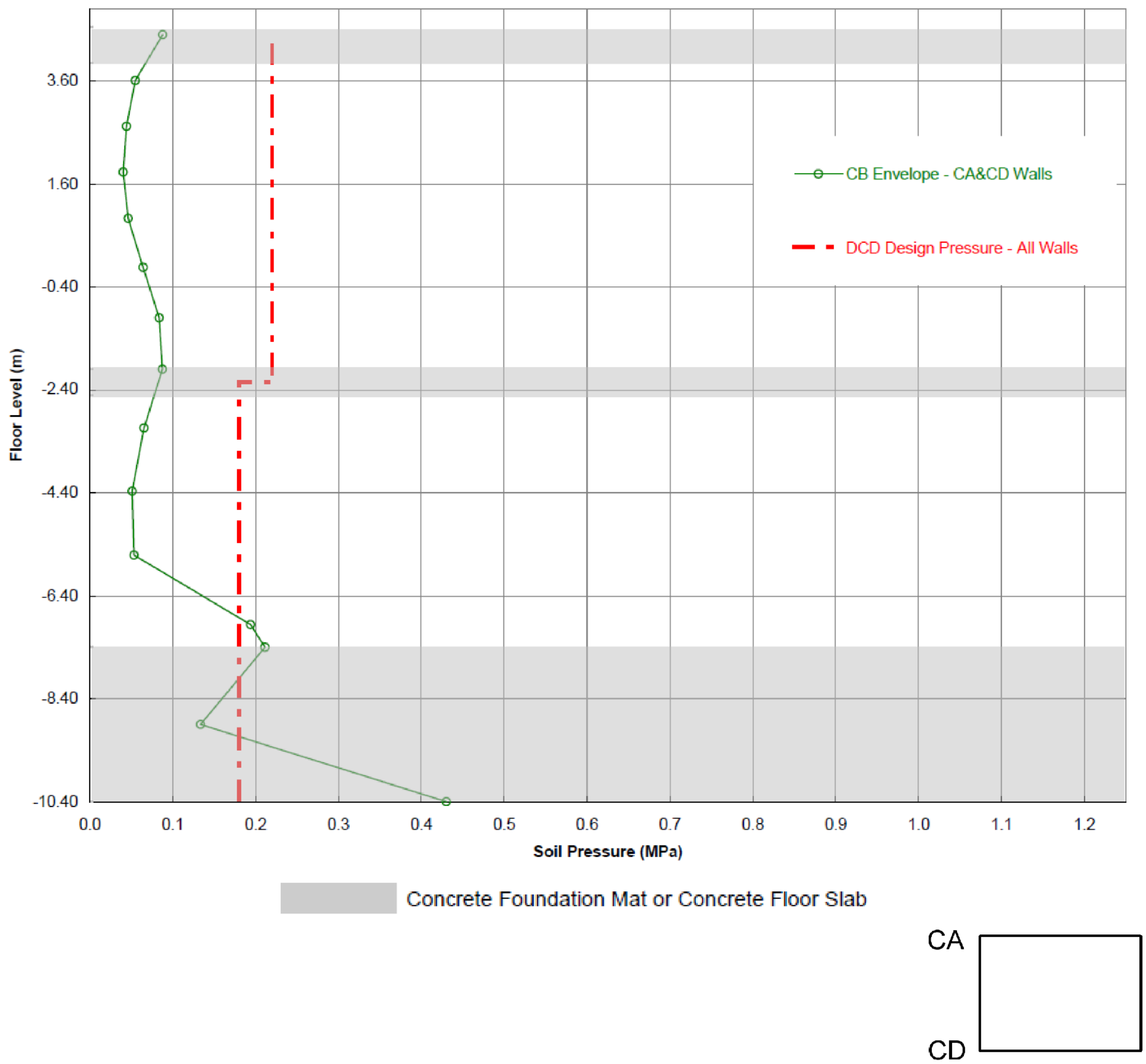
(d) Engineered Backfill Envelope – Walls on Column Rows CA and CD

Figure 3.8.4-203a SSSI Lateral Soil Pressure CB



(a) Engineered Backfill Envelope – Walls on Column Rows C1 and C5

Figure 3.8.4-203b SSSI Lateral Soil Pressure CB



(b) Engineered Backfill Envelope – Walls on Column Rows CA and CD

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

This subsection addresses information concerning methods of analysis for seismic Category I components and supports, including both those designated as ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1 Class 1, 2, 3, or CS and those not covered by the ASME B&PV Code as discussed in Standard Review Plan (SRP) 3.9.1. Information is also presented concerning design transients for ASME B&PV Code Class 1 and CS components and supports.

The plant design meets the relevant requirements of the following regulations:

1. General Design Criterion (GDC) 1 as it relates to safety-related components being designed, fabricated, erected, constructed, tested and inspected in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety-related function to be performed.
2. GDC 2 as it relates to safety-related mechanical components of systems being designed to withstand seismic events without loss of capability to perform their safety-related function.
3. GDC 14 as it relates to the reactor coolant pressure boundary (RCPB) being designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
4. GDC 15 as it relates to the mechanical components of the reactor coolant system being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).
5. Title 10, Code of Federal Regulations (10 CFR), Part 50, Appendix B as it relates to design quality control.
6. 10 CFR 50, Appendix S as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

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 caused by accidents, earthquakes and certain operating conditions. The number of cycles associated with each event for the design of the Reactor Pressure Vessel (RPV), as an example, is 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. Appropriate Service Levels (A, B, C, D or testing) as defined in the ASME B&PV Code, are designated for design limits. The design and analyses 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 specifications and/or stress reports of the respective equipment. Table 3.9-2 shows the load combinations and the standard acceptance criteria. Table 3.9-9 shows the specific load combinations and acceptance criteria for Class 1 piping systems.

3.9.1.2 Computer Programs Used in Analyses

The computer programs used in the analysis 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 General Electric Company (GE) or by outside computer program developers. In either case, the quality of the programs and the 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.

3.9.1.3 Experimental Stress Analysis

The following subsections list those Nuclear Steam Supply System components for which experimental stress analysis is performed in conjunction with analytical evaluation. The experimental stress analysis methods are used in compliance with the provisions of Appendix II of the ASME B&PV Code.

Piping Snubbers and Restraints

The following components have been tested to verify their design adequacy:

1. piping seismic snubbers, and
2. pipe whip restraints.

Descriptions of the snubber and whip restraint tests are contained in Subsection 3.9.3 and Section 3.6, respectively.

3.9.1.4 Considerations for the Evaluation of Faulted Condition

All Seismic Category I equipment is evaluated for the faulted (Service Level D) loading conditions identified in Table 3.9-1 and Table 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 discussion of faulted analysis can be found in Subsections 3.9.2, 3.9.3 and 3.9.5.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits, such as clearance limits, are satisfied.

Fine Motion Control Rod Drive

The Fine Motion Control Rod Drive (FMCRD) major components that are part of the RCPB are analyzed and evaluated for the faulted conditions in accordance with the ASME B&PV Code, Appendix F.

Hydraulic Control Unit

The Hydraulic Control Unit (HCU) is analyzed and tested for withstanding the faulted condition loads. Dynamic tests that are part of the seismic and dynamic qualification program establish the loads in the horizontal and vertical directions as the HCU capability for the frequency range that is likely to be experienced in the plant. These tests also ensure that the scram function of the HCU can be performed under these loads. Dynamic analysis of the HCU with the mounting beams is performed to assure that the maximum faulted condition loads remain below the HCU capability.

Reactor Pressure Vessel Assembly

The RPV assembly includes: (1) the RPV boundary out to and including the nozzles and housings for FMCRD and in-core instrumentation; (2) vessel sliding support and (3) the shroud support. The design and analysis of these three parts complies with Subsections NB, NF and NG, respectively, of the ASME B&PV Code. For faulted conditions, the reactor vessel is evaluated using elastic analysis. For the sliding supports and shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly.

Core Support Structures and Other Safety-Related Reactor Internal Components

The core support structures and other safety-related reactor internal components are evaluated for faulted conditions. The basis for determining the faulted loads for seismic events and other dynamic events is 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.

RPV Stabilizer and FMCRD and In-Core Housing Restraints (Supports)

The calculated maximum stresses meet the allowable stress limits based on the ASME B&PV Code, Subsection NF, for the RPV stabilizer and supports for the FMCRD housing and in-core housing for faulted conditions. These supports restrain the components during earthquake, pipe rupture or other Reactor Building Vibration (RBV) events.

There are eight RPV Stabilizers attached to the Reactor Shield Wall (RSW) that are equally spaced around the circumference of the RPV. The stabilizers interact with eight stabilizer brackets welded to the outside of the RPV wall to resist horizontal loads and limit RPV motion during an earthquake or a postulated pipe break. The lugs are free to move in the vertical and radial directions to accommodate RPV thermal growth and dilation due to pressure. The horizontal loads are resisted

by a series of Belleville washers (i.e., spring washers) on either side of the vessel brackets and transferred to the RSW.

Main Steam Isolation Valve, Safety Relief Valve and Other ASME Class 1 Valves

Elastic analysis methods and standard design rules, as defined in the ASME B&PV Code, are utilized in the analysis of the pressure boundary, Seismic Category I, ASME Class 1 valves. The ASME B&PV Code-allowable stresses are applied to assure integrity under applicable loading conditions including faulted condition. Subsection 3.9.3 discusses the operability qualification of the major active valves including main steam isolation valve (MSIV) and the main steam (MS) safety relief valve (SRV) for seismic and other dynamic conditions.

Fuel Storage and Refueling Equipment

Refueling and servicing equipment and other equipment, which in the case of a failure would degrade a safety-related component, are defined in Section 9.1, and are classified per Table 3.2-1. 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 Zero Period Acceleration (ZPA) defined in Subsection 3.7.2.7 in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific equipment, to Industrial Codes (ASME, ANSI), or Industrial Standards (AISC) allowables.

Fuel Assembly (Including Channel)

GE ESBWR fuel assembly (including channel) design bases, and analytical and evaluation methods including those applicable to the faulted conditions are similar to those contained in References 3.9-1 and 3.9-2.

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 the ASME B&PV Code. These allowables are above elastic limits.

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 the ASME B&PV Code. These allowables are above elastic limits.

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 the ASME B&PV Code. These allowables are above elastic limits.

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 Appendix F (for Class 1) and NC/ND-3600 (for Class 2 and 3 piping) of the ASME B&PV Code. These allowables are above elastic limits. The allowables for functional capability of the safety-related piping are provided in a footnote to Table 3.9-2.

Inelastic Analysis Methods

Inelastic analysis is only applied to ESBWR components to demonstrate the acceptability of two types of postulated events. Each event is an extremely low-probability occurrence and the equipment affected by these events would not be reused. These two events are as follows:

- postulated gross piping failure; and
- postulated blowout of a Control Rod Drive housing (CRDH) caused by 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. Except for pipe whip restraints, inelastic analysis methods are not used in the ESBWR piping design and analysis.

The mitigation of the CRDH attachment weld failure relies on components with regular functions to mitigate the weld failure effect. The components are specifically:

- core support plate;
- control rod guide tube (CRGT);
- CRDH;
- control rod drive (CRD) outer tube; and
- bayonet fingers.

Only the bodies of the CRGT, CRDH and CRD outer tube are analyzed for energy absorption by inelastic deformation.

Inelastic analyses for the CRDH attachment weld failure, together with the criteria used for evaluation, are consistent with the procedures described in Subsection 3.6.2 for the different components of a pipe whip restraint. Figure 3.9-1 shows the stress-strain curve used for the inelastic analysis.

3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment

This subsection presents the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor

internals, and their supports (including supports for conduit and cable trays, and ventilation ducts) under vibratory loadings, including those due to fluid flow and postulated seismic events discussed in SRP 3.9.2. Structural requirements for conduits and cable tray supports and Heating, Ventilation and Air Conditioning duct supports are specified in Subsections 3.8.4.1.6 and 3.8.4.1.7 respectively.

The plant meets the following requirements:

1. GDC 1 as it relates to the testing and analysis of systems, components, and equipment with appropriate safety-related functions being performed to appropriate quality standards.
2. GDC 2 as it relates to safety-related systems, components and equipment being designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena (safe shutdown earthquake [SSE]).
3. GDC 4 as it relates to safety-related systems and components being appropriately protected against the dynamic effects of discharging fluids.
4. GDC 14 as it relates to systems and components of the RCPB being designed to have an extremely low probability of rapidly propagating failure or of gross rupture.
5. GDC 15 as it relates to the reactor coolant system being designed with sufficient margin to ensure that the RCPB is not breached during normal operating conditions, including AOOs.

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 is performed during both of these phases as described in Chapter 14. 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 for the design and testing of the piping support system are described in Subsection 3.9.3.7.

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 (RG) 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.” More specific vibration testing requirements are defined in Part 3 of Reference 3.9-8, “Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems.” Detailed test specifications are in accordance with this standard and 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.

Measurement Techniques

There are essentially three methods available for determining the acceptability of steady state and transient vibration for the affected systems. These are visual observation, local measurements, and remotely monitored/recorded measurements. The technique used depends on such factors as the safety significance of the particular system, the expected mode and/or magnitude of the vibration, the accessibility of the system during designated testing conditions, or the need for a time-history recording of the vibratory behavior. Typically, the systems where vibration has the greatest safety implication are subject to more rigorous testing and precise instrumentation requirements and, therefore, 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 is less complex and of lower magnitude. Many systems that are accessible during the preoperational test phase and that do not show significant intersystem interactions fall into this category. Visual observations are used 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 accessibility is considered. Application of these measurement techniques is detailed in each testing specification consistent with the guidelines contained in Part 3 of Reference 3.9-8.

Monitoring Requirements

As described in Chapter 14, safety-related system critical components and piping runs are subjected to steady state and transient vibration measurements. The scope of such testing includes safety-related instrumentation piping and attached small-bore piping (branch piping). Monitoring location selection considerations 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 are detailed in each test specification. Monitored data includes actual deflections and frequencies as well as related system operating conditions. Time duration of data recording should be sufficient to indicate whether the vibration is continuous or transient. Steady state monitoring is performed at critical conditions such as minimum or maximum flow, or abnormal combinations or configurations of system pumps or valves. Transient monitoring includes 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 are detailed in the appropriate testing specification consistent with Reference 3.9-8 Part 3 guidelines.

Test Evaluation and Acceptance Criteria

The piping response to test conditions is 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 the ASME B&PV Code (NB, NC, ND-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 testing personnel protection, 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 is only 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 are required with appropriate quantitative acceptance criteria.

There are two stress levels of acceptance criteria for allowable vibration displacements/deflections. Level 1 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 Reference 3.9-8, Part 3, paragraph 3.2.1.2. For stainless steel, the Level 1 criterion is 75 MPa (10,880 psi). For carbon steel and low alloy steel, the Level 1 criterion is 53 MPa (7692 psi). The corresponding Level 2 criteria are based on one half of the Level 1 limits. For transient vibration, the Level 1 criteria are based on either the ASME B&PV Code upset primary stress limit or the applicable snubber load capacity. Level 2 criteria are based on a given tolerance about the expected deflection value.

Reconciliation and Corrective Actions

During the course of the tests, the remote measurements are regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements are monitored at more frequent intervals. The test is held for Level 2 criteria violations and terminated as soon as Level 1 criteria are violated. As soon as possible after the test hold or termination, appropriate investigative and corrective actions are taken. If practicable, a walkdown of the piping and suspension system is made in an attempt to identify potential obstructions, improperly operating suspension components, or sensor malfunction. 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 are investigated.

Instrumentation indicating criteria failure is 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 are verified against actual conditions and discrepancies noted are 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. This might include identification and reduction or elimination of offending forcing functions, detuning of resonant piping spans by modifications, addition of bracing, or changes in operating procedures to avoid troublesome conditions. Any such modifications require retest to verify that vibrations have been sufficiently reduced.

3.9.2.1.2 **Thermal Expansion Testing**

A thermal expansion preoperational and startup testing program verifies that normal unrestrained thermal movement occurs in specified safety-related high- and moderate-energy piping systems. The testing is performed through the use of visual observation and remote sensors. The purpose of this program is to ensure the following:

- 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.
- The piping system does shake down after a few thermal expansion cycles.
- The piping system is working in a manner consistent with the predictions of the stress analysis.
- There is adequate agreement between calculated values and measured values of displacements.
- 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 RG 1.68, “ Initial Test Programs for Water-Cooled Nuclear Power Plants.” More specific requirements are defined in Reference 3.9-8, Part 7 “Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems.” Detailed test specifications are prepared in full accordance with this standard and 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.

In addition to thermal expansion testing, thermal stratification testing for the feedwater system piping is also performed on the initial ESBWR plant. This testing is performed using external thermocouples on the pipe to confirm that the thermal stratification inputs to the piping analysis were conservative.

Measurement Techniques

Verification of acceptable thermal expansion of specified piping systems can be accomplished by several methods. One method is to walk down the piping system and verify visually 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 involves 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 uses permanent or temporary instrumentation that directly measures displacement, such as a lanyard potentiometer, that is monitored via a remote indicator or recording device. The technique used depends on such factors as the amount of movement predicted and the accessibility of the piping.

Measurement of piping temperature is also important when evaluating thermal expansion. This is accomplished either indirectly by measuring the temperature of the process fluid or by direct measurement of the piping wall temperature. Such measurements may be obtained either locally or remotely. The choice of technique used depends on such considerations as the accuracy required and the accessibility of the piping.

Monitoring Requirements

As described in Chapter 14, safety-related piping is included in the thermal expansion testing program. Thermal expansion of specified piping systems is measured at both the cold and hot extremes of their expected operating conditions. Walkdowns and recording of hanger and snubber positions are conducted where possible, considering accessibility and local environmental and radiological conditions in the hot and cold states. Displacements and appropriate piping/process temperatures are recorded for those systems and conditions specified. Sufficient time passes before taking such measurements to ensure the piping system is at a steady-state condition. In selecting locations for monitoring piping response, consideration is given to the maximum responses predicted by the piping analysis. Specific consideration is also given to the first run of pipe attached to component nozzles and pipe adjacent to structures requiring a controlled gap.

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 limits 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 is considered acceptable if the test results indicate that the piping responds in a manner consistent with the predictions of the stress report and/or that piping stresses are within the ASME B&PV Code (NB, NC, ND-3600) limits. Acceptable thermal expansion limits are determined after the completion of piping system stress analysis and are provided in the piping test specifications. Level 1 criteria are bounding based on ASME B&PV Code stress limits. Level 2 criteria are stricter based on the predicted movements using the calculated deflections plus a selected tolerance.

Reconciliation and Corrective Actions

During the course of the tests, the remote measurements are regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements are monitored at more frequent intervals. The test is held for Level 2 criteria violations, and terminated as soon as Level 1 criteria are violated. As soon as possible after the test hold or termination, investigative and corrective actions are taken. If practicable, a walkdown of the affected piping and suspension system is made to identify potential obstructions 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 are investigated.

Instrumentation indicating criteria failure is 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 are compared with actual test conditions. Discrepancies noted are 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 complete or partial removal of an interfering structure; replacing, readjusting, adding or repositioning piping system supports; modifying the pipe routing; or modifying system operating procedures to avoid the temperature conditions that resulted in the unacceptable thermal expansion.

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 the qualification testing and/or analysis applicable to the major components on a component by component basis. Seismic and other events that may induce RBV are considered. In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit (e.g., HCU). These modules are generally discussed completely in this subsection and Subsection 3.9.3 rather than providing a separate discussion of the 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-related 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, operability is demonstrated by testing. Otherwise, operability is demonstrated by mathematical analysis or by a combination between analysis and test.

Equipment that is large, simple, and/or consumes large amounts of power is usually qualified by analysis or static bend tests 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 there are no natural frequencies below ZPA defined in Subsection 3.7.2.7. If a natural frequency lower than ZPA defined in Subsection 3.7.2.7 in the case of other RBV induced loads is discovered, dynamic tests and/or mathematical dynamic 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) over the frequency range of interest. Whichever method is used, the input amplitude during testing envelops 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. Other interface loads (nozzle loads, weights of internal and external components attached) are simulated.

Equipment having an extended structure, such as a valve operator, is 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 static bend test is used to determine spring constant and operational capability at maximum equivalent dynamic load conditions.

Random Vibration Input

When random vibration input is used, the actual input motion envelops 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:

- the characteristics of the required input motion is dominated by one frequency.
- the anticipated response of the equipment is adequately represented by one mode.
- the input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra envelops the corresponding response spectra of the individual modes.

Application of Input Modes

When dynamic tests are performed, the input motion is applied to the 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.

Fixture Design

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

Prototype Testing

When possible, equipment testing is conducted on prototypes of the equipment to be installed in the plant. If not, a detailed inspection and justification of the capacity of the equipment tested is made.

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 B&PV Code equipment including equipment supports.

CRD and CRDH

The qualification of the CRDH (with enclosed CRD) is done analytically, and the stress results of the analysis establish the structural integrity of these components. Dynamic tests are conducted to verify the operability of the CRD 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.

The correlation of the test with analysis is via the channel deflection, not the housing structural analysis, because insertability is controlled by channel deflection, not housing deflection.

Core Support (Fuel Support and Control Rod 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.

Hydraulic Control Unit

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, the faulted condition loads are calculated to be below the HCU maximum capability.

Fuel Assembly (Including Channel)

ESBWR 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.

Standby Liquid Control Accumulator

The standby liquid control accumulator is a cylindrical vessel. The standby liquid control accumulator is qualified by analysis for seismic and other RBV loads.

The results of this analysis confirm that the calculated stresses at all investigated locations are less than their corresponding allowable values.

Main Steam Isolation Valves

The MSIVs are qualified for seismic and other RBV loads. The fundamental requirement of the MSIV following a safe shutdown earthquake (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 3.9.3.5.

Standby Liquid Control Valve (Injection Valve)

The standby liquid control injection valve is qualified by type test to IEEE 344 for seismic and other RBV loads. The qualification test as discussed in Subsection 3.9.3.5 demonstrates the ability to remain operable after the application of horizontal and vertical dynamic loading in excess of the required response spectra. The valve is 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.

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 analysis as discussed in Subsection 3.9.3.5 demonstrate the satisfactory operation of the valves during and after the test.

Other ASME B&PV Code, Section 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 seismic event.

Dynamic load qualification is done by testing, analysis, or both. 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 rigid and single frequency tests at every 1/3 octave frequency interval are acceptable. Also, if all natural frequencies of the equipment are greater than ZPA defined in Subsection 3.7.2.7, 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 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.

If 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 modal analysis techniques 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 multi-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 required 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 does not result in higher response than the operating condition. The analysis includes evaluation of the effects of the calculated stresses on mechanical strength, alignment, electrical performance

(microphonics, contact bounce, etc.) and non-interruption 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.

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 consoles, racks, etc., may be vibration tested without the equipment 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.

Alternatively, equipment may be qualified by presenting historical performance data, which demonstrate 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-related function, are also 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 for this purpose.

Historically, it has been shown that the main cause of 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 specified otherwise, anchorage devices are designed in accordance with the requirements of the ASME B&PV Code, Subsection NF, or ANSI/AISC-N690 and ACI 349.

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 are used. For the case of equipment having multiple supports with different dynamic motions, an

upper bound envelope of all the individual response spectra for these locations is used to calculate maximum inertial responses of items with multiple supports.

Refer to Subsection 3.9.3.5 for additional information on the dynamic qualification of valves.

Supports

Subsections 3.9.3.7 and 3.9.3.8 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 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. The vibration forcing functions for operational flow transients and steady state conditions are determined by first postulating the source of the forcing function, such as forces due to flow turbulence, symmetric and asymmetric vortex shedding, pressure waves from steady state and transient operations. Based on these postulates, prior startup and other test data from similar or identical components are examined for the evidence of the existence of such forcing functions. Special analysis of the response signals measured for reactor internals of many similar designs is performed to obtain the parameters, which determine the amplitude and modal contributions in the vibration responses. Based on these examinations, the magnitudes of the forcing functions and response amplitudes are derived. These magnitudes are then used to calculate the expected ESBWR responses for each component of interest during steady state and transient conditions. This study provides useful predictive information for extrapolating the results from tests of components with similar 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:

- *Dynamic modal analysis of major components and subassemblies is 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.*
- *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 Boiling Water Reactors (BWRs) of differing size and design.*

- *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.*
- *Correlation functions of the 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.*
- *Predicted vibration amplitudes for components of the prototype plants are obtained from these correlation functions based on applicable values of the parameters for the prototype plants. 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 modal analyses.*

The dynamic modal analysis forms the basis for interpretation of the initial startup test results (Subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and highest component stresses for each of the lower normal modes.

Details of the special signal analyses of the vibration sensors are given below:

The test data from sensors (accelerometers, strain gages, and pressure sensors) installed on reactor internal components are first analyzed through signal processing equipment to determine the spectral characteristics of these signals. The spectral peak magnitudes and the frequencies at the spectral peaks are then determined. These spectral peak frequencies are then classified as natural frequencies or forced frequencies. If a spectral peak is classified as being from a natural frequency, its amplitude is then determined using a band-pass filter if deemed necessary. The resultant amplitude is then identified as the modal response at that frequency. This process is used for all frequencies of interest. Thus the modal amplitudes at all frequencies of interest are determined. If a spectral peak is identified as being from a forced frequency, the source (such as the vane passing frequency of a pump) is identified. Again, its magnitude is determined using a band-pass filter if deemed necessary.

The modal amplitudes and the forced response amplitudes are then used to calculate the expected ESBWR amplitudes for the same component. These ESBWR expected amplitudes are determined by calculating the expected changes in the forcing function magnitudes from the test component to the ESBWR component. For example, for flow turbulence excited components, the magnitudes are determined by ratio with the flow velocity squared.

A flow chart of the above process is shown in Figure 3.9-6.

The allowable vibratory amplitude in each mode is that which produces a highest stress amplitude of ±68.95 MPa (±10,000 psi). For the steam dryer and its components, a higher allowable highest stress limit is used as explained in the following paragraphs.

Vibratory loads are continuously applied during normal operation and the stresses are limited to ±68.95 MPa (±10,000 psi), with the exception of the steam dryer, in order 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.

Extensive predictive evaluations have been performed for the steam dryer loading and structural evaluation. These evaluations are described in Appendix 3L.4. In the dryer design and in the development of the initial strain and accelerations acceptance limits used during startup, the fatigue analysis performed for the ESBWR steam dryer uses a fatigue limit stress amplitude of 93.7 MPa (13,600 psi). For additional conservatism in the predictive analysis, the analysis stress results will also meet a minimum alternating stress ratio (MASR) of 2.0 between the analysis results and the fatigue acceptance limit. This is verified for the predictive analysis of each as-built steam dryer through inspections, tests, and analyses acceptance criteria during construction.

The startup testing then uses the fatigue limit stress amplitude of 93.7 MPa (13,600 psi) with a MASR of 1.0 as the basis for the acceptance limits for the on-dryer instrument measurements during power ascension. Following the startup testing of the first unit, a confirmatory stress analysis will be performed based on the on-dryer instrument measurements following startup testing. The FIV load definition for this analysis is defined from the recorded on-dryer pressure measurements. A structural assessment is performed to benchmark the FE model strain and acceleration predictions against the measured data. The steam dryer stresses are determined using the on-dryer FIV load definition and adjusted for end-to-end bias and uncertainties determined from the FE model benchmark. A fatigue limit stress amplitude of 93.7 MPa (13,600 psi) with a MASR of 1.0 is used as the acceptance limit for this confirmatory stress analysis. If an acceptance limit is reached during power ascension, the same process will be used to confirm that the steam dryer stresses are below the fatigue limit stress amplitude of 93.7 MPa (13,600 psi) and to redefine the acceptance limits for the on-dryer instrument measurements before continuing with the power ascension.

The subsequent ESBWR steam dryers will follow the same process as the prototype ESBWR steam dryer, with the predictive analysis verified through inspections, tests, and analyses acceptance criteria; the FIV monitoring process using on-dryer instruments during startup testing; and a confirmatory analysis based on on-dryer measurements at full power following startup testing. The acceptance limits for steam dryers in subsequent plants are based on (1) the predictive analysis for the as-built steam dryer satisfying the 2.0 MASR to the 93.7 MPa (13,600 psi) fatigue stress limit, and (2) assuring that the steam dryer stresses remain less than 93.7 MPa (13,600 psi) with a MASR of 1.0 for the startup testing and confirmatory analysis based on on-dryer measured data. The steam dryer is a nonsafety-related component, performs no safety-related functions, and is only required to maintain its structural integrity (no loose parts generated) for normal, transient and accident conditions.

The dynamic loads caused by FIV of the steam separators have been determined using a full-scale separator test under reactor conditions. During the test, the flow rate through the steam separator was 226,000 kg/hr (499,000 lbm/hr) at 7% quality. This is higher than the ESBWR maximum separator flow of 100,700 kg/hr (222,000 lbm/hr) at rated power. Test results show a maximum FIV stress of less than 49.6 MPa (7200 psi), well below the GEH acceptance criterion of 68.9 MPa (10,000 psi). Thus it can be concluded that separator FIV effects are acceptable. Jet impingement from feedwater flow has no significant effect on the steam separator assembly since the separator outer-most cylindrical structure (also referred to as the separator "skirt") is above the feedwater flow impingement area.

3.9.2.4 Initial Startup Flow-Induced Vibration Testing of Reactor Internals

A reactor internals vibration measurement and inspection program is conducted only during initial startup testing. This meets the guidelines of RG 1.20 with the exception of those requirements related to pre-operational testing which cannot be performed for a natural circulation reactor.

Initial Startup Testing

Vibration measurements are made during reactor startup at conditions up to 100% rated flow and power. Steady state and transient conditions of natural circulation flow operation are evaluated. The primary purpose of this test series is to verify the anticipated effect of single- and two-phase flow on the vibration response of internals. Details of the initial startup vibration test program are described in Subsection 3L.4.6 for the steam dryer and Section 3L.5 for other reactor internals. A brief summary is given below.

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 are provided in Appendix 3L.

In all plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data are recorded and provision is made for selective on-line 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 that best approximates the observed mode.

The visual inspections conducted prior to and remote inspections conducted following startup testing are for damage, excessive wear, or loose parts. At the completion of initial startup testing, remote inspections of major components are performed on a selected basis. The remote inspections cover the steam dryer, chimney, chimney head, core support structures, the peripheral CRD and incore housings. Access is provided to the reactor lower plenum for these inspections.

The analysis, design and equipment that are to be utilized for ESBWR comply with RG 1.20, as explained below.

RG 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, Appendix A to 10 CFR 50. This RG is applicable to the core support structures and other reactor internals.

Vibration testing of reactor internals is performed on all GE-BWR plants. Since the original issue of RG 1.20, test programs for compliance have been instituted for preoperational and startup testing. The first ESBWR plant is instrumented for testing. However, it can be subjected to startup flow testing only to demonstrate that FIVs similar to those expected during operation do not cause damage. Subsequent plants, which have internals similar to those of the first plant, are also tested in compliance with the requirements of RG 1.20. GEH is committed to confirm the satisfactory vibration performance of the internals in these plants through startup flow testing followed by inspection. Extensive vibration measurements in prototype plants together with satisfactory operating experience in all BWR plants have established the adequacy of reactor internal designs. GEH continues these test programs for subsequent plants to verify structural integrity and to establish the margin of safety. The FIV evaluation program pertaining to reactor internal components is addressed in Appendix 3L. As part of the initial implementation of the vibration assessment program, RG 1.20 guidance in Section 2.4 states that if inspection of the reactor internals reveals defects, evidence of unacceptable motion, or excessive or undue wear; if results from the measurement program fail to satisfy the specified test acceptance criteria; or if results from the analysis, measurement, and inspection programs are inconsistent, then further evaluations, modifications, or other actions are taken to justify the structural adequacy of the reactor internals. Such results and actions are reported to the NRC as part of the final report documentation of results of the comprehensive vibration assessment program following testing.

For reactor internals other than the steam dryer, the vibration assessment program, as specified in Regulatory Guide (RG) 1.20, is provided in Appendix 3L and the following referenced GEH Report:

- NEDE-33259P-A, “Reactor Internals Flow Induced Vibration Program”

The classification of the Fermi 3 reactor internals in accordance with RG 1.20 is dependent on ESBWR status, i.e. if Fermi 3 is the initial ESBWR to perform testing of the reactor internals, or if testing is performed at another reactor prior to Fermi 3 testing. There are two different scenarios:

1. A valid prototype for the Fermi 3 reactor internals does not exist. Under this scenario, Fermi 3 reactor internals classification is a prototype per RG 1.20.
2. A valid prototype for Fermi 3 reactor internals does exist. If the prototype testing is performed outside the United States, the guidance in Regulatory Guide 1.20, Revision 3, Regulatory Position 1.2, would need to be satisfied in order for this reactor to be considered a “valid prototype.” Assuming that Fermi 3 reactor internals are substantially similar to the

valid prototype and that the valid prototype does not experience inservice problems that result in component or operational modifications, Fermi 3 reactor internals will be classified as non-prototype category I. If a change to classification for Fermi 3 reactor internals is later determined to be necessary, the classification change will be addressed at the time the change is proposed with proper evaluation/justification and documented in a revision to the UFSAR.

Specific to the steam dryer, the comprehensive vibration assessment program, as specified in RG 1.20, is provided in Appendix 3L and the following referenced GEH Reports:

- NEDE-33312P, "ESBWR Steam Dryer Acoustic Load Definition."
- NEDE-33313P, "ESBWR Steam Dryer Structural Evaluation."
- NEDE-33408P, "ESBWR Steam Dryer – Plant Based Load Evaluation Methodology, PBLE01 Model Description."

The steam dryer is classified as a prototype according to RG 1.20, Revision 3. Section 10.2 of NEDE-33313P provides four elements of a steam dryer Comprehensive Vibration Assessment Program that must be addressed. The following describes the approach for the steam dryer Comprehensive Vibration Assessment Program elements, consistent with Regulatory Guide 1.20 and Section 10.2 of NEDE-33313P:

1. The ESBWR steam dryer Comprehensive Vibration Assessment Program is described in Section 3.9, Appendix 3L, and NEDE-33313P, Section 10.0, which includes a description for preparing and submitting to the NRC a Steam Dryer Monitoring Plan no later than 90 days before startup.
2. The detailed design of the steam dryer will follow the methodology described in Appendix 3L and the incorporated engineering reports. As described in NEDE-33313P, Section 10.2(b), an example of a steam dryer predicted analysis that concludes the steam dryer will not exceed stress limits with applicable bias and uncertainties and the minimum alternating stress ratio of 2.0 is provided in NEDE-33408P. The final detailed design of the ESBWR steam dryer has not yet been completed. Therefore, the example of an as-designed steam dryer that has been subject to the predicted analysis process and successful startup testing described in NEDE-33408P serves as the design analysis report for the steam dryer and provides sufficient information for licensing. The post-licensing commitments in ITAAC and license conditions confirm the acceptability of the ESBWR steam dryer design.
3. The startup program and associated license conditions that include appropriate notification points during power ascension, providing data to the NRC at certain hold points and at full power, and providing to the NRC a full stress analysis report and evaluation within 90 days of reaching the full power level, are established in accordance with NEDE-33313P, Section 10.2(c).

4. Periodic steam dryer inspection during refueling outages is as described in NEDE-33313P, Section 10.2(d), and associated license conditions.

[START COM-FSAR-3.9-001]For reactor internals other than the steam dryer, the comprehensive vibration assessment program will be developed and implemented as described in Appendix 3L with no departures. The vibration measurement and inspection programs will comply with the guidance specified in RG 1.20, Revision 3, consistent with the Fermi 3 reactor internals classification. A summary of the vibration analysis program and description of the vibration measurement (including measurement locations and analysis predictions) and inspection phases of the comprehensive vibration inspection program will be submitted to the NRC six months prior to implementation. **[END COM-FSAR-3.9-001]**

[START COM-FSAR-3.9-006] For reactor internals other than the steam dryer, the preliminary and final reports (as necessary), which together summarize the results of the vibration analysis, measurement and inspection programs will be submitted to the NRC within 60 and 180 days, respectively, following the completion of the programs. **[END COM-FSAR-3.9-006]**

3.9.2.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

The faulted events that are evaluated are defined in Subsection 3.9.5.3. 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 (Table 3.9-3) due to an assumed break of a main steamline (MSL) or feedwater line (FWL) are determined by analysis as described in Subsection 3.9.5.3. 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 comprehensive horizontal and vertical dynamic models of the RPV and internals. 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 MSL, the FWL or the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) 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 loss-of-coolant-accident (LOCA) loads, are considered as shown in Table 3.9-1.

3. SRV Loads — The discharge of the SRVs results in RBVs due to suppression pool dynamics as described in 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 as described in Appendix 3B and the response 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 3.9-1.
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 are then determined by either the time-history method or the response-spectrum method. The loads on the reactor internals due to a 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 (Small Break LOCA (SBL), Intermediate Break LOCA (IBL) or Large Break LOCA (LBL)) and SSE loads as defined in Table 3.9-1 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 defined in Subsection 3.9.5.4.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Prior to initiation of the instrumented vibration measurement program for a 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 are 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 used 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 B&PV Code Class 1, 2 and 3 Components, Component Supports and Core Support Structures

This subsection discusses the structural integrity of pressure-retaining components, their supports, and core support structures which are designed in accordance with the rules of the ASME B&PV Code, Section III, Division 1 and GDC 1, 2, 4, 14, and 15 as discussed in SRP 3.9.3.

The plant design meets the relevant requirements of the following regulations:

1. 10 CFR Part 50.55a and GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety-related function to be performed.
2. GDC 2 as it relates to safety-related structures and components being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.
3. GDC 4 as it relates to safety-related structures and components being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions.
4. GDC 14 as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
5. GDC 15 as it relates to the reactor coolant system being designed with sufficient margin to ensure that the design conditions are not exceeded.

The ASME B&PV Code, Section III, requires that a design specification be prepared for ASME Class 1, 2 and 3 components. The design specifications for ASME Class 1, 2 and 3 components, supports, and appurtenances are prepared under administrative procedures that meet the ASME B&PV Code rules. The specifications conform to and are certified to the requirements of the applicable subsection of the ASME B&PV Code, Section III. The ASME B&PV Code also requires design reports for Class 1, 2 or 3 components be prepared which demonstrate that the as-built components satisfy the requirements of the respective ASME design specification for each component and the applicable ASME B&PV Code. These design specifications and the design reports are completed by the license holder, or the holder’s authorized agent, in accordance with the responsibilities outlined under the ASME B&PV Code, Section III. The ASME B&PV Code design reports include the record of as-built reconciliations, for example, the evaluations of changes to piping support locations, the pre-operational testing and results, and reported construction deviation resolutions, and also includes the small-bore piping analysis.

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 RBV events for the design of safety-related ASME B&PV 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 3.9-1 presents the plant events to be considered for the design and analysis of all ESBWR ASME B&PV Code Class 1, 2, and 3 components, component supports, core support structures and equipment. Specific loading combinations considered for evaluation of specific equipment are derived from Table 3.9-2 and are contained in the design specifications and design reports for the respective equipment. For Class I components where analysis for cyclic operation is evaluated in accordance with ASME B&PV Code, Section III, subarticle NB-3222.4, the fatigue usage evaluation shall include the use of environmental fatigue curves in accordance with RG 1.207 and NUREG/CR-6909.

Specific load combinations and acceptance criteria for Class 1 piping are shown in Table 3.9-9. Also for Class 1 piping, the operating temperatures above ambient or below ambient are included in the fatigue analysis. Even the ambient temperature is included as a load set with defined cycles. The stress-free state for the piping system is defined as a temperature of 21°C (70°F) for Class 1, 2, 3 or B31.1 piping. For Class 2, 3 or B31.1 piping, no thermal expansion analysis will be performed for a piping system operating at 65°C (150°F) or less.

The design life for the ESBWR Standard Plant is 60 years. A 60-year design life is a requirement for all major plant components.. However, all plant operational components and equipment except the reactor vessel are designed to be replaceable.. 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.

[START COM 3.9-002] The equipment stress reports identified in this section will be completed within six months of completion of DCD ITAAC Table 3.1-1. **[END COM 3.9-002] [START COM 3.9-004]** The UFSAR will be revised as necessary in a subsequent update to address the results of this analysis. **[END COM 3.9-004]**

In the event any non-Class 1 component is subjected to cyclic loadings of a magnitude and/or duration so severe that the 60-year design life cannot be assured by required ASME B&PV Code calculations, applicants referencing the ESBWR design shall 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. For example, thermal sleeves may be required to protect the pressure boundary from severe cyclic thermal stress, at points where mixing of hot and cold fluids occur. For ESBWR, these locations include the SRV discharge line going to the quencher and the feedwater pipe within the steam tunnel at the reactor water cleanup (RWCU) junction.

3.9.3.1.1 **Plant Conditions**

All events that the plant might credibly experience during a reactor-year are evaluated to establish the 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 below and correlated to service levels for design limits defined in the ASME B&PV Code Section III as shown in Table 3.9-1 and Table 3.9-2.

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.

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, i.e., AOOs, as defined in 10 CFR 50, Appendix A, which result from any single operator error or control malfunction, from a fault in a system component requiring its isolation from the system, or from a loss of load or power. Hot standby with the main condenser isolated is an upset condition.

Emergency Condition

An emergency condition includes deviations from normal conditions that require shutdown for correction of the condition(s) or repair of damage in the RCPB. Such conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system. Emergency condition events include but are not limited to infrequent operational transients (IOT), e.g., infrequent events, as defined in Subsection 15.0.1.2, caused by one of the following: (a) a multiple valve blowdown of the reactor vessel; (b) LOCA from a small break or crack (SBL) which does not depressurize the reactor systems, does not automatically actuate the Gravity-Driven Cooling System (GDCS) and Automatic Depressurization System (ADS), and does not result in leakage beyond normal make-up system capacity, but which requires the safety-related functions of isolation of containment and shutdown and may involve inadvertent actuation of the ADS; (c) improper assembly of the core during refueling; or (d) depressurization valve (DPV) blowdown. An

Anticipated Transient Without Scram (ATWS) or reactor overpressure with delayed scram (Table 3.9-1 and Table 3.9-2) is a special event, as defined in Subsection 15.0.1.2, that is classified as an emergency condition.

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 a LOCA, 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: (a) a fuel-handling accident; (b) a MSL or FWL break; (c) the combination of any SBL or IBL with the SSE, and a loss of off-site power; or (d) the SSE plus LBL plus a loss of off-site power.

The IBL classification covers those breaks for which the GDCS operation occurs during the blowdown. The LBL classification covers the sudden, double-ended severance of a MSL inside or outside the containment that results in transient reactor depressurization, or any pipe rupture of equivalent flow cross-sectional area with similar effects.

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 B&PV 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*	$10^{-4} > P > 10^{-6}$

Safety-Related Functional Criteria

For any normal or upset design condition event, safety-related equipment and piping (Section 3.2) are capable of accomplishing their safety-related functions as required by the event and incurring no permanent changes that could deteriorate their ability to accomplish safety-related functions as required by any subsequent design-condition event.

For any emergency or faulted design-condition event, safety-related equipment and piping are capable of accomplishing their safety-related functions as required by the event, but repairs could be required to ensure their ability to accomplish safety-related functions as required by any subsequent design-condition event.

3.9.3.1.2 Inspections/Testing Following the Reactor Coolant System Exceeding Service Level B Pressure Limit

If any abnormal event causes the reactor coolant system pressure to exceed 110% of its design value (i.e., exceed the ASME B&PV Code Service Level B pressure limit), an inspection program should be satisfactorily completed, before normal plant operations may proceed. Within ASME B&PV Code, Section XI, Subarticles IWB-2400 and IWB-2500 there are inspection specifications that can determine the structural integrity of the reactor coolant system components directly affected by the pressurization event. Therefore, if the pressure of the reactor coolant system exceeds its ASME B&PV Code Service Level B limit, then an inspection program will be established based on an assessment of all potentially affected safety-related reactor coolant system components, and subsequent inspections or testing per the appropriate portions of ASME B&PV Code, Section XI, Subarticles IWB-2400 and IWB-2500 will be performed and evaluated against the ASME B&PV Code acceptance criteria, prior to commencement of normal power operations.

3.9.3.2 Reactor Pressure Vessel Assembly

The reactor vessel assembly includes: the RPV boundary out to and including the nozzles and housings for FMCRD and in-core instrumentations; vessel sliding support, and shroud support.

The RPV, vessel sliding support, and shroud support are designed and constructed in accordance with the ASME B&PV Code. The shroud support consists of support legs and a support ring. The RPV is ASME Class 1, and the RPV internals are classified in Subsection 3.9.5. Complete stress reports on these components are prepared in accordance with the ASME B&PV Code requirements. The guidance from NUREG-0619 and associated Generic Letters 80-95 and 81-11 is factored into the feedwater nozzle and sparger design. The feedwater nozzle/sparger design does not allow incoming feedwater flow to have direct contact with the nozzle bore region, and the double thermal sleeve design adds further protection against thermal cycling on the nozzle. Also see Subsection 3.9.5.2 for additional information.

The stress analysis is performed on the RPV, vessel sliding support, and shroud support for various plant operating conditions (including faulted conditions) by using elastic methods, except as noted in Subsection 3.9.1.4. 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.

3.9.3.3 Main Steam System Piping

The piping systems extending from the RPV to and including the outboard MSIV are designed and constructed in accordance with the ASME B&PV Code Section III, Class 1 criteria. Stresses are

calculated on an elastic basis for each service level and evaluated in accordance with NB-3600 of the ASME B&PV Code. Table 3.9-9 *shows the specific load combinations and acceptance criteria for Class 1 piping that apply to this piping. For the MS Class 1 piping, the thermal loads per Equation 12 of NB-3600 are less than 2.4 S_m, and are more limiting than the dynamic loads that are required to be analyzed per Equation 13 of NB-3600.*

The MS system piping extending from the outboard MSIV to the turbine stop valve is constructed in accordance with the ASME B&PV Code, Class 2 Criteria.

3.9.3.4 Other Components

Standby Liquid Control (SLC) Accumulator

The standby liquid control accumulator is designed and constructed in accordance with the requirements of the ASME B&PV Code, as a Class 2 component.

SLC Injection Valve

The SLC injection valve is designed and constructed in accordance with the requirements for the ASME B&PV Code, as a Class 1 component.

GDCS Piping and Valves

The GDCS valves connected with the RPV, including squib valves, and up to and including the biased-open check valve are designed and constructed in accordance with the requirements of the ASME B&PV Code, as Class 1 components. Other valves in the system are Class 2 components.

Main Steamline Isolation, Safety Relief, and Depressurization Valves

The MSIVs, SRVs, and DPVs are designed and constructed in accordance with the ASME B&PV Code, NB-3500 requirements for Class 1 components.

Safety Relief Valve Piping

The relief valve discharge piping extending from the relief valve discharge flange to the vent wall penetration is designed and constructed in accordance with the ASME B&PV Code requirements for Class 3 components. The relief valve discharge piping extending from the diaphragm floor penetration to the quenchers is designed and constructed in accordance with the ASME B&PV Code requirements for Class 3 components.

Isolation Condenser System (ICS) Condenser and Piping

The ICS piping inside the primary containment between the RPV and the condenser isolation valve is designed and constructed in accordance with the ASME B&PV Code requirements for Class 1 piping. The isolation condenser and piping outside containment are designed and constructed in accordance with Class 2 requirements.

RWCU/SDC System Pump and Heat Exchangers

The RWCU/SDC pump and heat exchangers (regenerative and nonregenerative) are not part of a safety system. However, the pumps and heat exchanger are Seismic Category I equipment. The ASME B&PV Code requirements for Class 3 components are used in the design and construction of the RWCU System pump and heat exchanger components.

ASME Class 2 and 3 Vessels

The Class 2 and 3 vessels (all vessels not previously discussed) are constructed in accordance with the ASME B&PV Code. The stress analysis of these vessels is performed using elastic methods.

ASME Class 1, 2 and 3 Valves

The Class 1, 2, and 3 valves (all valves not previously discussed) are constructed in accordance with the ASME B&PV Code.

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. Refer to Subsection 3.9.3.5 for additional information on valve operability.

ASME Class 1, 2 and 3 Piping

The Class 1, 2 and 3 piping (all piping not previously discussed) is constructed in accordance with the ASME B&PV Code. For Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of the ASME B&PV Code, and fatigue usage is in accordance with RG 1.207 and NUREG/CR-6909. For Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NC/ND-3600 of the ASME B&PV Code. In the event that a NB-3600 analysis is performed for Class 2 or 3 pipe, all analyses required for Class 1 pipe as specified in this document and the ASME B&PV Code are performed. Table 3.9-9 shows the specific load combinations and acceptance criteria for Class 1 piping systems. *For the Class 1 piping that experiences the most significant stresses during operating conditions, the thermal loads per Equation 12 of NB-3600 are less than 2.4 S_m, and are more limiting than the dynamic loads that are required to be analyzed per Equation 13 of NB-3600. The piping considered in this category is the RWCU/SDC, feedwater, MS, and isolation condenser steam piping within the containment.* These were evaluated to be limiting based on differential thermal expansion, pipe size, transient thermal conditions and high energy line conditions. If ASME B&PV Code Case N-122-2 is used for analysis of a class 1 pipe, the analysis complying with this case is included in the design report for the piping system.

For submerged piping and associated supports, the applicable direct external loads (e.g., hydrodynamic) applied to the submerged components are included in the analysis.

3.9.3.5 Valve Operability Assurance

This subsection discusses operability assurance of active ASME B&PV Code valves, including actuators (Subsection 3.9.2.2).

Valves that perform an active safety-related function are functionally qualified to perform their required functions. For valve designs developed for the ESBWR that were not previously qualified, the qualification programs meet the requirements of QME-1-2007. For valve designs previously qualified to standards other than ASME QME-1-2007, the following approach is used.

- *The ESBWR general valve requirements specification includes requirements related to design and functional qualification of safety-related valves that incorporate lessons learned from nuclear power plant operations and research programs.*
- *Qualification specifications (e.g., design specifications) consistent with Appendices QV-I and QV-A of QME-1-2007 are prepared to ensure the operating conditions and safety functions for which the valves are to be qualified are communicated to the manufacturer or qualification facility.*
- *Suppliers are required to submit, for GE Hitachi Nuclear Energy (GEH) review and approval, application reports, as described in QME-1-2007, that describe the basis for the application of specific predictive methods and/or qualification test data to a valve application.*
- *GEH reviews the application reports provided by the suppliers for adherence to specification requirements to ensure the methods used are applicable and justified and to verify any extrapolation techniques used are justified. A gap analysis is performed to identify any deviations from QME-1-2007 in the valve qualification. Each deviation is evaluated for impact on the overall valve qualification. If the conclusion of the gap analysis is that the valve qualification is inadequate, then the valve may be qualified using a test-based methodology, as allowed by QME-1-2007.*
- *GEH performs independent sizing calculations, using bounding design parameters (such as sliding friction coefficients), to verify supplier actuator sizing.*

Functional qualification addresses key lessons learned from industry efforts, particularly on air- and motor-operated valves, many of which are discussed in Section QV-G of QME-1-2007. For example:

- Evaluation of valve performance is based on a combination of testing and analysis, using design similarity to apply test results to specific valve designs.
- Testing to verify proper valve setup and acceptable operating margin is performed using diagnostic equipment to measure stem thrust and torque, as appropriate.
- Sliding friction coefficients used to evaluate valve performance (e.g., disk-to-seat friction coefficients for gate valves and bearing coefficients for butterfly valves) account for the effects of temperature, cycle history, load and internal parts geometry.

- Actuator sizing allows margin for aging/degradation, test equipment accuracy and other uncertainties, as appropriate.
- Material combinations that may be susceptible to galling or other damage mechanisms under certain conditions are not used.

Subsection 3.9.2.2 and Section 3.10 provide details on the seismic qualification of valves. Section 3.11 provides details on the environmental qualification (EQ) of valves.

Section 4.4 of GEH's EQ Program (Reference 3.9-3) applies to this subsection, and the seismic qualification methodology presented therein is applicable to mechanical as well as electrical equipment.

3.9.3.5.1 **Major Active Valves**

Some of the major safety-related active valves (Table 3.2-1) discussed in this subsection for illustration are the MSIVs and SRVs, and SLC injection valves and DPVs. These valves are designed to meet the ASME B&PV Code 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 3.9.3.7). The dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

Main Steam Isolation Valves

The 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 design basis accident (DBA) and SSE.

The valve body is designed, analyzed and tested in accordance with the ASME B&PV Code, Class 1 requirements. The MSIVs are modeled mathematically in the MSL 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.

The MSIV and associated electrical equipment (wiring, solenoid valves, and position switches) are dynamically qualified to operate during an accident condition.

Main Steam Safety Relief Valves

The typical SRV design described in Subsection 5.2.2.2 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 the ASME B&PV Code Class 1 analysis and test.

- The 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.

- A production SRV is demonstrated for operability during a dynamic qualification (shake table) type test with moment and loads applied greater than the required equipment's design limit loads and conditions.

A mathematical model of this valve is included in the MSL system analysis, as with the MSIVs. This analysis ensures the equipment design limits are not exceeded.

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 the ASME B&PV Code, 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.

Depressurization Valves

The DPV design described in Subsection 6.3.2.8 is qualified by test to IEEE 344 for operability during a dynamic event. Structural integrity of the configuration during dynamic events is demonstrated by both the ASME B&PV Code Class 1 analysis and test.

- The valve is designed for maximum moments on the inlet that 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.
- A production DPV is demonstrated for operability after the performance of a dynamic qualification (shake table) type test with moment and loads applied greater than the required equipment's design limit loads and conditions.

A mathematical model of this valve is included in the ICS system analysis. These analyses assure that the equipment design limits are not exceeded.

3.9.3.5.2 Other Active Valves

Other safety-related active valves are ASME B&PV Code 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 operate during a dynamic seismic and other RBV event.

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 Section 3.10.

Tests

Prior to installation of the safety-related valves, the following tests are performed: (1) shell hydrostatic test to the ASME B&PV Code requirements; (2) back seat and main seat leakage tests; (3) disk hydrostatic test; (4) functional tests to verify that the valve opens and closes 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. EQ 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.

Dynamic Load Qualification

The functionality of an active valve during and after a seismic and other RBV event may be demonstrated by an analysis or by a combination of analysis and test. The qualification of electrical and instrumentation components controlling valve actuation is discussed in Section 3.10. The valves are designed using either stress analyses or the pressure temperature rating requirements based upon design conditions. An analysis of the extended structure is performed for static equivalent dynamic loads applied at the center of gravity of the extended structure. Refer to 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.

When qualification of mechanisms that must change position to complete their safety-related function is based on dynamic testing or equivalent static load testing, operability testing is performed for the loads defined by the applicable events and conditions per Subsection 3.9.1.1 and Table 3.9-1.

The dynamic qualification testing procedure for valve operability is outlined below. A subject valve assembly is mounted in a test stand or fixture in a manner that conservatively represents typical valve installation(s). Each test valve assembly includes the actuator and accessories that are attached to an inservice valve. Additional discussion of test criteria and method is provided in Subsection 3.9.2.2, and also in the portions of Subsections 3.10.1 and 3.10.2 applicable to active valve assemblies.

Dynamic load qualification is accomplished in the following way:

1. The active valves are designed to have a fundamental frequency that is greater than the high frequency asymptote 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 to 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. Powered valve actuators and other accessory components directly attached onto the valve or actuator that are necessary for operation are qualified as operable during a dynamic event by appropriate qualification tests prior to installation on the valve. The powered actuator assemblies then have individual Seismic Category I supports attached to decouple the dynamic loads between the actuators and valves themselves.

The piping, stress analysis, and pipe support designs maintain the actuator assembly 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 other accessories, is qualified by a shake table test.

Valves that 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:

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:

- Stress analysis including the dynamic loads where applicable
- In-shop hydrostatic tests
- In-shop seat leakage test
- Periodic in-situ valve exercising and inspection to assure the functional capability of the valve

Pressure-Relief Valves

The active pressure relief valves are qualified as follows. 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 inspection, as applicable, and periodic valve removal, refurbishment, performance testing, and reinstallation are performed to assure the functional capability of the valve. Tests of the relief valve 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.

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 may be followed.

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. Installing the non-operating devices assures that the test panel has representative structural characteristics. 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 are requalified to the higher levels.

Documentation

All of the preceding requirements 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 a designated quality assurance representative has validated the tests. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.6 **Design and Installation of Pressure Relief Devices**

Main Steam Safety Relief Valves

SRV lift in the 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 RPV 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 the ASME

B&PV Code flow rating, increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves in a MS line is assumed in the analysis because simultaneous discharge is considered to induce maximum stress in the piping. 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 MSL, 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 ASME B&PV Code stress limits for service levels corresponding to load combination classification as normal, upset, emergency, and faulted are applied to the MS and discharge pipe.

Many of the SRV design parameters and criteria are specified in Sections 5.2 and 15.2. The procurement specification for the SRV defines the SRV requirements that are necessary to be consistent with the SRV parameters used in the steam line stress analysis.

Other Safety Relief and Vacuum Breaker 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 3.9.3.5 applies to the SRVs.

ESBWR SRVs and vacuum breakers are designed and manufactured in accordance with the ASME B&PV Code requirements.

The design of ESBWR SRVs incorporates SRV opening and pipe reaction load considerations required by ASME B&PV Code, Appendix O, and including the additional criteria of SRP, Subsection 3.9.3, Paragraph II.2 and those identified under NB-3658 of the ASME B&PV Code for pressure and structural integrity. Safety relief and vacuum relief valve and vacuum relief 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.

Depressurization Valves

The instantaneous opening of a DPV due to the explosion of the DPV operator results in a transient that produces impact loads and momentary unbalanced forces acting on the ICS and DPV piping system. The impact load forcing functions associated with DPV operation used in the piping analyses are determined by test. From the test data a representative force time-history is developed and applied as input to a time-history analysis of the piping. If these loads are defined to act in each of the three orthogonal directions, the responses are combined by the SRSS method.

The momentary unbalanced forces acting on the piping system are calculated and analyzed using the methods described above for SRV lift analysis.

The resulting loads on the DPV, the ICS lines, and the DPV piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. In accordance with Table 3.9-1 and Table 3.9-2, the ASME B&PV Code stress limits for service levels corresponding to load combination classification as normal, upset, emergency, and faulted are applied to the ICS and DPV discharge piping.

3.9.3.7 **Component Supports**

The establishment of the design/service loadings and limits is in accordance with the ASME Section III, Division 1, Article NCA-2000 and Subsection NF. These loadings and stress limits apply to the structural integrity of components and supports when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events. The combination of loadings and stress limits are included in the Design Specification of each component and support. Where the design and service stress limits specified in the ASME B&PV Code do not necessarily provide direction for the proper consideration of operability requirements for conditions which warrant consideration, Section II.3 and Appendix A of SRP 3.9.3, and Regulatory Guides 1.124 and 1.130 are used for guidance. Where these stress limits apply, the treatment of functional capability, including collapse, deformation and deflection limits are evaluated and appropriate information is developed for inclusion into the Design Specification.

ASME Section III component supports shall be designed, manufactured, installed and tested in accordance with all applicable codes and standards. Supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers and limit stops. Pipe whip restraints are not considered as pipe supports.

The design of bolts for component supports is specified in the ASME B&PV Code, 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 68.95 MPa (10,000 psi) on the nominal bolt area in shear or tension.

The design and installation of all anchor bolts is performed in accordance with Appendix B to ACI 349 "Anchoring to Concrete," subject to the conditions and limitations specified in RG 1.199.

It is preferable to attach pipe supports to embedded plates; however, surface-mounted base plates with undercut anchor bolts can be used in the design and installation of supports for safety–related components.

3.9.3.7.1 **Piping Supports**

Supports and their attachments for safety-related Code 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 design of the nuclear power plant structures, systems, and components will provide access for the performance of inservice testing and inservice inspection as required by the applicable ASME Code. The building structure component supports (connecting the NF support boundary component to the existing building structure) are designed in accordance with ANSI/AISC N690, Nuclear Facilities-Steel Safety-Related Structures for Design, Fabrication and Erection, or the AISC Specification for the Design, Fabrication, and Erection of Structural Steel. The applicable loading combinations and allowables used for design of supports are shown on Tables 3.9-10, -11, and -12. The stress limits are per ASME B&PV Code, Subsection NF and Appendix F.

Maximum calculated static and dynamic deflections of the piping at support locations do not exceed the allowable limits specified in the piping design specification.

Seismic Category II pipe supports are designed so that the SSE would not cause unacceptable structural interaction or failure. Support design follows the intent and general requirement specified in ASME B&PV Code, Nonmandatory Appendix F. This is used to evaluate the total design load condition with respect to the requirements of the SSE condition to ensure the structural integrity of the pipe supports are maintained.

The design of supports for the non-nuclear piping satisfies the requirements of ASME B31.1 Power Piping Code, Paragraphs 120 and 121.

For the major active valves identified in Subsection 3.9.3.5, the valve operators are not used as attachment points for piping supports.

The friction loads caused by unrestricted motion of the piping due to thermal displacements are considered to act on the support with a friction coefficient of 0.3, in the case of steel-to-steel friction. For stainless steel, Teflon, and other materials, the friction coefficient could be less. The friction loads are not considered during seismic or dynamic loading evaluation of pipe support structures.

For the design of piping supports, a deflection limit of 1.6 mm (1/16 in.) for erection and operation loadings is used, based on Welding Resource Council bulletin WRC-353 paragraph 2.3.2. For the consideration of loads due to SSE and in the cases involving springs, the deflection limit is increased to 3.2 mm (1/8 in.).

For frame type supports, the total gap is limited to 3.2 mm (1/8 inch). In general, this gap is adequate to avoid thermal binding due to radial thermal expansion of the pipe. For large pipes with higher temperatures, this gap is evaluated to assure that no thermal binding occurs. The minimum total gap is specified to ensure that it is adequate for the thermal radial expansion of the pipe to avoid any thermal binding.

The small bore lines (e.g., small branch and instrumentation lines) are supported taking into account the flexibility, and thermal and dynamic motion requirements of the pipe to which they connect. Subsection 3.7.3.16 provides details for the support design and criteria for instrumentation lines 50 mm (2 in.) and less where it is acceptable practice by the regulatory agency to use piping handbook methodology.

The design criteria and dynamic testing requirements for the ASME B&PV Code 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 B&PV 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.
3. Snubbers—The operating loads on snubbers are the loads caused by dynamic events (e.g., seismic, RBV due to LOCA, SRV and DPV discharge, discharge through a relief valve line, or valve closure) during various operating conditions. Snubbers restrain piping against response to the dynamic 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 cannot exceed the snubber load capacity for design, normal, upset, emergency and faulted conditions.

      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 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 ASME B&PV 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. Inspection, 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 program for inservice examination and testing of snubbers in the completed ESBWR construction is prepared in accordance with the requirements of ASME OM Code, Subsection ISTD, and the applicable industry and regulatory guidance including RG 1.192. The intervals for visual examination are the subject of Code Case OMN-13, which is accepted under the RG 1.192. The preparation and submittal of a program for the inservice testing and examination of snubbers is addressed in Subsection 3.9.9.*

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.

A thermal motion monitoring program is established for verification of snubber movement, adequate clearance and gaps, including motion measurements and acceptance criteria to assure compliance with ASME B&PV Code, Section III Subsection NF.

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 the rules and regulations of the ASME B&PV Code, Section III, Subsection NF and consider the following:

- Design requirements include analysis for normal, upset, emergency and faulted loads. Calculated loads are then compared against allowable loads as established by snubber vendor.
- Swing angles, as supplied by the snubber vendor, are incorporated into the design. Pipe movements in the horizontal and vertical direction are taken into account to prevent end bracket/paddle plate binding.
- Snubber stiffness, as supplied by the snubber vendor, is included in the piping analysis. Other support components such as the pipe clamp, extension piece and transition tube and structural auxiliary steel stiffness values are incorporated into the final determination of the stiffness value used in the analysis.

In multiple snubber applications where mismatch of end fitting clearance and lost motion could possibly exist, the synchronism of activation level or release rate is evaluated, if deemed necessary, in the piping analysis model when this application could be considered critical to the functionality of the system, such as a multiple snubber application located near rotating equipment. Equal load sharing of multiple snubber supports is not assumed if a mismatch in end fitting clearances exists and is evaluated as a part of this assessment.

ii. A list of snubbers on systems which experience sufficient thermal movement to measure cold to hot position is provided as part of the testing program after the piping analysis has been completed.

iii. The snubbers are tested to ensure 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. Production and qualification test programs for both hydraulic and mechanical snubbers are carried out by the snubber vendors in accordance with the snubber installation instruction manual required to be furnished by the snubber supplier. Acceptance criteria to assure compliance with ASME B&PV Code Section III Subsection NF, and other applicable codes, standards and requirements are as follows:

- Snubber production and qualification test programs are carried out by strict adherence to the manufacturer's snubber installation and instruction manual, which is

prepared by the snubber manufacturer and subjected to review by the holder for compliance with the applicable provisions of the ASME B&PV Code of record. The test program is periodically audited during implementation by the holder for compliance.

- All snubbers will be inspected and tested for compliance with the design drawings and functional requirements of the procurement specifications.
- *All snubbers are inspected and tested. No sampling methods may be used in the qualification tests.*
- *All snubbers are load rated by testing in accordance with the snubber manufacturer's testing program and in compliance with the applicable sections of ASME QME-1-2007, Subsection QDR and ASME OM Code, Subsection ISTD.*
- *Design compliance of the snubbers per ASME B&PV Code Section III Paragraph NF-3128, and Subparagraphs NF-3411.3 and NF-3412.4.*
- *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. The functional parameters cited in Subparagraph NF-3412.4 are included in the snubber qualification and testing program. Other parameters in accordance with applicable ASME QME-1-2007 and the ASME OM Code will be incorporated.*
- *The codes and standards used for snubber qualification and production testing are as follows:*
  - *ASME B&PV Code Section III (Code of Record date) and Subsection NF.*
  - *ASME QME-1-2007, Subsection QDR and ASME OM Code, Subsection ISTD.*
- *All large bore hydraulic snubbers include full Service Level D load testing, including verifying bleed rates, control valve closure within the specified velocity ranges and*

drag forces/breakaway forces are acceptable in accordance with ASME QME-1-2007 and ASME OM Codes.

iv. All safety-related components that utilize snubbers in their support systems will be identified and inserted into the Final Safety Analysis Report in table format and will include the following:

- identification of systems and components.
- number of snubbers utilized in each system and on that component.
- snubber type (s) – (hydraulic or mechanical) – and name of supplier.
- constructed to ASME B&PV Code Section III, Subsection NF or other.
- snubber use such as shock, vibration, or dual purpose.
- those snubbers identified as dual purpose or vibration arrestor type, will include an indication if both snubber and component were evaluated for fatigue strength.

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 that 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 Preservice and Inservice Examination and Testing

Preservice Examination and Testing

The preservice examination plan for snubbers is prepared in accordance with the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Subsection ISTD, and the additional requirements of this section. The preservice examinations are made after snubber installation but not more than

6 months prior to initial system pre-operational testing. The preservice examination verifies the following:

i. There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

ii. The snubber load rating, 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 the recommended level and is not to be 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 preservice examination and initial system pre-operational tests exceeds 6 months, reexamination of Items i, iv, and v is performed. Snubbers that are installed incorrectly or otherwise fail to meet the above requirements are repaired or replaced and re-examined in accordance with the above criteria.

A preservice thermal movement examination is also performed; during initial system heatup and cooldown, for systems whose design operating temperature exceeds 121°C (250°F), snubber thermal movement is verified.

Additionally, preservice operational readiness testing is performed on all snubbers. The operational readiness test is performed to verify the parameters of ISTD-5120. Snubbers that fail the preservice operational readiness test are evaluated to determine the cause of failure, and are retested following completion of corrective action(s).

Snubbers that are installed incorrectly or otherwise fail preservice testing requirements are re-installed correctly, adjusted, modified, repaired or replaced, as required. Preservice examination and testing is re-performed on installation- corrected, adjusted, modified, repaired or replaced snubbers as required.

The preservice inspection and testing programs for snubbers will be completed in accordance with milestones described in Section 13.4.

Inservice Examination and Testing

Inservice examination and testing of all safety-related snubbers is conducted in accordance with the requirements of the ASME OM Code, Subsection ISTD. Inservice examination is initially performed not less than two months after attaining 5 percent reactor power operation and will be completed within 12 calendar months after attaining 5 percent reactor power. Subsequent examinations are performed at intervals defined by ISTD-4252 and Table ISTD-4252-1. Examination intervals, subsequent to the third

interval, are adjusted based on the number of unacceptable snubbers identified in the then current interval.

An inservice visual examination is performed on all snubbers to identify physical damage, leakage, corrosion, degradation, indication of binding, misalignment or deformation and potential defects generic to a particular design. Snubbers that do not meet visual examination requirements are evaluated to determine the root cause of the unacceptability, and appropriate corrective actions (e.g., snubber is adjusted, repaired, modified, or replaced) are taken. Snubbers evaluated as unacceptable during visual examination may be accepted for continued service by successful completion of an operational readiness test.

Snubbers are tested inservice to determine operational readiness during each fuel cycle, beginning no sooner than 60 days before the scheduled start of the applicable refueling outage. Snubber operational readiness tests are conducted with the snubber in the as-found condition, to the extent practical, either in place or on a test bench, to verify the test parameters of ISTD-5210. When an in-place test or bench test cannot be performed, snubber subcomponents that control the parameters to be verified are examined and tested. Preservice examinations are performed on snubbers after reinstallation when bench testing is used (ISTD-5224), or on snubbers where individual subcomponents are reinstalled after examination (ISTD-5225).

Defined test plan groups (DTPG) are established and the snubbers of each DTPG are tested according to an established sampling plan each fuel cycle. Sample plan size and composition are determined as required for the selected sample plan, with additional sampling as may be required for that sample plan based on test failures and failure modes identified. Snubbers that do not meet test requirements are evaluated to determine root cause of the failure, and are assigned to failure mode groups (FMG) based on the evaluation, unless the failure is considered unexplained or isolated. The number of unexplained snubber failures not assigned to an FMG determines the additional testing sample. Isolated failures do not require additional testing. For unacceptable snubbers, additional testing is conducted for the DTPG or FMG until the appropriate sample plan completion criteria are satisfied.

Unacceptable snubbers are adjusted, repaired, modified, or replaced. Replacement snubbers meet the requirements of ISTD-1600. Post-maintenance examination and testing, and examination and testing of repaired snubbers, is done to ensure that test

parameters that may have been affected by the repair or maintenance activity are verified acceptable.

Service life for snubbers is established, monitored and adjusted as required by ISTD-6000 and the guidance of ASME OM Code Nonmandatory Appendix F.

The inservice inspection and testing programs for snubbers will be completed in accordance with milestones described in Section 13.4.

The inservice examination and testing plan for snubbers is prepared in accordance with the requirements of the ASME OM Code, Subsection ISTD and is in conformance with the relevant requirements of 10 CFR 50 Part B, Appendix A, GDC 1.

f. Snubber Support Data

[START COM 3.9-003] For the ASME Class 1, 2, and 3 systems listed in Tier 1, Section 3.1, that contain snubbers, a plant specific table will be prepared in conjunction with the closure of the system-specific ITAAC for piping and component design and will include the following specific snubber information. **[END COM 3.9-003]**

i. the general functional requirement (i.e., shock, vibration, dual purpose) for each system and component using snubbers including the number and location of each snubber. If either dual-purpose or arrestor type, indicate whether the snubber or component was evaluated for fatigue strength.

ii. operating environment.

iii. applicable codes and standards.

iv. list type of snubber (i.e., hydraulic, mechanical), materials of construction, standards for hydraulic fluids and lubricants, and the corresponding supplier.

v. environmental, structural, and performance design verification tests.

vi. production unit functional verification tests and certification.

vii. packaging, shipping, handling, and storage requirements.

viii. description of provisions for attachments and installation.

ix. quality assurance and assembly quality control procedures for review and acceptance by the purchaser.

[START COM 3.9-005] This information will be included in the UFSAR as part of a subsequent UFSAR update. **[END COM 3.9-005]**

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 the ASME B&PV Code, Subsection NF-3000.

Struts are passive supports, requiring little maintenance and inservice inspection, and are normally 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 are not used at locations where restraint of pipe movement to thermal expansion significantly increases the secondary piping stress ranges or equipment nozzle loads.

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 are obtained from an analysis, and are confirmed not to exceed the design loads for various operating conditions.

5. Frame Type (Linear) Pipe Supports — Frame type pipe 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 pipe. They act as guides to allow axial and rotational movement of the pipe but act as rigid restraints to lateral movement in either one or two directions. Frame type pipe supports are designed in accordance with the ASME B&PV Code, NF-3000.

   Frame type pipe supports are passive supports, requiring little maintenance and inservice inspection, and are normally 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 are not used at locations where restraint of pipe movement to thermal expansion significantly increases the secondary piping stress ranges or equipment nozzle loads.

   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, which are assured not to exceed the design loads for various operating conditions.

   Any hot or cold gaps required by the qualifying pipe stress analysis results are incorporated in the design. Where friction between the pipe and frame support occurs as a result of sliding, an appropriate coefficient of friction is used in order to calculate friction loading on the support. Seismic inertia loads as well as static seismic loads are considered in the design of frame supports covered by ASME Section III Subsection NF.

   For insulated pipes, special pipe guides with one or two-way restraint (two or four trunnions welded to a pipe clamp) may be used in order to minimize the heat loss of piping systems. For small bore pipe guides, it could be acceptable to cut the insulation around the support frame, although this must be indicated in the support specification.

6. Special Engineered Pipe Supports are not used.

3.9.3.7.2 Reactor Pressure Vessel Sliding Supports

The ESBWR RPV sliding supports are sliding supports as defined by NF-3124 of the ASME B&PV Code and are designed as an ASME B&PV Code Class 1 component support per the requirements of the ASME B&PV Code, Subsection NF. The loading conditions and stress criteria are given in Table 3.9-1 and Table 3.9-2, and the calculated stresses meet the ASME B&PV Code allowable stresses at all locations for various plant operating conditions. The stress level margins assure the adequacy of the RPV sliding supports.

3.9.3.7.3 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as a safety-related linear type component support in accordance with the requirements of ASME B&PV Code Section III, Subsection NF. The stabilizer provides a reaction point near the upper end of the RPV to resist horizontal loads caused by effects such as earthquake, pipe rupture, and RBV. The design loading conditions and stress criteria are given in Table 3.9-2, and the calculated stresses meet the ASME B&PV Code allowable stresses in the critical support areas for various plant operating conditions.

3.9.3.7.4 Floor-Mounted Major Equipment

Because the major active valves are supported by piping and not tied to building structures, valve “supports” do not exist.

The condensers in the Isolation Condenser System (ICS) and Passive Containment Cooling System (PCCS) are analyzed to verify the adequacy of their support structure under various plant operating conditions. The analysis applies the maximum sheer, moment, and accelerations calculated from the seismic response analysis for the Reactor Building at the attachment locations on the pool floor. In the analysis, no credit is taken for damping effects of the pool water. Additionally, the mass of the condensers is increased by an amount equivalent to the weight of water they displace. This conservative factor accounts for the hydrodynamic effects that include impulsive loads and convective loads (sloshing of the pool water). In all cases, the load stresses in the critical support areas are within ASME B&PV Code allowables.

3.9.3.8 Other ASME B&PV Code Component Supports

The ASME B&PV Code component supports and their attachments (other than those discussed in the preceding subsection) are designed in accordance with Subsection NF of the ASME B&PV Code up to the interface with the building structure. The intermediate building structural steel component supports are designed in accordance with the codes as specified in Section 3.8. 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 3.9.3.7. The stress limits are per ASME B&PV Code, Subsection NF and Appendix F. The supports are evaluated for buckling in accordance with ASME B&PV Code.

3.9.3.9 Threaded Fasteners – ASME B&PV Code Class 1, 2 and 3

3.9.3.9.1 Material Selection

Material used for threaded fasteners complies with the requirements of ASME B&PV Code Section III NB-2000, NC-2000, ND-2000 or NF-2000 as appropriate. Fracture toughness testing is performed in accordance with ASME B&PV Code Section III NB-2300, NC-2300 or ND-2300, as appropriate. For verification of conformance to the applicable ASME B&PV Code requirements, a chemical analysis is required for each heat of material and testing for mechanical properties is required on samples representing each heat of material and, where applicable, each heat treat lot.

The criteria of ASME B&PV Code Section III NB-2200, NC-2200 or ND-2200 rather than the material specification criteria applicable to the mechanical testing shall be applied if there is a conflict between the two sets of criteria. For safety-related threaded fasteners, documentation related to fracture toughness (as applicable) and certified material test reports are provided as part of the ASME B&PV Code records that are provided at the time the parts are shipped, and are part of the required records that are maintained at the site.

Threaded fasteners are selected for compatibility with the materials of the component being joined and the piping system fluids. The selection process considers deterioration which may occur during service as a result of corrosion, radiation effects, or instability of material.

3.9.3.9.2 Special Materials Fabrication Processes and Special Controls

The design of threaded fasteners complies with ASME B&PV Code Section III NB-3000, NC-3000 or ND-3000, as appropriate. Fabrication of threaded fasteners complies with ASME B&PV Code Section III NB-4000, NC-4000 or ND-4000, as appropriate. Inspection of threaded fasteners complies with ASME B&PV Code Section III NB-2500, NC-2500 or ND-2500, as applicable.

Lubricants with deliberately added halogens, sulfur, or lead are not used for any RCPB components or other components in contact with reactor water. Lubricants containing molybdenum sulfide (disulfide or polysulfide) are not to be used for any safety-related application. For ferritic steel threaded fasteners, conversion coatings, such as the Parkerizing process are suitable and may be used. If fasteners are plated, low melting point materials, such as zinc, tin, cadmium, etc., are not used.

3.9.3.9.3 Preservice and Inservice Inspection Requirements

Preservice Inspection (PSI) and Inservice inspection is performed in accordance with ASME B&PV Code, Section XI. The requirements for pressure retaining Class 1 bolting are addressed as Category B-G-1 for bolting greater than 2 inches in diameter and B-G-2 for bolting with diameters 2 inches and less. The Class 1 pressure retaining bolting sample is limited to the bolting on the heat exchangers, piping, pumps, and valves that are selected for examination in the in-service inspection program.

Category B-G-2 requires visual, VT-1, examination of the selected bolting. For Class 1, 2 and 3 systems, the bolted connections are examined for leakage (VT-2) during the system pressure tests required by ASME Section XI. For safety-related threaded fasteners, documentation related to PSI is provided as part of the ASME B&PV Code records that are provided at the time the parts are shipped, and are part of the required records that are maintained at the site.

3.9.4 **Control Rod Drive System**

This subsection addresses the CRD system as discussed in SRP 3.9.4. The CRD system consists of the control rods and the related mechanical components that provide the means for mechanical movement. As discussed in GDC 26 and 27, the CRD system provides one of the independent reactivity control systems. The rods and the drive mechanisms are capable of reliably controlling reactivity changes either under conditions of AOOs, or under postulated accident conditions. A positive means for inserting the rods is always maintained to ensure appropriate margin for malfunction, such as stuck rods. Because the CRD system is a safety-related system and portions of the CRD system are a part of the RCPB, the system is designed, fabricated, and tested to quality standards commensurate with the safety-related functions to be performed. This provides an extremely high probability of accomplishing the safety-related functions either in the event of AOOs or in withstanding the effects of postulated accidents and natural phenomena such as earthquakes, as discussed in GDC 1, 2, 14, 26, 27 and 29 and 10 CFR 50.55a.

The plant design meets the requirements of the following regulations:

1. GDC 1 and 10 CFR 50.55a, as they relate to the CRD system being designed to quality standards commensurate with the importance of the safety-related functions to be performed.
2. GDC 2, as it relates to the CRD system being designed to withstand the effects of an earthquake without loss of capability to perform its safety-related functions.
3. GDC 14, as it relates to the RCPB portion of the CRD system being designed, constructed, and tested for the extremely low probability of leakage or gross rupture.
4. GDC 26, as it relates to the CRD system being one of the independent reactivity control systems that is designed with appropriate margin to assure its reactivity control function under conditions of normal operation including AOOs.
5. GDC 27, as it relates to the CRD system being designed with appropriate margin, and in conjunction with the emergency core cooling system, being capable of controlling reactivity and cooling the core under postulated accident conditions.
6. GDC 29, as its relates to the CRD system, in conjunction with reactor protection systems, being designed to assure an extremely high probability of accomplishing its safety-related functions in the event of AOOs.

The CRD system includes electrohydraulic FMCRD mechanisms, the HCU assemblies, the condensate supply system, and power for FMCRD motors. The system extends inside RPV to the coupling interface with the control rod blades.

3.9.4.1 Descriptive Information on Control Rod Drive System

Descriptive information on the FMCRDs as well as the entire CRD system is contained in Subsection 4.6.1.

3.9.4.2 Applicable Control Rod Drive System Design Specification

The CRD system, which is designed to meet the functional design criteria outlined in Subsection 4.6.1, consists of the following:

- electro-hydraulic fine motion control rod drive
- HCU
- hydraulic power supply (pumps)
- electric power supply (for FMCRD motors)
- interconnecting piping
- flow and pressure and isolation valves
- instrumentation and electrical controls

Those components of the CRD system forming part of the primary pressure boundary are designed according to the ASME B&PV Code, 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 3.2-3, 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, and seismic testing in Subsection 3.9.2.2.

3.9.4.3 Design Loads and Stress Limits

Allowable Deformations

The ASME B&PV Code, Section III components of the CRD system have been evaluated analytically and the design loading conditions and stress criteria are as given in Table 3.9-1 and Table 3.9-2, and the calculated stresses meet the ASME B&PV Code allowable stresses. For the non-ASME B&PV Code components, the ASME B&PV Code, Section III 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 **Control Rod Drive Performance Assurance Program**

The following CRD tests are described within Section 4.6.3:

- factory quality control tests
- functional tests
- operational tests
- acceptance tests
- surveillance tests

3.9.5 **Reactor Pressure Vessel Internals**

This subsection addresses the RPV internals as discussed in SRP 3.9.5. The RPV internals consist of all the structural and mechanical elements inside the reactor vessel. Safety-related structures and components are constructed and tested to quality standards commensurate with the importance of the safety-related functions to be performed, and designed with appropriate margins to withstand effects of AOOs, normal operation, natural phenomena such as earthquakes, postulated accidents including LOCA, and from events and conditions outside the nuclear power unit as discussed in GDC 1, 2, 4 and 10 and 10 CFR 50.55a.

The plant meets the requirements of the following regulations:

1. GDC 1 and 10 CFR 50.55a, as they relate to reactor internals; the reactor internals are designed to quality standards commensurate with the importance of the safety-related functions to be performed.
2. GDC 2, as it relates to reactor internals; the reactor internals are designed to withstand the effects of earthquakes without loss of capability to perform their safety-related functions.
3. GDC 4, as it relates to reactor internals; reactor internals are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated LOCA. Dynamic effects associated with postulated pipe ruptures are excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.
4. GDC 10, as it relates to reactor internals; reactor internals are 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 AOOs.

This subsection identifies and discusses the structural and functional integrity of the major RPV internals, including core support structures.

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and in-core nuclear instrumentation) are as follows:

- Core Support Structures
  - shroud
  - shroud support
  - core plate (and core plate hardware)
  - top guide (and top guide hardware)
  - fuel supports (orificed fuel supports and peripheral fuel supports)
  - CRGTs
  - non-pressure boundary portion of CRDHs
- Internal Structures (Components marked with an * are nonsafety-related.)
  - chimney* and partitions*
  - chimney head* and steam separator assembly*
  - steam dryer assembly*
  - feedwater spargers*
  - SLC header and spargers and piping
  - RPV vent assembly*
  - in-core guide tubes and stabilizers
  - surveillance sample holders*
  - non-pressure boundary portion of in-core housings

A general assembly drawing of the important reactor components is shown in Figure 3.9-7.

The floodable inner volume of the RPV can be seen in Figure 3.9-2. It is the volume up to the level of the GDCS equalizing nozzles.

The design arrangement of the reactor internals, such as the shroud, chimney, steam separators and guide tubes, is such that one end is unrestricted and thus free to expand.

3.9.5.1 Core Support Structures

The core support structures consist of those items listed in Subsection 3.9.5 and are safety-related as defined within 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 locate and support the fuel assemblies. Figure 3.9-3 shows the reactor vessel internal flow paths.

Shroud

The shroud and chimney 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 region or chimney surrounds the core discharge plenum, which is bounded by the chimney head on top and the top guide below. The central region of the shroud surrounds the active fuel. This section is bounded at the top by the top guide and at the bottom by the core plate. The lower region, surrounding part of the lower plenum, is welded to the support legs. The shroud provides the horizontal support for the core by supporting the core plate and top guide. A conceptual design of the connection between the shroud, chimney, and top guide is shown in Figure 3.9-8.

Shroud Support

The RPV shroud support is designed to support the shroud and the components connected to the shroud. The RPV shroud support is a ring supporting the core plate and series of vertical support legs supporting the ring. The support legs are welded to the vessel bottom head and the bottom of the support ring.

Core Plate

The core plate consists of a circular stainless steel plate with round openings. The core plate provides lateral support and guidance for the CRGTs, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core plate. The core plate is bolted between the support ring and shroud. A conceptual design of the connection between the core plate, support ring and shroud is shown in Figure 3.9-9.

Top Guide

The top guide consists of a circular plate with square openings for fuel. 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 in-core flux monitors and startup neutron sources. The top guide is mechanically attached to the top of the shroud and provides a flat surface for the chimney flange. The chimney is bolted to the top surface of the top guide as shown in Figure 3.9-8.

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 contains 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 a CRGT. 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.

CRGTs

The CRGTs located inside the vessel extend from the top of the CRDHs 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 CRDH, which, in turn, transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. The CRGTs also include coolant flow holes near the top that are aligned with the coolant flow holes in the orificed fuel supports.

3.9.5.2 Internal Structures

The internal structures consist of those items listed in Subsection 3.9.5, and are safety-related or nonsafety-related as noted. These components direct and control coolant flow through the core or support safety-related and nonsafety-related functions.

Chimney and Partitions

These components are nonsafety-related internal components. The chimney is a long cylinder mounted to the top guide that supports the steam separator assembly. The chimney provides the driving head necessary to sustain the natural circulation flow. The chimney forms the annulus separating the subcooled recirculation flow returning downward from the steam separators and feedwater from the upward steam-water mixture flow exiting the core. The chimney cylinder is flanged at the bottom and top for attachment to the top guide and the chimney head, respectively. Inside the chimney are partitions that separate groups of 16 fuel assemblies. These partitions act to channel the mixed steam and water flow exiting the core into smaller chimney sections, limiting cross flow and flow instabilities, which could result from a much larger diameter open chimney. The partitions do not extend to the top of the chimney, thereby forming a plenum or mixing chamber for the steam/water mixture prior to entering the steam separators.

Chimney Head and Steam Separator Assembly

The chimney head and standpipes/steam separators are nonsafety-related internal components. The chimney head and steam separators assembly includes the upper flanges and bolts, and forms the top of the core discharge mixture plenum. 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 chimney head. The steam separators have no moving parts. In each separator, the steam/water mixture rising through the standpipe passes through vanes that impart a spin and 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 separator assembly is removable from the RPV on a routine basis.

Steam Dryer Assembly

The steam dryer assembly is a nonsafety-related component. 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 drain ducts 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 dryer assembly includes the dryer banks, dryer supply and discharge ducting, drain collecting trough, drain duct, and a skirt that 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 RPV.

During normal refueling outages, the ESBWR steam dryer is supported from the floor of the equipment pool by the lower support ring that is located at the bottom edge of the skirt. The steam dryer is installed and removed from the RPV by the reactor building overhead crane. A steam separator and steam dryer lifting device, which attaches to four steam dryer lifting rod eyes, is used for lifting the steam dryer. Guide rods in the RPV are used to aid steam dryer installation and removal. Upper and lower guides on the steam dryer assembly are used to interface with the guide rods.

Feedwater Spargers

These are nonsafety-related components. Each of two feedwater lines is connected to spargers through three RPV nozzles. The feedwater spargers deliver makeup water to the reactor during plant start up, power generation and plant shutdown modes of operation. The RWCU/SDC system and CRD system, upon low water level, also utilize the feedwater spargers.

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 by a tee and is shaped to conform to the curve of the vessel wall. The sparger tee inlet is connected to the thermal sleeve arrangement. 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. The feedwater also serves to condense steam in the region above the downcomer annulus and to subcool the water flowing down the annulus region.

SLC Header and Sparger and Piping

These are safety-related components. Each of two SLC nozzles feeds vertical piping extending down from the SLC nozzles to a header. Each header feeds two distribution lines extending down from the header to about the bottom of the fuel, and four injection lines with nozzles penetrating the

shroud at four different levels (elevations). The injection lines enable the sodium pentaborate solution to be injected around the periphery of the core.

RPV Vent Assembly

This is designed as a nonsafety-related component. Only the piping external to the vessel is RCPB, and the vent function is a nonsafety-related operation.

The head vent assembly passes steam and noncondensable gases from the reactor head to the steamlines during startup and operation. During shutdown and filling for hydrostatic testing, 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.

In-Core Guide Tubes and Stabilizers

These are safety-related components. The guide tubes protect the in-core instrumentation from the flow of water in the bottom head plenum and provide a means of positioning fixed detectors in the core. The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing to the top of the core plate. The power range detectors for the power range monitoring units and the startup range neutron monitor detectors are inserted through the guide tubes. A conceptual design of the in-core guide tube and in-core monitor connection to the core plate is show in Figure 3.9-12.

A latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. A conceptual design of the In-core lateral supports is shown in Figures 3.9-10 and 3.9-11.

Surveillance Sample Holders

These are nonsafety-related components. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The baskets hang from the brackets that are attached to the inside of the reactor vessel wall and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor vessel itself.

3.9.5.3 Loading Conditions

Events to be Evaluated

Examination of the spectrum of conditions for which the safety-related design bases (Subsection 3.9.5.4) must be satisfied by core support structures and safety-related internal components reveals three significant load events:

- RPV Line Break Accident — a break in any one line between the reactor vessel nozzle and the isolation valve (the accident results in significant pressure differentials across some of the structures within the reactor and RBV caused by suppression pool dynamics).

- Earthquake — subjects the core support structures and reactor internals to significant forces as a result of ground motion and consequent RBV.
- SRV or DPV Discharge — RBV caused by suppression pool dynamics and structural feedback.

The faulted conditions for the RPV internals are discussed in Subsection 3.9.1.4. Loading combination and analysis for safety-related reactor internals including core support structures are discussed in Subsection 3.9.5.4.

Reactor Internal Pressure Differences

For reactor internal pressure differences, the events at normal, upset, emergency and faulted conditions are considered.

The TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs, infrequent events and accidents (e.g., LOCA). The analytical model of the vessel consists of axial and radial 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.

In order to determine the maximum pressure differences across the reactor internals, a two sigma statistical uncertainty study is performed to determine the upper bound pressure difference adders that are applied to the nominal pressure differences.

Table 3.9-3 summarizes the maximum pressure differentials that result from the limiting events among the AOOs, infrequent events and accidents (e.g., LOCA).

Seismic and Other RBV Events

The loads due to earthquake and other RBV acting on the structure within the reactor vessel are based on a dynamic analysis methods described in Section 3.7.

Steam Dryer Acoustic Loading Effects from Safety-Relief Valve Standpipes and Main Steam Piping

The safety relief valves (SRVs) and safety valves (SVs) standpipes and main steam branch lines in the ESBWR are specifically designed to preclude first and second shear layer wave acoustic resonance conditions from occurring and to avoid pressure loads on the steam dryer at plant normal operating conditions. Appropriate selection of SRV standpipes vertical height between main steam piping and the valves, along with selection of valve entrance effects, minimizes vortex generation. Design of piping is based on Strouhal numbers outside the range for which adverse impacts due to acoustic resonance would occur (based on resonance frequency determined using velocity at the speed of sound). Calculations performed as described below show that design features and selections can acceptably eliminate first and second shear wave resonances in SRV/SV standpipes and main steam piping.

For standpipe configuration, main steam and SRV/SV configuration (dimensions and arrangement) and attachment of standpipes to the main steam piping are considered. System geometry and flow rates are used to calculate critical Strouhal ranges. Then the Strouhal values for flow at 100% and 102% power levels are calculated to ensure that the values at normal operating flow rates are not within the critical range and that additional margin is provided.

As shown on Figure 5.2-2, the ESBWR main steam system has two longer main steam lines with 5 SRV/SVs, and two shorter main steam lines with 4 SRV/SVs, mounted perpendicular to the main steam pipe centerline. For eliminating main steam flow acoustic frequency at branchline connections, boundary conditions at the connections are evaluated to ensure flow disturbances are not on the same order as acoustic waves that may pass through the system. Design features such as dimensions, diameter ratios, lengths of piping from reactor vessel nozzles, entrance effects, and flow rate effects are considered in the evaluation.

Acceptance criteria are to demonstrate through the evaluation that the final as-built design of the main steam line and SRV/SV branch piping geometry precludes first and second shear layer wave acoustic resonance conditions from occurring and results in no significant pressure loads on the steam dryer at plant normal operating conditions. This is demonstrated by analysis showing that the final design maximum velocity, Strouhal range, and power level are outside the critical range of values where first and second shear resonances would occur. See Subsection 3.9.2.1 for piping vibration and dynamics effects testing during preoperational and startup testing. See NEDE-33313P (Reference 3.9-7), Appendix B, for information regarding this design approach.

3.9.5.4 Design Bases

Safety-Related Design Bases

The reactor internals, including core support structures, meet the following safety-related design bases:

- The reactor vessel nozzles and internals are 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.
- Deformation of internals is limited to assure that the control rods and core standby cooling systems can perform their safety-related functions.
- Mechanical design of applicable structures assures that the above safety-related design bases are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

Power Generation Design Bases

The reactor internals, including core support structures, are designed to the following power generation design bases:

- The internals provide the proper coolant distribution during all anticipated normal operating conditions to full power operation of the core without fuel damage.
- The internals are arranged to facilitate refueling operations.
- The internals are designed to facilitate inspection.

Design Loading Categories

The basis for determining faulted dynamic event loads on the reactor internals is shown in Section 3.7. Table 3.9-2 shows the load combinations used in the analysis.

Core support structures and safety class internals stress limits are consistent with the ASME B&PV Code, Subsection NG. For these components, 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 the following paragraphs.

Stress and Fatigue Limits for Core Support Structures

The design and construction of the core support structures are in accordance with the ASME B&PV Code, Subsection NG.

Stress, Deformation, and Fatigue Limits for Safety Class and Other Reactor Internals (Except Core Support Structures)

For safety-related 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 manufacturers' standards, or by empirical methods based on field experience and testing. For the quantity minimum safety factor (SF_{min}) 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 RPV such as control rods, which must move during accident conditions, are examined to determine if adequate clearances exist during emergency and faulted conditions. The forcing functions applicable to the reactor internals are discussed in Subsection 3.9.2.

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 paragraph NG-3000 of the ASME B&PV Code and are constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

The reactor internal structures classified as nonsafety-related in Subsection 3.9.5 are not ASME B&PV Code components, but their design complies with the requirements of ASME B&PV Code,

Subsection NG-3000 except for the weld quality and fatigue factors for secondary structural non-load bearing welds. Primary structural load bearing welds use quality and fatigue factors as given in NG-3000. The steam dryer assembly weld quality and fatigue factor methodology is discussed in Reference 3.9-7.

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of certain ASME B&PV Code, Section III, Class 1, 2, and 3 pumps and valves is performed in accordance with the ASME OM Code as required by 10 CFR 50.55a(f), including limitations and modifications set forth in 10 CFR 50.55a. The Inservice Testing (IST) Program does not include any non-ASME B&PV Code Class valves. The design of the nuclear power plant structures, systems, and components will provide access for the performance of inservice testing and in-service inspection as required by the applicable ASME B&PV Code.

Inservice testing of pumps and valves is in conformance with the relevant requirements of 10 CFR Part 50, Appendix A, GDC 1, 37, 40, 43, 46, 54, and 10 CFR 50.55a(f). The relevant requirements are as follows:

1. GDC 1, as it relates to testing safety-related components to quality standards commensurate with the importance of the safety-related functions to be performed.
2. GDC 37, as it relates to periodic functional testing of the emergency core cooling system to ensure the leak tight integrity and performance of its active components.
3. GDC 40, as it relates to periodic functional testing of the containment heat removal system to ensure the leak tight integrity and performance of its active components.
4. GDC 43, as it relates to periodic functional testing of the containment atmospheric cleanup systems to ensure the leak tight integrity and the performance of the active components, such as pumps and valves.
5. GDC 46, as it relates to periodic functional testing of the cooling water system to ensure the leak tight integrity and performance of the active components.
6. GDC 54, as it relates to piping systems penetrating containment being designed with the capability to test periodically the operability of the isolation and determine valve leakage acceptability.
7. Subsection 50.55a(f) of 10 CFR, as it relates to including pumps and valves whose function is required for safe operation in the IST Program to verify operational readiness by periodic testing.

The IST Program includes periodic tests and inspections that demonstrate the operational readiness of safety-related components and their capability to perform their safety-related functions. The IST Program is based on the requirements of the ASME OM Code, Subsections ISTA, ISTB, ISTC and (mandatory) Appendix I. The specific ASME OM Code requirements for functional testing of pumps are found in the ASME OM Code, Subsection ISTB, requirements for

inservice testing of valves are found in the ASME OM Code, Subsection ISTC, and requirements for inservice testing of pressure relief devices are found in ASME OM Code, (mandatory) Appendix I. General requirements for inservice testing are found in ASME OM Code, Subsection ISTA.

The requirements for system pressure testing are defined in ASME Code Section XI, Subsection IWA-5000; this testing, which verifies pressure boundary integrity, is included within the scope of the in-service inspection program described in Subsection 5.2.4 and Section 6.6.

The requirements for preservice and inservice examination and testing of dynamic restraints are defined in the ASME OM Code, Subsection ISTD. This program is described in Subsection 3.9.3.7.1.

Milestones for implementation of the ASME OM Code preservice and inservice testing programs are defined in Section 13.4.

3.9.6.1 **InService Testing Valves**

Certain ASME B&PV Code Class 1, 2, and 3 valves and pressure relief devices are subject to inservice testing in accordance with the ASME OM Code, Subsection ISTC and/or Appendix I, including the general requirements in ISTA. Inservice testing of valves assesses operational readiness including actuating and position-indicating systems. The valves that are subject to inservice testing include those valves that perform a specific function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident. In addition, pressure relief devices used for protecting systems or portions of systems that perform a function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident, are subject to inservice testing.

The IST Program does not require testing of nonsafety-related valves. Any nonsafety-related valves included in the IST Program as part of regulatory treatment of nonsafety-related systems (RTNSS, see Appendix 19A) are considered augmented components and tested commensurate with their functions.

Valves subject to inservice testing in accordance with the ASME OM Code are indicated in Table 3.9-8.

Each valve subject to inservice testing is also tested during the preservice test (PST) period. Preservice tests are conducted under conditions as near as practicable to those expected during subsequent inservice testing. Valves (or the control system) that have undergone maintenance that could affect performance, or valves that are repaired or replaced, are re-tested to verify performance parameters that could have been affected are within acceptable limits. Safety and relief valves and nonreclosing pressure relief devices are preservice tested in accordance with the requirements of the ASME OM Code, Mandatory Appendix I.

Active valve dynamic qualification and pre-installation testing requirements to assure valve operability are addressed in Subsection 3.9.3.5. Periodic operability (non-ASME OM Code) testing for power-operated valves is described in Subsection 3.9.6.8.

3.9.6.1.1 **Valve Exemptions**

ASME OM Code ISTC-1200 provides exemptions from the IST Program for certain ASME B&PV Code Class 1, 2, and 3 valves provided that they are not required to perform a specific function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident. The following valves are exempt from the ASME OM Code, Subsection ISTC:

1. valves used only for operating convenience such as vent, test, drain and instrument valves.
2. valves used only for system control, such as pressure regulating valves.
3. valves used only for system or component maintenance.
4. skid-mounted valves provided they are justified and adequately tested.
5. valves used for external control and protection systems responsible for sensing plant conditions and providing signals for valve operation (e.g., solenoid valves on air operated valves).

3.9.6.1.2 **Valve Categories**

Non-exempt ASME B&PV Code Class 1, 2 and 3 valves are categorized in accordance with the ASME OM Code Subsection ISTC-1300 as follows:

1. Category A – valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function(s).
2. Category B – valves for which seat leakage in the closed position is inconsequential for fulfillment of the required function(s).
3. Category C – valves that are self-actuating in response to some system characteristic, such as pressure (relief valve) or flow direction (check valve) for fulfillment of the required function(s).
4. Category D – valves that are actuated by an energy source capable of only one operation, such as rupture disks and explosively actuated valves.

When more than one distinguishing category characteristic is applicable, all requirements of each of the individual categories are applicable, although duplication or repetition of common testing requirements is not necessary.

3.9.6.1.3 **Valve Functions**

Valves in the IST Program are classified as either active or passive in accordance with the ASME OM Code ISTA-2000 as follows:

1. Active Valve – valves that are required to change obturator position to accomplish a specific function in shutting down a reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident.
2. Passive Valve – valves that maintain obturator position and are not required to change obturator position to accomplish the required function(s) in shutting down a reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident.

The IST Program identifies the safety-related functions for safety-related valves. The following are typical safety-related functions that are identified in the IST Program.

- Maintain closed (passive function)
- Maintain open (passive function)
- Transfer closed (active function)
- Transfer open (active function)

3.9.6.1.4 **Valve Testing**

Based on the valve category, active/passive function(s), and safety-related function(s) identified for each valve, the inservice tests to confirm the capability of the valve to perform these functions are identified in Table 3.9-8. ASME OM Code Table ISTC-3500-1, Inservice Test Requirements, specifies the required tests. Other specific testing requirements for power-operated valves include stroke-time testing and, as applicable, diagnostic testing to evaluate valve condition and to verify the valve will continue to function under design-basis conditions.

Table ISTC-3500-1 of the ASME OM Code requires the following four basic valve tests:

- exercise tests
- seat leakage tests
- remote position indicator tests
- special tests (i.e., fail-safe tests, explosive valve tests, rupture disc tests)

1. Valve Exercise Tests

   Active Category A valves, Category B valves, and Category C check valves are exercised periodically, except for self-actuated safety and relief valves. The ASME OM Code specifies a quarterly valve exercise frequency for all valves except power-operated safety and relief valves, which are required to be tested once per fuel cycle, and manual valves, as discussed in Subsection 3.9.6.1.5(2). Valves are tested by full-stroke exercising, during positions required to fulfill their functions. Where it is not practicable to exercise a valve during normal power operation, the valve exercise test is deferred to either cold shutdown or refueling outages. If full-stroke exercising is not practicable, part-stroke exercising is performed during operation at power or during cold shutdown. Valve exercise tests and

frequencies are identified in Table 3.9-8. In some cases, quarterly stroke testing is deferred to refueling outages or cold shutdown, as indicated in Table 3.9-8 Note g. The bases for deferral are consistent with NUREG-1482, Revision 1, considering the ESBWR is a new plant design. Where practical, the ESBWR is designed to accommodate quarterly stroke testing.

During extended shutdowns, valves that are required to be operable must remain capable of performing their intended safety function. Exercising valves during cold shutdown commences within 48 hours of achieving cold shutdown and continues until testing is complete or the plant is ready to return to operation at power. Valve testing required to be performed during a refueling outage is completed before returning the plant to operation at power.

During valve exercise tests, the necessary valve obturator movement is determined while observing an appropriate direct indicator, such as indicating lights that signal the required changes of obturator position, or by observing other evidence or positive means, such as changes in system pressure, flow, level, or temperature that reflects change of obturator position. Valve testing uses reference values determined from the results of PST or IST. These tests that establish reference values are performed under conditions as near as practicable to those expected during the IST. Stroke time is measured and compared to the reference value, except for valves classified as fast-acting (e.g., solenoid-operated valves (SOVs) with stroke time less than 2 seconds), for which a stroke time limit of 2 seconds is assigned.

Check valve exercise tests use direct observation or other positive means (ASME OM Code, ISTC 5221(a)) for verification of valve obturator position.

SOVs are tested to confirm the valves move to their energized positions and are maintained in those positions, and to confirm that the valves move to the appropriate failure mode positions when de-energized. Pre-conditioning of valves or their associated actuators or controls prior to IST undermines the purpose of IST and is prohibited. Pre-conditioning includes manipulation, pre-testing, maintenance, lubrication, cleaning, exercising, stroking, operating, or disturbing the valve to be tested in any way, except as may occur in an unscheduled, unplanned, and unanticipated manner during normal operation.

2. Valve Leakage Tests

Active and passive Category A containment isolation valves are tested to verify seat leakage is within limits in accordance with 10 CFR 50 Appendix J. Frequencies of containment isolation valve seat leakage tests are in accordance with the Appendix J requirements. All containment isolation valves and seat leakage tests are identified in Table 3.9-8.

Other Category A valves are required to be seat leakage tested at least once every two years as specified by the ASME OM Code ISTC-3630.

3. Remote Position Indicator Tests

Active and passive valves that are included in the IST Program and that are equipped with remote position indication require periodic verification of the remote position indication function in accordance with ASME OM Code ISTC-3700. Valves that require remote position indication testing are observed locally during valve exercising to verify proper operation of the position indication. The frequency for this position indication test is once every two years. Where local observation is not practicable, other methods are used for verification of valve position indicator operation.

Valves with remote position indicators are identified in Table 3.9-8.

4. Special Tests

- Valves with fail-safe actuators are tested by observing the operation of the actuator upon loss of valve actuating power (electrical power and/or pneumatic supply) in accordance with ASME OM Code ISTC-3560. These tests are performed in conjunction with the valve exercise test. Fail-safe test requirements are identified in Table 3.9-8.
- Category D explosively actuated valves are subject to periodic test firing of the explosive actuator charges. In accordance with ASME OM Code ISTC-5260, at least 20 percent of the charges installed in the plant in explosively actuated valves are fired and replaced at least once every two years. If a charge fails to fire, all charges within the same batch number are removed, discarded, and replaced with charges from a different batch. The firing of the explosive charge may be performed inside the valve or outside of the valve in a test fixture.

  The maintenance and review of the service life for charges for explosively actuated valves follows the requirements in the ASME OM Code ISTC-5260. Replacement charges are from batches from which a sample charge has been tested satisfactorily, and with a service life such that the requirements of ASME OM Code ISTC–5260(b) are met.

  Industry and regulatory guidance is considered in development of IST program for explosively actuated valves. In addition, the IST program for explosively actuated valves incorporates lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of explosively actuated valves to perform their safety functions.

  Category D explosively actuated valves are identified in Table 3.9-8.
- Category D rupture disks are replaced on a 5 year frequency unless historical data indicates a requirement for more frequent replacement, in accordance with Mandatory Appendix I of the ASME OM Code.

Category D rupture disks are identified in Table 3.9-8.

3.9.6.1.5 **Specific Valve Test Requirements**

1. Power-Operated Valve Tests

   Power-operated valves are tested in accordance with the ASME OM Code, Subsection ISTC. Specific testing activities for each valve are listed in Table 3.9-8. For active power-operated valves, stroke times will be measured during the exercise tests. Any abnormalities or erratic actions will be documented and evaluated. Test failures (e.g., failure to fully stroke or high stroke time measurements) are addressed per the ASME OM Code by repair, replacement or analysis.

   Subsection 3.9.6.8 describes additional (non-Code) testing of power-operated valves as discussed in Regulatory Issue Summary 2000-03.

2. Manual Valve Exercise Tests

   Active Category A and B manual valves are exercised once every two years in accordance with 10 CFR 50.55a(b)(3)(vi).

3. Check Valve Exercise Tests

   Category C check valves are exercised to both the open and closed positions regardless of safety function position in accordance with ASME OM Code ISTC-3522(a) using the methods of ISTC-5221. Check valves that have seat leakage requirements are leak tested in accordance with ASME OM Code ISTC-3600.

   Check valve testing requires verification that obturator movement is in the direction required for the valve to perform its safety function. During the exercise test, valve obturator position is verified by direct observation (position indicating lights) or by other positive means (i.e., changes in system pressure, temperature, flow rate, level, seat leakage or nonintrusive testing results).

   Check valves are exercised open with flow to either the full open position or to the position required to perform its intended open safety function. Check valve closure tests are performed by verifying that the obturator travels to the seat upon cessation of flow or reverse flow. Check valves with only an open safety function may be verified closed by other direct observations such as pressure, level, temperature, or seat leakage. This methodology meets the exercise requirements of ASME OM Code ISTC-5221.

   Acceptance criteria for this testing consider the specific system design and valve application. For example, a valve's safety function may require obturator movement in both open and closed directions. A mechanical exerciser may be used to operate a check valve for testing. Where a mechanical exerciser is used, acceptance criteria are provided for the force or torque required to move the check valve's obturator. Exercise tests also detect missing, sticking, or binding obturators.

If these test methods are impractical for certain check valves, or if sufficient flowcannot be achieved or verified, a sample disassembly examination program verifies valve obturator movement. The sample disassembly examination program groups check valves by category of similar design, application, and service condition.

During the disassembly process, the full-stroke motion of the obturator is verified. Nondestructive examination is performed on the hinge pin to assess wear, and seat contact surfaces are examined to verify adequate contact. Full-stroke motion of the obturator is re-verified immediately prior to completing reassembly. At least one valve from each group is disassembled and examined at each refueling outage, and all the valves in each group are disassembled and examined at least once every eight years. Before being returned to service, valves disassembled for examination or valves that received maintenance that could affect their performance are exercised with a full- or part-stroke. Details and bases of the sampling program are documented and recorded in the test plan.

When operating conditions, valve design, valve location, or other considerations prevent direct observation or measurements by use of conventional methods to determine adequate check valve function, diagnostic equipment and nonintrusive techniques are used to monitor internal conditions. Nonintrusive tests used are dependent on system and valve configuration, valve design and materials, and include methods such as ultrasonic (acoustic), magnetic, radiography, and use of accelerometers to measure system and valve operating parameters (e.g., fluid flow, disk position, disk movement, disk impact, and the presence or absence of cavitation and back-tapping). Nonintrusive techniques also detect valve degradation. Diagnostic equipment and techniques used for valve operability determinations are verified as effective and accurate under the PST program.

Testing is performed, to the extent practical, under normal operation, cold shutdown, or refueling conditions applicable to each check valve. Testing includes effects created by sudden starting and stopping of pumps, if applicable, or other conditions, such as flow reversal. When maintenance that could affect valve performance is performed on a valve in the IST program, post-maintenance testing is conducted prior to returning the valve to service.

Preoperational testing is performed during the initial test program (refer to Section 14.2) to verify that valves are installed in a configuration that allows correct operation, testing, and maintenance. Preoperational testing verifies that piping design features accommodate check valve testing requirements. Tests also verify disk movement to and from the seat and determine, without disassembly, that the valve disk positions correctly, fully opens or fully closes as expected, and remains stable in the open position under the full spectrum of system design-basis fluid flow conditions.

Data acquired during check valve testing and inspections, and the maintenance history of a valve or group of valves is collected and maintained in order to establish the basis for

specifying inservice testing, examination, and preventive maintenance activities that will identify and/or mitigate the failure of the check valves or groups of check valves tested. This data is also used to determine if certain check valve condition monitoring tests, such as nonintrusive tests, are feasible and effective in monitoring for these identified failure mechanisms, whether periodic disassembly and examination activities would be effective in monitoring for these failure mechanisms, as well as to determine possible valve groupings to implement in a future check valve condition monitoring program as allowed by ISTC-5222, the requirements of which are described in ASME OM Code, Appendix II.

Check valve exercise tests and frequencies are included in Table 3.9-8.

4. Vacuum Breaker Tests

   Vacuum breakers must meet the test requirements for both a Category C check valve (ASME OM Code ISTC-5220) and for a pressure relief device (Appendix I). Vacuum breaker tests and frequencies are included in Table 3.9-8.

5. Pressure Relief Valve Tests

   Pressure relief devices that protect systems or portions of system that are required to perform a function in shutting down the reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident, are subject to periodic inservice testing. The inservice tests for these valves are identified in ASME OM Code (mandatory) Appendix I.

   The periodic inservice testing includes visual inspection, seat tightness determination, set pressure determination, and operational determination of balancing devices, alarms, and position indication as appropriate. The frequency for this inservice test is every five years for ASME Class 1, and every 10 years for ASME Classes 2 and 3 devices. Pressure relief valves that require inservice testing are identified in Table 3.9-8.

3.9.6.2 **Inservice Testing of Pumps**

The ESBWR design does not require the use of pumps to mitigate the consequences of any DBA, or to achieve or maintain the safe shutdown condition. Therefore, there are no pumps required to be included in the IST Program. Table 3.9-8 does not list any pumps in the IST Program.

3.9.6.3 **Preservice Testing of Valves**

Category A, B, C (check valves), and D valves that are subject to periodic inservice testing are preservice tested in accordance with ASME OM Code Subsection ISTC-3100.

Category C pressure relief valves are preservice tested in accordance with ASME OM Code, Mandatory Appendix I.

3.9.6.4 **Deferred Testing Justifications**

In cases where it is not practicable to exercise category A, B or C (check) valves during normal power operations (quarterly), the valve is exercised during cold shutdown or refueling as permitted by ASME OM Code Subsections ISTC-3521 and ISTC-3522.

Valve exercise tests and associated frequencies are identified in Table 3.9-8. Justifications for deferred testing are detailed in Table 3.9-8.

3.9.6.5 **Valve Replacement, Repair and Maintenance**

Testing in accordance with ASME OM, ISTC-3310 and ISTC-5000 is performed after a valve is replaced, repaired, or has undergone maintenance that could affect the valve's performance. When a valve or its control system has been replaced, repaired, or has undergone maintenance that could affect valve performance, a new reference value is determined, or the previous value is reconfirmed by an inservice test. This test is performed before the valve is returned to service, or immediately if the valve is not removed from service. Deviations between the previous and new reference values are identified and analyzed. Verification that the new values represent acceptable operation is documented.

3.9.6.6 **10 CFR 50.55a Relief Requests and Code Cases**

Inservice testing of ASME B&PV Code Class 1, 2, and 3 pumps and valves is performed in accordance with the ASME OM Code except where specific relief has been granted by the NRC in accordance with 10 CFR 50.55a(f). Relief from the testing requirements of ASME OM Code is requested when compliance with requirements of the ASME OM Code is not practical. In such cases, specific information is provided which identifies the impractical code requirement, justification for the relief request, and the testing method to be used as an alternative. Demonstration of the impracticality of the testing required by the ASME OM Code, and justification for alternative testing proposed are provided. No relief from or alternative to the ASME OM Code is being requested.

The IST Program for valves does not invoke the use of any ASME Code Cases for inservice testing.

3.9.6.7 **Inservice Testing Program Implementation**

ASME OM Code inservice test intervals are as required by ISTA-3120; the initial 120-month test interval begins following the start of commercial service. The duration of each 120-month test interval may be modified by as much as one year as allowed by the ASME OM Code, provided these adjustments do not cause successive intervals to be altered by more than one year from the original pattern of intervals.

3.9.6.8 **Non-Code Testing of Power-Operated Valves**

Although the design basis capability of active, safety-related power-operated valves is verified as part of the design and qualification process, power-operated valves that perform an active safety

function are tested again after installation in the plant, as required, to ensure valve setup is acceptable to perform their required functions, consistent with valve qualification. These tests, which are typically performed under static (no flow or pressure) conditions, also document the "baseline" performance of the valves to support maintenance and trending programs. During the testing, critical parameters needed to ensure proper valve setup are measured. Depending on the valve and actuator type, these parameters may include seat load, running torque or thrust, valve travel, actuator spring rate, bench set and regulator supply pressure. Uncertainties associated with performance of these tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately.Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.

Additional testing is performed as part of the air-operated valve (AOV) program, which includes the key elements for an AOV Program as identified in the JOG AOV program document, Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000 Reference 3.9.1-201 and Reference 3.9.1-202. The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in RIS 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-related Power- Operated Valves Under Design Basis Conditions, Reference 3.9.1-203 by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air- and other power- operated valves included in the IST program. For example, key lessons learned addressed in the AOV program include:

- Valves are categorized according to their safety significance and risk ranking.
- Setpoints for AOVs are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).
- Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, to identify potential degradation, unless those valves are periodically cycled during normal plant operation under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If required based on valve qualification or operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform is required functions.
- Sufficient diagnostics are used to collect relevant data (e.g., valve stem thrust and torque, fluid pressure and temperature, stroke time, operating and/or control air pressure, etc.) to verify the valve meets the functional requirements of the qualification specification.
- Test frequency is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with Reference 3.9.1-201 and

Reference 3.9.1-202, with a minimum of 5 years (or 3 refueling cycles) of data collected and evaluated before extending test intervals.

- Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when maintenance on the valve, valve repair or replacement, have the potential to affect valve functional performance.
- Guidance is included to address lessons learned from other valve programs in procedures and training specific to the AOV program.
- Documentation from AOV testing, including maintenance records and records from the corrective action program are retained and periodically evaluated as a part of the AOV program.

The attributes of the AOV testing program described above, to the extent that they apply to and can be implemented on other safety-related power-operated valves, such as electro-hydraulic valves, are applied to those other power-operated valves.

3.9.7 **Risk-Informed Inservice Testing**

Risk informed inservice testing is not being utilized.

3.9.8 **Risk-Informed Inservice Inspection of Piping**

Risk informed inservice inspection is not being utilized.

3.9.9 **COL Information**

3.9.9-1-A **Reactor Internals Vibration Analysis, Measurement and Inspection Program**

This COL item is addressed in Subsection 3.9.2.4.

3.9.9-2-A ASME Class 2 or 3 or Quality Group D Components with 60 Year Design Life

This COL item is addressed in Subsection 3.9.3.1.

3.9.9.3-A **Inservice Testing Programs**

This COL item is addressed in Subsection 3.9.6.

3.9.9.4-A **Snubber Inspection and Test Program**

This COL item is addressed in Subsection 3.9.3.7.1, 3.e and Subsection 3.9.3.7.1, 3.f.

3.9.10 **References**

3.9-1 General Electric Company, “BWR Fuel Channel Mechanical Design and Deflection,” NEDE-21354-P, September 1976 (GE proprietary) and NEDO-21354, September 1976 (Non-proprietary).

3.9-2 General Electric Company, “BWR Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (Amendment No. 3),”

NEDE-21175-3-P-A, October 1984 (GE proprietary) and NEDO-21175-3-A, October 1984 (Non-proprietary).

3.9-3 General Electric Company, “General Electric Environmental Qualification Program,” NEDE-24326-1-P, Proprietary Document, January 1983.

3.9-4 M.A. Miner, “Cumulative Damage in Fatigue,” Journal of Applied Mechanics, Vol. 12, ASME, Vol. 67, pages A159-A164, September 1945.

3.9-5 American Society of Mechanical Engineers Code for Operation and Maintenance of Nuclear Power Plants, 2001 Edition with 2003 Addenda.

3.9-6 (Deleted)

3.9-7 *GE Hitachi Nuclear Energy, “ESBWR Steam Dryer Structural Evaluation,” NEDE-33313P, Revision 5, Class III (Proprietary), December 2013, and NEDO-33313, Revision 5, Class I (Non-Proprietary), December 2013.*

3.9-8 American Society of Mechanical Engineers OM-S/G-1990, Standards and Guides for Operation and Maintenance of Nuclear Power Plants.

3.9.1-201 Joint Owners Group Air Operated Valve Program Document, Revision1, December 13, 2000.Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000.

3.9.1-202 USNRC, Eugene V. Imbro, letter to Mr. David J. Modeen, Nuclear Energy Institute, Comments On Joint Owners' Group Air Operated Valve Program Document, October 8, 1999.

3.9.1-203 Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-related Power-Operated Valves Under Design Basis Conditions, March 15, 2000.

Table 3.9-1 Plant Events

		ASME B&PV Code Service Limit(8)	No. of Cycles
A. Plant Operating Events(1), (9)			
1.	Boltup(1)	A	45
2.	a. Hydrostatic Test (two test cycles for each boltup cycle)	Testing	90
	b. Hydrostatic Test (shop and field)	Testing	3
3.	Startup (55.6°C/hr [100°F/hr] Heatup Rate)(2)	A	180
4.	Turbine Roll and Increase to Rated Power	A	180
5.	Daily and Weekly Reduction to 50% Power(1)	A	20,200
6.	Control Rod Pattern Change(1)	A	300
7.	Loss of Feedwater Heaters	B	60
8.	Scram:		
	a. Turbine Generator Trip, Feedwater On, and Other Scrams	B	60
	b. Loss of Feedwater Flow, MSIV Closure	B	60
9.	Reduction to 0% Power, Hot Standby, Shutdown (55.6°C/hr [100°F/hr] Cooldown Rate)(2)	A	172
10.	Refueling Shutdown and Unbolt(1)	A	45
11.	Scram:		
	a. Reactor Overpressure with Delayed Scram (ATWS)	C	1(3)
	b. Automatic Blowdown	C	1(3)
12.	Improper Plant Startup	C	1(3)
B. Dynamic Loading Events(1), (6), (9)			
13.	Safe Shutdown Earthquake (SSE) at Rated Power Operating Conditions	B(4)	20(5)
14.	Safe Shutdown Earthquake (SSE) at Rated Power Operating Conditions	D(7)	1(3)
15.	a. Safety Relief Valve (SRV) actuation (one) with depressurization (scram)	B	8
	b. Depressurization Valve (DPV) actuation (one) with depressurization (scram)	C	1(3)
16.	Loss-of-Coolant-Accident (LOCA):		
	Worst of small break LOCA (SBL), intermediate break LOCA (IBL), or large break LOCA (LBL)	D(7)	1(3)

Notes:

1. Some events apply to RPV only. The number of events/cycles applies to RPV as an example.

2. Bulk average vessel coolant temperature change 55.6°C (100°F) in any one-hour period.
3. The annual encounter probability of a single event is < 10^{-2} for a Level C event and < 10^{-4} for a Level D event. Refer to Subsection 3.9.3.1.
4. The effects of displacement-limited, seismic anchor motions due to SSE are evaluated for safety-related ASME B&PV Code Class 1, 2, and 3 components and component supports. See Table 3.9-2 for stress limits to be used to evaluate the seismic anchor motion effects.
5. Use 20 peak SSE cycles for evaluation of ASME Class 1 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.
6. Table 3.9-2 shows the evaluation basis combination of these dynamic loadings.
7. ASME B&PV Code Appendix F or other appropriate requirements of the ASME B&PV Code are used to determine the Service Level D limits, as described in Subsection 3.9.1.4.
8. These ASME B&PV Code Service Limits apply to ASME B&PV 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.
9. Plant events listed are those expected to occur over the 60-year life of the plant.

Table 3.9-2 Load Combinations and Acceptance Criteria for Safety-Related, ASME B&PV 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*	*N*	*A*
2.	*Plant/System Operating Transients (SOT)*	*(a) N + TSV* *(b) N + SRV*(5)	*B* *B*
3.	*Normal Operation + SSE*	*N + SSE*	*B*(11), (12)
4.	*Infrequent Operating Transient, ATWS, DPV*	*(a) N*(6) *+ SRV*(5) *(b) N + DPV*(7)	*C*(13) *C*(13)
5.	*SBL*	*N + SRV*(8) *+ SBL*	*C*(13)
6.	*SBL or IBL + SSE*	*N + SBL (or IBL) + SSE + SRV*(8)	*D*(13)
7.	*LBL + SSE*	*N + LBL + SSE*	*D*(13)
8.	*NLF*	*N + SRV*(5) *+ TSV*(10)	*D*(13)

Notes:

(1) *See Legend on the following pages for definition of terms. Refer to* Table 3.9-1 *for plant events and cycles information.*

The service loading combination also applies to Seismic Category I instrumentation and electrical equipment (refer to Section 3.10).

(2) *For vessels, 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 MS and branch piping evaluation, additional loads associated with relief line clearing and blowdown into the suppression pool are included.*

(6) *The RCPB is evaluated using in the load combination the maximum pressure expected to occur during ATWS.*

(7) *This applies only to the MS and Isolation Condenser systems. The loads from this event are combined with loads associated with the pressure and temperature concurrent with the event.*

(8) *The most limiting load combination case among SRV(1), SRV(2) and SRV (ADS). See Note (5) for MS and branch piping.*

(9) *(Deleted)*

(10) *This applies only to the main steamlines and components mounted on it. The low probability that the TSV closure and SRV loads can exist at the same time results in this combination being considered under service level D.*

(11) *Applies only to fatigue evaluation of ASME B&PV Code Class 1 components and core support structures. See Dynamic Loading Event No. 13,* Table 3.9-1*, and Note 5 of* Table 3.9-1 *for number of cycles.*

(12) *For ASME B&PV Code Class 1, 2 and 3 piping the following changes and additions to ASME B&PV Code Section III NB-3600, NC-3600 and ND-3600 are necessary and are evaluated to meet the following stress limits:*

a. *ASME B&PV Code Class 1 Piping*

$$S_{SAM} = C_2 \frac{D_0}{2I} M_c \leq 6.0 S_m \qquad \text{Eq. (12a)}$$

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 B&PV Code NB-3600.*

S_m *is the tabulated value of allowable stress at temperature per the ASME B&PV Code or its equivalent*

SSE inertia and seismic anchor motion loads are included in the calculation of ASME B&PV Code NB-3600 Equations (10) and (11).

b. *For ASME B&PV Code Class 2 and 3 piping:*

$$S_{SAM} = i \frac{M_c}{Z} \leq 3.0 S_h \; (\leq 2.0 S_y) \qquad \text{Eq. (12b)}$$

Where: S_{SAM} *and* M_c *are defined in (a) above.*

i and Z are defined in ASME B&PV Code Subsections NC/ND-3600

SSE inertia and seismic anchor motion loads are not included in the calculation of ASME B&PV Code Subsections NC/ND-3600 Equation (9), Service Levels A and B and Equations (10) and (11).

(13) ASME B&PV 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. Piping system dynamic moments can be calculated using an elastic response spectrum or time history analysis.

Load Definition Legend for Table 3.9-2	
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.*
TSV	*Turbine stop valve closure induced loads in the MS piping and components integral to or mounted thereon.*
NLF	*Non-LOCA Fault.*
SSE	*RBV loads induced by safe shutdown earthquake.*
SRV(1), SRV(2)	*RBV loads induced by safety relief valve (SRV) discharge of one or two adjacent valves, respectively.*
SRV (ALL)	*RBV loads induced by actuation of all safety relief valves, which activate within milliseconds of each other (e.g., turbine trip operational transient).*
SRV (ADS)	*RBV loads induced by the actuation of safety relief valves in Automatic Depressurization System operation, which actuate within milliseconds of each other during the postulated small or intermediate break LOCA, or SSE.*
DPV	*Depressurization Valve opening induced loads in the Isolation Condenser system piping and pipe-mounted equipment.*
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 LOCA1 through LOCA7. LOCA events are grouped in three categories, SBL, IBL or LBL, as defined here.*
LOCA1	*Pool swell drag/fallback loads on safety-related piping and components located between the main vent discharge outlet and the suppression pool water upper surface.*
LOCA2	*Pool swell impact loads acting on safety-related piping and components located above the suppression pool water upper surface.*
LOCA3	*(a)Oscillating pressure induced loads on submerged safety-related piping and components during main vent clearing (VLC), condensation oscillations (COND), or chugging (CHUG), or* *(b)Jet impingement (JI) load on safety-related piping and components as a result of a postulated IBL or LBL event. Piping and components are defined safety-related, if they are required for shutdown of the reactor or to mitigate consequences of the postulated pipe failure without off-site power (refer to introduction to Section 3.6).*
LOCA4	*RBV load from main vent clearing (VLC).*
LOCA5	*RBV loads from condensation oscillations (COND).*
LOCA6	*RBV loads from chugging (CHUG).*
LOCA7	*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 3.9.2.4) 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.*

Load Definition Legend for Table 3.9-2	
SBL	*Loads induced by small break LOCA (Subsection 3.9.3.1); the loads are: LOCA3(a), LOCA4 and LOCA6.*
IBL	*Loads induced by intermediate break LOCA (Subsection 3.9.3.1); the loads are: LOCA3(a) or LOCA3(b), LOCA4, LOCA5 and LOCA6.*
LBL	*Loads induced by large break LOCA (Subsection 3.9.3.1); the loads are: LOCA1 through LOCA7.*

Table 3.9-3 Pressure Differentials Across Reactor Vessel Internals

Reactor Component[2]	Maximum Pressure Differences[1,3] kPaD, (psi)
1 Core plate and guide tube	76.7 (11.1)
2 Support legs and support ring (beneath the core plate)	51.3 (7.44)
3 Chimney head (at marked elevation)	84.9 (12.3)
4 Upper shroud (just below top guide)	122.6 (17.8)
5 Core averaged power fuel bundle (bulge at bottom of bundle)	44.8 (6.50)
6 Core averaged power fuel bundle (collapse at bottom of top guide)	66.6 (9.66)
7 Maximum power fuel bundle (bulge at bottom of bundle)	71.1 (10.3)
8 Top guide	74.6 (10.8)
9 Steam Dryer	15.8 (2.3)
• Chimney head to water level, for points (a) to (b), irreversible pressure drop	71.2 (10.3)
• Chimney head to water level, from points (a) to (b), elevation pressure drop	50.0 (7.25)

Notes:

1. At 100% rated core power, 100% rated steam flow, and 100% rated core flow with two sigma statistical calculations.
2. Item numbers in this column correspond to the location (node) numbers identified in Figure 3.9-5.
3. In application there is an additional 10% margin added to these values.

Table 3.9-4 Deformation Limit for Safety Class Reactor Internal Structures Only

Permissible deformation, DP	General Limit
a. Analyzed deformation causing loss of function, DL	≤ 0.90 SF_{min}

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 function.[1]

SF_{min} = Minimum safety factor (refer to Subsection 3.9.5.4).

Notes:

1. “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 may 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 CRD alignment and clearances for proper insertion, 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.	Elastic evaluated primary stresses, PE / Permissible primary stresses, PN	≤	2.25 SF_{min}
b.	Permissible load, LP / Largest lower bound limit load, CL	≤	1.5 SF_{min}
c.	Elastic evaluated primary stress, PE / Conventional ultimate strength at temperature, US	≤	0.75 SF_{min}
d.	Elastic-plastic evaluated nominal primary stress, EP / Conventional ultimate strength at temperature, US	≤	0.9 SF_{min}
e.(1)	Permissible load, LP / Plastic instability load, PL	≤	0.9 SF_{min}
f. (1)	Permissible load, LP / Ultimate load from fracture analysis, UF	≤	0.9 SF_{min}
g. (1)	Permissible load, LP / Ultimate load or loss of function load from test, LE	≤	1.0 SF_{min}

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 supports the external loading, is added to the membrane stresses at the section of interest.

PN = Permissible primary stress levels under service level A or B (normal or upset) conditions under ASME B&PV Code, Section III.

LP = Permissible load under stated conditions of service level A, B, C or D (normal, upset, emergency or faulted).

CL = Lower bound limit load with yield point equal to 1.5 Sm where Sm is the tabulated value of allowable stress at temperature per the ASME B&PV Code or its equivalent. The “lower bound limit load” is here 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.

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 is 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 is 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 3.9.5.4).

Notes:

1. Equations e, f, or g will not be used unless supporting data are provided to the NRC for review and approval.

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.	$\dfrac{\text{Permissible load, LP}}{\text{Service level A (normal) permissible load, PN}}$	$\leq \dfrac{2.25}{SF_{min}}$
b.	$\dfrac{\text{Permissible load, LP}}{\text{Stability analysis load, SL}}$	$\leq \dfrac{0.9}{SF_{min}}$
c.[1]	$\dfrac{\text{Permissible load, LP}}{\text{Ultimate buckling collapse load from test, SET}}$	$\leq \dfrac{1.0}{SF_{min}}$

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 are 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 is 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 is 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 (refer to Subsection 3.9.5.4)

Notes:

1. Equation c is not used unless supporting data are provided to the NRC.

Table 3.9-7 Fatigue Limit for Safety Class Reactor Internal Structures Only

Cumulative Damage From Fatigue[1]	Limit for Service Levels A&B (Normal and Upset Conditions)
Design fatigue cycle usage from analysis using the method of the ASME B&PV Code	≤ 1.0

Note:

1. Reference 3.9-4.

Table 3.9-8 Inservice Testing (Sheet 1 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
B21 Nuclear Boiler System Valves													
F710	1	Excess flow check valve – RPV shutdown range water level instrument reference leg line (7a)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L SO SC	App J RO RO
F701	4	Excess flow check valve – RPV water level instrument reference leg line (7a)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L SO SC	App J RO RO
F703	4	Excess flow check valve – RPV narrow range water level instrument sensing line (7a)	CK	SA	2	A, C	A	O	O/C	N/A	Y	SO SC L	RO RO App J
F705	4	Excess flow check valve – RPV wide range water level instrument sensing line (7a)	CK	SA	2	A, C	A	O	O/C	N/A	Y	SO SC L	RO RO App J
F707	4	Excess flow check valve – RPV fuel zone range water level instrument sensing line (7a)	CK	SA	2	A, C	A	O	O/C	N/A	Y	SC SO L	RO RO AppJ
F099	2	Feedwater (FW) supply line second outboard check valve (7i)	CK	SA	2	A, C	A	O	C	N/A	---	SO SC L	RO RO 2 yrs
F100	2	FW supply line second containment isolation valve (7t)	GT	PM	1	A	A	O	C	C	Y	SC FC L P	RO RO App J 2 yrs
F101	2	FW supply line outboard containment isolation valve (7t)	GT	PM	1	A	A	O	C	C	Y	L SC FC P	App J RO RO 2 yrs

Table 3.9-8 Inservice Testing (Sheet 2 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F102	2	FW supply line inboard containment isolation valve (7i)	CK	SA	1	A, C	A	O	O/C	N/A	Y	SO SC P L	RO RO 2 yrs App J
F111	2	RWCU/SDC to Feedwater outboard containment isolation valve (tu)	CK	SA with AO	1	A, C	A	O	O/C	O	Y	SO SC FO L P	RO RO RO App J 2 yrs
F001	4	Inboard MSIV (7j)	GT	PM	1	A	A	O	C	C	Y	L P SC FC	App J 2 yrs CS CS
F002	4	Outboard MSIV (7j)	GT	PM	1	A	A	O	C	C	Y	L P SC FC	App J 2 yrs CS CS
F006	10	Safety relief valve (SRV) (7a)	RV	SA with NO	1	A, C	A	C	O/C	N/A	--	R	5 yrs
F003	8	Safety Valve (7a)	RV	SA	1	A, C	A	C	O/C	N/A	--	R	5 yrs
F004	8	DPV	SQ	EX	1	D	A	C	O	as-is	--	X P	E2 2 yrs
F010	1	Inboard MSIV upstream drain line inboard containment isolation valve	QBL	NO	1	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo

Table 3.9-8 Inservice Testing (Sheet 3 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F011	1	Inboard MSIV upstream drain line outboard containment isolation valve	QBL	NO	1	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3mo
F016	4	Outboard MSIV upstream drain line outboard containment isolation valve	QBL	AO	1	A	A	O	O/C	C	Y	L P SC SO FC	App J 2 yrs 3 mo 3 mo 3 mo
F715	4	Excess flow check valve – MSL flow restrictor instrument line (7g)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L SO SC	App J RO RO
F713	4	Excess flow check valve – MSL flow restrictor instrument line (7f)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L SO SC	App J RO RO
F026	1	RPV top head vent inboard shutoff valve (7b)	QBL	NO	1	B	A	C	C	C	--	P SC FC	2 yrs CS CS
F027	1	RPV top head vent outboard shutoff valve (7b)	QBL	NO	1	B	A	C	C	C	--	P SC FC	2 yrs CS CS
F007	10	SRV discharge line inboard vacuum breaker (7k)	VB	SA	3	C	A	C	O/C	N/A	--	R SC SO	2 yrs RO RO
F008	10	SRV discharge line outboard vacuum breaker (7k)	VB	SA	3	C	A	C	O/C	N/A	--	R SC SO	2 yrs RO RO
F035	10	SRV pneumatic supply line check valve (7l)	CK	SA	3	C	A	C	C	N/A	--	SC SO	RO RO

Table 3.9-8 Inservice Testing (Sheet 4 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F031	4	Inboard MSIV air supply line check valve (7m)	CK	SA	3	C	A	C	C	N/A	--	SO SC	RO RO
F032	4	Outboard MSIV air supply line check valve (7m)	CK	SA	3	C	A	C	C	N/A	--	SC SO	RO RO
F028	1	RPV head vent discharge line vacuum breaker (7k)	VB	SA	3	C	A	C	O/C	N/A	--	R SC SO	2 yrs RO RO
B32 Isolation Condenser System Valves													
F001	4	Steam supply line isolation valve	QBL	EH	1	A	A	O	O/C	as-is	Y	L P SC SO	App J 2 yrs 3 mo 3 mo
F002	4	Steam supply line isolation valve	GT	NO	1	A	A	O	O/C	as-is	Y	L P SC SO	App J 2 yrs 3 mo 3 mo
F003	4	Condensate return line isolation valve	QBL	NO	1	A	A	O	O/C	as-is	Y	L P SC SO	App J 2 yrs 3 mo 3 mo
F004	4	Condensate return line isolation valve	GT	EH	1	A	A	O	O/C	as-is	Y	L P SC SO	App J 2 yrs 3 mo 3 mo
F005	4	Condensate return valve	QBL	EH	1	B	A	C	O	as-is	--	P SO	2 yrs 3 mo
F006	4	Condensate return bypass valve	QBF	NO	1	B	A	C	O	O	--	P SO FO	2 yrs 3 mo 3 mo

Table 3.9-8 Inservice Testing (Sheet 5 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F007	4	Condenser upper header vent valve	GB	SO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F008	4	Condenser upper header vent valve	GB	SO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F009	4	Condenser lower header vent valve	GB	SO	2	A	A	C	O/C	O	N	L P SC FC	App J 2 yrs 3 mo 3 mo
F010	4	Condenser lower header vent valve	GB	SO	2	A	A	C	O/C	O	N	L P SC FC	App J 2 yrs 3 mo 3 mo
F011	4	Bypass lower header vent valve	RV	SA	2	A	A	C	O/C	N/A	Y	R L	10 yrs App J
F012	4	Bypass lower header vent valve	GB	SO	2	A	A	C	O/C	O	Y	L P SC SO FO	App J 2 yrs 3 mo 3 mo 3 mo
F013	4	Condenser purge line isolation valve	GB	SO	1	A	A	O	O/C	C	Y	SO SC FC P L	3 mo 3 mo 3 mo 2 yrs App J

Table 3.9-8 Inservice Testing (Sheet 6 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F014	4	Condenser purge line isolation valve (g4)	CK	SA	1	A, C	A	O	O	N/A	Y	L P SO SC	App J 2 yrs RO RO
F104	2	Pool cross-connect valve	SQ	EX	3	D	A	C	O	as-is	--	X	E2
F105	2	Pool cross-connect valve	QBF	AO	3	B	A	C	O	as-is	--	SO	3 mo
F017	4	High Pressure Nitrogen check valve (g5)	CK	SA	2	C	A	C	C	N/A	--	SO SC	RO RO
F018	4	High Pressure Nitrogen check valve (7e)	CK	SA	2	C	A	C	C	N/A	--	SO SC	RO RO
F701	4	Excess flow check valve – steam supply line differential pressure instrument sensing line (7e)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L P SC SO	App J 2 yrs RO RO
F703	4	Excess flow check valve – steam supply line differential pressure instrument sensing line (7f)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L P SC SO	App J 2 yrs RO RO
F705	4	Excess flow check valve – steam supply line differential pressure instrument sensing line (7f)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L P SC SO	App J 2 yrs RO RO
F707	4	Excess flow check valve – steam supply line differential pressure instrument sensing line (7f)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L P SC SO	App J 2 yrs RO RO
F709	4	Excess flow check valve – condensate return line differential pressure instrument sensing line (7f)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L P SC SO	App J 2 yrs RO RO

Table 3.9-8 Inservice Testing (Sheet 7 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F711	4	Excess flow check valve – condensate return line differential pressure instrument sensing line (7f)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L P SC SO	App J 2 yrs RO RO
F713	4	Excess flow check valve – condensate return line differential pressure instrument sensing line (7f)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L P SC SO	App J 2 yrs RO RO
F715	4	Excess flow check valve – condensate return line differential pressure instrument sensing line (7f)	CK	SA	2	A, C	A	O	O/C	N/A	Y	L P SC SO	App J 2 yrs RO RO
C12 Control Rod Drive System Valves													
F022	1	High pressure makeup line check valve (7g)	CK	SA	2	C	A	O	O/C	N/A	--	SO SC	RO RO
D005	269	Ball check valve – CRD drive insert line (7g)	CK	SA	3	C	A	O	O/C	N/A	--	SO SC	RO RO
F071	1	High pressure makeup line isolation valve	QBL	AO	2	B	A	O	C	Closed	--	P FC SC	2 yrs 3 mo 3 mo
F072	1	High pressure makeup line isolation valve	QBL	AO	2	B	A	O	C	Closed	--	P FC SC	2 yrs 3 mo 3 mo
C41 Standby Liquid Control (SLC) System Valves													
F002	4	SLC injection line shutoff valve	QBL	AO	2	A	A	O	O/C	as-is	--	SO SC P L	3 mo 3 mo 2 yrs 2 yrs

Table 3.9-8 Inservice Testing (Sheet 8 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F003	4	SLC injection line squib valve	SQ	EX	1	A, D	A	C	O	as-is	Y	X L	E2 App J
F004	2	SLC injection line outboard check valve (7n)	CK	SA	1	A, C	A	C	O/C	N/A	Y	L SC SO	App J RO RO
F005	2	SLC injection line inboard check valve (7n)	CK	SA	1	A, C	A	C	O/C	N/A	Y	L SC SO	App J RO RO
F030	2	SLC accumulator tank relief valve	RV	SA	2	C	A	C	O/C	N/A	--	R	10 yrs
F507	2	SLC accumulator tank inboard vent valve	GB	AO	2	B	A	C	O/C	C	--	P SC SO FC	2 yrs 3 mo 3 mo 3 mo
F508	2	SLC accumulator tank outboard vent valve	GB	AO	2	B	A	C	O/C	C	--	P SC SO FC	2 yrs 3 mo 3 mo 3 mo
D11 Process Radiation Monitoring System Valves													
F001	1	Drywell Fission Product Monitoring Line Inboard isolation Valve	GB	SO	2	A	A	O	O	as-is	Y	SO P L	3 mo 2 yrs App J
F002	1	Drywell Fission Product Monitoring Line Outboard isolation Valve	GB	SO	2	A	A	O	O	as-is	Y	SO P L	3 mo 2 yrs App J
F003	1	Drywell Fission Product Monitoring Line Inboard isolation Valve	GB	SO	2	A	A	O	O	as-is	Y	SO P L	3 mo 2 yrs App J

Table 3.9-8 Inservice Testing (Sheet 9 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F004	1	Drywell Fission Product Monitoring Line Outboard isolation Valve	GB	SO	2	A	A	O	O	as-is	Y	SO P L	3 mo 2 yrs App J
T62 Containment Monitoring System Valves													
F001	2	Drywell to Sample Rack Inboard	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F002	2	Drywell to Sample Rack Outboard	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F006	2	Gas Sample Return to Wetwell Inboard	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F005	2	Gas Sample Return to Wetwell Outboard	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F003	2	Wetwell to Sample Rack Inboard	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F004	2	Wetwell to Sample Rack Outboard	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F701	4	Suppression Pool Level Narrow Range	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F703	4	Suppression Pool Level Narrow Range	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F705	2	Suppression Pool Level Wide Range	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F707	2	Suppression Pool Level Wide Range	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F709	2	Suppression Pool Level Wide Range	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F711	2	Suppression Pool Level Wide Range	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F713	2	Lower Drywell Level Post Accident Monitoring (PAM)	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J

Table 3.9-8 Inservice Testing (Sheet 11 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F715	2	Lower Drywell Level PAM	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F717	4	Lower Drywell Level Isolation	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F719	4	Lower Drywell Level Isolation	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F721	2	Drywell Upper Level Isolation	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F723	2	Drywell Upper Level Isolation	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F725	2	Drywell/Wetwell Delta Pressure	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F727	2	Drywell/Wetwell Delta Pressure	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F729	2	Drywell Pressure PAM	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F731	4	Drywell Pressure Narrow Range	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F733	4	Drywell Pressure Wide Range	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
F735	2	Wetwell Pressure	QBL	AO	2	A	A	O	O	O	Y	SO FO P L	3 mo 3 mo 2 yrs App J
E50 Gravity-Driven Cooling System Valves													
F001	8	GDCS injection line manual shutoff valve	QBL	M	1	B	P	O	O	N/A	--	P	2 yrs
F002	8	GDCS injection squib actuated valve	SQ	EX	1	D	A	C	O	as-is	--	X P	E2 2 yrs
F003	8	GDCS check valve (7h)	CK	SA	1	A, C	A	O	O/C	N/A	--	L SC SO P	RO RO RO 2 yrs
F004	4	GDCS manual shutoff valve	QBL	M	2	B	P	O	O	N/A	--	P	2 yrs
F005	4	GDCS equalization line manual shutoff valve	QBL	M	1	B	P	O	O	N/A	--	P	2 yrs

Table 3.9-8 Inservice Testing (Sheet 13 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F006	4	GDCS equalization squib actuated valve	SQ	EX	1	D	A	C	O	as-is	--	X P	E2 2 yrs
F007	4	GDCS check valve (7i)	CK	SA	1	A, C	A	O	O/C	N/A	--	L SO SC P	RO RO RO 2 yrs
F008	4	GDCS manual shutoff valve	QBL	M	2	B	P	O	O	N/A	--	P	2 yrs
F009	12	GDCS deluge squib valve	SQ	EX	2	D	P	C	C	as-is	--	X P	E2 2 yrs
F010	4	GDCS deluge line isolation valve	QBL	NO	2	B	P	O	O	as-is	--	P	2 yrs
G21 Fuel and Auxiliary Pools Cooling System (FAPCS) Valves													
F210	1	Emergency makeup spent fuel pool water line check valve	CK	SA	3	C	A	O	O/C	N/A	--	SO SC	3 mo 3 mo
F211	1	Emergency makeup spent fuel pool water line shutoff valve	QBL	M	3	B	A	C	O	N/A	--	SO	2 yrs
F212	1	Reactor well drain line containment isolation valve	QBL	M	2	A	P	C	C	N/A	Y	P L	2 yrs App J
F213	1	Reactor well drain line second containment isolation valve	QBL	M	2	A	P	C	C	N/A	Y	P L	2 yrs App J
F303	1	GDCS pool return line outboard isolation valve	QBL	AO	2	A	A	C	C	C	Y	SC FC L P	3 mo 3 mo App J 2 yrs
F304	1	GDCS pool return line inboard isolation check valve	CK	SA	2	A, C	A	C	C	N/A	Y	SO SC L	3 mo 3 mo App J

Table 3.9-8 Inservice Testing (Sheet 14 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F306	2	Suppression pool return line outboard isolation valve	QBL	AO	2	A	A	C	C	as-is	Y	SC L P	3 mo App J 2 yrs
F307	2	Suppression pool return line inboard isolation check valve	CK	SA	2	A, C	A	C	C	N/A	Y	SO SC L	3 mo 3 mo App J
F309	1	Drywell spray line outboard isolation valve	QBL	AO	2	A	A	C	C	C	Y	SC FC L P	3 mo 3 mo App J 2 yrs
F310	1	Drywell spray line inboard isolation check valve (7o)	CK	SA	2	A, C	A	C	C	N/A	Y	SO SC L	RO RO App J
F323	1	GDCS pool suction line inboard isolation valve	QBL	NO	2	A	A	C	C	C	Y	SC FC L P	3 mo 3 mo App J 2 yrs
F324	1	GDCS pool suction line outboard isolation valve	QBL	AO	2	A	A	C	C	C	Y	SC FC L P	3 mo 3 mo App J 2 yrs
F321	2	Suppression pool suction line outboard isolation valve	QBL	AO	2	A	A	C	C	as-is	Y	SC L P	3 mo App J 2 yrs
F322	2	Suppression pool suction line second isolation valve	QBL	AO	2	A	A	C	C	as-is	Y	SC L P	3 mo App J 2 yrs

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F333	2	Low Pressure Coolant Injection (LPCI) testable check valve(7p)	CK	AO	2	A, C	A	C	C	C	--	SC SO FC L P	RO RO RO 2 yrs 2 yrs
F420	1	Emergency makeup Isolation Condenser/Passive Containment Cooling System (IC/PCCS) pool water line shutoff valve	QBL	M	3	B	A	C	O	N/A	--	SO	2 yrs
F421	1	Emergency makeup IC/PCCS pool water line check valve	CK	SA	3	C	A	C	O/C	N/A	--	SO SC	3 mo 3 mo
F426	2	Fire Protection System (FPS) water makeup valve to IC/PCCS pool	QBL	M	3	B	A	C	O	N/A	--	SO	2 yrs
F427	2	FPS water makeup check valve to IC/PCCS pool	CK	SA	3	C	A	C	O/C	N/A	--	SO SC	3 mo 3 mo
F428	2	FPS water makeup valve to Spent Fuel Pool	QBL	M	3	B	A	C	O	N/A	--	SO	2 yrs
F429	2	FPS water makeup check valve to Spent Fuel Pool	CK	SA	3	C	A	C	O/C	N/A	--	SO SC	3 mo 3 mo
G31 Reactor Water Cleanup/Shutdown Cooling System Valves													
F002	2	RWCU/SDC mid-vessel suction line inboard isolation valve	QBL	NO	1	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F003	2	RWCU/SDC mid-vessel suction line outboard isolation valve	QBL	AO	1	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo

Table 3.9-8 Inservice Testing (Sheet 16 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F007	2	RWCU/SDC bottom head suction line inboard isolation valve	QBL	NO	1	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F008	2	RWCU/SDC bottom head suction line outboard isolation valve	QBL	AO	1	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F038	2	RWCU/SDC bottom head suction line sample line inboard isolation valve	GB	SO	1	A	A	C	O/C	C	Y	L P SO SC FC	App J 2 yrs 3 mo 3 mo 3 mo
F039	2	RWCU/SDC bottom head suction line sample line outboard isolation valve	GB	SO	1	A	A	C	O/C	C	Y	L P SO SC FC	App J 2 yrs 3 mo 3 mo 3 mo
U50 Equipment and Floor Drain System Valves													
F001	1	Drywell equipment drain (low conductivity waste [LCW]) sump discharge line inboard isolation valve	QBL	NO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F002	1	Drywell equipment drain (LCW) sump discharge line outboard isolation valve	QBL	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F003	1	Drywell floor drain (high conductivity waste [HCW]) sump discharge line inboard isolation valve	QBL	NO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo

Table 3.9-8 Inservice Testing (Sheet 17 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F004	1	Drywell floor drain (HCW) sump discharge line outboard isolation valve	QBL	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
P10 Makeup Water System													
F016	1	Demineralizer water drywell distribution system inboard containment isolation valve	CK	SA	2	A/C	A	C	C	N/A	Y	SO SC L P	3 mo 3 mo App J 2 yrs
F015	1	Demineralizer water drywell distribution system outboard containment isolation valve	QBL	M	2	A	P	C	C	N/A	Y	L P	App J 2 yrs
P25 Chilled Water System Valves													
F023	2	Chilled water supply line to drywell cooler outboard isolation valve	GT	SO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F024	2	Chilled water supply line to drywell cooler inboard isolation valve	QBL	NO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F025	2	Chilled water return line from drywell cooler inboard isolation valve	QBL	NO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F026	2	Chilled water return line from drywell cooler outboard isolation valve	GT	SO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
P51 Service Air System													

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F002	1	Service air system inboard containment isolation valve	QBL	M	2	A	P	C	C	N/A	Y	L P	App J 2 yrs
F001	1	Service air system outboard containment isolation valve	QBL	M	2	A	P	C	C	N/A	Y	L P	App J 2 yrs
P54 High Pressure Nitrogen Supply System Valves													
F026	1	Nitrogen supply line outboard isolation valve to MSIV and other uses	QBL	AO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F027	1	Nitrogen supply line inboard check valve to MSIV and other uses (7e)	CK	SA	2	A, C	A	O/C	C	N/A	Y	L SC SO	App J RO RO
F009	1	Nitrogen supply line outboard isolation valve to ADS, SRV and ICS isolation valve accumulators	QBL	AO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F010	1	Nitrogen supply line inboard isolation check valve to ADS, SRV and ICS isolation valve accumulators (7e)	CK	SA	2	A, C	A	O/C	C	N/A	Y	L SC SO	App J RO RO
T10 Containment													
F001	3	Drywell wetwell vacuum breaker isolation valve	QBF	NO	2	A	A	O	O/C	as-is	--	P L SO SC	2 yrs 2 yrs 3 mo 3 mo

Table 3.9-8 Inservice Testing (Sheet 19 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F002	3	Drywell wetwell vacuum breaker valve (7c)	VB	SA	2	A, C	A	C	O/C	N/A	--	SO SC L P R	RO RO 2 yrs 2 yrs RO
T15Passive Containment Cooling System Valves													
F001	6	Vent fan isolation valves	QBL	NO	2	A	P	C	O/C	As-Is	--	L	2 yrs
T31 Containment Inerting System Valves													
F012	1	Suppression pool exhaust line outboard isolation valve (7s)	QBF	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs RO RO
F007	1	Air/Nitrogen supply line to suppression pool outboard isolation valve (7s)	QBF	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs RO RO
F008	1	Air/N2 supply line to outboard isolation valve (7s)	QBF	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs RO RO
F009	1	Air/N2 supply line to upper drywell outboard isolation valve (7s)	QBF	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs RO RO
F023	1	N2 makeup line outboard isolation valve	QBL	AO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo

Table 3.9-8 Inservice Testing (Sheet 20 of 20)

Number	Quantity	Description(7)	Valve Type(9)	Actuator(2)	Code Class (1)	Code Category(3)	Valve Function(4)	Normal Position	Safety Position	Fail Safe Position	Containment Isolation Valve	Test Parameter(5)	Test Frequency(6)
F024	1	N2 makeup line to suppression pool outboard isolation valve	QBL	AO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F025	1	N2 makeup line to upper drywell outboard isolation valve	QBL	AO	2	A	A	O	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F010	1	Lower drywell exhaust line outboard isolation valve	QBF	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F011	1	Containment atmospheric exhaust line outboard isolation valve	QBF	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F014	1	Containment atmospheric bleed line outboard isolation valve	QBL	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo
F015	1	Containment atmospheric bleed line outboard isolation valve	QBL	AO	2	A	A	C	C	C	Y	L P SC FC	App J 2 yrs 3 mo 3 mo

Notes:

1. 1, 2 or 3 – ASME B&PV Code class per Section 3.2.
2. Valve actuators:

AO Air operated

EX Explosively actuated

NO Nitrogen operated

M Manually operated

MO Motor operated

PM Process Medium-actuated

SA Self-actuated

SO Solenoid operated

EH Electro-hydraulic operated

3. A, B, C or D – Valve category per ASME OM Code –Subsection ISTC-1300.

4. Valve Function:

   A or P – Active or passive per ASME OM Code – Paragraph ISTC-1300.

5. Valve test parameters per ASME OM Code – Subsection ISTC and Appendix I:

   L Seat leakage rate (Paragraph ISTC-3600 and Subsection 6.2.6.3)

   P Valve position verification (Paragraph ISTC-3700)

   R Safety and relief test including visual examination, set pressure determination and seat tightness testing in accordance with Appendix I of the ASME OM Code. Category A and B requirements for safety and relief valves of ISTC-3500 and ISTC-3700 are excluded per ISTC-1200.

   SO Open stroke tests for Category A and B valves (Paragraph ISTC-3521) and Category C valves (Paragraph ISTC-3522)

   SC Closure stroke tests for Category A and B valves (Paragraph ISTC-3521) and Category C valves (Paragraph ISTC-3522)

   FO Fail open tests for Category A and B valves (Paragraph ISTC-3560)

   FC Fail closed tests for Category A and B valves (Paragraph ISTC-3560)

   X Explosively actuated valve tests (Paragraph ISTC-5260)

6. Valve test frequency for the specified test parameter including summary of exclusions and alternatives per ASME OM Code – Subsection ISTC and Appendix I:

CS Cold shutdown

RO Refueling outages. For position verification: refueling outages, but in no case greater than two years.

E2 Fired and replaced per Paragraph ISTC-5260.

App J Per 10 CFR 50 Appendix J requirements

7. Justifications for ASME OM Code-defined testing exceptions or alternatives as allowed by Paragraphs ISTC-3510 of the ASME OM Code for exercising tests and ISTC-3630 for seat leakage rate tests are as follows.

   7a. Paragraph ISTC-3600 (leak testing requirements) is not applicable to these valves since they function in the course of plant operation in a manner that demonstrates functionally adequate seat leak-tightness.

   7b. Although these valves could be tested one at a time at power, there is a risk of depressurizing the reactor.

   7c. These valves cannot be tested at power because sufficient differential pressure/flow between the wetwell and the drywell cannot be created.

   7d. These valves cannot be tested at power because a reverse flow cannot be established.

   7e. These valves are installed in nitrogen supply lines to nitrogen-operated valves. If the main valve is tested quarterly, the opening function of the check valve will be tested as part of that test. Otherwise the check valve cannot be tested without

potentially stroking the main valve. The closing function cannot be tested at power because a reverse flow cannot be established.

7f. These valves are installed in sensing lines. Valve opening is verified by the continued operation of the sensor. High flow cannot be established through these valves at power to verify valve closure.

7g. These valves cannot be tested for opening at power because of the potential for moving the control rods and cannot be tested for closing at power because a reverse flow cannot be established.

7h. There are squib valves in series with these valves; therefore, normal flow cannot be established through the line. Since the valves are inside containment, an alternate test method using test connections cannot be used.

7i. Valve opening is verified during normal plant operation. Valve closing cannot be verified without stopping feedwater flow in the train.

7j. These valves cannot be stroked without interrupting main steam flow.

7k. Normal flow through these valves cannot be established at power. Since the valves are inside containment, an alternate test method using test connections cannot be used.

7l. These valves cannot be tested at power without potentially operating an SRV.

7m. These valves cannot be tested at power without potentially operating an MSIV.

7n. There are squib valves in series with these valves; therefore, normal flow cannot be established through the line. There is a test connection upstream of the

valves; however, using this connection to test at power would inject cold water into the reactor.

7o. Normal flow cannot be established without initiating Drywell Spray. Since the valves are inside containment, an alternate test method using test connections cannot be used.

7p. Normal flow cannot be established because RWCU/SDC system pressure exceeds FAPCS system pressure.

7q. (Deleted)

7r. (Deleted)

7s. Although these valves could be tested one-at-a-time during power operation, there is a risk of purging/venting the containment during this test.

7t. These valves cannot be tested at power without interrupting feedwater flow.

7u. Valve opening is verified during normal plant operation. Valve closing cannot be verified because a reverse flow cannot be established.

8. General Note on Check Valves: To satisfy the requirement for position verification of the ASME OM Code Paragraph ISTC-3700 for check valves, where local observation is not possible, other indications are used for verification of valve operation.

9. Valve Types (See Table 6.2-15 for a more detailed description of valve types):

GT Gate valve

GB Globe valve

QBL Quarter-turn ball valve

QBF Quarter-turn butterfly valve

CK Check valve

RV Safety and Relief valve

SQ Squib valve

VB Vacuum breaker

Table 3.9-9 Load Combinations and Acceptance Criteria for Class 1 Piping Systems

Condition	Load Combination for all terms[(1) (2)(3)]	Acceptance Criteria
Design	PD + WT	Eq 9 $\leq$ 1.5 S_m NB-3652
Service Level A & B	PP, TE, ΔT1, ΔT2 , TA-TB, RV_1, RV_2I, RV_2D, TSV, SSEI, SSED	Eq 12 & 13 $\leq$ 2.4 S_m Fatigue - NB-3653: U < 0.40[(4)]
Service Level B	PP + WT + (TSV) PP + WT + (RV_1) PP + WT + (RV_2I)	Eq 9 $\leq$ 1.8 S_m, but not greater than 1.5 S_y Pressure not to exceed 1.1P_a (NB-3654)
Service Level C	PP + WT + $[(CHUGI)^2 + (RV_1)^2]^{1/2}$ PP + WT + $[(CHUGI)^2 + (RV_2I)^2]^{1/2}$	Eq 9 $\leq$2.25 S_m, but not greater than 1.8 S_y Pressure not to exceed 1.5 P_a (NB-3655)
Service Level D	PP + WT + $[(SSEI)^2 + (TSV)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (CHUGI)^2 + (RV_1)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (CHUGI)^2 + (RV_2I)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (CONDI)^2 + (RV_1)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (CONDI)^2 + (RV_2I)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (API)^2]^{1/2}$	Eq 9 $\leq$ 3.0 S_m but not greater than 2.0 S_y Pressure not to exceed 2.0 P_a (NB-3656)

(1) RV1 and TSV loads are used for MS Lines only

(2) RV2 represents RV2 ALL (all valves), RV2SV (single Valve) and RV2 AD (Automatic Depressurization operation)

(3) For the SRV discharge piping, all direct loads for SRV and LOCA loads are evaluated for submerged piping.

(4) In conjunction with compliance with RG 1.207, the fatigue usage limit of $\leq$ 0.40 will be used as the criteria for piping locations exempt from pipe break consideration.

Where: API = Annulus Pressurization Loads (Inertia Effect)

CHUGI = Chugging Load (Inertia Effect)

CONDI = Condensation Oscillation (Inertia Effect)

PD = Design Pressure

PP = Peak Pressure or the Operating Pressure Associated with that transient

RV_1 = SRV Opening Loads (Acoustic Wave)

Table 3.9-10 Snubber Loads

Condition	Load Combination(1)(2)	Acceptance Criteria
Service Level B	(TSV) (RV_1) $[(RV_2I)^2 + (RV_2D)^2]^{1/2}$	Vendor Load Capacity Datasheet (LCD) or Vendor Design Report Summary (DRS)
Service Level C	$[(CHUGI)^2 + (CHUGD)^2 + (RV_1)^2]^{1/2}$ $[(CHUGI)^2 + (CHUGD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$	Vendor Load Capacity Datasheet (LCD) or Vendor Design Report Summary (DRS)
Service Level D	$[(SSEI)^2 + (SSED)^2 + (TSV)^2]^{1/2}$ $[(SSEI)^2 + (SSED)^2 + (CHUGI)^2 + (CHUGD)^2 + (RV_1)^2]^{1/2}$ $[(SSEI)^2 + (SSED)^2 + (CHUGI)^2 + (CHUGD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$ $[(SSEI)^2 + (SSED)^2 + (CONDI)^2 + (CONDD)^2 + (RV_1)^2]^{1/2}$ $[(SSEI)^2 + (SSED)^2 + (CONDI)^2 + (CONDD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$ $[(SSEI)^2 + (SSED)^2 + (API)^2 + (APD)^2]^{1/2}$	Vendor Load Capacity Datasheet (LCD) or Vendor Design Report Summary (DRS)

(1) RV_1 and TSV loads are used for MS Lines

(2) RV_2 represents RV_2 ALL (all valves), RV_2SV (single valve) and RV_2 AD (Automatic Depressurization Operation).

Where: TSV = Turbine Stop Valve closure loads

RV1 = SRV Opening Loads (Acoustic Wave)

RV2I = SRV Building Acceleration Loads (Inertia Effect) (all valves)

RV2D = SRV Building Acceleration Loads (Anchor Displacement Loads) (all valves)

CHUGI = Chugging Load (Inertia Effect)

CHUGD = Condensation Oscillation (Anchor Displacement Loads)

SSEI = Safe Shutdown Earthquake (Inertia Effect)

SSED = Safe Shutdown Earthquake (Anchor Displacement Loads)

CONDI = Condensation Oscillation (Inertia Effect)

CONDD = Condensation Oscillation (Anchor Displacement Loads)

API = Annulus Pressurization Loads (Inertia Effect)

APD = Annulus Pressurization Loads (Anchor Displacement Loads)

Table 3.9-11 Strut Loads

Condition	Load Combination(1)(2)(3)	Acceptance Criteria
Service Level A	*WT + TE*	*Vendor Load Capacity Datasheet (LCD) or Vendor Design Report Summary (DRS)*
Service Level B	$WT + TE + (TSV)$ $WT + TE + (RV_1)$ $WT + TE + [(RV_2I)^2 + (RV_2D)^2]^{1/2}$	*Vendor Load Capacity Datasheet (LCD) or Vendor Design Report Summary (DRS)*
Service Level C	$WT + TE + [(CHUGI)^2 + (CHUGD)^2 + (RV_1)^2]^{1/2}$ $WT + TE + [(CHUGI)^2 + (CHUGD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$	*Vendor Load Capacity Datasheet (LCD) or Vendor Design Report Summary (DRS)*
Service Level D	$WT + TE + [(SSEI)^2 + (SSED)^2 + (TSV)^2]^{1/2}$ $WT + TE + [(SSEI)^2 + (SSED)^2 + (CHUGI)^2 + (CHUGD)^2 + (RV_1)^2]^{1/2}$ $WT + TE + [(SSEI)^2 + (SSED)^2 + (CHUGI)^2 + (CHUGD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$ $WT + TE + [(SSEI)^2 + (SSED)^2 + (CONDI)^2 + (CONDD)^2 + (RV_1)^2]^{1/2}$ $WT + TE + [(SSEI)^2 + (SSED)^2 + (CONDI)^2 + (CONDD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$ $WT + TE + [(SSEI)^2 + (SSED)^2 + (API)^2 + (APD)^2]^{1/2}$	*Vendor Load Capacity Datasheet (LCD) or Vendor Design Report Summary (DRS)*

(1) RV_1 and TSV loads are used for MS Lines

(2) RV_2 represents RV_2 ALL (all valves), RV_2SV (single valve) and RV_2 AD (Automatic Depressurization Operation)

(3) TE = Thermal expansion case associated with the transient

Where: *TSV = Turbine Stop Valve closure loads*

WT = Dead Weight

TE = Thermal Expansion

RV_1 *= SRV Opening Loads (Acoustic Wave)*

RV_2I *= SRV Building Acceleration Loads (Inertia Effect) (all valves)*

RV_2D *= SRV Building Acceleration Loads (Anchor Displacement Loads) (all valves)*

CHUGI = Chugging Load (Inertia Effect)

CHUGD = Condensation Oscillation (Anchor Displacement Loads)

SSEI = Safe Shutdown Earthquake (Inertia Effect)

SSED = Safe Shutdown Earthquake (Anchor Displacement Loads)

CONDI = Condensation Oscillation (Inertia Effect)

CONDD = Condensation Oscillation (Anchor Displacement Loads)

API = Annulus Pressurization Loads (Inertia Effect)

APD = Annulus Pressurization Loads (Anchor Displacement Loads)

Table 3.9-12 Linear Type (Anchor and Guide) Main Steam Piping Support

Condition	Load Combination(1)(2)(3)	Acceptance Criteria(4)(5)
Service Level A	WT + TE	*Table NF-3131(a)-1 for Linear Supports*
Service Level B	WT + TE + (TSV) WT + TE + (RV_1) WT + TE + $[(RV_2I)^2 + (RV_2D)^2]^{1/2}$	Table NF-3131(a)-1 for Linear Supports
Service Level C	WT + TE + $[(CHUGI)^2 + (CHUGD)^2 + (RV_1)^2]^{1/2}$ WT + TE + $[(CHUGI)^2 + (CHUGD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$	Table NF-3131(a)-1 for Linear Supports
Service Level D	WT + TE + $[(SSEI)^2 + (SSED)^2 + (TSV)^2]^{1/2}$ WT + TE + $[(SSEI)^2 + (SSED)^2 + (CHUGI)^2 + (CHUGD)^2 + (RV_1)^2]^{1/2}$ WT + TE + $[(SSEI)^2 + (SSED)^2 + (CHUGI)^2 + (CHUGD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$ WT + TE + $[(SSEI)^2 + (SSED)^2 + (CONDI)^2 + (CONDD)^2 + (RV_1)^2]^{1/2}$ WT + TE + $[(SSEI)^2 + (SSED)^2 + (CONDI)^2 + (CONDD)^2 + (RV_2I)^2 + (RV_2D)^2]^{1/2}$ WT + TE + $[(SSEI)^2 + (SSED)^2 + (API)^2 + (APD)^2]^{1/2}$	Appendix F Subarticle F-1334

(1) RV_1 and TSV loads are used for MS Lines

(2) RV_2 represents RV_2 ALL (all valves), RV_2SV (single valve) and RV_2AD (Automatic Depressurization Operation)

(3) TE = Thermal expansion case associated with the transient.

(4) See Subsection 3.7.3.3.1 pertaining to the weights of the frame.

(5) See Subsection 3.9.3.7.1 regarding friction forces induced by thermal in unrestrained direction.

Where: TSV = Turbine Stop Valve closure loads

WT = Dead Weight

TE = Thermal Expansion

RV_1 = SRV Opening Loads (Acoustic Wave)

RV_2I = SRV Building Acceleration Loads (Inertia Effect) (all valves)

RV_2D = SRV Building Acceleration Loads (Anchor Displacement Loads) (all valves)

CHUGI = Chugging Load (Inertia Effect)

CHUGD = Condensation Oscillation (Anchor Displacement Loads)

SSEI = Safe Shutdown Earthquake (Inertia Effect)

SSED = Safe Shutdown Earthquake (Anchor Displacement Loads)

CONDI = Condensation Oscillation (Inertia Effect)

CONDD = Condensation Oscillation (Anchor Displacement Loads)

API = Annulus Pressurization Loads (Inertia Effect)

APD = Annulus Pressurization Loads (Anchor Displacement Loads)

Figure 3.9-1 Stress-Strain Curve for Blowout Restraints

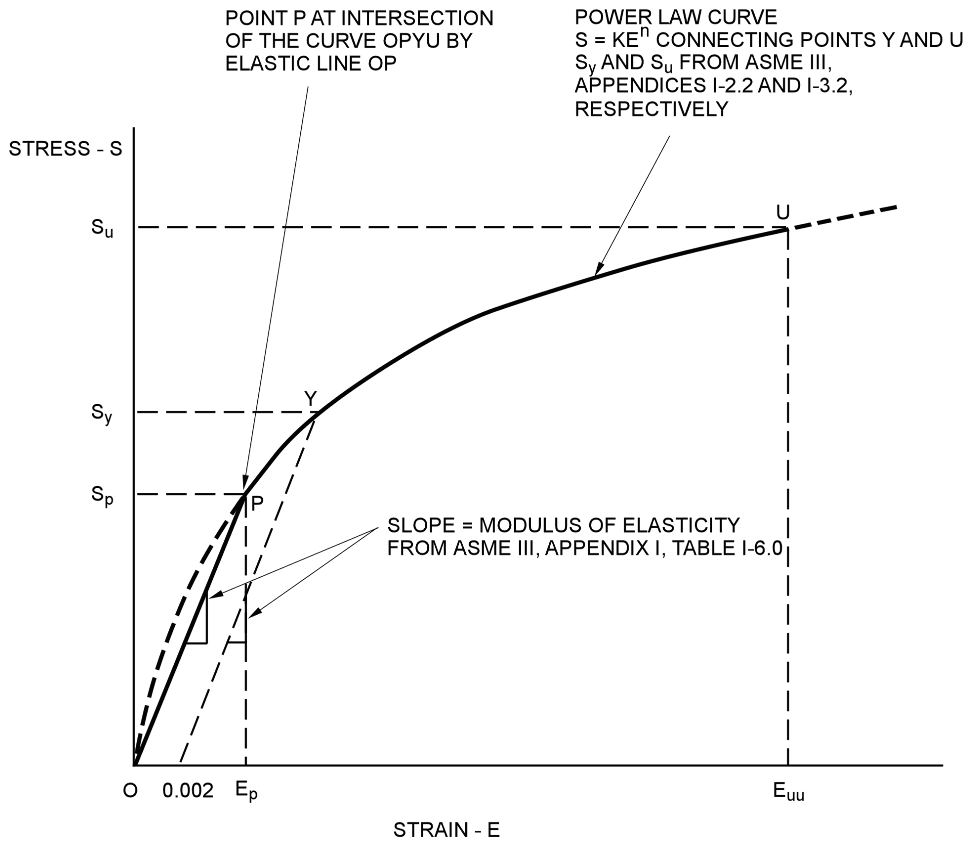


Figure 3.9-2 Minimum Floodable Volume

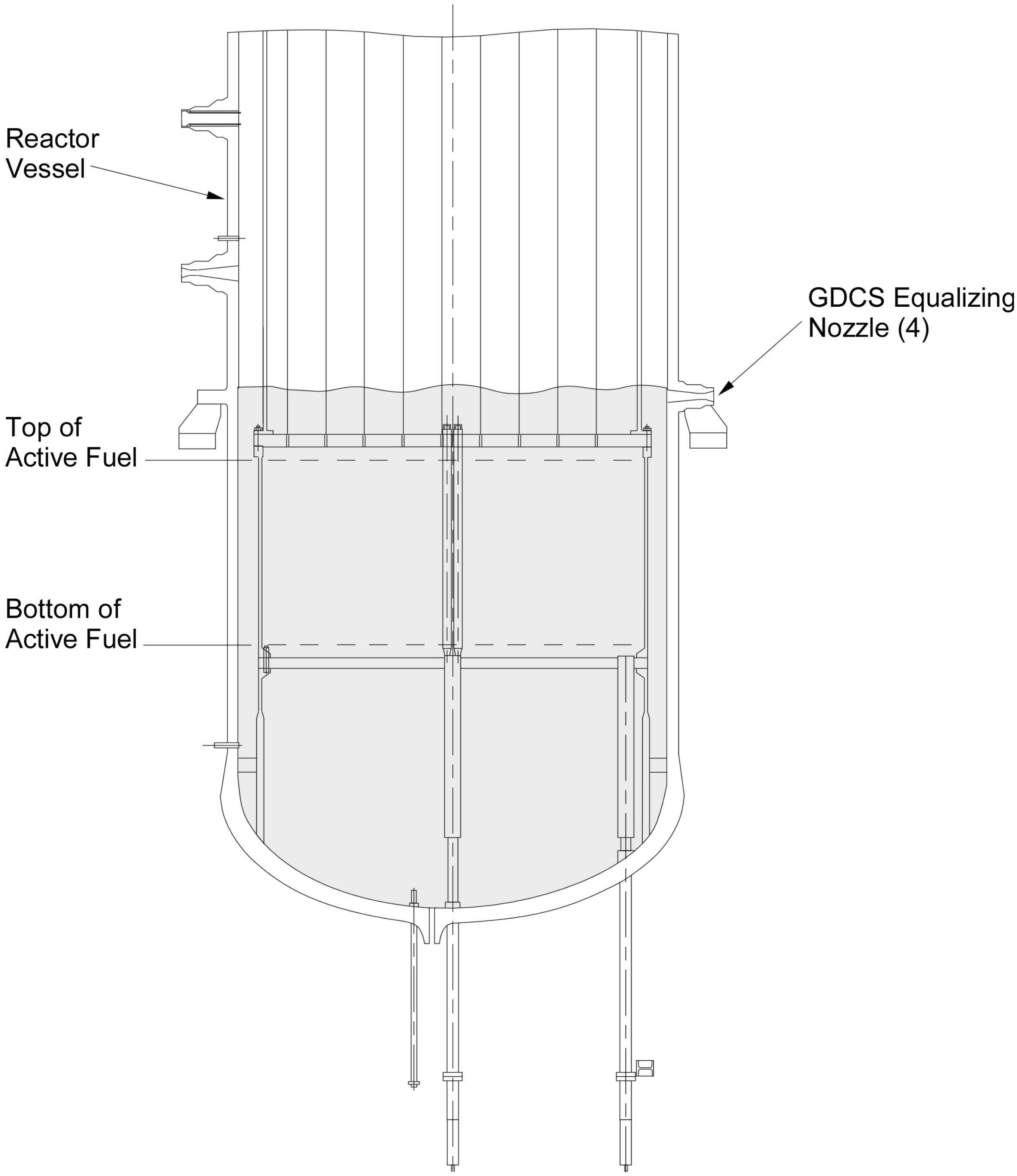


Figure 3.9-3 Recirculation Flow Path

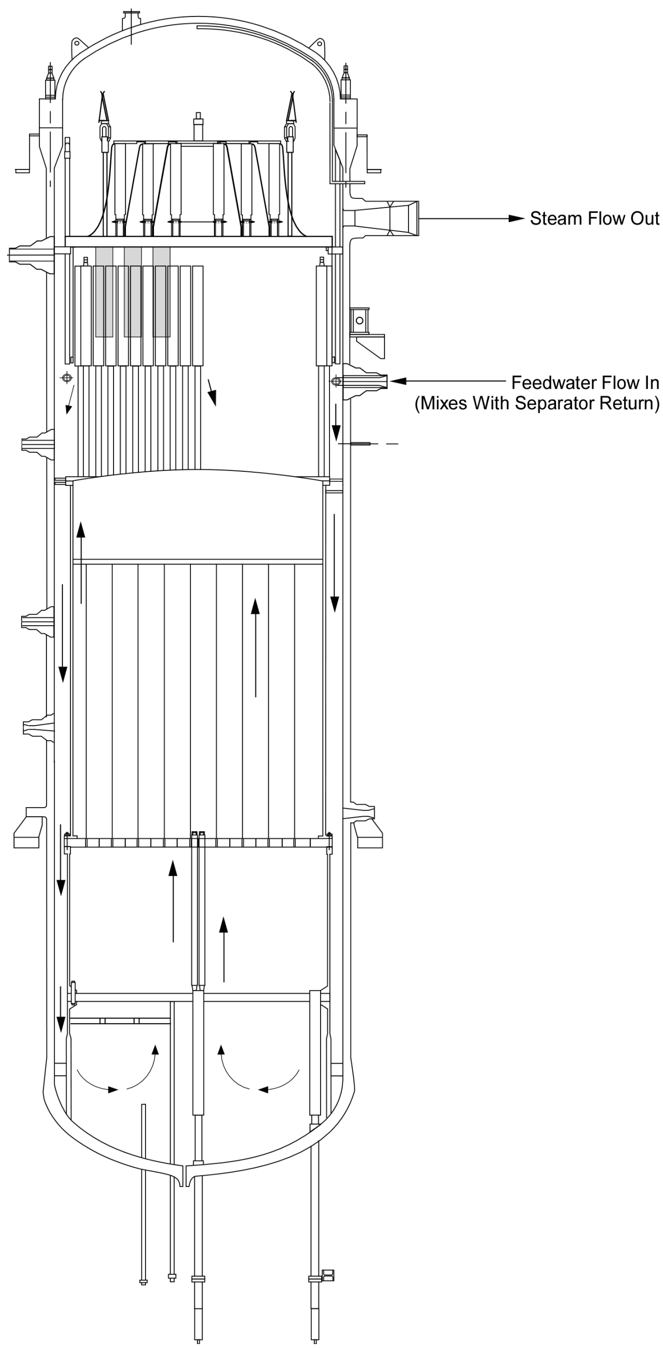


Figure 3.9-4 Fuel Support Pieces

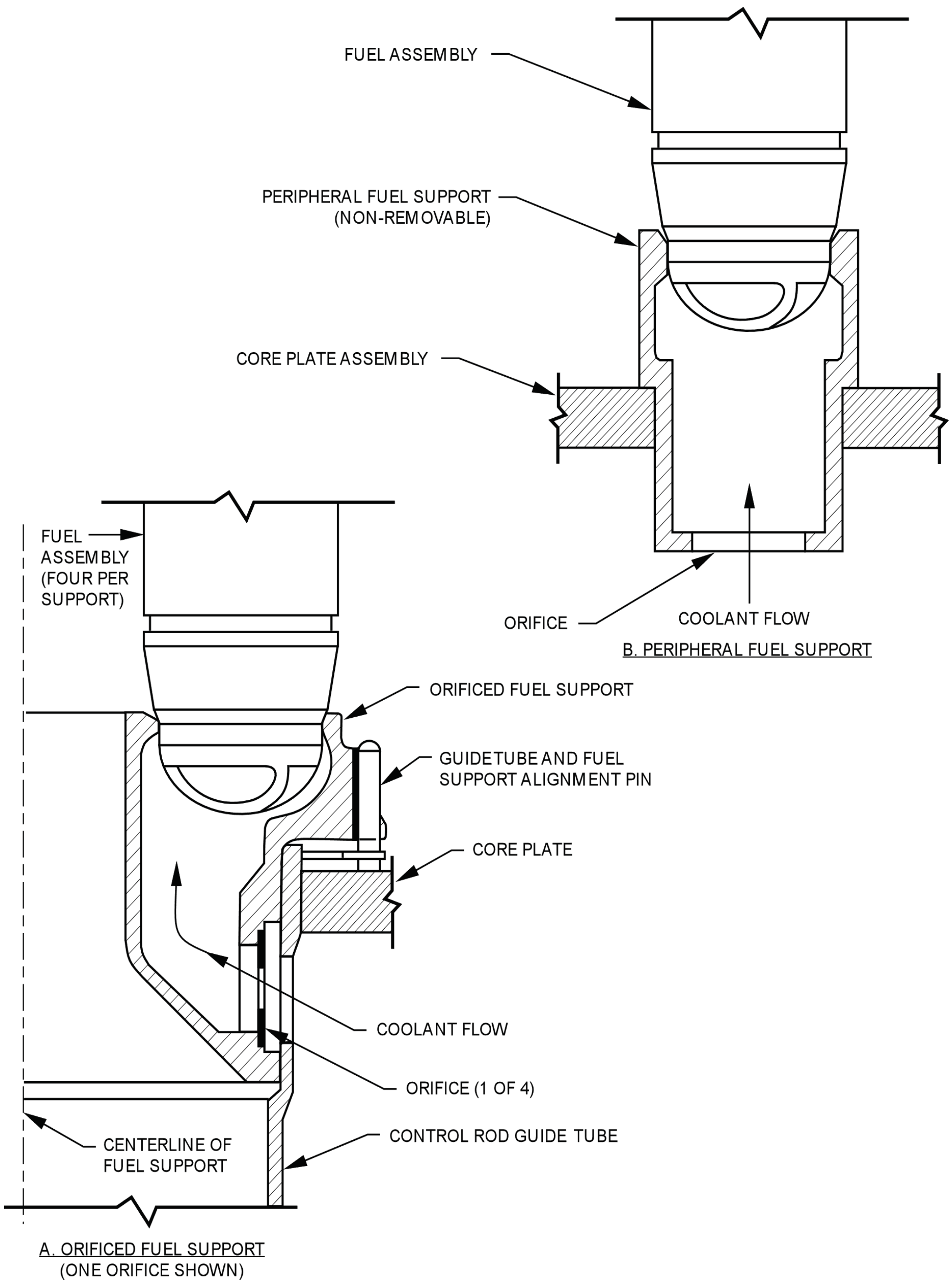


Figure 3.9-5 Pressure Nodes for Depressurization Analysis

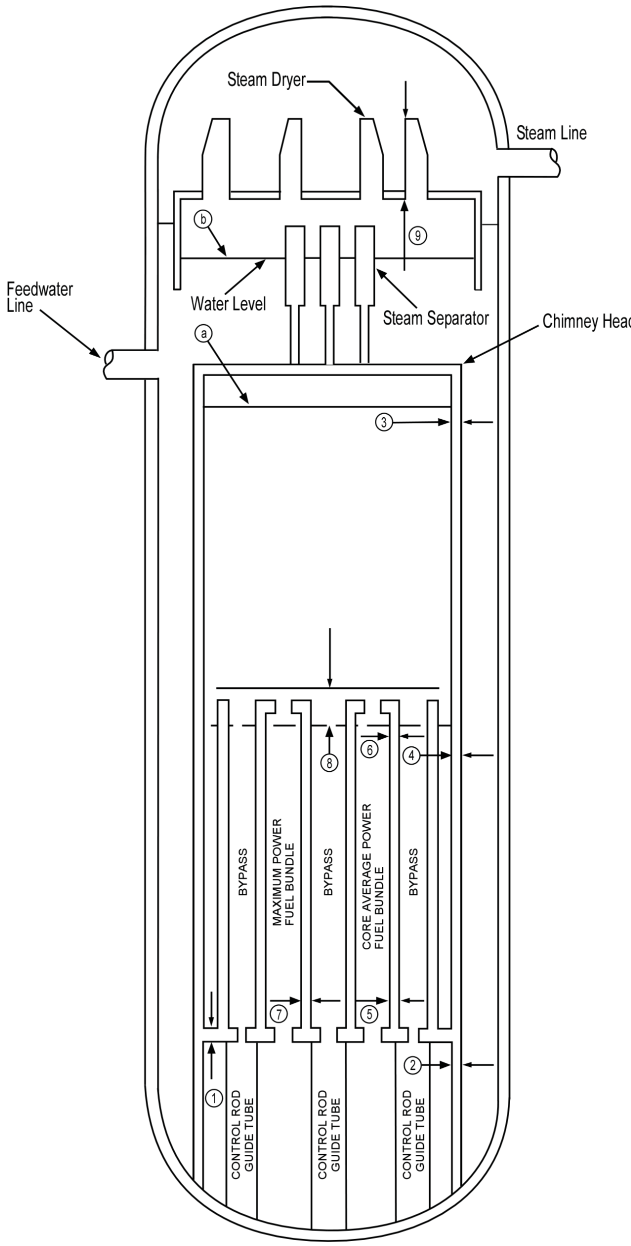


Figure 3.9-6 Flow Chart for Determining Test Data Frequency and Amplitude

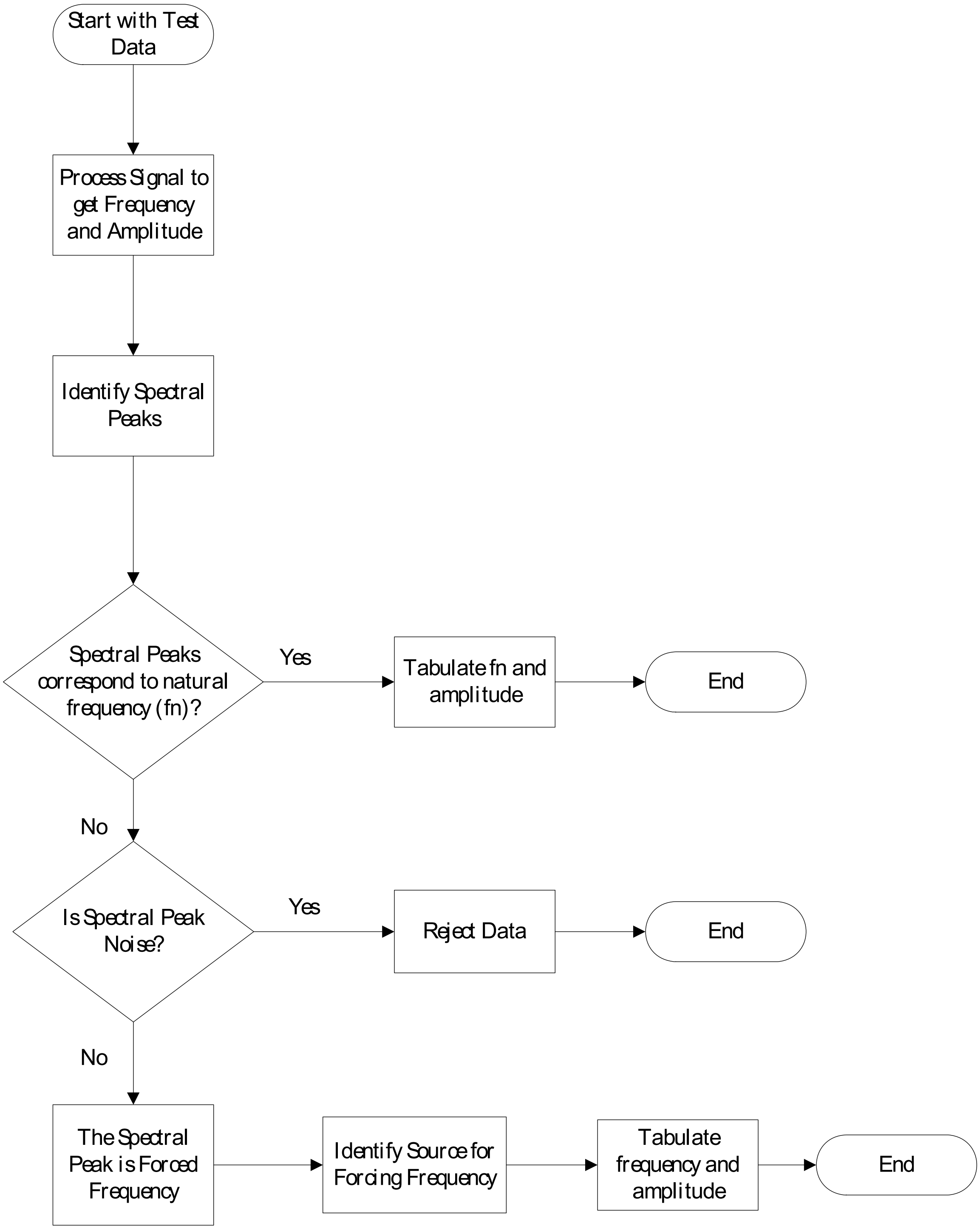


Figure 3.9-7 ESBWR Reactor Assembly Showing Reactor Internal Components

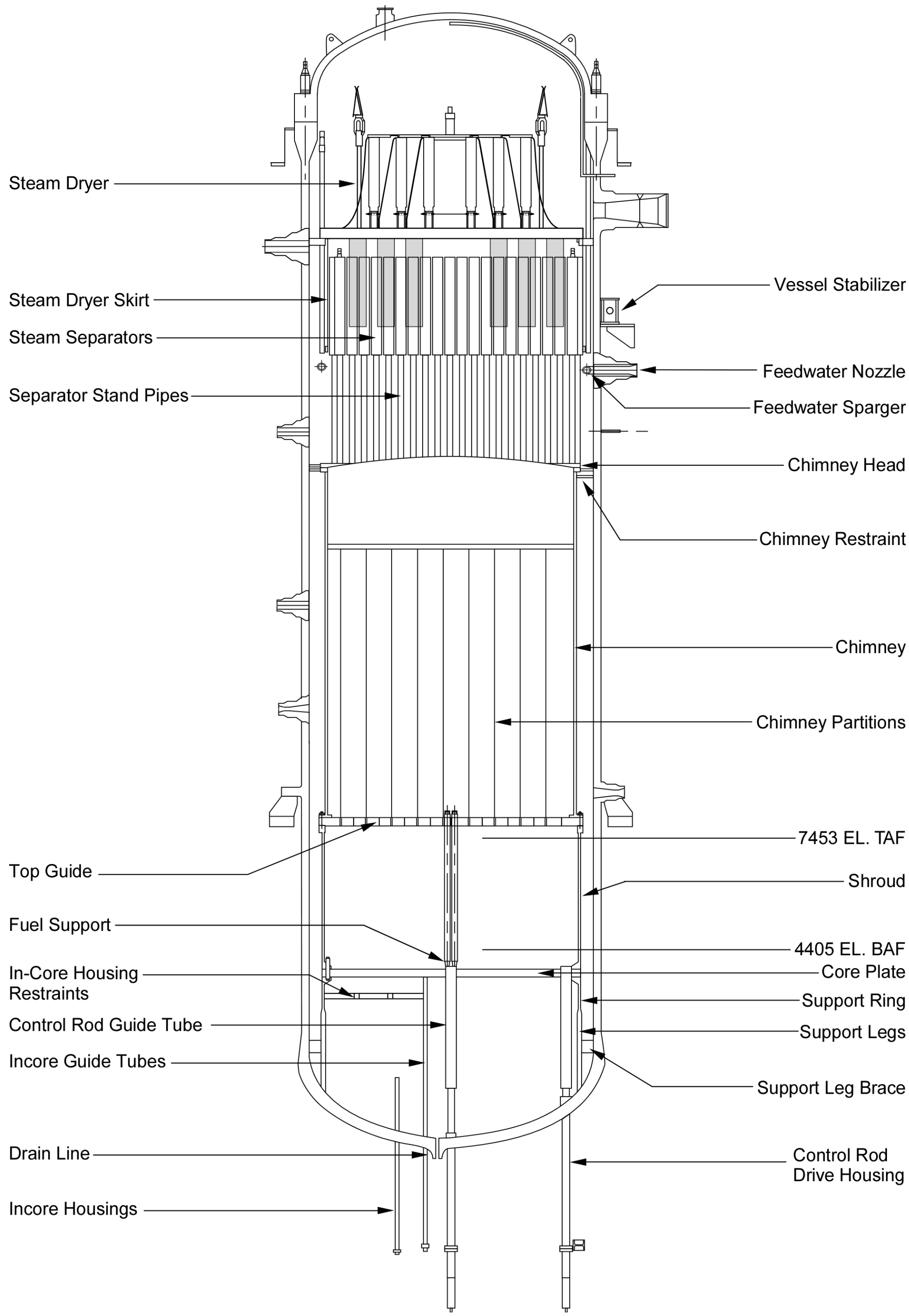


Figure 3.9-8 Typical Shroud, Chimney, and Top Guide Assembly

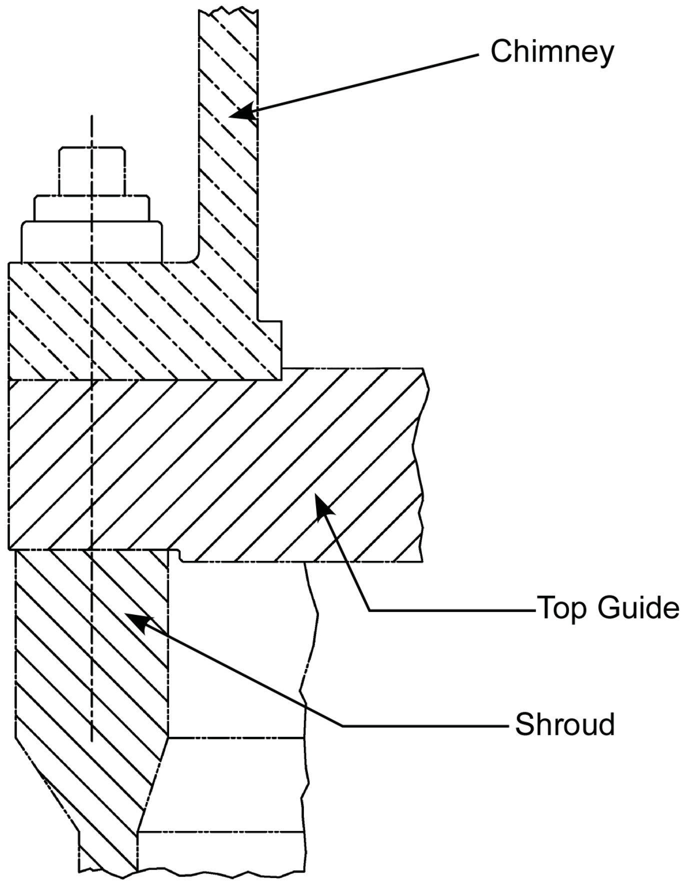


Figure 3.9-9 Typical Core Plate to Shroud Connection

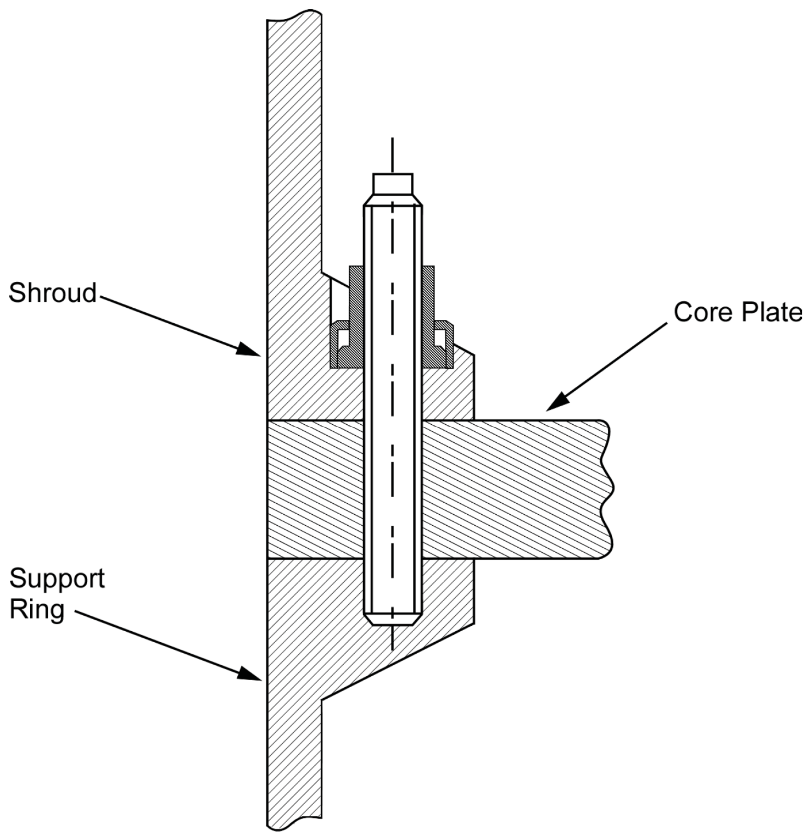


Figure 3.9-10 Typical In-core Guide Tube Lateral Support Connection to Support Ring

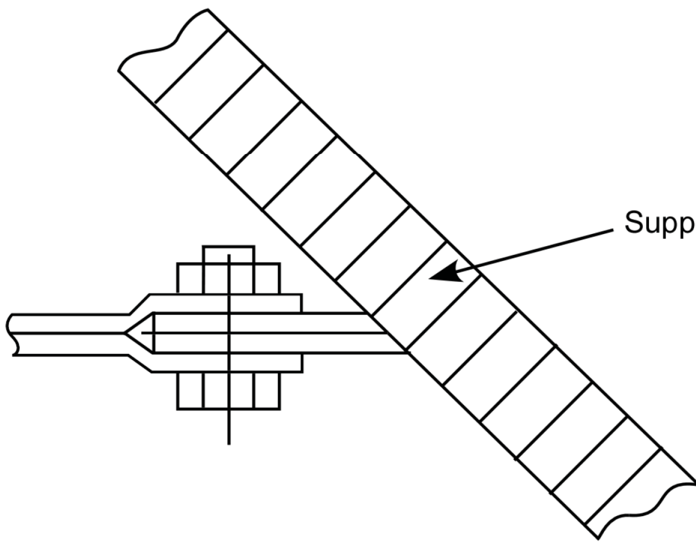


Figure 3.9-11 Typical Inter-Connection Between In-core Guide Tube Lateral Supports

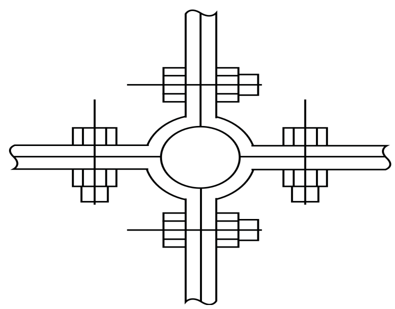
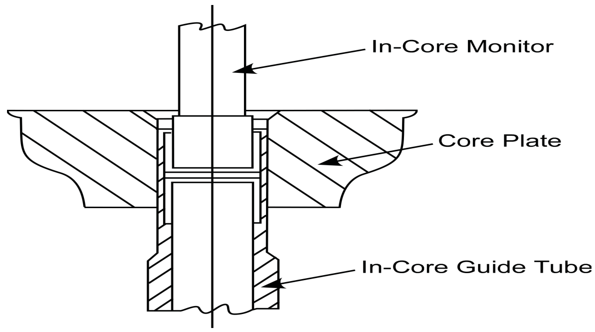


Figure 3.9-12 Typical Connection Between In-Core Guide Tube and Core Plate



3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

This section addresses methods of test and analysis employed to ensure the operability of mechanical and electrical equipment (includes instrumentation and control) under the full range of normal and accident loadings (including seismic), to ensure conformance with the requirements of General Design Criteria 1, 2, 4, 14 and 30 of Appendix A to 10 CFR 50, as well as Appendix B to 10 CFR Part 50, as discussed in SRP 3.10 Draft Revision 3 (Reference 3.10-1) and Appendix S to 10 CFR 50. Mechanical and electrical equipment are designed to withstand the effects of earthquakes, i.e., seismic Category I requirements, and other accident-related loadings. Mechanical and electrical equipment covered by this section include equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment. Also covered by this section is equipment (1) that performs the above functions automatically, (2) that is used by the operators to perform these functions manually, and (3) whose failure can prevent the satisfactory accomplishment of one or more of the above safety-related functions. Instrumentation that is needed to assess plant and environs conditions during and after an accident, as described in Regulatory Guide (RG) 1.97, is also covered by this section. Examples of mechanical equipment included in these systems are pumps, valves, fans, valve operators, battery and instrument racks, control consoles, cabinets, and panels. Examples of electrical equipment are valve operator motors, solenoid valves, pressure switches, level transmitters, electrical penetrations, and pump and fan motors.

The methods of test and analysis employed to ensure the operability of mechanical and electrical equipment meet the relevant requirements of the following regulations, industry codes and standards and Regulatory Guides:

1. Code Federal Regulations (CFR):
    a. 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants,” (Criteria 1, 2, 4, 14 and 30).
    b. 10 CFR 50 Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”
    c. 10 CFR 50 Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants.”
2. Institute of Electrical and Electronics Engineers (IEEE):
    a. IEEE-323-1974, “Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.”
    b. IEEE-382-1996 (R2004), “Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants.”

c. IEEE-344-1987 (R1993), "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

3. American Society of Mechanical Engineers (ASME):

   a. ASME Boiler and Pressure Vessel (B&PV) Code Section III, "Rules for Construction of Nuclear Facility Components."

   b. ASME NQA-1-1983, Addenda NQA-1a-1983, "Quality Assurance Requirements for Nuclear Facility Applications."

   c. ASME B&PV Code Section III, Division 1, Subsection NF, "Rules for Construction of Nuclear Facility Components."

4. U.S. Nuclear Regulatory Commission (NRC) Regulatory Guides:

   a. Regulatory Guide 1.63-1987, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants."

   b. Regulatory Guide 1.122-1978, "Development of Floor Design Response Spectra for Seismic Design of Floor–Supported Equipment or Components."

   c. Regulatory Guide 1.61-1973, "Damping Values for Seismic Design of Nuclear Power Plants."

   d. Regulatory Guide 1.92 – 2006, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

   e. Regulatory Guide 1.29-1978, "Seismic Design Classification."

   f. Regulatory Guide 1.100-1988, "Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants."

The dynamic loads may occur because of the Reactor Building Vibration (RBV) excited by the suppression pool dynamics when a loss-of-coolant-accident, a safety relief valve discharge or a depressurization valve discharge occurs. The non-seismic RBV dynamic loads are described in Table 3.9-1 and 3.9-2 and can be categorized as Service Level B, C, or D depending upon the excitation source.

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.

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. Seismic and dynamic

qualification methodology in Section 4.4 of GE's Environmental Qualification (EQ) Program (Reference 3.10-2) applies to mechanical as well as electrical equipment.

3.10.1 **Seismic and Dynamic Qualification Criteria**

3.10.1.1 **Selection of Qualification Method**

The qualification of Seismic Category I mechanical and electrical equipment is accomplished by test, analysis, or a combination of testing and analysis. Qualification by actual seismic experience as permitted by IEEE 344-1987, is not utilized.

In general, analysis is used to supplement test data although simple components may lend themselves to dynamic analysis in lieu of full scale testing. The deciding factors for choosing between tests or analysis include:

- Magnitude and frequency of seismic and RBV dynamic loadings
- Environmental conditions (Appendix 3H) associated with the dynamic loadings
- Nature of the safety-related function(s)
- Size and complexity of the equipment
- Dynamic characteristics of expected failure modes (structural or functional)
- Partial test data upon which to base the analysis

The selection of qualification method to be used is largely a matter of engineering judgment; however, tests 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).

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 building dynamic analysis, as described in Section 3.7. They are grouped by buildings and by elevations. This RRS definition incorporates the contribution of RBV dynamic loads as specified by the load combinations in Table 3.9-1 and Table 3.9-2. When one type of equipment is located at several elevations 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 GEH's EQ Program (Reference 3.10-2). The program conforms to the requirements of IEEE 323-1974 as modified and endorsed by the RG 1.89, and meets the criteria contained in IEEE 344-1987 as modified and endorsed by RG 1.100.

3.10.1.4 **Dynamic Qualification Report**

The Dynamic Qualification Report (DQR) identifies all Seismic Category I electrical and mechanical equipment and their supports. The DQR contains the following:

- 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 Material Parts List item number and name, the qualification method, the input motion, the supporting structure of the equipment, and the corresponding qualification summary table or vendor’s qualification report.
- The mode of safety-related operation (i.e., active, manual active or passive) of the equipment along with the manufacturer identification and model numbers shall also be tabulated in the DQR. The operational mode identifies the instrumentation, device, or equipment:
  - That performs the safety-related functions automatically;
  - That is used by the operators to perform the safety-related functions manually; or
  - Whose failure can prevent the satisfactory accomplishment of one or more safety-related functions.

[START COM 3.10-003] Detroit Edison shall submit to the NRC, no later than 1 year after issuance of the combined license or at the start of construction as defined in 10 CFR 50.10(a), whichever is later, its implementation schedules for completing of the following ITAACs. Detroit Edison shall submit updates to the ITAAC schedules every 6 months thereafter and, within 1 year of its scheduled date for initial loading of fuel, and shall submit updates to the ITAAC schedules every 30 days until the final notification is provided to the NRC under paragraph (c)(1) of this section." **[END COM 3.10-003]**

[START COM 3.10-001] The Dynamic Qualification Report and documentation that describe the seismic and dynamic qualification methods will be made available for NRC staff review, inspection, and audit. Information that verifies the seismic and dynamic qualification will be made available to the NRC to facilitate reviews, inspections, and audits throughout the process. **[END COM 3.10-001]** **[START COM 3.10-002]** UFSAR information will be revised, as necessary, as part of a subsequent UFSAR update. **[END COM 3.10-002]**

Section 17.5 defines the Quality Assurance Program requirements that are applied to equipment qualification files, including requirements for handling safety-related quality records, control of purchased material, equipment and services, test control, and other quality related processes.

3.10.2 **Methods and Procedures for Qualifying Mechanical and Electrical Equipment**

The following subsections describe the methods and procedures incorporated in the above mentioned dynamic qualification program. Described here are the general methods and procedures for qualifying by testing, analysis, or combined testing and analysis, the Seismic Category I mechanical and electrical equipment for operability during and after the safe shutdown earthquake

(SSE) loads and Service Level D RBV dynamic loads and for continued structural and functional integrity of the equipment after low level earthquake loading of lesser magnitude (Section 3.7) and Service Level B RBV dynamic loads.

3.10.2.1 **Qualification by Testing**

The testing methodology includes the hardware interface requirements and the test methods.

Interface Requirements

Intervening structures or components (such as interconnecting cables, bus ducts, conduits) that serve as interfaces between the equipment to be qualified and that is 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 due to the SSE loads including appropriate RBV dynamic loads. Embedment loads and mounting requirements for the equipment supports are also specified in this manner.

Test Methods

The test method is biaxial, random single- or multi-frequency excitation to envelop generic RRS levels in accordance with Section 7 of IEEE 344-1987. Past testing demonstrates that Seismic Category I electrical equipment has 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.

Biaxial testing applies input motions to both the vertical and one 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 performed:

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

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 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.

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 accelerations and any resonances at such interface locations are recorded during the equipment test and documented in the test report.

Dynamic Testing Sequence

The test sequence includes vibration conditioning, exploratory resonance search, low level earthquake loading including Service Level B RBV dynamic loads, and the SSE loading including Service Level D RBV dynamic loads.

Vibration Conditioning — If required by the applicable qualification standard for the equipment, vibration conditioning is performed at this point in the sequence and the vibration conditioning details are given.

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 0.2 g, 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.

Testing for Low Level Earthquake Loading and RBV Dynamic Loads — This test is performed on all test specimens. This test is conducted to demonstrate that the low level earthquake (as defined in Section 3.7) loads combined with Service Level B RBV dynamic loads do not degrade the continued structural and functional integrity of the equipment. Strong motion test inputs are applied for a minimum of 15 seconds in each orientation. Operability of equipment is verified as described below.

Testing for SSE Loading and RBV Dynamic Loads — An SSE test including other appropriate Service Level D RBV dynamic loads is performed on all test specimens. This test is conducted to demonstrate that equipment would perform its safety-related function through a SSE (as defined in Section 3.7) combined with Service Level D RBV dynamic loads. The strong motion of the test lasts a minimum of 15 seconds in each orientation. Operability of equipment is verified as described in the next subsection.

Qualification for Operability — In general, analyses are only used to supplement the 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 mechanical and electrical equipment have safety-related function requirements before, during, and after seismic events. Other equipment (such as plant status display equipment) have requirements only before and after seismic events. All equipment is operated at appropriate times to demonstrate the 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 is acceptable provided proper documentation of such tests and analyses is available.

Documentation of Testing

Qualification results are documented and include, but are not necessarily limited to the following:

- Locations of accelerometers.
- Resonant frequency, if any, and transmission ratios (if exploratory tests are applicable).
- Equipment damping coefficients if there is resonance in the 1-60 Hz range or over the range of the test response spectra (if exploratory tests are applicable).
- Test equipment used.
- Approval signature and dates.
- Description of test facility.
- Summary of results.
- Equipment seismic qualification conclusions (including RBV dynamic loads).
- Justification for using single axis or single frequency tests for all items that are tested in this manner.

See Subsection 3.10.1.4 for additional information on the documentation of test results.

3.10.2.2 **Qualification by Analysis**

The discussion presented in the following subsections applies to the qualification of equipment by analysis.

Analysis Methods

Dynamic analysis or an equivalent static analysis, described in Subsection 3.7.3, is employed to qualify the equipment. In general, the choice of the analysis is based on the expected design margin, because 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 (cutoff frequency of RRS) 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, at a conservative and justifiable value of damping.

If the equipment is determined to be flexible (i.e., with the fundamental frequency of the equipment within frequency range of the input spectra) and not simple enough for equivalent static analysis, a dynamic analysis method is applied.

Acceptance Criteria for Qualification by Analysis

The structural and functional integrity of the equipment is maintained under low level earthquake loads including appropriate RBV dynamic loads in combination with normal operating loads. Where applicable, normal operating and SSE loads including appropriate RBV dynamic loads do not result in failure of the equipment to perform its safety-related function(s).

Documentation of Analysis

Qualification results are documented and include, but are not necessarily limited to equipment specification requirements, a summary of qualification results, and justification that the methods used demonstrate that the equipment does not malfunction. See Subsection 3.10.1.4 for additional information on the documentation of qualification results.

3.10.2.3 **Qualification by Combined Testing and Analysis**

In some instances, it is not practical to qualify the 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.

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 that best excite the various modes of vibration of the equipment. Data are obtained from sensors for subsequent analysis of the equipment performance under seismic plus appropriate 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 the SSE loads including appropriate RBV dynamic loads, by using analysis to justify the excitation, and by utilizing the test data on modal frequencies to verify the mathematical model.

Extrapolation of Similar Equipment

As discussed in IEEE 344-1987, 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. Assurance is obtained that changes from originally tested equipment do not result in the formation of previously non-existent resonances.

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.

Extrapolation of Dynamic Loading Conditions.

Test results can be extrapolated for dynamic loading conditions in excess of or different from previous tests 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.

Documentation of Combined Testing and Analysis

Qualification results are documented and include, but are not necessarily limited to, equipment specification requirements, a summary of qualification results, and justification that the methods used demonstrate that the equipment does not malfunction.

If qualification is by analysis and testing or by extrapolation from similar equipment, the report includes:

- Reference to the specific method of combined analysis and testing used
- Description of equipment involved
- Analysis data
- Test data
- Justification of results

When extrapolation of data is made from similar equipment, a description of the differences between the equipment items involved is required. Justification that the differences do not degrade the seismic adequacy below acceptable limits and any additional supporting data shall be included.

See Subsection 3.10.1.4 for additional information on the documentation of qualification results.

3.10.2.4 (Deleted)

3.10.3 Analysis or Testing of Electrical Equipment Supports

The following subsections describe the general methods and procedures, as incorporated in the dynamic qualification program (see Subsection 3.10.1.3), for analysis and testing of supports of Seismic Category I 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) are tested with the equipment installed. Otherwise, a dummy is employed to simulate inertial mass effects and dynamic coupling to the support.

Combined stresses of the mechanically designed component supports are maintained within the limits of ASME B&PV 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 delineated in Section 3.8.

3.10.3.1 Nuclear Steam Supply System 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 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 Seismic 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 four basic generic panel types: vertical board, instrument panel, relay rack, and NEMA Type 12 enclosure. One or more of each type are tested to full acceleration levels and qualified using the above procedures. From these 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 Electrical Equipment Supports**

Supports for Battery Racks, Instrument Racks, Control Consoles, Cabinets, and Panels

Response spectra for floors where Seismic Category I equipment is located are supplied to each vendor. The vendor submits test data and/or calculations to verify that the equipment did 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 or with dummy loads. 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.

Cable Trays and Conduit Supports

Seismic Category I cable trays and conduit supports are designed by the response spectrum method. Analysis and dynamic load restraint measures are based on combined limiting values for static load, span length, and response to excitation at the natural frequency. Restraint against excessive lateral and longitudinal movement uses the structural capacity of the tray to determine the spacing of the fixed support points. Provisions for differential motion between buildings are made by breaks in the trays and flexible connections in the conduit.

The following loadings are used in the design and analysis of Seismic Category I cable tray and conduit supports:

Loads

- Dead loads and live loads - 112 kg/m (75 lbm/linear-ft) load used for 0.46 m (18 in) and wider trays; 75 kg/m (50 lbm/linear-ft) load used for 0.31 m (12 in) and narrower trays.
- Dynamic loads - SSE loads plus appropriate RBV dynamic loads.

Dynamic Analysis

- Regardless of cable tray function, all supports are designed to meet Seismic Category I requirements. Seismic and appropriate RBV dynamic loads are determined by dynamic analysis using appropriate response spectra.
- Floor Response Spectra — Floor response spectra used are those generated for the supporting floor. In case supports are attached to the walls or to two different locations, the upper bound envelope spectra are used. In many cases, to facilitate the design, several floor response spectra are combined by an upper bound envelope.

Structural requirements for Conduits and Cable Tray supports are also specified in Subsection 3.8.4.1.6.

Local Instrument Supports

For field-mounted Seismic Category I instruments, the following is applicable:

- The mounting structures for the instruments have a fundamental frequency above the excitation frequency of the RRS.
- 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.

Instrument Tubing Support

The following bases are used in the seismic and appropriate RBV dynamic loads design and analysis of Seismic Category I instrument tubing supports:

- The supports are qualified by the response spectrum method.

- Dynamic load restraint measures and analysis for the supports are based on combined limiting values for static load, span length, and computed dynamic response.
- The Seismic Category I instrument tubing systems are supported so that the allowable stresses permitted by Section III of ASME B&PV Code are not exceeded when the tubing is subjected to the loads specified in Subsection 3.9.2 for Class 2 and 3 piping.

3.10.3.3 **Documentation of Testing or Analysis of Electrical Supports**

Qualification results are documented and include, but are not necessarily limited to, equipment specification requirements, a summary of qualification results, and justification that the methods used demonstrate that the equipment does not malfunction. If qualification is by analysis, testing or extrapolation from similar equipment, the report includes:

- Reference to the specific method of combined analysis and testing used
- Description of equipment involved
- Analysis data
- Test data
- Justification of results

When extrapolation of data is made from similar equipment, a description of the differences between the equipment items involved is required. Justification that the differences do not degrade the seismic adequacy below acceptable limits and any additional supporting data shall be included.

See Subsection 3.10.1.4 for additional information on the documentation of qualification results.

3.10.4 **COL Information**

3.10.4.1-A **Dynamic Qualification Report**

This COL is addressed in Subsection 3.10.1.4.

3.10.4-2-H **Equipment Qualification Records (Deleted)**

3.10.5 **References**

3.10-1 USNRC, SRP 3.10 Draft Revision 3 (04/1996), “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.”

3.10-2 General Electric Co., “General Electric Environmental Qualification Program,” NEDE-24326-1-P, Proprietary Document, January 1983.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

3.11.1 Description Requirements

This section describes the requirements for the environmental qualification (EQ) elements of the equipment qualification program as related to electrical and mechanical equipment. The equipment qualification program also includes dynamic and seismic qualification of safety-related electrical and mechanical equipment. Dynamic qualification is addressed in Sections 3.9 and 3.10 for Seismic Category I mechanical and electrical equipment, respectively, and the discussion in this section focuses on the environmental qualification elements of the equipment qualification program.

The equipment qualification program includes safety-related electrical and mechanical equipment located in harsh and mild environments. Safety-related electrical equipment consists of all safety-related electrical power and instrumentation and control (I&C) equipment, which includes all safety-related analog (non-digital) and digital I&C components. Computer-based I&C equipment is a subset of digital I&C components.

Mechanical, electrical, and I&C equipment associated with systems described below are reviewed to determine whether they are designed to meet the requirements described under the acceptance criteria as follows:

1. Equipment associated with systems that are essential for emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment.
2. Equipment that initiates the above functions automatically.
3. Equipment that is used by the operators to initiate the above functions manually.
4. Equipment whose failure can prevent the satisfactory accomplishment of one or more of the above safety functions.
5. Other electrical equipment important to safety, as described in 10 CFR 50.49(b)(1) and (2).
6. Certain post-accident monitoring equipment, as described in 10 CFR 50.49(b)(3) and Regulatory Guide (RG) 1.97.

The electrical equipment identified in 10 CFR 50.49 as electric equipment important to safety covered by (b)(1), (b)(2), and (b)(3) is included in the equipment qualification program.

The equipment qualification program includes qualification of safety-related electrical and mechanical equipment for natural phenomena and external events, unless the adverse effects are precluded by design. For example, location of safety-related electrical and mechanical equipment within safety-related structures may preclude the adverse effects of flood, wind, tornados, and tornado missiles.

The equipment qualification program includes safety-related electrical equipment, including I&C equipment in a mild environment. Safety-Related Distributed Control and Information System equipment located in areas characterized as mild environments, also meets RG 1.209, "Guidelines for Environmental Qualification of Safety-Related Computer-based Instrumentation and Control Systems in Nuclear Power Plants," (Reference 3.11-4), and type testing is the preferred method of qualification.

Mild environments do not experience a loss-of-coolant-accident (LOCA), high energy line break (HELB), or main steamline break (MSLB) and have the environmental limits shown in Table 3H-13.

Equipment supporting RTNSS functions located inside containment are included in the equipment qualification program, and are qualified using the appropriate methods for their location. The remainder of the RTNSS equipment is qualified as outlined in Section 19A. Table 3.11-1 includes RTNSS equipment located inside containment.

The equipment in the EQ program is referred to as EQ equipment.

3.11.1.1 **Applicable Regulations and Standards**

The environmental qualification of electrical and mechanical equipment meets the relevant requirements of the following regulations:

1. Code Federal Regulations (CFR):
    a. 10 CFR 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."
    b. 10 CFR 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."
    c. 10 CFR 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."
    d. 10 CFR 50, Appendix A, General Design Criterion 23, "Protection System Failure Modes."
    e. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
    f. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Section III, "Design Control," Section XI, "Test Control," and Section XVII, "Quality Assurance Records."
2. Institute of Electrical and Electronics Engineers (IEEE):
    a. IEEE-323-2003, "Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." Note: Applies only to electrical equipment in a mild environment.
    b. IEEE-317-1983 (R2003), "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations."

c. IEEE-383-2003, “Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations.”

d. IEEE-420-2001, “Standard for the Design and Qualification of Class 1E Control Boards, Panels and Racks Used in Nuclear Power Generating Stations.”

e. IEEE-535-1986 (R1994), “Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations.”[1]

f. IEEE-603-1991, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations.”

g. (Deleted)

h. IEEE-638-1992 (R2006), “Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Stations.”

i. IEEE-649-1991 (R2004), “Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations.”

j. IEEE-650-1990 (R1998), “Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations.”

k. IEEE-382-1996 (R2004), “Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants.”

l. (Deleted)

m. IEEE-572-1985 (R2004), “Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations.”

n. IEEE-634-2004, “Standard Cable-Penetration Fire Stop Qualification Test.”

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

p. IEEE-334-1994 (R1999), “IEEE Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations.”

q. IEEE-344-1987 (R1993), “Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.”

r. IEEE-497-2002, “Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations.”

s. IEEE-7-4.3.2-2003, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations.”

t. IEEE-1202-2006, “Standard for Flame Propogation Testing of Wire and Cable.”

3. American Society of Mechanical Engineers (ASME):

---

1 Applies except that duty cycle is 72 hours.

a. ASME Boiler and Pressure Vessel (B&PV) Code Section III, “Rules for Construction of Nuclear Facility Components.”

b. ASME NQA-1-1983, Addenda NQA-1a-1983, “Quality Assurance Program Requirements for Nuclear Facilities.”

4. U.S. Nuclear Regulatory Commission (NRC) Regulatory Guides:

a. Regulatory Guide 1.63, “Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants.”

b. Regulatory Guide 1.73, “Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants.”

c. Regulatory Guide 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.”

d. Regulatory Guide 1.131, “Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants.”

e. Regulatory Guide 1.153, “Criteria for Safety Systems.”

f. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactor.”

g. Regulatory Guide 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.”

h. Regulatory Guide 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems.”

i. Regulatory Guide 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants.”

j. Regulatory Guide 1.40, “Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants.”

k. Regulatory Guide 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants.”

l. Regulatory Guide 1.156, “Environmental Qualification of Connection Assemblies for Nuclear Power Plants.”

m. Regulatory Guide 1.158, “Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants.”

5. Department of Defense (DOD) Military Standards (MIL-STD)

a. MIL-STD 461E, “Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment.”

6. International Electrotechnical Commission (IEC)

a. 61000-4 series, “Electromagnetic Compatibility (EMC): Testing and Measurement Techniques.”

3.11.1.2 General Requirements

Environmental design and qualification used to implement the relevant requirements of 10 CFR 50.49; General Design Criteria (GDC) 1, 2, 4 and 23; and 10 CFR 50, Appendix B, Quality Assurance Criteria III, XI, and XVII are as follows:

1. The equipment is designed to have the capability of performing its design safety-related functions under all anticipated operational occurrences (AOOs) and normal, accident, and post-accident environments and for the length of time for which its function is required.
2. The equipment environmental capability is demonstrated by appropriate testing and analyses.
3. A quality assurance program meeting the requirements of 10 CFR Part 50, Appendix B, is established and implemented to provide assurance that all requirements have been satisfactorily accomplished.

A review is performed to assure conformance with the environmental design basis requirements of 10 CFR Part 50, Appendix A, GDC 4 which states, in part, that “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.”

3.11.1.3 Definitions

Normal Operating Conditions — Planned, purposeful, reactor operating conditions including startup, power range, hot standby (condenser available), shutdown, and refueling.

Anticipated Operational Occurrences (AOOs) – Conditions of normal operation expected to occur one or more times during the life of the nuclear power unit and include but not limited to loss of the turbine generator set, isolation of the main condenser and loss of offsite power.

Test Conditions — Planned testing including pre-operational tests.

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 reactor coolant pressure boundary rupture may qualify as an accident; a fuel cladding defect does not).

Design Basis Event (DBE) or Design Basis Accident (DBA) – Postulated events used in the design to establish the acceptable performance requirements for structures, systems, and components.

Equipment – The physical envelope of structures, systems and components which is the device or represents a collection of physically connected devices that perform the required function, or a

group of related and representative functions, that as a whole unit is representative of how it will be installed in the plant. Specific to electrical equipment, including computer-based I&C, this physical envelope is defined as the cabinet or enclosure. The cabinet or enclosure contains representative and functional electrical or electronic assemblies and components (e.g.; chassis, controller packages, instruments, power supplies, video display units, fans, cabling, wiring, baffles, auxiliary devices). This does not include Interfaces which are defined separately.

Equipment Qualification – The generation and maintenance of evidence to ensure equipment will operate on demand to meet system performance requirements during normal and AOO service conditions and postulated design basis events.

Harsh Environment – An environment resulting from a design basis event, i.e., LOCA, HELB, and MSLB.

Interfaces – Physical attachments, mounting, auxiliary components, and connectors (electrical and mechanical) to the equipment at the equipment boundary.

Margin – The difference between service conditions and the conditions used for equipment qualification.

Mild Environment – An environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including AOOs.

Post-Accident Conditions — The environment after an accident where equipment must perform its safety-related function and must remain in a safe mode after the safety-related function is performed.

Qualified Life – The period of time, prior to the start of a design basis event, for which the equipment was demonstrated to meet the design requirements for the specified service conditions.

Service Conditions – Environmental, loading, power, and signal conditions expected as a result of normal operating requirements, expected extremes (abnormal) in operating requirements, and postulated conditions appropriate for the design basis events of the station.

Significant Aging Mechanism – An aging mechanism that, under normal and abnormal service conditions, causes degradation of equipment that progressively and appreciably renders the equipment vulnerable to failure to perform its safety-related function(s) during the design basis event conditions.

3.11.2 **Equipment Identification**

The equipment qualification program generates and maintains a list of EQ equipment located in harsh and mild environments. The systems containing EQ equipment are identified in Table 3.11-1.

The Environmental Qualification Document (EQD) summarizes the qualification results for all EQ equipment in the equipment qualification program. The EQD is current and in an auditable form for

the entire period during which the covered item is installed or is stored for future use to permit verification that each item meets the equipment qualification requirements.

3.11.3 Environmental Conditions

3.11.3.1 General Requirements

Environmental Design Bases

Analysis is performed to identify the environmental design bases including the definition of AOO and normal, accident, and post-accident environments. DBA and AOO define the temperature and pressure time-dependent information for areas subject to accidents and AOOs.

EQ equipment is qualified to the worst-case environmental conditions for the areas in which they are located for the duration that they are required to perform their safety-related function.

The environmental design basis includes the safety-related function for each item of safety-related equipment and its acceptance criteria; electromagnetic interference/radio frequency interference (EMI/RFI) and voltage Surges; environmental conditions including temperature, equipment heating, Heating, Ventilation and Air Conditioning (HVAC) and lack of HVAC, inside and outside maximum and minimum temperatures, and time dependency of temperatures.

The safety-related functions are either functional performance requirements or fail-safe requirements. A fail-safe safety-related function consists of not failing in a manner detrimental to plant safety, accident mitigation, or prevention of a safety-related function. The basis for the safety-related function is included in the qualification documentation.

The following provides detailed information on each environment included in the environmental design basis. The environments are considered for electrical and mechanical equipment in the EQ program.

Temperature

The temperature qualification for EQ equipment in harsh environments is by test. EQ equipment is demonstrated to perform as intended while exposed to the qualification temperature. The qualification temperature is 10°C higher than the maximum temperature to which the equipment is exposed for the worst-case DBA, while the equipment is under its maximum loading, to comply with margin requirements.

For EQ safety-related electrical equipment in mild environments, except for digital computer-based I&C, the temperature qualification methods are test or analysis. The maximum qualification temperature is 10°C higher than the maximum temperature to which the equipment is exposed for the worst-case AOO, while the equipment is under its maximum loading, to comply with margin requirements. The minimum qualification temperature is 10°C lower than the minimum temperature to which the equipment is exposed for the worst-case AOO.

For EQ safety-related digital computer-based I&C equipment in mild environments, the temperature qualification method is by test. The maximum qualification temperature is 10°C higher than the maximum temperature to which the equipment is exposed for the worst-case AOO, while the equipment is under its maximum loading, to comply with margin requirements. The minimum qualification temperature is 10°C lower than the minimum temperature to which the equipment is exposed for the worst-case AOO.

Since HVAC is nonsafety-related, AOO, including Station Blackout (SBO), and DBA conditions assume no HVAC cooling and assume worst case highest ambient temperature caused by lack of HVAC, maximum outside temperature, and maximum heat rise from collocated and adjacent heat sources. Additionally, AOO, including SBO, conditions assume no HVAC heating and assume worst case lowest ambient temperature caused by lack of HVAC heating, minimum outside temperature, and minimum heat rise from collocated and adjacent heat sources.

This ensures that EQ equipment is qualified for the worst-case temperatures with margin per the requirements of IEEE 323-1974 for Harsh Environment and IEEE 323-2003 for Mild Environment.

Pressure

The pressure qualification for EQ equipment in harsh environments is by test. The qualified pressure is 10% higher, to comply with margin requirements, than the maximum pressure to which the equipment is exposed for the worst-case DBA, while the equipment is under its maximum loading.

For EQ safety-related electrical equipment in mild environments, including digital computer-based I&C, the pressure qualification methods are by test or analysis. The qualified pressure is 10% higher, to comply with margin requirements, than the maximum pressure to which the equipment is exposed for the worst-case AOO, while the equipment is under its maximum loading.

This ensures that EQ equipment is qualified for the worst-case pressure with margin per the requirements of IEEE 323-1974 for Harsh Environment and IEEE 323-2003 for Mild Environment.

Humidity

Relative humidity requirements are defined for DBAs and AOOs, and EQ equipment is qualified for the applicable relative humidity conditions. The qualification for steam exposure for safety-related equipment in harsh environments is by test. The qualified steam conditions are those identified in the DBA analysis.

For EQ safety-related electrical equipment in mild environments, including digital computer-based I&C, the qualification methods for humidity are by test or analysis.

This ensures that EQ equipment is qualified for the worst-case humidity conditions per the requirements of IEEE 323-1974 for Harsh Environment and IEEE 323-2003 for Mild Environment.

Chemical effects

The assumed composition of chemicals is at least as severe as that resulting from the most limiting mode of plant operation (e.g., normal operation and borated water from the SLC system). There is no caustic containment spray in an ESBWR. The qualification for chemical exposure for EQ equipment in harsh environments is by test.

This chemical exposure test ensures that EQ equipment is qualified for the worst-case chemical conditions per the requirements of IEEE 323-1974. The EQ safety-related electrical equipment, including digital computer-based I&C, in mild environments is not exposed to chemicals.

Radiation

The radiation environment is based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe DBA during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near recirculating lines and including dose rate effects. Radiation exposure simulates radiation degradation for the total integrated dose applicable for the normal radiation dose. Accident dose may be added to the normal dose and a single radiation total dose applied by test. Equipment that has accident dose rate sensitivity is tested at the most degrading dose rate. EQ equipment is qualified for radiation. The qualification for radiation for EQ equipment in harsh environments is by test. The qualification radiation total integrated accident dose is 10% higher, to comply with margin requirements, than the maximum accident total integrated dose to which the equipment is exposed for the worst-case DBA.

EQ equipment that could be exposed to radiation is environmentally qualified to a radiation dose that simulates the calculated radiation environment (normal and accident) that the equipment should withstand prior to completion of its required safety-related functions. Such qualification considers that equipment damage is a function of total integrated dose and can be influenced by dose rate, energy spectrum, and particle type. The radiation qualification includes doses from all potential radiation sources at the equipment location. Specific plant design features to maintain radiation exposure to equipment less than the equipment qualification levels inside the containment during normal operations will be evaluated during the detailed design process. These analyses will evaluate each specific equipment location and determine design features needed to maintain integrated doses less than the qualification criteria for electronic equipment. If the integrated dose exceeds the equipment qualification values after the detailed calculations, shielding or other methods (e.g., equipment replacement program) to reduce the dose will be incorporated during the detailed design. Plant-specific analysis is used to justify any reductions in dose or dose rate resulting from component location or shielding. The foregoing defines how the qualification environment at the equipment location is established.

Shielded components are qualified only to the gamma radiation environment provided it can be demonstrated that the sensitive portions of the component or equipment are not exposed to significant beta radiation dose rates or that the effects of beta radiation, including heating and secondary radiation, have no deleterious effects on component performance. If, after considering the appropriate shielding factors, the total beta radiation dose contribution to the equipment or component is calculated to be less than 10% of the total gamma radiation dose to which the equipment or component has been qualified, the equipment or component is considered qualified for the beta and gamma radiation environment, based only on gamma radiation testing.

EQ equipment located outside containment that is exposed to the radiation from a recirculating fluid is qualified to withstand the radiation penetrating the containment plus the radiation from the recirculating fluid.

For EQ safety-related electrical equipment in mild environments, including digital computer-based I&C, the qualification methods for radiation are by test or analysis. A mild radiation environment for electronic equipment is a total integrated dose less than 10 Gy (1.0E3 rad), and a mild radiation environment for other equipment is less than 100 Gy (1.0E4 rad). Safety-related electronic and electrical equipment is tested with the equipment energized and performing its safety-related function, if the required total integrated dose exceeds the mild environment level.

This ensures EQ equipment is qualified for the worst-case radiation with DBA margin per the requirements of IEEE 323-1974 for Harsh Environment and IEEE 323-2003 for Mild Environment.

Operating Time

EQ equipment is qualified for its required operating time during DBA and post-accident conditions. Some mechanical and electrical equipment may be required to perform an intended function from within minutes of the occurrence of an event up to 10 hours into the event. Such equipment is shown to remain functional in the accident environment for a period of at least one hour in excess of the time assumed in the accident analysis unless a time margin of less than one hour can be justified. Such justification includes 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-related function is not detrimental to plant safety nor misleads the operator.
4. Determination that the margin applied to the minimum operability time, when combined with other test margins, accounts 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.

For EQ equipment with a required time of operation during accident and post-accident conditions of more than 10 hours, testing demonstrates that the EQ equipment remains functional under such conditions for a period of time at least 10% longer than the required time of operation.

Aging

EQ equipment in harsh environments is analyzed for significant aging mechanisms. If the equipment is determined to have a significant aging mechanism, then the mechanism is accounted for in the qualification program. Aging mechanisms include time-temperature degradation, cycle aging and normal radiation exposure. Artificial aging or natural aging simulate time-temperature degradation. Artificial aging is determined from the Arrhenius Equation. Cycle aging conservatively simulates the degradation during the required operating cycles. Use of synergistic effects is considered when these effects are believed to have a significant effect on equipment performance.

Age conditioning is not required for EQ equipment without significant aging mechanisms, or for EQ safety-related equipment in mild environments.

Equipment is reviewed in terms of design, function, materials, and environment for its specified application to identify potentially significant aging mechanisms. An aging mechanism is significant if subsequent to manufacture, while in storage, and/or in the normal and abnormal service environment, it results in degradation of the equipment that progressively and appreciably renders the equipment vulnerable to failure to perform its safety-related function(s) under DBE conditions.

Artificial accelerated aging simulates the significant aging mechanisms and a qualified life is established. For materials with a qualified life of less than 60 years, maintenance requirements are established to ensure that the material is replaced prior to the end of its qualified life. Alternatively, materials with a qualified life of less than 60 years may be evaluated with condition monitoring to ensure that the material degradation is less than the degradation, which was simulated in type tests, prior to the simulated DBA conditions.

Submergence

EQ equipment that is submerged during or after a design basis event is tested for the resulting worst-case submergence.

Synergistic effects

The age conditioning considers sequential, simultaneous, and synergistic effects in order to achieve the worst state of degradation. Synergistic effects are considered when they have a significant effect on equipment performance.

Electromagnetic interference (EMI)/radio frequency interference (RFI) and Voltage Surges

EQ equipment is qualified for EMI/RFI and voltage surge protection against the following:

- EMI

- RFI
- Electrostatic Discharge
- Electrical Surge

EMI qualifications follow the requirements defined in EPRI TR-102323 –1994 (as approved by NRC Safety Evaluation Report dated April 16 1996)or MIL-STD 461E or IEC 61000-4 series. The qualification for EMI/RFI and voltage surges for EQ equipment in harsh and mild environments is by test, consistent with RG 1.180. Nonsafety-related electrical and digital computer-based I&C equipment is tested for conducted emission via power leads and radiated emission from electric fields to ensure that emissions from nonsafety-related electrical and I&C equipment do not exceed allowable limits and do not affect the EQ equipment.

This ensures that safety-related equipment is qualified for EMI/RFI and voltage surges per the requirements of IEEE 323-2003.

3.11.3.2 **Environmental Requirements**

Environmental conditions for the zones where EQ equipment is located are calculated for normal, AOO, test, accident and post-accident conditions and are documented in Appendix 3H, Equipment Qualification Environmental Design Criteria. Environmental conditions are tabulated by zones contained in the referenced building arrangements. Typical equipment in the noted zones is shown in the referenced system design schematics.

Environmental parameters include thermodynamic parameters (temperature, pressure and relative humidity), radiation parameters (radiation type, dose rates and total integrated dose) and chemical spray parameters (chemical composition and the resulting pH).

AOO and test condition environments are bounded by the normal or accident conditions according to the Appendix 3H tables.

Margins are 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 3H tables do not include margins.

The environmental conditions shown in the Appendix 3H tables are upper-bound envelopes used to establish the environmental design and qualification bases for equipment. The upper bound envelopes indicate that the zone data reflect the worse case expected environment produced by a compendium of accident conditions.

3.11.4 **Qualification Program, Methods and Documentation**

3.11.4.1 **Harsh Environment Qualification**

Some EQ equipment is located in a harsh environment. All three categories of 10 CFR 50.49(b) electrical equipment that are located in a harsh environment are qualified by test or other methods

as described in IEEE-323-1974 and permitted by 10 CFR 50.49(f) (Reference 3.11-2). Equipment type test is the preferred method of qualification.

A type test subjects a representative sample of equipment, including interfaces, to a series of tests, simulating the effects of significant aging mechanisms during normal operation. The sample is subsequently subjected to DBA testing that simulates and thereby establishes the tested configuration for installed equipment service, including mounting, orientation, interfaces, conduit sealing, and expected environments. A type test demonstrates that the equipment performs the intended safety-related function(s) for the required operating time before, during, and/or following the DBA, as appropriate.

Performance data from equipment of similar design that has successfully operated under known service conditions may be used in qualifying other equipment to equal or less severe conditions. Applicability of this data depends on the adequacy of documentation establishing past service conditions, equipment performance, and similarity to the equipment to be qualified and upon which operating experience exists. A demonstration of required operability during applicable DBA(s) is included in equipment qualification programs based on operating experience, when harsh environment design basis event qualification is required.

Qualification by analysis requires a logical assessment or a valid mathematical model of the equipment to be qualified. The bases for analysis typically include physical laws of nature, results of test data, operating experience, and condition indicators. Analysis of data and tests for material properties, equipment rating, and environmental tolerance can be used to demonstrate qualification. However, analysis alone is not used to demonstrate the initial qualification for safety-related electrical equipment in a harsh environment.

EQ safety-related mechanical equipment qualified by analysis is consistent with ASME B&PV Code Section III, “Rules for Construction of Nuclear Facility Components.”

Active EQ safety-related mechanical equipment is qualified by the qualification methods of IEEE 323-1974.

EQ equipment located in harsh environments may be qualified by combinations of type test, operating experience, and analysis. For example, if a type test of a complete assembly is not possible, component testing supplemented by analysis may be used.

The ESBWR equipment qualification program meets the guidance of RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.” RG 1.89 endorses IEEE 323-1974. EQ equipment is qualified using the qualification methods of IEEE 323-1974.

The effects of chemical spray must be addressed. Containment spray, emergency core cooling initiation, and recirculation system operation are included in the qualification tests. The ESBWR SLC system injects borated water into the Reactor Pressure Vessel (RPV) during DBA LOCA.

Containment spray is not caustic; therefore the effect of the water spray is included in the equipment qualification.

The equipment qualification program includes safety-related mechanical equipment in harsh environment areas and verifies that they are designed to be compatible with postulated environmental conditions, including those associated with LOCA. Active safety-related mechanical equipment is qualified using test, analysis, or a combination of test and analysis.

In some instances, mechanical equipment loading under normal service is more severe than loading under DBA. The loading under normal service is documented with test and/or analysis. The loading and capability under DBA conditions is analyzed in the equipment qualification process to establish the suitability of materials, parts, and equipment needed for safety-related functions, and to verify that the design of such materials, parts, and equipment is adequate. The qualification of mechanical equipment includes materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms), required operating time, non-metallic subcomponents of such equipment; the environmental conditions and process parameters for which this equipment must be qualified; non-metallic material capabilities; and the evaluation of environmental effects.

The EQ equipment in a harsh environment has a maximum qualified life of 60 years. The qualified life is verified using methods and procedures of qualification as stated in IEEE-323-1974 and as addressed herein.

The duty cycle of safety-related batteries in ESBWR is different from the duty cycle basis in IEEE-535-1986. Safety-related batteries are qualified to meet IEEE-535-1986, with the exception that the duty cycle is 72 hours and supplemental discharge cycle testing is required to meet the harsh environment qualification process of IEEE-323-1974.

ESBWR's equipment qualification type test process for batteries includes evaluation of significant aging mechanisms that are related to failure mechanisms from radiation exposure, time-temperature aging, and cycle aging; age testing for significant aging mechanisms for a 20-year qualified life; seismic test; and performance testing for the 72-hour duty cycle (see Reference 3.11-6).

3.11.4.2 **Mild Environment Qualification**

EQ safety-related equipment located in a mild environment is qualified per guidelines of Regulatory Guide 1.209 – 2007 and IEEE 323-2003 for qualification of Safety Related or Important to Safety digital I&C to be installed in a mild environment.

For EMI/RFI qualification, as endorsed in Regulatory Guide 1.180, Rev 1, qualification methods shall be per EPRI TR-102323 –1994 (as approved by NRC Safety Evaluation Report dated April 16 1996) or MIL-STD- 461E or IEC 61000-4 series.

To assure EQ safety-related equipment located in a mild environment meets its safety-related functional requirements during normal environmental conditions and AOOs, the environmental design basis for normal environmental conditions and AOO requirements is specified in the design/purchase specifications. A qualified life is not required for equipment located in a mild environment that has no significant aging mechanisms.

For all EQ safety-related equipment, excluding EQ safety-related digital computer-based I&C systems, a Certificate of Conformance from the vendor of the safety-related equipment to be located in a mild environment needs to certify performance to the environmental design basis for normal environmental conditions and AOO requirements for the equipment location for the time that the safety-related function is required.

3.11.4.3 **Computer-based Instrumentation and Control Systems**

EQ safety-related digital computer-based I&C systems comply with RG 1.209 and RG 1.180. For all EQ safety-related digital computer-based I&C systems, located in a mild environment, type testing is the preferred qualification method to demonstrate performance to the environmental design basis for normal environmental conditions and AOO requirements for the equipment location for the time that the safety-related function is required.

Type tests may be separate laboratory or manufacturer's tests that document performance to the applicable service conditions with due consideration for synergistic effects, if applicable.

When digital computer-based I&C systems type testing is performed:

- The system under test functions and performs with safety-related software that has been validated and verified and is representative of the software to be installed in the nuclear power plant.
- Testing demonstrates performance of safety-related functions at the specified environmental service conditions, including AOOs.
- Testing exercises all portions of the system under test that are necessary to accomplish the safety-related functions and those portions whose operation or failure could impair the safety-related functions.
- Testing confirms the response of digital interfaces and verifies that the design accommodates the potential impact of environmental effects on the overall response of the system.
- Testing of representative sample of the equipment as a complete system contained within its cabinet or enclosure is preferred.
- When testing of a complete system is not practical, confirmation of the dynamic response to the most limiting environmental and operational conditions is based on type testing of the individual modules and analysis of the cumulative effects of environmental and operational stress on the entire system to demonstrate required safety-related performance.

In addition to Type Testing, analysis may be utilized per 10CFR50.49 to support digital I&C qualification in a mild environment via;

1. Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.
2. Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.
3. Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.
4. Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

The evidence of qualification in a mild environment is consistent with the guidance given in IEEE 323-2003 Section 7.2.

3.11.4.4 **Environmental Qualification Documentation**

The procedures and results of qualification by tests, analyses or other methods are documented, maintained, and reported in accordance with requirements of 10 CFR 50.49(j), RG 1.209, and IEEE 323-2003 Section 7.2. The EQD summarizes the qualification results for all equipment identified in Subsection 3.11.2. The EQD is developed during program implementation and includes the following:

- The environmental parameters and the methodology used to qualify the equipment for harsh and mild environments.
- The System Component Evaluation Work sheets which include a summary of environmental conditions and qualified conditions.

The compliance with the applicable portions of the GDC of 10 CFR 50, Appendix A, and the Quality Assurance Criteria of 10 CFR 50, Appendix B are described in the NRC approved Licensing Topical Report on GE's environmental qualification program (Reference 3.11-3).

The documentation necessary to support the continued qualification of the equipment installed in the plant that is within the Environmental Qualification (EQ) Program scope is available in accordance with 10 CFR 50 Appendix A, General Design Criterion 1. EQ files are maintained for equipment and certain post-accident monitoring devices that are subject to a harsh environment. The files are maintained for the operational life of the plant.

Central to the EQ Program is the EQ Master Equipment List (EQMEL). The EQMEL identifies the electrical and mechanical equipment or components that must be environmentally qualified for use in a harsh environment. The EQMEL consists of equipment that is essential to emergency reactor shutdown, containment isolation, reactor core cooling, or containment and reactor heat removal, or that is otherwise essential in preventing a significant release of radioactive material to the

environment. This list is developed from the equipment list provided in Table 3.11-1. The EQMEL and a summary of equipment qualification results are maintained as part of the equipment qualification file for the operational life of the plant.

Administrative programs are in place to control revisions to the EQ files and the EQMEL. When adding or modifying components in the EQ Program, EQ files are generated or revised to support qualification. The EQMEL is revised to reflect these new components. To delete a component from the EQ Program requires a deletion justification to be prepared that demonstrates why the component can be deleted. This justification consists of an analysis of the component, an associated circuit review, if appropriate and a safety evaluation. The justification is released and/or referenced on the appropriate change document.

For changes to the EQMEL, supporting documentation is completed and approved prior to issuing the changes. This documentation includes safety reviews and new or revised EQ files. Plant modifications and design basis changes are subject to change process reviews, e.g., reviews in accordance with 10 CFR 50.59 or the change control requirements of the ESBWR-specific appendix to 10 CFR Part 52, in accordance with appropriate plant procedures. These reviews address EQ issues associated with the activity. Any changes to the EQMEL that are not the result of a modification or design basis change are subject to a separate review that is accomplished and documented in accordance with plant procedures.

Engineering change documents or maintenance documents generated to document work performed on an EQ component are reviewed against the current revision of the EQ files for potential impact. Changes to EQ documentation may be due to, but not limited to, plant modifications, calculations, corrective maintenance, or other EQ concerns.

The operational aspects of the EQ program include:

- Evaluation of EQ results for design life to establish activities to support continued EQ.
- Determination of surveillance and preventive maintenance activities based on EQ results.
- Consideration of EQ maintenance recommendations from equipment vendors.
- Evaluation of operating experience in developing surveillance and preventive maintenance activities for specific equipment.
- Development of plant procedures that specify individual equipment identification, appropriate references, installation requirements, surveillance and maintenance requirements, post-maintenance testing requirements, condition monitoring requirements replacement part identification, and applicable design changes and modifications.
- Development of plant procedures for reviewing equipment performance and EQ operational activities, and for trending the results to incorporate lessons learned through appropriate modifications to the operational EQ program.
- Development of plant procedures for the control and maintenance of EQ records.

Implementation of the environmental qualification program, including development of the plant specific Environmental Qualification Document (EQD), will be in accordance with the milestone defined in Section 13.4.

3.11.5 Loss of Heating, Ventilating and Air Conditioning

Sections 6.4 and 9.4 describe the HVAC systems including their design evaluations. The loss of ventilation conditions are considered in Appendix 3H and the calculations are based on maximum heat loads assuming operation of all operable equipment regardless of safety-related classification.

3.11.6 Estimated Chemical and Radiation Environment

Chemical Environment

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 operating and accident conditions is reported in Appendix 3H.

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.

Radiation Environment

EQ equipment is designed to perform its safety-related function when exposed to the normal operational radiation levels and accident radiation levels.

Dose rates and integrated doses of radiation that are associated with normal plant operation and the DBA condition for various plant compartments are presented in Appendix 3H; these parameters are presented in terms of time-based profiles where applicable.

3.11.7 COL Information

3.11-1-A Environmental Qualification Document

This COL item is addressed in Subsection 3.11.4.4.

3.11-2-H Environmental Qualification Records (Deleted)

3.11.8 References

3.11-1 USNRC, Standard Review Plan, NUREG-0800, SRP 3.11, Revision 3, March 2007, “Environmental Qualification of Mechanical and Electrical Equipment.”

3.11-2 USNRC, Code of Federal Regulations, Title 10, Chapter I, Part 50, Paragraph 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

3.11-3 General Electric Co., “General Electric Environmental Qualification Program,” NEDE-24326-1-P, Proprietary Document, January 1983.

3.11-4 Regulatory Guide 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants,” March 2007.

3.11-5 NUREG 0588, USNRC, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” December 1979.

3.11-6 *GE Hitachi Nuclear Energy, “ESBWR Qualification Plan Requirements for a 72-Hour Duty Cycle Battery,” NEDE-33516P-A Revision 2, Proprietary Document, September 2010.*

Table 3.11-1 Electrical and Mechanical Equipment for Environmental Qualification (Sheet 1 of 8)

Components	Quantity	Location (note 1)	Function (note 2)	Required Operation Time (note 3)	Qualification Program (note 4)
B21 Nuclear Boiler System					
Depressurization Valves	8	CV	ESF	72 hr	MH
Safety Relief Valves	10	CV	ESF	72 hr	MH
Temperature element in DPV/SRV Discharge	18	CV	ESF	72 hr	EH
MSIV - Inboard	4	CV	PB	100 Days	MH
MSIV - Outboard	4	ST	PB	100 Days	MH
MSIV Drain Bypass Valve	2	ST	ESF	72 hr	MH
Steam Line Lowpoint Drain Bypass Valve	1	TB	ESF	72 hr	MH
Feedwater isolation valve	8	ST, CV	PB	100 Days	MH
RPV Level Transmitters	All	RB	ESF	100 Days	EH
RPV Temperature Elements	All	CV	ESF	100 Days	EH
RPV Pressure Transmitter	All	RB	ESF	100 Days	EH
Feed Piping Diff Pressure Transmitter	All	RB	ISOL	100 Days	EH
Steam Line Flow Transmitter	All	RB	ISOL	100 Days	EH
Electrical Modules and Cable	All	CV, RB, ST, TB	ESF	100 Days	EH
B32 Isolation Condenser System					
Isolation Valves	16	CV	PB	100 Days	MH
Isolation Valves Operator	16	CV	ESF	100 Days	MH
Condensate Return Valves	4	CV	ESF	100 Days	MH
Condensate Return Valves Operator	4	CV	ESF	100 Days	MH
Condensate Return Bypass Valve	4	CV	ESF	100 Days	MH
Condensate Return Bypass Valve Operator	4	CV	ESF	100 Days	MH
Upper Header Vent Valve	8	CV	ESF	100 Days	MH
Upper Header Vent Valve Actuator	8	CV	ESF	100 Days	MH
Lower Header Vent Valve	16	CV	ESF	100 Days	MH
Lower Header Vent Valve Actuator	12	CV	ESF	100 Days	MH
Pool Cross-Connect Valves	4	RB	ESF	100 Days	MH
Vent Line Temperature Element	8	CV	ESF	100 Days	EH

Table 3.11-1 Electrical and Mechanical Equipment for Environmental Qualification (Sheet 2 of 8)

Components	Quantity	Location (note 1)	Function (note 2)	Required Operation Time (note 3)	Qualification Program (note 4)
Condensate Drain Temperature Element	4	CV	ESF	100 Days	EH
Steam Piping Diff Pressure Transmitter	8	CV	ESF	100 Days	EH
Condensate Drain Diff Pressure Transmitter	8	CV	ESF	100 Days	EH
Electrical Modules and Cable	All	CV, RB	ESF	100 Days	EH
C11 Rod Control and Information System					
Electrical Modules and Cable	All	CB, RB	ESF	72 hr	EH
C12 Control Rod Drive System					
HCU Scram Solenoid Pilot Valve	135	RB	ESF	72 hr	MH
FMCRD Passive Holding Brake	269	CV	ESF	72 hr	MH
FMCRD Separation Switch	538	CV	ESF	72 hr	EH
Charging Water Header Pressure Transmitter	4	RB	ESF	72 hr	EH
Electrical Modules and Cable	All	CV, RB	ESF	72 hr	EH
High Pressure CRD Makeup Line Isolation Valves	2	RB	ESF	72 hr	MH
Backup Scram Valve Solenoids	2	RB	ESF	72 hr	EH
C21 Leak Detection and Isolation System					
Pressure Transmitters	All	CV, RB, CB	ESF	100 Days	EH
Temperature Sensors	All	CV, RB, CB	ESF	100 Days	EH
Electrical Modules and Cable	All	CV, RB, CB	ESF	100 Days	EH
C31 Feedwater Control System					
Electric Modules and Cable	All	CB, RB	ESF	72 hr	EH
C51 Neutron Monitoring System					
Detector and Tube Assembly	All	CV	ESF	72 hr	MH
Electrical Modules and Cable	All	CV, RB, CB	ESF	100 Days	EH
C61 Remote Shutdown System					
Electrical Panels, Modules and Cable	All	RB	ESF	100 Days	C
C63 Safety-Related Distributed Control and Information System (DCIS)					
Electrical Modules and Cable	All	RB, CB	ESF	100 Days	C
C71 Reactor Protection System					
Electrical Modules and Cable	All	CB, RB	ESF	100 Days	EH

Table 3.11-1 Electrical and Mechanical Equipment for Environmental Qualification (Sheet 3 of 8)

Components	Quantity	Location (note 1)	Function (note 2)	Required Operation Time (note 3)	Qualification Program (note 4)
C72 Diverse Protection System					
Electrical Modules and Cable	All	CB, RB, TB	ESF, ISOL	100 Days	EH
C74 Safety System Logic and Control					
Electrical Modules and Cable	All	CB, RB	ESF	100 Days	EH
C41 Standby Liquid Control System					
Isolation Check Valves	4	CV, RB	PB	100 days	MH
Squib Injection Valves	4	RB	ESF	72 hr	MH
Injection Shut-Off Valves Actuator	4	RB	ESF	100 Days	EH
Nitrogen Charging Globe Valve	2	RB	ESF	100 Days	MH
Nitrogen Charging Globe Valve Actuator	2	RB	ESF	100 Days	EH
Nitrogen Charging Check Valve	2	RB	ESF	72 hr	MH
Accumulator Depressurization Valves	4	RB	ESF	100 Days	MH
Accumulator Depressurization Valves Actuator	4	RB	ESF	100 Days	EH
Accumulator Relief Valve	2	RB	ESF	72 hr	MH
Injection Shut Off Valves	4	RB	ESF	100 Days	MH
Accumulator Level Instrumentation	8	RB	ESF	100 Days	EH
Accumulator Pressure Instrumentation	8	RB	ESF	100 Days	EH
Electrical Modules and Cable	All	CV, RB	ESF	100 Days	EH
D11 Process Radiation Monitoring System					
Isolation Valves	4	CV, RB, CB	ESF	100 Days	MH
Radiation Monitors, Sensors, Electrical Modules and Cable	All	CV, RB, CB	ESF	100 Days	EH
E50 Gravity-Driven Cooling System (GDCS)					
GDCS Pool Level Instrumentation	12	CV	ESF	100 Days	EH
GDCS Squib Valve to GDCS Pool	8	CV	ESF	72 hr	MH
GDCS Check Valve to GDCS Pool	8	CV	ESF	72 hr	MH
GDCS Squib Valve to Suppression Pool	4	CV	ESF	72 hr	MH
GDCS Check Valve to Suppression Pool	4	CV	ESF	72 hr	MH
GDCS Squib Valve to Lower Drywell (DW)	12	CV	ESF	72 hr	MH

Table 3.11-1 Electrical and Mechanical Equipment for Environmental Qualification (Sheet 4 of 8)

Components	Quantity	Location (note 1)	Function (note 2)	Required Operation Time (note 3)	Qualification Program (note 4)
Electrical Modules and Cable	All	CV, RB, CB	ESF	100 Days	EH
G21 Fuel and Auxiliary Pools Cooling System					
Containment Isolation Valve (CIV) - Drywell Spray - Outboard	1	RB	PB	100 Days	MH
CIV - Drywell Spray - Inboard	1	CV	PB	100 Days	MH
CIV – Suppression Pool Cooling (SPC) Suction - Outboard	4	RB	PB	100 Days	MH
CIV - SPC Return - Outboard	2	RB	PB	100 Days	MH
CIV - SPC Return - Inboard	2	CV	PB	100 Days	MH
CIV - GDCS Suction - Outboard	1	RB	PB	100 Days	MH
CIV - GDCS Suction - Inboard	1	CV	PB	100 Days	MH
CIV - GDCS Return - Outboard	1	RB	PB	100 Days	MH
CIV - GDCS Return - Inboard	1	CV	PB	100 Days	MH
LPCI Isolation	4	FB, RB	PB	100 Days	MH
IC/PCCS Pool Level Instrumentation	All	RB	ESF	100 Days	EH
Fuel Pool Level Instruments	2	FB	ESF	100 Days	EH
Electrical Modules and Cable	All	CV, FB, RB, CB	ESF	100 Days	EH
G31 Reactor Water Cleanup/Shutdown Cooling System					
CIV - Mid Vessel - Inboard	2	CV	PB, ISOL	100 Days	MH
CIV - Mid Vessel - Outboard	2	RB	PB, ISOL	100 Days	MH
CIV - Mid Vessel - Inboard Operator	2	CV	ISOL	72 hr	EH
CIV - Mid Vessel - Outboard Operator	2	RB	ISOL	72 hr	EH
CIV - Bottom Drain Inboard	2	CV	PB, ISOL	100 Days	MH
CIV - Bottom Drain Outboard	2	RB	PB, ISOL	100 Days	MH
CIV - Bottom Drain Inboard Operator	2	CV	ISOL	72 hr	EH
CIV - Bottom Drain Outboard Operator	2	RB	ISOL	72 hr	EH
CIV - Process Sampling Line -Inboard	2	CV	PB, PAMS	100 Days	MH
CIV - Process Sampling Line -Outboard	2	RB	PB, PAMS	100 Days	MH

Table 3.11-1 Electrical and Mechanical Equipment for Environmental Qualification (Sheet 5 of 8)

Components	Quantity	Location (note 1)	Function (note 2)	Required Operation Time (note 3)	Qualification Program (note 4)
CIV - Process Sampling Line -Inboard Operator	2	CV	ISOL, PAM	100 Days	EH
CIV - Process Sampling Line -Outboard Operator	2	RB	ISOL, PAM	100 Days	EH
Return Line Shutoff Valve	2	RB	ISOL	100 Days	MH
Check Valve to Feedwater	4	RB	ISOL	100 Days	MH
Mid-vessel Flow Instrumentation	All	CV	ISOL	100 Days	EH
Mid-vessel Temperature Instrumentation	All	CV	ISOL	100 Days	EH
Bottom Drain Flow Instrumentation	All	CV	ISOL	100 Days	EH
Bottom Drain Temperature Instrumentation	All	CV	ISOL	100 Days	EH
Return Line Flow Instrumentation	All	RB	ISOL	100 Days	EH
Return Line Temperature Instrumentation	All	RB	ISOL	100 Days	EH
Overboard Flow Instrumentation	All	RB	ISOL	100 Days	EH
Overboard Temperature Instrumentation	All	RB	ISOL	100 Days	EH
Electrical Modules and Cables	All	CV, RB	ESF	100 Days	EH
H11 Main Control Room (MCR) Panels					
Panels, Modules and Cables	All	CB	ESF	100 Days	C
H12 MCR Back Room Panels					
Panels, Mod□les and Cable	All	CB	ESF	100 Days	C
H21 Local Panels and Racks					
Panels, Modules and Cable	All	ALL	ESF	100 Days	EH
N21 Condensate and Feedwater System					
Feed Line Temperature Element	All	ST	ESF	100 Days	EH
Feed Piping Diff Pressure -Transmitter	All	ST	ISOL	100 Days	EH
Electrical Modules and Cable	All	ST, CB	ESF	100 Days	EH
P10 Makeup Water System					
Isolation Valves	All	CV, RB	ISOL	100 Days	MH
P25 Chilled Water System					
Isolation Valves	All	CV, RB	ISOL	100 Days	MH
P51 Service Air System					

Table 3.11-1 Electrical and Mechanical Equipment for Environmental Qualification (Sheet 6 of 8)

Components	Quantity	Location (note 1)	Function (note 2)	Required Operation Time (note 3)	Qualification Program (note 4)
Isolation Valves	All	CV, RB	ISOL	100 Days	MH
P54 High Pressure Nitrogen Supply System					
Isolation Valves	4	CV, RB	ISOL	100 Days	MH
R10 Electrical Power Distribution System (EPDS)					
Cable and Supports	All	CB, FB, RB	ESF	100 Days	EH
R13 Uninterruptible AC Power Supply					
Electrical Modules and Cable	All	CV, CB, RB	ESF	100 Days	EH
R16 Direct Current Power Supply					
Divisional 250 VDC Battery	8	RB	ESF	100 Days	E
Divisional 250 VDC Normal/Standby Battery Charger	12	RB	ESF	100 Days	E
Divisional 250 VDC Power Center	8	RB	ESF	100 Days	E
Divisional 250 VDC Transfer Switch Box	8	RB	ESF	100 Days	E
Isolation Power Center Normal Main Circuit Breaker	4	RB	ISOL	100 Days	E
Isolation Power Center Alternate Main Circuit Breaker	4	RB	ISOL	100 Days	E
Isolation Power Center Supply Breaker to Division 250 VDC Normal Battery Charger	12	RB	ISOL	100 Days	E
Electrical Modules and Cable	All	CV, CB, RB, TB	ESF	100 Days	E
R31 Raceway System					
Electrical Penetrations	All	CV	PB	100 Days	EH
Conduit, Cable Trays and Supports	All	CV, CB, RB, TB, FB	ESF	100 Days	EH
T10 Containment System					
Vacuum Breakers	3	CV	ESF	100 Days	MH
Vacuum Breaker Isolation Valves	3	CV	ESF	72 hr	MH
Instrumentation and Cables	All	CV	ESF	100 Days	EH
Basemat Internal Melt Arrest Coolability (BiMAC) Temperature Element	ALL	CV	ESF	100 Days	EH
BiMAC Temperature Switch	ALL	CV	ESF	100 Days	EH
T15 Passive Containment Cooling System					

Table 3.11-1 Electrical and Mechanical Equipment for Environmental Qualification (Sheet 7 of 8)

Components	Quantity	Location (note 1)	Function (note 2)	Required Operation Time (note 3)	Qualification Program (note 4)
Vent Fan Isolation Valves	6	CV	ESF	100 Days	MH
Passive Containment Cooling System (PCCS) Vent Fan	6	CV	ESF	100 Days	EH
Vent Line Catalyst Module	12	CV	ESF	100 Days	MH
T31 Containment Inerting System					
Isolation Valve	10	CV, RB	ISOL	100 Days	MH
Electrical Modules and Cable	All	CB, RB	ESF	100 Days	EH
T49 Passive Autocatalytic Recombiner System					
Passive Autocatalytic Recombiners	ALL	CV	ESF	100 Days	MH
T62 Containment Monitoring System					
Containment Isolation Valves	All	CV, RB	ISOL	100 Days	MH
Electrical Modules and Cable	All	CB, CV, RB	ESF	100 Days	EH
Drywell Pressure Transmitters	All	RB	ESF	100 days	EH
Differential Pressure Transmitters	All	RB	ESF	100 days	EH
Suppression Pool Temperature Element	All	CV	ESF	100 days	EH
Lower DW Level Transmitter	All	RB	ESF, PAMS	100 days	EH
Suppression Pool Level Transmitters	All	RB	PAMS	100 days	EH
Suppression Pool Pressure Transmitters	All	RB	PAMS	100 days	EH
Hydrogen Analyzers	All	RB	ESF, PAMS	100 days	EH
Oxygen Analyzers	All	RB	ESF, PAMS	100 days	EH
U40 Reactor Building HVAC					
Building Isolation Dampers	All	RB	ESF	100 Days	EH
Electrical Modules and Cable	All	RB	ESF	100 Days	EH
U77 Control Building HVAC					
Control Room Habitability Area (CRHA) Supply Air Isolation Dampers	All	CB	ESF	100 Days	E
Emergency Filter Unit (EFU) Downstream Isolation Dampers	All	CB	ESF	100 Days	E
CRHA Restroom Exhaust Isolation Dampers	All	CB	ESF	100 Days	E

Table 3.11-1 Electrical and Mechanical Equipment for Environmental Qualification (Sheet 8 of 8)

Components	Quantity	Location (note 1)	Function (note 2)	Required Operation Time (note 3)	Qualification Program (note 4)
CRHA Smoke Purge Intake Isolation Dampers	All	CB	ESF	100 Days	E
CRHA Smoke Purge Exhaust Isolation Dampers	All	CB	ESF	100 Days	E
Emergency Filter Unit (EFU)	All	CB	ESF	100 Days	E
Electrical Modules and Cable	All	CB	ESF	100 Days	E
U98 Fuel Building HVAC					
Fuel Building General Area HVAC Subsystem (FBGAVS) Building Supply Air Isolation Dampers	All	FB	ESF	100 Days	EH
FBGAVS Building Exhaust Air Isolation Dampers	All	FB	ESF	100 Days	EH
Fuel Building Fuel Pool Area HVAC Subsystem (FBFPVS) Building Supply Air Isolation Dampers	All	FB	ESF	100 Days	EH
FBFPVS Building Exhaust Air Isolation Dampers	All	FB	ESF	100 Days	EH
Electrical Modules and Cable	All	FB	ESF	100 Days	EH

Note 1: CV – Containment Vessel
ST – Steam Tunnel
RB – Reactor Building
FB – Fuel Building
CB – Control Building
TB – Turbine Building
OO – Outdoors Onsite

When multiple locations are listed, information in this table applies to equipment in all locations listed that also meets the other criteria shown.

Note 2: ESF – Engineered Safety Feature
PAMS – Post Accident Monitoring
ISOL – Containment Isolation
PB – Primary Pressure Boundary

When multiple functions are listed, information in this table applies to equipment associated with either function that also meets the other criteria shown.

Note 3: Required operation time refers to the period of time which the equipment must remain available or operational. Required operation times apply to equipment when all criteria shown in the first four columns of the table are met.

Note 4: E – Electrical Equipment Program
M – Mechanical Equipment Program
C – Computer Based I&C System Program
H – Harsh Environment (omission of H indicates Mild Environment)

Qualification program classifications apply to equipment when all criteria shown in the first four columns of the table are met.

Note 5: Valve operators/actuators are considered to be part of the valve assembly and are generally not listed separately in this table.

3.12 Piping Design Review

Information on seismic Category I and II, and nonseismic piping analysis and their associated supports is presented in Sections 3.7, 3.9, Appendix 3D, 3K, 5.2 and 5.4.

3.13 Threaded Fasteners - ASME Code Class 1, 2, and 3

Criteria applied to the selection of materials, design, inspection and testing of threaded fasteners (i.e., threaded bolts, studs, etc.) are presented in Subsection 3.9.3.9, with supporting information in Sections 4.5.1, 5.2.3, and 6.1.1.